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STATUS OF NEUTRON CROSS SECTIONS FOR REACTOR DOSIMETRY

by

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M. F. Vlasov,[*] A. Fabry,[**] and W. N. McElroy[+]

I. INTRODUCTION

Reactor dosimetry is aimed at providing the capability to properly correlate, interpolate and/or extrapolate integral quantities, such as flux and fluence, fission rates, burn up, damage rates, doses, heating rates, etc. [1-5]

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

Goal accuracies for determining such transfer functions are in the range of 2-5% (1σ) for integral and 2-15% (1σ) for differential results.[1-5] This in turn requires the development and use of a consistent set of differential-energy dependent cross sections for dosimetry nuclear reactions that have good relative and integral accuracy in the range of a few percent. The concept and the role of reference neutron fields in determining and validating the accuracy of cross sections and transfer functions are best illustrated (Fig. 1) by considering the crucial issue of applying high flux test reactor materials property-change data to commercial power plants.

Fluence neutron spectra for high flux test reactors are usually unfolded from a set of measured reaction rates R_i. Property changes observed in a series of such fluence neutron spectra are correlated by adjustment of theoretically based damage functions so as to provide consistent life time predictions for materials and components for commercial plants. Such an adjustment procedure will be biased if the unfolded fluence neutron spectra are not consistent with the ones resulting from design computations.

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The most practical and accurate way to avoid such bias is to adjust the differential-energy dependent cross sections of dosimetry reactions within their assigned uncertainties so as to reproduce the reaction rates observed in a set of benchmark reference neutron fields. This is possible provided that:

1) the reference neutron fields span a relevant range of spectral hardness and shape,

2) the neutron spectrum characterization of these reference neutron fields is complete and accurate, and that

3) the available integral reaction rate data are sufficiently accurate.

The status of current international efforts to develop standardized sets of evaluated energy-dependent (differential) neutron cross sections for reactor dosimetry is reviewed in subsequent sections of this paper. The status and availability of differential data are considered first, some recent results of the data testing of the ENDF/B-IV dosimetry file using ^{252}Cf and ^{235}U benchmark reference neutron fields are presented, and finally a brief review is given of the current efforts to characterize and identify dosimetry benchmark radiation fields.

II. DIFFERENTIAL-ENERGY DEPENDENT CROSS SECTIONS

International recognition of the need to develop standardized and consistent sets of evaluated energy-dependent dosimetry cross sections is evident in the recommendations of the IAEA Consultants' Meeting on Nuclear Data for Reactor Dosimetry [1], in the report on the U.S. Interlaboratory LMFBR Reaction Rate (ILRR) programme [3], in the proceedings of the First ASTM-Euratom Symposium on Reactor Dosimetry [4] and elsewhere [2;5].

An important step in establishing better consistency and accuracy will be the continued development of the ENDF/B file for dosimetry applications [3]. Version IV of this file [6] is now available through the four neutron data centres at CCDN Saclay, Obninsk, Brookhaven and IAEA.

Other files that are available are: SAND-II and DETAN-74 [7], UK [8], KEDAK [9], Chalk River [10], Argonne National Laboratory [11], USSR [12], Lawrence Livermore Laboratory [13].

The field of the differential data for reactor dosimetry has recently been reviewed by A. Paulsen and B. Magurno [4]. They indicated, in particular, that the situation in the field of fast neutron activation reactions is unsatisfactory and there has been little progress. The reasons for this unsatisfactory situation are the following [1]:

a) The lack of agreement for a limited set of reactions on which all measuring efforts are concentrated.

b) The failure to concentrate the differential measurement efforts on the most sensitive energy region for dosimetry purposes. In the case of (n,p), (n,α) and (n,2n) reactions this region extends from threshold to 5.5 - 6.5 MeV above threshold. It comprises 90 - 95 % of the total response in the fission neutron spectrum.

c) The lack of a sufficient number of laboratories equipped with accelerators which can produce monoenergetic neutrons in the 6-12 MeV region and which are used for neutron measurements.

In this section the status of this category of the data (threshold reactions) will be discussed briefly. In Appendix 1 excitation functions (energy-dependent cross-sections) are plotted for some threshold reactions commonly used for neutron spectra unfolding by the foil-activation technique. All published experimental data are plotted together with the available evaluations.

Many measurements in the past were performed with monoenergetic neutrons from inexpensive low-voltage Cockcroft-Walton neutron generators and electrostatic generators with a maximum energy of the accelerated protons (or deuterons) up to 3 MeV, using D $(d,n)^3He$, $T(d,n)^4He$ and $T(p,n)^3He$ reactions.

As a result of this historical development, three general observations can be made concerning the present status of knowledge of the excitation functions for many threshold reactions:

a) the gap between 6 and 12 MeV mentioned above could be covered with such accelerators;

b) shortage (or low quality) of the data near threshold where a high intensity of monoenergetic neutrons is needed due to the low values of the cross sections;

c) relatively many measurements of different quality between 13 and 15 MeV.

Measurements covering the total energy range of interest have been performed, e.g. at Chalk River (D.C. Santry, J.P. Butler), CBNM Geel (H. Liskien, A. Paulsen), Los Alamos (R.S. Prestwood, B.P. Bayhurst et al.) and more recently at Argonne (D.L. Smith, J.N. Meadows), and Bruyères-le-Chatel (J. Fréhaut, G. Mosinski).

The Geel group which is equipped with an ordinary electrostatic generator has obtained its results by such reactions as $^{9}Be(\alpha,n)^{12}C$ (6-8 MeV neutrons) $^{14}C(d,n)^{15}Ng$ (8.5 - 10 MeV), and $^{15}N(d,n)^{16}Og$ (10-11.5 MeV). However, the progress in this field is closely connected with big machines such as the Tandem accelerators in Chalk River and Bruyères-le-Chatel and especially with the Argonne Tandem Dynamitron accelerator which can provide high intensity monoenergetic neutrons in a wide energy range.

In the following we shall briefly consider some of the important reactions for which large discrepancies between measured and calculated integral results exist (Table 1):
$^{48}Ti(n,p)^{48}Sc$, $Ti(n,x)^{46}Sc$, $^{47}Ti(n,p)^{47}Sc$, $^{63}Cu(n,\alpha)^{60}Co$, $^{63}Cu(n,2n)^{62}Cu$, $^{90}Zr(n,2n)^{89}Zr$ and $^{58}Ni(n,2n)^{57}Ni$.

The status of the excitation functions of some other reactions can be seen from Appendix 1.

$^{48}Ti(n,p)^{48}Sc$, Fig. 8: This reaction is a good example of the importance of integral measurements. Its effective threshold (7.6 MeV) is close to that of the standard reactions $^{27}Al(n,\alpha)^{24}Na$ (7.2 MeV) and $^{56}Fe(n,p)^{56}Mn$ (6 MeV).

For these two reactions the excitation functions are established with an accuracy of about $\pm$ 5% and good agreement exists between the measured and calculated integral cross section $\langle\sigma\rangle$ (Table 1). This means that the neutron spectra used for the calculation of $\langle\sigma\rangle$ are reasonably well determined in the response range of these detectors. 90% response of the $^{48}Ti(n,p)^{48}Sc$ detector to these spectra is in about the same energy range as that of the two above mentioned standard reactions ($\sim$4.7 - 11 MeV, using the "preliminary evaluation" curve, Fig. 8).

Integral measurements of $^{48}Ti(n,p)^{48}Sc$ in the ^{235}U thermal fission spectrum are accurate to $\pm$ 6% (recommended value 0.300$\pm$0.018 mb [14]. (Precision measurement in the ^{252}Cf spontaneous fission spectrum yields 0.42 $\pm$ 0.01 mb [15] (Accuracy $\pm$2%).

The evaluation of this reaction for ENDF/B-IV has been done when no experimental data were available between 6 and 12.5 MeV. The $\langle\sigma\rangle$'s calculated from this evaluation are 70% lower than the measured integral values given above. Later accurate differential measurements performed at Argonne in the energy-range 6-10 MeV [Smith 75, Fig. 8] show much higher values.

The curve "preliminary evaluation" based on the new Argonne measurements gives $\langle\sigma\rangle$ values of 0.265 mb, 0.264 mb and 0.396 mb for the fission spectra of ^{235}U (NBS evaluation), ^{235}U (SAND-II adjusted) and of ^{252}Cf (NBS evaluation) respectively which are in much better agreement with the integral values (bias factors are 1.132, 1.136 and 1.061 for the above mentioned spectra). Consideration of the later renormalization of the Argonne data to the new ENDF/B-IV ^{235}U and ^{238}U fission cross section data [11], especially between 5.5 and 7 MeV will still improve this situation.

$^{46}Ti(n,p)^{46}Sc$, Fig. 6: For ^{235}U fission neutron spectrum this reaction is practically equal to $Ti(n,x)^{46}Sc$: contribution to the total production of ^{46}Sc in natural titanium is ~99%. The effective threshold is at 3.9 MeV and 90% response in fission spectrum covers the energy range between 3.4 and 9.1 MeV.

Between 6 and 8.5 MeV recent Argonne data [Smith 75] are slightly higher than ENDF/B-IV cross sections and lower between 8.5 - 10 MeV. No experimental data are available from threshold to 3.67 MeV and from 10 to 12.5 MeV. Both these energy intervals are important for determination of the excitation function with better accuracy in order to clarify the existing discrepancies between measured and calculated integral $\langle\sigma\rangle$ (~15 - 18 %, Table 1).

$^{47}Ti(n,p)^{47}Sc$, Fig. 7: This is an exoergic reaction with the effective threshold at ~2.2 MeV. The 90% response in the fission spectrum is in the 2.1 - 7 MeV energy range. The contribution of ^{48}Ti $(n,np)^{47}Sc$ reaction to the production of ^{47}Sc is important above 12 MeV, but is negligible ($<0.1\%$) in the case of the fission spectrum.

The ENDF/B-IV evaluation based on [Smith 73] data from threshold to 6 MeV is in good agreement with the recent [Smith 75] measurements (especially after renormalization, see section on ^{48}Ti $(n,p)^{48}Sc$.

Attention should be paid to the value of the β-branch to the 0.159 MeV level in ^{47}Ti. This quantity has been measured by many authors, e.g. Z.T. Bak (1968), S.C. Misra (1964), W.S. Lyon (1955), L. Marquez (1953) etc. [16], and is scattered between 66% and 74%. The value, recommended by Lederer et al. [17] is 73% and that used by the Argonne group is 68.5% [11]. If the Lederer value is accepted, then the Argonne results should be decreased by ~6%.

At present $\langle\sigma\rangle$ calculated from the Argonne differential data is about 12 - 15 % higher than the value recommended from integral measurements (Table 1).

$^{63}Cu(n,\alpha)^{60}Co$, Fig. 16: The effective threshold ~6.2 MeV, and 90% response in the fission spectrum is in the 6.1 - 11.3 MeV energy range. This reaction represents a classical example for the differential-integral discrepancies: $\langle\sigma\rangle$ obtained from integral measurements is systematically about 40% higher than the calculated value. The excitation function is determined by only one set of measurements. [Paulsen 67]. The ENDF/B-IV evaluation is practically an eye-guide curve through these experimental points. The Czapp 60, Barrall 69, Maslov 72 measurements between 14 and 15 MeV give higher values than Paulsen 67. The measurements are performed by activation technique.

A discussion of the possible reasons for differential-integral discrepancies (subthreshold contribution, underestimation of the Co impurities and thermal neutron flux in integral experiments, etc.) can be found elsewhere [1,18,19]. Theoretical considerations, however, indicate higher cross section values near threshold [Benzi 74, Fig. 16]. New differential measurements, especially near threshold and in the 90% response energy range, are needed to solve the present discrepancies.

$^{63}Cu(n,2n)^{62}Cu$, $^{90}Zr(n,2n)^{89}Zr$, $^{58}Ni(n,2n)^{57}Ni$, Fig. 15, 17, 13:

The effective thresholds of these reactions are at ~12.4 MeV, 13 MeV and 13.5 MeV and the 90% responses in the ^{235}U fission spectrum are between ~11.9 - 16.4 MeV, 12.5 - 16.7 MeV and 13.2 - 17 MeV respectively. In the case of these high threshold reactions the large discrepancies between calculated and measured integral values depend strongly upon the representation of the fission spectrum used for the calculation (Table 1).

$^{63}Cu\ (n,2n)^{62}Cu$, Fig. 15: The shape of the excitation function is established well by [Liskien 65] measurements; the slope between 13.5 - 15 MeV is supported by [Csikai 65] and [Cuzzocrea 68] data. In spite of the spread of experimental results in this energy range the value of the cross section is determined reasonably well. Only a few old measurements are available near threshold, where all evaluations are in disagreement. Theoretical calculations [Benzi 74] indicate higher values than the experiment.

$^{90}Zr(n,2n)^{89}Zr^{m+g}$, Fig. 17: Practically no data are available near threshold. The new [Bayhurst 75] measurements covering a wide energy range (from 13 to 28 MeV) are slightly lower than his previous data at lower energies [Prestwood 61]. They are in agreement with [Sigg 75] measurement at 14.8 MeV, performed with an enriched isotope sample. Data of [Kanda 72], [Araminowicz 73] and recent [Eapen 75] are strongly scattered and do not agree with those of Bayhurst and Sigg, while [Nethaway 72] data are systematically slightly higher (~5 %) than Bayhurst and Sigg results.

The [Eapen 75] absolute measurement at 14.8 MeV gives the cross section 50% higher than the available evaluations. The $\langle \sigma \rangle$ measured is at present 70 to 200 % higher than calculated from differential data (SAND-II, Fig. 17).

Accurate measurements between threshold and 15 MeV are needed to improve the reliability of this excitation function, especially to close the gap between threshold and 13.5 MeV.

$^{58}Ni(n,2n)^{57}Ni$, Fig. 13: The four available measurements (Jeronymo 63 is probably wrong) cover a large energy range and are strongly scattered above 15 MeV. Below this energy the shape of the excitation function seems to be rather well established.

The new [Bayhurst 75] data suggest a possible trend which is much higher than all other data and could possibly explain part of the differential-integral discrepancy. Confirmation of Bayhurst's results would therefore be very desirable, particularly as very recent preliminary measurements by Marcinkowski et al. [20] support the earlier lower data.

Because of the strong fission spectrum dependence of the calculated integral values for these high threshold reactions it is not expected that the differential-integral discrepancies are due to the unsatisfactory knowledge of the excitation functions alone.

Other reactions:

There are no measurements in a wide energy range near threshold for $^{55}Mn(n,2n)^{54}Mn$. Much worse is the situation for many other reactions not considered here. (cf. Fig. 9). Only several isolated measurements are available for $^{93}Nb(n,n')^{93}Nb^{m}$, $^{199}Hg(n,n')^{199}Hg^{m}$ etc.

In Section III, the consistency and accuracy of a number of high energy neutron reaction cross sections in the ENDF/B-IV file are studied using the ^{252}Cf and ^{235}U fission spectra reference benchmark fields.

III. DOSIMETRY CROSS SECTIONS VALIDATION AND ADJUSTMENT

Microscopic cross sections averaged over fundamental energy-distributed neutron spectra provide an important way to check and eventually improve differential-energy cross sections, because the convolution of the energy-dependent functions must reproduce (within uncertainties) the measured integral quantities. The measurement of microscopic integral cross sections does not involve the definition of an absolute flux as a function of neutron energy nor the establishment of an accurate energy calibration scale; but, to be useful, it must be performed in a well-known neutron spectral shape and the total absolute flux must be determined. Microscopic differential-energy and integral cross section measurements require both the accurate determination of absolute reaction rates and/or the total number of reactions as well as a number of corrections which must be thoroughly investigated. There are two main reasons for which integral data play a major role in terms of defining acceptable dosimetry cross section files:

1) The requirement of overall integral consistency, as outlined in Section I.

2) The fact that, as shown in Section II, differential-energy cross sections for many nuclear reactions of relevance to dosimetry are poorly established; e.g., they either display untolerable discrepancies in shape and magnitude from one experiment to another, or are characterized by a serious lack of data in significant energy ranges. [4,6 - 13].

By contrast, microscopic integral cross sections not only constitute the needed information for a number of engineering applications but also are often very reliable in terms of the agreement between experimenters: one reason for this lies in the relatively less sophisticated nature of an integral measurement; the other one is probably to be found in the more systematic amount of effort devoted internationally to such type of work in the field of reactor dosimetry (an example is the widely recognized U.S. Interlaboratory ILRR program [3]).

As already mentioned, the crucial requirement for an integral cross section result to be successfully used for the testing and/or improvement of differential-energy cross section files is that the neutron field in which measurements are obtained must be well characterized. Ideally, such a field should be a standard as defined in Section IV.

In practice, this is seldomly realized, but there exists a number of "benchmark" neutron fields -- categorized more specifically as standard, reference or controlled environments [4] -- for which the neutron spectral characterization is rather accurate (or will be), over at least most of the relevant energy range; outside of this range, and sometimes even within this range, some spectral adjustments are necessary and can be done by means of appropriate computer codes, provided they are accomplished with caution and with a serious amount of physical judgment [3]. In this section, the SAND-II Monte Carlo error analysis code [21] is applied to demonstrate the value of an integral approach to the validation and adjustment of neutron fields and dosimetry cross section files. The need and recommendations for such validation and adjustment are considered in more detail in references [1 to 5].

Consider the relationship

$$\bar{\sigma}_j = \int_0^\infty \phi(E)\ \sigma_j(E)\ dE \qquad j = 1...n \qquad (1)$$

between a set of measured integral cross sections $\bar{\sigma}_j$, in a benchmark neutron field of normalized spectral distribution $\phi^j(E)$, and their corresponding differential-energy cross sections $\sigma_j(E)$. Given a physically reasonable representation for the neutron spectrum $\phi(E)$, the SAND-II code iteratively modifies the input shape $\phi^o(E)$ until an adjusted shape $\phi'(E)$ is obtained such that the above relationship is satisfied for all reactions in the set j=1...n. The SAND-II Monte Carlo code is used to generate uncertainties for $\phi'(E)$ in a 15 group structure, based on a Monte Carlo propagation of the errors affecting the measured integral quantities as well as the cross sections $\sigma_j(E)$. [3, 22]

Similarly, considering one nuclear reaction for which integral cross sections have been measured in a set of well-known neutron spectra $\phi_k(E)$, k = 1...K, the code will adjust a reasonable input cross section shape $\sigma^o(E)$ until a solution $\sigma'(E)$ is found that satisfies the set of equations [19]

$$\bar{\sigma}_k = \int_0^\infty \phi_k(E)\ \sigma(E)\ dE. \qquad k = 1...K \qquad (2)$$

In reference 1, nuclear reactions are divided into two categories:

Category I: Those for which $\sigma(E)$ is considered to be well established.

Category II: Those for which $\sigma(E)$ is less well known.

A set of neutron spectra ϕ_k (E) can be adjusted and assigned errors by reference to the Category I reactions using Equation (1); imposing the constraint that the adjusted solutions must match all features of ϕ_k (E) which can be considered well established by:
1) Differential spectrum measurement results in the energy range where they are reliable and
2) theoretically based data.

Once the neutron spectral shapes ϕ_k (E) have been validated and/or improved and assigned errors, a set of Category II differential-energy cross sections can be adjusted, and, with appropriate constraints, can be validated and/or improved.

Such an approach, which was recommended to the IAEA by the 1973 Consultants' Meeting on Nuclear Data for Reaction Dosimetry[1], has been adopted for the present study. In this paper, however, attention will be limited to cross sections in the energy range above $\sim$ 0.1 MeV and the emphasis will primarily be on threshold reactions. Furthermore, and although microscopic integral cross sections are available in a number of benchmark neutron fields [3,4], only the two most fundamental fields are considered here: the spontaneous fission neutron spectrum of californium-252, denoted by χ_{82}, and the thermal neutron-induced fission neutron spectrum of uranium-235, χ_{25}.

The χ_{82} and χ_{25} neutron fields have been historically the most extensively studied since the eve of atomic power. A number of differential measurements of their spectral shapes have been performed and they have been recently evaluated [23] with great care at the U.S. National Bureau of Standards. Over the energy range, where most fission neutrons are emitted, these spectra are reasonably well described by Maxwellian functions [23] with average energies of 1.97 and 2.13 MeV for χ_{25} and χ_{82}, respectively. Departures from the Maxwellian reference shapes, however, are evident and the final NBS evaluated data are therefore presented as Maxwellian functions multiplied by segment-adjusted linear or exponential corrections specified in an energy grid involving a few discrete groups; group uncertainties are also given.

The χ_{82} evaluated differential data are extremely reliable from $\sim$250 keV up to $\sim$8 MeV. With regard to χ_{25}, however, corrections for neutron scattering and absorption within the generally thick fission source samples have been neglected by most experimenters and were not accounted for in the NBS evaluation. Recent and comprehensive time-of-flight measurements have examined this effect [24] and the correction in terms of the spectral averaged energy is of the order of +0.02 MeV, which corresponds to a 4 to 10% flux decrease below and 4 to 20% increase above $\sim$ 1.5 MeV, depending on the energy.

On the integral side, on the other hand, the bulk of the international efforts has been on measurements in χ_{25}, with few, but some recent and careful measurements in χ_{82}. For the present study, the evaluated measured microscopic integral cross sections recommended by Fabry, et al. [14] have been accepted. Thus, the work in reference [14] was used as the starting point of a systematic sensitivity study by means of the latest version [25] of the SAND-II Monte Carlo error analysis code, with solution weighting. The ENDF-B-IV dosimetry cross section file [6] was used as the reference set of differential-energy cross sections. The reference Maxwellian functions for χ_{25} and χ_{82} as defined above were accepted as input spectra and deviations from these forms as unfolded by the code were compared with the NBS evaluations. The sensitivity of the unfolded solution to a given reaction was established, simultaneously, in the two neutron fields by comparing successive computer runs in which the reaction was either withdrawn from the initial set or its cross section was rescaled by some fixed percent fraction. Monte Carlo error propagation was explicitly considered in each case.

A study was made of which combination of reactions could be assigned the label of "Category I" in terms of integral consistency. The guideline was that the NBS evaluation for χ_{82} had to be well reproduced within uncertainties; however, certain deviations were accepted for χ_{25} provided they were consistent with the shape biases expected from the neglect of the fission source sample finite size. In this manner, it was established that the reactions $^{239}Pu(n,f)$, $^{237}Np(n,f)$, $^{238}U(n,f)$, $^{56}Fe(n,p)$, and $^{27}Al(n,\alpha)$ could be assigned the "Category I" label.

The addition of the $^{235}U(n,f)$ reaction to the above set generated spectral distortions in the SAND-II adjusted solutions that were systematic and unacceptable, both for χ_{25} and χ_{82}, especially in the energy range 100-600 keV. The distortions could be removed if $\sigma_{f5}(E)$ in the range 0.1 - 1 MeV were decreased by at least 5% with respect to the present ENDF/B-IV evaluation.

The $^{58}Ni(n,p)^{58}Co$ reaction, a Category I candidate, was also used but its use supports the above concerns regarding the correctness of the χ_{25} spectral shape. Indeed, if it is added to the set identified above, it preserves more or less the consistency for χ_{82}, but induces a shape difference for χ_{25} that is outside of uncertainties with respect to the NBS evaluation.

The uranium-235 fission spectrum averaged cross section for the $^{58}Ni(n,p)^{58}Co$ reaction has been extensively measured and is well established, in particular with respect to the basic fission cross sections [14]; further, it is backed by the data obtained in the CFRMF benchmark. [3] The same quantity in the californium-252 fission spectrum has been determined carefully, but so far by only one experimental group. [4] If it is assumed that their measurement of (118±3) mb is too high by 3.5%, the SAND-II adjusted spectral shape then agrees extremely well with the NBS evaluation, as illustrated in Figure 2.

The same conclusion is reached of course, if instead, the $^{58}Ni(n,p)^{58}Co$ energy-dependent cross section were increased uniformly by 3.5%. In this case, the SAND-II adjusted χ_{25} spectral shape - dotted line in Figure 3 - using the higher nickel cross section agrees more closely with the NBS χ_{25} evaluation than the solution - solid line in Figure 3 - obtained with the unmodified ENDF/B-IV cross section.

It is worth noting that the spectral data shown in Figures 2 and 3 are expressed as ratios to the reference Maxwellian shapes and that the dashed lines represent the NBS evaluations. Further, the data are presented in the 15 group format used for SAND-II error propagation: the numerator is the adjusted or evaluated total flux in group g, χ_g, while the denominator is the integrated reference Maxwellian flux $\int_g \chi_M(E)\,dE$; the exact expressions accepted for the Maxwellian shapes $\chi_{M(E)}$ in the two fields are explicitly written on the ordinate of each figure.

The SAND-II uncertainties associated with the adjusted spectra are shown as error bars*, while for the NBS evaluations, they are displayed as shaded areas corresponding to the original NBS evaluated group segmentation.

At neutron energies above 13 MeV, differential spectroscopy data for χ_{25} and χ_{82} are extremely inaccurate. Consequently, previous integral measurements for the (n,2n) reactions for ^{63}Cu and ^{58}Ni have been used in the present study for adjustment of the χ_{25} high-energy tail (Fig. 3).

A number of previously observed trends in other dosimetry benchmark fields and in fast reactor physics critical assemblies are consistent with the harder χ_{25} fission spectrum, the solid line in Figure 3, for which the average energy is 2.01 MeV with an assigned Monte Carlo uncertainty of ± 0.08 MeV. Also, the deviations from the reference Maxwellian appear to be compatible, within uncertainties with Islam and Knitter's suggested corrections for finite fission source effects. [24] All of these conclusions are tentative, however, and depend on further study and to a large degree upon future interlaboratory integral measurements of the $^{58}Ni(n,p)^{58}Co$ average cross section in the ^{252}Cf fission neutron spectrum. The goal accuracy for this measurement should be 2% or better at the 1σ confidence level.

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* The input integral and differential cross section errors used here are essentially those defined in references 3, 22 and 14.

On the basis of the unfolded spectral shapes, bias factors can be derived for all dosimetry reactions. They are defined as ratios between measured and computed integral reaction rates. If, for a given nuclear reaction, these bias factors are independent of the benchmark fields considered (within uncertainties), their values averaged over the fields can be applied as normalization or scaling factors for correcting differential energy cross sections. This will be the case if three conditions are satisfied:

1) The spectral shapes are properly characterized,

2) The integral data are sufficiently accurate, and

3) The differential-energy cross sections are well established in shape and are mostly inaccurate in magnitude.

The judicious adjustments of neutron spectra on the basis of Category I reactions for the energy range considered here will satisfy condition (1), at least in principle. An important demonstration of the value of such adjustments can be given for the CFRMF benchmark: previous [3] as well as more recent adjusted spectra [14] depart significantly from the initial recommended spectral shape, [3] but agree well with subsequent neutronics computations using the ENDF/B-IV general cross section file; current ANC studies suggest that the previous differences are essentially traceable to improved uranium-238 inelastic scattering data.

It is not as yet clear that condition (2) is adequately fulfilled for all dosimetry reactions for the χ_{82} benchmark neutron field. As shown in reference [14], for both the ENDF/B-IV and SAND-II files, the bias factors for important dosimetry reactions are different in χ_{25} and χ_{82} if the NBS evaluations of their spectral shapes are accepted. As already stressed, the χ_{25} integral data are well established. If such is also the case for most χ_{82} integral measurements, two possibilities remain:

(a) Either condition (3) is not satisfied; in particular, unresolved energy scale calibration errors are present, which is of critical concern near a reaction's threshold, or

(b) the spectral shape inadequacies for χ_{25} suggested here as well as in reference [14] are real; in this context, it is important to note that bias factors based on SAND-II adjusted spectral shapes are indeed remarkably consistent for χ_{82}, χ_{25}, CFRMF and $\Sigma\Sigma$.

Condition (3) is however likely to be unfulfilled for a number of reactions. The $^{48}Ti(n,p)^{48}Sc$ is a straightforward example as discussed in Section II. The case of the $^{63}Cu\ (n,\alpha)^{60}Co$ reaction is, perhaps, the most illustrative of why caution is needed and why interdisciplinary cooperation is important when dealing with nuclear cross section data. Integral versus differential data discrepancies for this reaction -- which is a crucial fluence monitor for fuels and materials dosimetry -- have long been puzzling. It has been thought that, because of its negative Q value and/or ^{59}Co impurity, that this reaction would have some significant response to thermal and/or epithermal neutrons in softer spectra; but previous SAND-II analysis [19] and more recent measurements in the fast reactor TAPIRO [4] do not support such a conclusion. Differential measurements by CBNM were linked to integral ones at CEN-SCK without resolving this problem. The TAPIRO work plus that reported by McElroy, et al.[19], indicate the need for some near threshold cross section enhancement. This is further supported by the recent nuclear model computations of Mann and Schenter using the computer code Hauser-4 [26] and the analysis of radial traverse measurements in EBR-II by Lippincott, et al. [27].

The most complete array of international integral data at present has been obtained in χ_{25}. Table I gathers bias factors for this benchmark for 27 dosimetry reactions, arranged in order of increasing energy response as specified in terms of effective thresholds [28,29]. The bias factors are given both for the NBS evaluated spectral shape and for the one adjusted by SAND-II on the basis of the Category I cross section set identified in this paper. Table I largely reflects the state-of-the-art of integral data testing of the ENDF/B-IV dosimetry cross section file in the energy range above 0.1 MeV.

IV. DOSIMETRY BENCHMARK NEUTRON FIELDS: STATUS AND PERSPECTIVES

A number of fast reactor reference dosimetry benchmark neutron fields were identified for a study by the ILRR program in 1971 [3]. Additional thermal, fast and fusion reactor neutron fields were identified by the IAEA in 1973 [1]. The IAEA listing was updated in 1975 at the First ASTM-Euratom Symposium on reactor dosimetry [4] and the workshop on "Reactor Dosimetry Benchmarks" focused its attention essentially on delineating the concept, role and use of benchmark radiation fields for reactor dosimetry.

In recent years, most efforts in the area of dosimetry benchmarks have focused on fuels and materials development for fast breeder reactors (generally the LMFBR) [3], but there is a growing and justified emphasis on problems related to other types of reactors;

such as pressure vessel surveillance in light water (LWR) power reactors [5], dosimetry for controlled thermonuclear reactor (CTR) [4] research, and dosimetry of shielding experiments. This enlarged interest calls for a better definition and classification of dosimetry benchmarks and this means, furthermore, that both the neutron and gamma ray components of the radiation field must be taken into consideration when discussing benchmarks and associated programmes.

The word benchmark refers to at least three distinct types of radiation fields which may be termed respectively: standard and reference radiation fields, and controlled radiation environments. There is no universally accepted definition of a benchmark radiation field. Still, the three types mentioned above may be distinguished as follows: [4]

- Standard radiation field: a permanent, stable and reproducible radiation field (neutron or gamma or mixed) that is characterized to state-of-the-art accuracy in terms of flux intensity and energy spectra, and spatial and angular flux distribution. Important field quantities must be verified by interlaboratory measurement.

- Reference radiation field: a permanent and reproducible radiation field reasonably well characterized in the above terms and accepted as a reference by a community of users.

- Controlled radiation environment: a radiation field employed for a restricted set of well-defined experiments.

To be of use today, a measurement performed in any of these "benchmark" fields must be well documented and its accuracy thoroughly assessed [3,4]. Officially, there exist only a few standard radiation fields relevant for dosimetry; e.g., thermal neutron density standards, and Ra(Be) primary source strength standards. However, a number of neutron fields in various stages of development and use could become standards [4]. These include (1) the californium-252 and uranium-235 fission source fields; (2) the $\Sigma\Sigma$-type secondary standard radiation field [3]; (3) the Intermediate-Energy Standard Neutron Field (ISNF) [23]; and (4) reactor neutron beams of known flux, gradient and energy spectrum (e.g., thermal beams, filtered beams, etc.)

For the present investigation, the discussion has been limited to the ^{252}Cf and ^{235}U benchmark neutron fields; however, preliminary results of data testing using a number of other neutron fields are currently available and are under study [3,4]. The status of these efforts and recommendations for future direction will be the subject of the IAEA "Consultants' Meeting on Integral Cross Section Measurements in Standard Neutron Fields for Reactor Dosimetry,"* to be convened at Vienna, November 15-19, 1976, and supported by the International Working Group on Reactor Radiation Measurement (IWGRRM) and the International Nuclear Data Committee (INDC).

V. CONCLUSIONS

Reactor dosimetry is an international undertaking aimed at providing the capability to properly correlate and apply irradiation effects data. Goal accuracies for such correlation and application are in the range of a few percent. This requires the development and standardization of evaluated energy-dependent dosimetry cross section files and their validation in benchmark neutron fields. The results reported herein demonstrate the value of the dosimetry data testing approach recommended by the 1973 IAEA "Consultants' Meeting on Nuclear Data for Reactor Neutron Dosimetry" and adopted earlier by the ILRR program.

It is apparent, that interlaboratory integral reaction rate measurements in benchmark neutron fields play a crucial role in establishing consistent differential dosimetry cross section files. The present work has shown that the consistency of one of these files, ENDF/B-IV, in the MeV energy range is good to within 5 % for many reactions, but for some - particularly the very high energy reactions, there are inconsistencies up to as high as 100% , depending on the accepted benchmark spectral shapes. These results, as well as others, justify recent initiatives of the IAEA to organize an international work program in this field.

* The Summary Report of this Meeting is being published as INDC(NDS)-81/L+M; the Proceedings will be issued as an IAEA Technical Report in June 1977.

TABLE I. INTEGRAL TESTING OF DOSIMETRY CROSS SECTION FILES[a] IN THE URANIUM-235 THERMAL FISSION SPECTRUM NEUTRON FIELD, χ_{25}

REACTION[b]	EFFECTIVE THRESHOLD (MeV)	BIAS FACTOR[c] FOR $\chi_{25}(E)$: NBS EVALUATION (E = 1.98 MeV)	BIAS FACTOR[c] FOR $\chi_{25}(E)$: SAND-II ADJUSTED (E = 2.01 MeV)	RELIABILITY[d]
$^{115}In(n,\gamma)^{116m}In$	-	0.990	0.996	
$^{197}Au(n,\gamma)^{198}Au$	-	0.987	0.972	
$^{63}Cu(n,\gamma)^{64}Cu$	-	0.846	0.834	x
$^{235}U(n,f)$	-	0.969	0.968	?
$^{239}Pu(n,f)$	-	1.017	1.017	
$^{237}Np(n,f)$	0.6	0.994	1.000	
$^{115}In(n,n')^{115m}In$	1.2	1.037	1.019	
$^{232}Th(n,f)$	1.4	1.174	1.149	x
$^{238}U(n,f)$	1.5	1.031	1.010	
$^{47}Ti(n,p)^{47}Sc$	2.2	0.888	0.852	x
$^{31}P(n,p)^{31}Si$	2.4	[1.092]	[1.041]	x
$^{58}Ni(n,p)^{58}Co$	2.8	1.068	1.020	
$^{32}S(n,p)^{32}P$	2.9	1.042	0.987	
$^{54}Fe(n,p)^{54}Mn$	3.1	1.026	0.977	
$Ti(n,x)^{46}Sc$	3.9	1.181	1.155	x
$^{27}Al(n,p)^{27}Mg$	4.4	0.937	0.917	x
$^{56}Fe(n,p)^{56}Mn$	6.0	0.983	1.004	
$^{59}Co(n,\alpha)^{56}Mn$	6.8	0.966	0.973	
$^{63}Cu(n,\alpha)^{60}Co$	6.8	1.420	1.445	x
$^{24}Mg(n,p)^{24}Na$	6.8	[0.977]	[0.993]	
$^{27}Al(n,\alpha)^{24}Na$	7.2	1.017	1.022	
$^{48}Ti(n,p)^{48}Sc$	7.6	1.734	1.714	x
$^{127}I(n,2n)^{126}I$	10.5	0.885	0.778	x
$^{55}Mn(n,2n)^{54}Mn$	11.6	0.996	0.803	x
$^{63}Cu(n,2n)^{62}Cu$	12.4	[1.407]	[0.897]	
$^{90}Zr(n,2n)^{89}Zr$	13	[2.951]	[1.715]	x
$^{58}Ni(n,2n)^{57}Ni$	13.5	2.050	1.120	

(a) ENDF/B-IV, except for bias factors within brackets where the SAND-II file was used.
(b) Underlined reactions are considered Category I and form the basis for χ_{25} spectral shape adjustments.
(c) Measured/computed integral cross sections.
(d) Crosses (x) if $\sigma(E)$ in file is deemed seriously unreliable for the energy response range of the reaction in χ_{25}.

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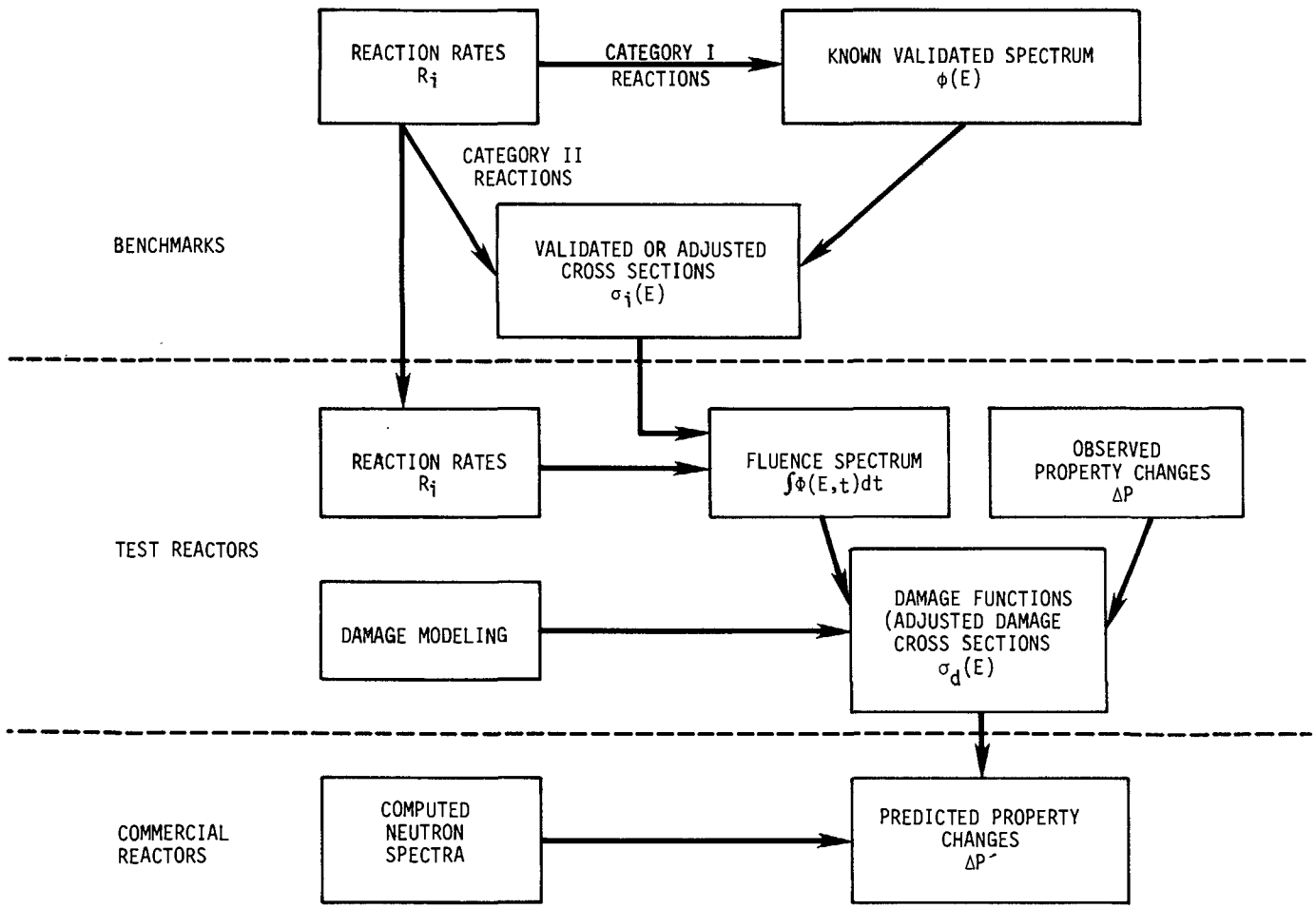

REACTION RATES
R_i
CATEGORY I
REACTIONS
KNOWN VALIDATED SPECTRUM
$\phi(E)$
CATEGORY II
REACTIONS
BENCHMARKS
VALIDATED OR ADJUSTED
CROSS SECTIONS
$\sigma_i(E)$
REACTION RATES
R_i
FLUENCE SPECTRUM
$\int\Phi(E,t)dt$
OBSERVED
PROPERTY CHANGES
ΔP
TEST REACTORS
DAMAGE MODELING
DAMAGE FUNCTIONS
(ADJUSTED DAMAGE
CROSS SECTIONS
$\sigma_d(E)$
COMMERCIAL
REACTORS
COMPUTED
NEUTRON
SPECTRA
PREDICTED PROPERTY
CHANGES
$\Delta P'$

Fig. 1

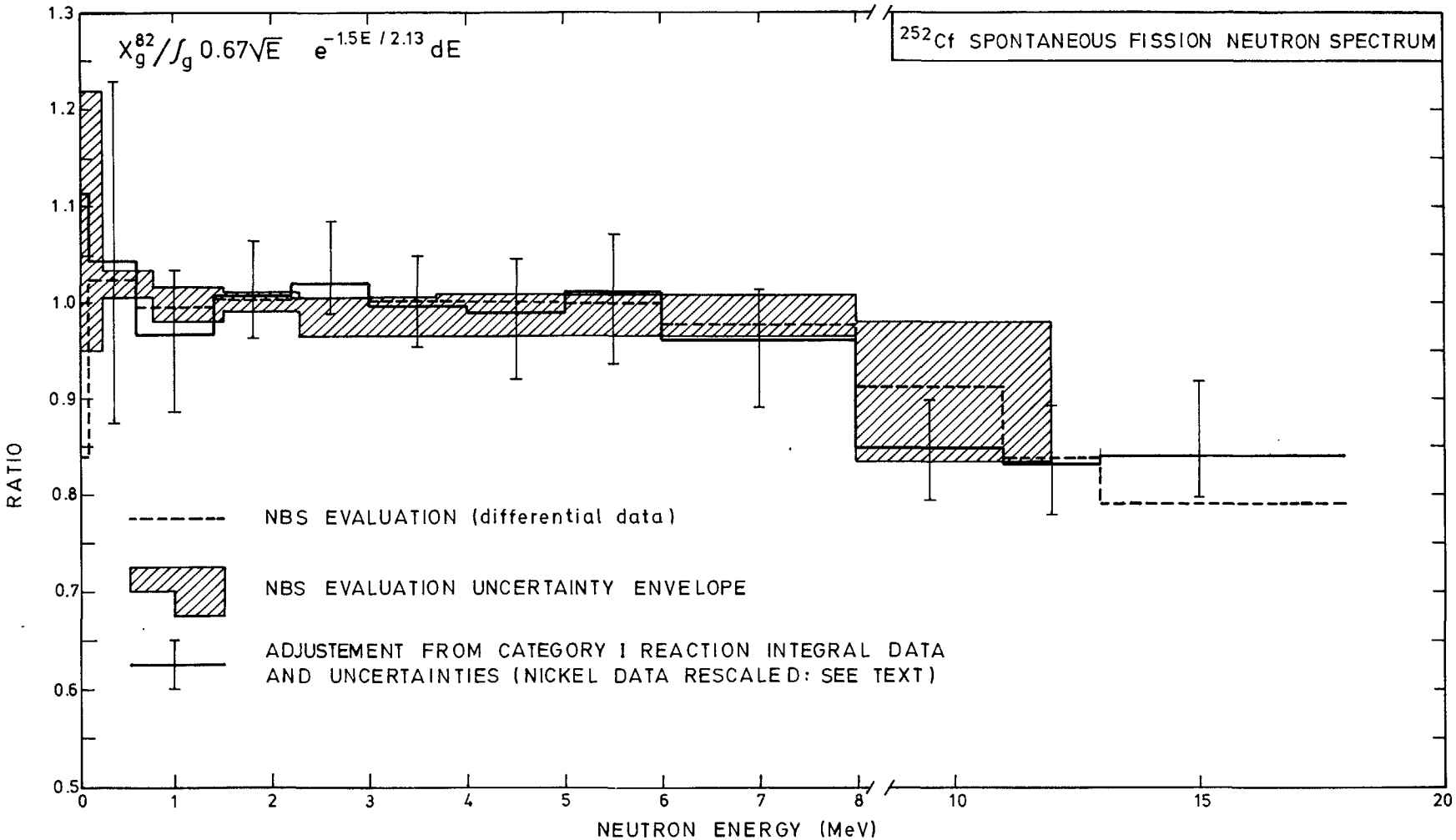


Fig. 2 DEVIATION OF THE ^{252}Cf SPONTANEOUS FISSION SPECTRUM NEUTRON DISTRIBUTION FROM A REFERENCE MAXWELLIAN SHAPE

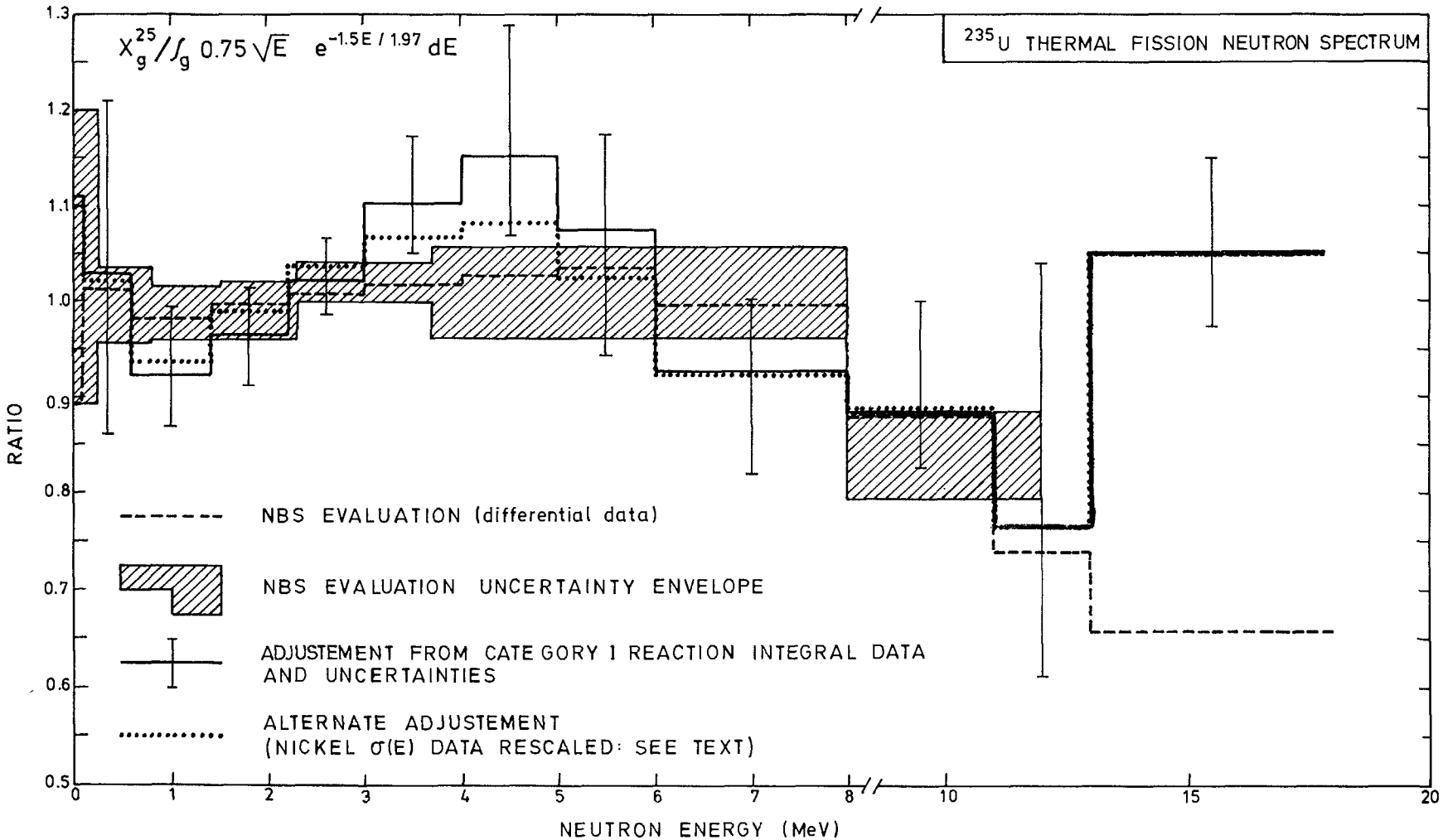
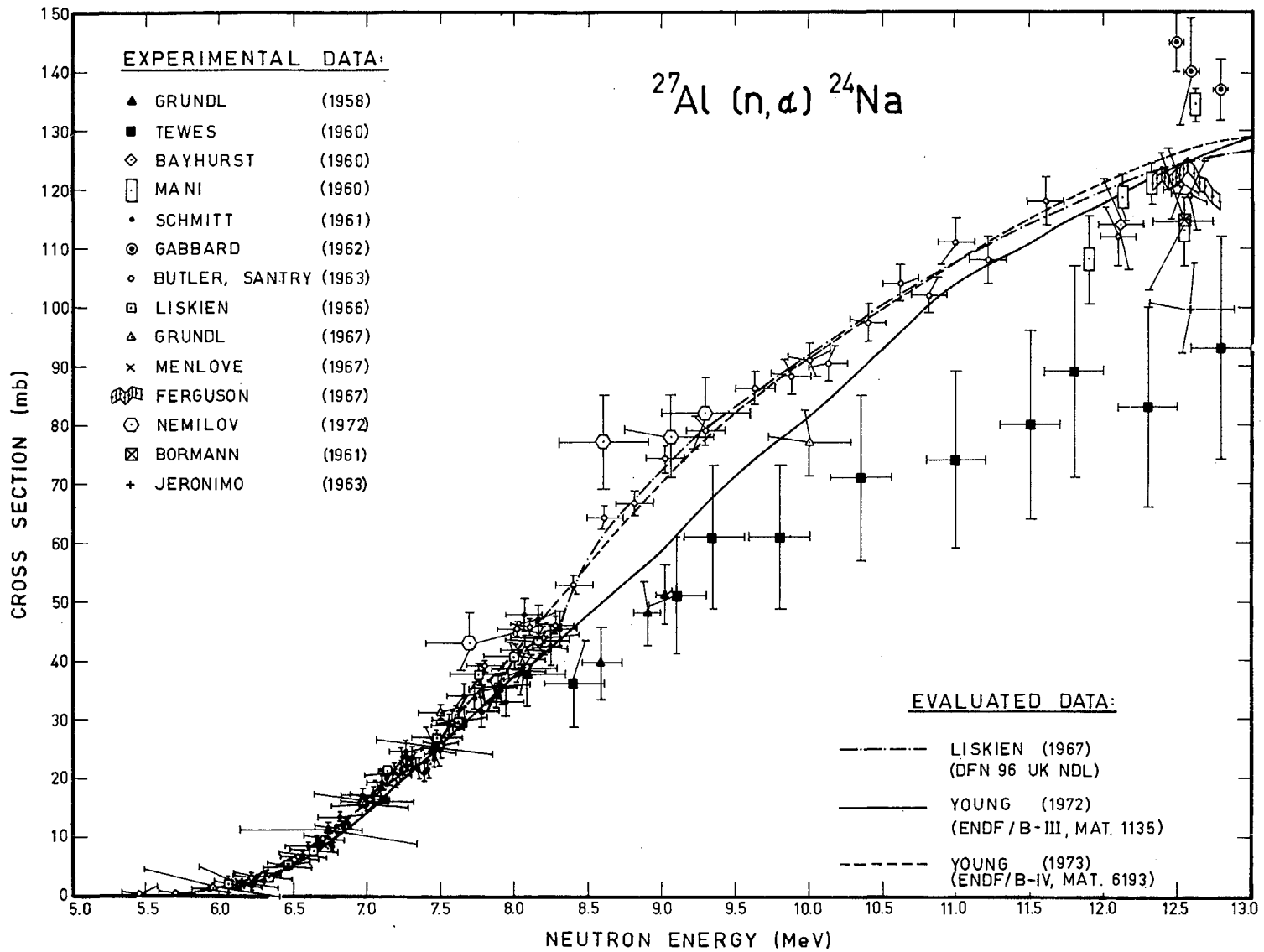


Fig. 3 DEVIATION OF THE ^{235}U FISSION SPECTRUM NEUTRON DISTRIBUTION FROM A REFERENCE MAXWELLIAN SHAPE

EXCITATION FUNCTIONS

(Status as of November 1976)

Reaction	Figure
$^{27}Al\ (n,\alpha)^{24}Na$	1A, 1B, 1C, 1D
$^{31}P\ (n,p)^{31}Si$	2,3
$^{32}S\ (n,p)^{32}P$	4,5
$^{46}Ti\ (n,p)^{46}Sc$	6
$^{47}Ti\ (n,p)^{47}Sc$	7
$^{48}Ti\ (n,p)^{48}Sc$	8
$^{55}Mn\ (n,2n)^{54}Mn$	9
$^{56}Fe\ (n,p)^{56}Mn$	10, 11, 12
$^{58}Ni\ (n,2n)^{57}N$	13
$^{59}Co\ (n,\alpha)^{56}Mn$	14
$^{63}Cu\ (n,2n)^{62}Cu$	15
$^{63}Cu\ (n,\alpha)^{60}Co$	16
$^{90}Zr\ (n,2n)^{89}Zr^{m+g}$	17
$^{93}Nb\ (n,2n)^{92}Nb^{m}$	18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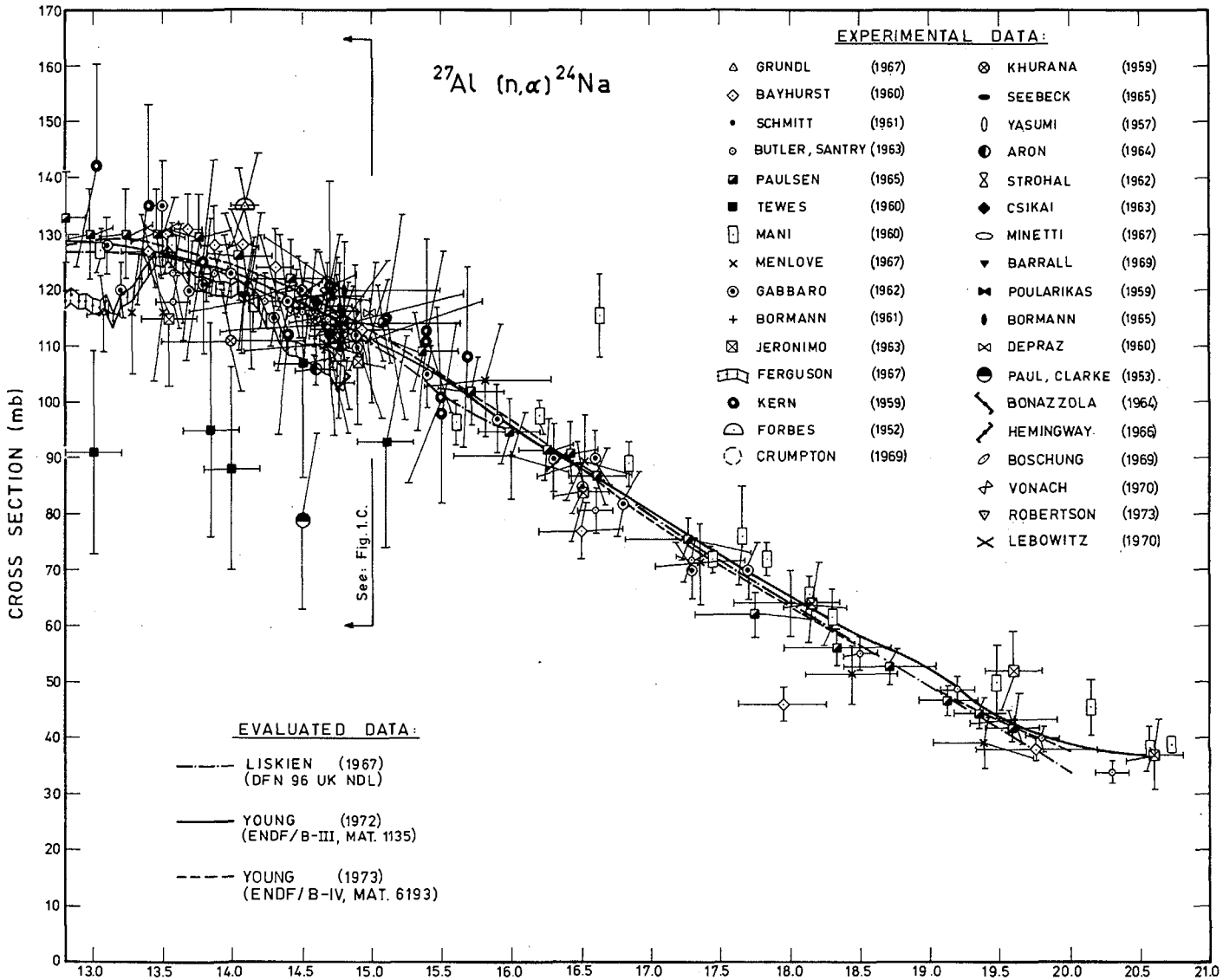
TEWES (1960)
BAYHURST (1960)
MANI (1960)
SCHMITT (1961)
GABBARD (1962)
BUTLER, SANTRY (1963)
LISKIEN (1966)
GRUNDL (1967)
MENLOVE (1967)
FERGUSON (1967)
NEMILOV (1972)
BORMANN (1961)
JERONIMO (1963)
$^{27}Al\ (n,\alpha)\ ^{24}Na$
EVALUATED DATA:
LISKIEN (1967)
(DFN 96 UK NDL)
YOUNG (1972)
(ENDF/B-III, MAT. 1135)
YOUNG (1973)
(ENDF/B-IV, MAT. 6193)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
5.0
5.5
6.0
6.5
7.0
7.5
8.0
8.5
9.0
9.5
10.0
10.5
11.0
11.5
12.0
12.5
13.0

Fig. 1A.


$^{27}Al\ (n,\alpha)\ ^{24}Na$
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PAULSEN (1965)
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MANI (1960)
MENLOVE (1967)
GABBARO (1962)
BORMANN (1961)
JERONIMO (1963)
FERGUSON (1967)
KERN (1959)
FORBES (1952)
CRUMPTON (1969)
KHURANA (1959)
SEEBECK (1965)
YASUMI (1957)
ARON (1964)
STROHAL (1962)
CSIKAI (1963)
MINETTI (1967)
BARRALL (1969)
POULARIKAS (1959)
BORMANN (1965)
DEPRAZ (1960)
PAUL, CLARKE (1953)
BONAZZOLA (1964)
HEMINGWAY (1966)
BOSCHUNG (1969)
VONACH (1970)
ROBERTSON (1973)
LEBOWITZ (1970)
See: Fig. 1.C.
EVALUATED DATA:
LISKIEN (1967)
(DFN 96 UK NDL)
YOUNG (1972)
(ENDF/B-III, MAT. 1135)
YOUNG (1973)
(ENDF/B-IV, MAT. 6193)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
10
20
30
40
50
60
70
80
90
100
110
120
130
140
150
160
170
13.0
13.5
14.0
14.5
15.0
15.5
16.0
16.5
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17.5
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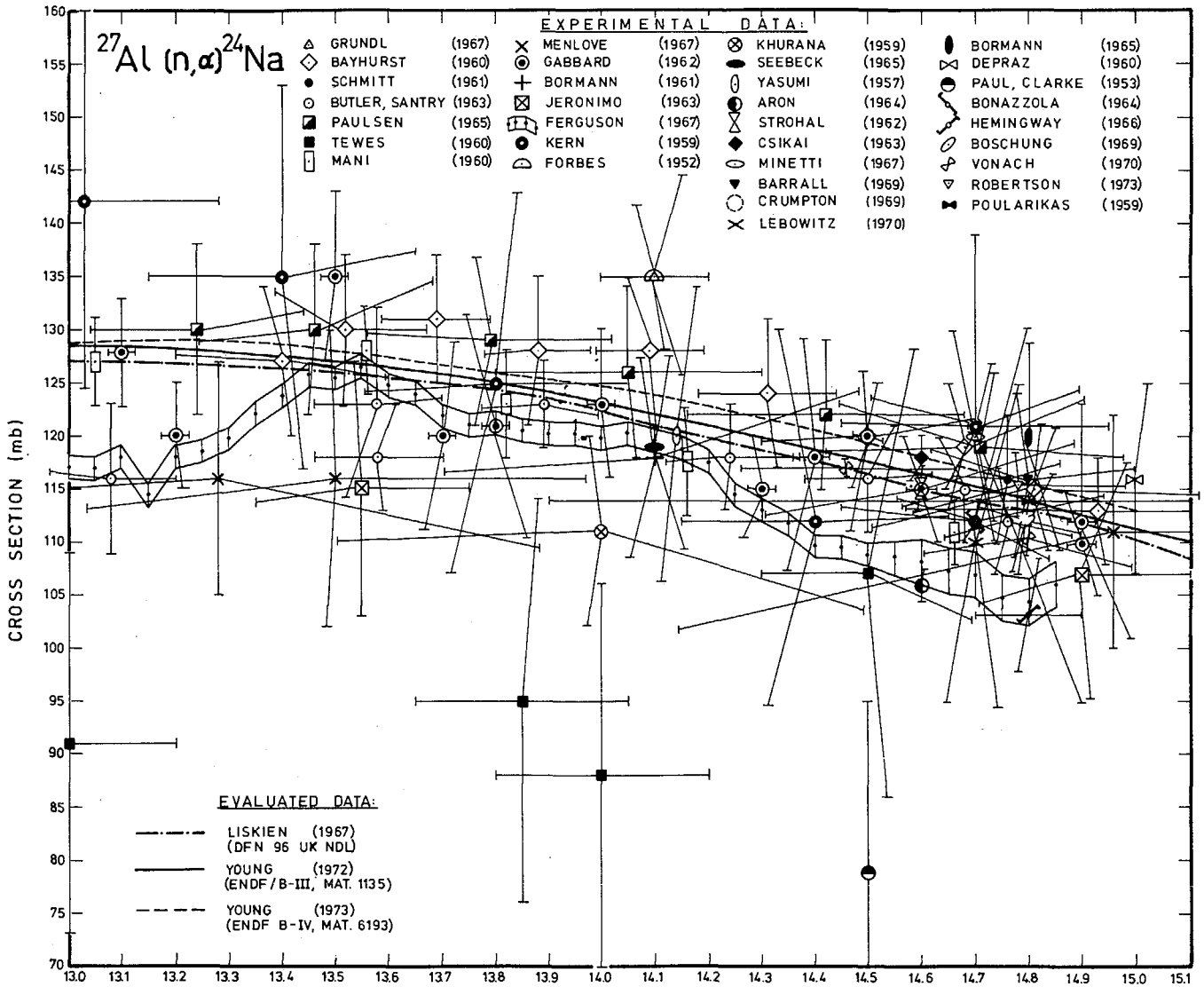
18.5
19.0
19.5
20.0
20.5
21.0

Fig. 1 B.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EXPERIMENTAL DATA:
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BAYHURST (1960)
SCHMITT (1961)
BUTLER, SANTRY (1963)
PAULSEN (1965)
TEWES (1960)
MANI (1960)
MENLOVE (1967)
GABBARD (1962)
BORMANN (1961)
JERONIMO (1963)
FERGUSON (1967)
KERN (1959)
FORBES (1952)
KHURANA (1959)
SEEBECK (1965)
YASUMI (1957)
ARON (1964)
STROHAL (1962)
CSIKAI (1963)
MINETTI (1967)
BARRALL (1969)
CRUMPTON (1969)
LEBOWITZ (1970)
BORMANN (1965)
DEPRAZ (1960)
PAUL, CLARKE (1953)
BONAZZOLA (1964)
HEMINGWAY (1966)
BOSCHUNG (1969)
VONACH (1970)
ROBERTSON (1973)
POULARIKAS (1959)
EVALUATED DATA:
LISKIEN (1967) (DFN 96 UK NDL)
YOUNG (1972) (ENDF/B-III, MAT. 1135)
YOUNG (1973) (ENDF B-IV, MAT. 6193)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)

Fig. 1C.

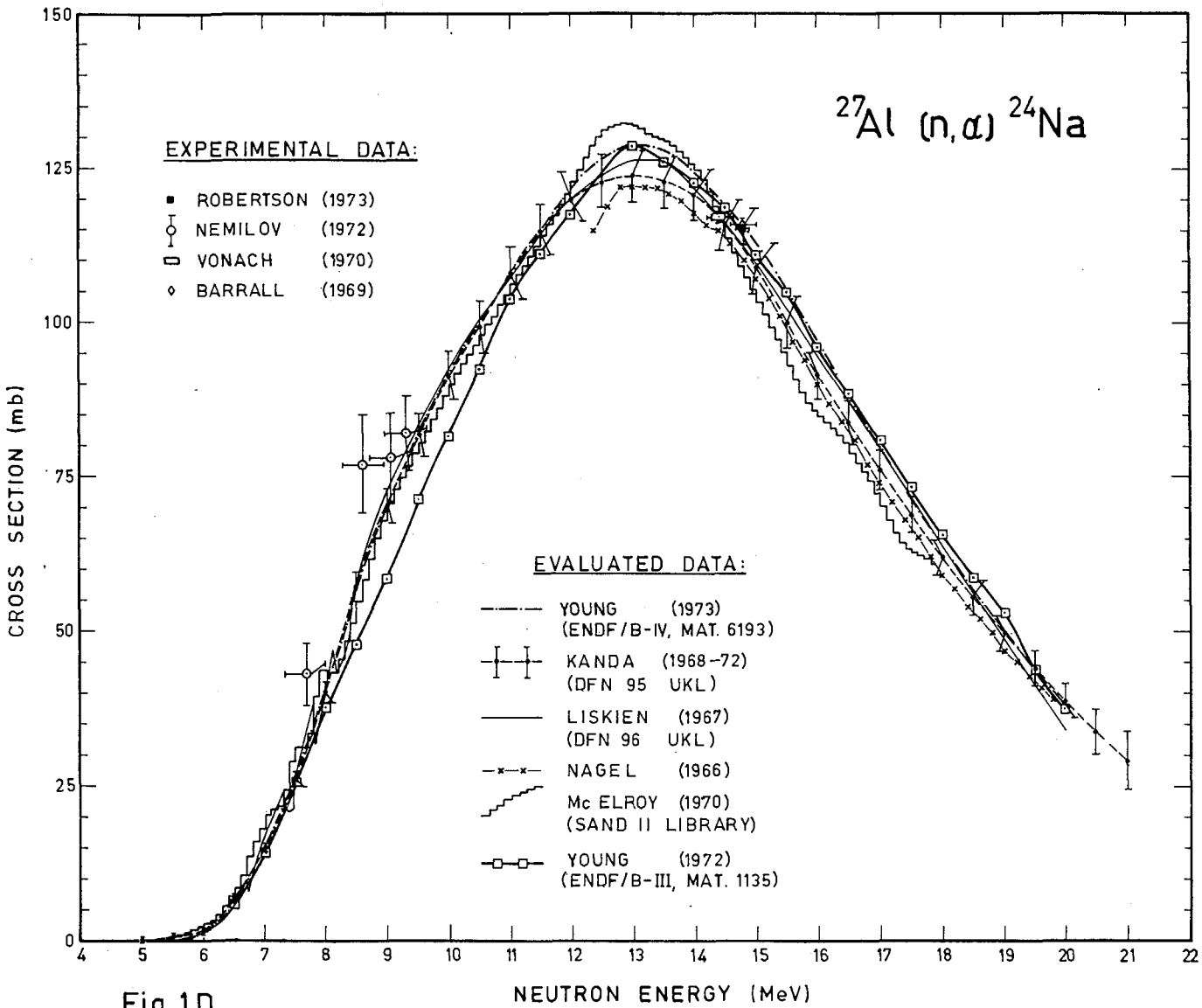

$^{27}Al\ (n,\alpha)\ ^{24}Na$
EXPERIMENTAL DATA:
ROBERTSON (1973)
NEMILOV (1972)
VONACH (1970)
BARRALL (1969)
EVALUATED DATA:
YOUNG (1973)
(ENDF/B-IV, MAT. 6193)
KANDA (1968-72)
(DFN 95 UKL)
LISKIEN (1967)
(DFN 96 UKL)
NAGEL (1966)
Mc ELROY (1970)
(SAND II LIBRARY)
YOUNG (1972)
(ENDF/B-III, MAT. 1135)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
25
50
75
100
125
150
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22

Fig. 1D.

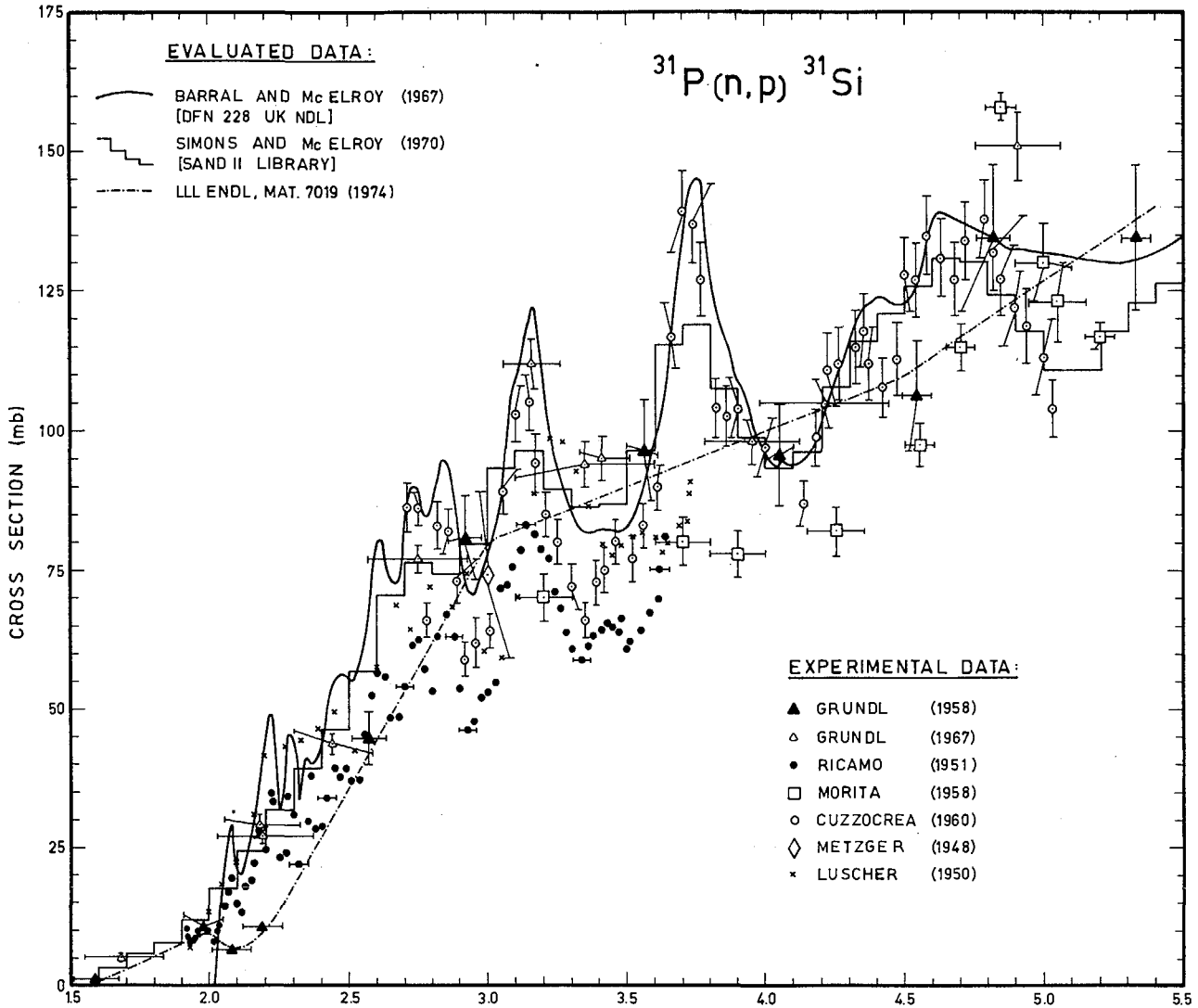
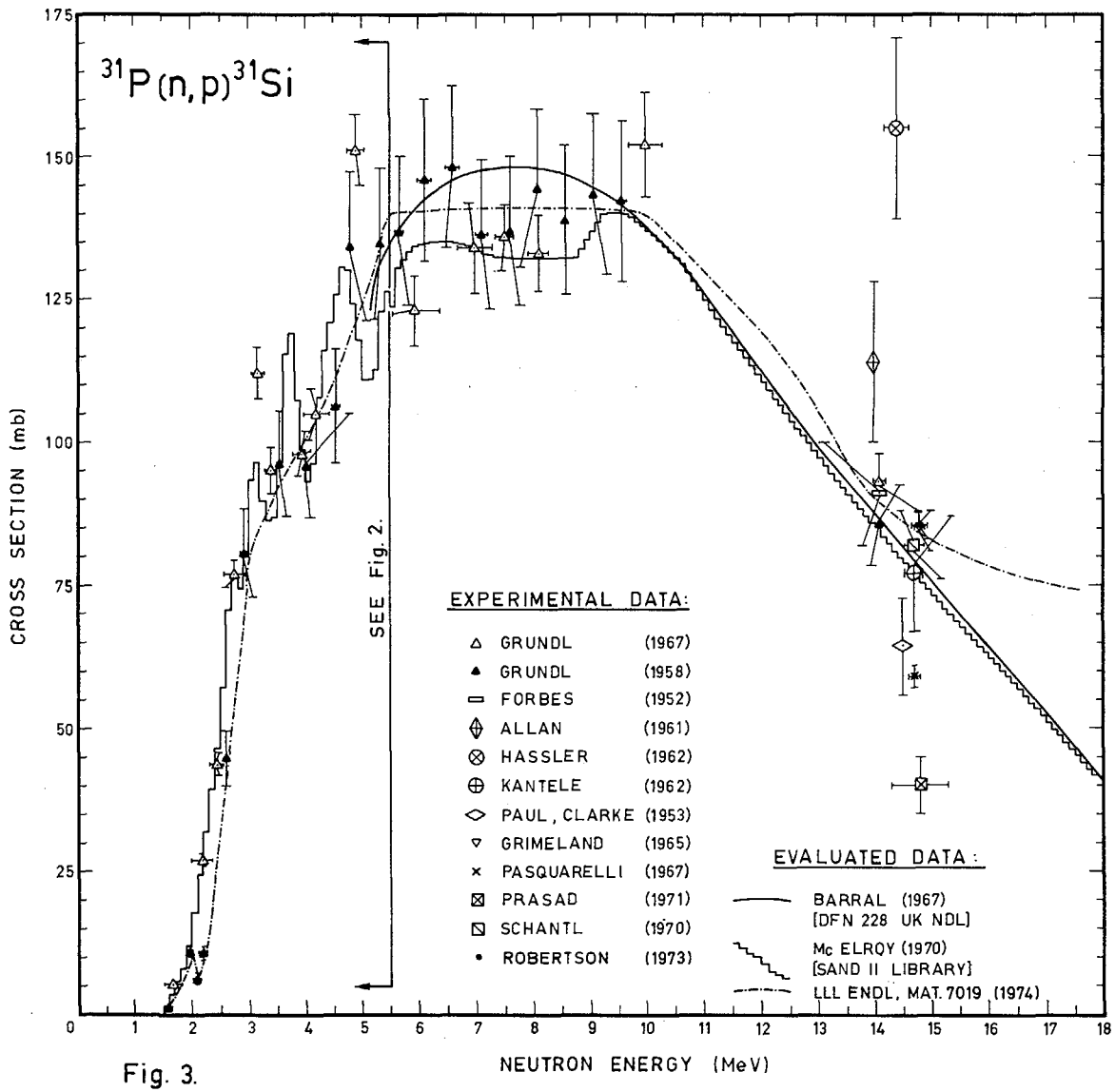

EVALUATED DATA:
BARRAL AND Mc ELROY (1967) [DFN 228 UK NDL]
SIMONS AND Mc ELROY (1970) [SAND II LIBRARY]
LLL ENDL, MAT. 7019 (1974)
$^{31}P(n,p)\ ^{31}Si$
EXPERIMENTAL DATA:
GRUNDL (1958)
GRUNDL (1967)
RICAMO (1951)
MORITA (1958)
CUZZOCREA (1960)
METZGER (1948)
LUSCHER (1950)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
25
50
75
100
125
150
175
1.5
2.0
2.5
3.0
3.5
4.0
4.5
5.0
5.5

Fig. 2.


$^{31}P(n,p)^{31}Si$
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
SEE Fig. 2.
EXPERIMENTAL DATA:
GRUNDL (1967)
GRUNDL (1958)
FORBES (1952)
ALLAN (1961)
HASSLER (1962)
KANTELE (1962)
PAUL, CLARKE (1953)
GRIMELAND (1965)
PASQUARELLI (1967)
PRASAD (1971)
SCHANTL (1970)
ROBERTSON (1973)
EVALUATED DATA:
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Mc ELROY (1970) [SAND II LIBRARY]
LLL ENDL, MAT. 7019 (1974)

Fig. 3.

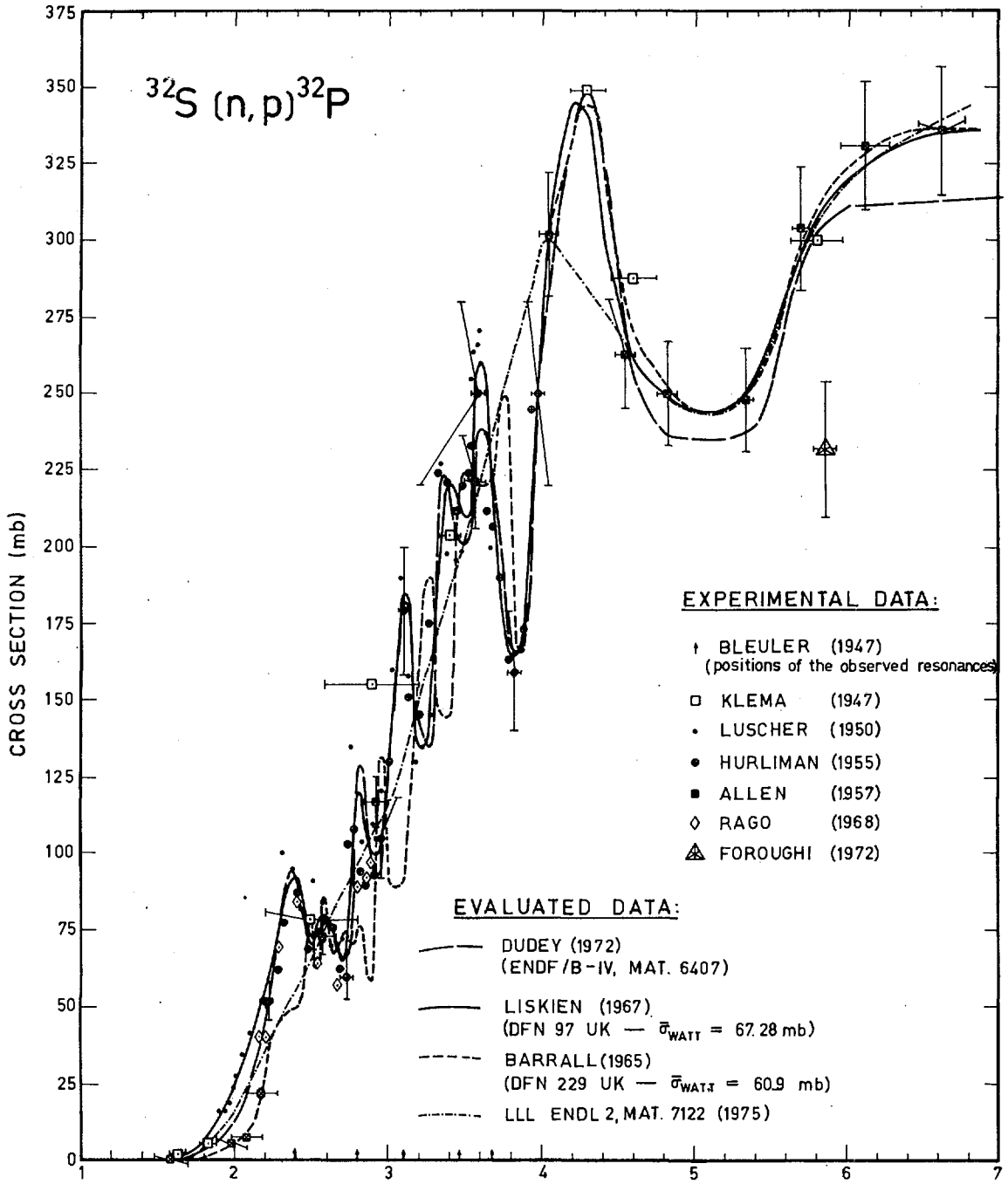
^{32}S (n, p) ^{32}P
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
EXPERIMENTAL DATA:
BLEULER (1947)
(positions of the observed resonances)
KLEMA (1947)
LUSCHER (1950)
HURLIMAN (1955)
ALLEN (1957)
RAGO (1968)
FOROUGHI (1972)
EVALUATED DATA:
DUDEY (1972)
(ENDF/B-IV, MAT. 6407)
LISKIEN (1967)
(DFN 97 UK — $\bar{\sigma}_{WATT}$ = 67.28 mb)
BARRALL (1965)
(DFN 229 UK — $\bar{\sigma}_{WATT}$ = 60.9 mb)
LLL ENDL 2, MAT. 7122 (1975)

Fig. 4.

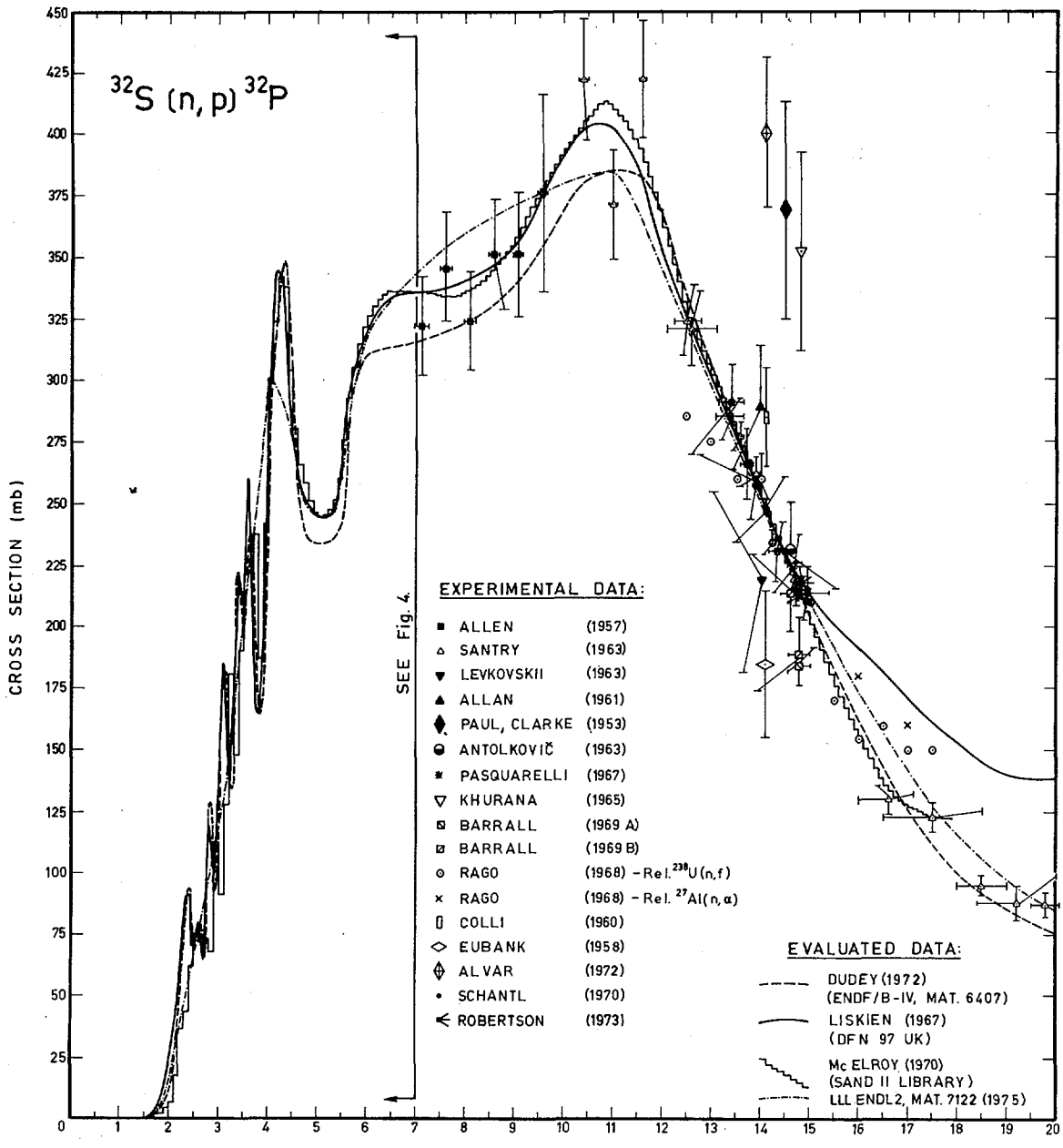
^{32}S (n, p) ^{32}P
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
SEE Fig. 4.
EXPERIMENTAL DATA:
ALLEN (1957)
SANTRY (1963)
LEVKOVSKII (1963)
ALLAN (1961)
PAUL, CLARKE (1953)
ANTOLKOVIČ (1963)
PASQUARELLI (1967)
KHURANA (1965)
BARRALL (1969 A)
BARRALL (1969 B)
RAGO (1968) - Rel. ^{238}U(n,f)
RAGO (1968) - Rel. ^{27}Al(n,α)
COLLI (1960)
EUBANK (1958)
ALVAR (1972)
SCHANTL (1970)
ROBERTSON (1973)
EVALUATED DATA:
DUDEY (1972) (ENDF/B-IV, MAT. 6407)
LISKIEN (1967) (DFN 97 UK)
Mc ELROY (1970) (SAND II LIBRARY)
LLL ENDL2, MAT. 7122 (1975)

Fig. 5.

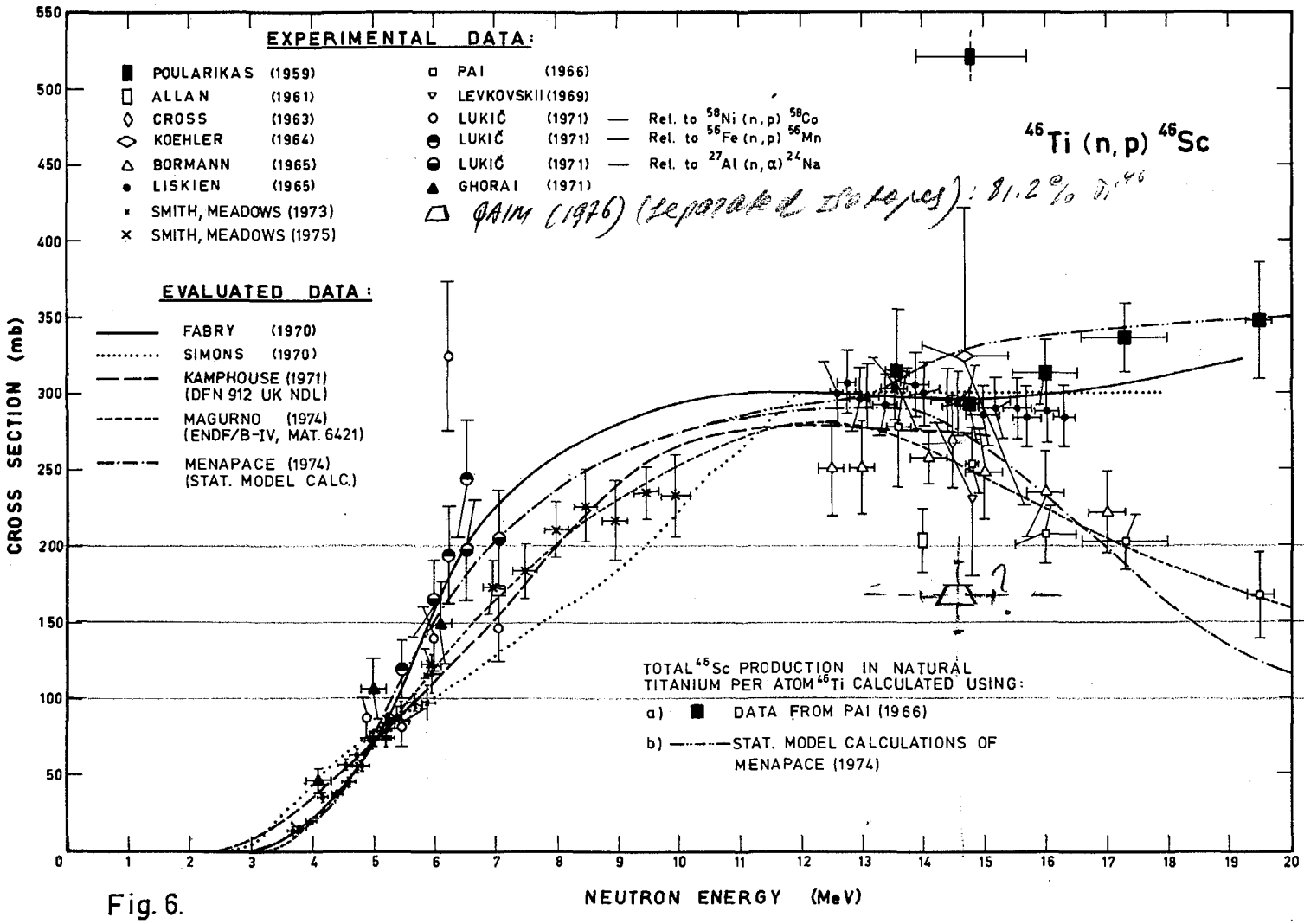
EXPERIMENTAL DATA:
POULARIKAS (1959)
ALLAN (1961)
CROSS (1963)
KOEHLER (1964)
BORMANN (1965)
LISKIEN (1965)
SMITH, MEADOWS (1973)
SMITH, MEADOWS (1975)
PAI (1966)
LEVKOVSKII (1969)
LUKIČ (1971) — Rel. to $^{58}Ni(n,p)^{58}Co$
LUKIČ (1971) — Rel. to $^{56}Fe(n,p)^{56}Mn$
LUKIČ (1971) — Rel. to $^{27}Al(n,\alpha)^{24}Na$
GHORAI (1971)
PAIM (1976) (separated isotopes): 81.2 % Ti-46
$^{46}Ti(n,p)^{46}Sc$
EVALUATED DATA:
FABRY (1970)
SIMONS (1970)
KAMPHOUSE (1971) (DFN 912 UK NDL)
MAGURNO (1974) (ENDF/B-IV, MAT. 6421)
MENAPACE (1974) (STAT. MODEL CALC.)
TOTAL ^{46}Sc PRODUCTION IN NATURAL TITANIUM PER ATOM ^{46}Ti CALCULATED USING:
a) DATA FROM PAI (1966)
b) STAT. MODEL CALCULATIONS OF MENAPACE (1974)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20
50 100 150 200 250 300 350 400 450 500 550

Fig. 6.

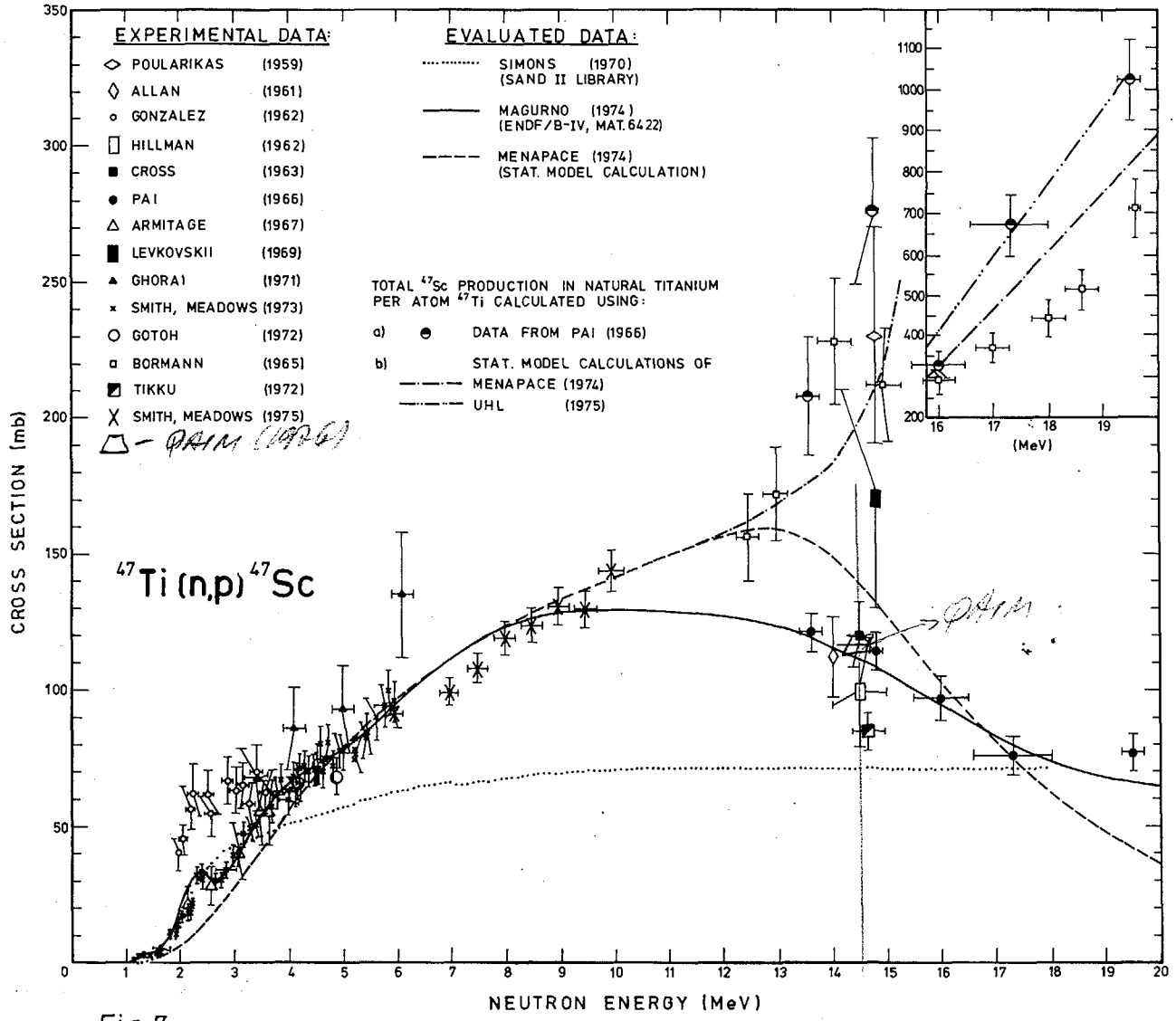
EXPERIMENTAL DATA:
POULARIKAS (1959)
ALLAN (1961)
GONZALEZ (1962)
HILLMAN (1962)
CROSS (1963)
PAI (1966)
ARMITAGE (1967)
LEVKOVSKII (1969)
GHORAI (1971)
SMITH, MEADOWS (1973)
GOTOH (1972)
BORMANN (1965)
TIKKU (1972)
SMITH, MEADOWS (1975)
EVALUATED DATA:
SIMONS (1970) (SAND II LIBRARY)
MAGURNO (1974) (ENDF/B-IV, MAT. 6422)
MENAPACE (1974) (STAT. MODEL CALCULATION)
TOTAL 47Sc PRODUCTION IN NATURAL TITANIUM PER ATOM 47Ti CALCULATED USING:
a) DATA FROM PAI (1966)
b) STAT. MODEL CALCULATIONS OF
MENAPACE (1974)
UHL (1975)
47Ti (n,p) 47Sc
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
(MeV)

Fig. 7

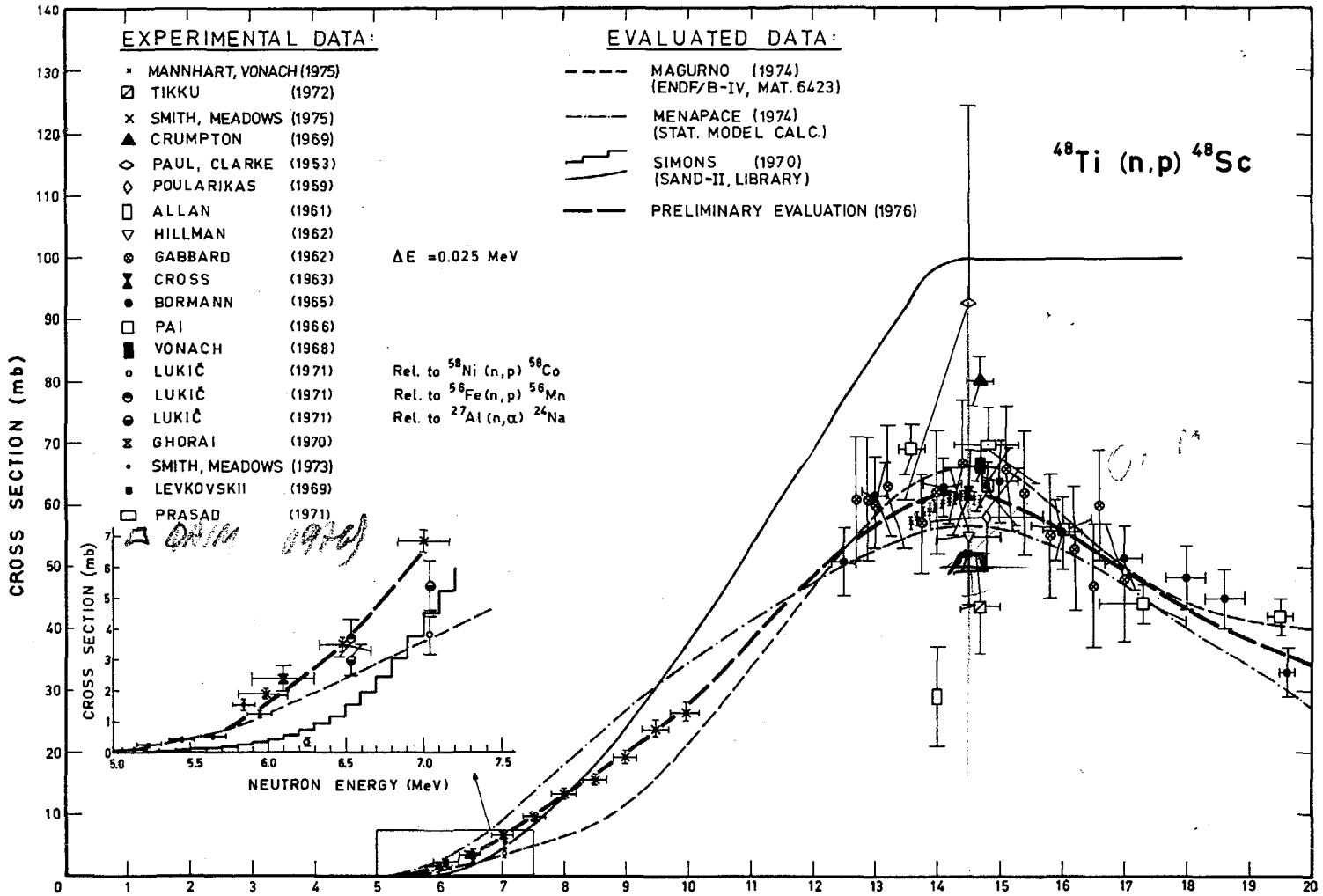
EXPERIMENTAL DATA:
MANNHART, VONACH (1975)
TIKKU (1972)
SMITH, MEADOWS (1975)
CRUMPTON (1969)
PAUL, CLARKE (1953)
POULARIKAS (1959)
ALLAN (1961)
HILLMAN (1962)
GABBARD (1962)
CROSS (1963)
BORMANN (1965)
PAI (1966)
VONACH (1968)
LUKIČ (1971)
LUKIČ (1971)
LUKIČ (1971)
GHORAI (1970)
SMITH, MEADOWS (1973)
LEVKOVSKII (1969)
PRASAD (1971)
ΔE =0.025 MeV
Rel. to 58Ni (n,p) 58Co
Rel. to 56Fe(n,p) 56Mn
Rel. to 27Al (n,α) 24Na
EVALUATED DATA:
MAGURNO (1974) (ENDF/B-IV, MAT. 6423)
MENAPACE (1974) (STAT. MODEL CALC.)
SIMONS (1970) (SAND-II, LIBRARY)
PRELIMINARY EVALUATION (1976)
48Ti (n,p) 48Sc
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)

Fig. 8.

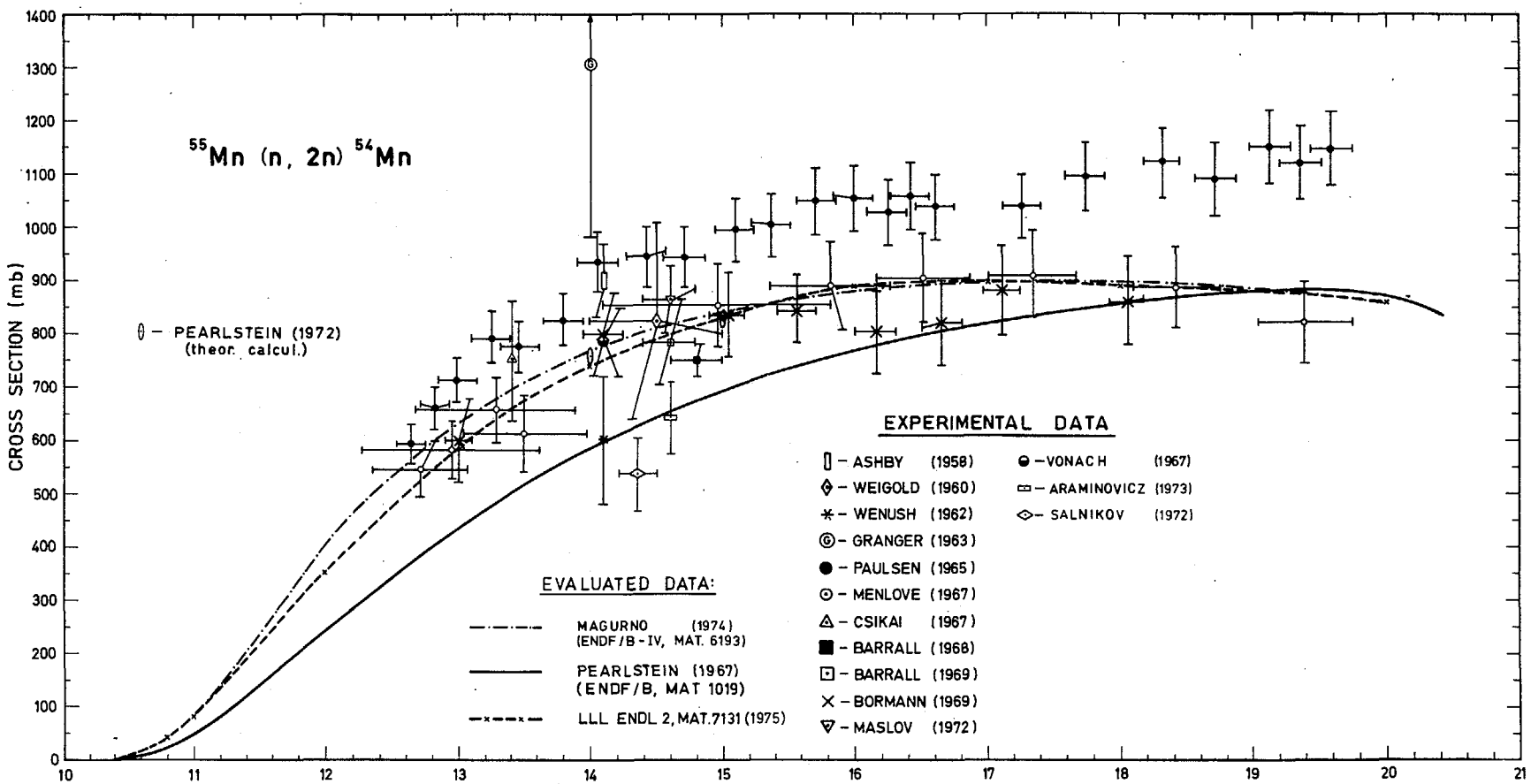
^{55}Mn (n, 2n) ^{54}Mn
θ – PEARLSTEIN (1972)
(theor. calcul.)
EXPERIMENTAL DATA
– ASHBY (1958)
– WEIGOLD (1960)
– WENUSH (1962)
– GRANGER (1963)
– PAULSEN (1965)
– MENLOVE (1967)
– CSIKAI (1967)
– BARRALL (1968)
– BARRALL (1969)
– BORMANN (1969)
– MASLOV (1972)
– VONACH (1967)
– ARAMINOVICZ (1973)
– SALNIKOV (1972)
EVALUATED DATA:
MAGURNO (1974)
(ENDF/B-IV, MAT. 6193)
PEARLSTEIN (1967)
(ENDF/B, MAT 1019)
LLL ENDL 2, MAT. 7131 (1975)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
100
200
300
400
500
600
700
800
900
1000
1100
1200
1300
1400
10
11
12
13
14
15
16
17
18
19
20
21

Fig. 9.

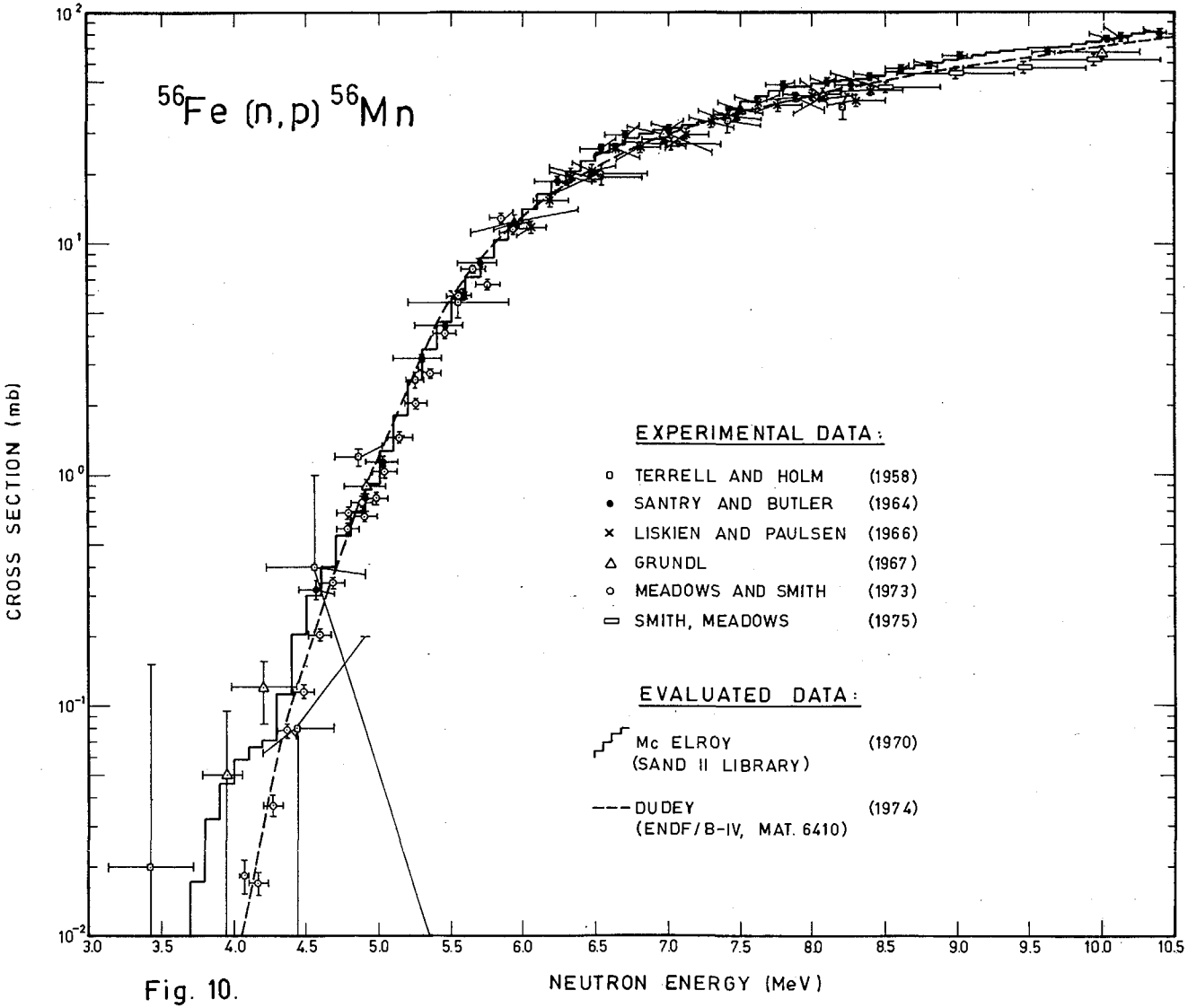
$^{56}Fe\ (n,p)\ ^{56}Mn$
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
EXPERIMENTAL DATA:
TERRELL AND HOLM (1958)
SANTRY AND BUTLER (1964)
LISKIEN AND PAULSEN (1966)
GRUNDL (1967)
MEADOWS AND SMITH (1973)
SMITH, MEADOWS (1975)
EVALUATED DATA:
Mc ELROY (1970)
(SAND II LIBRARY)
DUDEY (1974)
(ENDF/B-IV, MAT. 6410)

Fig. 10.

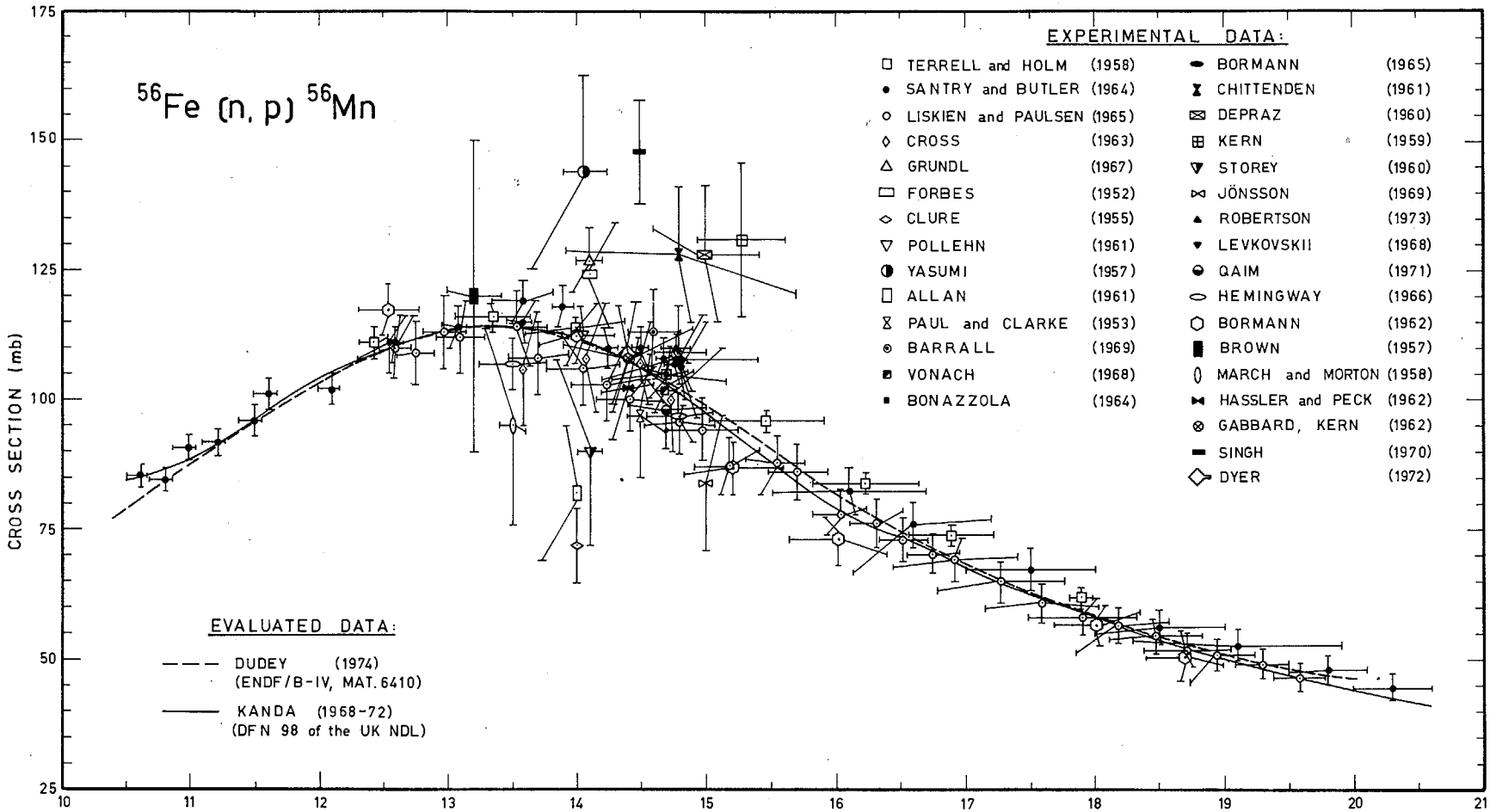
^{56}Fe (n, p) ^{56}Mn
EXPERIMENTAL DATA:
TERRELL and HOLM (1958)
SANTRY and BUTLER (1964)
LISKIEN and PAULSEN (1965)
CROSS (1963)
GRUNDL (1967)
FORBES (1952)
CLURE (1955)
POLLEHN (1961)
YASUMI (1957)
ALLAN (1961)
PAUL and CLARKE (1953)
BARRALL (1969)
VONACH (1968)
BONAZZOLA (1964)
BORMANN (1965)
CHITTENDEN (1961)
DEPRAZ (1960)
KERN (1959)
STOREY (1960)
JÖNSSON (1969)
ROBERTSON (1973)
LEVKOVSKII (1968)
QAIM (1971)
HEMINGWAY (1966)
BORMANN (1962)
BROWN (1957)
MARCH and MORTON (1958)
HASSLER and PECK (1962)
GABBARD, KERN (1962)
SINGH (1970)
DYER (1972)
EVALUATED DATA:
DUDEY (1974)
(ENDF/B-IV, MAT. 6410)
KANDA (1968-72)
(DFN 98 of the UK NDL)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
175
150
125
100
75
50
25
10
11
12
13
14
15
16
17
18
19
20
21

Fig. 11.

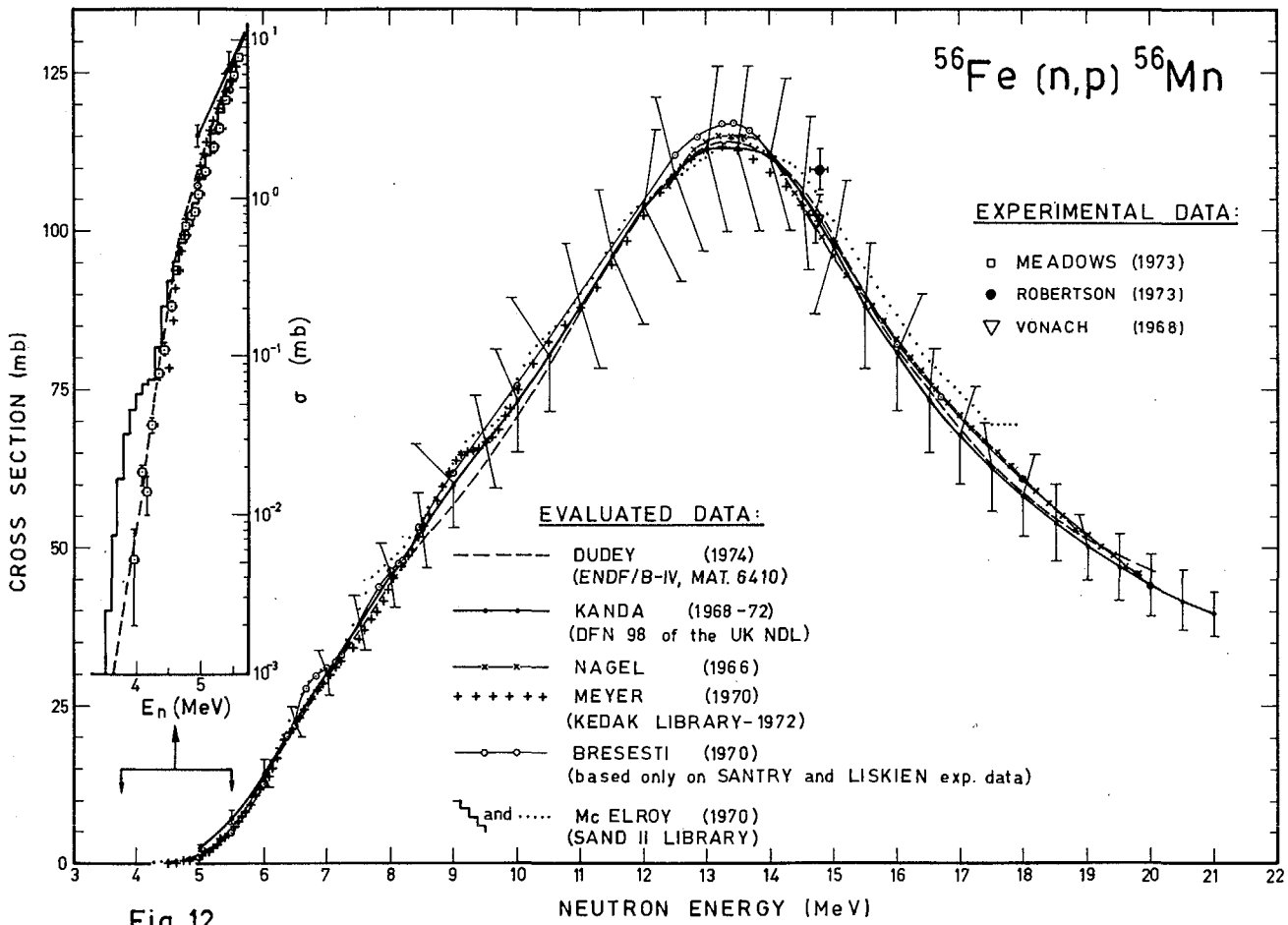
^{56}Fe (n,p) ^{56}Mn
EXPERIMENTAL DATA:
MEADOWS (1973)
ROBERTSON (1973)
VONACH (1968)
EVALUATED DATA:
DUDEY (1974)
(ENDF/B-IV, MAT. 6410)
KANDA (1968-72)
(DFN 98 of the UK NDL)
NAGEL (1966)
MEYER (1970)
(KEDAK LIBRARY-1972)
BRESESTI (1970)
(based only on SANTRY and LISKIEN exp. data)
and ..... Mc ELROY (1970)
(SAND II LIBRARY)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
σ (mb)
E_n (MeV)

Fig. 12.

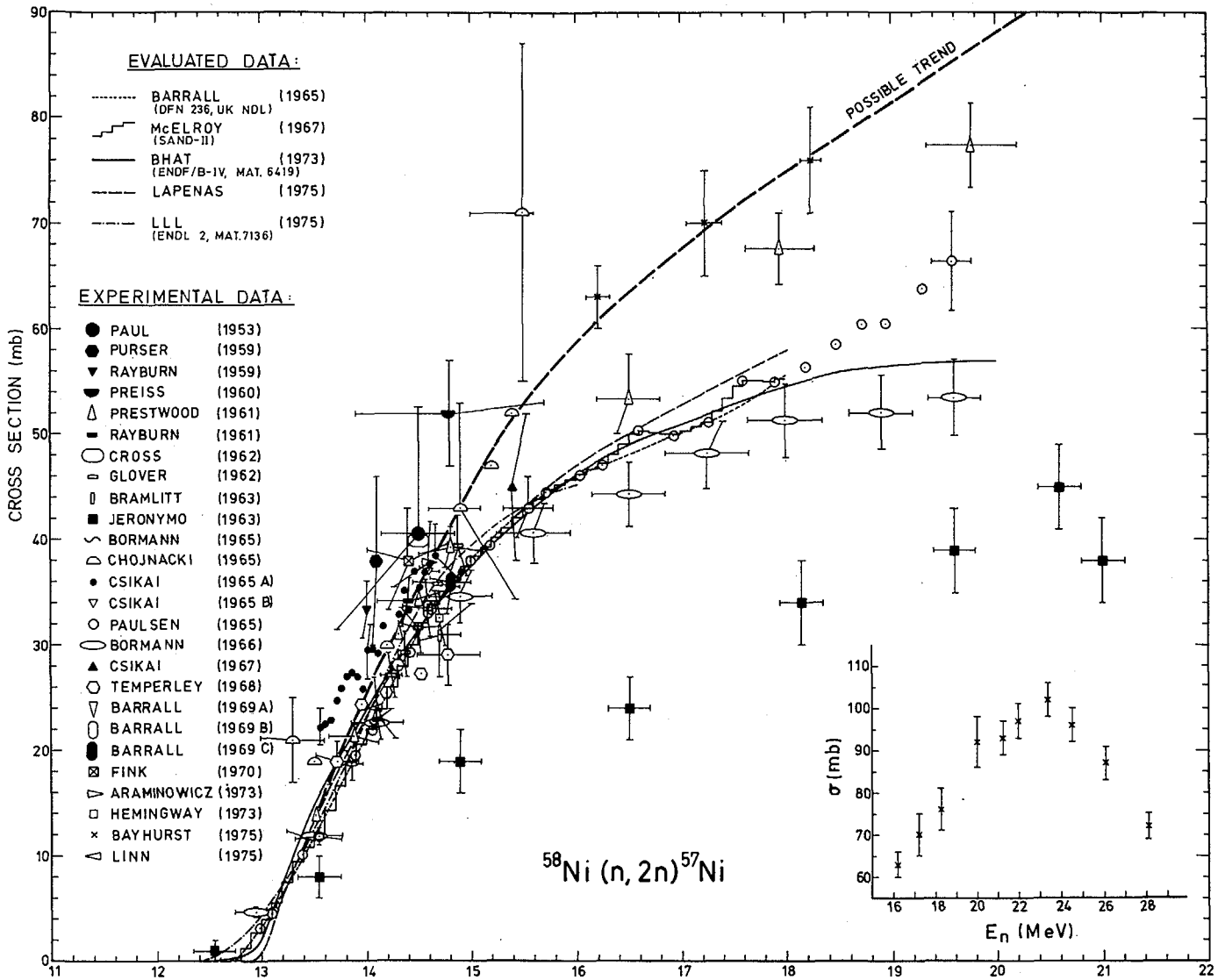
EVALUATED DATA:
BARRALL (1965) (DFN 236, UK NDL)
McELROY (1967) (SAND-II)
BHAT (1973) (ENDF/B-IV, MAT. 6419)
LAPENAS (1975)
LLL (1975) (ENDL 2, MAT.7136)
EXPERIMENTAL DATA:
PAUL (1953)
PURSER (1959)
RAYBURN (1959)
PREISS (1960)
PRESTWOOD (1961)
RAYBURN (1961)
CROSS (1962)
GLOVER (1962)
BRAMLITT (1963)
JERONYMO (1963)
BORMANN (1965)
CHOJNACKI (1965)
CSIKAI (1965 A)
CSIKAI (1965 B)
PAULSEN (1965)
BORMANN (1966)
CSIKAI (1967)
TEMPERLEY (1968)
BARRALL (1969 A)
BARRALL (1969 B)
BARRALL (1969 C)
FINK (1970)
ARAMINOWICZ (1973)
HEMINGWAY (1973)
BAYHURST (1975)
LINN (1975)
POSSIBLE TREND
58Ni (n, 2n)57Ni
σ (mb)
En (MeV)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)

Fig. 13

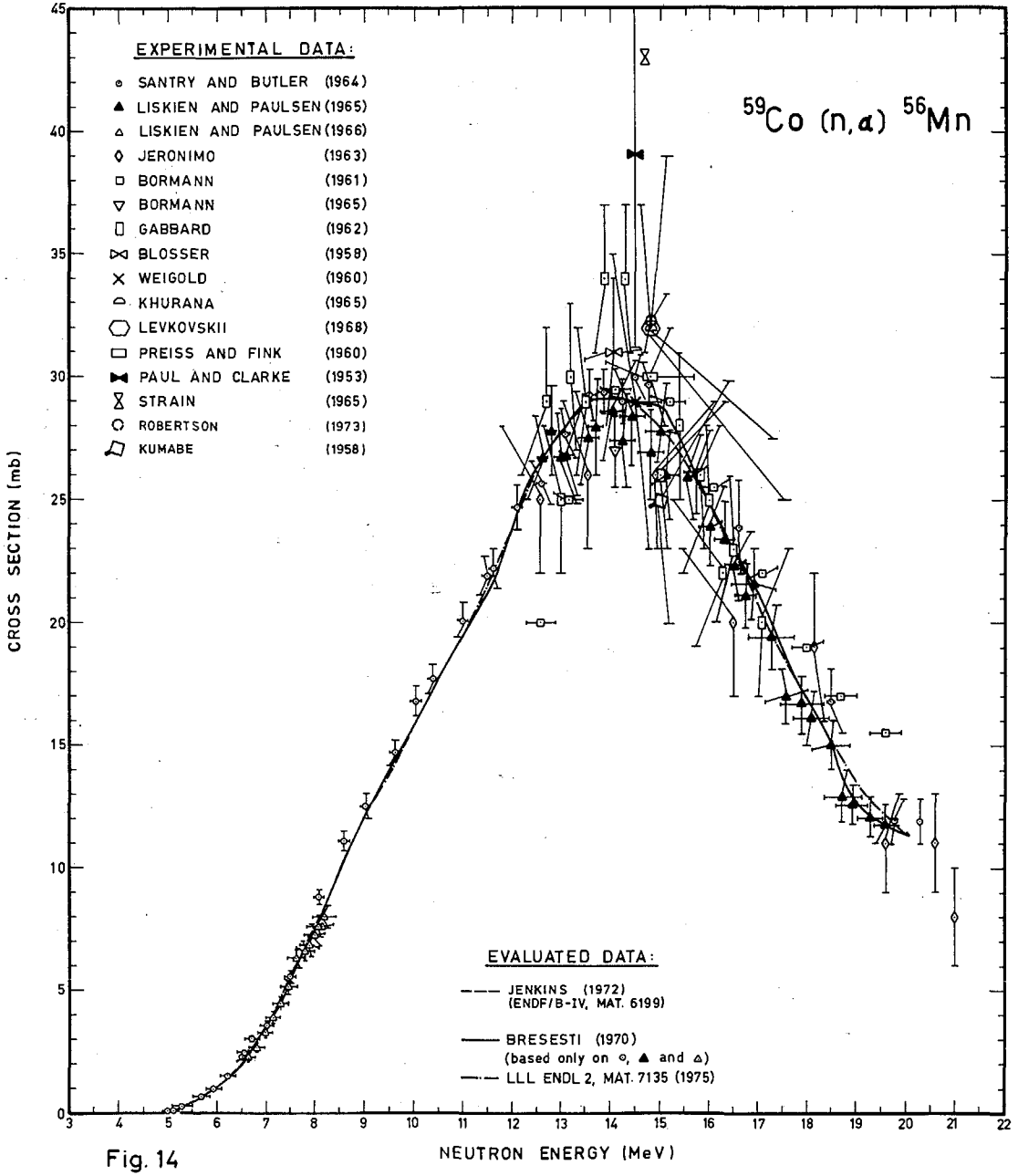
EXPERIMENTAL DATA:
SANTRY AND BUTLER (1964)
LISKIEN AND PAULSEN (1965)
LISKIEN AND PAULSEN (1966)
JERONIMO (1963)
BORMANN (1961)
BORMANN (1965)
GABBARD (1962)
BLOSSER (1958)
WEIGOLD (1960)
KHURANA (1965)
LEVKOVSKII (1968)
PREISS AND FINK (1960)
PAUL AND CLARKE (1953)
STRAIN (1965)
ROBERTSON (1973)
KUMABE (1958)
59Co (n,α) 56Mn
EVALUATED DATA:
JENKINS (1972)
(ENDF/B-IV, MAT. 6199)
BRESESTI (1970)
(based only on ⊙, ▲ and △)
LLL ENDL 2, MAT. 7135 (1975)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
5
10
15
20
25
30
35
40
45
3
4
5
6
7
8
9
10
11
12
13
14
15
16
17
18
19
20
21
22

Fig. 14

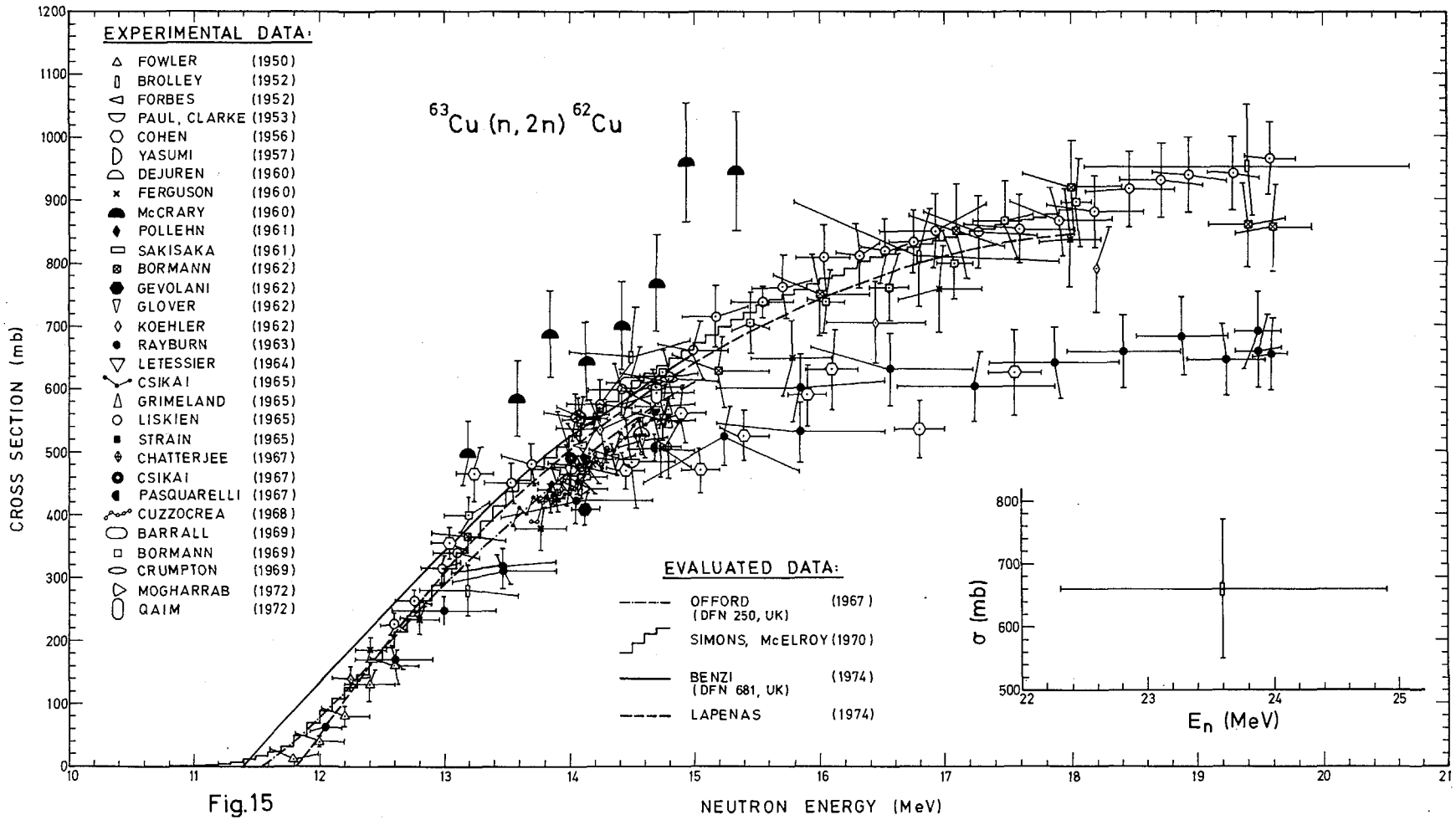
EXPERIMENTAL DATA:
FOWLER (1950)
BROLLEY (1952)
FORBES (1952)
PAUL, CLARKE (1953)
COHEN (1956)
YASUMI (1957)
DEJUREN (1960)
FERGUSON (1960)
McCRARY (1960)
POLLEHN (1961)
SAKISAKA (1961)
BORMANN (1962)
GEVOLANI (1962)
GLOVER (1962)
KOEHLER (1962)
RAYBURN (1963)
LETESSIER (1964)
CSIKAI (1965)
GRIMELAND (1965)
LISKIEN (1965)
STRAIN (1965)
CHATTERJEE (1967)
CSIKAI (1967)
PASQUARELLI (1967)
CUZZOCREA (1968)
BARRALL (1969)
BORMANN (1969)
CRUMPTON (1969)
MOGHARRAB (1972)
QAIM (1972)
63Cu (n, 2n) 62Cu
EVALUATED DATA:
OFFORD (1967) (DFN 250, UK)
SIMONS, McELROY (1970)
BENZI (1974) (DFN 681, UK)
LAPENAS (1974)
σ (mb)
En (MeV)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)

Fig.15

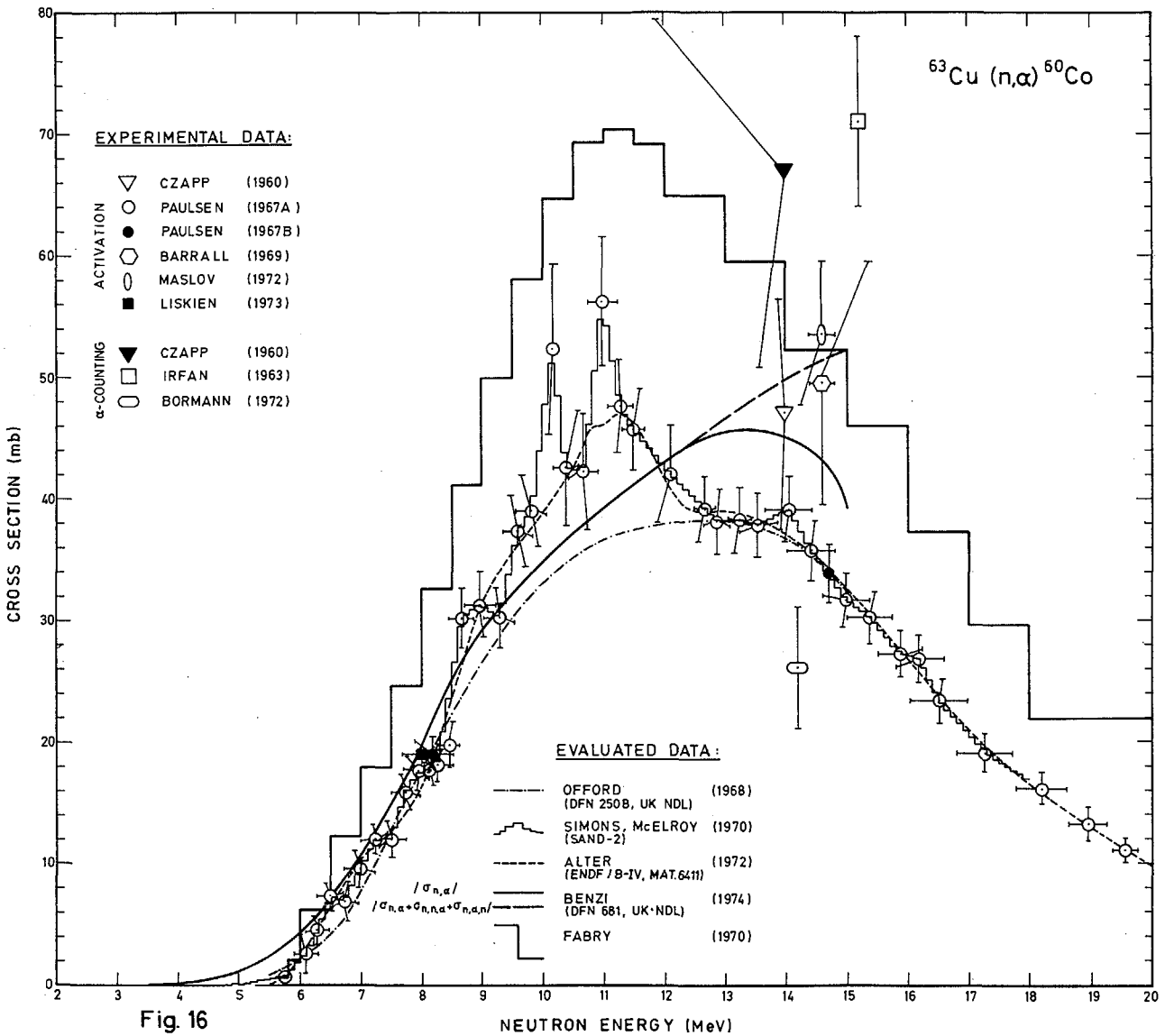
$^{63}Cu\ (n,\alpha)\ ^{60}Co$
EXPERIMENTAL DATA:
ACTIVATION
CZAPP (1960)
PAULSEN (1967A)
PAULSEN (1967B)
BARRALL (1969)
MASLOV (1972)
LISKIEN (1973)
α-COUNTING
CZAPP (1960)
IRFAN (1963)
BORMANN (1972)
EVALUATED DATA:
OFFORD (1968) (DFN 2508, UK NDL)
SIMONS, McELROY (1970) (SAND-2)
ALTER (1972) (ENDF/B-IV, MAT.6411)
BENZI (1974) (DFN 681, UK·NDL)
FABRY (1970)
$/\sigma_{n,\alpha}/$
$/\sigma_{n,\alpha}+\sigma_{n,n\alpha}+\sigma_{n,\alpha n}/$
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)

Fig. 16

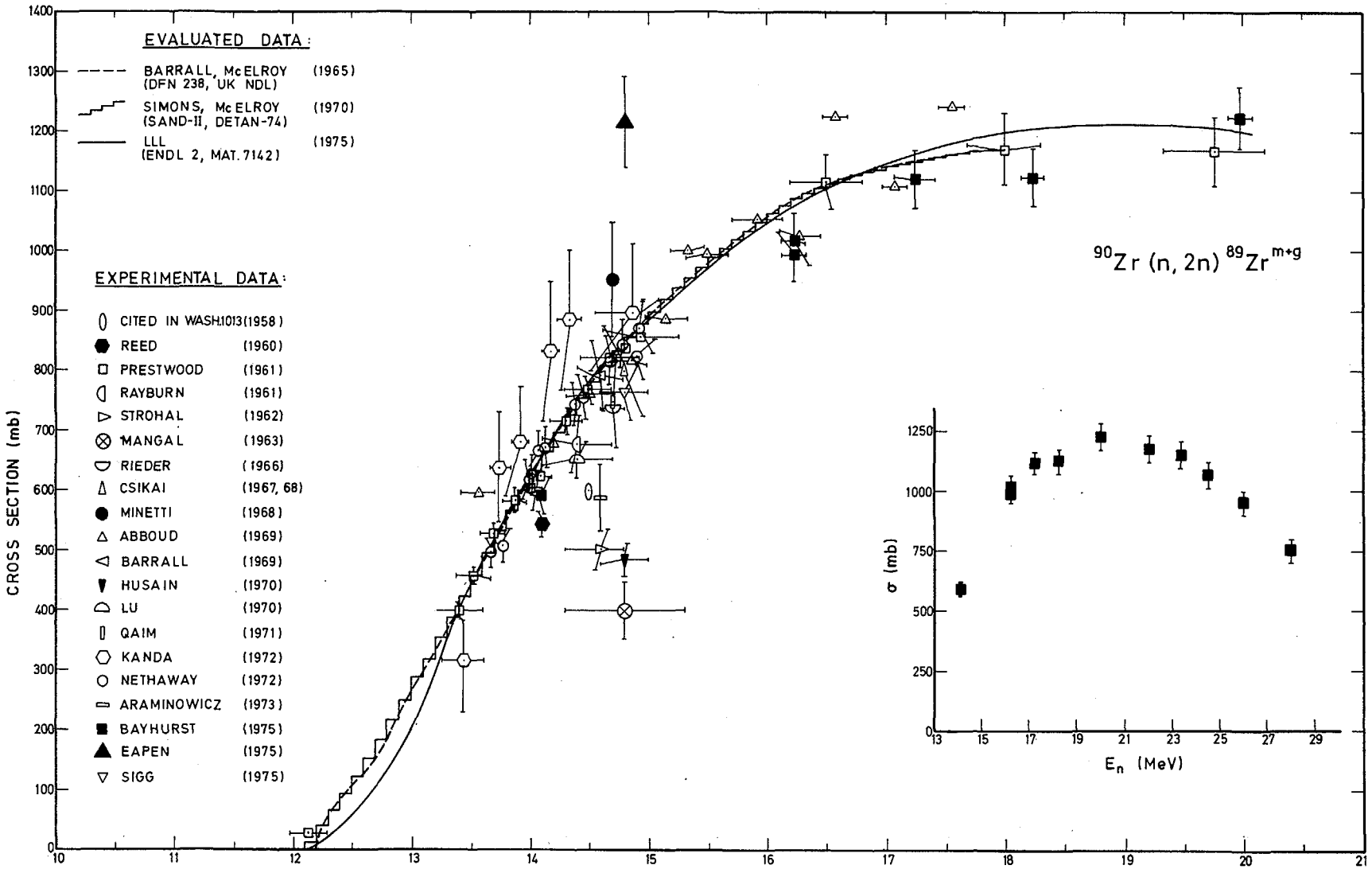
EVALUATED DATA:
BARRALL, McELROY (1965)
(DFN 238, UK NDL)
SIMONS, McELROY (1970)
(SAND-II, DETAN-74)
LLL (1975)
(ENDL 2, MAT. 7142)
EXPERIMENTAL DATA:
CITED IN WASH1013 (1958)
REED (1960)
PRESTWOOD (1961)
RAYBURN (1961)
STROHAL (1962)
MANGAL (1963)
RIEDER (1966)
CSIKAI (1967, 68)
MINETTI (1968)
ABBOUD (1969)
BARRALL (1969)
HUSAIN (1970)
LU (1970)
QAIM (1971)
KANDA (1972)
NETHAWAY (1972)
ARAMINOWICZ (1973)
BAYHURST (1975)
EAPEN (1975)
SIGG (1975)
$^{90}Zr\ (n, 2n)\ ^{89}Zr^{m+g}$
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
σ (mb)
E_n (MeV)

Fig. 17

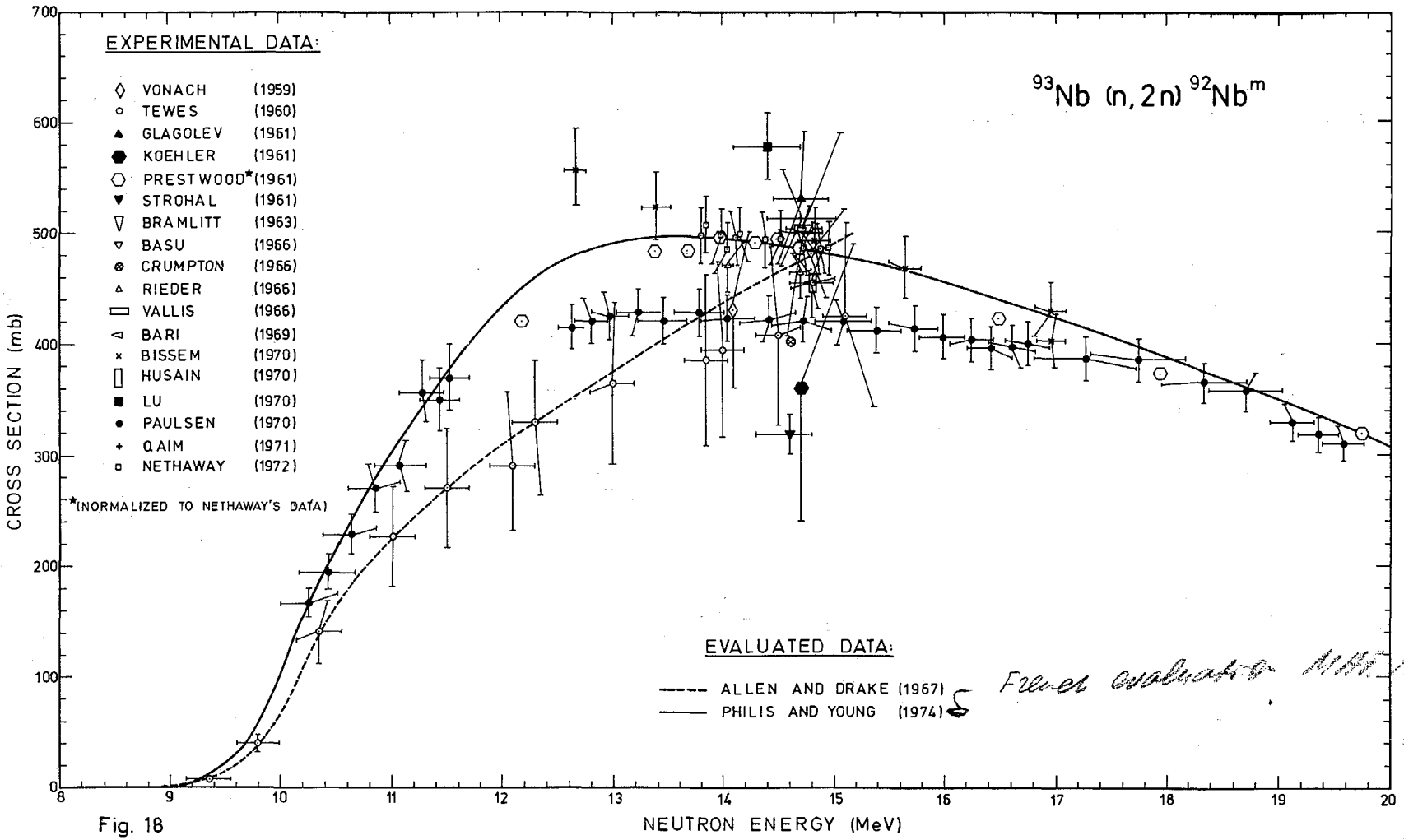
EXPERIMENTAL DATA:
VONACH (1959)
TEWES (1960)
GLAGOLEV (1961)
KOEHLER (1961)
PRESTWOOD* (1961)
STROHAL (1961)
BRAMLITT (1963)
BASU (1966)
CRUMPTON (1966)
RIEDER (1966)
VALLIS (1966)
BARI (1969)
BISSEM (1970)
HUSAIN (1970)
LU (1970)
PAULSEN (1970)
QAIM (1971)
NETHAWAY (1972)
*(NORMALIZED TO NETHAWAY'S DATA)
93Nb (n, 2n) 92Nbm
EVALUATED DATA:
ALLEN AND DRAKE (1967)
PHILIS AND YOUNG (1974)
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
0
100
200
300
400
500
600
700
8
9
10
11
12
13
14
15
16
17
18
19
20

Fig. 18

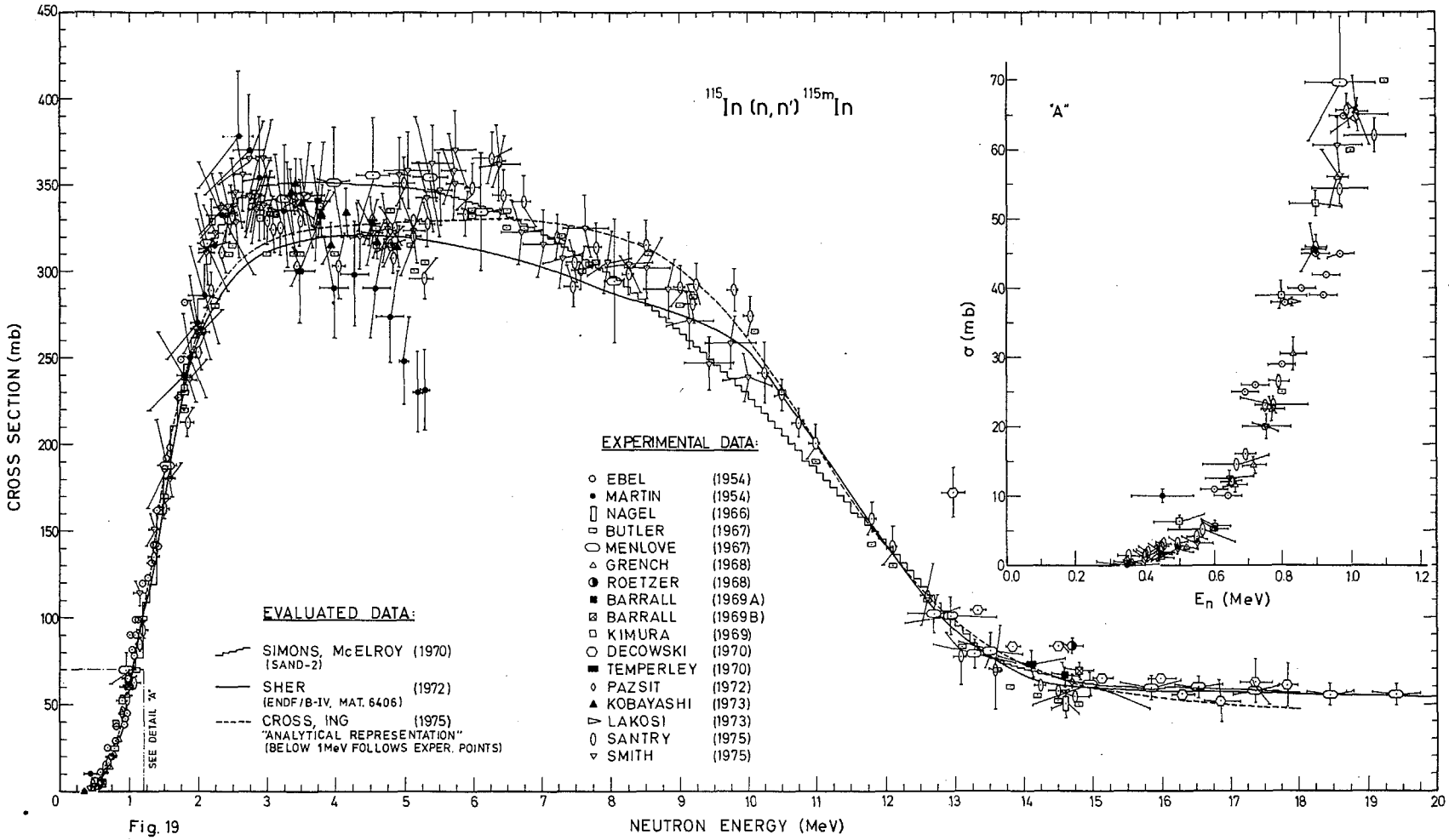
^{115}In (n, n') ^{115m}In
CROSS SECTION (mb)
NEUTRON ENERGY (MeV)
SEE DETAIL "A"
"A"
σ (mb)
E_n (MeV)
EVALUATED DATA:
SIMONS, McELROY (1970) (SAND-2)
SHER (1972) (ENDF/B-IV, MAT. 6406)
CROSS, ING (1975) "ANALYTICAL REPRESENTATION" (BELOW 1MeV FOLLOWS EXPER. POINTS)
EXPERIMENTAL DATA:
EBEL (1954)
MARTIN (1954)
NAGEL (1966)
BUTLER (1967)
MENLOVE (1967)
GRENCH (1968)
ROETZER (1968)
BARRALL (1969A)
BARRALL (1969B)
KIMURA (1969)
DECOWSKI (1970)
TEMPERLEY (1970)
PAZSIT (1972)
KOBAYASHI (1973)
LAKOSI (1973)
SANTRY (1975)
SMITH (1975)

Fig. 19

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