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FISSION PRODUCT NUCLEAR DATA (FPND) - 1977

VOL.I

PROCEEDINGS OF THE SECOND ADVISORY GROUP MEETING
ON FISSION PRODUCT NUCLEAR DATA
ORGANIZED BY THE
INTERNATIONAL ATOMIC ENERGY AGENCY
AND HELD AT THE
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Summary

The Second IAEA Advisory Group Meeting on Fission Product Nuclear Data (FPND) was a follow-up meeting of the first Panel on the same subject which had been organized by IAEA/NDS in Bologna, Italy, in November 1973; the Proceedings of this Panel are published as IAEA-report in three volumes, IAEA-169 (1974).

The main purpose of the Second AGM on FPND was to re-convene users and measurers of FPND in order to review the present state of requirements for FPND as well as the development and progress in FPND research since the Bologna Panel.

Fifteen review papers were presented at this meeting, which covered the full scope of FPND and their applications, and which formed the basis for the subsequent discussions.

The principal results of this meeting were:

- detailed comparisons were performed between the accuracy status and the current requirements for FPND;
- those user areas were clearly delimited which still require an improvement in the status and accuracy of FPND;
- many detailed recommendations for future work on FPND, including coordinating activities to be performed by the IAEA, were formulated.

The meeting was attended by 52 participants from 13 Member States and 3 international organisations. Appendix A contains the list of participants, Appendix B the meeting agenda and Appendix C the working groups which were formed after the presentation of the review papers in order to discuss specific subjects.

Selected contributions to review papers are published separately as INDC(NDS)-87 report.

The scientific secretaries wish to thank the participants of the meeting for their efficient work, and ECN Petten and its staff for the hospitality and the excellent organization.

Table of Contents

Page

Volume 1

Volume 2

INTRODUCTION

G. Lammer, IAEA Vienna

1. Objectives and scope of the meeting

The importance of fission products rests on the fact that practically all stages of the nuclear fuel cycle are affected by their presence. Their behaviour in the reactor as well as at the storage sites, in the processing plants and in the environment must therefore be known as accurately as possible. In addition, fission products are often used for industrial, medical and other scientific purposes.

For the assessment, prediction and control of the effects of fission products, in many cases an accurate knowledge of fission product nuclear data (FPND) is required, the kind of data to be known (and their accuracy) varying with the application area for which they are needed.

A Panel meeting on FPND had been convened by the IAEA in Bologna, Italy, in November 1973, which reviewed for the first time the requirements of FPND in the various application fields and compared them to the status of the data.

In the four years since this Panel, not only a considerable improvement of the knowledge and accuracy of FPND has been achieved, but also new requirements for FPND have emerged, partly as a result of sensitivity studies which have been performed in the meantime, partly because the accuracy limits have been changed, and partly because new nuclear technologies have been conceived or developed.

In order to review all the development in the field of FPND since the Bologna Panel, the IAEA convened the present "Second Advisory Group Meeting on Fission Product Nuclear Data". The specific aims of this meeting are:

- to assess the development of the applications of FPND as well as of the status of the data themselves;
- to clearly define the importance of individual fission products in the various fields of applications and for the different existing calculational methods;
- to issue a list of well defined FPND requirements reflecting the present status of knowledge;
- to agree on the priorities of the requirements and to issue technical recommendations about how to fulfill the requirements according to their priorities; and

- to find out and recommend ways of appropriate cooperation and satisfactory intercommunication between data users, measurers and evaluators.

The fission product nuclear data to be reviewed for these purposes comprise

- fission yields;
- decay data;
- delayed neutron data;
- neutron cross section data.

2. Review of the recommendations of the Bologna Panel

(i) A large number of detailed recommendations concerned the performance of sensitivity studies and the improvement of FPND accuracies by new measurements or evaluations. The progress made in this respect will be reviewed in the corresponding review papers.

(ii) The Panel recommended to solicit measurers and evaluators for publication of all details of their work which are relevant for the interpretation and judgement of the results. In accordance with this recommendation, IAEA/NDS had asked several evaluators of FPND to list all the experimental details which they require for a careful evaluation. These lists were included in a circular which IAEA/NDS distributed to measurers of FPND.

(iii)In order to simplify the evaluation procedures, and for intercomparison of different evaluations, the establishment of a common computerized experimental data base was recommended as well as the use of a common computer format for evaluated data, preferably ENDF/B. It may be discussed at this meeting whether the situation with respect to computer formats has improved, and if it is now satisfactory.

(iv) The decay heat emission after reactor shutdown was considered as one of the most serious problems in connection with safety measures. In order to eliminate existing discrepancies between afterheat measurements and calculations, and to establish a data base sufficiently accurate for safety requirements, the performance of coordinated international benchmark experiments and calculations was strongly recommended by the Bologna Panel. The International Nuclear Data Committee (INDC) however noted that a number of afterheat experiments were already in progress at that time, and proposed to await and evaluate their results before deciding about a stimulation of new experiments.

In RP 15, R.E. Schenter and T.R. England are reviewing all afterheat experiments and calculations which have been performed to date, and with this information it should be possible to draw conclusions about the presently obtainable accuracies and about further needs for FPND or afterheat experiments.

(v) It was suggested that IAEA/NDS initiate an international request list for FPND, which should include a detailed justification for each requirement, so that the requirements identified by the Bologna Panel would be updated regularly and completed. The INDC proposed however to include all FPND requirements in the existing "World Request List for Nuclear Data" (WRENDA). I would like to invite the meeting participants to discuss this question further in order to find a solution which may satisfy both users and 'producers' of FPND.

(vi) An international newsletter containing information about all activities concerning FPND had been recommended by the Bologna Panel. Consequently IAEA/NDS initiated the annual report "Progress in Fission Product Nuclear Data". So far, 3 issues have been sent to measurers, evaluators and users of FPND, the distribution list including at present about 350 names. I would like to ask the opinion of the meeting participants about usefulness and possible improvements of these reports.

(vii)Finally, the Bologna Panel had recommended that the list of FPND compilations and evaluations, which was presented as RP 1b at Bologna, should be continuously updated and published regularly. This had so far not been done, but the RP 1 presented to this meeting should partially fulfill the recommendation. If requested by this meeting, IAEA/NDS may in future update and complete the list of evaluations, and publish it periodically, either in the FPND-progress report or separately.

3. Outlook

The review papers which will be presented in the meeting show that a considerable amount of research work has been performed since the time of the Bologna Panel. A lot of new data have accumulated, especially in the fields of fission yields and decay properties of short lived fission products; theoretical approaches and semi-empirical systematics have been developed, and methods for and results of adjustments of differential quantities with the help of integral data will be reported. There are, however, still a number of data which are either not measured or for which higher accuracies are required, and the meeting will have to define the justification and urgency for further measurements and/or evaluations.

A number of studies concerning the sensitivity of bulk properties to FPND accuracies have been performed since the last meeting, particularly for afterheat problems, which should enable the formulation of well defined requests.

In conclusion, it can be expected that the meeting will be in a position to establish a list of detailed and well justified requirements and to recommend ways how these requirements could be fulfilled. Finally, this meeting should stimulate further improvement of the intercommunication between users, measurers and evaluators, as to facilitate the solution of problems still existing or open to development.

Review Paper 1

REVIEW OF EXISTING COMPILATIONS AND EVALUATIONS OF FISSION PRODUCT NUCLEAR DATA

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Abstract

The intention of the present paper is to give a survey of all FPND libraries that have been published or have become available after 1970. As such, it is a complement to RP 1b of the Bologna Panel which listed all compilations and evaluations of FPND that existed in 1973.

For each library, the kind and number of data and the evaluation procedure are briefly described, which may help the user to find the most suitable library for his purpose. No attempt is made in this review to judge or compare the qualities of the different evaluations.

I. Introduction

This is a survey of nuclear data libraries which have become available after 1970 and which include the following data types:

- fission product yields;
- neutron cross sections of fission products;
- fission product decay data;
- delayed neutron data.

The description of each library includes: an artificial name of the library, composed of two figures denoting the year of the last publication (or update) and a five characters abbreviation of the library's original name or of the name of one of the authors; the titles of and references to the publications related to the library; a short description of the evaluation procedure applied or of other special features; a brief enumeration of the contents, either of the whole library or of each publication; occasionally the deadline of literature coverage and a remark if the data are available on tape.

The following Sections are each devoted to one of the above mentioned data types. Within a Section, the libraries are arranged in decreasing 'last publication year' order, and then in increasing alphabetic order.

II. Review of fission yield evaluations

II.1. List of libraries and publications

"Name"	Publication (year)	Author (comments)
77Crouc	ADNDT [1] 19 (5) 419 (1977)	E.A.C. Crouch
	(AERE-R-8152 (1976))	
	(AERE-R-7680 (1974))	" (all superseded by the above publication)
	(AERE-R-7394 (1973))	
	(AERE-R-7209 (1973))	
	(73 Paris, vol.1, 393 = IAEA/SM-170/94)	same data as AERE-R-7209
77Madla	LA-6783-MS (1977)	D.G. Madland, L. Stewart (ternary light charged particle yields)
77Meek	NEDO-12154-2 (1977)	M.E. Meek, B.F. Rider
	(NEDO-12154-1 (1974))	" " (superseded)
76BIBGR	ZJE-188 (1976)	J. Hep, V. Valenta
76Ford	LA-6129 (1976)	G.P. Ford, A.E. Norris (radiochem. measured yields)
76Madla	LA-6595-MS (1976)	D.G. Madland, T.R. England (indep. yields to isomeric states)
	LA-6430 (1976)	D.G. Madland, T.R. England (pairing effects)
75Amiel	Phys. Rev. C11,845 (1975)	S. Amiel, H. Feldstein
	73Rochester, vol.2, 65 = IAEA/SM-174/25	" "

1) ADNDT ... Atomic Data and Nuclear Data Tables

"Name"	Publication (year)	Author (comments)
74Daroc	IAEA-169, III, 281 (1974) (= Bologna Panel Proceedings)	S. Daroczy, P. Raics, S. Nagy (14 MeV U238 yields)
74Wolfs	LA-5553-MS (1974)	K. Wolfsberg (fract. indep. yields)
73Lamme	73Paris,505 (=IAEA/SM-170/13)	M. Lammer, O.J. Eder
73Netha	UCRL-51458 (1973)	D.R. Nethaway, G.W. Barton
73Walke	AECL-3037, Part II (1973)	W.H. Walker
72Sideb	TRG-report 2143 (R) (1972)	E.W. Sidebotham

II.2. Description of the fission yield libraries [2)]

77Crouc: a) At. Data and Nucl. Data Tables 19 (5) 419 (1977), E.A.C. Crouch: "Fission product yields from neutron-induced fission".

This paper replaces and makes obsolete the references b) to e).

Literature covered: up to 1975.

Availability: the compilation is updated continuously; compiled and evaluated data are available on tape, evaluated data in ENDF/B-IV format.

2) abbreviations:

A	...	atomic number
CHY	...	chain yield
g	...	subscript referring to ground state
IY	...	independent yield
m	...	subscript referring to metastable state
ms	...	mass spectrometric
rms	...	'root mean square'
UCD	...	unchanged charge distribution
w.a.	..	weighted average
Y	...	yield
Z	...	element number
Z_p	...	most probable charge
σ	...	Gaussian width

Evaluation procedure:

(i) Recommended cumulative and independent yields:

All published experimental cumulative and independent yields are renormalized when necessary and possible. An uncertainty is attached to each value, which is in most cases the uncertainty given by the authors, or an estimate when no error is given in the publication; in a few cases, the quoted uncertainties are adjusted to comply with statistics.

To obtain recommended chain, cumulative and independent yields, w.a. of the renormalized experimental values are calculated. Those independent yields for which no experimental data are available are obtained by interpolation, assuming a Gaussian charge dispersion model and unchanged charge distribution for the determination of $Z_p(A)$ and $\sigma(A)$.

(ii) Adjusted chain and independent yields:

For some important fissioning systems, consistent sets of FP chain and independent yields are calculated. With the recommended chain and independent yields (see above) as input data, a least squares fit of the whole mass distribution curve ($72 \leq A \leq 161$) of a fissioning system to some "physical laws is performed. The following conservation rules are used: the normalization rule, Σ_A CHY(A) = 200%, the mass conservation rule, Σ_A CHY(A).A = $(A_{fissile}+1-\bar{\nu})$ x 100, and the charge conservation rule, Σ_A CHY(A).$\bar{Z}(A) = Z_{fissile}$, $\bar{Z}(A)$ being the mean atomic number of the decay chain with mass A.

Contents:

(i) all experimental cumulative and independent FP yields found in the literature, and the deduced recommended values and uncertainties. The following neutron-fission systems are included:

Ac227 :	fast	Pu239 :	thermal, fast, 14MeV
Th227 :	thermal	Pu241 :	thermal
Th229 :	"	Am241 :	"
Th232 :	fast, 3MeV, 11MeV, 14MeV	Am242 :	"
Pa231 :	fast, 3MeV, 14MeV	Cm245 :	"
U233 :	thermal, fast, 14MeV	Cm249 :	"
U235 :	" " "	Cf251 :	"
U238 :	fast, 3MeV, 14MeV	Es254 :	"
Np237 :	fast, 1.1MeV, 14MeV	Fm255 :	"

(ii) adjusted chain and fractional independent yields and their uncertainties in the mass range $72 \leq A \leq 161$ and from the following fission reactions:

Th232	:	fast, 14MeV
U233	:	thermal, fast, 14MeV
U235	:	" " "
U238	:	fast, 14MeV
Pu239	:	thermal, fast
Pu240	:	fast
Pu241	:	thermal, fast

The following publications are superseded by the above:

b) AERE-R-7209 (1973), E.A.C. Crouch: "Fission product chain yields from experiments in thermal reactors". (= paper IAEA/SM-170/94, Proceedings of the Symposium on Nuclear Data in Science and Technology, Paris 12-16 March 1973, vol.I, p.393)

c) AERE-R-7394 (1973), E.A.C. Crouch: "Fission product chain yields from experiments in reactors and accelerators producing fast neutrons of energies up to 14MeV".

d) AERE-R-7680 (May 1974), E.A.C. Crouch: "Assessment of known independent yields and the calculation of those unknown in the fission of ^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu and ^{241}Pu".

e) AERE-R-8152 (Jan. 1976), E.A.C. Crouch: "Chain and independent fission product yields adjusted to conform with physical conservation laws. Part 2". (= improvement of part 1, AERE-R-7785).

77Madla: LA-6783-MS (1977) (ENDF-247), D.G. Madland, L. Stewart: "Light ternary fission products: probabilities and charge distributions".

Evaluation procedure:

(i) emission probabilities: experimental results for N=number of light charged particles per 1000 fissions are collected. The general dependence of N from the fissioning compound nucleus and from the excitation energy is estimated by fitting linear functions of $(4Z-A)_{compound}$ resp. of ϵ, the excitation energy with respect to the outer fission barrier, to the data.

(ii) charge distribution: relative data compiled by Halpern [3] have been normalized with values for N from the compilation described above. Resulting absolute light charged particle (Z≤4, A≤ 10) yields are listed.

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3) I. Halpern, "Three Fragment Fission", Annual Review of Nucl. Sci. 21, E. Segrê, Ed. (Annual Reviews Inc., Palo Alto, Calif. 1971), p. 245.

Contents:

(i) emission probabilities: evaluated N-values and uncertainties for neutron-induced fission, at various incoming energies, of the following fissionable nuclides:

Th232, U233, U235, U238, Pu239, Pu241,

and formulae for interpolation with respect to fissioning nuclide and excitation energy.

(ii) charge distribution: absolute yields and uncertainties of light charged particles ($Z \leq 4$, $A \leq 10$) for thermal fission of U233, U235 and spontaneous fission of Cf252.

77Meek: NEDO-12154-2 (1977), M.E. Meek and B.F. Rider: "Compilation of Fission Product Yields" (supersedes NEDO-12154-1 (1974)).

Availability: the compilation is continuing; the data are available on tape in intermediate ENDF/B-V format.

Evaluation procedure: w.a. of corrected and renormalized experimental independent and cumulative yields is calculated. Resulting independent and cumulative yields, complemented by calculated ones where experimental data are missing, are evaluated in a common treatment which takes into account the 'odd-even effect' and the isomeric yield distribution (76Madla), and which is similar to the maximum likelihood method. The resulting values are finally adjusted in order to give Σ CHY (light) = Σ CHY (heavy) = 100%. The adjusted yields are given as the recommended values.

Contents:

Recommended independent and cumulative yields in the mass range $66 \leq A \leq 172$ and from the following fission reactions:

Th232	:	fast, 14MeV
U233	:	thermal, fast, 14MeV
U235	:	" " "
U236	:	fast
U238	:	fast, 14MeV
Np237	:	" "
Pu239	:	thermal, 14MeV
Pu240	:	fast
Pu241	:	thermal, fast
Pu242	:	fast
(Cf252	:	spontaneous)

76BIBGR: ZJE-188 (1976), J. Hep, V. Valenta: "Group library of fission products - BIBGRFP".

Availability: the data are available on tape.

The evaluation procedure is not given in the document, reference is made to an internal report (in Czech): Ac-3771/Dok-Zpr.ZVJE about a "YIELDS"-library, where more information may be contained.

Contents:

Evaluated(?) independent yields in the mass range $71 \leq A \leq 165$ from the following fission reactions:

U235 : thermal, 1MeV
U238 : 1MeV
Pu239 : thermal, 1MeV
Pu241 : thermal, 1MeV

76Ford: LA-6129 (Feb. 1976), G.D. Ford, A.E. Norris: "A compilation of yields from neutron-induced fission of ^{232}Th, ^{235}U, ^{236}U, ^{237}Np, ^{238}U, and ^{239}Pu measured radiochemically at Los Alamos".

Evaluation: the evaluation consists in converting the measured relative yields to absolute yields, after a consistency check by the 2-mode-of-fission hypothesis ($Y_{ji} = a_i Y_{j,111} + b_i Y_{j,99}$; a_i, b_i ...constant j ... fissioning nucleus, i ... FP) and careful investigation of the reasons for inconsistencies.

Contents:

R-values (based on mass 99 yield from U235 thermal fission) and their uncertainties, and the chain yields, for those fissioning nuclei given in the title and for different neutron sources.

76Madla: a) LA-6595-MS (Nov. 1976), D.G. Madland, T.R. England: "Distribution of independent fission product yields to isomeric states".

b) Nucl. Sci. and Engg. 64 (1977) 859, D.G. Madland, T.R. England: "The influence of isomeric states on independent fission product yields".

Calculation method: The simplifying assumption is made that the ratio IY(meta)/IY(ground) is only dependent on the differences between the angular momentum parameter J_{rms} of the initial fragment and J_m and J_g respectively; and that J_{rms} = const. x E_n (incoming neutron-energy), where 'const.' refers to both actinides and fission fragments.

Contents:

$$\text{Calculated } R = \frac{\text{IY (meta)}}{\text{IY (ground)}}$$ for isotopes in the mass range

$69 \leq A \leq 170$ and for the incoming energies: ≤ 0.5 MeV, 14MeV (valid for all actinides); compared to the very few experimental values available.

c) LA-6430-MS (July 1976), D.G. Madland, T.R. England: "The influence of pairing on the distribution of independent yield strengths in neutron-induced fission".

Evaluation procedure: From an analysis of the U235 thermal fission yield data from 75Amiel, the average fractional enhancements relative to the "normal" yield due to proton and neutron pairing, X and Y respectively, are derived.

By comparison and analysis of different studies of other fission systems, an empirical (simple) relation between excitation energy and X,Y is proposed.

Contents: X and Y values (and uncertainties) for neutron-induced fission in 17 nuclides are tabulated:

Th229,232; U233-238; Pu238-242,244; Cm242,244,249.

75Amiel: a) IAEA-SM-174/25 (1973), S. Amiel and H. Feldstein: Proceedings of the Symposium on Physics and Chemistry of Fission, Rochester, 13-17 August 1973, vol.2 p.65: "A systematic odd-even effect in the independent yield distributions of nuclides from thermal-neutron-induced fission of ^{235}U".

Evaluation procedure: published experimental fractional cumulative and fractional independent yields are compiled. Best values for fractional independent yields and missing data are chosen, respectively calculated according to Wahl's Gaussian distribution multiplied by an odd-even factor (which was derived from available experimental element or isotopic independent yield distributions).

Contents:

"Corrected" fractional independent yields and their uncertainties in the mass ranges $83 \leq A \leq 97$ and $130 \leq A \leq 145$ from thermal fission of U235; compilation of experimental data in the same mass-range.

b) Phys. Rev. C11, 845 (1975), S. Amiel, H. Feldstein: "Odd-even systematics in neutron fission yields of ^{233}U and ^{235}U".

Evaluation procedure: similar to a) above.

Contents:

"Corrected" fractional independent yields and their uncertainties for U235 thermal fission as in a), for U233 thermal fission and $83 \leq A \leq 94$, $131 \leq A \leq 143$, and for a few mass chains from U235 fast fission.

74Daroc: IAEA-169, vol.III, 281 (1974), S. Daroczy, P. Raics, S. Nagy (Proceedings of the Bologna Panel): "Compilation of fission product yields of U-238 for 14MeV neutrons".

Literature covered up to 1972.

Evaluation procedure: After evaluation (w.a.) of the Mo99 and Ba140 yields, experimental values for chain (or cumulative) yields are renormalized to these standards, where possible. Missing data are obtained from exponential interpolation.

Contents:

Complete set of evaluated chain yields and uncertainties in the mass range $66 \leq A \leq 172$ from 14MeV fission of U238. Compilation of experimental cumulative and independent yields.

74Wolfs: LA-5553-MS (May 1974), K. Wolfsberg: "Estimated values of fractional yields from low energy fission and a compilation of measured fractional yields".

Estimation: A Gaussian charge dispersion with $\sigma=0.56 \pm 0.06$ is assumed for all primary nuclides considered, and Z_p is calculated according to 2 different formulae for $\Delta Z = Z_{UCD} - Z_p$. Odd-even factors are taken from 75Amiel or extrapolated.

Contents:

Estimated fractional independent yields and their uncertainties in the mass range $74 \leq A \leq 161$, and comparison to compiled experimental data, for the following fission reactions:

U233 : thermal
U235 : thermal, 14MeV
U238 : fast, 14MeV
Pu239 : thermal, fast, 14MeV

73Lamme: IAEA-SM-170/13 (1973), M. Lammer, O.J. Eder, Proceedings of the Symposium on Nuclear Data in Science and Technology, Paris, 12-16 March 1973: "Discussion of fission product yield evaluation methods and a new evaluation".

Evaluation procedure: Published experimental yields were checked carefully and corrected for more recent nuclear data, where possible. Yields suffering from uncertain or wrong corrections were rejected. Relative isotopic yields from mass-spectrometric (ms) measurements are calculated first, then linked by element yields, to establish the yield curve. Non-ms yields are used for the final normalization to make the sum of chain yields in each mass peak 100%.

Contents:

Experimental cumulative and recommended chain yields for the following fissioning systems:

$$\left.\begin{array}{r} \text{U233} \\ \text{U235} \\ \text{Pu239} \end{array}\right\} \text{thermal}$$

and Th232 : fast (revised in IAEA-169, Vol.III, 245 (1974)).

73Netha: UCRL-51458 (Oct. 1973), D.R. Nethaway, G.W. Barton: "A compilation of fission product yields in use at Lawrence Livermore Laboratory".

Procedure: Data from NEDO-12154 (1972) (see 77Meek) are used as far as available. Other data are compiled and averaged (w.a.), the errors had often to be assigned arbitrarily. Fractional yields are calculated from Gaussian charge dispersion with $\sigma = 0.56$, with a simple account for odd-even effect.

Contents:

Cumulative yields, R-values (rel. Mo99) and 'Q-values'

$$\left(Q(A) = \frac{Y_i\ (A)}{Y_{standard}\ (A)} \qquad i \text{ ...fissionable nucleus under consideration}\right)$$

of the 111 most important nuclides between Ni^{66} and Yb^{175}, and the respective uncertainties, from the following fission reactions:

$$\left.\begin{array}{r} \text{Th232} \\ \text{U233} \\ \text{U235} \\ \text{U238} \\ \text{Pu239} \end{array}\right\} \text{thermal, fast, 15MeV}$$

73Walke: AECL-3037, Part 2 (April 1973), W.H. Walker: "Fission product nuclear data for thermal reactors".

Literature covered up to 1970 (1971).

Evaluation procedure: similar to 73Lamme, but taking w.a. of isotopic yields.

Contents:

Compilation of experimental cululative yields and recommended mass yields and uncertainties in the range $77 \leq A \leq 160$ from the following fission reactions:

U233, U235, Pu239, Pu241 } thermal

72Sideb: TRG-report 2143 (R) (1972), E.W. Sidebotham: "Fission product yield data extrapolated for some actinides".

Literature covered: (used as basis for extrapolation) up to 1969.

Extrapolation: the mean masses of heavy and light pre-neutron emission fragments are calculated according to empirical formulae and the mass conservation law, and the mean masses after neutron emission (~ peak-positions) are derived from the (energy + mass)-conservation rule and empirical values for parameters involved ($\bar{\nu}$, $\bar{E}_\gamma$, etc); yields of actinides with unknown yields are then derived from the assumption that the shape of their distribution is the same as for the closest actinide with known yields.

Contents:

"Known" chain yields (i.e. taken from other compilations) in the mass range $72 \leq A \leq 162$ for the fission reactions:

U233 : thermal
U235 : " , fast
U238 : fast
Pu239 : thermal, fast
Pu241 : thermal

and extrapolated chain yields (in the same mass range) from:

Th232 :	fast	Pu240 :	thermal, fast
U234 :	fast	Pu241 :	fast
U236 :	fast	Pu242 :	thermal, fast
U237 :	thermal, fast	Am241 :	" "
Np237 :	" "	Am243 :	fast
Np238 :	fast	Cm242 :	"
Pu238 :	fast		

II.3. Survey of fission yield libraries (compilations and evaluations), sorted by actinides:

t ... thermal neutrons   f ... fission neutrons   h ... high energy = 14 MeV neutrons

M ... 'medium' energy = MeV neutrons

Ac227 f	Th227 t	Th229 t	Th232 t	Th232 f	Th232 M	Th232 h
77Crouc	77Crouc	77Crouc	73Netha	77Crouc 77Meek 76Ford 73Netha 73Lamme 72Sideb	77Crouc	77Crouc 77Meek 73Netha
Pa231 f	**Pa231 M**	**Pa231 h**	**U233 t**	**U233 f**	**U233 h**	**U234 f**
77Crouc	77Crouc	77Crouc	77Crouc 77Meek 75Amiel 74Wolfs 73 Lamme 73 Netha 73Walke 72Sideb	77Crouc 77Meek 73Netha	77Crouc 77Meek 73Netha	72Sideb
U235 t	**U235 M**	**U235 f**	**U235 h**	**U236 f**	**U237 t**	**U237 f**
77Crouc 77Meek 76Ford 76BIBGR 75Amiel 74Wolfs 73Lamme 73Netha 73Walke 72Sideb	76BIBGR	77Crouc 77Meek 75Amiel 73Netha 72Sideb	77Crouc 77Meek 74Wolfs 73Netha	77Meek 76Ford 72Sideb	72Sideb	72Sideb
U238 t	**U238 f**	**U238 M**	**U238 h**	**Np237 t**	**Np237 f**	**Np237 M**
73Netha	77Crouc 77Meek 76Ford 74Wolfs 73Netha 72Sideb	77Crouc 76BIBGR	77Crouc 77Meek 74Wolfs 74Daroc 73Netha	76Ford 72Sideb	77Crouc 77Meek 76Ford 72Sideb	77Crouc

Np 237 h	Np 238 f	Pu 238 f	Pu239 t	Pu239 f	Pu239 M	Pu239 h
77Crouc	72Sideb	72Sideb	77Crouc 77Meek 76Ford 76BIBGR 74Wolfs 73Lamme 73Netha 73Walke 72Sideb	77Crouc 76Ford 74Wolfs 72Sideb	77Crouc 76BIBGR	77Crouc 77Meek 74Wolfs
Pu240 t	Pu240 f	Pu240 h	Pu241 t	Pu241 f	Pu241 M	Pu242 t
77Crouc 72Sideb	77Crouc 77Meek 72Sideb	77Crouc	77Crouc 77Meek 76BIBGR 73Walke 72Sideb	77Crouc 77Meek 72Sideb	76BIBGR	72Sideb
Pu242 f	Am241 t	Am241 h	Am242m t	Am243 f	Cm242 f	Cm245 t
77Meek 72Sideb	77Crouc 72Sideb	72Sideb	77Crouc	72Sideb	72Sideb	77Crouc
Cf249 t	Cf252 spon	Es254 t	Fm255 t			
77Crouc	77Meek	77Crouc	77Crouc			

III. Review of evaluations of FP cross sections

III.1. List of libraries and publications

"Name"	Publication (year)	Author (comments)
77Benzi	(INDC(NDS)-75,p.44 (1976))	V. Benzi et al. (40 FP)
77JENDL	J. Nucl.Sci.Tech. 14,161(1977)	S. Iijima et al. (detailed description of method)
	NBS-Spec.Pub. 425 (=75 Wash.), p. 320	S. Igarasi et al.
	JAERI-M-6001 (1975)	Y. Kikuchi et al. (preliminary version)
	JAERI-M-5752 (1974)	S. Igarasi et al. (preliminary)
77RCN-2	ECN-13 (1976) ECN-14 (1977)	H. Gruppelaar J.W.M. Decker } (24 nuclides)

"Name"	Publication (year)	Author (comments)
76ENDF	BNL-NCS-50545(ENDF-243)(1976)	P.F. Rose, T.W. Burrows (summary)
	LA-6116-MS(ENDF-223 (1975)	T.R. England, R.E. Schenter (summary)
75Ribon	CEA-N-1832 (1975)	P. Ribon et al. (26 nuclides)
74Cook	AAEC/TM-619 (1972)	E. Clayton (thermal and res.int.
	AAEC/E-214 (1971)	W.K. Bertram et al. (128-group cross sections)
	AAEC/TM-549 (1970)	J.L. Cook (list of nuclides, no data)
	AAEC/E-198, Suppl.1 (1970)	A.R. de L. Musgrove (res.-params, only few FP)
	(AAEC/E-198 (1969))	(superseded by Suppl.1)
73RCN-1	RCN-191 (1973)	G. Lautenbach (RCN-1 set, partly basis for 77RCN-2)
72Walke	AECL-3037, Part I (rev. 1972)	W.H. Walker (thermal and res. int.)
71Benzi	RT/FI(72)6 (=CEC(71)-9) (1972)	V. Benzi et al.
	CEC(71)2 (1971	" "
	CCDN-NW/10 (1969)	" " (includes detailed description of evaluation)

III.2. Description of FP cross section libraries

III.2.a. Evaluation methods in general

Most evaluations of neutron reaction cross sections provide recommended point or group data which cover incoming neutron energies up to about 15MeV. Usually, three energy ranges are discerned for which the evaluation procedures are different:

Thermal and resolved resonance region: the cross section is calculated from the resonance parameters with a Breit-Wigner formalism (single or multilevel); then the thermal cross section is adjusted to the experimental value by an additive residual cross section (for which resonance at negative energy may be introduced).

Statistical region (~ 1 to 500keV): the cross section is calculated with the statistical model in which the penetrability factor is obtained from a level density formula. Dobs, Γ_γ and S are obtained either from experiments, systematics, theories or other evaluations (like BNL 325), and often adjusted to experimental data.

above about 500keV: the statistical model with width fluctuations is used in which the transmission coefficient is obtained from the optical model. The result is often checked by inelastic scattering data.

III.3.b. Publications

77Benzi: INDC(NDS)-75, p.44 (1976), V. Benzi et al; only an abstract is given.

Availability: The files for the 63 FP nuclides contained in the joint "CNEN-CEA Preliminary Evaluation, 1977" (see also 75Ribon) are available in ENDF/B-format.

Contents:

The CNEN Bologna evaluation includes evaluated resolved and mean resonance parameters, total, elastic inelastic, (n,2n), (n,γ), (n,α) cross sections, energy and angular distributions (energy range: 10^{-5}eV to 15MeV) for the following 41 FP nuclides:

Rb85; Y91; Zr91-96; Nb95; Ru100,106; Pd104,106,108,110; Cd111; In115; Te128; Ba138,140; La139; Ce140-142,144; Pr143; Nd144,146-148, 150; Sm147,150152,154; Eu154,155; Gd156,157; Tb159.

77JENDL: a) J. Nucl. Sci. and Techn. 14, 161 (1977), S. Iijima, T. Nakagawa, Y. Kikuchi, M. Kawai, H. Matsunobu, K. Maki, S. Igarasi: "Evaluation of neutron cross sections of 27 fission product nuclides important for fast reactors".

b) NBS-Spec. Publ. 425, 320 (1975) (Proceedings of the Conference on Nuclear Cross Sections and Technology, Washington, 3-7 March 1976) S. Igarasi, S. Iijima, M. Kawai T. Nakagawa, Y. Kikuchi, K. Maki, H. Matsunobu: "Evaluation of fission product nuclear data for 28 important nuclides".

Evaluation method: the optical model is also used for the unresolved resonance region; special care is given to the connection between resonance region and statistical region.

Availability: the JENDL-1 FPND library, which at present contains data for the 27 nuclides included in the publications a) and b) and for additional 34 FP nuclides (see below) are available on tape in ENDF/B-IV format. Evaluation for 29 other FP nuclides is being prepared.

Contents:

Total, elastic, inelastic, capture cross sections from thermal to 15MeV for the following nuclides (those marked with * are included in the publications a) and b)):

Br81; Kr84; Rb85,87; Y89; Zr90,91,92,93*,94,96; Nb93; Mo95*; Mo97*; Tc99*; Ru101*,102*,104*,106*; Rh103*; Pd104,105*,107*,108,110; Ag107,109*; Te128,130; I127,129*; Xe131*; Cs133*,135*,137*; Ba138; La139; Ce140,142,144*; Pr141; Nd143*,144*,145*,146,148,150; Pm147* Sm147*,148,149*,150,151*,152,154; Eu151,153*,155*; Gd155,156,157.

c) JAERI-M-5752 (1974), S. Igarasi et al.: "Evaluation of fission product nuclear data for fast reactors (neutron cross sections for 28 nuclides)".
Same cross sections as above, superseded by the above evaluation.

d) JAERI-M-6001 (1975), Y. Kikuchi, A. Hasegawa, K. Tasaka, H. Nishimura, I. Otake, S. Katsuragi: "JNDC Fission Product Group Constants - Preliminary version".

Contents:

70- and 25-group cross sections (total, elastic, inelastic, capture) weighted by 1/E flux for energies below 1MeV, by fission spectrum above 1MeV, based on evaluated data of JAERI-M-5752.

77RCN-2: a) ECN-13 (1976), H. Gruppelaar: "Tables of RCN-2 fission product cross section evaluation, part 1" (24 nuclides)

b) ECN-14 (1977), J.W.M. Dekker: "Tables and figures of adjusted and unadjusted capture group cross sections based on the RCN-2 evaluation and integral measurements in STEK, part 1".

Availability: the RCN-2 library contains at present point and 26-group data for 24 nuclides (see below) in KEDAK-format and will be available soon; evaluation for other 30 FP nuclides is ongoing.

Contents:

Evaluated total, elastic, capture, inelastic, (n,2n) cross sections (ECN-13) and capture cross sections adjusted to integral measurements (ECN-14) up to 15 MeV and covariance matrices for the 26-group constants, for 24 nuclides:

Nb93; Mo92,94,95,96,97,98,100; Tc99; Ru101,102,104; Rh103; Pd102,104,105,106,107,108,110; I127; Cs133; La139; Pr141.

76ENDF: a) BNL-NCS-50545 (ENDF-243) (1976), P.F.Rose, T.W. Burrows: "ENDF/B Fission Product Decay Data". This report contains the thermal total and capture cross sections and the total and capture resonance integrals as extracted from the ENDF/B-IV files.

b) LA-6116-MS (ENDF-223)(1975), T.R. England, R.E.Schenter: "ENDF/B-IV Fission Product Files - Summary of major nuclide data". This report contains the thermal capture cross sections and resonance integrals as extracted from the ENDF/B-IV files.

Availability: The ENDF/B-IV FP cross section data are available on tape.

Contents:

The ENDF/B-IV library contains evaluated total, elastic, inelastic and capture cross sections, and energy distributions of inelastically scattered neutrons for 181 nuclides, i.e. all stable and some radioactive fission products; energy-range: 10^{-5}eV to 20MeV.

75Ribon: CEA-N-1832 (1975), P. Ribon, E. Fort, J. Krebs, T.Quoc Thuan: "Evaluation of capture and inelastic cross sections of 26 fission products".

Availability: the data for 22 nuclides are available within the joint "CNEN-CEA Preliminary Evaluation, 1977" (see also 77Benzi), in ENDF/B-format.

Evaluation method: the resonance parameters are also evaluated in this work.

Contents:

Recommended resonance parameters and capture cross sections and calculated inelastic cross sections in the energy range 1keV-1MeV for the following FP-isotopes:

Mo95,97,97,100; Tc99; Ru101,102,103,104; Rh103; Pd105,107,108; Ag109; I127,129; Cs133,135; La139; Pr141; Nd143,145; Pm147; Sm149,151; Tb159.

74Cook: a) AAEC/TM-619 (1972), E. Clayton: "Thermal capture cross sections and resonance integrals for the AAEC fission product library".

b) AAEC/E-214 (1971), W.K. Bertram, E. Clayton, J.L. Cook, H.D. Ferguson, A.R. Musgrove, E.K. Rose: "A Fission Product Cross Section Library". Containing tables of 128-group cross sections (total, elastic, inelastic, (n,γ)).

c) AAEC/TM-549 (1970), J.L. Cook: "Fission Product Cross Sections". No data are given in this reference.

Availability: The Australian FP library has been updated in 1973 and 1974. The updated versions in UKNDL-format, are available on tape.

Contents:

For 192 FP nuclei (184 ground states and 8 isomeric states), comprising almost all stable isotopes and the important radioactive ones, the evaluated total, elastic, non-elastic, capture, inelastic and transport cross sections are given between 10^{-3}eV and 15MeV, with 223 data points for each reaction.

73RCN-1: RCN-191 (1973), G. Lautenbach: "Calculated neutron absorption cross sections of 75 fission products".

Evaluation method: the level density formula was applied to the whole energy region above the resolved resonances.

Contents:

RCN-1 set= 26-group (n,γ) cross sections for the following 75 fission products:

Br81; Kr83,84,85,86; Rb85,87; Sr88,90; Y89; Zr91,92,93,94,96; Mo95,97,98,100; Tc99; Ru101, 102, 104, 106; Rh103; Pd105,106,107, 108,110; Ag109, Cd111,112,113,114; In115; Te126,128,130; I127,129; Xe131,132,134,136; Cs133,135,137; Ba138; La139; Ce140,142; Pr141; Nd143,144,145,146,148,150; Pm147; Sm147,148,149,150,151,152,154; Eu153,154,155; Gd155,156,157,158; Tb159.

72Walke: AECL-3037, Part 1 (Dec. 1969, revised Jan. 1972), W.H. Walker: "Fission Product Data for Thermal Reactors, Part 1 - Cross Sections".

Evaluation procedure: calculation of w.a. of consistent experimental values, elimination or correction of 'outlying' data; estimates for nuclides for which no measured cross sections exist.

Contents:

Experimental and recommended (n γ) cross sections for 2200m/sec neutrons and resonance integrals and their uncertainties for FP with half-lives $\gtrsim$ 5h, A = 76-165.

71Benzi: a) CCDN-NW/10, p.6 (Dec. 1969), V. Benzi, G. Reffo: "Fast neutron capture cross sections of stable nuclei with $32 \leq Z \leq 66$".

b) CEC(71)2 (1971), V. Benzi, G.C. Panini, G. Reffo, M. Vaccari: "An estimate of (n,n'), (n,2n) and (n,γ) excitation functions for some fission product nuclei".

c) CEC(71)9 (1971) (= RT/FI(71)6), V. Benzi, R. D'Orazi, G. Reffo, M. Vaccari: "Fast neutron radioactive capture cross sections of stable nuclei with $29 \leq Z \leq 79$".

Contents:

Tables of parameters adopted for calculations, adopted level schemes, (n,γ) cross sections of all stable nuclei with $29 \leq Z \leq 79$, (n,n') and (n,2n) cross sections of some of them; energy range: 1keV to 10MeV.

IV. Review of evaluations of FP decay data [4]

IV.1. List of libraries and publications

Only such evaluations are included that contain at least half-lives, total decay energies, or radiation energies and intensities. Some of them are specifically FP-oriented, and only those usually include the very neutron rich nuclei.

"Name"	Publication (year)	Author (comments)
77Tobia	CEGB-RD/B/N4053 (1977)	A. Tobias (library composed of 76ENDF and 73Tobia data)
76BIBGR	ZJE-188 (1976)	J. Hep, V. Valenta
76ENDF	BNL-NCS-50545(ENDF-243)(1976)	P.F. Rose, T.W. Burrows

4) abbreviations:
ADNDT ... Atomic Data and Nuclear Data Tables
Augere. ... Auger electrons
BR ... branching ratio
c.e. ... conversion electrons
Es ... energies
HL ... half life
Ints ... intensities
IT ... internal transition
Q ... total decay energy
s ... second(s)
w.a. ... weighted average
y ... year(s)

"Name"	Publication (year)	Author (comments)
76ENDF	LA-6116-MS (1975)	T.R. England, R.E. Schenter
76Marti	ORNL-5114 (1976)	M.J. Martin (compilation of data from Nuclear Data Sheets)
	(Nucl. Data Tables A8, 1; ORNL-4923 (1973))	(Superseded by the above)
76Wapst	ADNDT 17, 474 (1976)	A.H. Wapstra, K. Bos (mass evaluation)
	(Nuclear Data Tables 9, 267 (1971))	(A.H. Wapstra, N.B. Gove; superseded by the above)
75Devil	CEA-N-1822 (1975)	J. Blachot et al (French file)
	(CEA-N-1526)	(Superseded by the above)
74Bowma	ADNDT 13 (2-3),204 (1974)	W.W. Bowman, K.W. Macmurdo
	(Nuclear Data Tables 8, 445 (1971))	(M.A. Wakat, superseded by the above)
74Legra	'Table of radionuclides' (Lab. de Métrol. de Rayonnements Ionisants) (1974)	J. Legrand et al
74Meixn	JÜL-1087-RX (1974)	Ch. Meixner
73Erdtm	JÜL-1003-AC (1973)	G. Erdtmann, W. Soyka
73Tobia	CEGB-RD/B/M-2669 (1973)	A. Tobias
71Sangiu	'Table of γ-Rays' (CESNEF, 1971)	V. Sangiust, M. Terrani

IV.2. Description of the decay data libraries

IV.2.a. Evaluation procedures

Recommended values for half-lives, and energies and intensities of the principal radiations (α, β, γ) are mostly obtained either from the reference considered to be the 'most reliable' one, or as weighted averages of statistically consistent data.

Other recommended decay parameters, like the Q-values, conversion electron coefficients, X-rays and Auger electrons etc, are usually derived from a combination of available experimental data and theoretical evaluations (the results of which are often available as numerical data tables).

IV.2.b. Publications

77Tobia: CEGB-RD/B/N4053 (1977), A. Tobias: "An ordered table of gamma radiation derived from an ENDF/B-IV fission product data file".

Availability: the data are available in ENDF/B-IV format.

Contents:

The data from the US ENDF/B-IV fission product file (see 75ENDF) and from the UK evaluated data file (see 73Tobia) have been combined into one library in ENDF/B-IV format.

The report includes, for more than 700 radioactive isotopes with $27 \leq Z \leq 68$ the following parameters:

HL, $\langle E_\beta \rangle$, $\langle E_\gamma \rangle$, Es+Ints of γ-rays.

76BIBGR: ZJE-188 (1976), J. Hep, V. Valenta: "Group library of fission-products - BIBGRFP". (see also Sect. II.2.)

Contents:

γ-intensities of 13 energy-groups for 500 radioactive FP nuclides (including 150 metastable states). The data are on tape.

76ENDF: a) BNL-NCS-50545 (ENDF-243) (Aug. 1976), P.F. Rose, T.W. Burrows: "ENDF/B Fission Product Decay Data".

Availability: the ENDF/B-IV FP decay data are available on tape.

Contents:

For 711 radioactive nuclides of 96 mass-chains ($72 \leq A \leq 167$) the following evaluated decay properties and their uncertainties are included:

HL, Q, BR, $\langle E_\gamma \rangle$, $\langle E_\beta \rangle$, $\langle E_\alpha \rangle$; Es+Ints of various radiations (γ, β, X-ray, c.e., Auger e.) for 180 nuclides.

b) LA-6116-MS (ENDF-223) (Oct. 1975), T.R. England, R.E. Schenter: "ENDF/B-IV Fission Product Files: Summary of Major Nuclide Data".

This report includes, for the same FP (see above):

HL, Q, BR, $\langle E_\beta \rangle$, total E_γ (including c.e.), $\langle E_\alpha \rangle$.

76Marti: ORNL-5114 (March 1976), M.J. Martin: "Nuclear Decay Data for Selected Radionuclides".

Availability: the evaluations for the Evaluated Nuclear Structure Data File (ENSDF) are continuing, the data are available on tape.

Contents:

Summary of the mass-chain evaluations performed by the Nuclear Data Project (and published in the Nuclear Data Sheets) listing for 194 important radioactive nuclides, which include about 60 FP, the following decay properties and their uncertainties:

HL, Es+Ints of α-, β-, γ-rays, c.e., X-rays, Auger e. (K-,L-).

76Wapst: Atomic Data and Nucl. Data Tables 17 (5/6) 474 (1976), A.H. Wapstra and K. Bos: "A 1975 midstream atomic mass evaluation".

An interim update of the mass evaluation presented in Nuclear Data Tables 9, 267 (1971) by A.H. Wapstra and N.G. Gove. A more careful analysis of the recent experimental data has been performed in the meantime and the results are published in At. Data and Nuclear Data Tables 19(3) 177 (1977).

Contents:

Table of evaluated mass-excess values (= M(A,Z) [amu]-a) for 1300 nuclides, compared to the results obtained from calculations with 9 different mass-formulae.

75Devil: CEA-N-1822 (1975), J. Blachot, C. Devillers, R.de Tourreil, C. Fiche, B. Nimal, J.C. Noël: "Library of Data for Fission Products".

Availability: the report lists the contents of the 'French file', which is available on tape. The evaluation is continuing.

Method: Essentially the data from the Nuclear Data Sheets are taken over, but more recent publications are also considered and, when their data are more accurate, they are selected for inclusion in the library.

Contents:

For 635 FP ($71 \leq A \leq 170$) the following decay parameters, and - where available - the uncertainties, are given:

HL, Q, BR, Es+Ints of β^+, β^-, γ rays.

74Bowma: a) At. Data and Nucl. Data Tables 13(2-3), 204 (Feb. 1974), W.W. Bowmann, K.W. MacMurdo: "Gamma rays ordered by energy and nuclide".

b) Nucl. Data Tables 8, 445 (1971), M.A. Wakat: "Catalogue of γ-rays emitted by radionuclide" is superseded by the above reference.

Contents:

Fo all known radioactive nuclides with HL>0.1 sec, the following decay parameters are given, as selected from the 'most realiable' reference.

HL, Es+Ints (relative or absplute, as given in original reference) of γ- and X-rays.

74Legra: "Table of radionuclides" (1974), Lab. de Métrologie de Rayonnements Ionisants; J. Legrand, J.P. Pérolat, F. Lagoutine, Y. Le Gallic.

Evaluation method: careful analysis of published data, which are only taken into account when uncertainties are given. Determination of best values by calculating the w.a. and adjusting them to give $\sum$BR = 1 and $\sum$Es = Q; in case of incompatible values, a new measurement is undertaken at the laboratory.

The evaluation is continuing.

Contents:

For 24 nuclides, including 13 FP (Ru-103, Rh-103m, Ag110, Ag-110m, Sb-125, Te-125m, Xe-133, Xe-133m, Cs-234, Cs-134m, Ce-144, Pr-144, Er-169) the following evaluated data are given:

HL; Es+Ints of α-, β-, γ-rays, c.e., X-rays and Auger e.; Level spins and parities (mainly from 'Table of Isotopes', 6th ed., 1967, C.M. Lederer et al).

74Meixn: JÜL-1087-RX (July 1974), Ch. Meixner: "Gamma Energien".

Contents:

For all important radioactive nuclides, w.a. of the following data are given:

HL, γ-Es+Ints (converted to absolute where possible).

73Erdtm: JUL-1003-AC (Sept. 1973, 2nd unrevised ed. Apr. 1974), G. Erdtmann, W. Soyka: "Die Gammalinien der Radionuklide".

Contents:

For all important radioactive nuclide, w.a. (of consistent data) are given for the following quantities:

HL, Es+Ints (converted to absolute where possible) of γ- and X-rays.

73Tobia: CEGB/RD/B/M-2669 (June 1973), A. Tobias: "Data for the calculation of gamma radiation spectra and beta heating from fission products (rev. 3)".

Availability: the file has been translated into ENDF/B-IV format (see also 77Tobia). An updating of the evaluation is planned.

Contents:

For all identified FP in the mass range $72 \leq A \leq 166$, selected values are given for:

HL, Q_β, γ-Es+Ints (converted to absolute for almost all nuclides); $\langle E_\beta \rangle$.

71Sangi: "Table of γ-Rays" (1971) (Politecnico di Milano), V. Sangiust, M. Terrani.

Contents:

Data taken from selected references for all FP and nuclides produced by (n,γ) reaction, with 0.1 sec $\leq$ HL $\leq 10^7$y:

HL, γ-Es+Ints.

V. Review of delayed neutron data evaluations [5)]

V.1. List of libraries and publications

"Name"	Publication (year)	Author (comments)
77Saphi	Nucl. Sci. and Engg. 62, 660 (1977)	D. Saphier et al (6-groups d.n. spectra)
76Rudst	LF-70 (1976, revised)	G. Rudstam (d.n. spectra of 25 precursors)
75Amiel	Nucl. Phys. preprint (1974)	Y. Nir-El, S. Amiel (P_n-values)
	Nucl. Sci. and Engg. 57, 117 (1975)	T. Izak-Biran, S. Amiel (P_n-values)
	IAEA-169, vol.II, p.33 (1974) (73Bologna)	S. Amiel (superseded by the above)
75Tuttl	Nucl. Sci. and Engg. 56, 37 (1975)	R.J. Tuttle (absolute d.n. yields)
74Cox	ANL-NDM-5 (1974)	S.A. Cox
73Tomli	Atomic Data and Nucl. Data Tables 12 (2) 179 (1973)	L. Tomlinson (HL and P_n-values)
	AERE-R-6993 (1972)	(updated by the above)
72Maner	At. Energy Review 10 (4) 637 (1972)	F. Manero and V. Konshin

V.2. Description of delayed neutron data libraries

77Saphi: Nucl. Sci. and Engg. 62, 660 (1977), D. Saphier, D. Ilberg S. Shalev, S. Yiftah: "Evaluated delayed neutron spectra and their importance in reactor calculations".

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5) abbreviations: d.n. ... delayed neutrons
P_n ... delayed neutron emission probability

Evaluation method: based on the measurement performed by Rudstam and Shalev, spectra of 6 HL-groups as emitted from various fissionable isotopes are constructed by least square fitting.

Contents: evaluated d.n. spectra for HL-groups, emitted in:

thermal fission of: U233,235; Pu239,241;
fast fission of: Th232; U235,238; Pu239;
14.7 MeV fission of: U235,238.

76Rudst: LF-70 (1976, revised), G. Rudstam: "Characterization of delayed neutron spectra".

Evaluation method: d.n. spectra for 25 precursors measured by Rudstam et al and Kratz et al are analyzed for gross and fine structure. The gross structure is represented as parameter formula and fitted to the experimental data.

Contents:

Parameter representation of the d.n. spectra and graphs showing the comparison of the measured and the calculated spectra for the following 25 precursors:

Ga79,80,81; As85; Br87,88,89,90,91; Rb93,94,95; In129,130; Sn134; Sb135; Te136; I137,138,139,140,141; Cs142,143,144.

75Amiel: a) Nucl. Phys. preprint (1974), Y.Nir-El, S. Amiel: "Systematics of delayed neutron emission probabilities in medium mass nuclides (fission products)".

Evaluation method: Least square fitting of a semiempirical formula for P_n-values: $P_n = C(Q_\beta - B_n)^m$ (C,m are constants) to the available experimental data, separated into odd-odd, odd-even etc nuclides. Prediction of unknown P_n-values with the obtained parameters.

Contents:

Compiled HL, $Q_\beta - B_n$, and P_n for 39 precursors;
Predicted P_n-values for:

Ga79,80; Ge83,84; As87; Kr94,95; Rb99; Sr97,98; Y98; Xe143,144.

b) Nucl. Sci. and Engg. 57, 117 (1975), T. Izak-Biran, S. Amiel: "Reevaluation of the emission probabilities of delayed neutrons from fission products".

Evaluation method: calculation of P_n-values from experimental d.n. yields and cumulative FP yields from U233,235 thermal and fast fission reevaluated on the basis of the odd-even effect.

Contents:

Evaluated P_n-values and uncertainties of 38 precursors, compared to available directly measured values. Delayed neutron yields and uncertainties for 6 half life groups and total d.n. yields from U233 and 235 thermal fission obtained by summation.

c) IAEA-169, vol.II, p.33 (1974), S. Amiel: "Status of delayed neutron data".

Contents:

Evaluated P_n-values for 18 precursors, superseded by Nucl. Sci. and Engg. 57, 117 (1975) (see above).

75Tuttl: Nucl. Sci. and Engg. 56, 37 (1975), R.J. Tuttle: "Delayed neutron data for reactor physics analysis".

Evaluation method: all data on absolute d.n. yields available up to August 1974 are corrected for systematic errors, converted to absolute values and the uncertainties adjusted to become consistent with each other. W.a. of the resulting values are calculted.

Contents:

Recommended total d.n. yields and uncertainties for the following fission processes:

thermal fission of:	U233,235; Pu239,241;
fast fission of:	Th232; U233,234,235,236,238; Pu238,239, 240,241,242.

74Cox: ANL/NDM-5 (1974), S.A. Cox: "Delayed neutron data - review and evaluation".

Evaluation method: A simple formula for the dependence of the d.n. yield on the incoming neutron energy is used, fitted to the available - renormalized - experimental data and extrapolated to 20MeV incoming energy. On this basis, d.n. yields of 6 half life groups as a function of incoming neutron energy are recommended for inclusion in ENDF/B-IV.

Contents:

d.n. yields of 6 HL-groups for 3 incoming energy regions (constant for $E_n \lesssim 4$MeV, linear decrease for 4MeV $\lesssim E_n \lesssim$ 7MeV, constant for $7 \lesssim E_n \lesssim$ 20MeV) for the following fissioning nuclides:

Th232; U233,235,238; Pu239,240,241.

Recommended d.n. spectra are taken over from G. Fieg [J. of Nucl. Energy 26, 585 (1972)].

73Tomli: a) Atomic Data and Nuclear Tables 12 (2) 179 (1973), L. Tomlinson: "Delayed neutrons precursors".

Contents:

Available experimental data and recommended values (= weighted - or unweighted in case of inconsistencies - averages of experimental data) of the half lives (and uncertainties) of 57 precursor FP and of the P_n-values (and uncertainties) of 34 precursors.

b) AERE-R-6993 (1972), L. Tomlinson: "Delayed neutrons from fission (A compilation and evaluation of experimental data)". Updated (and extended to new FP precursors) by a).

Contents:

Compiled and evaluated HL and uncertainties for 45 d.n. precursors, P_n-values and uncertainties for 34 of them.

72Maner: Atomic Energy Review. 10 (4) 637 (1972), F. Manero and V. Konshin: "Status of the energy dependent $\bar{\nu}$-values for the heavy isotopes (Z>90) from thermal to 15MeV and of $\bar{\nu}$-values for spontaneous fission".

Contents:

$\bar{\nu}_{prompt}$ and $\bar{\nu}_{total}$ vs. incoming energy for Th232, U233,235,238, Pu239:

d.n. yield = $\bar{\nu}_{total} - \bar{\nu}_{prompt}$ (not varying with energy).

Review paper 2

NEEDS AND ACCURACY REQUIREMENTS FOR FISSION PRODUCT NUCLEAR DATA IN THE ASSESSMENTS OF ENVIRONMENTAL IMPACTS

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ABSTRACT

This is a review of the needs and accuracy requirements for fission products nuclear data (FPND) in the assessments of environmental impacts from the UNSCEAR point of view. The main information source has been UNSCEAR reports (1, 2 and 3). The review gives definitions of some of the quantities used in assessments of the detriments from the use of nuclear explosions and nuclear power. Examples of uncertainties in such assessments will also be given. These are generally dominated by other uncertainties than those coming from FPND.

SOME QUANTITIES USED IN ENVIRONMENTAL ASSESSMENTS OF DETRIMENTS DUE TO IONIZING RADIATION

The mean absorbed dose in various human organs or tissues is the physical quantity usually taken as a basis for radiation risk estimates and a linear relationship without threshold between dose and probability of an effect is usually assumed to be valid at small doses (4). The absorbed dose D is defined as the mean energy imparted per unit mass at a certain point.

For the assessments of the relative importance of the potential hazards from various sources it is of interest to define some quantities which take into account the distribution of doses in exposed populations for different periods of time. The quantities can be source-related for use in assessments of the total hazards from a certain source - these are here symbolized with S - or they can be individual-related for use in assessments of the hazards to individuals - these quantities are here indicated with D.

1. Collective dose

The collective dose rate from a source k exposing a population is defined by the expression

$$\dot{S}_k = \int_0^\infty \dot{D}_k N(\dot{D}_k) d\dot{D}_k \quad \text{..........................(1)}$$

where $N(\dot{D}_k)$ is the population spectrum in dose rate.

The collective dose S_k accumulated in the population over a specified period of time, t_1 to t_2, is the time integral of the collective dose rate

$$S_k = \int_{t_1}^{t_2} \dot{S}_k(t)dt \quad \text{................(2)}$$

The unit of collective dose is manGy.

For the determination of the collective dose rate , eq (1), both the dose and the number of people in areas exposed to the source must be known. For the collective dose calculation, eq (2), information concerning the changes in dose with time as well as changes in the population spectrum with time are necessary.

When the collective dose includes all individuals exposed to a source it is usually called the total collective dose.

2. Collective dose commitment

If the integration limits in eq (2) - t_1 and t_2 - are changed to give the infinite time integral, the collective dose commitment S_k^c from source k is obtained

$$S_k^c = \int_0^\infty S_k(t)dt \quad \text{......................(3)}$$

3. Dose commitment

The individual-related quantity corresponding to the source-related collective dose commitment is the dose commitment D_k^c which is defined as the infinite time integral of the "per caput" dose rate due to source k

$$D_k^c = \int_0^{\infty} \overline{\dot{D}_k(t)}\,dt$$

where the "per caput" dose rate is the quotient of the collective dose rate and the population size at time t

$$\overline{\dot{D}_k(t)} = \frac{\dot{S}_k(t)}{N(t)}$$

If the population size remains unchanged over the period contributing to the integral then $S_k^c = D_k^c N$

(In the cases when the dose commitments are to be related to ICRP dose limits, it is rather the dose equivalent commitments that would be calculated and the unit will consequently be manSv.)

Long-lived radionuclides will continue to contribute to the dose commitment (and collective dose commitment) for very long periods, and will give rise to large dose commitments even though the future annual dose will be small. To get a more relevant measure of the radiological impact from such sources the time integral has been suggested to be limited to the duration of the practice (5). This integral is often referred to as the "incomplete" dose commitment. It can be shown that as a rule the "incomplete" dose commitment from one year´s practice is numerically approximately the same as the maximum annual dose resulting from the practice.

The quantities given above can be used for a single source such as a certain radioactive nuclide released from a nuclear power station or for the total releases from all nuclear installations.

ENVIRONMENTAL MODELS USED IN THE ASSESSMENTS OF DOSE COMMITMENTS

The chain of events leading from the primary release of radioactive substances to the irradiation of human tissues can be schematically represented by compartment models in which the rates of transfer of radioactivity between compartments are specified by constants or time functions. An indication of such a compartment model is given in Figure 1.

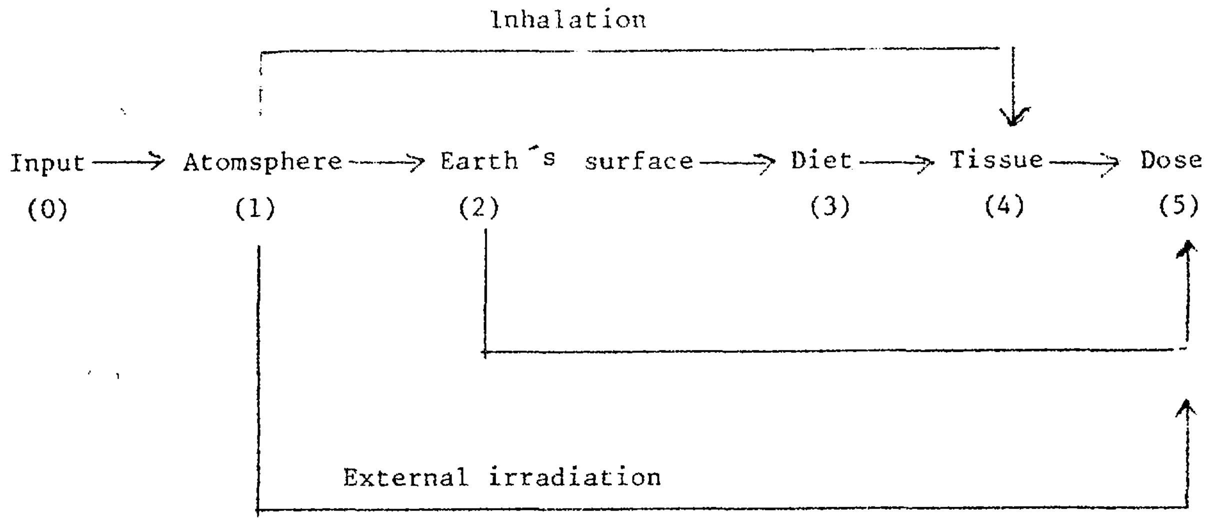

Figure 1

Since the dose commitment from a given source is the integral over infinite time of the "per caput" dose rate resulting from the input, it is practical to define transfer factors P_{ij} which are the quotient of the infinite time-integral of the appropriate quantity in step i to the infinite time-integral of the appropriate quantity in the preceding step j.

If the transfer factors in the pathways of the model are known the dose commitment D_k^c can be calculated as

$$D_k^c = Y \sum_{\text{parallels}} \prod_{\text{series}} P_i$$

where Y is the input and ΠP is the product of factors in series and where products from parallel series are summed up. Transfer factors and functions for a number of nuclides have been discussed by UNSCEAR (6,7).

FPND could play a role in the input Y (production of the nuclide) or in the final dosimetric factors P_{45} or for example P_{25} (through the decay data). However, according to UNSCEAR the uncertainties in the calculations of the pathways to man are usually much larger than those associated with calculating the input of the nuclides. This is in agreement with (8) and (9) with the exception of tritium in which case Findlay et coll. state that tritium yields from fission are inadequately known. According to the authors there are therefore requirements for measurements of this quantity accurate to $\pm$ 10 % for U-235, U-238 and Pu-239 in both thermal and fast reactor (LMFBR) spectra, Pu-240 in a fast reactor spectrum and Pu-241 in a thermal reactor spectrum. However, these requirements should be seen against the background that the total collective dose commitment to the public due to the release of tritium from nuclear power production has been estimated to be about 10 per cent of the total collective dose commitment (3).

An example of the environmental uncertainties involved can be taken from the UNSCEAR report (2) concerning the transfer factor P_{23} of ^{90}Sr from fallout to diet. ^{90}Sr has been measured frequently at a number of stations in both hemispheres for long periods and the pathways can be considered as relatively well known. After deposition on the earth's surface ^{90}Sr will enter into various components of the diet to different degrees. Since different regions of the world have different agricultural practices and soil conditions, P_{23} values will have to be established for different regions. In the UNSCEAR-report such values for the total diet are reported for Argentina, Australia, Denmark and New York. The reported values are 6.49, 7.47, 4.01 and 4.87 respectively in relative units. UNSCEAR makes use of a value of 5. However, an estimation of the influence of different diets showed that this value could lead to an underestimation of the dose commitment by a factor not exceeding 2. In comparison with such a figure uncertainties from FPND as reviewed in (10) will usually become negligible.

In the summary of the main dose commitments from nuclear tests, UNSCEAR reports the total uncertainties for internal irradiation to be within a factor 2. For external irradiation and for lung dose estimates, the uncertainties are reported to be within a factor of 5. These figures are usually dominated by uncertainties from environmental factors.

As regards dose commitment from nuclear power production (3,11) the uncertainties are difficult to estimate. For many nuclides of relevance, they are higher than in the case of nuclear test explosions. The dominating uncertainties relate to release data and environmental transport models.

CONCLUSION

According to the opinion of UNSCEAR, uncertainties of nuclear data are generally of small importance as compared to other sources of uncertainty in the assessment of environmental effects. In making this assessment, the radionuclides considered have been those relevant for fallout measurements and for assessment of the radiological consequences of nuclear power production and could be found in references (2) and (3). The conclusion seems to be in good agreement with conclusions from others.

ACKNOWLEDGEMENTS

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Review paper 3

NEEDS AND ACCURACY REQUIREMENTS FOR FISSION PRODUCT NUCLEAR DATA IN THE PHYSICS DESIGN OF POWER REACTOR CORES

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ABSTRACT

The fission product nuclear data accuracy requirements for fast and thermal reactor core performance predictions were reviewed by Tyror at the Bologna FPND Meeting. The status of the data was assessed at the Meeting and it was concluded that the requirements of thermal reactors were largely met, and the yield data requirements of fast reactors, but not the cross section requirements, were met. However, the World Request List for Nuclear Data (WRENDA) contains a number of requests for fission product capture cross sections in the energy range of interest for thermal reactors. Recent reports indicate that the fast reactor reactivity requirements might have been met by integral measurements made in zero power critical assemblies. However, there are requests for the differential cross sections of the individual isotopes to be determined in addition to the integral data requirements.

The fast reactor requirements are reviewed, taking into account some more recent studies of the effects of fission products. The sodium void reactivity effect depends on the fission product cross sections in a different way to the fission product reactivity effect in a normal core. This requirement might call for different types of measurement. There is currently an interest in high burnup fuel cycles and alternative fuel cycles. These might require more accurate fission product data, data for individual isotopes and data for capture products. Recent calculations of the time dependence of fission product reactivity effects show that this is dependent upon the data set used and there are significant uncertainties.

Some recent thermal reactor studies on approximations in the treatment of decay chains and the importance of xenon and samarium poisoning are also summarised.

1. INTRODUCTION

The importance of fission product nuclear data in the physics design of power reactor cores was reviewed by Tyror (1) in his paper (RP3) to the Bologna FPND Meeting.

For thermal reactors the primary requirement was identified as for data on cross sections determining the reactivity effects of fission products to enable the fuel lifetime to be determined to 2%. The most important fission products are Xe135, Sm149, Nd143, Rh103, Pm147, Xe131, Cs133, Sm151, Sm152 and Tc99 and the paper identified the accuracy requirements on yields, yi, thermal capture cross sections, σ_0, and capture resonance integrals, RI.

These requirements are summarised in Appendix 1. Shielding in fission product resonances and associated Doppler effects are negligible but the energy dependence of fission product cross sections affects the temperature coefficient, and there is a requirement for this effect to be known to $\pm$ 10%. About half the effect is contributed by Xe135. Sm149 is also important in this context. There is a requirement for the energy shape of this cross section to be known to a sufficient accuracy. There is also a requirement for the half lives of I135 and Xe135 to be known to $\pm$ 5% for the design of the control of Xenon instabilities.

The Bologna Meeting compared the status of the data with the users requirements and concluded that most of the thermal reactor core performance requirements were met. The yield requirements were met, apart from the chain yield for mass number 103 and, for Pu239, for mass number 131 for which there were discrepancies in the data. The cross section requirements were met apart from the thermal cross section of Sm151. The half life requirements have been met.

Since there is flexibility in the way the overall accuracy requirement is partitioned between the data requirements for individual isotopes, and for yields and cross sections, a factor of up to about 2 relaxation in the requirement on any one item of data (apart from the requirements for Xe135) can be permitted provided that the other data are known to a somewhat higher accuracy than that requested. Consequently we may conclude that the thermal reactor core performance FPND requirements proposed by Tyror have essentially been met, although integral measurements made to confirm the accuracy of the differential cross section data might be considered desirable. A study of the accuracy of the energy dependence of f.p capture cross sections, in particular for Xe135 and Sm149, and the accuracy of prediction of the temperature dependence of the reactivity effects of these isotopes might be required before we can conclude that all the requirements have been met.

Some recent studies of the importance of f.p data in thermal reactor core performance predictions are described in a later section.

For fast reactor core performance the primary requirement was identified as for f.p cross section data to be determined to an accuracy which would enable the reactivity effects to be estimated to $\pm$ 10% precision. Because of the customary use of the lumped cross section or pseudo fission product model in fast reactor calculations (rather than allowing for the time dependence of individual F.P's) it was proposed that the accuracy requirement for the pseudo f.p. capture cross section should be increased to $\pm$ 8% to allow for uncertainties in the model. Target precisions for the capture cross sections of individual isotopes were derived on the assumption that the errors between isotopes (and between yields and cross sections) would be uncorrelated. The accuracies requested for the yields and capture cross sections were in the range $\pm$ 20% to $\pm$ 55%. There is also a requirement for the shape of the cross sections to be known sufficiently well for the effect on sodium and Doppler effects to be determined. The Bologna Meeting considered these requirements and concluded that the yield requirements had been met but that most of the capture cross section requirements had not been.

Several comprehensive reviews of fission product data for fast reactor neutronics calculations have been issued recently. Two of these consider the status of pseudo fission product cross sections and the accuracies they estimate for the data are close to the requirement proposed by Tyror. Heijboer and Janssen (2) conclude that pseudo fission product capture effects calculated using their recommended data set (which has been adjusted to fit integral measurements made in the STEK facility) have an accuracy in the range 7 to 10%. When all the reactivity worth measurements made in the STEK facility have been taken into account in adjusting the cross sections the uncertainty in the capture cross section will be about 7%, (the main component of the uncertainty being the systematic errors associated with the interpretation of the STEK measurements). Langlet, Coppe and Doat (3) conclude that the uncertainty in the pseudo fission product reactivity effect is ± 16% (at the 2σ level) the major components being the uncertainty in the cross sections (mainly capture) of ± 10% (2σ) and in the yields of ± 8% (2σ). First results of measurements in Rapsodie and Phenix indicate that a higher accuracy will be obtained from these. Thus it appears that the requirements for fast reactor core performance predictions have now been met apart, possibly, from the energy dependence of the cross sections required to predict sodium void and Doppler effects. There is a need for the different types of integral measurements to be intercompared so that any systematic errors can be found and also for uncertainties in cross sections other than (n,γ) to be assessed.

Delayed neutron data are required for both thermal and fast reactor applications to an accuracy of 3% to 5%. Some uncertainties remain.

The fission product data requirements for studying the performance of fusion-fission systems have not yet been assessed but they are expected to be less stringent than those for thermal and fast reactors. However, it is possible that there could be a need for more accurate high energy data for the design of these systems.

The current interest in alternative fuel cycles (such as U233-Thorium fuelled fast reactors) could lead to a requirement for improved yield data for these isotopes. High burnup reactors, designed for improved fuel utilisation or to minimise reprocessing, could require more accurate f.p. data, because of the higher f.p. fractions, and also data for the capture products.

In the following sections the effects of fission products on fast reactor core performance predictions are reviewed and some recent thermal reactor studies are briefly summarised.

2. EFFECTS OF FISSION PRODUCTS ON FAST REACTOR CORE PERFORMANCE

2.1 Fast Reactor Core Physics Properties Affected by Fission Products

The effects of fission products on the following core properties must be estimated so that the corresponding design parameters can be specified.

(a) Effect on Reactivity at the End of the Fuel Cycle

At the end of the fuel cycle the control rods which compensate burn up effects are at their minimum operational worth positions (having only the margins required for stability control and because of differences in the end of cycle reactivity in different cycles) and the fuel feed enrichment is chosen to give criticality in this condition. Thus the fuel feed enrichment depends on the reactivity invested in fission products at the end of the fuel cycle (a 1% reduction in reactivity requiring a 1.8% increase in the mean fuel feed enrichment, ($\delta e/e$) to compensate). The accuracy with which the reactivity of the core should be predicted at the end of a cycle and at operating temperatures is 0.5% to 1.0%($\delta k/k$).

(b) Control Rod Reactivity Investment Required to Compensate for the Variation in Reactivity with Burnup Through the Fuel Cycle

The operating rods compensate for the variation of reactivity with burnup through the fuel cycle. The incorporation of more control absorbers or control rod channels than are needed worsens the neutronics performance, reducing the breeding gain and requiring a higher fissile material investment. Uncertainties in prediction of control requirements could cause the inclusion of more control rods than are needed or the use of expensive enriched boron instead of natural boron. The reactivity control requirements should be predicted to an accuracy of $\pm$ 5%.

(c) Reactivity Coefficients

Uncertainties in the Doppler and sodium void reactivity coefficients are dependent on uncertainties in fission product data. The importance of these uncertainties depends on the type of design and the safety philosophy. For example, if the safety criterion were to be adopted that the maximum positive sodium void coefficient must be less than 2$ then the uncertainty in estimating this coefficient would be more important than for a conventional design of fast reactor (which has a larger maximum positive void coefficient). As a guide accuracy requirements on Doppler and sodium void effects of $\pm$ 15% are taken.

(d) Delayed Neutron Effects

The effective delayed neutron fraction and the time dependence of delayed neutron emission following fission are needed for safety studies. However, the main requirement for accurate data arises from the use of kinetic response measurements to determine reactivities and the need for accurate reactivity scales. The requirement is for an accuracy of 3% to 5% in the effective delayed neutron fraction and in the period-reactivity relationship (for periods of about 1 sec to 1 min).

(e) Fission Product Gamma Emission Data

Fission product gamma emission data are used to predict the neutron sources from the (γ,n) reaction in reactors containing deuterium or beryllium. Fission product gammas can also affect the neutron emission from sources such as antimony-beryllium sources and affect the time dependence of these sources during the reactor shut down period.

2.2 Pseudo Fission Product Cross Sections and the Reactivity Effects of Fission Products in Fast Reactors

Fission products reduce the reactivity of a fast reactor, the main effect being neutron absorption by the (n,γ) cross section, which contributes about

90% of the reactivity effect. Inelastic scattering contributes most of the remaining 10%, with elastic moderation contributing about 2%. The effect of the f.p. transport cross sections on leakage are small and (n,p), (n,α) and (n,2n) reactions are usually neglected.

The usual approximation made in fast reactor calculations is to neglect the time dependence of the fission products, due to decay and burnup, and to take a set of time independent effective yields. Studies have been made recently of this time dependence, defining the pseudo fission product cross section for reaction r in group g for fission in isotope j by

$$\tilde{\sigma}^{j}_{r,g}(t) = \sum_{i} y^{j}_{i}(t) \cdot \sigma_{r,g,i}$$

where $y^{j}_{i}(t)$ is the concentration of fission product i per fission in isotope j at time t after fission, and

$\sigma_{r,g,i}$ is the cross section for reaction r in group g for f.p. i.

The values of the time dependent yields or concentrations $y^{j}_{i}(t)$ are calculated from the time dependent equations describing the effects of fission product decay and neutron capture.

The fluence corresponding to a burnup of 10% of heavy atoms in CFR is about

$\phi.t \approx 0.3/\text{barn}$

and the average fluence for a fission product will be about half of this (the flux at the centre of the reactor being about $1 \times 10^{16}/(\text{cm}^2\text{sec})$). Most important fission products have spectrum averaged 1 group capture cross sections larger than 0.5 barns and several have capture cross sections larger than 1 barn (in a typical commercial sized fast reactor spectrum). For these of the order of 10% will undergo neutron capture reactions. However, the change in the pseudo fission product one group capture cross section associated with these capture reactions is small because they are replaced by f.ps with similar cross sections on average. The time dependence of the one group pseudo fission product capture cross section has been studied as part of the comprehensive reviews made by Kikuchi, Hasegawa, Nishimura and Tasaka (4) and by Heijboer and Jannssen (2).

The JAERI study did not include the effect of neutron capture (which had been shown to be small in an earlier study). For Pu239 fission the effective one group pseudo f.p. capture cross section is 32% smaller after one day decay than the value after 1 year decay and is 6% smaller after 30 days decay than the 1 year value. Values are given for both a 1000 MWe FBR and a small (JOYO class) FBR and the time dependence is similar in these two spectra. The JAERI study found similar trends for other f.p. cross section sets (JNDC, JNDC-P Cook and ENDF/B-4). The thermal fission yields of Meek and Rider (1972) (5) were used.

The study by Heijboer and Janssen included the effects of f.p. capture and used the fast fission yields of Crouch (1976) (6). Instead of increasing continuously in time, the one group capture cross section decreases after about 200 days, being 1% lower after 464 days (compared with the ~ 1% increase found in the JAERI study for this period). The initial value is 14% lower than the value after 1 year decay and irradiation and is 6% lower after 10 days decay and 3% lower after 30 days decay than the 1 year decay value. The initial time variation is thus about one half of that found in the JAERI study. The RCN study also found that the variation obtained using the CNEN f.p. cross section set was different, the one group pseudo f.p. capture cross section decreasing beyond the first time step (58 days) used in the calculations. This difference was found to be mainly due to the differences in the Ru103 cross section relative to the Rh103 cross section(Ru103 half life: 39.6 days).

In CFR at the end of the fuel cycle about 10% of the f.ps will have been produced by fissions occuring in the previous 20 days and so an uncertainty of about 20% in the average pseudo f.p. capture cross section in this period could be acceptable. The error involved in using the concentrations corresponding to the average irradiation time is probably less than 5% but some further studies of the important factors in the time dependence would improve confidence in predictions of f.p. effects. It should be noted that both the JAERI and RCN studies approximated the cross sections for some short lived f.ps by the longer lived members of the same chain.

The JAERI study shows that the inelastic and elastic scattering cross sections have a smaller time variation than the capture cross section (10% and 1% respectively) and so these variations can be neglected.

2.3 Fission Product Migration Effects

Stable and long lived gaseous fission products (isotopes of Br, Kr, I and Xe) contribute about 5% of the f.p. capture effect. Cs133 and Cs135, which have gaseous precursors, contribute 6% to 10% (depending on the data set used).

The fraction of these which diffuse out of the fuel is uncertain and Langlet et al (3) associate an uncertainty of $\pm$ 3% (2σ) in the pseudo fission product reactivity effect because of this.

2.4 Accuracy Requirements for f.p. Reactivity Effects

In Table 1 the reactivity effects of fission products at the end of a fuel cycle stage in CFR and Super Phenix are given. This is one of the quantities which determines the enrichment of the fuel supplied to the reactor. On the basis of integral measurements it is now estimated that the reactivity of a fast reactor containing fresh fuel at 300°K can be predicted to an accuracy of 0.3% to 0.5%. (See, for example (7)). It would seem a reasonable aim to make the uncertainty contributed by other factors (such as burnup effects and temperature effects) each less than about 0.3% in reactivity. This would lead to a f.p. requirement of $\pm$ 10% to meet the highest f.p. reactivity in Table 1 (3% δK).

Table 1 contains data for 3 types of fuel cycle being considered for CFR (2 batch, 3 batch and 6 batch). Initially the maximum heavy atom burnup proposed for Super Phenix is 8.5% with the aim of progressing to 12%. This results in an increase of the f.p. effect by 35%. The reactors must be designed with flexibility and control margins which would enable these different modes of operation to be achieved. These control margins make the consequences of an error in the prediction of the reactivity effect of fission products less serious (at worst limiting the maximum burnup achievable or requiring a change in the mode of refuelling until the feed fuel enrichment can be adjusted). Information on the requirements for high burnup for Super Phenix will be obtained during the period of operation at lower burnup when the control margins will be larger. Similarly, the mode of batch refuelling in CFR could be altered until fuel of the required enrichment can be supplied.

TABLE 1

REACTIVITY INVESTED IN FISSION PRODUCTS AT THE END OF A FUEL CYCLE STAGE

CFR (Maximum burnup 10^5 MWd/t)	
Type of cycle	F.P. reactivity
2 batch	2.7%
3 batch	2.4%
6 batch	2.1%
Super Phenix (2 batch cycle)	
Maximum burnup	F.P. reactivity
8.5% heavy atoms	2.1%
12% heavy atoms	3.0%

TABLE 2

VARIATION OF FUEL REACTIVITY THROUGH A FUEL CYCLE AT EQUILIBRIUM

CFR (with a maximum burnup of 10^5 MWd/t)		
Type of cycle	Total variation	F.P. contribution
2 batch	4.2%	1.8%
3 batch	2.8%	1.2%
6 batch	1.4%	0.6%
Super Phenix (2 batch cycle)		
Maximum burnup	Total variation	F.P. contribution
8.5% (7.7×10^4 MWd/t)	2.5%	1.4%
12% (1.1×10^5 MWd/t)	3.5%	2.0%

Table 2 summarises the variation of fuel reactivity through a fuel cycle at equilibrium for CFR and Super Phenix.

The control rod investment in Super Phenix is 8% $\delta k/k$. If it is required to predict this to an accuracy of $\pm$ 5% an accuracy in the total reactivity compensation of $\pm$0.4% $\delta k/k$ is required. Limiting the contribution from any one effect to $\pm$ 0.2% $\delta k/k$ results in a f.p. reactivity accuracy requirement of $\pm$ 10%.

2.5 Requirements for the Prediction of Sodium Void and Doppler Effects

The sodium void reactivity effect can be separated into two components, the central component and the leakage component. The central component is due mainly to the change of neutron spectrum when sodium is removed and is positive in plutonium fuelled fast reactors. The leakage component is negative.

The spatial dependences of the two terms are different, the central term bring approximately proportional to the square of the flux and the leakage term to the square of the flux gradient. The central term is thus larger in the inner core regions and the leakage term larger in the outer core regions. One of the quantities of interest is the maximum positive void effect.

The effect of fission products is to increase the central term, the increase at end of cycle being about 10%. Butland (9) has calculated the effect on the void coefficient in CFR at equilibrium burn up (~ 3.5% heavy atoms). The increase is 0.25% $\delta k/k$ or 13% when the Cook (10) data are used and 0.4% $\delta k/k$ or 21% when the older UKNDL (11) data are used. The difference between the reactivity worths of fission products in the two f.p. sets is 20% and yet the difference between the effects on the maximum void worth is 60%. Ilberg, Saphier and Yiftah (12), in their study of the sensitivity of fast reactor static and dynamic parameters to different f.p. cross section data, found that the most significant changes were in the sodium void reactivity, and the effect of different f.p. data sets on reactivity was quite different from the effect on sodium void reactivity (see Table 3). Thus although the Cook (capture only) data set had the same effect on reactivity as the Benzi set (capture only) the effects on sodium void reactivity (combined with the fuel burn up effect) were 0.54% $\delta k/k$ and 0.3% $\delta k/k$ respectively (at 50,000 MWd/t). The inclusion of f.p. scattering has the effect of increasing the reactivity reduction due to fission products but reducing the effect of f.p's on the sodium void reactivity effect (by 0.14% $\delta k/k$).

Although the accuracy required for the effect of fission products on the sodium void reactivity effect is not stringent (~ ± 30% giving an uncertainty contribution to the central term of about ± 3% and to the total core voidage of about ± 7% at the end of cycle condition) it cannot be defined easily in terms of cross section accuracies and cannot be related to the accuracy of integral measurements in a single spectrum. The relative accuracies of integral measurements of f.p. reactivity effects in normal and voided cores are required. The reactivity worth of fission products is about 15% less in a voided core than in a normal core. To obtain this difference to an accuracy of 30% would require an accuracy of 3.5% in the worths in the two cores.

For CFR the control rods required to compensate the variation of reactivity with burnup might be designed differently and separately controlled (although there could be some flexibility to interchange the roles of different types of control rod). A margin in the control requirement must be included to allow flexibility in the refuelling and in the plutonium composition of the feed and a typical estimate of the required reactivity worth of the operating rods is 6.5% $\delta K/k$. A reactivity accuracy requirement of ± 10% would contribute less than about ± 3% to the uncertainty in the operating rod reactivity requirements.

TABLE 3

CHANGE IN REACTIVITY AND SODIUM VOID EFFECT WITH BURNUP CALCULATED USING DIFFERENT DATA SETS
(Ilberg, Saphier and Yiftah (12) calculations for a 1000 MW(e) fast reactor, burnup 50000 MWD/t)

FP Data	Relative Absorption Cross Section	Change in Reactivity With Burnup (Per Cent $\delta k/k$)	Change in Sodium Void Reactivity Effect with Burnup (Per Cent $\delta k/k$)
COOK + scattering data	100	-5.6	0.40
COOK (capture only)	100	-4.7	0.54
BENZI (capture only)	101	-4.8	0.30
UKNDL (capture only)	120	-5.3	0.74
ENDF/BIII + scattering data	80	-4.8	0.41

The sodium void reactivity in the clean core is 1.32% $\delta k/k$

The f.p. reactivity effect is probably the major source of uncertainty in the variation of reactivity with burnup and could be permitted to contribute this uncertainty to the operating rod reactivity requirements.

In a heterogeneous fast reactor the variation of reactivity with burn up could be different from the values for CFR and Super Phenix. For example, Sicard et al (8) describe a design of heterogeneous core in which the reactivity loss due to fuel burnup is 4.2 pcm/day compared with 13.3 pcm/day for the homogeneous design. In such a heterogeneous design the fission product effect is relatively more important in determining the operating rod requirements and could lead to a higher accuracy requirement.

To summarise, an accuracy of ± 10% in the reactivity effect of fission products is a reasonable requirement for current designs of fast reactor. For more advanced designs, in which the reactivity variation with burnup is smaller, a higher accuracy could be required (if the operating rod requirements must be separately identified).

To meet this ± 10% accuracy requirement on the reactivity effect an accuracy of ± 10% for the spectrum averaged pseudo f.p. capture cross section and ± 30% for the inelastic scattering contribution are required and a higher accuracy would be preferable to allow for uncertainties in effects such as the time dependence and f.p. migration.

Because no f.p. contributes more than 10% of the reactivity effect a ± 20% uncertainty on yields and spectrum averaged capture cross sections and ± 60% on the inelastic scattering contribution of individual f.ps is permissable, provided that the uncertainties are uncorrelated. When the the data are derived from integral measurements systematic errors are significant and can determine the bulk f.p. reactivity effect. The accuracy requirements on the individual f.p. capture cross sections proposed by Tyror (1) can be taken when the uncertainties are uncorrelated.

The effects of fission products data uncertainties on the Doppler effect can be separated into the scattering effect which increases the fraction of neutrons in the Doppler energy region and the components of the capture effect from above the Doppler region which decreases this fraction.

The net effect of fission products on the Doppler effect in the average burn up condition in CFR (~ 3.5% heavy atom) has been calculated by Butland (13) to be a 10% reduction. At the end of cycle the effect would be 15% (for a 2 batch scheme). For an overall accuracy of ± 15% in the Doppler effect the f.p. contribution to the uncertainty should be less than about ± 7% and so the aim should be to determine this component to ± 50%. In order to assess the accuracy requirements on the different cross sections the relative importance of inelastic moderation, elastic moderation and capture above the Doppler energy region would need to be assessed. However it seems likely that the requirements specified for the determination of reactivity (± 30% on inelastic scattering and ± 10% on capture) will meet this requirement.

2.6 Delayed Neutron Data Requirements

Delayed neutron data are required for the determination of reactivities by measurements of the kinetic response of the reactor to the reactivity change. The accuracy required is 3% to 5%.

Three items of delayed neutron data are required:

(a) The total yields per fission in Th232, U233, U235, U238 and Pu239 to ± 3% (Pu240 and Pu241 to a lower accuracy).

(b) The time dependence, to enable the period-reactivity relationship to be determined to ± 2% for periods in the range 1 to 100 seconds.

(c) The spectra of the delayed neutron emission, to enable the reactivity worth of delayed neutrons relative to prompt neutrons to be determined to ± 2%.

The dependence of delayed neutron yields on incident neutron energy needs to be known so that the accuracy of using an average of yields measured in fast and thermal reactor spectra can be assessed. (This is the method used at present to obtain data for use in reactor calculations.)

In a plutonium fuelled fast reactor about half the delayed neutrons arise from fission in Pu239 and the other half from U238. In recent years the discrepancies in the U238 total yield measurements have been a significant source of uncertainty. There are also some significant differences between different measurements of the delayed neutron spectra.

The Pu239 thermal fission data are less accurate than the fast fission data and the fast data have been averaged with the thermal data to obtain data for thermal reactor calculations.

The methods used to represent delayed neutron data in reactor calculations are those described by Keepin (14). The yield data are represented as a sum of six exponential decays:

$$n_j(t) = \beta_j \sum_{i=1}^{6} a_{ij} \, e^{-\lambda_{ij} \cdot t}$$

where $n_j(t)$ is the yield at time t for fission in isotope j.

β_j is the total delayed neutron yield for fissioning isotope j.

a_{ij} is the relative yield in group i.

and λ_{ij} is the decay constant for group i.

It is possible to obtain a satisfactory fit to the U235, U238, Pu239, Pu240 and Pu241 data by using a single set of λ values, (λ_i), and this is currently done in UK reactor calculations. Associated with each of the six delayed neutron groups is a delayed neutron spectrum (the same spectrum being used for fission in all isotopes). If the precursors contributing to a delayed neutron group differ significantly between fission in different isotopes this assumption could be unsatisfactory.

3. SOME RECENT STUDIES OF THE EFFECTS OF FISSION PRODUCTS IN THERMAL REACTORS

3.1 Importance of Short Lived Fission Products in the Xe135 and Sm149 Chains

Ottewitte and Seiber (15) have made a study of approximations in the treatment of the decay of nuclides precursor to Xe135 and Sm149.

Yield data are now available for 11 isotopes in the 135 mass chain and for 10 isotopes in the 149 mass chain.

Because of the complexity of the extended chains Ottewitte and Seiber have investigated the accuracy of simplified chains which use effective half-lives and yields. Approximating the 149 chain by a one step (Am→ Sm) chain gives a decay rate error of $\sim 3\%$ and approximating it by a two step chain (Nd→ Pm→ Sm) reduces the error to 1%.

3.2 Reactivity Transients in the Czechoslovakian Reactor KS150 Due to Fission Product Poisoning

Rana (18) has investigated the reactivity transients in the thermal reactor KS150 due to fission product poisoning. The reactor is fuelled with natural uranium, moderated by heavy water and cooled by carbon dioxide gas. The reactivity transients, which are mainly due to Xe135 and Sm149, necessitate rigorous operation according to prescribed power diagrams.

Reactivity transients due to xenon poisoning determine and limit the permissable changes in power level. A method of reducing the xenon peak after shut down is to first reduce the power for a few hours then to raise the power for a few minutes.

From the change of control rod positions during reactivity transients the control rods can be calibrated from accurate calculations of the f.p. poisoning and balancing reactivity. This is found to be more convenient than dynamic methods.

Accurate calculations of fission product poisoning are important for neutron economy.

3.3 Thermal Reactor f.p. Poisoning Data Requirements in WRENDA 76/77

Although the conclusions of the Bologna FPND Meeting suggest that the thermal reactor capture cross section requirements for the prediction of core performance have been met (apart from Sm151 thermal cross section) WRENDA 76/77 contains a number of requests. These are summarised in Appendix 2.

There are different views on the target accuracies for the prediction of core properties and different ways in which accuracy requirements can be allocated to individual items of nuclear data.

There are also different views on the investigations necessary before one can be satisfied that the data are to the required accuracy. The reactor physicist will often wish to have confirmation of the differential data by integral measurements. Comparisons of different differential measurements and careful consideration of the energy dependence and its significance are also required before the data can be accepted as meeting the requirements.

4. CONCLUSIONS

4.1 Thermal Reactor Data Requirements

The requirements for the prediction of thermal reactor core performance proposed by Tyror at the Bologna meeting, and the status of the required data, as assessed at the Bologna meeting, are summarised in Appendix 1. The yield requirements were considered to be met, apart from discrepancies in the 103 chain yield and in the Pu239 131 chain yield. The cross section requirements were met, apart from the Sm151 thermal capture cross section, and the possible need for further investigations of the accuracy of the energy dependences of the Xe135 and Sm149 capture cross sections. The half life requirements were met.

Studies by Ottewitte and Seiber using the recently produced data on the short lived precursors of Xe135 and Sm149 show that these are significant in determining the rate of production of Xe135 and Sm149, although the use of effective decay constants with simplified decay chains can provide adequate accuracy.

Recent studies by Rana of the importance of xenon and samarium poisoning on the methods of operating the Czechoslovakian natural uranium fuelled D_2O moderated, CO_2 cooled reactor KS150, and the use of the poisoning for calibrating control rods, emphasise the need for accurate data and comprehensive calculations.

WRENDA 76/77 contains requests for capture cross section measurements in the range 1 meV to 1 keV (to an accuracy of typically $\pm$ 10%) for 32 f.p.s. These are summarised in Appendix 2. This list could reflect differences of views on the accuracy requirements for the prediction of core properties and on the status of the cross section data to those presented at the Bologna meeting.

4.2 Fast Reactor Data Requirements

The requirements have been reviewed and are summarised in Appendix 3. The reactivity effects of fission products are required to an accuracy of $\pm$ 10%. Recent reviews of the status of the data by Heijboer and Janssen and by Langlet et al suggest that this requirement has been met by data adjusted to fit integral measurements. The requirement is for an accuracy of $\pm$ 10% on the capture component and $\pm$ 30% on the scattering component and some further studies might be needed to show that this requirement has been met.

The variation of the reactivity effect of fission products with the time after fission has a large uncertainty up to times of about 20 days and some further studies of the uncertainties in this variation and its significance are desirable.

The effects of f.p.s on sodium void coefficients are different from the effects on reactivity in a normal core (containing sodium). The accuracy required for the prediction of the effect is $\pm$ 30% but this cannot be simply converted into cross section accuracy requirements. Integral reactivity measurements in normal and sodium voided cores would be required to an accuracy of $\pm$ 3.5% to enable the 30% accuracy in the effect on sodium void reactivity to be achieved.

The accuracy required for the effect on the Doppler coefficient is $\pm$ 50% and although this cannot be simply related to the cross section accuracy requirements it seems probable that the requirements for the determination of reactivity (and the accuracy of currently available data) are sufficient.

Appendix 3 includes the capture cross section accuracy requirements proposed by Tyror (assuming no correlation of uncertainties between f.p.s.-an assumption which does not apply to the integral measurements). The requirements in WRENDA76/77 are also tabulated there.

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APPENDIX 1

FP DATA REQUIREMENTS FOR THERMAL REACTOR CORE PERFORMANCE AND THE STATUS OF THE DATA

CONCLUSIONS OF THE BOLOGNA FPND MEETING

1. Yield requirements and status

FP Isotope	Accuracy Requirement (Per Cent)	Comments on Requirements not Met
Tc99	11	
Rh103	6[a]	Discrepancies between measurements
Xe131	8	Discrepancies for Pu239 fission
Xe135	5	
Cs133	8	
Nd143	6	
Pm147	7	
Sm149	8	
Sm151	8[b]	
Sm152	9[b]	

Notes: a Yield is effectively that of Ru103.

b Effective yield including decay and neutron capture yield.

2. Capture cross section requirements and status

(a) Thermal values and resonance integrals

FP Isotope	Accuracy Requirements (Per Cent)		Status (Per Cent)	
	Thermal	RI	Thermal	RI
Tc99	20	15	10	10
Rh103	6	(50)	4	5
Xe131	15	10	11[a]	5[a]
Xe135	8	(100)	3	7
Cs133	15	10	5	7
Nd143	6	30	3	20
Pm147	15	8	7	7
Sm149	(20)		3	
Sm151	8	40	15[b]	20
Sm152	20	10	3	5

Notes: a Values from BNL325 Third Edition Volume 1

b Accuracy in BNL 325 Third Edition Volume 1 is 12%

The only requirement not met is the Sm151 thermal value.

(b) Energy dependence of the cross sections (to enable the variation of the reactivity effect with temperature to be estimated to $\pm$ 10%).

Xe135 variation with spectrum to $\pm$ 10%
Sm149 variation with spectrum to $\pm$20%

(The status was not assessed explicitly although the accuracies of the thermal values and resonance data suggests that the requirements are met.)

3. Half life requirements and status

FP Isotope	Accuracy Requirement (Per Cent)	Status (Per Cent)
I135	5	0.1
Xe135	5	0.1
Pm149		0.1

APPENDIX 2

REQUESTS IN WRENDA76/77 FOR FISSION PRODUCT CAPTURE CROSS SECTIONS FOR THERMAL REACTOR POISON CALCULATIONS

Isotope	Energy Range		Accuracy (Per Cent)	Category
	(meV)	(KeV)		
Kr83	1	1	10	2
Zr95	500	1		2
Nb95	25.3			2
Mo99	1	1		2
Ru101	1	10	10	1
Ru103	1	1		2
Rh103	1	1	10	2
Rh105	1	1	10	1
Pa107	1	10	10	2
Ag107	1	5	10	2
Ag109	1	1eV	10	2
(Sn126)	25.3			2)(Unknown
(Sb125)	25.3			3)cross section)
(Sb127)	25.3			3)
Te127	1	1eV	20	2
(Te129)	25.3			3 (Unknown σ)
Te132	25.3	1eV	20	2
Xe131	1	1	10	2
Xe133	1	5	3	2
Xe135	1	5	5	1
Cs133	1	1	10	1
Cs135	1	10	10	1
Pr141	1	10	10	2
Nd143	1	1	10	1
Nd145	1	1	10	1
Nd147	1	5	10	1
Pm147	1	1	10	1
Pm148	1	1	10	1
Pm149	1	1	20	1
Pm151	1	1	10	2
Sm150	1	1		1
Sm151	1	1	5	1
Sm152	1	1	10	2
Sm153	1	1		1
Eu153	1	5	10	2
Eu154	1	1	10	2
Eu155	1	1	10	2

APPENDIX 3

REQUIREMENTS FOR THE PREDICTION OF FAST REACTOR CORE PERFORMANCE

1. The net reactivity effect of fission products: ± 10%.

   (i) The net capture effect of fission products ± 10%

   (ii) The effect of scattering on fission product reactivity ± 30%

   (iii) An investigation of uncertainties in the variation of the reactivity effect of fission products with time at short times would be valuable.

   (iv) The requirements on the cross sections of individual f.p. isotopes are summarised in Section 4.

2. The effect of fission products on sodium void reactivity: ± 30%.

   (i) This would require an accuracy of about ±3.5% on the measurement of the reactivity effect of fission products in a normal core and a sodium void core.

   (ii) The cross section requirements are not easily defined. Some insight into the current uncertainties can be obtained by comparing the results of calculations made using different data sets.

3. The effect of fission products on Doppler effects: ± 50%.

   (i) This requirement is different from 1 and 2 but these are thought to be more stringent.

   (ii) The requirement is most probably met by the available data and some calculations using different data sets could help to illuminate this.

4. Requirements for individual f.p. capture cross sections in fast reactor spectra proposed at the Bologna meeting and included in WRENDA 76/77.

Isotope	Accuracy Requirements (in Per Cent) and Priorities (in Brackets)						
	Bologna Meeting	Japanese Requests		US Requests		French Requests	
Zr93	-	30	(2)	20	(2)		
Mo95	30	30	(2)	-			
97	30	20	(1)	-			
98	35	-		-			
100	35	-		-			
Tc99	20	20	(1)	-			
Ru101	20	20	(1)	10	(1)		
102	25	30	(2)	-			
103	-	-		20	(2)		
104	30	30	(2)	10	(1)		
106	35	-		10	(1)		
Rh103	20	-		-			
Pd105	20	20	(1)	10	(1)		
107	25	20	(1)	10	(1)		
Ag109	40	30	(2)	10	(2)		
I 129	-	-		20	(2)		
Xe131	30	20	(1)	-			
Cs133	20	20	(1)	-			
135	30	20		10	(1)		
137	40	-		-			
La139	40	-		-			
Ce144	-	-		10	(1)		
Pr141	35	-		-			
Nd143	30	20	(1)	-			
145	35	20	(1)	10	(1)		
146	-	-		-		20	(2)
148	-	-		-		20	(2)
Pm147	25	20	(1)	-		-	
Sm147	-	-		-		20	(1)
149	30	20	(1)	10	(1)		
151	25	30	(2)	10	(1)		
152	55	-		-			
Eu153	40	30	(2)	-			
155	55	-		20	(2)		

Review Paper 4

THE IMPORTANCE OF FISSION PRODUCT NUCLEAR DATA IN REACTOR DESIGN AND OPERATION

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ABSTRACT

This paper is devoted to the updating of nuclear data requirements concerning fission products in the three application fields of contamination of reactor components, detection of fuel failures and decay heat.

It is concluded that almost all requirements related to contamination and fuel failure detection are met. For decay heat following a long irradiation time, nuclear data requirements are supported by sensitivity analysis for the five top priority cases of ^{233}U, ^{235}U, ^{239}Pu thermal fission and ^{235}U, ^{239}Pu fast fission.

From error estimates associated with decay heat summation calculations, contributions to the total error come, in decreasing order, from uncertainties in effective decay energies ($\bar{E}_\beta + \bar{E}_\gamma$), yields (independent yields for cooling times $<10^4$ s, chain yields beyond) and half-lives of individual isotopes.

Comparisons made between predictions and calorimetric measurements as well as between different summation calculations seem to confirm theoretical error estimates. Further work in evaluating error bars in decay energies and yields of contributing isotopes (see Annex to this paper [16]) is however necessary.

1. INTRODUCTION

The fission product nuclear data(FPND) requirements for reactor design and operation were described in detail in review paper No.4 presented at the Bologna Panel in November 1973. The fields of application considered were:

- The escape of fission products from the fuel and the contamination of reactor components;
- The detection and location of cladding cracks;
- The residual heat emitted by fission products after shutdown.

This paper is an attempt to bring review paper No.4 up-to-date, account being taken of the conclusions and recommendations of the Bologna Panel, which may be summarized as follows:

- The accuracies aimed at in inventory calculations of the nuclides involved in contamination, in the detection of cladding cracks and in fuel design have been virtually achieved with present data (there are some exceptions which have to be looked into);

- The nuclear data and accuracy requirements for residual heat calculations will have to be established through sensitivity studies and on the basis of a programme of benchmark experiments.

Possible requirements in connection with inventory calculations will be studied in section 2. The methodology of studies of the sensitivity of residual heat to FPND will be dealt with in section 3 and estimates of errors in residual heat calculations will be presented in section 4 for the five main types of fission (thermal fission of ^{233}U, ^{235}U and ^{239}Pu; fast fission of ^{235}U and ^{239}Pu).

2. NUCLEAR DATA REQUIREMENTS FOR FISSION PRODUCT INVENTORY CALCULATIONS

2.1. Contamination of reactor components by fission products

The precisions aimed at have not changed since the last Panel. An uncertainty of $\pm$ 40% (1σ) in the inventory of important isotopes is acceptable as there are major unknowns in the estimate of leaks from fuel and in the deposition of fission products on cycle components.

Inventory error calculations performed for the Bologna Panel ([1], vol.1, p.99) already showed that this degree of precision had been achieve for most important isotopes - in particular:

- (Short half-lives): ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{89}Kr, ^{131}I, ^{133}I, ^{133}Xe, ^{135m}Xe, ^{135}Xe, ^{137}Xe

- (Long half-lives): ^{85}Kr, ^{90}Sr, ^{95}Zr, ^{103}Ru, ^{106}Ru, ^{110m}Ag, ^{125}Sb, ^{134}Cs, ^{136}Cs, ^{137}Cs, ^{140}Ba

except for ^{110m}Ag, ^{125}Sb, ^{134}Cs and ^{136}Cs in the case of sodium-cooled fast reactors and ^{129m}Te in the case of all reactors.

The present situation with regard to these isotopes is summarized in Table I. The accuracy for yields is taken from the compilation of Meek and Rider [2]; the status with regard to capture cross-sections is taken from Ref. [1] (vol.2, p.318).

Table I - ACCURACY OF THE INVENTORY OF ISOTOPES INVOLVED IN THE CONTAMINATION OF COMPONENTS

Isotope	Quantity determining the accuracy a)	Type of reactor	Accuracy
^{110m}Ag	σ_{γ} ^{109}Ag	LMFBR	< 30 %
^{125}Sb	y_c	LMFBR	< 20 %
^{129m}Te	y_c	PWR	< 15 %
		HTGR	30 %
		LMFBR	< 30 %
^{134}Cs	σ_{γ} ^{133}Cs	LMFBR	30 %
^{136}Cs	σ_{γ} ^{135}Cs	LMFBR	30% fact. 2

a) y_c ... cumulative yield

It will be noted that the $\pm$ 40% accuracy aimed at is achieved, except perhaps in the case of ^{136}Cs, where it would be necessary to know the ^{135}Cs capture cross-section in a fast reactor spectrum to within about $\pm$ 40%.

2.2. Detection and location of cladding cracks

Detection of cladding cracks by counting the gamma activity of gaseous isotopes requires an accuracy of $\pm$ 40% (1σ) in the calculation of the inventory of these isotopes.

As we saw in section 2.1, this accuracy has been achieved in the case of ^{85m}Kr, ^{87}Kr, ^{88}Kr, ^{89}Kr, ^{133}Xe, ^{135m}Xe and ^{135}Xe.

With regard to the other isotopes of importance in cladding crack detection (^{90}Kr, ^{91}Kr, ^{138}Xe, ^{139}Xe, ^{140}Xe and ^{141}Xe), Table II shows that this accuracy has now also been achieved.

Table II - ACCURACY OF THE INVENTORY OF ISOTOPES INVOLVED IN CLADDING CRACK DETECTION (%)

Isotope	$\frac{\Delta\lambda}{\lambda}$ (%)	PWR	HTGR	LMFBR
^{90}Kr	0.3 a/	6.	6.	20.
^{91}Kr	2.3 b/	7.	7.	30.
^{138}Xe	0.5 a/	3.	4.	9.
^{139}Xe	1.8 a/	5.	6.	13.
^{140}Xe	1.2 b/	5.	5.	12.
^{141}Xe	0.8 a/	6.	6.	29.

a/ Based on Ref. [3].

b/ Based on Ref. [1], vol.2, p.302.

Generally, the inventory accuracy calculated with the formula

$$\frac{\Delta C}{C} = \frac{\Delta y}{y} + \frac{\Delta\lambda}{\lambda}$$

is governed by the uncertainty in the cumulative yields.

The cumulative yields for the different reactor types under consideration are determined by a formula of the type

$$y = \sum_i \alpha_i \, y_i$$

where the sum extends to the different fissile isotopes.

The coefficients α_i adopted are the same as in [1,RP 4].

The uncertainties in the yields are taken from Ref. [2].

As regards detection by measuring delayed neutrons, the accuracies aimed at for the yields and half-lives of the delayed neutron groups are ± 20%.

According to Ref. [1] (vol.2, p.299), these accuracies have been achieved for the most important groups and for the half-lives of the most important precursors.

3. STUDIES OF THE SENSITIVITY OF RESIDUAL HEAT TO UNCERTAINTIES IN THE DATA

3.1. Accuracies aimed at for residual heat

Knowledge of the residual heat released in a fuel after reactor shutdown is necessary in connection with three fields of application:

- Removal of the residual power as a function of the time elapsed since reactor shutdown for normal operation and emergency shutdown conditions after a cooling time ranging from 0 to several days;

- The handling of irradiated fuel and its temporary storage at the reactor site after a cooling time ranging from a few hours to several months (or even years);

- Fuel transport and reprocessing and waste packaging.

The last of these three fields is dealt with in review paper No.5 to be presented at this meeting.

The accuracies aimed at for residual power are summarized in Table III. The only difference from the values of the Bologna Panel is in the accuracy required for the integral of the residual power from 0 to 24 hours in the case of sodium-cooled fast reactors, which has changed from 15% to 10%.

Table III – ACCURACY AIMED AT FOR THE TOTAL RESIDUAL POWER a)

Reactor type	Cooling time						
	0	1 m.	10 m	8 h	24 h	days	months
PWR-BWR	←		10(5)	→			
HTGR			←		10 (5)	→	
fast		←		10 (5)	→		
fast	← integrated (0–24 h)			10	→		
handling / storage				←	10	(5) →	≤ 5 →

a) The accuracy is given in %, with the long term requirements given in parantheses.

Desired accuracies for residual heat emitted by fission products, which are compatible with the values in Table III, are indicated in Table IV. To determine these accuracies, account is taken of the contribution of fission products to the total residual power in a water-cooled reactor: ~40% at 1 sec, ~50% at 10 sec and ~90% at 100 sec and beyond.

For reactor applications the needs may be classified as follows:

Priority 1 – residual head from ^{233}U, ^{235}U and ^{239}Pu thermal fission, and ^{235}U and ^{239}Pu fast fission;

Priority 2 – residual heat from the thermal (Pu recycling) and fast fission of ^{241}Pu and the fast fission of ^{238}U.

The accuracy aimed at in the case of priority 2 is about one third of that aimed at in the case of priority 1.

Table IV – ACCURACY REQUIRED FOR THE RESIDUAL HEAT EMITTED BY FISSION PRODUCTS [a)]

Fission	Cooling time							
	1 s	10 s	100 s 1.7 m	10^4 s 2.8 h	10^5 s 28 h	10^6 s 12 d	10^7 s 116 d	10^8 s 3.2 y
Thermal ^{235}U, ^{239}Pu	25(12)	20(1)	←	10 (5)		→	≤ 5	→
Thermal ^{233}U			←	10 (5)		→	≤ 5	→
Fast ^{235}U, ^{239}Pu			←	10 (5)		→	≤ 5	→
	← integrated		(0–24 h)	10		→		

a) The accuracy is given in %, with the long term requirements given in parentheses.

As it is difficult to perform precise measurements of the residual power in a reactor, the only way of verifying whether the desired accuracies have been achieved is to perform sensitivity calculations followed by error estimates on the basis of the uncertainties in the data. However, benchmark experiments are necessary in order to confirm that the error estimates are realistic. In accordance with the recommendations of the Bologna Panel, sensitivity studies and attempts at error calculation have been made at different laboratories [4-6].

We shall confine ourselves to the results of French studies, review paper No. 15 being supposed to give a detailed account of all the work relating to this subject (sensitivity studies and benchmark experiments).

3.2. Principle of sensitivity calculations

For the wide range of cooling times considered, a large number of fission products contribute to the residual heat. An early sensitivity study consisted in preparing lists showing (in descending order) the individual contributions of fission products to the residual heat as a function of cooling time (Ref. [1], vol. 1, p. 121); the relative contribution of a fission product represents the sensitivity of the residual heat to the energy emitted through decay of that fission product. Later, a general code for sensitivity calculations (DPEPIN [7]) was developed. This code is able to calculate, without any approximation, the sensitivity of the residual heat to the independent yields, the half-lives and the decay energies.

The method used in the sensitivity calculations may be summarized as follows.

The residual heat at time t, f(t), after one fission burst is calculated by the summation method:

$$f\ (t) = \sum_{i=1}^{M} E_i\ \lambda_i\ \ N_i\ (t)\ \ \text{MeV/s per fission}$$

where M is the number of fission products taken into account (M = 635 nuclides, of which 514 are radioactive), E_i is the mean beta plus gamma energy (MeV) emitted per disintegration of the nuclide (i), λ_i is the decay constant of nuclide i (s^{-1}), and $N_i(t)$ is the number of atoms of nuclide i at time t after fission.

The variation - as a function of time - in the number of nuclei of each nuclide is given by the Bateman method:

$$N_i\ (t) = \sum_{k=1}^{i} P_{ik}\ \exp\ (-\ \lambda_k t)$$

with $$P_{ik} = -\frac{1}{\lambda_i - \lambda_k} \sum_{j=k}^{i-1} \lambda_{ij} \; P_{jk}$$

where the summation extends to all possible precursors of nuclide i in the same chain (mass A), including the metastable ones, or in the A+1 chain in the case of delayed neutron emission,

and $$P_{ii} = y_i - \sum_{k=1}^{i-1} P_{ik}$$

where y_i is the independent yield of nuclide i, $\lambda_{ij} = b_{ij}\,\lambda_j$, and b_{ij} is the branching ratio from j towards i.

The sensitivity of the residual heat to a particular quantity x_i, s_x^i, can then be calculated for each cooling time by writing

$$s_x^i = \frac{\partial f}{\partial x_i} \; \frac{x_i}{f} \, ,$$

which gives the following values:

(a) Sensitivity to the independent yield

$$s_y^i\,(t) = \sum_{k=i}^{M} E_k \; \lambda_k \; \frac{\partial N_k\,(t)}{\partial y_i} \; \frac{y_i}{f\,(t)} \, ;$$

(b) Sensitivity to the half-life

$$s_{T\,1/2}^i\,(t) = -\sum_{k=i}^{M} E_k \left[\lambda_k \frac{\partial N_k\,(t)}{\partial \lambda_i} + \delta_{ki} N_k(t) \right] \frac{\lambda_i}{f\,(t)}$$

with $\delta_{ki} = 1$ for $k = i$, $\delta_{ki} = 0$ for $k \neq i$;

(c) Sensitivity to the energy emitted per disintegration

$$s_E^i\,(t) = \frac{E_i \;\; \lambda_i \;\; N_i\,(t)}{f\,(t)}$$

The sensitivity to the energy emitted per disintegration of nuclide i is in fact the relative contribution of this nuclide to the total residual heat.

The sensitivity calculations were performed with the help of the DPEPIN code [7] using the following data:

- Independent yields recommended by Meek and Rider [2];

- Decay data (half-lives, branching, beta transition probabilities, gamma spectra) from the 4th edition of the French library [3] compiled by J. Blachot.

Nevertheless, it is conceivable that, for the data under consideration, the sensitivities obtained are only slightly dependent on the origin of the nuclear data and have a kind of universal character.

The sensitivities obtained in the case of a fission burst can now be used to determine sets of sensitivities corresponding to irradiation of any duration.

In particular, irradiations of long duration (from one month to several years) are interesting from the point of view of reactor operation. In this case, the residual heat can be determined by integration of the function f(t) obtained for fission burst

$$F\ (\theta,t) = \int_{t}^{\theta+t} f\ (u)\ du \qquad \text{MeV/s per fission/s,}$$

where θ is the irradiation time, and t is the cooling time.

This approach does not take into account the effect of neutron capture on residual heat. In practice, there is an effect only with a very limited number of nuclides, in particular through the appearance of ^{134}Cs; this is so whatever type of reactor is involved. Sensitivity to capture cross sections will be examined in section 4.

For the other data, the sensitivities are obtained from logarithmic integration formulas

$$F\ (\theta,\ t) = \sum_{k=k_1}^{k_2} \frac{(f.t)_{k+1} - (f.t)_k}{\alpha_{k+1}}$$

with $$\alpha_k = \frac{\ln\ (f_{k+1}/f_k)}{\ln\ (t_{k+1}/t_k)}$$

and $$t_{k1} = t\ ,\ t_{k2} = \theta + t$$

The sensitivities of the function f to the data being known, it is easy to deduce those of the function F. These calculations are performed with the PEPER code. The formula, which are cumbersome, are not presented here.

Comment:

This calculation technique can be used with advantage for short irradiations, especially within the framework of benchmark experiments.

With it one can find the experimental conditions (θ',t' couples) which best reproduce the sensitivity spectra obtained in realistic cases associated with reactor operation (θ',t couples).

The most important values for sensitivities to individual data (sensitivity > 0.02) for 900-day irradiation are presented in the Annex [16] for different cooling times and the five top-priority cases: the thermal fission of ^{233}U, ^{235}U and ^{239}Pu and the fast fission of ^{235}U and ^{239}Pu.

4. ESTIMATION OF THE ERRORS IN RESIDUAL HEAT CALCULATIONS

4.1. General

The influence of errors on independent yields, half-lives and disintegration energies has been studied.

Moreover, the contribution of a number of nuclides with short half-lives which have been ignored in calculations has been estimated, as has the influence of neutron capture.

The difficulties associated with error calculations lie in the fact that:

- Not all experimental data are accompanied by an indication of the error limits;
- The errors in the evaluated data are difficult to estimate;
- Little is known about the error introduced by the use of models (e.g. calculation of the mean energy $\bar{E}_\nu$ of neutrinos);
- There is no information on possible systematic errors.

4.2. Errors from yields

The independent yields recommended by Meek and Rider [2] have been used with their associated errors. The number of nuclides considered is 635. Table V gives an idea of the distribution of the errors in the independent yields for the five top-priority cases.

To allow for the negative correlations between errors in the independent yields resulting from more precise knowledge of the chain yields, the errors in the chain yields have also been introduced. The distribution of these errors is illustrated in Table VI.

Table V - DISTRIBUTION OF ERRORS IN THE INDEPENDENT YIELDS

Error (%)	Number of nuclides				
	^{233}U thermal	^{235}U thermal	^{239}Pu thermal	^{235}U fast	^{239}Pu fast
2.	0	1	0	0	0
2.8	0	4	0	0	0
4.	2	9	0	0	1
6.	8	14	8	2	2
8.	13	17	11	6	1
11.	13	21	10	3	7
16.	20	18	6	7	6
23.	22	28	26	23	20
32.	35	45	47	52	51
45.	24	23	25	22	20
64.	498	465	502	520	527

Table VI - DISTRIBUTION OF ERRORS IN THE CHAIN YIELDS

Error (%)	Number of chains				
	^{233}U thermal	^{235}U thermal	^{239}Pu thermal	^{235}U fast	^{239}Pu fast
0.35	1	3	0	0	0
0.5	1	6	1	0	0
0.7	5	16	5	2	0
1.	21	10	4	7	0
1.4	4	7	11	13	6
2.	3	4	13	19	9
2.8	5	2	8	8	27
4.	6	6	3	1	6
6.	4	5	11	2	6
8.	3	3	2	7	4
11.	9	12	6	13	12
16.	18	12	12	5	10
23.	11	10	12	22	13
32.	9	4	12	1	7
45.	0	0	0	0	0
64.	0	0	0	0	0

To ensure a correct balance of 200 fission products for 100 fissions, cumulative fractional yields have been assigned to each nuclide at the start of a chain. The contributions of the ignored precursors were estimated separately.

The independent yields of Meek and Rider allow for an even-odd effect. However, residual heat calculations performed using a Gaussian charge distribution with the recommended Z_p and σ values give results which are not very different, the discrepancy not exceeding 2% whatever the cooling time. It is therefore the chain yields which are the important quantities.

The error in the calculation of the residual heat was determined by the expression

$$\left(\frac{\Delta F}{F}\right)_y = \left[\sum_i \left(\frac{S_y^i}{y_i}\right)^2 \mu_{ii} + \sum_i \sum_{j \neq i} \frac{S_y^i}{y_i}\,\frac{S_y^j}{y_j}\,\mu_{ij}\right]^{1/2}$$

where S_y^i is the sensitivity of the residual heat to the independent yield y_i, and μ_{ij} is the covariance matrix coefficient of the independent yields y_i and y_j.

Comparison of Tables V and VI shows that the chain yields are much more accurate than the independent yields. The errors among independent yields of the same chain therefore have a strong negative correlation.

Applying Bayes' theorem it is possible to calculate [8] the covariances among the different nuclides of a single chain resulting from knowledge of the chain yields

$$\mu_{ii} = \sigma_i^2 \left(1 - \frac{\sigma_i^2}{\sigma^2 + \sum\limits_{j=1}^{n} \sigma_j^2}\right)$$

$$\mu_{ij} = - \frac{\sigma_i^2 \, \sigma_j^2}{\sigma^2 + \sum\limits_{j=1}^{n} \sigma_j^2}$$

where σ_i^2 is the standard deviation of the independent yield y_i, σ^2 is the standard deviation of the chain yield, and n is the number of nuclides in the chain.

If the chain yield were unknown ($\sigma^2 = \infty$), one can see that $\mu_{ii} = \sigma_i^2$ while $\mu_{ij} = 0$.

On the other hand, if the chain yield is known perfectly ($\sigma^2 = 0$), there is a strong correlation.

The correlations between chains due to the laws of mass conservations have been studied elsewhere [5]. The effect is important in cases of long irradiations, but is limited to short cooling times ($< 10^4$ s) where most chains contribute to the residual heat.

Therefore, our estimate of the error from yields is certainly on the high side for $t < 10^4$s.

4.3. Errors from half-lives

The library compiled by Blachot [3] contains the experimental error limits for about two thirds of the half-lives.

To complete the list of errors, the half-lives were grouped according to time range and the average of the known errors was determined in each range (Table VII).

Table VII - MEAN ERRORS IN HALF-LIVES GROUPED ACCORDING TO TIME RANGE

time range	number of nuclides	mean experimental error (%)
0 - 3 s	89	13.
3 s - 1 m	146	8.
1 m - 5 m	48	5.
5 m - 1 h	82	2.
1 h - 5 h	32	1.4
5 h - 1 d	40	1.5
1 d - 20 d	22	0.85
20 d - 1 y	23	0.50
1 y - 3 y	3	0.90
> 3 y	29	3.5
stable	121	

In cases where the errors were not known, the following rules were applied:

0 - 3 s	:	error = $\pm$ 30%
3 s - 1 m	:	error = $\pm$ 10%
Other ranges	:	error = mean error in the range

For about 30 nuclides, the half-life had to be calculated and an error of 100% was adopted. However, the sensitivity of the residual heat to these data is very low.

The error in the residual heat calculations due to uncertainties associated with the half-lives, assumed to be independent, was calculated using the expression

$$\left(\frac{\Delta F}{F}\right)_{T1/2} = \left[\sum_i \left(S^i_{T1/2}\right)^2 \left(\frac{\Delta T1/2}{T1/2}\right)^2_i \right]^{1/2}$$

where $S^i_{T1/2}$ is the sensitivity of the residual heat to the half-life of nuclide i.

4.4. Errors from decay energies

In isomeric transitions, the energy emitted per decay event is simply $E = Q$, so that one simply has $\Delta E = \Delta Q$.

For beta transitions, the effective energy emitted is

$$E = Q - E_\nu$$

where E_ν is the mean energy carried off by neutrinos.

One then has two cases:

- The decay schemes (i.e. the beta transition energies and probabilities) are known. E is then calculated using the Fermi model [9] for each beta transition and averaged over the transition probability spectrum. The error is then calculated from

  $$\Delta E = \left[(\Delta Q)^2 + (\Delta E_\nu)^2 \right]^{1/2}$$

  When the errors ΔQ are not available, they are taken from the compilation of Wapstra [10] or that of Garvey [11]. When no error is available at all, a value of $\pm$ 15% is adopted. The error in the calculation of E_ν was estimated to be $\pm$ 5%.

- When the decay schemes are unknown, E_V is calculated by analogy with nuclides having the same atomic number (Z) whose mass is of the same parity (mean value E_V/Q for the nuclides (Z,A-2), (Z,A-4) ...).

For nuclides with a short half-life, we have $E_V \simeq 0.4Q$, the coefficient varying in general between 0.28 and 0.52 (uncertainty of $\pm$ 30%); hence, $E \simeq \alpha Q$ with α varying from 0.72 to 0.48 (uncertainty of $\pm$ 20%). In this case, the error will therefore be calculated from the expression

$$\frac{\Delta E}{E} = 0.2 + \frac{\Delta Q}{Q}$$

When several decay modes are in competition, the errors corresponding to each mode have been combined statistically.

The error in the residual heat calculations from uncertainties associated with the emitted energies, assumed to be independent, was calculated from the expression

$$\left(\frac{\Delta F}{F}\right)_E = \left[\sum_i \left(S_E^i\right)^2 \left(\frac{\Delta E}{E}\right)_i^2\right]^{1/2}$$

where S_E^i is the sensitivity of the residual heat to the energy emitted per decay event of nuclide i (or the relative contribution of nuclide i to the residual heat).

4.5. Total errors

The errors from yields, half-lives and energies have been combined statistically:

$$\frac{\Delta F}{F} = \left[\left(\frac{\Delta F}{F}\right)_y^2 + \left(\frac{\Delta F}{F}\right)_{T1/2}^2 + \left(\frac{\Delta F}{F}\right)_E^2\right]^{1/2}$$

All the calculations described above were performed for the priority 1 cases, i.e. the thermal fission of ^{233}U, ^{235}U and ^{239}Pu and the fast fission of ^{235}U and ^{239}Pu for an irradiation time of 900 days ($\simeq 8 \times 10^7$ s).

The overall results are presented in Figs. 1-5, which reveal the following general trends in the five cases under consideration:

- The uncertainties associated with the yields play a relatively minor role, except in the case of ^{239}Pu fast fission;

- The uncertainties associated with the decay energies are the determining ones for all cooling times;

- The uncertainties associated with the half-lives play only a very small role.

It can also be seen that the overall error is small- always less than $\pm$ 5% (1σ). Also, the total error curve has a minimum around 10^4 seconds, where the nuclides which contribute to the residual heat are both numerous (whence error compensation) and well known - conditions which do not occur together for either short or long cooling times.

The tablesin the Annex [16] give the partial errors induced by the data to which the residual heat is sensitive.

To complete this error analysis, one should evaluate the residual heat deficiency - for very short cooling times - due to the nuclides ignored in the calculations.

The most important of these nuclides are: ^{99}Sr, ^{100}Y, ^{101}Y, ^{103}Zr, ^{104}Nb, ^{111}Ru, ^{147}Ba, ^{150}Ce and ^{151}Ce. The increase in residual heat due to these nuclides is significant only at a cooling time of one second and amounts to about 1%.

Lastly, the effect of neutron capture must be studied. This was dealt with in review paper No.4 presented at the Bologna Panel ([1], vol.1, pp.115 and 117). The orders of magnitude of this effect, confirmed by Tasaka [12], are shown in Table VIII.

Table VIII - INCREASE IN RESIDUAL HEAT DUE TO NEUTRON CAPTURE (%)

Cooling time (s)	Thermal reactor $\phi_{th}=4.4\times10^{13} n/cm^2\ s$ irradiation for 900 days	Fast reactor $\phi_t=4.3\times10^{15} n/cm^2\ s$ irradiation for two years
10^0	< 1	< 1
10^1	< 1	< 1
10^2	< 1	< 1
10^3	< 1	< 1
10^4	1.4	1
10^5	4.3	2
10^6	5.3	3
10^7	8.4	4

It is therefore necessary to take account of neutron capture for t $\geq 10^4$s, especially in connection with fuel assembly transport and storage problems. The increase in residual heat is due mainly to the appearance of ^{134}Cs as a result of neutron capture by ^{133}Cs; the formation of ^{148m}Pm due to neutron capture by ^{147}Pm also plays a role up to 10^7 s.

If one accepts an error of 1% at t = 10^6 s and 10^7 s in the calculation of residual heat due to the uncertainty associated with neutron capture, the following uncertainties are needed:

^{133}Cs	σ_γ pile	thermal:	± 10%
		fast:	± 20%
^{134}Cs	σ_γ pile	thermal and fast:	factor of 2
^{147}Pm	σ_γ pile	thermal and fast:	factor of 2
	branching	thermal:	± 30%
		fast:	± 60%
^{148m}Pm	σ_γ pile	thermal:	± 30%
		fast:	± 60%

These accuracies have been achieved for the most part, or at least they are relatively easy to achieve ([1], vol.2, pp.312 and 318).

5. DISCUSSION AND CONCLUSION

If one is satisfied with the above-mentioned error calculation results, then the desired accuracies for residual heat calculations (Table IV) have been achieved, including long-term requirements, for the five priority 1 cases. There are accordingly no further nuclear data requirements.

However, in view of the reservations expressed in section 4.1, this conclusion should be confirmed by the interpretation of benchmark experiments which are sufficiently precise and representative and which have been performed at several laboratories. Few experimental results relating to benchmarks are published nowadays. The latest experiments will certainly be dealt with in review paper No.15.

Calorimetric measurements by Lott [13] have made it possible to generate a residual heat function for ^{235}U thermal fission between 100 s and 10^5 s. The results are given with an uncertainty of ± 9% at 100 s cooling time and ± 6% for cooling times of 1000 s or more.

The error in the residual heat calculated for ^{235}U thermal fission was estimated using the techniques described in section 4. The results of the error calculation are presented in Table IX.

Table IX – ESTIMATE OF THE UNCERTAINTY IN CALCULATIONS OF THE RESIDUAL HEAT AFTER A ^{235}U THERMAL FISSION BURST

cooling time (s)	residual heat (MeV/fis. s)	error (%) due to			total error (%)
		Y	T1/2	E	
1	0.624	2.84	2.82	4.65	6.14
10	0.125	1.22	1.24	4.10	4.45
100	0.0129	1.04	1.38	3.15	3.59
10^3	$9.06\ 10^{-4}$	0.40	1.12	1.83	2.19
10^4	$6.00\ 10^{-5}$	0.41	0.37	0.95	1.10
10^5	$2.60\ 10^{-6}$	0.48	1.38	1.25	1.92
10^6	$1.57\ 10^{-7}$	0.45	0.31	1.19	1.31

As in the case of irradiation of long duration, the error derives essentially from the energies E; the half-lives obviously play a much greater role here than in the case of lengthy irradiation. The comparison between measurement and calculation is shown in Fig.6 for $10^2 \leq t < 10^5$ s.

It can be seen that, beyond 100 s, the calculated values lie within the experimental errors. This suggest that the error calculation is realistic, at least for ^{235}U.

It is interesting to note that the test for the residual heat after fission burst between 10^2 and 10^5 seconds is an indirect test for the residual heat after lengthy irradiation for cooling times $\leq 10^2$ s.

$$t \leq 10^2 s : F(\theta, t) = \int_t^{\theta + t} f(u)\, du$$

$$= \int_t^{10^2} f(u)\, du + \int_{10^2}^{10^5} f(u)\, du + \int_{10^5}^{\theta + t} f(u)\, du$$

The integral between 10^2 and 10^5 contributes in the following manner to the function $F(\theta,t)$ for $\theta >> 10^5$ s:

t (s)	$\int_{10^2}^{10^5} f(u)\,du \Big/ F(\theta,t)$
1	45 %
10	56 %
100	84 %

Comparison between calculation results provided by different laboratories for lengthy irradiations generally reveals discrepancies little different from the calculated error limits (e.g. see Ref. [14]). Another example is given in Table X, where it can be seen that the discrepancy does not exceed 2%.

Table X - RESIDUAL HEAT FROM THE THERMAL FISSION OF ^{235}U (IRRADIATION FOR 1 YEAR)

Cooling time (s)	Calculation (MeV/fission)		CEA / LASL
	CEA	LASL [15]	
1	11.28	11.46	0.984
10	9.045	9.167	0.987
100	6.007	5.998	1.002
10^3	3.594	3.616	0.994
10^4	1.757	1.752	1.003
10^5	0.813	0.818	0.994
10^6	0.395	0.399	0.990
10^7	0.101	0.102	0.990

Despite the proximity of the set of calculation results, there may still be doubt owing to the fact that some data are the same.

If one wishes to improve the data further, with the tables in the Annex [16] it is possible to identify the data to which the residual heat is most sensitive and hence those for which an improvement in accuracy would be most beneficial.

For the residual heat problems associated with reactor operation, the most important data are the following (in order of importance):

- The effective energies emitted per decay event: $Q - E_v$;
- The yields; and
- The half-lives.

The error in the residual heat due to the uncertainties associated with the energies ($Q-E_v$) derives essentially - for short cooling times ($\leq 10^2$ s) - from the fact that beta spectra of many nuclides with short half-lives are unknown. The contribution of these nuclides is indicated in Table XI.

Table XI - CONTRIBUTION OF NUCLIDES WITH UNKNOWN BETA SPECTRUM TO THE RESIDUAL HEAT (%) (irradiation 900 days)

cooling time (s)	^{233}U thermal	^{235}U thermal	^{239}Pu thermal	^{235}U fast	^{239}Pu fast
1.	19.6	26.6	26.4	25.3	23.8
10.	15.0	19.8	20.7	19.1	18.8
100.	5.2	6.6	8.7	6.7	7.9
10^3	0.8	0.9	1.6	0.9	1.5
10^4	1.2	1.1	1.2	1.2	1.3
10^5	2.8	2.5	2.4	2.6	2.6
10^6	4.1	3.7	3.6	3.9	3.8
10^7	1.2	1.0	0.8	1.1	0.8
10^8	0.	0.	0.	0.	0.

It can be seen that nuclides with unknown beta spectrum contribute to the residual heat in almost the same way in the five cases, and especially for $t < 100$ s.

The uncertainty associated with the energy $E = Q - E_\nu$ emitted per decay for one of these nuclides is assumed to be (see section 4.4):

$$\frac{\Delta E}{E} = 0.2 + \frac{\Delta Q}{Q}$$

If all the uncertainties were in the same direction and taking $\frac{\Delta Q}{Q} = 15\%$, which is pessimistic, the error associated with the residual heat would be about 9% at 1 second, 7% at 10 s and 3% at 100 s. In reality, as the uncertainties are independent and the number of nuclides contributing to the residual heat in a short time is large, the estimated errors are much less: on average 1.7% at 1 second, 1.6% at 10 s and 1.3% at 100 s. Among the nuclides with unknown spectrum, the following are the most important:

$^{88,89}Br$, $^{95,96}Sr$, ^{96}Y, ^{100}Zr, ^{102}Nb, $^{103,104,105}Mo$, $^{105,107}Tc$, ^{135}Te, $^{137,138}I$, $^{141,142}Cs$, $^{143,144}Ba$ and $^{144,145,146}La$.

An accuracy of 30% in $E = Q - E_\nu$ is sufficient for these nuclides.

For longer cooling times most of the contributing nuclides have a known beta spectrum, therefore, the error is due above all to the uncertainty in the calculation of E_ν, assumed to be $\pm$ 5%, especially when E_ν is a large fraction of Q.

In all cases considered, for $t \geq 10^5$ s the nuclides responsible for most of the error are: ^{89}Sr, ^{90}Y, ^{91}Y, ^{140}Ba, ^{140}La, ^{143}Pr and ^{144}Pr to which ^{106}Rh must be added in the case of ^{239}Pu.

For $t \simeq 10^6$ s, ^{141}Ce - the beta spectrum of which is unknown - also exerts an influence.

For these nuclides, it would be useful to have values for $E = Q - E_\nu$ with an uncertainty of 5-10%.

The error due to the uncertainty associated with the yields is relatively small. For ^{235}U thermal fission, the error from yields is $\leq 0.5\%$; it is of the order of 1% for the other cases except ^{239}Pu fast fission, where it reaches 3% for $t \geq 10^6$ s because of the uncertainty associated with the 106 chain yield.

In order that one may appreciate the indirect effect of an improvement in the accuracy of the chain yields, the chain yield errors have been systematically divided by 2. The independent yields and their errors are unchanged, but the negative correlations along a chain are thus strengthened. Table XII shows the resulting reduction in the error from yields.

Table XII – REDUCTION IN THE ERROR FROM YIELDS WHEN THE ERRORS IN THE CHAIN YIELDS ARE DIVIDED BY 2 (irradiation 900 days)

cooling time (sec)	^{233}U thermal	^{235}U thermal	^{239}Pu thermal	^{235}U fast	^{239}Pu fast
1.	0.95	0.96	0.89	0.95	0.91
10.	0.91	0.95	0.86	0.92	0.88
100.	0.83	0.84	0.76	0.82	0.79
10^3	0.84	0.77	0.75	0.81	0.78
10^4	0.75	0.74	0.63	0.74	0.74
10^5	0.60	0.76	0.60	0.64	0.69
10^6	0.52	0.78	0.52	0.56	0.52
10^7	0.51	0.63	0.51	0.54	0.51
10^8	0.52	0.67	0.51	0.54	0.52

The error reduction is small for $t \leq 10^4$ s as the nuclides which play a role in the residual heat are at the head of the chain; the uncertainties in their independent yields are therefore the determining ones and an improvement in the accuracy of the chain yields does not make much difference.

On the other hand, the more the cooling time increases, the more the nuclides which count are at the end of the chain; an improvement by a factor of 0.5 in the accuracy of the chain yields is therefore reflected directly in a decrease of the residual heat error from yields (a factor of $\simeq 0.5$ for $t \geq 10^6$ s). It should be noted that, in the case of ^{235}U thermal fission, for which the independent yields are better known, the gain in precision is less.

Concluding the question of yields, it would appear therefore that, if the uncertainties evaluated by Meek and Rider are realistic, no new demand is justified. It is desirable therefore that the uncertainties be confirmed, on one hand for the independent yields of the nuclides at the head of a chain which contribute to the residual heat for $t \leq 10^4$ s (see table in Annex [16]) and on the other for the chain yields for the nuclides which contribute to the residual heat for $t > 10^4$ s.

As regards the half-lives, the error introduced by the associated uncertainties is of the order of 0.5%, so that the present data are adequate.

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FIGURE CAPTIONS

FIGURE 1 - ^{233}U Thermal Fission - Decay Heat Error Estimate - Irradiation 900 days, ϕ=0

FIGURE 2 - ^{235}U Thermal Fission - Decay Heat Error Estimate - Irradiation 900 day, ϕ=0

FIGURE 3 - ^{239}Pu Thermal Fission - Decay Heat Error Estimate - Irradiation 900 days, ϕ=0

FIGURE 4 - ^{235}U Fast Fission - Decay Heat Error Estimate - Irradiation 900 days, ϕ=0

FIGURE 5 - ^{239}Pu Fast Fission - Decay Heat Error Estimate - Irradiation 900 days, ϕ=0

FIGURE 6 - Decay Heat From a Fission Burst - ^{235}U Thermal Fission -

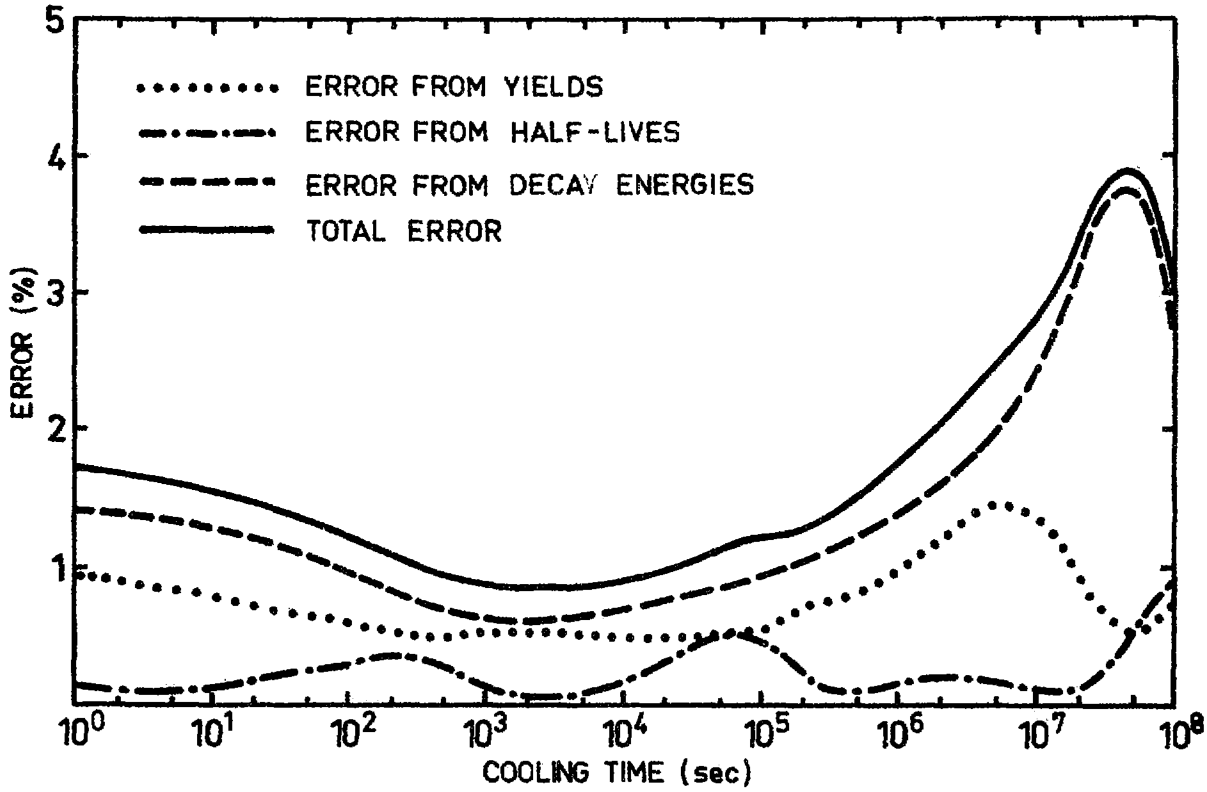

Fig.1_ ^{233}U THERMAL FISSION DECAY HEAT ERROR ESTIMATE
Irradiation 900 days, Ø = 0

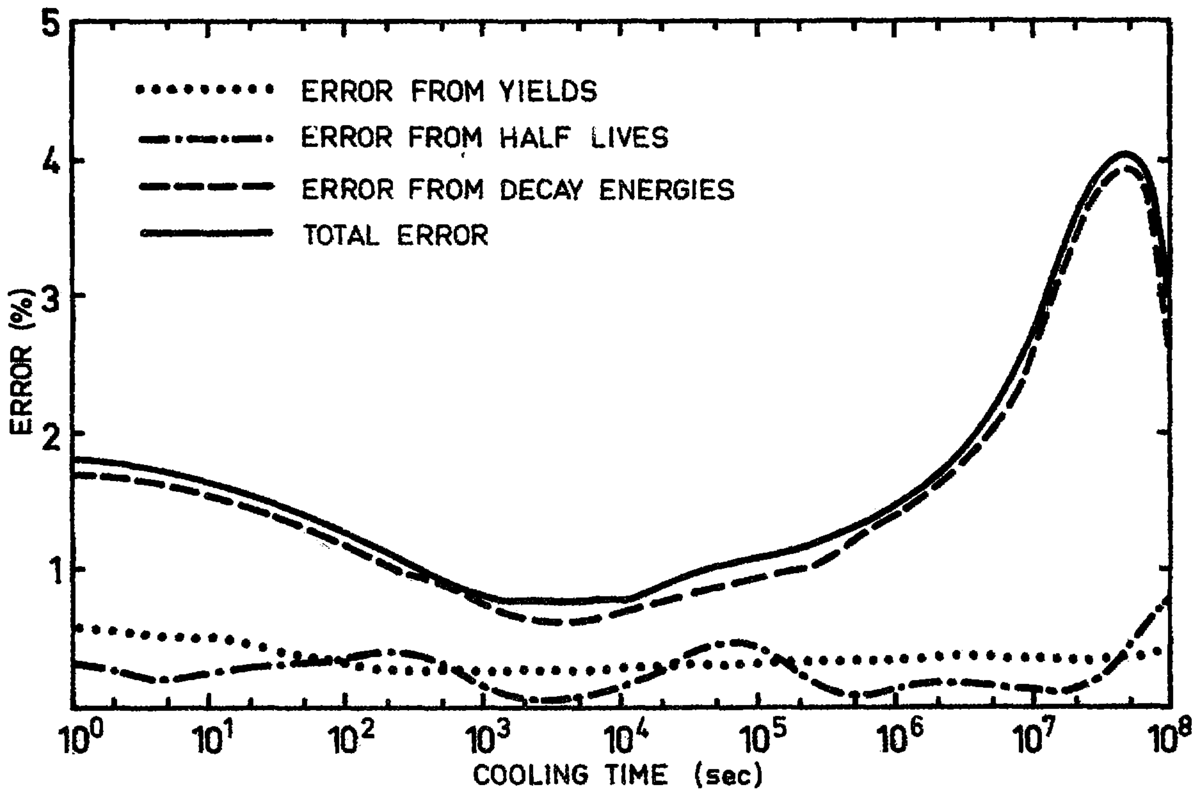

Fig.2_ ^{235}U THERMAL FISSION DECAY HEAT ERROR ESTIMATE
Irradiation 900 days, Ø = 0

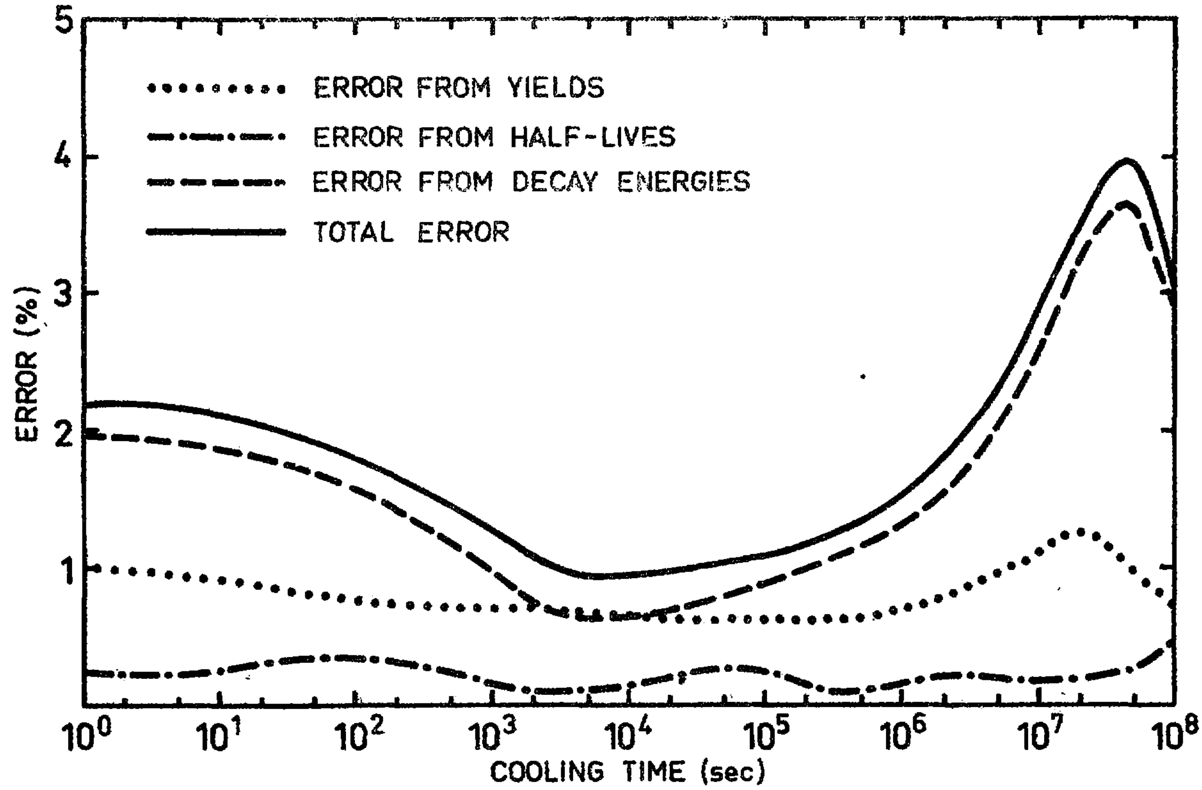

Fig. 3 _ ^{239}PU THERMAL FISSION DECAY HEAT ERROR ESTIMATE
Irradiation 900 days, Ø = 0

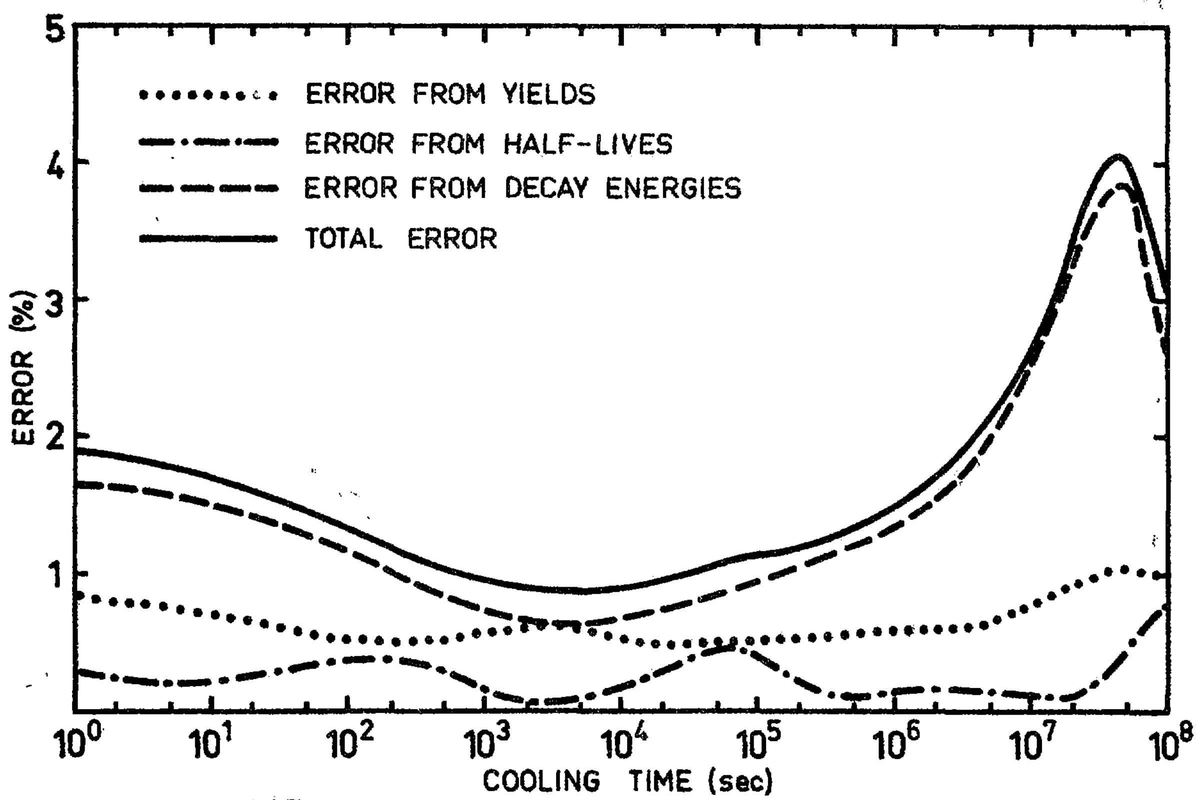

Fig. 4 _ ^{235}U FAST FISSION DECAY HEAT ERROR ESTIMATE
Irradiation 900 days, Ø = 0

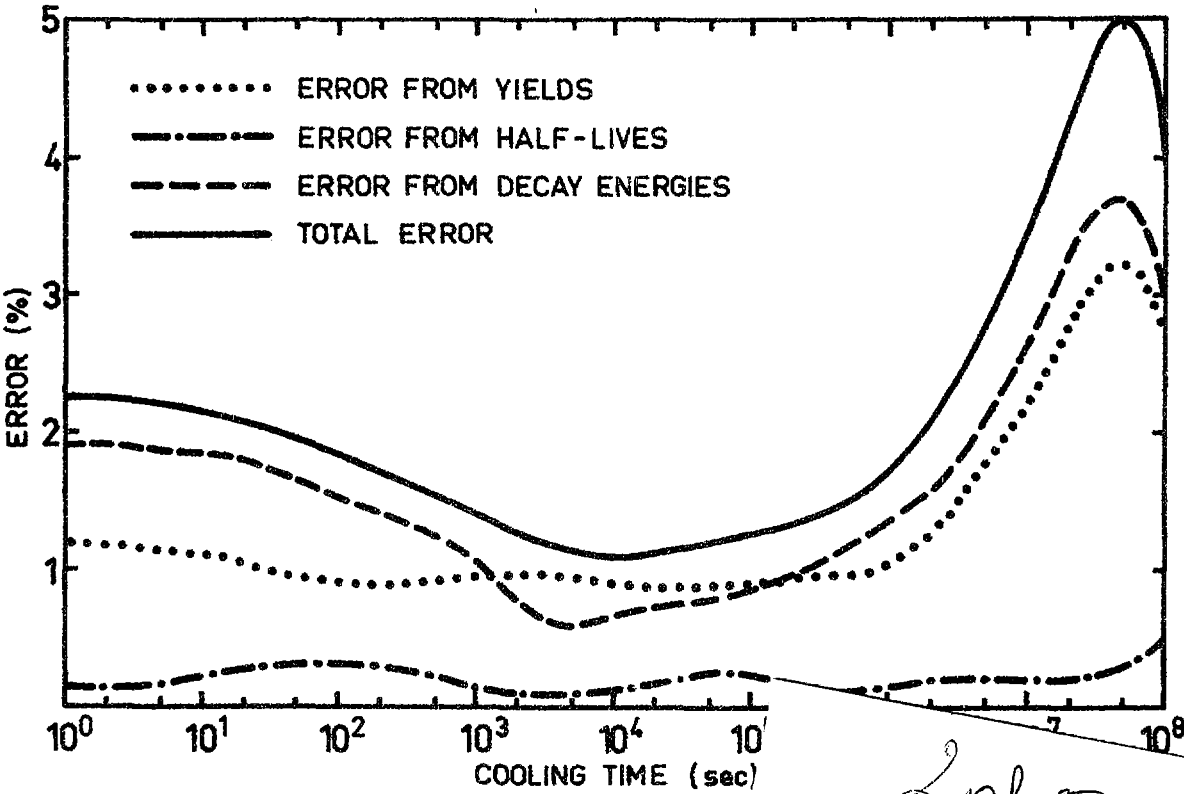

Fig. 5 _ ^{239}PU FAST FISSION DECAY I
Irradiation 900 days,

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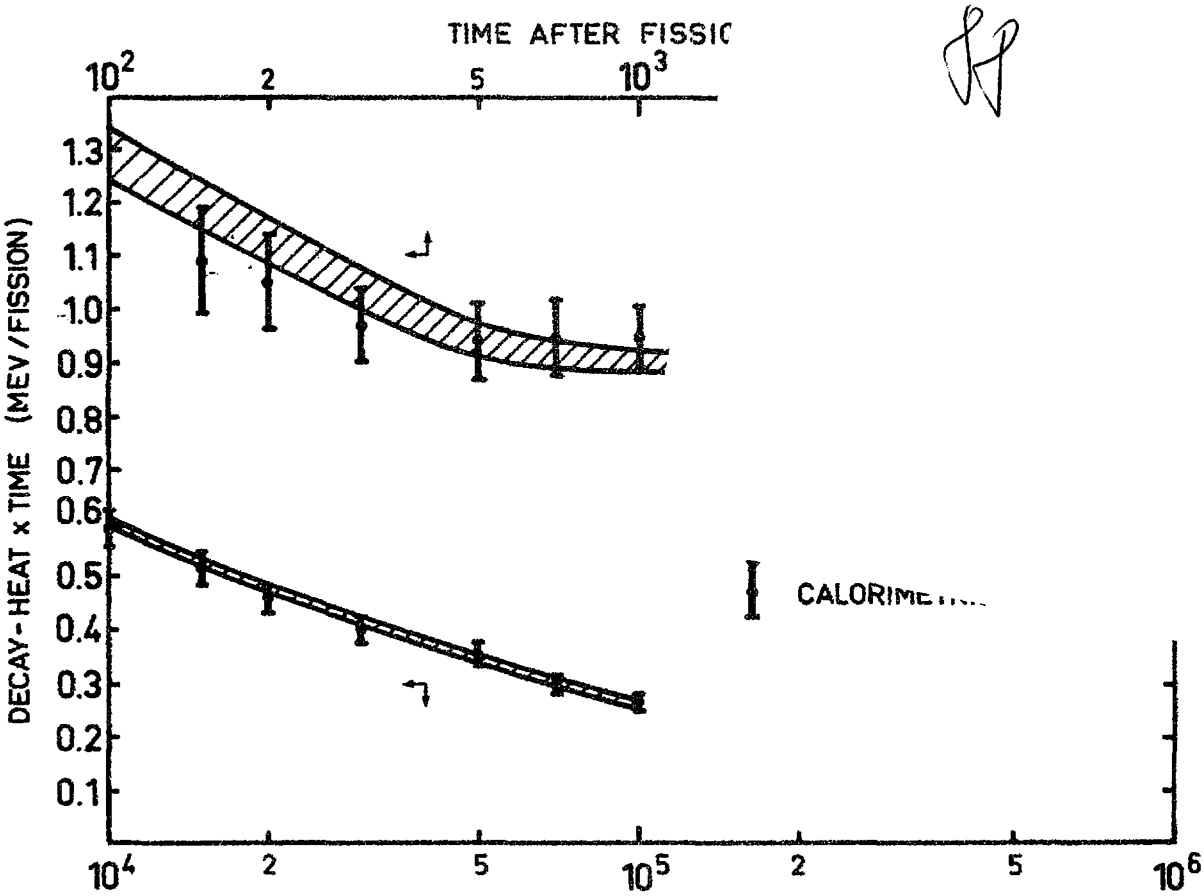

Fig. 6 _ DECAY HEAT FROM A FISSION BURST
^{235}U THERMAL FISSION

Review Paper 5

NEEDS AND ACCURACY REQUIREMENTS FOR FISSION PRODUCT NUCLEAR DATA IN THE OUT-OF-PILE FUEL CYCLE

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ABSTRACT

Both the conventional fuel cycles and a variety of alternative fuel cycles are outlined. The assessment of FPND requirements made at Bologna for the conventional cycles is confirmed in general, but cooling times of less than 6 months for FBR's are now regarded as unrealistic, so that certain FPND are no longer required for fuel cycle purposes. A preliminary assessment has been made of additional FPND requirements for alternative fuel cycles.

Particular topics discussed include:

- Decay heat calculations, for which the decay energies of ^{89}Sr, ^{137}Cs, ^{141}Ce and ^{144}Pr are in doubt.

- Radiation from spent fuel elements.

- Fission yields of ^{3}H, ^{129}I, $^{95}Zr/Nb$, ^{103}Ru, $^{106}Ru/Rh$, and stable and long-lived Mo, Tc and Pd, for all of which an accuracy of at least $\pm$ 10% is desirable for striking a mass-balance in reactors and reprocessing plants.

- The ^{133}Cs (nγ) cross-section, required in connection with burn-up measurements.

Introduction

At the Bologna meeting on Fission Product Nuclear Data (FPND) in 1973, there was a paper by E. Merz and M. Laser [1] on the importance of FPND for fuel handling. Since they interpreted fuel handling as "fuel storage and transport, reprocessing and refabrication, as well as possible isolation of actinide elements", it will be clear that they aimed to cover essentially the same ground as the present review. Their paper did not discuss waste management, since this was included in the review of the environmental significance of FPND, though they did themselves make a contribution on this topic [2], which was published in the Bologna proceedings. At the present meeting, waste management is again listed among the subjects to be dealt with in the environmental review, but obviously cannot be entirely excluded from discussion of fuel cycles.

There seems little that needs modifying in what was presented at Bologna, relevant to FPND and conventional fuel cycles, except that very short cooling times are no longer regarded as realistic for the FBR (Fast Breeder Reactor). This and other points requiring further consideration are dealt with below. Perhaps more important is to outline the developments that are taking place in our ideas about the fuel cycles themselves, and then to examine their significance for FPND.

In the four years since Bologna there has been an immense upsurge in public concern about the nuclear industry. There is anxiety about radioactive pollution, about proliferation of nuclear weapons, and about terrorist theft of fissile material. Anti-nuclear demonstrations, President Carter's nuclear policy, and the Windscale inquiry, illustrate different aspects of this concern.

There has as a result been a great ferment in the nuclear industry itself. Existing fuel-cycle strategies are being questioned, and alternatives are being proposed. Believing that nuclear power is the only hope of averting a world energy famine, those in the industry are making considerable endeavours to minimize the potential hazards. It is therefore an appropriate time to review the stores of information, including FPND, required for fuel cycle assessments.

Conventional Fuel Cycle

The title given for this review speaks of "the out-of-pile fuel cycle" in the singular, as if there was only one such cycle. Presumably it was the conventional U/Pu cycle involving thermal reactors that was intended. This is shown in outline in Fig. 1. The reprocessing plant here aims at a very high degree of separation of U, Pu and FP's (with the trivalent actinides, Am, Cm etc. accompanying the FP's), so as to provide:

(1) U that can be handled with no greater precautions than are needed for natural U, and can be fed to an enrichment (diffusion) plant.

(2) Pu that can be handled through gloves.

(3) FP's in a form suitable for disposal.

This is customarily achieved by wet processing, involving especially solvent extraction with tri-n-butyl phosphate (TBP); there is no reason at the present time to consider any alternatives.

Besides the main FP waste stream there are of course further waste streams, and those from the reprocessing plant especially are all contaminated in varying degrees. They include the discarded fuel element cladding ("hulls"), the FP insolubles from the dissolver, solvent wash wastes etc. (The FP insolubles are the FP's in the insoluble residue left when irradiated oxide or carbide fuel is dissolved in acid. They consist mainly of alloys of Mo, Tc, Ru, Rh, and Pd.)

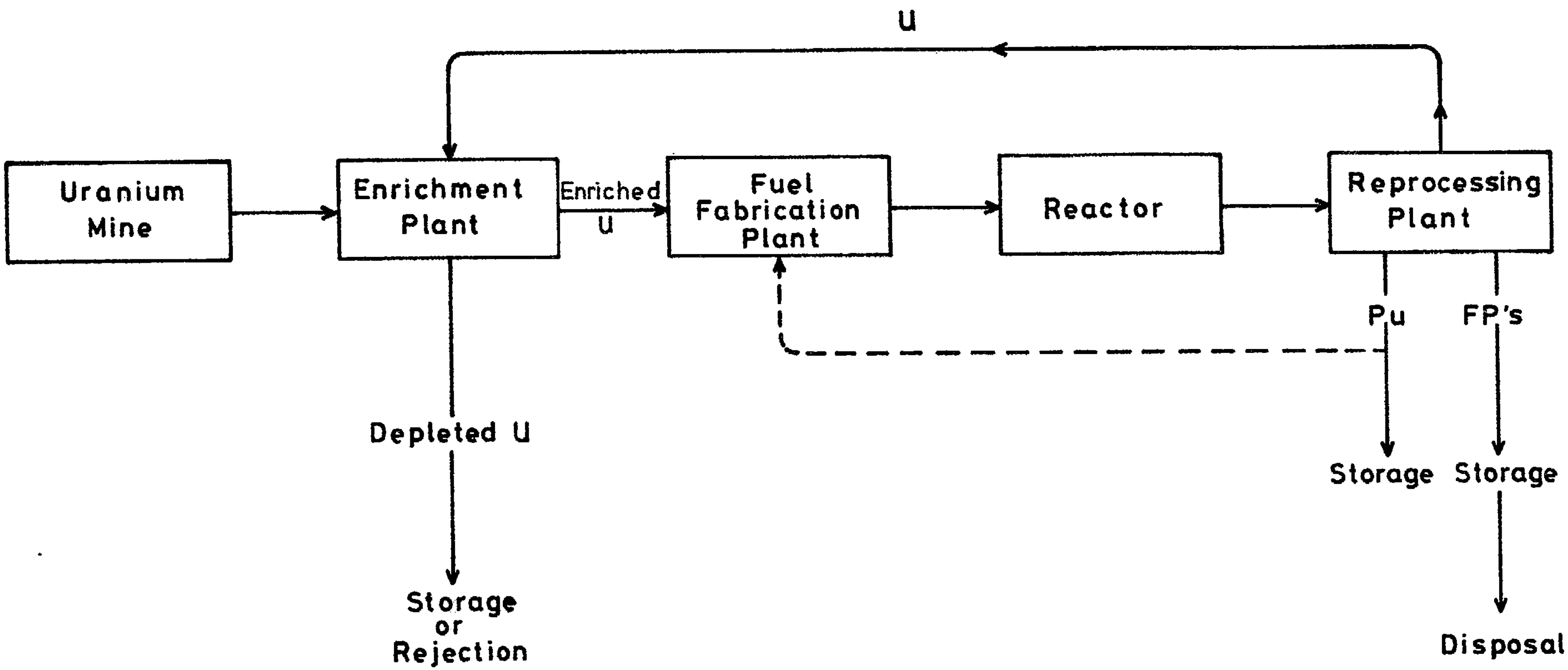

FIG.1. THERMAL REACTOR FUEL CYCLE

Although Fig. 1 is described here as the conventional fuel cycle, the complete cycle is only in small-scale operation anywhere in the world. Some reactors use natural U, viz. the British and French Magnox reactors and the Canadian CANDU reactors; in such cases there is no enrichment plant in the cycle. In the CANDU case, the reprocessing plant is also missing, the spent fuel being merely stored. For LWR (Light Water Reactor) fuel, enrichment is necessary, but reprocessing can be dispensed with, the spent fuel again being stored. In the U.S., whose power reactors are virtually all LWR's, there is in fact a moratorium on commercial reprocessing and only in Europe does the complete cycle function, and then only at low tonnages. However, both in Europe and Japan plans are being made to operate the full cycle on a substantial scale in a few years' time.

The dashed line in Fig. 1 represents a possible development of the conventional cycle: re-cycling of Pu to thermal reactors, especially LWR's.

If and when FBR's are introduced, the cycle in Fig. 1 will be coupled to a further cycle, in which its Pu and depleted U products are fabricated into FBR fuel and blanket material. The conventional form of this further cycle consists of fabrication plants for fuel and blanket, reactor, and reprocessing plant; the Pu and U are completely separated in the reprocessing plant and are totally recycled to the fabrication plants (Fig. 2). The Pu after recycling through FBR's will become too active (through growth of ^{241}Am) to handle through gloves, so the fuel fabrication plant will have to operate remotely.

A possible variation of Fig. 2 would have a separate reprocessing plant for the blanket material.

<u>Alternative Fuel Cycles</u>

The foremost reason at the present time for considering alternatives to the conventional cycles just described is to avoid the production of pure fissile material, especially Pu, at any stage, which might lead to proliferation of nuclear weapons or to thefts by terrorists. Other reasons are concerned with economics, maximum exploitation of uranium, and reduction of potential hazards.

First, there are alternatives which involve no major changes in the reactors or their fuels. These include the so-called "throw-away" cycle in which reprocessing is completely eliminated, the spent fuel being stored indefinitely; and cycles with modified reprocessing that does not yield a pure fissile material stream.

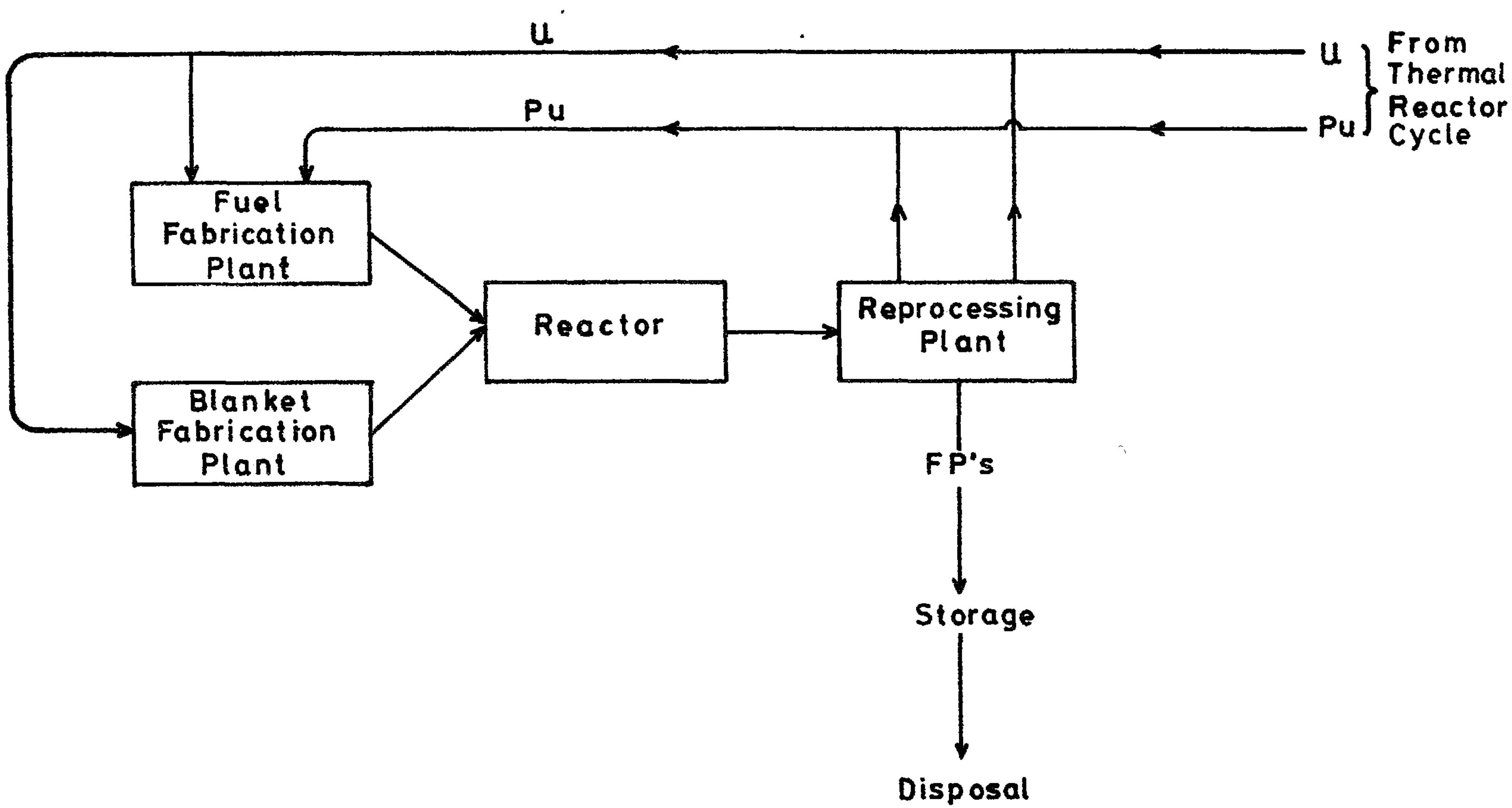

FIG. 2. FAST REACTOR FUEL CYCLE

There are three "throw-away" cycle concepts:

(1) The existing cycle terminating with spent fuel storage, as already practised in North America.

(2) The same cycle, but with much higher fuel burn-up to improve its U usage. This might for instance be achieved by re-design of the fuel elements.

(3) The tandem cycle, in which a high burn-up is obtained by passing the fuel through two reactors in succession. The chief opportunity for this lies with LWR's followed by CANDU's. The option is only attractive if the irradiated fuel elements can be transferred from one reactor to the other without alteration.

There are two principal ways in which reprocessing could be modified to avoid producing pure Pu:

(1) Co-processing, yielding a mixed U/Pu product instead of separate U and Pu streams. Either a single U/Pu stream, or a pure U stream and a U/Pu stream, could be produced.

(2) Incomplete decontamination of Pu, or mixed U/Pu, from FP's.

Either or both of these concepts might be introduced to simplify reprocessing and/or to render the Pu less attractive to terrorists. They might complicate fuel refabrication, but might be acceptable in certain circumstances, e.g. if remote fabrication were necessary for other reasons, such as the problem of handling recycled Pu from FBR's.

Secondly, there are various ways of modifying the reactors and their fuels, or using different reactor types, usually with the aim of extracting more energy from the U before it is discarded, and so making the "throw-away" option more attractive. Here there are three broad possibilities:

(1) Use of heavy instead of light water. It is the improved neutron economy so achieved that enables CANDU to use U about 50% more efficiently than an LWR. A similar benefit can be achieved by modifying LWR's to operate with a light/heavy water mixture, and changing the ratio as the fuel burns (the "spectral-shift reactor").

(2) Use of the Th/U cycle. There are many possibilities, all tending to economise U, which can be divided into two main groups:

(a) Use in existing reactor types. LWR's, CANDU's and AGR's (Advanced Gas-cooled Reactors) can in principle operate with a $Th/^{235}U$ mixture, or even with a Th/Pu mixture. An FBR might similarly have a Th blanket, and breed ^{233}U instead of Pu.

(b) Use in specially-designed reactors. The main contenders are HTR's (High Temperature Reactors), which can operate on a "throw-away" cycle.

(3) Use of homogeneous reactors. The fuel itself circulates outside the core through heat exchangers and a clean-up circuit to remove FP's. Pure fissile material need not appear at any point in the fuel cycle.

This is not the place to discuss the relative merits of these alternatives, but only to draw attention to their needs for FPND.

FP Behaviour

In conventional U/Pu fuel cycles, spent fuel discharged from the reactor is stored at the reactor site; transported to the reprocessing plant; stored for a further period; and then reprocessed. The U and Pu products of reprocessing are stored, and may then be routed to fuel fabrication plants or, in the case of U, to an enrichment plant.

Up to the reprocessing stage, the totality of the FP's are present, with minor exceptions, but after that, separations occur. The majority of the FP's go into the high-level waste (HLW), but other plant streams may contain particular FP's:

Table 1 - Radioactive FP's in Reprocessing Plant Streams

Stream	Radioactive FP's which may be present
Off-gases	^{3}H, ^{85}Kr, ^{125}Sb, ^{129}I, ^{131}I
Zircaloy hulls	^{3}H, traces of all FP's
SS hulls	Traces of all FP's
FP insolubles	^{95}Zr/Nb (traces), ^{99}Tc, ^{103}Ru, ^{106}Ru/Rh, ^{107}Pd
High-level waste	All non-volatile FP's
Medium- and low-level wastes	^{3}H, ^{95}Zr/Nb, ^{103}Ru, ^{106}Ru/Rh
U and Pu products	^{95}Zr/Nb, ^{99}Tc, ^{103}Ru, ^{106}Ru/Rh
Fuel pond storage water	^{89}Sr, ^{90}Sr/Y, ^{103}Ru, ^{106}Ru/Rh, ^{134}Cs, ^{137}Cs/Ba

Inactive isotopes of the elements mentioned will, of course, also be present in the plant streams, and also stable Xe in the off-gases, and stable Mo in the FP insolubles.

Merz and Laser [1] suggested that shorter cooling times down to a month might be adopted for FBR's and that a further range of FP's with half-lives generally of a few days might therefore have to be considered. The consensus among reprocessing chemists is now, however, that cooling times of less than 6 months are unrealistic.

The situation just outlined would remain qualitatively the same in many respects in the alternative strategies. Exceptions are:

(1) If there is no reprocessing the FP's remain with the spent fuel, apart from small leakages of volatiles to the gas phase, and of soluble species, particularly ^{137}Cs, to the fuel pond water.

(2) There are additional complications in the Th/U cycle, e.g. a Pa removal step in reprocessing, when some of the FP's may accompany the Pa.

(3) There is a totally different situation with homogeneous reactors, requiring detailed consideration in each partiuclar case.

Quantitatively, of course, there would be many differences between strategies. Particularly significant may be the strategies involving very high fuel burn-up, when the FP composition will be considerably modified.

General Comments on FPND Requirements

FPND requirements fall into three categories:

(1) Requirements related to the radioactivity of the total FP mixture, or at least to the non-volatile fraction of the mixture. The principal effects which arise in this context are shielding, radiolysis, and decay heating. They have to be considered during storage and transport of spent fuel; in the head-end stages of reprocessing; and in storage or disposal of the HLW.

(2) Requirements related to the radioactivity of individual FP's, including those listed in Table 1. Here again there are shielding and radiolysis problems, though not in most cases decay heating problems, and in addition individual FP's cause handling and disposal problems for the materials listed in Table 1. All such problems are largely eliminated in throw-away fuel cycles, though it may still be necessary to consider the behaviour of individual volatile and leachable species.

(3) Requirements related to the total quantities (stable plus radioactive) of individual fission products. Here the problems are mainly chemical in nature, and arise in connection with the metallurgy and chemistry of the discharged fuel; the off-gases; the FP insolubles; and the HLW.

The FPND accuracy requirements for fuel cycle purposes are less severe than in other connections, and broadly speaking existing data are adequate, as Merz and Laser [1] indicated at Bologna. They stated that the total decay heat should be known to ± 5%, and it can be agreed that this is the most crucial feature. Otherwise, for longer-lived r dioactive species (say with half-lives > 5 days) they suggested an accuracy of ± 5% to ± 10%, and for shorter-lived species, of ± 10% to ± 20%. Although these are somewhat vague statements, they can generally be accepted as adequate, because there is so little strigency in most of the requirements. Indeed for many purposes, uncertainties amounting to a factor of 2 are tolerable. A further reason for accepting relatively low accuracies is that uncertainties in irradiation histories, chemical behaviour, etc. under practical conditions are frequently more important than errors due to inaccurate FPND.

It should be noted that fission yield data are sometimes required for minor actinide species, such as ^{236}U, ^{237}Np, ^{241}Am etc., which can on occasion make a significant contribution to the total yields of some FP's. These contributions will be particularly important in very high burn-up fuel cycles, and of course in nuclear incineration to destroy the minor actinides by recycling them to reactors. The accuracy required obviously depends on the magnitude of the contributions in relation to those from the major actinides. Usually only low accuracy is necessary, but in some circumstances the accuracy needs to be as high as Merz and Laser proposed for FP arisings from the major actinides.

Decay Heat Calculations

Only for ^{89}Sr, ^{137}Cs, ^{141}Ce, and ^{144}Pr among the longer-lived FP's are the discrepancies between different calculations of the decay heat contributions greater than ± 5% [3], and this is broadly speaking encouraging, since the consequent discrepancies in the total decay heat will generally be much less than ± 5%. It should be noted, however, that ^{144}Pr is a major contributor at 100 days cooling. The discrepancies appear to arise from the values recorded in different libraries for the mean β-energies and, in the case of ^{141}Ce, also from that for the mean γ-energy. The uncertainties moreover suggest the possibility that the absolute errors in the values in the libraries may be larger than supposed.

The ± 5% accuracy requirement applies principally to interim storage and transport of spent fuel before reprocessing, transport being probably the more critical. It is important to optimise transport conditions, in view of the costs involved, and the decay heat plays a significant part in this.

Handling and shearing (or other treatment) of the fuel in the head-end of the plant is another critical area as regards heat evolution, especially for FBR fuel. The problems are difficult to assess, so that large safety margins must be provided; in these circumstances $\pm$ 10% accuracy in calculating the decay heat ma be regarded as adequate

Similarly, $\pm$ 10% may be sufficient accuracy in assessing the decay heat from the FP insolubles, due principally to ^{103}Ru and ^{106}Ru/Rh [4]. The question is important, especially in FBR reprocessing, because several kilowatts of heat may be evolved from a small mass of material in a single day's throughput of a typical FBR reprocessing plant.

Decay heats are also required in connection with very long-term storage and disposal, whether of untreated spent fuel or of vitrified HLW. Initially such material will be provided with cooling, but eventually it may be transferred to a permanent repository without special cooling. The dominant FP's at this stage will be ^{90}Sr/Y and ^{137}Cs/Ba, whose FPND are well-known, so no particular difficulties are expected.

Radiation from Spent Fuel Elements

Large quantities of spent fuel elements will be available in future, especially if "throw-away" fuel cycles are adopted. The question of whether these might be employed as radiation sources has recently been considered by D.F. Sangster [5]. Their use would involve detailed information on the γ-energies emitted as a function of time; existing FPND are probably satisfactory for this purpose, though this would need investigation. However, Sangster concludes against spent fuel elements becoming a major source of radiation for processing.

Yields of Particular FP's

Tritium. Tritium fission yield measurements have been reviewed by J.G. Cuninghame [6] and by E.A.C. Crouch [7]. The measurements may be made by chemical means (isolation and counting of the tritium) or physical (counting the energetic light particles from a thin source subject to fission). The latter give the ^{3}H/^{4}He ratio, and the value for the tritium yield depends on the value taken for the ^{4}He yield. The ^{4}He yields, however, do not appear to be well-established.

For thermal neutron fission of ^{235}U, the weighted mean of 6 published "chemical" measurements is

$$0.89 \pm 0.04 \text{ tritons per } 10^4 \text{ fissions,}$$

while a provisional unpublished measurement by Crouch gives 0.92 tritons per 10^4 fissions. The "physical" measurements depend on assumed values for the ^{4}He yield, which appear to be underestimated by the technique used; a factor of 1.6 is needed to bring them into line with the "chemical" results.

For fast neutron fission of ^{235}U there is only one set of published results, from which the authors propose a value of

$$2.0 \pm 0.2 \text{ tritons per } 10^4 \text{ fissions}$$

over the range 0.17-0.70 MeV, but the results show considerable scatter. Both "chemical" and "physical" methods were used, and appear to be in agreement without the introduction of the 1.6 factor required at thermal energies.

Only thermal fission has been investigated for ^{239}Pu and ^{233}U, and only "physical" measurements have been made for ^{239}Pu. Applying the 1.6 factor to the mean of 4 "physical measurements for the thermal neutron fission of ^{239}Pu, we obtain

$$1.02 \pm 0.07 \text{ tritons per } 10^4 \text{ fissions}$$

but this value cannot be regarded as well established.

For thermal fission of ^{233}U, Cuninghame [6] quotes one "chemical" measurement (0.91 ± 0.06) and 2 "physical" measurements (mean = 0.98), but inadvertently indicates all 3 as "physical". If the factor 1.6 is applied to the "physical" results, a value of 0.61 is obtained. It is difficult to recommend a value from this evidence, but

$$0.8 \text{ tritons per } 10^4 \text{ fissions}$$

may be suggested.

For thermal reactors, so long as fission is principally in ^{235}U, tritium production by fission can be calculated to within ± 5%. As burn-up increases, and a significant proportion of the fissions are in ^{239}Pu, the uncertainties become larger. ± 5% is sufficient for striking a mass-balance between tritium production and tritium appearing in different plant streams; and amply sufficient for environmental purposes, when results are not required to better than ± 20% [8].

For fast reactors, for thermal reactors with Pu recycling, and for reactors using the Th/U cycle, the data are clearly altogether inadequate. On the basis of the fast neutron data for ^{235}U it is often assumed that all the Pu isotopes yield 2 to 3 tritons per 10^4 fast neutron fissions, but this is guesswork, since there is no theory to indicate how the yield varies with nuclear properties and neutron energy.

Measurements are in progress by Crouch at AERE, Harwell for thermal fission in ^{235}U and ^{239}Pu, and for fast fission in ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu.

<u>Iodine 129.</u> One other volatile FP, ^{129}I, may be mentioned. For environmental purposes its yield may be required to ± 20%, in order to make due allowance for the effect of isotope dilution by natural ^{127}I, while ± 10% is desirable for mass-balance calculations [8].

Zirconium and ruthenium. ^{95}Zr/Nb, ^{103}Ru, and ^{106}Ru/Rh are a major preoccupation of reprocessing chemists, as Table 1 shows. After the first solvent extraction contactor, they are the only significant FP's remaining with the U/Pu stream(s), and their removal is a principal objective of the later parts of the plant. It is therefore desirable to know the quantities of these species produced in the fuel. For flowsheeting purposes, an accuracy of a factor of 2 is generally sufficient, though if a mass-balance is to be struck, ± 10% is indicated [8].

In some connections the yield of the stable as well as the radioactive Zr and Ru species are required. This applies in the production of FP insolubles, and in the precipitation of Zr compounds, e.g. from the HLW; an accuracy of ± 10% is desirable.

Other species significant in reprocessing. Besides volatile species, Zr, and Ru, the FP's requiring special consideration in reprocessing include Mo, Tc, and Pd. All three occur in the FP insolubles, and Mo is also often an important constituent of precipitates in the HLW. Stable as well as radioactive species must be considered. An accuracy of ± 10% in the yields is again desirable [8].

The ^{134}Cs/^{137}Cs ratio. Since ^{134}Cs is a second-order product, formed by neutron-capture in the FP ^{133}Cs, and ^{137}Cs is a first-order product, the ratio of the quantities of the two isotopes is to a first approximation proportional to burn-up, and the ratio is therefore used for burn-up measurements. However, the value of the ratio for dissolved Magnox fuel, as measured by A.J. Fudge [9], was only about half the calculated value. He has also observed considerable variations in the ratio for different fast reactor fuel specimens, after allowing for the different burn-ups. He believes that the source of the discrepancies is the cross-sections used for the ^{133}Cs(nγ) reaction. There is a large resonance in the epithermal region, and this could lead to errors in the three-group averaging used in the calculations. The ^{134}Cs/^{137}Cs ratio is a function not only of the burn-up, but also of the hardness of the neutron flux, and accurate knowledge of the ^{133}Cs(nγ) cross-section is required to make the necessary corrections. It seems that the ratio can be used satisfactorily to determine burn-up in thermal reactors, provided the method is first calibrated under the conditions concerned, but not at present in fast reactors.

The ^{134}Cs/^{137}Cs ratio is also of significance in connection with the shielding requirements for radiocaesium. For large sources (say > 1 kCi) 0.5 cm extra of lead shielding is needed for each factor of 2 in the ratio; this is because ^{134}Cs emits a proportion of rather hard γ-rays. However, there is no longer much interest in large radiocaesium sources, radiocobalt being preferred. For nuclear fuel cooling pond water, ^{137}Cs dominates the picture, and the small proportion of ^{134}Cs can be neglected so far as radiological effects are concerned.

FPND and Alternative Fuel Cycles

Some of the consequences of introducing alternative fuel cycles have already been mentioned. Additional comments follow.

Effect of very high burn-up. Several of the alternative fuel cycles envisage very high burn-up. The effect of this on FP composition does not appear to have been closely studied, but it is desirable that it should be, to identify resultant FPND needs. Among the possibilities are requirements for more accurate cross-section data on radioactive FP's of medium half-life, and fission yield data for the higher actinides, which may, as already noted, begin to make significant contributions to the total FP yields. It may also be important to define the chemical composition of the highly burned-up fuel with some precision, and this may give rise to requirements for accurate calculations of the yield of different FP elements.

Effect of introducing the Th/U cycle. When the Th/U cycle is employed, data for ^{233}U fission constitute an additional requirement.

Use of homogeneous reactors. In the operation of a homogeneous reactor it is necessary to consider the behaviour of a number of short-lived FP's in the FP clean-up loop, in the core cover gas etc. However, such items should be regarded as part of the reactor itself, rather than the out-of-pile fuel cycle. The same applies, of course, to the behaviour of FP's in the fluid fuel itself, e.g. their precipitation. Major processing of the fuel, would, so far as possible, be carried out similarly to conventional reprocessing, and would not then introduce any novel FPND requirements; however this would need study in relation to specific homogeneous reactor proposals.

Acknowledgements. Communications and comments have been received from those listed below in refs. [3-8], and also from A.L. Mills of Dounreay.

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Review paper 6

A REVIEW OF FISSION PRODUCT NUCLEAR DATA REQUIREMENTS FOR INVESTIGATION OF IRRADIATED NUCLEAR FUEL:

BURNUP MEASUREMENTS
NEUTRON DOSIMETRY
NUCLEAR SAFEGUARDS

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ABSTRACT

The needs and accuracy requirements for fission product nuclear data (FPND) important to burnup measurements, neutron dosimetry, and safeguards have been reviewed. For burnup, the most pressing needs are for more accurate fission yields and a knowledge of the change in yields with neutron energy. For neutron dosimetry, improvements in certain fission yields and decay schemes are required. The needs for improved FPND for safeguards are not well defined. Until a firm plan for safeguarding nuclear materials is implemented and the exact data of interest are stated, current requests for improved FPND are assigned a lower priority than those for burnup, neutron dosimetry, and reactor physics.

1. INTRODUCTION

This review paper presents the needs and accuracy requirements for fission product nuclear data (FPND) important to the fields of burnup analysis, neutron dosimetry, and nuclear safeguards. These subjects have been combined into a single review because they are becoming highly correlated with respect to the use of the results and the requirements for similar data and accuracies.

This review is a follow-up to that presented at the Panel on FPND, which was held in Bologna, Italy in 1973[1]. Therefore, much of the introductory material presented at the Bologna meeting has been omitted. Of primary concern, in this review, is the listing of prior FPND requirements which still have not been satisfied, updated requirements, especially with respect to increased accuracy needs, and new requirements which have surfaced in the interim.

2. FPND REQUIREMENTS FOR BURNUP MEASUREMENTS

2.1 Uses of Burnup Data

In order to formulate the requirements of precision and accuracy as related to the measurement of burnup, it is first necessary to identify the uses to which the final results will be put. Listed below are many uses of burnup data. While this list may not be all inclusive, it does indicate the wide variety of uses to which burnup data are put.

BU-1, Measurement of the integrated number of fissions.

BU-2, Measurement of the integrated individual sources of fission.

BU-3, Determination of the energy release per unit mass or volume of fuel.

BU-4, Determination of the fission rate, integrated and terminal.

BU-5, Correlation of fuel temperature with fuel melting and fission gas retention.

BU-6, Decay heat calculations as related to nuclear safety considerations.

BU-7, Shielding, coolant requirements, and transportation calculations.

BU-8, Estimation of radiolysis and solvent damage in fuel processing facilities.

BU-9, Waste Management studies.

BU-10, Verification of nuclear physics reactor prediction codes.

BU-11, Calculation of residual fuel content and reactivity.

BU-12, Contractual agreements for fissionable and fissile element content in the reprocessing of fuel.

BU-13, Calibration of non-destructive burnup analysis techniques.

Burnup can be defined in several ways, depending upon the user and his needs. To avoid confusion, in this review, burnup will always mean:

$$\text{Burnup} = \text{Atom \% fission} = \frac{\text{number of fission X 100}}{\text{initial number of total heavy elements atoms}}$$

This definition relates directly the number of events of interest, fissions, to the quantity of most interest, the number of fissionable and fertile atoms in the fuel at the beginning of the irradiation.

2.2 Measurement of Burnup

Before discussing the needs and accuracy requirements for the determination of burnup, it might be well to briefly review the measurement techniques and sources of error associated with a burnup analysis.

In Review Paper No. 5 (RP-5), presented at the Bologna Conference[2], the various methods, both destructive and non-destructive, for the determination of nuclear fuel burnup, were discussed. Of these, the most accurate and widely applicable is the fission product monitor-residual heavy atom technique. In this method, the fuel specimen is dissolved and the fission product monitor and heavy atoms are determined. Burnup is computed from the relationship

$$\text{Burnup} = a/oF = 100\,\frac{A/Y}{H+A/Y}$$

where a/oF = atom percent fission

A/Y = number of fission,in which A is the determined atoms of the fission product monitor nuclide, and Y is the effective fractional fission yield of A

H = determined number of residual heavy atoms.

The successful application of this technique requires accurate measurements of the fission product monitor and heavy atoms and an accurate value for the effective fission yield. The most accurate technique for the measurement of the fission product monitor and the heavy elements is isotope dilution mass spectrometry. Currently, errors of $\sim$0.5% on the measurement of ^{148}Nd atoms and $\sim$0.2% for uranium and plutonium are achievable on a semi-routine basis using highly qualified individuals. Thus, the propagated uncertainty associated with the measurements is less than 1% relative. The dominant error in the final burnup value lies with the value for the effective fission yield. To attain an uncertainty in the final burnup value of 1.5-2.0% relative, requires that the uncertainty in the effective fission yield be in the range of 1-1.5% relative.

The interplay of the random and systematic errors in the measurements and the error associated with the value for the fission yields was discussed in some detail in RP-5 of the Bologna Conference.[2]

2.3 Selection of Fission Product Monitors

2.3.1. Thermal Fission

For many years, ^{148}Nd has been considered as the near ideal burnup monitor for thermal light water reactor fuels because 1) neodymium possessed the desirable chemical characteristics and behavior, both in the fuel and in solution, 2) its fission yield is essentially identical for ^{235}U and ^{239}Pu thermal fission, and 3) it was believed that ^{148}Nd was not subject to extensive neutron capture corrections.

In RP-5 at the Bologna Conference[2], it was suggested that a large, then unknown thermal neutron capture cross section existed for 11d-^{147}Nd, which could give rise to an abnormally high amount of ^{148}Nd, especially in a high flux reactor. The result would be a high biased value for the burnup. A recommendation of the Bologna Panel[1] was that the ^{147}Nd thermal capture cross section should be measured. In 1974, Heck, Borner, Pinston, and Rousille[3] reported a measured thermal neutron capture cross section for ^{147}Nd of 440±150b. In support of this high ^{147}Nd cross section, Table I gives data obtained in the reviewer's laboratory[4] from a series of analyzed sample punchings from a single fuel plate (93% enriched ^{235}U) which had been irradiated in the core of the high-flux Advanced Test Reactor in Idaho. Also given in Table II is the isotopic abundance of ^{148}Nd obtained from a separate sample irradiated in a lower flux for a shorter period of time. That the ^{148}Nd relative atom abundance increases with flux and burnup is evident.

TABLE I

MEASURED ^{148}Nd ATOM FRACTION ABUNDANCE FOR INCREASING NEUTRON FLUX

	Flux, n/cm /sec[a]	Burnup	^{148}Nd, Atom Fraction Abundance
ATR FUEL PLATE	0.56×10^{14}	8.8	0.09186
	1.40×10^{14}	20.1	0.09476
	2.08×10^{14}	28.2	0.09748
	3.54×10^{14}	42.4	0.1029
	0.2×10^{14}	∿1	0.08079

[a]Includes thermal and resonance.

The impact of this high ^{147}Nd cross section is two-fold. First, it may mean that previously measured ^{148}Nd atom abundances, and hence, fission yields may be too high compared to the instantaneous values. In another publication relative to this subject[4], this reviewer has suggested that the fission yield for ^{148}Nd from ^{235}U thermal fission may be ∿1.65% rather than 1.68%. This subject is discussed in more detail by J. G. Cuninghame RP-10 at this conference.

The second impact relates to the measurement of the number of fissions and burnup where ^{148}Nd is used as the fission monitor. It is now believed that some of the reported data for the number of fissions and, hence burnup, are biased high. That is, the reported number of fissions exceeded the actual number of fissions because of excess ^{148}Nd being produced from capture on ^{147}Nd. The magnitude of this error is a function of the flux and fluence.

To provide the analytical chemist who invariably measures and reports the number of fissions, and the user who evaluates the data, with a guide as to the magnitude of the corrections involved, the excess ^{148}Nd produced by capture on ^{147}Nd has been calculated as a function of neutron flux and fluence. These results given in Figure 1 are for the following conditions: ^{147}Nd σ_c = 450 b, ^{148}Nd σ_c = 2.5 b, ^{235}U σ_f = 580 b, 147/148 fission yield ratio f 1.34, and continuous neutron exposure. These data (Figure 1) indicate corrections ranging from a few percent for power reactor fuels to 20% or more for high flux irradiations. The latter is especially significant for constant prolonged high flux irradiations, which could be the case for experimental fuel development and clad failure studies. The reader is cautioned that Figure 1 is only a guide because the resonance capture cross section is unknown and a significant fraction of reactor down time will reduce the magnitude of the capture effect.

For burnup analyses on highly enriched fuels where there is only one major source of fission, ^{235}U or ^{239}Pu, it is recommended that the sum of ^{145}Nd + ^{146}Nd be used to obtain the most nearly correct value for the number of fissions[4]. Although ^{145}Nd has a significant capture cross section (∿50 b), the ^{145}Nd production cross section from ^{144}Ce and ^{144}Nd is small (∿3 b) and approximately equivalent to the burnout cross section on ^{146}Nd. For the data given in Table I, the measured

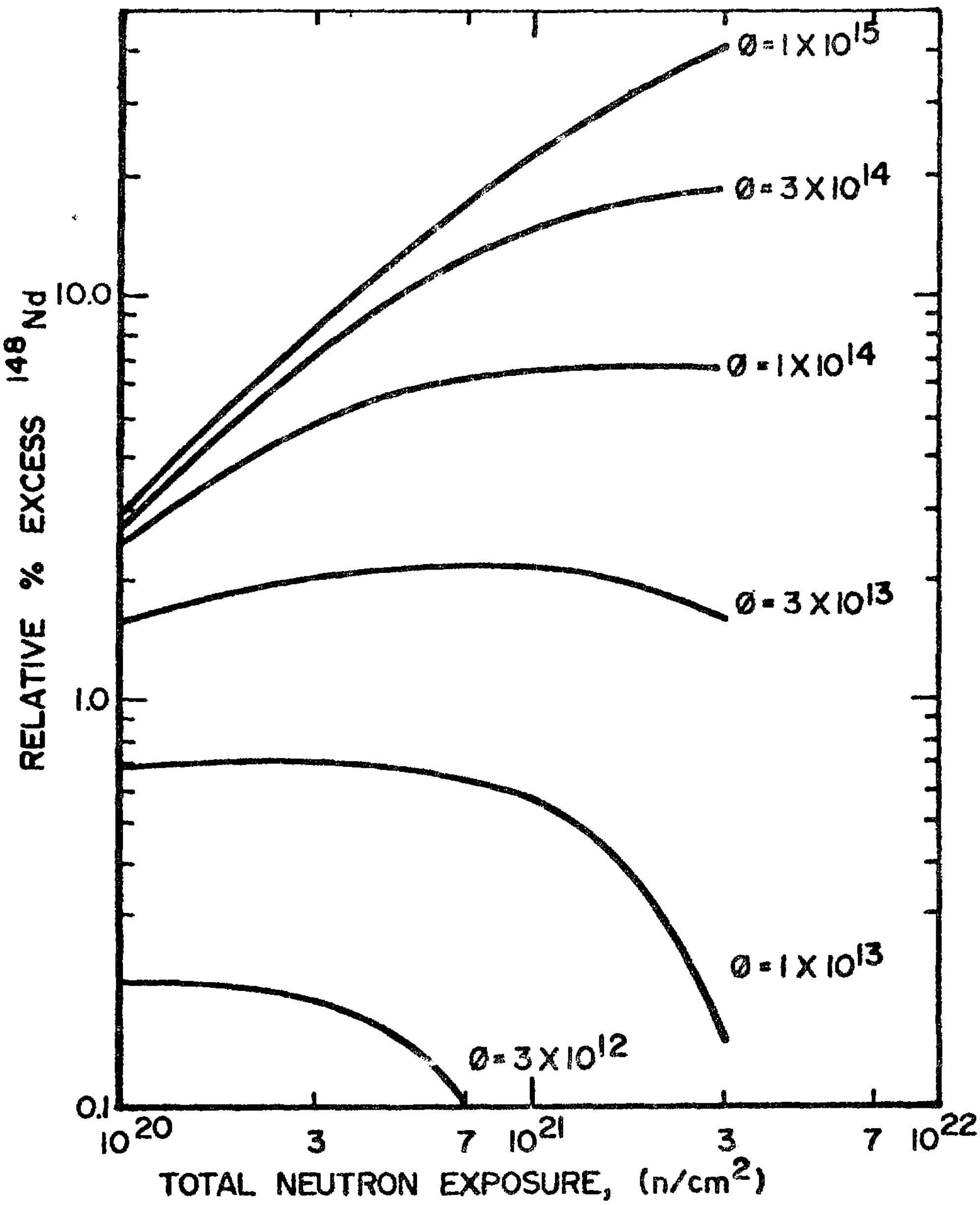

Fig. 1 Effect of Neutron Flux and Fluence on Measured Abundance of ^{148}Nd

atom ratio of ^{145}Nd + ^{146}Nd to the sum of the total neodymium atoms was constant within 1% relative over the flux range of the five samples.

For low enrichment, moderate-flux power reactor fuels, the correction for neutron capture on ^{147}Nd is small (generally 2-5%). Unfortunately, the ^{145}Nd + ^{146}Nd summing technique discussed above for single source fissioning fuels is not applicable because the fission yields of ^{145}Nd and ^{146}Nd for ^{235}U and ^{239}Pu are different. Therefore, when ^{148}Nd is used as the fission monitor, the requester should provide to the measurer an estimate of the neutron flux and irradiation history especially with respect to reactor up and down time such that the appropriate correction factor can be calculated. The reactor cycling time could be important if the down times were frequent and the durations significant compared to the 11-d half-life of ^{147}Nd. The fraction of down time will reduce the magnitude of the correction.

For thermal reactor fuels in which ^{233}U and ^{235}U are the principal sources of fission, ^{140}Ce appears more suitable as a fission monitor than any of the isotopes of neodymium. Cerium retains the desirable chemical properties of neodymium and the fission yield for mass 140 is essentially the same ($\sim$6.35%) for ^{233}U and ^{235}U thermal fission. The major deterrent to the use of ^{140}Ce is that the mass spectrometric measurement is less precise than that for neodymium because it is more difficult to correct for natural contamination.

2.3.2 Fast Fission

For fast reactor fuels, the selection of an ideal fission monitor is more complex than for thermal fuels. This results from 1) more widely varying fuel compositions, 2) varying sources of fission, especially in fuel development studies, and 3) until recently, the lack of reliable fission yield data and a knowledge of the changes in the fission yields with neutron energy. To fulfil the needs for burnup measurements on fast reactor fuels, many laboratories have continued to use ^{148}Nd, recognizing that in many cases it is not the ideal monitor.

This problem was recognized at the Bologna Conference[1] and one of the major conclusions of that conference was the need for more accurate fission yield data for a large number of the heavy nuclides.

Since that time we have reported[5,6] new absolute fast reactor fission yields for ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu while continuing to work on new yields for ^{240}Pu, ^{242}Pu, ^{241}Am, and ^{237}Np. The ^{237}Np results will be available in September 1977[7], the ^{240}Pu and ^{242}Pu yields early to mid 1978, and the ^{241}Am data late, in 1978. These data are from samples irradiated in the Experimental Breeder Reactor-II (EBR-II) in a neutron spectrum characteristic of a large mixed-oxide fueled LMFBR.

To provide a mechanism to assist in the selection of burnup monitors for fast reactor fuels, the fission yields for the isotopes of Nd are given in Table II for ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, and ^{241}Pu From these data, it is quite evident that the fission yield of ^{148}Nd is too variable to be used as a universal fission monitor, and that other neodymium isotopes may be a more ideal monitor depending on the sources of fission.

If it is assumed, for most fast reactor fuels that two nuclides contribute the bulk of fissions (∿90%), then the selection of the monitor can be based on the difference in the individual fission yields between the two major source of fission. To illustrate this, listed in Table III directly below Table II, are the differences in the neodymium fission yields for four different primary sources of fission. The percent difference in the individual fission yields was calculated by subtracting the smaller of the two yields from the larger, dividing by the average of the two yields, and multiplying by 100. From Table III, it is quite evident that the monitor nuclide having the most identical yields, or least affected by the different sources of fission, is variable and dependent upon the primary sources of fission involved. For each case, the promising monitor has been underlined. Three of the cases listed in Table III are for experimental fuels with varying heavy element composition and one, the ^{239}Pu-^{241}Pu couple, is applicable to a plutonium fueled fast breeder. From these data, the use of ^{143}Nd for a fission monitor for plutonium fuels appears promising. For other mixtures of isotopes, a similar approach is suggested.

To further evaluate the use of ^{143}Nd as a burnup monitor for plutonium fueled fast breeders, the effective ^{143}Nd fission yield for ^{239}Pu and ^{241}Pu has been plotted versus the fractional sources of fission from these two isotopes (Figure 2). For a reference point, it has been assumed that in a fast breeder fueled with recycle plutonium, that the ratio of ^{239}Pu fission to ^{241}Pu fission is ∿4 to 1. For this case, the effective fission yield of ^{143}Nd is 4.42%. The ±1% relative error lines (Figure 2) placed about this value shows that it is applicable over nearly any reasonable ratio of ^{239}Pu to ^{241}Pu fission within this limit. Also plotted on the right side of Figure 2 is the fast reactor yield for ^{238}U which can be expected to be the third largest source of fission

TABLE II

FAST REACTOR FISSION YIELDS FOR THE NEODYMIUM ISOTOPES[5,6]

	FISSIONING NUCLIDE				
Isotope	^{233}U	^{235}U	^{238}U	^{239}Pu	^{241}Pu
^{143}Nd	5.54	5.80	4.56	4.38	4.60
^{144}Nd	4.40	5.27	4.48	3.72	4.20
^{145}Nd	3.20	3.83	3.76	3.01	3.27
^{146}Nd	2.39	2.94	3.40	2.47	2.74
^{148}Nd	1.20	1.68	2.08	1.65	1.91
^{150}Nd	0.465	0.672	1.25	0.982	1.19
Σ Nd-144	12.80	14.92	15.05	12.49	13.71

TABLE III

FISSION YIELD DIFFERENCES FOR PRIMARY SOURCES OF FISSION

	% Difference in Fission Yields for Two Sources of Fission			
	^{233}U-^{235}U	^{233}U-^{239}Pu	^{235}U-^{239}Pu	^{239}Pu-^{241}Pu
^{143}Nd	4.6	23.4	24.4	4.9
^{145}Nd	17.8	16.1	24.0	8.3
^{146}Nd	20.7	3.3	17.4	10.4
^{148}Nd	33.3	31.6	1.8	14.6
^{150}Nd	36.4	71.5	37.5	19.2
Σ Nd-144	15.3	2.4	17.7	9.3

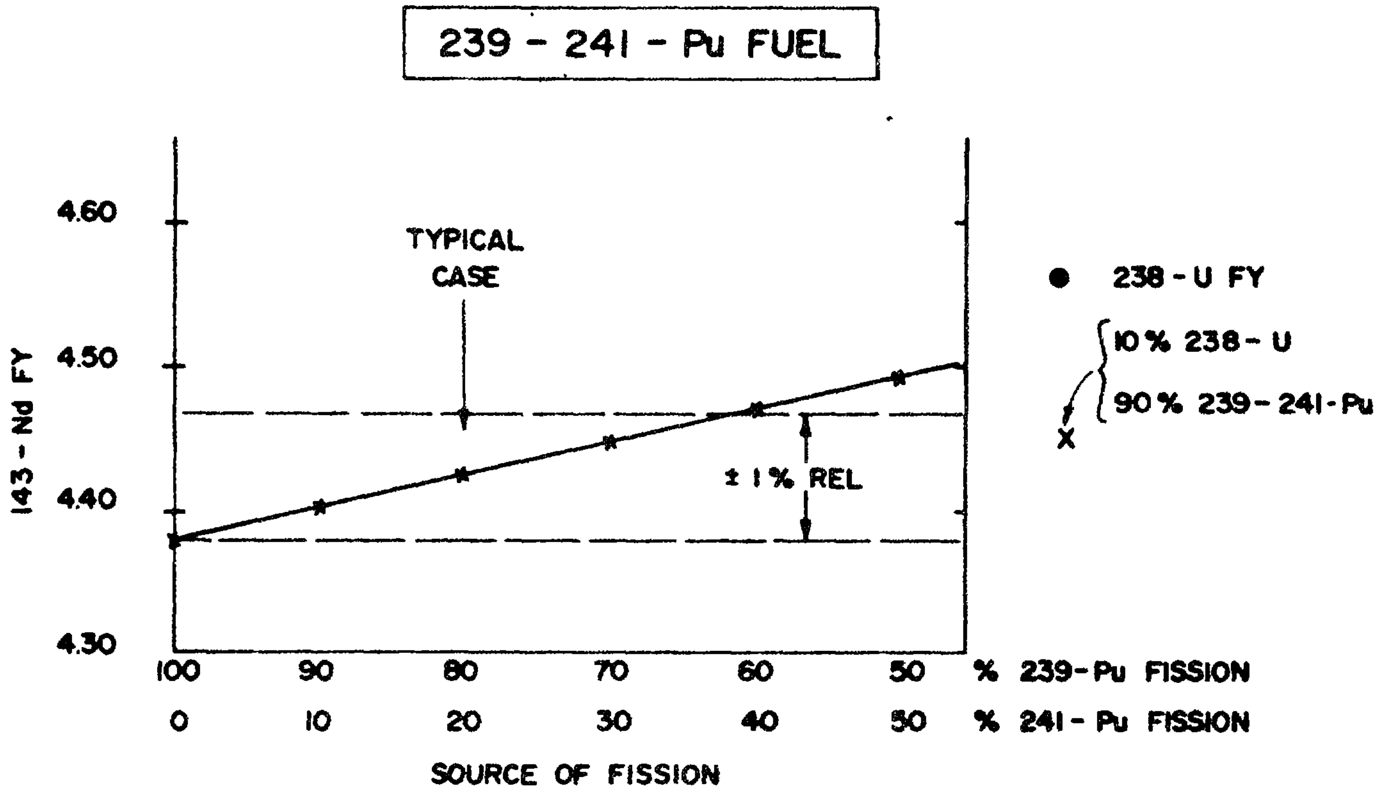

Fig. 2. Effective Fast Reactor Fission Yields for ^{143}Nd as a Function of Source of Fission

if depleted uranium is used in the fuel. For this rather simple evaluation, it has been further assumed that 10% of all of the fission could arise from ^{238}U. A recalculation of the ^{143}Nd effective fission yield for this case gives a value which still resides with the ±1% error bars. For this more likely case, the isotope ^{143}Nd and an effective fission yield of 4.43% is suggested for use in the measurement of burnup.

It is recognized that at least three other factors must be considered in the use of ^{143}Nd as a fission monitor for plutonium fuels. First, are the fission yields of ^{240}Pu and ^{242}Pu, second, is the effect of neutron energy on the ^{143}Nd yield, and third, is the neutron capture cross section of ^{143}Nd.

With regard to the fast reactor fission yields for ^{240}Pu and ^{242}Pu, preliminary yield data produced in the reviewer's laboratory, indicates that the fissioning of these two nuclides will have little or no significant effect on the effective ^{143}Nd value of 4.43%, provided that the number of fissions from these two sources does not exceed 5-7% of the total. For both nuclides the yields for ^{143}Nd appear to be slightly larger than for ^{241}Pu (^{240}Pu = ∿4.74%, and ^{242}Pu = ∿4.65%).

Thus, a firm FPND requirement for the selection of a burnup monitor for plutonium fuels is a better knowledge of the neodymium fission yields for ^{240}Pu and ^{242}Pu fast fission.

The effect of neutron energy on the ^{143}Nd fission yield for the major sources of fission in plutonium fuel remains to be determined. Based on the data available to this reviewer, the change in the ^{143}Nd yields with neutron energy for ^{239}Pu fast fission is less than that for ^{235}U fast fission; and for ^{241}Pu fast fission, less than that for ^{239}Pu. For the range of neutron energies expected in large FBR cores, it appears that the ^{143}Nd fission yield for ^{239}Pu changes ∿2%. This value must be better defined and is another request for improved FPND.

Determining the effect of neutron energy on fission yields is not a simple matter, especially when the change is in the 1-5% range.

Attempts to determine the change in yields as a function of neutron energy by comparing or plotting of reported literature yields versus neutron energy will not produce the desired results, because most reported yield data are biased as a result of systematic errors associated either with the measured number of atoms and/or the measured number of fissions. This is especially true when the change in the yields with neutron energy is small. Only by a comparison of the relative isotopic abundances, which can be determined with a high degree of reliability, can a meaningful indication of the degree of change in a fission yield with neutron energy be established. This point is most clearly shown with an illustration. In the top portion of Figure 3, literature values for ^{143}Nd fission yields for ^{235}U fast fission are plotted as a function of the $^{150}Nd/^{143}Nd$ isotopic ratio, which has been shown to be correlated with neutron energy[2]. Because of the spread in these data, it is next to impossible to make a definitive statement regarding the energy

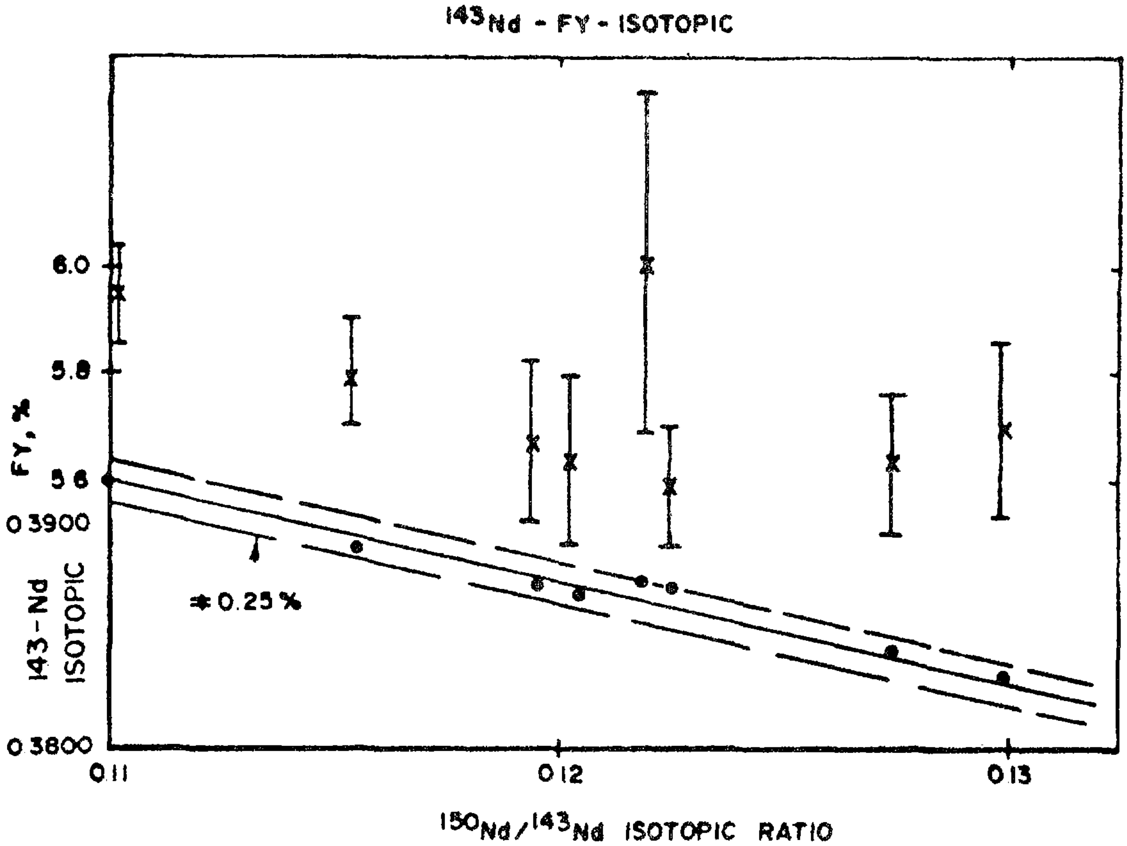

Fig. 3. ^{143}Nd Fast Reactor Fission Yield and Isotope Composition as a Function of Neutron Energy

dependency of the ^{143}Nd fission yield. Plotted in the lower portion of Figure 3, are the relative ^{143}Nd isotopic abundances for the same data. These values which are not compounded with systematic errors in the measurement of the number of ^{143}Nd atoms nor in the number of fissions, clearly show a decrease in the relative isotopic abundance of fission product ^{143}Nd with increasing neutron energy. The dashed error lines are ±0.25% relative limits about a best line fit to the data. In the reviewer's opinion, this is the preferred technique to determine if significant change in the yields as a function of neutron energy can be expected.

The major problem now remaining is the assignment of the correct fission yields to these data. The correlation of yields with neutron energy is a continuing effort in the reviewer's laboratory.

For a typical LMFBR core, the capture cross section for ^{143}Nd is estimated to be $\sim$300 mb[8]. For this value, the amount of burnout of ^{143}Nd is $\sim$1% at a burnup level of 10%. A correction of this amount is on the border of being significant. <u>Therefore, it is requested that the neutron capture cross section for ^{143}Nd in a fast reactor be better defined, preferably as a function of neutron energy</u>. An accuracy of ±20% should be acceptable.

2.4 Accuracy Requirements for Burnup Measurements

In Section 2.1 of this review, the various uses of burnup data were listed. The following is a discussion of the accuracy requirements for the various applications of burnup data and identification of the FPND needs to achieve these accuracies. In this discussion, the reference to BU-1, BU-9, etc., refer to the applications identified in Section 2.1.

2.4.1 Measurement of the Total Number of Fissions (BU-1)

It is becoming increasing evident that for many applications, burnup measurements are required with an accuracy of 1.5-2.0%. This means that the yield of the fission product monitor nuclide used to establish the number of fissions must be known to 1-1.5% relative. For thermal fission, this accuracy requirement for fission yields applies to ^{233}U, ^{235}U, ^{239}Pu, and ^{241}Pu, and for fast fission to ^{235}U, ^{239}Pu, and ^{241}Pu. Because ^{238}U, ^{240}Pu, and ^{242}Pu generally contribute less than 10% to the total number of fissions, the accuracy of the yield data for these nuclides can be relaxed to 3-5% relative.

An accuracy requirement of 1.5-2.0% for a burnup measurement represents the practical limit of request, because it requires that the fission yields be known to $\sim$1.0-1.5%. Based on years of experience in measuring fission yields, it is essentially fruitless to expect to determine fission yields to better than 1%. Even to attain this level requires a dedicated effort and a considerable expenditure of funds.

The major error component associated with the measurement of absolute fission yields is generally the uncertainty associated with the number of fissions. Techniques for the measurement of absolute fission yields, and especially the number of fissions have recently been reviewed[9] and is also a subject covered by Cunninghame in RP-10 of this conference

Of equal importance to the requirement of accurate fission yields to produce the desired accuracy for burnup data is a knowledge of the degree of change in the fission yields with neutron energy. This point was strongly stressed at the Bologna conference[2], and in the opinion of this reviewer, is still an important requirement. See Section 2.3 relative to the selection of burnup monitors.

2.4.2 Measurement of the Sources of Fission (BU-2)

A knowledge of the sources of fission in a fuel sample is important in the study of the fuel and cladding performance. For example, in the study of fission gas release and retention, which is important in fuel design, the source of the fission must be known to establish the amount of fission gas retained in the fuel.

A commonly used technique to estimate the fractional sources of fission is to measure the isotopic composition (either by mass spectrometry or gamma ray spectrometry) of selected fission products whose fission yields are different for the different fissioning nuclides. Fission products which are useful for differentiating the sources of fission are the stable isotopes of krypton (mass 83, 84, and 86), ruthenium (mass 101, 102, and 104), neodymium (mass 143, 148, and 150), samarium (mass 147, 149, 152, 154), and the radioactive isotopes, ^{106}Ru and ^{144}Ce. The use of the neodymium and samarium isotopes is primarily limited to fast reactor fuels. For example, if the neutron spectrum is reasonably well known, the $^{143}Nd/^{150}Nd$ ratio is useful in differentiating ^{235}U and ^{239}Pu fissions.

To estimate the source of fission to ±5% requires that the fission yield data be known to ±3%.

Krtil and coworkers[11,12] have conducted extensive studies, both experimental and theoretical, on low enrichment ^{235}U metal fuels, relative to the determination of the total number of fissions and the source of fissions. These studies include both gamma ray spectrometric measurements for ^{134}Cs, ^{137}Cs, ^{144}Ce, and ^{106}Ru, and mass spectrometric measurements of the neodymium isotopes.

2.4.3 Determination of the Energy Release Per Unit or Volume of Fuel (BU-3)

The same fission yield accuracy requirement, 1.0-1.5% relative, as specified for the determination of the total number of fission is desired.

2.4.4 Determination of the Fission Rate, Integrated and Terminal (BU-4)

Fission rate estimations are made from the measured total number of fissions, reactor power ratings, and calculations. The fission yield accuracy requirements expressed in 2.4.1 are more than adequate for this application of burnup data. Terminal rating calculations are based on the measurement of the shorter lived fission product nuclides such as $^{140}Ba/^{140}La$ and $^{95}Zr/^{95}Nb$. Fission yields and decay schemes for the monitor nuclides are required to ±2% relative for the thermal and fast fission for ^{235}U and ^{239}Pu.

2.4.5 Correlation of Fuel Temperature with Fuel Melting and Fission Gas Retention (BU-5)

The evaluation of the temperature of the fuel during reactor operation is an important use of burnup data. At the present time, considerable conservatism must be designed into the fuel elements to allow for gas release and fuel melting because of the uncertainties in the knowledge of the fuel temperature. For example, a 5% increase in the fission rate can result in a 30% increase in the degree of fuel pellet radial melting and a 70% increase in the amount of gas release near the melting temperature for a fuel[10]. At the present time, uncertainties in the data required to convert the fission rate into temperature are limiting. If these uncertainties can be overcome, the requirement for accurate burnup and fission yield data may become limiting. However, as has been pointed out previously, obtaining fission yields to better than ±1% is highly unlikely.

2.4.6 Decay Heat Calculations (BU-6)

For decay heat calculations, the FPND requirements vary. For fuel cooled several days, the fission yield accuracy requirements stated in 2.4.1 are sufficient. However, for short cool times, 0-10,000 seconds, the need is for independent yield data and accurate decay scheme data for the shorter lived nuclides. This subject is discussed in considerable detail by Schenter in RP-15 at this conference.

2.4.7 Fuel Storage and Processing (BU-7,8)

Knowledge of burnup is important in the calculation of shielding requirements, coolant requirements, transportation restrictions, solvent radiolysis and solvent damage. The burnup and fission yield accuracy requirements stated in Section 2.4.1 are probably more than adequate for these applications. In fact, the accuracy of existing data is probably sufficient.

2.4.8 Waste Management Studies (BU-9)

Although the FPND requirements for this area of study are not too well defined at this time, burnup and fission yield data are required for calculating the fission product inventory of the fuel processing plant waste. Existing fission yield and decay scheme data are probably adequate for this application.

2.4.9 Verification of Reactor Physics Reactor Codes (BU-10)

A variety of calculational codes have been developed and are in use for the prediction of burnup. Because the data being generated by these codes are used in a wide variety of disciplines, ranging from reactor design to safeguards, the need for an independent verification of the results is great. The only true verification of the predicted burnup is achieved by a destructive analysis for burnup on well characterized fuel specimens. The data of greatest importance are the actinide abundances, the total number of fissions, and the fractional sources of fission. The FPND requirements discussed in Section 2.4.1 directly apply.

2.4.10 Calculation of Residual Fuel Content and Reactivity (BU-11)

Burnup data provide the primary input for calculating the residual fissionable and fissile nuclide content of a fuel, and for estimating the reactivity worth of the fuel. This information is important in reactor operation and fuel development studies. The accuracy limits specified in Section 2.4.1 are adequate.

2.4.11 Contractual Agreements Relative to the Fissionable and Fissile Element Content of Spent Fuels (BU-12)

At one time, early in the development of commercial nuclear power, one of the prime uses of burnup data was in the settlement of contractual agreements. In the U.S. at least, the need for accurate burnup data was largely based on this use. As more experience in reactor operation and fuel utilization was gained, the need for highly accurate burnup analyses has diminished. These agreements are now almost entirely made and settled on the basis of reactor physics code calculations and extensive measurements of burnup on dissolved fuel solutions are seldom required. The FPND requirements for reactor physics code calculations have been previously discussed (Section 2.4.9).

2.4.12 Calibration of Non-Destructive Burnup Analysis Techniques (BU-13)

Several different non-destructive analysis techniques have been proposed, investigated, and used for the determination of burnup[2]. Of these, gamma ray scanning of irradiated fuel is the most widely used. In the strict definition of burnup, these techniques do not provide absolute burnup values, because the residual heavy atom content of the fuel is not measured. The principal advantage of non-destructive assay, especially gamma scanning is to provide a rapid technique for the measurement of relative burnup or fission.

The preferred method for obtaining reliable burnup data from the integrated number of fissions determined from gamma scanning is through an extensive calibration program. This requires a destructive analysis of many selected fuel specimens which previously have been measured by gamma scanning. The number of fissions which has occurred in the fuel sample is determined from a destructive fission product analysis, and calibration factors are established relative to the intensity of the gamma rays selected for analysis. This procedure eliminates many FPND uncertainties and the errors now start to approach the uncertainties in the fission yields used to establish the number of fissions. For this application, fission yields accurate to ±1.5-2.0% are required.

The application of non-destructive gamma-ray scanning methodology to safeguard studies is discussed in Section 4 of this review.

2.4.13 Measurement of Fission Product Spectrum Averaged Capture Cross Sections

One method of determining a fission product spectrum averaged cross section is to conduct two irradiations in the same neutron environment, one at a low fluence and the other at a high fluence. A comparison of the relative isotopic composition of a given fission product element will give an indication of the magnitude of neutron capture cross section for each isotope of the element. If the fission yields of the particular nuclide of interest are well known, a reasonable value for the capture cross section can be calculated. The yield accuracy requirements given in 2.4.1 are adequate for this purpose.

2.5 Summary

The accuracy requirement for burnup has been established to be in the range of 1.5-2.0% relative. This requires that the fission yields for the most promising fission monitors be known to 1.0-1.5%, and that only the most accurate and precise chemical measurement techniques can be used.

For thermal fission, caution must be exercised in the selection of the fission monitor. The neodymium fission yields for ^{235}U, and especially ^{239}Pu, must be reevaluated based on new data which are discrepant with the other data [4]. The neutron capture cross section for ^{147}Nd needs to be better established.

The selection of fission monitors for fast reactor fuels has been reviewed. The request for fast fission yields accurate to 1.0-1.5% requires the consideration of other factors if the uncertainty in the burnup is to be limited to 1.5–2.0%. These are, the change in the yields with neutron energy and the fast reactor neutron capture cross section of the selected monitor nuclide.

FPND requirements for burnup are listed in Table IV.

3. FPND REQUIREMENTS FOR NEUTRON DOSIMETRY

Reactor neutron dosimetry performed in research and power reactors provides information relative to neutron flux densities, fluence values, and neutron spectra. This information is considered to be of the primary type from which, secondary information can be derived relative to fission rates, burnup, damage rates, heating rates, transmutation rates, helium production, and other factors important to reactor operation and safety.

The interaction of the numerical data derived from dosimetry measurements and the end use is discussed in the following excerpt from a report on the Status of Neutron Cross Sections for Reactor Dosimetry[14]

> "The determination of flux-fluence neutron spectra is not a primary objective of reactor neutron dosimetry, but a necessary intermediate step in a more general correlation scheme between different, independent integral quantities; i.e., the damage rate in a given material exposed at a given temperature in a test reactor to the damage rate of the same material under other exposure conditions. The reaction rates and total number of reactions observed in neutron dosimeters are the basic correlation parameters in this scheme, and the flux-fluence neutron spectra are the corresponding transfer functions. The goal accuracies for determining such transfer functions are in the range of 2-5% for integral and 2-15% for differential results".

3.1 Uses of Dosimetry Data

In-core dosimetry data have a variety of uses which can be classified according to materials.

3.1.1 Fuel Material

Of primary importance here is information relative to fission rate density (i.e., number of fissions per unit volume) and its axial and radial distribution. Equally important is a knowledge of the flux density within a cluster or assembly of rods, and from cluster to cluster. By extensive flux density mapping of the entire core, the following information can be obtained:

a. macroscopic flux density pattern over the core.

b. microscopic flux density in selected regions of the core.

c. peak flux densities which can provide information relative to hot spots and possible fuel cladding failure.

These items are important to the smooth, safe, and optimum operation of a reactor.

Localized flux density measurements are used to provide information relative to radionuclide production in irradiation facilities and for fuel research and development studies.

3.1.2 Fuel Cladding

In this case, neutron dosimetry data are used to correlate damage effects in the fuel with neutron exposure. These data can be used to extrapolate the effects observed in one location to that of another if the neutron spectrum is comparable.

3.1.3 Structural Material

Included in this classification are such items as reactor reflector materials, reactor tanks, pressure vessels, structural materials, and biological shields. Of prime importance is a knowlege of how long these materials can be exposed in a given location of the reactor before radiation damage prohibits or seriously reduces the safe use of these materials in the reactor. The need for accurate dosimetry surveillance programs is of prime economic importance when one considers the cost and useful lifetime of a reactor.

Damage effects in structural material is primarily induced by neutrons with energies greater than 10 keV.

3.2 Accuracy Requirements of Dosimetry Data

The accuracy requirements for neutron dosimetry are set by the quantities of final interest (i.e., flux density, structural damage, etc.).

In 1973, at a Consultants Meeting on Nuclear Data for Reactor Neutron Dosimetry[15], the most stringent requirement was for accuracies to 5% for special cases like fuel and graphite irradiations in high temperature gas cooled reactors. However, accuracies less than this were acceptable for most other applications.

At the 1975 ASTM-Euratom Symposium on Reactor Dosimetry, the accuracy requirements were stated[16] to be in the range of 2-5% for fast breeder reactors, and somewhat less for light water reactors and controlled thermonuclear reactors.

Present state-of-the-art accuracies are estimated to be in the range of 2-30%. Most long-term reactor fuel and materials development programs will not accept an uncertainty larger than 5%. Therefore, an accuracy level of 2-5% is a reasonable goal if 5% is to be achieved on a routine basis.

3.3 Dosimetry Measurements and FPND Requirements

At the present time, multiple foil-activation is the only practical means for achieving 2-5% accuracies in fission rate measurements. This technique involves the irradiation of selected materials, which have known neutron-induced reaction thresholds, followed by a gamma-ray assay of the reaction products to determine the number of reactions produced in each monitor. From this information, the flux spectra and fluences can be deduced from codes such as SAND-II[17], SPECTRA[18], RDMN[19], and others[20].

Materials and reactions which have been used or proposed for use in dosimetry measurements are: $^{235}U(n,f)FP$, $^{239}Pu(n,f)FP$, $^{237}Np(n,f)FP$, $^{232}Th(n,f)FP$, $^{238}U(n,f)FP$, $^{238}U(n,\gamma)^{239}U$, $^{55}Mn(n,\gamma)^{56}Mn$, $^{45}Sc(n,\gamma)^{46}Sc$, $^{63}Cu(n,\gamma)^{64}Cu$, $^{59}Co(n,\gamma)^{60}Co$, $^{115}In(n,\gamma)^{116m}In$, $^{197}Au(n,\gamma)^{198}Au$, $^{27}Al(n,p)^{27}Mg$, $^{27}Al(n,\alpha)^{24}Na$, $^{56}Fe(n,p)^{56}Mn$, $^{58}Ni(n,p)^{58}Co$, $^{48}Ti(n,p)^{48}Sc$, $^{47}Ti(n,p)^{47}Sc$, and $^{46}Ti(n,p)^{46}Sc$. The measurement of reaction rates based on the use of the fissionable detectors (^{235}U, ^{238}U, ^{239}Pu, ^{237}Np, and ^{232}Th) require the analysis of the number of fission product(s) atoms contained in the sample. The fission rate is then calculated from the following relationship:

$$F_i = R_j/Y_{ij}$$

where, F_i is the fission rate (fissions per atom per second) R_j is the measured rate of production of fission product j (in units of atoms per second per fissionable atom) and Y_{ij} is the spectrum-averaged fission yield for isotope j produced in the fission of isotope i.

The most commonly measured fission products, because of their half-lives and strong gamma rays are: ^{95}Zr, ^{97}Zr, ^{103}Ru, ^{131}I, ^{132}Te, ^{137}Cs, ^{140}Ba, ^{143}Ce, and ^{144}Ce. In addition, the stable isotope, ^{148}Nd, is frequently measured but this requires a destructive analysis of the sample.

The primary FPND requirements are for fission yields, decay schemes, and half-lives. Fission yields accurate to ∿2% are required for the fast breeder reactor programs, while accuracies ranging up to ∿10% are adequate for other reactor programs. Because one of the uses of dosimetry data is to determine neutron spectra, a knowledge of the neutron energy effect on the fission yields is imperative. For CTR application, fission yields for neutron energies up to 20 MeV may be required.

The accuracy requirement for decay scheme data, in terms of gammas per disintegration is ±1% for the major gamma rays. In 1975, Helmer and Greenwood[21] showed that the uncertainties in the branching ratios for ^{103}Ru, ^{132}Te, and ^{144}Ce were too large to meet a goal of ±2.5% uncertainty in the fission rate. Improved decay scheme data are needed for these three nuclides.

The accuracy requirements for the half-life values cannot be simply stated, because several factors enter into the sensitivity calculation. Factors such as the counting time, irradiation time, and out of reactor time relative to the half-life must be considered. Helmer and Greenwood[21] have reviewed this problem in considerable detail.

3.4 Summary

The most significant FPND requirements relate to fission yields and decay schemes. The effect of neutron energy on the yields must be determined. Table V lists the various requirements.

4. FPND REQUIREMENTS FOR SAFEGUARDS

4.1 The Use of FPND in Safeguards

The importance and the accuracy requirements of FPND for nuclear material safeguards are not nearly as well defined as they are in the related fields of reactor physics and burnup. This primarily results from the fact that most safeguard programs are still in the planning and development stages and that the specific FPND requirements for safeguards are not well defined. Before the definitive needs for FPND can be identified, the following questions need to be answered.

A. Is there any use or dependence on FPND in the safeguards effort?

B. How can more accurate FPND benefit the safeguarding of virgin fuel material, scrap material, unirradiated fuel, spent fuel, dissolver solutions, recovered fuel, and waste? More specifically, what improvement in nuclear data will result in a decrease in the material unaccounted for, or MUF; which FPND are usable to reduce the potential for diversion of special nuclear material; what improvement in FPND is required for verification of nuclear material accountability measurement?

C. Which are the exact data of interest?

D. How sensitive is a measurement technique to the uncertainties in the existing FPND data?

The answer to the first question is always a resounding 'yes'. However, the answer to the other question are quite nebulous. Until these questions are answered, and a firm plan to safeguard special nuclear material is implemented, it is difficult to identify specific and urgent needs for improved FPND for safeguards.

It is recognized that safeguards, as opposed to the other nuclear disciplines discussed in this review, is a relatively new field and in the process of rapid and diversified development. Therefore, it can be expected that the requests for improved FPND will rapidly change, depending upon the then current measurement or calculational method being investigated. In 1974, Weitkamp[22] reviewed this problem rather extensively and posed several interesting questions relative to the need for better FPND for safeguards. Many of his comments and questions are still relevant today.

It appears to this reviewer that the need for improved FPND in safeguards is highly dependent upon the method of safeguards to be employed. If the basic techniques are to be physical constraint, locked doors, automated plants, by-difference accounting, tag inventory, or digital accounting rather than measurement, then there is no need for better or additional FPND. If measurements are made to determine the change in the material balance as a result of decay, burnup, or burnin, then some form of FPND will be required. These data may be useful in the actual measurement, and/or in the data reduction process. In certain cases, the measurement of selected fission products may provide the means for an independent verification of other measurements. However before a high priority request for improved FPND can be placed before this conference, the specific need must be identified.

4.2 Safeguards Measurement Programs

Development programs employing various measurement and computational techniques, which depend on the use of FPND are being conducted in several areas. The three most important and the methodology being investigated are the following:

A. Analysis of fresh fuel during the enrichment, chemical processing, fabrication, and reprocessing stages of the nuclear fuel cycle. The preferred analysis techniques are those based on nondestructive analysis (NDA). Currently under investigation are methods based on gamma-ray resonance fluorescence, neutron capture gamma-ray spectrometry, neutron activation analysis, passive neutron counting techniques, passive gamma-ray assay, and calorimetry. Although not strictly a NDA technique, alpha spectrometry also is being investigated. Details relative to these techniques, their potential use in safeguards, and the nuclear data required for these various methods are given in references [22,23, and 24].

B. Spent fuel analysis - The NDA of spent fuel for safeguards is of less importance than the analysis of fresh and recovered fuel. In fact, NDA of spent fuel is of more importance to reactor operation and design, fuel reprocessing, and waste management. Of the several NDA methods (neutron activation, gamma and resonance neutron transmission, calorimetry, and gamma-ray spectrometry) which have been proposed for the analysis of spent fuel, only high resolution gamma-ray spectrometry shows promise for field use.

C. Isotope correlations - This method is primarily based on empirical and computational data which show that certain fission product and heavy element ratios can be correlated with burnup, fission rates, fluence, cooling times, Pu-to-U fission ratios, and initial and final fuel enrichment. The methods for obtaining the desired ratios included both destructive and non-destructive analysis. Because most of the correlations are empirical, there is little need for FPND. The need for FPND arises when the correlations are used to verify other measurements through calculational procedures.

The most extensive isotope correlation studies are being conducted in the Euratom community, primarily at Karlsruhe and Ispra. This work includes sensitivity studies and the building of an isotope correlation data base. This latter effort involves correlating the fission product and heavy element isotopic data with known reactor operating history. This has required an extensive measurement program and the collection of much detailed reactor history. While this data base may be useful for safeguards, if sufficient reactor operating history is known, its usefulness in deducing reactor history is still questionable.

Lammer[25], in an extensive contribution to this review, evaluated the use of selected activity ratios for the deduction of cooling time, irradiation time, fluence, flux, source of fission, fission rates, breeding rates, and initial enrichments. In this study, he considered such factors as reactor type, initial enrichment, flux, irradiation time, and the neutron spectrum. While in many cases, the use of selected fission product activity ratios appeared to be useful in deducing certain parameters, most were subject to the variations in reactor operation. For example, the use of the $^{144}Ce/^{137}Cs$ activity ratio for deducing cooling time was shown to be dependent upon the irradiation time. In this contribution, Lammer identified several improvements in FPND accuracies which would be of value in future studies.

Returning to the subject of instrument and methods development for safeguards, one item stands out, especially with respect to NDA techniques. That is, in the transfer of technology from the laboratory to the field, the uncertainties associated with the mechanical operation of the instrument dominate, and render insignificant the uncertainties associated with the FPND now available. This is not meant to imply that FPND are not important, but rather, that the current needs are for improved technology. Once this has been accomplished, then the FPND uncertainties may become limiting and specific needs for improved FPND can be identified.

The fact that FPND currently do not appear to be a limiting factor in developing technology for safeguards does not preclude the need for data improvements. It is just that the same priority cannot be assigned to the requests compared to those associated with burnup, neutron dosimetry, and reactor physics. Because the three fields of study covered in this report are so interrelated, it is becoming increasingly difficult to separate the requirements for improved FPND, and it is quite possible that those for safeguards will be achieved through the others.

At the present time, most NDA safeguard instrumentation is standardized based on calibration procedures and destructive analysis rather than the use of nuclear data. While this may be an expensive and tedious procedure, I believe it is a more correct procedure, because in most methods there are just too many systematic errors which cannot be determined.

It is quite probable that the greatest safeguard improvements will come through more accurate destructive analysis methods, improved certified U and Pu isotopic and chemical standards, interlaboratory comparison studies, and better volume measurements.

4.3 Summary

The need for improved FPND for safeguards has been reviewed. Although many of the requests cannot be assigned a high priority because of questionable benefit to the safeguards effort, some work in this area is certainly justified. Listed in Table VI are the request for improved FPND and the accuracy requirements submitted to this reviewer.

TABLE IV

SUMMARY - FPND REQUIREMENTS FOR DETERMINATION OF BURNUP

FPND REQUIREMENT		STATUS	NEEDS	
I. Fission Yields				
A. Thermal Reactors				
1. General	^{233}U	~±2% for stable and long-lived nuclides	May require 1.0-1.5% accuracy for ^{233}U-Th reactors	
	^{235}U	Requires reevaluation, especially of Nd isotopes	1.0-1.5%, major stable and long-lived nuclides	
	^{239}Pu	Significant discrepancies reported, especially for Nd and Xe isotopes	1.0-1.5%, major stable and long-lived nuclides	
	^{241}Pu	2-3%	Probably adequate	
2. Specific				
^{134}Cs independent yield		Poor	5-10%	
^{103}Ru, ^{106}Ru and stable Ru isotopes		Ru data poor in comparison to other yields	±2%	
^{137}Cs, ^{144}Ce, ^{95}Zr		±2% for end member of chain	Probably adequate	
B. Fast Reactors				
1. General	^{232}Th	Poor	3-5%; may increase to 1.5%	
	^{233}U	~2% for typical LMFBR spectrum	1-2%	For all cases, the effect of neutron energy needs to be established.
	^{235}U	~2% for typical LMFBR spectrum	1-2%	
	^{238}U	~3% for typical LMFBR spectrum	3-5%	
	^{239}Pu	1.0-1.5% for typical LMFBR spectrum[6]	1-2%	
	^{240}Pu*	Poor; however see ref. [13].	3-5%	

TABLE IV (Cont'd)

FPND REQUIREMENT	STATUS	NEEDS
B. Fast Reactors		
^{241}Pu	1.0–1.5% for typical LMFBR spectrum[6]	1–2%
^{242}Pu*	Poor; however see Ref. [13].	3–5%
^{237}Np	∿1.5% for typical LMFBR spectrum[7]	2%
^{241}Am	Not available	5–10%

*Work recently completed in reviewer's laboratory indicates error on ^{240}Pu yields will be ∿3% and for ^{242}Pu ∿2%. See Ref. [13].

FPND REQUIREMENT	STATUS	NEEDS
2. Specific: Of prime importance are Nd yields	1–1.5% for typical LMFBR spectrum. Effect of neutron energy needs better definition.	1–2% well characterized spectrum over range of thermal to 1–2 MeV.
II. Neutron Capture Cross Sections		
A. Fission Products		
1. Thermal Neutrons		
^{147}Nd	±30%, one measurement.	±10%, thermal and resonance
^{133}Cs, ^{134}Cs, ^{141}Pr, ^{143}Nd, ^{145}Nd, ^{153}Sm, ^{153}Eu, ^{154}Eu.	Variable	3–5% thermal, if greater than 20 b. 10% resonance, if greater than 50 b.
2. Fast Neutrons		
All isotopes of Nd	Variable or not known	±10% if greater than 100 mb, differential data preferred.
Major stable and long-lived fission products	Variable or not known	±15% if greater than 100 mb.
III. Decay Schemes		
^{95}Zr-Nb, ^{103}Ru, ^{106}Ru, ^{137}Cs, ^{140}Ba-La, ^{141}Ce, ^{144}Ce-Pr, ^{147}Nd, ^{154}Eu, ^{155}Eu.	Significant improvement since Bologna conference.	1% relative uncertainty for major (>10% rel I) γ-rays. Prefer results in form of γ/dis., 1% rel. uncertainty.

TABLE V

FPND REQUIREMENTS FOR NEUTRON DOSIMETRY

FPND REQUIREMENT	STATUS	NEEDS
1. Fission Yields		
^{95}Zr, ^{97}Zr, ^{103}Ru, ^{132}Te, ^{137}Cs, ^{140}Ba, ^{143}Ce, ^{144}Ce, ^{148}Nd	Of these the least well-known is ^{103}Ru.	Require ^{103}Ru yields to 1.5% for all fissionable isotopes.
for		
^{235}U	∿2%	Primary need is for information relative to neutron energy effect ∿2%
^{238}U	3-5%	
^{239}Pu	1-2%	
^{237}Np	∿1.5%	
^{232}Th	Poor	
2. Decay Schemes		
^{95}Zr	adequate	
^{97}Zr	adequate	
^{103}Ru	inadequate	1%
^{132}Te	inadequate	1%
^{137}Cs	adequate	
^{140}Ba	adequate	
^{143}Ce	adequate	
^{144}Ce	inadequate	1%
3. Half-Life		
^{95}Zr, ^{97}Zr, ^{103}Ru, ^{132}Te, ^{137}Cs, ^{140}Ba, ^{143}Ce, ^{144}Ce.	Depends on specific case	Generally better than 1%.

TABLE VI

FPND REQUIREMENTS FOR SAFEGUARDS

FPND REQUIREMENT	DESIRED ACCURACY
1. Fission Yields	
Thermal	
^{95}Zr - ^{239}Pu	1.5%
^{103}Ru - ^{235}U, ^{239}Pu, ^{241}Pu	2%
$^{153,4,5,6}Eu$ - ^{235}U, ^{239}Pu	2%
A=103-115 - ^{235}U, ^{239}Pu	2%
Fast	
^{103}Ru - ^{235}U, ^{239}Pu, ^{241}Pu, ^{238}U	2%
Independent	
^{134}Cs - ^{235}U, ^{239}Pu, ^{241}Pu-thermal	3%
- ^{235}U, ^{239}Pu, ^{241}Pu, ^{238}U-fast	3%
2. Half-Lives	
^{154}Ru	1%
3. Gamma-Ray Intensities	
^{154}Eu	1%
^{156}Eu	1%
^{134}Cs	1%
4. Neutron Capture Cross Sections	
^{153}Eu	2% pile
^{154}Eu	2% pile
^{155}Eu	2% pile
^{156}Eu	±300 b

TABLE VI (Cont'd)

FPND REQUIREMENTS FOR SAFEGUARDS

NON-FPND REQUIREMENTS	DESIRED ACCURACY
1. Half-Life	
Pu isotopes	0.1%
2. Mass Adsorption Coefficients	
UO_2, PuO_2, mixed oxide pellets	
Al, Zr, stainless steel	

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Review paper 7

STATUS OF NEUTRON REACTION CROSS SECTIONS OF FISSION PRODUCTS IN THE ENERGY RANGE OF RESOLVED AND UNRESOLVED RESONANCES

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ABSTRACT

The purpose of this paper is to summarize the present "State of Art" concerning the data on fission products cross sections in the energy range varying from thermal energy to about 100 Kev.

The advantages and inconveniences of the various experimental methods and evaluation techniques have been examined in detail. As a result some methods are considered more suitable than others. In the case of sever disagreements between results, some solutions have been proposed.

It would appear from this review that the needs of reactor physicists concerning informations on the most important fission products in thermal and fast reactors are, in general, largely satisfied.

CONTENTS

IV - Formalisms

IV.1 - Resolved resonance region

IV.2 - Unresolved resonance region

V - Mean parameters determination

V.1 - Average resonance spacing determination

V.2 - Determination of mean radiative widths

VI - Local systematics

VI.1 - Small "a" systematics

VI.2 - Neutron strength function systematics

VI.3 - Radiative widths or radiative strenght function systematics

VII - Uncertainties estimates

C INTERCOMPARISON OF CAPTURE CROSS SECTIONS FROM RECENT EVALUATIONS OF SOME MOST IMPORTANT FISSION PRODUCTS. RECOMMENDED PARAMETERS.

CONCLUDING REMARKS

- INTRODUCTION -

The purpose of this paper is to review the status of fission products evaluations in the energy range E_{th} - 100 Kev and to examine whether the available evaluations meet the requested accuracies.

If we refer to WRENDA list 76-77 or to informations available from the litterature (Tyror, Ribon,...) the order of the FP in the lists of importance and the requested accuracies differ from one author to an other.

In this review paper we will center our interest on the 22-25 isotopes which are the most important for both thermal and fast reactors according to all the requestors. We shall also presume that the requested accuracy on the capture cross section is ±10% (standard deviation) for all these nuclides, which is a reasonable mean value.

The reasons for this apparently arbitrary choice are as follows :

The importance of fission products depends on the application foreseen but the main justification for data is their contribution to the reactivity change with burn-up which depends on various factors : nature of the fuel and fuel cycle, decay periods and cross sections. In fact, the capture cross sections appear to be the most important data and they contribute by about 60% to the total reactivity loss, according to LANGLET et al [LA 77] ; in commercial fast reactor in the 1000 MWe range. The order in the list of importance depends on the reference library, and thus, a relative character has to be attached to it.
The situation is similar for the requested accuracies:

If the effects of the separated FP can be grouped in a lumped FP the calculated uncertainty on each factor (cross-sections, Yields,...) allows to calculate the uncertainty on the global effect. But in a reciprocal way if the uncertainty on this global effect is imposed, the requested accuracy on each parameter can be fixed only by using some assumptions. In fact it is difficult to distinguish for a same factor the contribution due to random errors and that which can be reduced by improvement, due to systematic errors. And for what concerns more precisely the cross sections, the uncertainties on the various FP must be distributed according to a law which has to be defined. A law of equal distribution is interesting since it does not refer to any cross section library. That is precisely what we have considered . An accuracy of ±10% on the cross section corresponds to an accuracy of 0,2% on the reactivity effect $\frac{\Delta K}{K}$ for fast reactors in the 1200 MWe power range and for cycles of the order of a year.

A. TECHNIQUES AND EXPERIMENTAL METHODS REVIEW

We will divide the energy range E_{th} - 100 Kev into three parts :

I. Thermal and Epithermal Energy
II. Resolved Resonance Region
III. Unresolved Resonances Region

It may seem excessive to classify experimental methods in function of energy since same methods can be applied in different ranges. However, often, the application conditions are characteristic of the energy range of interest.

I. Thermal energy : Maxwellian averaged cross sections. Capture resonance integral.

The two methods the most utilized for measuring the absorption cross sections averaged in a thermal flux and the cross section at 2200m/sec are : the activation method and the in pile oscillation method. The first is more directly interpretable. The various corrections make the second one less accurate.

I.1.1 . Activation method

The neutron capture rate of a sample in a thermal flux is given by

$$A = n\, v_0 \times F\ (T)$$

where A is the capture per atom, $n\ v_0$ the conventional definition of the neutron flux, F (T) is a function of the thermal neutron temperature and of r the epithermal neutron flux index.

If the epithermal flux is proportional to 1/E, Westcott's definition must be applied.

$$F(T) = G_{th} \times g(T) + G_r \times r \times s(T)$$

In the above, $g(T)$ and $s(T)$ are tabulated functions. G_{th} and G_r are thermal and resonance self-shielding factors.

According to Ryves (Ry 69), the above situation is theoretical and it would be better to express the epithermal flux/unit energy by $\Phi(E) = \frac{A}{E^{1+\beta}}$

In this new convention, r is independent of the target sample and of the neutron flux shape. The β constant can be extracted from the ratio of activities per atom of two Au and Mn samples of standard thickness (0.0025 cm) in the same thermal flux. $\Omega\left(^{M}n/A_u\right) = 0.0214 - 0.051\ \beta + 0.13\ \beta^2$. In such a case, a new F (T) function must be used.

$$F(T) = G_{th} \times g(T) + r \times C\ (\beta,T) \times G_r \times h\ (\beta,G_r \times h\ (\beta,G_r) \times {}^{I'}/\sigma_0$$

In this relation, C (β,T) is a renormalization constant of r at temperature T. $h(\beta,G_r)$ is a correction to the resonance integral defined for a 1/E slowing down flux. σ_0 and I' are respectively the thermal cross section and the reduced integral resonance (less the 1/v part).

The measurement can be absolute or relative. The activitu of the sample is compared either to the activity of a similar sample in a case of Cd or to the activity of an Au sample.

The accuracy is between 0.5 % and 1 % for σ_0, between 1.7 % and 3 % for I' (without ever exceeding 7 %).

In general, the values found in the literature for σ_0 are in agreement (Ry 69, Ry 71, Ry 74). Important disagreements (up to a factor of 2 in the extreme of ^{81}Br) exist for I'. Ryves attributes the differences to the approximation of the epithermal flux to a 1/E law leading to an uncorrected perturbation which can reach 30 %, especially for isotopes with many large resonances.

1.1.2. On pile oscillation method

This method, which gives accurate raw data, has the advantage of being independent of the time of life of the produced nucleus.

Unfortunately, thick samples are required for which self-shielding corrections are important, and large signals due to possible predominant scattering resonances are difficult to separate from the capture. In addition, in thermal reactors, the epithermal neutron flux is distorted because of 238U fuel (large resonance at low energy) and moderator geometry.

Other methods exist : e.g scattering cross section substraction from total cross section capture.

II. Resolved resonance region

The aim of the experiments in this energy range is a precise description of the cross section as a function of the energy. It can be obtained only by a T.O.F method by Linac.

The experimental techniques do not differ from those used at higher energy (transmission, prompt γ detection).

The following experimental effects make it difficult for resonance parameters to be extracted from experimental raw data :

- Doppler broadening, generally treated as a Gaussian function.
- Broadening due to the experimental resolution.
- Self-absorption and multiple scattering effects which are calculated by analytical and Monte Carlo techniques respectively.
- Overlapping and interference effects which are taken into account by utilizing a MLBW formalism.

Two kinds of methods are used in order to solve these problems :

II. 1. Area analysis.

This method can be applied to all types of cross sections. The total area of the resonance is used as a basic data. This area is expressed in a reciprocal way, in terms of the partial widths of the resonance and of the sample thickness. If the latter is changed, a system of curves F $(\Gamma_n, \Gamma_\gamma)$ is obtained whose convergence area determines the Γ_n and Γ_γ widths with more or less accurate results.

The advantage of this method is that the resonance area is not sensitive to the Doppler and resolution effects.

The disadvantages are : it is difficult to apply this method in the case of non isolated resonances, and to detect background errors. Furthermore, the method is time consuming (several samples).

II. 2. Shape analysis

It is applied to total and reaction cross sections.

Provided that the samples used are thin enough so that the multiple scattering corrections can be neglected, the principle is to formulate a function such as :

$$\sigma_{eff}(E) = \sigma(E) \times D(E) * R(E)$$

In this formula, $\sigma(E)$ is the cross section calculated by a formalism which can be sophisticated (multichannel - multilevel, Reich-Moore, Adler-Adler, Vogt formalisms). $D(E)$ and $R(E)$ are respectively the Doppler broadening and resolution functions. Then, the function $\sigma_{eff}(E)$ is adjusted on the experimental points using a least squares method. In such a way, the resonance parameter can be obtained.

Inconveniences : the method needs thin samples.

Appli ation requirement : $D(E)$ and $R(E)$ functions have to be accurately known. Doppler and resolution widths must be small in relation to the resonance total width.

Advantages:

The method is fast and allows the detection of systematic errors (normalization, background). Applying this procedure to raw data from various laboratories (Geel-Saclay-Columbia), l'Heriteau, Derrien, Ribon (LH 71, DE 75, DE 75 a) have shown that the resonance parameters for 238 U and 232 Th extracted from total cross section measurements are in fact, coherent, in oposition to the results obtained by area analysis.

III. Unresolved Resonance Region

To measure capture cross sections, the methods are numerous. For some of them, different techniques can be employed.

III.1 . Spherical shell transmission method

This method was proposed by Bethe and al(Be 56). It consists of placing an isotropic neutron source in the center of a spherical shell and of measuring the transmitted flux. If the neutron source is not isotropic, the detector and source are interchanged. The latter is placed at a great distance.

Advantages :

- No sophisticated electronic setup necessary. Absolute values for cross section result without having to know the neutron flux.

Disadvantages :

- Important quantities of material needed, which restricts the field of application.

- The multiple scattering and self-shielding corrections are difficult to calculate. They require an exact knowledge of total and elastic scattering cross sections.

- Due to systematic errors, this method has resulted in overestimated values. Therefore, it has been rarely utilized during the last 15 years (Be 58, Le 58).

Accuracy on cross section was : 10 % to 12 %. Accuracy on energy : that of the neutron source.

III. 2 . Detection of prompt gamma rays following neutron capture

This method necessitates detectors whose efficiency is independent of the gamma ray cascade mode. This independance is obtained by making the efficiency close to 100 % (large detectors) or proportional to the total energy (Moxon Rae or Maier-Leibnitz type).

The large detector technique permits a good time resolution, but the electronic background noise is important due to the detector size. Therefore, a high electronic threshold leads to a non negligible efficiency correction which can be subject to a large margin of uncertainty.

This technique is designed for nuclides which emit a large total gamma energy (Gi 61, Ma 63, Ko 69, Fr 70).

The accuracy on cross section varies between 3 % and 4 % for energies lower than 1 Kev (white neutron sources), between 5 % and 7 % in the MeV region. The energy resolution depends on the neutron source.

The two other techniques, very similar (Moxon-RAE, Maier-Leibnitz (with a weighting function applied to each pulse)) employ detectors of small volume.

However, their efficiencies are different : a few percents for the first, 15 to 25 % for the second. The applicability is limited to energies lower than the inelastic threshold. The good timing allows experiments where the detector is placed close to the neutron source (Mo 63, LR 75).

The neutron flux is measured by a calibrated detector or by counting the γ emitted by a "black" Boron sample ($^{10}B\ (n, \alpha_\gamma)\ ^7Li$ reaction at low energy).

An efficiency correction must be applied because of the liquid scintillator activation by the sample scattered neutrons.

Accuracy on cross section : 4 to 6 %. Accuracy on energy : 0.3 Kev at 100 Kev.

Finally, reference is to be made to slowing down spectrometers. They use the neutron slowing down by elastic and inelastic processes on a heavy element. Their main advantage is a non-sophiticated setup. In addition, an important neutron flux can be obtained. The obtained results must be normalized on thermal values or on known resonance areas. (Ka 63, Po 62, Ko 67, Ch 73).

Some corrections are peculiar to the method :

- elastic scattering of the neutrons by the detector on the sample
- poor energy resolution in the upper energy range
- error in the γ background because of neutron flux distortion.

According to Moxon (Mo73), this technique could underestimate the capture cross section in the vicinity of the large scattering resonances.

Accuracy on cross section : ± 10%
Accuracy on energy : ± 20%

Of all the procedures using prompt detection, that of Maier-Leibnitz seems the most interesting.

III.3. Activation Method

This technique is based on the counting of the beta or gamma emission characteristic of the radioactive nuclide.

Two types are possible :

III.3.1 An absolute technique in which the incident neutron flux and the beta or gamma induced activity are both measured by calibrated detectors. The calibration for the beta or gamma counter is facilitated by an adequate decay scheme (beta-gamma coincidence).

$$\sigma_c(E) = k \frac{\varepsilon_n \ (E)}{\varepsilon_\beta \ (E_\beta)}$$

Actually, this procedure has rarely been applied on fission products (Ma 57).

III.3.2 A "relative" technique with two versions

a. The activity at the energy of interest is compared to that of a reference energy (in general, thermal energy).

There are then two possibilities :

1) A reference neutron detector is used. In this case, a relative measurement is sufficient since the neutron flux ratio is used. If the neutron flux is measured by fission chambers, with ^{235}U deposits for example, the flux ratio is close to the cross section ratio [Jo 59]

$$\sigma_c(E) = k \ \sigma_c \ (E_{th}) \ x \ \frac{\varepsilon_n \ (E)}{\varepsilon_n \ (E_{th})} = k' \ \sigma_c(E_{th}) \ x \ \frac{\sigma_{n1f}(E)}{\sigma_{n1f}(E_{th})}$$

2) The activity is referred to that of a standard at two different energies. That is, the so-called "double comparison" method. (Bo58, Le58).

$$\sigma_c(E) = k\,\sigma_c(E_{th}) \times \frac{\sigma_{ref}(E)}{\sigma_{ref}(E_{th})}$$

A knowledge of the beta or gamma detection efficiency is not needed for techniques 1) and 2).

b. The activity is referred to that of a standard at the energy of interest. The variation of the beta or gamma efficiency versus the energy must be known (Ma 57, Po 65, Ch 66)

$$\sigma_c(E) = k\,\sigma_c^{ref}(E) \times \frac{\varepsilon_\beta(E_\beta^{ref})}{\varepsilon_\beta(E_\beta)}$$

Accuracy on cross section : 3 % - 6 %

Accuracy on energy : that of the neutron source

The activation methods are accurate. Although requiring meticulous handling, they do not impose a complex electronic set-up.

The disadvantages are :

- Sensitivity to scattered background neutrons which must be imperatively eliminated.
- Important neutron flux needed so that the obtained cross section values correspond to a large energy spectrum.

Conclusions from experimental data

Most of the experimental data are outdated and were obtained by relative activation methods. There is an important disagreement regarding not only the shape but also the magnitude. This situation has been only slightly improved by a renormalization based on the recent evaluations of fission and capture standards (So 74, Fo 76).

The recent data are in better agreement. It is to be noted, however, that the most important isotopes have stimulated an unequal interest which is partially explainable by the difficulty of obtaining samples.

B. Evaluation Techniques Review

Most of the various authors use the same procedure.

The starting point is the choice or eventually the calculation of a set of resonance parameters.

1. Resonance Parameters

1.1. Nuclei with experimental data

There are two extreme possibilities for the choice.

- to average the available experimental data of the parameters of each resonance.
- to choose one single set of parameters.

We recommend the second solution, the other data being used as a check or a guide for modification.

In the choice of a set of parameters, the following criteria should be observed :

i) Shape analysis of raw data instead of area analysis as much as possible

ii) A parameter set as complete as possible

iii) The most recent data with an equal condition of accuracy

iv) The data which are not suspicious because of possible systematic errors.

Usually, the published data have to be completed for what concerns ℓ orbital momentum assignment. It seems worthwhile to utilize successively the two following rules :

1) The resonances at energies E_λ such as $g\Gamma n(E_\lambda) > 10\ \overline{g\Gamma n(E_\lambda)}^{\,\ell=1}$ must be classified as sure "s" wave resonances (Ri 75).

2) The "p" wave resonances, extracted from the remaining resonances, can be grouped, making use of Baye's Criterion (based on the size of the resonance) [Sc73]

$$P_p\,(g\Gamma n) = \left\{ 1 + \frac{\pi_s}{\pi_p} \sqrt{\frac{\overline{g\Gamma n}^{\,\ell=1}\,(E_\lambda)}{\overline{g\Gamma n}^{\,\ell=o}\,(E_\lambda)}}\ \exp\left[\frac{g\Gamma n}{2}\left(\frac{1}{\overline{g\Gamma n}^{\,\ell=1}(E_\lambda)} - \frac{1}{\overline{g\Gamma n}^{\,\ell=o}\,(E_\lambda)}\right)\right]\right\}^{-1} > 0.5$$

It must be noted that :

- This argument does not work in the case of "s" and "p" neutron width distributions which overlap.

- The quantity $P_p(g\Gamma n)$ is very sensitive to $D^{\ell=o}$.

1.2. Nuclei without experimental data (^{103}Ru, ^{107}Pd, ^{135}Cs)

In the cooperation between CNEN and CEA, the following method was used :

The resonance energies and the Γ_n were generated at random according to Wigner and Porter-Thomas distributions with averaged parameters extracted from systematics. The number of generated resonances depended on the magnitude of $D^{\ell=o}$ (between 10 and 50). The thermal cross section and resonance integral adopted values were reproduced (within the error bars) either by generating several sets of resonance parameters or by modifying one generated set.

Not all the authors use this method. The Japanese team, for example, assumes a $1/v$ behaviour for the cross section between thermal energy and the lower limit of the statistical region. This procedure is valid if the self-shielding is small.

1.3. Determination of the connecting energy between statistical and resolved resonance regions. Missing resonance contribution

In order to determine the connecting energy E_c, two solutions are proposed :

- In the first, E_c is calculed by $E_c = E_{max} + \frac{D^{\ell=0}}{2}$ where E_{max} is the energy of the last observed or generated resonance and a "background" cross section is added, representing the contribution of the missing resonances.

- The second solution is :

--In the case of no experimental data : $E_c = \frac{D^{\ell=0}}{2}$

--When these data exist two possibilities must be considered :

a) Few resonances are observed $E_c = E_{max} + \frac{D^{\ell=0}}{2}$

b) Many resonances are observed :

Thus, the connecting energy is determined as the energy above which resonances are obviously missing (utilization of a statistical criterion).

Regarding this method, some comments may be made :

. The position of E_c can be sensitive to the quality of the statistical model parameters. An extrapolation to low energy is possible only if the parameters of the optical model yield a good interpretation of the "s" wave neutron strength function.

A lowering of the connecting energy corresponds to a loss of information needed for self-shielding calculation.

The background cross sections are a statistical estimate of the contribution due to the missing resonances. For the elastic scattering process, this residual contribution can be neglected.

The estimate for the radiative capture process is made by comparing the values of cross sections, in some energy groups, calculated from the resonance parameters, with those given by an extrapolation of the statistical model. But it is difficult to distinguish between statistical fluctuations and systematic loss of resonances.

A solution useful for the reactor physicist would consist of generating the missing levels according to the distribution of the observed resonances. As far as we know, this has not been employed.

II. Thermal Cross Sections

II.1. Capture cross sections

The cross section calculated from the resonance parameters is always lower or equal to the value recommended by specialists. One or more negative resonances are added with parameters $g\Gamma n = \overline{g\Gamma n}$ and $\Gamma_\gamma = \overline{\Gamma_\gamma}$ so as to reproduce the thermal cross section. These supplementary resonances may be considered as bound levels. Their contribution can be described by an $\sqrt{\frac{0.0253}{E}}$ law or an $\sqrt{\frac{0.0253}{E}}\left(\frac{0.0253 - E_\lambda}{E - E_\lambda}\right)^2$ according as the capture cross section has a $1/v$ behaviour or not.

II.2. Elastic scattering cross section

In this process, the cross section depends on the potential scattering radius R'. The values generally adopted for this parameter result from a systematics, combining precise experimental data and optical model calculations (OM of Chase [Ch 58] , OM of Moldauer [Mo 53]).

III. Capture Resonance Integral RI_c

Walker's formula is generally adopted for capture resonance integral calculation for resonances above few ev.

Some problems can appear.

In addition to the nuclides without experimental data (see C.I.2), they affect those for which the negative resonances are close to 0 energy and those for which too few levels have been determined.

III.1 - Nuclei with negative resonances close to 0 energy (^{149}Sm, ^{151}Sm)

The value of RI_c strongly depends on this resonance energy. In this case, the RI_c recommended experimental value is a check of the resonance parameters.

III.2. Nuclei with few resonance parameters (^{108}Pd)

If the experimental value of RI_c is accurately known, the resonance parameters can be adjusted.

IV. Formalisms

IV.1. Resolved resonance region

SLBW and MLBW formalisms are used in the resolved resonance region. Sometimes, some negative peaks appear in the elastic scattering cross sections. They can be due not only to the interference between resonances which is not taken into account in the SLBW formalism, but also to the dissymmetry of the resonance number on both sides of the energy of interest. That is why according to RIBON (Ri 76), a corrective term has to be added to the MLBW formalism which, for "s" resonances takes the form (ANNEX I)

$$\sigma_{n,n}^{dys} = 4 \pi \lambda\!\!\!^{-} R' \sqrt{E} \times S^0 \times \log \left| \frac{E - E_s}{E - E_i} \right|$$

where E_i and E_s are lower and upper limits of the resolved resonance region. It should be noted that a similar resonance potential interference terms is included in the formulation proposed by JAMES and STORY [Ja 66].

IV.2 - Unresolved resonance region

Hauser-Feshbach's statistical formalism with width fluctuation correction is used. The cross sections, as average values, are calculated from average resonance parameters, target nucleus level scheme (if the inelastic scattering process is possible) and neutron transmission factors or neutron strength functions. The use of neutron transmission factors or neutron strength functions is equivalent from a theoretical point of view. In practice, the OM available from the literature are not adequate to describe simultaneously the neutron strength functions S^0 and S^1 and neutron cross sections. But we have to notice that at high energies and for nuclei with large $D^{\ell=0}$ also at low energies the capture cross sections are not very much sensitive to the values of S^0 and S^1.

The strength function formalisn is utilized by most of the authors, except by the Japanese team which employs the O.M. formalism on the overall energy range, and in ENDF/BIV for some nuclei.

The important problem of the width fluctuation correction has been extensively studied by Moldauer [Mo 75], Tepel and al [Te 74] and Gruppelaar and Reffo [Gr 77] and should be considered as solved. Several approximations have been indicated in order to save computer time. We do not enter the matter and refer the review by Gruppelaar and Reffo which is complete and the most recent one.

The radiative widths are usually estimated by means of Weisskopf or Brink-Axel model. These models predict two different energy dependences of Γ_γ which diverge at increasing energies but are very close up to some hundred KeV. Because Brink-Axel model is based on the experimental evidence of the giant resonance phenomenon it is considered preferable by an increasing number of evaluators after the recommendation of Benzi and Reffo (last Panel).

As far as we are informed the valence neutron process has been introduced only in RCN evaluations.

Finally, the Gilbert-Cameron formula, with an adjustement coefficient, is used for describing the level density. The adopted formula for the spin cut-off factor is mostly the same but different (Ri 75, Mo 75, Gr 76) constants are used with little consequence on the cross sections in the reviewed energy range. (There are theoretical arguments [Fa68] in favour of the higher value for the constant ($k = 0.147$)).

V. Mean Parameter Determination

The mean parameters are estimated by an analysis of the resonance parameter distribution. Then, they are adjusted within the error bars so as to give a good fit of the experimental data in the Kev region.

Differences occur, sometimes important, in the published values resulting from different analysis methods as well as from the priority accorded to the various parameters during the adjustment procedure.

V.1. Average resonance spacing determination

The D^{obs} parameter is, by far, the most important. It is basic to the level density determination.

It also has an important influence on the ℓ assignment for the resonances. Indirectly it plays a role in the "background" cross section determination due to missing levels.

At least, five methods exist for D^{obs} determination. Two of them employ the resonance energy distribution as basic data and the three others the reduced neutron width distribution.

1) The classical staircase method
2) The Dyson-Mehta Δ_3 parameter method (Dy)
3) The so-called "missing level estimator " method (Ke 76) (Annex II)
4) A method of fitting the experimental reduced neutron width distribution to a χ^2 law with one degree of freedom (Ro 76)
5) The so-called "ESTIMA" method (Ri 75) (Annex III)

The staircase method permits the determination of the energy above which resonances are obviously missing in the total (s & p resonances) but does not give any information on compensation for missing "s" resonances by "p" resonances. A loss of "s" resonances or a contamination by "p" resonance according to a linear law of energy, is not experimentally excluded.

The Δ_3 parameter method does not seem to be precise enough since, for example, i. tolerates an error of `1 % for a set of 00 resonances (even-even nuclei). In fact, this method appears to give a necessary but not sufficient condition for a set of resonances to be complete.

The "missing level estimator" method applies some Porter-Thomas law properties to a truncated experimental neutron width distribution (see Annex II). Unfortunately, this technique is sensitive to the quality of the largest width determination.

Methods 4 and 5 are very similar, since both fit a truncated experimental neutron width distribution to a χ^2 law with one degree of freedom. In such a way, $D^{\ell=0}$ and $\overline{g\Gamma n}$ values are obtained. The consequent $S^{\ell=0}$ value is compared to $\frac{\sum g\Gamma^0_{n_i}}{\Delta E}$ which sometimes allows the possibility of detecting some experimental systematic errors on the largest neutron widths.

A contamination by spurious resonances is also detected.

In method 4, the threshold is located so that only the s resonances are considered.

In method 5, all the information coming from the variation of the threshold is used (See Annex III).

To illustrate the disagreements which can occur in the use of these five methods, two examples have been chosen :

D^{obs} (eV) Fig. 2-3-4-5-6-7

Nucleus / Method	1	2	3	4	5
^{151}Sm	1.62	1.73	1.58	1.11	1.05
^{148}Nd	91.	113.	158.	171.	191.

Method 5 has the advantage of providing an "a posteriori" check of the $D^{\ell=0}$ estimated value. This is the one we recommend.

V.2 Determination of mean radiative widths

The mean value for the gamma width is obtained as an average over the known resonances.

When the experimental data are accurate enough it seems to appear some spin and parity dependence which is accounted for by taking into account spin and parity selection rules and by using a realistic level density description as indicated in the last Panel by Benzi and Reffo and applied by Ribon and al [Ri 75] .

VI. Local Systematics

They concern all the average statistical parameters. They are very useful for the completion of an evaluation program of vast dimensions. Actually, they are the only source of information which can be referred to in the case of absent experimental data. Furthermore, they can be used as guides (time saver) during the adjustment operations.

They have to be considered as semi-empirical laws since :

they are based on estimated data whose values depend on the technique used ($D^{\ell=0}$, for example).

- they depend on the weight given to the information coming respectively from the resonance parameters and from the capture data in the Kev region.

They can only be used locally and always for interpolations.

In the case where the systematics concern non-independent parameters ($D^{\ell=0}$, $\overline{\Gamma_\gamma}$), they must be realized simultaneously only from adjusted experimental data. It could be risky to impose to either of them a variation law theoretical or empirical, which would not have been previously verified on a large number of similar nuclei.

VI.1. Small "a" systematics (Fig. 8-9)

Since the last Panel, two semi-empirical systematics (Ri 75, Gr 76) have been published and a global theoretical description (Ch 76b) as well.

For the nuclei belonging to the "light" peak of the FP mass distribution ($42 \leq Z \leq 46, 52 \leq N \leq 65$), the two systematics give very close trends. They differ only by the absolute values due to different choices for the σ^2 normalization constant. They exhibit a nearly linear structure for "a" against the neutron number N_c in the compound nucleus

For the nuclei of the "heavy" peak, an odd-even effect seems to appear. As the neutron number move away from the magic numbers ($N = 52, 80$), the trend of "a" deviates more and more from the linear law and can decrease after having increased. An explanation for these two effects can be found in the deformation of nuclei (Ki 75, general trend of a) and in the presence of vibrationnal or rotational (spin effect) states (Fe 75).

A CNEN/CEA joint effort concerning more than 60 isotopes is underway to extend this systematics.

Owing to new, richer experimental data (Ch 76) on resonance parameters and more refined analysis techniques, more certain values for $D^{\ell=0}$ are presented which can support sophisticated theories on level density formulae.

The improvements in $D^{\ell=0}$ knowledge realized since 1973 are significant (Ex. ^{103}Rh).

For what concerns the investigated families, one can estimate that the accuracy on D^{obs} lies between 10 % and 30 % (1σ) for the nuclei with experimental data and between 50 % and 100 % for the nuclei which are relevant to systematics. Further improvements have to be made since this basic parameter must be known with an error bar no greater than 20 % for all the nuclei.

VI.2 Neutron strength function systematics

From the experimental point of view one should say the situation has been improved since last Panel in that better sets of consistent parameters are available and improved methodology in statistical analysis of resonance parameters is being used by experimentalists and evaluators.

From the theoretical point of view the situation has not significantly changed in the last years. As far as we know no overall optical model parameter set able to reproduce simultaneously both strength functions and neutron cross sections is available. Only recently [La 76] a local spherical OMP set has been found for ^{98}Mo which fits S^0, S^1, R' and total cross section. A sizeable improvement would be obtained if this method were successfull for large classes of nuclei.

As an alternative, at Bologna, [Re 77] is underway a study on the effective possibility to find an overall spherical local potential non local equivalent able to fit all relevant data in the whole energy range.

On the other hand the presented systematics are affected by large uncertainties. This could be due besides to experimental errors, also to the fact that a more complicate parametrization should be used. For instance odd-even fluctuations are observed (Ri 75, Gr 77) and interpreted by Kirouac (Ki 75).

Many authors (Ri 75, Mu 75, Gr 76, Ko 77) have observed, in a general way, that $S_2 > S_0$. This fact is in opposition with Musgrove's indications.

VI.3. Radiative width or radiative strength function systematics (Fig. 10,11,12)

Because the theoretical expression of Γ_γ depends through the E_1 selection rule on spin and parity distribution of levels, particularly low lying levels, it seems "a priori" rather difficult to find a complete parametrization suitable for a systematic analysis of the trend of the behaviour of this important parameter.

In addition it should be also noticed that, according to available experimental data, not all the nuclei exhibit a sizeable spin and parity dependance.

A tentative of taking into account spin and parity effect has been made by Ribon and al [Ri 75].

Exception made for the even-odd target nuclei this tentative seems not to be completely satisfying

A particular mention ought to be made of Musgrove's overall systematics of Γ_γ which proved to give, in general, good results where the spin and parity effectis are negligible.

VII. Uncertainties estimate

In the procedure of adjust-ments on integral data a knowledge of error bars on evaluated data is necessary. The solution of this important problem corresponds to a recommendation of the previous Panel but seems to have been studied in a complete way only in RCN evaluations. The adopted method is described in reference [Gr75]. It could be useful to compare the calculated uncertainties with the experimental errors and the spread of experimental values in order to test the assumptions and the error estimates on the mean parameters of the statistical model.

C - Intercomparison of capture cross sections from recent evaluations of some most important fission products - Recommended average parameters

(Tables 1 to 7)

In the resolved resonance region, it is obvious that the difference between evaluated group cross sections result of differences in the resolved resonance parameters (including resonance energies in the case of few resonances) or of a different treat ment in the thermal range (1/v behaviour, negative resonances, statistically generated resonances). This argument will not be repeated in the following, where R.R and U.R will refer to resolved resonance region and to unresolved resonance region respectively.
On the figures the curves numbered x x x are calculated with the recommended parameters.

$\boxed{95Mo}$ Fig. 13

R.R : acceptable agreement

U.R

Experiments : good agreement between KAPCHIGASHEV's (SDS) and MUSGROVE 's (Maïer - Leibnitz detector) values.

Evaluations : overall agreement (better than 10%) between all the reviewed evaluations.

Recommended evaluation : CNEN/CEA - recommended average parameters

$$D^{obs} = 86 \text{ ev} \quad S_0 = 0.5, \quad S_1 = 7.$$
$$\Gamma_\gamma^{\ell=0} = 170 \text{ mev}, \quad \Gamma_\gamma^{\ell=1} = 280 \text{ mev}$$

Presence of non statistical effects.
Requested accuracy : achieved.

$\boxed{^{97}Mo}$ Fig. 14

R.R : Appreciable difference between the two extremes JAERI and RCN, probably due to radiation width.

U.R

Experiment : Somewhat good agreement between MUSGROVE (Maïer - Leibnitz detector) and KAPCHIGASHEV (SDS) except above ≃ 20 Kev where the latter values cannot be fitted in using the same average parameters as at lower energy.

Evaluation : strong desagreement JAERI-ENDF/BIV below ≈ 15 Kev. Most evaluations are in agreement for E > 15 Kev. Due to better agreement with integral results (La76) R C N evaluation is prefered rather than CNEN/CEA evaluation (very close).
Recommended parameters : D^{obs} = 47ev, So = 0.5, S_1 = 7.5 $\Gamma_\gamma^{\ell=0}$ = 100 mev $\Gamma_\gamma^{\ell=1}$ = 145 mev). Presence of non statistical effects.
Requested accuracy : achieved.

$\boxed{^{98}Mo}$ Fig. 15

R.R : in general close results from RCN and ENDF/BIV evaluation. Reasonable agreement with the others up to ≈10 Kev. There is a not completely explainable discrepancy between Musgrove's values' and group cross sections calculated by Gruppelaar from CHRIEN'S resonance parameters.
U.R : Numerous data obtained by all the possible techniques (except sphere transmission) but in strong desagreement (up to a factor 3) which cannot be explained (statistical fluctuations) by considering that the resolved region is extending to energies close to 50 Kev. All the investigated evaluations are in good agreement but are lower than the experimental data.
An experiment by activation with a bad energy resolution, around 40 Kev could help to solve this discrepancy.
The recommended curve (affected by an estimated uncertainty of ± 30%) corresponds to that obtained with the following recommended parameters (very close to Musgrove's (Mu 76) values).
(D^{obs} = 910 ev, So = 0.6, S_1 = 6. $\Gamma_\gamma^{\ell=0}$ = 88 Mev, $\Gamma_\gamma^{\ell=1}$ = 107 Mev)
Presence of non statistical effects.

$\boxed{^{100}Mo}$ Fig. 16

R.R

Reasonable agreement (except around 60 ev) between all the evaluations. After Gruppelaar (Gr76) a new experiment of Weigman (76) indicates smaller neutron widths for s wave resonances between 1 and 2.5 Kev than those given in BNL325 and used in some evaluations.

U.R

Experiments : Similar situation to that of ^{98}Mo.

Evaluations : Similar shape but disagreement of ≈ 40% between the extremes. The evaluated curves are generally lower than the experimental values. For this nuclide, also, an experiment by activation (and bad energy resolution), around 40 Kev could be useful.

The adjustement based on STEK measurement is in favour of ENDF/BIV curve which is recommended with an accuracy of ± 30%.

Gruppelaar indicates opposition between differential and integral results.

Recommended parameters : (D^{obs} = 700 ev, S_0 = .35, S_1 = 5. $\Gamma_\gamma^{\ell=0}$ = 70 mev, $\Gamma_\gamma^{\ell=1}$ = 75 mev)

$\boxed{^{99}Tc}$ Fig. 17

R.R

Reasonable agreement, but CNEN/CEA or RCN group cross section values should be prefered because based on recent values of Adamchuck [Ad73] .

U.R

Experiment : only one set of data from Chou and Werle (Ch73)(SDS).

Evaluations : good agreement between all the investigated evaluations.

But since all the adjustements on integral data [La76, Gr76] indicate an increase, JAERI curve is prefered affected by an accuracy better than ± 10%.

Recommended parameters : (D^{obs} = 17.6 ev, S_0 = .47, S_1 = 7. $\Gamma_\gamma^{\ell=0}$ = 142 mev, $\Gamma_\gamma^{\ell=1}$ = 142 mev)

COMMENT : A measurement is underway in GEESTHACHT (University of KIEL - GERMANY, F.R.).

$\boxed{^{109}Ag}$ Fig. 26

R.R Good agreement of all the evaluations

U.R Few experimental values in disagreement. They have been obtained by the relative activation method, except those of

Kononov who used the large detector technique.
The evaluations differ considerably. All the adjustements (La76, Gr76) show a tendancy to support the microscopic values of Poenitz. That is why we recommend RCN evaluation affected by an accuracy of ±20%.
Recommended parameters D^{obs} = 17.7 ev, S_0 = 0.8, S_1 = 3.8
$\Gamma_\gamma^{\ell=0}$ = 135 mev, $\Gamma_\gamma^{\ell=1}$ = 135 mev.

$\boxed{^{133}Cs}$ Fig. 29

R.R

Rather good agreement of all the evaluations

U.R

The available data obtained by many methods (SDS, activation, big detector technique etc) are very discrepant. This imporant discrepancy explains the spread of the evaluations which are more or less compromises. Because of integral experiments indications [La 76, Gr 76], an average between RCN and CNEN/CEA should be adopted, which would rather nicely reproduce the experimental points of Booth and Kompe.
Assigned uncertainty : ± 10%.

Recommended parameters : D^{obs} = 23.4 ev, So = 0.8, S_1 = 3.4
$\Gamma_\gamma^{\ell=0}$ = 133 mev, $\Gamma_\gamma^{\ell=1}$ = 145 mev.

$\boxed{^{101}Ru}$ Fig. 18

R.R : Reasonable agreement except for ENDF/BIV which shows a large discrepancy between 100 ev and 465 ev due to different resonance parameters (Gr76) .

U.R

Experiment : no experimental data available at present time.
After Gruppelaar, recent measurements by Hockenbury and al (Ho76) not yet published.

Evaluations : Acceptable agreement between RCN, JAERI and CNEN/CEA curves - ENDF/BIV lower by 20%.
According to adjust ments on integral experiments, an average between RCN and CNEN/CEA curves should be adopted.
Recommended parameters : $D^{obs} = 16.7$ ev, $S_0 = 0.47$, $S_1 = 7$.
$\Gamma_\gamma^{\ell=0} = 170$ mev, $\Gamma_\gamma^{\ell=1} = 190$ mev.
Requested accuracy achieved.

$\boxed{^{102}Ru}$ Fig. 19

<u>R.R</u> : 3 observed resonances. Large discrepancy in the ABBN scheme between CNEN/CEA group constants and the others. The amplitude of the discrepancy is largely reduced in the CARNAVAL energy scheme The adjustements [L.76] on integral experiment go in direction of RCN values. (A value of 210 mev instead of 181 mev should be used in CNEN/CEA calculations for the gamma width).

<u>U.R</u> : the presented evaluations agree with microscopic data and with each other. The integral experiments require large amplitude of adjust ment.
The recommended curve should result from the use of the following average parameters. $D^{obs} = 550$ ev, $S_0 = 0.3$, $S_1 = 4.5$
$\Gamma_\gamma^{\ell=0} = 210$ mev, $\Gamma_\gamma^{\ell=1} = 230$ mev.
It is obvious that measurements in both resolved and unresolved resonance regions are required.

$\boxed{^{103}Ru}$ Fig. 20

There are no experimental data for this nuclide. Not too high a value for the capture cross section is likely otherwise it would have been detected in integral experiment. A cross section similar in amplitude to that of ^{101}Ru can reasonably be accepted. An other argument in favour of CNEN/CEA cross section is that the average parameters are relevant of more recent systematics.
The recommended curve is the CNEN/CEA curve, but reduced between 1 ev and 47 ev by a factor 0.55. The attributed uncertainty is ± 40%.

Recommended parameters : D^{obs} = 7.5 ev, S_0 = 0.45, S_1 = 6.
$\Gamma_\gamma^{\ell=0}$ = 96 mev, $\Gamma_\gamma^{\ell=1}$ = 111 mev.

$\boxed{^{104}Ru}$ Fig. 21

R.R : Only 4 observed resonances are given. The group cross sections of the various evaluations exhibit some differences which can reach 40%-50%. Due to their better agreement with integral data (Gr76) RCN values have to be prefered.

U.R : Only few microscopic measurements exist which are not coherent (factor 3 between Macklin's and Chaubey's values at 25 Kev (activation method)). The evaluations are in reasonable agreement.
Recommended curve : ENDF/BIV, accuracy : ± 15%
Recommended parameters : D^{obs} = 300 ev, S_0 = 0.34, S_1 = 4.
$\Gamma_\gamma^{\ell=0}$ = 75 mev, $\Gamma_\gamma^{\ell=1}$ = 75 mev.
New experiments are requested in both resolved and unresolved resonance regions.

$\boxed{^{103}Rh}$ Fig. 22

R.R : The set of R.P. given by BNL 325 [Mu73] can be considered as complete and coherent. Some differences occur with respect to this reference. For example, as reported by Gruppelaar (Gr76) the capture width for the resonance at 2.6689 Kev in ENDF/BIV is too large by a factor of 10 and some Γ_n for many resonances with unknow spins are too high by a factor2. The evaluations are in good agreement except for the group 20 (ABBN scheme). The results of integral experiments [L 76] seem to indicate that the CNEN/CEA evaluation should be prefered.
It must be mentionned that DILG and MANHORT [Di70] give at 2200m/s for σ_t a value of 144.8±0.7 b in contradiction with the value adopted for σ_c = 149.1 b, the latter being supported by the value obtained from resonance parameters.

U.R : Many microscopic results (by all techniques except sphere transmission) sometimes in disagreement. Most evaluations are in good agreement (10% or better) (ENDF/BIV is lower by 30%).

According to all adjust-ments JAERI curve should be retained. The requested accuracy is achieved, but since ^{103}Rh is often used as standard the opposition between the tendancies indicated by integral experiments and the most recent (confident) microscopic (Le75, Ho76) results has to be solved.

Recommended parameters : D^{obs} = 26.4 ev, S_0 = 0.57, S_1 = 7.

$\Gamma_\gamma^{\ell=0}$ = 160 mev, $\Gamma_\gamma^{\ell=1}$ = 160 mev.

The curve calculated with the recommended parameters is a better compromise between microscopic integral data than the recommended curve.

$\boxed{^{105}Pd}$ Fig. 23

R.R : The differences in the evaluated data, between 10^{-1} ev and 10^{1} ev are due to different treatements in the thermal range.

U.R : One single set of data is available [Ho75] (large detector linac) which have not been taken into account by most evaluations. The differences observed in the lower part of the energy range are due to different values of So. Above 1 Kev and up to 50 Kev, RCN, JAERI and CNEN/CEA evaluations are in good agreement and well supported by the experimental data. At higher energy an important disagreement between experimental and evaluated values begins to appear. Because it is in a better overall agreement with integral data, the CNEN/CEA evaluation has to be prefered, affected by an accuracy of ± 10%.

Recommended parameters : D^{obs} = 10.2 ev, S_0 = 0.45, S_1 = 6.

$\Gamma_\gamma^{\ell=0}$ = 154 mev, $\Gamma_\gamma^{\ell=1}$ = 164 mev.

COMMENT : A measurement is planned at GEEL.

$\boxed{^{107}Pd}$ Fig. 24

There are no experimental data at all. The differences observed in the resolved resonance region are due to different treat-ments (generation of resonances or not) and to different systematics, especially for the parameter "a". The latter reason explains also the disagreement up to 100 Kev, which can be completely reduced only by new microscopic experiments in both R.R and U.R for

^{106}Pd, ^{108}Pd, ^{110}Pd (this possibility is considered at Rensselaer Polytechnic Institute) and by the informations from integral experiments. RCN and CNEN/CEA evaluations are very close.
Thanks to indications from STEK experiments, they can be equally recommended, affected by an uncertainty of ± 30%.
Recommended parameters : D^{obs} = 5 ev, S_0 = 0.4, S_1 = 5.5
$\Gamma_\gamma^{\ell=0}$ = 137 mev, $\Gamma_\gamma^{\ell=1}$ = 160 mev.

COMMENT : The results of a new measurement (RR) performed at RPI will be published at the ANS WINTER Meeting (Nov. 27 - Dec. 2, 1977).

$\boxed{^{135}Cs}$ x

The only available experimental data is the capture cross section at thermal energy.
The differences, sometimes very important (factor 5 between ENDF/BIV and CEA) between the evaluations in R.R and U.R can be explained by the adopted value for D^{obs} whose determination is depending on systematics. The latter could be improved by new experiments (R.R) concerning ^{135}Cs.
A mean curve between JAERI 2 and CNEN/CEA curves affected by an uncertainty of ± 50% could be retained.
Recommended parameters : D^{obs} = 70 ev, S_0 = 0.7, S_1 = 1.7
$\Gamma_\gamma^{\ell=0}$ = 125 mev, $\Gamma_\gamma^{\ell=1}$ = 125 mev

COMMENT : According to Priesmeyer Dobs value is likely under-estimated. A new measurement is planned for next year in university of KIEL (Germany, F.R.)

$\boxed{^{141}Pr}$ Fig. 34

R.R : RCN and CNEN/CEA evaluation are in reasonable agreement.
ENDF/BIV differs appreciably sometimes, (in the ABBN groups 21-19) as a consequence of a different method of fitting the thermal region.

U.R

Experiment : The available microscopic data have been obtained mainly by activation. They differ largely (factor 2), even after a renormalization. ENDF/BIV evaluation is systematically higher than the other evaluations and the experimental points. RCN and CNEN/CEA evaluations agree well and can be equally recommended since they are in accordance with the indications of integral experiments (Gr76).
Assigned uncertainty ± 10%.

Recommended parameters : D^{obs} = 132 ev, S_0 = 1.6, S_1 = 1.1
$\Gamma_\gamma^{\ell=0}$ = 92 mev, $\Gamma_\gamma^{\ell=1}$ = 95 mev

^{149}Sm Fig. 35

R.R : this isotope is characterized by a very large thermal capture cross section. Rather large disagreements above 10 ev.

U.R : there is only one experimental result from MACKLIN (Ma63) obtained with total energy detector (MOXON-RAE). So, the evaluations are mainly based on average parameters resulting from resonance parameter distribution analysis and systematics. That explains the observed (≃ 20%) differences. Following Gruppelaar's argumentations we recommend CNEN/CEA evaluation affected by an uncertainty of ± 15%.
Recommended parameters : D^{obs} = 1.9 ev, S_0 = 5.0, S_1 = 1.4
$\Gamma_\gamma^{\ell=0}$ = 63 mev, $\Gamma_\gamma^{\ell=1}$ = 60 mev
Comments : due to its "importance" an experimental effort (mainly integral experiment) should be made for this isotope.

^{151}Sm Fig. 36

This isotope is characterized by a large thermal capture cross section. There are only three recent evaluations. The JAERI evaluation is about 30% lower than the CNEN/CEA evaluation in the overall reviewed energy range. ENDF/BIV is lower in R.R but systematically larger by a factor 2 in U.R with respect to the CNEN/CEA evaluation. This differences can be explained mainly by differences in D^{obs} estimation.
Since we are very confident in the "ESTIMA" method we think that the CNEN/CEA value for this parameter is probably the more realistic one.
Integral experiment (La76) results support an increase of CNEN/CEA curve. We recommend the CNEN/CEA curve with an assigned uncertainty of ± 20%.
Recommended parameters : D^{obs} = 1.05 ev, S_0 = 3.8, S_1 = 1.4
$\Gamma_\gamma^{\ell=0}$ = 96 mev, $\Gamma_\gamma^{\ell=1}$ = 96 mev.
Comments : An experimental effort has to be made for this isotope. A microscopic experiment is possible up to ≃ 70 Kev using Maier-Leibnitz detector, since the inelastic level at 4.8 Kev will

provide γ rays of same energy which will be canceled by the electronic set-up threshold.

$\boxed{^{108}Pd}$ Fig. 25

R.R

There are 3 observed resonances so that an accurate determination of D^{obs} is problematic. ENDF/BIV and RCN evaluations are in good agreement, STEK experiments indicate a tendancy for lower values.

U.R

Experiments : very few values, all obtained by activation.
Evaluations : the evaluations are very close and fit into the experimental data.
As a consequence of the informations from STEK experiments, ENDF/BIV should be prefered, affected by an accuracy of ± 20%.
Recommended parameters D^{obs} = 200 ev S_0 = 0.4 S_1 = 3.
$\Gamma_\gamma^{\ell=0}$ = 68 mev $\Gamma_\gamma^{\ell=1}$ = 75 mev.

$\boxed{^{127}I}$ Fig. 27

R.R.

A lot of resonances have been detected. There are no experimental data for Γ_γ, except in the most recent experiment of Rohr (Ro 76) There are appreciable differences in the group cross-sections.

U.R.

The discrepancy between the two extremes, evaluations CNEN/CEA and RCN, can reach 20%. That is a surprising situation for an isotope which has been considered often as a standard. RCN evaluation, based on adjusted mean parameters deduced from Rohr's experiment, is smaller than nearly all the experimental points. Since some weight has to be given to the "old" experiments and since it fits perfectly into the recent (confident) result of Rimawi and Chrien

/Ri 76/, ENDF B-IV has to be prefered, affected by an uncertainty no better than 10%.

Recommended parameters : D^{obs}=14.2 eV ; So = 0.8 ; S_1 = 1.8 $\Gamma_\gamma^{\ell=0}$ = 120 mev ; $\Gamma_\gamma^{\ell=1}$ = 120 mev.

$\boxed{^{129}I}$ Fig. 28

R.R.

There are few exprimental resonance parameters given (no data for Γ_γ). Discrepancies are present in the energy range of the observed resonances. Good agreement for σ_{th} and RI_c.

U.R.

No data given. It is difficult to recommend one evaluation since "a" and Γ_γ (mostly) systematics are rather uncertain. The differences observed in the evaluations are mainly due to different values adopted for Γ_γ. With respect to ^{127}I, RCN and CEA systematics on Γ_γ show variations in opposite sense, although they are based on similar formalisms. It seems more likely that Γ_γ for ^{129}I is smaller than for ^{127}I. Starting from the recommended Γ_γ value for ^{127}I and assuming a variation of the same amplitude (but in opposite sense) as calculated by Grupellaar, it follows a value of $\approx$ 110 mev very close to that used in RCN evaluation which is recommended, affected by an uncertainty of $\pm$ 40%. It is obvious that a careful study on the systematics will improve the evaluation of this nuclide.

Suggested parameters : D^{obs} = 30 ev, So = 0.5, S_1 = 2. $\Gamma_\gamma^{\ell=0}$=110 mev, $\Gamma_\gamma^{\ell=1}$=110 mev.

$\boxed{^{143}Nd}$ Fig. 31

R.R.

For this nuclide thereis a rather complete experimental set of resonance parameters. (E < 2 KeV). There is no evident J, π dependence for Γ_γ.

The observed difference between ENDF/BIV, JAERI 77 and CNEN/CEA group cross sections (19-21 groups - ABBN scheme) is probably due to a difference in resonance parameters of the first resonance.

As indicated by integral experiment, CNEN/CEA data seem preferable.

U.R.

There are no experimental data.

ENDF/BIV is much higher than JAERI and CNEN/CEA evaluations which should be adopted on the basis of integral data [Ta 76].

Assigned accuracy ± 20%.

Recommended parameters : D^{obs} = 39 ev ; So = 3.3 ; S_1 = 0.8 ; $\Gamma_\gamma^{\ell=0}$ = 70 mev ; $\Gamma_\gamma^{\ell=1}$ = 70 mev.

Comment : a microscopic experiment (Maier-Leibnitz detector) is possible up to 750 KeV.

$\boxed{{}^{145}Nd}$ Fig. 32

R.R.

For this nuclide, the conclusions are similar to the above one.

U.R.

There are no experimental data (microscopic or integral).
The available recent evaluationsare in very good agreement and can be equally recommended, with an assigned accuracy of ± 20%.

Recommended parameters : D^{obs} = 19.2 ev ; So = 3,5 ; S_1 = 0.8 ; Γ_γ = 53mev ; Γ_γ = 46 mev.

Comment : a microscopic experiment (Maier-Leibnitz detector) is possible up to 70 KeV.

^{139}La Fig. 30

R.R.

Important discrepancies between ENDF/BIV and the other evaluations. Very few data for Γ_γ.

U.R.

Numerous experimental data in big disagreement, especially for energies around 25 KeV. Although the scarcity of information about Γ_γ, there is a good agreement between all the evaluations which can be equally recommended affected by an uncertainty of $\pm$ 15%.

Recommended parameters : D^{obs} = 270 ev ; So = 0.72 ; S_1 = 17 ; $\Gamma_\gamma^{\ell=0}$ = 60 mev ; $\Gamma_\gamma^{\ell=1}$ = 50 mev.

^{147}Pm Fig. 33

R.R.

There is a good agreement between all the available evaluations.

U.R.

No experimental data given. There is a discrepancy of about 30% between the two extremes evaluations ENDF/BIV and CEA. This difference is explained by the fact that ENDF evaluation has been calculated using Moldauer's O.M. Due to the small value of D^{obs}, σ_c is very sensitive to the choice of OMP.
The agreement between CNEN/CEA and JAERI 77 is very good and supported by the integral experiment data [La 76]. These two evaluations are equally recommended affected by an uncertainty of $\pm$ 15%.

Recommended parameters : D^{obs} = 5 ev ; So = 3.5 S_1 = 0.7 ; $\Gamma_\gamma^{\ell=0}$ = 73 mev ; $\Gamma_\gamma^{\ell=1}$ = 73 mev.

Concluding remarks - Suggestions for improvements

The suggestions can be different if the fields of the pure nuclear physics and of the reactor physics are considered separately ; this situation is due mainly to the fact that the accuracy assigned to an evaluation can be different in magnitude to that assigned to the ajusted curve :

For the most important FP, the reactor physicists claim that, thanks to the joint information from evaluations and integral experiments, their needs are nearly satisfied, with few exceptions concerning, besides the nuclei whose evaluation is relying only on systematics (Ru^{103}, Pd^{107}, Cs^{135}) the following nuclei : ^{102}Ru, ^{143}Nd, ^{109}Ag, ^{151}Sm.

- For the evaluators the strong opposition between microscopic and integral data which occur for some nuclei (^{98}Mo, ^{100}Mo) must be solved by new experiments. Furthermore the lack of data in the Kev region imposes a limitation to the accuracy (^{149}Sm, ^{151}Sm, ^{147}Pm).

On the other side, for the nuclei of intermediate importance, the ajustements on integral data can be questionable since a more and more large contribution to the reactivity loss comes from neutron slowing down and accurate data for elastic and inelastic cross sections are needed.

The conclusions we present here are a trial to conciliate different appreciations of the needs.

Microscopic data

In general, the needs for information in the resolved resonance region concern mainly Γ_γ data, which are at present very scarce, spin and orbital momentum assignments by experimental procedure. That, in order to make possible more precise determination of D^{obs} and to deduce the J, π dependence and the sharing between the compound and direct components for Γ_γ. This recommendation concerns most nuclei.

Appreciation of the situation of the main neutron standards

The most recent evaluations of ^{197}Au (n,γ) cross section (ENDF/B-V and Fo 75) agree very well in the common energy range. A study should look at the impact of GWIN'S (Gw 75) results on the evaluation of Sowerby of the ^{235}U (n,f) cross section for energies below 100 Kev.

Formalisms and techniques for evaluation

1) The resonances parameters presented in the evaluation should result (as in Ri 75) from a critical analysis of the available experimental sets. Such important work could, profitably, be performed in the frame of a cooperation between specialized laboratories.

2) The use of "ESTIMA" method is presently recommended for D^{obs} determination.

3) Neutron strength functions S_0 and S_1 have to be taken into account in the procedure to determine the OMP. As a starting point the "SPRT" method is recommended, [De75a] but improvements have not to be excluded (for example a mixture of the classical method with the SPRT method).

4) Presently, D^{obs} can be considered as better determined than Γ_γ. As consequences of this fact :

- In the statistical region the adjustments procedure should concern with priority the radiative width rather than D^{obs}.

- New theoretical developments are required to improve the treatment of Γ_γ and to find realistic interpretation of the reliable systematics in view of their extension to the following nuclei in the list of importance for which there are generally few experimental data.

5) It is recommended that, in the future evaluations, the confidence interval be estimated.

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ANNEX. I

- CONTRIBUED PAPER FROM P. RIBON -

It is well known that computation of cross sections of non fissile nuclei from resonance parameters can provide negative values of scattering and total cross sections. As a fact, there are 2 distinct origins of this defect-which add their effects :

1. The use of the Single Level Breit and Wigner formalism - i.e. the neglect of the scattering interference between levels. This leads to negative total and scattering cross sections when there are 2 (or more) very close resonances, of the same spin, with great neutron widths : a typical case is the group of 2 resonances of 232 Th at 21.8 and 23.4 eV.

This effect can be corrected by the use of the Multi Level Breit and Wigner formalism, i.e. by the addition of the term :

$$\sigma_{IR} = 4\pi \lambda\!\!\!^{-2} \sum_{J,\ell} \sum_{\lambda} \sum_{\lambda' \neq \lambda} g \frac{\Gamma_{n\lambda}}{\Gamma_\lambda} \frac{\Gamma_{n\lambda'}}{\Gamma_{\lambda'}} \frac{x x' - 1}{(1+x^2)(1+x'^2)} \quad [1]$$

with : $x = \dfrac{2\,(E - E_\lambda)}{\Gamma_\lambda}$

This term can be written as :

$$\sigma_{IR} = 4\pi \lambda\!\!\!^{-2} \sum_{J,\ell} \sum_{\lambda} g \frac{\Gamma_{n\lambda}}{\Gamma_\lambda} \sum_{\lambda' \neq \lambda} \Gamma_{n\lambda'} \frac{x\,(E_\lambda - E_{\lambda'}) + \frac{\Gamma_\lambda - \Gamma_{\lambda'}}{2}}{(1+x^2)\left[(E_\lambda - E_{\lambda'})^2 + \frac{(\Gamma_\lambda - \Gamma_{\lambda'})^2}{4}\right]} \quad [2]$$

It can be incorporated into the usual terms of the S L B W formalism by writing the resonance term as :

$$\sigma_R = 4\pi \lambda\!\!\!^{-2} \sum_{J,\ell} \sum_{\lambda} g \frac{\Gamma_{n\lambda}}{\Gamma_\lambda} \frac{1}{1+x^2} \left[1 + \sum_{\lambda' \neq \lambda} \frac{\Gamma_{n\lambda'} \frac{\Gamma_\lambda - \Gamma_{\lambda'}}{2}}{(E_\lambda - E_{\lambda'})^2 + \frac{(\Gamma_\lambda - \Gamma_{\lambda'})^2}{4}} \right] \quad [3]$$

and the potential scattering resonance interference term as :

$$\sigma_{IP} = 8\pi \lambda\!\!\!^{-} \sum_{J,\ell} \sum_{\lambda} g \frac{\Gamma_{n\lambda}}{\Gamma} \frac{x}{1+x^2} \left[R + \sum_{\lambda' \neq \lambda} \frac{\lambda\!\!\!^{-}}{2} \Gamma_{n\lambda'} \frac{(E_\lambda - E_{\lambda'})}{(E_\lambda - E_{\lambda'})^2 + \frac{(\Gamma_\lambda - \Gamma_{\lambda'})^2}{4}} \right] \quad [4]$$

This form of expression /1/ introduces into the SLBW formula corrections which are either constant, or small and smoothly varying with energy, and allows the use of the functions ψ, ϕ for Doppler broadening calculations. More precisely, as $\Gamma_n \sim \sqrt{E}$ for $\ell = 0$ resonances, the corrective term in /4/ is energy independent, while the corrective term in /3/ is energy dependent, but can be neglected since it is small.

2. The neglect of far-away levels in the calculation of the scattering interference between potential and resonance scattering, when there are not as many levels taken into account above and below the energy of interest.

This term is (for $\ell = 0$) :

$$\sigma_{IP} = 8\pi \lambda\!\!\!^{-} R \sum_J \sum_\lambda g \frac{\Gamma_{n\lambda}}{\Gamma_\lambda} \frac{x}{1+x^2}$$

The contribution of resonances is either positive or negative, and is decreasing as $\frac{1}{E-E_\lambda}$; assuming a simple model (constant values of $\bar{D}$ and S^o), the sum $\sum_{E_o}^{E\to\infty}$ is not finite. In order to provide an exact value of σ_{IP}, the same energy range should be taken into account above and below the energy E (i.e. approximately the same number of resonances are to be taken into account above and below E).

If :

$$\sigma_{IP(E)} = \sum_{E_\lambda = E_i}^{E_\lambda = E_s} \quad \text{(resonances } \lambda\text{)}$$

there is a systematic error - the cross sections are underevaluated if $E_s - E > E - E_i$ (what happens generally at low energy, when there are many resolved resonances).

This effect can be corrected by adding to the scattering and to the total cross section a term :

$$\delta\sigma_{IP} = 4\pi \lambda\!\!\!^{-} R \sqrt{E}\; S^o \; Log \left| \frac{E - E_s}{E - E_i} \right|$$

E_i being the lower energy, and E_s the higher, of the interval where resolved resonances are taken into account to calculate σ_{IP} .

- CONCLUSION -

At low energy these two approximations of the exact formalism add their effects to induce negative values of cross sections computed from resonance parameters, and it is recomm nded :

1/ to use the MLBW formalism ;
2/ either to add $\delta\sigma_{IP}$ as a smooth cross section in the evaluated file, or to take into account the correction $\delta\sigma_{IP}$ in the formula utilized by users to compute the resonance cross-section.

ANNEX II

- "MISSING LEVEL ESTIMATOR" METHOD EXTRACTED FROM ANL 7690 ARGONNE CONFERENCE - 1976 -

Authors : G.A. KEYWORTH, M.S. MOORE and J.D. MOSES.

The method is based on the assumption that : the neutron width distribution follows a Porter Thomas law, the larger widths are accurately known. It uses the properties of the Porter Thomas law resulting from a partial integration. For example, if a threshold equal to $\frac{g\overline{\Gamma_n^\circ}}{4}$ is applied to the experimental distribution, it gives :

$$\int_{1/4}^{\infty} f(x)\,dx = 0{,}617\,,\quad \int_{1/4}^{\infty} \sqrt{\Gamma_n^\circ}\, f(x)\,dx = 0{,}704\left(\overline{\Gamma_n^\circ}\right)^{1/2},\quad \int_{1/4}^{\infty} \Gamma_n^\circ\, f(x)\,dx = 0{,}969\,\overline{\Gamma_n^\circ}$$

with $x = \dfrac{\Gamma_n^\circ}{\overline{\Gamma_n^\circ}}$ and $f(x) = \dfrac{1}{\sqrt{2\pi x}} \exp\left(-x/2\right)$

So the ratio

$$\sum_{\overline{\Gamma_n^\circ}/4}^{\infty} g\Gamma_n^\circ \Big/ \left(\sum_{\overline{\Gamma_n^\circ}/4}^{\infty} \sqrt{g\Gamma_n^\circ}\right)^2 \text{ is equal to : } \frac{0{,}969}{(0{,}704)^2} \times \frac{0{,}617}{n} = \frac{1{,}206}{n}$$

In the pratice the method consists in "calculating the quantity $n \sum g\Gamma_n^\circ / \left(\sum \sqrt{g\Gamma_n^\circ}\right)^2$ starting with the largest value of $g\Gamma_n^\circ$ in the interval and adding additional levels, one at a time, going from larger to smaller in the ordered array of observed values of $g\Gamma_n^\circ$; when this quantity equals 1.206, the total number of levels in the interval is $\frac{n}{0.617}$".

A N N E X III

-"ESTIMA METHOD"- See reference (Ri 75).

The basic assumption is that the neutron width distribution follows a Porter-Thomas law. The principle is to fit a truncated experimental distribution with a χ^2 law using a maximum likelihood method (Ri69-Tr73). The latter permits to obtain $\overline{g\Gamma_n^o}$ and the number N_t of resonances of the complete distribution in the energy interval, once the degree of freedom has been fixed (equal to unity). The threshold is varied. Each value of the threshold is characterized by one value of $\overline{g\Gamma_n^o}$ and of N_t. The adopted values for these two quantities are those which remain constant when the threshold varies, as it would be the case for a pure distribution of numerous neutron widths (see figure relative to ^{127}I).

The contamination by spurious resonances leads to a quick variation (and always in the same sense) of both $\overline{g\Gamma_n^o}$ and N_t when the threshold varies and is easily detected.

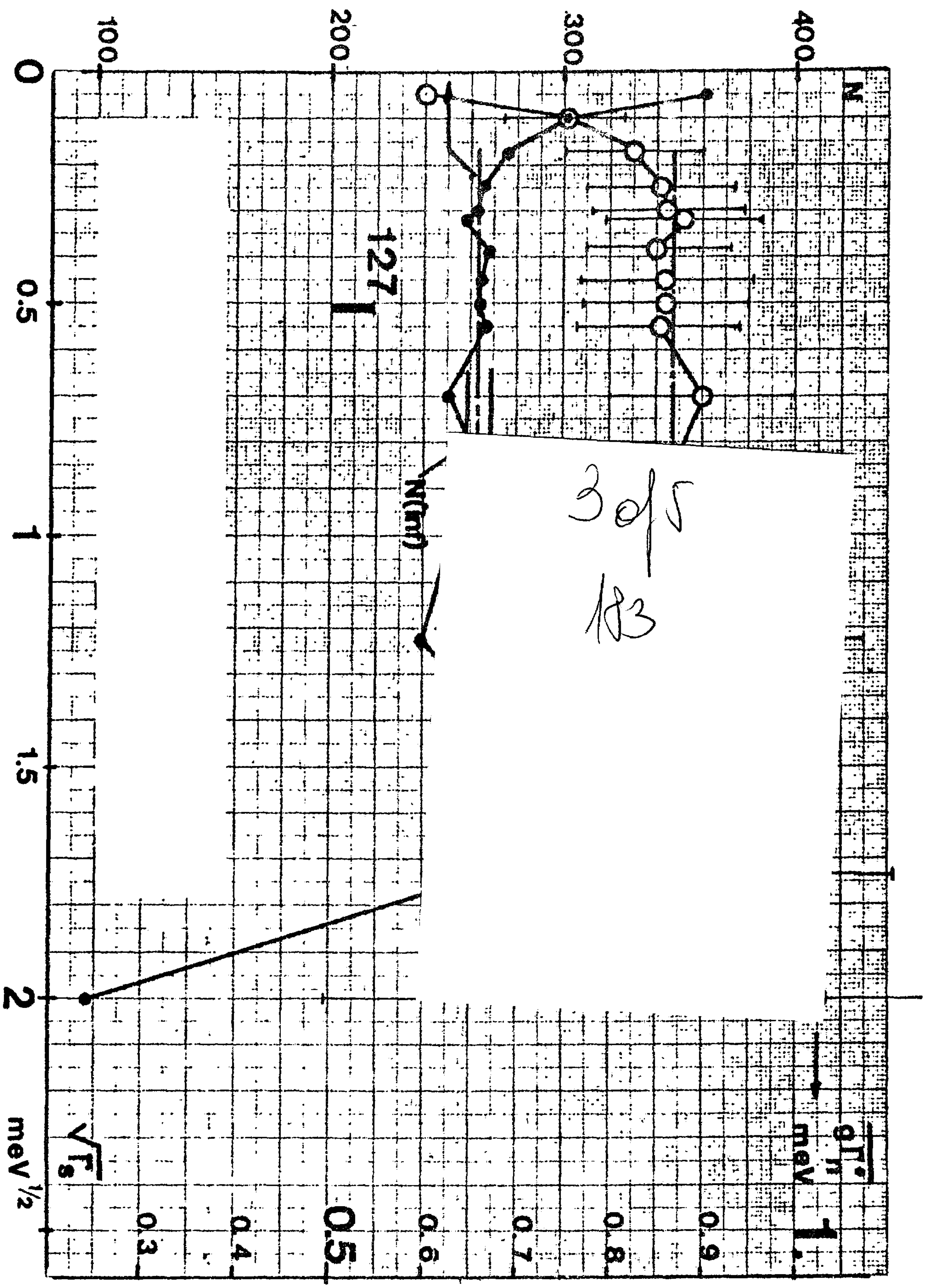
The relative error on the estimate of N_t is quadratically combined with the relative sampling error (given by Dyson and MEHTA equal to $\frac{0.45}{N}\sqrt{\log(2\pi N)+0.343}$) in order to obtain the total relative error on D^{obs}.

A check of D^{obs} value is provided by the following considerations : looking, in an adequate subinterval $\delta\varepsilon$ of the energy interval of interest, to the resonances declared as sure "s wave" resonances (See B.1.1. page 9) a counting is successively made of the resonances having reduced neutron widths $g\Gamma_n$ between $\overline{g\Gamma_n^o}$ and infinite $\rightarrow(N_1)$, between $0.5\overline{g\Gamma_n^o}$ and $\overline{g\Gamma_n^o}\rightarrow(N_2)$, between $0.2\overline{g\Gamma_n^o}$ and $0.5\overline{g\Gamma_n^o}\rightarrow(N_3)$, between $0.05\overline{g\Gamma_n^o}$ and $0.2\overline{g\Gamma_n^o}\rightarrow(N_4)$. (See fig. 4,7).

Since $\int_1^{-\infty} f(x)\,dx = 0.32$; $\int_{0.5}^{1} f(x)\,dx = 0.17$; $\int_{0.2}^{0.5} f(x)\,dx = 0.18$; $\int_{0.05}^{0.2} f(x)\,dx = 0.17$;

with $x = \frac{g\Gamma n^o}{\overline{g\Gamma n^o}}$ and $f(x) = \frac{1}{\sqrt{2\pi x}}\exp(-x/2)$

the following criteria have to be satisfied in a statistical way : $N_1 = 2N_2 = 2N_3 = 2N_4$. If the first equality, at least, is observed, then $D^{obs}_{chech} = \frac{\delta\varepsilon^4 x\ 0.32}{N_1} \simeq D^{obs}$; if not, a study is performed with a different value of $\overline{g\Gamma_n^o}$.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meV$^{1/2}$
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1

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LIST OF TABLES

LIST OF FIGURES

TABLE 1 σ^c_{th} (barn)

TARGET NUCLIDE	I^π	Exp BNL 325 III ed.	JAERI 77	ENDF/ BIV	RCN 76	CNEN/ CEA 76	MAXIMUM OF DIFFERENCE %	ACCURACY %(1σ)
94Mo				0.010				
95Mo		14.5±0.5	14.42	14.47	14.385	14.8	2.9	
96Mo				1.00				
97Mo		2.2±0.7	2.18	2.17	2.297	2.5	15.2	
98Mo		0.130±0.006		0.13	0.146	0.16	23	
100Mo		0.199±0.003		0.20	0.198	0.18	10	
99Tc		19±2	17.7	19.03	19.	20.5	15.8	
101Ru		3.1±0.9	3.34	3.10	2.88	3.0	15.7	
102Ru		1.3±0.15	1.310	1.30	1.1058	1.4	26	
103Ru*				7.70		65.5	750	
104Ru		0.47±0.2	0.411	0.44	0.47	0.47	14.4	
103Rh		150±5	146.3	148.24	147.44	158.3	8	
102Pd		4.8±1.5						
104Pd				0.39	4.	1.04	900	
105Pd		14	13.99	14.00	14.212	14.7	5	
106Pd		0.30±0.03		0.24	0.2957	0.305	3	
107Pd			10	10.00	9.983	20.8	110	
108Pd		12.2±0.24		12.21	12.2	9.7	25	
110Pd				0.22	0.2099	0.22	4.8	
107Ag		37.2±1.2		36.85				
109Ag		91±3	91	91.79	89.98	92.	2.2	
127I		6.2±0.2		6.20		6.2	0	
129I		27±3		27.00		27.	0	
131Xe				90.03				
133Cs		29±1.5	29	29.51	29.14	30.1	3.8	
135Cs*		8.7±0.5	8.7	8.7		9.	3.4	
137Cs		0.11±0.033	0.11	0.11		0.14	26	
141Pr				11.50	11.49	11.5	0	
143Nd		325±10	325.	325.08		317.	2.5	
145Nd		42±2	41.85	42.02			0.4	
147Sm		64±5	68.2	64.02	64.09	56.5	21	
148Sm				2.70				
149Sm		11000±2000	41500.	41191.	40997.9	40637.	2	
150Sm				102.00	108.573	108.	6.4	
151Sm		15000±1800	15600.	15008.	15147.4	15127.	3	
152Sm				206.10	205.971	209.	1.4	
153Eu		390±30	391.9	452.62		459.	17.6	
155Eu		4040±125	4040.	4040.		362.	11.	
139La		9±0.3		9.00	9.55	9.04	6.1	
147Pm		181±7	187.	181.98		198.	8.8	

TABLE 2 RI_c (barn) Experimental data

TARGET NUCLIDE	I^{π}	BNL 325	WALKER WA 72	COOK CO 70	POPE (Po73) STORY	LAUTENBACH LA 73	MAXIMUM OF DIFFERENCE %	ACCURACY % (1σ)
94 Mo		0.57±12						
95 Mo		105 ± 7	100	106	109	117	17	
96 Mo		20 ± 5						
97 Mo		13 ± 3	15	15	16.1	15.0	7.3	
98 Mo		6.2±0.3						
100 Mo		3.75±0.15						
99 Tc		340±20	200	197	353	368	87	
101 Ru		85±12	76	85.7	79.6	82.0	8	
102 Ru		4.1±0.4	4.20	10.6	6.9	14.5	245	
103 Ru*								
104 Ru		4.6±1.0	4.4	5.41	3.73	10.1	170	
103 Rh		1100±50	1100	1066	1048	1038	6	
102 Pd								
104 Pd								
105 Pd		90±10	85	74.5	86.3	90.8	22	
106 Pd		5.73±0.57						
107 Pd				80.1	79.7	68	18	
108 Pd								
110 Pd								
107 Ag		94±8						
109 Ag		1450±40	1450	1422	1457	1470	3.4	
127 I		147±6						
129 I		36±4	23	25.5		43	87	
131 Xe			830	787	890	898	14	
133 Cs		115±15	450	377	380	387	19	
135 Cs			58	58.1	30.2	20.5	183	
137 Cs				0.41	0.23	0.6	160	
141 Pr		14.1±0.2						
143 Nd			60	64.5	136	137	128	
145 Nd		240±35	250	272	298	237	25	
147 Sm		714±50	600	566	623	601	10	
148 Sm		27±14						
149 Sm				3594	3183	5541	54	
150 Sm		310±15						
151 Sm		3300±700	3100	2170	3772	2277	73	
152 Sm		3000±200						
153 Eu		1635±200	1500	1203	1512	1635	36	
155 Eu				1223	1817	3629	99	
139La		12.2						
147Pm		2300±80	2200		2236	2270	3	

TABLE 3 RI_c (barn) Evaluated data

TARGET NUCLIDE	I^{π}	JAERI 77	ENDF/B-IV BNL/NCS 50545	RCN 76	CNEN/CEA 76	MAXIMUM OF DIFFERENCE %	ACCURACY % (1σ)
94 Mo			0.9				
95 Mo		119.2	113.20		121.	6.7	
96 Mo			19.46				
97 Mo		17.09	16.12		15.7	8.9	
98 Mo			6.875		8.51	25	
100 Mo			3.85		5.40	40	
99 Tc		206.9	353.4	368.759	331.	71	
101 Ru		106.7	95.22	94.636	98.5	12	
102 Ru		4.141	4.03	3.886	3.79	9.2	
103 Ru*			70.31		567.1	715	
104 Ru		7.75	6.53	6.9589	7.51	18.7	
103 Rh		1034.	1048.0		1030.	1.75	
102 Pd							
104 Pd			17.95		20.41	13.7	
105 Pd		94.3	91.68		92	2.8	
106 Pd			7.18		6.23	15.3	
107 Pd		120.	69.9		255.	264	
108 Pd			224.40		252.	12.2	
110 Pd			7.05		4.27	65	
107 Ag			116.30			0	
109 Ag		1470.	1468.0		1427.	3	
127 I			155.20		150.7	3	
129 I		44.2	36.44		29.2	51.4	
131 Xe		904.	876.40			3.15	
133 Cs		398.	380.30	397.87	378.4	5	
135 Cs		62.	61.84		59.95	3.4	
137 Cs		0.59	0.488		0.76	29	
141 Pr			19.45		16.35	19	
143 Nd		134.	131.30		125.	63.6	
145 Nd		266.	231.3		225.5	18	
147 Sm		763.	748.20		736	3.7	
148 Sm			27.64				
149 Sm		3454.	3183.		3351.	8.5	
150 Sm			320.90		313	2.2	
151 Sm		4039.	3405.		3115.	29.7	
152 Sm			2996.0		2608	15.3	
153 Eu		1529.	1569.0		1396.	12.4	
155 Eu		3218.	1856.000		2180.	73.4	
139La			12.04	11.482	11.75	2.5	
147Pm		2206.	2283.0		2100.	8.7	

TABLE 4

D^{obs}_{eV}

TARGET NUCLIDE	I^{π}	SCHMITTROTH 73	JAERI 77	ENDF/BIV	RCN 76	CNEN/CEA 76	RECOMMENDED	ACCURACY %(1σ)
94 Mo					1740.			
95 Mo		114.	69.3	127	82.	85	86	10
96 Mo		1387.			1300			
97 Mo		77.5	72.3	66.5	68.	65	47.	15
98 Mo		1014		1275.	1000.	730	910.	15
100 Mo		1339		1200.	700.	520.	700.	25
99 Tc			16.2	14.?	18.6	18.6	17.6	6
101 Ru		18.3	13.8	14.	16.7	16.7	16.7	15
102 Ru			290.5	264.	573	550.	550.	30
103 Ru*				23.		7.5	7.5	50
104 Ru		285	588	784	265	270.	300.	25
103 Rh		27.4	26.1	20.3	26.1	26.4	26.4	5
102 Pd					1130.	880		
104 Pd					530.	460		
105 Pd*		10.1	11.1	8.8	10.	10.2	10.2	15
106 Pd				463	330.	270		
107 Pd			10.0	10.4	4.2	5.5	5	50
108 Pd				290.	200.	200.	200.	40
110 Pd				900.	146.			
107 Ag		32.2			19.			
109 Ag		19.5	12.7		17.5	18.	17.7	2.5
127 I		14.7		14.7	12.2	15.	13.7	3
129 I		26.1	21.	26.	30.	30.	30.	25.
131 Xe		39.2	33.2					
133 Cs		20.2	23.2		20.	23.4	23.4	4
135 Cs			60.0	328.		82.	70.	50
137 Cs			1100.	1930.				
141 Pr					120	132.	132	
143 Nd		32.0	46.4			39.	39	10
145 Nd		18.9	24.2			19.	19.	10
147 Sm		8.18	4.26		6.3	7.4	7.2	
148 Sm					107	114.	110.	20
149 Sm		2.88	1.63	1.7	2.0	1.97	1.9	10
150 Sm					56.5	66	64	
151 Sm			1.50		1.72	1.05	1.05	10
152 Sm					53.8			
153 Eu			1.46	1.3		1.05		
155 Eu			2.50	.86		0.92		
139 La				262.	286.	270.	270.	10
147 Pm			4.7	6.8		4.7	5.	15

TABLE 5

$S^0 \times 10^4$

TARGET NUCLIDE	I^{π}	MUSGROVE 70	JAERI 77	ENDF/BIV	RCN 76	CNEN/CEA 76	RECOMMENDED	ACCURACY %(1σ)
94 Mo		1.0			0.8		0.5	30
95 Mo		0.55	0.373		0.8	0.48	0.5	30
96 Mo		1.20			0.36		0.5	30
97 Mo		0.6	0.352		0.75	0.5	0.75	30
98 Mo		0.8			0.35	0.7	0.5	30
100 Mo		0.9			0.3	0.35	0.35	30
99 Tc		0.5	0.336	.44	0.47	0.45	0.47	20
101 Ru		0.4	0.328		0.56	0.56	0.6	20
102 Ru		0.4	0.325		0.322	0.25	0.3	30
103 Ru*		0.4				0.45	0.45	30
104 Ru		0.4	0.326		0.321	0.34	0.34	20
103 Rh		0.4	0.324	0.43	0.47	0.49	0.57	10
102 Pd		0.4			0.4			
104 Pd		0.4			0.4	0.4	0.4	50
105 Pd		0.35	0.327		0.5	0.45	0.45	25
106 Pd		0.4			0.4	0.4	0.4	50
107 Pd		0.4	0.336		0.4	0.45	0.4	50
108 Pd		0.6			0.4	0.6	0.6	40
110 Pd		0.6			0.4	0.6	0.6	50
107 Ag		0.4			0.37			
109 Ag		0.8	0.354		0.6	0.8	0.8	20
127 I		0.6			0.8	0.58	0.795	10
129 I		0.3	1.06		0.8	0.44	0.5	30
131 Xe		0.7	1.22					
133 Cs		0.7	1.42		0.8	0.85	0.8	12
135 Cs		0.8	1.61			0.7	0.7	
137 Cs		0.5	1.83			0.7	0.7	
141 Pr					1.72	1.5	1.6	
143 Nd		4.	2.82			3.3	3.3	20
145 Nd		3.	3.19			3.3	3.3	15
147 Sm		4.3	4.02		4.3		3.6	20
148 Sm		3.5			3.		3.6	
149 Sm		3.	3.88	3.2	5.1	5.3	5.0	20
150 Sm		3.			3.3	3.3	4.6	30
151 Sm		3.	3.80		3.65	3.67	3.8	15
152 Sm		2.5			2.2	2.3	2.2	20
153 Eu		2.2	4.20	2.5		2.8		
155 Eu		2.2	4.13	2.2		2.2		
139 La					0.64	0.8	0.72	25
147 Pm			3.48			3.2	3.3	20

TABLE 6

$S^1 \times 10^4$

TARGET NUCLIDE	I^{π}	MUSGROVE 70	JAERI 77	ENDF/BIV	RCN 76	CNEN/CEA 76	RECOMMENDED	ACCURACY %(1σ)
94Mo		5.3			4.5			
95Mo		5.06	5.41		5.4	7.	7.	30
96Mo		6.2			5.5		7.	30
97Mo		7.	5.86		6.	6.	6.	30
98Mo		7.			6.1	6.	6.	30
100Mo		6.5			6.2	7.	5.	
99Tc		7.00	6.1	3.9	6.	6.	7.	30
101Ru		6.5	6.19		7.3	7.3	7.	40
102Ru		6.5	6.17		7.3	5.	4.5	60
103Ru*		6.6				6.	6.	30
104Ru		5.	6.04		7.0	4.2	4.0	40
103Rh		6.6	6.12	5.1	6.5	6.	7.	15
102Pd		6.5			6.3	5.	5.5	
104Pd		5.			6.1	5.	5.5	
105Pd		4.	5.94		5.4	5.5	6.	30
106Pd		3.			5.7	3.	3.	50
107Pd		1.81	5.64		5.5	5.5	5.5	
108Pd		1.75			5.3	4.	3.	50
110Pd		1.65			4.9	1.65		
107Ag		1.81			3.8			
109Ag		1.7	5.26		3.8	3.	3.8	30
127I		1.64			2.	2.	1.6	40
129I		1.80	1.76		2.	2.1	2.	30
131Xe		1.7	1.58					
133Cs		1.30	1.42		3.9	3.	3.4	30
135Cs		1.10	1.26			2.	1.7	50
137Cs		1.	1.13			1.	1.5	50
141Pr					1.05	1.2	1.1	
143Nd		0.80	0.818			0.8	0.8	40
145Nd		0.70	0.744			0.7	0.7	40
147Sm		0.58	0.528		1.8	1.	1.4	40
148Sm		0.1			1.2		1.2	
149Sm		0.35	0.54	.5	1.8	0.6	1.4	40
150Sm		0.10			1.2	1.	1.3	
151Sm		0.50	0.549		1.2	0.8	1.4	40
152Sm		0.22			1.2	1.4	1.4	40
153Eu		0.25	0.487	.6		0.1		40
155Eu		0.1	0.486	.65	1.05	1.2	1.1	40
139La					2.	1.4	1.7	40
147Pm			0.6			0.7	0.7	40

TABLE 7 $\Gamma_\gamma^{\ell=0}$ mev / $\Gamma_\gamma^{\ell=1}$ mev

TARGET NUCLIDE	I^π	MUSGROVE 70	JAERI 77	ENDF/BIV	RCN 76	CNEN/CEA 76	RECOMMENDED	ACCURACY % (1σ)
94Mo		310			169 254			25
95Mo		227	180	350.	154 281	170/280	170/280	25
96Mo		267			152 202			25
97Mo		200	170.3	220.	134 155	134/190	100/145	25
98Mo		181		100.	86 138	85/106	88/107	25
100Mo		148		150.	58 85+20	60/90	70/105	25
99Tc		170	112	122	130 130	136/136	142/142	25
101Ru		191	165	192	174 194	171/191	170/190	25
102Ru		186	165	290	275 275	230/320	210/230	25
103Ru*		194		170		96/111	96/111	40
104Ru		210	165	160	97 97	97/100	75/75	25
103Rh			164	153	161 161	161/161	160/160	10
102Pd		204			250 275			25
104Pd		120			210* 190*			25
105Pd		155	155	153	155 155	156/170	154/164	25
106Pd		137		145.	120* 130*			25
107Pd		135	140	140.	100* 110*	123/155	137/160	40
108Pd		124		98.	70 80	65/76	68/75	25
110Pd		123		150.	50 55			25
107Ag		137			140 140			25
109Ag		157	130		129 129	126/126	135/135	25
127I				120.	95 95	143/150	120/120	20
129I			100	117.	107* 107*	68/68	110/110	25
131Xe			114	96.	85 85	92/90	92/95	25
133Cs		126	118		125 125	127/140	133/145	40
135Cs			125	105.		125/125	125/125	25
137Cs			100	95.				25
141Pr					85 85	92/90	92/95	25
143Nd			85			70/70	70/70	25
145Nd			60			53/46	53/46	25
147Sm					100			25
148Sm					60			25
149Sm			61	62	76	60/57	63/60	25
150Sm					60			25
151Sm			75		96	90/90	96/96	25
152Sm					70			25
153Eu			94	105		90/90		25
155Eu			100	96.		128/128		25
139La				55.	50.	62/48	60/50	25
147Pm			66	70.		69/69	73/73	25

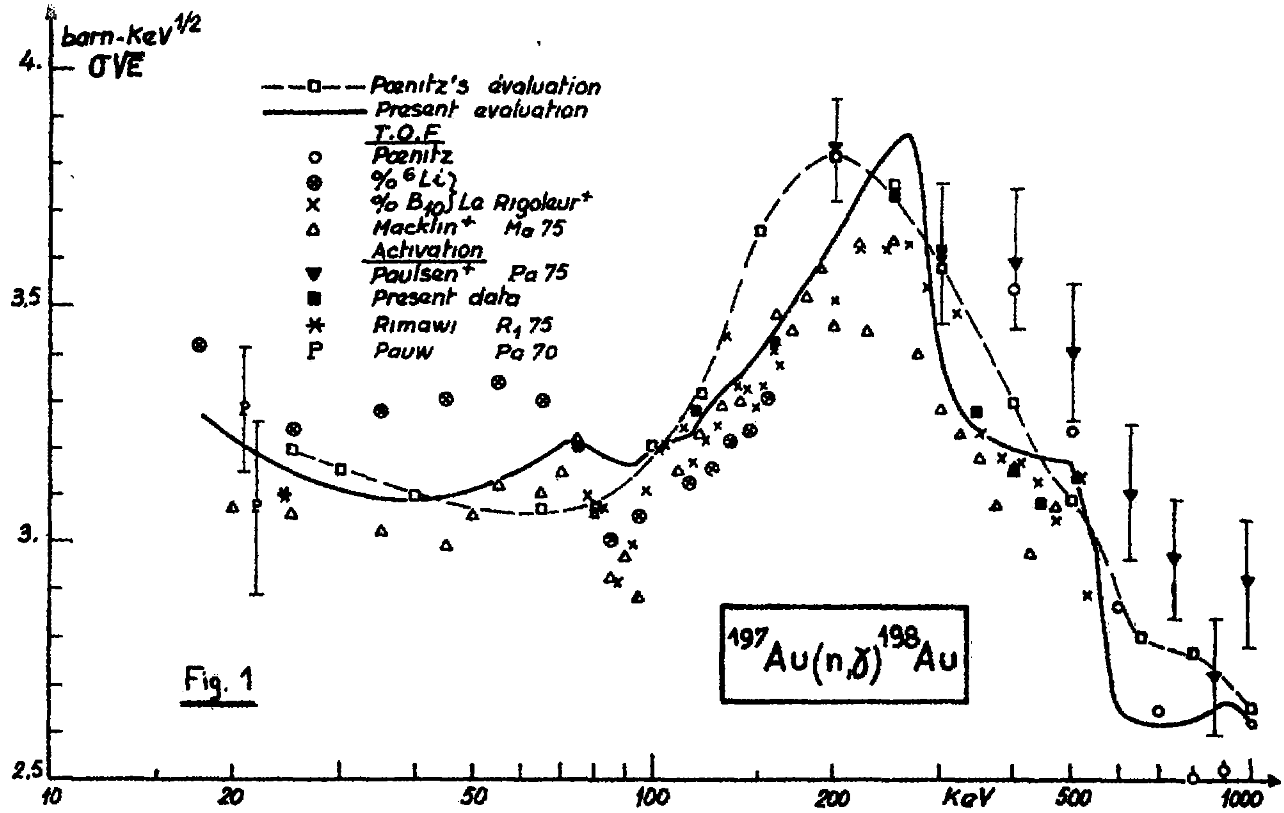


Fig. 1

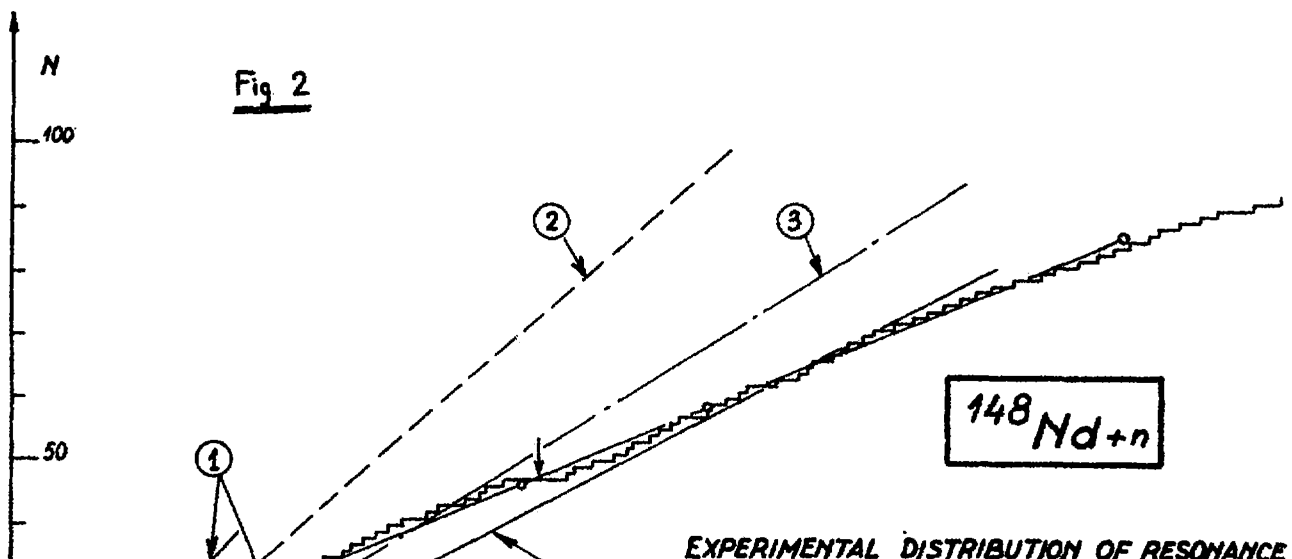


Fig. 2

EXPERIMENTAL DISTRIBUTION OF RESONANCE ENERGIES COMPARED TO THE RESULTS OBTAINED WITH THE VARIOUS METHODS OF D^{OBS} DETERMINATION

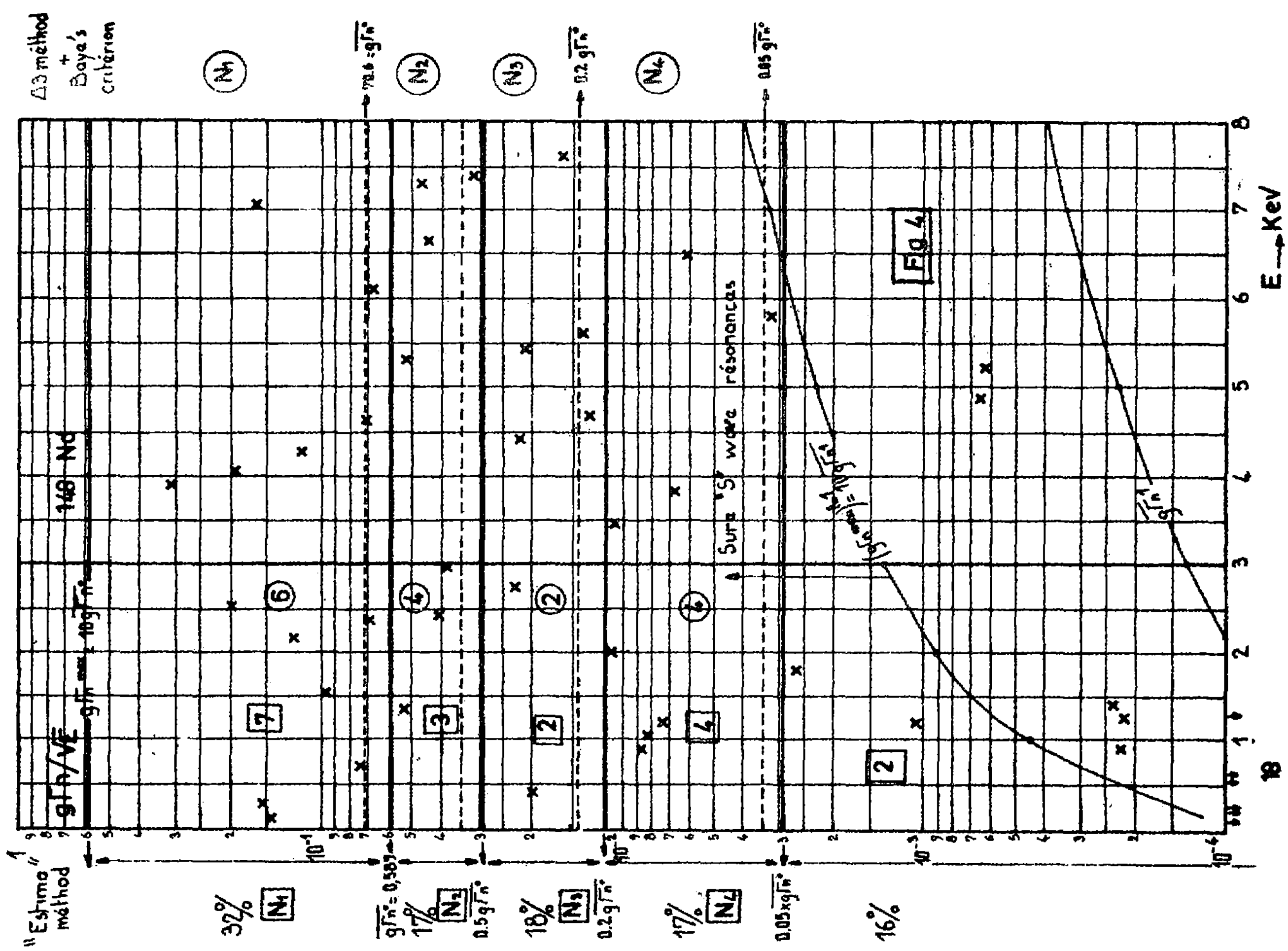

148 Nd
E → KeV
Δ3 method + Baye's criterion
"Estima" method
32%
17%
18%
17%
16%
Sure "S" wave résonances

Fig 4

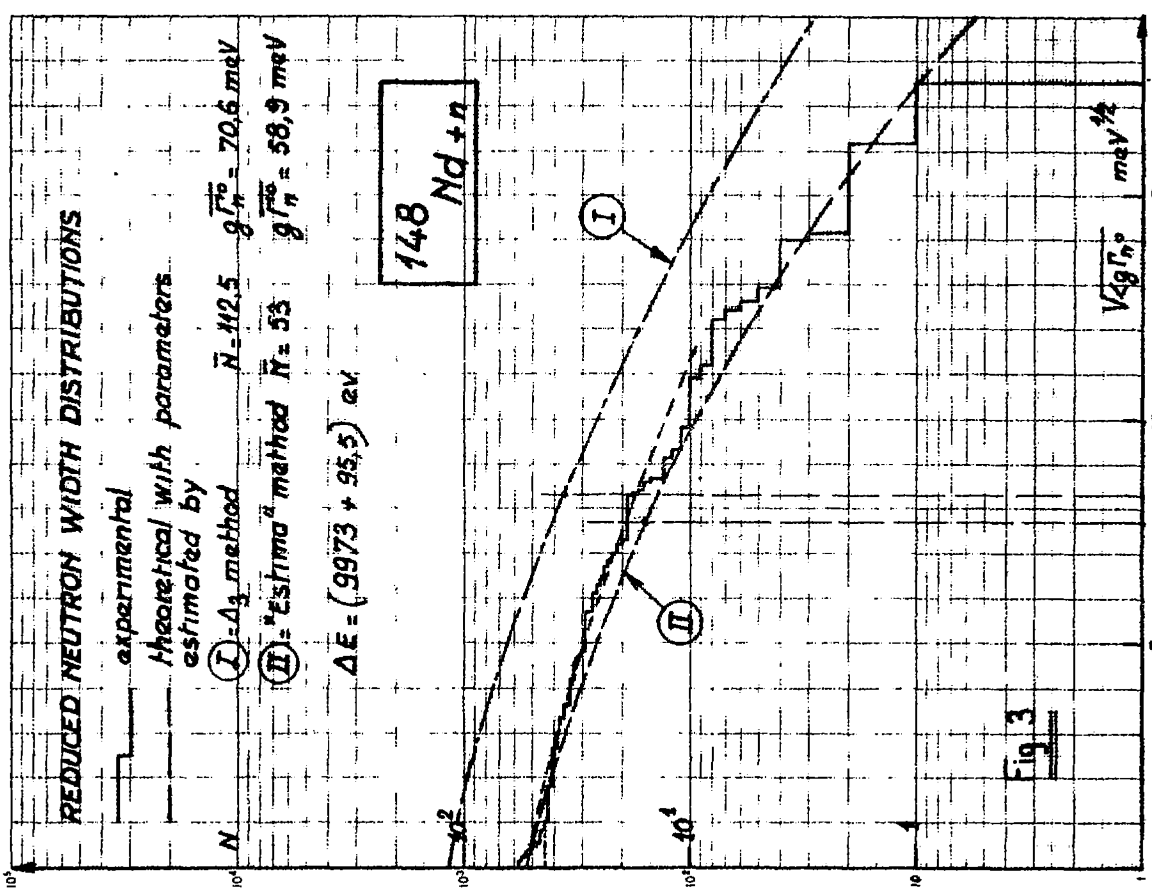

REDUCED NEUTRON WIDTH DISTRIBUTIONS
experimental
theoretical with parameters estimated by
I = Δ3 method N̄ = 112,5 gΓn0 = 70,6 meV
II = "Estima" method N̄ = 55 gΓn0 = 58,9 meV
ΔE = (9973 + 95,5) eV
148 Nd + n
meV 1/2

Fig 3

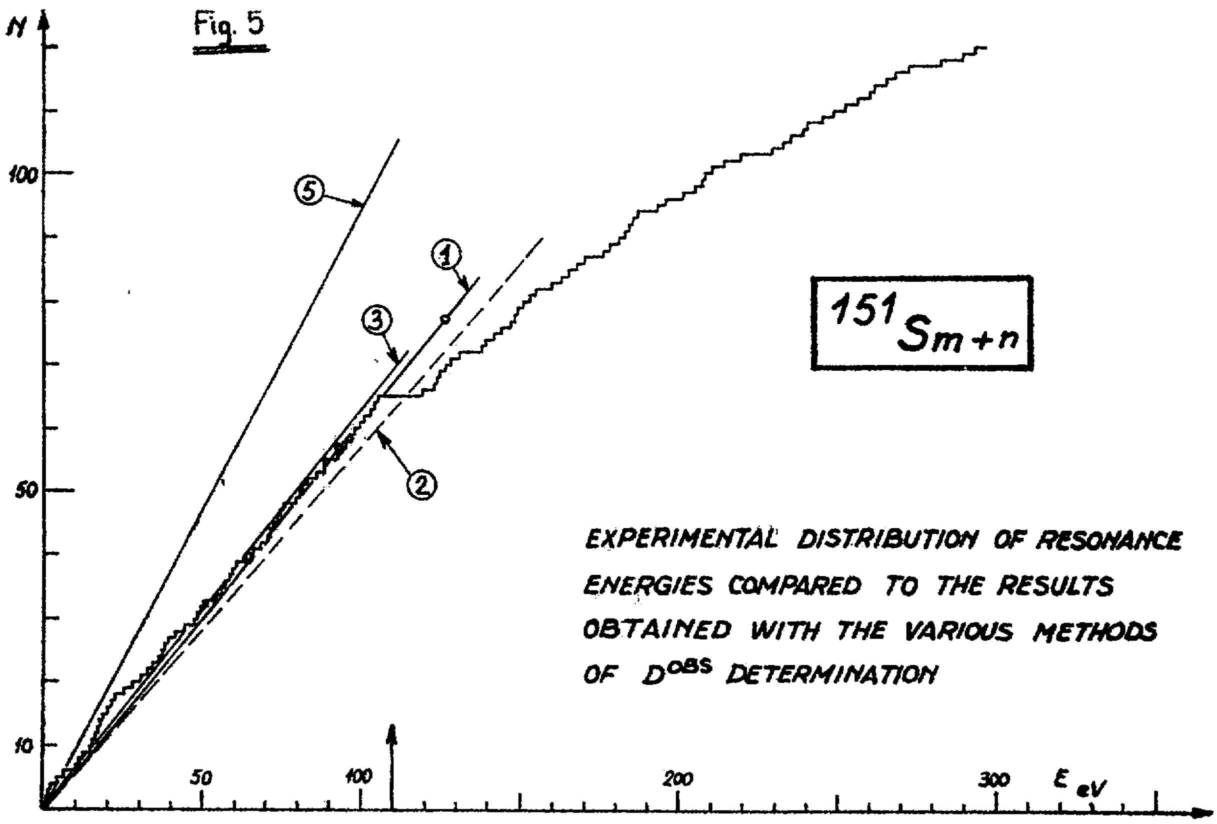
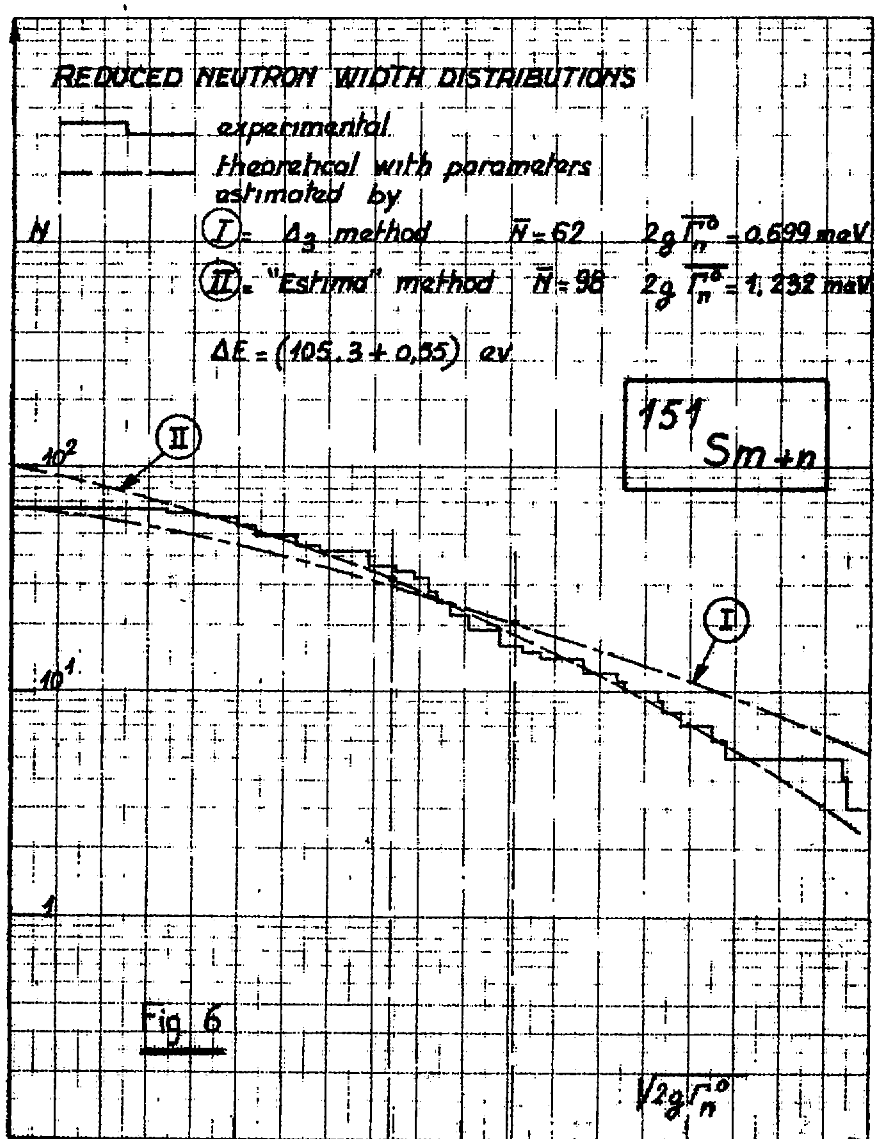
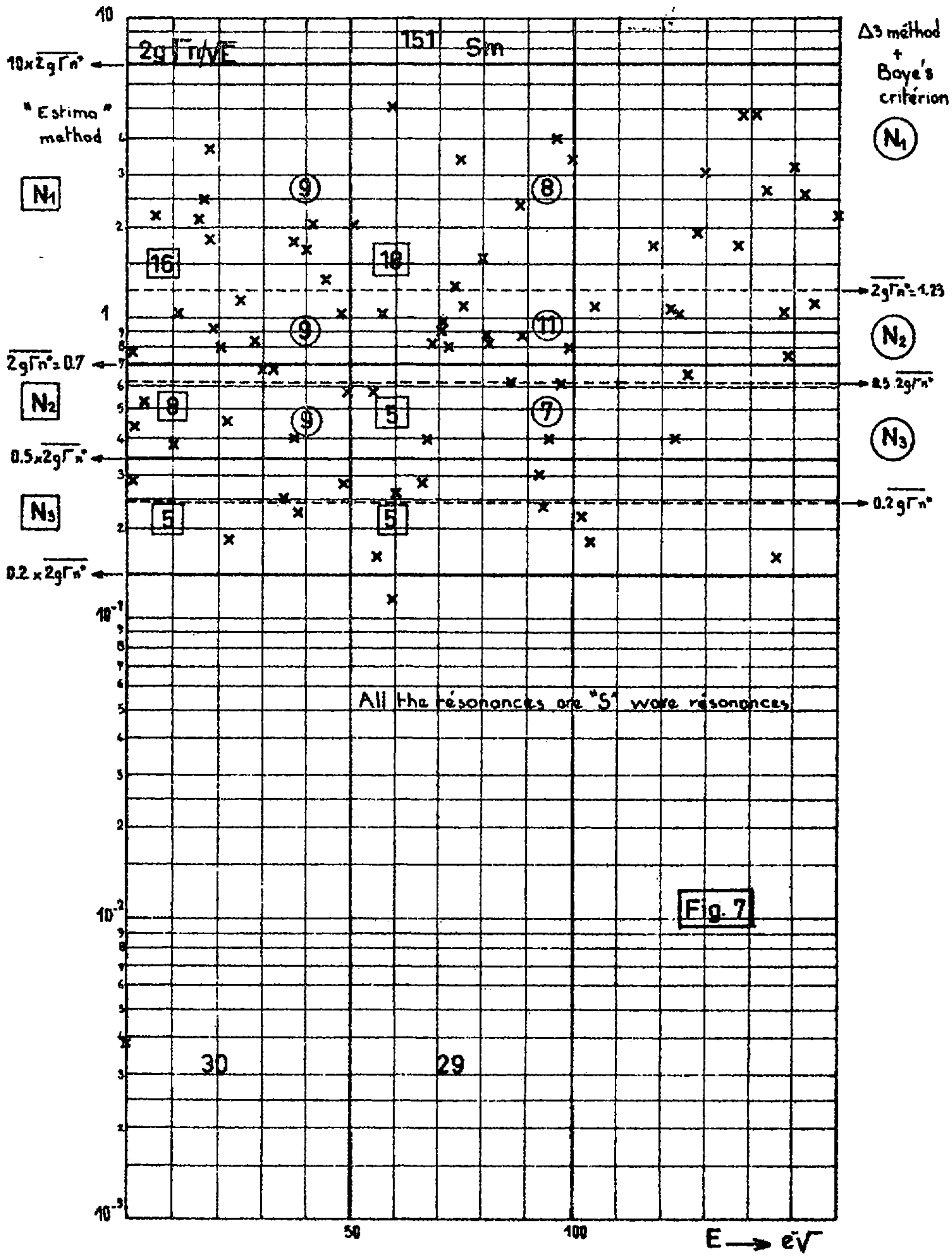


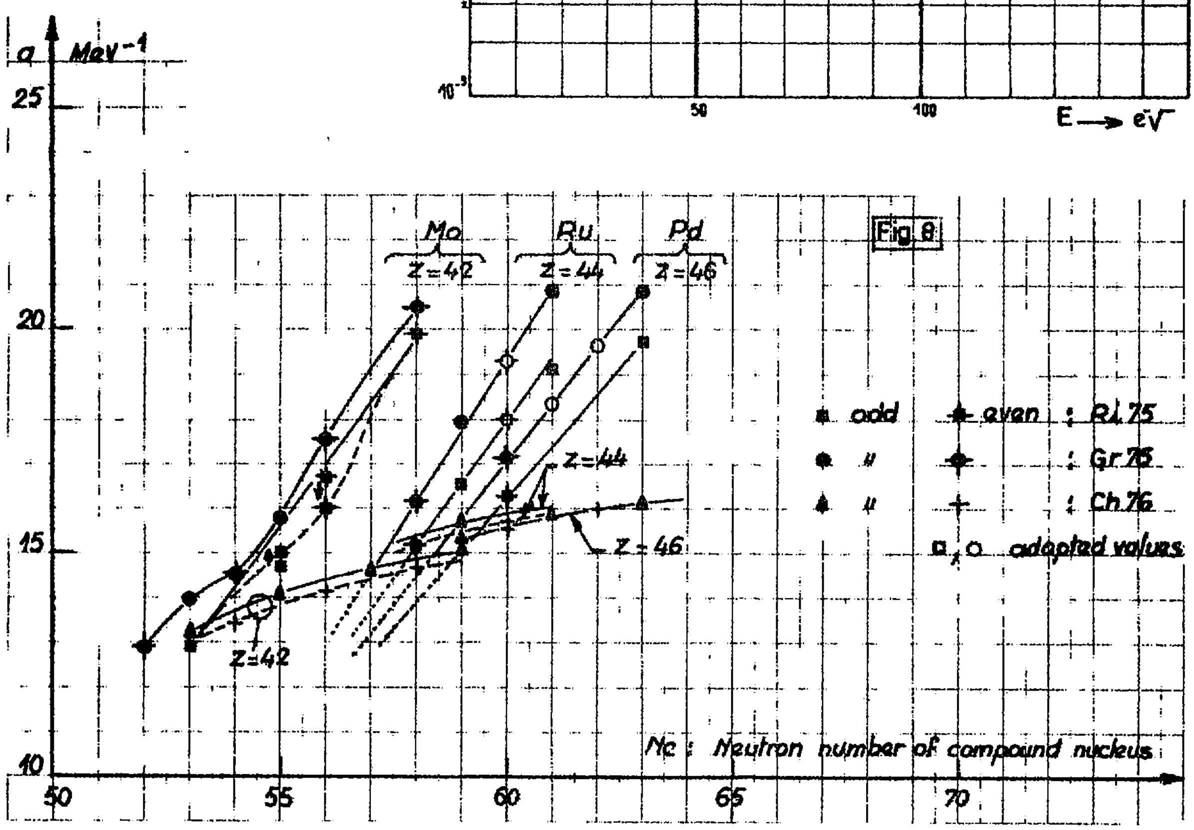
Fig. 5
N
100
50
10
50
100
200
300
E eV
151 Sm+n
EXPERIMENTAL DISTRIBUTION OF RESONANCE ENERGIES COMPARED TO THE RESULTS OBTAINED WITH THE VARIOUS METHODS OF D^OBS DETERMINATION


REDUCED NEUTRON WIDTH DISTRIBUTIONS
experimental
theoretical with parameters estimated by
N
I = Δ3 method N̄ = 62 2g Γn0 = 0.699 meV
II = "Estima" method N̄ = 98 2g Γn0 = 1.232 meV
ΔE = (105.3 ± 0.55) eV
151 Sm+n
10^2
10^1
1
Fig 6
√2gΓn0
0,5
0,836
1
1,11
1,5
meV 1/2



2g Γn/√E
151 Sm
"Estima" method
Δ3 méthod + Boye's critérion
N1
N2
N3
All the résonances are "S" wave résonances
E → eV

Fig. 7



a Mev-1
Mo
Z=42
Ru
Z=44
Pd
Z=46
Z=44
Z=46
Z=42
odd
even : RL 75
: Gr 75
: Ch 76
adapted values
Nc : Neutron number of compound nucleus

Fig. 8

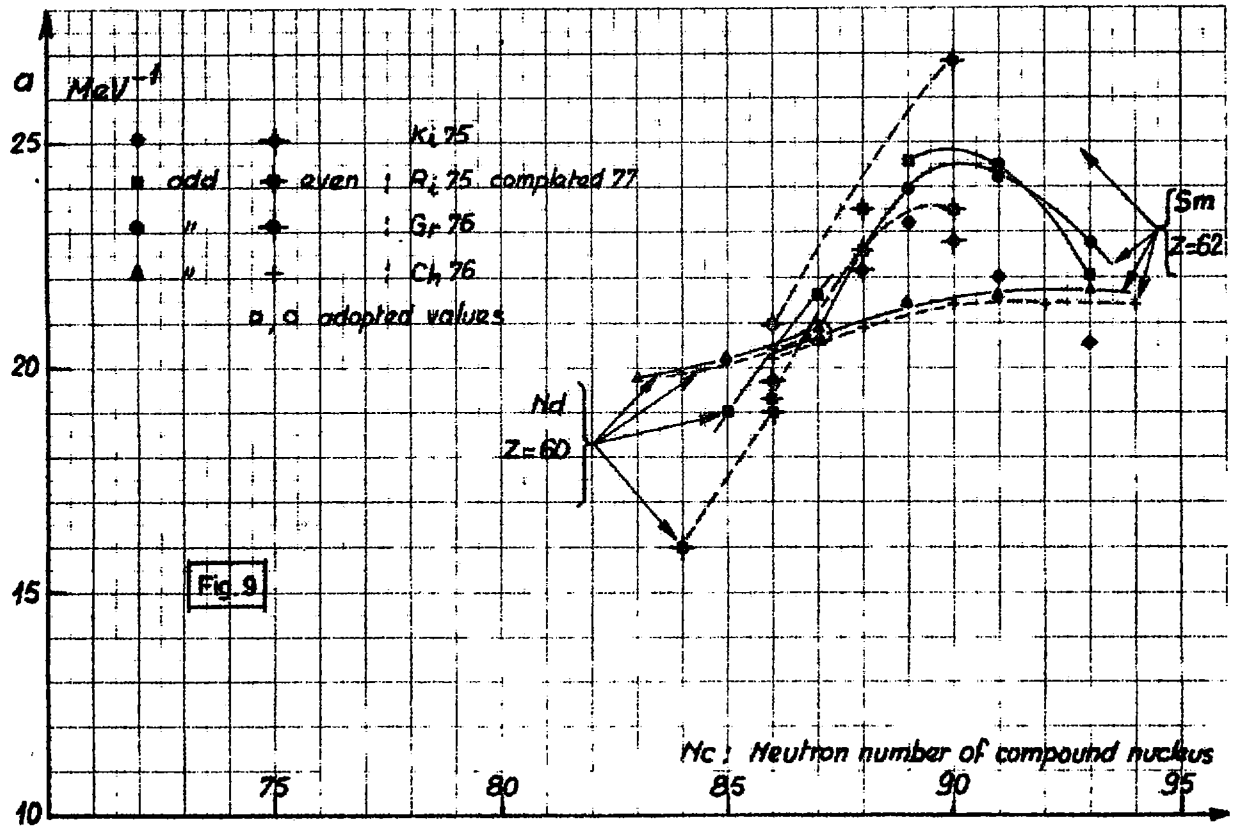

a MeV^-1
25
20
15
10
odd
even
Ki 75
Ri 75 completed 77
Gr 76
Ch 76
a, a adopted values
Nd
Z=60
Sm
Z=62
Nc : Neutron number of compound nucleus
75
80
85
90
95

Fig. 9

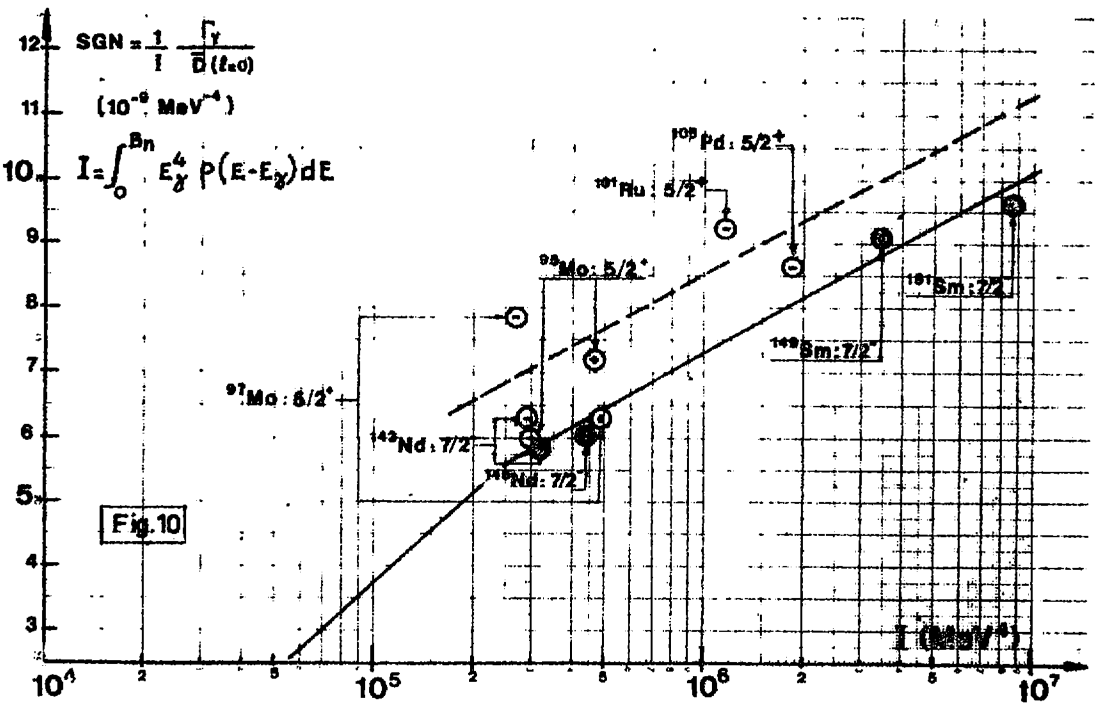

$SGN = \frac{1}{I} \frac{\Gamma_\gamma}{\bar{D}(\ell=0)}$
(10^{-9} MeV^{-4})
$I = \int_0^{B_n} E_\gamma^4 \, \rho(E-E_\gamma) \, dE$
^{105}Pd : 5/2+
^{101}Ru : 5/2+
^{95}Mo : 5/2+
^{97}Mo : 5/2+
^{143}Nd : 7/2-
^{145}Nd : 7/2-
^{151}Sm : 7/2-
^{149}Sm : 7/2-
I (MeV^4)
10^4
10^5
10^6
10^7

Fig. 10

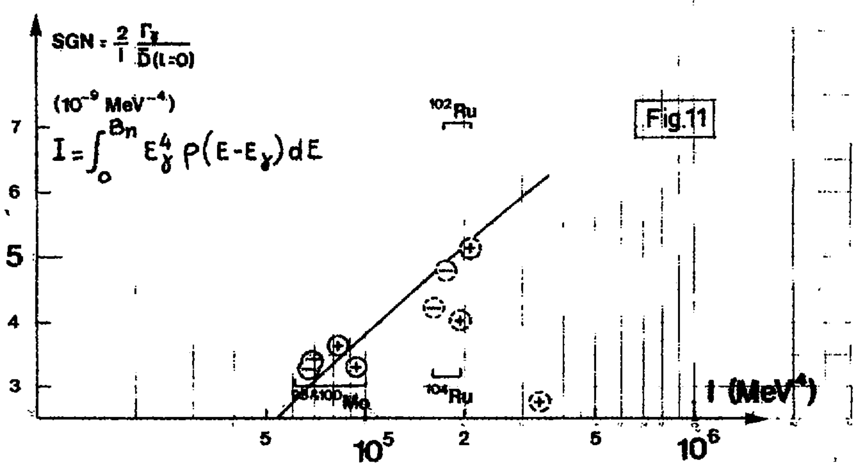

SGN = $\frac{2}{I}\frac{\Gamma_\gamma}{\bar{D}(l=0)}$
(10^{-9} MeV^{-4})
$I=\int_0^{B_n} E_\gamma^4\,\rho(E-E_\gamma)\,dE$
^{102}Ru
^{104}Ru
$^{98+100}$Mo
I (MeV^4)
7
6
5
4
3
5
10^5
2
5
10^6

Fig.11

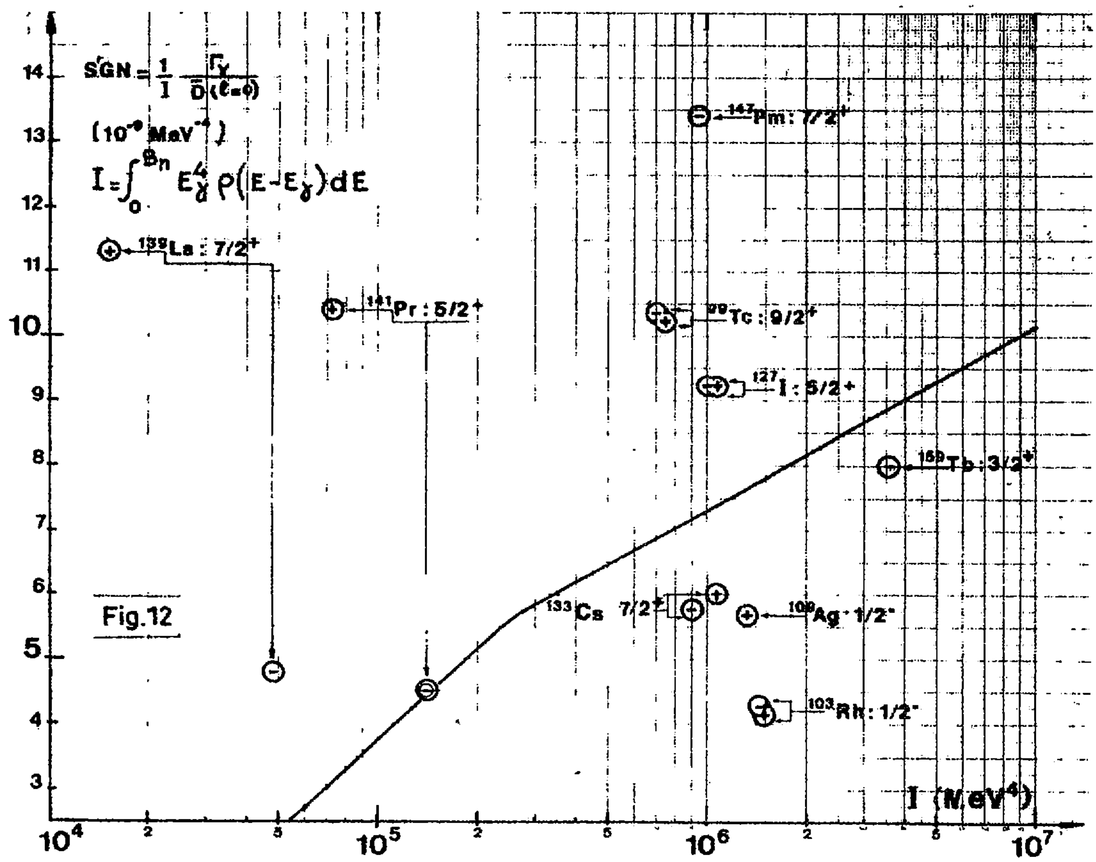
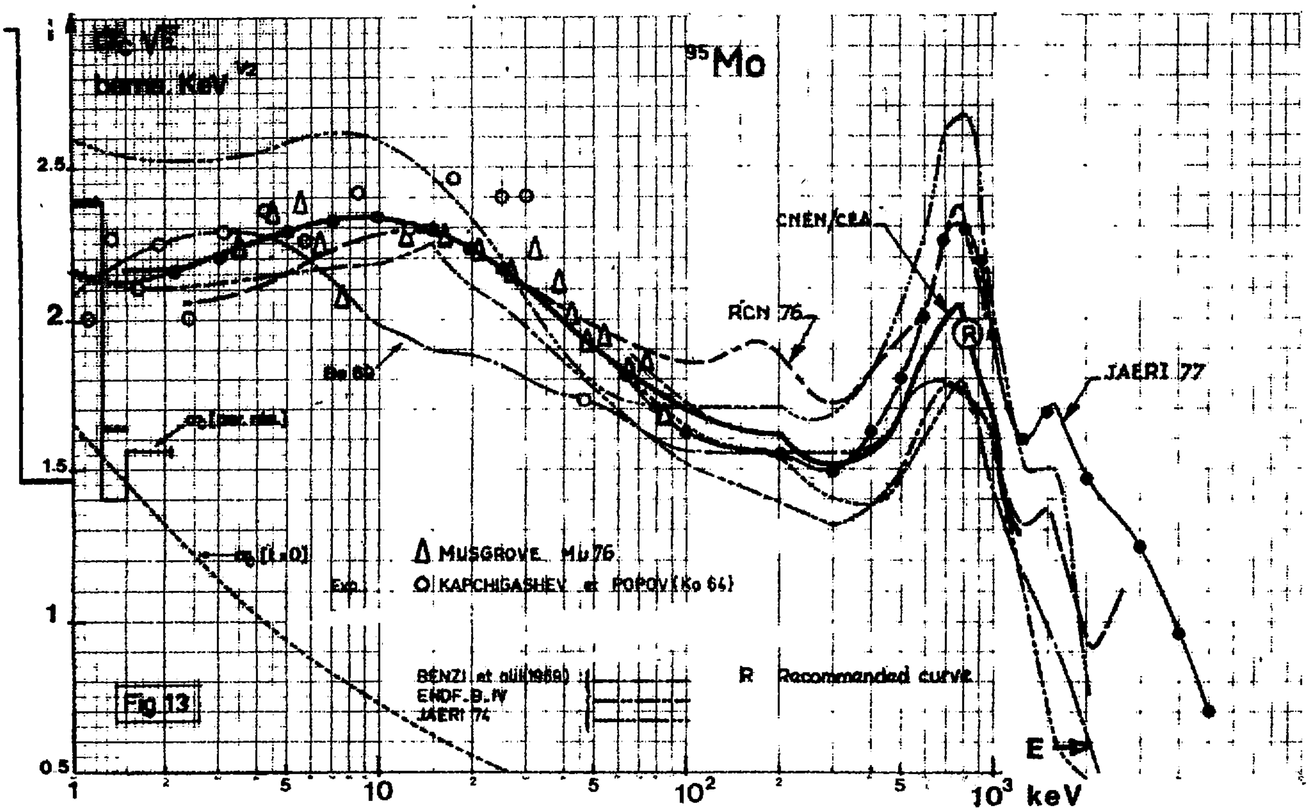
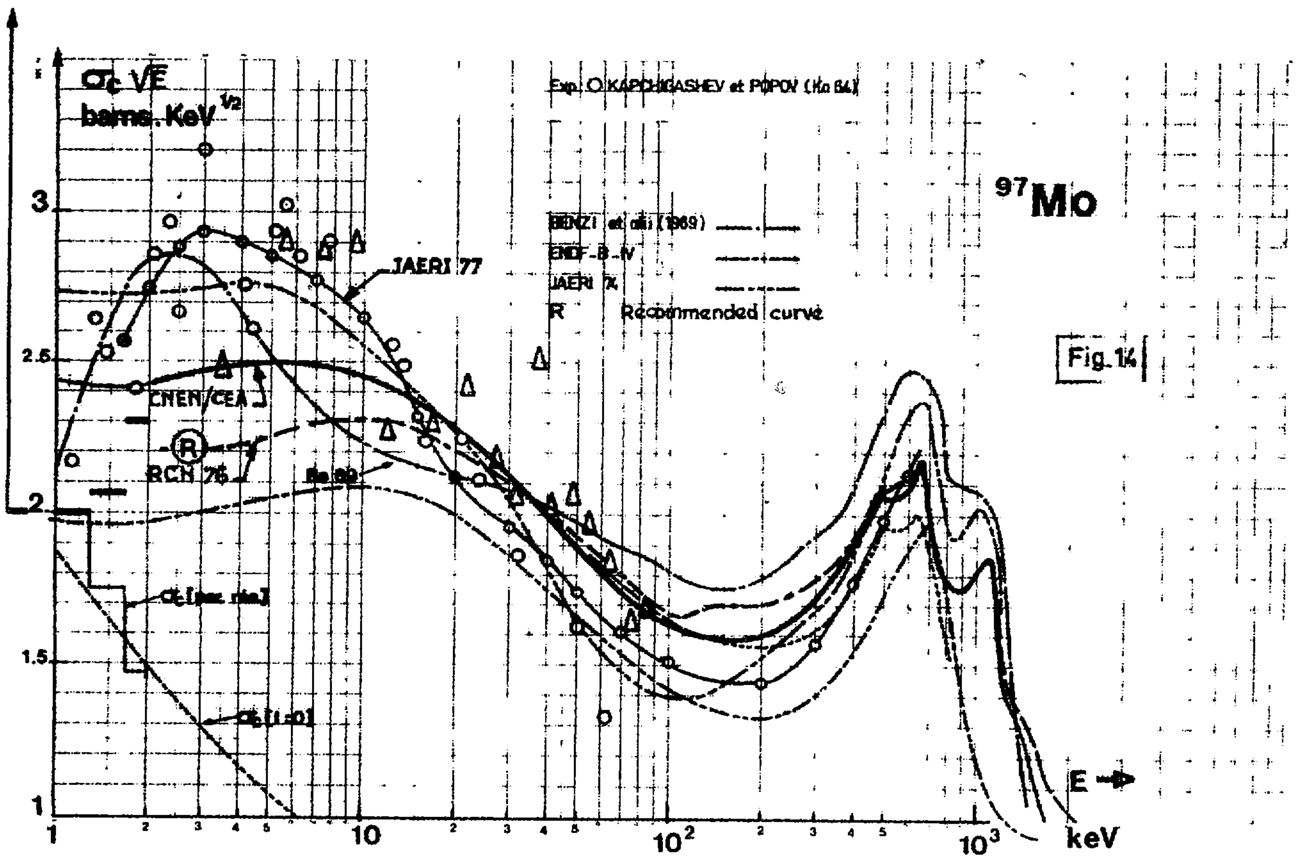

SGN = $\frac{1}{I}\frac{\Gamma_\gamma}{\bar{D}(\ell=0)}$
(10^{-9} MeV^{-4})
$I=\int_0^{B_n} E_\gamma^4\,\rho(E-E_\gamma)\,dE$
^{139}La : 7/2$^+$
^{141}Pr : 5/2$^+$
^{147}Pm : 7/2$^+$
^{99}Tc : 9/2$^+$
^{127}I : 5/2$^+$
^{159}Tb : 3/2$^+$
^{133}Cs 7/2$^+$
^{109}Ag : 1/2$^-$
^{103}Rh : 1/2$^-$
I (MeV^4)
14
13
12
11
10
9
8
7
6
5
4
3
10^4
2
5
10^5
2
5
10^6
2
5
10^7

Fig.12



$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
^{95}Mo
CNEN/CEA
RCN 76
JAERI 77
Be 69
σ_c [par. min.]
σ_c (l = 0)
Δ MUSGROVE Mu 76
Exp. O KAPCHIGASHEV et POPOV (Ko 64)
BENZI et alii (1969)
ENDF.B.IV
JAERI 74
R Recommanded curve
Fig 13
E
10
10^2
10^3 keV



$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
Exp. O KAPCHIGASHEV et POPOV (Ko 64)
^{97}Mo
BENZI et alii (1969)
ENDF-B-IV
JAERI 74
R Recommended curve
JAERI 77
CNEN/CEA
RCN 76
Be 69
Fig.14
σ_c [par. min.]
σ_c (l = 0)
E
10
10^2
10^3
keV

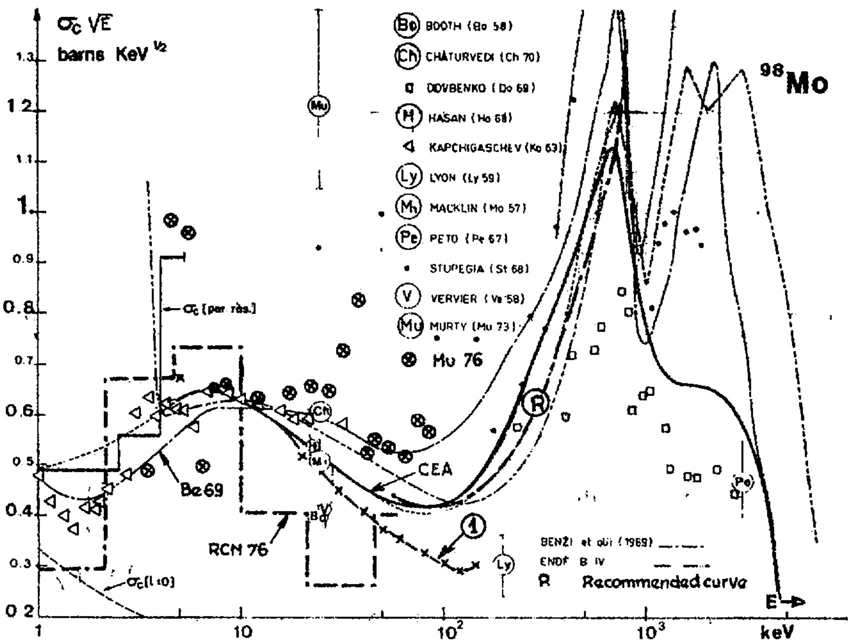


Fig. 15

① Curve calculated with recommended average parameters

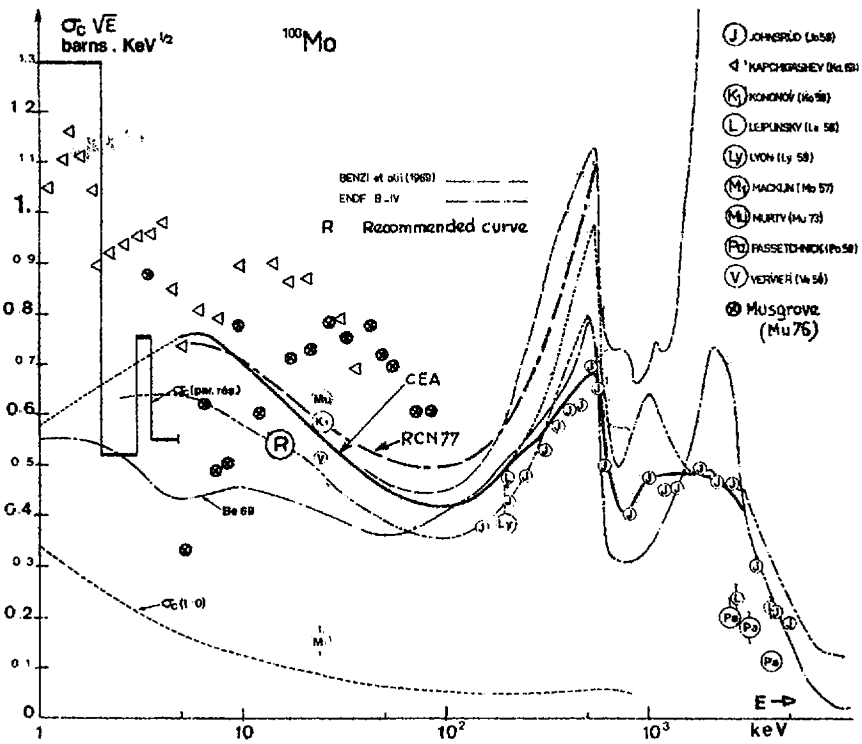
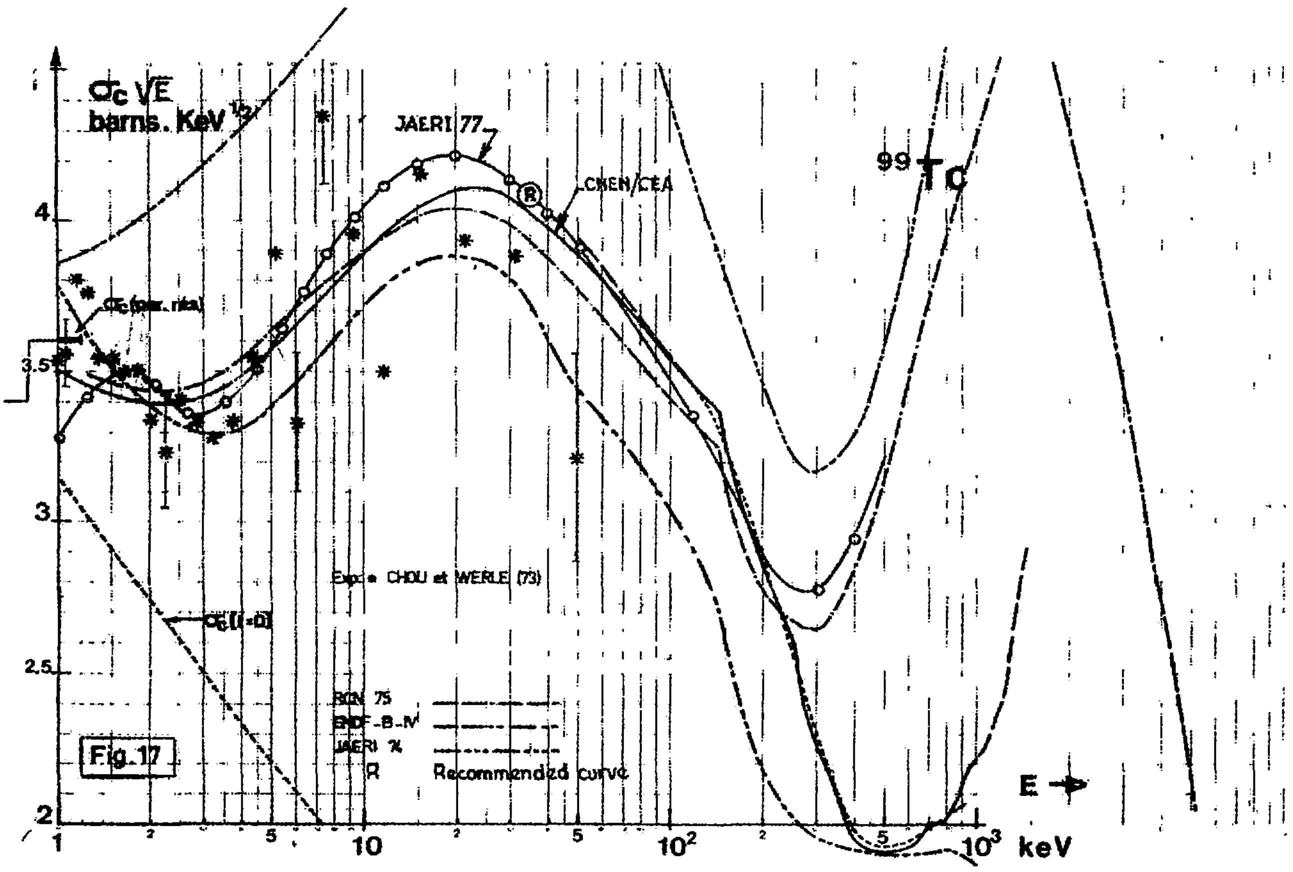
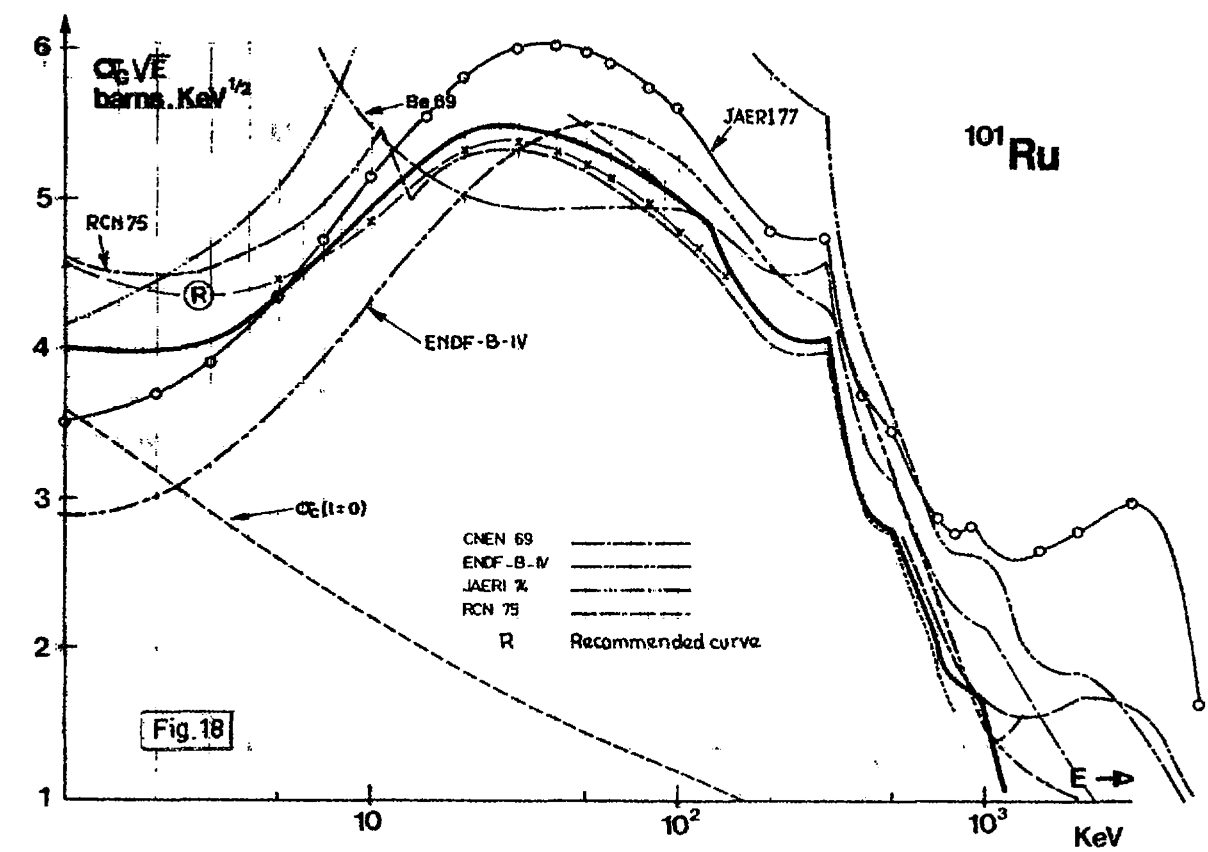


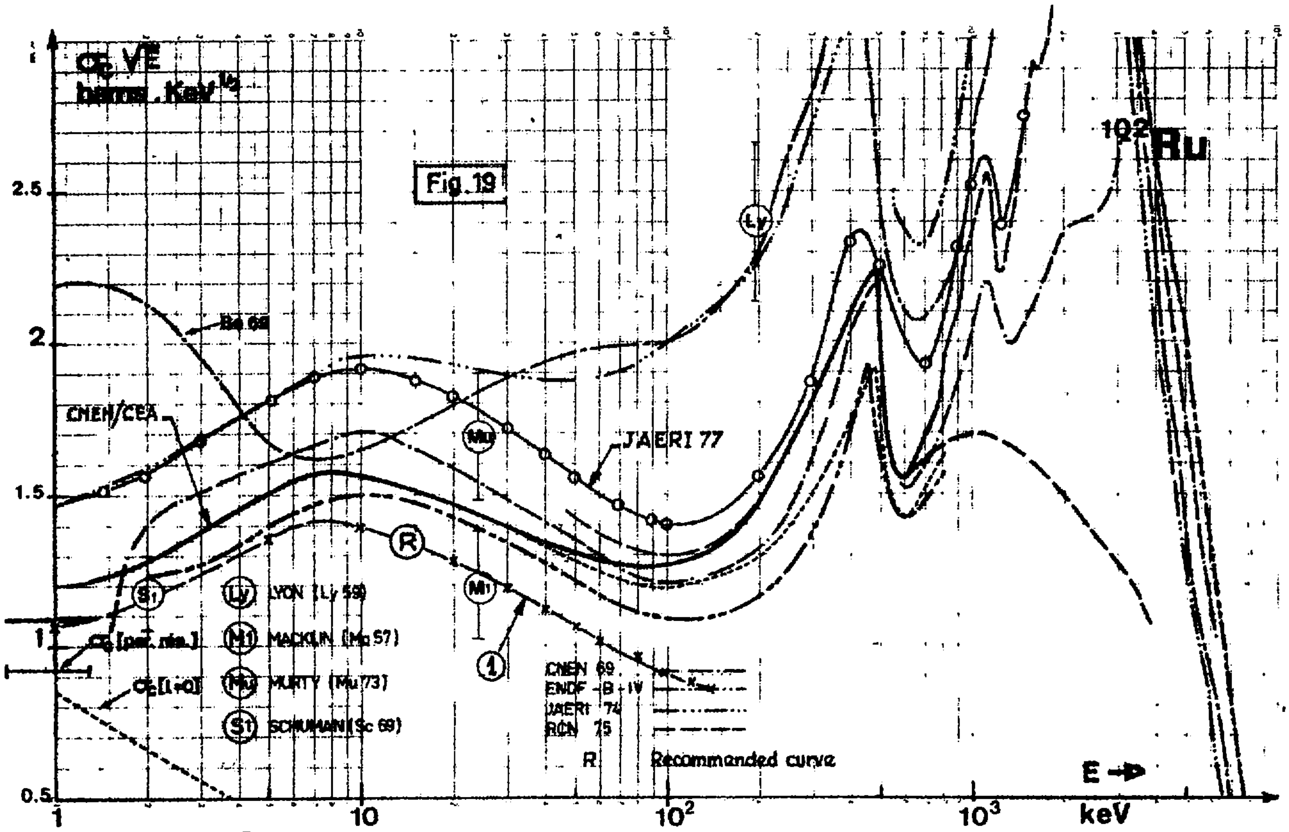
Fig. 16



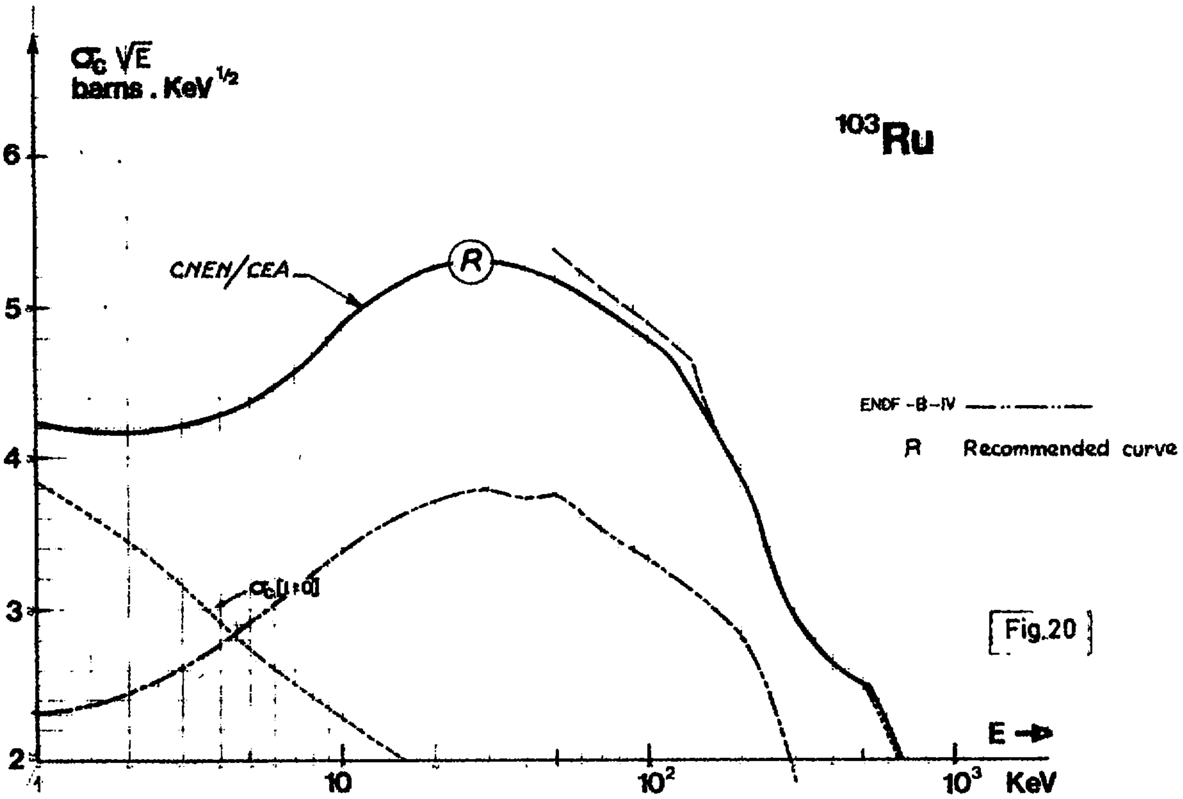
$\sigma_c\sqrt{E}$ barns. KeV$^{1/2}$
JAERI 77
CNEN/CEA
99 Tc
σ_c(par. nits)
Exp: * CHOU et WERLE (73)
σ_c(l=0)
RCN 75
ENDF-B-IV
JAERI 76
R Recommended curve
Fig.17
E
keV

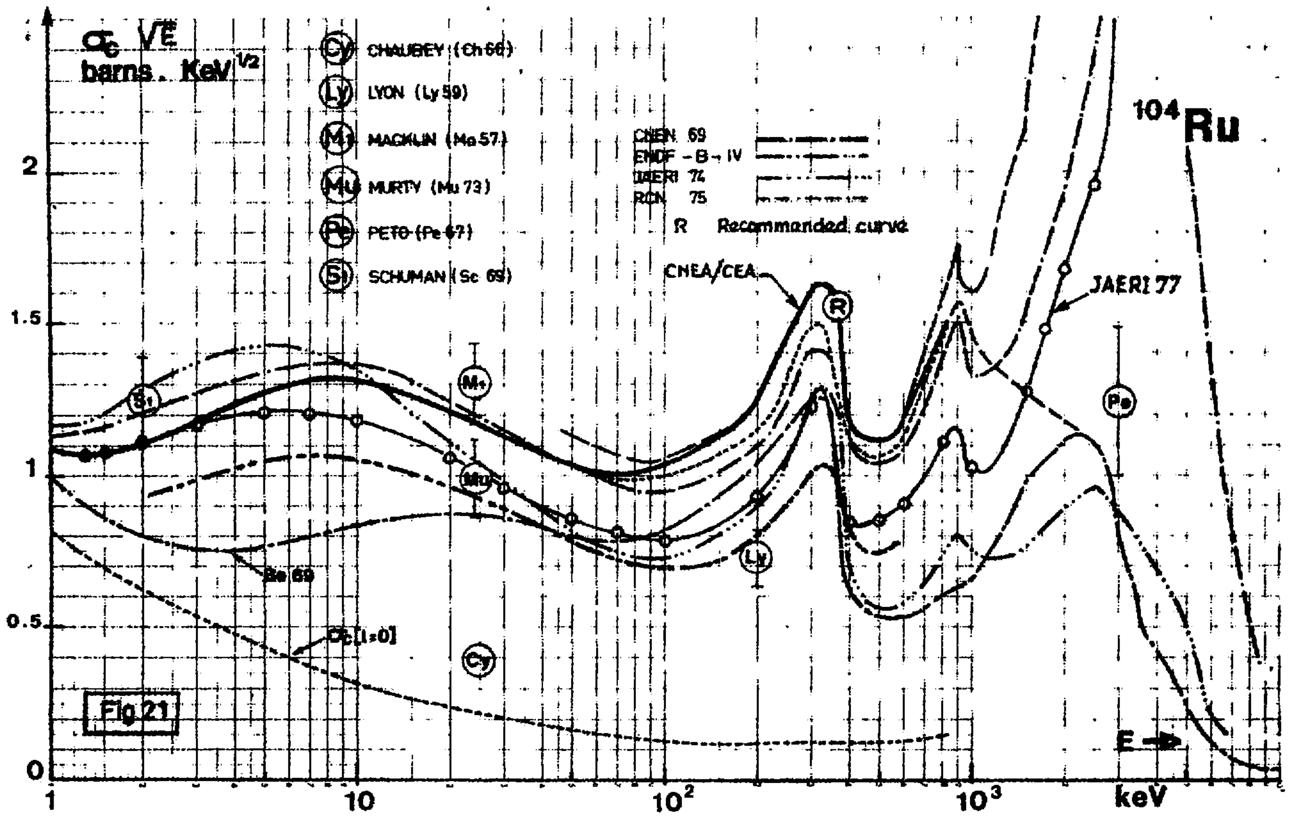


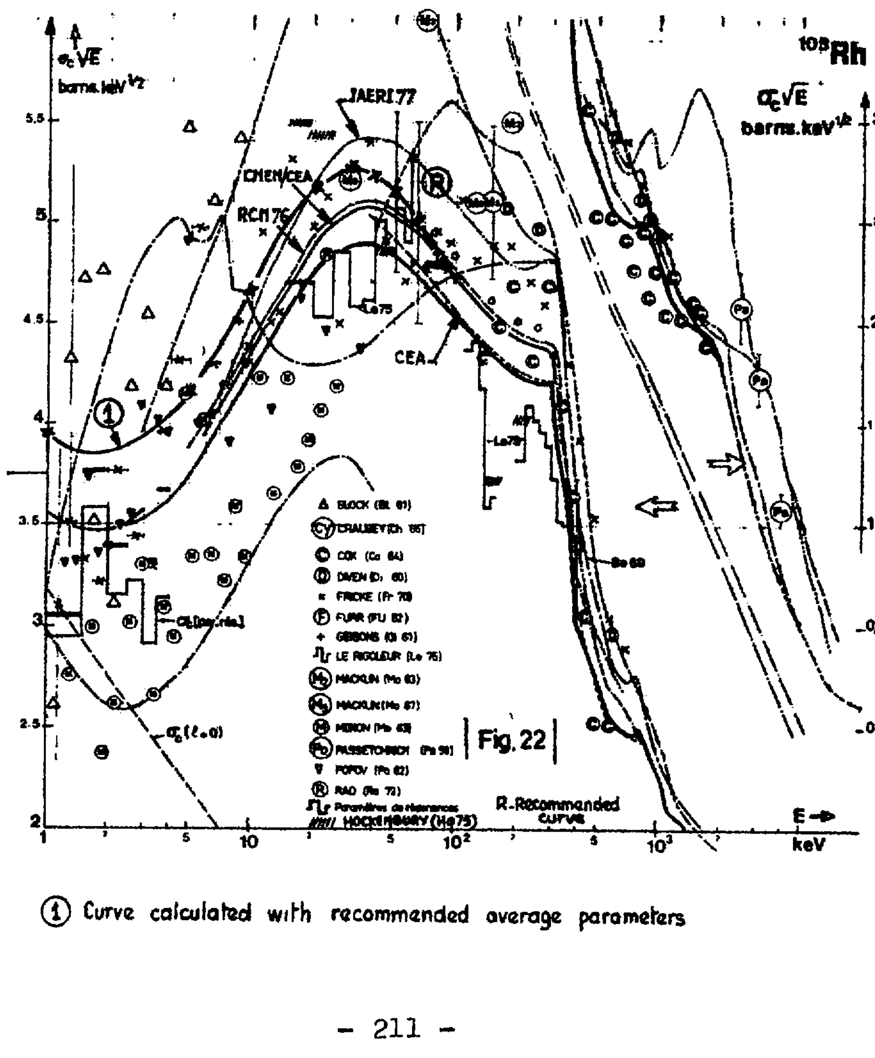
$\sigma_c\sqrt{E}$ barns. KeV$^{1/2}$
Be 69
JAERI 77
101 Ru
RCN 75
ENDF-B-IV
σ_c(l=0)
CNEN 69
ENDF-B-IV
JAERI 76
RCN 75
R Recommended curve
Fig.18
E
KeV



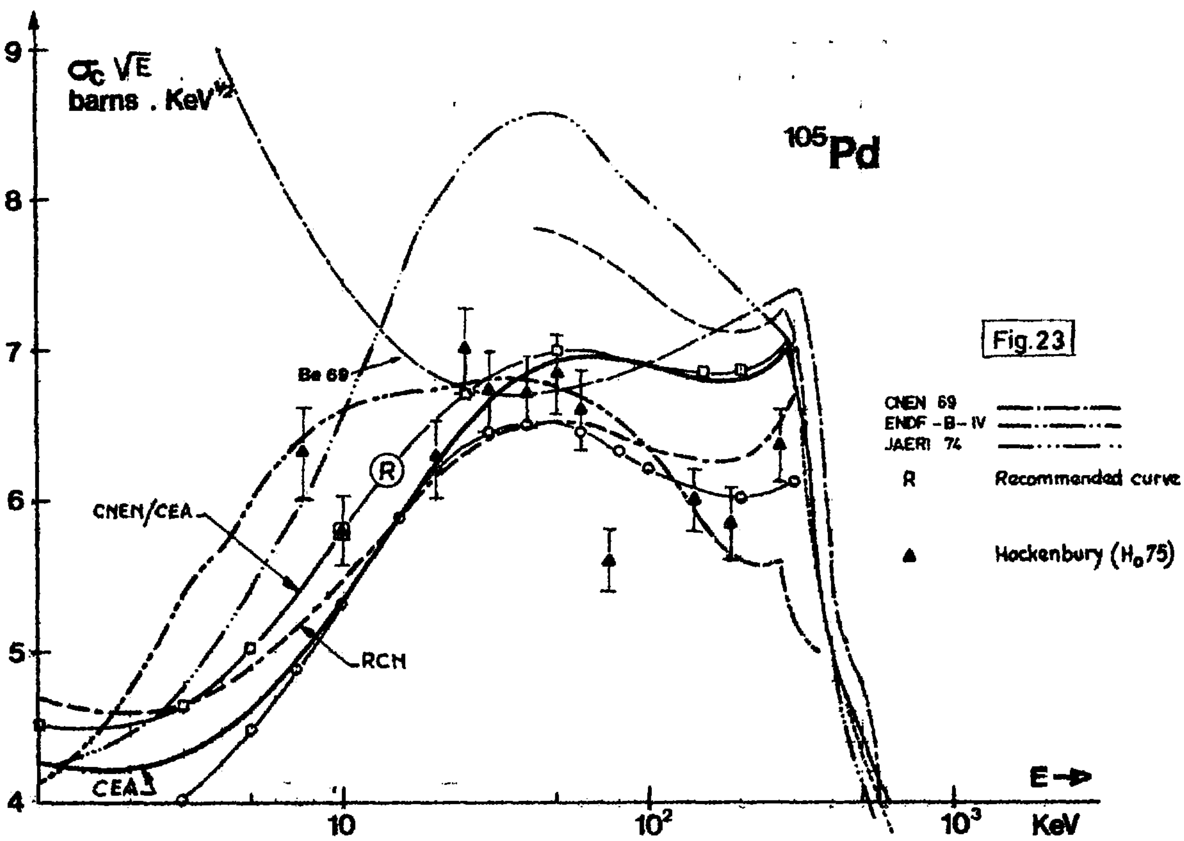
① curve calculated with recommended average parameters



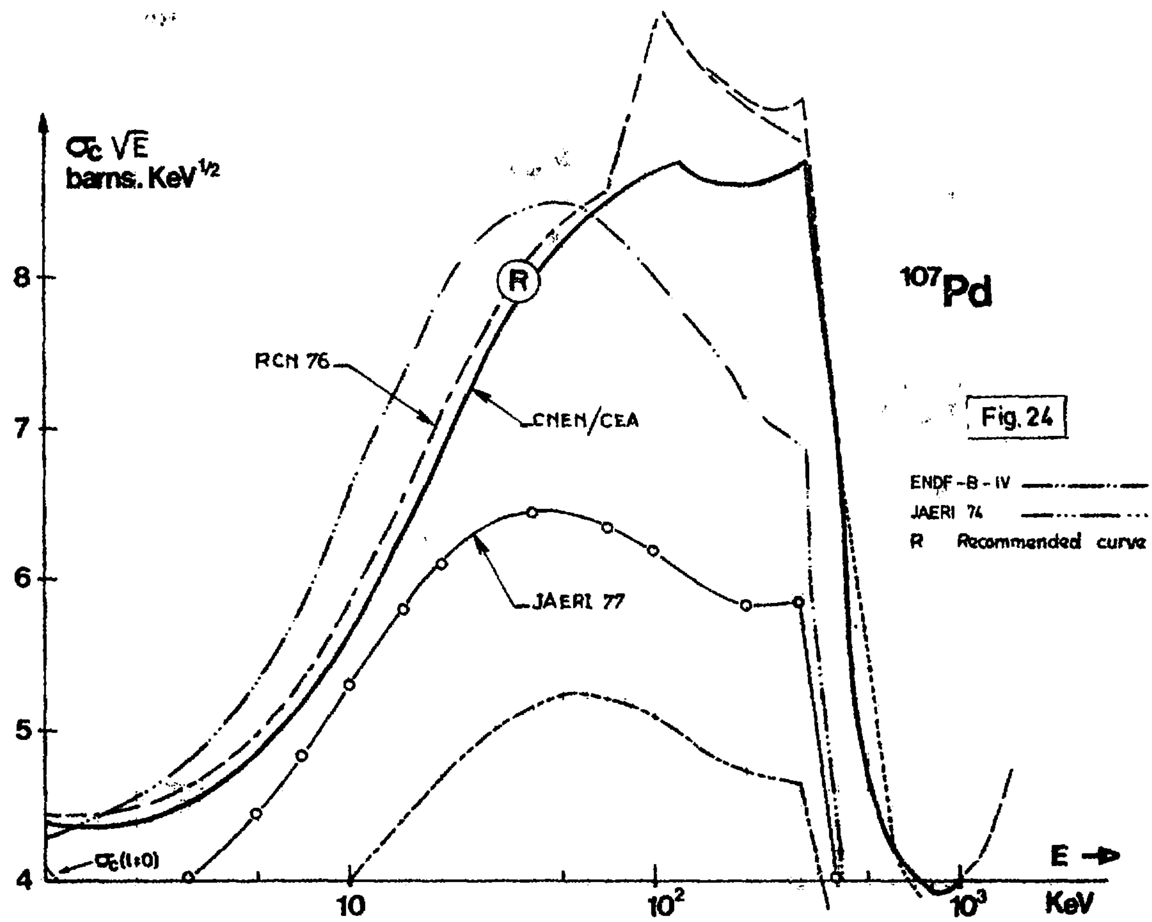




(1) Curve calculated with recommended average parameters



$\sigma_c \sqrt{E}$ barns . KeV$^{1/2}$
^{105}Pd
Fig.23
CNEN 69
ENDF - B - IV
JAERI 74
R Recommended curve
Hockenbury (H_o75)
Be 69
CNEN/CEA
RCN
CEA
E
10
10^2
10^3
KeV



$\sigma_c \sqrt{E}$ barns. KeV$^{1/2}$
^{107}Pd
Fig. 24
ENDF - B - IV
JAERI 74
R Recommended curve
RCN 76
CNEN/CEA
JAERI 77
$\sigma_c(l=0)$
E
10
10^2
10^3
KeV

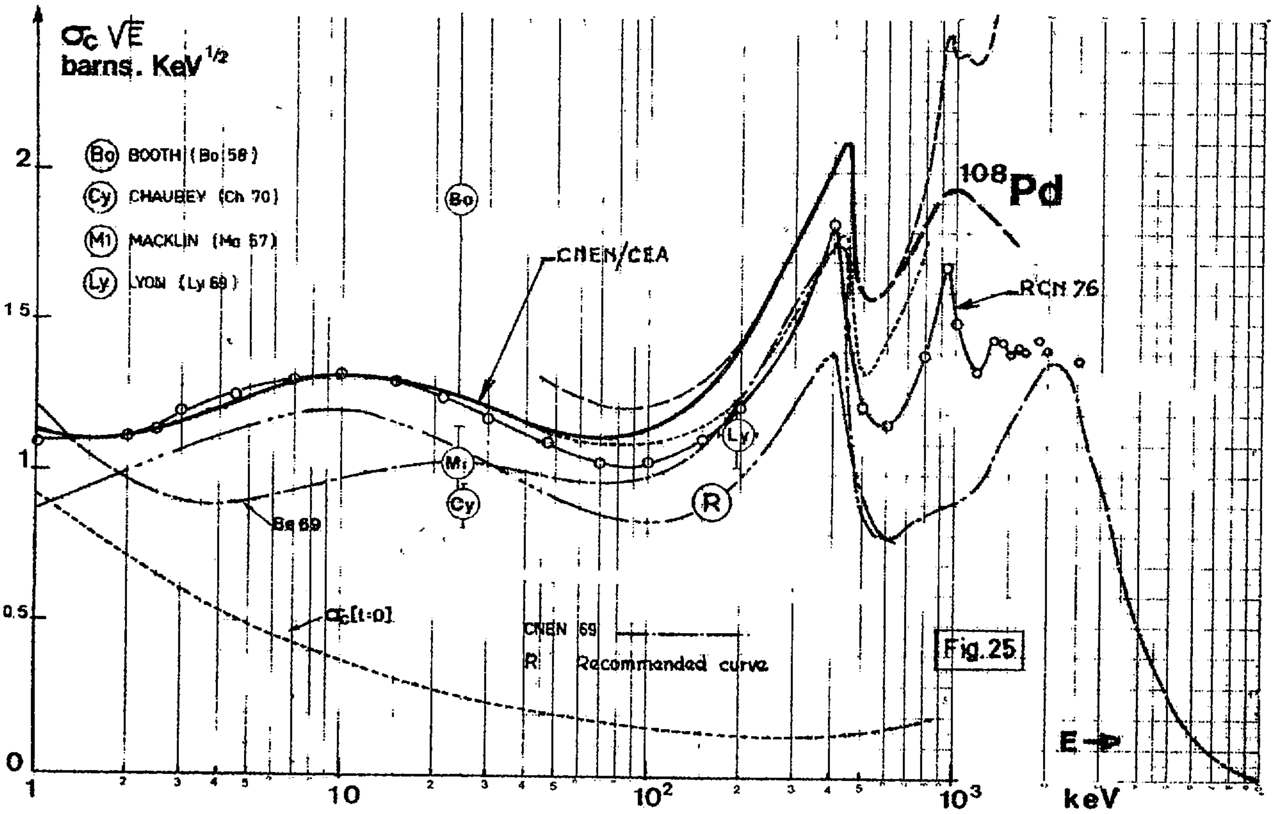

$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
Bo BOOTH (Bo 58)
Cy CHAUBEY (Ch 70)
M1 MACKLIN (Ma 67)
Ly LYON (Ly 69)
CHEN/CEA
^{108}Pd
RCN 76
Be 69
σ_c[l:0]
CNEN 69
R Recommended curve
E
keV

Fig.25

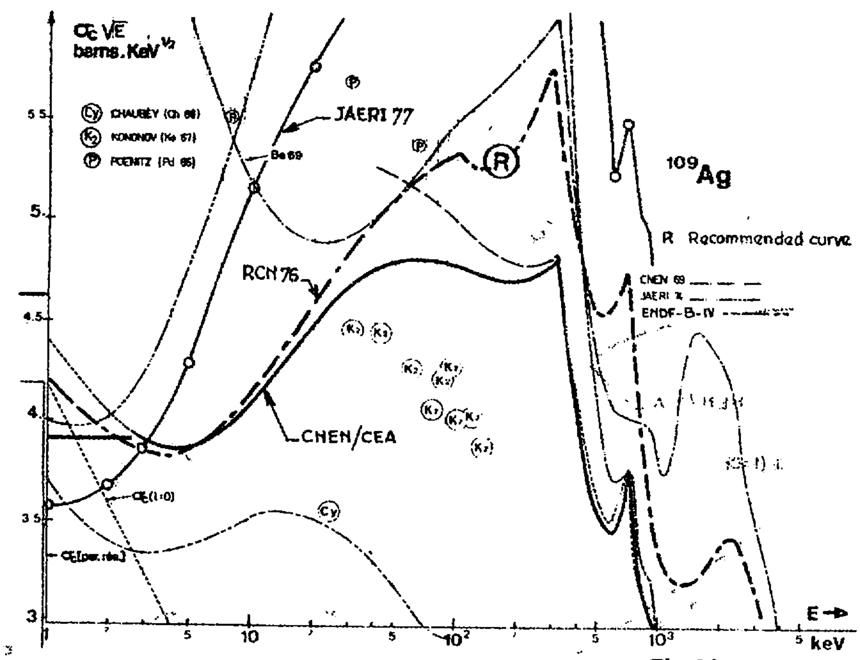
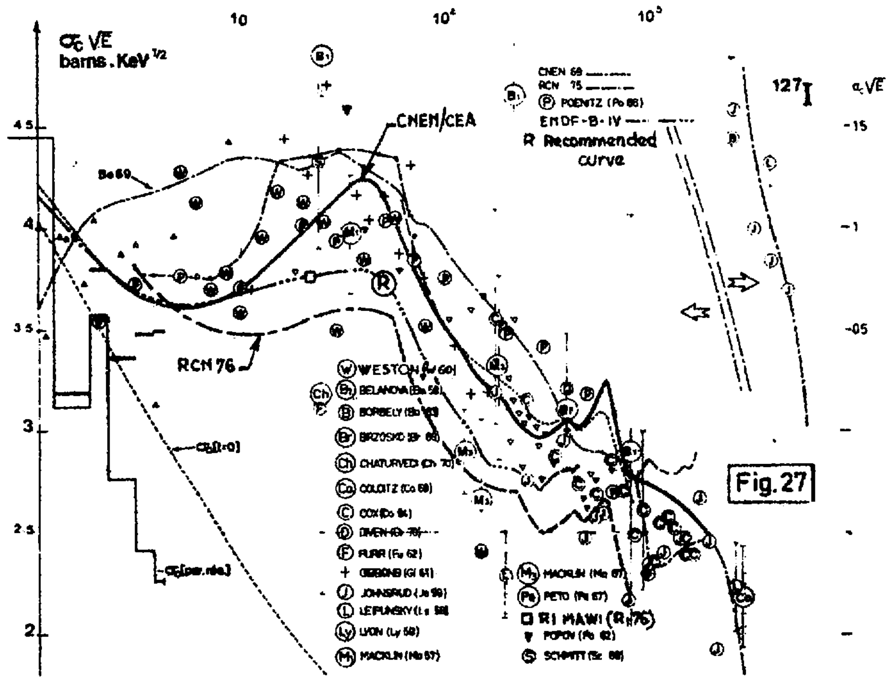
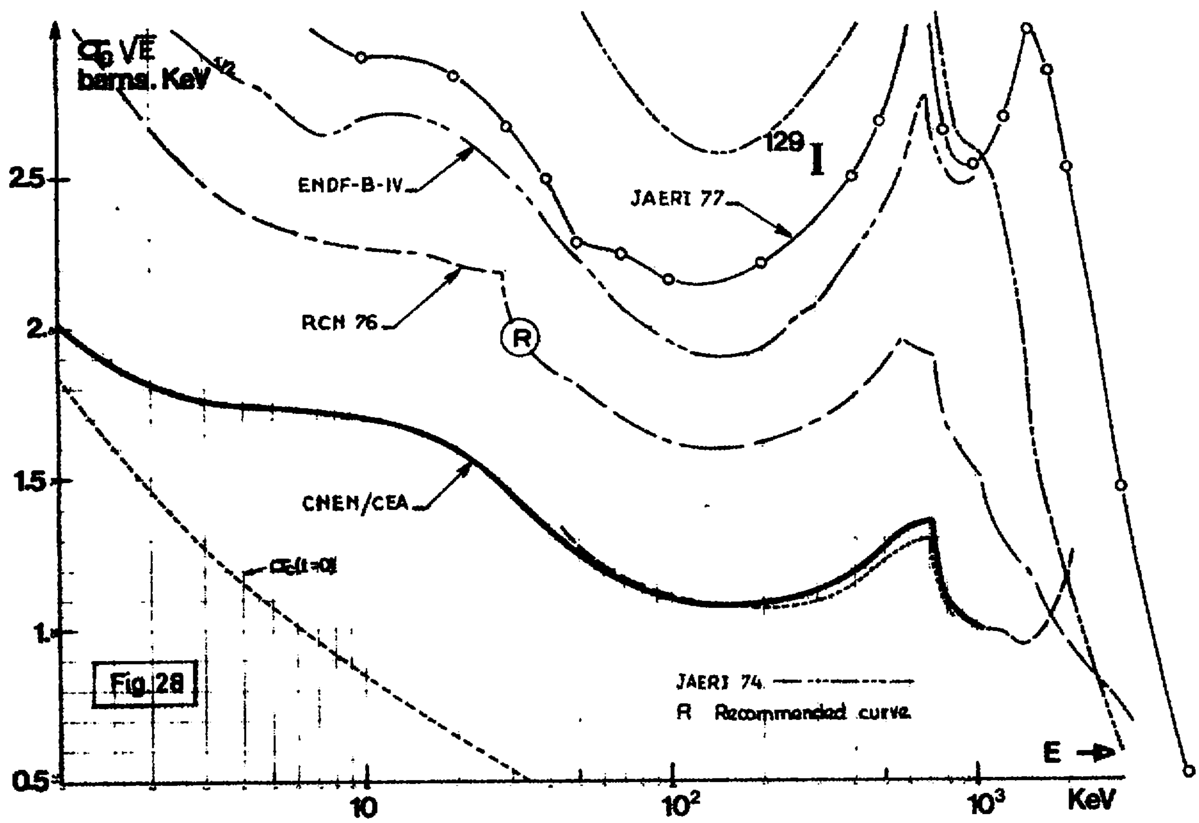

$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
JAERI 77
Be 69
RCN 76
CNEN/CEA
^{109}Ag
R Recommended curve
CNEN 69
ENDF-B-IV
E
keV

Fig.26


$\sigma_c \sqrt{E}$ barns . KeV$^{1/2}$
CNEN/CEA
RCN 76
ENDF-B-IV
R Recommended curve
127I
Fig.27


$\sigma_c \sqrt{E}$ barns. KeV$^{1/2}$
129I
ENDF-B-IV
JAERI 77
RCN 76
CNEN/CEA
$\sigma_c(l=0)$
JAERI 74
R Recommended curve
E
KeV
Fig.28

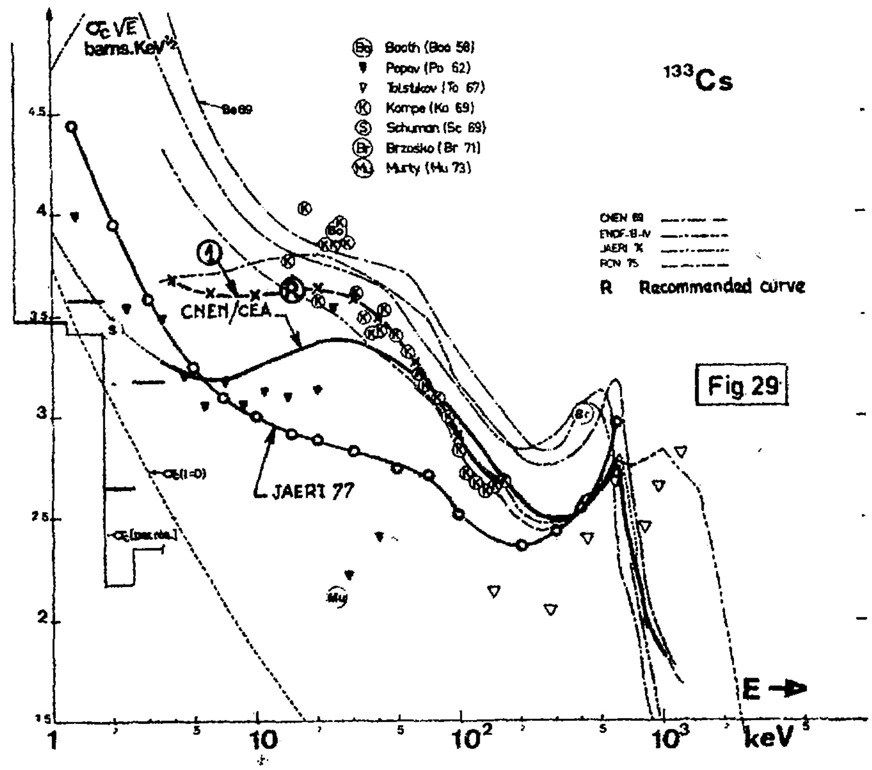


Fig 29

① Curve calculated with recommended average parameters

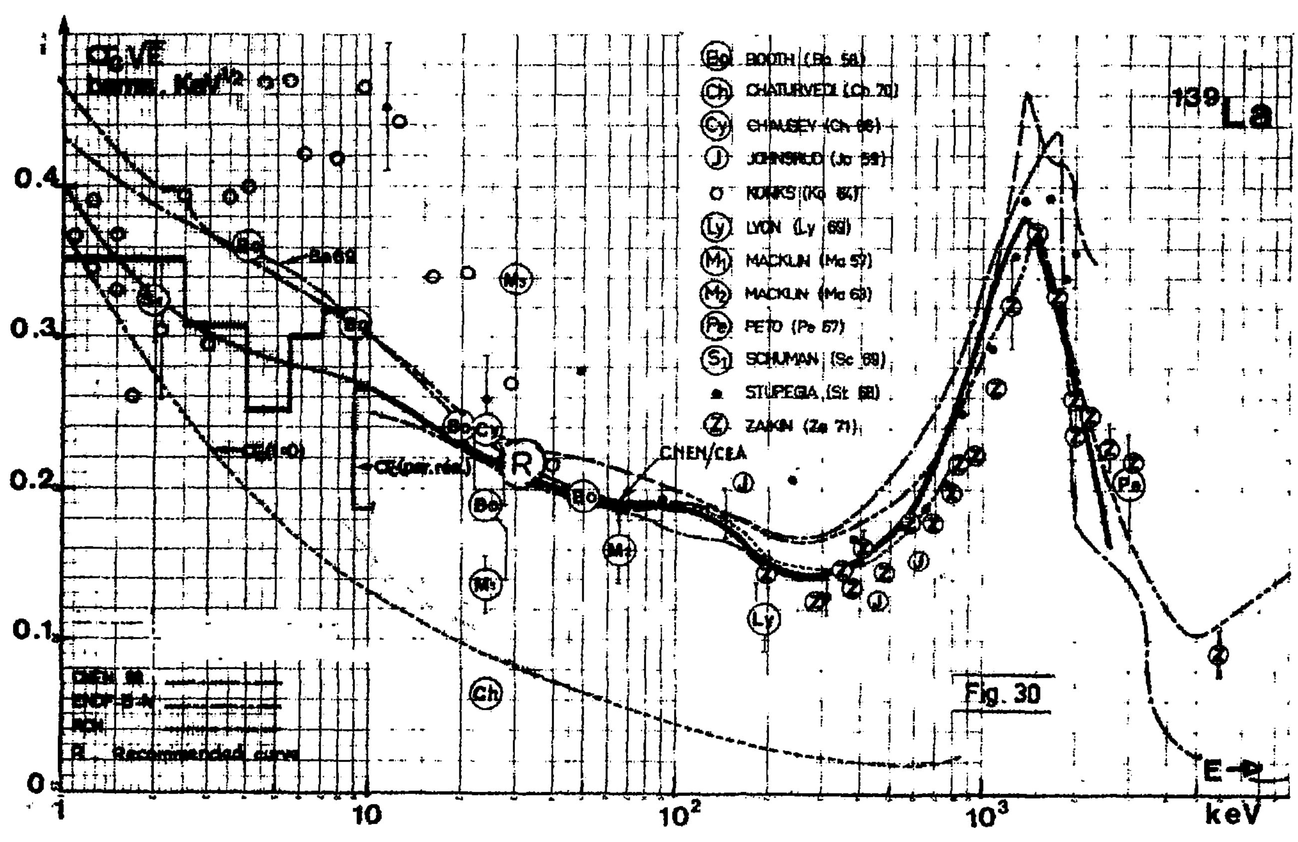


Fig. 30

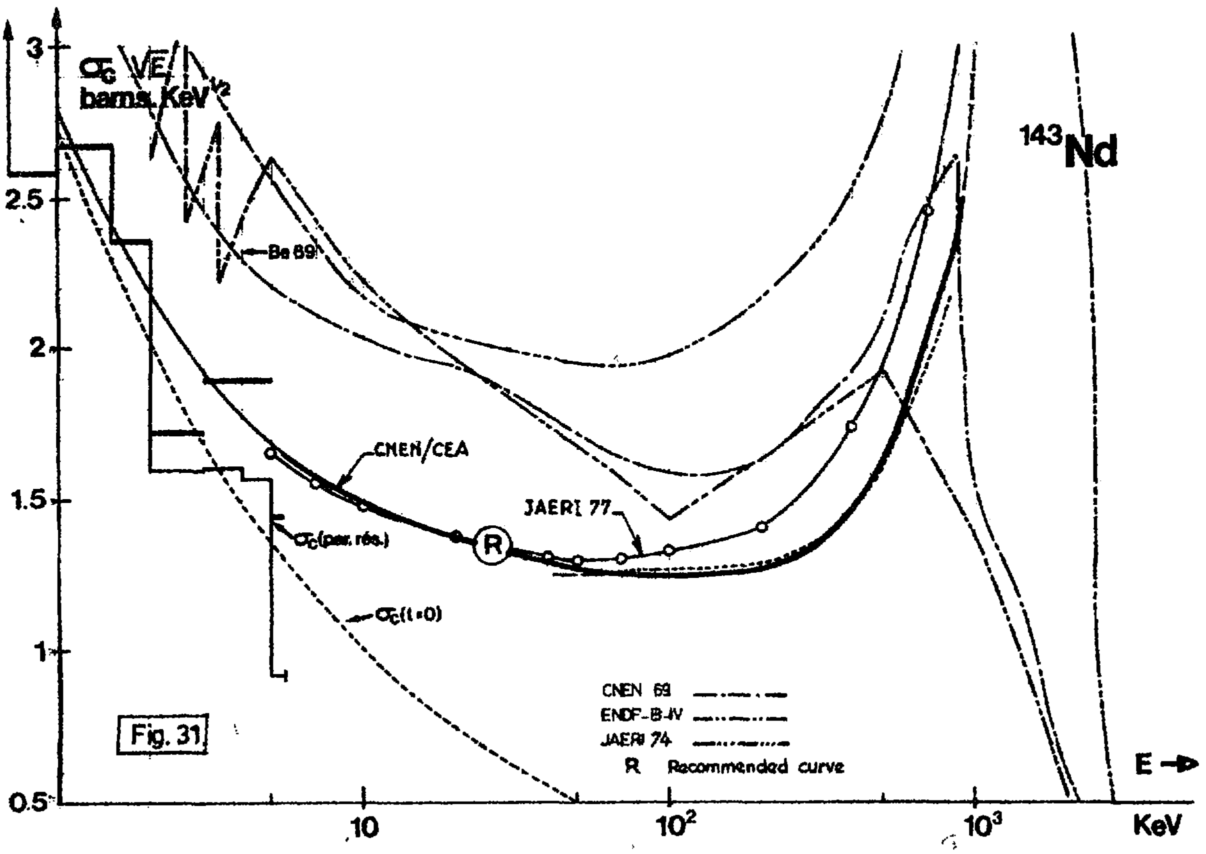

3
2.5
2
1.5
1
0.5
$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
^{143}Nd
Be 69
CNEN/CEA
JAERI 77
R
σ_c(par. rés.)
σ_c(l=0)
CNEN 69
ENDF-B-IV
JAERI 74
R Recommended curve
E
10
10^2
10^3
KeV

Fig. 31

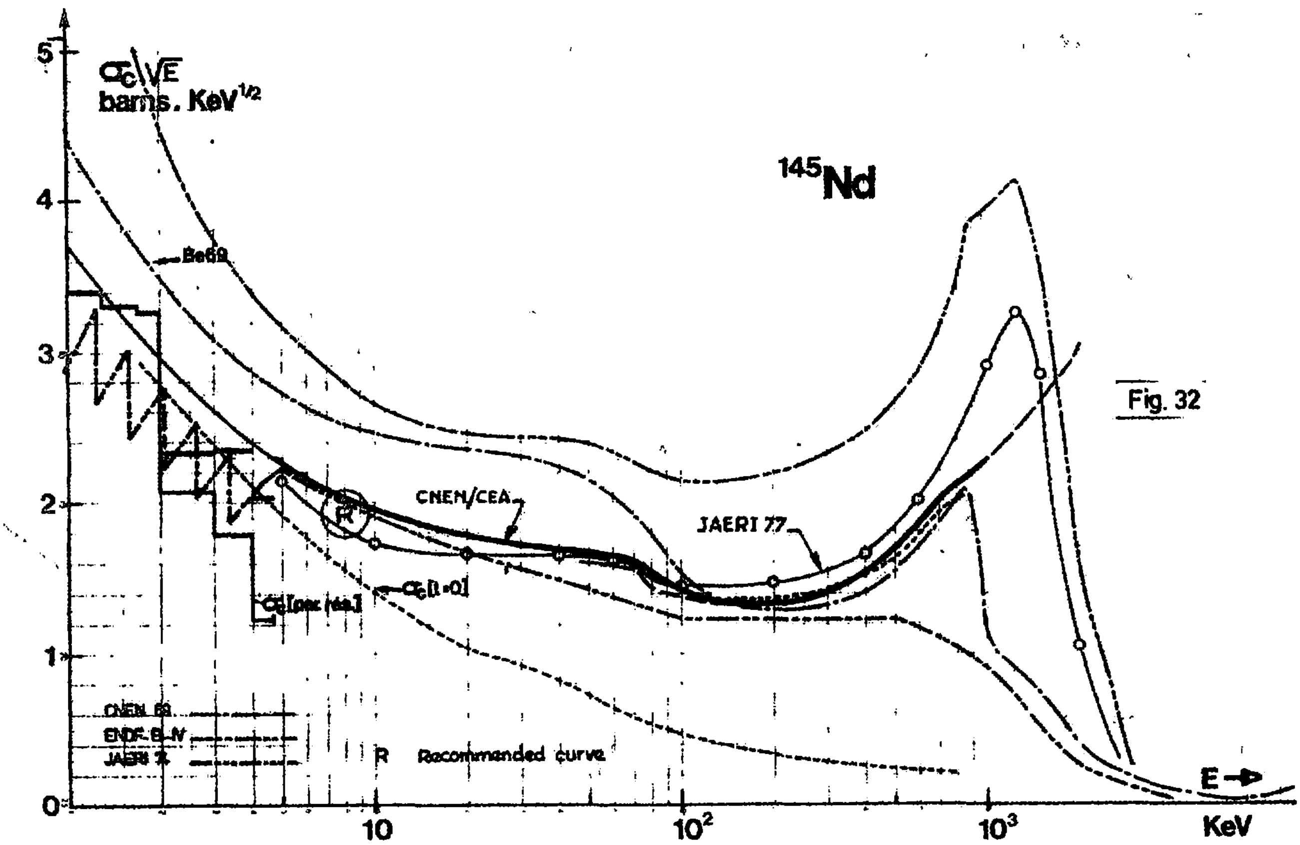
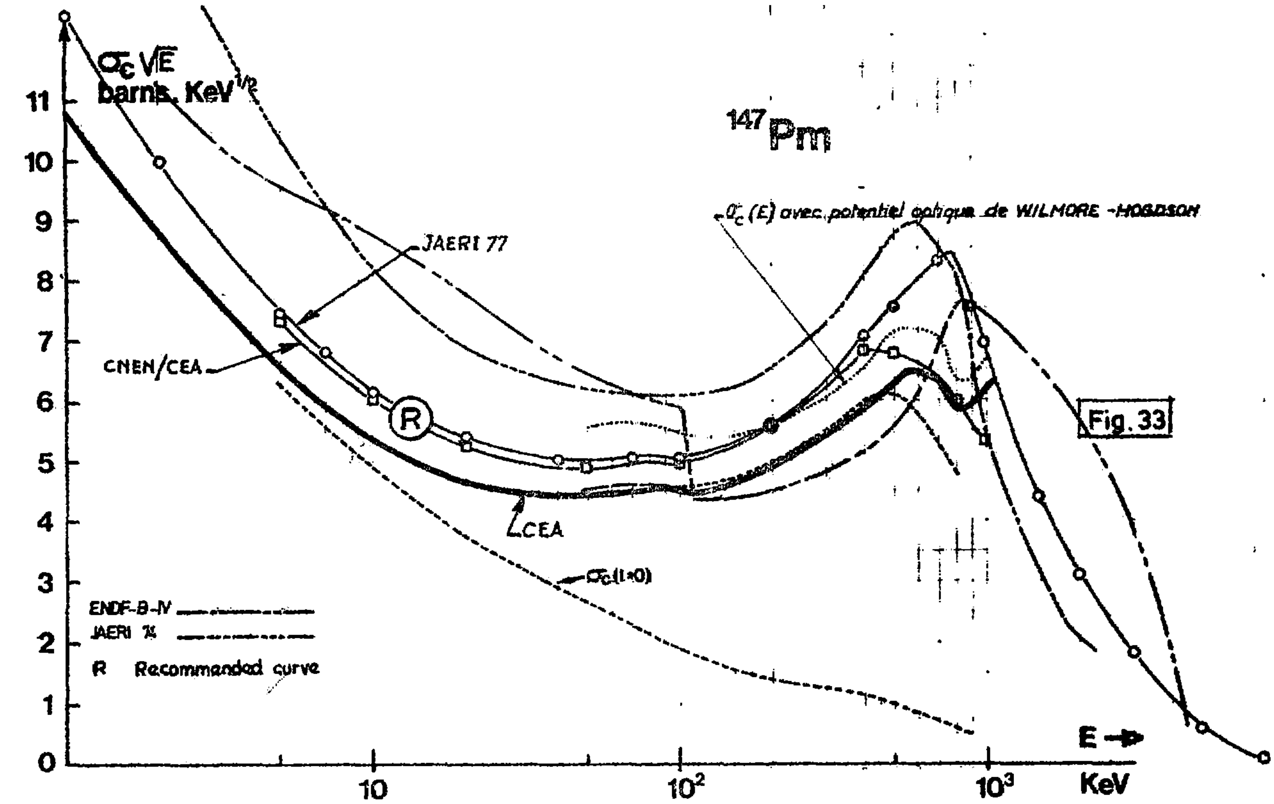
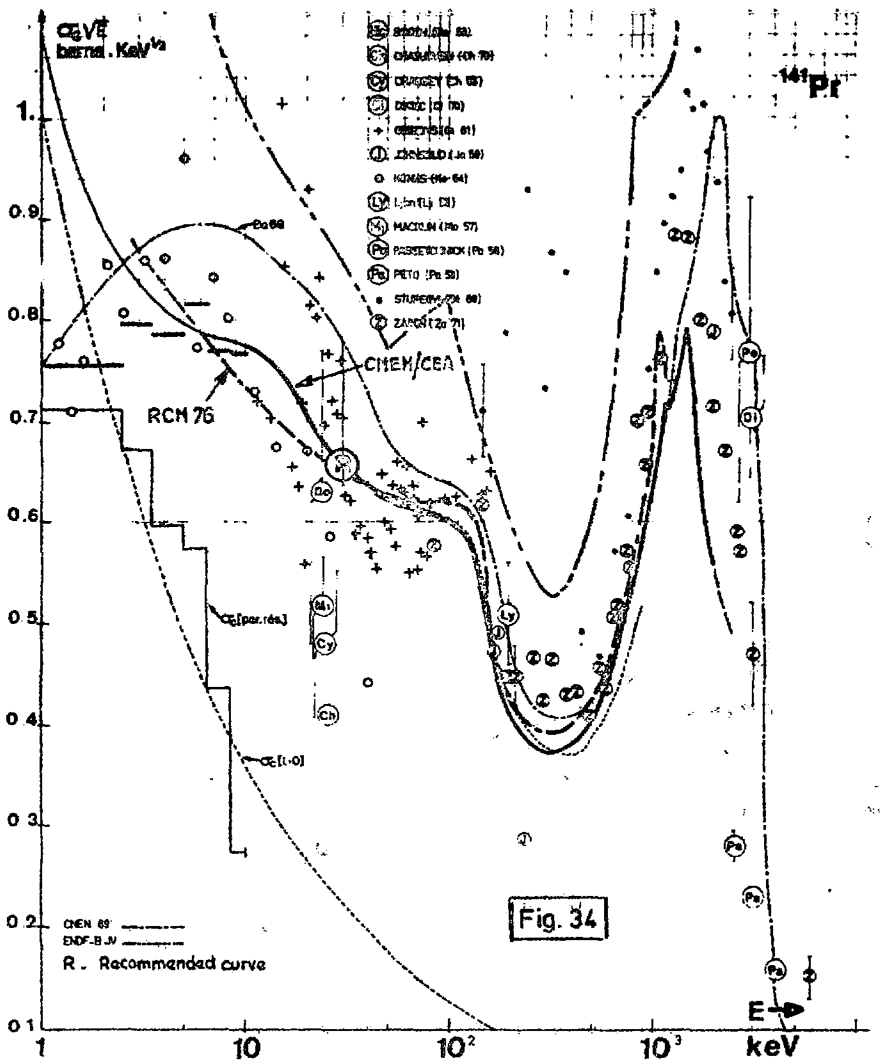

5
4
3
2
1
0
$\sigma_c \sqrt{E}$
barns. KeV$^{1/2}$
^{145}Nd
Be69
R
CNEN/CEA
JAERI 77
σ_c(l=0)
σ_c[par. rés.]
CNEN 69
ENDF-B-IV
JAERI 74
R Recommended curve
E
10
10^2
10^3
KeV

Fig. 32



σc √E
barns. KeV^1/2
147Pm
σc (E) avec potentiel optique de WILMORE - HODGSON
JAERI 77
CNEN/CEA
R
CEA
σc(l=0)
ENDF-B-IV
JAERI 74
R Recommanded curve
Fig. 33
E
KeV
10
10²
10³
0
1
2
3
4
5
6
7
8
9
10
11



σc √E
barns . KeV^1/2
141Pr
CNEN/CEA
RCN 76
Da 69
σc(l=0)
CNEN 69
ENDF-B-IV
R . Recommanded curve
Fig. 34
E
keV
1
10
10²
10³
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.

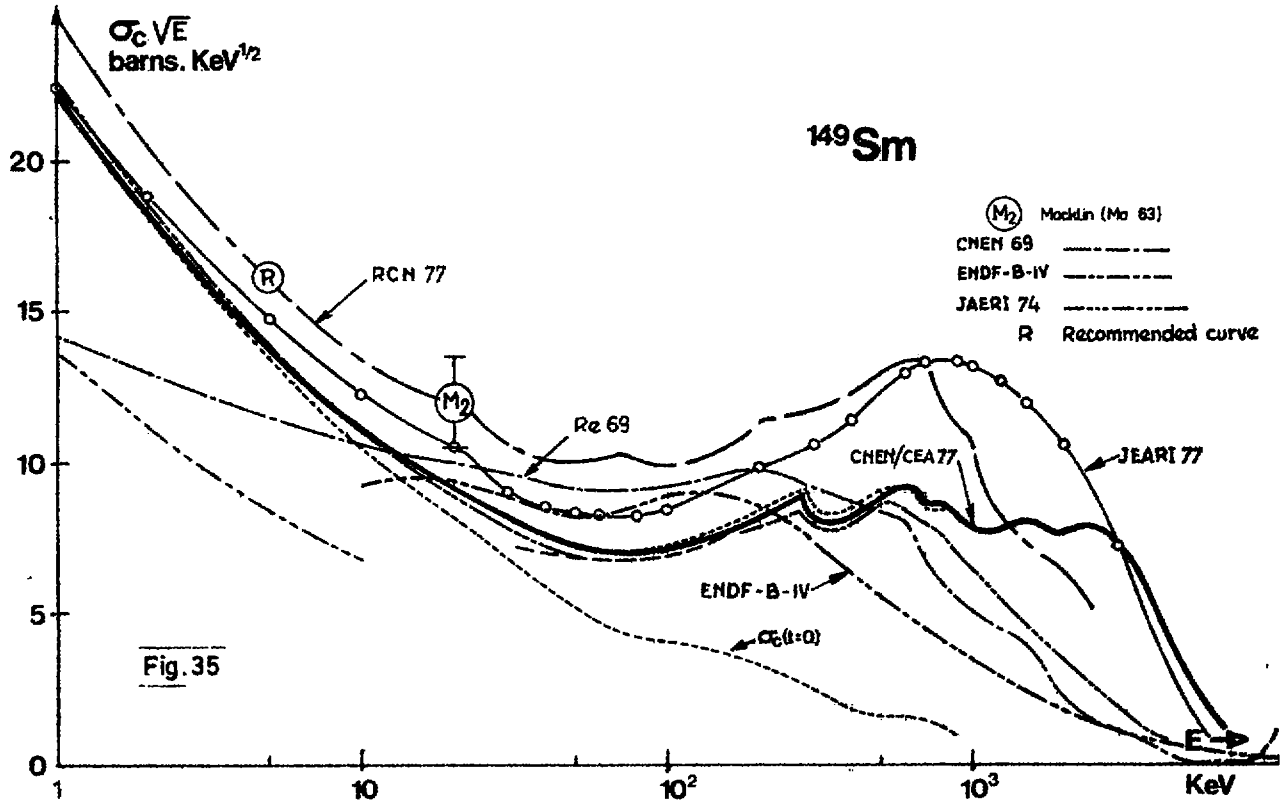


Fig. 35

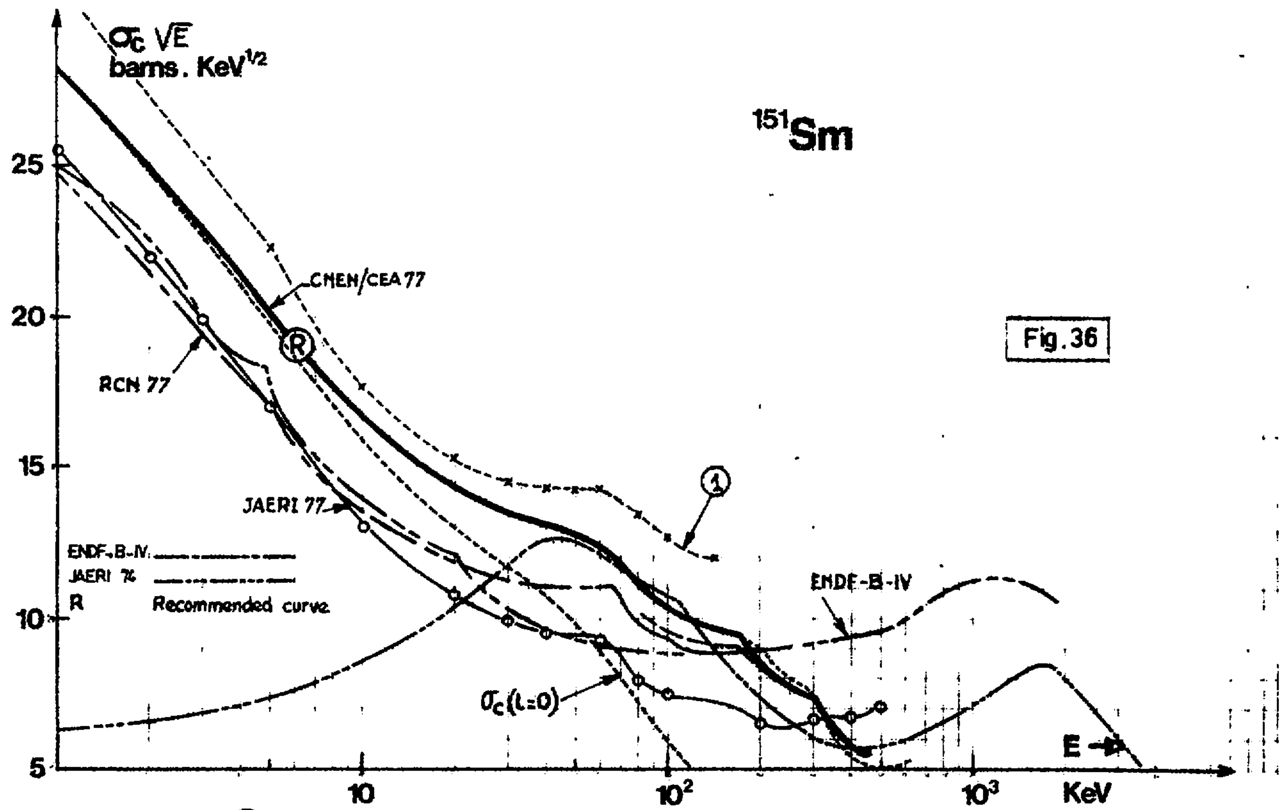


Fig. 36

(1) Curve calculated with recommended average parameters

Review paper 8

IMPACT OF INTEGRAL MEASUREMENTS ON THE CAPTURE CROSS SECTION EVALUATIONS OF INDIVIDUAL FISSION PRODUCT ISOTOPES

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ABSTRACT

In a number of fast reactor facilities integral data, such as capture rates and central reactivity worths, have been measured for a series of samples of individual fission-product isotopes. These integral data provide valuable information for the evaluation of neutron cross sections. In this paper the impact of integral measurements on the capture cross section evaluations is discussed. First, a short summary is given on adopted methods to incorporate these data into the calculated cross sections. This is mostly performed for group cross sections, by means of a least-squares adjustment technique. Next, some results are given of adjusted capture cross sections based upon integral STEK and CFRMF measurements. The adjustments can be translated sometimes into changes in statistical-model parameters such as neutron strength functions, the average observed level spacing or the mean radiation width. The systematics of the last two parameters is shortly discussed. A rather extensive discussion is devoted to the changes in the cross sections of a number of important fission-product isotopes. Finally a summary and some conclusions are given.

1. INTRODUCTION

Integral cross sections of fission-product (f.p.) nuclides measured in fast reactor spectra have been obtained in France (ERMINE, MASURCA, PHENIX), Holland (STEK), Sweden (FRO) and the USA (CFRMF, EBR-2).
These data have been reviewed in a paper contributed to this meeting |1|. Integral data, i.e. transmutation or activation capture rates and central reactivity worths, can be used to *test* evaluated capture cross sections.

Recent evaluations (independent of integral data) of f.p. neutron point cross sections are: CNEN/CEA |2|, ENDF/B-IV |3|, JENDL-1 |4| and RCN-2 |5|. In this paper the outcomes of such tests are discussed. The results show that it is advisable to use the integral data to obtain *adjusted* evaluations of f.p. capture cross sections. In this context we use the word "evaluation" both for multi-group cross section sets and for point cross section sets. However, up till now adjustments based on integral data for fission-products have been applied only to *group* cross sections. Known adjusted f.p. group cross section sets are the French CARNAVAL-4 set and the RCN-2A set |6|. The a-priori data for these sets are the evaluations |2| and |5|, respectively. In the near future the new f.p. *point* cross section file of ENDF/B-V will also be partly based upon integral data. Also an adjusted version of the RCN-2 point cross section set is being prepared.

In this paper the *methods* and *results* of cross section adjustments based on STEK measurements |6| and those based on a combination of STEK data and CFRMF activation data |7| are reviewed with emphasis on the impact of these measurements on σ_c evaluations of individual f.p. nuclides. The CARNAVAL-4 set, which is based upon integral measurements in both French and foreign facilities, has not been released for publication. A considerable amount of integral data has not yet been used for group cross section adjustment. Therefore the scope of this paper is rather limited. Methods or results which have not been published or are not very clear in existing literature are accentuated in this review. Well-published work is very shortly summarized. Finally some conclusions and recommendations are given.

2. METHODS

2.1. Corrections and uncertainties in integral data and spectra

The experimental integral data often have to be corrected for contributions of contaminants or of nuclides which are no fission product. Corrections for self-shielding effects in the measured samples are also needed. Futhermore, if one is interested in capture only, one has to correct reactivity worths for possible contributions of scattering effects (not relevant for reaction rates). The uncertainties in these corrections have to be added to the uncertainties in the measured data; see the discussion in |1, 9|.

Sometimes the corrections are so large and uncertain that it is better to consider the uncorrected data. For instance, the *self-shielding effect* of samples measured in STEK is taken into account, not by extrapolation of measured effects of a number of samples of different sizes to zero thickness ("infinite dilution"), but by introducing all samples measured for a certain nuclide together in the adjustment calculation. The cross sections used for these samples are then macroscopic self-shielded cross sections. The self-shielding factors and the infinite dilution cross sections are of course dependent on the resolved and unresolved resonance parameters used. Therefore the macroscopic cross sections of these samples are heavily correlated (see further 2.3. and |9|). One could also try to correct the measured data with calculated self-shielding corrections. However, the latter method has the draw-back that the corrections depend on the cross sections which have to be adjusted.

In experiments where *inelastic scattering* gives a high contribution to the reactivity effect (e.g. for many CFRMF reactivity measurements) one could try to adjust simultaneously the capture and the inelastic scattering cross sections. These cross sections are not independent as they are calculated both from the statistical model. Therefore, correlations between σ_c and $\sigma_{nn'}$ should be taken into account. However this type of adjustment calculations has not yet been performed for f.p. cross sections.

Uncertainties in the *flux spectra* and/or *adjoint flux spectra* also need to be known. These errors (see |1|) give an additional uncertainty margin to the calculated integral effects. Moreover, this error is correlated for all nuclides measured in the same reactor core; see discussion in sect. 2.3.

2.2. Uncertainties and correlations in cross sections

In the technique of applying adjustments to capture cross sections it is needed to know the uncertainties (i.e. standard deviations) in these cross sections. In the f.p. mass-range most evaluations are - at least partly - based upon nuclear model calculations. For an assessment of the uncertainty in σ_c one could therefore estimate the uncertainty in the model parameters and calculate the propagation of these errors in the cross sections. The model parameters are Breit-Wigner resolved resonance parameters and statistical-model parameters. The important statistical model parameters are based upon resolved resonance data or are derived from nuclear systematics. The role of differential cross section measurements is to tune the various parameters and their uncertainties in

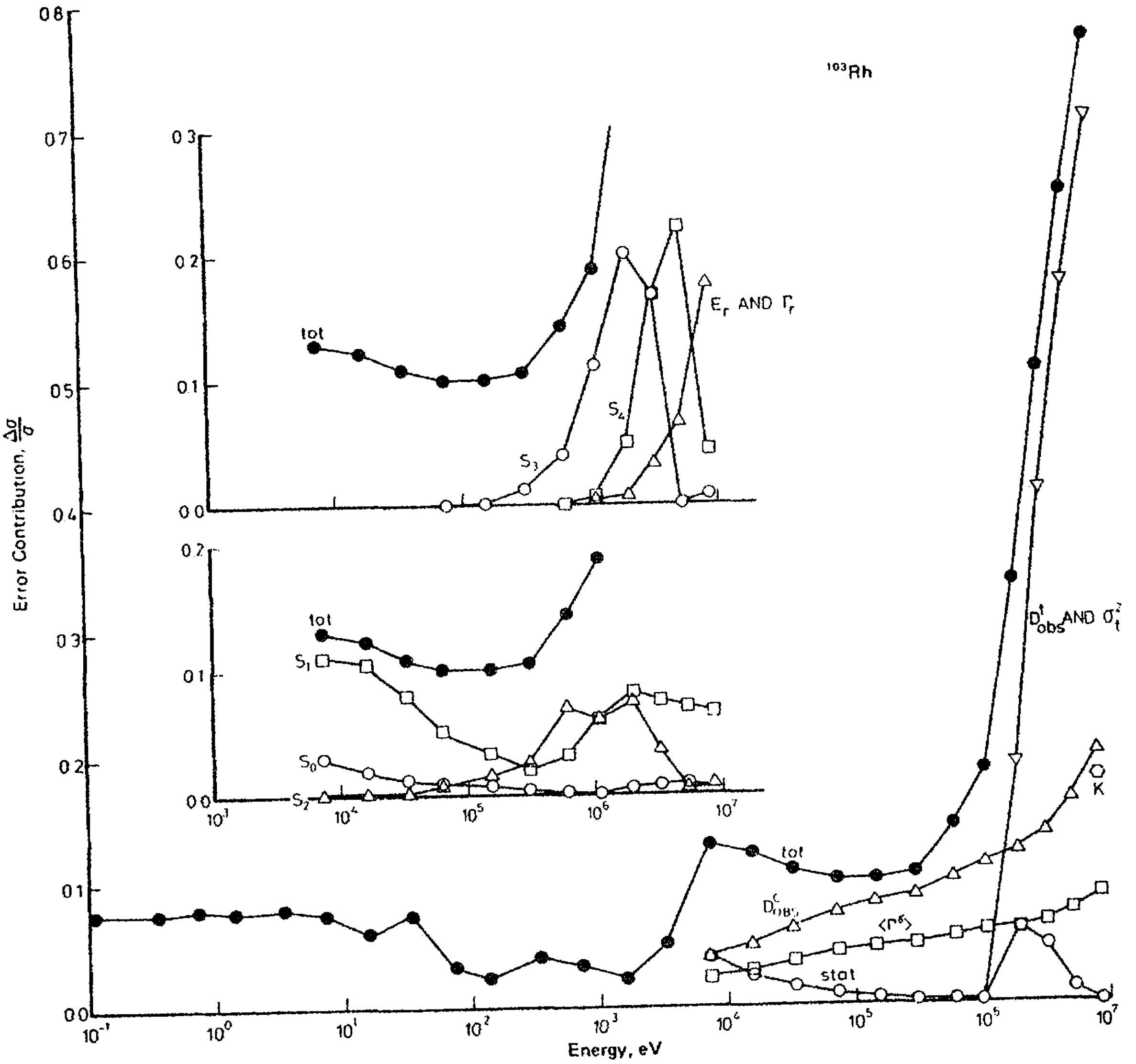

Fig. 1 Example of contributions to the relative total standard deviation (tot) in the capture group cross section of ^{103}Rh. See text for explanation of symbols. Figure taken from ref. |9|.

order to obtain agreement between calculated and measured cross sections before adjustment to integral data. Such an approach has been followed in the evaluation of uncertainties for the RCN-2 group cross section set (26-groups structure |8|). The methods of error evaluation applied to this set have been described in refs. |9-11|; results are given in ref. |6| and summarized in table 1, which is discussed at the end of this section. Apart from uncertainties, correlation coefficients between the group cross sections need to be known, except when very wide groups are used. For the RCN-2 error evaluation a 26×26 co-variance matrix is calculated for every nuclide.

In fig. 1 (taken from |9|) an *example* is given of how the different contributions add to the uncertainty of σ_c of ^{103}Rh. Below E = 4 keV the total relative standard deviation ("tot") is due to uncertainties in resolved-resonance parameters. For many nuclides the uncertainty in the capture width of each resonance is assumed to be fully correlated with Γ_γ values of all other resonances. No further correlations between parameters have been taken into account. In the figure the uncertainties refer to infinite dilution only. Above 4 keV the different error components are indicated separately.

The points indicated with "stat" are statistical errors not connected with uncertainties in the parameters, but are due to the statistical nature of the model |10, 11|, resulting from: uncertainties in the number of levels per energy interval, fluctuations in the level widths and uncertainties in the number of target levels which can be excited by inelastic scattering. The other parameters (with assumed uncertainties in parenthesis) are:

$\langle\Gamma\gamma\rangle$, average capture width (6%), $\langle\Gamma_\gamma\rangle = \langle\Gamma_\gamma(\ell=0)\rangle = \langle\Gamma_\gamma(\ell=1)\rangle$;

D^c_{obs} , s-wave level spacing of compound nucleus at the neutron binding energy (11%);

S_o , s-wave neutron strength function (15%);

S_1 , p-wave neutron strength function (30%);

S_2, S_3, S_4, d-, f-, g-wave neutron strength functions (100%);

D^t_{obs} , s-wave level spacing of target nucleus at the neutron binding energy (30%);

$\sigma^2{}_t$, spin cut-off parameter of target nucleus at low excitation energy (30%);

Table 1

Estimated uncertainties in fast capture cross sections for a number of important fission-product isotopes (relative standard deviations in %).

Nuclide	CEA [13]	unadjusted RCN-2 [6]				
	25 keV	10-21.5 keV	100-200 keV	0.8-1.4 MeV	6.5-10 MeV	average spect. d)
^{95}Mo	12	21	23	31	50	18
^{97}Mo	13	21	26	30	52	17
^{98}Mo	12	4 a)	29	37	46	9
^{100}Mo	12	41	42	60	60	27
^{99}Tc	10	19	22	23	84	16
^{101}Ru	20	28	22	37	84	16
^{102}Ru b)	15	42	44	49	110	35
^{103}Ru c)	25	--	--	--	--	--
^{104}Ru b)	15	36	37	48	140	30
^{103}Rh	3	12	10	19	71	9
^{105}Pd	25	16	18	30	55	16
^{106}Pd b)	--	64	74	100	150	60
^{107}Pd c)	30	33	88	170	220	55
^{108}Pd b)	20	82	100	160	200	85
^{109}Ag	15	13	16	27	48	12
^{127}I	5	12	16	29	45	9
^{129}I	35	25	31	42	58	25
^{133}Cs	12	14	20	26	76	12
^{135}Cs c)	35	--	--	--	--	--
^{139}La	20	24	25	24	52	16
^{141}Pr	15	17	18	20	77	12
^{143}Nd	18	--	--	--	--	--
^{145}Nd	20	--	--	--	--	--
^{147}Pm	22	--	-	--	--	--
^{149}Sm	10	16	19	25	40	15
^{151}Sm	20	9	16	16	65	9
^{152}Sm	--	18	22	21	39	12

a) Calculated from resolved resonance parameters (Breit-Wigner formula).

b) Very few resolved resonance parameters known (not more than 3).

c) No resolved resonance resonance parameters known.

d) Uncertainty in average capture cross section in a fast breeder reactor spectrum (SNR-300).

Table 2

Uncertainties in a number of important statistical model parameters and in σ_c at 25 keV according to Ribon et al. |13|.

	$\bar{D}$ (l = 0)	l = 0		l = 1		S_0	S_1	σ_c(25keV)
		$\bar{\Gamma}_\gamma$	$\bar{\Gamma}_\gamma/\bar{D}$	$\bar{\Gamma}_\gamma$	$\bar{\Gamma}_\gamma/\bar{D}$			
95 Mo	10 %	3 %	11 %	20 %	15 %	25 %	25 %	12 %
97 Mo	15 %	5 %	16 %	20 %	18 %	30 %	30 %	13 %
98 Mo	30 %	10 %	12 %	15 %	12 %	30 %	30 %	12 %
100 Mo	15 %	15 %	12 %	15 %	12 %	30 %	30 %	12 %
99 Tc	6 %	4 %	7 %	15 %	10 %	20 %	25 %	10 %
101 Ru	15 %	3 %	15 %	25 %	15 %	25 %	30 %	20 %
102 Ru	25 %	30 %	15 %	30 %	15 %	40 %	30 %	15 %
103 Ru	40 %	45 %	20 %	45 %	20 %	30 %	30 %	25 %
104 Ru	25 %	30 %	15 %	30 %	15 %	30 %	35 %	15 %
103 Rh	3 %	1.5 %	3.5 %	3 %	4 %	11 %	20 %	3 %
105 Pd	20 %	6 %	15 %	30 %	20 %	25 %	35 %	25 %
107 Pd	40 %	40 %	20 %	45 %	20 %	35 %	35 %	30 %
108 Pd	35 %	40 %	20 %	45 %	20 %	40 %	40 %	20 %
109 Ag	2.5 %	4 %	5 %	10 %	10 %	18 %	30 %	15 %
127 I	1.5 %	10 %	10 %	10 %	10 %	9 %	40 %	5 %
129 I	25 %	45 %	30 %	45 %	30 %	45 %	40 %	35 %
133 Cs	3.5 %	5 %	6 %	16 %	15 %	15 %	50 %	12 %
135 Cs	40 %	50 %	30 %	50 %	30 %	30 %	50 %	35 %
139 La	9 %	26 %	25 %	26 %	25 %	25 %	45 %	20 %
141 Pr	6 %	21 %	20 %	21 %	20 %	25 %	45 %	15 %
143 Nd	8 %	3 %	9 %	7 %	11 %	20 %	40 %	18 %
145 Nd	9 %	10 %	15 %	2[illegible] %	22 %	15 %	40 %	20 %
147 Pm	15 %	3 %	16 %	28 %	22 %	22 %	40 %	22 %
149 Sm	8 %	2.5 %	9 %	20 %	18 %	14 %	40 %	10 %
151 Sm	20 %	7 %	8 %	25 %	22 %	20 %	40 %	20 %
159 Tb	7 %	3 %	8 %	16 %	15 %	11 %	50 %	25 %

$\overline{E_r}$, giant dipole resonance energy (30%);
$\overline{\Gamma_r}$, giant dipole resonance width (40%);
$\overline{K}$, constant for direct and collective capture (100%).

From the figure it is obvious that in a relatively large energy range (up to about 500 keV) the most important statistical-model parameters are $<\Gamma_\gamma>$, D^c_{obs}, S_o and S_1. In fact, at low neutron energies (below 50 keV) S_o and S_1 are very sensitive parameters, whereas at higher energies the ratio $<\Gamma_\gamma>/D^c_{obs}$ is most important. At energies below 10 keV the statistical model errors (stat) can become the most important errors (not in the case of ^{103}Rh). At energies above 500 keV the uncertainty in D^t_{obs} dominates.

The result of an error calculation of σ_c at 25 keV based upon uncertainties in S_0, S_1, $<\Gamma_\gamma>$ and D^c_{obs} is given in table 2 which has been taken from the work of Ribon et al. |13|. A difference with the RCN-2 error evaluation is that correlations between $<\Gamma_\gamma> = \overline{\Gamma_\gamma}$ and $D^c_{obs} = \overline{D}$ (important when $<\Gamma_\gamma>$ is estimated from theory) are included in table 2. The distinction between s- and p-wave capture widths is also made in the RCN-2 error evaluation for a number of nuclides (with the assumption that these widths are fully correlated).

In table 5 (section 3.2.) uncertainties in the parameters S_o, S_1, $<\Gamma_\gamma>$ and D^c_{obs} are given according to the RCN-2 error evaluation.

An *intercomparison* between the results of two uncertainty estimates is given in table 1, second and third columns. The RCN-2 error evaluation gives much more conservative estimates, in particular for nuclides with masses around A≈100. For these nuclides many of the known resolved resonances might have p-wave character, which complicates the determination of D^c_{obs}. Moreover, in the case of Mo isotopes non-statistical effects (e.g. valency capture; see App. 3 of |26|) enlarges the uncertainty in σ_c. Therefore, additional uncertainty has been assigned to a number of parameters in the RCN-2 error evaluation. Attention has also to be paid to the fact that the uncertainty in σ_c for ^{107}Pd (with no resolved resonances known) is smaller than for ^{106}Pd and ^{108}Pd, although the uncertainties in the parameters for ^{107}Pd were assumed to be larger or equal to those for $^{106,108}Pd$. The explanation is that at low neutron energies a large value of $<\Gamma_\gamma>/D^c_{obs}$ (as for all odd targets) reduces the sensitivity of σ_c to variations in this ratio, thereby increasing the sensitivity to variations in S_o and S_1. This follows from the fact that at low energies σ_c is roughly proportional to

$$\sigma_c \sim \frac{\langle\Gamma_\gamma\rangle/D^c_{obs} \times (S_o + S_1)}{\langle\Gamma_\gamma\rangle/D^c_{obs} + S_o + S_1}. \qquad (1)$$

Concluding this section we can say that for all nuclides the uncertainty in the statistical model region up to about 500 keV is mainly determined by $\langle\Gamma_{\gamma_c}\rangle/D^{c}_{obs}$. Below 50 keV the parameters S_o and S_1 compete with $\langle\Gamma_\gamma\rangle/D^c_{obs}$; for odd targets σ_c is less sensitive to $\langle\Gamma_\gamma\rangle/D^c_{obs}$ than for even targets. Below 10 keV the uncertainties due to the statistical model itself can become important. In the MeV range the parameter D^t_{obs} dominates.

2.3. Group cross section adjustment

Adjustments of group cross sections for fission-product nuclides based upon integral measurements have been performed in France (CARNAVAL-4 set) and the Netherlands (RCN-2A set). In Japan the STEK data have been used extensively to test the JENDL-1 set |21|. However, there is as yet no unified view in the FPND working group of the Japanese Nuclear Data Committee about the adjustment of capture (group) cross sections of fission products |37|.

Various *methods* for adjustment of group cross sections have been described in literature |9, 14-18|. Recent reviews have been given by Gandini |19| and by Kuroi and Mitani |20|. Therefore it is not necessary to discuss all these methods in this report.

The method given in ref. |9| has been applied to the analysis of the *STEK measurements*. This method can be interpreted as a least-squares minimization in which both integral data (collected in a vector R) and (macroscopic) group cross sections (collected in a vector Σ) are involved:

$$q^2 = (R' - R)\, V^{-1}\, (R' - R) + (\Sigma' - \Sigma)\, M^{-1}\, (\Sigma' - \Sigma) = \text{minimum}. \qquad (2)$$

The quantities to be adjusted are indicated with a prime in eq. (2). The matrices V and M contain the co-variances of the experimental integral data and a-priori cross section data, respectively. The relation between the adjustments of integral data and those of the cross sections is given by

$$R' - R = G\,(\Sigma' - \Sigma), \tag{3}$$

where G is a sensitivity matrix, to be calculated from the flux and adjoint flux spectra |9|. Eq. (2) can be interpreted as a least-squares fit of integral quantities to measured data by a variation of group cross sections, thereby considering the a-priori values as fictitious integral measurements. In the usual least-squares fitting terminology the "parameters" are the group cross sections and the "measurements" are the values of the integral data as well as the a-priori values of the group cross sections. The number of "free parameters" equals that of the integral measurements, therefore. The minimum value of q^2 is called χ^2 and this value can be used for a consistency test.
For the calculation of the error matrices V and M, the sensitivity matrix G, the adjusted cross section vector Σ' and the adjusted integral data vector R' the reader is referred to |9|.

The final aim of the adjustment procedure is to find adjusted microscopic capture group cross sections (without *self-shielding*) per isotope. In practice however, one has to deal with samples of different isotopic composition and of different weights, measured in different reactor cores. Therefore the macroscopic group cross section vectors Σ and Σ' are extended with isotopic group cross sections in infinite dilution. Correspondingly, the matrix M is extended to correlate the infinite dilution data to the self-shielded macroscopic cross sections. Though there are no experimental integral data R which correspond with these extensions, the adjustment procedure still can be performed and gives the correct adjusted values of infinite dilution cross sections |9|. This method has proven to be much more reliable than to correct the measured data prior to the adjustment calculation for self-shielding effects by means of an extrapolation of measured data for different sample sizes to zero thickness.

Another experience from the analysis of STEK measurements is related to the role of the *uncertainties in the flux and adjoint spectra*. These uncertainties were initially (|9|,|26|) attached to the co-variance matrix V of the integral data, taking into account correlations, e.g. for different samples in the same reactor core. As a result of the adjustment procedure the integral data, the cross sections, but also the spectra were adjusted. These (implicit) spectrum adjustments are in principle correct. However, for each set of samples treated in one adjustment calculation different implicit spectrum adjustments occurred, which obstructed the systematic interpretation of cross section adjustments. Therefore in the final adjustment calculations |6|, the values of the spectra were "frozen" in the adjustment procedure; however, uncertainties and correlation coefficients originating from errors in the spectra were attached afterwards to the adjusted error matrices V' and M'. This means that the adjusted dat |6| for all nuclides have a common systematic error originating from spectrum uncertainties ($\approx$5% in $<\sigma_c>$).

The possibility to use *different types of measurements* in one adjustment calculation has been utilized in ref. |7| by a simultaneous adjustment of both STEK reactivity measurements and CFRMF activation measurements. In ref. |7| also CFRMF activation data have been used without STEK data. However, it is more attractive to combine data which have been measured in different spectra. In France the adjustments for the CARNAVAL-4 set are also based on common adjustments of integral data from various sources. It is also possible to include differential measurements in the adjustment procedure (instead of taking these data into account in the evaluation). This can be accomplished very easily by a condensation of differential data into groups and considering these group constants as integral measurements.

2.4. Parameter adjustment

In the statistical model region the group cross sections depend on a relatively small number of nuclear model parameters. The cross section adjustments in this energy range can be translated into adjustments of the underlying parameters, by means of an assumed linear relationship

$$\Sigma' - \Sigma = S\ (P' - P), \qquad (4)$$

where P denotes a parameter vector and S is a sensitivity matrix of partial derivatives of Σ with respect to P (see ref. |9| for relation between M and S). In the application to be discussed in sect. 3.2 most a-priori parameters are assumed to be independent. However, for a number of nuclides (see table 5b) different values of $<\Gamma_\gamma>$ for $\ell = 0$ and $\ell = 1$ have been used, which were assumed to be fully correlated. The parameters are as given in sect. 2.2, i.e. $<\Gamma_\gamma>$, D^c_{obs}, S_ℓ ($\ell = 0$-4), D^t_{obs}, σ^2_t, E_r, Γ_r and K. However, it turned out that for most STEK results only the parameters $<\Gamma_\gamma>$, D^c_{obs}, S_o and S_1 show significant adjustments (see sect. 3.2). For the CFRMF measurements the parameter S_o shows smaller adjustments, whereas the parameter S_2 shows larger adjustments. The parameters exclusively related to the MeV range (i.e. D^t_{obs}, σ^2_t, E_r, Γ_r, K) are not adjusted significantly. This can be ascribed to the low sensitivity of most integral measurements for cross section changes in that energy range.

2.5. Point cross section adjustment

Adjusted point cross section sets, based upon integral measurements have not yet been published for f.p. nuclides. In future evaluations of ENDF/B-V such adjustments will be applied |22,23|. Work on an adjusted version of the RCN-point cross section set is in progress.

A general reference in which adjustment is applied to point cross section data is ref. |24|. However, since most codes have been developed for group section adjustments rather than for point cross section adjustment, it seems more practical to try to translate the group cross section adjustments into point cross section adjustments. In the smooth statistical model region this can be easily performed since the results of most adjustments are simply renormalizations of the capture cross sections (see graphs in |6|). Instead of renormalizing σ_c as suggested in |22| it is also possible to translate the differences between unadjusted and adjusted group constants into a mathematical function, e.g. a polynomial |33|.

A somewhat more sophisticated approach could be the use of adjusted parameters S_0, S_1, $<\Gamma_\gamma\ (\ell=0)>$, $<\Gamma_\gamma\ (\ell=1)>$ and D^c_{obs} (see sect. 2.4.) to recalculate σ_c. An example of this procedure is given in fig. 2 for σ_c of ^{152}Sm. The parameters before and after adjustment are as given in table 5 (next chapter). Group cross sections calculated from the adjusted point cross section set are not completely the same as the directly adjusted group cross sections from ref. |6|; see table 3. The main differences are: no adjustments in the resolved resonance range (groups 14-26) and too large adjustments above 1 MeV (groups 1-5). This last mentioned difference originates from the use of a strength function model in the uncertainty calculation for all energies, whereas in the cross section calculation s- and p-wave strength functions are only used

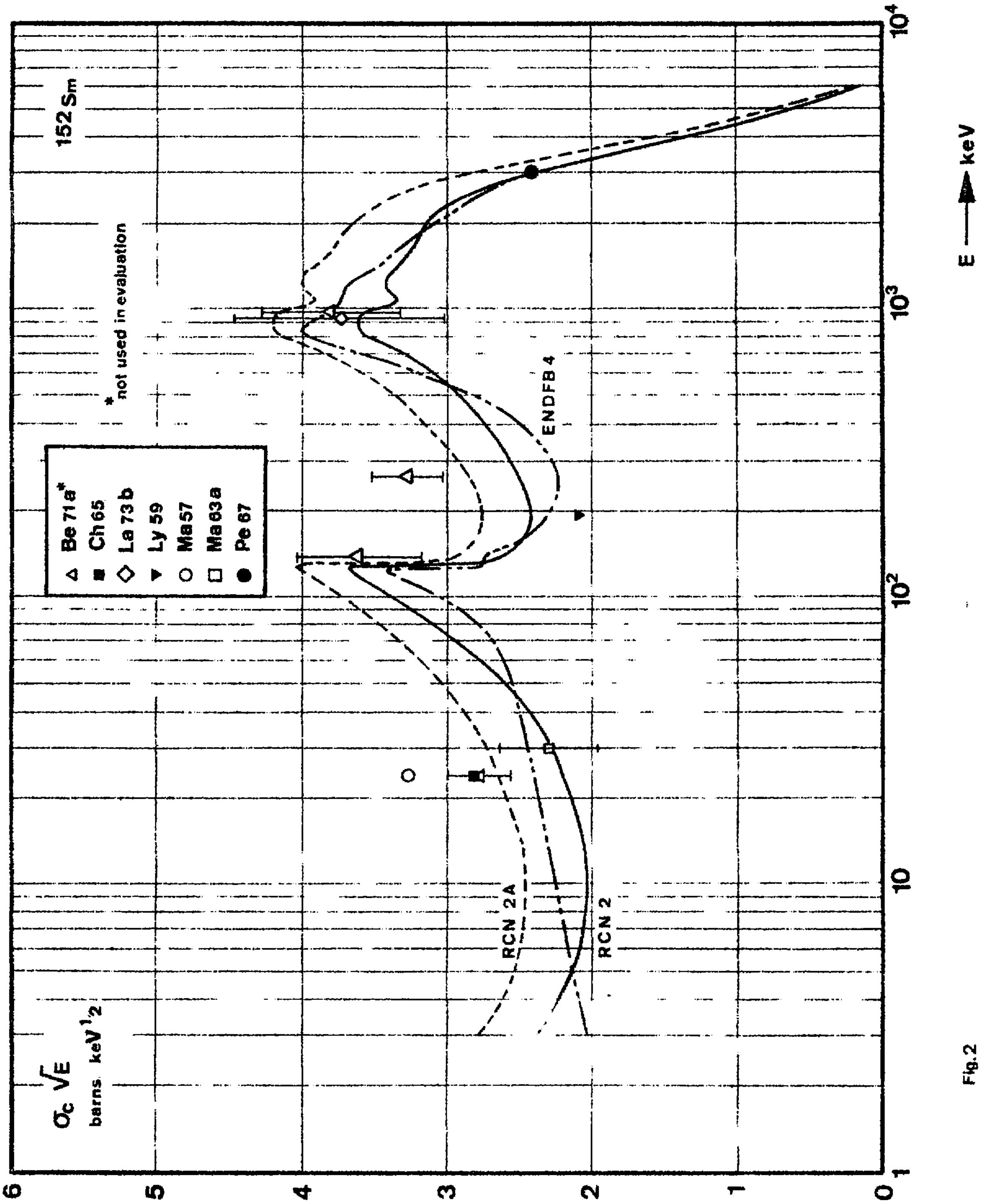

Fig. 2 Capture cross section of ^{152}Sm according to the unadjusted RCN-2 and adjusted RCN-2A evaluations. For comparison available capture data and results from the ENDF/B-IV evaluation are also plotted.

up to 70 keV. Above 1 MeV an increase of S_0, S_1 and S_2 (S_2 is also of some importance here) leads to a decrease of σ_c as a result of inelastic scattering competition. From this example it is clear that this method to obtain adjusted point cross sections works well in a large statistical-model energy range, where the adjustments are significant, but is inappropriate in the resolved resonance region. So far, results of this kind have been obtained for 11 nuclides. In table 5 adjusted parameters are given for 32 nuclides (see sect. 3.2.), for use in future re-evaluations.

Table 3

Relative adjustments (%) of capture group constants of ^{152}Sm, calculated from adjusted point cross sections (P) compared with directly adjusted group cross sections (G) from ref. |6|.

ABBN group	Relative adjustments (%) P	G	ABBN group	Relative adjustments (%) P	G
1	17	3	14	0	10
2	13	5	15	0	10
3	16	6	16	0	10
4	17	7	17	0	8
5	14	10	18	0	20
6	13	12	19	0	10
7	14	13	20	0	6
8	11	14	21	0	4
9	15	15	22	0	6
10	20	17	23	0	7
11	21	18	24	0	7
12	19	16	25	0	7
13	12	13	26	0	7

2.6. Interpretation of adjusted data

Integral data

Adjusted data (i.e. integral quantities, group cross sections and model parameters) are the result of a least-squares procedure in which all available information is used in an optimal way. The most reliable results are obtained for adjusted integral data (e.g. reaction rates), calculated from adjusted cross sections in spectra which are not too much different from those in which the integral measurements have been performed. For these data the uncertainties are mostly reduced drastically and there is only a *very small dependence* on the a-priori data. This is so because the uncertainties in the a-priori integral data of f.p. are almost always larger than the uncertainties in the integral measurements. Therefore, the adjusted integral data are about equal to the *weighted average of (corrected) experimental integral data*, almost independent of the a-priori data. This follows from plots in ref. |6| of calculated, experimental and adjusted integral data.

Cross sections

The adjusted multi group cross sections depend of course much stronger on the a-priori group constants and covariances. In energy regions where the uncertainty is small, mostly also small adjustments take place. Moreover, the sensitivity of cross sections to various integral quantities plays an important role. For instance, the adjustments in the *thermal energy range* are small and have not much meaning. There is also not much sensitivity in the *MeV range*, but adjustments in that region are mostly correlated with those in the keV range. In the *resolved-resonance range* the uncertainties are often small; likewise adjustments are small in that region Therefore, an appreciable part of the significant adjustments occur in the energy range from 1 keV to about 500 keV. In this *statistical model energy range* it is often found that the shape of the calculated cross section agrees with the curve through well-measured experimental points. Thus, in many cases the main effect of adjustments should be a renormalization of σ_c, which is easily performed when the uncertainty in the ratio $\langle\Gamma_\gamma\rangle/D_{obs}$ is not too small. This is indeed observed for many nuclides |6|.

Nevertheless, in general there is a *non-negligible dependence* on a-priori data and one has to interprete the results of adjusted (group) cross sections with care, in particular when the a-priori uncertainties are small.

For most nuclides there is an appreciable reduction of errors in the group constants of the statistical model region |6|. It has to be noticed that by the adjustment quite different correlations between the group constants are introduced.

Parameters

For the adjusted parameters the dependence on the a-priori values and uncertainties is *rather strong*. Moreover, correlations are introduced between these parameters. See further the discussion in the previous section.

Systematical errors

Systematical errors in the adjusted cross section data could occur due to systematical errors in the (normalization of) integral measurements and in the sensitivities (spectra) |1|, or due to systematical errors in the evaluation. These last-mentioned errors are particularly important when the statistical model fails (Mo-isotopes, sect. 3.5.2) or when resolved resonances have been missed. Estimated systematical errors caused by errors in normalizations and in sensitivities of integral experimental data are about 5% in $<\sigma_c>$ |1|. There are no indications for larger systematic errors, as follows from χ^2-tests |6| or from comparisons of integral experimental data with well-known capture cross sections, e.g. σ_c of ^{103}Rh (see sect. 3.5.1) and ^{93}Nb |6|. The capture cross section of ^{10}B and the fission cross section of ^{235}U are also well-known, but integral data for these nuclides have been used for STEK spectrum adjustments |35|. For many f.p. nuclides the capture cross section in the resolved resonance region is fairly well-known. There is a slight tendency for positive adjustments in this energy range, but it is difficult to ascribe this to a systematical effect.

3. RESULTS

3.1. Classification

In this chapter the effect of adjustments to cross sections of the most important (i.e. 0.5% or larger contribution to the total f.p. capture rate in a fast breeder reactor) f.p. nuclides is discussed. A possible classification of nuclides is as follows:

a) Odd-Z f.p. nuclides

All important stable or long-lived odd-Z f.p. nuclides have an even value of N. The level density of these nuclides is much larger than for even mass nuclides. Therefore, the capture cross section is relatively large for these f.p. Many of the odd-Z elements have only one stable isotope. For this reason there are relatively many differential capture measurements for these nuclides. However, there is also a number of unstable nuclides for which these measurements are scarce or even lacking (e.g. for ^{135}Cs). For many odd-Z f.p. nuclides total scattering cross sections and elastic or inelastic scattering cross sections are also known. Most nuclides are spherical or show a soft vibrational character. The nuclides are listed in table 4a, together with some global characteristics.

b) Even-Z f.p. nuclides with Z<50

The most important nuclides in this class (see table 4b) are the isotopes of Mo, Ru and Pd. These nuclides have many charachteristics in common. The number of differential capture cross section measurements in the unresolved resonance region is relatively small. For the evaluation of neutron cross sections one can make use of "local systematics" in the estimation of the level density parameter a, see sect. 3.2. For most of the important f.p. nuclides in this class the p-wave strength function is very large compared to the s-wave strength function. As a result, the determination of average resonance parameters from resolved resonances is rather difficult, since many weak resonances with unknown value of ℓ might have p-wave character.

Moreover, non-statistical capture effects like valency capture and doorway state mechanisms play a role for nuclides with masses near $A = 100$. These features apply particularly to the Zr and Mo isotopes, which are nearly spherical (close to magic $N = 50$). For the neutron-rich Ru and Pd isotopes slight deformation effects of vibrational character are to be expected.

c) <u>Even-Z f.p. nuclides with Z>50</u>

Most of the STEK measurements with the nuclides in this class have not been analysed yet (see table 4c). It seems that this class of nuclides gives a significantly smaller contribution to the total capture rate of f.p. in a fast breeder reactor than classes (a) or (b). In thermal reactors the opposite is true because of the high thermal capture cross sections of Sm and Gd isotopes.
The explanation cannot be that the average capture cross sections at high energies for heavy f.p. are smaller than for the light f.p. because the mass dependence of the photon strength function (which is roughly proportional to the capture cross section) gives evidence for an opposite effect:

$$\frac{<\Gamma_\gamma>}{D} = k_{E_1} \, A^{2/3} \, \varepsilon_\gamma^3 \, , \tag{5}$$

where k_{E_1} is about constant |25|.

The isotopes of Xe, Nd and Sm have neutron numbers rather close to the magic number $N = 82$. However, for many isotopes of Nd and Sm deformation effects play a role, see also sect. 3.2. For instance the Sm isotopes cover a range from a closed-neutron shell nucleus (^{144}Sm) with a spherical shape to a strongly deformed rotational nucleus (^{154}Sm). The level schemes of the transitional nuclei ^{147}Sm to ^{151}Sm show the characteristics of a soft vibrator.

In general the level density of nuclides in class (c) is higher than that of class (b); the opposite is true for $<\Gamma_\gamma>$. The s-wave neutron strength function is much larger than the p-wave strength function for $A \approx 150$. This is also opposite to the situation in class (b).

Table 4a

Average radiative capture cross sections $<\sigma_c>$ for important stable or long-lived odd-Z nuclides ($<\sigma_c>$ in b; flux spectrum SNR-300; uncertainties in %)

Nuclide[a)]	Natural abun- dance (%)	RCN-2[b)] (unadjusted)		RCN-2A[b)] (STEK)		RCN-2A[c)] (STEK+CFRMF)		ENDF/B-4	CNEN/ CEA	JENDL-1
^{99}Tc	0	0.54	(16%)[d)]	0.64	(7%)	0.59	(6%)	0.49	0.55	0.54
^{103}Rh	100	0.64	(9%)[d)]	0.64	(6%)	0.64	(6%)	0.70	0.63	0.65
^{109}Ag	48	0.68	(12%)	0.71	(7%)	0.73	(6%)	0.48	0.65	0.81
^{127}I	100	0.52	(9%)[d)]	0.57	(7%)	0.57	(7%)	0.54	-	-
^{129}I	0	0.34	(25%)	0.30	(12%)	0.30	(8%)	0.38	-	0.44
^{133}Cs	100	0.51	(12%)	0.51	(7%)	0.49	(6%)	0.48	0.49	0.45
^{135}Cs	0	e)		e)		-		0.067	0.21	0.27
^{139}La	99.9	0.031	(16%)	0.035	(12%)	0.031	(7%)	0.038	0.028	-
^{141}Pr	100	0.13	(12%)	0.12	(8%)	0.12	(7%)	0.16	0.13	-
^{147}Pm	0	e)		e)		e)		1.25	1.08	1.08
^{153}Eu	52	e)		e)		e)		2.29	2.48	2.4

a) STEK reactivity measurements also for ^{93}Nb, ^{137}Cs, ^{151}Eu, ^{159}Tb; CFRMF activation measurements also for ^{87}Rb, ^{89}Y, ^{93}Nb, ^{107}Ag, ^{115}In, $^{121,123}Sb$, $^{151,153}Eu$.

b) Taken from tables in ref. |6| and supplements to |6|; adjustments based on STEK reactivity measurements.

c) Taken from tables in ref. |7|; adjustments based on STEK reactivity data and CFRMF activation data.

d) Revised evaluations, not given in refs. |5,6|.

e) To be analysed in the near future.

Table 4b

Average radiative capture cross sections $\langle\sigma_c\rangle$ for important stable or long-lived even-Z nuclides with Z<50 ($\langle\sigma_c\rangle$ in b; flux spectrum SNR-300, uncertainties in %)

Nuclide[a)]	Natural abundance (%)	RCN-2[b)] (unadjusted)	RCN-2A[b)] (STEK)	RCN-2A[c)] (STEK+CFRMF)	ENDF/B-4	CNEN/CEA	JENDL-1
^{93}Zr	0	d)	d)	-	0.086	0.11	0.16
^{95}Mo	16	0.30 (18%)	0.28 (8%)	-	0.29	0.27	0.30
^{97}Mo	10	0.30 (17%)	0.30 (9%)	-	0.28	0.28	0.31
^{98}Mo	24	0.086 (9%)	0.084 (9%)	0.087 (6%)	0.101	0.104	-
^{100}Mo	20	0.10 (27%)	0.080 (21%)	0.074 (8%)	0.078	0.082	-
^{101}Ru	17	0.69 (16%)	0.68 (8%)	-	0.53	0.76	0.71
^{102}Ru	32	0.20 (35%)	0.15 (18%)	0.16 (8%)	0.19	0.22	0.22
^{104}Ru	19	0.17 (30%)	0.14 (10%)	0.14 (7%)	0.14	0.18	0.16
^{105}Pd	22	0.81 (16%)	0.88 (7%)	-	0.83	0.85	0.76
^{106}Pd	27	0.19 (60%)	0.22 (12%)	-	0.16	0.19	-
^{107}Pd	0	0.96 (55%)	0.93 (10%)	-	0.57	0.79	0.75
^{108}Pd	27	0.18 (85%)	0.17 (19%)	0.17 (17%)	0.16	0.20	-

a) STEK reactivity measurements also for ^{90,91,92,94,96}Zr, ^{92,94,96}Mo, ^{104,110}Pd, ^{111}Cd, ^{128,130}Te;
CFRMF activation measurements also for ^{110}Pd.

b) See footnote b) of table 4a.

c) See footnote c) of table 4a.

d) To be analysed in the near future.

Table 4c

Average radiative capture cross sections $<\sigma_c>$ for important stable or long-lived even-Z nuclides with Z>50 ($<\sigma_c>$ in b; flux spectrum SNR-300; uncertainties in %)

Nuclide[a)]	Natural abun-dance (%)	RCN-2[b)] (unadjusted)	RCN-2A[b)] (STEK)	RCN-2A (STEK+CFRMF)	ENDF/B-4	CNEN/ CEA	JENDL-1
^{131}Xe	21	c)	c)	-	0.21	-	0.37
^{132}Xe	27	c)	c)	c)	0.69	-	-
^{134}Xe	11	c)	c)	c)	0.35	-	-
^{143}Nd	12	c)	c)	-	0.30	0.34	0.29
^{145}Nd	8	c)	c)	-	0.33	0.36	0.34
^{146}Nd	17	c)	c)	c)	0.13	0.071	-
^{148}Nd	11	c)	c)	c)	0.18	0.16	-
^{150}Nd	6	c)	c)	c)	0.22	0.21	-
^{149}Sm	14	2.24 (15%)	2.21 (9%)	-	1.41	1.76	1.99
^{151}Sm	0	2.13 (9%)	1.80 (14%)	-	2.21	2.11	2.07
^{152}Sm	27	0.41 (12%)	0.47 (9%)	c)	0.40	-	-
^{157}Gd	16	c)	c)	-	3.54	1.13	-

a) STEK reactivity measurements also for ^{140,142}Ce, ^{142,144}Nd, ^{147,148,150,154}Sm, ^{156}Gd;

CFRMF activation measurements also for ^{140,142}Ce, ^{154}Sm, ^{158,160}Gd.

b) See footnote b) of table 4a.

c) To be analysed in the near future.

3.2. Global results of parameter adjustment

Unadjusted and adjusted values of parameters S_o, S_1, $D_{obs}^{\dagger}$, $<\Gamma_\gamma\ (\ell = 0)>$ and $<\Gamma_\gamma\ (\ell = 1)>$ have been listed in table 5.
The unadjusted parameters are from the RCN-2 library |5|; the adjusted data are based on the STEK reactivity measurements. The other parameters (see sect. 2.4.) show no appreciable changes, with a few exceptions, i.e. S_2 is of importance for ^{92}Mo, ^{102}Ru, $^{148,\ 149,\ 150,\ 152,\ 154}$Sm.
The parameters $<\Gamma_\gamma\ (\ell = 0)>$ and $<\Gamma_\gamma\ (\ell = 1)>$ have been assumed to be fully correlated in the adjustment procedure.

In general the *s-wave strength function* (table 5a) is not adjusted very much; also the uncertainty is not reduced significantly. Exceptions are found for a few odd-mass nuclides for which the statistical model region commences at a relatively low energy. For nuclides of class (c) these adjustments are most important, due to the relatively high value of S_o. It is interesting to note that the observed "odd-even effect" in S_o for the Sm isotopes |29| disappears after adjustment (see sect. 3.5.3). In fig. 3 the ^{105}Pd cross section adjustments are shown as an example. The value of S_o is adjusted by 16% which corresponds to an increase of σ_c for E = 160 eV to about 10 keV. The remaining adjustment of σ_c has to be attributed to a change in D_{obs}.

The *p-wave strength function* (table 5a) is adjusted in more cases than the s-wave strength function; moreover, the uncertainty is reduced in many cases. In the case of ^{101}Ru the 12% adjustment of S_1 corresponds with a 3% to 4% increase of σ_c from 0.2 to 20 keV (see |6|). Thus the influence of S_1 adjustment on σ_c is relatively small.

Adjustments of s- and p-wave *capture widths* (table 5b) are important, because σ_c is rather sensitive to a hange of $<\Gamma_\gamma>$ over a large energy range. Adjustments of more than 10% occur for ^{99}Tc, ^{95}Mo, $^{102,\ 104}$Ru, $^{147,\ 150,\ 152,\ 154}$Sm. Mostly the uncertainty is reduced significantly.

† The index "c" is dropped in the remaining part of this paper.

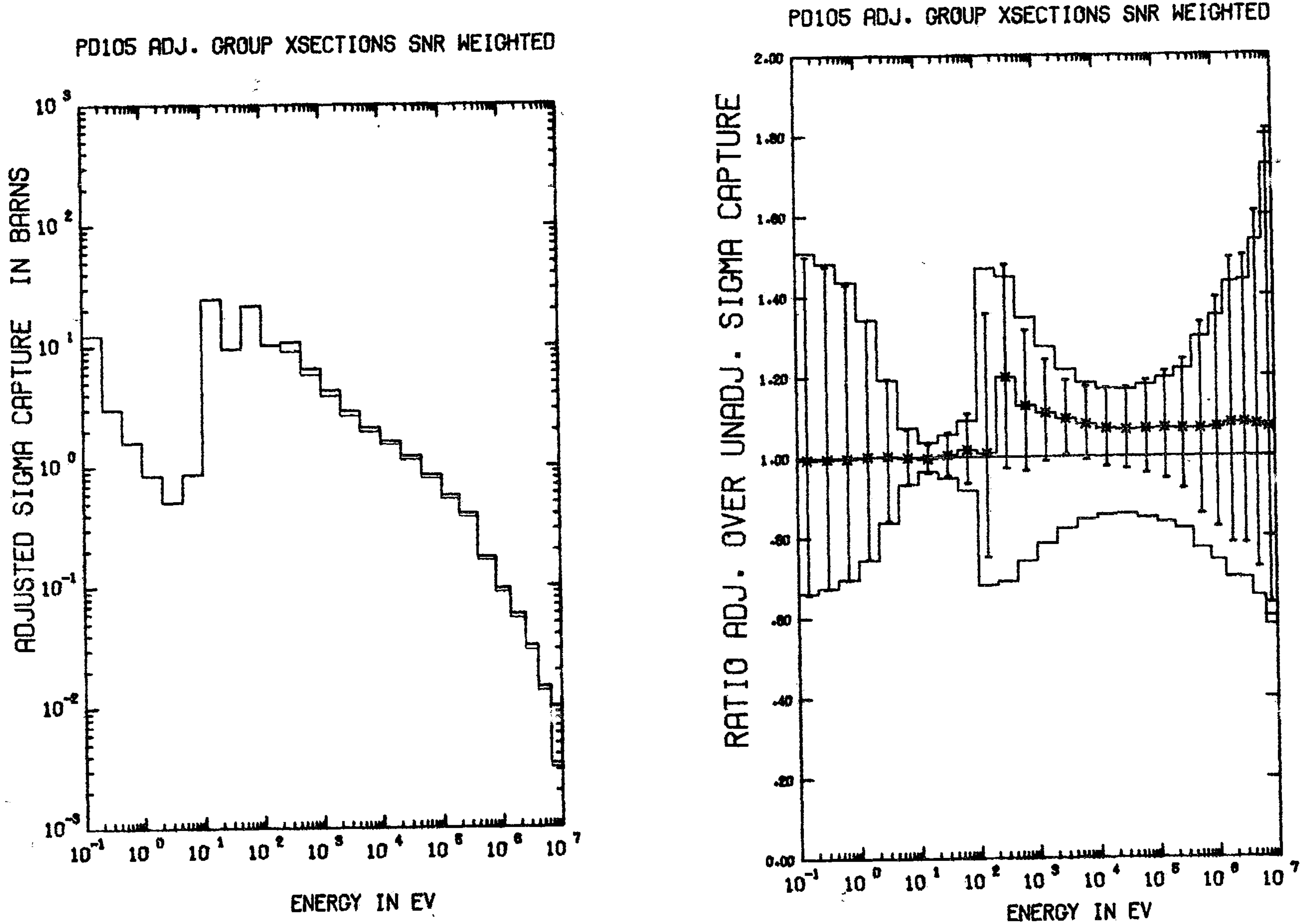

Fig. 3. Adjusted capture group cross sections of ^{105}Pd. Left-hand figure: dashed curve is the unadjusted cross section. Right-hand figure: envelope histograms represent unadjusted uncertainty margins; asterisk-marked histogram + error bars represent relative group cross section adjustments (figure from ref. |6|).

Table 5a

S- and p-wave strength function adjustment based on RCN-2 evaluation and STEK integral measurements

Class	Nuclide	S_o (10^4 $eV^{\frac{1}{2}}$) unadjusted ref. [5]		S_o adjusted[a)]		S_1 (10^4 $eV^{\frac{1}{2}}$) unadjusted ref. [5]		S_1 adjusted[a)]	
class (a)	^{93}Nb	0.36	(17%)	--	--	7.0	(50%)	--	--
	^{99}Tc[b)]	0.47	(30%)	0.52	(27%)	6.0	(50%)	7.7	(39%)
	^{103}Rh[b)]	0.47	(15%)	--	--	6.5	(30%)	6.3	(19%)
	^{109}Ag	0.60	(25%)	0.66	(22%)	3.8	(27%)	4.0	(24%)
	^{127}I[b)]	0.80	(20%)	0.87	--	2.0	(25%)	2.2	--
	^{129}I	0.80	(50%)	0.52	(38%)	2.0	(50%)	2.1	(48%)
	^{133}Cs	0.80	(13%)	--	(12%)	3.9	(26%)	3.7	(24%)
	^{139}La	0.64	(23%)	--	--	2.0	(50%)	--	--
	^{141}Pr	1.72	(20%)	--	--	1.1	(50%)	1.0	(50%)
class (b)	^{92}Mo	0.65	(40%)	--	--	3.3	(33%)	--	--
	^{94}Mo	0.80	(50%)	--	--	4.5	(33%)	--	--
	^{95}Mo	0.80	(50%)	--	--	5.0	(33%)	--	(30%)
	^{96}Mo	0.36	(50%)	--	--	5.4	(33%)	--	--
	^{97}Mo	0.75	(50%)	--	--	6.0	(33%)	--	(30%)
	^{98}Mo	0.35	(50%)	--	--	6.1	(33%)	--	--
	^{100}Mo	0.30	(50%)	--	--	6.2	(33%)	--	--
	^{101}Ru	0.56	(27%)	--	(23%)	7.3	(100%)	6.4	(52%)
	^{102}Ru	0.32	(100%)	0.37	--	7.3	(100%)	10.0	(75%)
	^{104}Ru	0.32	(100%)	0.30	--	7.0	(100%)	5.0	(7%)
	^{104}Pd	0.40	(50%)	--	--	6.1	(30%)	--	--
	^{105}Pd	0.50	(40%)	0.58	(10%)	5.4	(30%)	--	(26%)
	^{106}Pd	0.40	(50%)	--	--	5.7	(30%)	--	--
	^{107}Pd	0.40	(50%)	0.45	(60%)	5.5	(30%)	5.8	--
	^{108}Pd	0.40	(50%)	0.43	--	5.3	(20%)	--	--
	^{110}Pd	0.40	(50%)	0.36	--	4.9	(30%)	--	--
class (c)	^{147}Sm	4.3	(30%)	--	(22%)	1.8	(50%)	--	(46%)
	^{148}Sm	3.0	(33%)	--	--	1.2	(50%)	--	(47%)
	^{149}Sm	5.1	(18%)	4.1	(12%)	1.8	(50%)	2.3	(47%)
	^{150}Sm	3.3	(33%)	3.5	--	1.2	(50%)	1.5	(47%)
	^{151}Sm	3.7	(14%)	2.6	(23%)	1.2	(50%)	1.1	--
	^{152}Sm	2.2	(18%)	2.4	(16%)	1.2	(50%)	1.6	(45%)
	^{154}Sm	1.8	(28%)	--	--	1.2	(50%)	1.3	--

a) Only adjusted data or errors which differ more than 5% from the unadjusted values have been denoted.

b) Revised evaluation.

<u>Table 5b</u>

Adjustment of D_{obs} and $<\Gamma_\gamma>$ based on the RCN-2 evaluation and STEK integral measurements.

class	nuclide	D_{obs} (eV)[a] unadjusted ref. |5|	D_{obs} (eV)[a] adjusted[b]	$<\Gamma_\gamma(\ell=0)>$ (meV) unadjusted ref. |5|	$<\Gamma_\gamma(\ell=0)>$ (meV) adjusted[b]	$<\Gamma_\gamma(\ell=1)>$ (meV) unadjusted ref. |5|	$<\Gamma_\gamma(\ell=1)>$ (meV) adjusted[b]
class (a)	^{93}Nb	100 (10%)	-- --	146 (3.5%)	-- --	195 (10%)	192 (9%)
	^{99}Tc[c]	18.6(12%)	17.8(10%)	130 (25%)	154 (15%)		
	^{103}Rh[c]	26.1(11%)	-- (9%)	161 (6%)	-- --		
	^{109}Ag	17.5(18%)	17.0(12%)	129 (10%)	-- (9%)		
	^{127}I[c]	12.2(15%)	11.1(12%)	95 (10%)	99 (9%)		
	^{129}I	30 (33%)	31 (17%)	107 (20%)	-- (16%)		
	^{133}Cs	20 (11%)	-- (10%)	125 (20%)	121 (12%)		
	^{139}La	286 (10%)	-- --	50 (25%)	51 --		
	^{141}Pr	120 (18%)	124 (16%)	85 (6%)	-- --		
class (b)	^{92}Mo	3950 (25%)	3540 --	178 (8%)	182 --	285 (20%)	302 --
	^{94}Mo	1740 (60%)	1940 (20%)	169 (18%)	-- --	254 (20%)	-- --
	^{95}Mo	82 (13%)	84 (11%)	154 (40%)	129 (17%)	281 (20%)	258 (8%)
	^{96}Mo	1300 (76%)	2020 (15%)	152 (26%)	143 (24%)	202 (20%)	193 (19%)
	^{97}Mo	66 (25%)	-- (13%)	134 (10%)	-- (8%)	190 (20%)	-- (17%)
	^{98}Mo	1000 (28%)	1110 (26%)	86 (19%)	81 (18%)	138 (22%)	130 (21%)
	^{100}Mo	690 (51%)	990 (18%)	58 (16%)	55 (15%)	115 (17%)	109 (16%)
	^{101}Ru	16.7(10%)	-- --	172 (27%)	-- (17%)		
	^{102}Ru	570 (30%)	650 (22%)	275 (50%)	170 [illegible]		
	^{104}Ru	265 (40%)	325 (20%)	97 (25%)	88 (21[illegible])		
	^{104}Pd	530 (100%)	490 (20%)	190 (30%)	-- (28%)	210 (30%)	-- (28%)
	^{105}Pd	9.9(22%)	9.3(15%)	155 (10%)	-- (9%)	155 (10%)	-- --
	^{106}Pd	330 (100%)	280 (13%)	120 (30%)	-- (27%)	130 (30%)	-- (27%)
	^{107}Pd	4.2(190%)	4.9(12%)	100 (30%)	-- --	110 (30%)	-- --
	^{108}Pd	200 (150%)	220 (10%)	70 (30%)	-- --	80 (30%)	-- --
	^{110}Pd	146 (150%)	280 (10%)	50 [illegible]	48 --	55 (30%)	53 --
class (c)	^{147}Sm	6.3(11%)	-- (10%)	100 [illegible]0%)	87 (10%)		
	^{148}Sm	107 (50%)	-- (15%)	60 (33%)	57 (25%)		
	^{149}Sm	2.0(15%)	-- (11%)	76 (20%)	-- (12%)		
	^{150}Sm	56.5(18%)	50.1(12%)	60 (33%)	69 (13%)		
	^{151}Sm	1.72(9%)	1.8(10%)	96 (8%)	91 (9%)		
	^{152}Sm	53.8 (8%)	52.6 --	70 (17%)	77 (13%)		
	^{154}Sm	130 (13%)	-- --	70 (33%)	79 (26%)		

[a] The uncertainty in D_{obs} is mostly asymmetric. Only the error margin for a positive variation is indicated.

[b] Only adjusted values which differ more than 2% (uncertainties more than 5%) from the unadjusted values have been denoted.

[c] Revised evaluation.

The largest adjustments and uncertainty reductions occur for the *level spacing* D_{obs} (table 5b), which has about the same influence (in opposite direction) on σ_c as the parameter $<\Gamma_\gamma>$. However, the adjustments of D_{obs} are larger than those of $<\Gamma_\gamma>$ due to the fact that the a-priori uncertainties of D_{obs} mostly exceed those of $<\Gamma_\gamma>$. The direction of adjustment of D_{obs} is always opposite to that of $<\Gamma_\gamma>$, which means that the corresponding cross section adjustments are in the same direction.

3.3. Systematics of a

The results of parameter adjustment can be used to improve the *systematics of the level-density parameter a*. This has been discussed in App. 2 of ref. |26|. A short outline of this discussion together with updated results is given below. The parameter a is defined as given by Gilbert and Cameron |27|, however with a modified expression for the spin cut-off factor, i.e.

$$\sigma^2 = 0.146 \sqrt{aU} \; A^{2/3} , \tag{5}$$

where U is the neutron energy corrected for pairing energy |27|. The value of a is deduced from D_{obs}. In table 6 the experimental, adopted and adjusted values of a are given. The adopted and the adjusted values of a correspond with values of D_{obs} of table 5b. The references of the experimental values of D_{obs} have been given in |5|. Level-density parameters for nuclides of class (a) are not discussed since the adjustments are mostly small.

From the phenomenological *systematics of a for nuclides of class (b)*, plotted in fig. 4, it is very clear that for different families of isotopes the curves of a versus N are shifted. The adopted values of a for the Pd isotopes have been obtained by drawing a straight line through the measured point for ^{106}Pd with a slope deduced from the a-curve of the Mo-isotopes. The curve for the Mo isotopes is based on experimental values of a, while taking into account the slope of the curves for the Ru isotopes.

The *shape* of the adopted curves (fig. 4) can be understood from theory - at least qualitatively - by considering the level density of a single-particle level scheme, thereby considering neutrons and protons separately..

For nuclides with N close to the magic number 50, the level density is very low. The level density increases about linearly with N when more shells become involved. The a-priori systematics is based upon this assumption. However, it is clear that for high values of N this behaviour of a versus N should change, since there is another magic number at $N = 82$. Moreover, the effect of deformations (Nilsson model) is to smooth the shell effects in a.

The *adjusted values of a for class (b)* (see fig. 4) indeed show that for many neutron-rich isotopes the a-priori a-values have been overestimated. One could of course criticize the adjusted results by objecting that these data are not independent of adjustments of other parameters. However, it follows from sect. 3.2 that from the four important parameters for nuclides of class (b) S_o and S_1 are far less important than D_{obs} and $\langle\Gamma_\gamma\rangle$. Moreover, variations in S_o and S_1 influence σ_c only in local energy ranges, which means that adjustments of D_{obs} and $\langle\Gamma_\gamma\rangle$ are not very much dependent on those of S_o and S_1. The adjustments of D_{obs} and $\langle\Gamma_\gamma\rangle$ are heavily correlated, i.e. the uncertainty in the *ratio* of adjusted D_{obs} and $\langle\Gamma_\gamma\rangle$ is much smaller than what would follow from the adjusted uncertainties given in table 5b. Therefore large adjustments in D_{obs} have to be interpreted with care when the uncertainty in $\langle\Gamma_\gamma\rangle$ is also large, i.e. for the nuclides ^{102}Ru, ^{104}Ru, ^{110}Pd, and most Mo-isotopes (sect. 3.5.2).

In fig. 5 the *systematics of a for nuclides of class (c)* is shown. The references for the experimental points of Sm are given in |5|; the experimental values of a for the Nd and Gd isotopes have been taken from the work of Benzi et al. |28|. The full curves have been drawn through experimental a-values for the odd-mass nuclides; the dashed curves are for the even-mass nuclides. For a number of Sm isotopes the values of D_{obs} are very accurately known. Therefore, the adjustments are small for most nuclides. It is also evident from the figure that there is an odd-even effect in a. The adjusted a-values are slightly more accurate and support this conclusion. This effect has not been noticed in the work of Kirouac et al. |29| on the systematics of a. The explanation of this odd-even effect is probably that there is no correction for deformation effects in the Gilbert-Cameron formula. Preliminary calculations |30| with a formula given by Felvinci et al. |31| (based on the work of Ericson |32|) support this explanation.

Table 6

Systematics of level-density parameter a (MeV^{-1})

compound nuclide	N	experimental[a)]	adopted, unadjusted[b)] ref. |5|	adjusted[b)]
^{94}Nb	53	12.51± 0.14	12.58± 0.14	12.48± 0.11
^{93}Mo	51	10.53± 0.3	10.53± 0.3	10.69± 0.3
^{95}Mo	53	12.9 ± 0.8	12.9 ± 0.8	12.73± 0.2
^{96}Mo	54	13.9 ± 0.2	14.0 ± 0.2	13.96± 0.17
^{97}Mo	55	--	14.6 ± 1.0	13.81± 0.3
^{98}Mo	56	15.5 ± 0.4	15.8 ± 0.4	15.77± 0.2
^{99}Mo	57	17.4 ± 0.5	17.62± 0.5	17.40± 0.5
^{101}Mo	59	19.2 ± 1.0	20.55± 1.0	19.67± 0.4
^{100}Tc	57	16.42± 0.1	16.38± 0.20	16.45± 0.18
^{102}Ru	58	16.2 ± 0.2	16.20± 0.16	16.20± 0.15
^{103}Ru	59	17.94± 0.5	17.94± 0.5	17.69± 0.4
^{105}Ru	61	20.85± 0.8	20.85± 0.8	20.38± 0.4
^{104}Rh	59	17.13± 0.05	17.13± 0.17	17.13± 0.15
^{105}Pd	59	(16.4)	15.9 ± 1.3	16.05± 0.3
^{106}Pd	60	17.2 ± 0.3	17.2 ± 0.3	17.30± 0.2
^{107}Pd	61	(17.2)	18.4 ± 1.5	18.73± 0.3
^{108}Pd	62	--	19.6 ± 2.0	19.31± 0.2
^{109}Pd	63	(23.9)	20.9 ± 2.	20.69± 0.2
^{111}Pd	65	--	23.4 ± 2.3	21.74± 0.2
^{110}Ag	63	18.38± 0.3	18.38± 0.3	18.43± 0.2
^{128}I	75	17.32± 0.13	17.46± 0.2	17.62± 0.2
^{130}I	79	16.41± 0.5	16.41± 0.5	16.33± 0.3
^{134}Cs	79	16.19± 0.18	16.19± 0.18	16.17± 0.15
^{140}La	83	15.56± 0.18	15.56± 0.19	15.57± 0.18
^{142}Pr	83	15.60± 0.11	15.85± 0.3	15.79± 0.3
^{148}Sm	86	20.56± 0.2	20.77± 0.2	20.74± 0.19
^{149}Sm	87	--	23.78± 1.0	23.80± 0.4
^{150}Sm	88	23.62± 0.3	24.00± 0.3	24.00± 0.2
^{151}Sm	89	26.88± 0.4	26.88± 0.4	27.21± 0.3
^{152}Sm	90	24.23± 0.18	24.23± 0.18	24.10± 0.2
^{153}Sm	91	25.63± 0.2	25.63± 0.2	25.69± 0.2
^{155}Sm	93	24.03± 0.3	23.66± 0.3	23.71± 0.3

a) Deduced from experimental values of D_{obs}; see references in |5|.

b) Corresponds to table 5b.

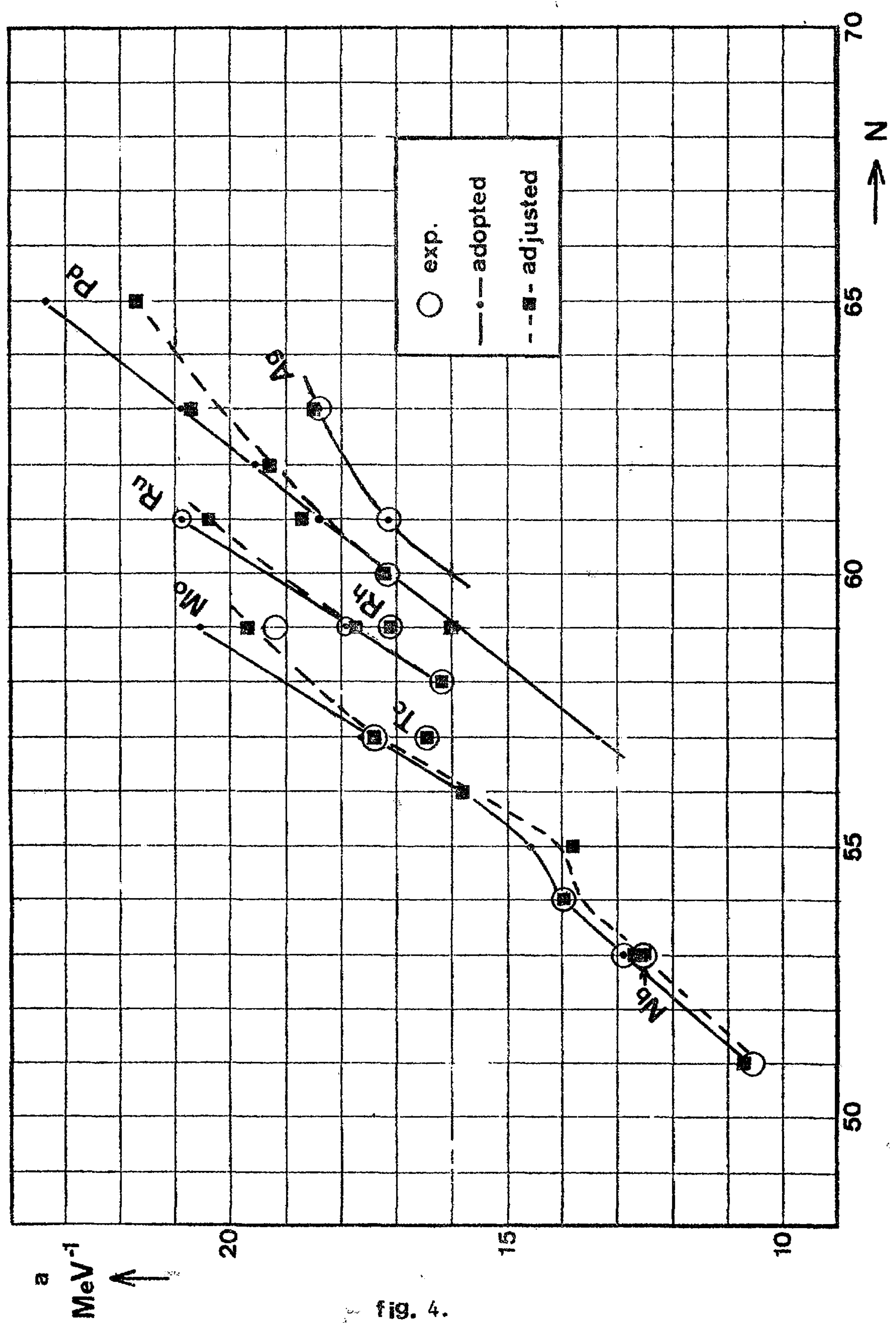

fig. 4.

The nuclear level density parameter a as a function of the neutron number N for nuclides of class (b). See sect. 3.3 for further explanation.

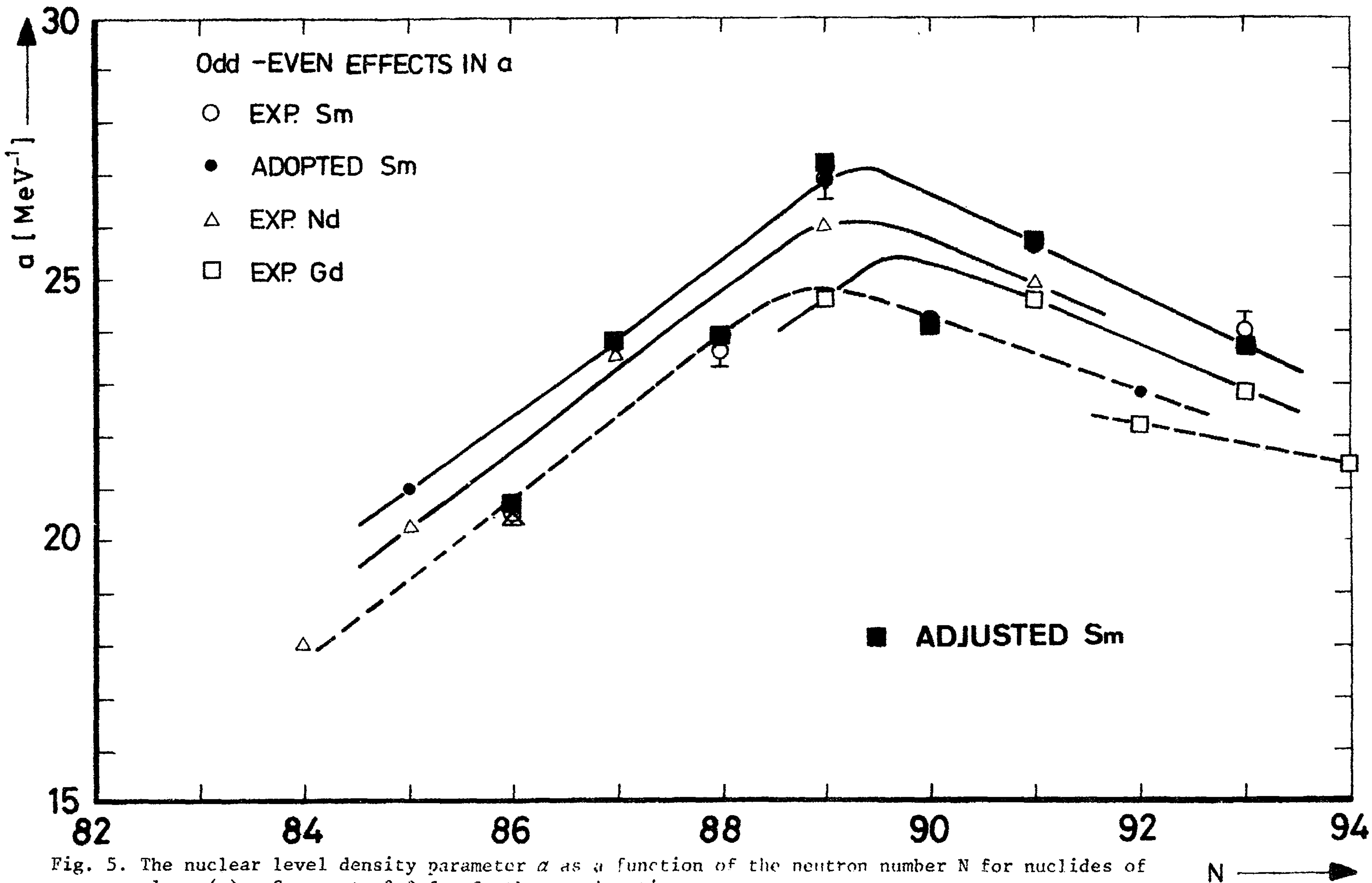

Fig. 5. The nuclear level density parameter a as a function of the neutron number N for nuclides of class (c). See sect. 3.3 for further explanation.

We *conclude* this section by saying that the adjusted values of a give useful and additional information to the evaluator who has to rely on systematics for the prediction of the level density of nuclides for which no (sufficient) resolved-resonance data exist, e.g. for important f.p. nuclides as ^{103}Ru and for some Pd isotopes.

3.4. Systematics of $<\Gamma_\gamma>$

In ref. |28| calculations of $<\Gamma_\gamma>$ have been reported based upon the Brink-Axel estimate, in which the primary γ-ray transitions were summed over levels with known spin and parity, whilst a "continuum" contribution was added. We have repeated these calculations with input parameters (level scheme data and a-parameters) as in the RCN-2 evaluation. In these calculations we have also introduced various estimates for the spin and parity distributions of levels in the continuum. Both experimental giant resonance parameters and those obtained from systematics have been used. Moreover we have performed calculations with the single-particle estimate, i.e. $<\Gamma_\gamma>/D = k_{E_1}\ \varepsilon_\gamma^3\ A^{2/3}$ for each primary γ-ray transition of energy ε_γ. From experimental data obtained for nuclides with masses from A = 50 to A = 250 the value $k_{E_1} \approx 2.5\times10^{-9}$ MeV^{-3} has been deduced |25|. The M_1 capture width is about a factor of 7 smaller than the E_1 capture width |25| and has been neglected in our calculations.

None of the results obtained was very satisfactory. It appeared that there were quite large differences between calculated and experimental capture widths, in particular for the odd-Z isotopes (class a). An interesting result of these calculations was that when an equal-parity distribution was used for all levels in the continuum, the experimental ratio $<\Gamma_\gamma(\ell=0)>/<\Gamma_\gamma(\ell=1)>$ for the Mo-isotopes was more or less reproduced, without any assumption about other reaction mechanisms like valency capture, etc. Thus the differences between $<\Gamma_\gamma(\ell=0)>_J$ and $<\Gamma_\gamma(\ell=1)>_J$ in the calculations were only caused by differences in the spins and parities of the discrete level schemes, Use of the parity distribution function for levels in the continuum as given by Igarasi |36| led to too large differences between s- and p-wave capture widths.
The overall comparison between experimental and calculated capture widths was very disappointing. It appeared that there was no clear preference for either the Brink-Axel model or the single-particle model, even with a modified value of k_{E_1}. From a least-squares fit of k_{E_1} to experimental

capture widths of 22 nuclides ($90 < A < 155$) the value $k_{E_1} = (1.5\pm0.4) \times 10^{-9}$ was found, with $\chi^2/21 = 40$. This value of k_{E_1} is much lower than the value given before. For nuclides of classes (b) and (c) the fits were slightly better: $k_{E_1} = (1.5\pm0.4) \times 10^{-9}$ with $\chi^2/8 = 7.2$ and $k_{E_1} = (2.2\pm1.7) \times 10^{-9}$ with $\chi^2/5 = 4.7$, respectively. A small improvement was obtained when adjusted a-parameters (table 5b) were used.

From the above mentioned experience with theoretical $\langle\Gamma_\gamma\rangle$ calculations it is clear that these calculations have limited value. Only when theoretical values of $\langle\Gamma_\gamma\rangle$ are fitted to experimental data in a small mass range one might hope to have some success in the prediction of capture widths. This is in fact the same situation as holds for the systematics of a. Therefore we expect that also for $\langle\Gamma_\gamma\rangle$ the use of adjusted capture widths as given in table 5b might be of some help to the evaluator.

3.5. Results for individual nuclides

Some global results of adjustment of σ_c have been given in table 4. Note the large reduction in uncertainty of $\langle\sigma_c\rangle$ after adjustment, in particular when also CFRMF measurements are utilized. This is partly so because the CFRMF spectrum is relatively close to the SNR-300 spectrum which was used in table 4 to average σ_c. Further general conclusions are given in sect.4.

A rather detailed discussion of results of adjustments has been given in ref.|26| for the following f.p. nuclides:
class(a): ^{99}Tc, ^{103}Rh, ^{127}I, ^{133}Cs, ^{139}La, ^{141}Pr;
class(b): ^{95,97,98,100}Mo, ^{101,102,104}Ru, ^{105,107,108}Pd.
It has to be noticed that the adjustments given in ref.|26| are mostly slightly different (viz. with other spectrum and implicit spectrum adjustment, see also sect. 2.3) from those given in the more recent work of ref.|6|; however, all qualitative conclusions remain the same. Meanwhile, data for more nuclides have been analysed and results from adjustments with combined STEK and CFRMF data are available |7|. The main observations in ref.|26| are repeated here with some additional remarks and extensions. The discussion is restricted to the energy range above about 1 keV. References to differential measurements are given in a different notation, see sect. 5.2. Plots of these data have been given in refs. |26,34|.

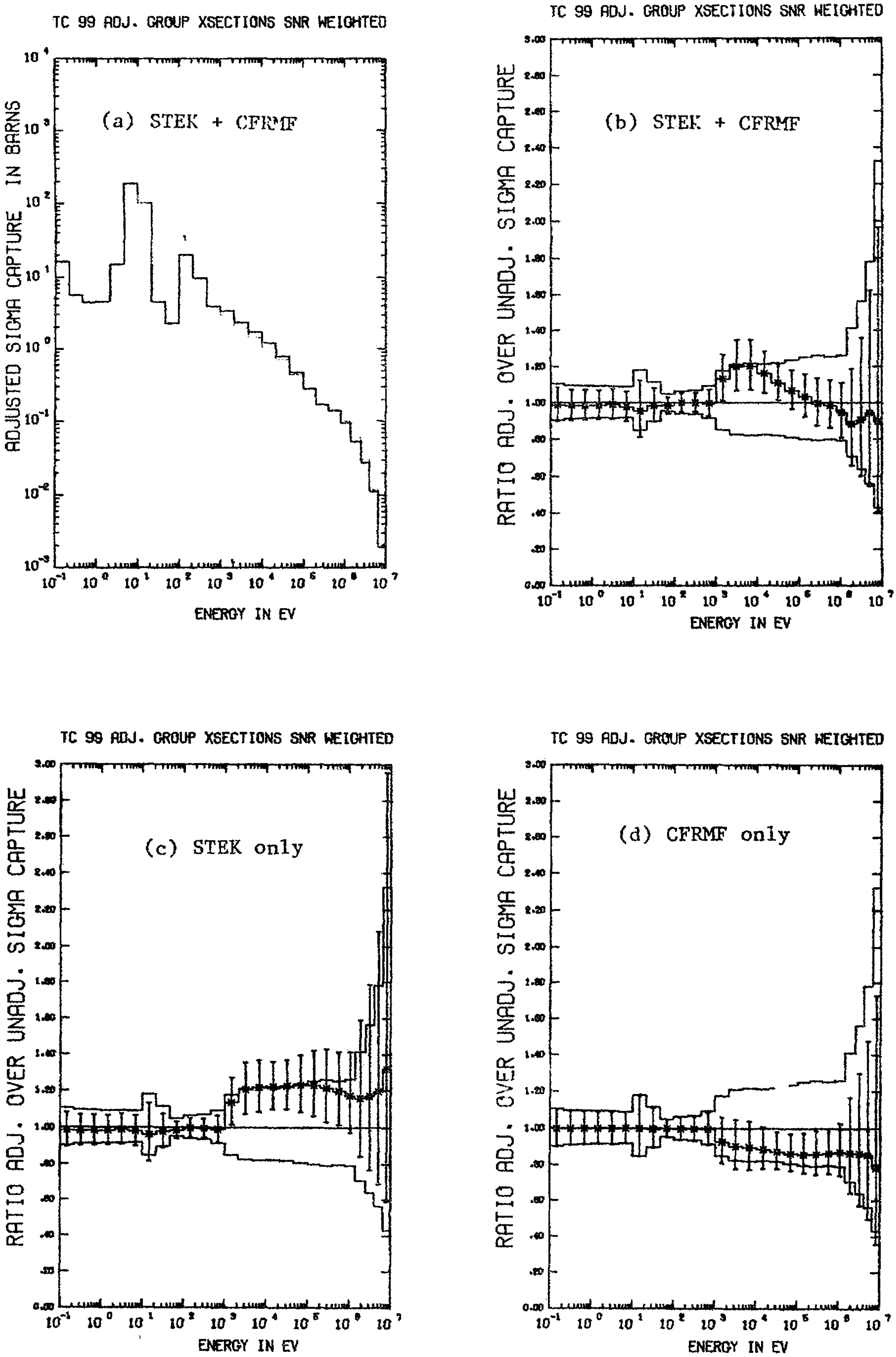

Fig. 6. Adjusted capture cross sections of ^{99}Tc based upon STEK + CFRMF data (a, b), STEK data only (c), or CFRMF data only (d). See fig. 3 for explanation of symbols (Figure from ref. |7|).

3.5.1. Class (a)

^{99}Tc

Up to 50 keV all evaluations |26| are in reasonable agreement with experimental points |Ch73|. At higher energies the evaluations become more and more different. In the MeV range the differences are extremely large due to uncertainties in the level scheme of ^{99}Tc: ENDF/B-IV gives the lowest; JENDL-1 gives the highest value of σ_c. In the (revised) RCN-2 evaluation the most recent level scheme data |Sv76| have been used, which lead to much lower values of σ_c than previously adopted, but not so low as in the ENDF/B-IV evaluation. From combined STEK and CFRMF measurements if follows |7| that at low energies (up to 100 keV) σ_c of RCN-2 is too small, whereas at high energies σ_c of RCN-2 is too large; see fig. 6. It is interesting to note that when STEK and CFRMF measurements are used separately in the adjustment calculation, quite different results are obtained. This indicates a discrepancy between STEK and CFRMF measurements, although the sensitivities to σ_c of these measurements are rather different. The ENDF/B-IV cross section appears to be much too low, however.

^{103}Rh

Except for the ENDF/B-IV evaluation, which is clearly too high compared with most differential experimental data, the evaluations of RCN-2 (revised), JENDL-1 and CNEN/CEA are about the same up to 0.8 MeV |26|. In the MeV range the differences are much larger; here the ENDF/B-IV evaluation is the lowest. Both STEK and CFRMF data are in perfect agreement with the RCN-2 evaluation |7|. In the energy range from 1 keV to 10 keV σ_c has a relatively high uncertainty (15% to 20%); from 10 keV to 0.5 MeV the uncertainty is 10% to 15%. It has to be noted that the group cross sections for ENDF/B-IV and CNEN/CEA as given in fig. 3b of |26| are wrong for group 13 (2.15 - 4.65 keV), due to an error in the processing code. The mutual differences between RCN-2 (revised), CNEN/CEA and JENDL-1 are in fact rather small for group 13. The ENDF/B-IV cross section still seems to be too high.

^{109}Ag

There are appreciable differences between the various evaluations |26| due to the fact that there are different series of differential measurements which seriously disagree with each other |We60 , Po65 , Ko66|. The adjustments |7| based upon STEK and CFRMF (7% to 10% upwards) exclude the low capture cross section data measured by Kononov et al. |Ko66|. The ENDF/B-IV evaluation is almost certainly much too low.

The adjustments for σ_c of <u>^{107}Ag</u> (based upon CFRMF only |7|) are also in upward direction (≈5%). The adjusted capture cross section for <u>natural silver</u> is therefore also about 5% to 10% higher, which only improves the already reasonably good agreement between σ_c of RCN-2 and the recently measured differential data: |Ch73a|, |He73| (normalized to σ_c of Au from ENDF/B-IV), and |Ya75|.

^{127}I

The most recent measurements of resolved resonance parameters performed at CBNM |Ro76| lead to the following parameters: $S_o = (0.80 \pm 0.09) \times 10^{-4}$, $S_1 = (3.4 \pm 1.4) \times 10^{-4}$, $<\Gamma_\gamma> = 86 \pm 9$ meV and $D_{obs} = 13.3 \pm 1.0$ eV. Use of these parameters in a statistical model calculation leads to too low values of σ_c compared with most available differential capture data. Moreover, the value of S_1 obtained from total cross section data is much lower: $S_1 = (1.6 \pm 0.5) \times 10^{-4}$ |Ca74|. Therefore, in the (revised) RCN-2 evaluation the parameters have been tuned to the values (see table 5): $S_o = (0.80 \pm 0.16) \times 10^{-4}$, $S_1 = (2.0 \pm 0.5) \times 10^{-4}$, $<\Gamma_\gamma> = 95 \pm 10$ meV, and $D_{obs} = 12.2 \pm 1.8$ eV. The RCN-2 capture cross section calculated with these adjusted parameters is still below most experimental points and is about 7% lower than the ENDF/B-IV curve. However, the adjusted curve |7| based upon STEK and CFRMF data is approximately 10% higher in the energy range from 2 keV to 10 MeV, again in agreement with most experimental points |34| and relatively close to ENDF/B-IV values.

As a conclusion we could say that the statistical model calculations based on the recent average parameters of Rohr et al.|Ro76| give too low capture cross sections. The adopted parameters in the RCN-2 evaluation still give too low values of σ_c. After adjustment there is reasonable agreement with ENDF/B-IV and differential capture data.

^{129}I

For this nuclide there are no differential σ_c measurements in the region above 168 eV. The adopted RCN-2 values for S_o, S_1 and $<\Gamma_\gamma>$ (see table 5) are based upon values for ^{127}I and some systematics. The value of D_{obs} = 30 ± 10 eV follows from resolved resonance parameters. The RCN-2 curve

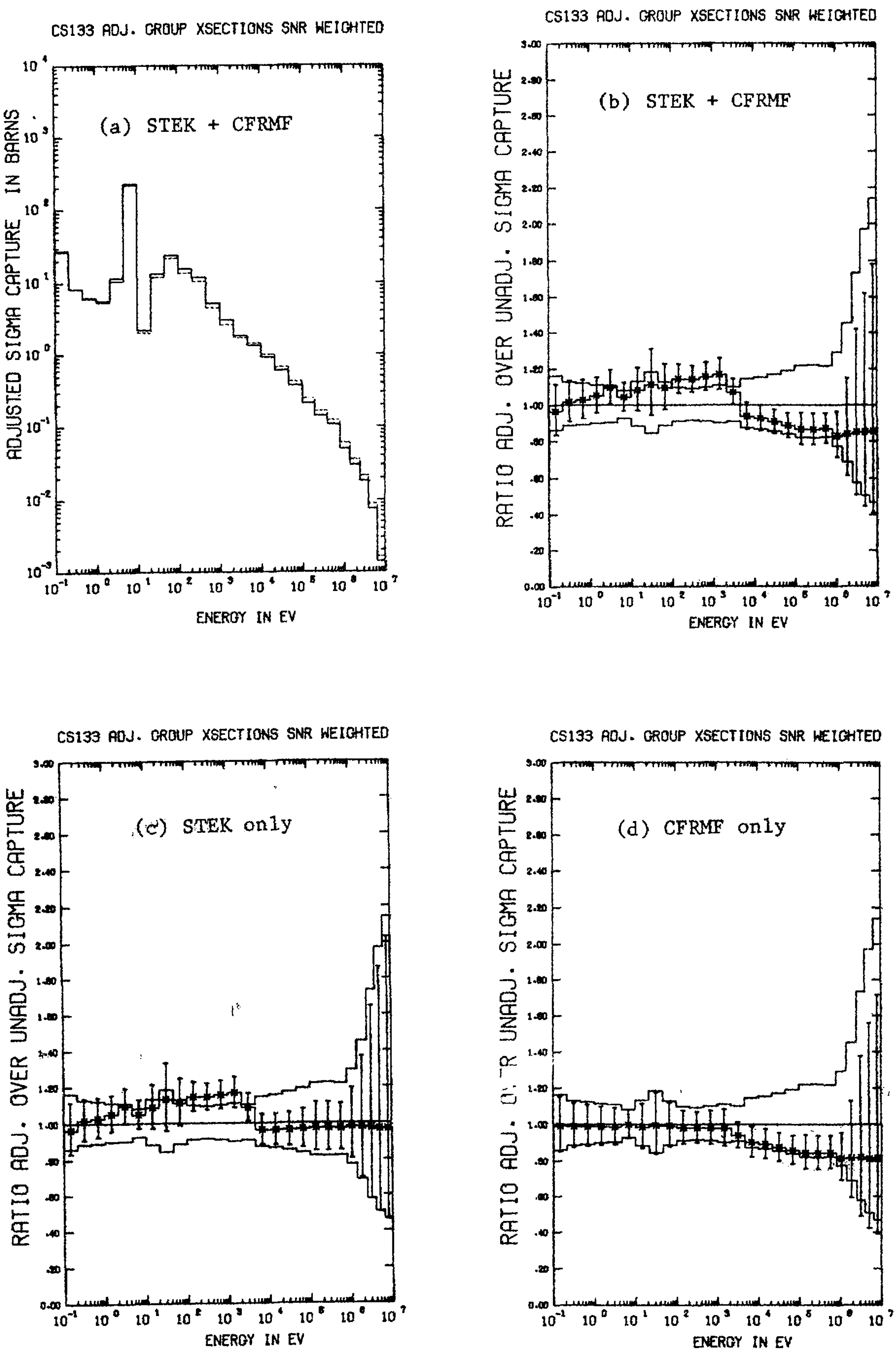

Fig. 7. Adjusted capture cross sections of ^{133}Cs based upon STEK + CFRMF data (a, b), STEK data only (c), or CFRMF data only (d). See fig. 3 for explanation of symbols (Figure from ref. |7|).

calculated with these parameters is much lower than those of JENDL-1 and ENDF/B-IV. The adjusted curve based on STEK and CFRMF |7| is still lower. At energies just above the resolved resonance region the adjustment (via parameter S_0, see table 5) is even more than -50%. The ENDF/B-IV and JENDL-1 evaluations are almost certainly much too high.

<u>^{133}Cs</u>

For σ_c of ^{133}Cs (up to 500 keV) there are two series of measurements which are discrepant, those of Kompe |Ko69| and Russian data |Po62, To67|. The ENDF/B-IV, CNEN/CEA and the RCN-2 evaluations are close to the relatively high data of Kompe; the JENDL-1 evaluation is much closer to the Russian data |26|. In the MeV range the evaluations are rather discrepant. The adjustments based upon CFRMF integral data |7| give much lower σ_c values than those of STEK |6| in the statistical model energy range (above 3.5 keV). It is rather curious that there is a relatively large upward adjustment in the resolved resonance range (see fig. 7); possibly resolved resonances have been missed in the experiments. The JENDL-1 evaluation is almost certainly too low.

<u>^{139}La</u>

This nuclide has a magic number of neutrons. The capture cross section is relatively low therefore. The scattering reactivity worth for the hard spectrum of STEK-500 is about 50% larger than the calculated capture worth. Therefore, the adjustments for σ_c of RCN-2 based upon STEK measurements |6| in the hardest spectra were not used. In the highest part of the resolved resonance region (1-10 keV) the STEK measurements indicate an upward adjustment |6| of about 25% to 30%, which is in agreement with low-resolution capture data |Ko64|. Thus, probably resolved resonances have been missed in that energy region. The adjustments based upon CFRMF |7| indicate about 7% lower capture cross sections than those of RCN-2. The adjusted σ_c is in agreement with data of Zaikin et al. |Za71|. For $E > 10$ keV the ENDF/B-IV values of σ_c are about 20% to 30% higher, whereas the CNEN/CEA values are about equal to the adjusted cross sections. The RCN-2A cross section based upon CFRMF+STEK is recommended. The ENDF/B-IV curve is almost certainly too high.

^{141}Pr

Like for ^{139}La this nuclide has magic N, the capture cross section is low and the scattering effect in STEK-500 is large (about 60% of the capture worth). The RCN-2 evaluation is in very good agreement with both STEK and CFRMF |6,7| measurements; only in the resolved resonance region from 0.1 to 1 keV there is a -10% adjustment. There are three recent evaluations available |26|: ENDF/B-IV, CNEN/CEA and RCN-2. The ENDF/B-IV curve is the highest, the RCN-2 curve is the lowest one, close to the CNEN/CEA evaluation. From combined STEK and CFRMF measurements it follows that ENDF/B-IV capture cross sections are too high. This is also in agreement with the majority of the differential experimental data, see fig. 13a of |26|. The data of Stupegia et al.|St68| seem to be somewhat too high.

3.5.2. Class (b)

Even Mo-isotopes

The capture cross sections of the (even) Mo-isotopes are difficult to evaluate, for reasons already mentioned in sect. 3.1.(b), see also |26|. From comparisons between calculated and experimental point cross sections it follows that for the even Mo-isotopes the statistical model fails at neutron energies from 1 keV to about 100 keV. Nevertheless the statistical model has been used in all available evaluations to fit σ_c as good as possible to the measured data. In the RCN-2 evaluation the average resonance parameters (mostly deduced from BNL-325 |Mu73|; for ^{98}Mo from |Ch76|) have been modified to fit these data. In particular the p-wave capture width has been increased (probably too much) in order to take into account valency effects (see also Appendix 3 of |26|). The results of adjustments are also difficult to interprete due to the fact that the scattering corrections to the reactivity worths of even Mo-isotopes in the hardest STEK cores are very high. Moreover there are discrepancies between the adjusted capture cross sections and recent high-resolution data of Musgrove et al. |Mu76a|, measured for the stable Mo-isotopes from 3 to 90 keV.

The problems can be summarized as follows:

(a) The statistical model is not adequate.
(b) High scattering corrections for hard STEK cores.
(c) Adjusted cross sections based upon STEK |6| have become smaller, whereas recent high-resolution data for $E < 90$ keV indicate *higher* values of σ_c.
(d) Adjusted cross sections based upon STEK |6| are closer to experimental data points for ^{98}Mo and ^{100}Mo at $E > 90$ keV |26|.

(e) Adjusted average parameters D_{obs} and $<\Gamma_\gamma>$ (table 5b) are in disagreement with values given by Musgrove et al. |Mu76a|.

(f) CFRMF activation results |7| disagree with STEK results for ^{98}Mo, but agree with STEK results for ^{100}Mo in the direction of the adjustment.

We conclude that the cross sections of the even Mo-isotopes need to be re-evaluated utilizing the most recent differential data |Wa73, Mu76a, Ch76, We76| and integral measurements. Probably it is better to discard the statistical model for $E < 90$ keV and to draw an "eye-guided" curve through the measurements of Musgrove et al. Above 90 keV the capture cross sections should be much lower than those of RCN-2. See also the discussion on ^{98,100}Mo capture cross sections in |26| and the discussion for natural Mo.

^{95,97}Mo

The problems mentioned for the even Mo-isotopes are not so serious for the odd ones. The evaluators have tried to fit the data of Kapchigashev and Popov |Ka64| for E = 1 to 60 keV. These data are in good agreement with recent high-resolution measurements of Musgrove et al. |Mu76a|. However, it is not very well possible to fit the high capture data of ^{97}Mo for E = 1 to 10 keV with the statistical model. Therefore an "eye-guided" curve was adopted in the JENDL-1 evaluation. The other evaluations of σ_c for ^{97}Mo are probably too low in that energy range (in particular the ENDF/B-IV evaluation). For ^{95}Mo all evaluations are in reasonably good agreement with the measurements for E = 1 to 90 keV. Both for ^{95}Mo and ^{97}Mo the differences between the various evaluated σ_c curves become somewhat more serious above 100 keV. The adjustments based on STEK are about -10% for ^{95}Mo at energies above 2 keV, but can be neglected for ^{97}Mo. The overall differences between RCN-2A, ENDF/B-IV, CNEN/CEA and JENDL-1 are not important (see table 4b). Nevertheless similar conclusions as given for the even Mo-isotopes apply.

Natural Mo

The RCN-2 capture cross group sections for natural Mo (of interest for construction materials), calculated by summation of the isotopic contributions, have been plotted in fig. 8, together with recent experimental differential data and the ENDF/B-IV and KEDAK-3 evaluated curves. The recent data measured by Musgrove et al. |Mu76a| are not shown in

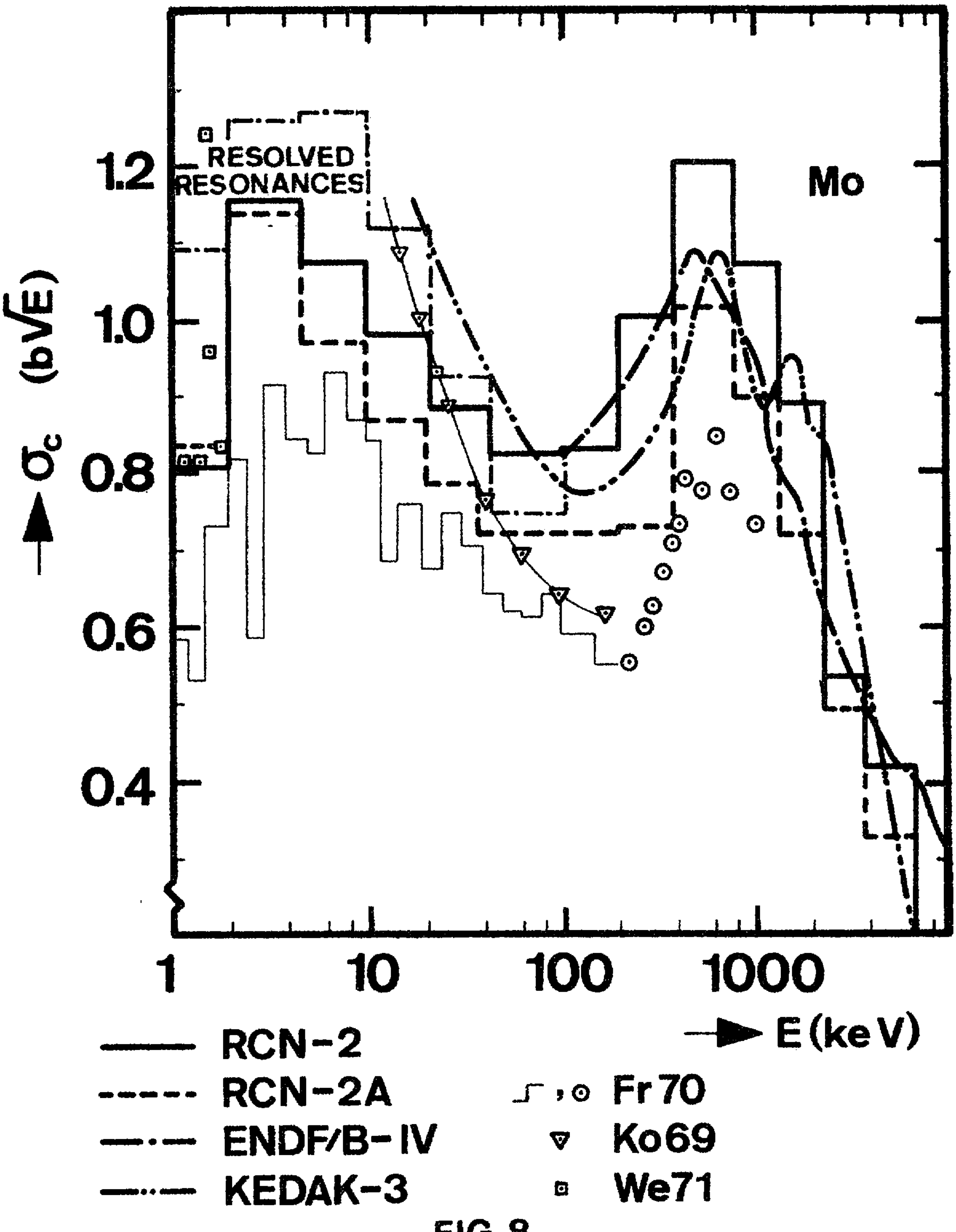

FIG. 8

Capture (group) cross sections for natural Mo according to RCN-2, RCN-2A, ENDF/B-IV and KEDAK-3 evaluations.
For comparison the most recent capture data have been plotted.

this figure, but they are rather close to the KEDAK-3 values. The adjusted RCN-2A curve is much lower than the data from existing evaluations, more in the direction of the data of Fricke et al. |Fr70|. Most data plotted in fig. 8 suggest that all evaluated σ_c are too high above about 50 keV. The RCN-2A adjustments in that energy region are correlated with those for $E < 50$ keV. This could explain the too low RCN-2A values in the keV range. A new evaluation of σ_c with relatively high values for $E < 50$ keV and relatively low values for $E > 50$ keV is recommended.

^{101}Ru

The evaluations for this nuclide are completely based upon statistical-model calculations with parameters obtained from average resolved-resonance parameters. In ENDF/B-IV resolved-resonance parameters up to 113 eV have been used, in all other evaluations new parameters up to 666 eV have been used |26|. Above 1 MeV the JENDL-1 and CNEN/CEA capture cross sections are relatively high, which is possibly due to a different treatment of the target spin cut-off parameter σ^2 (see Appendix 1 of |26|). The differences between integral data calculated with the existing evaluations are large (see table 4b). The adjustments based upon STEK integral measurements are very small |6, 12|. ENDF/B-IV gives probably too low values for σ_c.

^{102,104}Ru

There are rather large differences between the various evaluations for σ_c of ^{102}Ru and ^{104}Ru, because there are only three or four resolved resonances known and, moreover, only very few - discrepant - point cross sections are available |26|. See table 4b for a global view of the differences. The adjustments based upon STEK |6, 12| and/or CFRMF |7| show that the capture cross section for these nuclides must be very low, relatively close to the ENDF/B-IV evaluation. Meanwhile, cross section measurements have been performed by Hockenbury et al. |Ho76|, which also yielded very low capture cross sections. Hockenbury et al. assume that $<\Gamma_\gamma>$ is about 60 meV, both for ^{102}Ru and ^{104}Ru. For the (unadjusted) RCN-2 evaluations values of 275 meV (fitted to experimental cross sections |Ly59, Ma57, Mu73b, Sc69|) and 79 meV were adopted for $<\Gamma_\gamma>$ of ^{102}Ru and ^{104}Ru, respectively. The adjusted parameters are much lower: $<\Gamma_\gamma> = 170$ meV and $<\Gamma_\gamma> = 88$ meV (from table 5b). The RCN-2A capture cross sections are recommended. For ^{104}Ru also the ENDF/B-IV cross sections could be used.

^{105}Pd

For this very important fission product nuclide there is a number of resolved resonance parameters available |Mu73| as well as - quite recent - capture data from Knox et al. (plotted in ref.|34|), Hockenbury et al.|Ho75| and Macklin et al.|Ma75| (unpublished). The existing evaluations of σ_c |26| are in very good agreement with each other for E = 10 keV to 100 keV. Large discrepancies between these evaluations exist however outside this region. The data of Knox et al. and of Hockenbury et al. are in reasonably good agreement with most evaluations, except from 70 to 150 keV, where the experimental points seem to be wrong (see plot in |34|). From adjustments based upon STEK integral data an increase of about 7% follows for σ_c at energies greater than 10 keV and even more (up to 20%) for energies from 150 eV to 10 keV, see fig. 3. It has to be noted that in this low energy range the RCN-2 evaluation was already very much higher than the other evaluations. This has to be attributed to the relatively high adopted value of S_o, see table 5a. The adjustment of $<\sigma_c>$ (see table 4b) is appreciably higher than reported previously |26|.

^{107}Pd

There are no capture data available for this nuclide. Even the thermal capture cross section is unknown. Therefore appreciable differences exist between the various σ_c evaluations, of which the RCN-2 values are the highest and the ENDF/B-IV values are the lowest |26|. The main uncertainty in σ_c arises from the parameter α which has been estimated from systematics (see fig. 4).

In the STEK facility a fission-product sample was used containing about 16% ^{107}Pd and 49% ^{105}Pd. Therefore, the results have to be interpreted with care. The adjustments are positive (<11%) for E = 50 eV to 20 keV and negative (-2 to -28%) for higher energies. The same f.p. Pd sample has been used for differential total and capture cross section measurements performed at Rensselaer Polytechnic Institute (USA). Results are not yet available. Meanwhile it seems that the relatively high RCN-2A cross section has to be recommended. Note the very large reduction of error in $<\sigma_c>$ after adjustment (see table 4b).

Even Pd-isotopes

Evaluations for the stable even Pd-isotopes (ENDF/B-IV, CNEN/CEA, RCN-2) in the energy range above 1 keV are based on statistical-model calculations with parameters obtained from systematics, because there are only very few resolved resonances known and only for σ_c of ^{108}Pd and ^{110}Pd there are some -rather old- measured points |34|.

For the values of a and $<\Gamma_\gamma>$ adopted in the RCN-2 evaluation we refer to table 5b.

The results of adjustments of σ_c can be summarized as follows:

^{104}Pd: small adjustments; RCN-2A is much lower than the other evaluations.

^{106}Pd: positive adjustments of about 15% to 30%; RCN-2(A) in between other evaluations.

^{108}Pd: small adjustments, CFRMF adjustments in perfect agreement with STEK adjustments; RCN-2(A) in agreement with 24 keV and 195 keV data of Lyon and Macklin |Ly59|, but 60% lower than measured data of Weston et al. |We60|; other evaluations not very much different |26|.

^{110}Pd: large negative adjustments of about -50%, also the CFRMF data indicate very low capture cross sections; RCN-2A is in agreement with 24 keV meausrement of Chaubey and Sehgal |Ch66| and ENDF/B-IV.

Natural Pd

The RCN-2 capture cross section for natural Pd, calculated by summation of the isotopic contributions, agrees rather well with the data of Kompe |Ko69| (10 keV to 200 keV). There are no significant adjustments to σ_c. For energies from 1 keV to 10 keV the RCN-2(A) curve is somewhat below the data points of Block et al. |B161|.

3.5.3. Class (c)

$^{147,149}Sm$

For these isotopes the resolved resonances are well known. Recently, also point cross sections from about 20 keV to 300 keV have been measured by Yurelov et al. |Yu75| and - only for ^{149}Sm - by Hockenbury et al. |Ho76a| (unpublished). In the past only the 24 keV data of Macklin et al. |Ma63a| were known. However, it is not possible to fit the calculated capture cross sections to the - very high - recent data

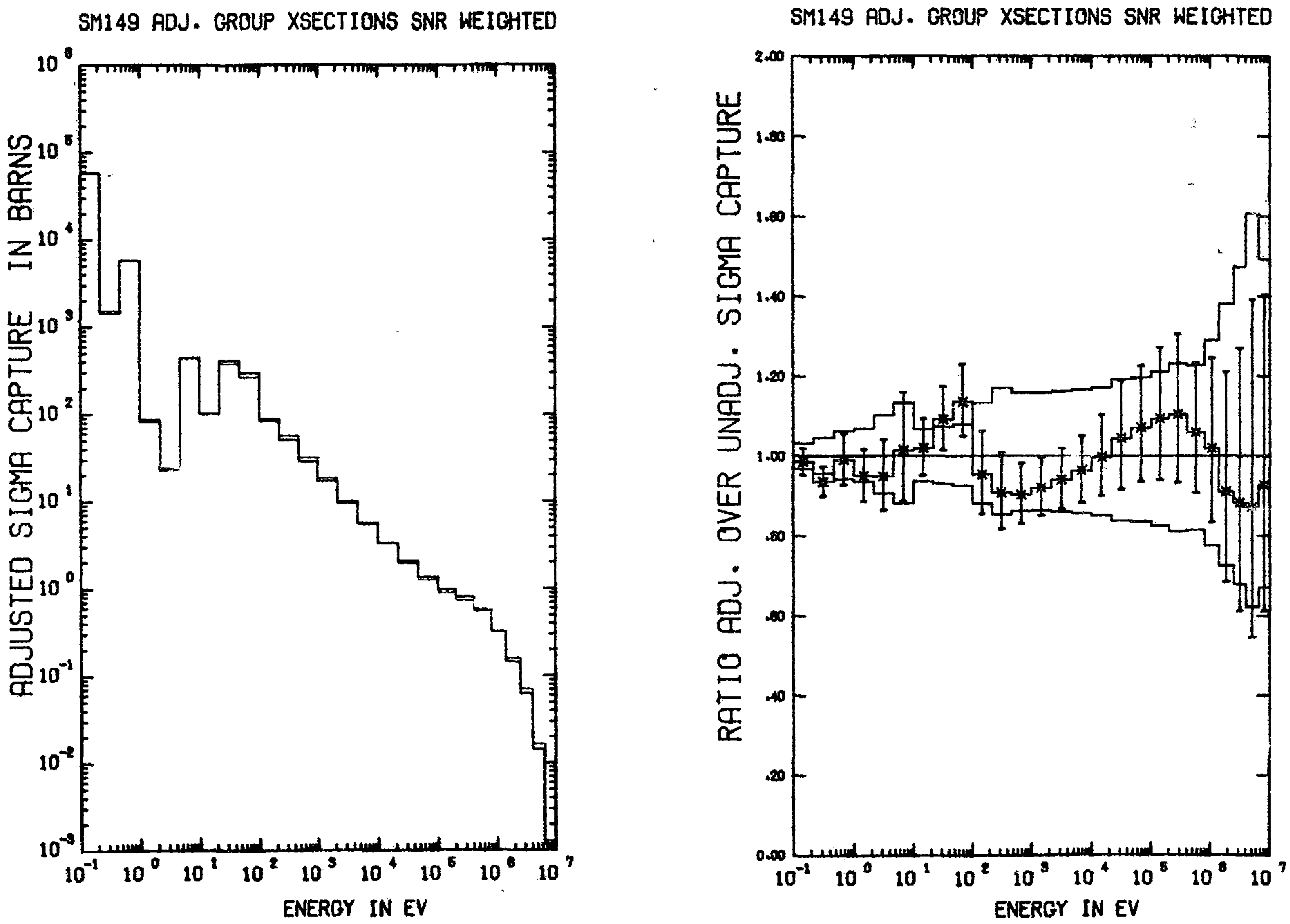

Fig. 9. Adjusted capture group cross sections of ^{149}Sm. See fig. 3 for explanation of symbols.

when using realistic average resolved-resonance parameters. The RCN-2 adopted values of S_1, D_{obs} and $\langle\Gamma_\gamma\rangle$ (see table 5) are about one standard deviation different from the most likely values deduced from optical model (S_1) or resolved resonances (D_{obs}, $\langle\Gamma_\gamma\rangle$). As a result the RCN-2 σ_c evaluation is higher than all other evaluations, but still 30% to 50% lower than the recent data. The calculated values of σ_c at 24 keV for ^{147}Sm and ^{149}Sm are still close to the old values of Macklin et al.

From STEK integral data it follows that σ_c of ^{147}Sm is adjusted by about -10% for E > 100 eV. The capture cross section of ^{149}Sm is adjusted by about -10% for E around 1 keV and by about +10% for E around 100 keV (see fig. 9). Therefore the results from STEK are in disagreement with the most recent experimental differential data. The adjustment for σ_c of ^{147}Sm is connected with an adjustment of $\langle\Gamma_\gamma\rangle$ (see table 5b). For ^{149}Sm the values of the S_o, S_1 and S_2 strength functions are adjusted.

Also after adjustment the RCN-2 capture cross sections are higher than those of the other evaluations, in particular ENDF/B-IV (see also table 4c).

^{151}Sm

No capture measurements for energies above the resolved-resonance region are known. Yet, the uncertainty in σ_c based on the statistical-model calculation is small (fig. 10). The is due to the fact that the average parameters based on the recent resolved-resonance parameters of Kirouac and Eiland |Ki75| have rather small uncertainties. he differences between the various (unadjusted) evaluations are also very small for E = 100 eV to 100 keV (see also table 4c). Above a few hundred keV there are very large differences in σ_c due to the use of different level schemes of ^{151}Sm or due to differences in the continuum calculation of σ_c (Appendix 1 of |26|). Of all evaluations RCN-2 is the lowest one in the MeV range.

From STEK measurements rather large negative adjustments of -10% to -30% result (fig. 10), mainly by a decrease of S_o. The adjustments are - and this is an exception - larger than the a-priori uncertainty limits. Moreover, the uncertainty in a number of adjusted parameters is larger than before adjustment (table 5). The adjusted value of S_o for ^{151}Sm

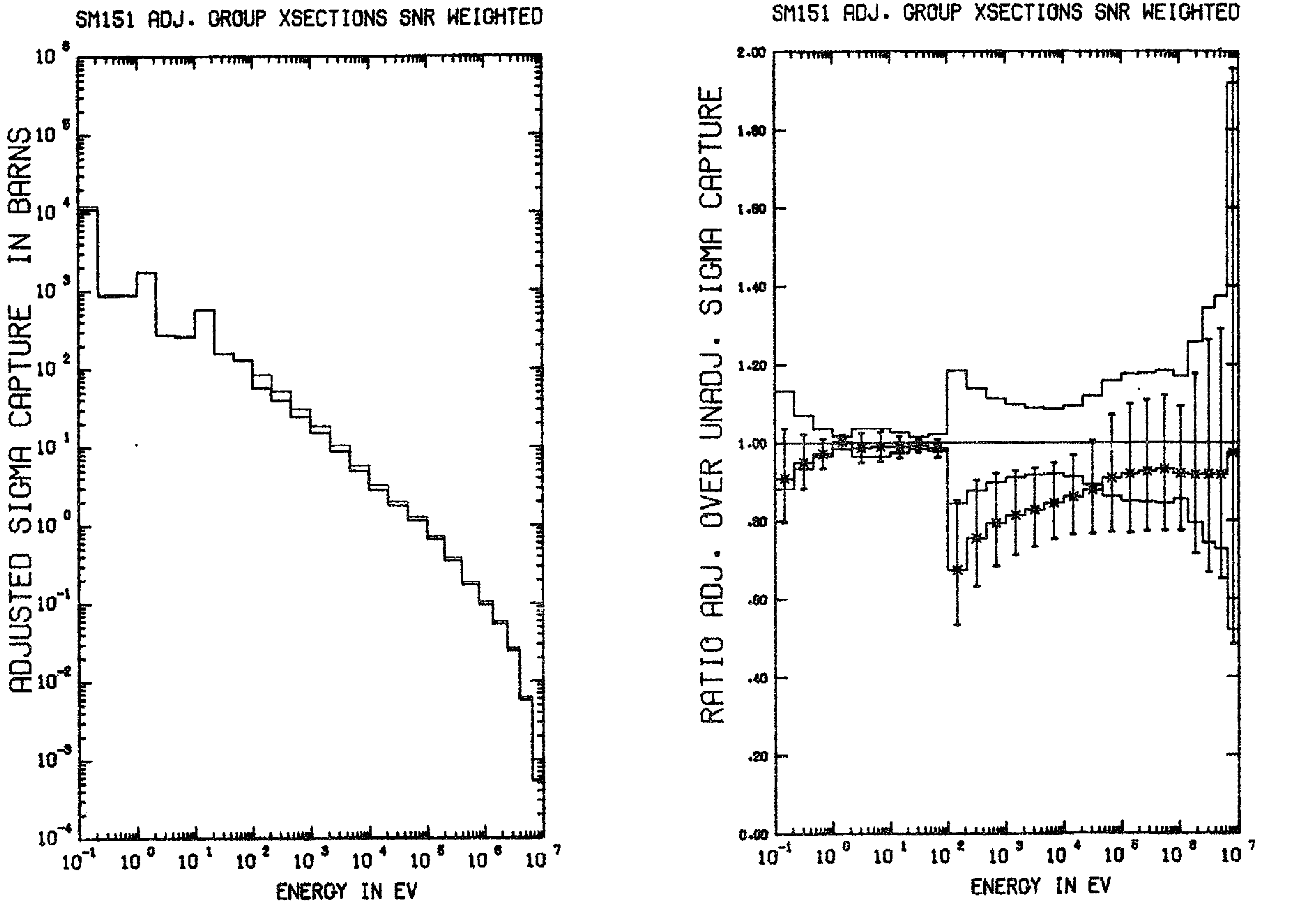

Fig.10. Adjusted capture group cross sections of ^{151}Sm. See fig. 3 for explanation of symbols.

(and for ^{150,152}Sm) is not in agreement with the observed "odd-even effect" in S_o |Ki75|.

Even Sm-isotopes

For the isotopes ^{148,150,152,154}Sm there are only two evaluations available: ENDF/B-IV and RCN-2. These evaluations are primarily based upon statistical model calculations using average resolved resonance parameters. For ^{148}Sm no resolved resonances are known; therefore nuclear systematics has to be used to determine the statistical model parameters, see sect. 3.3. At 24 keV there are capture cross section measurements for all of these isotopes |Ma63a|; for ^{152}Sm (fig. 2) and ^{154}Sm (fig. 11) there are more point cross section data. In the RCN-2 evaluation the adopted value of D_{obs} for ^{148}Sm and the values of $<\Gamma_\gamma>$ for ^{152,154}Sm have been slightly modified to fit these data. The fits for σ_c of ^{152,154}Sm are not very good, however.

After adjustment based upon STEK data the cross section of ^{148}Sm is not changed very much, whereas adjustments of +25%, 15% and 15% are found for ^{150}Sm, ^{152}Sm and ^{154}Sm, respectively. Due to the adjustment the discrepancies with ENDF/B-IV have been increased.

For σ_c of ^{154}Sm (see fig. 11) there are large discrepancies with (recent) data. These data could be explained much better if there were an additional level of ^{154}Sm around 30 keV, because the data of Fawcett et al. show a threshold at that energy. However there are no (other) indications for such a hypothesis.

Natural Sm

In fig. 12 the adjusted and unadjusted (RCN-2) capture cross sections for natural Sm, calculated by a summation over all isotopic contributions (except ^{144}Sm), have been plotted together with experimental data, see references in |34|. In addition the experimental value of Macklin et al. (obtained by summing isotopic data |Ma63a|) and some recent data of Yurelov et al. (obtained from a graph in |Yu75|) have been plotted. The adjusted curve is obtained with methods described in sect. 2.5. The differences between the unadjusted and adjusted curves are very small. There is a serious discrepancy between the evaluated points (still in agreement with KFK values) and most other data from about 10 to 50 keV.

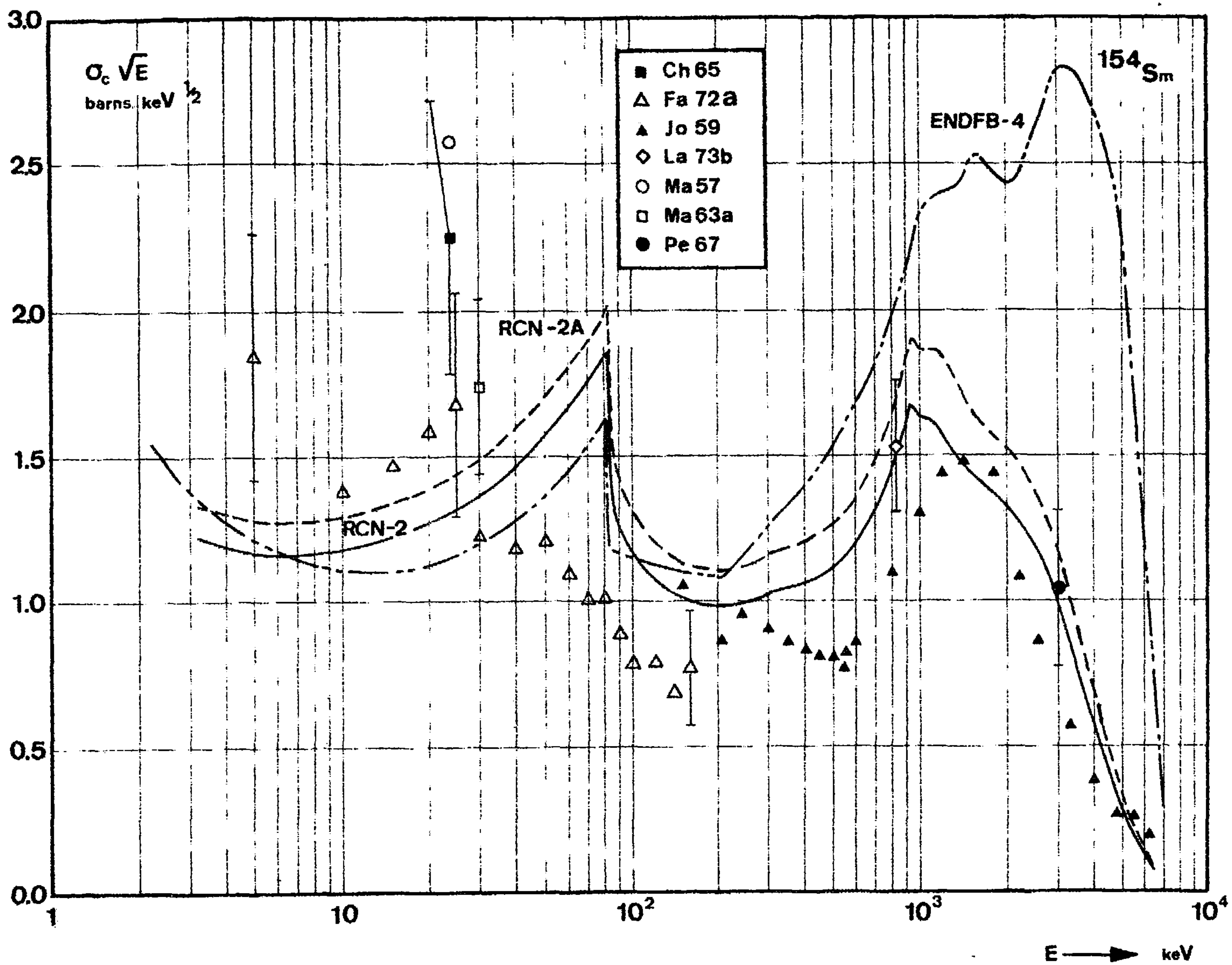

Fig.11. Adjusted (RCN-2A) and unadjusted (RCN-2, ENDF/B-IV) evaluated capture cross sections for ^{154}Sm. Also plotted are available experimental points.

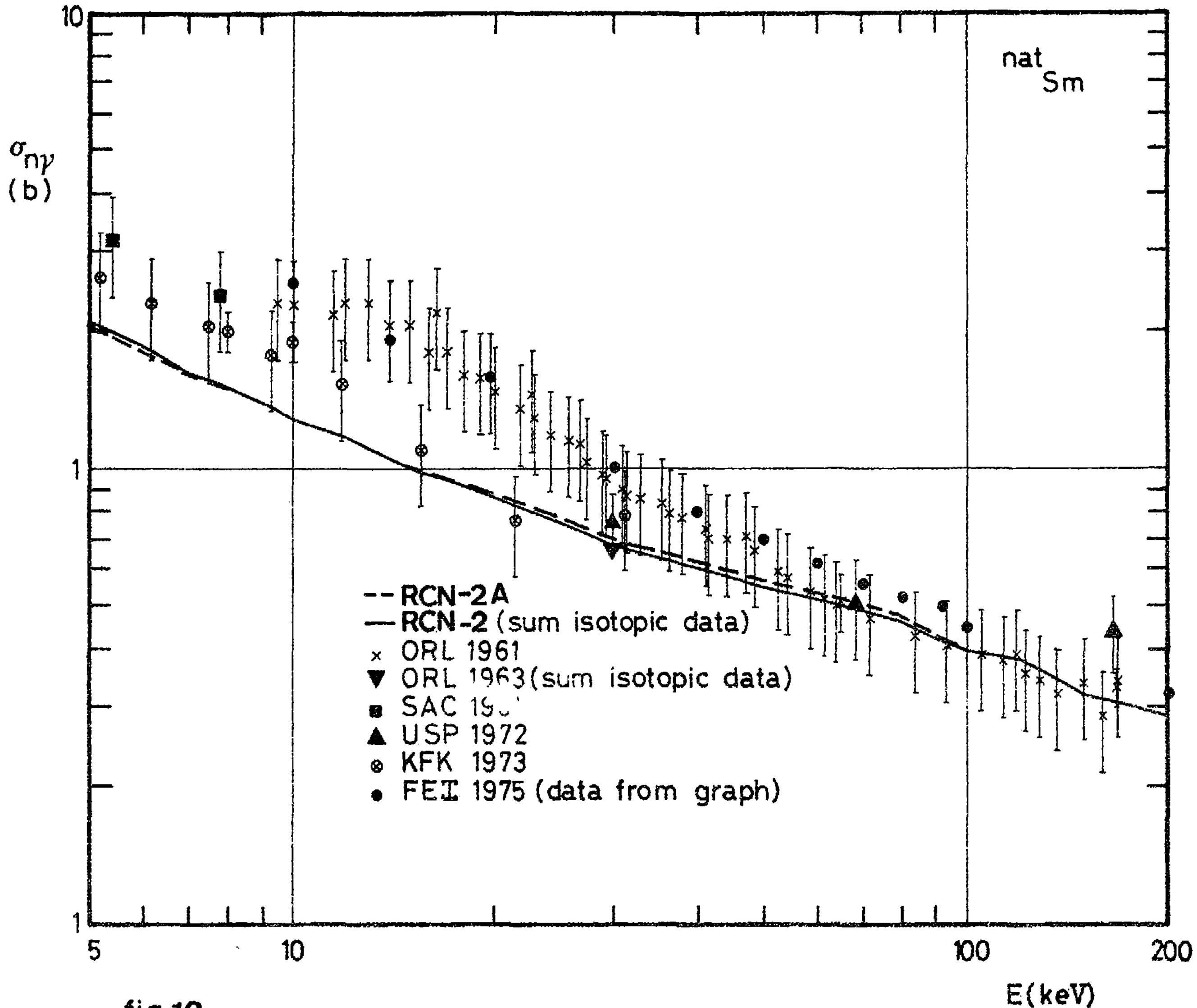

fig.12

Adjusted (RCN-2A) and unadjusted (RCN-2) evaluated capture cross sections for natural samarium. Also plotted are experimental points as given in ref. |34| and recent data from FEI |Yu75|.

4. SUMMARY AND CONCLUSIONS

(a) In this paper results obtained from adjustments of capture cross sections of f.p. nuclides have been reviewed. These adjustments are based upon integral data, i.e. STEK reactivity worths and CFRMF reaction rates. Only part of the available integral data have been analysed as yet. Before applying adjustments, the STEK data were *corrected* for non-f.p. admixtures and (inelastic) scattering effects, whereas the CFRMF data were corrected for (small) self-shielding effects. The experimental STEK data were not corrected for self-shielding, because this relatively large effect is a function of the cross sections which have to be adjusted (see discussion in sects. 2.1, 2.3).

(b) In the adopted *adjustment procedure* four levels are distinguished: adjustment of integral data |1|, group cross section adjustment (sect. 2.3), model-parameter adjustment (sect. 2.4) and point cross section adjustment (sect. 2.5). The following data (with uncertainties and correlation coefficients) have been used for these adjustments: experimental integral data |1|, sensitivities |1| (normalized neutron spectra), a-priori multi-group cross sections |6| and model parameters |5| used to calculate σ_c.

(c) An important role in the adjustment procedure play the *uncertainties in the nuclear model parameters* which induce uncertainties and correlations in the group cross sections (sect. 2.2). In the adopted model correlations between most a-priori parameters are neglected. The adopted uncertainties in these parameters are rather conservative, compared to estimates of Ribon et al., see sect. 2.2. In a large statistical model region from about 1 to 500 keV, the following parameters are most important: S_0, S_1, $<\Gamma_\gamma>$ and D_{obs} (see table 5). In a few cases also S_2 is of some importance. Uncertainties in unadjusted group cross sections have been given in table 1.

(d) The *dependence of adjusted data to the a-priori values and uncertainties* has been discussed in sect. 2.6. Adjusted integral data, multigroup constants (or point cross sections) and model parameters are almost independent, moderately dependent or rather strongly dependent on these a-priori values, respectively.

(e) *Systematical errors* in the adjusted cross section data could occur due to systematical errors in the (normalization of) integral measurements and in the sensitivities (spectra), which give estimated uncertainties of about 5% in the average cross sections. From comparisons with well-known capture cross sections and χ^2-tests there are no indications for larger systematic errors (sect. 2.6 and discussions in sect. 3.5).

(f) *Results of adjusted average cross sections* (capture rates in SNR-300) are given in table 4 for three classes of f.p. nuclides: (a) Odd-Z f.p. nuclides, (b) Even-Z f.p. nuclides with $Z < 50$, (c) Even-Z f.p. nuclides for $Z > 50$. These adjusted data probably do not depend very much on the a-priori values (sect. 2.6). After adjustment a large reduction in uncertainty is observed; the adjustments based on CFRMF data are mostly in agreement with those based on STEK. Some data of other evaluations are more than three standard deviations apart from the adjusted values (see table 4 and discussion in |1|).

(g) *Global results of parameter adjustment* have been given in sect. 3.2, table 5. The *s-wave strength function* is not adjusted very much; also the uncertainty is not reduced significantly. Exceptions are found for a few odd-mass nuclides for which the statistical model region commences at a relatively low energy and for nuclides of class (c). The "odd-even" effect experimentally observed in S_0 for the Sm isotopes has disappeared after adjustment. The *p-wave strength function* is adjusted for more nuclides than the s-wave strength function; its uncertainty is reduced in many cases, but the influence on σ_c is relatively small.
The largest adjustments and uncertainty reductions occur for the *level spacing* D_{obs}. The adjusted values of the level-density parameter a (table 6), derived from adjusted values of D_{obs}, give useful indications to improve the systematics of a (sect. 3.3, figs. 4,5). This is particularly of importance for the prediction of the level density of nuclides for which no (sufficient) resolved-resonance data exist, e.g. ^{103}Ru and some Pd isotopes. Another result of a-parameter adjustment is the confirmation of an odd-even effect

in a for the Sm-isotopes (fig. 5).
The *capture width* $<\Gamma_\gamma>$ is mostly known somewhat better than D_{obs}. The adjustments in $<\Gamma_\gamma>$ are less than those for D_{obs}; the uncertainty is reduced significantly for many nuclides. The results can be used to try to improve the systematics of $<\Gamma_\gamma>$, see the discussion in sect. 3.4.

(h) *Results of adjustments of group cross sections* have been discussed in sect. 3.5 for the following nuclides:
class (a): ^{99}Tc, ^{103}Rh, ^{107,109}Ag, natural Ag, ^{127,129}I, ^{133}Cs, ^{139}La, ^{141}Pr;
class (b): even Mo-isotopes, ^{95,97}Mo, natural Mo, ^{101,102,104}Ru, ^{105,107}Pd, even Pd isotopes, natural Pd;
class (c): ^{147,149,151}Sm, even Sm-isotopes, natural Sm.
The results have been compared with 26-group constants calculated from the unadjusted RCN-2 evaluation and from other recent evaluations: CNEN/CEA, ENDF/B-IV and JENDL-1. Moreover, the adjusted capture cross sections have been compared with available point cross sections.

In the thermal energy range adjustments are small and have not much meaning, because the sensitivities of STEK and CFRMF integral measurements are small in that region.
In the resolved resonance region the adjustments are mostly small with a preference for positive adjustments. For ^{133}Cs and ^{139}La there are indications that resolved resonances have been missed in the evaluation; $<\Gamma_\gamma>$ is also not known very well for these nuclides.

In the statistical-model region the adjustments are small for nuclides with reasonably well-known cross sections (like ^{93}Nb, ^{103}Rh). For many other isotopes the adjustments are not very large either (i.e. less than 10%). Larger adjustments occur for ^{99}Tc, ^{129}I, ^{133}Cs, ^{139}La and for some even isotopes of Mo, Ru, Pd and Sm. In most cases, (except perhaps for ^{99}Tc, ^{98}Mo, ^{133}Cs, ^{139}La) the adjusted data of STEK and CFRMF are in perfect agreement with each other. It is strongly recommended to combine integral data obtained at different facilities in the adjustment calculation. In many cases the results of adjusted capture cross sections are also useful to select between discrepant series of differential measurements (e.g. for ^{109}Ag, natural Mo, ^{133}Cs, ^{141}Pr and natural Sm).

For some nuclides there are *recent differential data* which were not used in the a-priori evaluation, viz. for Mo isotopes, $^{102,104}Ru$, ^{105}Pd and $^{147,149}Sm$ and most data for natural elements (because in the evaluation only cross sections for the separate isotopes were considered). For the Mo isotopes the new data of Musgrove et al. in the energy range from 3 keV to 90 keV clearly show that the statistical model fails in that energy region. Therefore it is recommended to re-evaluate these cross sections utilizing all available differential and integral data. For natural Mo the adjustments indicate much lower capture cross sections above 90 keV than adopted in the evaluations of ENDF/B-IV and KEDAK-3. For $^{102,104}Ru$, ^{105}Pd, natural silver and natural Pd there is good agreement with recent measurements. The recent measurements of Yurelov et al. on $^{147,149}Sm$ and natural Sm are not in agreement with the adjusted data.

Suggestions for *new capture measurements* in the keV range follow from the discussion in sect. 3.5. In a number of cases differential measurements have already been performed, but results have not been published yet. Of particular importance are data for ^{99}Tc and ^{133}Cs (discrepancy STEK/CFRMF) and ^{149}Sm (discrepancy STEK and recent data).

In the MeV region the results of adjustments mostly follow those for the keV region because of correlations between the group constants and because most integral data are not very sensitive in this energy range. This is a pity because there are large and often systematic differences between the various evaluations in that energy region, due to differences in the adopted models (e.g. the distribution of spins of target levels). On the other hand for most applications these differences are not important.

(i) *Adjusted point cross sections* have been calculated in a number of cases by making use of adjusted model parameters (sect. 2.5, figs. 2, 11, 12). This method is useful in the statistical model energy range. In the process of recalculating the capture cross section it is recommended to include also new differential data, if these data are available.

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Review paper 9

STATUS OF FAST NEUTRON REACTION CROSS SECTIONS OF FISSION PRODUCTS

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Abstract

Status of the measurements, evaluations, and WRENDA requests on fast neutron capture cross sections is surveyed for about 170 nuclides in fission product mass region. Nuclides are classified according to the order of importance for fast reactor applications, and 42 nuclides are selected. Experimental data on capture cross sections for these 42 nuclides are reviewed and discrepancies are commented.
The calculational methods and parameter determinations adopted in recent evaluations are described and discussed. Evaluated capture cross sections and inelastic scattering cross sections are compared and discussed briefly.

1. Introduction

Evaluation of fission product neutron cross sections is one of the important long term subjects of fast reactor physics.
The difficulty in evaluation is that a great number of nuclides must be treated whose experimental data are quite scarce or discrepant, and then, the evaluators depend largely on gross systematics of parameters.

Since the last IAEA meeting on fission product nuclear data at Bologna [1, 2], there has been remarkable progress in experiments, compilations, and evaluations in this field. Many new data on resonance and keV capture cross sections for separated isotopes have been obtained quite recently. Local systematics of parameters such as strength functions, average level spacings, etc. have been established more firmly than before.

Evaluations are now in progress in several countries. Much effort has been devoted to the compilation of level scheme and cross section data, the critical evaluation of existing experimental data, the parameter determination, and the improvement of calculational methods. Progress in the evaluation field is significant.

It would not be possible to review all of these works on all range of nuclides in fission products mass region. We shall select about 40 nuclides important for fast reactor applications, and shall confine ourselves mainly to capture cross sections in the energy range from a few tens of keV to several MeV, where optical model and the statistical theory are applicable.

In the next section a general status is reviewed of measurements, evaluations, and WRENDA requests on data. Nuclides are classified according to their importance for applications to fast reactors.

In section 3, the status of measurements of capture cross sections is described. Activation data at 25 keV, which are still of considerable importance, are discussed. Discrepancies in the keV capture data are commented.

In section 4 are discussed some details of calculational methods and the parameter determination in the frame work of the optical model and the statistical theory. Recent results on non-statistical aspect of capture process are described with their implications to cross section evaluation.

In section 5 we give some limited comparisons of evaluated capture cross sections and inelastic scattering cross sections.

2. Survey of measurements, evaluations, and WRENDA requests

In Table 1 is given a general status of the measured, evaluated, and requested data of resonance and keV-MeV capture cross sections for about 170 nuclides ranging from Z = 32 (Ge) to 65 (Tb). Nuclides of half-lives shorter than a few days are omitted. Only several short-lived nuclides such as ^{105}Rh, ^{135}Xe and Pm isotopes, whose thermal cross sections or resonance integrals are particularly large, are included in the table for the sake of completeness.

Classification of nuclides

In column 4 of Table 1 we have given the classification [3] of nuclides according to the order of importance for prediction of burnup reactivity in large fast reactors. The classification was based on a burnup at one year of a typical 1000 MWe fast reactor now under study in Japan. Calculation was made with fission products nuclear data file of JENDL-1 (1975) containing 28 nuclides, complemented by Cook's library (1971).

Class I nuclides (11 nuclides) are those contributing more than 3% each to total capture rate of all fission products. In the same way, the contributions are estimated to be 3-1%, 1-0.4%, and 0.4-0.1%, respectively, from each nuclide in class II (16 nuclides), class III (11 nuclides), and class IV (18 nuclides). The 38 nuclides of class I, II and III contribute about 93% to total capture. These are considered as sufficient for burnup calculation of fast reactor. For further applications to fast reactor we added ^{93}Nb, ^{151}Eu, ^{152}Eu and ^{154}Eu to the above 38 nuclides, making up 44 nuclides in all. Natural elements of Zr, Mo, Eu and some others are also important for reactor calculation and consistency check of evaluation. We have not included them in the present review.

In the summary report (2) of the previous meeting at Bologna, ^{134}Cs, ^{137}Cs, ^{144}Ce, and ^{144}Nd have been considered as important for fast reactor applications. But these nuclides may be excluded on the basis of studies by Hasegawa et al. (3) and Gruppelaar et al. described below.

Use of other fission product data libraries naturally results in a different order of importance. Fig. 1 is reproduced from Heijboer et al. [53], which shows the results of calculation of effective cross sections multiplied by the normalized isotopic concentrations in SNR-300 at a burnup of 42 MWd/kg fuel. Discrepancies among different evaluated data sets are demonstrated very clearly. It is seen that the classification given above is not changed significantly. Significant changes of order of importance are seen only for ^{96}Zr, ^{134}Xe, ^{135}Cs, ^{142}Ce, ^{150}Nd and ^{157}Gd.

Evaluation activities

The nuclides contained in recent evaluated data files are marked by a cross in Table 1. The pioneering work of Benzi et al. (4-6) is still a good standard for comparison with more recent evaluations because of the consistent evaluation method and good recording of the results. This evaluation will be called CNEN-1 here. The fission product data file of ENDF/B-4 (7) was released in 1974 and contains cross sections of about 180 stable and radioactive nuclides. ENDF/B-5 will be completed in 1978. In Europe, very detailed evaluations are going on with strong cooperation in parameter determination between the Netherlands (RCN-2, and adjusted file RCN-2A) [8-11], Italy (CNEN-2 or Bologna library) [12], and France (CEA) [13, 14]. There has been much progress in methodology and parameterology of evaluation through this cooperative work.

In Japan the evaluation of 28 isotopes was completed in spring 1975 (JENDL-1) [15, 16]. Five Mo isotopes were supplied from an other group. Preliminary evaluation of additional 34 nuclides was completed quite recently. These data of 67 nuclides are called here JENDL-1 data.

<u>WRENDA requests on capture cross section</u>

The requests on fast neutron capture data are mostly for fast reactor applications. The accuracies of 10-30% are required for burnup calculations. Requests for thermal reactor burnup calculations are not included in the table, since these requests are made mainly on data below a few keV. These requests will be discussed in RP7 of this meeting. Europium data are requested with 5-10% accuracy for reactor control. Nb and Mo cross sections are requested as those of structural materials of fusion and fast reactors with 10-30% accuracy. There is also need of Sb data for neutron source design.

The energy range required differs depending on requestors, some up to a few hundred keV, and some up to 10 MeV. But we think that the energy range important for direct applications to fast reactor core calculations is 100 eV - 500 keV.

Other requests are made for astrophysics applications (^{96}Mo, ^{110}Cd, ^{135}Cs, ^{136}Ba) with rather high accuracy of 10-20%, for tests of nuclear theory (Zr isotopes resonance parameters), for activation analysis (^{84}Kr) and dosimetry (^{93}Nb) with 5-10% accuracy, and for thermal reactor control material (Gd isotopes resonance parameters).

Comparison of WRENDA requests and the "ongoing" measurements in Table 1 shows that many of the requests are expected to be satisfied in very near future, at least, in somewhat limited energy range. However, there seem to be no plans in near future to measure the capture cross sections for class I and II radioactive nuclides (^{93}Zr, ^{103}Ru, ^{129}I , ^{135}Cs, ^{151}Sm) and rare gas nuclides (^{131}Xe, ^{132}Xe).

WRENDA requests on other cross sections

There are few requests on data other than capture. The elastic and inelastic scattering data for Zr isotopes are required for tests of nuclear theory with 10-15% accuracy. All data of ^{93}Nb and Mo isotopes are requested for fusion and fast reactors. Except these requests, there are only two requests for activation analysis (^{107}Ag (n, α), ^{144}Sm (n, 2n)), and five requests for dosimetry (^{85}Rb (n, 2n), ^{93}Nb, ^{103}Rh (n, n'), ^{115}In (n, n'), ^{127}I (n, 2n)). The requests for dosimetry are made chiefly from EUR dosimetry group, and with very high precision (2-5% mostly).

There are recent complilations of (n, 2n) cross sections by Davey et al. [17], and of (n, p), (n, alpha) and (n, 2n) cross sections by Bormann et al. [18], and by Garber and Kinsey [20].

3. Status of measured capture cross sections

3.1 Recent experimental data

The status of available experimental data in Table 1 is based on ref. [19, 20], NEUDADA 1976, CINDA 76/77, and the compilation by Matsunobu and Watanabe [21]. Experiments in the column "ongoing" are those recently published or completed or in progress, the informations of which are based on mostly ref. [22, 23, 52] and CINDA.

Capture cross section data in keV-MeV range are currently measured by direct detection of prompt gamma rays emitted after capture event. Such measurements are in progress in USA (ORNL, RPI), USSR (Lebedev), Japan (JAERI, Tokyo Institute of Technology/Kyoto University), and so on with improved apparatus and techniques.

An extensive series of measurements of capture cross sections are going on at Oak Ridge electron linear accelerator (ORELA) for nuclei ranging from Li to U in 3 - 700 keV. The measurements on the following nuclides in fission product mass range have been recently published or completed.

Sr-(86,87,88), Zr-(90,91,92,94), Mo-(92,94,95,96,97,98,100), Ru-(100,101,102,104), Pd-(104,105,106,108,110), Cd-(106,108,110, 111,112,113,114,116), Te-(122,123,124,125,126,128,130), Ba-(134,135,136,137,138), Nd-(142,143,144,145,146,148), Y-89, Nb-93, Rh-103, La-139, Ce-140, Pr-141, Tb-159, Ho-165, Tm-169.

These ORELA data are supplying us with considerable amounts of new information, filling gaps or lack of existing data and giving improved systematics of average resonance parameters. Also, much information is being obtained concerning the non-statistical capture process. However, the measured capture cross sections are sometimes in significant disagreement with older data. The case of Nb cross section is shown in

Fig. 2. It is seen that there is an excellent agreement between the data of Macklin [Ma 76a] and those of Yamamuro et al. [Ya 75,77]. Below 30 keV appreciable difference exists between these two recent data sets and those of Kompe [Ko 69], and Popov and Shapiro [Po 62]. The agreement becomes better at higher energies.

At RPI, the capture measurements are going on for $^{95,97}Mo$, ^{99}Tc, $^{101,102,104}Ru$, ^{103}Rh, ^{107}Pd (mixed fission-product sample), ^{133}Cs, ^{141}Pr, $^{143,144,145}Nd$, and ^{149}Sm [22,23]. These are mostly the nuclides for which no data or only single data sets exist, or the existing data are discrepant.

At Kyoto University and JAERI, measurements were completed for ^{93}Nb, ^{127}I, ^{133}Cs (Tokyo Inst. Technol./Kyoto Univ.), and $^{151,153}Eu$ (JAERI). The measured ^{133}Cs cross section [Ya 75,77] is shown in Fig. 3 together with other data sets. Recent measurement at RPI is reported to give cross sections which are about 10 % higher than the data of Popov et al., and therefore, in agreement with ENDF/B-4 evaluation. It is of interest to note that the data of Yamamuro et al. in Fig. 3 and those of Macklin and Yamamuro et al. in Fig. 2 for ^{93}Nb show the similar trends relative to the data of Popov and Shapiro and of Kompe.

Very recently Kononov et al. at Lebedev institute measured the capture cross sections for In, Ta, Au, and for isotopes of Sm and Eu [52]. (See also Yurelov et al. [Yu 75].) Since there have been few data for Sm isotopes these Lebedev data will be of great value for cross section evaluation.

At Universität Kiel the resonance parameters of ^{135}Cs and ^{137}Cs were obtained recently [22] in the energy range 42 - 880 eV using the mixed sample of fission-product Cs isotopes. These data are of considerable interest since no data have been available at all and there has been very large disagreement among the evaluated data (see Fig. 1). It is also reported that the measurements of resonance parameters of ^{152}Eu, ^{154}Eu and ^{155}Eu are planned using a mixture of Eu isotopes.

3.2 Activation cross sections

Although the current measurements of capture cross sections in keV range are performed mostly with the method of direct detection of capture γ-rays, the classical activation data are still of considerable value in giving the isotopic cross sections, which are not easily obtained by direct detection method. However, existing activation cross section data are often very discrepant. In most cross section evaluations the calculation is adjusted by available capture data. The disagreement between evaluated data is often a result of the different choice of experimental data. Here we shall review briefly the status of activation cross sections near 25 keV.
Other than 25 keV data, there are typical data for wide range of fission products isotopes at 2 keV by Schumann [Sh 69], at 195 keV by Lyon and Macklin [Ly 59], and at 3 MeV by Petö et al. [Pe 67]. Typical poly-energetic data are those of Stupegia et al. [St 65a, 66b, 68] in 5 keV-3MeV, Johnsrud et al. [Jo 59] in 0.15-6.2 MeV, Pasechnik et al. [Pa 58] at 2.5, 3.1 and 4.1 MeV, and Leipunsky et al. [Le 58] at 0.2, 2.7 and 4 MeV. At 14 MeV there exist a number of recent experiments. The 14 MeV data are used in cross section evaluation as the normalization of calculation of direct and collective capture. But we shall not consider them further in this review.

<u>Capture cross sections for ^{115}In, ^{127}I and ^{197}Au near 25 keV</u>

Most of the activation cross sections near 25 keV are measured relative to the value of ^{127}I, and in some experiments relative to ^{115}In or Au. In Table 2 the existing activation data for these nuclides are listed together with the ones obtained by other techniques.

It has been remarked that as a general trend the newer activation experiments give the smaller cross sections. This is interpreted [Ch 75] as that the improved source and techniques reduced the effect of resonance and thermal capture. Recent data of Rimawi et al. [Ri 75], Yamamuro et al. [Ya 75], and Macklin et al. [Ma 75] agree within 10% for all three nuclides. We may assume the value of 763mb for ^{127}I (n, γ) cross section at 24 keV, taking the average of recent two data. The Au cross section of 654mb at 24 keV recommended by Pönitz [24] is reasonable.

Activation cross sections of fission products near 25 keV

In Table 3 are listed the activation cross sections near 25 keV for important fission products classified in section 2. The data with mono-energy neutron near 25 keV measured by other techniques are also listed. Values of standards adopted in the measurements are briefly described in the continuation of Table 3. Renormalization factors based on the alteration of these standard data are given in parentheses.

Ribon et al. [13] have applied to all data of Macklin et al. [Ma 57] a base renormalization factor 0.61. This factor is the ratio of the recommended (654mb) to the measured (1120mb) Au cross sections, corrected further by alteration of branching ratio data. Since Macklin et al. adopted ^{127}I cross section as standard we do not agree with the renormalization by Ribon et al. They have also taken the value of 800mb as standard for ^{127}I cross section at 24 keV, which is probably a little too high if one looks at Table 2.

Status of capture data of important nuclides

The status of capture data for 42 nuclides of class I, II and III is summarized in Table 4. Discrepancies are commented. The (n,γ) data at 14 MeV are not included in the table.

Recent experiments completed or ongoing are rapidly filling the lack or discrepancies of data. For unstable nuclides and rare gas isotopes there seem to be no immediate plans of measurement. The measurements of resonance parameters as going on at Kiel and Jülich are expected to be very useful for the evaluation of cross sections.

4. Method of evaluation and parameter determination

We start with the Hauser-Feshbach-Margolis theory of fast neutron cross sections. The capture cross section is given by

$$\sigma_{n\gamma}(E_n) = \frac{\pi}{k^2} \sum_{J\pi} g_J \sum_{lj}{}' \frac{T_\gamma^{J\pi} T_c}{T_{\gamma T}^{J\pi} + \sum_{c''} T_{c''}} W_{c,cap} \qquad (1)$$

Here, $T_{\gamma T}^{J\pi} = 2\pi \Gamma_{\gamma T}^{J\pi}$, $\Gamma_{\gamma T}^{J\pi}$ being the total radiation width. Correspondingly, $\Gamma_\gamma^{J\pi}$ is the capture width. T_c is the neutron transmission coefficient in channel c which is obtained by the optical model or the strength function model. $W_{c,\ cap}$ is the neutron width fluctuation correction factor.

The calculation of capture cross section is usually adjusted by capture data in keV region and so as to obtain satisfactory overlapping with averaged resonance cross sections. The most crucial parameter is Γ_γ / D_s and in some cases neutron strength functions. This process usually gives a reasonable fit to experimental data up to 1$\sim$2 MeV, provided that the target level scheme is well known. Above about

1 MeV, the target levels are not known any more, the calculated cross section depends rather sensitively on various parameters of level density.

4.1 Optical model parameters

The neutron optical model potential is defined as follows.

$$V(r) = -V_0\, f_1(r; R_1, a_1) - i W_V\, f_1(r; R_2, a_2) - i W_s\, f_s(r; R_s, a_s)$$
$$- V_{so} (\hbar/m_\pi c)^2 \frac{1}{r} \left| d f_1(r; R_{so}, a_{so})/dr \right| (\vec{\sigma} \cdot \vec{\ell}), \qquad (2)$$

where

$$f_1(r; R, a) = 1 \Big/ \left[1 + \exp\left(\frac{r-R}{a}\right)\right],$$
$$f_s(r; R, a) = 4 \exp\left(\frac{r-R}{a}\right) \Big/ \left[1 + \exp\left(\frac{r-R}{a}\right)\right]^2,$$
$$\text{or} \quad = \exp\left[-\left(\frac{r-R}{a}\right)^2\right],$$
$$(\hbar/m_\pi c)^2 = 2.00\ (\mathrm{fm})^2.$$

Of a number of optical model parameter sets, the potentials of Moldauer [25] and Igarasi et al. [15] have been used frequently for the evaluation of fission product cross sections. Potentials of Rosen et al. [26], Becchetti and Greenless [27], and Wilmore and Hodgson [28] were also used by some evaluators. These optical model parameter sets are listed in Table 5.

Delaroche et al. [29] suggested the SPRT method to determine the optical model parameters. We can conversely test the above five global parameter sets by SPRT check, that is, by comparison of the calculations and measurements of strength functions for s-and p-waves, scattering radii and total cross sections, all of which are obtained directly by optical model. This comparison is shown in Fig. 4, 5 and 6.

We observe the following from this test.

(1) None of the single global parameter set studied here gives satisfactory fit to all the experimental data in A=80-160 simultaneously.

(2) The sharp peaks of p-wave strength functions near A=95 (Zr, Mo) and of s-wave strength functions near A=145 (Nd, Sm) are not calculated well. The s-wave strength functions for A=133-142 (Cs, Ba, La, Ce, Pr) are considerably overpredicted, resulting in appreciable overprediction of total cross sections for these nuclides at low energy.

(3) Above 1 MeV the calculated total cross sections with different parameter sets agree within 20% and also with experimental values. On the contrary, the disagreement of compound nucleus formation cross sections at low energy, if it exists, does not vanish up to high energy. Disagreement is about a factor of 1.3 to 1.5 in MeV region. Disagreement in the calculation of inelastic scattering cross section could be of the same order of magnitude.

(4) Scattering radii are predicted rather well on the average. The calculated values are systematically smaller than the measured values by 10-30% for A=110-125. For A>150, the measured scattering radii vary considerably (up to a factor of 2) from nucleus to nucleus. Potentials of Moldauer and Igarasi et al. give too small values in this mass region.

To conclude, we must choose the potential parameter set carefully to reproduce the local systematics. Global parameter set is probably not successful.

The strength function model is used often in low energy region and the optical model in high energy. But, if the optical potential parameter set adopted does not reproduce the measured strength function data with reasonable accuracy, there occurs difficulty in determining the dividing energy between two models. We might be forced to assume different dividing energies for capture cross section and elastic scattering cross section respectively.

Lagrange and Delaroche [55,56] have recently performed coherent optical model calculations for ^{98}Mo and Sm isotopes. In ref. [56] is demonstrated that all relevant cross sections for ^{98}Mo are predicted successfully by optical model in the energy range 1 keV to 10 MeV.

4.2 Level density parameters

The composite level density formula of Gilbert and Cameron (30) is,

$$\rho_{J\pi}(E) = p_\pi R_J \rho_0(E) \tag{3}$$

$$R_J = \frac{2J+1}{2\sigma^2} \exp\left[-(J+\tfrac{1}{2})^2/2\sigma^2\right] \tag{4}$$

$$\rho_0(E) = \frac{\exp(2\sqrt{aU})}{12\sqrt{2}\,\sigma a^{1/4} U^{5/4}}, \quad E > E_x \tag{5}$$

$$= \frac{1}{T}\exp\left[(E-E_0)/T\right], \quad E \leq E_x. \tag{6}$$

Here, p_π and R_J are parity and spin distributions, respectively. $\rho_0(E)$ is the total level density (except the degeneracy 2J+1 in magnetic quantum numbers). U is the effective excitation energy given by U=E-P, where P is the pairing energy correction taken relative to odd-odd nucleus. Other symbols are those commonly used.

The parity distribution p_π is often assumed as 1/2 for both parities. In JENDL-1 evaluation, the expression by Igarasi [31] which approaches to 1/2 asymptotically is used.

$$P_\pi(E) = \frac{f_\pi + 0.5\exp[(U-U_0)/\delta]}{1+\exp[(U-U_0)/\delta]} \tag{7}$$

where f_π is the fraction of low lying levels with parity π. U_0 and δ are somewhat arbitrarily determined parameters.

Spin cut-off parameter

The spin cut-off parameter σ is expressed by

$$\sigma^2 = c\sqrt{aU}\,A^{2/3}, \quad E > E_x \tag{8}$$

The value c=0.0888 was first derived by Jensen and Luttinger [32] and was used by Gilbert and Cameron. Facchini et al. [33] obtained c=0.146 and stated that c=0.0888 was in error. The choice of two different values of c gives different values of level density parameter a by about 0.5 MeV^{-1}. c=0.0888 is adopted in ENDF/B-4, CEA and JENDL-1 evaluations, while c=0.146 is adopted in CNEN-2 and RCN-2 evaluations.

For $E<E_x$, Gilbert and Cameron did not give the expression for the spin cut-off parameter. There are several expressions for this parameter currently used in evaluations.

$$\sigma^2 = c\sqrt{aU}\,A^{2/3} \qquad U>0, \tag{9a}$$

$$\text{or} \quad = \sigma^2(E_x) = c\sqrt{aU_x}\,A^{2/3} = \text{constant} \tag{9b}$$

$$\text{or} \quad = \sigma^2_{exp} = \text{constant}, \tag{9c}$$

$$\text{or} \quad = \sigma^2_{exp} + (\sigma^2(E_x) - \sigma^2_{exp})\,\frac{E}{E_x} \tag{9d}$$

For expressions (9c,d) the spin data of low lying levels are to be used. Schmittroth [34] has given an expression for σ^2_{exp} by using the maximum likelihood method:

$$\sigma^2_{exp} = \frac{1}{2N} \sum_{i=1}^{N} (I_i + \frac{1}{2})^2 \tag{10}$$

where I_i ($i=1$, --, N) are the spins of low lying levels.

A feature of σ_{exp} is that it is usually smaller than $\sigma(E_x)$, and is nearly constant ($\sigma_{exp}=2\sim3$) for wide range of mass number. Spin distribution function with σ^2 as a parameter is shown in Fig. 7. Formulas (9a)-(9d) are illustrated in Fig. 8, for target states of ^{109}Ag. In Fig. 9 is shown the effect of σ^2-value on calculated capture cross section for ^{109}Ag. These figures are reproduced from the work of Gruppelaar [10]. The calculated capture cross section depends on σ^2-value in rather complex manner, but the effect is significant in the case of ^{109}Ag for energy above E_c (the highest target level).

The formula (9d) is used in RCN-2 evaluation. In ENDF/B-4 and JENDL-1 evaluations eq. (9b) is used.

<u>Level density parameter a</u>

Gilbert and Cameron have given the gross systematics of a parameter as a function of shell energy $S = S(Z)+S(N)$.

$$\begin{aligned} a/A &= 0.142 + 0.00917\ S,\ \mathrm{MeV}^{-1} \quad \text{(undeformed nuclei)} \\ &= 0.120 + 0.00917\ S,\ \mathrm{MeV}^{-1} \quad \text{(deformed nuclei)} \end{aligned} \tag{11}$$

The shell energy was given by Cameron's mass formula.

There are a number of more recent attempts to predict a values theoretically or phenomenologically. We quote here merely the works of Weigmann and Rohr [35], Schmittroth [36], and the references therein. (See also a very recent investigation by Reffo [57].)

In recent experiments of resonances the angular momentum of each resonance is more clearly assigned, and the local systematics of level density parameters is being obtained more firmly than before. In Figs. 10a and 10 b are shown the plots of the values of a/A for isotopes of Zr, Mo, Ba, and Nd as a function of Cameron's average shell correction energy. The data compiled by Schmittroth [36] and the recent data at ORELA are compared, assuming c = 0.0888 for spin cut-off parameter. The error of 1 % in a parameter causes typically the error of 7 % in the value of D_s. Therefore, it will not be an easy task to predict the unknown values of D_s within accuracy of 30 % even if the local systematics is utilized.

In Fig. 10c is shown the a value for Sm isotopes taken from ref. [8]. (c = 0.146 is assumed.) The even-odd effect is seen clearly. The upper figure is the plot of a/A versus shell correction energy, and the lower figure is that of a versus neutron number N of the compound nucleus. The even-odd effect and the convex curve of a/A versus N are tried to be interpreted in terms of the deformation effects in [8] and the references therein. It is reported also that the similar trend is observed for isotopes of Nd and Gd [8]. In the upper figure of Fig. 10c, a/A versus S, the upward convex curve is not observed, since the deformation effects have been absorbed, at least partly, in the phenomenological average shell correction energy. It will be easier with this form of plot of a parameter in order to predict the unknown values of a.

4.3 Level scheme

Many new data on level scheme are obtained every year. Yet, it is rather exceptional case that the spins and parities are assigned for the first 10 levels without any ambiguity. There are often the cases where only the energies of levels are known, or the levels may be missed.

Recently, Matumoto et al. [37] have completed the compilation of level schemes of 100 fission product nuclides for JENDL, although some of the adopted data were based on older Nuclear Data Sheets. They have compared the adopted data with those of RCN-2 evaluation [9]. Appreciable disagreements are found for 11 nuclides out of 27 nuclides compared, both in spin-parity assignment and the number of levels adopted.

In Tables 6a and 6b examples are shown of the level schemes adopted in JENDL-1, the revised JENDL data, and RCN-2 data. Calculated cross sections are also shown in the table. Calculation was performed (S. Igarasi, priv. comm.) with the potential parameters of Igarasi et al. In case of ^{103}Rh, all the level schemes (JENDL-1, the revised one and RCN-2) are rather similar below 920 keV, except the assignment of levels between 607 - 848 keV. The effect of these levels on the inelastic total cross section is small, but the capture cross section at 1 MeV is seen to be affected by about 20 %. ^{139}La is the case where there are only few levels below 1 MeV. In JENDL-1 evaluation, four levels at 570, 830, 930, and 1070 keV levels were assigned based on Nuclear Data Sheets 1974, but were discarded in the revision, since these levels have not been identified in the later publications. The effects of discarding these four levels on the calculated cross sections are significant even at E_n = 1.75 MeV.

Lagrange studied also the effect of the choice of the level scheme on the calculated capture and inelastic scattering cross sections for ^{98}Mo [56]. He compare the level scheme of Smith et al. [58] and the one adopted by Ribon et al. [13]. He found a disagreement of about 40% in the calculated capture cross sections near 2 MeV.

Above the highest energy E_c of known levels the target levels are approximated by the constant temperature level density formula, eq. (6), up to energy E_x. Since $E_c \lesssim$ 2 MeV and $E_x \cong$ 5 - 7 MeV in most cases, the capture cross section in MeV range is mostly governed by the level density of eq. (6).via competition of inelastic scattering. Schmittroth [34] examined the correlation between the calculated capture cross section and the target level density above E_c. His finding was that the agreement between the calculated and the measured capture cross sections was improved significantly if the level density parameters were adjusted so as to give a good overall fit with the low-lying target levels. In Fig. 11 the case of ^{75}As is reproduced from the work of Schmittroth.

These examples discussed above will show that we must carefully examine the level scheme and the level density to calculate the capture cross sections correctly in MeV region. Also, we must constantly watch the newer data of level scheme. Old data are often obsolete.

In case that the spin-parity assignment is not possible for certain levels, Schmittroth has given a useful approximate analytical expression for inelastic channels leading to these levels.

4.4 Radiation width

From the detail balance, the partial dipole radiation width Γ_γ (b $\rightarrow$ a) from state b to a is related to the photo-nuclear excitation cross section σ_γ(a $\rightarrow$ b).

$$\sigma_\gamma(a \rightarrow b) = \frac{3\pi^2}{k_\gamma^2} \frac{\Gamma_\gamma(b \rightarrow a)}{D(E_b)} , \qquad (12)$$

where D (E_b) is the spacing of levels near $E=E_b$ with appropriate spin and parity. (The factor $3\pi^2$ appears instead of $2\pi^2$ after summation over direction of polarization and integration over direction of emitted photons.)

Statistical theory

With Brink's hypothesis, total capture strength function is given by the statistical theory using Axel's esimate of E1 dipole resonance.

$$T_\gamma^{J\pi} = \frac{2\pi \Gamma_\gamma^{J\pi}(E_n)}{D_{J\pi}(E_n+B_n)} = \frac{2}{3\pi}\sum_{J'\pi'}{}' \int_{E_n}^{E_n+B_n} d\varepsilon_\gamma\, k_\gamma^2\, \sigma_\gamma(\varepsilon_\gamma)\, \rho_{J'\pi'}(E_n+B_n-\varepsilon_\gamma), \quad (13)$$

$$\text{with} \quad \sigma_\gamma(\varepsilon_\gamma) = \frac{2\pi^2 e^2 \hbar}{M_n c}\frac{NZ}{A}(1+0.8x)\,\varepsilon_\gamma f(\varepsilon_\gamma), \quad (14)$$

$$f(\varepsilon_\gamma) = \frac{2\Gamma_R}{\pi}\frac{\varepsilon_\gamma}{(\varepsilon_\gamma - \Gamma_R)^2 + (\Gamma_R \varepsilon_\gamma)^2}$$

Here, E_R and Γ_R are the energy and width of giant dipole resonance, and x ($\approx$ 0.5) is the fraction of exchange force of total(n, p) force. Axel [38] has given $E_R = 77A^{-1/3}$ MeV and $\Gamma_R = 5$ MeV.

Experimental data of photonuclear cross section are fitted to a Lorentz curve,

$$\sigma_\gamma(\varepsilon_\gamma) = \sum_{i=1,2} \frac{\sigma_{Ri}}{1+[(\varepsilon_\gamma^2 - E_{Ri}^2)^2/\Gamma_{Ri}^2 \varepsilon_\gamma^2]} \quad (15)$$

Two Lorentz curves are necessary for deformed spheroidal nuclei, corresponding to oscillations along each of the non-degenerate axis of spheroid. The parameters σ_{Ri}, E_{Ri} and Γ_{Ri} are tabulated by Berman [39]. Carlos et al. [40] has given the systematics of Γ_R in graphical form.

The dipole sum rule states

$$\int_0^\infty \sigma_\gamma(\varepsilon_\gamma)\, d\varepsilon_\gamma = \frac{2\pi^2 e^2 \hbar}{M_n c}\frac{NZ}{A}(1+0.8x) = 60\frac{NZ}{A}(1+0.8x) \quad \text{MeV-mb}. \quad (16)$$

Eq. (13) is usually used to calculate the energy dependence of Γ_γ , normalizing its value by slow neutron resonance data. Neglecting the spin cut-off parameter in level density, Γ_γ is considered as spin independent. With inclusion of spin cut-off parameter Gruppelaar [41] obtained an approximate spin dependence of Γ_γ as

$$\Gamma_\gamma^{J\pi} \sim \exp\left[+(J+\tfrac{1}{2})^2\left(\frac{1}{2\sigma_c^2}-\frac{1}{2\sigma_b^2}\right)\right]$$

where σ_c^2 is the spin cut-off parameter of the compound state and σ_b^2 is an effective spin cut-off parameter of the bound levels.

Benzi et al. [42] used eq. (13) directly to calculate the radiation widths at neutron binding energy. They took into account the transitions to discrete states, and used experimental photo-nuclear cross sections rather than sum-rule. The results were generally in fair agreement with experiments, but there were also many cases where disagreement was beyond a factor of 2. This disagreement is not unexpected since only the 1 % of the total dipole strength occurs below the photo-neutron threshold region. Besides, the Lorentzian fit is made for relatively narrow range near resonance peak [39]. The plot of experiment-to-calculation ratio shows gross size structure against mass number, as observed by Cameron many years ago. There are gross peaks near 3p and 4s size resonances. This has been explained partly as the valency neutron capture effect [43, 44-46].

Two very interesting works were reported to this meeting by Reffo [57] and Benzi[59]. Reffo has recalculated the radiation width using eq. (13), based on recent level scheme. Although the work is under way, he found a strong spin and parity dependence of radiation width for some nuclei, which is caused by the transitions to discrete states. Benzi has proposed a very simple formula of radiation width based on a black body radiation model of nucleus. He obtained the expression :

$$\Gamma_\gamma(B) = 1222\ A^{-1/3}\ T^3\ \text{mV} \qquad (T \text{ in MeV}) \qquad (17)$$

where T is the thermodynamic temperature in energy unit at excitation energy B and is numerically very close to the nuclear temperature given by a level density formula. This simple formula was shown to reproduce the experimental data remarkably well except in the mass region where the non-statistical capture effect is expected to be dominant.

Non-statistical effect

There are now many quantitative data of non-statistical capture process in eV-keV region. Especially, from recent experiments at ORELA and their analyses on resonance and keV-capture cross sections for isotopes of Sr, Mo, Ba and Nd, quantitative data have been obtained concerning the difference in magnitude of total radiation widths for s-wave and p-wave captures, and the correlation between radiation width and neutron strength function. Calculations of valency neutron capture have been tried to interprete these data. Some typical results are as follows.

(1) Sr and Zr isotopes [Bo 75,76] : There are large differences between Γ_γ^s and Γ_γ^p, and these are explained successfully by valency theory. Very strong correlation between Γ_γ^p and Γ_n^1 is found.

(2) Mo isotopes [Mu 76a] : The difference between Γ_γ^s and Γ_γ^p decreases as neutron number increases from major closed shell. Valency theory was proved to be qualitatively successful. High correlation between Γ_γ^p and Γ_n^1 exists for ^{92}Mo.

(3) ^{138}Ba [Mu 74,75,76b] : Γ_γ^s is anomalously large (310 ± 25 mV) compared with the ones for neighboring nuclei (120 mV). Valency theory fails to explain this anomaly. Rather high correlation $\rho(\Gamma_n^0, \Gamma_\gamma^s)$ exists.

(4) Nd isotopes [Mu 77] : Γ_γ^s of odd isotopes are systematically larger than those of even isotopes. In case of ^{142}Nd, Γ_γ^p(46 mV) is considerably smaller than Γ_γ^s (78 mV), and high correlation $\rho(\Gamma_n^0, \Gamma_\gamma^s)$ exists.

Valency theory fails to account for the difference between Γ_γ^s and Γ_γ^p, and the measured correlation coefficients.

These new results should be taken into account in the cross section evaluation. Strong initial state correlation between Γ_γ^ℓ and Γ_n^ℓ implies that the calculation of width fluctuation correction should be modified for this part of the partial wave capture. This was done in RCN-2 evaluation of Mo isotopes.[9]

Weigmann and Rohr [35] have fitted the measured radiation widths by semi-empirical formula including the valency neutron capture.

$$\Gamma_\gamma(B) = \Gamma_\gamma^{stat} + s g' A^{2/3} D_s \Gamma_n^1 ,$$

where Γ_γ^{stat} is the statistical contribution, g' is a spin statistical factor and s is the parameter fitted to reproduce $\Gamma_\gamma^{exp} - \Gamma_\gamma^{stat}$.
They have given $s = 3.82 \times 10^{-4}$ for p-waves, $88 \leq A \leq 125$.
Based on expression by Weigmann and Rohr, Gruppelaar [9] considered the formula,

$$(T_\gamma^{val})_{\ell J} = r_1 \beta_J \bar{\varepsilon}_\gamma^2 A^{2/3} T_{\ell j}^n \tag{18}$$

where $\bar{\varepsilon}_\gamma$ is an effective gamma-ray energy, and β_J is the spin statistical factor taking the value 1,2, and 3 according to J =0, 1/2 and ≥ 1, respectively. A parameter r_1 was fitted to the valency capture calculation for ^{90}Zr, ^{92}Mo and ^{98}Mo. In RCN-2 evaluation the parameter r_1 was further adjusted so as to obtain a better agreement between calculated and experimental capture cross sections in the keV region.

4.5 Width fluctuation correction

Classical integration method

The width fluctuation factor is defined by

$$W_{cc'} = \left\langle \frac{T_c T_{c'}}{T} \right\rangle \Big/ \frac{\langle T_c \rangle \langle T_{c'} \rangle}{\langle T \rangle} \tag{19}$$

Based on the statistical assumption that each channel width is completely uncorrelated with each other, we obtain the following expressions.

$$W_{c,cap} = \langle T \rangle \int_0^\infty \frac{e^{-\langle T_{\gamma T} \rangle t}\, dt}{\left(1+\frac{2\langle T_c \rangle t}{\nu_c}\right)^{\nu_c/2+1} \prod_{c''(\neq c)} \left(1+\frac{2\langle T_{c''} \rangle t}{\nu_{c''}}\right)^{\nu_{c''}/2}} \tag{20a}$$

for capture cross section, and

$$W_{cc'} = \langle T \rangle \left(1+\frac{2}{\nu_c}\delta_{cc'}\right) \int_0^\infty \frac{e^{-\langle T_{\gamma T} \rangle t}\, dt}{\left(1+\frac{2\langle T_c \rangle t}{\nu_c}\right)^{\frac{\nu_c}{2}+1} \left(1+\frac{2\langle T_{c'} \rangle t}{\nu_{c'}}\right)^{\frac{\nu_{c'}}{2}+1} \prod_{c''(\neq c,c')} \left(1+\frac{2\langle T_{c''} \rangle t}{\nu_{c''}}\right)^{\frac{\nu_{c''}}{2}}} \tag{20b}$$

for particle channel.

Here, ν is the number of degrees of freedom of width distribution function (χ^2 distribution). The radiation width is assumed as not fluctuating. The factor $(1 + 2/\nu_c)$ in eq.(20b) is an elastic enhancement factor.

<u>Method of Tepel et al.</u>

There is another approach to calculate the width fluctuation correction by Tepel, Hoffmann and Weidenmüller [47, 48]. Very recently, Gruppelaar and Reffo [49] investigated the validity of various approximations to width fluctuation correction calculation. We shall follow their work below.

V_c is defined by

$$\langle T_c \rangle = V_c + (w_c - 1) V_c \Big/ \sum V_{c''} \tag{21}$$

where $w_c = 1 + 2/\nu_c^{eff}$. Width fluctuation correction is given approximately by

$$W_{cc'} = \frac{\mathcal{V}_c \mathcal{V}_{c'}}{\sum \mathcal{V}_{c''}} \left[1 + (w_c - 1)\, \delta_{cc'} \right] \Big/ \frac{\langle T_c \rangle \langle T_{c'} \rangle}{\langle T \rangle} \qquad (22)$$

This approximation was verified by statistical computer experiments [48]. Computer experiments have given at the same time an empirical relation for w_c, which is a function of T_c, T, and $\bar{T}_c$ (arithmetic mean of all T_c).

When many channels ($\gtrsim$ 10 or 20) are open, eq. (21) is solved to a very good approximation,

$$\mathcal{V}_c = \langle T_c \rangle \Big/ \left[1 + (w_c - 1) \frac{\langle T_c \rangle}{\langle T \rangle} \right] \qquad (23)$$

Using eqs. (22) and (23), the width fluctuation correction is calculated very easily and fast.

For lumped gamma-ray channel, Gruppelaar and Reffo suggested the following substitution:

$$\mathcal{V}_\gamma^{tot} = T_\gamma^{tot}, \qquad w_\gamma = 1 \qquad (24)$$

They have studied the cases of the capture cross sections for ^{100}Mo and the elastic and inelastic cross sections for ^{96}Mo. Monte Carlo calculations as performed by Moldauer [50], Tepel at al. [47] and Hoffmann et al. [48] were considered to give correct answers. The results are the followings.

(1) For neutron capture calculations, the classical integration method, eq. (20a), with $\nu_c = 1$ seems to be good enough for many nuclides at all energies.

(2) For compound elastic and inelastic scattering cross sections above inelastic threshold, there are appreciable differences

(10-20%) between the classical integration method and the method of Tepel et al. Method of Tepel et al. with semi-empirical relation for W_c has to be preferred.

From these results, Gruppelaar and Reffo recommended a practical prescription to calculate the width fluctuation correction.

5. Comparison of evaluated data

Gruppelaar, Janssen and Dekker [10] have made the extensive comparison of evaluated capture cross sections of ENDF/B-4, JENDL-1, CNEN-2, CEA, and RCN-2 and 2A files. They have given graphs of point-wise cross sections and group cross sections in ABBN group structure. In Fig. 1 was already shown the comparison of the macroscopic cross sections averaged over SNR-300 spectrum. Comparison of point-wise cross sections is also given in review paper No. 7 of this meeting [62].

We present in this section some limited comparison of evaluated cross sections for 42 nuclides defined in Sec. 2. We compare CNEN-1, ENDF/B-4, JENDL-1, CEA and RCN-2 evaluations. Unpublished CNEN-2 file was not available to us in time. CNEN-2 data are cited only through ref.[10].

In Table 7 are listed the average s-wave level spacings and the average radiation widths adopted in the evaluated data files. The ratio $S_\gamma = \Gamma_\gamma / D_s$ determines the magnitude of capture cross section in medium energy range. Many of the D_s values in Table 7 are more or less the adjusted values in accordance with available capture data. For some nuclides in ENDF/B-4 only the value $T_\gamma = 2\pi S_\gamma$ was given. The experimental data of average level spacings are also listed in the last column of the table. It is seen that the experimental data are still quite lacking and discrepant.

5.1 Capture cross sections

In Table 8 are given the capture cross sections at 30 keV and those averaged over SNR-300 spectrum [10] and ^{235}U fission spectrum. The fission spectrum averaged values of CNEN-1 data were taken from ref.[1].

In the last 3 columns of Table 8 we give the factor of disagreement (maximum value divided by minimum value) for the evaluated data sets. We shall first discuss the low energy cross sections. It is seen from this table that a good agreement is obtained for ^{93}Nb, $^{95,97,98}Mo$, ^{99}Tc, ^{103}Rh, $^{105,108}Pd$, ^{127}I, ^{139}La, ^{142}Ce, ^{148}Nd, ^{152}Sm, and $^{151,153,155}Eu$. These are mostly the nuclides for which there exist many capture data sets or only a single data set (see Table 4). Even then, the agreement among evaluated data is often not better than 20 %. There exixt also many data sets for ^{141}Pr. ENDF/B-4 value seems to be too high for this nuclide. (See Fig. 13a of ref.[10], and Harker's contribution to [22] pp.77-83 (1977).) Excepting ENDF/B-4 value, the agreement is improved appreciably for this nuclide.

Poor agreement (disagreement factor exceeding 1.6) is observed for ^{93}Zr, ^{107}Pd, ^{109}Ag, ^{129}I, ^{131}Xe, ^{135}Cs, ^{146}Nd, ^{150}Nd, and ^{147}Sm. These are mostly the nuclides for which there exixt no experimental data or the existing data are discrepant. For ^{109}Ag there exist systematic differences among the capture data in keV region. The disagreement in evaluated cross sections at 30 keV is the reflection of evaluator's choice of experimental data sets. The values of CNEN-1, CEA and RCN-2 are more consistent with the natural silver data than those of ENDF/B-4 and JENDL-1. Large disagreement for ^{150}Nd at 30 keV seems to be also the result of choice of data. IN CNEN-1 evaluation the data of Johnsrud et al. [Jo59] above 170 keV seem to have been adopted as the normalization of the calculation, while in JENDL-1 the activation data at 24 keV [Ha68, Th70] were adopted.

In all of the evaluations the very recent experimental data or on-going experiments described in sec. 3 have not yet been taken into account. Re-evaluation is necessary based on these new experimental data.

The fission spectrum averaged cross sections are, in general, more discrepant than low energy cross sections. These cross sections are sensitive to the method of calculation and the level scheme data as given in the last section.

5.2 Inelastic scattering cross sections

In Tables 9(a) and 9(b) the evaluated inelastic scattering cross sections are compared in the following two integral forms.

$$\langle\sigma_{in}\rangle_{Bethe} = \iint dE\,dE'\,\chi^{25}(E)\left[\sigma_f^{28}(E) - \sigma_f^{28}(E')\right]\sigma_{in}(E\to E') \Big/ \int dE\,\chi^{25}\sigma_f^{28}(E),$$

$$\langle\sigma_{in}\rangle_{\phi} = \int \sigma_{in}(E)\,\phi(E)\,dE \Big/ \int \phi\,dE \qquad (26)$$

Here, $\chi^{25}(E)$ is the ^{235}U thermal fission spectrum and $\sigma_f^{28}(E)$ is the ^{238}U fission cross section. $\phi(E)$ is the average neutron spectrum in the inner core of a typical 1000 MWe fast reactor studied in Japan.

Eq. (25) is given by Bethe, Beyster and Carter [51] and expresses the effective inelastic scattering cross section obtained by sphere transmission experiment. This gives the behavior of energy transfer kernel of inelastic scattering above 1 MeV. Eq. (26) does not correspond to the measurable physical quantities, but simply gives some idea of the magnitude of inelastic scattering cross section at low energy ($\lesssim$ 1 MeV).

From Table 9(b) we note the even-odd effect and shell closure effect on the calculated inelastic cross sections as expected. We may also anticipate that the total inelastic scattering cross sections can be predicted within 30 - 40 % provided the low-lying levels are known. But, as we see from the table, there are cases where disagreement is far greater than that. This is typically the case for ^{149}Sm and ^{151}Sm, especially at low energy. The inelastic threshold energies of these nuclides are very low (as revealed by the large magnitude of $\langle\sigma_{in}\rangle_{\phi}$), and the capture cross sections are large compared to inelastic cross sections at low energy. This implies that the calculation of inelastic scattering cross section is affected appreciably by relatively small differences in the estimation of capture cross section. The effect of width fluctuation correction is also very significant on the inelastic cross sections for these nuclides in low energy region [16].

For more detailed discussions of evaluated inelastic scattering cross sections, a systematic comparison is necessary concerning the adopted optical model parameters, level schemes, etc. An easier, and meaningful comparison may be done by inter-comparison of central perturbation cross sections in well-defined fast reactor benchmark cores.

6. Conclusions

Compared with the status reviewed in 1973 Bologna meeting, there has been cosiderable progress both in the experimental data and the method of evaluation. Yet, there remains much work to be done for the future. Large disagreements between the evaluated cross sections are mostly due to the fact that experimental data are absent or discrepant. Re-evaluation is required taking account of the new experimental data. As to the discrepant data, an international co-operation may be helpful to resolve the discrepancies. Integral data will provide a good indication as to what data set is preferable[60,61].

As to the evaluations, we feel the necessity for examination of optical model parameters. A global parameter set is probably not successful. Optical model parameters are better determined to reproduce the local systematics, especially that of strength functions. This is because the disagreement in compound nuclear formation cross section at low energy does persist up to high energy.

The data for average level spacings are still lacking or discrepant. More data on resonance parameters are required. It may be worth trying a theoretical approach to predict the local systematics of average level spacings or the level density parameter a. The same is also true for the radiation widths.

Finally, there exist appreciable disagreement among the adopted level schemes for a number of nuclides. Inter-comparison of level scheme and studies about its effect on the calculation of cross sections are quite laborious work. But we think this work is necessary and useful for the construction of good evaluated data file.

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List of Tables

Figure Captions

Fig. 1 SNR-300 spectrum averaged capture cross sections multiplied by isotopic concentrations. (reproduced from Heijiboer et al.[53])

Fig. 2 Capture cross section of ^{93}Nb.

Fig. 3 Capture cross section of ^{133}Cs.

Fig. 4 Scattering radii. Comparison of optical model calculations and measurements. Experimental data were taken from [19].

Fig. 5 Neutron strength functions. Comparison of optical model calculations and measurements. Experimental data are from [19] and recent ORELA results or those adopted therein ([Bo75,76], [Mu74,75,76a,76b,77]). Calculated S_0 values with potential of Becchetti et al. are about 1.5×10^{-4} in Fig. 5a and are not plotted in this Figure.

Fig. 6 Total and compound nucleus formation cross sections. Comparison of optical model calculations with measurements. Experimental values of toal cross sections were read from graphs in [20] at selected energy points.

Fig. 7 Spin distribution function. (reproduced from [10])

Fig. 8 Energy dependence of σ^2 for ^{109}Ag. (reproduced from [10])

Fig. 9 Dependence of calculated capture cross sections for ^{109}Ag on spin cut-off parameter (reproduced from [10]).

Fig. 10 Local systematics of level density parameter <u>a</u>.

Fig. 11 Capture cross section for ^{75}As.(reproduced from Schmittroth [34].) The figure shows the correlation between the calculated capture cross section and the target level density. Stair-case plot of nuclear levels is shown at the bottom of figure. ——— : reference calc. Target level density was determined by total number of known levels. ---: modified calc. Target density was determined so as to obtain overall fit to low-lying levels.

Table 1　　Status of fast neutron capture cross sections of fission products

Isotope	Half-life or % Abn	Pu-239 % cum. Yield	Class	Experimental Data					Evaluated Data [a)]						WRENDA requests [b)]		
				Resonance Parameters			keV σ (n, γ)										
				N_{Res}(Emax,keV)	Γ_γ	Ongoing	Available	Ongoing	B	US	It.	Fr.	Ne.	Ja.	%	P	Object
Ge 70	20.7	-		21 (38.4)	X				X	X							
Ge 72	27.5	$1.1\cdot10^{-4}$		16 (39.5)	X				X	X							
Ge 73	7.7	$2.5\cdot10^{-4}$		47 (8.53)	X				X	X							
Ge 74	36.4	$5.9\cdot10^{-4}$		10 (61.0)	X		X		X	X							
Ge 76	7.7	$3.1\cdot10^{-3}$		8 (48.7)	X				X	X							
As 75	100	$1.4\cdot10^{-3}$		135 (9.69)	X		X		X	X							
Se 74	0.9	-		8 (7.22)	X		X		X	X							
Se 76	9.0	-		22 (24.2)	X				X	X							
Se 77	7.5	$8.6\cdot10^{-3}$		37 (3.92)	X				X	X							
Se 78	23.5	0.029		20 (40.5)	X				X	X							
Se 79	6.5×10^{4}y	0.025		-	-												
Se 80	50.0	0.048		15 (39.9)	X		X		X	X							
Se 82	9.0	0.16		4 (26.6)	-		X		X	X							
Br 79	50.69			7 (0.39)	X		X		X	X							
Br 81	49.31	0.18	IV	3 (0.21)	X		X		X	X				X			
Kr 80	2.25	~ 0		2 (0.64)	X				X	X							
Kr 82	11.6	~ 0		1 (0.04)	-				X	X							
Kr 83	11.5	0.29	IV	2 (0.23)	X				X	X							
Kr 84	57.0	0.47		2 (0.58)	-				X	X				X	10	1	ACT
Kr 85G	10.7y	0.14		-	-	KIL				X							
Kr 86	17.3	0.74		-	-				X	X							
Rb 85	72.17	0.60	IV	43 (17.2)	X		X		X	X				X			
Rb 87	27.83	0.95		10 (23.5)	X		X		X	X				X			
Sr 84	0.56	-		10 (3.35)	-				X								
Sr 86	9.9	~ 0		24 (23.1)	-		X	ORL	X	X							
Sr 87	7.0	-		37 (9.97)	X		X	ORL	X	X							
Sr 88	82.6	1.35		54 (700)	X			ORL	X	X							
Sr 89	51d	1.67		-	-					X							
Sr 90	29y	2.12		-	-					X				X			
Y 89	100	1.67		37 (76.8)	-		X	ORL	X	X				X			

Table 1 (cont'd)

Isotope	Half-life or % Abn	Pu-239 % cum. Yield	Class	Experimental Data: Resonance Parameters N_{Res} (Emax,keV)	Γ_γ	Ongoing	keV $\sigma(n,\gamma)$ Available	Ongoing	Evaluated Data: B	US	It.	Fr.	Ne.	Ja.	WRENDA requests: %	P	Object
Y 91	59d	2.39		-	-					X							
Zr 90	51.46	2.12		18 (68.0)	X		X	ORL	X	X				X	10	2	RP
Zr 91	11.23	2.43	IV	20 (4.28)	X	CNEN/GEL	X	ORL	X	X	X			X	10	1	RP
Zr 92	17.11	2.93		18 (47.0)	X		X	ORL	X	X	X			X	10	1	RP
Zr 93	$9.5x10^5$y	3.78	II	1 (0.11)	-				X	X	X			X	20-30	2	BU
Zr 94	17.40	4.30		26 (41.6)	X		X	ORL	X	X				X	10	2	RP
Zr 95	65d	4.89	IV	-	-					X					20	2	BU
Zr 96	2.80	4.93	III	18 (58.3)	X	CNEN/GEL	X		X	X				X	10	1	RP
Nb 93	100	-		many (7.32)	X	GEL	X	ORL,KTO/TIT	X	X			X	X	10-25 5	1-3 2	FUS,FBR, DOS
Nb 94	$2.0x10^4$y	-		2 (0.023)	X					X					10	3	FUS
Nb 95	35d	4.89	IV	-	-					X							
Mo 92	15.84	-		43 (31.1)	X			ORL	X				X	X	20 10	2 2	FBR FUS
Mo 94	9.04	-		13 (5.38)	X			ORL	X	X			X	X	20	2	FBR
Mo 95	15.72	4.89	II	55 (2.14)	X		X	ORL,RPI	X	X	X	X	X	X	30	2	BU,FBR
Mo 96	16.53	-		24 (7.05)	X		X	ORL	X	X			X	X	10-20	2	FBR,ASTRO
Mo 97	9.46	5.58	I	64 (1.94)	X		X	ORL,RPI	X	X	X	X	X	X	20	1	BU
Mo 98	23.78	5.71	II	24 (9.05)	X		X	ORL	X	X		X	X	X			
Mo 100	9.63	6.87	II	25 (4.73)	X		X	ORL	X	X		X	X	X			
Tc 99	$2.1x10^5$y	6.43	I	11 (0.28)	X	KUR,KIL RPI	X	RPI	X	X	X	X	X	X	10-20	1	BU
Ru 96	5.7	-		-	-		X		X								
Ru 98	2.2	-		-	-				X								
Ru 99	12.72	-		13 (0.51)	X				X	X							
Ru 100	12.62	-		1 (0.23)				ORL	X	X	X						
Ru 101	17.07	6.04	I	30 (0.67)	X			ORL,RPI	X	X	X	X	X	X	10-20	1	BU
Ru 102	31.61	6.09	II	3 (1.3)	X		X	ORL,RPI	X	X	X	X	X	X	30	2	BU
Ru 103	40d	6.97	II	-	-					X	X	X			20	2	BU
Ru 104	18.58	6.03	II	4 (1.06)	-		X	ORL,RPI	X	X	X	X	X	X	10-30	1-2	BU
Ru 106	369d	4.25	III	-	-					X	X			X	10	1	BU
Rh 103	100	6.97	I	many (4.14)	X		X	ORL,RPI	X	X	X	X	X	X			
Rh 105	36h	5.39		-	-					X							

Table 1 (cont'd)

Isotope	Half-life or % Abn	Pu-239 % cum. Yield	Class	Experimental Data: Resonance Parameters N_{Res}(Emax,keV)	Γ_γ	Ongoing	keV σ (n, γ): Available	Ongoing	Evaluated Data [a]: B	US	It.	Fr.	Ne.	Ja.	WRENDA requests [b]: %	P	Object
Pd 102	0.96	-		1 (0.19)	-				X				X				
Pd 104	10.97	-		1 (0.19)	-		X	ORL	X	X			X				
Pd 105	22.23	5.39	I	58 (0.81)	X	RPI	X	ORL	X	X	X	X	X	X	10-20	1	BU
Pd 106	27.33	4.26	IV	1 (0.28)	-			ORL	X	X			X				
Pd 107	7×10^6y	3.00	I	-	-	RPI		RPI	X	X	X	X	X	X	10-20	1	BU
Pd 108	26.71	2.53	III	3 (0.092)	X		X	ORL	X	X		X	X	X			
Pd 110	13.5	0.74		-	-		X	ORL	X	X			X	X			
Ag 107	51.82	-		75 (2.66)	X		X	KTO (natural)	X	X			X	X			
Ag 109	48.18	1.38	II	81 (2.51)	X		X		X	X	X	X	X	X	30 15	2 2	BU FUS
Ag 110m	253d	-		-	-												
Ag 111	7.5d	0.27		-	-					X							
Cd 106	1.22	-		-	-	KIL		ORL	X								
Cd 108	0.87	-		-	-	KIL		ORL	X	X							
Cd 110	12.39	-		80 (9.90)	X	COL		ORL	X	X					10	2	ASTRO
Cd 111	12.75	0.27	IV	many (2.29)	X	COL		ORL,NSW	X	X							
Cd 112	24.07	0.12		98 (11.5)	X	COL		ORL	X	X							
Cd 113	12.26	0.084		38 (2.24)	X	COL		ORL	X								
Cd 114	28.86	0.054		54 (10.1)	X	COL		ORL	X	X							
Cd 116	7.58	0.037		21 (8.82)	X	COL		ORL	X	X							
In 113	4.28	-		42 (2.04)	X		X		X	X							
In 115	95.72	0.035		233 (1.98)	X	COL	X	JAE	X	X							
Sn 112	0.96	-		12 (1.42)	X				X								
Sn 114	0.66	-		5 (1.98)	-				X								
Sn 115	0.35	-		4 (0.87)	-	ORL			X	X							
Sn 116	14.3	-		11 (4.64)	X		X		X	X							
Sn 117	7.61	0.035		56 (2.98)	X	ORL	X		X	X							
Sn 118	24.03	0.035		12 (4.73)	X	ORL	X		X	X							
Sn 119	8.58	0.037		16 (1.26)	-	ORL	X		X	X							
Sn 120	32.85	0.037		169 (57.17)	-	ORL	X		X	X							
Sn 122	4.72	0.047		6 (6.87)	-	ORL	X		X	X							

Table 1 (cont'd)

Isotope	Half-life or % Abn	Pu-239 % cum. Yield	Class	Experimental Data					Evaluated Data[a)]						WRENDA requests[b)]		
				Resonance Parameters			keV σ (n, γ)										
				N_{Res}(Emax.keV)	Γ_γ	Ongoing	Available	Ongoing	B	US	It.	Fr.	Ne.	Ja.	%	P	Object
Sn 123	129d	0.028		-	-					X							
Sn 124	5.94	0.069		7 (9.97)	X	ORL	X		X	X							
Sn 126	1x10^{5}y			-	-					X							
Sb 121	57.25	0.041		134 (2.53)	X		X		X	X					15	2	FBR
Sb 123	42.75	0.056		107 (4.17)	X		nat. Sb		X	X					15	2	FBR
Sb 125	2.7y	0.11		-	-					X							
Sb 126	12d	0.22		-	-					X							
Te 122	2.46	-		42 (10.82)	X		X	ORL	X	X							
Te 123	0.87	-		43 (2.00)	X		X	ORL	X	X							
Te 124	4.61	~0		84 (28.1)	X		X	ORL	X	X							
Te 125	6.99	0.11		114 (7.75)	X	JUL	X	ORL	X	X							
Te 126	18.71	0.22		65 (17.8)	X		X	ORL	X	X							
Te 127m	109d	0.090		-	-					X							
Te 128	31.79	0.84		38 (21.8)	X		X	ORL	X	X				X			
Te 129m	34d	0.28		-						X							
Te 130	34.48	2.68		22 (30.2)	-		X	ORL	X	X				X			
Te 132	78h	5.09		-	-												
I 127	100	0.53	III	many (4.00)	-	GEL	X	TIT/KTO	X	X		X	X	X			
I 129	1.4x10^{7}y	1.69	II	5 (0.153)	-				X	X		X	X	X	20	2	BU
I 131	8.1d	3.89		-	-					X							
Xe 128	1.92	-		9 (3.44)	-				X	X							
Xe 129	26.44	-		69 (4.08)	X				X	X							
Xe 130	4.08	~0		11 (3.56)	-				X	X							
Xe 131	21.18	3.89	II	39 (3.95)	X				X	X				X	20	1	BU
Xe 132	26.89	5.16	II	3 (3.85)	-				X	X							
Xe 133	5.3d	6.84		-	-					X							
Xe 134	10.44	7.22	IV	1 (1.01)	-				X	X							
Xe 135	9.2h	7.22		1 (8.4x10^{-5})	X					X							
Xe 136	8.87	6.55		-	-				X	X							
Cs 133	100	6.84	I	160 (3.50)	X	KIL	X	RPI, TIT/KTO	X	X	X	X	X	X	10-30	1-2	BU

Table 1 (cont'd)

Isotope	Half-live or % Abn	Pu-239 % cum. Yield	Class	Experimental Data: Resonance Parameters N_{Res}(Emax,keV)	Γ_γ	Ongoing	keV $\sigma(n,\gamma)$ Available	Ongoing	Evaluated Data a): B	US	It.	Fr.	Ne.	Ja.	WRENDA requests b): %	P	Object
Cs 134	2.1y	–		–	–					X							
Cs 135	$2{,}3\times10^6$y	7.22	I	–	–	KIL			X	X	X	X		X	10-20	1	ASTRO
Cs 137	30.1y	6.53	IV	–	–	KIL			X	X				X			
Ba 134	2.42	–		8 (1.89)	X			ORL	X	X							
Ba 135	6.59	–		40 (2.00)	X			ORL	X	X							
Ba 136	7.81	0.10		8 (7.31)	–			ORL	X	X					10		ASTRO
Ba 137	11.32	–		8 (1.74)	–			ORL	X	X							
Ba 138	71.66	5.69		41 (192)	–		X	ORL	X	X	X			X			
Ba 140	12.8d	5.49	IV	–	–					X	X						
La 139	99.91	5.84	III	76 (10.4)	X	COL	X	ORL	X	X	X	X	X	X			
Ce 140	88.48	5.50	IV	3 (56)	–	COL	X	ORL	X	X	X	X		X			
Ce 141	32.5d	5.66	IV	–	–					X	X						
Ce 142	11.07	4.98	III	4 (4.38)	–		X		X	X	X			X			
Ce 144	284d	3.77	IV	–	–				X	X	X			X	10	1	BU
Pr 141	100	5.96	II	120 (10.0)	X		X	ORL	X	X		X	X	X			
Pr 143	13.6d	4.37	IV	–	–					X	X	X					
Nd 142	27.11	–		37 (31.1)	–			ORL	X	X							
Nd 143	12.17	4.46	II	111 (5.50)	X	RPI,GEL		ORL, RPI	X	X		X		X	20	1	BU
Nd 144	23.85	3.77	IV	35 (19.4)	X			ORL,RPI	X	X				X			
Nd 145	8.30	3.02	II	179 (4.64)	X	ANL		ORL,RPI	X	X		X		X	10-20	1	BU
Nd 146	17.22	2.48	III	44 (17.3)	X		X	ORL	X	X				X	20	2	BU
Nd 147	11d	1.95		–	–					X	X						
Nd 148	5.73	1.66	III	67 (11.9)	X		X	ORL	X	X				X	20	2	BU
Nd 150	5.62	0.97	II	79 (13.8)	X		X		X	X				X			
Pm 147	2.62y	1.95	I	39 (0.32)	X	JUL			X	X		X		X	20	1	BU
Pm 148g	5.37d	–		–	–					X							
Pm 148m	41.3d	–		1 (1.69×10^{-4})	X					X							
Pm 149	53h	1.24		–	–					X							
Sm 144	3.16	–		–	–		X		X								
Sm 147	15.07	1.95	III	131 (1.16)	X		X	USSR	X	X			X	X	20	1	BU

Table 1 (cont'd)

Isotope	Half-life or % Abn	Pu-239 % cum. Yield	Class	Experimental Data					Evaluated Data [a]						WRENDA requests [b]		
				Resonance Parameters			keV σ (n, γ)										
				N_{Res}(Emax,keV)	Γ_γ	Ongoing	Available	Ongoing	B	US	It.	Fr.	Ne.	Ja.	%	P	Object
Sm 148	11.27	-		-	-		X		X	X			X	X			
Sm 149	13.84	1.25	I	87 (0.25)	X		X	RPI USSR	X	X	X	X	X	X	10-20	1	BU
Sm 150	7.47	0.03		22 (1.56)	X		X		X	X			X	X			
Sm 151	93y	0.77	I	10 (0.013)	X	KAP			X	X		X	X	X	10-30	1	BU
Sm 152	26.63	0.58	III	90 (5.10)	X	COL	X		X	X			X	X			
Sm 154	22.53	0.27	IV	33 (5.08)	X	COL	X		X	X			X	X			
Eu 151	47.77	0.77		105 (0.99)	X	KIL	X	JAE	X	X				X	5-10	1	FBR
Eu 152	4.8y	-		-	-	KIL,JUL				X							
Eu 153	52.23	0.38	II	76 (0.097)	X	KIL	X	JAE	X	X				X	5-30	1 2	FBR BU
Eu 154	16y	-		-	-	KIL				X	X						
Eu 155	4.65y	0.21	III	-	-	KIL				X	X			X	20	2	BU
Eu 156	15d	0.082		-	-					X							
Gd 154	2.15	-		48 (0.99)	X				X	X							
Gd 155	14.73	0.21		92 (0.18)	X		X		X	X				X			
Gd 156	20.47	0.083		31 (1.43)	X	CNEN/GEL	X		X	X				X	5	1	RP
Gd 157	15.68	0.075	IV	56 (0.31)	X		X		X	X				X			
Gd 158	24.87	0.042		93 (9.98)	X		X		X	X					10	1	RP
Gd 159	18h	0.022		-	-					X							
Gd 160	21.90	0.010		44 (9.66)	X		X		X						10	1	RP
Tb 159	100	0.022		25 (0.11)	X		X	ORL	X	X		X					
Tb 160	72.3d	0.002		-	-					X							

a) B ... 'CNEN-1' evaluation (Benzi et al) /4-6/
US ... ENDF/B-4 /7/
It ... 'CNEN-2' (Bologna library) /12/
Fr ... CEA-evaluation /13,14/
Ne ... RCN-2 and RCN-2A /8-14/
Ja ... JENDL-1 /15,16/

b) P ... priority
ACT ... activation
ASTRO... astrophysics
BU ... burnup
DOS ... dosimetry
FBR ... fast breeder reactors
FUS ... fusion
RP ... resonance parameters

Table 2 Measured capture cross sections at 24 keV for ^{115}In, ^{127}I and ^{197}Au (in mb)

Neutron Energy, Reference	^{115}In	^{127}I	^{197}Au	Method
Activation				
24 keV Macklin + [Ma 57]	805±80	820±60[a]	1120±110	Sb-Be source, γ-line intensity measurement.
20 keV Booth + [Bo 58]	980±220	(820±60)[b]	890±190	Sb-Be source, double-ratio comparison. $\beta+\gamma$.
21 keV Kononov + [Ko 59]	590±20	(820)[b]	960±6	Sb-Be source, double-ratio comparison. β-count.
24 keV Chaubey + [Ch 65]	580±40(54*m*) 800(54*m*+13s)	(820)[b]	500±35	Sb-Be source, γ-line intensity measurement $+\beta$-count.
24 keV Robertson [Ro 65]	—	832±26	—	Calibrated Sb-Be source.
24.3 keV Rimawi + [Ri 75]	528±32 (54*m*)	767±50	630±17[a]	Fe-filter. ~400 keV γ-line counting rel. to 412 keV Au line.
Other methods				
30 keV Macklin + [Ma 63]	(763)[b]	733(±9%)	515(±9%)	Xelene liq-scint., 3MV VdG. T(p,n).
24 keV Schmitt + [Sh 60]	823±60	800±80[c]	630±60[c].	Sphere transmission, Sb-Be source
24 keV Belanova + [Be 65]	776±66	—	660±60[c]	Sphere transmission, Sb-Be source
24 keV Yamamuro + [Ya 75]	750±45	760±35	680±35	Fe filter. C_6F_6 scint., $^{10}B(n,\alpha)$ flux monitor.
24 keV[d] Macklin + [Ma 75]	—	—	622[d]	ORELA. C_6F_6 scint., 4.9 eV Au resonance. $^6Li(n,\alpha)$ flux monitor.

a) Used as standard, measured also in this experiment.

b) Used as standard, assumed.

c) Re-analyzed by Bogart and Semler [Bo 66].

d) Cited in Rimawi et al. [Ri 75], the value converted for Fe-filtered neutron spectrum.

Table 3. Activation data near 25keV for FBR applications

Nuclide	Macklin + (Ma57) 24keV	Booth + (Bo58) 20keV	Kononov + (Ko58) 21keV	Chaubey + (Ch66,Ha68) 24keV	Stupegia + (St68) 24keV	Chaturvedi+ (Ch70) 24keV	Thirumala+ (Th70) 25keV	Murty + (Mu73) 24keV	Rao + (Ra72) 25keV	Data obtained by other methods
Zr 96	22±4(?) (0.95)									40 ±12 (Ma63)
Nb 93			120±12 (1.0-1.1)							270±15 (Be65) 330±17 (Ya75)
Mo 98	209±12 (0.73)	390±120 (0.27)			190±20	121.5		252±38 (0.92)		
Mo100	38±4(5.14) 45±9(1.12)		112±3 (1.0-1.1)	110±15 (0.93)				131±20 (0.92)		
Ru102	386±39 (0.97)							350±42 (0.92)		
Ru104	211±21 (2.09)			80±10 (0.93)				204±25 (0.92)		
Rh103				510±52 (0.93)					645±90	
Pd108	290±35 (0.86)	580±200 (0.93)		185±15 (0.93)						
Ag109				690±60 (0.93)						
Cs133		900±300 (1.06)								580±35 (Ya75)
La139	50±7 (0.86)	49±15 (1.0)		50±10 (0.93)	53±5	13				
Ce142	425±43 (0.85)			525±50 (0.93)						
Pr141	155±15 (1.03)	170±40 (0.96)		100±15 (0.93)		82				
Nd146							89±10 (0.71)			
Nd148				165±35 (0.93)			195±20 (0.62)			
Nd150				125±25 (0.93)			85±9 (0.73)			
Sm147										1173±192 (Ma63)
Sm149										1622±279 (Ma63)
Sm152	668±100 (1.17)			575±60 (0.93)						411±71 (Ma63)
Eu151									3260±15	

Note : The number in parenthesis is the renormalization factor which should be multiplied to the listed experimental data. The renormalization actor was estimated based on the recent values of reference cross sections [19] and the decay data (G. Erdtmann [54]). See also Table 3 (cont'd).

Table 3 (cont'd) Method and Standards used in activation measurements

Author	Energy	Source	Method	Standards
Macklin + [Ma57]	24keV	Sb-Be	γ	^{127}I : σ_{24} = 820 mb, γ/dis. given
Booth + [Bo58]	20keV	Sb-Be	β +γ, D.C.[a)]	^{127}I : σ_{24} = 820 mb, σ_{th} = 5.5b.[b)] σ_{th} (target) given.
Kononov + [Ko58]	21keV	Sb-Be	β , D.C.	^{127}I : σ_{25} = 820 mb, σ_{th} = 6.7b. σ_{th} (target) from BNL-325 ('55,'59)
Chaubey + [Ch66]	24keV	Sb-Be	β	^{127}I : σ_{24} = 820 mb
Hasan + [Ha68]	Same as for [Ch66]			
Stupegia + [St68]	24keV	$^{7}Li(p,n)$	β or γ	$^{235}\sigma(n,f)$(White) at high energy, long counter at low energy.
Chaturvedi + [Ch70]	24keV	Sb-Be	γ	^{197}Au : σ_{24} = 640 mb, γ/dis. from Table of Isotopes, 6-th ed.
Thirumala + [Th70]	25keV	Sb-Be	D.C.	^{127}I : σ_{25} = 832 mb, σ_{th} = 6.17b, σ_{th} (target) given.
Murty + [Ma73]	24keV	Sb-Be		^{127}I : σ_{24} = 832 mb
Rao + [Ra72]	25keV	Sb-Be	γ	^{127}I : σ_{24} = 832 mb
Siddapa [Si73]	23keV	Sb-Be	γ	^{127}I : σ_{23} = 836 mb, Table of Isotopes, 6-th ed.

a) double-ratio comparison. b) $\sigma_{th}(^{127}I) = 6.2 \pm 0.2$b from BNL-325, third ed. vol.1 (1973)

Table 4 Status of capture data for fast reactor applications

Nuclide	Number of expt.	Energy range	Discrepancies and comments
Zr 93	-		Only one resonance is measured.
Zr 96	3	24,30,195keV	24keV data are discrepant by a factor 2.
Nb 93	10	11keV-2.5MeV	ORELA data are smaller by 20% than Kompe's data below 20 keV.
Mo 95	1	30eV-46keV	ORELA data agree with data of Kapchigashev et al.
Mo 97	1	30eV-61keV	Same as for Mo-95.
Mo 98	10	5eV-3.0MeV	ORELA data agree with data of Kapchigashev, but not with data of Stupegia.
Mo 100	10	5eV-6.2MeV	ORELA data agree with data of Kapchigashev, but not with data of Weston et al. and Tolstikov et al.
Tc 99	1	1eV-50keV	RPI exp. is ongoing.
Ru 101	-		ORNL exp. completed.
Ru 102	4	2,24,195keV	Discrepancy in energy variation. ORNL exp. in progress.
Ru 103	-		
Ru 104	6	24,195keV,3MeV	24keV data are very discrepant.
Ru 106	-		
Rh 103	19	0.4eV-4.7MeV	Large scattering among data sets below 10 keV.
Pd 105	1	20eV-200keV	RPI data (1971). ORNL exp. completed.
Pd 107	-		RPI exp. is ongoing.
Pd 108	4	20,24,195keV	24keV data are very discrepant. ORELA exp. completed.
Ag 109	6	2keV-200keV	Sets of data are systematically discrepant.
I 127	23	14eV-5.5MeV	New resolved res. pars., GEEL.
I 129	-		5 resonances up to 153 eV.
Xe 131	-		31 resonances up to 4keV.

Table 4 (cont'd)

Nuclide	Number of expt.	Energy range	Discrepancies and comments
Xe 132	-		3 resonances up to 3.9keV. No Γ_γ data given.
Cs 133	9	1eV-1.2MeV	Systematic dicrepancy between data of Popov+, Kompe, Yamamuro+.
Cs 135	-		No data at all. Res. pars. exp. is ongoing, Kiel.
La 139	12	1eV-5.9MeV	ORELA data published.
Ce 142	6	24,195keV,3Mev	
Pr 141	14	20eV-5.9MeV	ORELA data published. RPI exp. ongoing.
Nd 143	-		ORELA data published. RPI exp. ongoing.
Nd 145	-		ORELA data published.
Nd 146	2	23,25keV	ORELA data agree with two previous data at 25keV.
Nd 148	5	24keV,0.15-3MeV	ORELA data are smaller by 40% than 24keV activation data.
Nd 150	4	24keV,0.15-eMeV	24keV data are discrepant by 50%. Discrepancy in energy variation.
Pm 147	-		
Sm 147	1	30keV	Data of Yurelov+ disagree with 30 keV data.
Sm 149	1	30keV	Data of Yurelov+ and Hockenbury+ agree each other, but no with 30 keV data.
Sm 151	-		Res. pars. up to 13eV was extented to 300eV at KAPL/RPI.
Sm 152	7	24keV-3MeV	
Eu 151	5	0.8eV-2.5MeV	10-20% discrepancy in 3-30keV.
Eu 152	-		No resonance data. Res. pars. exp. is planned at Kiel.
Eu 153	6	1eV-200keV	Systematic 20% discrepancy in 0.8-12keV.
Eu 154	-		No resonance data. Res. pars. exp. is planned at Kiel.
Eu 155	-		No resonance data. Res. pars. exp. is planned at Kiel.

Table 5. Optical model parameters

		Igarasi et al.	Moldauer	Rosen et al.	Becchetti et al.	Wilmore and Hodgson
VOLUME TERM	R_1	$1.16A^{1/3}+0.6$	$1.16A^{1/3}+0.6$	$1.25A^{1/3}$	$1.17A^{1/3}$	$r_o^* A^{1/3}$ a)
	R_2	R_1	—	—	$1.26A^{1/3}$	—
	a_1	0.62	0.62	0.65	0.75	0.66
	a_2	a_1	—	—	0.58	—
	V_o	$\{46-0.25E_n, (A<147)$ $52.5-0.25E_n-40(N-Z)/A$ $(A\geq 147)$	46.0	$49.3-0.33E_n$	$56.3-0.32E_n$ $-24(N-Z)/A$	$47.0-0.267E_n$ $-0.00018E_n^2$
	W_v	$0.125E_n-0.0004E_n^2$	—	—	$0.22E_n-1.56$	—
SURFACE TERM	R_s	$\{1.16A^{1/3}+1.1\ (A<147)$ $1.16A^{1/3}+1.3\ (A\geq 147)$	$1.16A^{1/3}+1.1$ (Gaussian)	R_1	$1.26A^{1/3}$	$r_s^* A^{1/3}$ b)
	a_s	0.35	0.5	0.70	0.58	0.48
	W_s	7.0	14.0	5.75	$13.0-0.25E_n$ $-12(N-Z)/A$	$9.52-0.053E_n$
SPIN·ORBIT	R_{so}	R_1	R_1	R_1	$1.01A^{1/3}$	—
	a_{so}	a_1	a_1	a_1	a_1	—
	V_{so}	7.0	7.0	5.5	6.2	—

a) $r_o^* = 1.322 - 7.6\times10^{-4}A + 4\times10^{-6}A^2 - 8\times10^{-9}A^3$, b) $r_s^* = 1.266 - 3.7\times10^{-4}A + 2\times10^{-6}A^2 - 4\times10^{-9}A^3$

Table 6a. Level Scheme of ^{103}Rh and the Effect on Cross Sections

JENDL-1		Revised		RCN-2	
0.0 MeV	1/2-	0.0 MeV	1/2-	0.0 MeV	1/2-
0.040	7/2+	0.039750	7/2+	0.0398	7/2+
0.093	9/2+	0.093035	9/2+	0.0930	9/2+
0.298	3/2-	0.29498	3/2-	0.2949	3/2-
0.360	5/2-	0.37546	5/2-	0.3574	5/2-
0.537	5/2+	0.53684	5/2+	0.5368	5/2+
		0.60763	7/2+	0.6072	7/2+
0.651	7/2+	0.65009	7/2+	0.6500	5/2+
		0.65180	3/2+	0.6517	3/2+
0.798	5/2+			0.7980	9/2+
		0.8036	3/2-	0.8031	3/2-
0.843	3/2-	0.8477	7/2-	0.8475	7/2-
0.877	5/2-	0.8804	5/2-	0.8806	5/2-
0.915	5/2-	0.9200	9/2-	0.9200	9/2-
				0.9680	5/2-
				1.0100	5/2+
		continuum		1.0350	9/2+
				1.0800	7/2-
1.102	7/2+			1.1070	5/2-
				1.1400	5/2+
				1.1970	9/2-
				1.220	3/2+
1.247	9/2-				
				1.2520	5/2+
				1.252	5/2-
1.270	1/2-			1.2770	3/2-

Level Scheme	E_n = 0.5 MeV			E_n = 0.7 MeV			E_n = 1.0 MeV		
	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$
JENDL-1	1891	1043	131	1324	1192	105	802	1369	88.1
Revised	1884	1051	130	1300	1222	98.6	767	1415	76.7
RCN-2	1884	1051	130	1293	1230	97.5	756	1429	74.7

(cross sections in mb)

Table 6b. Level Scheme of ^{139}La and the Effect on Cross Sections

JENDL-1		Revised		RCN-2	
0.0 MeV	7/2+	0.0 MeV	7/2+	0.0 MeV	7/2+
0.1658	5/2+	0.1658	5/2+	0.1660	5/2+
0.570	3/2+				
0.830	3/2+				
0.930	9/2+				
1.070	7/2+				
1.206	1/2+	1.206	1/2+	1.206	1/2+
1.2191	9/2+	1.2191	9/2+	1.219	9/2+
1.2566	5/2+	1.2566	5/2+	1.257	5/2+
1.3813	7/2+	1.3813	7/2+	1.382	7/2+
1.4205	7/2+	1.4205	7/2+	1.421	7/2+
1.439	11/2-	1.439	11/2-	1.439	11/2-
1.4764	5/2+	1.4764	7/2+	1.477	7/2+
1.5363	7/2+	1.5363	7/2+	1.536	7/2+
1.5582	3/2+	1.5582	3/2+	1.558	3/2+
1.5782	9/2+	1.5782	9/2+	1.578	9/2+
1.6831	7/2+	1.6831	7/2+	1.683	7/2+
				1.714	5/2+
continuum		continuum		1.756	7/2+
				1.762	3/2+
				1.767	9/2+
				1.775	1/2+
				1.820	5/2+
				1.838	7/2-
				1.857	3/2+
				1.894	11/2+
				1.922	5/2+
				1.943	13/2+

Level Scheme	E_n = 1.0 MeV			E_n = 1.5 MeV			E_n = 1.75 MeV		
	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$	σ_{ce}	σ_{in}	$\sigma_{n\gamma}$
JENDL-1	1732	759	12.1	1146	1362	11.3	843	1623	9.86
Revised	1929	561	13.6	1510	995	14.6	1099	1364	12.3
RCN-2	1929	561	13.6	1511	994	14.6	1107	1355	12.9

(cross sections in mb)

Table 7 Average s-wave level spacings (ev) and radiation widths (meV)
(Number in paranthesis is the p-wave radiation width.)

Nuclide	CNEN-1	ENDF/B4	JEDL-1	CEA	RCN-2	Ds, Exp. data
Zr 93 D_s	220	379	300			
Γ_γ	160	194	300			
Zr 96 D_s	4117	3550	4790			
Γ_γ	120	220	250			
Nb 93 D_s	87.3	69.3	83.5		100	32.5(35,42)
Γ_γ	140	200(240)	144		146(195)	89.7±16.0(36)
Mo 95 D_s	94.9	114	69.2	89	82	102(35),114±35.2(36)
Γ_γ	205	350	180	160.5(299)	154(281)	51(42),80±25(Mu76a)
Mo 97 D_s	74.9	77.5	72.3	65	66	80(35),77.5±17.7(36)
Γ_γ	190	220	170	138(199)	134(190)	50(42),42±15(Mu76a)
Mo 98 D_s	1156	1010	640	730	1000	940(35),1014±112(36)
Γ_γ	160	150	93	85(106)	86(138)	940(42),950±150(Mu76a)
Mo100 D_s	1561	1340	680	520	700	770(35),1339±1040(36)
Γ_γ	135	150	75	58(77)	58(115)	770(42),420±100(Mu76a)
Tc 99 D_s	20.9	24.4	16.2	18.6	18.6	24.4(35,42)
Γ_γ	180	112	112	137	130	
Ru101 D_s	17	18.3	13.8	16.7	16.7	25(35),18.3±3.8(36)
Γ_γ	160	192	165	174	172	22.3(42)
Ru102 D_s	229	611	290.5	550	573	540(35)
Γ_γ	170	290	165	240	275	
Ru103 D_s		16		7.5		
Γ_γ		170		96		
Ru104 D_s	662	570	588	270	265	280(35)
Γ_γ	150	160	165	96	97	285±81(36)
Ru106 D_s		1230	1000			
Γ_γ		145	150			

Table 7 (cont'd)

Nuclide		CNEN-1	ENDF/B4	JEDL-1	CEA	RCN-2	Ds, Exp. data
Rh103	D_s	33.0	15.9	26.1	26.4	26.1	27 [35], 27.4±3.07 [36]
	Γ_r	195	171	164	162	161	33.8 [42], 12±1 [19]
Pd105	D_s	9.55	10.1	11.1	10	10.0	10.1±1.6 [36]
	Γ_r	150	153	155	157(196)	155	14.7 [42], 10 [19]
Pd107	D_s	8.66	10.9	10.0	5.5	4.2	
	Γ_r	140	140	140	123(160)	100(110)	
Pd108	D_s	41.5	290	291	165	200	
	Γ_r	120	98	100	65(76)	70(80)	
Ag109	D_s	18.8	T_r=0.02	12.7	17.9	17.5	12.8 [35], 19.5±2.54 [36]
	Γ_r	135		130	132(125)	129	19 [42]
I 127	D_s	13.0	14.7	13.2	15.05	12.2	13.0 [35], 14.7±2.44 [36]
	Γ_r	100	120	140	143(150)	95	13.0±0.5 [19]
I 129	D_s	17.9	26.0	21.0	30	30	26.1±6.66 [36]
	Γ_r	105	117	100	68(68)	107	
Xe131	D_s	33.8	T_r=0.0126	33.2			39.2±7.62 [36]
	Γ_r	110	117	114			
Xe132	D_s	735	T_r=8.6x10^{-4}				
	Γ_r	110	100				
Cs133	D_s	21.7	T_r=0.0383	23.2		20.0	20 [35], 20.2±3.32 [36]
	Γ_r	125	110	118		125	19.2 [42], 20±2 [19]
Cs135	D_s	64.1		60.0			
	Γ_r	110		125			
La139	D_s	484	312	311	270	286	100 [35], 312±45.8 [36]
	Γ_r	80	75	60	62(48)	50	240±10 [19]

Table 7 (cont'd)

Nuclide	CNEN-1	ENDF/B4	JEDL-1	CEA	RCN-2	Ds, Exp. data
Ce142 D_s	1000	1520	902			~1000 (19)
Γ_γ	50	79	66			
Pr141 D_s	114	75	135	132	120	75(35),63.9±10.4(36)
Γ_γ	75	83	80	83(99)	85	31.2(42),75±4(19)
Nd143 D_s	34.3		46.4	39		32(35),32.0±2.39(36)
Γ_γ	70		85	73(68)		32(42),32±3(19)
Nd145 D_s	24.1		24.2			18.9 (35),18.9±1.10(36)
Γ_γ	65		60			17.7(42),19±2(19)
Nd146 D_s	370		404			310(35),211±24.9(36)
Γ_γ	67		55			310(42),210±25(19)
Nd148 D_s	217		179			72.0±6.86(36)
Γ_γ	50		64			258(42),72±6(19)
Nd150 D_s	113		115			247(42)
Γ_γ	76		70			120±8(19)
Pm147 D_s	6.26		4.70	5.3		4.76(35),6.8±1.5(19)
Γ_γ	62		66	67(67)		4.76(42)
Sm147 D_s	6.54	7.4	4.26		6.3	8.18±1.43(36),7.4(35,42)
Γ_γ	95	63	67		100	7.4±0.7(19),7.00(Ki75)
Sm149 D_s	3.07	3.43	1.63	1.95	2.0	2.88±0.345(36),2.8(42)
Γ_γ	95	62	61	60.5(56.5)	76	2.9±0.3(19),2.38(Ki75)
Sm150 D_s	97.1		37		56.5	68(35),68±10(19)
Γ_γ	80		60		60	56.5(Ki75)
Sm151 D_s	1.305	1.3	1.50	0.90	1.72	1.3±0.2(19)
Γ_γ	70	74	75	78	96	1.72(Ki75)

Table 7 (cont'd)

Nuclide	CNEN-1	ENDF/B4	JEDL-1	CEA	RCN-2	Ds, Exp. data
Sm152 D_s	58.5		35.2		53.8	52(35), 52.5(42)
Γ_γ	66		63		70	51.8 ± 4.3(19), 53.8(Ki75)
Eu151 D_s	0.974	0.655	0.72			1.04(35)
Γ_γ	90	92	88			0.7 ± 0.2(19)
Eu152 D_s		0.444				
Γ_γ		92				
Eu153 D_s	0.999	1.3	1.46			1.45(35,42)
Γ_γ	95	114	94			1.3 ± 0.2(19)
Eu154 D_s		0.803				
Γ_γ		96				
Eu155 D_s			2.50			
Γ_γ			100			

Table 8 Inter-comparison of evaluated capture cross sections

Nuclide	Class	σ_c (30keV), mb					$<\sigma>_c$ (SNR-300), mb				$<\sigma_c>$(FSA), mb				Disagreement ratio		
		CNEN1	ENDF/B4	JENDL1	CEA	RCN2	ENDF/B4	JENDL1	CNEN2	RCN2	CNEN1	ENDF/B4	JENDL1	RCN2	at 30keV	SNR-300	FSA
Zr 93	II	144	73.2	177			86	160	110		11.0	21.2	44.9		2.4	1.86	4.1
Zr 96	III	27.7		40							8.0	23.8	28.8		1.44		3.6
Nb 93	(I)	262	229	274		283	210			210		26.9	34.0	28.9	1.20	1.0	1.26
Mo 95	II	336	393	384	385	396	290	300	270	300	36.5	47.1	51.2	50.7	1.18	1.11	1.40
Mo 97	I	378	372	359	389	396	280	310	280	300	41.8	43.8	46.9	43.3	1.10	1.11	1.12
Mo 98	II	104	94	101	90		101		104	86	27.3	28.6	29.3	21.2	1.16	1.22	1.39
Mo100	II	70.0	74.2	79.7	95	105	78		82	100	14.2	13.8	15.6	18.9	1.50	1.28	1.37
Tc 99	I	844	700	761	744	726	490	540	550	540	93.0	56.8	113	114	1.21	1.12	2.0
Ru101	I	902	851	1097	1000	970	530	710	760	690	74.5	61.0	105	76.4	1.29	1.42	1.72
Ru102	II	348	244	314	254	273	190	220	220	200	86.5	70.8	71.7	60.1	1.43	1.16	1.44
Ru103	II		711		968							52.7			1.36		
Ru104	II	155	161	177	206	210	140	160	180	170	24.6	33.7	37.2	48.4	1.33	1.29	1.97
Ru106	III		77.4	104								11.4	21.8		1.34		1.91
Rh103	I	793	960	984	890	927	700	650	630	640	85	90.5	98.8	87.1	1.24	1.11	1.10
Pd105	I	1221	1230	1178	1218	1163	830	760	840	810	106	135	124	110	1.04	1.11	1.27
Pd107	I	1245	895	1165	1386	1416	570	750	790	960	121	101	113	147	1.58	1.68	1.46
Pd108	III	188	185	200	223	216	160		200	180	32.1	45.4	57.8	41.2	1.21	1.25	1.80

Table 8 (cont'd)

Nuclide	Class	σ_c (30keV), mb					$<\sigma_c>$(SNR-300), mb				$<\sigma_c>$(FSA), mb				Disagreement ratio		
		CNEN1	ENDF/B4	JENDL1	CEA	RCN2	ENDF/B4	JENDL1	CNEN2	RCN2	CNEN1	ENDF/B4	JENDL1	RCN2	at 30keV	SNR-300	FSA
Ag109	II	897	624	1215	855	891	480	810	650	680	130	63.6	160	124	2.0	1.68	2.5
I 127	III	800	691	700	780	653	540			520	82.0	79.5	88.8	72.6	1.23	1.04	1.22
I 129	II	669	460	490	265	372	380	440		340	50	64.3	73.5	43.3	2.5	1.39	1.70
Xe131	II	491	300	424			210	370			39	26.2	47.5		1.64		1.81
Xe132	II	116	83.0				690(?)				27	23.0			1.40		1.17
Cs133	I	693	626	518	615	680	480	450	490	510	55	72.1	59.6	67.9	1.34	1.13	1.31
Cs135	I	294	64.5	275	334						27	5.8	39.2		5.2		6.8
La139	III	39.6	47.9	40	39	38	38		28	31	6.9	7.3	8.2	6.7	1.26	1.36	1.22
Ce142	III	47.1	43.0	55							14.2	15.9	22.9		1.28		1.61
Pr141	II	139	155	105	120	117	160		130	130	19.0	27.0	17.2	16.1	1.48	1.23	1.67
Nd143	II	339	277	243	242		300	290	340		42.3	65.5	86.7		1.40		2.1
Nd145	II	419	305	300	314		330	340	360		32.4	69.9	53.8		1.40		2.2
Nd146	III	138	157	85			130		71		30.1	58.8	28.5		1.85		2.1
Nd148	III	168	173	160			180		160		29.9	123	58.0		1.08	1.12	4.1
Nd150	II	361	281	100			220		210		53.3	57.4	60.6		3.6	1.05	1.14
Pm147	I	999	1300	938			1250	1080	1080		83	171	162		1.39	1.16	2.1
Sm147	III	1236	776	1005		1221					209	146	197	239	1.59		1.63

Table 8 (cont'd)

Nuclide	Class	σ_c (30keV), mb					$\langle\sigma_c\rangle$(SNR-300)[a], mb				$\langle\sigma_c\rangle$(FSA)[b], mb				Disagreement ratio		
		CNEN1	ENDF/B4	JENDL1	CEA	RCN2	ENDF/B4	JENDL1	CNEN2	RCN2	CNEN1	ENDF/B4	JENDL1	RCN2	at 30keV	SNR-300	FSA
Sm149	I	1687	1620	1645	1507	1947	1410	1990	1760	2240	168	133	331	285	1.29	1.59	2.5
Sm151	I	2100	1967	1825	2465	2062	2210	2070	2110	2130	157	293	259	135	1.35	1.07	2.2
Sm152	III	496	447	400		417	400			410	86	89.5	178	89.5	1.24	1.02	2.1
Eu151	(I)		3500	2983								534	517		1.17		1.04
Eu152	(I)		5196									444					
Eu153	II	2703	2550	2566			2290	2400	2480		303	311	354		1.06	1.08	1.17
Eu154	(I)		2920									227					
Eu155	III		2163	1885								790	349		1.15		2.3

a) capture cross section averaged over the SNR-300(fast breeder reactor) spectrum

b) " " " " " " thermal ^{235}U fission spectrum

Table 9a : Intercomparison of evaluated inelastic cross sections. ($\langle \sigma_{in} \rangle$ Bethe)

Nuclide	ENDF/B-4	JENDL-1	Nuclide	ENDF/B-4	JENDL-1
^{93}Zr	0.71	0.92	^{135}Cs	0.91	1.63
^{96}Zr	0.59	0.90	^{139}La	1.25	1.54
^{93}Nb	1.56	1.32	^{142}Ce	0.61	1.43
^{95}Mo	1.13	1.34	^{141}Pr	1.13	1.61
^{97}Mo	1.11	1.43	^{143}Nd	1.36	1.42
^{98}Mo	0.96	1.27	^{145}Nd	1.64	1.83
^{100}Mo	0.96	1.31	^{146}Nd	1.65	1.52
^{99}Tc	0.94	1.26	^{148}Nd	1.73	1.52
^{101}Ru	1.15	1.33	^{150}Nd	2.07	1.42
^{102}Ru	0.89	1.24	^{147}Pm	2.05	1.98
^{103}Ru	0.99		^{147}Sm	1.95	1.91
^{104}Ru	0.64	1.25	^{149}Sm	1.60	1.81
^{106}Ru	0.76	1.16	^{151}Sm	1.63	1.69
^{103}Rh	1.48	1.43	^{152}Sm	2.08	1.46
^{105}Pd	0.94	1.35	^{151}Eu	1.69	1.65
^{107}Pd	1.52	1.37	^{152}Eu	1.79	
^{108}Pd	0.93	1.20	^{153}Eu	1.87	1.75
^{109}Ag	1.30	1.41	^{154}Eu	1.89	
^{127}I	1.21	1.47			
^{129}I	1.16	1.46			
^{131}Xe	1.54	1.56			
^{132}Xe	1.36				
^{133}Cs	1.62	1.62			
^{134}Cs	1.47				

Table 96 Intercomparison of evaluated inelastic Cross sections. [$\langle \sigma_{in} \rangle_{\phi}$]

Nuclide	ENDF/B-4	JENDL-1	RCN-2	Nuclide	ENDF/B-4	JENDL-1	RCN-2
^{93}Zr	0.16	0.21		^{135}Cs	0.24	0.27	
^{96}Zr	0.051	0.077		^{139}La	0.28	0.28	0.25
^{93}Nb	0.22	0.17	0.18	^{142}Ce	0.084	0.21	
^{95}Mo	0.25	0.32	0.30	^{141}Pr	0.35	0.34	0.33
^{97}Mo	0.20	0.24	0.27	^{143}Nd	0.14	0.13	
^{98}Mo	0.14	0.18	0.19	^{145}Nd	0.48	0.48	
^{100}Mo	0.18	0.23	0.24	^{146}Nd	0.30	0.26	
^{99}Tc	0.40	0.40	0.38				
^{101}Ru	0.48	0.54	0.56	^{148}Nd	0.38	0.35	
^{102}Ru	0.18	0.24	0.21	^{150}Nd	0.56	0.49	
^{103}Ru	0.48			^{147}Pm	0.52	0.47	
^{104}Ru	0.19	0.29	0.26	^{147}Sm	0.46	0.41	0.51
^{106}Ru	0.26	0.34		^{149}Sm	1.15	0.67	0.88
^{103}Rh	0.46	0.39	0.38	^{151}Sm	2.23	1.38	1.71
^{105}Pd	0.35	0.42	0.45	^{152}Sm	0.58	0.45	0.70
^{107}Pd	0.43	0.48	0.49	^{151}Eu	0.83	0.73	
^{108}Pd	0.20	0.23	0.24	^{152}Eu	0.80		
^{109}Ag	0.40	0.34	0.42	^{153}Eu	0.96	0.76	
^{127}I	0.54	0.47	0.50	^{154}Eu	0.84		
^{129}I	0.39	0.46	0.45				
^{131}Xe	0.47	0.43					
^{132}Xe	0.20						
^{133}Cs	0.50	0.44	0.56				
^{134}Cs	0.73						

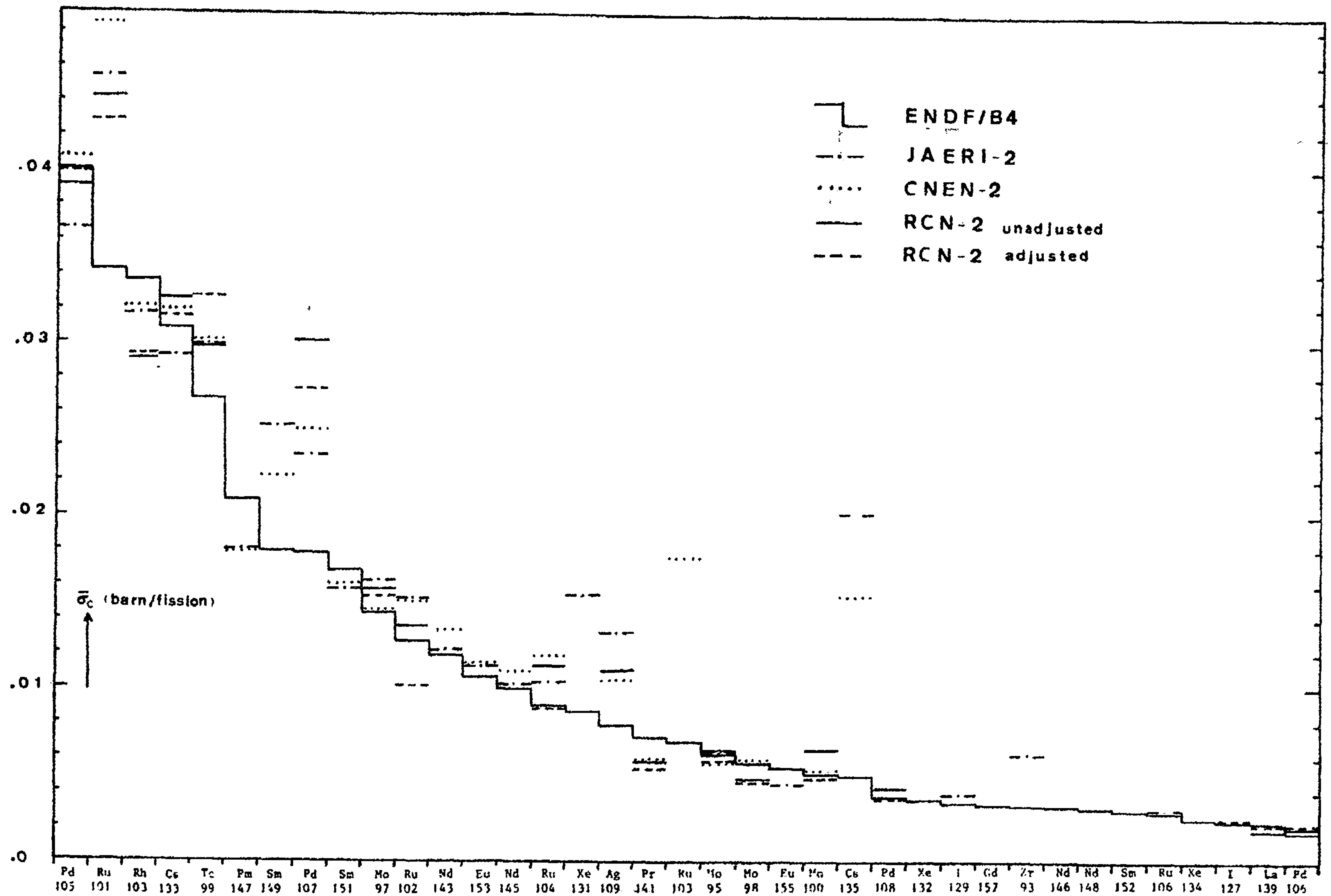


Fig. 1 : SNR-300 spectrum averaged cross sections multiplied by isotopic concentrations (from [53])

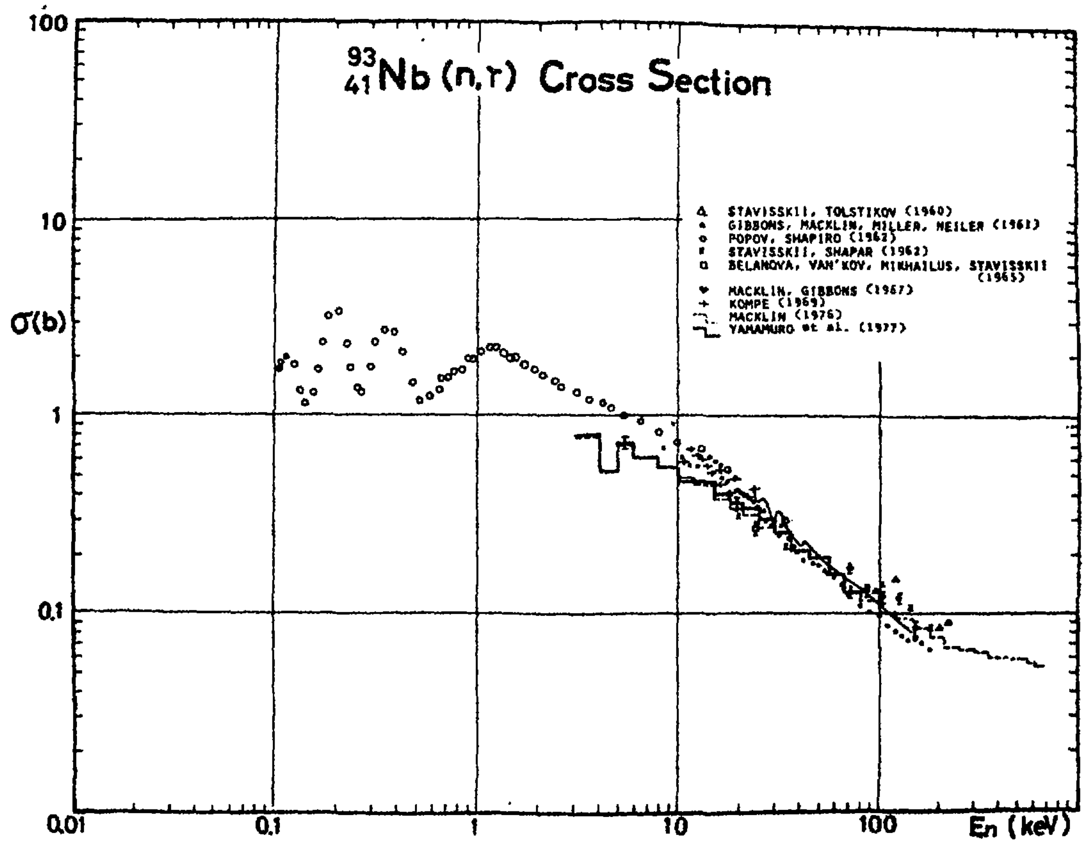


Fig. 2 (from[Ya 77])

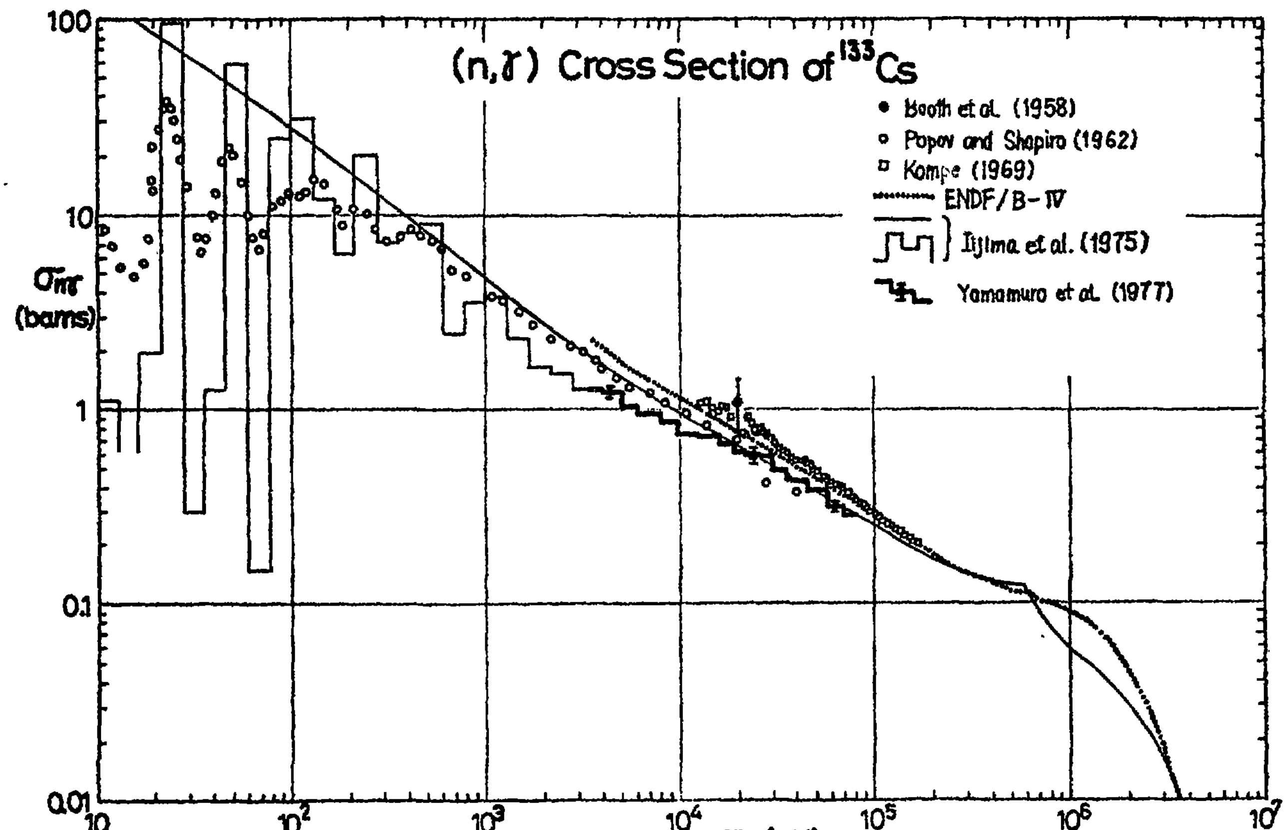
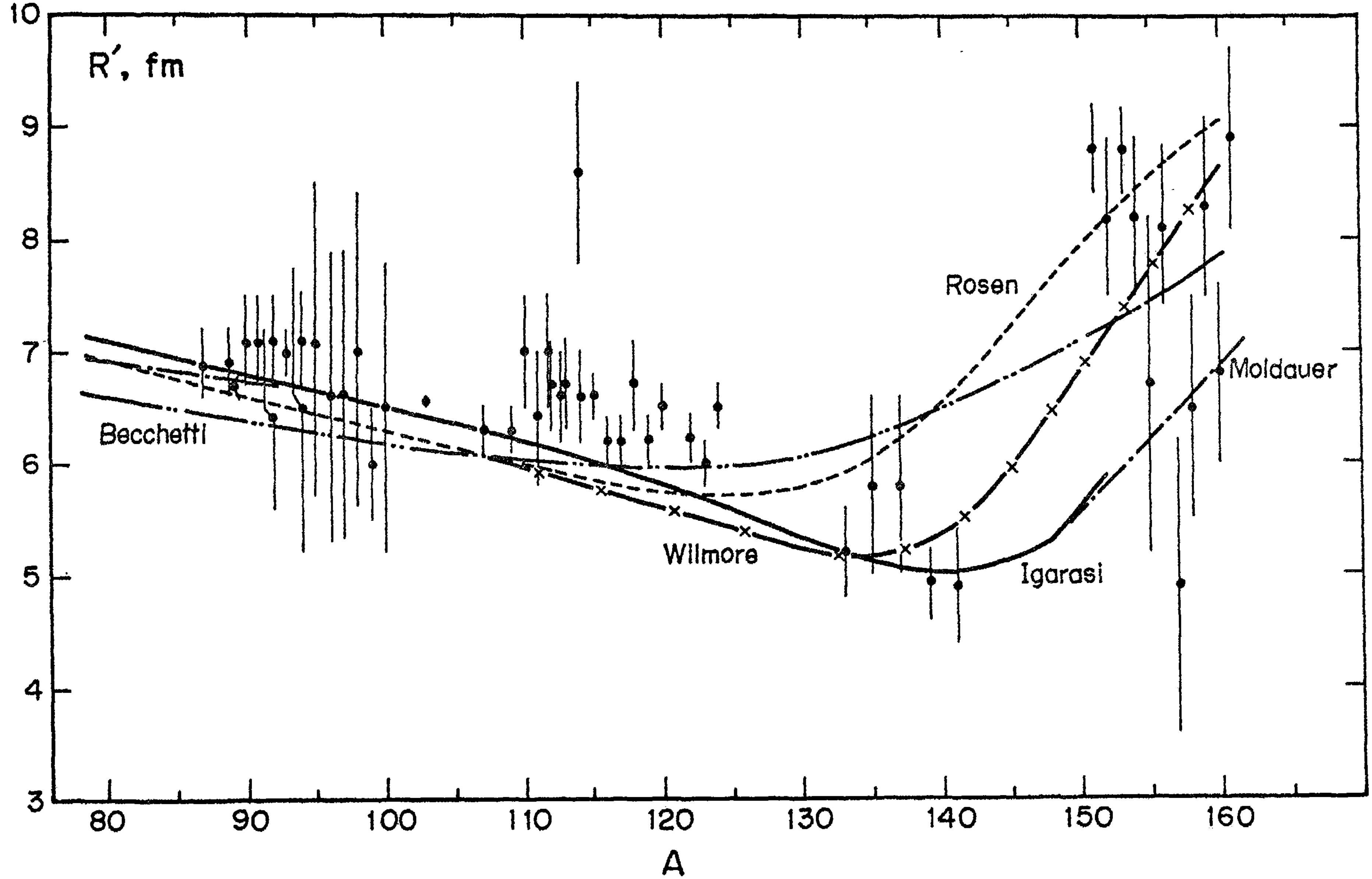


Fig. 3 (from [Ya 77])

Fig. 4 : Scattering radii. Comparison of optical model calculations and experimental data from /19/



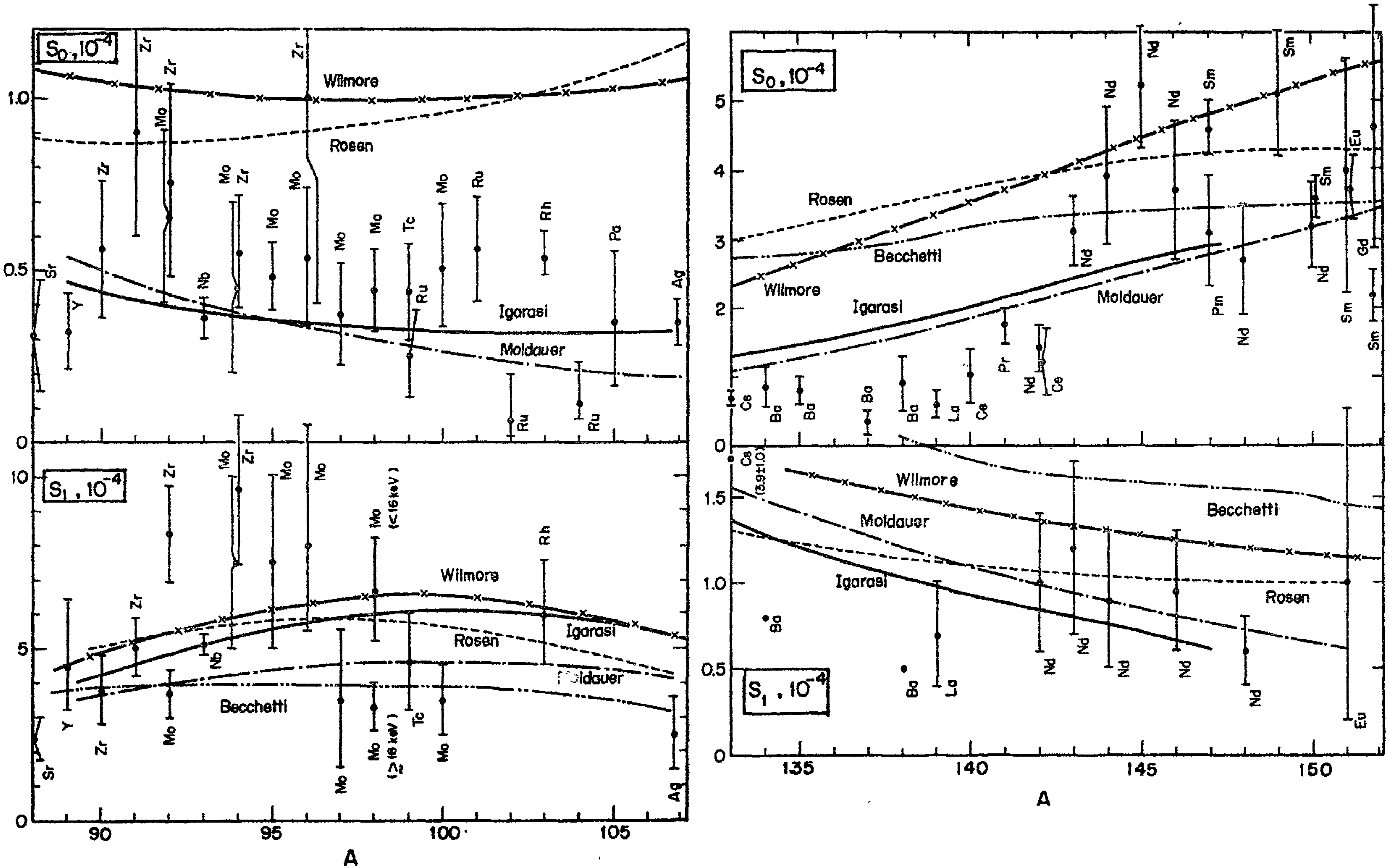
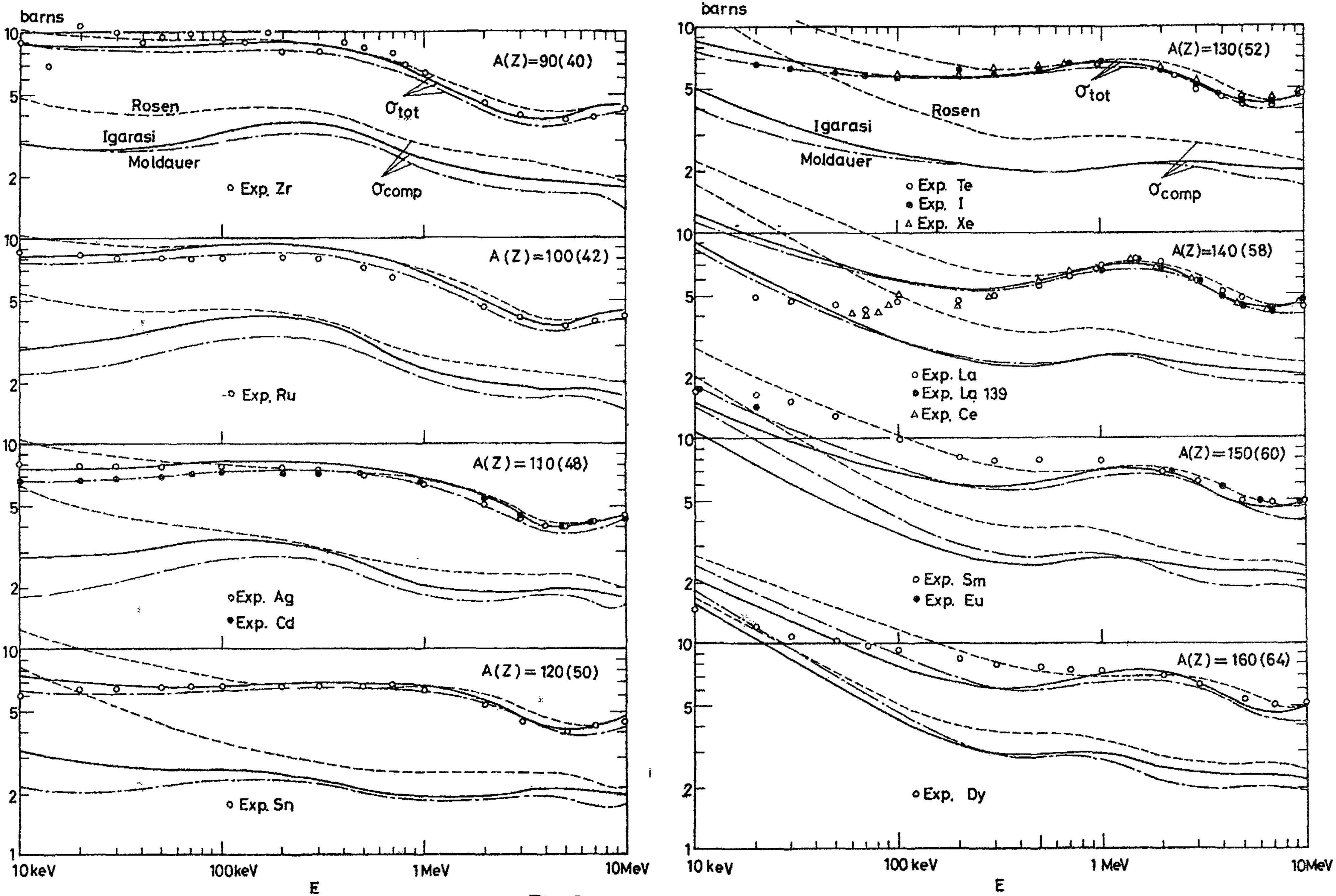

S0, 10-4
S1, 10-4
Wilmore
Rosen
Igarasi
Moldauer
Becchetti
A
90
95
100
105
135
140
145
150

Fig. 5

- 345 -



barns
A(Z)=90(40)
Rosen
Igarasi
Moldauer
σtot
σcomp
Exp. Zr
A(Z)=100(42)
Exp. Ru
A(Z)=110(48)
Exp. Ag
Exp. Cd
A(Z)=120(50)
Exp. Sn
10keV
100keV
1MeV
10MeV
E
barns
A(Z)=130(52)
σtot
Rosen
Igarasi
Moldauer
σcomp
Exp. Te
Exp. I
Exp. Xe
A(Z)=140(58)
Exp. La
Exp. La 139
Exp. Ce
A(Z)=150(60)
Exp. Sm
Exp. Eu
A(Z)=160(64)
Exp. Dy
10 keV
100 keV
1 MeV
10MeV
E

Fig. 6

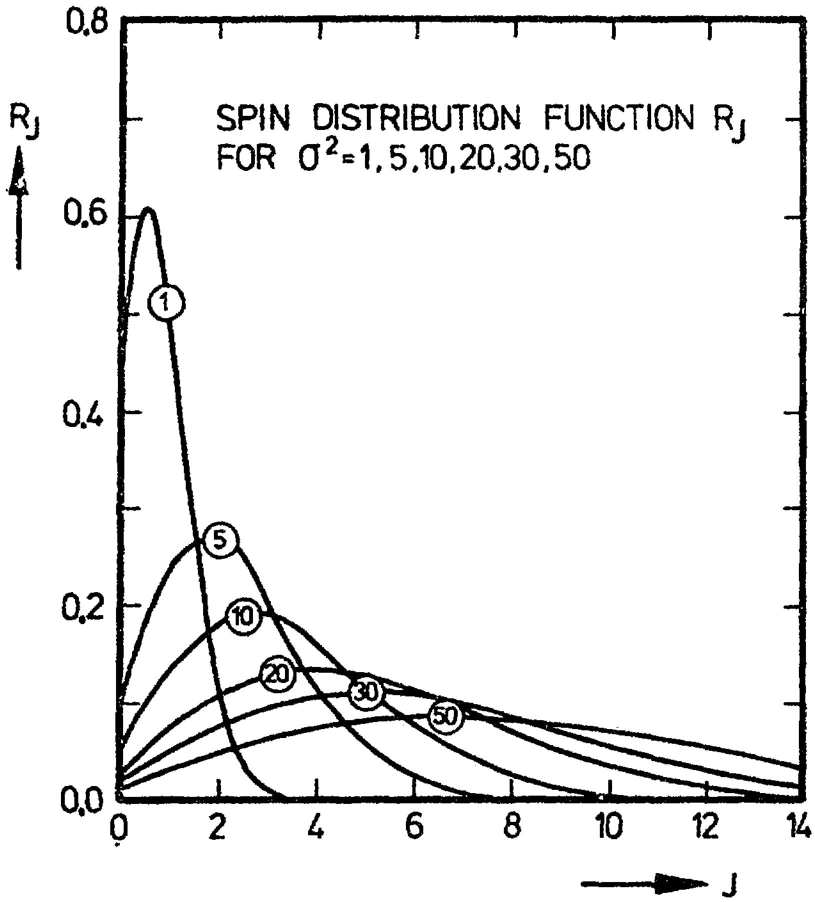


Fig. 7 (from [10])

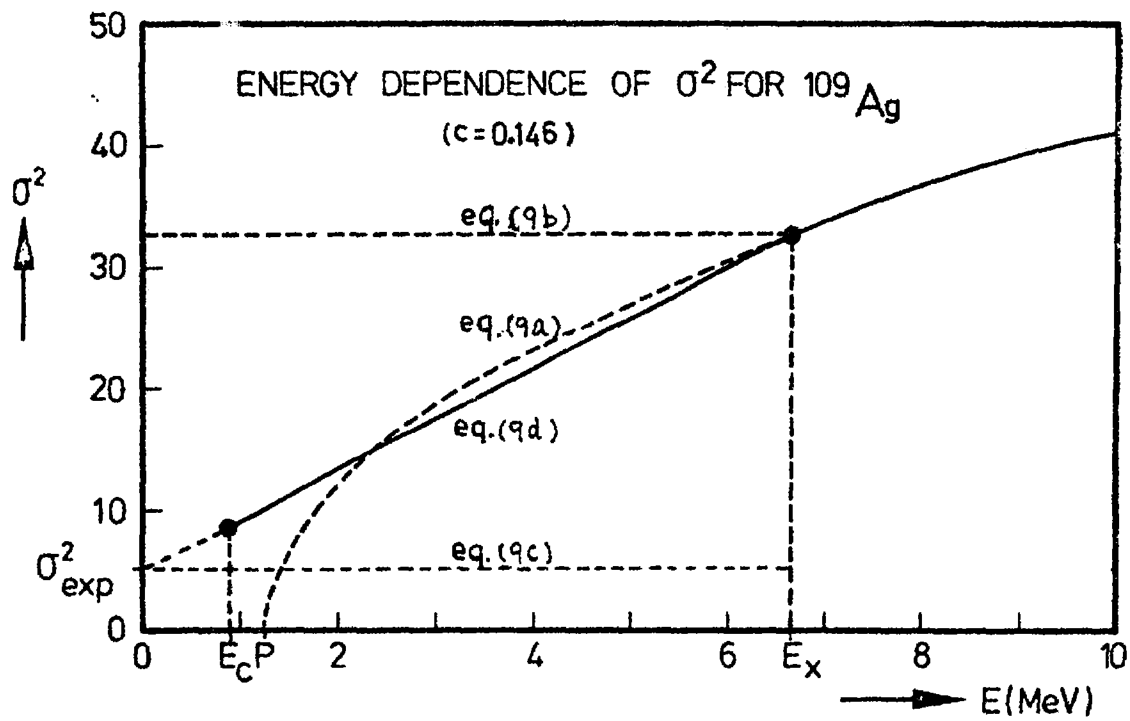
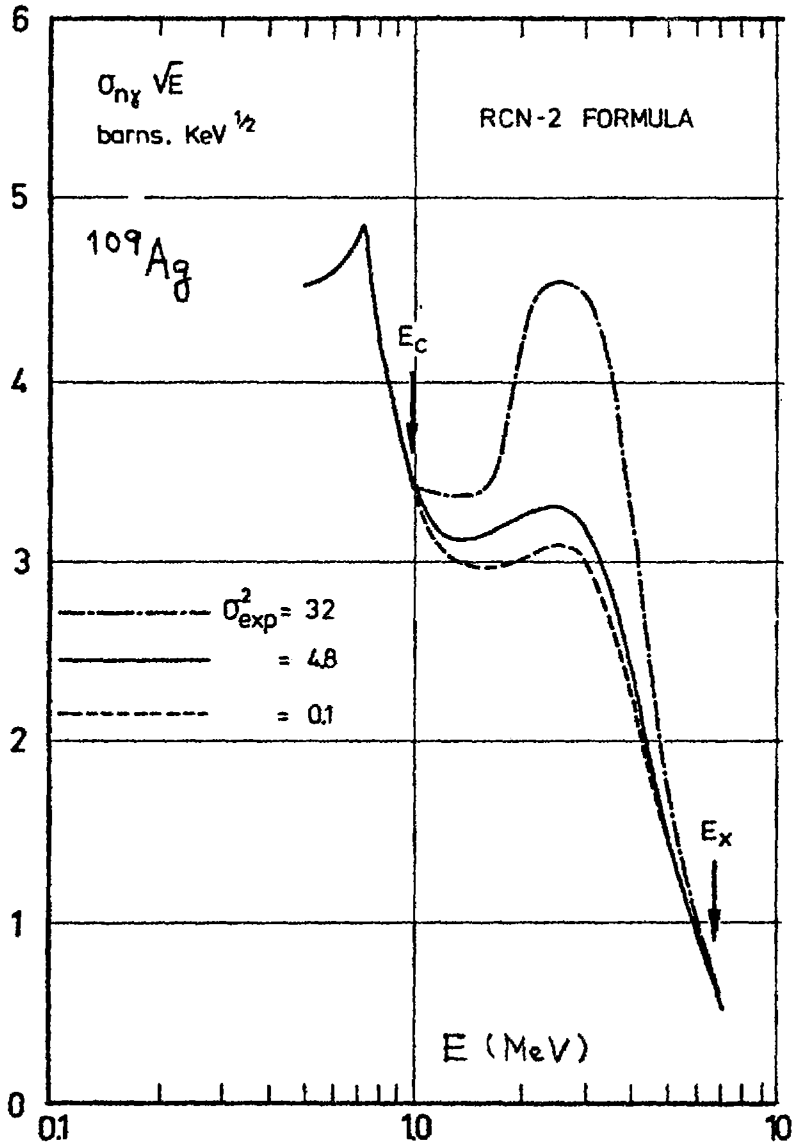


Fig. 8 (from [10])



$\sigma_{n\gamma}\sqrt{E}$
barns. KeV $^{1/2}$
RCN-2 FORMULA
^{109}Ag
E_c
E_x
σ^2_{exp} = 32
= 4.8
= 0.1
E (MeV)
6
5
4
3
2
1
0
0.1
1.0
10

Fig. 9 (from [10])

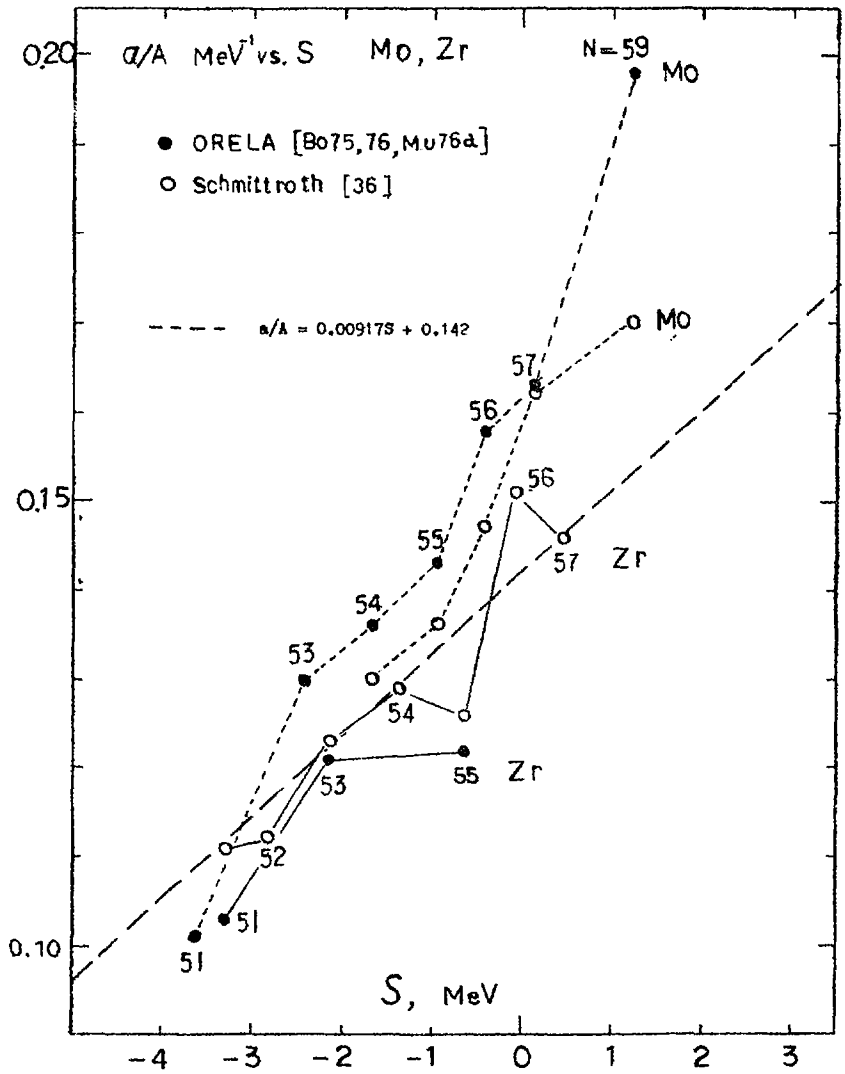

a/A MeV⁻¹ vs. S Mo, Zr
N=59
Mo
● ORELA [Bo75,76,Mu76a]
○ Schmittroth [36]
a/A = 0.00917S + 0.142
Mo
57
56
56
55
57 Zr
54
53
54
53
55 Zr
52
51
51
0.20
0.15
0.10
S, MeV
-4 -3 -2 -1 0 1 2 3

Fig. 10a

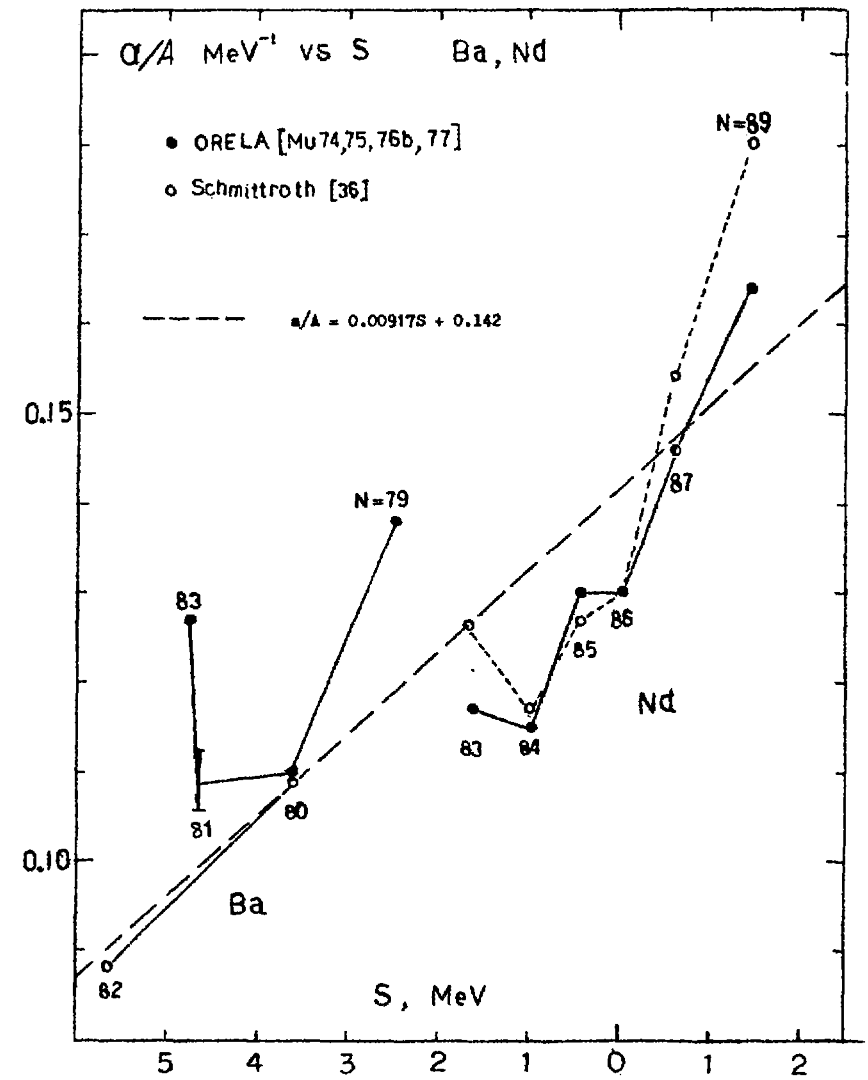


Fig. 10b

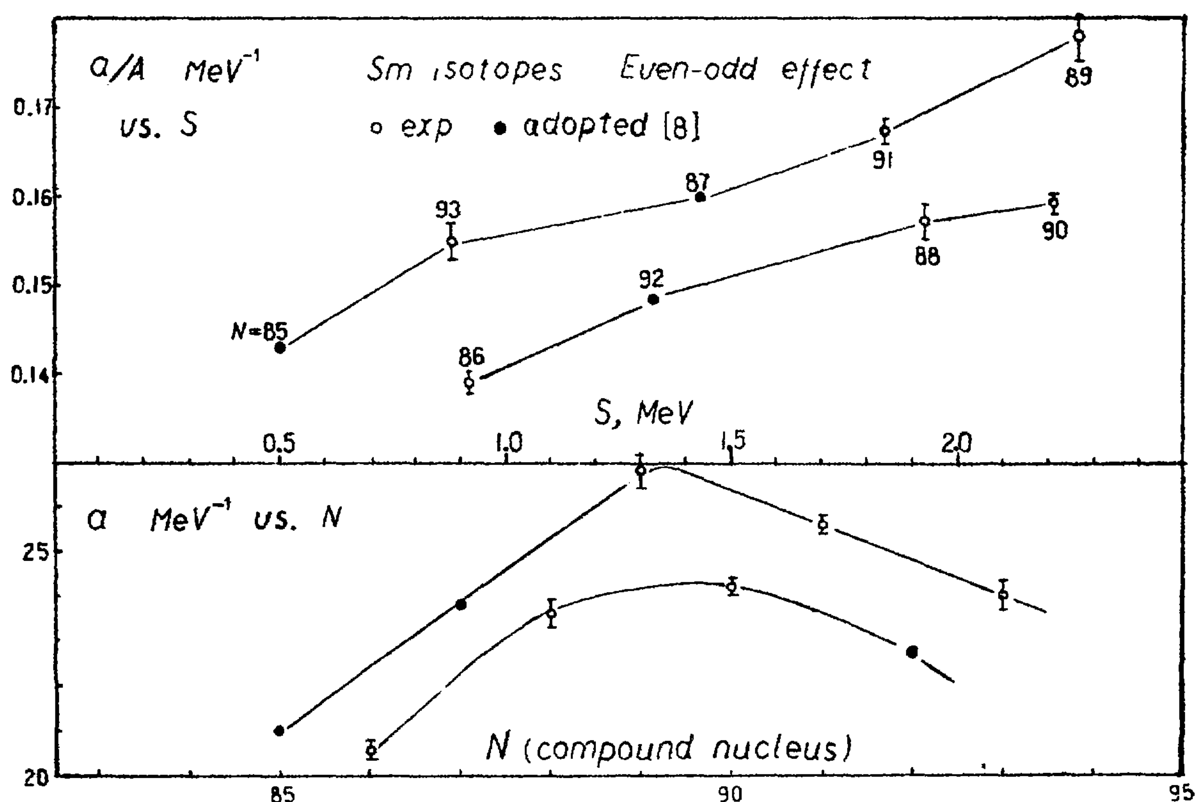


Fig. 10c

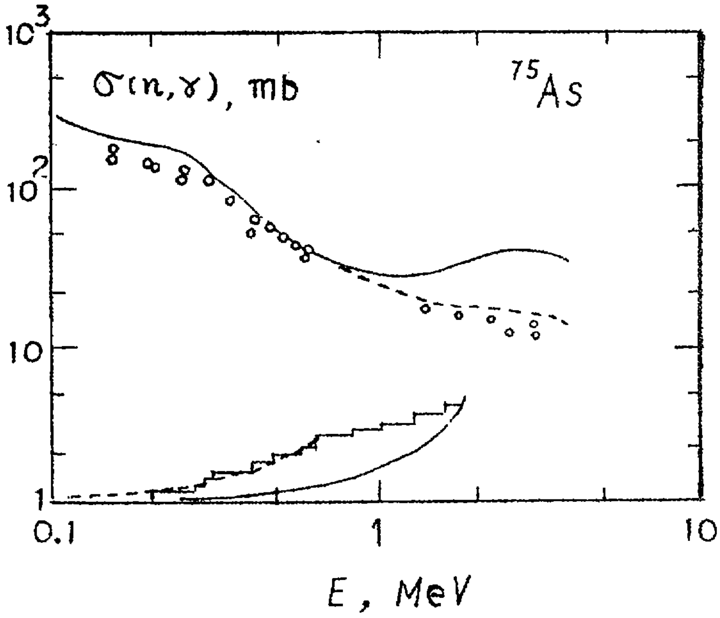


Fig. 11 (from /34/)
$^{75}As(n,\gamma)$ cross section:
Correlation between calculated capture cross section and target level density.

Staircase: nuclear levels;
——— : reference calc., i.e. level dens. determined by tot. number of levels;
----- : modified calc., i.e. level dens. determined so as to obtain overall fit to low-lying levels.

Review paper 10

STATUS OF FISSION PRODUCT YIELD DATA

by

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ABSTRACT

The topics covered in this paper are:-

(a) cumulative yields in thermal neutron fission and in fast fission up to 14 MeV incident neutron energy,

(b) dependence of the yields on incident neutron energy and spectrum,

(c) independent yields,

(d) charge dispersion and distribution, and

(e) yields of light particles from ternary fission.

The paper reviews information on these subjects for fission of actinides from ^{232}Th upwards with special emphasis on data published since the 1973 Bologna FPND Panel, compares data sets, and discusses the gaps still to be found in them.

INTRODUCTION

1. Purpose of the paper

The purpose of this paper is to review neutron fission yield data available at the end of March 1977, and thereby come to definite conclusions regarding the best yields to use today, and to point to areas of work where either more, or less, effort is needed.

At the first meeting of the IAEA Panel on FPND, held at Bologna in 1973(1), papers were presented which reviewed thermal (2) and fast (3) fission yields and my intention is to take these two papers as a starting point and confine this work mainly to developments since that time. Methods of evaluation, errors, methods of measurement of yields, etc, were all covered in these papers and will only be discussed now if there has been some significant new development.

2. Procedure used to achieve the purpose

The procedure used in this paper is to compare all current independent evaluations of yields and, from this comparison, make reasonable estimates of uncertainties still remaining in them. This comparison is supplemented by information obtained by searching the literature from the date of the latest evaluation, or taken from contributions made directly to me especially for this paper.

3. Topics covered

The topics covered are:-

a) cumulative yields in thermal neutron fission and in fast fission up to 14 MeV incident neutron energy,

b) dependence of the yields on incident neutron energy and spectrum,

c) independent yields,

d) charge dispersion and distribution, and

e) yields of light particles from ternary fission.

The terms "fast fission" and "14 MeV fission" were discussed in ref. 3 and have again been used as stated there.

4. Evaluations considered in this paper

Table 1 is a list of the evaluations considered, arranged so that the reader can see at a glance what kinds of evaluations are available for any particular fissioning system. There are several kinds of evaluated fission yields and I have split the table up to show the main ones. These are as follows:-

a) Unadjusted experimental chain yields (UC)

These are averaged and normalised chain yields for all experimental results accepted by the evaluator. Each mass chain is independent of every other one. These yields should be the best available values for particular mass numbers, provided there are sufficient high quality experimental data.

b) Adjusted experimental chain yields (AC)

These are the same as UC in that averaged normalised yields are used, but all the yields in a particular set (i.e. for one fissioning nuclide at one energy) are constrained to obey certain physical laws, e.g. to total to 200%. Where the experimental data for a particular mass chain are of inferior quality the adjusted value will probably be better than the unadjusted, but for mass chains with high quality experimental data the adjusted value may be somewhat less accurate. Notwithstanding this, however, adjusted yields should always be used for calculations which require a complete set of yields.

c) Calculated chain yields, (not directly based on experiment) (CC)

These are sets of chain yields based either solely on theory, or on a mixture of theory and parameters whose values are derived from experimental results from other sets of chain yields. They should only be used where good experimental sets (either UC or AC) are not available.

d) Unadjusted experimental independent yields (UI)

These are averaged and normalised independent yields for all accepted experimental results. Except in a limited number of cases they are usually inferior to calculated values.

e) Calculated independent yields (CI)

These are nearly always calculated from some simple hypothesis (e.g. the assumption of a gaussian yield distribution along a mass chain) with parameters obtained from measured data, although purely calculated results may also sometimes be found. They are to be preferred to experimental values in most cases.

CUMULATIVE FISSION YIELDS

5. Objects of this section

For most (although not all) purposes what the user requires is the total yield of a mass chain and this is fortunate since it is also the kind of yield most readily and frequently produced by the measurer. Cumulative yields, therefore, make up by far the greater part of this paper. I have attempted to provide answers to the following questions:-

a) What are the best values of the yields?

b) How do purely calculated yields fit into the picture?

c) Have the various evaluations moved closer to each other since 1973? Since the evaluators are all, presumably, using virtually the same experimental data input, this question relates to the quality of the evaluations (and evaluators!) themselves.

d) What errors are being reported by the main evaluators now and how do they compare with errors reported in 1973? This question relates primarily to the quality of the experimental data and should throw light on whether or not they are improving.

e) The overall conclusion in 1973 was that thermal yields were largely good enough but fast yields were not. Should we bother with fast yields at all, or are they close enough to thermal values for any difference to be swallowed up in the errors? If they are close enough, and if the fast yield errors are regarded as being tolerable, as were the thermal errors in 1973, then we are wasting our time in measuring and evaluating fast yields for thermally fissile fuel nuclides.

All the above questions are discussed with reference to the adjusted evaluated sets of yields available by the end of March 1977. The results therefore embrace at least 95% of all measurements made up to that date. In the final section of this part of the paper I refer to measurements which have not been included in evaluations, many of them sent to me as contributions especially for this panel meeting.

6. Best values for cumulative yields (adjusted evaluated data)

(a) Tables of yields

Tables 2 - 18 give thermal yield values for 4 fuel nuclides, fast (pile) values for 8 and 14 MeV values for 5. The general plan is the same for all the tables. The adjusted yield values from all "current" independent evaluations, i.e. those listed at the foot of Table 1, are given and a simple mean is calculated since there is no possible justification for weighting the yields. Where calculated yields exist from Sidebotham's paper[5] these are included in a separate column for reference, but they are NOT included in the mean. The letters heading the columns are the codes for the evaluators given at the foot of Table 1.

The adjustment methods used by the evaluators differ but all of them insist that the yields total to 200%. Similarly, the simple means also total to 200% and it is probably reasonable to assert that they represent the best evaluated values available, given the assumption made here that all the evaluators are equally competent.

(b) Mass-yield curves

The mean yields of Tables 2 - 18 are plotted, all on the same scale, in Figs. 1 - 17. Where applicable, Sidebotham's yields are plotted also.

7. Conclusions about calculated yields

From those cases where Sidebotham's calculated yields are plotted (Figs. 5, 6, 10, 11, 12) we are led to the same conclusions as in 1973, namely that they are unsatisfactory except where the measured data are totally inadequate. One such area is in fast (pile) fission yields of ^{240}Pu and ^{242}Pu; Koch[4] has compared his, as yet unevaluated, results for these nuclides irradiated in RAPSODIE with Sidebotham's calculated values and finds good agreement with the latter's thermal values but, in general, experimental and calculated yields are not very close. Koch's new results are further discussed in para 11 c) iii below.

8. Have the evaluations improved since 1973?

It seems reasonable to assert that if a number of people each independently evaluate a set of measurements and arrive at identical answers, then these ought to be the correct answers; at the very least, they must be the best that can be produced at that time. Bearing this in mind I have adopted as my criterion for measuring evaluation "quality" the spreads of values found for the yields, expressed as percentages of those mean values given in Tables 2 - 18. These spreads are plotted in Figs. 18 - 23, circles for 1973 and crosses for 1977, for six different fissioning systems.

The conclusion reached from a study of these plots is that, at least on this criterion, there has indeed been some improvement in evaluation BUT that it is confined largely to ^{235}U thermal yields and to the light peak yields of the other fissioning systems.

9. Errors in yields

It seems reasonable to think that if the general quality of yield measurements is improving then the errors quoted by the evaluators will become smaller, always assuming that the evaluators do not change their basis for estimating them. Tables 19 and 20 compare the errors reported by the two main evaluators, Meek and Rider[6][7] and Crouch Crouch[8] [9] [10] in 1973 and 1977. In Table 19 I have split up the yield curves into their five main parts and averaged the errors over the stated mass numbers, while in Table 20 I have picked out certain important yields. In both cases I have chosen the same six representative fissioning nuclides as for the yield spread discussion (para 8).

One major difference between the two evaluators immediately becomes evident; Crouch has, in most cases, increased his errors since 1973 while Meek and Rider have reduced theirs. The effect of these changes is that the errors of the two evaluations are now in reasonable agreement for the wings and valleys but in sharp disagreement on the peaks where

Meek and Rider show errors 2 - 3 times smaller than Crouch. This same disagreement is shown in the individual yield errors given in Table 20, except at mass 141.

It is difficult to know just what to make of this situation; the safest conclusion seems to be that there may have been some improvement in the quality of the yield measurements, but if so it has been masked by a change in the method of estimation of errors by the evaluators.

As to the question of what errors to use for the 1977 evaluated yields, the only reasonable course to take seems to be to assume that the two principal evaluators are equally competent and so to conclude that 1977 yield errors should be a mean of their estimates; values calculated in this way are given in Tables 19 and 20.

10. Is it worth measuring fast yields of thermally fissile nuclides?

The extra excitation given to the fissioning compound nucleus in a fast reactor environment is, on average, much less than 1 MeV and hence the variation in fission yields can not be very great except for rare fission modes on the wings and valley of the mass yield curve. The question which arises is whether or not thermal yields are accurate enough for practical use in nuclear power calculations, having regard to the errors in the evaluated yields and the accuracy required of them.

I have attempted to throw light on this problem in two ways. First, in Figs. 24-31, I have plotted the light and heavy peak yields, both fast and thermal for ^{233}U, ^{235}U, ^{239}Pu and ^{241}Pu. The yields are shown as histograms and for each one the upper and lower limit was calculated by taking the mean yields from Tables 2-18 and applying mean values of the 1977 errors (1σ) of Crouch and of Meek and Rider to them. We can immediately see that even though only 1σ errors have been used, the fast and thermal yields usually overlap.

To compare fast and thermal yields for the same four fissile nuclides over the whole mass range I have adopted a different treatment. In Figs. 32-35 the solid lines indicate the difference between fast and thermal yields, calculated from the simple mean figures of Tables 2-18. The broken lines are the sum of the 1σ fast and thermal errors, calculated using the same mean yields and averaging the 1σ errors (for each mass number separately) given in 1977 by Crouch and by Meek and Rider. Both yield differences and sums of errors are expressed as actual percentage fission yield and hence if the error line lies above the difference line the fast and thermal yields are statistically indistinguishable from each other. We can immediately see that, apart from rare fission modes (extreme wings and valleys) and for a few isolated yields on the peaks, thermal and fast yields are the same.

The conclusion to be drawn from these two series of calculations is that, with errors as they are today, we might just as well use thermal fission yields except in special cases. It is true that a different conclusion might emerge if sufficient effort was put into further fast yield measurements so as to reduce substantially the errors on these yields, but it is by no means certain that this would be so, and we should certainly ask whether or not the effort and expense required make it worth while. Such a conclusion cannot, of course, apply to those nuclides which are not thermally fissile.

11. New cumulative yield measurements

a) Purpose of this section

Between them, the two main evaluators[(7)(10)] have included all published (and some unpublished) yields up to the end of 1976. The purpose of this section is to bring to the attention of readers new data from the literature or sent to me for this paper, and to assess their likely effect on the questions discussed in paragraphs 5-10 above.

b) Thermal fission yield papers

(i) "Discrepancies and comments regarding ^{235}U and ^{239}Pu thermal fission yields and the use of ^{148}Nd as a burnup monitor". W.J. Maeck et al[(11)]

Although the data in this report have been included in both main evaluations[(7)(10)] in at least preliminary form, their importance is such that all users of fission yields, particularly for burnup purposes, should be aware of them. The authors report large changes in a number of their previously reported yield values, particularly in ^{239}Pu fission (14% higher for ^{138}Ba, for example) which are such that all their ^{239}Pu fission yields must be altered to preserve the mass balance. Many of the changes are far outside the accuracy they previously claimed and the authors are to be commended for bringing them to notice. As an example of the effect of these changes on evaluations, the current Meek and Rider yield for ^{138}Ba in ^{239}Pu thermal fission is ~ 6% higher than their 1973 value, this alteration being three times the old quoted error.

(ii) "Direct physical measurement of mass yields for ^{235}U thermal neutron fission". G. Diiorio and B.W. Wehring[(12)]

The yields in this paper have been included in the evaluations but the paper is mentioned here because they have been measured directly by the fission fragment recoil mass spectrometer HIAWATHA, the first reported measurements of this type by this machine. The authors claim that their results "give the most accurate mass yield data for thermal neutron fission of ^{235}U," apparently on the grounds that they are not subject to the same type of errors as those measured by other means. There is no evidence whatsoever that this claim is correct.

(iii) "Cumulative yields of rare earth elements in the thermal neutron induced fission of ^{249}Cf. H. Gäggeler and H.R. von Gunten[(13)]

Although as yet of no importance in nuclear power, yield measurements in ^{249}Cf fission add to our knowledge of fission yield systematics. Table 21 summarises these results.

(iv) "Fission product nuclear data obtained by use of an on-line mass spectrometer" P.L. Reeder, J.F. Wright, R.A. Anderl[(14)]

This paper gives ^{235}U thermal fission yields obtained using the SOLAR, on-line fission fragment mass spectrometer. The yields have been included in evaluations but are mentioned since they are another example of the comparatively new on-line technique for measuring cumulative yields.

c) Fast fission yields

(i) "Mass distribution in fission of five uranium isotopes (233, 234, 235, 236, 238) irradiated by fission spectrum nuetrons, by γ-spectrometry". A. Ferrieu, J. Blachot, A. Moussa[15]

The target nuclei were placed in the CARAMEL facility in the MELUSINE reactor at Grenoble; the neutrons are close to "fission spectrum". The results, which are included in Meek and Rider's evaluation but not in Crouch's, are given in Table 22. On the whole, these yields are in line with mean values given in this paper; however, ^{234}U yields have not yet been evaluated and these new measurements constitute a useful addition to yield data.

(ii) "Cumulative fission yields for fast neutron fission of ^{236}Np". J. Blachot, A. Ferrieu, G. Lhospice[16]

These yields have not yet been included in the evaluations and form a useful addition to the scanty data existing; they are given in Table 23.

(iii) "Cumulative fission yields for fast neutron fission of ^{240}Pu and ^{242}Pu". L. Koch[4]

These yields have not yet been included in the evaluations and should improve the existing ones. The irradiations were carried out in the fast reactor RAPSODIE during the TACO series of measurements; the results are given in Table 24. An interesting point about these new values is that they agree better with the "thermal" calculated yields of Sidebotham[5] than with his fast ones. Since both Crouch and Meek and Rider make heavy use of Sidebotham's "fast" yields in their evaluations, these new values should have a considerable effect on the latter when they are included. See also para 11 c) iv below.

(iv) "Fission of ^{240}Pu with neutrons having a fission spectrum energy distribution". W.A. Myers, M.V. Kantelo, A.L. Prindle, D.R. Nethaway[17]

Here we have another large set of ^{240}Pu fast yields which have not yet been included in the evaluations. The irradiations were carried out in the FLATTOP critical assembly at Los Alamos and the neutron characteristic is described as "fission spectrum"; the $\sigma_f(^{238}U)/\sigma_f(^{235}U)$ index for this assembly has a value of 0.172 and the spectrum is thus very hard. Table 25 gives the results as presented by the authors, and also compared with those of Koch[4], normalised to his value for mass 143. Taken together these two sets of yields should materially alter the ^{240}Pu evaluations.

d) 14 MeV fission yields

(i) "Fission of ^{240}Pu with 14.8 MeV neutrons" D.R. Nethaway, A.L. Prindle, W.A. Myers, W.C. Fuqua, M.V. Kantelo[18]

This is a substantial work in which 49 chain yields were measured giving, for the first time, a mass yield curve for 14 MeV fission of ^{240}Pu; the results are shown in Fig. 36.

(ii) "Experimental results on the mass distribution of ^{238}U fission induced by 14 MeV neutrons". S. Daróczy, S. Nagy, P. Raics, L. Kövér, I. Hamvas[19].

This paper, whose results have been included in Meek and Rider's evaluation, reports cumulative yields of 47 mass chains. It also gives a useful detailed account of the γ-spectrometric analytical procedures, claimed by the authors to be as accurate as mass-spectrometry.

(iii) "Fission of ^{237}Np by medium energy protons". P. Polak[20]

This paper discusses the systematics of fission yields at an excitation energy of 20 MeV, which corresponds roughly to 14 MeV neutron fission.

DEPENDENCE OF YIELDS ON INCIDENT NEUTRON ENERGY

12. General discussion

The question of whether the dependence of yields on energy has any practical importance for fast reactor work has been discussed in para 10, the conclusion arrived at being that, in general, it has not. This overall conclusion must, however, be modified by the following considerations:-

a) Some yields are required to a much higher accuracy than the general run and it may be necessary to make special measurements of these in different spectra.

b) Yields on the wings and valley of mass yield curves do change by a greater amount than the sum of the errors of fast and thermal.

c) "Fine structure" yields may be affected more than other peak fission yields.

d) Some important nuclides are not thermally fissile.

e) Once we start considering fission at energies above "fast (pile)" fission the variation with energy rapidly becomes significant.

f) The variation in shape of the mass yield curve with energy has fundamental importance.

The remainder of this section of the paper consists of a brief review of the evidence on this matter which has appeared since the Bologna Panel.

13. Monoenergetic neutron bombardment measurements

Four sets of measurements made since the 1973 Panel have been found:-

a) Flynn et al[21] have measured yields of over 40 mass chains in fission of ^{238}U by both 2 and 8 MeV neutrons. Their results are shown in Fig. 37.

b) Cuninghame and Willis[22] have reported measurements of absolute fission yields of ^{99}Mo, ^{111}Ag, ^{140}Ba, ^{147}Nd and ^{153}Sm at 6 different energies from 130 to 1700 KeV in ^{239}Pu fission; their results are shown in Fig. 38.

c) Kaiser[23] reports the yields of 6 mass chains in the 0.3 eV resonance of ^{239}Pu as follows:-

Mass Number	Chain Yield %
91	2.43 ± 0.04
111	0.28 ± 0.02
112	0.130 ± 0.011
113	0.076 ± 0.005
115	0.024
117	0.0181 ± 0.004

d) Edward(24) has measured yields of 18 mass chains in 2.95 MeV neutron fission of ^{238}U and of 46 in 14.8 MeV fission of ^{238}U. It has not proved possible to obtain a copy of his thesis in time for inclusion in this paper.

14. Integral measurements

Some new measurements bearing on the energy effect problem have been made in known fast neutron spectra since 1973:-

a) Larsen et al(25) measured yields of 5 nuclides in fission of ^{235}U and ^{239}Pu by both thermal and fast neutrons and of ^{238}U by fast. The irradiations were carried out in various positions in the ZPR-3 reactor so as to give different median neutron energies and also in a thermal reactor. Fig. 39 shows the effect of changing median neutron energy on the yields of two of the five nuclides in ^{235}U fission.

b) Debertin(26) has measured the ratio of fast yields, where ^{235}U was fissioned by the neutrons from a ^{252}Cf source (i.e. a very hard neutron spectrum), to thermal ^{235}U yields. His results are shown in Fig. 40. The continuous line in this figure shows the ratios from Meek and Rider's earlier evaluation(27).

15. Calculated mass-yield curves

One paper, by Cook et al(28) has appeared, in which they fit mass yield curves empirically by superposition of two pairs of asymmetric gaussian curves and a single symmetric gaussian; they use the procedure to compare yields at different energies. Unfortunately, the accuracy is only of the order of 20% and this is by no means good enough for the method to be used for fast reactor yields, although it may be helpful for predicting unknown yield curves.

16. Experimental points fitted to spectral indices

Work on correlating fission yields of nuclides on the wings or valley of mass yield curves with some suitable reactor spectral index continues and a paper may be presented at this panel by Maeck(29). An example of his earlier work(30) is given in Fig. 41. This clearly shows the effect of neutron energy on the yield ratio $^{150}Nd/^{143}Nd$, the majority of the change taking place in the ^{150}Nd yield.

The only other new work found is a paper by G. and M. Lammer(31). They plotted the measured yields in fission of ^{235}U, ^{238}U and ^{239}Pu of a number of nuclides against median neutron energy. Their results, which form a useful addition to the limited number of such correlations, are shown in Figs. 42, 43 and 44.

17. Summing up

The measurements and correlations reported above do represent an advance in our knowledge since 1973 and do reveal significant changes in yields with neutron energy in some cases. However, overall consideration of these data reinforces the statements made in paras 10 and 12; the variations in fission yields between typical fast reactor fission spectra and thermal are not large enough to be of practical importance except for extreme wing and valley yields, for yields required to high accuracy and for special purposes. For higher energy fission the differences start to become quite large and may need to be taken into account.

INDEPENDENT FISSION YIELDS

18. Charge distribution, charge dispersion and independent fission yields

It is not easy to know how best to write about these three closely related and interleaved topics. All evaluations of adjusted chain yields make use of sets of independent yields which have been calculated assuming some form of charge distribution and dispersion. To prepare these sets the evaluators have to use empirical parameters which are ultimately derived from experimental measurements.

I shall briefly discuss charge distribution and dispersion in the next part of this paper and will now only indicate where independent yield data are to be found and compare some of the current experimental and calculated results, with the intention of throwing light on the problem of the reliability of the sets calculated and used by the evaluators.

19. Evaluation and calculation of independent yields

Since the 1973 panel meeting several evaluations of experimental independent yields and several sets of calculated yields have appeared(7)(10)(32)(33). Table 26 shows the coverage of these papers.

20. Measurement of independent yields

Most of the latest published measurements are incorporated in the Crouch(10) and Meek and Rider(7) evaluations and so there is little point in repeating the data here, but I will mention a few of the latest and most interesting ones.

Since 1973 there has been a surge of publications reporting new measurements of independent yields made using fission fragment mass spectrometers. Most of the studies were on ^{235}U thermal fission(14)(34)(35)(36)(37)(39) but there was one with ^{239}Pu thermal fission(38). The very limited choice of fission reaction types reflects the fact that highly intense sources are at present required for these measurements. Because of this limited choice, the results are invaluable for investigating the systematics of independent yields and charge distribution but are inadequate as sources of data.

There have also been a few measurements made by what are essentially very fast ($\approx$ 1 sec for separation) nuclear chemical techniques. Examples are Brissot et al(40)(41) who have measured yields in thermal neutron fission of ^{239}Pu and ^{241}Pu and Rudolph et al(42) who measured some in thermal fission of ^{235}U.

Conventional radiochemical measurements have continued and would seem to be essential for the provision of data for the less common types of fission. Some examples are to be found in ref.(18)(^{240}Pu fast fission-see para 11 c)iv above), ref.(43)(^{233}U, ^{235}U, ^{239}Pu thermal fission), ref.(13)(^{249}Cf thermal fission) and ref.(44)(yields of ^{131}I, ^{132m}I, ^{132g}I, ^{133}I, ^{134m}I, ^{134g}I in thermal fission of ^{233}U, ^{235}U and ^{239}Pu).

21. A comparison of independent yields for ^{235}U thermal fission measured by on-line mass spectrometry with evaluated experimental yields and with calculated yields

The earliest mass spectrometric independent yields differed seriously from the existing radiochemical ones but this situation has now been resolved, at least for ^{235}U thermal fission, as we can see from Table 27. In this table some of the fractional independent yields from mass spectrometric measurements are compared with both experimental ones evaluated from all types of measurements and with the calculated yields (based on experimental parameters) used by the two main yield evaluators. We can see that agreement is now quite good, although one does have the impression that the errors quoted for the mass spectrometric yields may be on the optimistic side.

22. Conclusions about independent yields

The agreement now existing between esperimental and calculated yields for ^{235}U thermal fission encourages us to think that calculations of complete sets of independent yields are now satisfactory for this fissioning system. However, data for most other types of fission are too sparse for us to be certain that this is always the case. The calculations depend on the use of experimentally measured parameters (see next section on charge dispersion and distribution) and on extrapolations of these from types of fission where the experimental data are good to those where they may not necessarily be completely valid. What is needed is more measurements of independent yields in the less common types of fission to confirm that the parameters used for calculating sets of yields for these cases are correct.

CHARGE DISPERSION AND DISTRIBUTION

23. Importance for FPND

Charge dispersion, i.e. the distribution of independent yields along individual mass chains, and charge distribution, the way in which the protons are divided over the whole range of mass splits, have considerable fundamental interest for fission theory. However, from the point of view of their importance for FPND, the interest is mainly practical in that an understanding of these matters allows us to calculate independent yields.

24. Calculation of independent yields

a) General comments

To calculate fractional independent yields it is necessary to know Z_p, the most probable charge for a particular mass chain, and σ, the charge dispersion parameter which defines the shape of the gaussian distribution of the yields along the mass chain concerned. Because there are more experimental data for ^{235}U thermal neutron fission than for any other, it is usual to start with Z_p and σ values for this system and modify them for others as required. With these two parameters established, fractional independent or fractional cumulative yields are calculated and are then corrected for nuclear structure effects.

b) Calculation of Z_p values

Z_p values for ^{235}U thermal neutron fission are calculated by means of some formula of the type:-

$$Zp = A' \left[\frac{Z_f}{A_f}\right] + [\text{correction factor}]$$

A' is the pre-neutron emission mass of the fission product concerned and is obtained from:-

$$A' = A + \bar{\nu}$$

where A is the post-neutron fission product mass and $\bar{\nu}$ the average number of prompt fission neutrons emitted at that mass. Z_f and A_f are the charge and mass of the fissioning compound nucleus, while the correction factor shows the variation from the Unchanged Charge Distribution prescription for Zp. $\bar{\nu}$ and the correction factor both depend on experimental measurements.

Having established a set of values of Zp for ^{235}U thermal fission, sets are calculated for other systems. Since 1973 one new prescription for doing this has been published (by Nethaway[45][46]). A useful summary has been written by Wolfsberg[32].

c) Estimation of σ

σ is estimated from charge dispersion measurements and it now seems to be accepted that a value of 0.560 is satisfactory for all of the kinds of fission dealt with in this report; we should recognise, however, that this acceptance is not firmly based on practical measurements of σ except in the case of ^{235}U thermal fission.

d) Correction for nuclear structure effects (pairing)

At the time of the Bologna meeting it was just becoming accepted that there was an odd-even effect of Z. Several papers have appeared on this topic since then, e.g. by Amiel and Feldstein[33][47] and by Madland and England[48]; an odd-even N effect has also been established by these authors. The magnitude of the corrections can be seen in Table 28 which gives the Z and N effect corrections used in ref.[7]. By appropriate summing of these factors the corrections for the four possible pairing situations, i.e. odd Z-odd N, odd Z-even N, even Z-odd N and even Z-even N, are obtained.

25. Recommendations for future work

Conclusions in this section are inseparable from those on independent yields (para 22) and may be summarised by reiterating that what is needed is work directed to the establishment of better values for Zp, σ, the pairing corrections, and $\bar{\nu}$ as a function of mass. Such needs clearly require the continued accurate measurement of independent yields an of $\bar{\nu}$ versus mass particularly in those fissioning systems where the data are at present scanty.

YIELDS OF LIGHT PARTICLES FROM TERNARY FISSION

26. Summary of progress since 1973

There has been very little work reported on ternary fission since 1973 which is germane to the problems of fission product nuclear data. Wagemans and de Ruytter[49] have made measurements of light particle emission in some resonances of ^{239}Pu fission; Bischof et al[50] have carried out a similar study for ^{235}U fission; Mirell[51] has looked at ^{252}Cf spontaneous fission; Pik-Pichak[52][53] has written two papers on the theory of α-emission in fission. No papers have been found dealing with fission into three fragments of comparable size.

The only known measurements which have a direct bearing on FPND are those of Crouch[54]. His work is primarily directed towards measurements of ^{3}H is fast reactor fuels for waste disposal and other purposes. As a preliminary, to test his methods, he has re-measured the ^{3}H/fission ratio for ^{235}U thermal fission and finds a mean value for three measurements to be 0.92/10,000 fissions (errors not yet established). This compares with earlier values ranging from 0.5 to 1.08 (mean 0.93 ± 0.13) reported in ref.[3].

27. Recommendations on ternary fission

The data base on ternary fission does not seem to have expanded much since 1973. It seems desirable for more measurements to be made for, at least, ^{3}H yields and for a wider range of both fissioning nuclei and reactor neutron spectra. Some such measurements are in progress at Harwell.

SUMMARY OF CONCLUSIONS AND RECOMMENDATIONS

28. The following is a summary of the main points brought out in the text of this paper; the figures in square brackets refer to the relevant paragraph number:-

a) The simple means given in Tables 2-18 and plotted in Figs. 1-17 represent the best adjusted evaluated sets available today [6a].

b) The calculated yields of Sidebotham are unsatisfactory, except where the measured data are very sparse [7].

c) Because of a shortage of experimental data, evaluated yields for fast fission of ^{240}Pu, ^{241}Pu and ^{242}Pu rely heavily on Sidebotham's calculated values and are therefore in a different category from all other sets. [Notes to Tables 11,12,13]. However new experimental data should improve this actuation [11 c iii, 11 c iv].

d) Since 1973 the spread of evaluated yield values has decreased for ^{235}U thermal yields and for light peak yields for other fissile nuclides. This suggests an improvement in evaluation reliability for these cases.

e) The two chief evaluators are, in the main, in disagreement on yield errors; Crouch has usually increased his since 1973 while Meek and Rider have reduced theirs. The result is that Crouch's are, on the whole, about twice as large as Meek and Rider's. I suggest a mean between these two sets of errors be taken [9].

f) This disagreement between evaluators masks any general improvement in yield measurements which might have been inferred if the evaluators had agreed on a general reduction in errors [9].

g) Within the envelope of 1σ errors (taken as the average of the two main evaluators) thermal and fast fission yields are indistinguishable except in the case of the extreme wings and valleys of mass yield curves. This suggests that fast yield measurements should be concentrated on rare fission modes, on special yields where high accuracy is required, on experiments to establish systematic relationships between yield and energy and on non-thermally fissile nuclides [10].

h) The values of some mass spectrometric yields for ^{235}U and ^{239}Pu reported by one measurer have been changed by amounts far in excess of the quoted errors; this has had a marked effect on the evaluations and suggests that more attention should be given to possible systematic errors before very small errors are quoted [11 b i] .

i) A considerable number of new measurements have yet to be included in evaluations; those for ^{240}Pu and ^{242}Pu are particularly welcome [11c iii, 11 c iv, 11 d i].

j) Evidence is presented which shows a clear dependence of some yields on neutron energy, but the differences are only important in exceptional cases and for high energy fission [17].

k) There is now quite good agreement between calculated and measured independent yields for ^{235}U thermal fission which encourages us to believe that calculated sets for other types of fission may also be satisfactory. It is, however, important to obtain more high quality data to confirm this [21, 22].

l) The most important development on charge dispersion since 1973 has been experimental /confirmation of both proton and neutron pairing effects [24d].

m) To ensure that calculated independent yield sets are reliable, more measurements of independent yields and of $\bar{\nu}$ as a function of mass are required, particularly for fissioning systems other than ^{235}U thermal [25].

n) Relevant progress in ternary fission has been negligible, but a programme of ^{3}H measurements on fast reactor fuels is under way [26].

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TABLES

ILLUSTRATIONS

TABLE 1

Summary of evaluations considered in this paper

Type of yield / Type of fission	Unadjusted experimental chain UC	Adjusted experimental chain AC	Calculated chain CC	Unadjusted experimental independent UI	Calculated Independent CI
[Letters in these columns below refer to the reference letters of the evaluations, given at the foot of the table]					
SPONTANEOUS					
^{252}Cf	F	F		F	F
THERMAL					
^{227}Th	E				
^{229}Th	E				
^{233}U	E,F	B,C,E,F		E,F	E,F
^{235}U	E,F	B,C,E,F		E,F	E,F
^{237}U			A		
^{237}Np			A		
^{239}Pu	E,F	B,C,E,F		E,F	E,F
^{240}Pu			A		
^{241}Pu	E,F	B, E,F		F	E,F
^{242}Pu			A		
^{241}Am	E		A		
^{242}Am	E				
^{245}Cm	E				
^{249}Cm	E				
^{251}Cf	E				
^{254}Es	E				
^{255}Fm	E				
"FAST"(PILE)					
^{227}Ac	E				
^{231}Pa	E				
^{232}Th	E,F	C,E,F	A	F	E,F
^{233}U	E,F	E,F	A	F	E,F
^{234}U			A		
^{235}U	E,F	E,F		E,F	E,F
^{236}U	F	F	A	F	F
^{237}U	E		A		
^{238}U	E,F	E,F		E,F	E,F
^{237}Np	F	F	A	F	F
^{238}Np			A		
^{238}Pu			A		
^{239}Pu	E,F	E,F		F	E,F
^{240}Pu	F	(E)*(F)*	A	F	E,F
^{241}Pu	F	(E)*(F)*	A	F	E,F
^{242}Pu	F	(F)*	A	F	F
^{241}Am			A		
^{243}Am			A		
^{242}Cm			A		
"FAST"(MISC)					
1.1 MeV ^{237}Np	E				
3 MeV ^{231}Pa	E				
^{232}Th	E				
^{238}U	E				
8 MeV ^{232}Th	E				
11 MeV ^{232}Th	E				
"14" MeV					
^{231}Pa	E				
^{232}Th	E,F	E,F		E,F	E,F
^{233}U	E,F	E,F		E,F	E,F
^{235}U	E,F	E,F		E,F	E,F
^{238}U	D,E,F	E,F		E,F	E,F
^{237}Np	E				
^{239}Pu	E,F	F		E,F	F

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ENDF B V }

*Note: The adjusted chain yields for fast (pile) fission of ^{240}Pu and ^{241}Pu produced by Crouch take Sidebotham's calculated yields as their starting point. The process of adjustment does cause some changes in value, but these yields cannot be counted as being a true evaluation

Meek and Rider also rely heavily on Sidebotham's calculated data for ^{240}Pu, ^{241}Pu and ^{242}Pu fast (pile) fission yields, but they have also folded in such experimental data as do exist and so their yields can be regarded as being some way towards a true evaluation.

TABLE 2

Adjusted chain fission yields for ^{235}U thermal fission

Mass No.	Fission yield % B	C	E	F	Simple mean	Mass No.	Fission yield % B	C	E	F	Simple mean	Mass No.	Fission yield % B	C	E	F	Simple mean
70)		-	-	.0000392	.0000392	109	.047	.04	.0504	.0446	.0453	148	1.30	1.30	1.28	1.28	1.29
71)		-	-	.000174	.000174	110	(.029)	(.03)	.0310	.0261	.0290	149	.766	.76	.769	.766	.765
72)		(.0001)	-	.000503	.000301	111	.023	.021	.0198	.0191	.0207	150	.508	.501	.496	.502	.502
73)	(.01)	(.00033)	-	.00110	.000715	112	.015	.015	.0131	.0145	.0144	151	.314	.32	.348	.322	.326
74)		(.0009)	.00087	.00274	.00151	113	(.015)	(.015)	.0130	.0137	.0142	152	.213	.22	.196	.208	.209
75)		(.0023)	.0022	.00822	.00424	114	(.014)	(.020)	.0152	.0134	.0151	153	.105	.107	.104	.105	.105
76)		(.0068)	.0069	.0146	.00943	115	.017	.021	.0182	.0123	.0171	154	.0456	.0449	.0463	.0457	.0456
77	(.02)	.02	.0193	.0260	.0213	116	(.014)	(.021)	.0142	.0147	.0160	155	(.023)	.026	.0230	.0222	.0235
78	(.06)	(.04)	.0668	.0610	.0569	117	.014	.022	.0150	.0122	.0158	156	.0116	.012	.0112	.0113	.0115
79	(.16)	(.09)	.134	.150	.133	118	.0145	.022	.0155	.0128	.0162	157	.0065	.0072	.00669	.00638	.00669
80	(.26)	(.18)	.266	.237	.236	119	.015	.023	.0158	.0132	.0167	158)		(.0024)	.00230	.00237	.00236
81	.34	.33	.330	.381	.345	120	.016	.025	.0175	.0145	.0182	159)	(.004)	.00091	.000951	.000883	.000915
82	(.60)	(.60)	.683	.545	.607	121	.018	(.027)	.0193	.0153	.0199	160)		(.00035)	.000300	.000294	.000315
83	1.00	1.02	1.01	1.02	1.01	122	(.025)	.030	.0195	.0157	.0225						
84	1.66	1.69	1.76	1.71	1.71	123	(.037)	(.038)	.0204	.0202	.0289						
85	2.18	2.19	2.27	2.20	2.21	124	(.060)	.050	.0322	.0259	.0420						
86	2.80	2.86	2.93	2.87	2.86	125	.116	.110	.116	.114	.114						
87	3.98	4.01	4.08	4.04	4.03	126	(.26)	(.18)	.247	.252	.235						
88	5.53	5.54	5.45	5.52	5.51	127	.62	(.50)	.512	.570	.550						
89	6.33	6.41	6.12	6.29	6.29	128	(1.0)	(1.00)	.902	.781	.921						
90	6.81	6.88	6.46	6.85	6.75	129	(1.7)	(1.56)	1.14	1.66	1.51						
91	6.49	6.52	6.41	6.56	6.49	130	(2.5)	(2.40)	2.35	1.90	2.29						
92	6.67	6.65	6.63	6.62	6.64	131	3.53	3.54	3.48	3.64	3.55						
93	7.05	7.04	7.02	6.87	6.99	132	4.82	4.84	4.80	4.96	4.85						
94	6.75	6.81	6.80	6.84	6.80	133	5.99	6.03	5.96	6.04	6.00						
95	6.19	6.21	6.22	6.22	6.21	134	6.14	6.15	6.17	6.29	6.19						
96	5.66	5.73	5.77	5.68	5.71	135	6.21	6.27	6.01	6.25	6.18						
97	5.36	5.39	5.57	5.48	5.45	136	6.88	6.82	7.54	7.06	7.07						
98	5.10	5.14	5.20	5.18	5.08	137	6.76	6.85	6.64	6.82	6.77						
99	5.01	4.89	5.10	4.88	4.97	138	5.84	6.00	5.89	5.93	5.91						
100	4.36	4.38	4.46	4.43	4.41	139	6.41	6.34	6.45	6.37	6.39						
101	3.21	3.19	3.16	3.24	3.20	140	6.39	6.45	6.35	6.46	6.41						
102	2.44	2.42	2.37	2.46	2.42	141	6.62	6.56	6.74	6.58	6.62						
103	1.80	1.60	1.71	1.65	1.69	142	6.60	6.61	6.74	6.68	6.66						
104	1.03	1.02	1.02	1.03	1.02	143	5.85	5.88	5.92	5.90	5.89						
105	.53	(.54)	.549	.483	.525	144	4.62	4.64	4.58	4.65	4.62						
106	.253	.255	.245	.260	.253	145	3.38	3.39	3.38	3.41	3.39						
107	(.13)	(.12)	.145	.117	.128	146	2.55	2.53	2.51	2.55	2.53						
108	(.070)	(.065)	.0840	.0632	.0705	147	1.70	1.80	1.87	1.76	1.78						

NB. Yields in brackets are estimated by the evaluator concerned.

TABLE 3

Adjusted chain fission yields for ^{235}U thermal fission

Mass No.	Fission yield %				Simple mean	Mass No.	Fission yield %				Simple mean	Mass No.	Fission yield %				Simple mean
	B	C	E	F			B	C	E	F			B	C	E	F	
74)		.00035	.000340	.000338	.000344	113	(.012)	.004	.00981	.0117	.00938	152	.268	.262	.266	.268	.266
75)	(.005)	(.001)	.00110	.00109	.00106	114	(.0110)	.012	.0101	.0111	.0110	153	.167	.163	.159	.161	.162
76)		(.003)	.00310	.00345	.00318	115	.0104	.011	.0103	.0105	.0105	154	.0743	.072	.0753	.0737	.0738
77	.0083	.008	.00772	.00813	.00804	116	(.0105)	.011	.0102	.0114	.0108	155	.0321	.032	.0324	.0320	.0321
78	.020	.02	.0205	.0176	.0195	117	(.0105)	.011	.0102	.0111	.0107	156	.013	.014	.0127	.0132	.0132
79	.056	.056	.0431	.0450	.0500	118	(.0105)	.011	.0106	.00983	.0105	157	.0061	.0062	.00631	.00613	.00618
80	(.12)	(.11)	.0953	.125	.113	119	(.0105)	.012	.0112	.0102	.0110	158	.003	.0031	.00282	.00286	.00294
81	.20	.22	.209	.208	.209	120	(.0110)	.013	.0118	.0106	.0116	159	.001	.00105	.00106	.00100	.00103
82	(.33)	.35	.334	.384	.349	121	.013	.014	.0120	.0134	.0131	160	–	(.00035)	.000330	.000306	.000329
83	.535	.532	.530	.537	.533	122	(.013)	.015	.0140	.0121	.0135						
84	.986	1.000	.993	1.000	.995	123	.016	.0164	.0158	.0158	.0160						
85	1.33	1.33	1.32	1.31	1.32	124	(.020)	.024	.0210	.0190	.0210						
86	1.96	1.97	1.95	1.96	1.96	125	.029	.028	.0287	.0291	.0287						
87	2.53	2.56	2.54	2.55	2.54	126	(.053)	.063	.0583	.0550	.0573						
88	3.59	3.62	3.60	3.64	3.61	127	.124	.11	.115	.125	.118						
89	4.74	4.84	4.76	4.81	4.79	128	.34	.36	.286	.344	.332						
90	5.82	5.91	5.85	5.92	5.87	129	.88	.64	.612	.695	.707						
91	5.95	5.93	5.91	5.93	5.93	130	1.7	2.00	1.57	1.73	1.75						
92	5.98	5.98	5.99	5.98	5.98	131	2.80	2.82	2.78	2.89	2.82						
93	6.41	6.39	6.38	6.38	6.39	132	4.17	4.20	4.29	4.29	4.24						
94	6.43	6.45	6.44	6.44	6.44	133	6.79	6.73	6.75	6.70	6.74						
95	6.53	6.54	6.51	6.50	6.52	134	7.61	7.67	7.74	7.81	7.71						
96	6.30	6.29	6.27	6.27	6.28	135	6.60	6.55	6.58	6.56	6.57						
97	6.07	6.00	6.03	5.94	6.01	136	6.13	6.18	6.33	6.23	6.22						
98	5.81	5.81	5.81	5.80	5.81	137	6.24	6.26	6.22	6.23	6.24						
99	6.14	6.11	6.14	6.15	6.14	138	6.76	6.82	6.91	6.77	6.81						
100	6.31	6.32	6.34	6.22	6.30	139	6.55	6.55	6.65	6.42	6.54						
101	5.07	5.05	5.12	5.06	5.07	140	6.36	6.37	6.31	6.31	6.34						
102	4.19	4.19	4.20	4.21	4.20	141	5.87	5.85	5.81	5.84	5.84						
103	3.05	2.95	3.15	3.03	3.04	142	5.96	5.91	5.91	5.90	5.92						
104	1.83	1.83	1.85	1.83	1.83	143	5.95	5.92	5.94	5.95	5.94						
105	.95	.90	.983	1.02	.963	144	5.43	5.44	5.38	5.49	5.43						
106	.39	.387	.394	.392	.391	145	3.93	3.91	3.92	3.93	3.92						
107	.16	(.17)	.106	.186	.155	146	2.98	2.96	2.96	2.97	2.97						
108	(.070)	(.057)	.0633	.0706	.0652	147	2.26	2.22	2.31	2.24	2.26						
109	.030	.024	.0296	.0275	.0278	148	1.68	1.67	1.67	1.67	1.67						
110	(.022)	(.017)	.0235	.0221	.0211	149	1.08	1.05	1.08	1.08	1.07						
111	.018	.014	.0193	.0188	.0175	150	.652	.644	.645	.649	.647						
112	(.014)	.010	.00958	.0118	.0113	151	.419	.407	.424	.417	.417						

NB. Yields in brackets are estimated by the evaluator concerned.

TABLE 4

Adjusted chain fission yields for ^{239}Pu thermal fission

Mass No.	Fission yield % B	C	E	F	Simple mean
75)		(.0016)	[illegible]	[illegible]	[illegible]
76)	(.004)	(.0035)	[illegible]	[illegible]	[illegible]
77	.0073	.0075	[illegible]	[illegible]	[illegible]
78	.025	.028	[illegible]	[illegible]	[illegible]
79	(.050)	(.06)	[illegible]	[illegible]	[illegible]
80	(.12)	(.11)	[illegible]	[illegible]	[illegible]
81	.18	.186	[illegible]	[illegible]	[illegible]
82	(.22)	(.24)	[illegible]	[illegible]	[illegible]
83	.295	.298	[illegible]	[illegible]	[illegible]
84	.477	.482	[illegible]	[illegible]	[illegible]
85	.558	.566	[illegible]	[illegible]	[illegible]
86	.758	.764	[illegible]	[illegible]	[illegible]
87	.970	[illegible]	[illegible]	[illegible]	[illegible]
88	1.37	1.38	[illegible]	[illegible]	[illegible]
89	1.74	1.74	[illegible]	[illegible]	[illegible]
90	2.11	2.13	[illegible]	[illegible]	[illegible]
91	2.54	2.53	[illegible]	[illegible]	[illegible]
92	3.06	[illegible]	[illegible]	[illegible]	[illegible]
93	3.92	[illegible]	[illegible]	[illegible]	[illegible]
94	4.45	[illegible]	[illegible]	[illegible]	[illegible]
95	4.98	[illegible]	[illegible]	[illegible]	[illegible]
96	5.12	[illegible]	[illegible]	[illegible]	[illegible]
97	5.58	[illegible]	[illegible]	[illegible]	[illegible]
98	5.81	[illegible]	[illegible]	[illegible]	[illegible]
99	6.10	[illegible]	[illegible]	[illegible]	[illegible]
100	7.00	[illegible]	[illegible]	[illegible]	[illegible]
101	6.04	6.01	[illegible]	[illegible]	[illegible]
102	6.15	6.09	6.23	[illegible]	[illegible]
103	5.94	5.86	6.79	6.94	6.38
104	6.10	6.03	6.18	5.91	6.05
105	5.47	5.47	5.54	5.36	5.46
106	4.45	4.64	4.29	4.24	4.40
107	(3.5)	(3.3)	2.91	3.37	3.27
108	2.3	(2.0)	1.90	2.18	2.09
109	1.3	1.13	1.10	1.65	1.29
110	(.65)	(.57)	.540	.601	.590
111	.28	.27	.267	.308	.281
112	.11	.11	.104	.110	.108
113	.076	.065	.0770	.0775	.0739

Mass No.	Fission yield % B	C	E	F	Simple mean
114	(.049)	(.041)	.0540	.0452	.0473
[illegible]	[illegible]	.036	.0377	.0374	.0373
[illegible]	[illegible]	[illegible]	[illegible]	.0294	.0346
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0353
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0356
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0362
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0368
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0397
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0431
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0525
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.0678
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.108
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.211
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.499
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	.868
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	1.77
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	2.48
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	3.77
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.25
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	6.95
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	7.44
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	7.43
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	6.71
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	6.65
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.76
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.56
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.56
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	5.27
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	4.98
143	4.53	[illegible]	[illegible]	[illegible]	4.47
144	3.81	3.76	3.82	3.73	3.78
145	3.06	3.04	3.12	2.98	3.05
146	2.53	2.49	2.53	2.45	2.50
147	2.16	2.09	2.11	2.01	2.09
148	1.69	1.68	1.71	1.63	1.68
149	1.30	1.24	1.28	1.22	1.26
150	.989	.97	.991	.961	.978
151	.814	.76	.801	.760	.784
152	.619	.58	.593	.574	.591

Mass No.	Fission yield % B	C	E	F	Simple mean
153	.38	.44	.395	.357	.393
154	.286	.273	.278	.267	.276
155	.17	.17	.218	.160	.179
156	.120	.12	.112	.117	.117
157	.076	.080	.0760	.0728	.0762
158	.041	(.045)	.0400	.0381	.0410
159	.021	.022	.0212	.0202	.0211
160)		(.011)	.00850	.00877	.00942
161)		.0051	.00460	.00476	.00482
162)	(.015)	(.0025)	-	.00216	.00233
163)		-	-	.000878	.000878
164)		-	-	.000332	.000332
165)		-	-	.000129	.000129

N.B. Yields in brackets are estimated by the evaluator concerned.

TABLE 5

Adjusted chain fission yields for ^{241}Pu thermal fission

Mass No.	Fission Yield % B	Fission Yield % E	Fission Yield % F	Simple Mean	Mass No.	Fission Yield % B	Fission Yield % E	Fission Yield % F	Simple Mean	Mass No.	Fission Yield % B	Fission Yield % E	Fission Yield % F	Simple Mean
78	.0082	.00910	.00771	.00834	117	(.026)	.0260	.0256	.0259	156	.167	.165	.173	.168
79	(.016)	.0150	.0163	.0158	118	(.025)	.0250	.0241	.0247	157	.130	.131	.134	.132
80	(.033)	.0280	.0326	.0312	119	(.025)	.0250	.0243	.0248	158	(.090)	.0800	.0849	.0850
81	(.065)	.0640	.0632	.0641	120	(.025)	.0240	.0235	.0242	159	.046	.0463	.0476	.0466
82	(.120)	.150	.113	.128	121	(.025)	.0240	.0230	.0240	160	(.020)	.0180	.0189	.0190
83	.202	.202	.210	.205	122	(.025)	.0240	.0229	.0240	161		.00816	.00839	.00827
84	.360	.353	.365	.359	123	(.027)	.0250	.0244	.0255					
85	.392	.389	.405	.395	124	(.031)	.0280	.0284	.0291					
86	.608	.601	.631	.613	125	.042	.0416	.0418	.0418					
87	.750	.741	.764	.752	126	(.080)	.110	.0756	.0885					
88	.966	.955	1.00	.974	127	(.17)	.238	.230	.213					
89	(1.20)	1.21	1.19	1.20	128	(.37)	.472	.349	.397					
90	1.55	1.49	1.56	1.53	129	(.80)	.937	.768	.835					
91	1.84	1.79	1.87	1.83	130	(1.70)	2.50	1.64	1.95					
92	2.26	2.23	2.34	2.28	131	3.12	3.20	2.98	3.10					
93	2.93	2.90	3.05	2.96	132	4.64	4.58	4.42	4.55					
94	3.37	3.34	3.50	3.40	133	6.72	6.55	6.76	6.68					
95	3.98	4.02	4.08	4.03	134	8.08	8.03	7.73	7.95					
96	4.39	4.34	4.55	4.43	135	7.06	7.47	7.23	7.25					
97	4.73	4.85	4.84	4.81	136	7.30	7.13	6.93	7.12					
98	(5.2)	5.66	5.16	5.34	137	6.50	6.50	6.79	6.60					
99	6.20	6.17	6.27	6.21	138	6.71	6.49	6.72	6.64					
100	(6.2)	6.32	6.16	6.23	139	6.3	6.48	5.99	6.26					
101	5.91	5.96	6.03	5.97	140	5.91	5.82	6.02	5.92					
102	6.29	6.34	6.42	6.35	141	4.98	4.79	4.91	4.89					
103	(6.65)	6.87	6.20	6.57	142	4.84	4.79	4.93	4.85					
104	6.77	6.83	6.90	6.83	143	4.52	4.42	4.63	4.52					
105	(6.75)	6.16	6.17	6.36	144	4.18	4.09	4.25	4.17					
106	6.05	6.10	6.21	6.12	145	3.22	3.14	3.29	3.22					
107	(5.3)	4.89	5.25	5.15	146	2.72	2.66	2.81	2.73					
108	(4.0)	3.60	3.96	3.85	147	2.20	2.27	2.32	2.26					
109	(2.5)	2.48	2.26	2.41	148	1.92	1.88	1.96	1.92					
110	(1.20)	1.21	1.18	1.20	149	1.44	1.46	1.50	1.47					
111	.55	.563	.573	.562	150	1.17	1.16	1.23	1.19					
112	(.20)	.213	.282	.232	151	.882	.898	.925	.902					
113	.153	.155	.155	.154	152	.697	.731	.736	.721					
114	(.075)	.0861	.0731	.0781	153	.522	.521	.537	.527					
115	(.040)	.0440	.0442	.0427	154	.378	.376	.390	.381					
116	(.030)	.0290	.0294	.0295	155	.231	.232	.235	.233					

N.B. Yields in brackets are estimated by the evaluator concerned.

TABLE 6

Adjusted chain fission yields for ^{232}Th fast (pile) fission

Mass No.	Fission yield %			Simple mean		Mass No.	Fission yield %			Simple mean		Mass No.	Fission yield %			Simple mean	
	C	E	F		A		C	E	F		A		C	E	F		A
74	-	.00110	.00115	.00112	.0011	113	.06	.0460	.0692	.0584	.045	152	.32	.256	.0756	.217	.24
75	-	.00240	.00293	.00266	.0026	114	(.06)	.0480	.0761	.0614	.04	153	.21	.207	.0328	.150	.15
76	-	.00490	.00716	.00603	.006	115	.057	.0626	.0850	.0682	.060	154	(.06)	.0974	.00764	.0550	.083
77	.010	.0116	.0120	.0112	.015	116	(.05)	.0500	.0742	.0581	.05	155	(.01)	.0468	.00401	.0203	.046
78	(.035)	.0240	.0362	.0317	.032	117	.049	.0490	.0612	.0531	.05	156	.0026	.00245	.00258	.00254	.003
79	(.08)	.0542	.0855	.0732	.076	118	(.05)	.0520	.0648	.0556	.04	157	-	.000610	.000981	.000795	.012
80	(.2)	.131	.206	.179	.17	119	(.05)	.0530	.0586	.0539	.04						
81	(.4)	.335	.425	.387	.39	120	.05	.0340	.0556	.0532	.04						
82	(1.0)	.766	1.09	.952	.88	121	.055	.0480	.0524	.0518	.03						
83	1.87	2.00	2.19	2.02	2.14	122	(.04)	.0559	.0379	.0446	.03						
84	3.44	3.80	4.04	3.76	3.11	123	.031	.0579	.0313	.0401	.03						
85	4.0	4.22	4.19	4.14	4.09	124	(.03)	.0599	.0278	.0392	.02						
86	5.66	6.37	6.61	6.21	6.31	125	.033	.0629	.0321	.0427	.026						
87	5.99	7.00	7.05	6.68	6.64	126	(.05)	.0679	.0507	.0562	.05						
88	6.32	7.08	7.38	6.93	6.78	127	.09	.0782	.0915	.0866	.11						
89	6.72	7.37	7.35	7.15	7.09	128	(.18)	.189	.189	.186	.22						
90	7.40	7.75	7.74	7.63	7.46	129	(.36)	.386	.395	.380	.44						
91	7.26	7.22	7.28	7.25	7.19	130	(.8)	.801	.842	.814	.86						
92	7.49	7.06	6.82	7.12	6.99	131	1.52	1.75	1.62	1.63	1.7						
93	7.21	6.57	7.42	7.07	6.55	132	2.70	2.76	2.89	2.78	2.9						
94	(6.23)	6.70	5.73	6.22	6.10	133	3.74	3.86	3.81	3.80	3.3						
95	5.30	5.73	5.33	5.45	5.64	134	5.06	5.26	5.40	5.24	5.4						
96	(4.8)	5.83	4.42	5.02	5.53	135	4.65	5.92	5.40	5.32	5.6						
97	3.96	4.75	4.33	4.35	5.45	136	5.30	5.47	5.64	5.47	5.7						
98	(3.4)	3.89	3.74	3.68	4.86	137	4.61	6.50	6.68	5.93	6.5						
99	2.76	2.98	2.91	2.88	2.92	138	(6.02)	5.00	7.10	6.04	6.8						
100	(1.9)	1.33	1.40	1.54	1.5	139	7.38	7.02	7.04	7.15	6.0						
101	(1.14)	.627	.742	.836	.7	140	8.31	7.60	7.77	7.89	7.35						
102	(.5)	.312	.375	.397	.35	141	7.28	7.05	7.27	7.20	7.25						
103	.146	.167	.161	.158	.17	142	(7.22)	6.55	6.15	6.64	5.86						
104	(.08)	.0791	.0896	.0829	.11	143	7.12	6.82	6.67	6.87	7.25						
105	.05	.0385	.0350	.0412	.07	144	7.66	7.62	7.83	7.70	7.8						
106	.041	.0483	.0442	.0445	.06	145	5.78	5.32	5.40	5.50	6.0						
107	(.04)	.0530	.0523	.0484	.06	146	4.95	4.58	4.61	4.71	4.6						
108	(.04)	.0500	.0620	.0507	.06	147	2.97	3.02	3.09	3.03	3.8						
109	.041	.0525	.0508	.0547	.052	148	2.18	2.05	2.02	2.08	1.85						
110	(.045)	.0480	.0711	.0547	.06	149	1.44	1.05	.893	1.13	.9						
111	.045	.0511	.0691	.0551	.07	150	1.09	1.03	.340	.820	.59						
112	.062	.0629	.0693	.0647	.07	151	.41	.359	.173	.314	.37						

NB. Yields in brackets are estimated by the evaluator concerned.

TABLE 7

Adjusted chain fission yields for ^{233}U fast (Pile) fission

Mass No.	Fission Yield %		Simple Mean	A	Mass No.	Fission Yield %		Simple Mean	A	Mass No.	Fission Yield %		Simple Mean	A
	E	F				E	F				E	F		
74	.00100	.00411	.00255	.0028	113	.0695	.0653	.0674	.08	152	.192	.191	.191	.20
75	.00220	.0123	.00725	.007	114	.0644	.0615	.0629	.08	153	.109	.116	.112	.12
76	.00480	.0217	.0132	.02	115	.0561	.0579	.0570	.10	154	.0330	.0618	.0474	.064
77	.0110	.0400	.0255	.067	116	.0604	.0629	.0616	.08	155	.0210	.0327	.0268	.033
78	.0231	.0684	.0457	.097	117	.0601	.0532	.0565	.09	156	.0163	.0178	.0170	.018
79	.0503	.108	.0791	.16	118	.0601	.0530	.0565	.04	157	.0107	.00968	.0102	.011
80	.106	.196	.151	.23	119	.0742	.0617	.0679	.04	158	.00410	.00314	.00362	.0044
81	.226	.370	.298	.39	120	.0832	.0681	.0756	.04	159	.00183	.00169	.00176	.0018
82	.509	.579	.544	.84	121	.0755	.0767	.0761	.25	160	.000870	.000396	.000633	.0009
83	1.00	.995	.997	1.75	122	.0832	.0703	.0767	.31					
84	1.66	1.64	1.65	2.78	123	.0948	.0739	.0843	.40					
85	2.15	2.16	2.15	3.26	124	.120	.101	.110	.50					
86	2.84	2.78	2.81	4.84	125	.142	.138	.140	.60					
87	3.95	3.81	3.88	4.95	126	.203	.266	.234	.79					
88	5.32	3.93	4.62	5.53	127	.318	.505	.411	1.0					
89	6.67	5.87	6.27	6.13	128	.467	1.11	.788	1.3					
90	6.76	6.45	6.60	6.44	129	.822	1.69	1.26	1.57					
91	6.72	6.46	6.59	6.64	130	1.62	2.52	2.07	3.0					
92	6.85	6.50	6.67	6.65	131	3.80	3.73	3.76	3.6					
93	7.27	6.89	7.08	6.75	132	5.06	4.97	5.01	5.2					
94	7.07	6.74	6.90	6.61	133	6.18	6.00	6.09	4.7					
95	6.57	6.27	6.42	6.47	134	6.49	6.23	6.36	5.9					
96	5.93	5.70	5.81	6.02	135	6.56	6.37	6.46	6.1					
97	5.65	5.45	5.55	5.96	136	7.10	6.82	6.96	6.2					
98	5.32	5.14	5.23	5.42	137	6.88	6.64	6.76	6.35					
99	5.15	4.69	4.92	4.75	138	6.71	6.48	6.59	6.4					
100	4.51	4.38	4.44	2.81	139	6.55	6.34	6.44	5.95					
101	4.30	3.74	4.02	1.4	140	6.39	6.20	6.29	6.35					
102	1.35	2.83	2.09	.75	141	7.00	6.39	6.69	6.45					
103	.418	1.77	1.09	.41	142	6.53	6.45	6.49	5.4					
104	.309	1.24	.774	.30	143	5.66	5.53	5.59	5.0					
105	.225	.886	.555	.22	144	4.44	4.47	4.45	4.2					
106	.161	.295	.228	.16	145	3.24	3.18	3.21	4.1					
107	.147	.150	.148	.15	146	2.41	2.38	2.39	3.2					
108	.128	.107	.117	.13	147	1.69	1.68	1.68	1.6					
109	.111	.0921	.102	.12	148	1.21	1.19	1.20	1.14					
110	.0949	.0882	.0915	.11	149	.714	.704	.709	.82					
111	.0884	.0769	.0826	.11	150	.466	.462	.464	.53					
112	.0766	.0684	.0725	.10	151	.311	.305	.308	.35					

TABLE 8

Adjusted chain fission yields for ^{235}U fast (Pile) fission

Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean
	E	F			E	F			E	F	
74	.00145	.00191	.00168	113	.0342	.0329	.0335	152	.312	.281	.296
75	.00430	.0162	.0103	114	.0342	.0323	.0332	153	.196	.181	.188
76	.0120	.0269	.0194	115	.0220	.0294	.0257	154	.0840	.0759	.0799
77	.0250	.0432	.0341	116	.0359	.0340	.0350	155	.0430	.0549	.0489
78	.0521	.0683	.0602	117	.0371	.0331	.0351	156	.0204	.0211	.0207
79	.0904	.108	.0992	118	.0381	.0316	.0348	157	.0110	.0139	.0124
80	.107	.171	.139	119	.0381	.0322	.0351	158	.00540	.00782	.00661
81	.253	.279	.266	120	.0426	.0329	.0377	159	.00315	.00292	.00303
82	.398	.386	.392	121	.0451	.0345	.0398	160	.00100	.00129	.00114
83	.578	.572	.575	122	.0491	.0344	.0417	161	.000398	.000331	.000364
84	1.04	1.02	1.03	123	.0541	.0441	.0491				
85	1.33	1.33	1.33	124	.0601	.0539	.0570				
86	1.96	1.92	1.94	125	.0728	.0747	.0738				
87	2.46	2.46	2.46	126	.126	.135	.130				
88	3.62	3.48	3.55	127	.212	.270	.241				
89	4.36	4.52	4.44	128	.354	.661	.507				
90	5.24	5.44	5.34	129	.553	.921	.737				
91	5.72	5.66	5.69	130	1.57	1.73	1.65				
92	5.83	5.73	5.78	131	3.23	3.20	3.21				
93	6.05	6.12	6.08	132	4.63	4.61	4.62				
94	6.22	6.17	6.19	133	6.65	6.61	6.63				
95	6.43	6.37	6.40	134	7.71	7.61	7.66				
96	6.24	6.08	6.16	135	6.28	6.34	6.31				
97	5.88	5.95	5.91	136	6.17	6.17	6.17				
98	6.18	5.86	6.02	137	6.18	6.17	6.17				
99	5.71	5.77	5.74	138	6.72	6.55	6.64				
100	6.48	6.27	6.37	139	6.65	6.34	6.49				
101	5.41	5.34	5.37	140	5.95	6.11	6.03				
102	4.90	4.51	4.70	141	5.93	6.01	5.97				
103	3.29	3.24	3.26	142	5.74	5.65	5.70				
104	2.01	2.26	2.14	143	5.74	5.70	5.72				
105	1.30	1.19	1.25	144	5.34	5.27	5.30				
106	.450	.559	.504	145	3.78	3.75	3.76				
107	.242	.186	.214	146	2.90	2.90	2.90				
108	.151	.166	.158	147	2.22	2.10	2.16				
109	.0863	.123	.104	148	1.68	1.67	1.67				
110	.0521	.0904	.0712	149	1.07	1.03	1.05				
111	.0407	.0434	.0420	150	.687	.685	.686				
112	.0392	.0369	.0380	151	.435	.407	.421				

TABLE 9

Adjusted chain fission yields for ^{238}U fast (pile) fission

Mass No.	Fission yield %		Simple mean	Mass No.	Fission yield %		Simple mean	Mass No.	Fission yield %		Simple mean
	E	F			E	F			E	F	
75	.000750	.000244	.000497	114	.0500	.0399	.0449	153	.439	.394	.416
76	.00165	.000807	.00123	115	.0420	.0459	.0439	154	.223	.212	.217
77	.00351	.00336	.00343	116	.0440	.0414	.0427	155	.116	.129	.122
78	.00810	.0113	.00970	117	.0681	.0372	.0526	156	.0663	.0678	.0670
79	.0180	.0330	.0255	118	.0460	.0399	.0429	157	.0311	.0376	.0343
80	.0379	.0696	.0537	119	.0470	.0351	.0410	158	.0150	.0164	.0157
81	.0847	.147	.116	120	.0480	.0348	.0414	159	.00831	.00757	.00794
82	.169	.248	.158	121	.0520	.0444	.0482	160	.00280	.00288	.00284
83	.395	.394	.394	122	.0561	.0367	.0464	161	.000968	.00127	.00112
84	.815	.814	.815	123	.0621	.0394	.0507				
85	.739	.729	.734	124	.0691	.0432	.0561				
86	1.28	1.28	1.28	125	.168	.0467	.107				
87	1.60	1.58	1.59	126	.0952	.0636	.0794				
88	1.95	2.07	2.01	127	.121	.123	.122				
89	3.01	2.82	2.91	128	.181	.464	.322				
90	3.19	3.20	3.19	129	.261	.916	.588				
91	4.41	4.18	4.30	130	1.14	1.85	1.49				
92	4.91	4.39	4.65	131	3.19	3.21	3.20				
93	5.22	4.98	5.10	132	5.11	5.18	5.14				
94	5.00	4.98	4.99	133	6.47	6.65	6.56				
95	5.28	5.13	5.20	134	7.49	7.88	7.68				
96	6.01	5.94	5.98	135	6.71	6.84	6.77				
97	5.55	5.51	5.53	136	6.63	7.06	6.84				
98	5.79	5.81	5.80	137	6.04	6.01	6.03				
99	6.20	6.25	6.22	138	5.96	5.65	5.80				
100	6.60	6.61	6.61	139	6.04	5.93	5.98				
101	6.05	6.09	6.07	140	6.07	5.94	6.01				
102	6.02	6.33	6.17	141	6.76	5.38	6.07				
103	6.38	6.19	6.28	142	4.86	4.71	4.78				
104	4.75	4.99	4.87	143	4.68	4.53	4.60				
105	3.48	3.99	3.73	144	4.65	4.52	4.58				
106	2.86	2.56	2.71	145	3.90	3.74	3.82				
107	.898	1.31	1.10	146	3.54	3.38	3.46				
108	.290	.603	.446	147	2.56	2.52	2.54				
109	.130	.219	.174	148	2.13	2.08	2.11				
110	.0960	.137	.116	149	1.67	1.60	1.64				
111	.0780	.0793	.0786	150	1.32	1.26	1.29				
112	.0700	.0611	.0655	151	.831	.792	.811				
113	.0560	.0524	.0542	152	.537	.518	.527				

TABLE 10

Adjusted chain fission yields for ^{239}Pu fast (pile) fission

Mass No.	Fission yield % E	Fission yield % F	Simple mean	Mass No.	Fission yield % E	Fission yield % F	Simple mean	Mass No.	Fission yield % E	Fission yield % F	Simple mean
77	.0150	.0136	.0143	116	.0636	.0594	.0615	155	.207	.213	.210
78	.0260	.0297	.0278	117	.0450	.0600	.0525	156	.107	.152	.129
79	.0419	.0563	.0491	118	.0750	.0599	.0675	157	.105	.108	.106
80	.0717	.0844	.0780	119	.0820	.0574	.0697	158	.0681	.0676	.0678
81	.116	.152	.134	120	.0900	.0576	.0738	159	.0371	.0388	.0379
82	.193	.215	.204	121	.100	.0656	.0828	160	.0245	.0246	.0245
83	.317	.308	.312	122	.109	.0659	.0874	161	.0110	.00954	.0103
84	.500	.486	.493	123	.121	.0815	.101				
85	.614	.597	.605	124	.133	.106	.119				
86	.791	.771	.781	125	.145	.163	.154				
87	1.05	1.03	1.04	126	.205	.276	.240				
88	1.33	1.33	1.33	127	.301	.541	.421				
89	1.80	1.76	1.78	128	.422	.907	.664				
90	2.00	2.01	2.00	129	.624	1.70	1.16				
91	2.45	2.44	2.45	130	1.90	2.47	2.18				
92	2.99	2.97	2.98	131	4.12	3.86	3.99				
93	3.69	3.53	3.61	132	5.34	5.22	5.28				
94	4.26	4.19	4.22	133	6.97	6.89	6.93				
95	4.71	4.66	4.69	134	7.09	7.24	7.15				
96	4.85	4.78	4.81	135	7.33	7.46	7.39				
97	5.21	5.24	5.23	136	7.04	6.84	6.94				
98	5.64	5.54	5.59	137	6.70	6.50	6.60				
99	5.77	5.99	5.88	138	6.04	6.04	6.04				
100	6.59	6.52	6.55	139	5.83	5.51	5.67				
101	6.84	6.45	6.65	140	5.23	5.35	5.29				
102	6.65	6.54	6.59	141	5.62	5.29	5.45				
103	6.61	6.76	6.68	142	4.93	4.83	4.88				
104	6.50	6.40	6.45	143	4.38	4.30	4.34				
105	5.51	5.57	5.54	144	3.73	3.68	3.70				
106	4.66	4.29	4.47	145	3.05	2.97	3.01				
107	3.26	3.23	3.24	146	2.54	2.44	2.49				
108	2.39	2.28	2.34	147	1.97	1.98	1.98				
109	1.77	1.44	1.60	148	1.69	1.63	1.66				
110	.771	.653	.712	149	1.29	1.26	1.28				
111	.398	.358	.378	150	1.02	.980	1.00				
112	.207	.192	.200	151	.795	.778	.786				
113	.133	.126	.130	152	.629	.606	.617				
114	.0987	.0925	.0956	153	.515	.432	.473				
115	.0668	.0808	.0738	154	.289	.277	.283				

TABLE 11

Adjusted chain fission yields for ^{240}Pu fast (pile) fission

Mass No.	Fission yield % F	Fission yield % E	Simple mean	A	Mass No.	Fission yield % F	Fission yield % E	Simple mean	A	Mass No.	Fission yield % F	Fission yield % E	Simple mean	A
75	.00340	.000200	.000270	.0002	114	.0990	.100	.099	.10	153	.561	.559	.560	.56
76	.00630	.000700	.00665	.0007	115	.0990	.100	.100	.10	154	.411	.409	.410	.41
77	.0139	.0140	.0139	.014	116	.0842	.0850	.0846	.085	155	.301	.300	.301	.30
78	.0287	.0290	.0289	.029	117	.0844	.0850	.0847	.085	156	.211	.210	.210	.21
79	.0535	.0540	.0537	.054	118	.0792	.0800	.0796	.08	157	.141	.140	.141	.14
80	.0891	.0900	.0895	.09	119	.0793	.0800	.0796	.08	158	.0903	.0900	.0901	.09
81	.148	.150	.149	.15	120	.0853	.0850	.0852	.085	159	.0552	.0550	.0551	.055
82	.208	.210	.209	.21	121	.0853	.0850	.0851	.085	160	.0331	.0330	.0331	.033
83	.317	.320	.318	.32	122	.0953	.0950	.0952	.095	161	.0190	.0190	.0190	.019
84	.446	.451	.449	.45	123	.110	.110	.110	.11					
85	.595	.601	.598	.60	124	.120	.120	.120	.12					
86	.784	.792	.788	.79	125	.161	.160	.160	.16					
87	1.01	1.02	1.01	1.02	126	.321	.320	.321	.32					
88	1.27	1.28	1.27	1.27	127	.371	.370	.370	.37					
89	1.60	1.61	1.61	1.60	128	.622	.620	.621	.62					
90	1.97	1.98	1.97	1.97	129	1.01	1.01	1.01	1.01					
91	2.38	2.39	2.39	2.37	130	1.89	1.89	1.89	1.89					
92	2.90	2.93	2.91	2.90	131	3.45	3.44	3.45	3.45					
93	3.51	3.54	3.53	3.50	132	4.93	4.91	4.92	4.94					
94	4.18	4.22	4.20	4.17	133	5.70	5.68	5.69	5.72					
95	4.62	4.66	4.64	4.60	134	6.31	6.28	6.29	6.34					
96	4.97	5.01	4.99	4.95	135	6.82	6.79	6.81	6.86					
97	5.31	5.35	5.33	5.28	136	7.20	7.16	7.18	7.25					
98	5.46	5.51	5.48	5.44	137	7.04	7.00	7.02	7.09					
99	5.75	5.78	5.77	5.71	138	6.49	6.46	6.47	6.54					
100	6.02	6.07	6.04	5.99	139	5.84	5.81	5.83	5.88					
101	5.97	6.02	6.00	5.95	140	5.41	5.39	5.40	5.45					
102	5.97	6.02	5.99	5.95	141	4.84	4.77	4.80	4.86					
103	5.99	6.04	6.02	5.97	142	5.34	5.31	5.33	5.38					
104	5.74	5.79	5.76	5.73	143	4.94	5.04	5.00	5.10					
105	5.41	5.45	5.43	5.40	144	4.00	3.98	3.99	4.02					
106	5.05	5.09	5.07	5.05	145	3.41	3.31	3.36	3.34					
107	4.24	4.27	4.26	4.24	146	2.88	2.81	2.84	2.83					
108	3.13	3.15	3.14	3.14	147	2.18	2.17	2.18	2.18					
109	2.15	2.16	2.15	2.15	148	1.98	1.83	1.90	1.84					
110	1.20	1.21	1.21	1.21	149	1.52	1.51	1.52	1.52					
111	.583	.590	.586	.59	150	1.21	1.17	1.19	1.17					
112	.277	.280	.279	.28	151	.941	.937	.939	.94					
113	.148	.150	.149	.15	152	.771	.768	.769	.77					

N.B. Because experimental measurements for ^{240}Pu are sparse, both Crouch, and Meek and Rider have relied very strongly on estimates; in almost all cases these estimates are based on the calculations of Sidebotham[4], even though they may give the appearance of being independent.

However, the two evaluators have treated Sidebotham's estimates somewhat differently:-

(a) Crouch has taken them in toto as if they were a set of experimental yields, estimated independent yields from them, and applied the conservation laws to produce his final adjusted set of yields

(b) Meek and Rider have folded in any experimental data which do exist, using the Sidebotham yields as if they were experimental ones for the remainder (ie for the majority). They then produce their recommended values as for any other fissile nucleus.

In both cases, the extensive use of the Sidebotham data must be considered to put the ^{240}Pu yield sets in a different category (shared by ^{241}Pu and, to a lesser extent, ^{242}Pu) from all the other yield sets.

TABLE 12

Adjusted chain fission yields for ^{241}Pu fast (Pile) fission

Mass No.	Fission Yield % F	Fission Yield % E	Simple Mean	A	Mass No.	Fission Yield % F	Fission Yield % E	Simple Mean	A	Mass No.	Fission Yield % F	Fission Yield % E	Simple Mean	A
75	.00197	.000100	.000144	.0001	114	.139	.100	.115	.10	153	.630	.632	.632	.63
76	.00295	.000300	.00298	.0003	115	.110	.110	.110	.11	154	.453	.450	.451	.45
77	.00996	.0100	.00998	.010	116	.0997	.100	.0999	.10	155	.332	.330	.331	.33
78	.0189	.0190	.0189	.019	117	.0899	.0900	.0899	.09	156	.242	.240	.241	.24
79	.0369	.0370	.0370	.037	118	.0858	.0860	.0859	.086	157	.161	.160	.161	.16
80	.0668	.0670	.0669	.067	119	.0848	.0850	.0849	.085	158	.111	.110	.110	.11
81	.110	.110	.110	.11	120	.0857	.0850	.0854	.085	159	.0645	.0640	.0642	.064
82	.169	.170	.170	.17	121	.0906	.0900	.0903	.09	160	.0393	.0390	.0392	.039
83	.249	.250	.249	.25	122	.0937	.0930	.0933	.093	161	.0232	.0230	.0231	.023
84	.369	.370	.370	.37	123	.101	.100	.101	.10					
85	.509	.510	.509	.51	124	.111	.110	.110	.11					
86	.669	.671	.670	.67	125	.141	.140	.141	.14					
87	.880	.881	.881	.88	126	.242	.240	.241	.24					
88	1.13	1.13	1.13	1.13	127	.322	.320	.321	.32					
89	1.38	1.38	1.38	1.38	128	.524	.520	.522	.52					
90	1.76	1.76	1.76	1.76	129	.857	.850	.853	.85					
91	2.12	2.13	2.12	2.12	130	1.54	1.53	1.54	1.53					
92	2.56	2.57	2.57	2.56	131	2.87	2.85	2.86	2.85					
93	3.15	3.15	3.15	3.14	132	4.50	4.47	4.48	4.48					
94	3.74	3.75	3.74	3.74	133	5.56	5.52	5.54	5.53					
95	4.49	4.50	4.50	4.48	134	6.13	6.08	6.11	6.10					
96	4.70	4.71	4.70	4.69	135	6.74	6.69	6.70	6.71					
97	5.12	5.13	5.13	5.11	136	7.15	7.10	7.13	7.12					
98	5.41	5.40	5.40	5.40	137	7.26	7.21	7.23	7.24					
99	5.49	5.48	5.49	5.45	138	6.75	6.70	6.78	6.73					
100	5.95	5.96	5.95	5.93	139	6.12	6.08	6.10	6.10					
101	6.02	6.03	6.03	6.0	140	5.57	5.53	5.55	5.55					
102	5.92	5.94	5.93	5.91	141	5.13	5.09	5.11	5.11					
103	6.00	6.02	6.01	5.99	142	5.01	4.98	4.99	5.0					
104	5.86	5.88	5.87	5.86	143	4.94	5.44	5.19	5.46					
105	5.63	5.64	5.63	5.62	144	4.37	4.34	4.36	4.35					
106	5.24	5.26	5.25	5.24	145	3.50	3.51	3.50	3.52					
107	4.91	4.92	4.92	4.91	146	2.95	3.00	2.98	3.01					
108	3.72	3.73	3.72	3.72	147	2.44	2.41	2.42	2.42					
109	2.78	2.78	2.78	2.78	148	2.05	1.94	2.00	1.94					
110	1.70	1.70	1.70	1.7	149	1.64	1.63	1.63	1.63					
111	.895	.900	.898	.9	150	1.30	1.30	1.30	1.30					
112	.419	.420	.419	.42	151	1.00	.999	1.00	1.00					
113	.219	.220	.220	.22	152	.845	.839	.842	.84					

N.B. See note at foot of tables 11 for ^{240}Pu Fast (Pile) Fission which is equally applicable to this case.

TABLE 13

Adjusted chain fission yields for ^{242}Pu fast (Pile) fission

Mass No.	Fission Yield % F	A	Mass No.	Fission Yield % F	A	Mass No.	Fission Yield % F	A
77	.00958	.005	116	.0945	.10	155	.363	.36
78	.0179	.014	117	.0975	.10	156	.262	.26
79	.0350	.028	118	.0863	.09	157	.181	.18
80	.0629	.053	119	.0866	.09	158	.120	.12
81	.103	.088	120	.0813	.085	159	.0722	.072
82	.160	.13	121	.0907	.09	160	.0451	.045
83	.240	.21	122	.0921	.092	161	.0262	.026
84	.345	.31	123	.0970	.097	162	.0141	.014
85	.411	.43	124	.110	.11	163	.00603	.006
86	.642	.59	125	.131	.13			
87	.859	.78	126	.169	.17			
88	1.09	1.00	127	.271	.27			
89	1.34	1.26	128	.431	.43			
90	1.73	1.58	129	.650	.69			
91	2.09	1.95	130	1.18	1.17			
92	2.50	2.34	131	2.27	2.25			
93	3.09	2.35	132	4.08	4.05			
94	3.65	3.46	133	5.38	5.35			
95	4.03	4.13	134	5.91	5.88			
96	4.48	4.60	135	6.61	6.57			
97	4.87	4.92	136	7.04	7.0			
98	5.15	5.27	137	7.42	7.38			
99	5.38	5.45	138	6.97	6.92			
100	5.63	5.73	139	6.38	6.33			
101	5.89	6.0	140	5.66	5.63			
102	5.85	5.99	141	5.41	5.36			
103	5.89	5.99	142	5.11	4.6			
104	5.82	5.96	143	5.04	5.82			
105	5.68	5.78	144	4.40	4.68			
106	5.33	5.46	145	3.72	3.7			
107	5.03	5.12	146	3.20	3.18			
108	4.24	4.34	147	2.67	2.66			
109	3.25	3.29	148	2.05	2.04			
110	2.20	2.26	149	1.74	1.73			
111	1.29	1.32	150	1.44	1.43			
112	.641	.67	151	1.06	1.05			
113	.305	.32	152	.900	.90			
114	.151	.16	153	.702	.7			
115	.105	.11	154	.482	.48			

N.B. See note at foot of table 11 for ^{240}Pu Fast (Pile) Fission which is partially applicable to this case.

TABLE 14

Adjusted chain fission yields for ^{232}Th 14 MeV fission

Mass No.	Fission yield %		Simple mean	Mass No.	Fission yield %		Simple mean	Mass No.	Fission yield %		Simple mean
	E	F			E	F			E	F	
72	.00133	.00743	.00438	111	1.20	1.24	1.21	150	.550	.359	.454
73	.0110	.0153	.0131	112	1.25	1.27	1.26	151	.293	.239	.266
74	.0180	.0261	.0220	113	1.18	1.22	1.20	152	.151	.129	.140
75	.0351	.0458	.0404	114	1.18	1.30	1.24	153	.0861	.0855	.0858
76	.0662	.0782	.0722	115	1.14	1.31	1.22	154	.0541	.0566	.0553
77	.124	.114	.119	116	1.09	1.46	1.27	155	.0320	.0311	.0315
78	.298	.225	.261	117	1.07	1.41	1.24	156	.0190	.0179	.0184
79	.932	1.11	1.02	118	1.02	1.40	1.21	157	.0120	.0105	.0112
80	1.09	1.20	1.14	119	1.00	1.33	1.17	158	.00710	.00649	.00479
81	1.20	1.40	1.30	120	.968	1.28	1.12	159	.00440	.00437	.00438
82	1.49	2.02	1.75	121	.923	1.02	.971	160	.00225	.00179	.00202
83	1.63	1.70	1.66	122	.989	1.16	1.07	161	.00106	.00101	.00104
84	2.15	2.17	2.16	123	1.02	1.15	1.08				
85	4.32	3.57	3.94	124	1.06	1.19	1.13				
86	5.42	4.35	4.88	125	1.11	1.10	1.10				
87	5.56	4.72	5.14	126	1.15	1.17	1.16				
88	4.62	5.27	4.94	127	1.12	1.17	1.14				
89	6.10	5.74	5.92	128	1.26	1.38	1.32				
90	6.56	5.79	6.17	129	1.28	1.56	1.42				
91	5.98	5.70	5.84	130	1.93	1.94	1.94				
92	5.35	6.03	5.69	131	2.55	2.53	2.51				
93	5.50	5.77	5.63	132	3.18	2.85	3.01				
94	6.42	6.53	6.47	133	4.58	3.87	4.22				
95	5.68	4.63	5.15	134	6.63	6.64	6.63				
96	4.52	3.76	4.14	135	4.67	5.04	4.85				
97	3.33	3.17	3.25	136	5.71	5.44	5.57				
98	2.75	2.92	2.83	137	6.12	5.63	5.87				
99	1.97	2.00	1.98	138	6.54	5.67	6.10				
100	1.84	1.67	1.75	139	5.86	5.58	5.72				
101	1.58	1.57	1.58	140	5.75	5.82	5.78				
102	.696	1.31	1.00	141	5.89	6.03	5.96				
103	.940	1.03	.985	142	5.19	5.20	5.19				
104	.993	1.12	1.06	143	4.94	5.31	5.12				
105	.990	.987	.988	144	3.58	4.22	3.90				
106	1.08	1.11	1.10	145	2.32	3.38	2.85				
107	1.15	1.10	1.12	146	2.20	2.57	2.38				
108	1.17	1.12	1.15	147	1.82	2.07	1.94				
109	1.15	1.21	1.17	148	1.38	.977	1.18				
110	1.22	1.21	1.22	149	.994	.609	.801				

TABLE 15

Adjusted chain fission yields for ^{233}U 14 MeV fission

Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean
	E	F			E	F			E	F	
72	.0135	.0136	.0135	111	1.36	1.19	1.27	150	.429	.446	.437
73	.0195	.0225	.0210	112	1.78	1.13	1.46	151	.310	.304	.307
74	.0290	.0338	.0314	113	1.27	1.06	1.16	152	.223	.223	.223
75	.0450	.0588	.0519	114	1.33	.992	1.16	153	.144	.148	.146
76	.0699	.0834	.0767	115	1.44	1.00	1.22	154	.0985	.102	.101
77	.108	.117	.112	116	1.37	1.03	1.20	155	.0662	.0676	.0669
78	.165	.185	.175	117	1.31	1.03	1.17	156	.0440	.0430	.0435
79	.249	.260	.255	118	1.27	1.06	1.16	157	.0280	.0278	.0279
80	.368	.385	.376	119	1.21	1.09	1.15	158	.0170	.0192	.0181
81	.586	.603	.595	120	1.15	1.11	1.13	159	.0107	.0111	.0109
82	.892	.916	.904	121	1.15	1.11	1.13	160	.00700	.00805	.00752
83	1.33	1.31	1.32	122	1.22	1.20	1.21	161	.00470	.00496	.00483
84	2.02	2.06	2.04	123	1.34	1.30	1.32				
85	2.36	2.32	2.34	124	1.47	1.44	1.45				
86	2.80	2.69	2.75	125	1.52	1.53	1.53				
87	3.33	3.24	3.29	126	1.95	1.85	1.90				
88	3.92	3.93	3.93	127	2.17	2.23	2.20				
89	4.79	4.75	4.77	128	2.52	2.57	2.54				
90	4.82	4.98	4.90	129	2.81	2.89	2.85				
91	5.12	5.53	5.32	130	3.11	3.31	3.21				
92	5.71	5.78	5.75	131	3.46	3.48	3.47				
93	5.49	5.60	5.55	132	3.80	4.28	4.04				
94	5.31	5.41	5.36	133	4.47	4.66	4.57				
95	5.20	5.42	5.31	134	4.72	5.04	4.88				
96	5.06	4.96	5.01	135	5.11	5.08	5.10				
97	4.81	4.96	4.89	136	5.81	5.52	5.67				
98	4.17	4.15	4.16	137	5.05	5.16	5.10				
99	3.63	3.62	3.62	138	6.56	5.85	6.20				
100	3.26	3.06	3.16	139	5.88	5.87	5.88				
101	2.90	2.78	2.84	140	4.57	5.68	5.12				
102	2.65	2.50	2.58	141	4.69	4.97	4.83				
103	2.33	2.34	2.33	142	4.36	4.48	4.42				
104	2.09	1.96	2.03	143	3.34	3.56	3.45				
105	1.84	1.80	1.82	144	2.59	2.55	2.57				
106	1.53	1.47	1.50	145	2.12	2.03	2.08				
107	1.54	1.60	1.57	146	1.61	1.61	1.61				
108	1.55	1.06	1.31	147	1.20	1.22	1.21				
109	1.50	1.17	1.33	148	.874	.959	.916				
110	1.43	1.25	1.34	149	.618	.623	.620				

TABLE 16

Adjusted chain fission yields for ^{235}U 14 MeV fission

Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean	Mass No.	Fission Yield %		Simple Mean
	E	F			E	F			E	F	
72	.00651	.00606	.00628	111	1.04	1.17	1.11	150	.613	.539	.576
73	.0111	.0111	.0111	112	.748	1.02	.884	151	.427	.357	.392
74	.0187	.0169	.0178	113	1.01	1.05	1.03	152	.314	.275	.294
75	.0310	.0267	.0288	114	1.05	.968	1.01	153	.220	.208	.214
76	.0490	.0396	.0443	115	1.08	1.04	1.06	154	.131	.0832	.107
77	.0680	.0657	.0668	116	1.05	.965	1.01	155	.0762	.0663	.0712
78	.113	.0985	.106	117	1.05	1.07	1.06	156	.0519	.0545	.0532
79	.165	.170	.168	118	1.05	1.10	1.08	157	.0310	.0393	.0351
80	.245	.257	.251	119	1.05	1.11	1.08	158	.0195	.0242	.0218
81	.351	.288	.319	120	1.05	1.13	1.09	159	.0127	.0121	.0124
82	.509	.619	.564	121	1.01	1.06	1.03	160	.00780	.00741	.00760
83	.959	1.13	1.04	122	1.19	1.21	1.20	161	.00510	.00525	.00517
84	1.03	1.55	1.29	123	1.31	1.25	1.28				
85	1.39	1.68	1.53	124	1.46	1.30	1.38				
86	1.89	2.53	2.21	125	1.82	1.42	1.62				
87	2.51	2.47	2.49	126	1.81	1.90	1.85				
88	3.33	3.50	3.41	127	1.95	2.27	2.11				
89	4.24	4.28	4.26	128	2.37	2.44	2.41				
90	4.44	4.81	4.62	129	1.47	3.53	2.50				
91	4.79	5.01	4.90	130	3.39	3.60	3.50				
92	4.98	5.30	5.16	131	4.09	4.03	4.06				
93	5.41	5.38	5.40	132	4.29	4.64	4.46				
94	5.09	5.06	5.07	133	4.91	5.57	5.24				
95	4.84	5.01	4.93	134	5.08	5.91	5.49				
96	5.35	5.11	5.23	135	4.59	5.79	5.19				
97	5.77	5.77	5.77	136	5.77	4.58	5.17				
98	5.54	4.16	4.85	137	6.15	4.84	5.49				
99	5.18	5.04	5.11	138	6.06	4.62	5.34				
100	4.73	4.00	4.36	139	4.92	4.78	4.85				
101	4.26	3.49	3.87	140	4.42	4.56	4.49				
102	3.75	3.29	3.52	141	3.91	4.34	4.12				
103	3.48	3.24	3.36	142	4.39	4.43	4.41				
104	2.55	2.06	2.31	143	3.93	3.92	3.92				
105	1.70	1.98	1.84	144	3.09	3.20	3.15				
106	1.73	1.57	1.65	145	2.71	2.66	2.69				
107	1.60	1.29	1.45	146	2.14	2.25	2.20				
108	1.44	1.07	1.25	147	1.67	1.64	1.65				
109	1.30	1.26	1.28	148	1.20	1.25	1.23				
110	1.18	1.04	1.11	149	.844	.644	.744				

TABLE 17

Adjusted chain fission yields for ^{238}U 14 MeV fission

Mass No.	Fission yield % E	Fission yield % F	Simple mean	Mass No.	Fission yield % E	Fission yield % F	Simple mean	Mass No.	Fission yield % E	Fission yield % F	Simple mean
72	.00300	.00298	.00299	111	.905	1.04	.972	150	1.02	1.07	1.04
73	.00540	.00518	.00529	112	1.03	1.01	1.02	151	.734	.817	.775
74	.00820	.00798	.00811	113	.914	.930	.921	152	.526	.573	.549
75	.0130	.0267	.0198	114	.935	.712	.823	153	.399	.391	.395
76	.0205	.0219	.0212	115	.948	.885	.916	154	.263	.252	.257
77	.0320	.0313	.0317	116	.880	.669	.774	155	.166	.156	.161
78	.0412	.0408	.0410	117	.830	.732	.781	156	.111	.111	.111
79	.184	.169	.176	118	.888	.812	.850	157	.0825	.0822	.0824
80	.239	.212	.228	119	.913	.705	.809	158	.0381	.0426	.0403
81	.339	.331	.335	120	.924	.762	.843	159	.0190	.0260	.0225
82	.477	.451	.464	121	.934	.831	.882	160	.0130	.0158	.0144
83	.719	.650	.683	122	1.04	.832	.936	161	.00784	.00847	.00815
84	1.26	1.09	1.18	123	1.14	.904	1.02				
85	1.07	.970	.989	124	1.04	1.01	1.02				
86	1.76	1.31	1.53	125	.921	1.20	1.06				
87	2.09	1.64	1.87	126	1.22	1.34	1.28				
88	2.71	2.12	2.47	127	1.48	1.44	1.46				
89	2.98	2.96	2.97	128	1.51	1.85	1.68				
90	3.16	3.26	3.21	129	1.32	2.03	1.67				
91	3.65	3.77	3.71	130	2.54	3.35	2.94				
92	3.82	3.94	3.88	131	3.92	4.18	4.05				
93	4.35	4.96	4.65	132	4.65	4.91	4.78				
94	4.90	4.81	4.86	133	6.25	6.21	6.23				
95	4.97	5.03	5.00	134	6.72	6.63	6.68				
96	4.93	5.56	5.25	135	5.74	5.88	5.81				
97	5.18	5.33	5.25	136	5.75	5.74	5.75				
98	5.36	5.49	5.43	137	6.14	4.87	5.50				
99	5.67	5.69	5.68	138	5.01	4.43	4.72				
100	5.63	5.09	5.36	139	4.83	5.13	4.98				
101	4.42	5.61	5.01	140	4.62	4.68	4.65				
102	3.76	4.60	4.18	141	4.42	4.41	4.41				
103	4.80	4.66	4.73	142	4.00	4.10	4.05				
104	4.06	3.54	3.80	143	3.70	3.94	3.82				
105	3.00	3.23	3.11	144	3.80	3.70	3.75				
106	2.41	2.43	2.42	145	2.98	3.00	2.99				
107	1.76	1.73	1.75	146	2.68	2.31	2.50				
108	1.53	1.25	1.39	147	2.08	2.07	2.08				
109	1.33	1.22	1.27	148	1.87	1.73	1.80				
110	1.04	1.02	1.03	149	1.45	1.42	1.44				

TABLE 18

Adjusted chain fission yields for ^{239}Pu 14 MeV fission

Mass No.	Fission Yield % F	Mass No.	Fission Yield % F	Mass No.	Fission Yield % F
72	.00207	111	1.57	150	1.04
73	.00356	112	1.44	151	.808
74	.00558	113	1.38	152	.612
75	.00972	114	1.34	153	.501
76	.0155	115	1.17	154	.364
77	.0224	116	1.23	155	.256
78	.0289	117	1.23	156	.221
79	.0827	118	1.34	157	.123
80	.149	119	1.24	158	.0828
81	.258	120	1.24	159	.0572
82	.320	121	1.50	160	.0414
83	.449	122	1.25	161	.0191
84	.756	123	1.44		
85	.994	124	1.64		
86	1.06	125	1.98		
87	1.35	126	2.05		
88	1.99	127	2.25		
89	2.08	128	2.68		
90	2.33	129	3.35		
91	2.07	130	4.08		
92	2.75	131	4.77		
93	3.04	132	5.22		
94	3.36	133	5.24		
95	3.58	134	5.53		
96	4.07	135	5.74		
97	4.45	136	4.45		
98	4.60	137	4.92		
99	4.16	138	4.66		
100	4.98	139	4.38		
101	5.32	140	4.03		
102	5.82	141	3.63		
103	5.42	142	3.28		
104	5.62	143	2.78		
105	4.40	144	2.79		
106	3.72	145	2.30		
107	3.04	146	1.94		
108	2.55	147	1.84		
109	2.52	148	1.44		
110	1.85	149	1.18		

TABLE 19

Means of Percentage Errors in Fission Yields (1σ) Shown in Some of the Evaluations Considered for the 1973 and 1977 FPND Panels

Fissile Nuclide	Fission Energy	Section of Mass Yield Curve	Mass Range	Mean of Percentage Errors Reported (1σ)				Suggested 1977 1σ Errors
				Crouch 1973	Meek and Rider 1973	Crouch 1977	Meek and Rider 1977	
^{235}U	Thermal	Light wing	72-84	10.9	15.1	20.5	17.6	19.0
		Light peak	85-104	3.4	1.0	2.7	0.9	1.8
		Valley	105-129	3.7	10.8	16.9	9.6	13.2
		Heavy peak	130-150	1.8	1.1	2.8	1.2	2.0
		Heavy wing	151-161	5.6	8.7	8.2	7.9	8.1
^{239}Pu	Thermal	Light wing	72-87	8.3	15.2	16.2	15.6	15.9
		Light peak	88-109	5.4	3.9	8.2	3.6	5.9
		Valley	110-129	11.7	13.8	17.4	15.1	16.2
		Heavy peak	130-150	5.1	1.7	6.2	1.2	3.7
		Heavy wing	151-161	11.3	9.1	13.5	8.8	11.1
^{235}U	Fast (pile)	Light wing	72-83	-	20.6	18.4	21.2	19.8
		Light peak	84-105	3.8	1.9	5.2	1.4	3.3
		Valley	106-129	10.1	9.3	17.0	10.1	13.5
		Heavy peak	130-150	3.4	1.8	3.8	1.4	2.6
		Heavy wing	151-161	11.3	12.6	15.3	12.2	13.7
^{238}U	Fast (pile)	Light wing	72-85	-	19.6	18.5	18.3	18.4
		Light peak	86-106	8.3	8.4	7.4	3.2	5.3
		Valley	107-129	16.9	13.0	20.5	11.4	16.0
		Heavy peak	130-150	8.6	3.9	6.0	1.8	3.9
		Heavy wing	151-161	13.0	10.7	15.5	9.0	12.2
^{239}Pu	Fast (pile)	Light wing	72-86	9.3	16.4	21.5	11.5	15.5
		Light peak	87-109	6.9	4.1	7.3	2.4	4.9
		Valley	110-129	24.1	10.1	22.1	9.6	15.8
		Heavy peak	130-150	4.7	3.1	4.9	1.6	3.3
		Heavy wing	151-161	8.2	9.8	12.6	8.2	10.4
^{235}U	14 MeV	Light wing	72-83	12.5	21.4	17.0	10.2	13.6
		Light peak	84-110	10.8	9.0	15.5	6.5	11.0
		Valley	111-129	13.6	9.6	16.1	7.8	12.0
		Heavy peak	130-150	8.0	8.4	12.3	5.7	9.0
		Heavy wing	151-161	9.2	14.9	16.0	9.0	12.5

TABLE 20

Percentage Errors in Fission Yields (1σ) of Certain Important Nuclides Given in Some of the Evaluations Considered for the 1973 and 1977 FPND Panels

Fissile Nuclide	Fission Energy	Evaluation	Percentage Errors Reported (1σ) for mass:-							
			95	103	106	133	137	140	141	Mean for:- 143, 144, 145 146, 148, 150
^{235}U	Thermal	Crouch 1973	2.0	6.0	12.0	0.5	1.0	0.5	3.0	1.4
		M & R 1973	0.7	2.0	1.4	0.5	0.5	0.5	1.4	0.5
		Crouch 1977	1.6	6.4	6.6	2.8	1.3	1.2	1.9	1.2
		M & R 1977	0.7	1.4	1.4	0.5	0.35	0.5	1.0	0.4
Suggested 1977 1σ error			1.1	3.9	4.0	1.6	0.8	0.9	1.5	0.8
^{239}Pu	Thermal	Crouch 1973	5.0	7.0	4.0	5.0	6.0	5.0	4.0	5.5
		M & R 1973	2.0	2.8	2.8	1.4	1.0	1.0	2.8	0.7
		Crouch 1977	2.9	4.3	3.8	9.5	2.9	5.9	3.3	7.0
		M & R 1977	2.0	2.0	2.8	0.7	0.5	1.0	2.8	0.5
Suggested 1977 1σ error			2.5	3.2	3.3	5.1	1.7	3.5	3.0	3.8
^{235}U	Fast (pile)	Crouch 1973	2.5	5.5	27.0	3.0	5.5	2.0	3.0	3.2
		M & R 1973	1.0	2.0	6.0	1.4	1.0	1.4	2.0	1.1
		Crouch 1977	1.8	2.6	27.4	2.3	4.6	1.5	2.7	2.5
		M & R 1977	1.0	1.4	6.0	1.4	0.7	0.7	2.0	0.8
Suggested 1977 1σ error			1.4	2.0	15.7	1.8	2.7	1.1	2.4	1.6
^{238}U	Fast (pile)	Crouch 1973	6.0	13.0	9.0	-	7.0	2.5	-	9.7
		M & R 1973	2.8	2.8	8.0	2.8	4.0	2.0	8.0	2.3
		Crouch 1977	4.3	6.4	7.7	6.1	5.6	2.1	20.0	4.6
		M & R 1977	1.4	2.0	4.0	1.4	1.0	1.4	2.8	1.1
Suggested 1977 1σ error			2.9	4.2	5.8	3.3	3.3	1.7	11.4	2.9
^{239}Pu	Fast (pile)	Crouch 1973	3.0	4.5	10.0	5.0	10.0	1.5	4.0	3.8
		M & R 1973	2.0	2.0	6.0	2.0	2.0	1.4	4.0	1.6
		Crouch 1977	3.3	6.4	10.5	3.3	8.6	1.9	3.6	3.5
		M & R 1977	1.4	2.0	2.8	1.4	0.7	1.0	2.8	0.8
Suggested 1977 1σ error			2.4	4.2	6.6	2.3	4.7	1.4	3.2	2.2
^{235}U	14 MeV	Crouch 1973	9.0	7.0	20.0	10.0	10.0	5.0	10.0	-
		M & R 1973	6.0	4.0	6.0	6.0	2.8	4.0	8.0	11.0
		Crouch 1977	7.6	5.7	17.6	11.7	10.0	2.6	10.0	15.0
		M & R 1977	6.0	4.0	4.0	6.0	2.8	2.8	6.0	7.8
Suggested 1977 1σ error			6.8	4.9	10.8	8.8	6.4	2.7	8.0	11.4

TABLE 21

Fission Yields in ^{249}Cf Thermal Neutron Fission

Nuclide	Mean Yield (this work)	Best Average Yield: All Authors
^{141}Ce	5.9 ± 0.6	5.57 ± 0.55
^{142}La	4.3 ± 0.4	4.88 ± 0.39
^{143}Ce	4.55 ± 0.17	4.46 ± 0.19
^{147}Nd	3.27 ± 0.16	3.16 ± 0.25
^{149}Nd	2.59 ± 0.16	2.48 ± 0.12
^{151}Pm	1.98 ± 0.25	1.96 ± 0.04
^{153}Sm	1.22 ± 0.15	1.24 ± 0.02
^{156}Sm	0.84 ± 0.09	0.84 ± 0.18
^{156}Eu	0.60 ± 0.09	0.63 ± 0.02
^{157}Eu	0.43 ± 0.02	0.45 ± 0.05
^{159}Gd	0.38 ± 0.03	0.37 ± 0.02
^{161}Tb	0.19 ± 0.02	0.20 ± 0.01
^{167}Ho	$<2.1 \times 10^{-2}$	$<2.1 \times 10^{-2}$
^{171}Er	$<1.3 \times 10^{-3}$	$<1.3 \times 10^{-3}$

TABLE 22

Fission Yields for Uranium Isotopes Irradiated by "Fission Spectrum" Neutrons

Mass Number: A	FISSION YIELDS %									
	^{233}U		^{234}U		^{235}U		^{236}U		^{238}U	
	Yield	Error	Yield	Error	Yield	Error	Yield	Error	Yield	Error
85	2.61	0.09	2.20	0.09	1.52	0.05	1.54	0.09	1.13	0.09
87	3.52	0.11	2.96	0.13	2.59	0.07	2.32	0.11	1.61	0.06
88	5.15	0.08	4.00	0.07	3.35	0.04	2.97	0.09	2.10	0.07
91	6.79	0.07	6.55	0.06	6.01	0.05	5.70	0.05	4.33	0.07
92	6.18	0.09	6.21	0.12	5.32	0.10	6.29	0.09	4.39	0.12
93	6.61	0.11	6.26	0.10	5.95	0.08	5.70	0.12	5.07	0.08
95	6.18	0.06	6.44	0.09	6.52	0.07	6.47	0.07	5.20	0.06
97	5.64	0.12	6.15	0.12	6.33	0.09	5.05	0.08	5.69	0.13
99	4.61	0.10	5.16	0.11	6.08	0.08	5.94	0.10	6.19	0.10
103	1.64	0.04	2.49	0.04	3.23	0.04	4.22	0.07	6.25	0.04
105	1.03	0.06	1.28	0.06	1.33	0.06	2.50	0.07	4.15	0.07
106	0.29	0.01	0.44	0.04	0.57	0.03	1.01	0.06	2.60	0.04
111	0.06	0.01	0.04	0.01	0.03	0.01	0.10	0.03	0.07	0.01
115	0.05	0.01	0.07	0.01	0.03	0.01	0.06	0.01	0.06	0.01
125	0.15	0.04	0.15	0.03	0.11	0.04	0.19	0.15	0.08	0.06
127	0.49	0.01	0.40	0.03	0.19	0.03	0.20	0.03	0.14	0.03
129	1.69	0.08	1.66	0.09	0.89	0.05	1.04	0.08	0.62	0.05
131	4.07	0.05	3.85	0.07	3.23	0.06	3.05	0.08	3.26	0.04
132	4.57	0.08	4.45	0.06	4.35	0.07	4.32	0.04	4.62	0.07
133	6.38	0.10	6.70	0.11	6.96	0.10	7.14	0.12	7.27	0.12
134	4.63	0.12	5.87	0.12	6.53	0.05	8.17	0.15	6.96	0.15
135	4.94	0.06	4.98	0.08	6.03	0.04	5.81	0.08	5.79	0.10
137	6.19	0.11	5.99	0.06	6.22	0.09	6.16	0.07	6.53	0.06
140	6.26	-	5.93	-	6.22	-	5.76	-	6.16	-
141	7.07	0.12	6.70	0.11	6.40	0.05	5.58	0.12	5.85	0.09
142	7.82	0.10	6.17	0.10	6.42	0.09	5.87	0.09	5.54	0.09
143	5.85	0.09	6.34	0.08	5.89	0.07	6.16	0.08	4.78	0.08
144	5.51	0.09	5.63	0.09	5.91	0.06	5.41	0.09	5.29	0.16
147	1.81	0.03	2.08	0.05	1.93	0.06	2.36	0.05	2.65	0.04
149	0.75	?	1.07	0.09	1.23	0.09	1.38	0.11	-	-
151	0.27	0.02	0.30	0.05	0.37	0.05	0.40	0.05	0.41	0.02
153	0.16	?	0.15	?	0.16	(0.05)	0.04	?	0.31	?

TABLE 23

Yields in the "Fast" Fission of ^{237}Np

Element	Mass Number: A	Yield	Error
Kr	85	0.79	0.05
Kr	87	1.59	0.22
Kr	88	1.76	0.11
Sr, Y	91	4.02	0.28
Sr	92	4.36	0.30
Y	93	5.78	0.28
Zr	95	5.78	0.12
Zr, Nb	97	6.51	0.28
Mo	99	5.44	0.16
Ru	103	5.50	0.16
Ru, Rh	105	2.94	0.12
Ru	106	1.76	0.06
Ag	111	0.06	0.01
In, Cd	115	0.04	0.02
Sb	125	0.16	0.01
Sb	127	0.30	0.02
Sb	129	1.56	0.05
I	131	4.07	0.16
I	132	5.30	0.33
I	133	6.70	0.4
I	134	5.85	0.28
I	135	6.60	0.40
Cs	137	6.08	0.33
Ba	139	5.74	0.28
Ba	140	5.58	-
Ce	141	6.30	0.16
La	142	4.40	0.39
Ce	143	4.92	0.16
Ce	144	4.30	0.11
Nd	147	2.29	0.06
Nd	149	1.67	0.11
Pm	151	0.61	0.11
Pm	153	0.44	0.02

TABLE 24

"Fast" Fission Yields of ^{240}Pu and ^{242}Pu

Mass Number: A	^{240}Pu Yield	^{242}Pu Yield
125	0.038	0.05
131	4.23	3.32
132	5.57	4.52
133	6.90	6.49
134	8.17	7.36
135	7.27	6.89
136	6.85	6.56
137	6.41	6.21
143	4.23	4.52
144	3.72	4.16
145	2.96	3.38
146	2.48	2.92
148	1.68	1.98
149	1.32	1.64
150	1.02	1.32
151	0.86	1.05
152	0.70	0.84
154	0.43	0.48

TABLE 25

"Fast" Fission Yields of ^{240}Pu

Mass Number: A	Total Chain* Fission Yield	Error (1σ)	Yield Normalised to Mass 143 = 4.23%	Koch Value (4)
67	3.8×10^{-7}	0.3×10^{-7}	3.62×10^{-5}	
72	1.16×10^{-6}	0.04×10^{-6}	1.11×10^{-4}	
89	0.0149	0.0010	1.42	
90	0.0193	0.0006	1.84	
91	0.0234	0.0007	2.23	
93	0.0398	0.0012	3.79	
95	0.0449	0.0013	4.28	
97	0.0530	0.0016	5.05	
99	[0.0618]	0.0019	5.89	
103	0.0709	0.0021	6.75	
105	0.0574	0.0017	5.47	
106	0.0512	0.0015	4.88	
109	0.0180	0.0008	1.72	
111	5.01×10^{-3}	0.15×10^{-3}	0.477	
112	2.34×10^{-3}	0.07×10^{-3}	0.223	
113	1.59×10^{-3}	0.15×10^{-3}	0.151	
115	6.49×10^{-4}	0.19×10^{-4}	0.0618	
127	4.47×10^{-3}	0.13×10^{-3}	0.426	
131	0.0392	0.0020	3.73	4.23
132	0.0517	0.0016	4.92	5.57
133	0.0757	0.0027	7.21	6.90
135	0.0760	0.0035	7.24	7.27
137	0.0606	0.0018	5.77	6.41
140	0.0546	0.0016	5.20	
141	0.0490	0.0015	4.67	
143	0.0444	0.0013	[4.23]	4.23
144	0.0397	0.0012	3.78	3.72
147	0.0227	0.0007	2.16	
149	0.0141	0.0004	1.34	1.32
151	8.42×10^{-3}	0.25×10^{-3}	0.802	0.86
153	6.19×10^{-3}	0.19×10^{-3}	0.590	
155	2.42×10^{-3}	0.07×10^{-3}	0.231	
156	1.72×10^{-3}	0.05×10^{-3}	0.164	
157	1.33×10^{-3}	0.10×10^{-3}	0.127	
159	3.24×10^{-4}	0.10×10^{-4}	0.0309	
161	1.12×10^{-4}	0.03×10^{-4}	0.0107	
169	$<1.1 \times 10^{-6}$	-	<0.000105	

* Normalised to ^{99}Mo = 0.0618

TABLE 26

Compilations of independent fission yields

Compiler / Nuclide	Ref. 7		Ref. 10		Ref. 32		Ref. 33	
	Experimental yields	Adjusted sets	Experimental yields	Adjusted sets	Experimental yields	Adjusted sets	Experimental yields	Adjusted sets
Spontaneous fission								
^{252}Cf	✓	✓						
Thermal fission								
^{233}U	✓	✓	✓	✓	✓	✓	✓	✓
^{235}U	✓	✓	✓	✓	✓	✓	✓	✓
^{239}Pu	✓	✓	✓	✓	✓	✓		
^{241}Pu	✓	✓		✓				
Fast (pile) fission								
^{237}Th	✓	✓		✓				
^{233}U	✓	✓		✓				
^{235}U	✓	✓	✓	✓	✓	✓		
^{236}U	✓	✓						
^{238}U	✓	✓	✓	✓	✓			
^{237}Np	✓	✓						
^{239}Pu	✓	✓		✓	✓	✓		
^{240}Pu	✓	✓		✓				
^{241}Pu	✓	✓		✓				
^{242}Pu	✓	✓						
14 M.e.V. fission								
^{232}Th	✓	✓	✓	✓				
^{233}U	✓	✓	✓	✓				
^{235}U	✓	✓	✓	✓	✓	✓		
^{238}U	✓	✓	✓	✓	✓	✓		
^{239}Pu	✓	✓	✓		✓	✓		

TABLE 27

A partial comparison of mass separated (MS), evaluated experimental (E) and calculated (C) fractional independent yields (as a percentage) for ^{235}U thermal neutron fission

Mass No.	Type	Zn	Ga	Ge	As	Se	Br	Kr	Rb	Sr	Y	Zr	Nb	Mo
		Element												
79	MS ref. 37	3.7 ± 1.5	37.8 ± 4.0	52.5 ± 5.0	6.0 ± 2.0									
	E ref. 33													
	C ref. 10	6.08	28.95	58.73	5.99									
	C ref. 7	5.11	41.52	49.49	3.76									
80	MS		9.7 ± 2.5	83.2 ± 4.0	7.1 ± 2.0									
	E													
	C10		9.37	82.43	8.12									
	C 7		7.16	82.22	10.04									
81	MS		4.9 ± 1.3	64.4 ± 3.0	26.6 ± 2.5	4.1 ± 1.5								
	E													
	C10		5.40	61.43	28.51	4.51								
	C7		1.50	62.58	33.79	1.68								
82	MS		2.1 ± 1.0	39.5 ± 3.0	47.2 ± 3.0	9.5 ± 1.5	1.7 ± 1.0							
	E													
	C10		2.84	39.50	36.79	20.05	0.71							
	C7		1.02	38.20	49.88	10.88	0.02							
83	MS			0.7 ± 1.0	58.0 ± 2.0	31.2 ± 2.0	3.1 ± 1.0							
	E													
	C10			12.00	37.29	47.64	2.86							
	C 7			8.41	54.81	31.99	4.70							
84	MS			2.3 ± 1.7	23.5 ± 1.4	69.2 ± 2.5	5.0 ± 1.0							
	E			3.0 ± ?	37 ± 6	57 ± 6	3 ± 2							
	C10			1.36	21.95	70.60	6.00							
	C 7			3.17	21.31	69.91	5.58							
85	MS			1.2 ± 0.5	11.4 ± 0.8	67.6 ± 1.5	17.7 ± 1.3	2.1 ± 0.7						
	E				13 ± 5	69 ± 7	18 ± 7							
	C10			1.06	14.10	68.32	15.28	1.25						
	C7			0.57	15.95	86.46	16.75	0.27						
86	MS				4.8 ± 1.0	60.4 ± 2.0	29.3 ± 1.5	5.5 ± 1.0						
	E				4 ± 1	62 ± 10	31 ± 10	3 ± 1						
	C10				5.53	57.52	30.01	6.67						
	C 7				6.96	67.46	20.09	5.40						
87	MS				2.0 ± 1.0	27.8 ± 1.2	51.4 ± 1.8	17.2 ± 1.0	1.6 ± 1.0					
	E				1 ± ?	39 ± 7	44 ± 7	15 ± 1						
	C10				2.27	33.37	36.62	26.22	1.39					
	C 7				2.85	28.91	50.15	17.89	0.08					
88	MS				1.0 ± 0.5	10.1 ± 1.0	39.9 ± 2.0	46.0 ± 2.0	2.4 ± 0.8					
	E					11 ± 1	51 ± 5	37 ± 3	1 ± ?					
	C10				0 59	15.12	31.36	46.63	5.78					
	C 7				1.91	8.28	48.02	41.08	0.70					
89	MS					2.2 ± 0.8	34.8 ± 1.4	67.5 ± 2.0	4.8 ± 0.9	0.76± 0.6				
	E					2.3 ± 1	26 ± 4	67 ± 4	4.7 ± 1.6					
	C10					5.57	28.67	59.56	5.93	0.22				
	C 7					2.58	23.75	71.67	1.95	0.04				
90	MS						10.0 ± 0.7	75.9 ± 1.0	14.1 ± 0.7					
	E						11 ± 2.5	76 ± 3	13 ± 2					
	C10						10.58	78.51	10.58					
	C 7						11.39	75.41	10.82					
91	MS						4.0 ± 0.3	53.9 ± 1.0	37.6 ± 1.0	4.5 ± 0.5				
	E						4 ± 1.6	55 ± 1.5	38 ± 3	3 ± 3				
	C10						4.49	55.27	32.31	7.69				
	C 7						4.05	54.92	36.45	4.53				
92	MS						1.8 ± 0.8	31.5 ± 1.4	50.4 ± 1.8	15.2 ± 1.4	0.9 ± 0.6	0.2 ± 0.2		
	E							31 ± 1	55 ± 3	14 ± 3				
	C10						2.32	30.24	34.61	30.24	2.32	0.13		
	C7						0.61	27.78	53.88	17.60	0.13	0.00		
93	MS							10.7 ± 1.2	48.6 ± 1.7	37.9 ± 1.5	2.6 ± 0.7			
	E							8 ± 1	49 ± 3	41 ± 3	1.6 ± 0.2			
	C10							11.51	34.71	49.56	3.87			
	C7							8.12	49.63	40.13	2.03			
94	MS							3.1 ± 0.8	24.8 ± 1.2	65.2 ± 1.6	6.3 ± 0.8	0.6 ± 0.5		
	E							2.7 ± 0.5	25 ± 1.5	66 ± 2.5	6 ± 2			
	C10							2.75	21.37	65.56	9.73	0.54		
	C 7							3.94	25.72	64.52	5.77	0.04		
95	MS							0.9 ± 0.5	13.7 ± 1.5	68.4 ± 2.0	15.4 ± 1.5	1.6 ± 0.8		
	E								10 ± 0.5	77 ± 6	13 ± 6			
	C 10							0.74	13.69	71.10	13.69	0.74		
	C 7							0.11	12.70	69.75	16.51	0.92		
96	MS								4.9 ± 1.0	58.4 ± 2.5	32.2 ± 2.5	3.5 ± 0.8	1.0 ± 0.8	
	E								2.1 ± 0.1					
	C10								4.96	53.73	31.99	8.97	0.13	
	C 7								3.12	53.59	39.29	3.97	0.01	
97	MS								3.5 ± 1.0	30.7 ± 2.0	34.1 ± 2.0	10.6 ± 1.5	1.1 ± 0.6	
	E								0.6 ± 0.1					
	C10								4.07	32.56	33.78	9.32	0.12	
	C7								0.93	34.47	51.54	12.85	0.04	
98	MS								0.9 ± 0.6	15.0 ± 2.2	37.7 ± 3.0	43.4 ± 3.0	3.0 ± 1.5	
	E													
	C10								0.89	18.63	32.73	42.56	4.75	
	C7								0.05	14.35	38.98	45.73	0.88	
99	MS									6.2 ± 1.2	34.3 ± 2.3	53.6 ± 2.5	4.6 ± 1.2	1.3 ± 0.8
	E													
	C10									6.15	25.51	58.66	8.85	0.71
	C7									5.43	34.22	53.38	6.92	0.03

TABLE 28

The odd-even Z and N effects

Nuclide	Energy	Proton effect	Neutron effect
^{252}Cf	Spont.	0.050	0.010
^{233}U	Thermal	0.210	0.041
^{235}U	Thermal	0.228	0.044
^{239}Pu	Thermal	0.171	0.033
^{241}Pu	Thermal	0.206	0.040
^{232}Th	Pile	0.327	0.063
^{233}U	Pile	0.143	0.028
^{235}U	Pile	0.151	0.029
^{236}U	Pile	0.166	0.032
^{238}U	Pile	0.329	0.063
^{237}Np	Pile	0.000	0.000
^{239}Pu	Pile	0.124	0.024
^{240}Pu	Pile	0.244	0.047
^{241}Pu	Pile	0.141	0.027
^{242}Pu	Pile	0.364	0.070
^{232}Th	14 M.e.V.	0.018	0.003
^{233}U	14 M.e.V.	0.015	0.003
^{235}U	14 M.e.V.	0.015	0.003
^{238}U	14 M.e.V.	0.018	0.003
^{239}Pu	14 M.e.V.	0.015	0.003

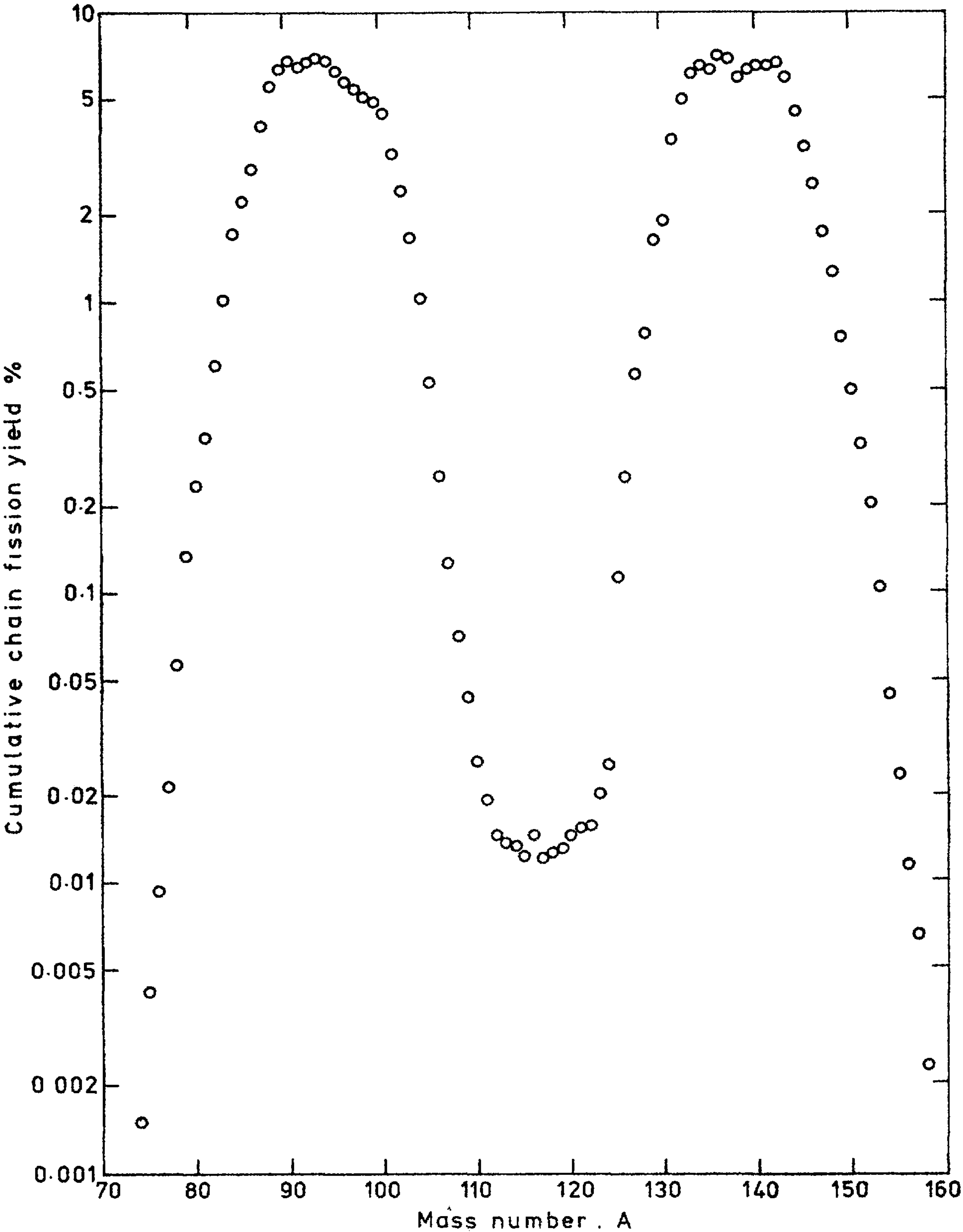


FIG. 1. ADJUSTED CHAIN FISSION YIELDS FOR ^{233}U THERMAL FISSION.

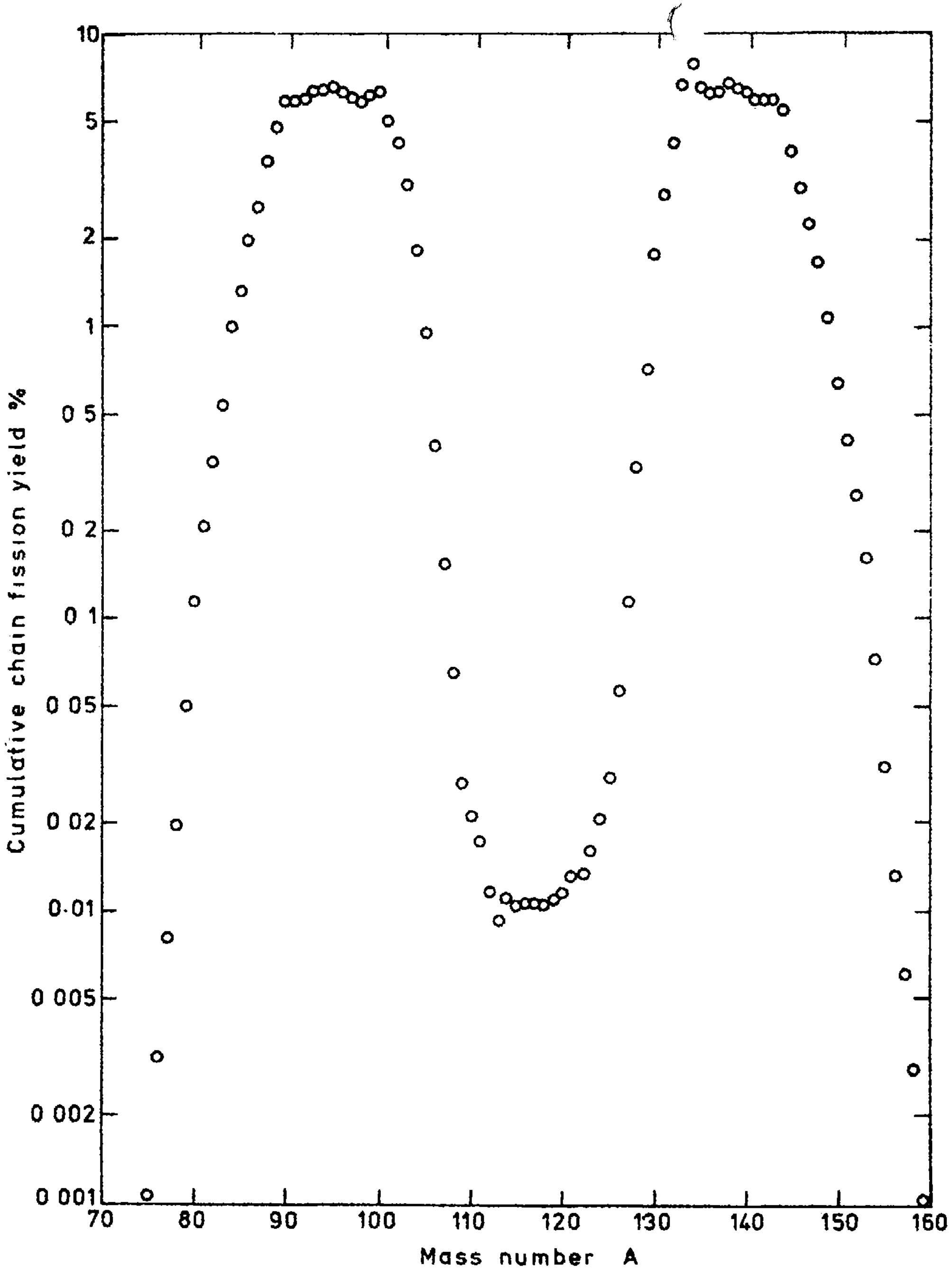


FIG 2 ADJUSTED CHAIN FISSION YIELDS FOR ^{235}U THERMAL FISSION

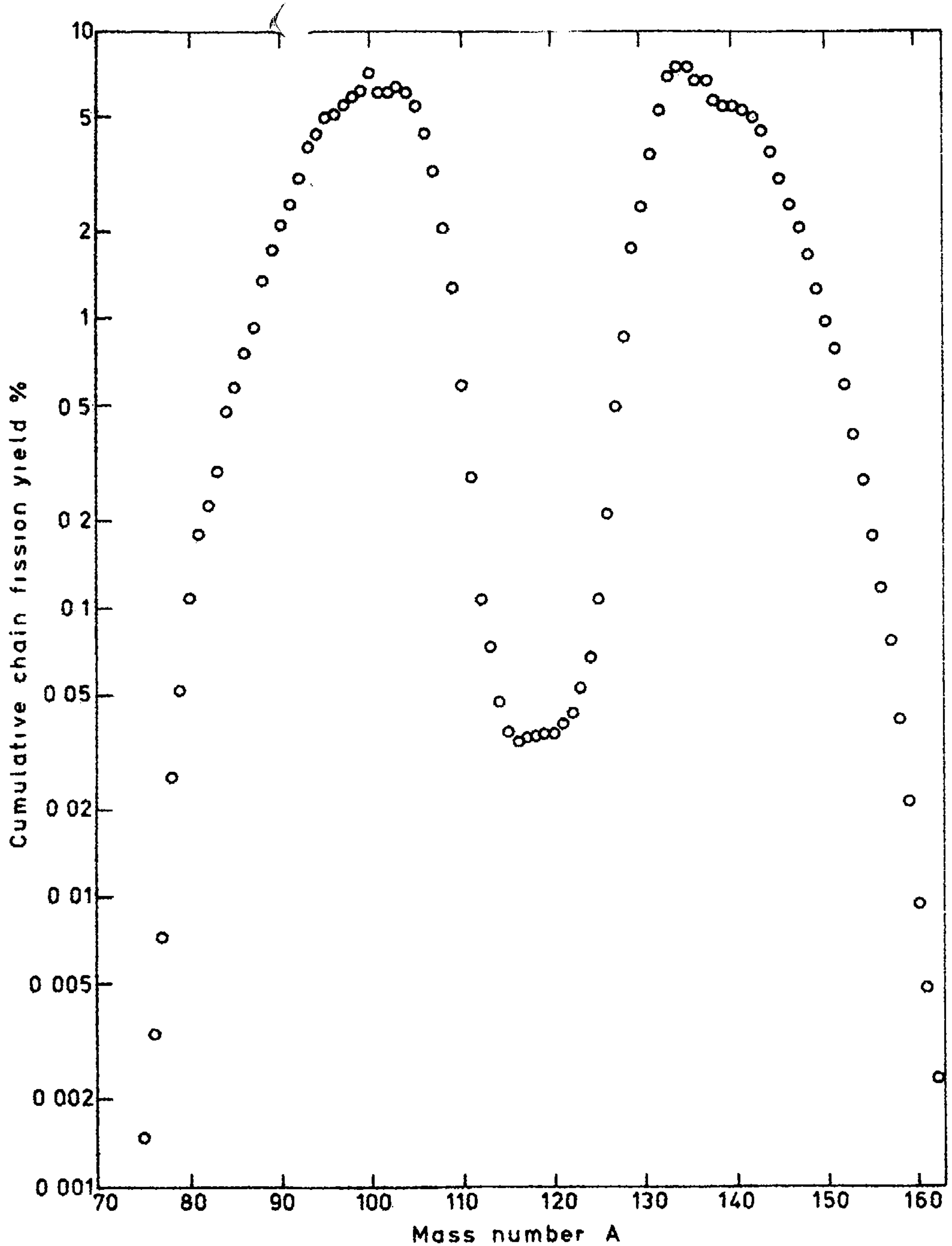


FIG 3 ADJUSTED CHAIN FISSION YIELDS FOR ^{239}Pu THERMAL FISSION

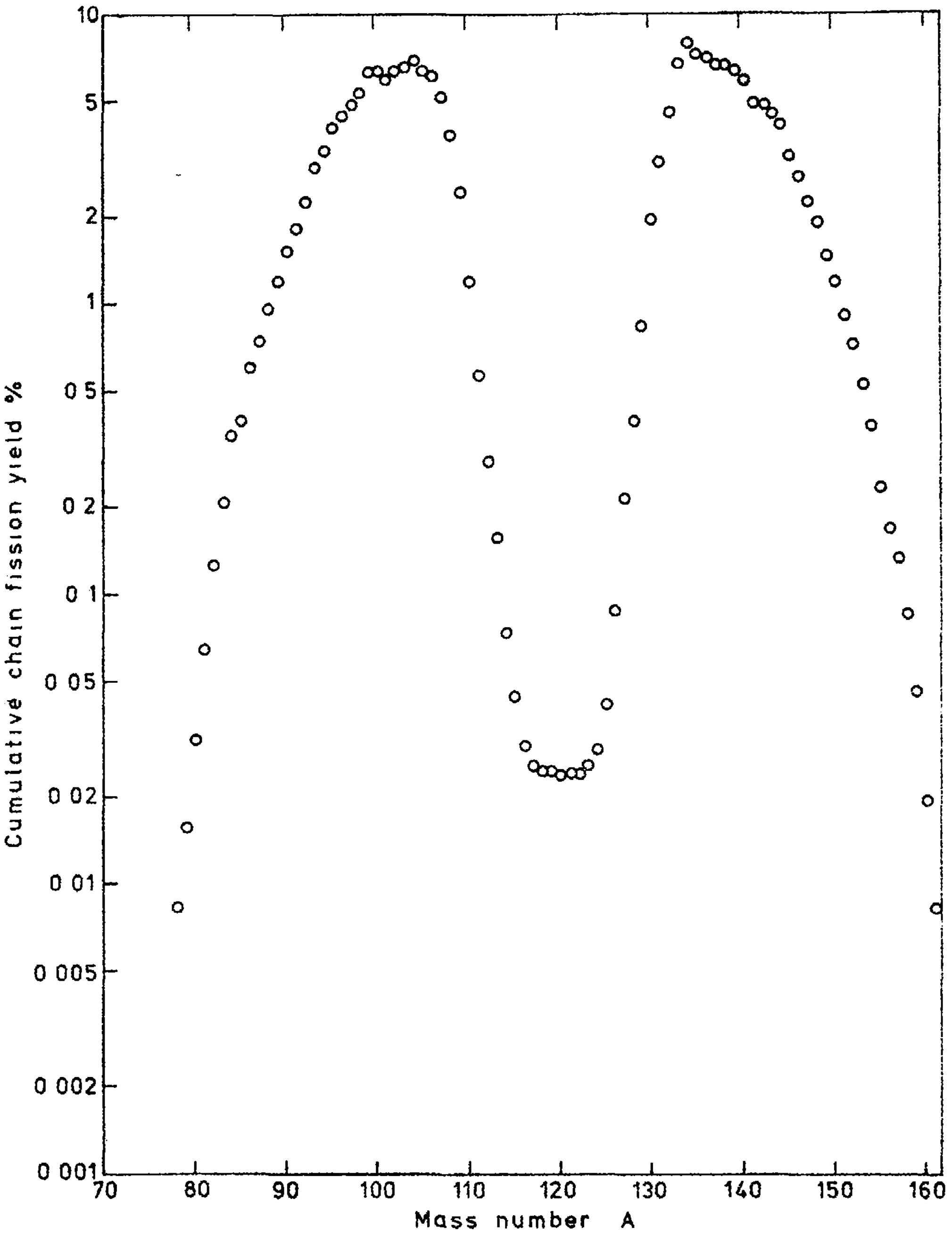


FIG 4 ADJUSTED CHAIN FISSION YIELDS FOR ^{241}Pu THERMAL FISSION

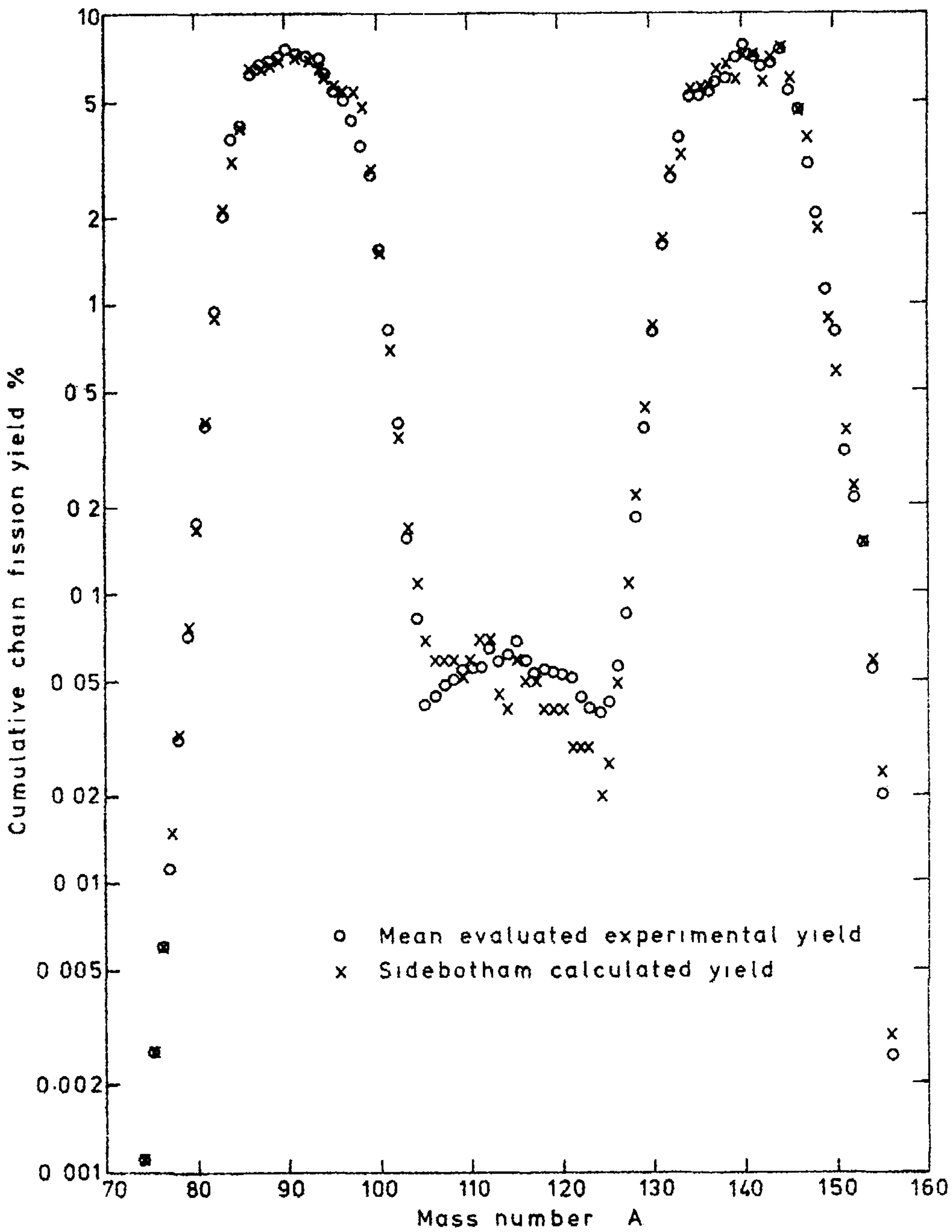


FIG 5 ADJUSTED CHAIN FISSION YIELDS FOR ^{232}Th FAST (PILE) FISSION

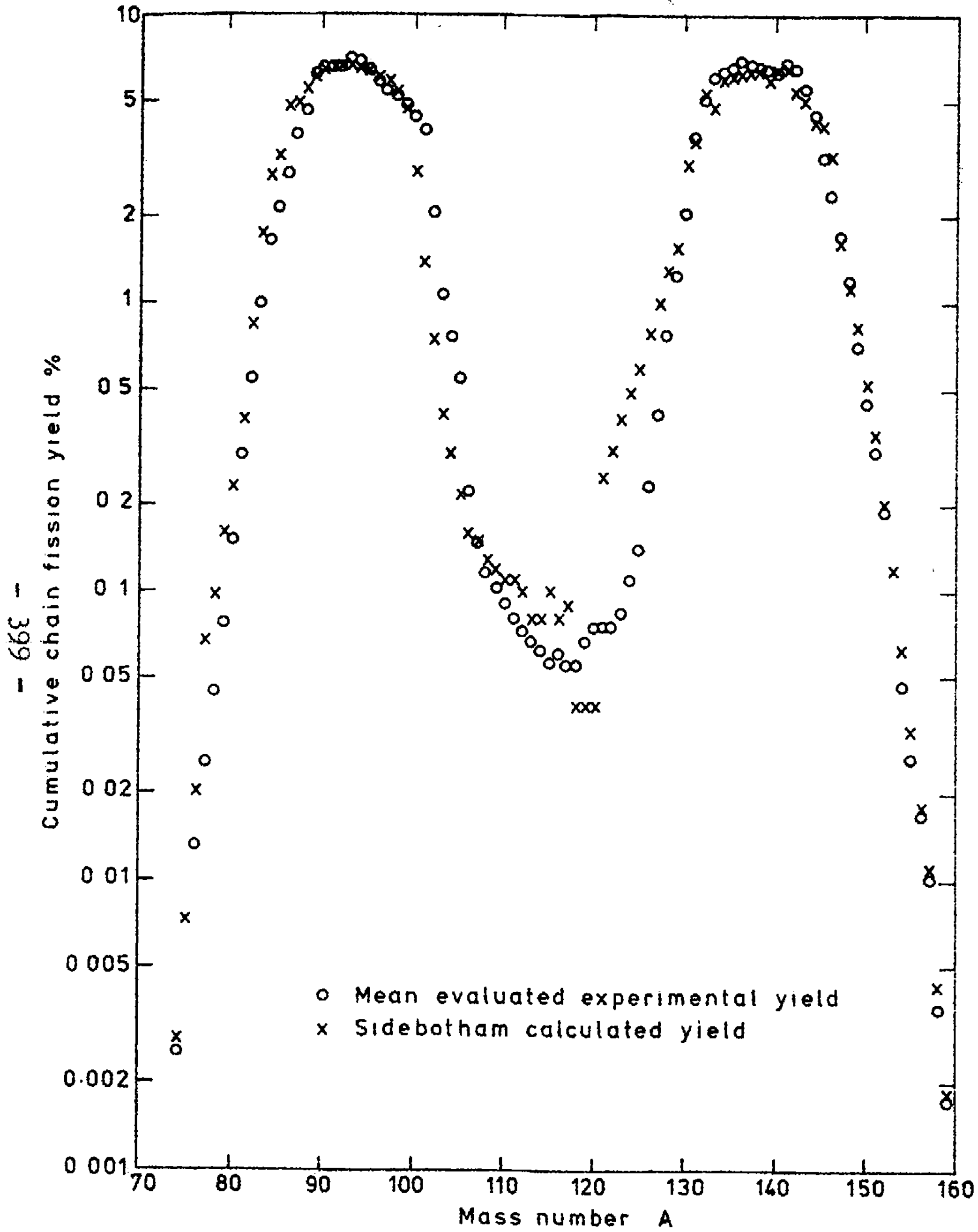


FIG. 6 ADJUSTED CHAIN FISSION YIELDS FOR ^{233}U FAST (PILE) FISSION

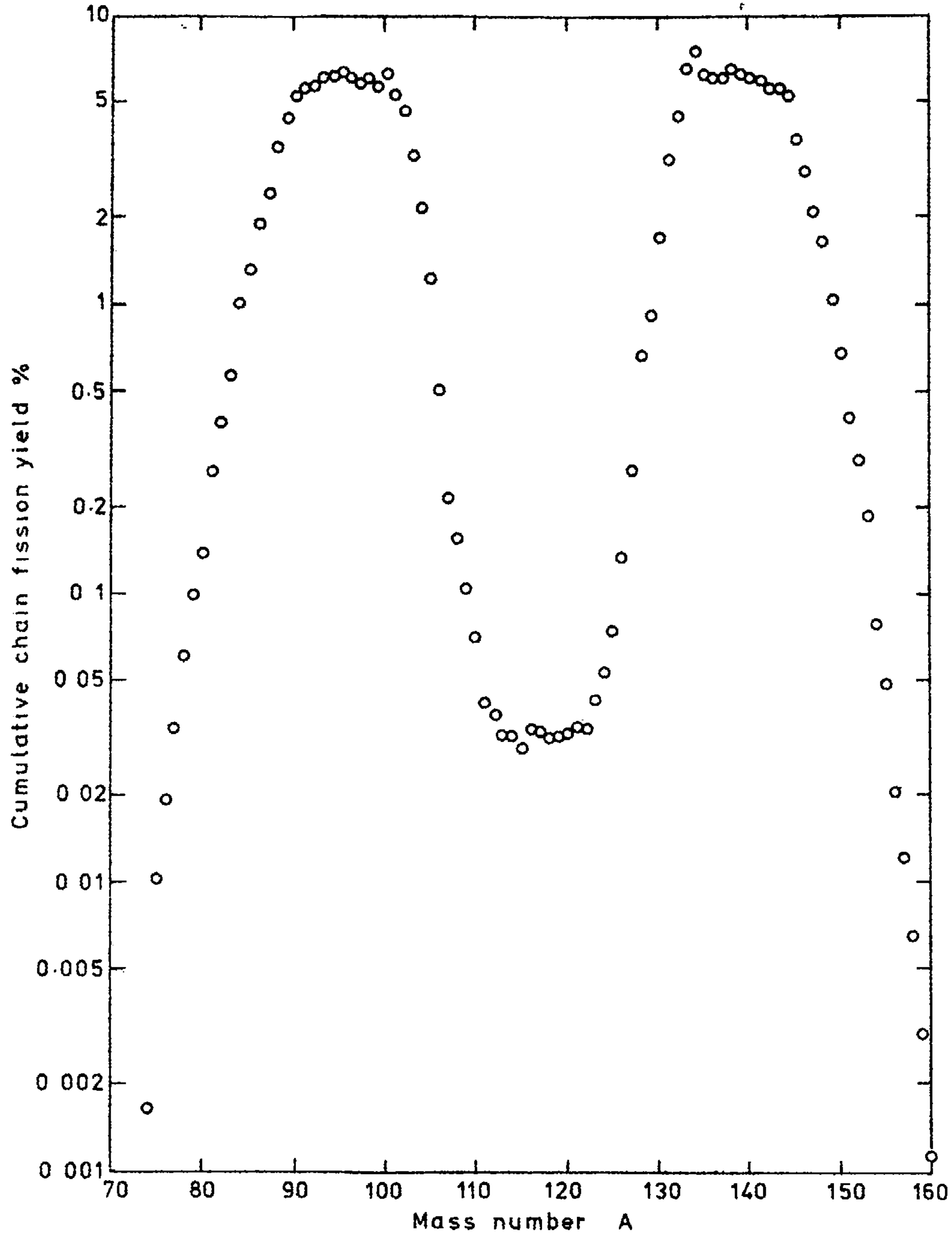


FIG 7 ADJUSTED CHAIN FISSION YIELDS FOR ^{235}U FAST (PILE) FISSION

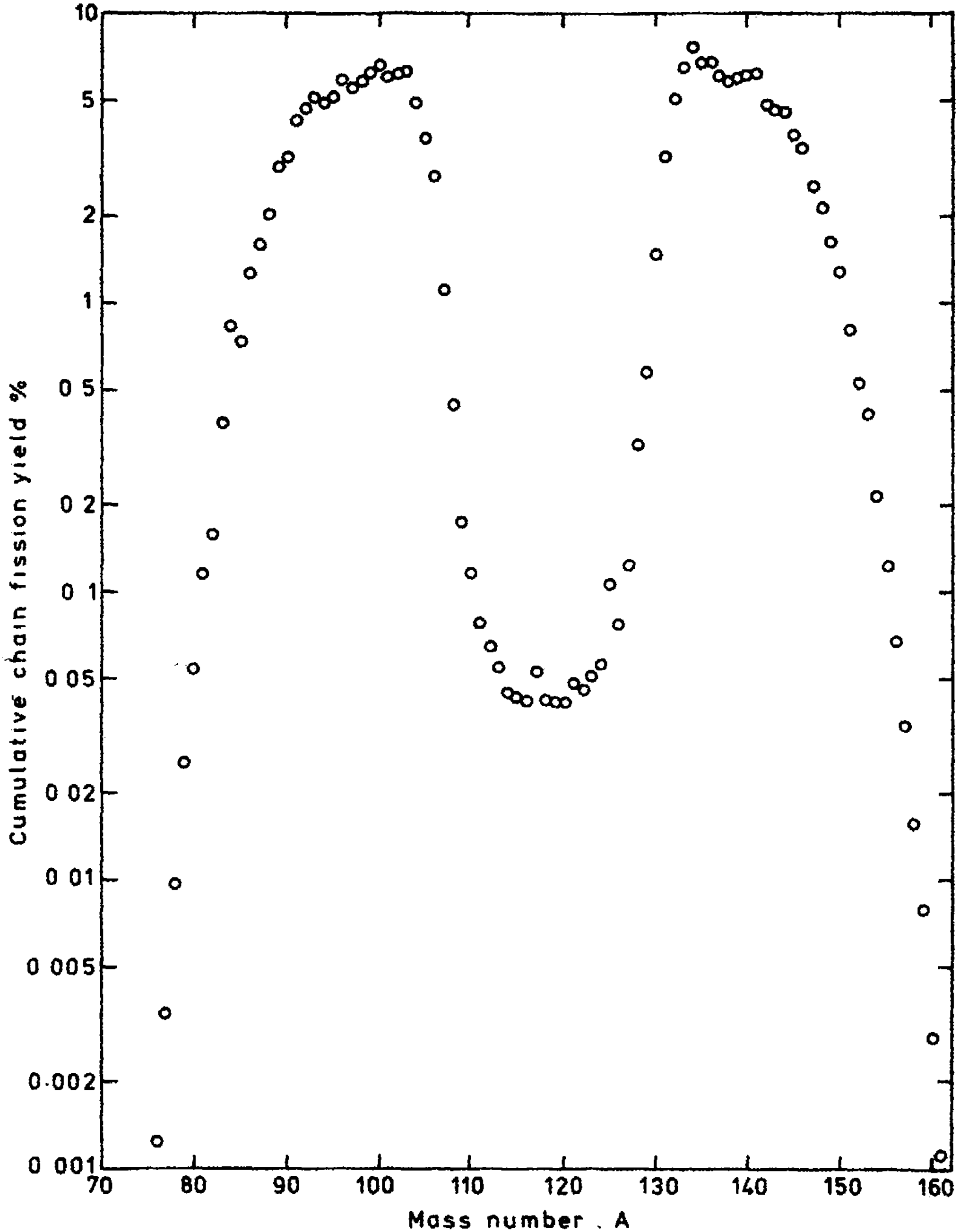


FIG. 8. ADJUSTED CHAIN FISSION YIELDS FOR ^{238}U FAST (PILE) FISSION.

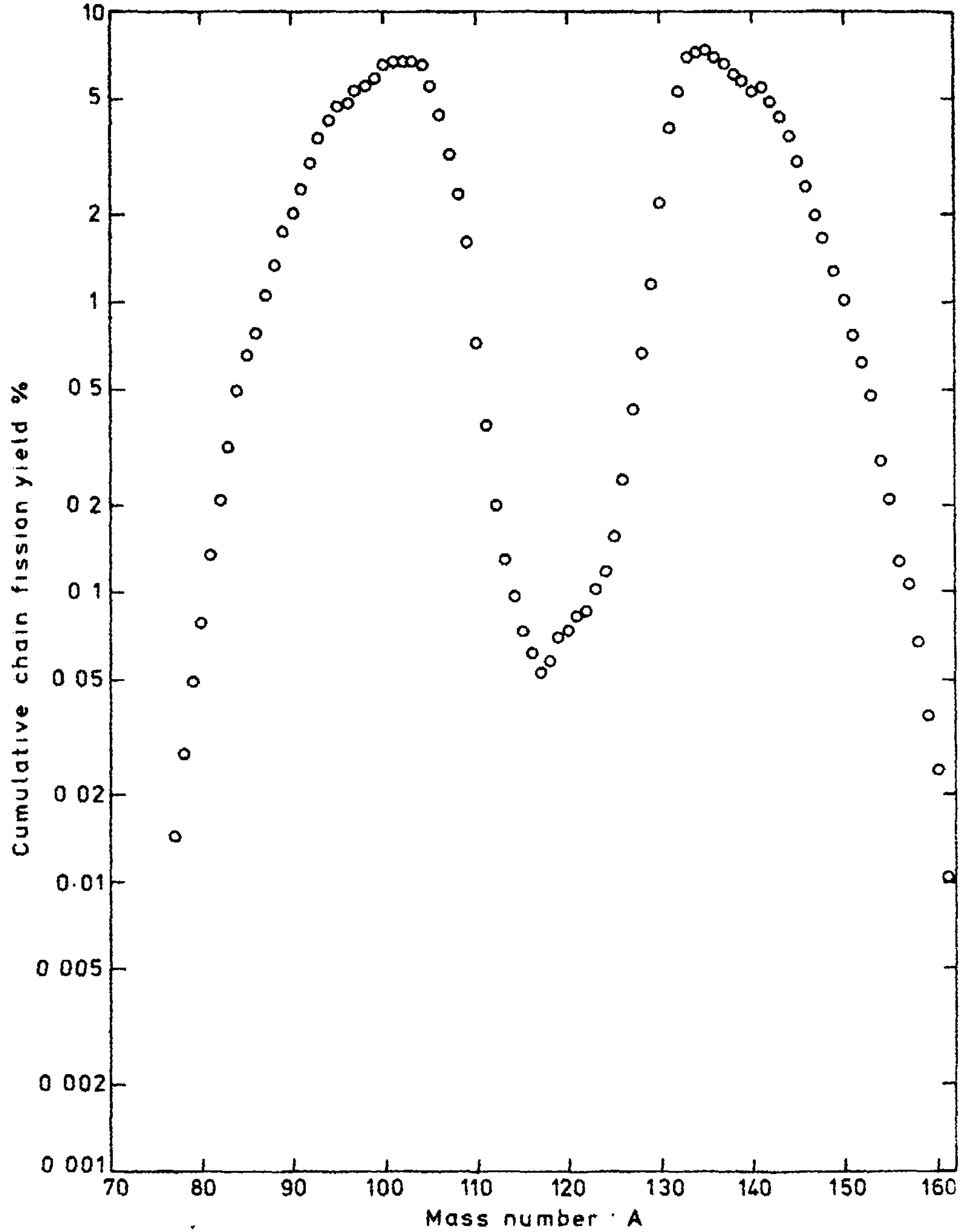


FIG 9. ADJUSTED CHAIN FISSION YIELDS FOR ^{239}Pu FAST (PILE) FISSION

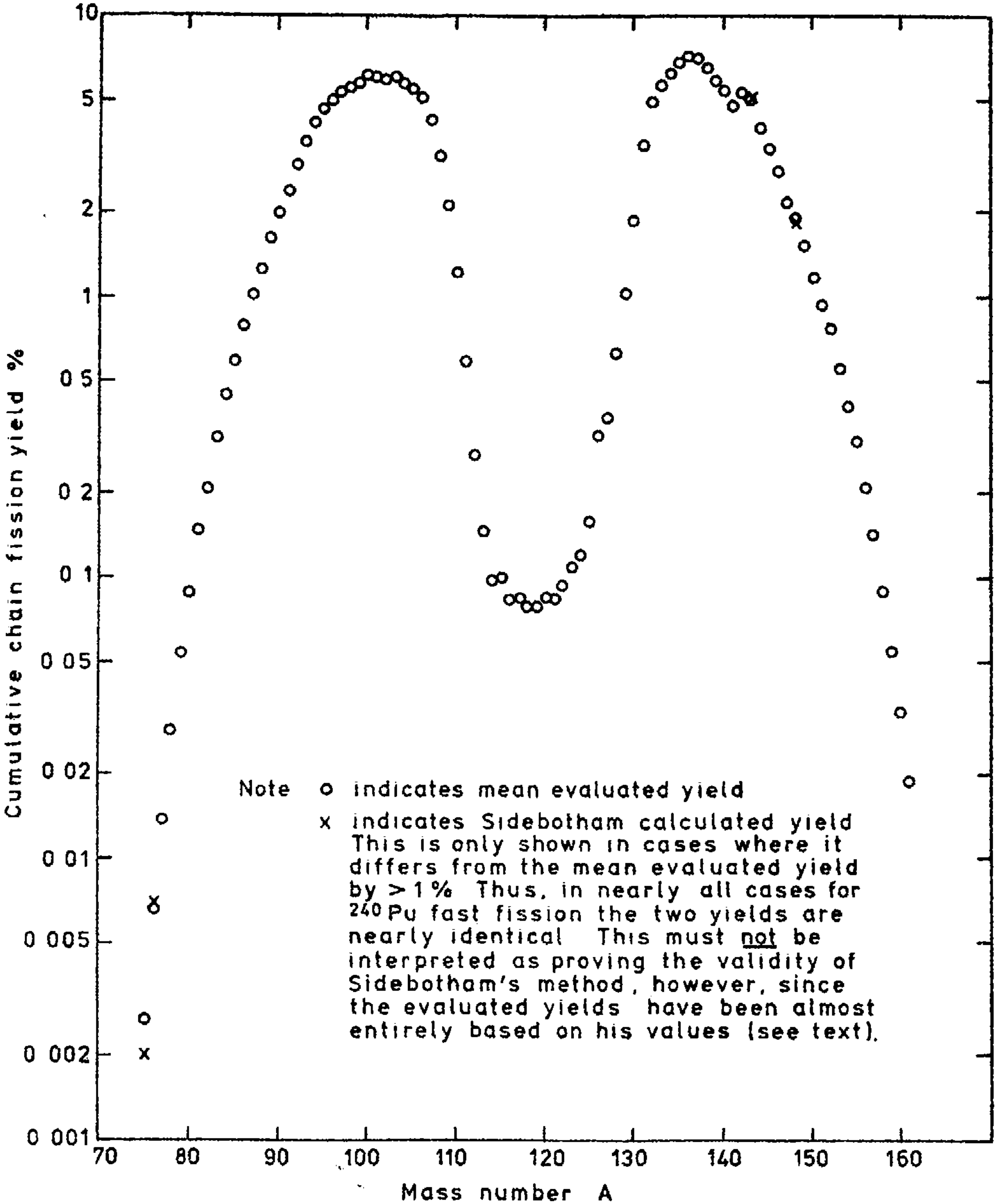


FIG 10 ADJUSTED CHAIN FISSION YIELDS FOR ^{240}Pu FAST (PILE) FISSION

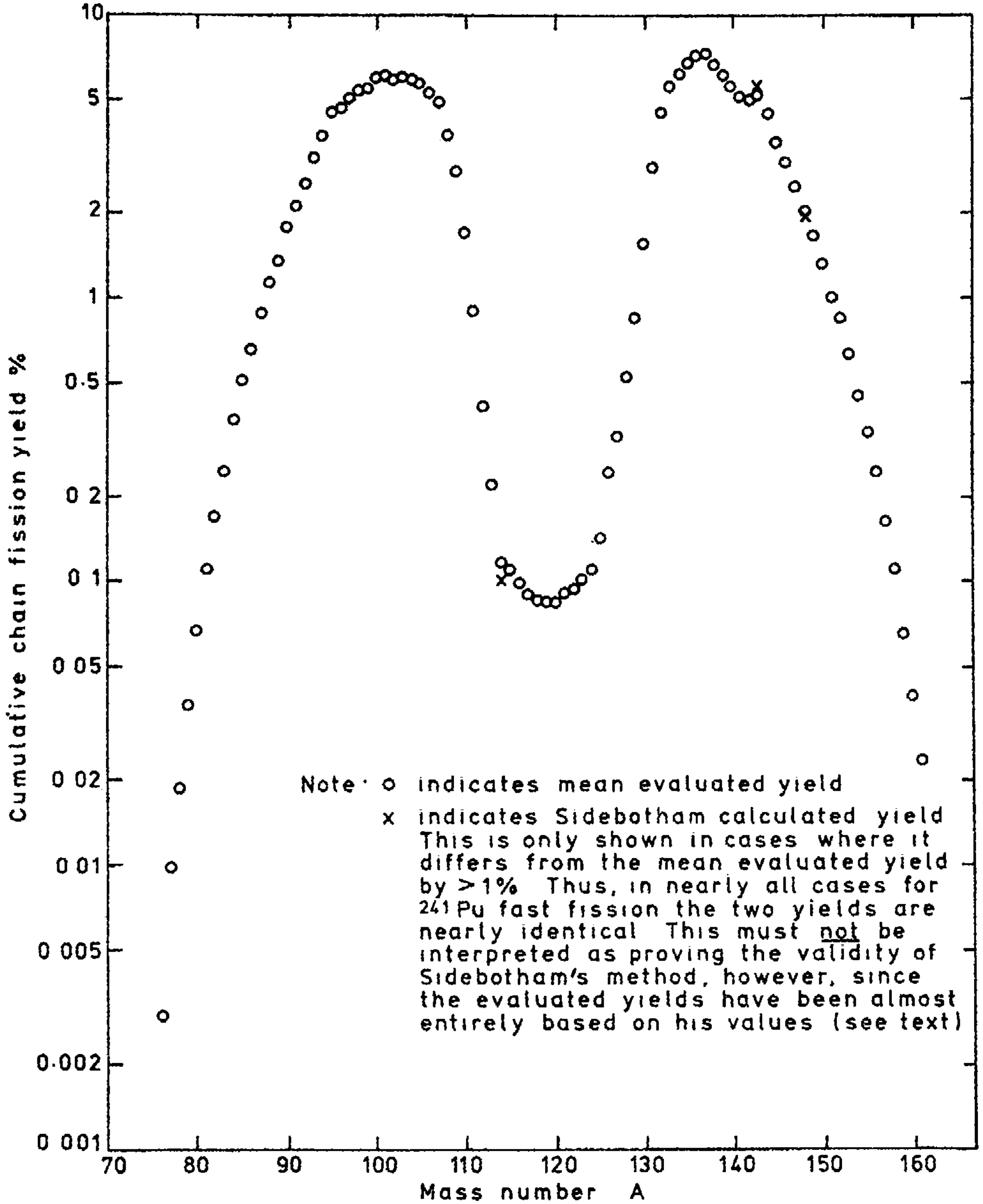


FIG. 11 ADJUSTED CHAIN FISSION YIELDS FOR ^{241}Pu

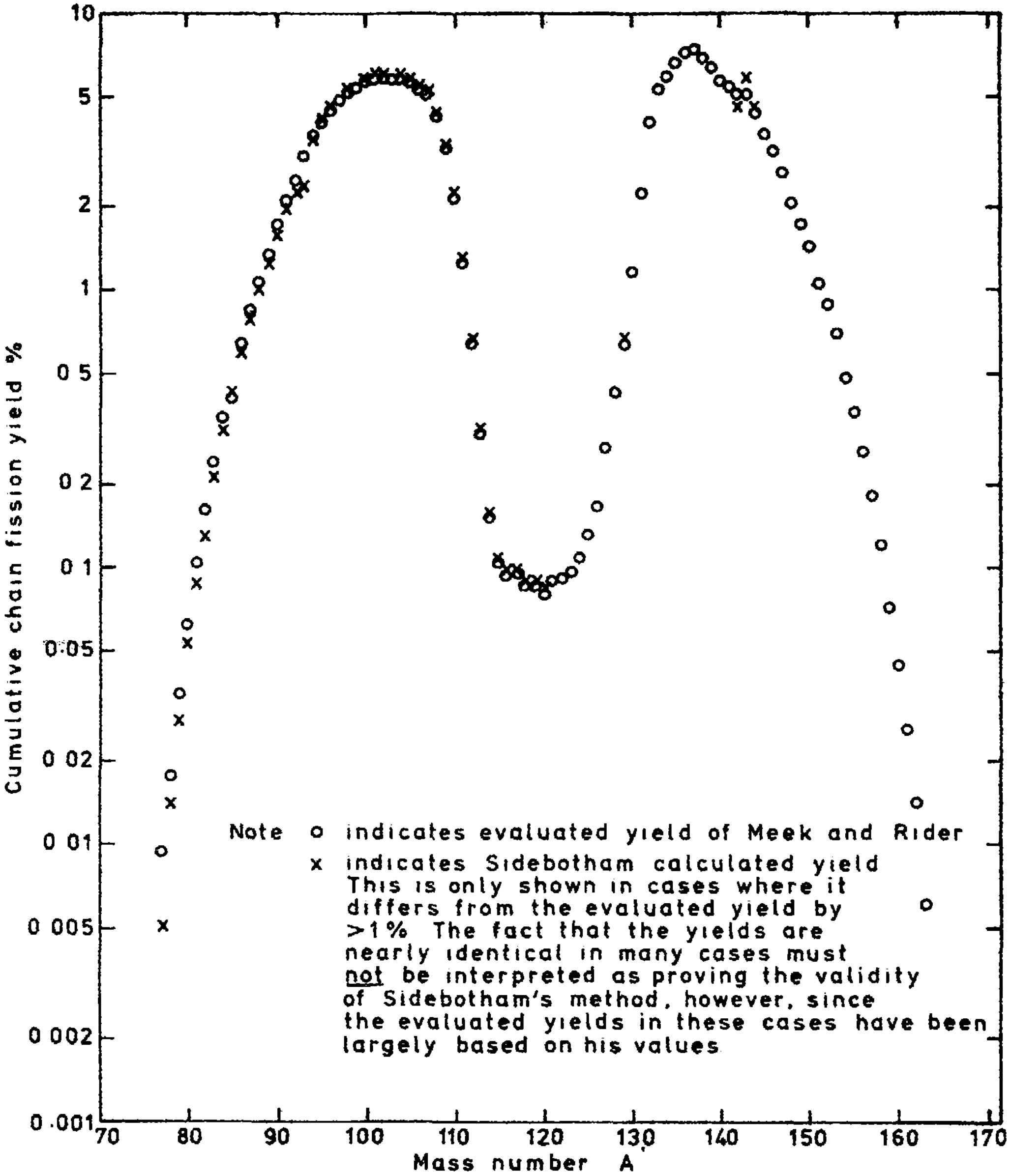


FIG. 12 ADJUSTED CHAIN FISSION YIELDS FOR ^{242}Pu FAST (PILE) FISSION

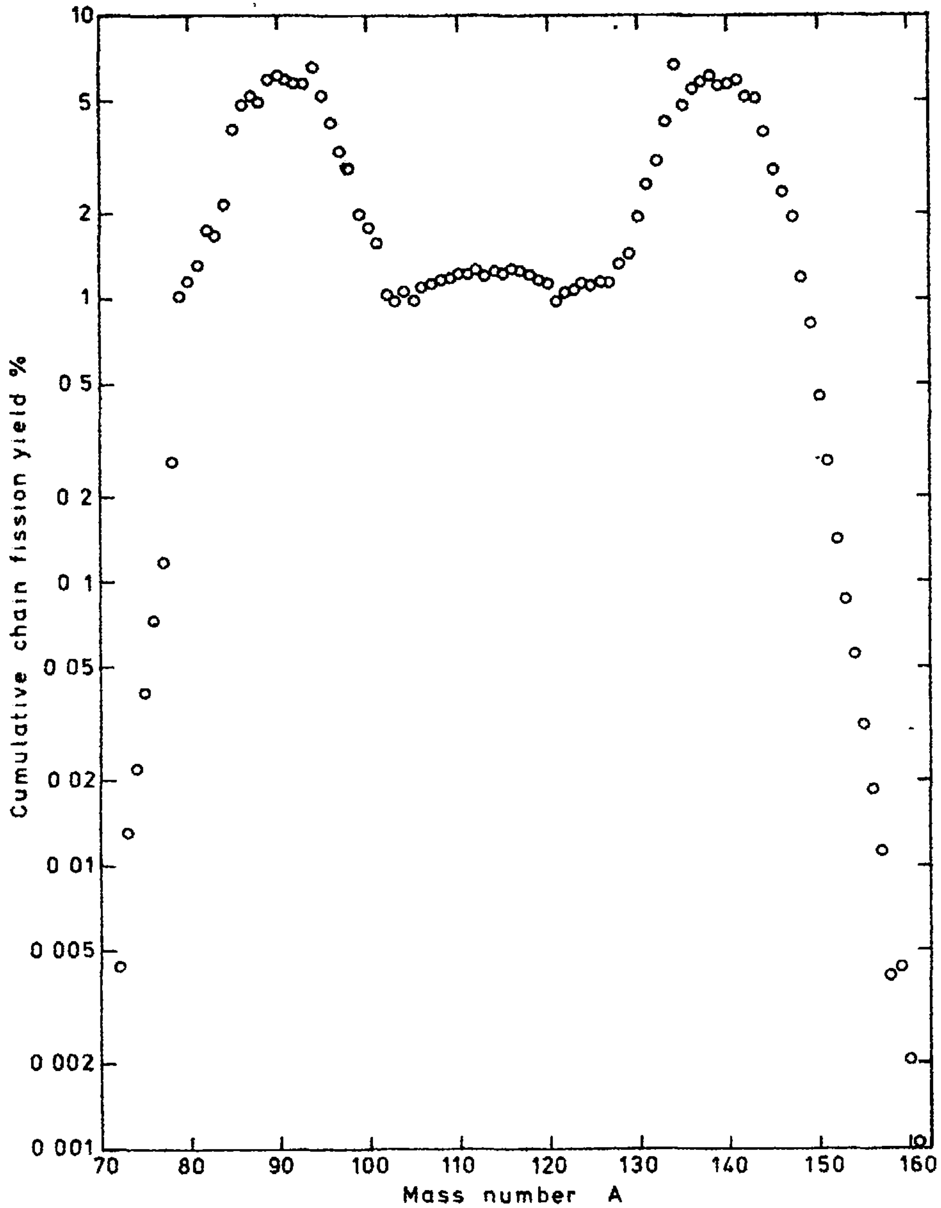


FIG 13 ADJUSTED CHAIN FISSION YIELDS FOR ^{232}Th 14 MEV FISSION.

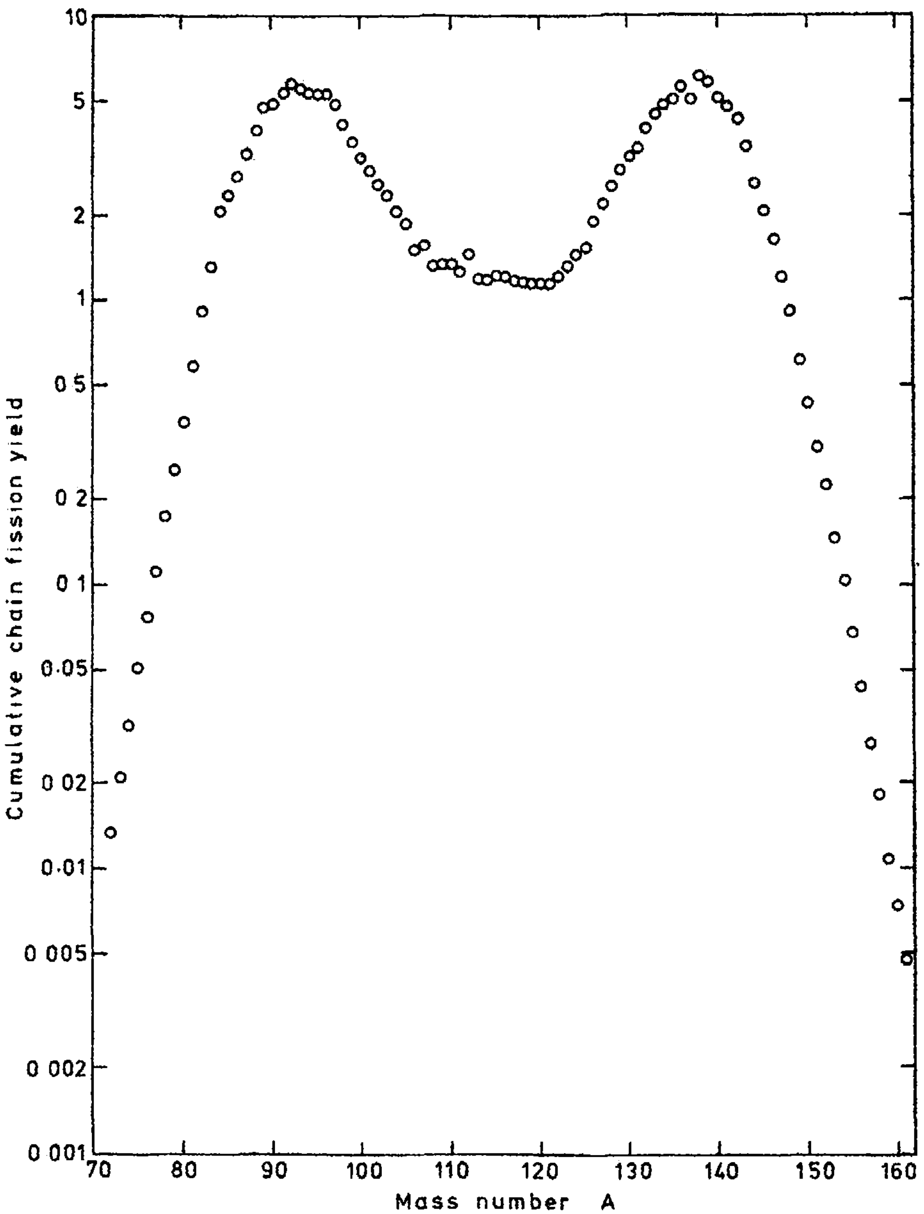


FIG 14 ADJUSTED CHAIN FISSION YIELDS FOR ^{233}U 14 MEV FISSION.

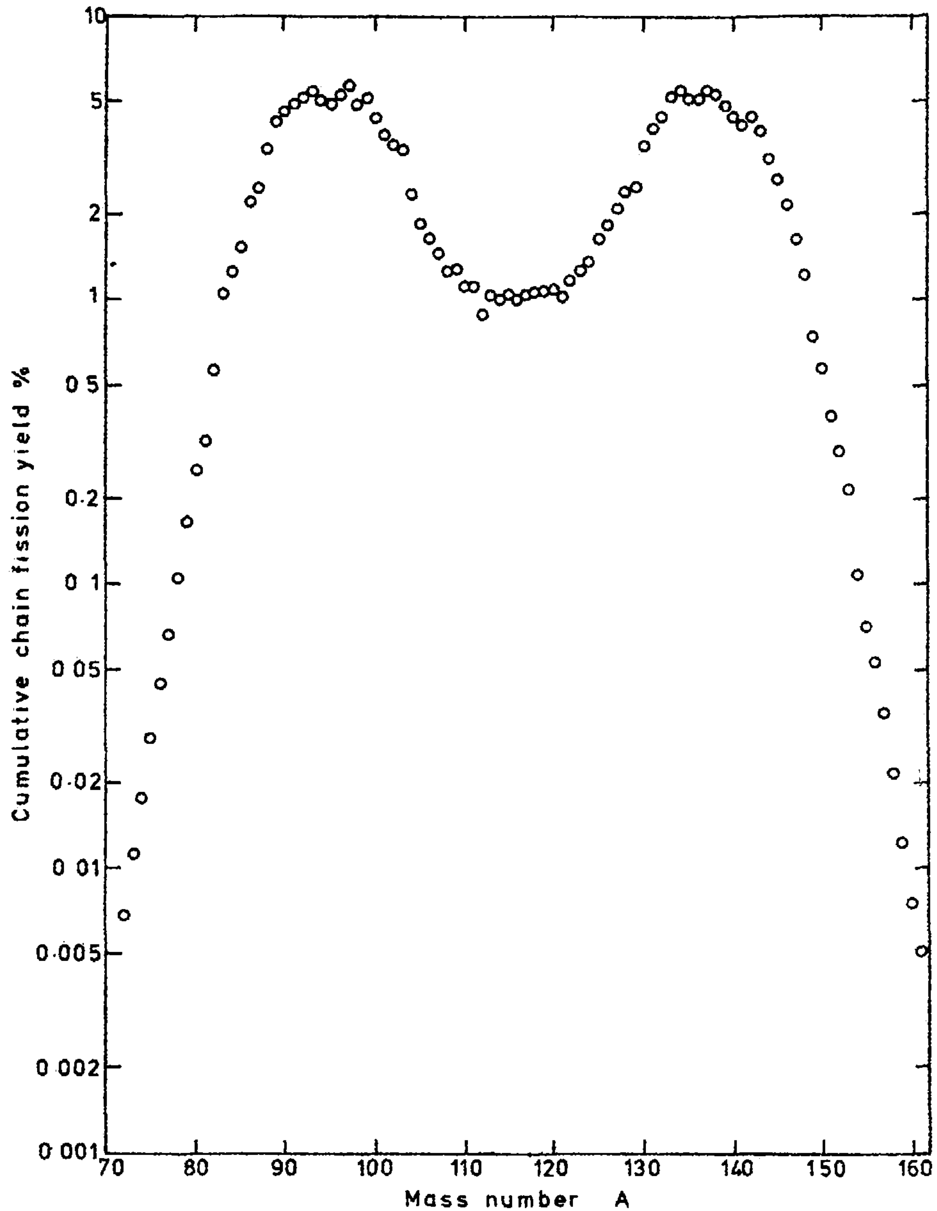


FIG 15 ADJUSTED CHAIN FISSION YIELDS FOR ^{235}U 14 MeV FISSION

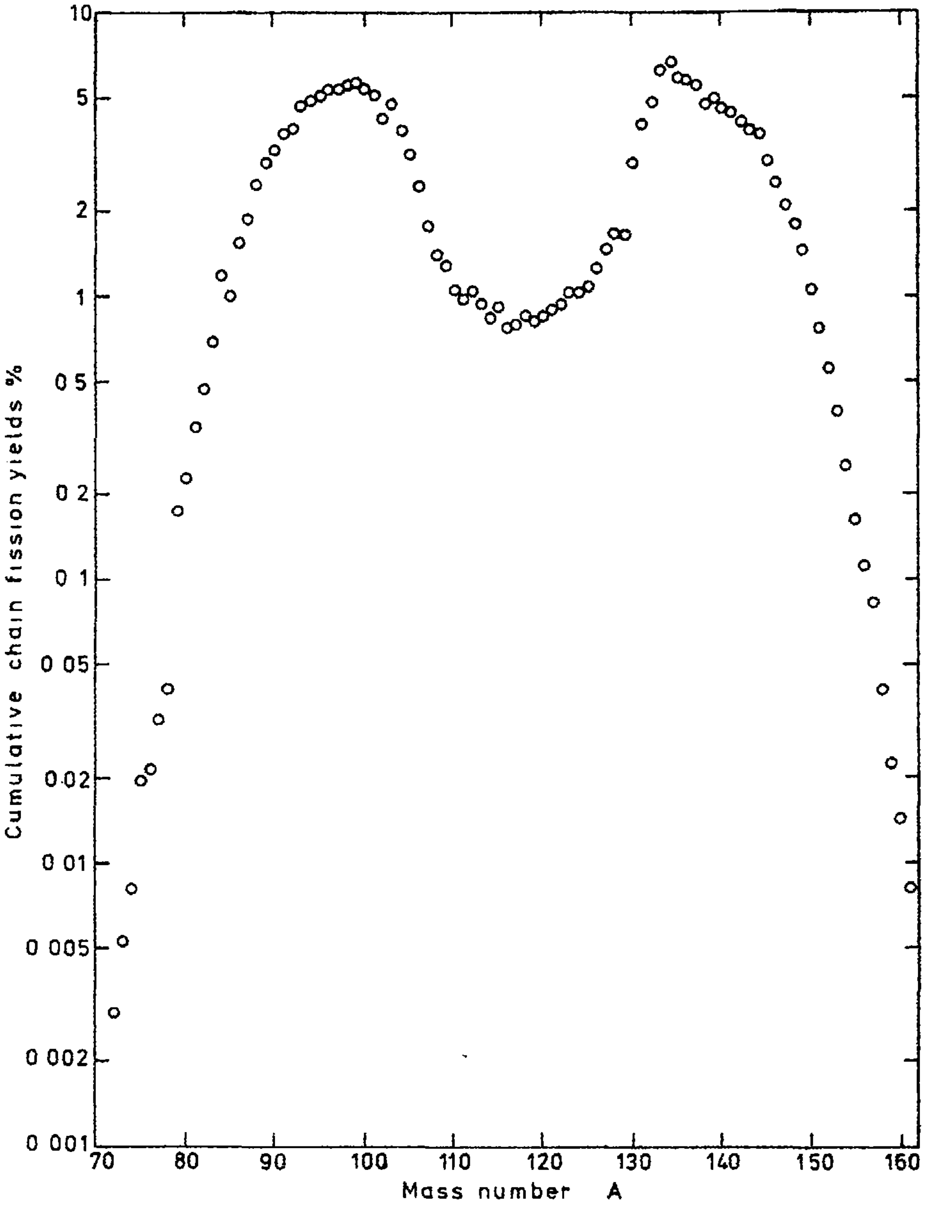


FIG 16. ADJUSTED CHAIN FISSION YIELDS FOR ^{238}U
14 MeV FISSION

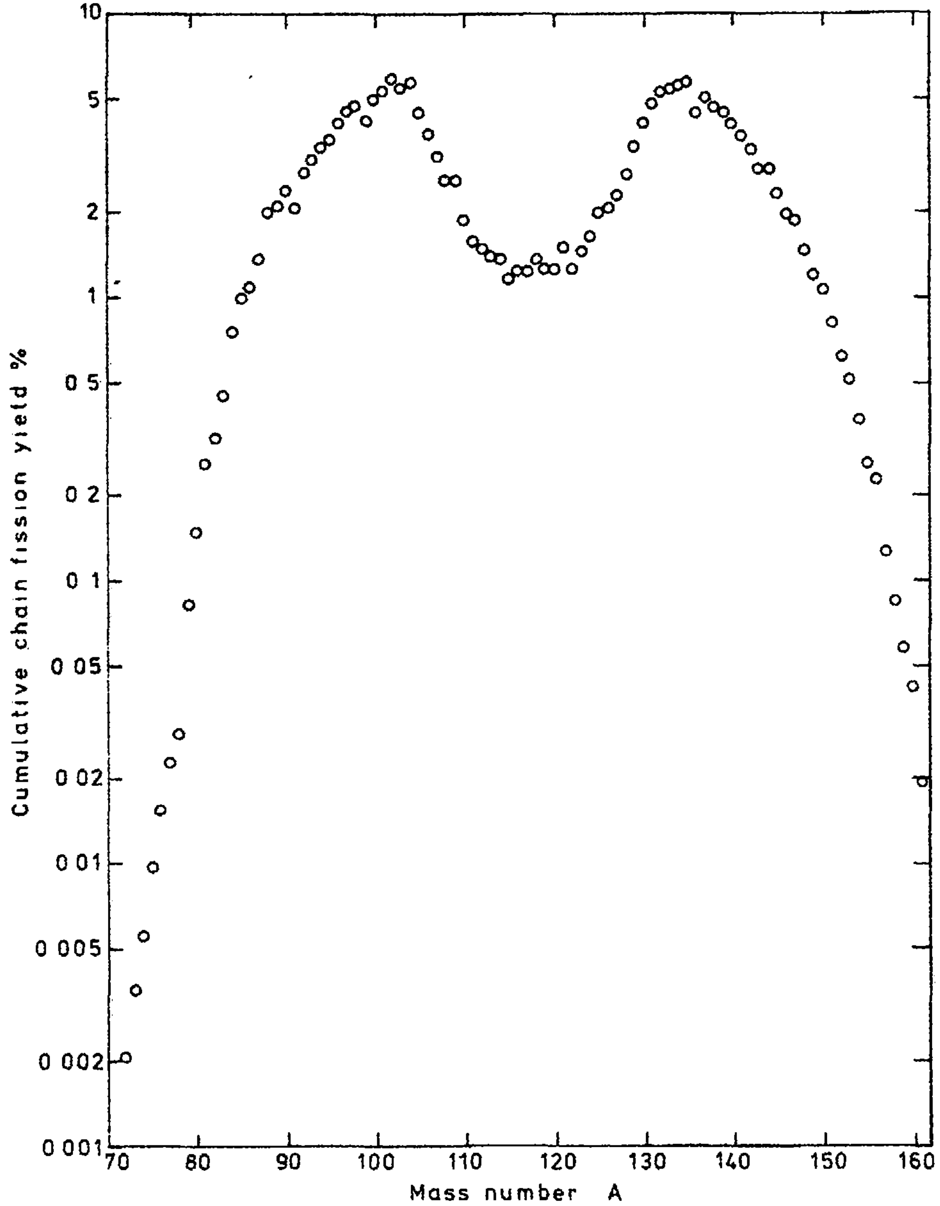


FIG 17 ADJUSTED CHAIN FISSION YIELDS FOR ^{239}Pu
14 MeV FISSION

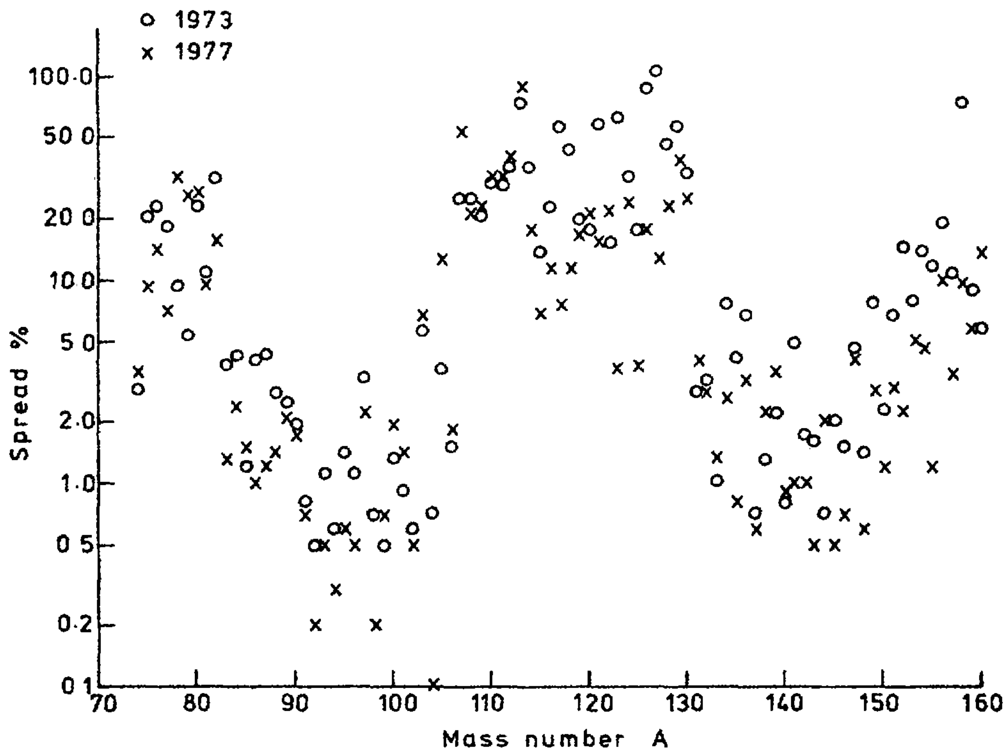


FIG. 18. SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YIELDS. ^{235}U THERMAL NEUTRON FISSION.

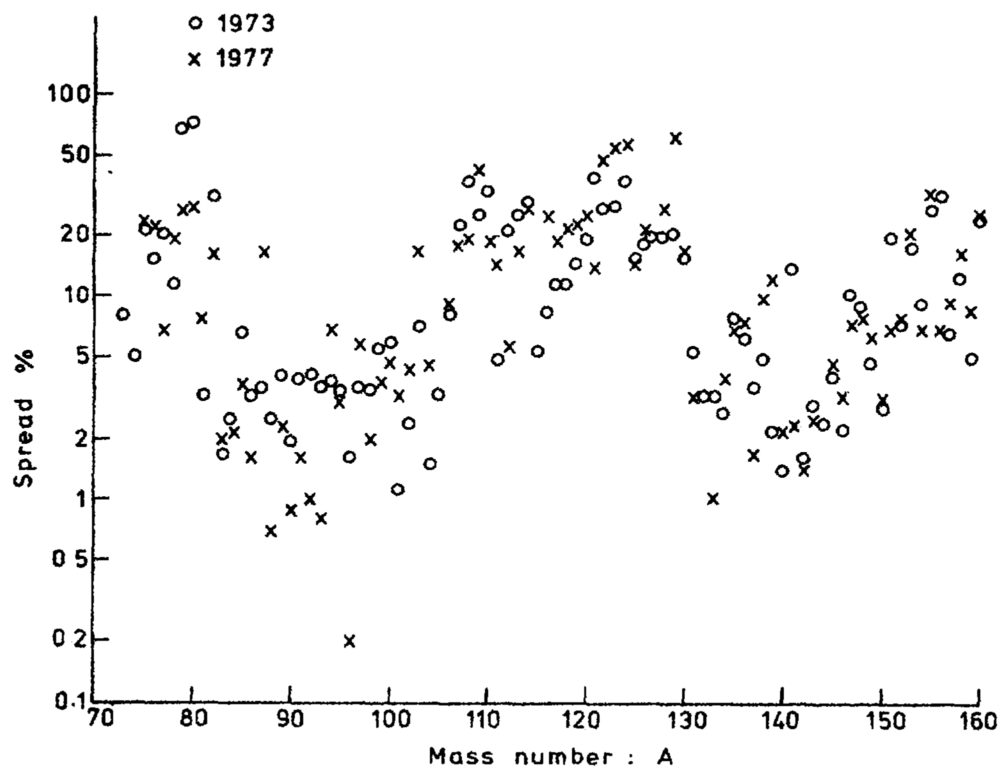


FIG. 19. SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YEILDS, ^{239}Pu THERMAL NEUTRON FISSION

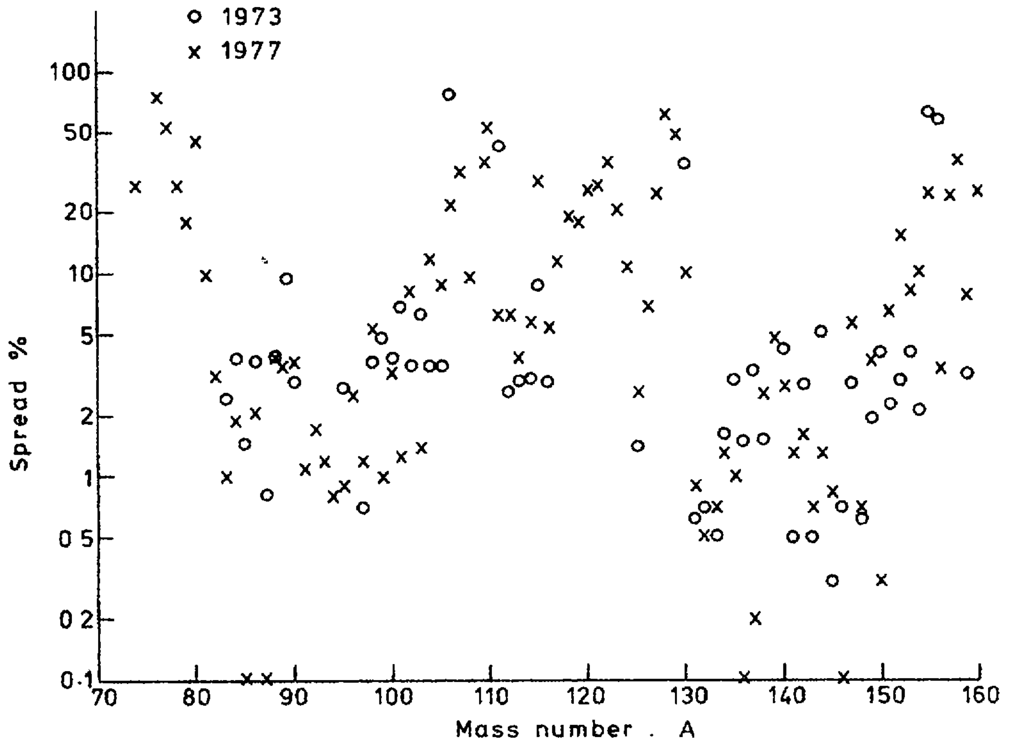


FIG 20 SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 ,AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YIELDS, ^{235}U FAST (PILE) NEUTRON FISSION

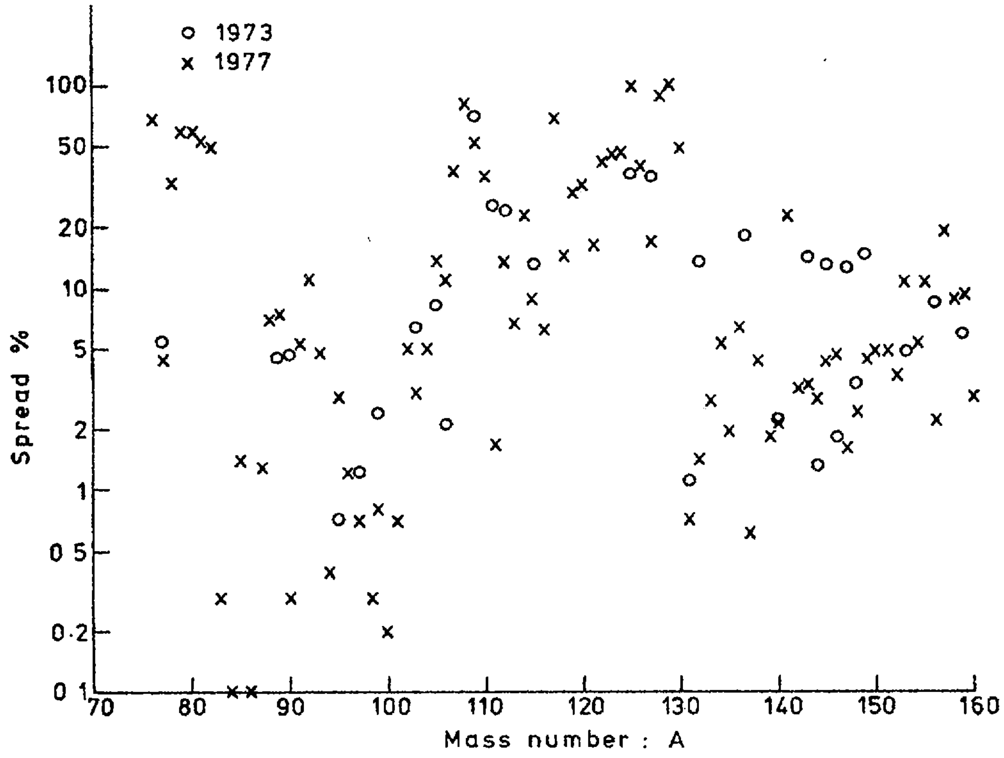


FIG. 21. SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YIELDS; ^{238}U FAST (PILE) NEUTRON FISSION.

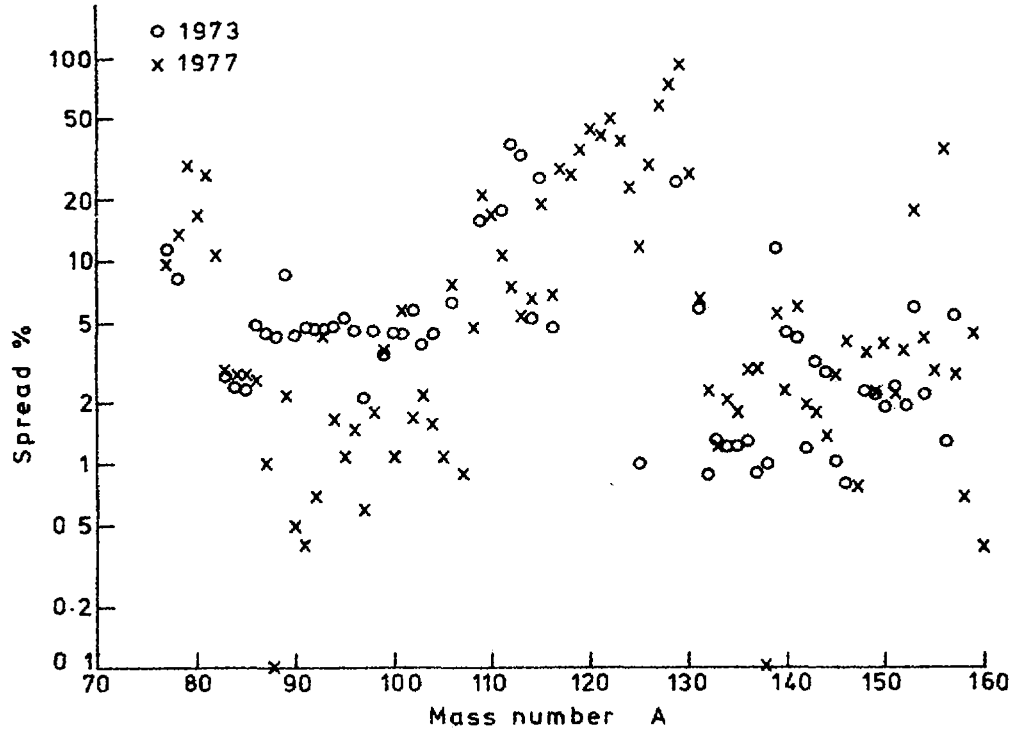


FIG 22 SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YIELDS, ^{239}Pu FAST (PILE) NEUTRON FISSION

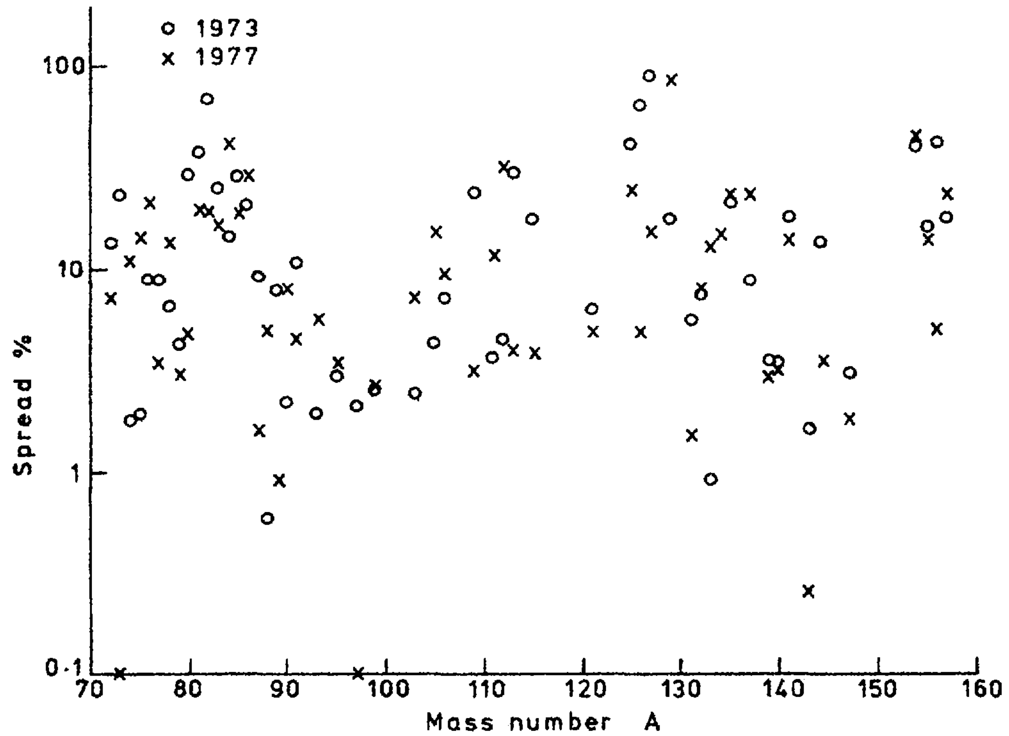


FIG 23 SPREAD OF EVALUATED FISSION YIELDS CONSIDERED IN 1973 AND 1977, EXPRESSED AS A PERCENTAGE OF THE RECOMMENDED YIELDS ^{235}U 14 MeV NEUTRON FISSION

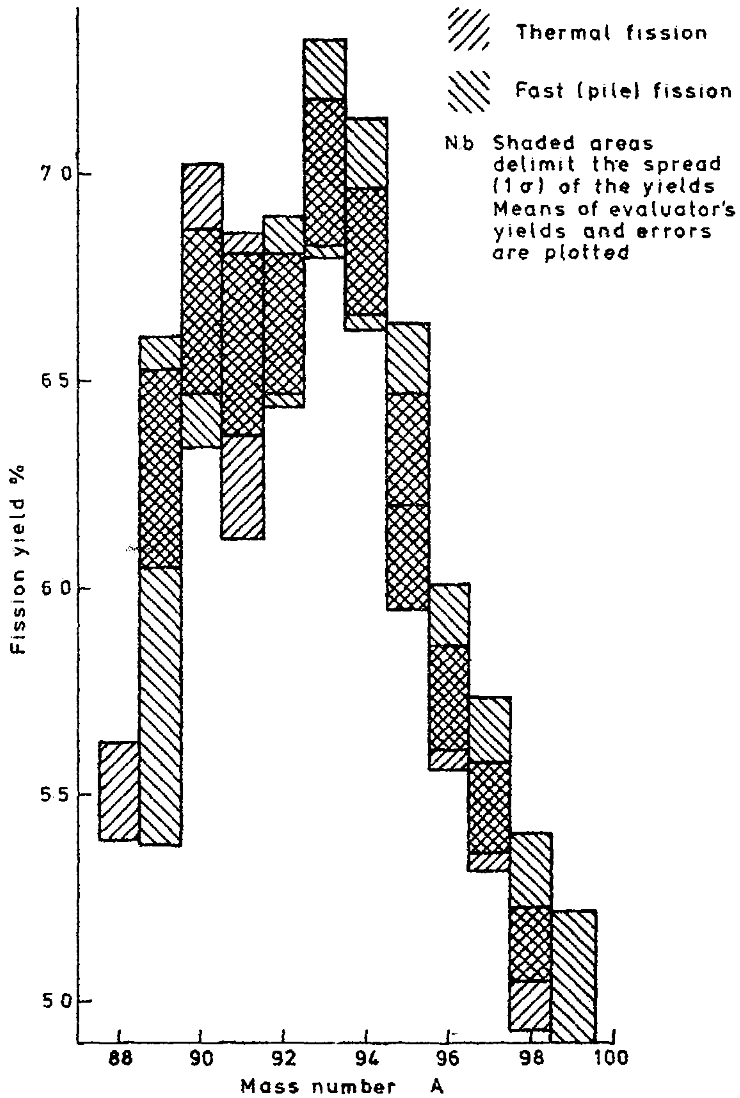


FIG 24. COMPARISON OF LIGHT PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{233}U

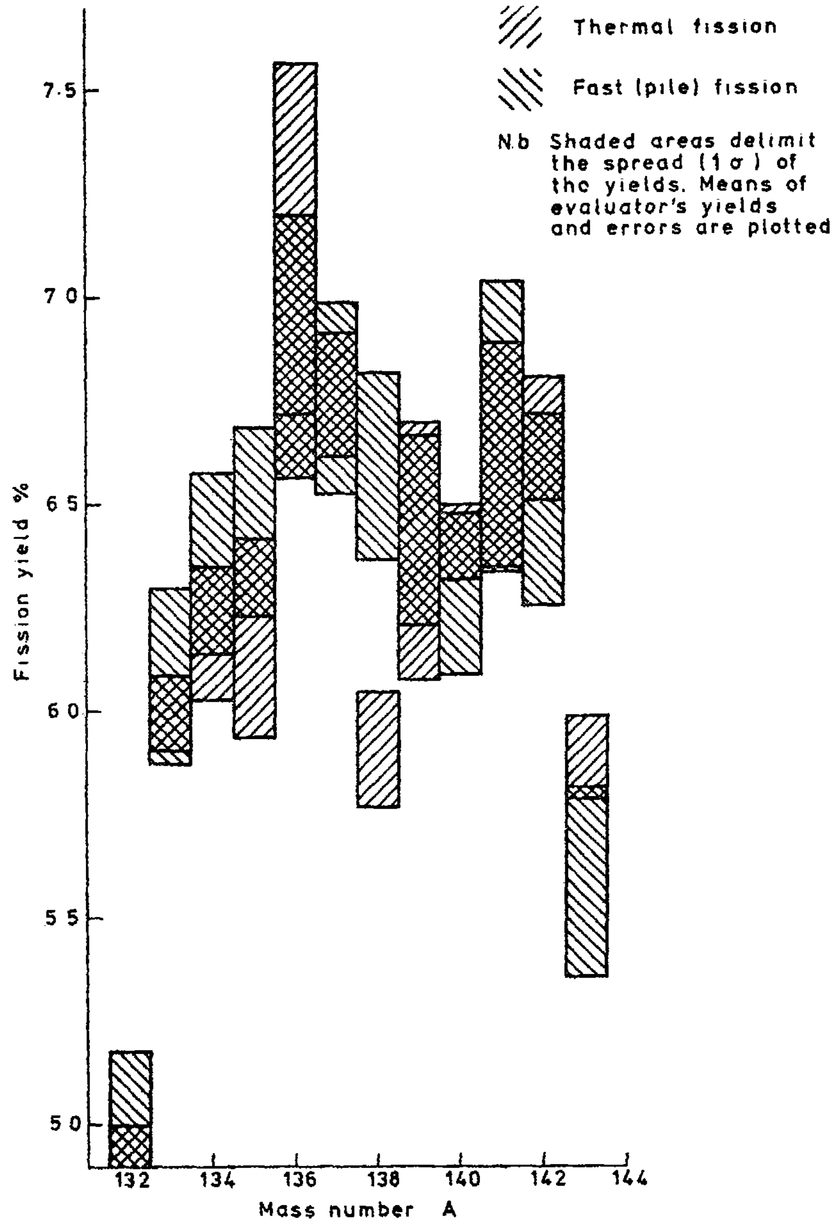


FIG 25 COMPARISON OF HEAVY PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{233}U

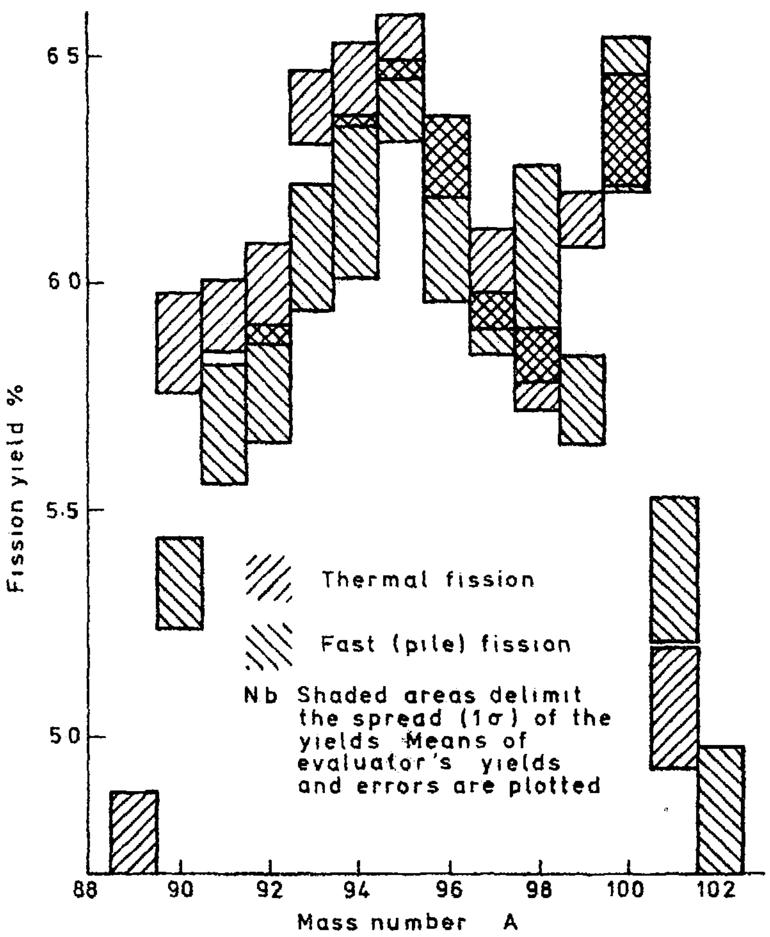


FIG 26 COMPARISON OF LIGHT PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{235}U

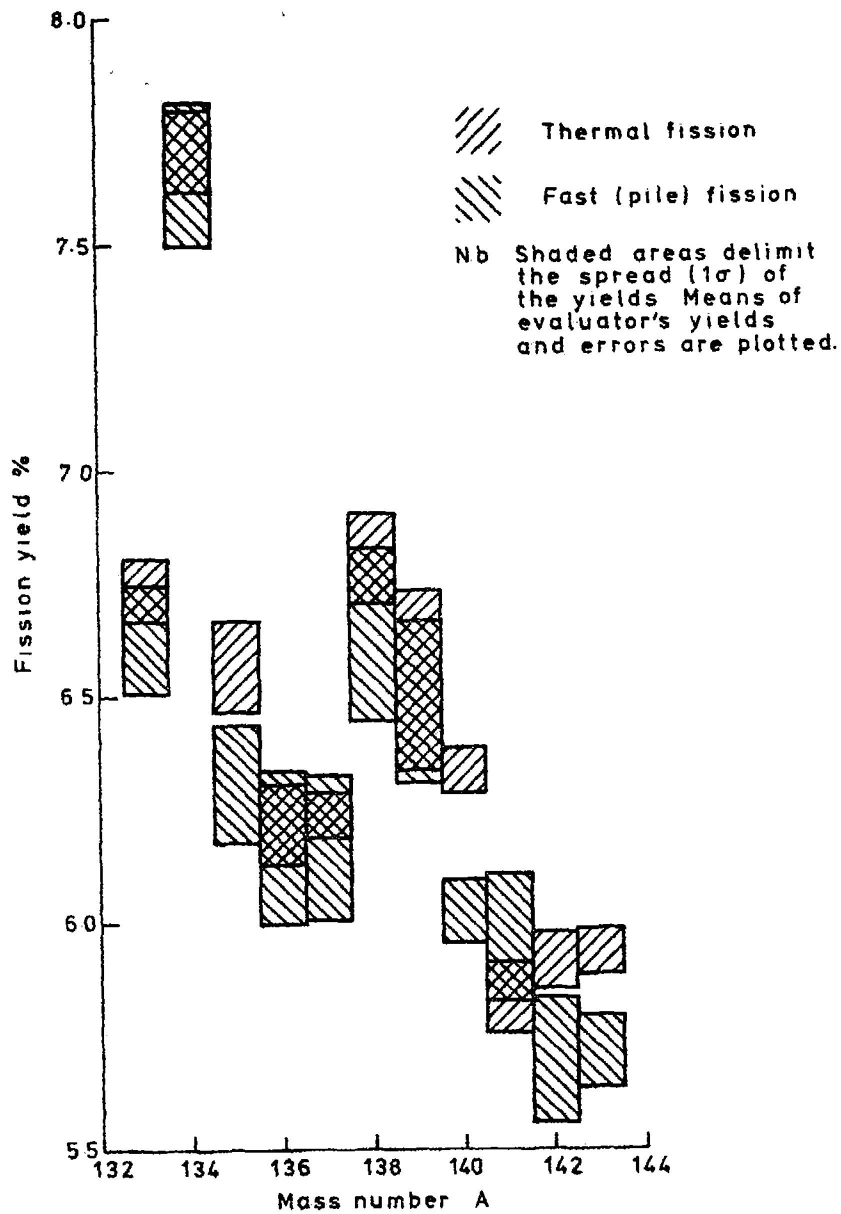


FIG 27 COMPARISON OF HEAVY PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{235}U

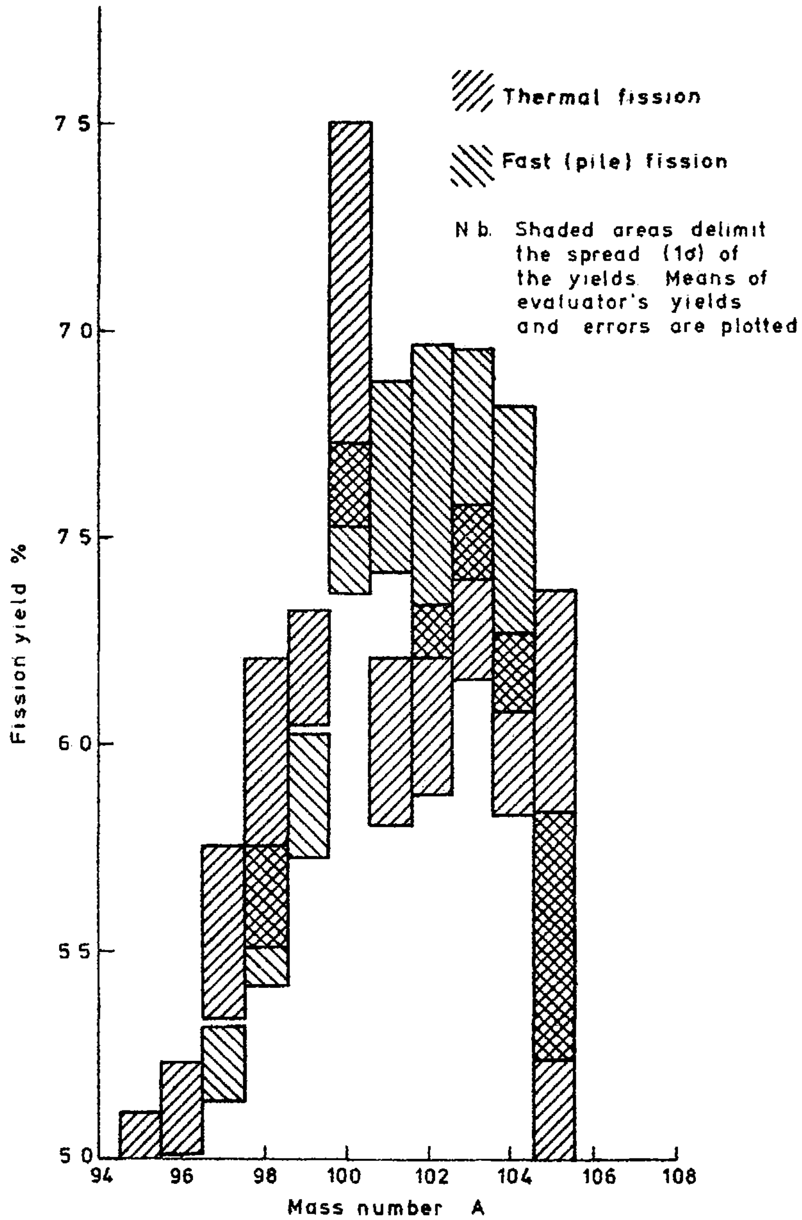


FIG 28 COMPARISON OF LIGHT PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{239}Pu

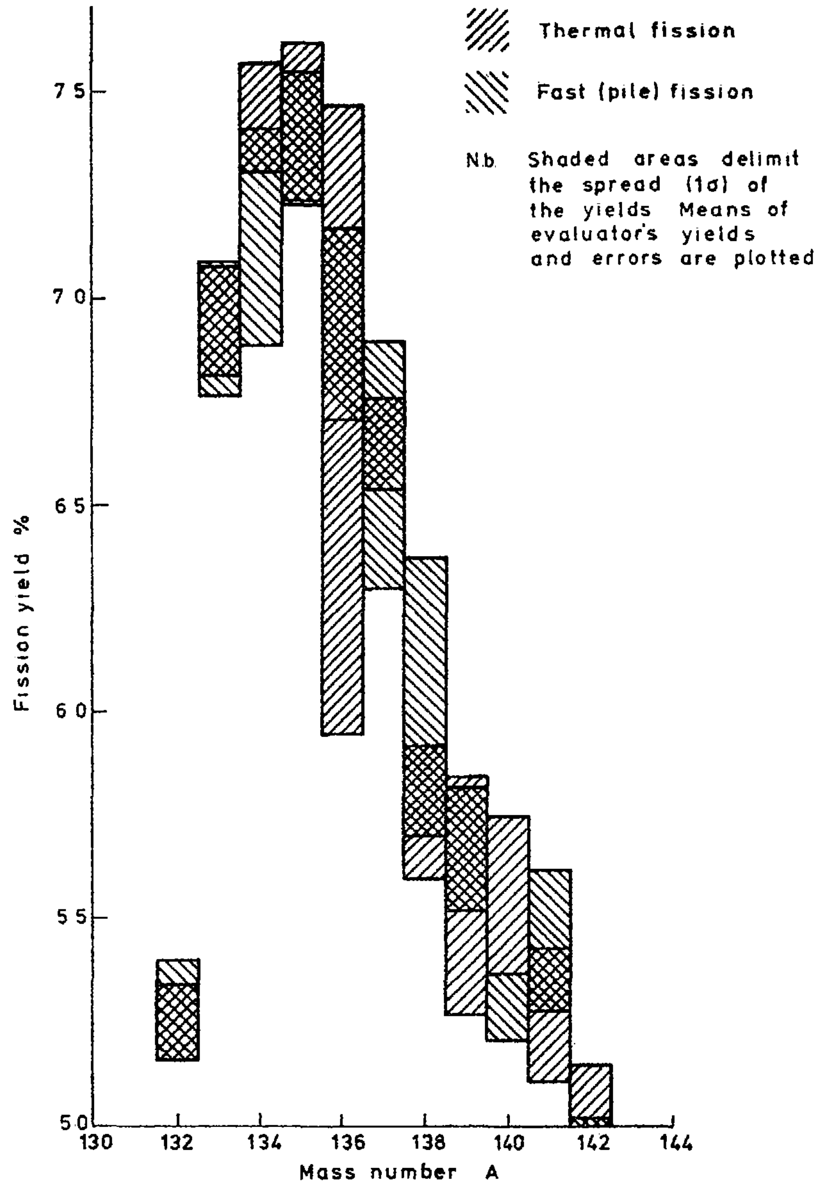


FIG 29 COMPARISON OF HEAVY PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{239}Pu

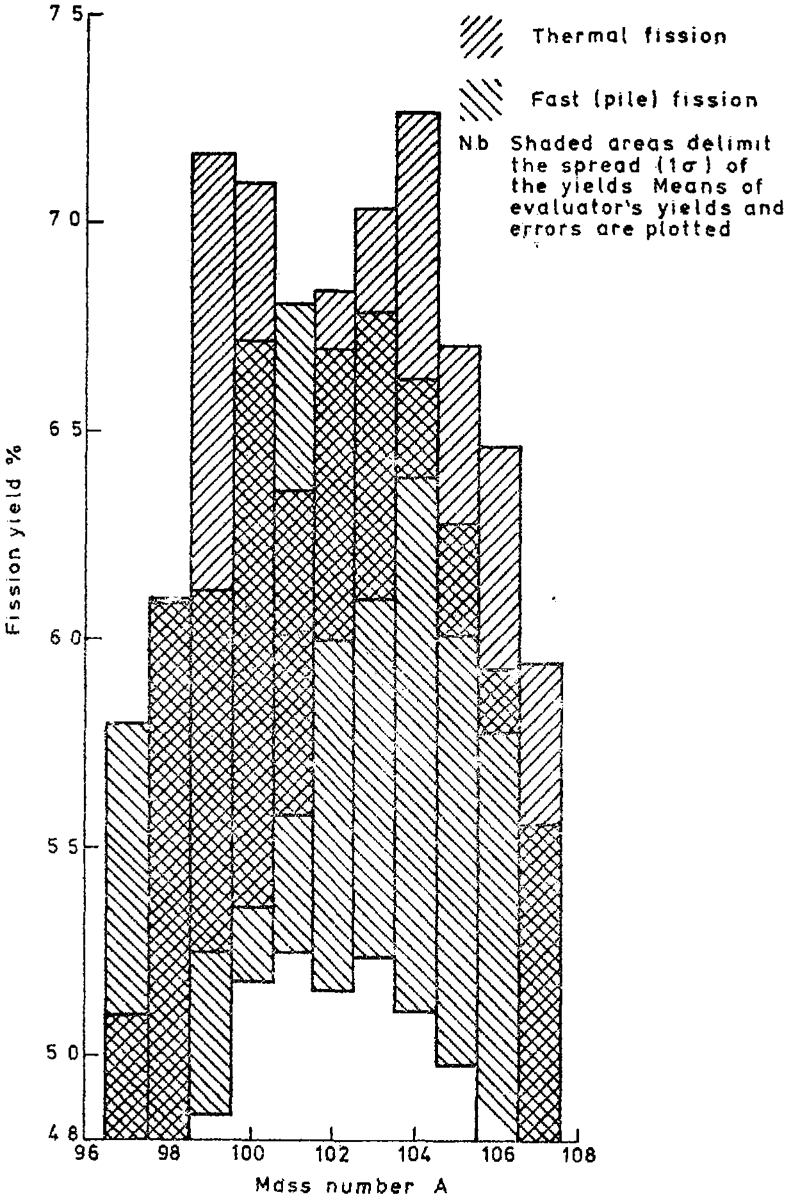


FIG 30 COMPARISON OF LIGHT PEAK FISSION YIELDS FOR THERMAL AND FAST (PILE) FISSION OF ^{241}Pu

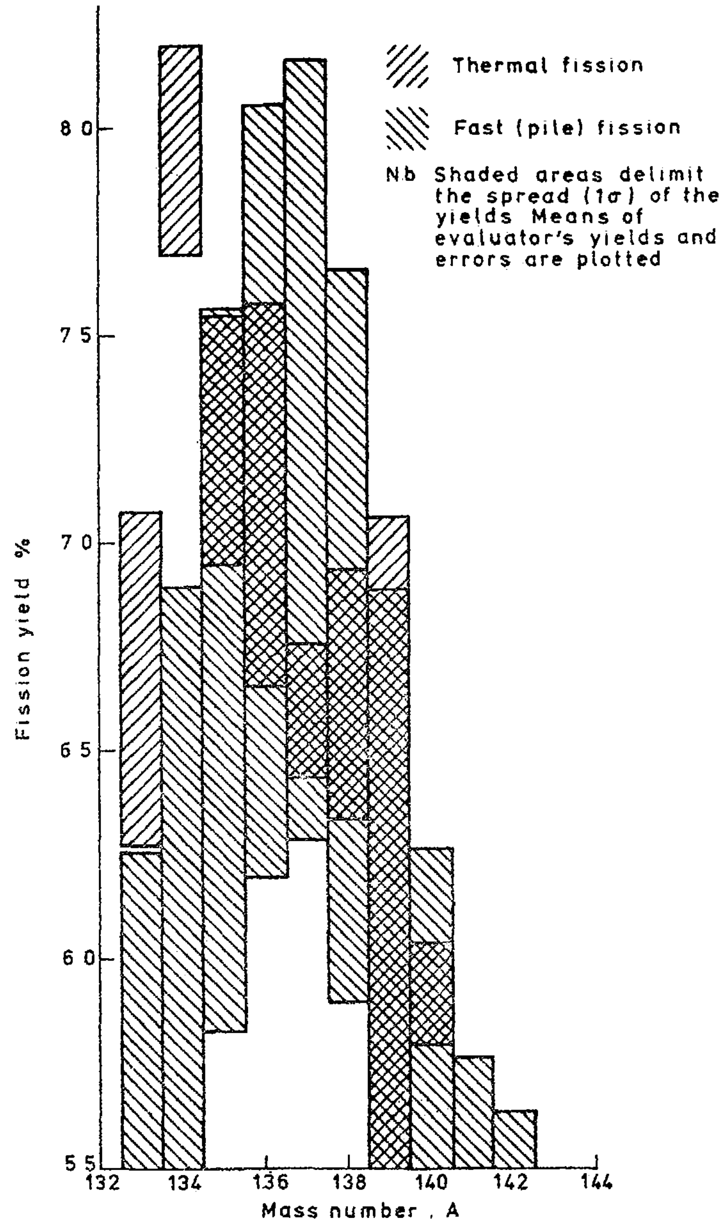


FIG 31 COMPARISON OF HEAVY PEAK FISSION YIELD FOR THERMAL AND FAST (PILE) FISSION OF ^{241}Pu

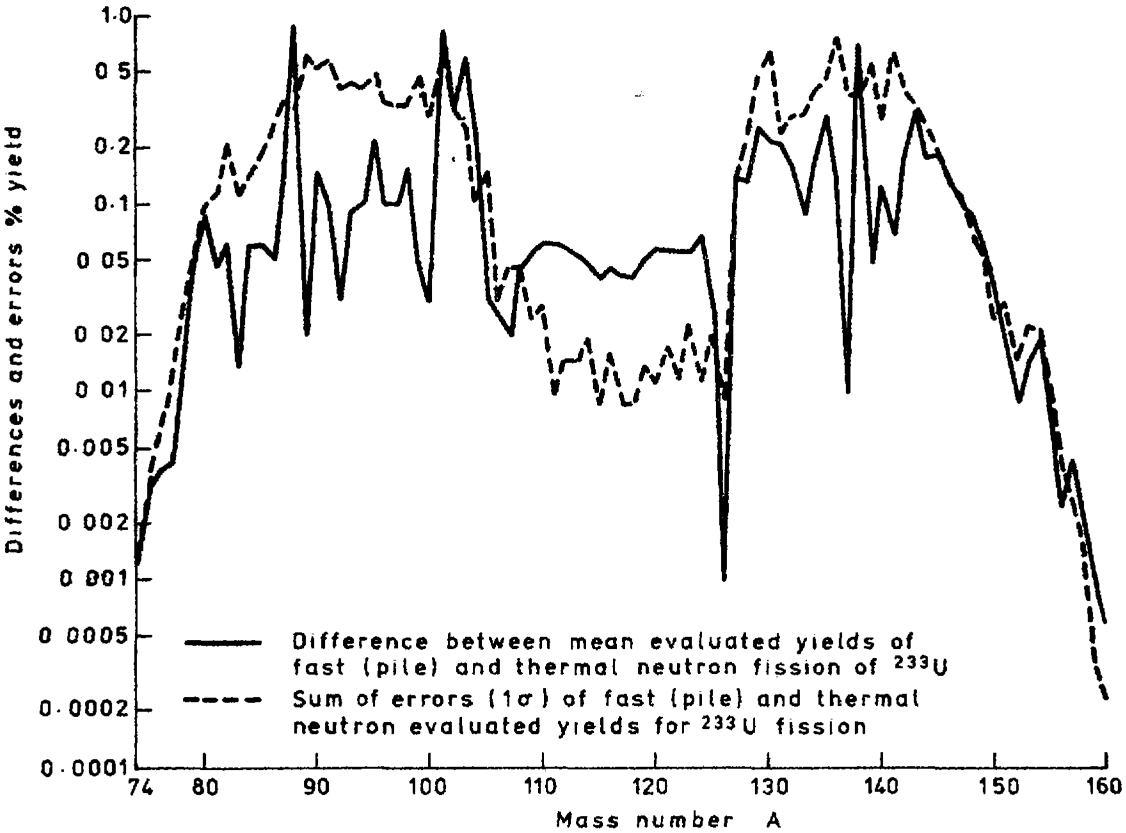


FIG. 32. COMPARISON OF THE DIFFERENCES BETWEEN THERMAL AND FAST (PILE) NEUTRON FISSION YIELDS WITH THE ERRORS ON THE YIELDS ^{233}U FISSION

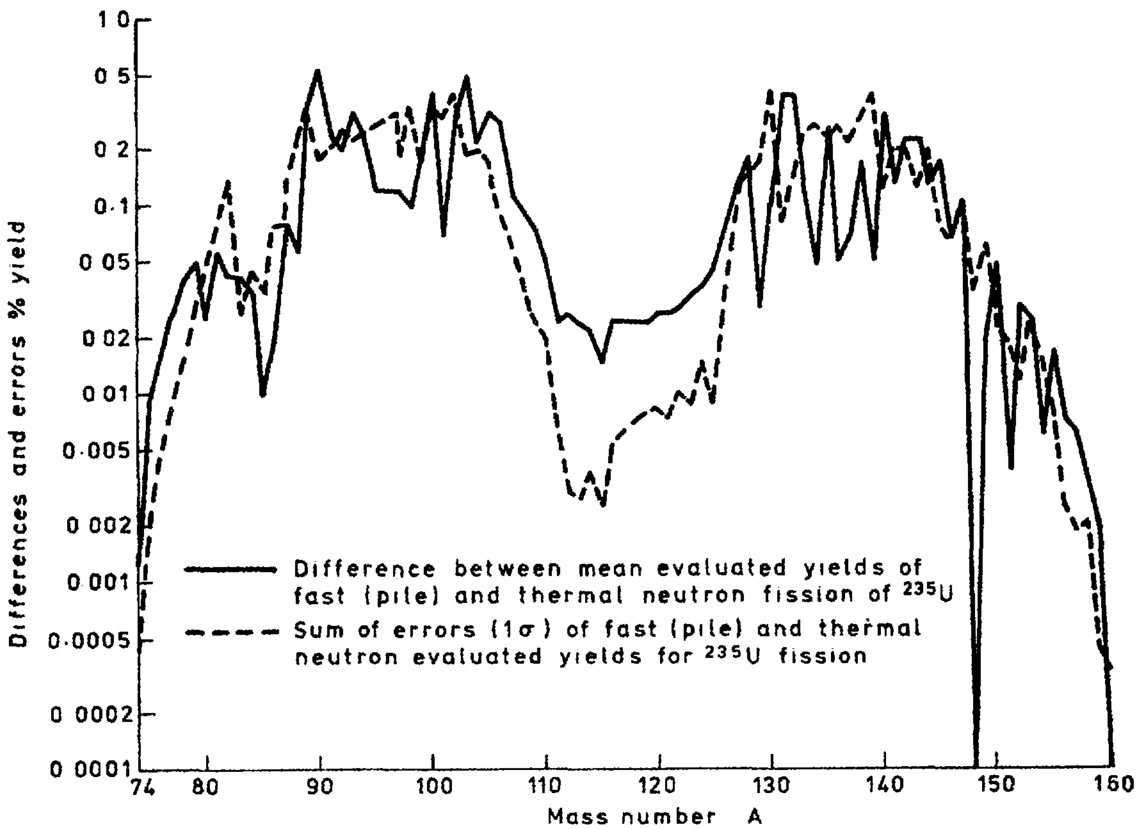


FIG 33. COMPARISON OF THE DIFFERENCES BETWEEN THERMAL AND FAST (PILE) NEUTRON FISSION YIELDS WITH THE ERRORS ON THE YIELDS ^{235}U FISSION

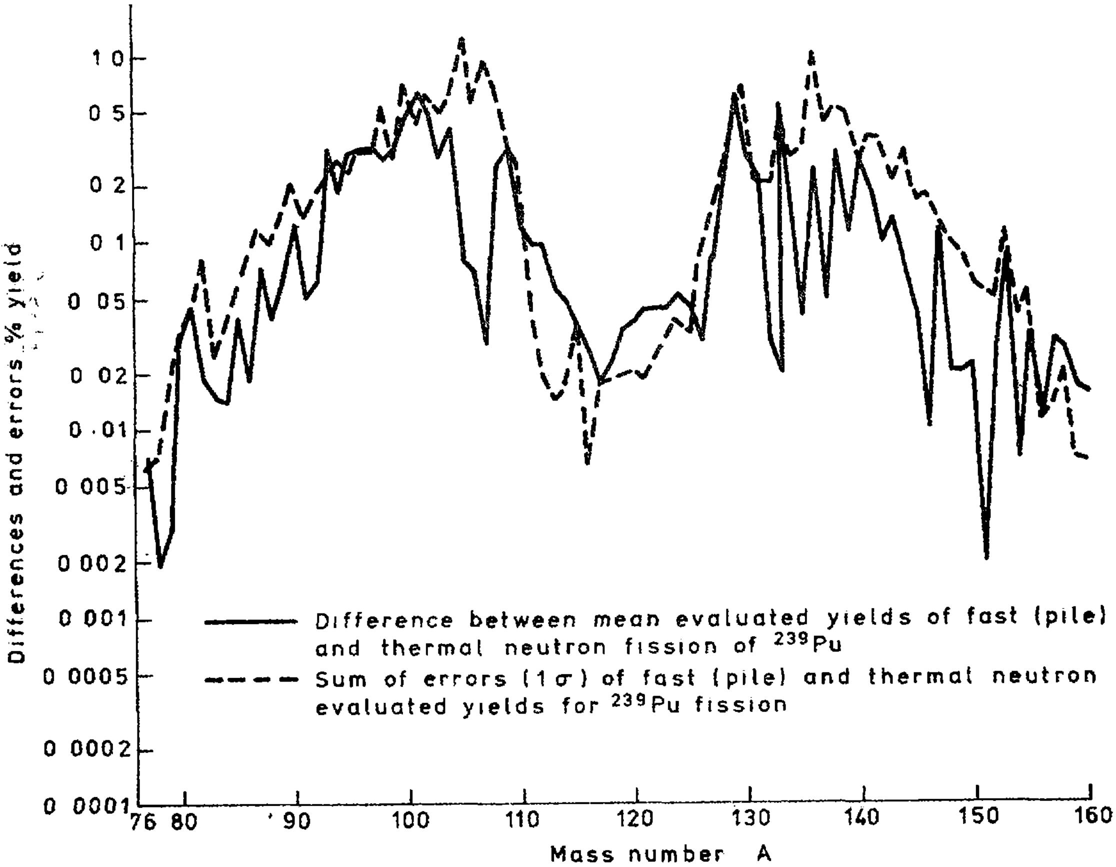


FIG 34 COMPARISON OF THE DIFFERENCES BETWEEN THERMAL AND FAST (PILE) NEUTRON FISSION YIELDS WITH THE ERRORS ON THE YIELDS ^{239}Pu FISSION

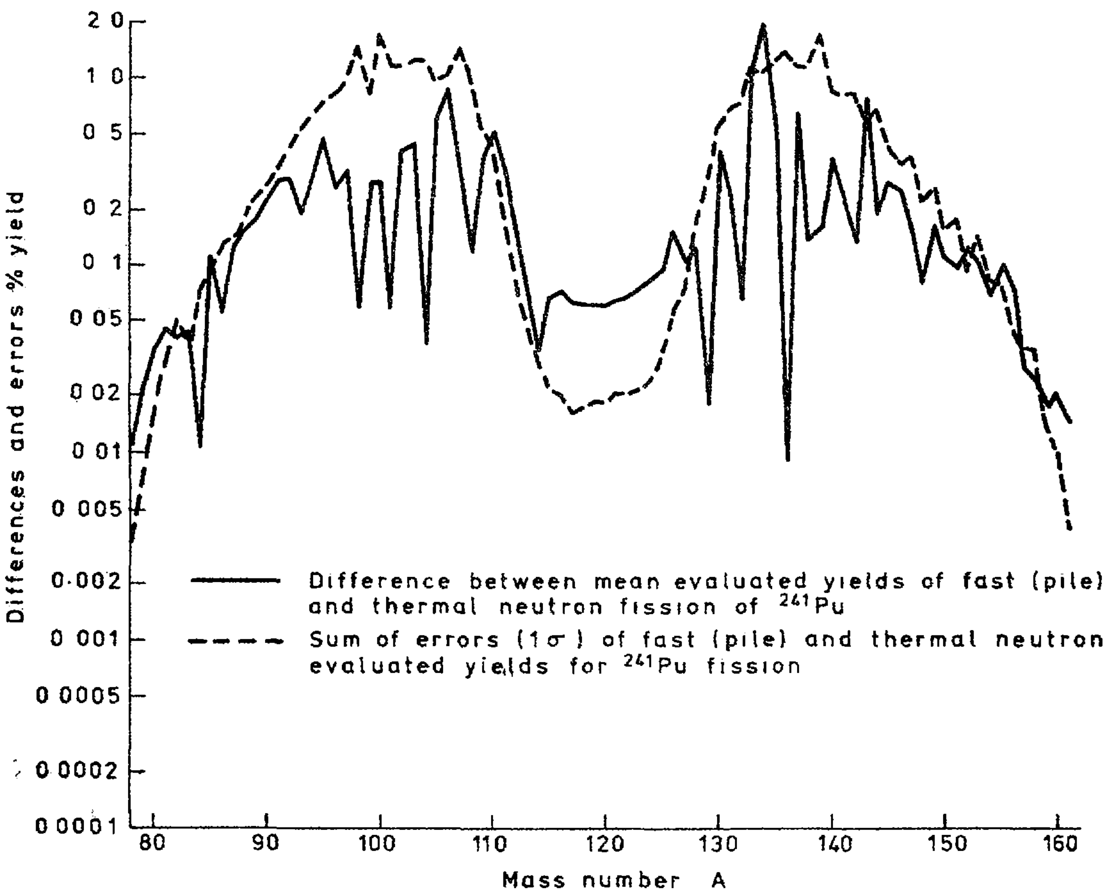


FIG 35 COMPARISON OF THE DIFFERENCES BETWEEN THERMAL AND FAST (PILE) NEUTRON FISSION YIELDS WITH THE ERRORS ON THE YIELDS ^{241}Pu FISSION

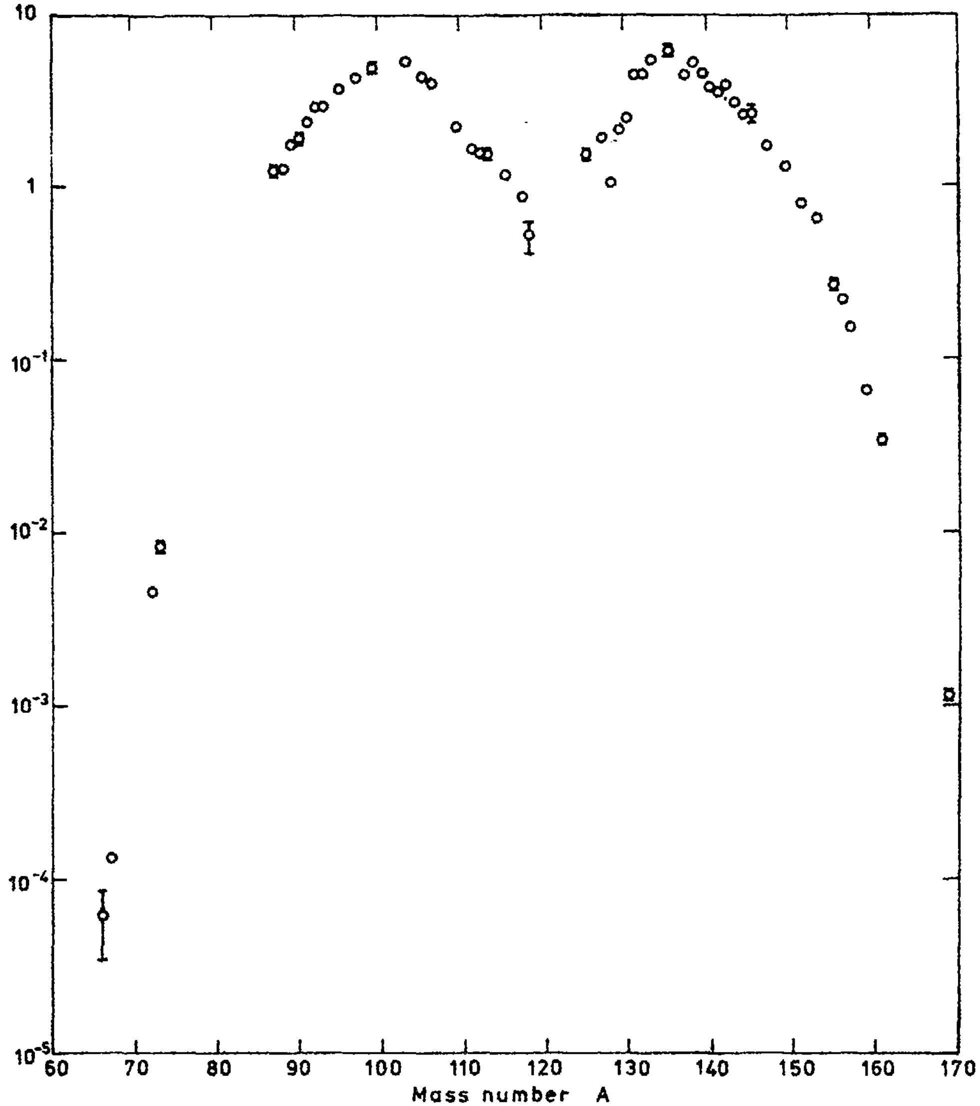


FIG. 36 MASS-YIELD CURVE FOR ^{240}Pu 14.8 MeV NEUTRON FISSION (REF 18)

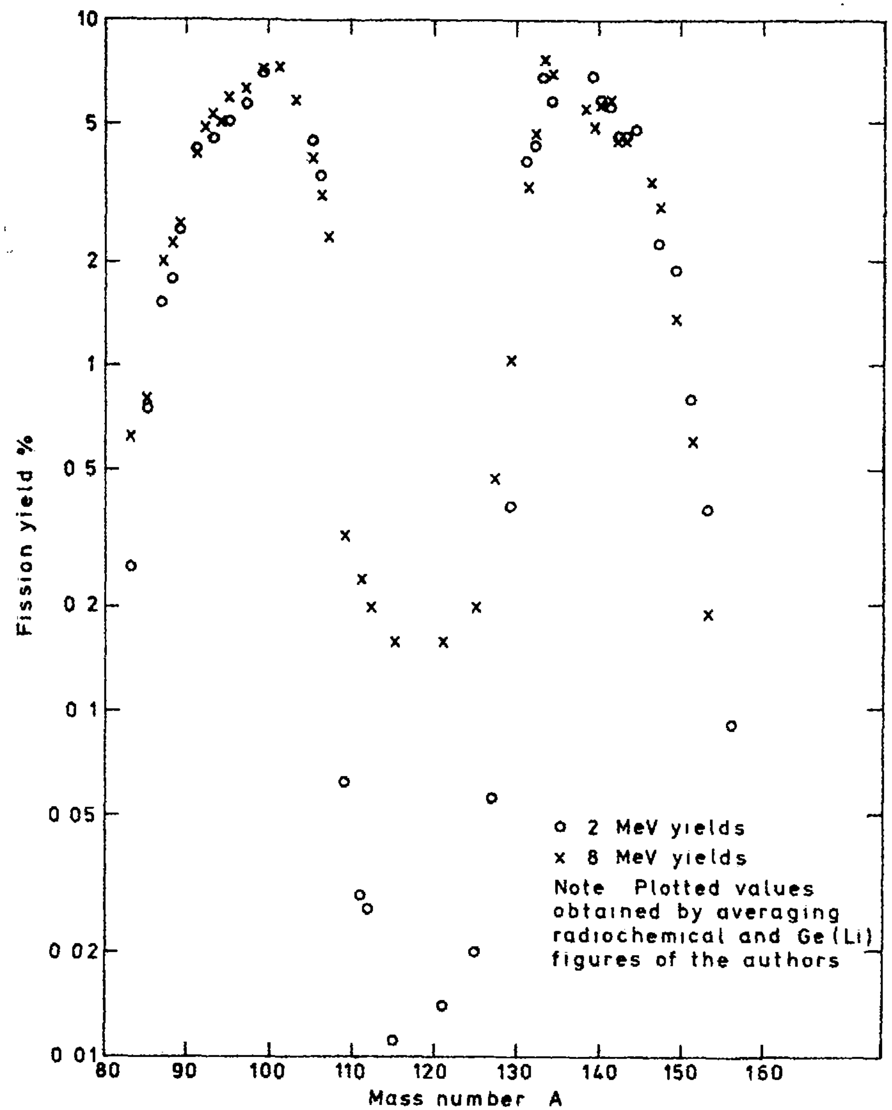


FIG 37 MASS-YIELD CURVES FOR FISSION OF ^{238}U BY 2 AND 8 MeV NEUTRONS (REF 21)

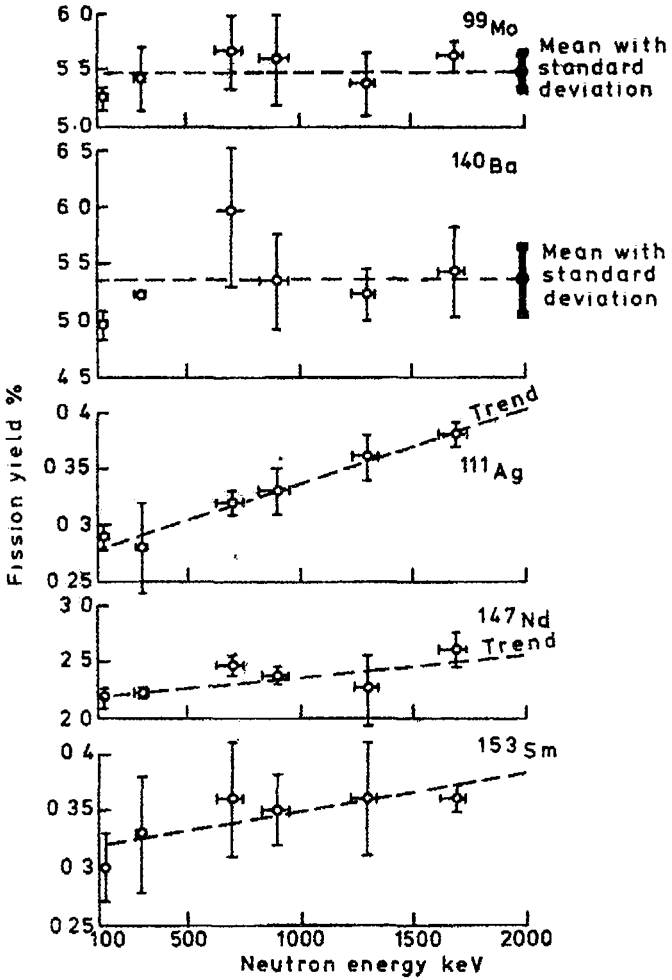


FIG 38. ABSOLUTE FISSION YIELDS FOR ^{239}Pu FISSION BY MONOENERGETIC NEUTRONS (REF 22)

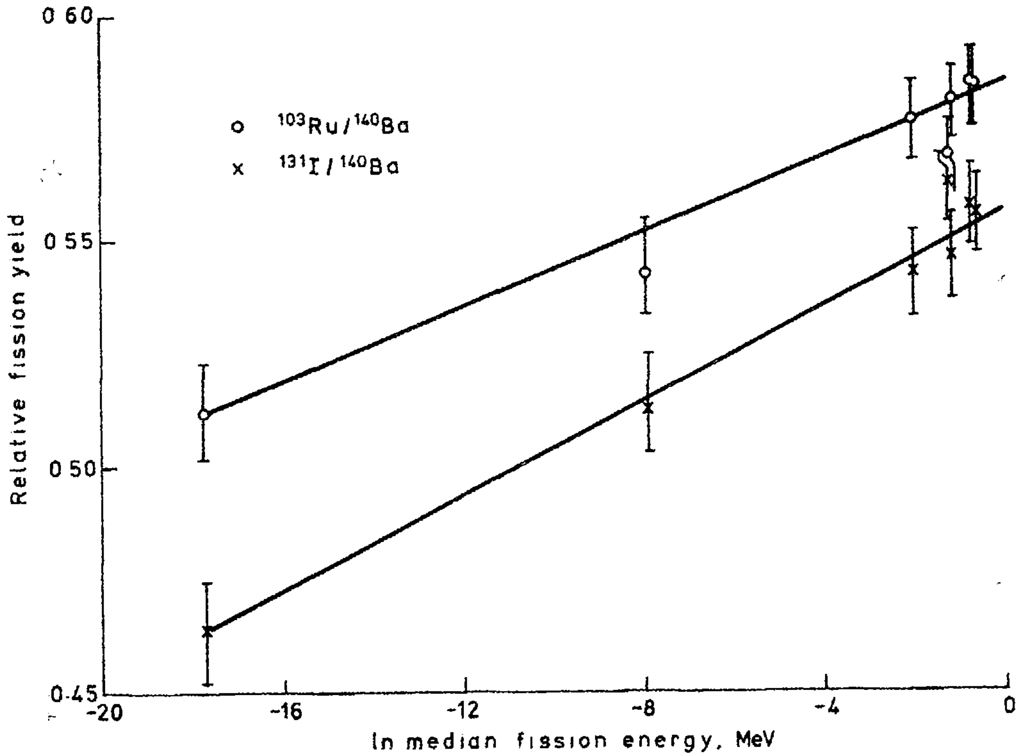


FIG 39 EFFECT OF CHANGE OF MEDIAN NEUTRON ENERGY ON SOME FISSION YIELDS FOR ^{235}U (REF 25)

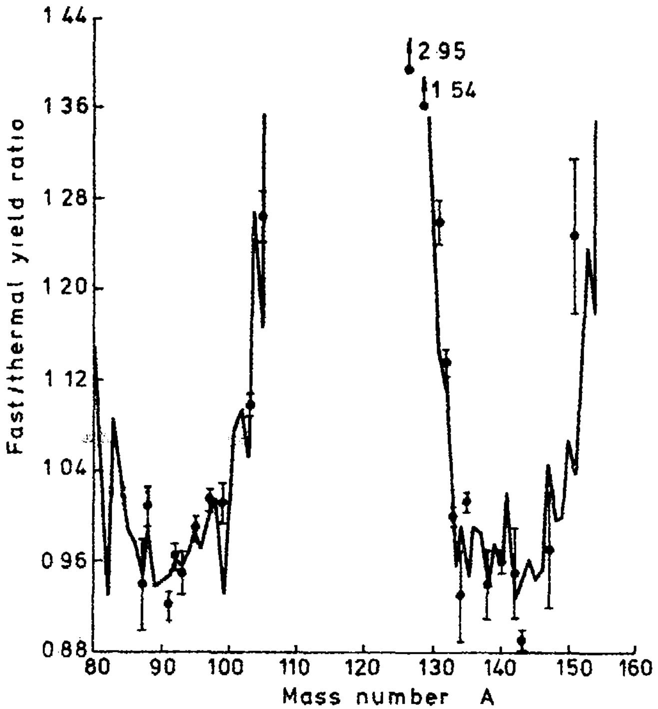


FIG 40 RATIO OF YIELDS FOR ^{235}U FISSIONED BY VERY HARD NEUTRONS (^{252}Cf SOURCE) AND BY THERMAL NEUTRONS THE CONTINUOUS LINE SHOWS THE RATIOS TAKEN FROM MEEK AND RIDER'S EARLIER EVALUATION (REF 26,27)

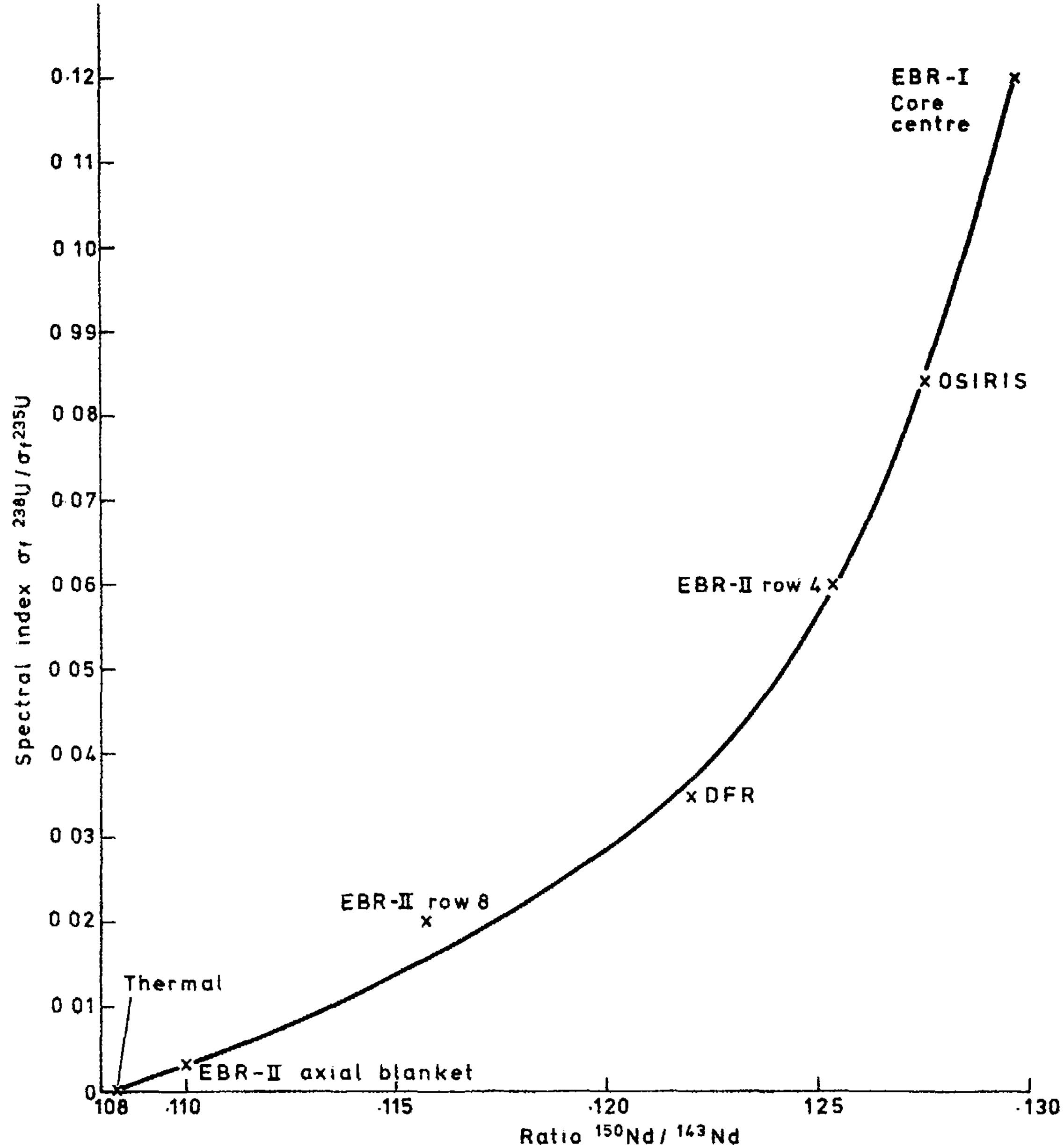
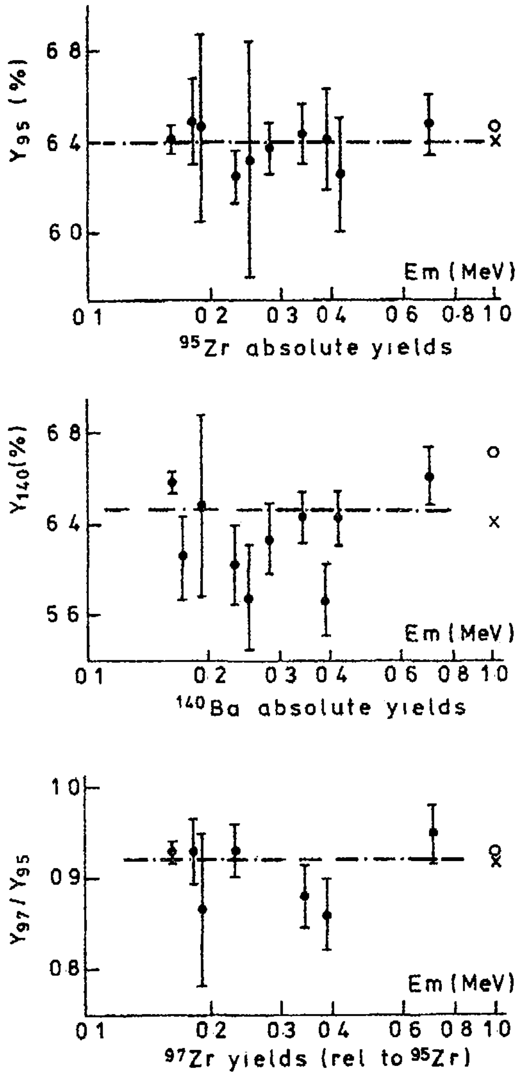


FIG 41 EFFECT OF CHANGE OF NEUTRON SPECTRUM ON THE YIELD RATES $^{150}Nd/^{143}Nd$ (REF 30)



Key: experimental data
x fast yield recommended in ref 27
o thermal yield recommended in ref 27
—·—·— weighted average

FIG 42 ^{235}U FAST FISSION YIELDS AS A FUNCTION OF ENERGY (REF 31)

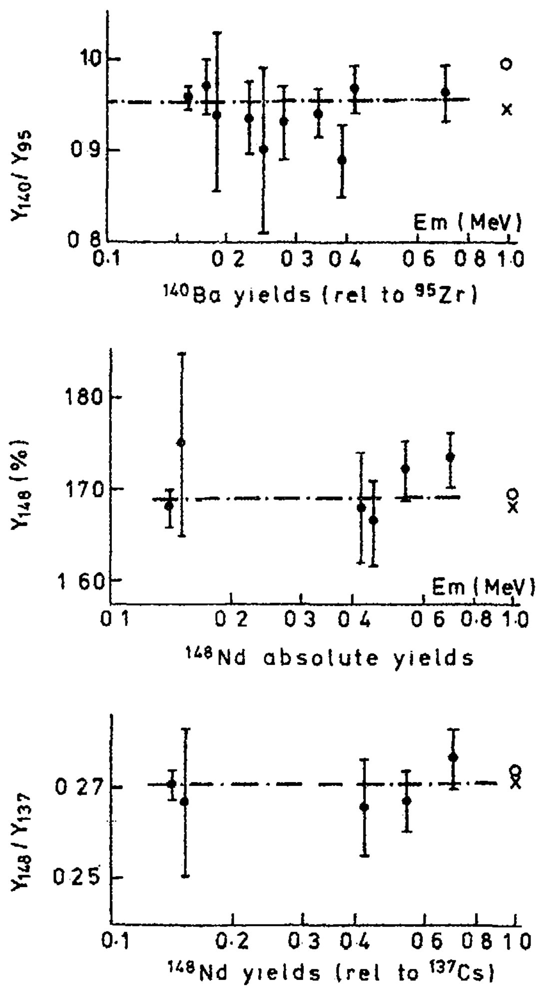


FIG 42. ^{235}U FAST FISSION YIELDS AS A FUNCTION OF ENERGY (CONCLUDED) (REF 31)

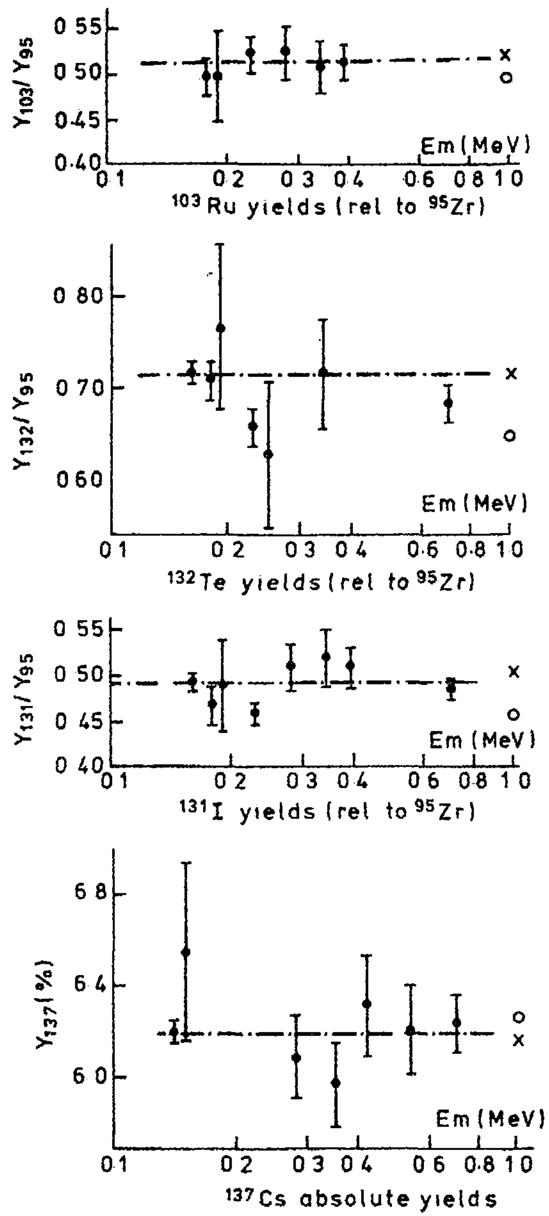
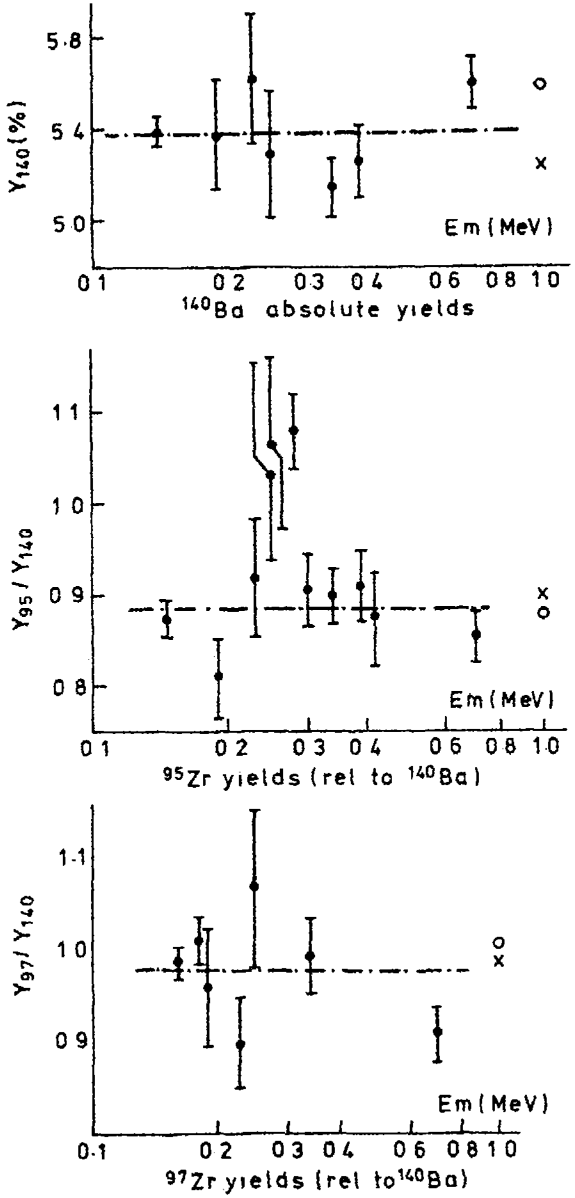


FIG. 42 ^{235}U FAST FISSION YIELDS AS A FUNCTION OF ENERGY (CONTINUED) (REF 31)



Key as fig 42

FIG. 43 ^{239}Pu FAST FISSION YIELDS AS A FUNCTION OF ENERGY (REF 31)

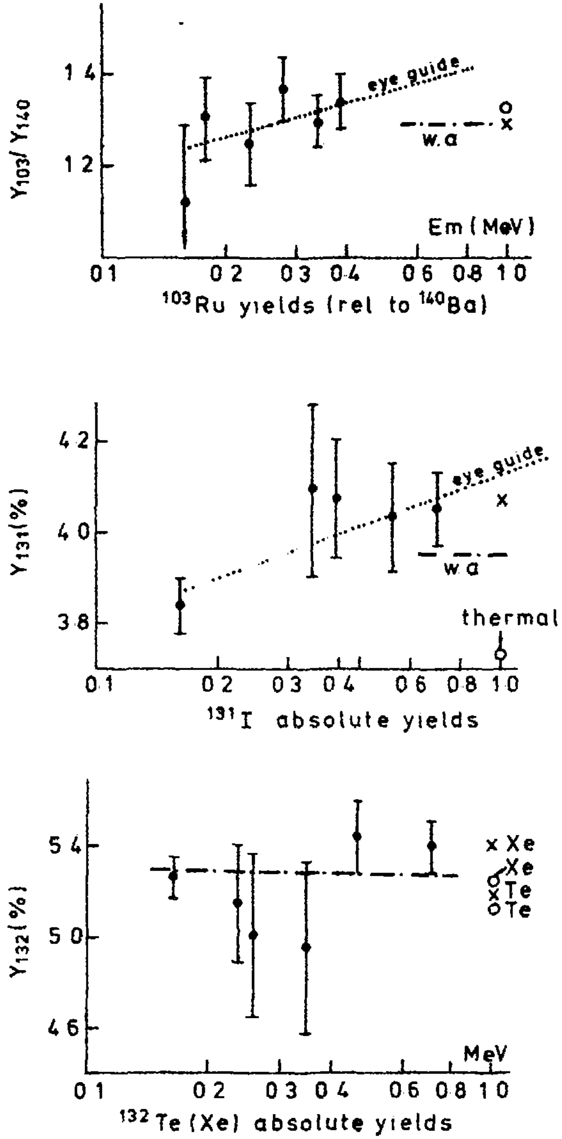


FIG 43 ^{239}Pu FAST FISSION YIELDS AS A FUNCTION OF ENERGY (CONTINUED) (REF 31)

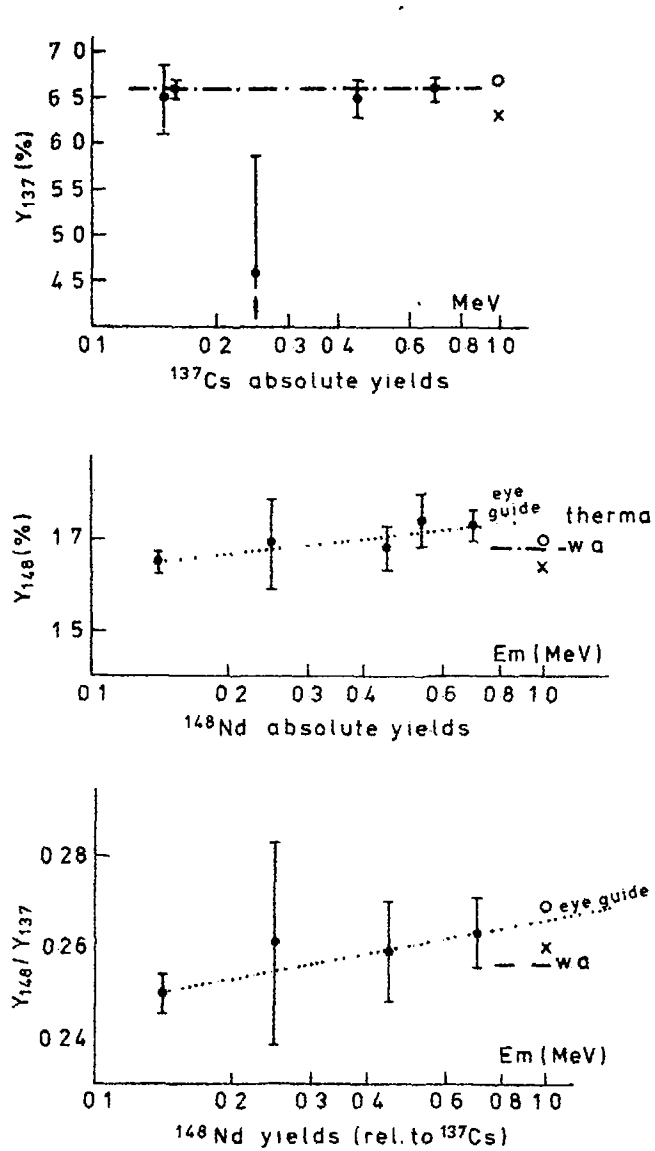
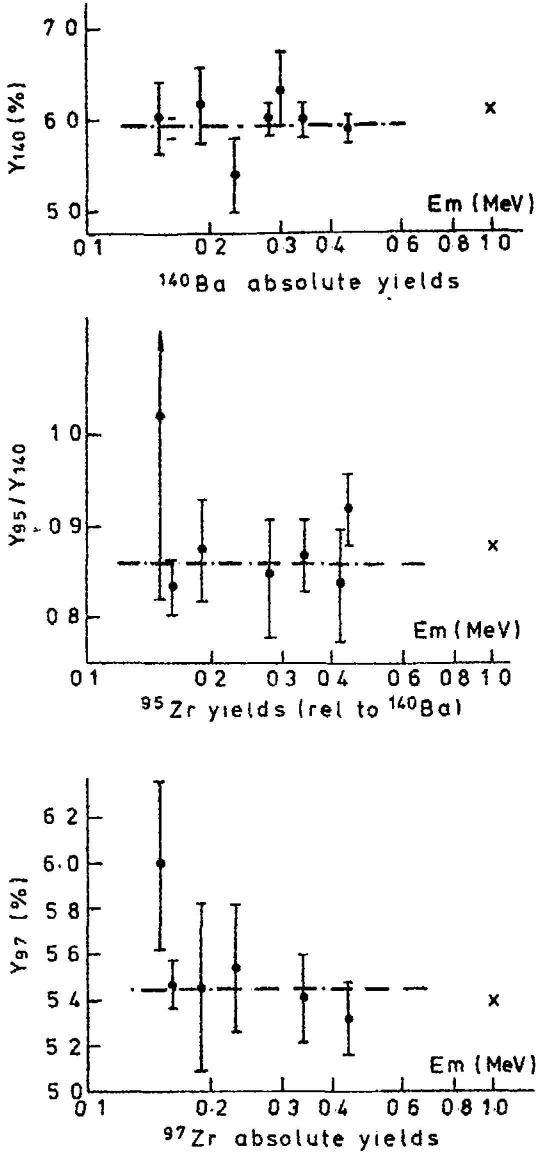


FIG 43 ^{239}Pu FAST FISSION YIELDS AS A FUNCTION OF ENERGY (CONCLUDED) (REF 31)



Key as fig. 42

FIG 44 ^{238}U FAST FISSION YIELDS AS A FUNCTION OF ENERGY (REF 31)

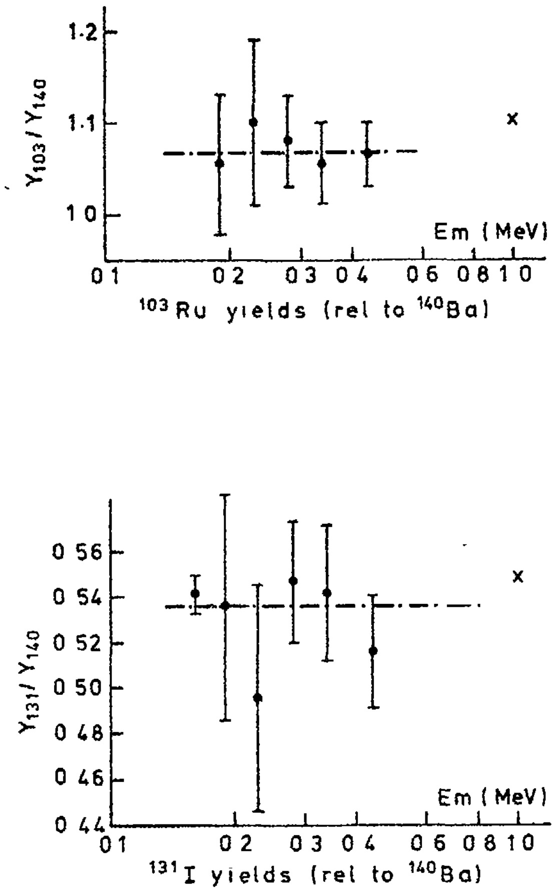


FIG 44 ^{238}U FAST FISSION YIELDS AS A FUNCTION OF ENERGY (CONCLUDED) (REF 31)