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INDC(UK)-33/G

Review of Important Nuclear Data Discrepancies: An NEANDC contribution to the INDC/NEANDC Discrepancy File

May 1980

Compiled by M. G. Sowerby

Introduction

At the 21st meeting of the NEANDC in September 1979 it was agreed that the NEANDC and INDC Discrepancy sub-committees would co-operate and maintain a joint discrepancy file. It is hoped that the file (or report) drawn up by the NEANDC sub-committee will be available for updating and comment by the INDC at its next meeting (9 months later). An updated report will then be prepared by the INDC sub-committee which can then be considered by the NEANDC at its next meeting. This process will then continue so that a discrepancy report is produced every 9 months and the INDC and NEANDC will always have an up to date report available for consideration at their meetings.

The present report is the first of this series of discrepancy reports. It is basically similar to the discrepancy section of the report issued in 1976 by the NEANDC sub-committee on Standards and Discrepancies (ANL/ND-77-1, NEANDC-105/L). However it is incomplete, as not all entries have been received in time for inclusion. However, it is felt that it is better to have an incomplete report than a delayed and possibly complete one.

The members of the NEANDC sub-committee on discrepancies at the Geel meeting in September 1979 were as follows

K. H. Böckhoff	CEC Geel
S. W. Cierjacks	Germany
H. Condé	Sweden
E. Fort	France
F. H. Fröhner*	Germany
H. Liskien*	Euratom
H. T. Motz	U.S.A.
F. G. J. Perey	U.S.A.
B. Rose*	CEC Geel
J. J. Schmidt*	IAEA
M. G. Sowerby	U.K. (Chairman)
K. Tsukada	Japan
E. Wattecamps*	CEC Geel

*observer

List of Discrepancies

Data Considered	Reviewer	
Li-7(n,n'αT) cross-section	M. T. Swinhoe	Germany
Capture cross-sections for Cr, Fe and Ni	G. Rohr and F. Corvi	CEC Geel
Cu-63(n,α)Co-60 reaction	G. Winkler	Austria
Nb-93(n,n')Nb-93m reaction	H. Vonach	Austria
Fast neutron capture in Th-232	W. Poenitz and A. Smith	U.S.A.
Fast neutron fission cross-sections of Th-232	A. Smith	U.S.A.
Fission cross-section and fission cross-section ratios for U-233	E. Fort	France
U-235 fission cross-section	M. G. Sowerby	U.K.
U-238(n,γ) cross-section below 100 keV and U-238 resonance parameters	G. De Saussure	U.S.A.
Inelastic neutron scattering from U-238	A. B. Smith	U.S.A.
Np-237(n,2n) cross-section	B. H. Patrick	U.K.
Pu-239 decay power discrepancy	T. R. England and P. G. Young	U.S.A.
Am-241 fission resonance integral	B. H. Patrick	U.K.
U-235, U-238, Pu-239 resonance parameters	H. Derrien	France
Delayed neutrons	E. Lund and G. Rudstam	Sweden

No entry has been made in the file for "Γ_γ for the 2.85 keV resonance for Na-23" as no new data have been produced since the entry given in the previous Standards and Discrepancy report (ANL/ND-77-1, NEANDC-105/L) was compiled. A file on the Pu-239(n,f) reaction should have been included in this report but is not available at the present time.

The ^{7}Li (n,n'αt) cross-section

1. Description of data and its application

The required cross-section is the total tritium production cross-section for ^{7}Li. That is the sum of the three reaction channels:

$$^{7}Li + n \longrightarrow {}^{7}Li + n' \quad \downarrow \alpha + t$$

(inelastic scattering excluding 1st excited state)

$$^{7}Li + n \longrightarrow {}^{5}He + t \quad \downarrow \alpha + n'$$

$$^{7}Li + n \longrightarrow n' + \alpha + t$$

(quasi-elastic scattering)

The application of this cross-section is in the calculation of the tritium breeding ratio of various fusion reactor blanket designs. The required accuracy of 1 % in the calculated tritium breeding ratio requires an uncertainty of less than 5 % in the ^{7}Li (n,n'αt) cross-section in the case of a natural lithium metal breeding blanket.[1,2] WRENDA (79/80)[3] has four priority one requests for cross-section data of this accuracy over an incident neutron energy range from 2.82 MeV (threshold) to 15 MeV.

2. Nature of the discrepancy

There are three types of evidence for a discrepancy in the size of the cross-section.

(i) A direct differential measurement carried out at Harwell.[4,5] This covered the energy range from 5 to 14 MeV using mono-energetic neutrons to produce tritium in lithium hydroxide samples. (> 99.5 % ^{7}Li). The tritium was measured using a liquid scintillation technique. The neutron fluence was checked by a simultaneous measurement of the $^{27}Al(n,\alpha)^{24}Na$ cross-section which gave results in agreement with other available measurements. The tritium measurement technique was checked with a determination of the $^{6}Li(n,\alpha t)$ thermal cross-section, and the result

was in agreement with the standard value.
The results of the ^7Li measurement are 26 % below the ENDF/B IV evaluation with average standard deviation of 6 %.

(ii) Three integral measurements using 14 MeV neutron source have recently been performed and compared with calculations. Firstly one at Jülich[6] using a natural lithium metal cylindrical blanket. The tritium production was measured by a liquid scintillation technique in samples of $^7Li_2CO_3$, $^6Li_2CO_3$ and natural Li_2CO_3. The measurements for $^6Li_2CO_3$ and natural Li_2CO_3 gave results close to the values calculated using ENDF/B III data, whereas those for $^7Li_2CO_3$ are approximately 15 % below the calculation. However the absolute error in this experiment is limited by the source strength calibration of ± 15 %.

Secondly at Karlsruhe[7] an integral experiment using a natural lithium metal sphere and the same tritium counting technique as above has been performed. Natural Li_2CO_3 samples were used. The measurements in this case are 35 % below the calculations which use ^7Li data from ENDF/B III. The standard deviation of the measurements is 5.1 %. The authors suggest that 13 - 20 % of this effect is due to an incorrect energy distribution of neutrons from in-elastic reactions (especially $^7Li(n,n'\alpha t)$) and the remainder is due to an evaluated $^7Li(n,n'\alpha t)$ cross-section which is too high by 15 - 20 %.
Thirdly an experiment at Los Alamos[8] used a ^6LiD sphere and ^7LiH samples. The tritium produced in these was measured using gas proportional counters. The results are up to 15 % lower than the calculations (using ENDF/B III data) with uncertainties of about 5 %. A previous experiment at Los Alamos[9] suggested that the evaluated $^7Li(n,n'\alpha t)$ cross-section was too high at 14 MeV and too low at lower energies, but a more recent analysis[10] has suggested that a reduction in the cross-section over the whole range 7.5 - 15 MeV (with a 13 % reduction at 14 MeV) would give a better representation of the experiment.

(iii) A measurement of the differential cross-section of $^7Li(n,xn)$ processes for secondary neutrons above 0.76 MeV has been carried out at ORNL.[11] The results for $\theta = 50^o$ and 126° in the laboratory frame are 20 % less than the ENDF/B IV library values over the range where the $^7Li(n,n'\alpha t)$ reaction is important. However these results could be explained in terms of incorrect angular distributions in the data file.

3. Status

The standard evaluated cross-section for this reaction (ENDF/B III, IV (V)) is that proposed by Pendlebury[12] in 1964. This has an estimated uncertainty of ± 15[13] - 20[1] %. No precise values for the required cross-section can be determined from the Oak Ridge measurements or the integral experiments, and although a complete report of the Harwell measurements is still in preparation, the preliminary values are given in Table 1.

There is another measurement of the cross-section at discrete energies in progress at Argonne,[14] and results from that are expected soon. Another experiment with monoenergetic neutrons has begun in Europe as a collaboration between the laboratories at Jülich and Geel.[15]

4. Comments and recommendations

This discrepancy leads to very significant differences in the predicted tritium breeding ratios of most fusion reactor blanket designs. A change of cross-section of this size can probably be counteracted by a suitable change in design, such as a thicker blanket or enrichment of the blanket in ^{6}Li or the use of a neutron multiplier. However such studies are severely limited by the current status of the microscopic data. This situation cannot be improved to a satisfactory level merely by a new evaluation. New differential (discrete energy) measurements are necessary. It may be that the new American and European experiments may resolve the discrepancy, however it is unlikely that they will provide sufficient data to satisfy the requirements outlined in section 1. The cross-section is required from threshold to 15 MeV for breeding studies, however the results are most sensitive to the 14 MeV value. This is one energy at which many laboratories are able to carry out experiments, and measurements at this energy would aid the resolution of the discrepancy, and help toward the production of a data set with the required uncertainty.

It should also be remembered that the energy spectrum of the secondary neutrons from the ^{7}Li (n,n'αt) reaction has considerable effect on the tritium breeding ratio of fusion reactor blankets, and large uncertainties exist in this area also.

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Incident neutron energy (MeV)	^{7}Li (n,n'αt) cross-section (millibarns)
4.7	68.6 ± 15
5.6	269 ± 27
7.5	318 ± 26
9.7	314 ± 19
9.8	310 ± 21
11.8	289 ± 15
14.1	228 ± 9
14.1	235 ± 9

Table 1. Preliminary values for the ^{7}Li(n,n'αt) cross-section from Harwell experiment.

1. D. Steiner and M. Tobias. Nuclear Fusion 14 153 (1974)

2. R.G. Alsmiller, R.T. Santoro, J. Barish, and T.A. Gabriel Nuc. Sci and Eng. 57 122 (1975)

3. INDC (SEC)-73/URSF (1979)

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12. E.D. Pendlebury AWRE-O-61/64 (1964)

13. L. Stewart USNDC-CTR-I (1974) Ed. D. Steiner

14. A.B. Smith Private Communication (1980)

15. S. Quaim Private Communication (1980)

Capture cross sections for Cr, Fe and Ni

Description of data and their application

The neutron capture data of primary structural materials (Fe, Ni and Cr) influence in fast reactors the critical enrichment, the breeding gain and, the Doppler coefficient. At present it is believed that steel contributes ~10% to the Doppler effect and that 60 - 80% of this is due to the 1.15 keV p-wave resonance in ^{56}Fe.

The most important energy range according to the European Request List (NEACRP-A-314) covers the region 100 eV < En < 100 keV and accuracies of 5 - 10%, 10% and 20% are required for Fe, Ni and Cr respectively. In the region 100 keV < En < 1 MeV an accuracy of 10 - 20%, 20% and 30% has been asked.

The capture cross section of structural materials has a pronounced resonance structure represented by a few very broad s-wave resonances ($\Gamma_n \gg \Gamma_\gamma$) with a small peak cross section and very sharp p- and d-wave resonances ($\ell \geq 1$). The number of degrees of freedom in the capture process is rather small ($\nu \sim 10$-15), which leads to a very strong fluctuation of the γ-ray spectrum from resonance to resonance.

Status

The data on the capture cross sections have been reviewed in NEANDC-105L (Discrepancy File 1976) and at the NEANDC/NEACRP Specialists' Meeting on Neutron Data of Structural Materials for Fast Reactors held in 1977 at Geel. Since then no results for capture cross sections of Ni and Cr have been published. Therefore the review of the status in this report is limited to Fe measurements only, as they were performed in recent years :

1. Cooperation with ORNL and Lucas Heights (Australia) :

   Measurements performed with a hydrogen-free liquid scintillator (C_6F_6 - detectors)using the weighting method, a pulsed linac neutron source (2.5 < En < 600 keV), and separated isotopes.

1a Results of ^{54}Fe : B.J. Allen et al. (1977) AAEC 403

1b Results of ^{56}Fe : B.J. Allen et al. Nucl. Phys. A 269 (1976) p. 408

1c Results of ^{57}Fe : B.J. Allen et al. Proc. Specialists' Meeting Geel (1977) p. 476

The results of broad s-wave resonances given in (1a) and (1b) are revised in (1d), considering the large neutron sensitivity of the ORNL detector system.

1d B.J. Allen and A.R. de L. Musgrove, Proc. Specialists' Meeting Geel (1977) p. 447

2. Harwell :

Measurements performed with a small liquid scintillator tank (700 l) : (efficiency determined by a semi-empirical method) on natural iron sample. Resonance parameters analysed for principal resonances up to 30 keV.

Preliminary results :

2a D.B. Gayther et al. Proc. Specialists' Meeting Geel (1977) p. 547

Final results :

2b D.B. Gayther and M.C. Moxon - private communication

3. Central Bureau for Nuclear Measurements - Geel :

Measurements performed with C_6D_6 detectors using the weighting method, pulsed linac neutron source ($0.5 < En < 600$ keV), and separated isotopes.

3a Preliminary results of ^{56}Fe

A. Brusegan et al. Int. Conf. on Neutron Cross Sections for Technology, Knoxville, 1979, paper BB6

3b Preliminary results of ^{54}Fe

G. Rohr et al. - private communication

4. KFK Karlsruhe :

Moxon-Rae γ-ray detector measurements of the broad 27.7 keV s-wave resonance, performed with a pulsed Van de Graaff accelerator using time of flight discrimination of γ-rays and scattered neutrons :

4a K. Wisshak and F. Käppeler Int. Conf. on Neutron Cross Sections for Technology, Knoxville (1979) paper BB6

Lucas Heights : same technique as described in 4

4b B.J. Allen et al. - private communication

Discrepancies

The capture area of small ^{56}Fe s-wave resonances and resonances with $\ell \geqslant 1$ obtained in the three main data sets mentioned under 'Status' shows significant systematic deviations. The Harwell data (2b) in the energy range $0 < En < 39$ keV are on the average by 8% lower and the ORNL-Lucas Heights data taken in the energy region 2.5 keV $< En < 90$ keV are by 20% higher than the Geel data analysed in the range $0 < En < 90$ keV. If the Harwell data would be normalized to the area of the 1.15 keV resonance using the same value of the neutron width as in Geel ($\Gamma_n = 58$ meV), then the disagreement would be reduced to about 2%.

A large effort has been put into the measurement of the radiative width of the very broad s-wave resonance at 27.7 keV. Excluding the results of paper ((1b),(1d)), an average value of $\Gamma_\gamma = 0.85$ eV is obtained, which is considerably smaller (70-80%) than those obtained before the Geel Meeting.

The older, larger values are ascribed to the neutron sensitivity of γ-ray detectors; a correction for this effect has been tried using Monte Carlo codes (1d).

The ^{54}Fe capture area (for resonances $\Gamma_n \lesssim \Gamma_\gamma$) of ORNL-Lucas Heights' results (1a) below 40 keV are systematically by 5% higher than those of Geel (3b), a difference considerably smaller than in the ^{56}Fe case.

Nature of Discrepancies

a) Flux and Normalization

The flux shape has been measured with Li-glass, boron-slab and ^{235}U detectors and should not introduce problems below 100 keV.

In Oak Ridge and Harwell the calibration has been performed using the black resonance technique at 4.9. eV, whereas in Geel the normalization is based on the 1.15 keV resonance parameters obtained in a transmission experiment. In the latter case the need to know the neutron flux behaviour down to the eV region is avoided.

b) Capture Detector Efficiency

The γ-ray spectra of resonances in structural materials are much harder than the resonances of Au and Ag normally used for calibration of the neutron flux. Further, the γ-ray spectra of different resonances belonging to the same isotope varies greatly (3a) and therefore procedures to correct the efficiency or the use of the correct weighting functions becomes very important in order to obtain an accuracy of 5-10% for individual resonances.

c) Scattered neutrons due to broad s-wave resonances ($\Gamma_n / \Gamma_\gamma \sim 10^4$) and captured in the detector or its surroundings cause a time dependent background, which is hard to distinguish from capture events of the sample. First trials to correct for this effect using computer programs are not very convincing (^{56}Fe Eo = 27.7 keV Γ_γ = 1.6 eV) (1d)

d) Multiple Scattering

Measurements with thick samples and broad s-wave resonances need corrections for multiple scattering, which are exclusively performed with computer programs using Monte Carlo codes. The codes seem to have improved : The capture area of the 27.7 keV resonance in ^{56}Fe calculated with FANAC (KFK, Fröhner) differs only by 5% from the results obtained with REFIT (Lynn and Moxon). This is quite remarkable in view of the fact that the multiple scattering correction in the case of a 2 mm thick natural sample is as high as 100% .

Comments and recommendations

1. In view of the remarks made in the previous paragraph a) and b) the most straightforward way of normalizing capture data of structural materials is to refer to some resonances with $\Gamma_n \ll \Gamma_\gamma$ and very well known parameters. For

this reason, accurate transmission measurements of the 1.15 keV resonance in ^{56}Fe and the 1.63 keV in ^{52}Cr are recommended.

2. Use detectors with low neutron sensitivity. The sensitivity should be measured experimentally and interpreted theoretically using Monte Carlo methods.

3. The capture detector efficiency should be checked up to 10 MeV γ-ray energy (see also point b).

4. All high resolution capture measurements should be supplemented by high resolution total cross section measurements in order to perform a correct reduction of resonance capture yields to capture cross sections and accurate Γ_γ determination.

5. The influence of non-isotropic neutron angular distribution should be checked for the treatment of multiple scattering corrections.

6. The importance of non-isotropic photon angular distribution on the sensitivity of capture gamma ray detectors should be investigated.

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The $^{63}Cu(n,\alpha)^{60}Co$ Reaction

A. Significance

The threshold-reaction $^{63}Cu(n,\alpha)^{60}Co$ is of continuing interest as a long-term fast neutron flux integrator (fluence monitor) in reactor dosimetry for the response range $E_n > 4.7$ MeV, because of the convenient residual-product decay permitting accurate detector calibrations for γ-detection, its long half-life, and the availability of sufficiently pure sample foils. Due to the particular response range of this reaction (the 5% - 95% response range in a ^{235}U-thermal-fission neutron spectrum is 4.7 - 10.9 MeV) its importance especially for light water reactor pressure vessel surveillance has been indicated in several articles. (See refs.1 in ref.1).

B. Status

The inconsistency in the available differential and integral cross section data base has limited the applicability of that reaction as a reliable monitor.[2,3] The ^{235}U fission spectrum averaged cross section as obtained from integral measurements was ~40% higher than the value calculated from existing differential data, essentially those of A. Paulsen,[4] which also were the basis of the ENDF/B-IV evaluation. The ENDF/B-V evaluation takes into account the recent result of a precise 14 MeV measurement[5] thus increasing the cross section by the ratio of the new 14 MeV value to the ENDF/B-IV 14 MeV cross section (i.e. ~17%) in the energy region 6.5 - 20 MeV, essentially retaining the shape of the original excitation function.[6] In the threshold region, i.e. below 5.5 MeV, ENDF/B-V is an extrapolation of the evaluation, guided by a Hauser-Feshbach calculation by C.Y. Fu and F.G. Perey from ORNL.[7] Using the ENDF/B-V Watts spectrum representation for the ^{235}U thermal fission neutron spectrum the calculated spectrum averaged cross section $\langle\sigma\rangle^{Cu}_{U5,calc.}$ changes from 0.364 mb (ENDF/B-IV cross section representation) to 0.557 mb (ENDF/B-V cross section representation). Several requests concerning the $^{63}Cu(n,\alpha)$ reaction are posted in WRENDA-79/80 for data with an accuracy of 5 - 10% in the energy region 6.0 - 18.0 MeV.

C. New Measurements and Evaluations

To solve the differential-integral inconsistency the excitation function has been remeasured at the Argonne National Laboratory from ~3 MeV to 10 MeV, the region of maximum response in a fission spectrum.[8] The measurements were done by activation relative to the well known $^{238}U(n,f)$ cross section, employing a high-sensitivity detector configuration in an extremely low background counting-facility, achieving an average accuracy of about 7%.

The new results differ significantly from the previously reported experimental data.[4] They represent higher values in the region near threshold below 6 MeV, lower values around 10 MeV, and show a cross over with the ENDF/B-IV values at about 8 MeV. Below 5 MeV they agree very well with the ENDF/B-V evaluation.

Based on these new differential results and some assumptions of minor importance for the energy region above 10 MeV (see ref.1), the ^{235}U fission spectrum averaged cross section was calculated to be
$\langle\sigma\rangle^{Cu}_{U5,calc.} = 0.507 \pm 0.049$ mb ($\pm$ 9.7%)
considering the effect of uncertainties in the assigned cross section and in the energy scale and in the chosen fission neutron spectum representation.[1] This value agrees with the result from a reevaluation of the experimental integral data base, providing a weighted average of the renormalized ^{235}U fission spectrum averaged cross sections $\langle\sigma\rangle^{Cu}_{U5}$ from the literature:
$\langle\sigma\rangle^{Cu}_{U5,eval.} = 0.534 \pm 0.015$.[1]

D. Conclusions and Recommendations

Comparing the ^{235}U thermal neutron fission spectrum averaged cross section for the reaction $^{63}Cu(n,\alpha)$, as derived from a new measurement of excitation function from threshold to ~10 MeV, with the result of a reevaluation of the integral experimental data, shows that the discrepancy between differential data and integral data for the reaction $^{63}Cu(n,\alpha)^{60}Co$ has been eliminated within the given error limits.

A check of the excitation function at few energy points, especially between 10 and 14 MeV, and between 15 and 20 MeV, to satisfy the WRENDA-requests and to compare with model-calculations

would be desirable. It seems very hard to achieve an accuracy of better 7% for the $^{63}Cu(n,\alpha)^{60}Co$ differential data over a broader energy region using the activation method with currently available neutron sources. Measurements involving the direct observation of the alpha-particles emitted from a ^{63}Cu-foil is recommended as an alternative method.

In cooperation with the National Bureau of Standards, USA, work on measuring the spectrum averaged cross section in the ^{252}Cf-fission neutron field and remeasuring the ^{235}U fission spectrum averaged cross section is in progress.

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The $^{93}Nb(n,n')^{93m}Nb$ reaction

A.) Significance

Because of its low threshold (30 keV) and long half-live (~ 15 years) the $^{93}Nb(n,n')^{93m}Nb$ reaction is potentially an excellent long-term fast neutron fluence monitor especially for radiation damage studies.

B.) Status

The present status of our knowledge of the excitation function of the $^{93}Nb(n,n')^{93m}Nb$ reaction is still extremely unsatisfactory.

No activation cross-section measurements for monoenergetic neutrons exist to date, only measurements in different fast reaction spectra have been reported [1].

Some information on the $^{93}Nb(n,n')^{93m}Nb$ cross-section in the neutron energy range 0.8 - 2.7 MeV can be derived from (nn'γ) measurements by summing up the gamma production cross-sections for all transitions populating the isomer. However the existing data [2 - 5] strongly disagree within each other and also with the existing data on the ^{93}Nb nonelastic cross-sections derived from sphere-transmission [6] and (n,n') measurements [7 - 9]. Thus at best cross-section values accurate to about 30% are derivable from this information.

Thus at present it seems, that the most accurate method of prediction of this cross-section are nuclear reaction calculations based on the statistical model of nuclear reactions including pre-equilibrium particle emission.

Calculations of this kind are being performed at present at the Institut für Radiumforschung und Kernphysik. An evaluation of the $^{93}Nb(n,n')^{93m}Nb$ cross-section up to 20 MeV based largely on the above calculations will probably be completed in summer 1980 and published in Physics Data by Fachinformationszentrum Karlsruhe, Germany and included in International Neutron Dosimetry File.

C.) Conclusions and recommendations

In order to improve the situation the following two measurements which seem possible would be extremely helpful.

1) An activation measurement of the $^{93}Nb(n,n')^{93m}Nb$ cross-section at 14 MeV, where the sufficiently intense neutron sources are available. A measurement of this kind would not only provide one reliable data point but it would also greatly improve the accuracy of the calculated cross-sections in the whole range 10 - 20 MeV by drastically reducing the admissible parameter variations.

2) New accurate (nnγ') measurements in the 0.8 - 3 MeV range should be performed in order to remove the mentioned discrepancies.

Literature

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FAST-NEUTRON CAPTURE IN ^{232}Th*

Requirements for fast-neutron capture cross sections of ^{232}Th are similar to those for ^{238}U assuming the need for Th/U fast-reactor cycles. In such cycles the objective is typically 0.5-1.0% in k_{eff} and 2% in breeding ratio implying cross-section accuracies of a few percent depending upon the exact design configuration. The available data base is far more limited than for the analogous ^{238}U capture process. The newer and more detailed experimental results are given in Refs. 1-6. Older data are, without exception, higher than the more recent values by 10-20%. As is evident in Fig. 1, the more recent experimental results spread over a range of at least ±10%, i.e. the uncertainty is at least a factor of two worse than for the comparable ^{238}U process.

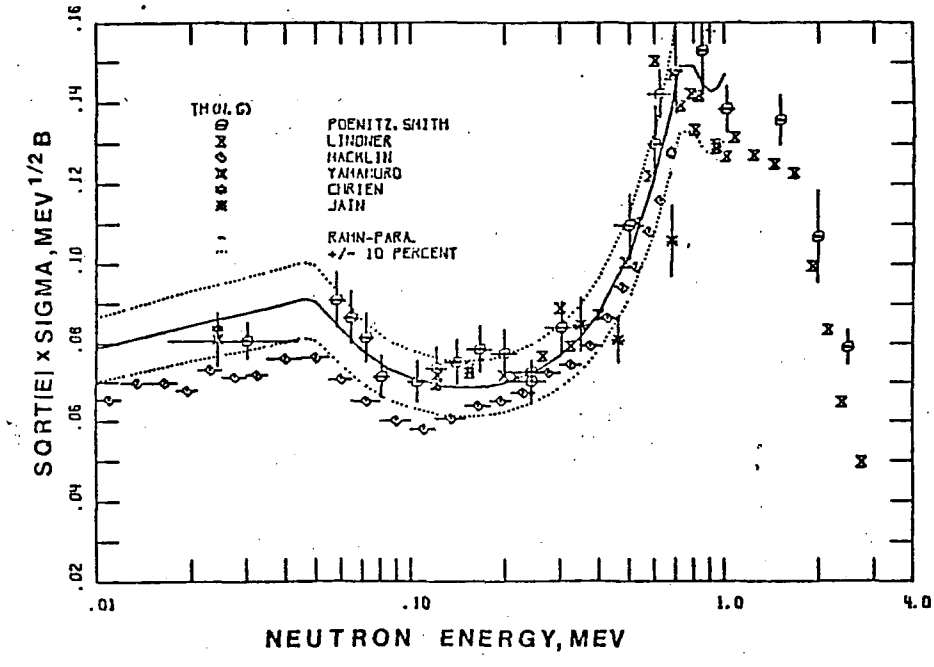

Fig. 1. Recent measured cross sections for the $^{232}Th(n,\gamma)$ reaction. Data points are taken from Refs. 1-6 and the curve is the result of statistical calculations described in Ref. 8. The dotted curves represent the calculated result ±10%.

The results of Lindner et al.[2] and of Poenitz and Smith[1] tend to be consistent and systematically higher than the values reported by Macklin and Halperin.[3] The differences may well be outside the respective experimental uncertainties. In any event, the measured values spread by 10-15%. The data of Macklin and Halperin was obtained using a spectral-weighing technique while those of Lindner et al. and Poenitz and Smith were obtained using activation methods relative to the $^{235}U(n;f)$ cross section. In addition, some of the measurements of Ref. 1

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*Detailed discussions of this problem area are given in Argonne Natl. Lab. Report, ANL/NDM-35 (1978) J. Meadows et al. and by W. Poenitz, Bull. Am. Phys. Soc., 24 872 (1979).

employed a large liquid scintillator for relative results which were subsequently normalized to the activation values. Data were obtained at 24 keV by Chrien et al.[5] and Yamamuro et al.[4] using an iron filter. These latter results tend to be somewhat larger than suggested by the values of Ref. 3. All of the newer experimental values suggest a capture cross section lower than given in ENDF/B-IV, and those of Ref. 3 strongly so.

The cross section can be calculated from average $\langle\Gamma\rangle$ and $\langle D\rangle$ parameters using the statistical model. Using the average parameters given by Rahn et al.[7], the calculated results, indicated by the solid curves in Fig. 1, were obtained.[8] The calculated results are reasonably consistent with the measured cross sections supporting the average parameters given in Ref. 7. The same calculated result is compared with the cross section as given in the ENDF/B-V evaluation in Fig. 2. The calculated (and measured) results and the evaluation are consistent to within about ~±10%. An uncertainty in the calculation was the detailed treatment of the competing inelastic scattering process.

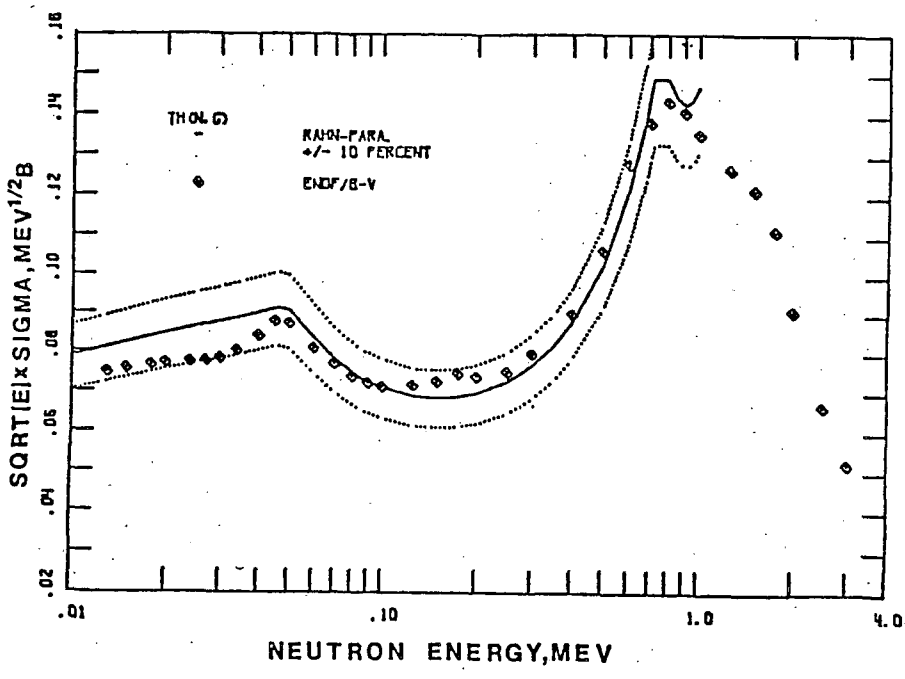

Fig. 2. Comparison of the evaluated $^{232}Th(n;\gamma)$ cross sections as given in ENDF/B-V with the results of statistical calculations (curve). The curves and their uncertainties are identical to those shown in Fig. 1.

It is concluded that the present experimental data base does not define the fast-neutron capture cross section of ^{232}Th to better than ±10%. The ENDF/B-V evaluation is a reasonable description of the available information. The present uncertainties will not be reduced without additional detailed measurements. The techniques and methods involved are very similar to those employed in the analogous ^{238}U measurements and should provide at least a factor of two improvement in accuracies with relative ease. Such measurements could well benefit from a correlated microscopic-macroscopic measurement approach to the problem.

W. Poenitz/A. Smith
Argonne National Laboratory
April 1980

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FAST-NEUTRON FISSION CROSS SECTIONS OF ^{232}Th*

The ^{232}Th(n;f) data base is less complete than that of the fission cross sections of the other common actinides. The early work is summarized in Ref. 1. The more recent values are given in Refs. 2-7. Some of the latter are of a preliminary nature. The older data are particularly subject to uncertainties as to; reference standard, error specification and experimental definition. Some of the reported measurements had the objective of basic physical studies (e.g. character of sub-threshold fission) and did not give particular attention to cross section definition.

The status up to 1978 is reasonably summarized by ENDF/B-V[1]. The evaluation first independently treats the ratio values ^{232}Th/^{235}U and ^{232}Th/^{238}U and the absolute ^{232}Th(n;f) cross section determinations. These three individual components are then combined to obtain the evaluation using the Version-V ^{235}U(n;f) and ^{238}U(n;f) reference values where appropriate. The latter two reference cross sections are not entirely independent but the effect of their correlation on the ^{232}Th result is small. The evaluation pointed up problems in both the energy scale near threshold and in the absolute normalization notably in the 14 MeV region (see Ref. 1). The Version-V evaluated result is similar to that of Version-IV as illustrated in Fig. 1.

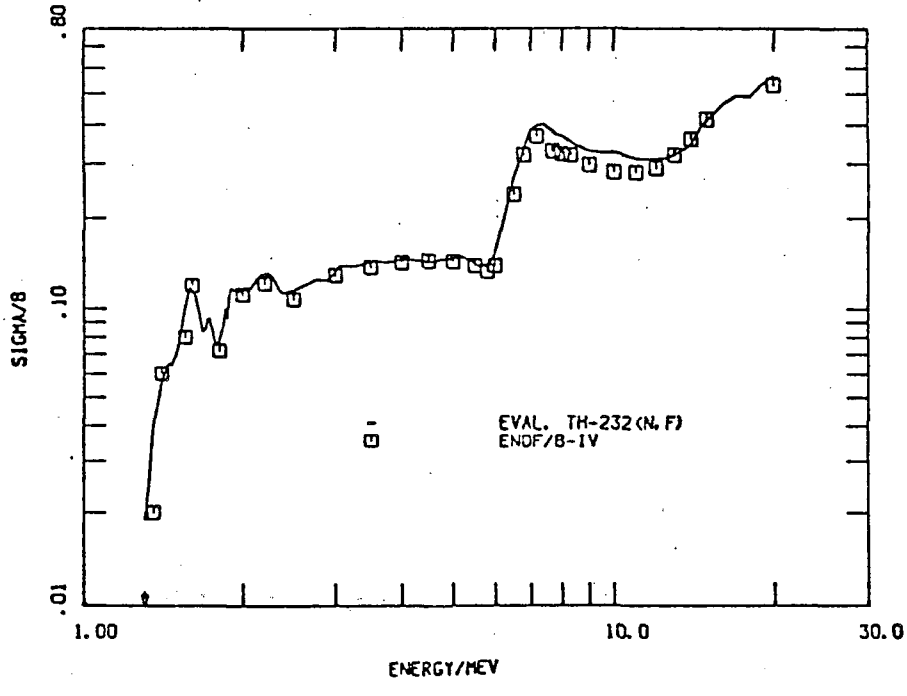

Fig. 1. Comparison of ENDF/B Version-V (curve) and Version-IV (boxes) evaluations of the ^{232}Th(n;f) cross section.

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*A detailed discussion of this problem area contemporary with 1978 is given in Ref. 1.

Subsequent to the above evaluation, new data has become available from Meadows[7] and from Nordborg et al.[6] Both of these data sets were obtained using monoenergetic source techniques. These two new sets, combined with the previously reported white-source results of Behrens et al.[4], give the most comprehensive coverage of the cross section in the region of most applied interest. All three sets of data are absolute ratios relative to the $^{235}U(n;f)$ cross section. These sets of data are compared in Fig. 2.

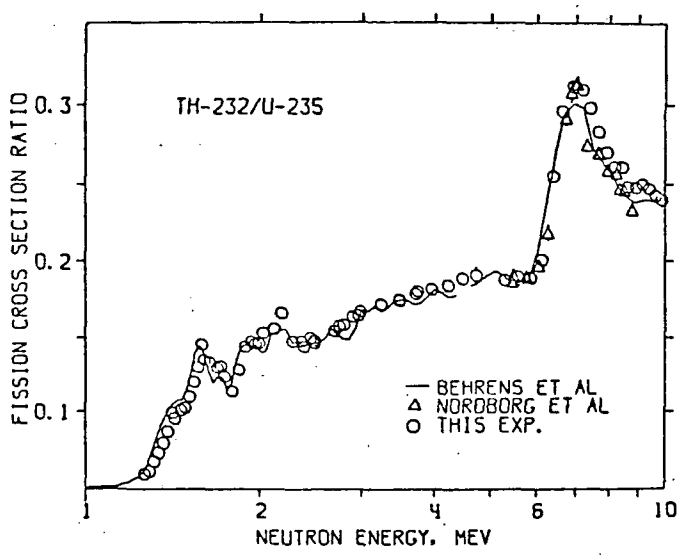

Fig. 2. The $^{232}Th/^{235}U$ (n;f) ratio data as reported by Behrens et al.[4] (curve), Nordborg et al.[6] (triangles) and Meadows[7] (circles).

The agreement between the above measured ratio values is good excepting a small energy-scale problem near threshold and some scatter in the 8-10 MeV region. These newer results are somewhat different from the ratio values implied by ENDF/B-V as illustrated in Fig. 3.

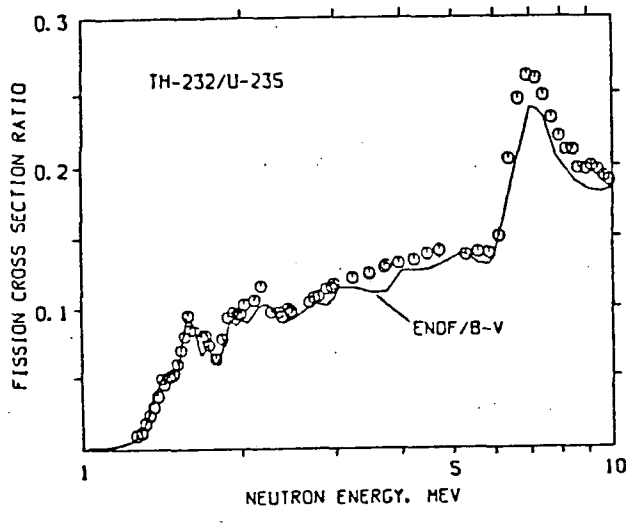

Fig. 3. The $^{232}Th/^{235}U$ (n;f) ratio implied by ENDF/B-V (curve) compared with the recent measurements of Meadows[7] (circles).

These are ratio comparisons therefore the discrepancies may be attributed to ^{232}Th, ^{235}U or both. It is noted that the 3-5 MeV region is an area where the $^{235}U(n;f)$ cross section is relatively uncertain.[8]

Persistent problems are; 1) discrepancies between cross section magnitudes determined from ratio and from absolute cross section measurements, and 2) small differences in the energy scales between measurements particularly near threshold.

A. Smith
Argonne, April 1980.

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Fission cross section and fission cross section ratios for ^{233}U

A comparison of the versions 76/77 and 79/80 of WRENDA list shows an increasing interest for ^{233}U fission cross section since the publication of a previous report[1] on this subject, in November 1976.

At that time the status was characterized by :

- old experimental data, of doubtfull quality, obtained essentially in a relative way
- few evaluations which didn't fullfil the requested conditions of accuracies.

Published results

I - Experimental data

There are many new experimental data since last report

a) Fission cross section.

The data of GWIN et al[2] were obtained on ORELA between 5 and 200 KeV.

POENITZ presented data obtained on electrostatic accelerators between 0.13 and 8 MeV. While POENITZ's values are pure absolute values, those given by GWIN and al are normalized on resonance integral calculated by WESTON and al[4] (5% accuracy on normalization) and that can possibly explain the discrepancy (8%) observed in the overlapping region of the two measurements.

The data of POENITZ and the old ones given by ALLEN and FERGUSON[14] seem to differ by only a constant factor.

An other absolute value, very accurate (accuracy < 2%), obtained at 14.8 MeV by ALKOSOV et al[5] using the associated particule method is beyond the scope of this report.

b) Ratio $^{233}U\ \sigma(n,f)/^{235}U\ \sigma(n,f)$

Obtained between 0.001 KeV and 30 MeV, on the 100 MEV LIVERMORE LINAC the data of BEHRENS and CARISON were omitted in the last report, but they have been reconfirmed recently in a complete document[6].

The datum at 144 KeV published by CHURALEV et al[7] is smaller (7.6%) than the one of GWIN et al, but in excellent agreement with those mentionned just above.

FOURSOV et al[8] propose values between 0.024 KeV and 7.4 MeV using an experimental technics which has some similarities with POENITZ's one. These values are systematically lower than those of MEADOWS[9] and those of CARLSON and BEHRENS.

The most recent data concerning the ratio of ^{233}U to ^{235}U fission cross section are due to JAMES et al[10]. Extended over the energy range 0.1 to 20 MeV they agree fairly well (± 3%) with CARLSON and BEHRENS' results up to 7 MeV. Above this energy these two sets disagree in shape and there is a difference of (5%) with inversion of sign around 10 MeV.

II - Evaluated data

There are no new evaluations in the energy range under consideration.

Comments

The present status is as follows :

- there is a large amount of recent ratio values of good quality
- according to pictures from PATRICK[11] there is a significant discrepancy between values of $\sigma(n,f)$ and what can be extracted from ratio values using ENDF/BV for ^{235}U fission cross section. That is essentially true for energies between 0.1 MeV and 1 MeV (an energy range of importance for fast breeders based on the U-Th fuel cycle).

The difference can amount to more than 5% around 0.3 MeV and is greater if FOURSOV's et al values are refered to.

It is not excluded that the actual difference can even be greater, if the very recent experimental indications from MEIER et al[12], concerning an overestimation by 3% of ^{235}U fission cross section in ENDF/BV between 0.2 MeV and 1.2 MeV, are taken into account

Thus, for evaluation purpose, it seems difficult to choose between absolute and relative data on the basis of arguments of experimental technics. One element for this choice could be given in calculating the fission cross section ($E < 0.5$ MeV) from the compound cross section (expected accu-

racy = 4%) resulting from deformed optical model calculations (parametrization obtained by the so called "SPRT" method) and the fission probability measured (3% accuracy) by BACK et al[13] with the ^{233}U (d,pf) ^{234}U reaction.

As final concluding remarks it can be said :

- the observed improvement is still of limited character, since the knowledge on the ^{233}U fission cross section above 1.2 MeV is certainly not better than 3%-4%.

- an accuracy of 1%, as it is sometimes requested, is probably out of reach.

- more absolute data are suitable for confident evaluations.

E. FORT, January 1980

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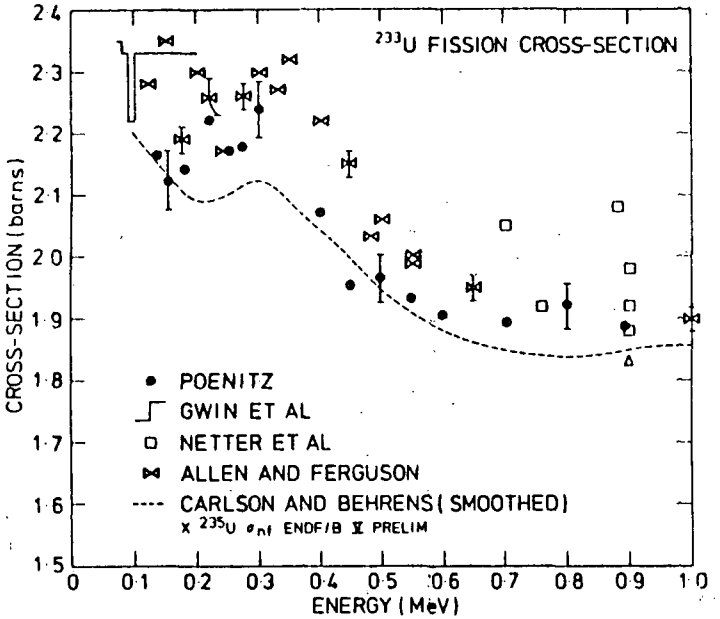

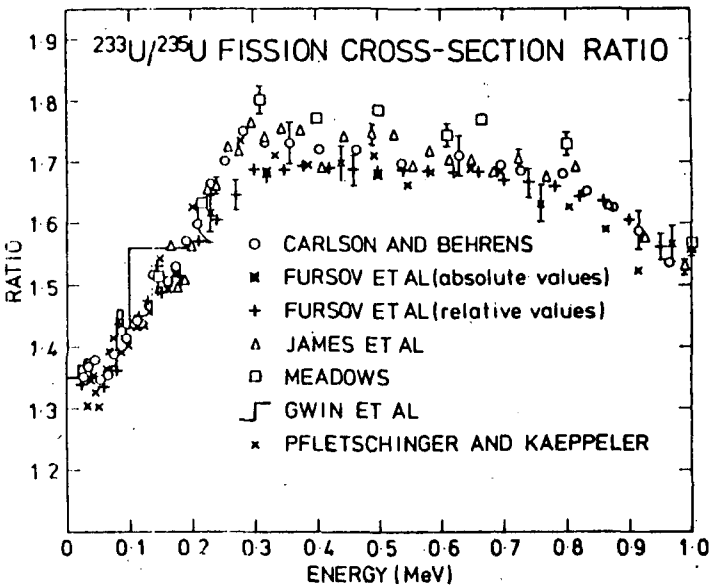

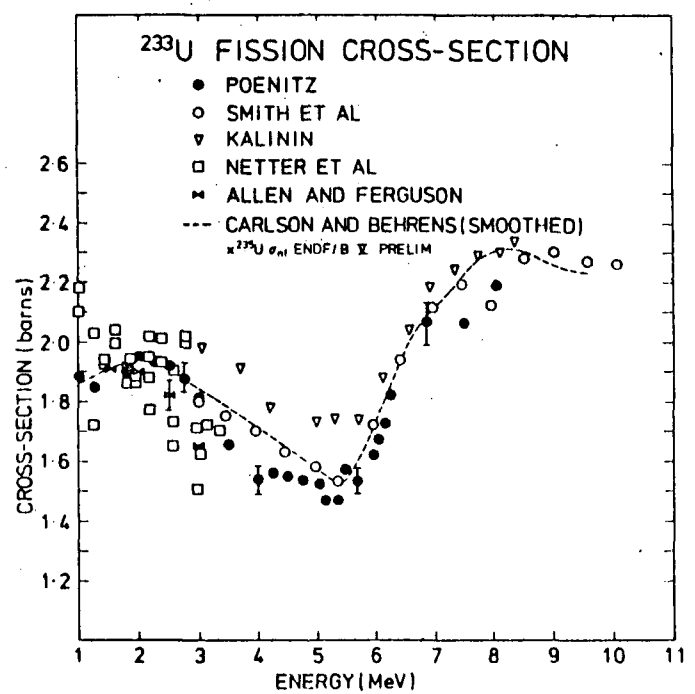

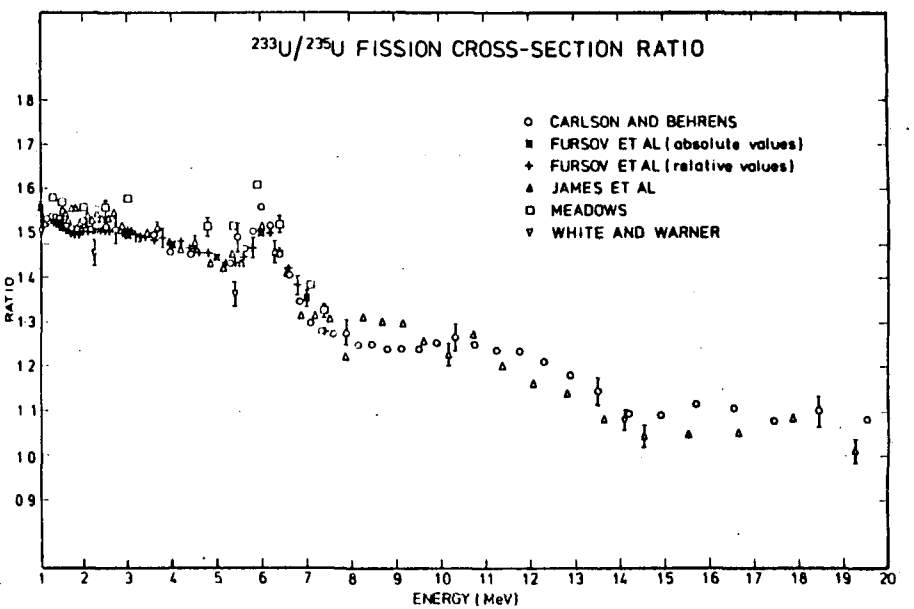

THE PICTURES ARE EXTRACTED FROM REFERENCE [11]

U-235 Fission Cross-section

Description of data and its application

Fission cross-section of U-235 between 100 eV and 20 MeV.

The data are required for the calculation of reactor properties and for use as a standard. Because of structure in the cross-section its use as a standard below 200 to 300 keV can lead to difficulties.

Nature of discrepancies

The cross-section has been reviewed or evaluated by a number of authors in recent years (e.g. Konshin et al INDC(CCP)-132 (1979), Poenitz ANL/NDM-45 (1979), Bhat BNL-NCS-51123 (1979), Kuzminov, Report of INDC/NEANDC Nuclear Standards File INDC-30/L+Sp 1980). There is general agreement that the cross-section is known to an accuracy of ∿±3% (1SD) below ∿15 MeV. At higher energies the error increases to ∿±6%. These accuracies meet some of the less stringent requests in WRENDA 79/80 but are a long way from meeting the requests for a 1 to 2% accuracy from the U.S.A., U.S.S.R., Sweden, Germany and France associated with obtaining improved standards and better reactor calculations.

However, though the overall accuracy is ±3% there are energy regions where the resolution of certain discrepancies could make significant improvement to the present uncertainties.

(1) In the keV energy range there are inconsistencies in the shapes of the measured cross-sections (for example see Bhat BNL-NCS-51123, also ANL-76-90, p.307).

(2) In the 0.3 to 0.8 MeV energy range the data of Wasson (ANL-76-90, p.183), Wasson and Meier (1979 Knoxville) and Poenitz (GND, NSE 53, 370 (1974)) all tend to lie low compared to the other data.

(3) Between 2 and 5 MeV the values of Carlson and Patrick (78 Harwell 880) and Poenitz (NSE 64, 894 (1977)) are low while those of Czirr and Sidhu (NSE 57, 18 (1975)) and Kari (KFK 2673 (1978)) tend to be high.

(4) Above 8 MeV there is a tendency for data to divide into a high and a low group.

Status

Only a limited number of new measurements are known to be in

progress or planned; most measurements that have been completed are now fully documented. The measurements known to be planned or in progress are listed in Table 1.

Table 1

U-235 fission cross-section measurements planned or in progress

Author	Establishment*	Reference	Energy Range	Comments
Engdahl	MHG	DoE/NDC-15/U, p.149	14 MeV	To be done
Wasson +	NBS		14.4 MeV	In progress
Gayther	HAR		<20 MeV	Planned on New Harwell Linac
Wagemans	GEL	NEANDC-202U, Vol.III,p.23	Thermal-30 keV	Data being analysed

*Using CINDA abbreviations

Comments and recommendations

In order to meet the most stringent requirements in WRENDA 79/80 and to sort out the detailed discrepancies in the present data more measurements are required. However, in view of the large number of existing measurements, new data should only be obtained if either (a) the accuracy is likely to be significantly better than previously or (b) a new technique is used or (c) some errors have been identified in earlier work. In view of the difficulty in finding good values for normalisation it is recommended that with white spectrum neutron sources only absolute measurements should be made unless (a) the relative measurements will be very accurate (±1 to 2%) or (b) they can be extended to the thermal or any other equally well known energy range in the resonance region.

The present uncertainties could well be associated with the errors inherent in the measurement of neutron flux. In view of this both the development of flux measurement techniques and the comparison of flux measurements between laboratories should be encouraged. In particular the comparison programme organised by BIPM and based on the fission chamber designed by Gayther should be supported (see AERE-PR/NP 26,p.35, INDC-30/L+Sp,p.107).

The background at neutron energies above a few MeV when using white spectrum neutron sources is often small but it is not well determined.

The investigation of the cause and magnitude of these backgrounds is desirable.

It is clear that some of the present uncertainties could be due to errors in the determination of fission foil masses. To obtain fission cross-sections accurate to ±1% requires that foils be assayed to ∿±0.3%. The proposals to have a U.S. national U-235 fissile foil reference mass standard and to compare this internationally with other standard foils should therefore be strongly supported.

M. G. Sowerby

$^{238}U(n,\gamma)$ Cross Section Below 100 keV and ^{238}U Resonance Parameters

G. de Saussure

June 29, 1979

DESCRIPTION

The ^{238}U capture cross section and resonance parameters are of major importance for the calculation of performance parameters of thermal and fast reactors, such as the effective multiplication constants, the breeding ratio as well as the Doppler coefficient of reactivity.

In most recent evaluations the ^{238}U cross sections are represented by resolved resonance parameters up to about 4 keV and by unresolved (statistical) parameters above 4 keV, up to 45 keV in ENDF/B-IV and up to 149 keV in ENDF/B-V.

The resolved resonance parameters are obtained by a consistent analysis of transmission, self-indication, capture, and scattering high-resolution measurements in conjunction with theoretical models of statistical properties and whatever other information may be available on the properties of specific resonances.

The unresolved parameters in ENDF/B versions IV and V were generated by using "conventional values" for the average s-wave parameters and adjusting the average p-wave neutron width to "fit" evaluated average capture and inelastic-scattering cross-sections. In this procedure the average p-wave neutron widths are redefined every few hundred eV.

STATUS

I. Resolved Range (below 4 keV)

A large number of important new measurements of the low energy ^{238}U cross sections have been reported in the past five years, and some older measurements have been carefully reexamined. Much of this work was stimulated by the apparent inability of ENDF/B-IV, and other evaluations, to predict the ^{238}U capture rate in thermal critical lattices. The problem was extensively discussed at a "Seminar on ^{238}U Resonance Capture" held in Brookhaven National Laboratory on March 18-20, 1975.[1]

The recent measurements and reanalyses of older data are discussed in "Evaluation of the ^{238}U Neutron Cross Sections for Incident Neutron Energies up to 4 keV,"[2] a paper describing the ENDF/B-V evaluation of the ^{238}U cross sections below 4 keV, where detailed references to the measurements are

given. The most significant changes suggested by the recent work is a reduction by about 15% of the capture widths of the first three s-wave levels and an increase of from 10 to 20% of the strength function above 1.5 keV. These changes have reduced but not completely eliminated the discrepancy between the computed and measured ^{238}U capture rates in thermal critical lattices.[3]

II. Unresolved Range (4 to 100 keV)

Recent measurements of the $^{238}U(n,\gamma)$ cross section above a few keV are discussed by Poenitz et al.[4] in "Evaluated Fast Neutron Cross Sections of Uranium-238," a document describing in particular the ENDF/B-V evaluation of the infinitely dilute ^{238}U capture cross section above 20 keV. The capture cross section measurements have generally large uncertainties (of the order of 6%), and show significant discrepancies even among the most recent data. In the range 20 to 100 keV the new data suggest a higher $^{238}U(n,\gamma)$ cross section than ENDF/B-IV, and indeed ENDF/B-V is higher, by amounts ranging from a fraction of 1% to 10%. On the other hand the analysis of integral benchmark experiments suggests[5] lower group cross sections than obtained with ENDF/B-IV.

CONCLUSION AND RECOMMENDATIONS

I. Resolved Range (below 4 keV)

Very recent transmission measurements performed at Harwell[6] have yielded resonance parameters up to 520 eV in substantial agreement with the ENDF/B-V evaluation. Further work will include an investigation of the systematic errors and possibly an extension of the analysis to higher energy regions.

As previously stated, even with ENDF/B-V the discrepancy between the computed and measured ^{238}U capture rates in thermal critical lattices is not completely eliminated. This suggests additional experimental and evaluation work; however, it is perhaps not completely clear, at present, that the discrepancy implies inadequacy of the ^{238}U resolved resonance parameters.

II. Unresolved Range (4 to 100 keV)

The large differences between the various measurements of the $^{238}U(n,\gamma)$ cross section in the unresolved region are very unfortunate, in view of the importance of the data to the nuclear energy programs. However, the large

uncertainties in the measurements and the discrepancies result from the inherent difficulties of capture measurements in the 1 to 100 keV range. The difficulties result in part from the low value of the ^{238}U binding energy, and also from the necessity to perform important background, efficiency, and multiple scattering corrections to the raw data. Additional measurements of the $^{238}U(n,\gamma)$ cross section should probably attempt to reduce the uncertainties associated with these corrections by stressing new approaches and better techniques.

Perhaps an even more important problem, particularly below 10 or 20 keV, is to test the validity of the representation of the unresolved resonance parameters. The technique used in ENDF/B-IV and V is straightforward from a "mechanical" viewpoint, but it is not unique and there is very little experimental confirmation of the adequacy of the model. Sowerby[7] and others[8] have recently discussed the problems associated with finding adequate unresolved parameters for ^{238}U. Recent experiments at Harwell[9] and at the University of Missouri[10] are designed to test the use of unresolved parameters to predict resonance self-shielding and Doppler effect. Probably the most efficient method to improve the ^{238}U cross section description would consist of extending the resolved range representation to energies above 4 keV. New measurements of the $^{238}U(n,\gamma)$ cross section below 10 keV are planned at ORNL. It is hoped that the result of these measurements combined with recent transmission measurements[11,12] will allow such an extension of the resolved region.

The importance of improving the representation of the ^{238}U cross sections in the keV region is confirmed by a large number of recent studies of fast reactors,[13] thermal reactors,[14] and Doppler effect.[15]

TABLES

In Table I is a comparison of infinitely dilute and strongly self-shielded (σ_0 = 10 b) group cross sections computed with ENDF/B versions IV and V. The comparison is over a somewhat arbitrary 8 group structure covering the resolved range. The values were obtained by R. Q. Wright, at ORNL, using MAT 1262 and 398, respectively. The two last columns of the table indicate that the differences between version IV and version V are typically a few percent for either unshielded or shielded group constants. However, it is important to note that the changes in the shielded group constants are not proportional to the changes in the dilute group constants (the signs are not even the same in most cases). This indicates that an overprediction of the ^{238}U capture in strongly self-shielded critical lattices does not necessarily imply that the evaluated infinitely dilute capture cross section is too high.

TABLE I. Comparison of ENDF/B-IV and V $^{238}U(n,\gamma)$ Group Cross Sections Over the Resolved Energy Range

Group	EL, EH eV	Dilute (1) IV b	Dilute (1) V b	Shielded (2) IV b	Shielded (2) V b	$\frac{V - IV}{IV}$ Dilute %	$\frac{V - IV}{IV}$ Shielded %
1	.4 - 100	45.55	45.65	1.700	1.636	+0.22	-3.8
2	100 - 170	23.80	22.91	1.220	1.235	-3.7	+1.2
3	170 - 280	11.46	10.96	.8472	.8867	-4.4	+4.7
4	280 - 450	3.578	3.489	.6686	.6754	-2.5	+1.0
5	450 - 750	3.538	3.521	.7720	.8120	-0.48	+5.2
6*	750 - 1230	2.692	2.777	.8167	.8066	+3.2	-1.2
7	1230 - 2040	1.753	1.774	.7037	.6889	+1.2	-2.1
8	2040 - 3360	1.352	1.401	.7541	.7874	+3.6	+4.4

(1) $$\frac{\int_{EL}^{EH} \sigma_{n\gamma} \frac{dE}{E}}{\int_{EL}^{EH} \frac{dE}{E}}$$

(2) $$\frac{\int_{EL}^{EH} \frac{\sigma_{n\gamma}}{\sigma_t + \sigma_o} \frac{dE}{E}}{\int_{EL}^{EH} \frac{1}{\sigma_t + \sigma_o} \frac{dE}{E}} \quad (\sigma_o = 10 \text{ b})$$

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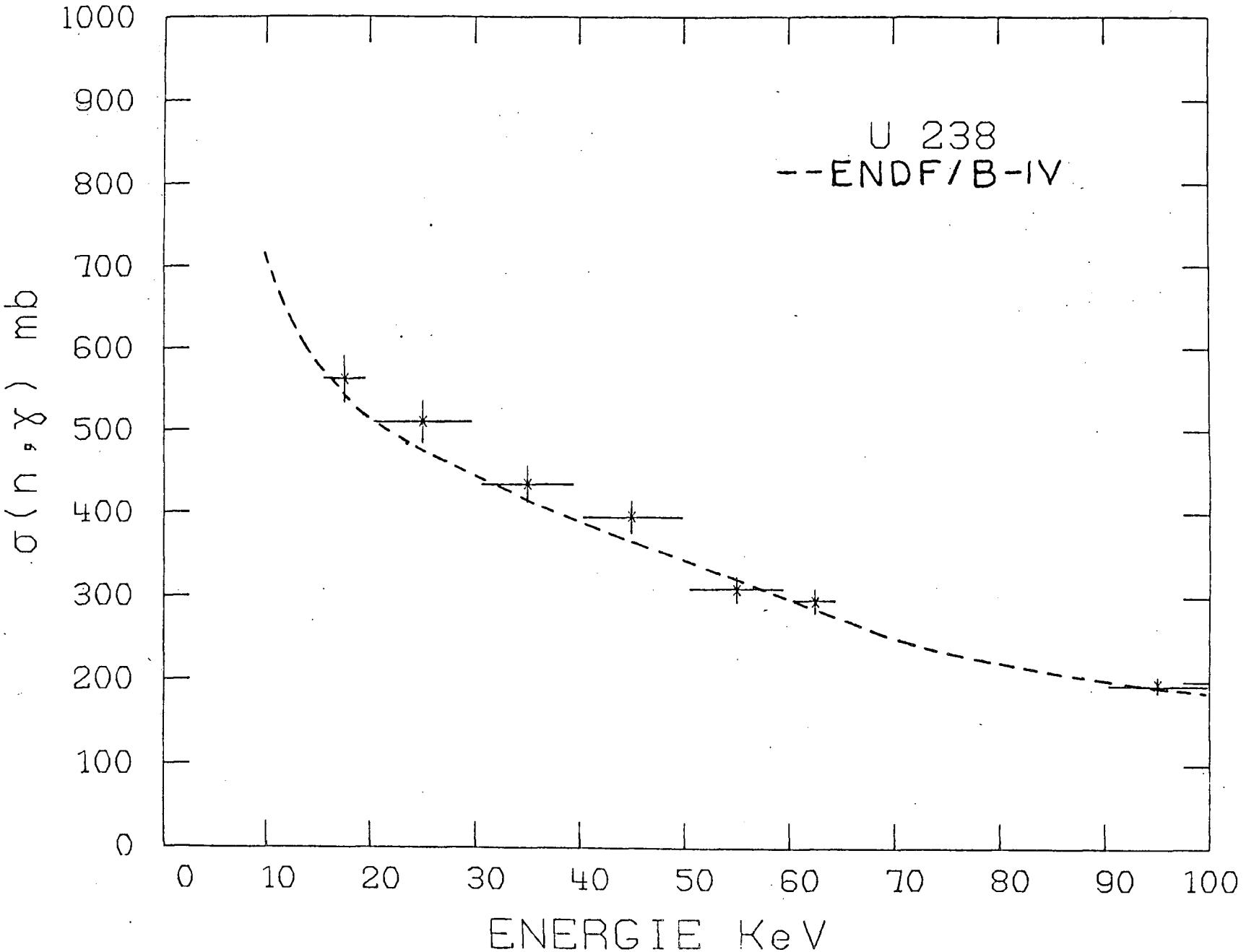
FIGURE

The figure shows a comparison between the recent data of Le Rigoleur et al. and ENDF/B-IV. The figure is from the reference:

> CEA-R-4788 - Le Rigoleur Claude - ARNAUD André - Taste Jean
> Absolute Measurements of Neutron Radiative Capture Cross Sections for ^{23}Na, Cr, ^{55}Mn, Fe, Ni, ^{103}Rh, Ta, ^{197}Au, ^{238}U in the keV Energy Range.

This reference also contains a detailed discussion of the experimental uncertainties of the measurements.

Other data are compared to ENDF/B-V in the report quoted in ref. 5.


U 238
--ENDF/B-IV
σ(n,γ) mb
ENERGIE KeV
0
100
200
300
400
500
600
700
800
900
1000
0
10
20
30
40
50
60
70
80
90
100

INELASTIC NEUTRON SCATTERING FROM ^{238}U*

Experimental knowledge of the energy-averaged neutron total and elastic scattering cross sections of ^{238}U has considerably improved in the last few years [1,2]. This, aided by improved model calculations[3], has resulted in a better understanding of the non-elastic cross section. The result, known to ~7%, is shown in Fig. 1 and indicates an increase in the non-elastic cross section from that implied by ENDF/B-IV. Version-V is consistent with the higher values. Below the onset of the (n;2n') cross section the non-elastic cross section, corrected for the relatively small fission component, is essentially the neutron inelastic scattering cross section. Thus the latter follows to within ~10% over the energy range of primary interest.

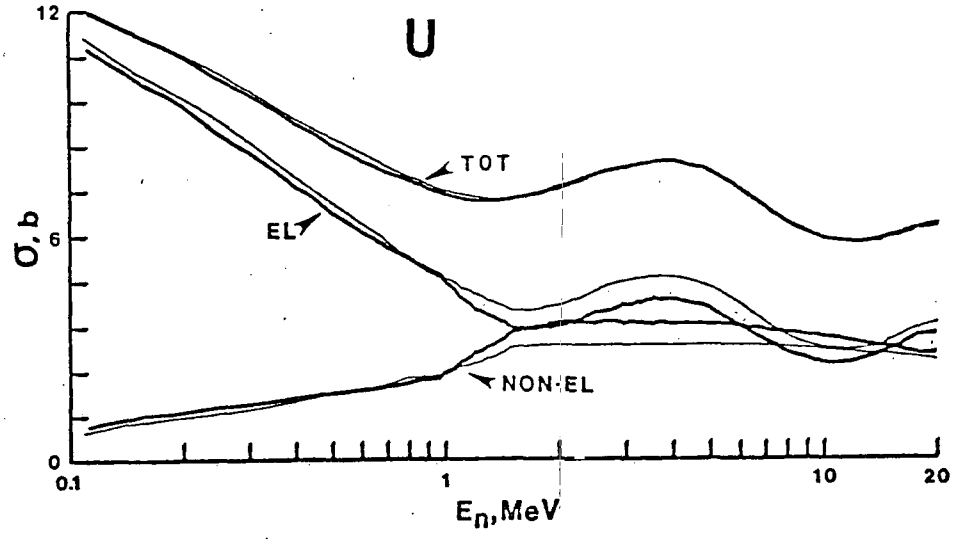


Fig. 1. Neutron total, elastic and non-elastic cross sections of ^{238}U. The light curves denote ENDF/B-IV values, the heavy curves ENDF/B-V results.

ENDF/B-V treats neutron inelastic scattering processes as; the excitation of discrete states, the excitation of composite contributions from a number of poorly resolved states and as the excitation of a continuum of unresolved states. The cross sections for excitations to energies of ~1.0 MeV are reasonably known and are individually treated in the Version-V evaluation. The contributions from states at excitation energies of $\gtrsim$1.0 MeV are more complex and uncertain. The Version-V groups such excitations into 10 groups to energies of 2.5 MeV. States with excitations in the range 2.5-4.0 MeV are represented by a simple ladder model with level-density increasing with energy. This representation is physically reasonable and blends smoothly into the continuum distribution starting at excitations of 2.5 MeV. The initial inelastic scattering cross sections were slightly adjusted to obtain improved agreement between measured and calculated integral benchmarks with such adjustments confined to reasonable uncertainties in the microscopic values and consistency with the measured non-elastic cross section.

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*Detailed discussion of this process is found in the Argonne National Laboratory Report, ANL/NDM-32, W. P. Poenitz et al., (1977).

The low-energy inelastic scattering cross section is dominated by the ground-state rotational band consisting of 45, 148 and 308 keV states. The latter contribution is small thus higher order states of this band were ignored in the Version-V evaluation. The Version-V evaluation is based upon a correlated application of measurement results and calculation as outlined in the footnote. The experimental results are reasonably consistent as illustrated in Fig. 2.

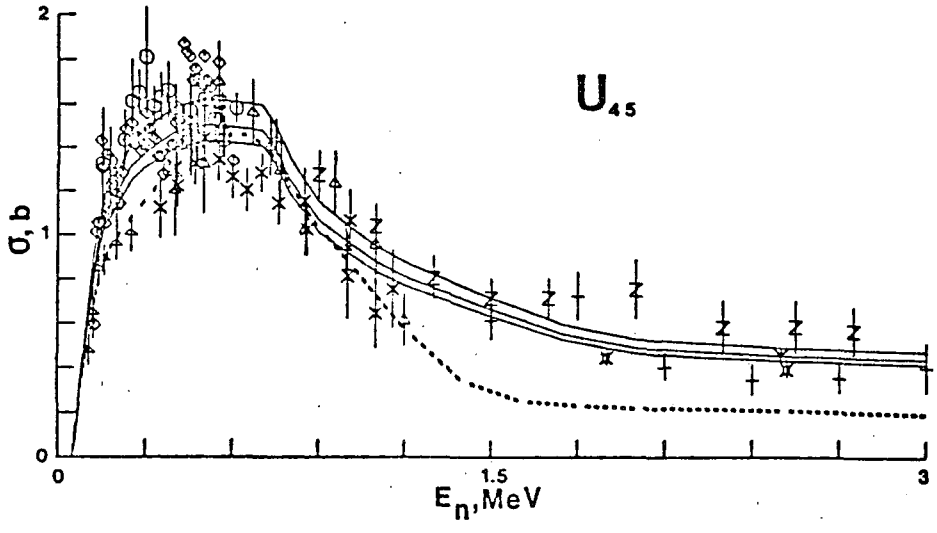

Fig. 2. Cross sections for the excitation of the 45 keV (2+) state. Measured values are indicated by symbols. The solid curve denotes the ENDF/B-V result with respective ± uncertainties. The dotted curve is from Version-IV.

Generally, the uncertainty in the evaluation of the prominent components is 5-10%. The most significant uncertainties are at low energies where the evaluation relied primarily on calculational extrapolation. Measured values at an incident energy of 85 keV are to be reported by Winters et al.[4] and should help resolve the low energy uncertainties. The ground-state-excitations of Version-V are much larger than those of Version-IV in the few MeV range in accord with measurements and calculations[2,3]. This results in a sharply larger total inelastic scattering cross section but has little impact on the typical fast-reactor multi-group transfer matrix.

The significant contributions from the K=0 band consist of 680, 732 and 827 keV states. The respective cross sections have been deduced from both direct neutron detection measurements and (n;n',gamma) measurements. The first two of these states make the major contribution to the cross section and the experimental results are reasonably consistent. The Version-V evaluation is primarily based upon the direct neutron measurements. The respective evaluation uncertainties are ~10%.

At excitations above ~1.0 MeV the Version-V evaluation combines discrete excitations into composite groups made up of the contributions from a number of states. The groups structure is a compromise between the resolutions available from the experiments and the definition needed for applications. The uncertainty associated with any one excitation function in this region can be large but the non-elastic cross section limits the cumulative uncertainty to 10-15%.

Recent (n;n',gamma) measurements by Olsen et al.[5] have improved the definition of the excitation of states at energies of ~0.7-1.5 MeV. A white source was used providing very good energy detail that makes possible the quantitative determination of the inelastic scattering cross sections in this energy range. Illustrative results for the 680 keV (1-) state are shown in Fig. 3.

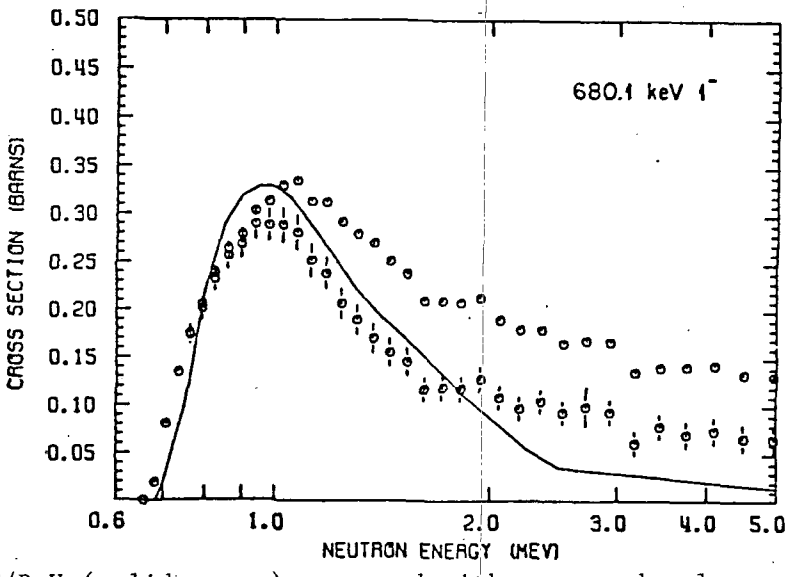

Fig. 3. ENDF/B-V (solid curve) compared with measured values of Ref. 5 for the excitation of the 680 keV state. The data points with bars are corrected for feeding from higher-energy levels and are to be compared with the evaluated cross section.

Similar comparisons can be made in a broader energy-average as illustrated in Fig. 4. Generally, the Olsen et al. results tend to be slightly smaller than the evaluation in this region but the difference is well within the respective uncertainties. Similar (n;n',gamma) measurements have recently been reported by Mittler et al.[6] A monoenergetic source was used in an energy range comparable to that of Ref. 5. Again, inelastic cross section values were deduced from the measurements. The results tend to be somewhat larger than given in Ref. 5 and in the Version-V evaluation. Thus the Version-V evaluation remains a reasonable representation of present experimental data base over the energy range of these most recent measurements.

The magnitude of the continuum inelastic cross section in Version-V is defined by the non-elastic component and the remaining independently-defined contributions. In addition, the evaluation was guided by macroscopic "benchmark" trials at energies above ~10 MeV. The resulting continuum contribution is considerably smaller in Version-V than Version-IV.

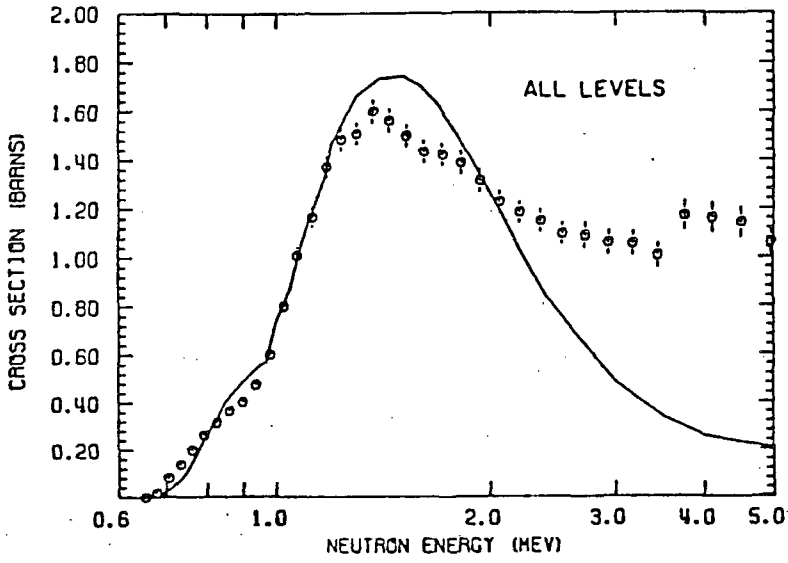

Fig. 4. ENDF/B-V compared with the corresponding GS-transition gamma-ray production and EO cross sections for all levels from 680-1169 keV of excitation. Measured values of Ref. 5 are indicated by data points, the evaluation by the curve.

The Version-V evaluation is summarized in Fig. 5 and compared with the total inelastic scattering cross section as given in Version-IV. Over wide energy ranges Version-V total inelastic scattering cross sections are much larger than those given in Version-IV. This difference can be deceptive as the transfer matrix involved in many applications insignificantly changed. Indeed, in some applications the newer evaluation may even lead to reduced energy transfer in the inelastic scattering process due to the re-arrangement of the higher-energy excitations.

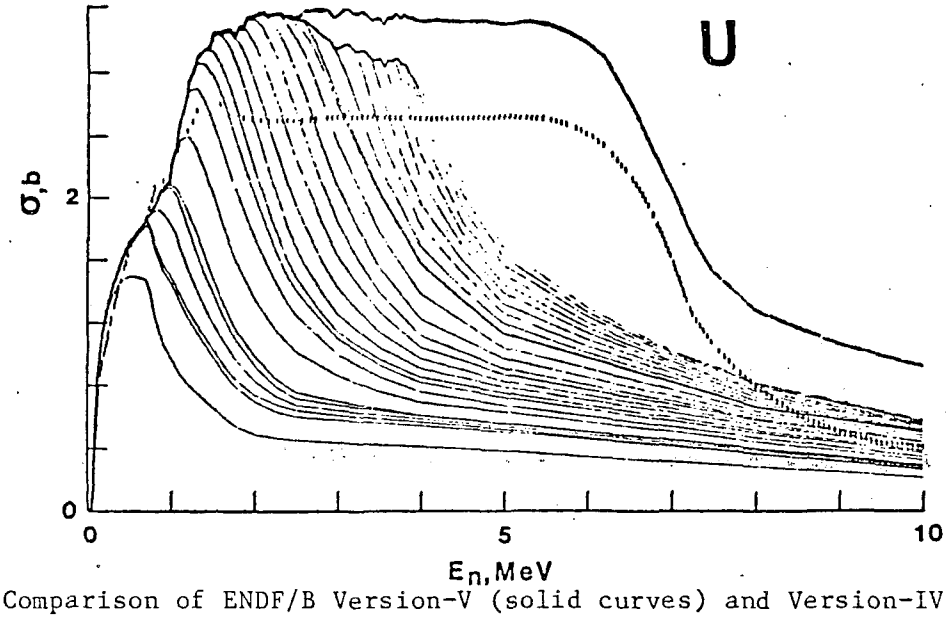

Fig. 5. Comparison of ENDF/B Version-V (solid curves) and Version-IV (dashed curves). The Version-V individual excitation functions are cumulatively summed to obtain the total inelastic scattering cross section. The corresponding total inelastic scattering cross section from Version-IV is shown.

Future work could well emphasize the following areas.

1. Experimental determination of the cross section at ~500 keV to ± 3% or better. Necessary for normalizing other measured and calculated shapes.

2. Resolution of question of fluctuations at energies of less than 500 keV.

3. Several measured values in the range 100-300 keV to accuracies of 10%.

4. Precision measurements of the differential-elastic-scattering cross section such that the non-elastic cross section is determined to ~5% from 1-20 MeV.

5. Relatively broad group excitations (e.g. ΔE_x ~250 keV) to accuracies of 10% for incident neutron energies in the range 1-5 MeV.

6. Several detailed measurements of the emission spectrum at incident neutron energies in the range 5-15 MeV with particular attention to precompound "tails".

7. Theoretical study of the excitation of the ground-state band particularly as relevant to the magnitude of enhancement factors. Present models generally underestimate the cross section at energies below 500-1000 keV.

A. B. Smith
Argonne, April 1980.

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3. Ch. LeGrange, NEANDC Report, NEANDC(J)-38L (1975).

4. R. Winters et al., Bull. Am. Phys. Soc., 25 543 (1980).

5. D. Olsen et al., Oak Ridge Natl. Lab. Report, ORNL/TM-6832 (1979).

6. A. Mittler et al., Bull. Am. Phys. Soc., 24 882 (1979).

$^{237}Np(n,2n)$ cross-section

Description of data and its application

The isotope ^{238}Pu has found considerable use as a heat source in a variety of applications, including flashing beacons in remote locations, heart pacemakers and space exploration. The isotope is generally produced by irradiating ^{237}Np in a thermal reactor, the reaction being $^{237}Np(n,\gamma)^{238}Np \xrightarrow{\beta^-} {}^{238}Pu$. In some cases it is important to produce the ^{238}Pu with as little ^{236}Pu as possible because of the hard (2.6 MeV) gamma-rays emitted by a daughter product (^{208}Tl). For this reason also, the production of ^{236}Pu is of some concern in the fabrication of plutonium fuel for fast reactors. The ^{236}Pu is mainly produced by the reaction $^{237}Np(n,2n)^{236}Np \xrightarrow{\beta^-} {}^{236}Pu$ (in fast reactor fuel the ^{237}Np is produced in the first place by (n,2n) reactions on ^{238}U) and hence a knowledge of the $^{237}Np(n,2n)$ cross-section is required.

The important quantities in the route from ^{237}Np to ^{236}Pu are illustrated in Fig. 1. The first point to note is that the (n,2n) reaction feeds two states in ^{236}Np, the ground state and an isomeric state, the exact assignment of these states being still in some doubt. In early papers the long lived ($T_{\frac{1}{2}} = 1.15 \times 10^5$ y) state was taken to be the isomer and the short lived ($T_{\frac{1}{2}} = 22$ h) state was assumed to be the ground state but in the most recent evaluation (Schmorak 1977) from the Nuclear Data Project, the assignments are reversed. This has no effect on the arguments which follow but for the sake of consistency in this note, the short lived state will be taken to be the isomer and the long lived one the ground state and these assignments will be used to describe results, even if in the original papers the reverse assignments were assumed.

The second important point to note is that from a practical point of view, ^{236g}Np can be considered stable and does not contribute to the ^{236}Pu production. Of course, this depends on there being a significant branch of the (n,2n) reaction leading to the isomer and in practice this seems to be the case.

The third quantity of importance is the β^- branch of the decay of ^{236m}Np.

All the known $^{237}Np(n,2n)$ measurements have used the activation technique in which the amount of ^{236}Pu produced in an irradiation is determined. It is clear from the considerations above and from Fig. 1

that the (n,2n) cross-section can be given in several different ways, including

(a) the total (n,2n) cross-section

(b) the (n,2n) cross-section for production of ^{236m}Np

(c) the (n,2n) cross-section for production of ^{236}Pu

the three forms being related through the appropriate branching ratios.

The β^- branch in the decay of ^{236m}Np is given as 48±1 % by Schmorak (1977) and although we have not examined the evidence, we shall accept this value without question. On the other hand, the value of the $^{236m}Np/^{236g}Np$ isomer ratio is known with much lower accuracy and, in particular, the energy dependence is totally unknown. Therefore any attempt to correct back from measurements of ^{236}Pu production to the total (n,2n) cross-section runs into difficulty over the isomer ratio. For these reasons, we shall convert the relevant experimental results to values of the $^{237}Np(n,2n)^{236m}Np$ cross-section so that comparisons can be made. However, the isomer ratio problem cannot be entirely ignored since theoretical calculations generally give data on the total (n,2n) cross-section and isomer ratio values have to be adopted before comparisons with the experimental data can be made.

Nature of discrepancy

There is a possible discrepancy between differential measurements of the $^{237}Np(n,2n)^{236m}Np$ cross-section and an integral measurement which claims an accuracy of ±10%. The results of a calculation based on systematics appear to support the integral measurement while another application of the statistical model seems to agree with the differential data and leads to an integral value which is approximately a factor of two larger than measured.

Requests in WRENDA 79/80 for the $^{237}Np(n,2n)$ cross-section indicate a required accuracy of 10-15%. Although it is not completely clear, it is presumed that this refers to the cross-section for production of ^{236}Pu.

Status

There are four known measurements of the differential (n,2n) cross-section for production of ^{236}Pu, all of which used the activation technique. The results have been converted to give values of the $^{237}Np(n,2n)^{236m}Np$ cross-section and these are plotted in Fig. 2. Where

necessary, the data were corrected to take account of the up-to-date β^-/EC branching ratio of ^{236m}Np (Schmorak 1977) and the half-life of ^{236}Pu which was taken to be 2.851±0.008 y (Lorenz 1979). It is to be noted that, with the exception of the 9.6 MeV point by Nichi et al (1975), all the measurements have been done in the region of 14 MeV and, furthermore, the agreement between the measurements is reasonably good.

The main integral evidence arises from a measurement by Paulson and Hennelly (1974) in which samples of highly enriched uranium fuel were irradiated to ∿30% burn-up in a heavy water moderated reactor and the ^{236}Pu/^{238}Pu ratio was determined by alpha spectrometry. The ^{237}Np(n,2n) cross-section was adjusted in the burn-up code, CASPER, until the calculated value of the ^{236}Pu/^{238}Pu ratio agreed with the measurement. If we accept that the spectrum causing the (n,2n) reactions is described by the ^{235}U fission neutron spectrum (the authors used a Watt form in their analysis) we can deduce a value of 1.10±0.10 mb for the ^{237}Np(n,2n)^{236m}Np cross-section averaged over the fission spectrum, a correction having been made for the currently accepted β^-/EC branching ratio of ^{236m}Np.

There is one other integral measurement, by Halperin et al (1968), which appears to have been issued in preliminary form only. ^{237}Np samples were irradiated in the ETR at Idaho and the ^{236}Pu produced was determined by alpha spectrometry. The result of the measurement gives a value for the ^{237}Np(n,2n)^{236m}Np cross-section of 2.5 mb (±25%) averaged over the reactor spectrum which is expected to be similar to the ^{235}U fission neutron spectrum in the region of interest (>6.8 MeV). It is to be noted that some of the ^{236}Pu may have been produced by the ^{237}Np(γ,n) reaction, the gamma-rays arising from neutron capture in the cadmium surrounding the samples. This possibility, coupled with the large error, means that little weight can be given to this measurement.

Now in order to see if there is really a disagreement between the differential and integral data, we require to know the energy dependence of the ^{237}Np(n,2n)^{236m}Np cross-section so that the fission neutron spectrum can be folded in and the average cross-section calculated. Theoretical estimates of the cross-section as a function of energy, not requiring normalisation to measured ^{237}Np(n,2n) values, have been performed by Pearlstein (1965) and Jary (1979). Both of these give values for the total (n,2n) cross-section and therefore before these can be converted to the cross-section for production of ^{236m}Np, the isomer

ratio $[\sigma(n,2n) \rightarrow {}^{236m}Np]/[\sigma(n,2n)_{total}]$ must be determined.

There appears to be only one measurement of the isomer ratio, by Myers et al (1975). Samples of ^{237}Np were placed in or near a number of thermonuclear devices and irradiated with a neutron spectrum, which was believed to be predominantly 14 MeV when the devices were actuated. The extent of the contribution from lower energy neutrons resulting from scattering is not known. The isomer ratio was measured using a combination of mass spectrometric and alpha-spectrometric techniques. Measurements were made in 9 separate experiments, up to 11 samples being analysed in a single experiment, and using the current value for the β^-/EC branching ratio, the average value of the (n,2n) cross-section feeding ^{236m}Np relative to the total (n,2n) cross-section is found to be 0.75.

Making the assumption that the isomer ratio is independent of energy, the total (n,2n) cross-sections obtained in the calculations of Pearlstein[+] and Jary have been converted to values for the cross-section leading to ^{236m}Np and the results are plotted in Fig. 2. Clearly, the calculation of Jary seems to agree rather well with most of the differential data whereas the ENDF/B V calculation is generally more than a factor of two lower.

The cross-section for the production of ^{236m}Np, averaged in a ^{235}U fission neutron spectrum, is estimated to be 1.10 mb from the ENDF/B V data and 1.88 mb from the Jary calculation. This latter value is significantly larger than the measurement of Paulson and Hennelly (1974). However, before concluding that there is a discrepancy in the strict sense of the word, it should be noted that ~95% of the average arises from the energy region between 7 and 12 MeV where there are almost no measurements. The 14 MeV region, where most of the measurements have been made, plays

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[+]In practice, the ENDF/B V values of the total (n,2n) cross-section were used, these being based on the calculations of Pearlstein (1965), renormalised by a factor of approximately 0.89 to give agreement with the integral measurement of Paulson and Hennelly (1974).

an essentially negligible part. The "discrepancy" is therefore not between the integral data and the measurements in the region of 14 MeV, but between the integral data and the integral value calculated by folding together the fission neutron spectrum and the (n,2n) cross-section obtained essentially entirely from theory. Moreover, the isomer formation ratio, and particularly its energy dependence, is a key parameter in this comparison and yet there is only one measured value, and that relates to the 14 MeV region which is of no great importance.

It would seem therefore that there may be no real discrepancy (except for the obvious one between Pearlstein and Jary) but rather a lack of good data on which to base evaluations. The problems of apparent inconsistency among the various data may simply be the result of wrong assumptions in the data analysis or comparisons. More specifically, the problem could be due to one or more of the following:

(a) The spectrum in which Paulson and Hennelly made their measurement may have deviated from a true ^{235}U fission spectrum. Scattering could have caused the spectrum to be degraded, shifting some neutrons to lower energies, giving rise to a lower average cross-section.

(b) The isomer formation ratio obtained by Myers et al (1975) may not be independent of energy. If the ground state were fed more strongly below ∿12 MeV than at 14 MeV, then the integral measurement of Paulson and Hennelly could possibly be brought into line with calculations like that of Jary.

(c) The fission spectrum averaged cross-section, calculated from differential data, is very susceptible to errors in the high energy component of the assumed spectrum, this being a region in which the uncertainties are quite large.

(d) The calculations of the $^{237}Np(n,2n)$ cross-section could be in error by up to ∿30%.

Comments and recommendations

The reasonably good agreement and the relative unimportance of the data in the region of 14 MeV leads to the conclusion that there is little point in doing further measurements in this region, at least by activation techniques. However, as all the measurements have been performed by activation methods, a measurement of the (n,2n) cross-section by counting neutrons would be valuable. This would serve to check the

activation measurements and the $^{236g}Np/^{236m}Np$ formation ratio.

Measurements of the $^{237}Np(n,2n)^{236m}Np$ cross-section are required between ∿8 and 11 MeV. If measurements of the total (n,2n) cross-section could also be made in this energy range, say by counting neutrons, then the isomer formation ratio could be determined.

There is a need to perform an integral measurement of the (n,2n) cross-section in a well characterised spectrum to check on the measurement of Paulson and Hennelly and to compare with the average cross-section derived from differential data and a knowledge of the spectrum.

Acknowledgement

The preparation of this report was aided by a contribution, from Dr. A. Michaudon to Dr. M. G. Sowerby, on the present status of the $^{237}Np(n,2n)$ cross-section, and the use of this is gratefully acknowledged.

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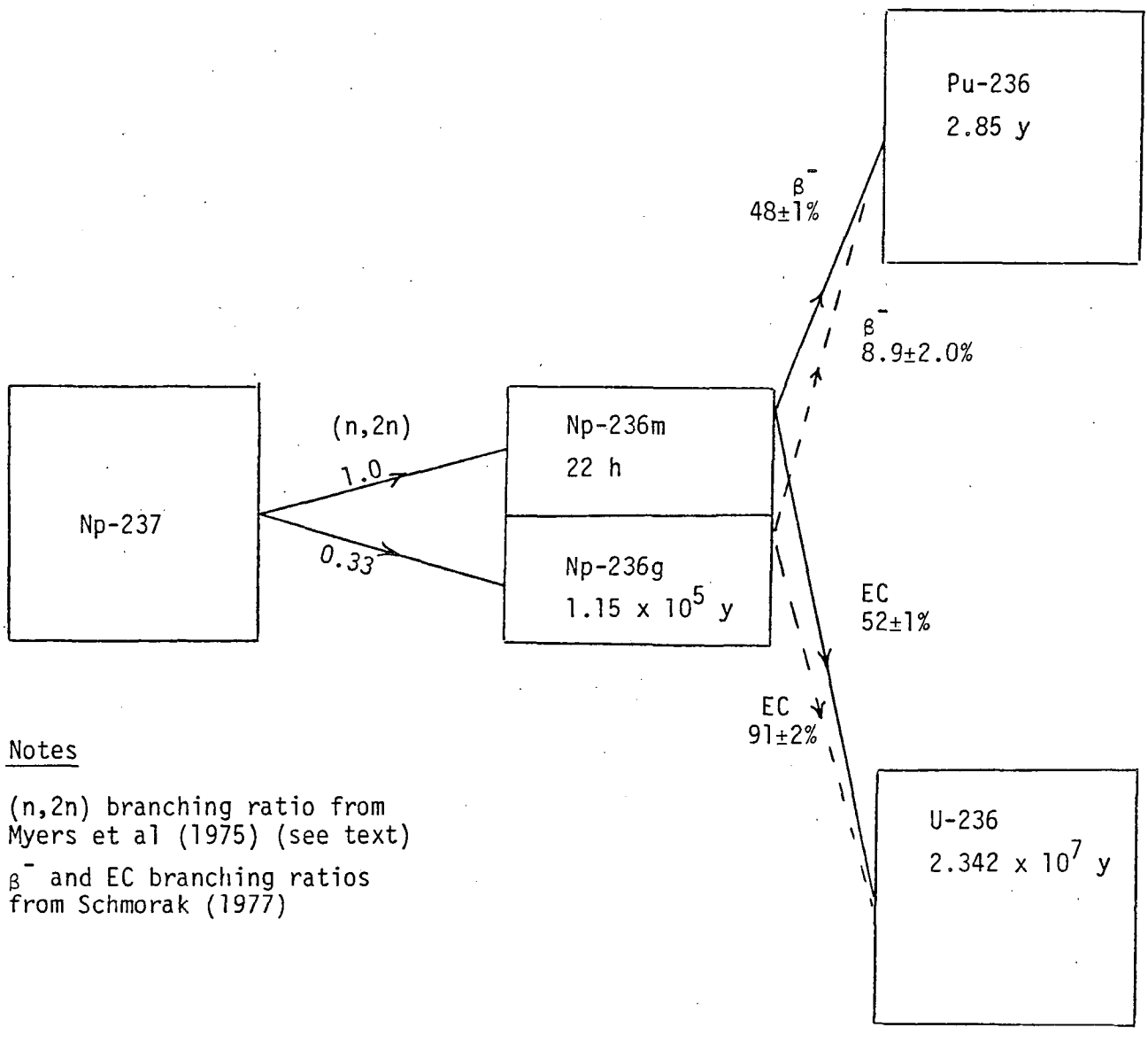

Notes

(n,2n) branching ratio from Myers et al (1975) (see text)

β^- and EC branching ratios from Schmorak (1977)

Fig. 1 A simplified decay scheme of the $^{237}Np(n,2n)$ reaction

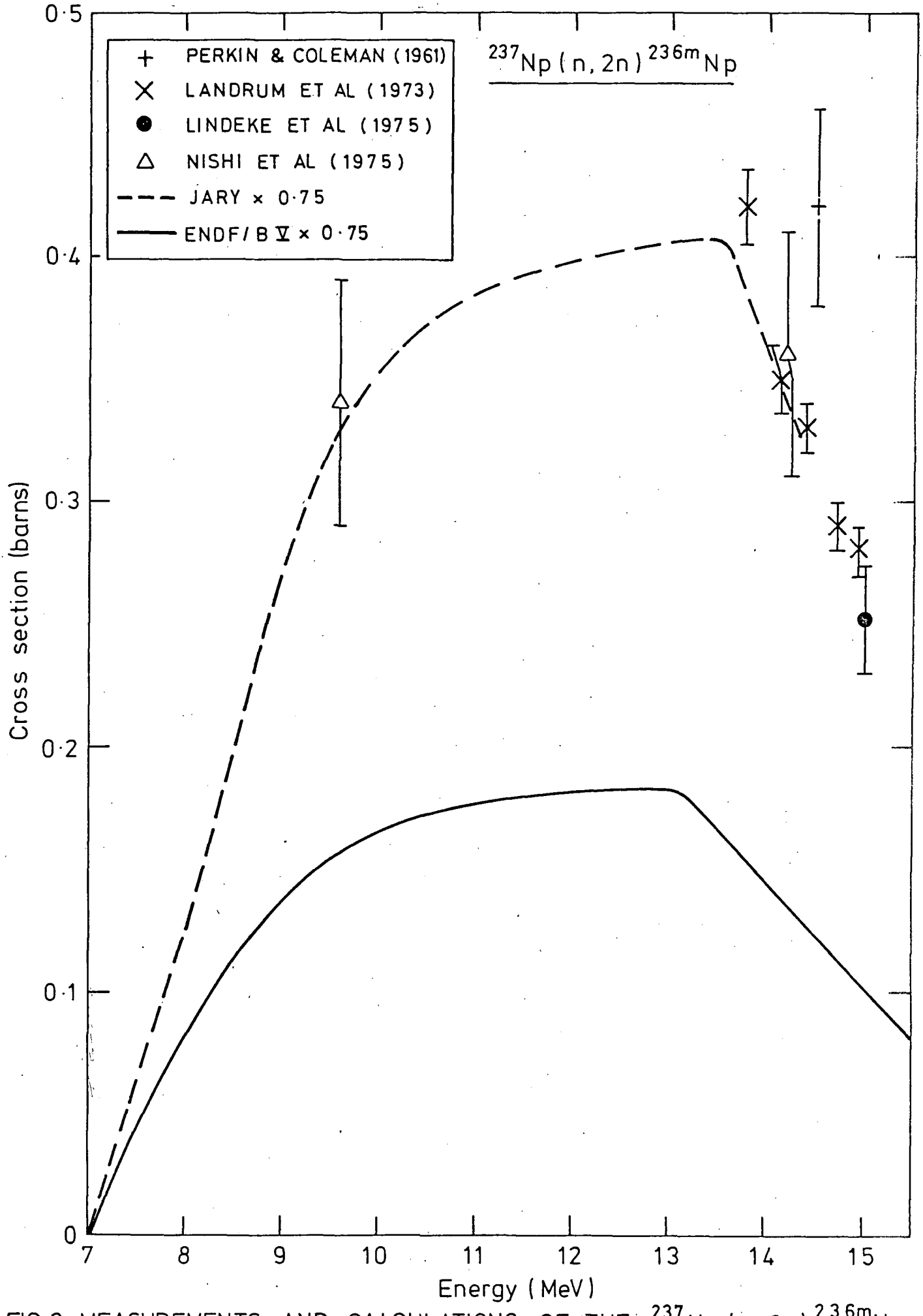

FIG.2. MEASUREMENTS AND CALCULATIONS OF THE $^{237}Np(n,2n)^{236m}Np$ CROSS SECTION.

^{239}Pu DECAY POWER DISCREPANCY

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A. BACKGROUND

An extensive effort was undertaken in the time period 1973-1978 to improve the basic decay-power standard used in U. S. reactor calculations. The following activities were included in this effort and were completed in 1978:

1. The U. S. Department of Energy (DOE--then ERDA) formed a task force to add fission-product decay, cross-section, and yield data for 825 nuclides to its evaluated data file (ENDF/B-IV) and supported efforts to combine analyses using these data with experiments to produce an improved standard.

2. The U. S. Nuclear Regulatory Commission (NRC) supported new benchmark experiments at the Los Alamos Scientific Laboratory (LASL) and Oak Ridge National Laboratory (ORNL) to accurately measure decay heat following fission of ^{233}U,* ^{235}U, ^{239}Pu, and ^{241}Pu.+

3. The Electric Power Reseach Institute (EPRI) supported new decay-heat experiments at the University of California (Berkeley) and at Intelcom Rad Tech (IRT).

The participants and results of these activities were brought together in an American Nuclear Society (ANS) standards committee, and a new ANS Decay Power Standard[1] was approved in 1979. The final standard was determined from a generalized least-squares analysis that combines summation calculations using the ENDF/B-IV data base with these and other decay-heat measurements. In obtaining the standard, a concerted effort was made to reconcile the various decay-heat measurements and the yield and decay data in the ENDF/B-IV data file. The development of the standard, the experimental data that were used, and the ^{239}Pu discrepancy noted below are described in Refs. 2 and 3.

B. ^{239}Pu DISCREPANCY

The decay-heat experiments carried out at LASL[4] and ORNL[5,6] are the most precise of the various measurements and therefore exert the most influence on the new decay-heat standard. The two experiments employ entirely different techniques. In the LASL experiment the decay heat of irradiated fissile samples was measured directly in a precision calorimeter, whereas energy spectra of decay gamma rays and beta particles were measured at various times following relatively short irradiations in the ORNL experiment. The uncertainties in the two experiments range from 2-1/2 to 4%; however, the results differ by approximately 10% for the ^{239}Pu measurements. (The ^{235}U results also differ but by a smaller amount.) The discrepancy between the LASL and ORNL measurements is shown in Fig. 1, where the plots have been derived for a common irradiation time.[2] The measurement in France by Lott and Fiche tends to support the LASL result but has a relatively large uncertainty.

*LASL only.
+ORNL only.

Summation calculations using the ENDF/B-IV data base (and calculations using an independent U.K. data base by Tobias) do not clarify the discrepancy. There is excellent agreement between the calculations and the LASL measurement of ^{235}U decay heating, but for ^{239}Pu the calculations agree better with the ORNL results. Figure 2 shows the ratio of calculated to measured decay heat for the LASL ^{235}U and ^{239}Pu results following a 20 000-s irradiation.

References 2 and 3 discuss the unsuccessful efforts to resolve this discrepancy in deriving the new decay-heat standard. The new standard for ^{239}Pu is intermediate between the LASL and ORNL measurements, whereas the ^{235}U standard is consistent with the measurements from both labs. However, the ^{239}Pu discrepancy could also reflect problems for the ^{235}U standard and should be resolved as soon as possible.

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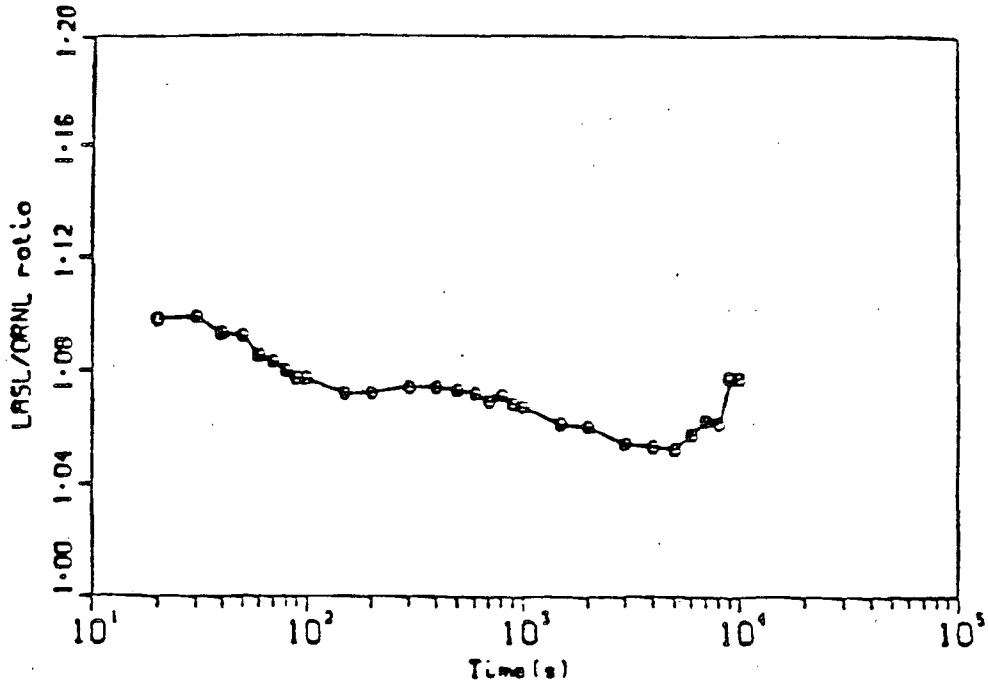

Fig. 1. ^{239}Pu decay-heat ratio of LASL to ORNL following a 20 000-s irradiation.

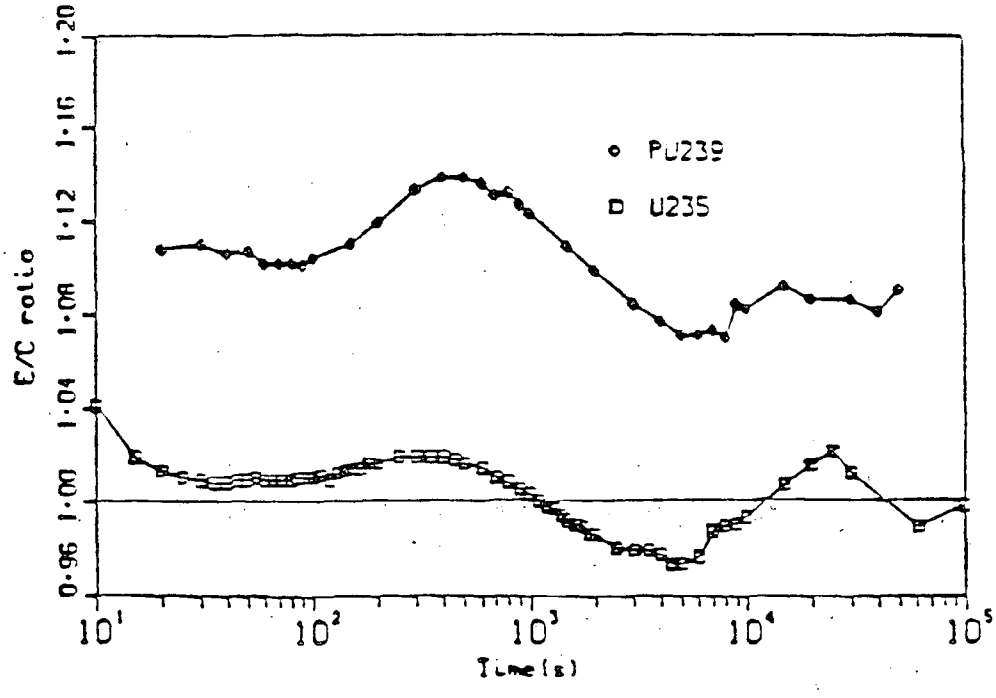

Fig. 2. Ratio of LASL decay-heat measurements to ENDF/B-IV calculations (20 000-s irradiation).

Am-241 fission resonance integral

Description of data and its application

Resonance integrals have an important application in the optimisation or calculation of the production of higher actinides in thermal reactors. Since measurements of resonance integrals can be made with fairly good precision, they can also act as a useful check on the low energy resonance parameters from which they can be calculated. This is probably the most important use for the fission resonance integral of Am-241 as the capture resonance integral is more than 50 times larger and therefore dominates.

Nature of discrepancy

The measurements of the Am-241 fission resonance integral, which claim an accuracy of between ±6% and ±10%, are about a factor of two higher than the value calculated from measured resonance parameters. It is to be noted that the presence of a resonance at 0.576 eV causes some difficulty in the choice of the cadmium cut-off energy to be used. However, a recent evaluation by Lynn et al (1979) has shown that no reasonable value of the cut-off energy can produce a calculated resonance integral which is in agreement with the measurements and therefore the discrepancy is not simply due to an incorrect cut-off energy.

Status

Table 1 lists the known measured values of the fission resonance integral together with the values obtained from various evaluations of the differential data. For the purposes of calculation, the cadmium cut-off energy used in the estimation of the resonance integral from the ENDF/B V and UKNDL (Lynn et al 1979) evaluations was 0.5 eV. To illustrate the effect of changes to this value, the resonance integral calculated from the recent UKNDL evaluation using a cut-off energy of 0.4 eV is 11.1 barns.

The resonance integral value arises from the contributions of the low-lying resonances (excluding the one at 0.308 eV as this is below the cadmium cut-off), the resonances at 0.576 and 1.276 eV accounting for approximately two-thirds of the integral. The important resonance parameters are Γ_f and $g\Gamma_n$. There are several sets of measured $g\Gamma_n$ values, including those by Derrien and Lucas (1975), Weston and Todd (1976) and Kalebin et al (1976), and the agreement between these is good which suggests that wrong $g\Gamma_n$ values are unlikely to be responsible for the discrepancy.

There are four known measurements of the fission widths of the low-lying resonances, by Derrien and Lucas (1975), Gayther and Thomas (1977), Bowman et al (1965) and Gerasimov (1967). Only the last two extend down to resonances below 1 eV and therefore cover the important resonance at 0.576 eV. On the whole, there is reasonable agreement between the various sets of Γ_f values up to ~15 eV, the maximum energy for which comparison is possible and certainly there is no sign of a factor of two error which would explain the resonance integral discrepancy.

As a result of these comparisons, further details of which are to be found in Lynn et al (1979), including the values of the resonance parameters, the source of the discrepancy would not appear to be the result of incorrect resonance parameters.

Turning to the measurements of the resonance integral, it is hard to see how these could be wrong by about a factor of two. A small amount of impurity with a high resonance integral could possibly have a significant effect and Am-242m, with I_f~1570 barns, would seem to be the most obvious candidate. But it is extremely unlikely that the Am-241 samples used would contain the 0.8% or so of Am-242m which would be required and in any case the measurers were totally aware of the problems of impurities. Furthermore, in each of the three resonance integral measurements, the sub-cadmium thermal average fission cross-section was also determined and found to be in reasonable agreement with other measurements, thus supporting the believed purity of the samples.

The low value of 3.1 barns for the Am-241 fission cross-section at 0.0253 eV would seem to exclude the possibility that epithermal neutrons, which passed through the cadmium enclosure and subsequently thermalised, could be responsible for enhancing the resonance integral. Similarly, it seems inconceivable that sufficient neutrons, having penetrated the cadmium, could scatter into the resonance at 0.308 eV giving a significant contribution to the resonance integral.

We therefore appear to have reached an impasse with both sides having about the same claim to being correct and therefore equally impregnable.

Recommendations

Possibly the weakest link in the comparison lies in the fission width determinations of the low-lying resonances and this area might therefore be the most rewarding to investigate. It might also be

useful to make a sample suitable for a resonance integral measurement from the same material as that used for the fission measurement. In this way, if the differential measurement showed no significant impurities to be present, and if the fission widths emerged unchanged, the integral measurement could be performed in the knowledge that impurities could not affect the result. If, after this programme has been completed, the situation remains unresolved then it will be necessary to investigate the $g\Gamma_n$ values.

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Table 1. Values of the Am-241 fission resonance integral (I_{nf}) and the assumed cadmium cut-off energy (E_{CD})

Measured

Author	I_{nf} (barns)	E_{CD} (eV)
Bak et al (1967)	21 ±2	N.S.
Zhuravlev et al (1976)	27.7±1.6	0.52
Gavrilov et al (1977)	22.5±1.7	0.68

Evaluated

Author or evaluation	I_{nf} (barns)	E_{CD} (eV)
ENDF/B V	8.2	0.5
Lynn et al (1979)	10.6	0.5
Maino et al (1979)	8.1	N.S.

N.S. = not specified

^{235}U, ^{238}U, ^{239}Pu RESONANCE PARAMETERS

Introductory comments

The status of neutron cross-section measurements and evaluated data of U 235, U238, Pu 239 in the resolved resonance region has been reviewed recently by G.A KEYWORTH and M.S MOORE (1) in an invited paper given at "the international conference on Neutron Physics and Nuclear Data for reactors and other applied purposes". This paper contain the references to the main works connected with these actinides. The authors have examined the results of the recent experiments and evaluations and have proposed some important extensions due to their own works ; they have concluded about the future experiments which should be undertaken to obtain some improvements in the existing set of data. Since nothing new had been obtained after the HARWELL conference, it does not seem necessary to return to an other review of the available data. The author of this note will only give his own opinion on the conclusions of KEYWORTH et al. and will, if possible, bring some new conclusions firstly concerning each isotopes and secondly about some general problems.

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U 235

When evaluating the resonance parameters of U 235, two sets of data should be reasonably used as the starting point of the evaluation. These data are the following :

1°) The results of the old total cross-section measurements of MICHAUDON et al.(2) performed at liquid nitrogene temperature with a very high quality of resolution ; these data still remain the most important set of $2g\Gamma^{o}_{n}$ values available up to 150 ev neutron energy ;

2°) The results of the fission measurements of KEYWORTH et al with polarized neutron beam and polarized target recently published by MOORE, KEYWORTH et al. (3).

A simple examination of these two sets of data shows immediatly what improvements can be obtained when using polarized neutron beam and polarized target. The spin-separated fission cross-sections show off a large number of small resonances (14 in the 0 - 50 ev energy range) and about the same number of relatively large resonances, which are not apparent in the total cross-section. The former correspond to the 20 % of missed weak levels (small value of $2g\Gamma^{o}_{n}$) foreseen by MICHAUDON et al. ; the later were not expected from the examination of the total cross-section of Saclay, and correspond to doublets in the non spin-separated data. As a consequence, the corrected mean level spacing of 0.54 ev obtained by MICHAUDON et al. is much larger than the value of 0.44 ev proposed by MOORE et al. Nevertheless, the 20 % of missed levels resulting from the Saclay analysis were confirmed by a Monte-Carlo simulation of the cross-sections. But the average fission width used to calculate the cross-sections was too small (44 mev) compared to the values of 196 mev (spin 3) and 91 mev (spin 4) obtained by MOORE et al. from the spin-separated fission cross-sections. Consequently the effect of the resonance overlapping was reduced and a large number of unresolved doublets was not observed in the Saclay simulated cross-sections.

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The existence of this large number of doublets explains why the average Γ_γ value obtained at Saclay is 25 % larger than the one used by MOORE et al. It is now possible to identify the few well isolated resonances in the total cross-section for which the Saclay shape analysis should give an accurate value of Γ_γ. There are only six resonances of this type :

ENERGY (ev)	Γ_γ (mev) of MICHAUDON et al.
2.03	36
4.84	37
7.07	36
10.18	37
11.66	36
16.09	37

These values are in excellent agreement with the value of (35 ± 2) mev choosen by MOORE (4) from a systematic study of s and p wave capture widths.

The set of resonance parameters recommended by MOORE et al does not contain the entire accuracy that should be expected from the analysis of the spin-separated fission cross-sections. Two recommandations should be made :

1°) A new shape analysis of the Saclay total data should be undertaken by using as starting point the new informations obtained from the spin-separated data, i.e the exact position of the resonances and more accurate values of Γ_f and Γ_γ. This new analysis should lead to very accurate values of $2g\Gamma_n^\circ$ which could then be used to analyse KEYWORTH et al. data ;

2°) A multi-level analysis could be done easily on the spin-separated fission cross-sections, at least on the spin 3 cross-section. For this spin state the fission widths are larger and the number of fission channels smaller ; consequently the level-level interferences are more important than in the spin 2 state ; that is clearly seen on the spin-separated fission cross-sections. Such a multi-level analysis (using two fission channels

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for the spin 3 state) should lead to more accurate fission widths and should give more confidence in the identification of some very weak resonances.

U 238

Concerning the resonance parameters of U 238, the situation seems to be seriously improved since the evaluation presented by MOXON at the 1974 specialist meeting of Saclay (5).

New data are available and the very recent evaluation of DE SAUSSURE et al (6) complemented by some remarks of KEYWORTH et al. seems to provide a definitive answer to several questions. The method used by DE SAUSSURE et al. shows how it is possible to conciliate several large sets of experimenal data apparently conflicting by a careful study of the possible systématic errors.

Pu 239

This nucleus has always been considered as a nice exemple for the study of the nuclear properties in the resonance region especially for the spin assignments and the fission widths. However the review by KEYWORTH et al. brings in some new problems which need to be solved ; these new problems concern the mean level spacing and the spin assignments. KEYWORTH et al. have used the method of moments to compare against the Porter-Thomas distribution for neutron widths of the levels assumed to be 1^+. They suggest that the 1^+ average spacing is (2.62 ± 0.24) ev instead of (3.2 ± 0.20) ev as given by the Saclay group (7). That means that one must admit 25 % of missed levels in the total cross-section measured at Saclay with a very high quality of resolution, while these experimentators predict only 5 %. KEYWORTH and al. have the feeling that 20 % of weak 1^+ levels could be missed and that a large part of the levels assigned 0^+ should be 1^+ levels. Indead, it is likely that some of the resonance spins are not assigned correctly, but the existence of 20 % of missed weak levels is questionnable. On the other hand, it is hard to believe that the problem concerning the resonance identification for Pu 239 is as

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difficult as that for U 235. It would therefore seem that the problem is still far from being completely solved. It is also obvious, that a better answer should be obtained by undertaking a fission or a total cross-section measurement of Pu 239 by using polarized neutrons and polarized target.

Some general recommendations

1°) One of the main problems arising from the study of the resonance parameters of the actinides is the exact determination of the s-wave level spacing. It is obvious that it is impossible to identify all the resonances in the experimental cross-section, even if the resolution is excellent. A correction needs to be applied to the observed level spacing to obtain a value as close as possible to the unknown value. The methods used to obtain the corrected value are numerous : least square fitting or maximum of likelihood on the Porter-Thomas distribution, Δ 3 statistics, examination of the Wigner distribution, simulation of cross-sections by Monte-Carlo method etc... But, having regard to the disparity of the results obtained by the different methods applied to the same data, it is urgent to consider the following points :

a) Should the level spacings and the reduced neutron widths follow strictly the Wigner and the Porter-Thomas law. If not, what deviation should we expected or accepted ?

b) Is it possible to "standardize" the methods used in the investigation of the missed levels ? One should establish some rigorous conditions of utilisation of these methods. For instance, it appears that one method works well when applied to a particular case and give inconsistant results when applied to another case. To avoid misunderstanding in the interpretation of the results, a maximum of details of the analysis should be given.

2°) The data abtained from a good transmission measurement should be the obligatory starting point for the evaluation of the resonance parameters of a fissile nucleus. Now, in the resonance

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region, thereare few transmission measurements available whith very good quality of resolution. It is regrettable that some U 235 or Pu 239 transmission measurements similar to those performed at Saclay 15 years ago have not been undertaken on ORELA or GELINA.

3°) As pointed out by DE SAUSSURE et al. one should avoid to recommend a set of resonance parameters obtained by averaging all the data available in the litterature. This type of evaluation leads to a set of data which does not represent the reality and which is not representative of a particular experiment. In such a data set it is impossible to preserve the correlations which exist in a serie of mesurements or in several series of measurements, unless all the data sets are consistent. In the last case, averaging the data or choosing a particular data set will lead to about the same results. In the case of large discrepancies, one must try to show off the systematic errors which are the cause of the discrepancies ; it is then possible to "adjust" the data and to obtain a consistent serie of values on which the average procedure could be applied. This method has been used by DE SAUSSURE et al. for the evaluation of U 238 on the large sets of data from Oak-Ridge, Columbia and Geel. KEYWORTH et al. have also "adjusted" the Columbia Th 232 data to the Saclay data by correcting the Columbia data. One must point out that this kind of "adjustment" has been already suggested in 1970 by RIBON et al. (8) who have shown, by applying the least square shape analysis method to the Columbia data, that the discrepancies between Columbia and Saclay were mainly due to an underestimation of the background in the Columbia experiment. One should also mention that the results of KEYWORTH et al. Th 232 evaluation are consistent with the evaluation performed at Saclay in 1973 (9) and based on the Saclay transmission data.

H. DERRIEN, February 1979

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DELAYED NEUTRONS

The following conclusions are largely extracted from the results of the IAEA Consultants' *Meeting in* Vienna, 26 - 30 March 1979[1].

Integral properties, i.e. those resulting from the bulk of fission products in nuclear fuel, and properties of individual products are treated separately.

1. Integral properties of delayed neutrons in nuclear fuel

1.1 Delayed neutron yields

The most stringent reqirements are for the interpretation of critical experiments which demand an accuracy of about ± 2 % of the yields. We are close to this goal for ^{235}U and ^{238}U, but more work is needed especially for ^{232}Th, ^{233}U, and ^{239}Pu. Moreover, the dependence of the yields on neutron energy should be better investigated.

Since the properties of individual delayed-neutron precursors, for instance the branching ratios, are often quite well known, the yield can be obtained not only by integral measurements but also by summing the contributions from the precursors. This might well be the best way to obtain yields for the heavy plutonium isotopes and for isotopes of americium and still heavier elements. It must be borne in mind, however, that this procedure throws the difficulties over to the field of fission yields. The accuracy of the evaluation will strongly depend on how well the fission yield pattern is known.

1.2 Delayed-neutron energy spectra

Data on the time dependence of group spectra is sparse and should be improved by new measurements. As an alternative approach, especially for fissionable materials which are difficult to measure, group spectra can be constructed from individual precursor data.

2. Properties of individual delayed-neutron precursors

2.1 Branching ratios

There are a number of cases for which several determinations exist but where the values and errors given clearly indicate systematic deviations between different laboratories/methods. The reason for such deviations should be tracked down and the measurements, if possible, corrected. To this group belong: ^{85}As, ^{88}Se, ^{89}Br, ^{93}Rb, ^{97}Rb, ^{138}I, $^{141-145}Cs$, $^{127-131}In$.

In certain cases only one experimental determination is available, which calls for at least one independent determination. To this group belong: $^{79-82}Ga$, ^{84}As, ^{87}As, ^{89}Se, ^{91}Se, ^{99}Rb, $^{97-99}Sr$, ^{97}Y, ^{98}Y, ^{134}Sn, ^{137}Te, ^{138}Te, ^{147}Ba, ^{148}Ba, ^{147}La.

For a few precursors - ^{83}Ga, ^{83}Ge, ^{84}Ge, ^{123}Ag, ^{133}Sn, ^{143}Xe - the branching ratio still remains to be measured.

New precursors of importance should be expected among the isotopes of light rare-earth elements.

2.2 Energy spectra

Energy spectra of delayed-neutrons have been measured using different techniques. There is a definite disagreement between results obtained, however, especially between the ^{3}He-spectrometer results and results obtained with a proton-recoil spectrometer. This probem has to be resolved. Also alternative techniques should be tried.

Most of the measured spectra extend from about 80-100 keV up to a few hundred keV below the upper limit. Both the low energy part and the high energy part are missing which means that complementary measurements should be carried out. Among the more important precursors to be further studied are $^{88-91}Br$, ^{134}Sn, ^{139}I, and ^{140}I.

No information about the neutron spectrum is available for the following group of precursors:

^{92}Br, ^{93}Kr, ^{94}Kr, $^{98-100}Sr$, $^{97-101}Y$, $^{137-139}Te$, and ^{141}I.

It is important that the experimentalists evaluate the uncertainty in the measured spectra.

2.3 Mean neutron energies

If the spectrum is known the mean neutron energy can be evaluated. The mean energy can also be measured directly. This has been done for a few cases[2)], and the work should be extended. Directly measured mean values are valuable as consistency checks on the spectra.

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