NUCLEAR DATA AND MEASUREMENTS SERIES

ANL/NDM-34

Graphical Representation of Neutron Differential Cross Section Data for Reactor Dosimetry Applications

by

Donald L. Smith

June 1977

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ARGONNE, ILLINOIS 60439, U.S.A.

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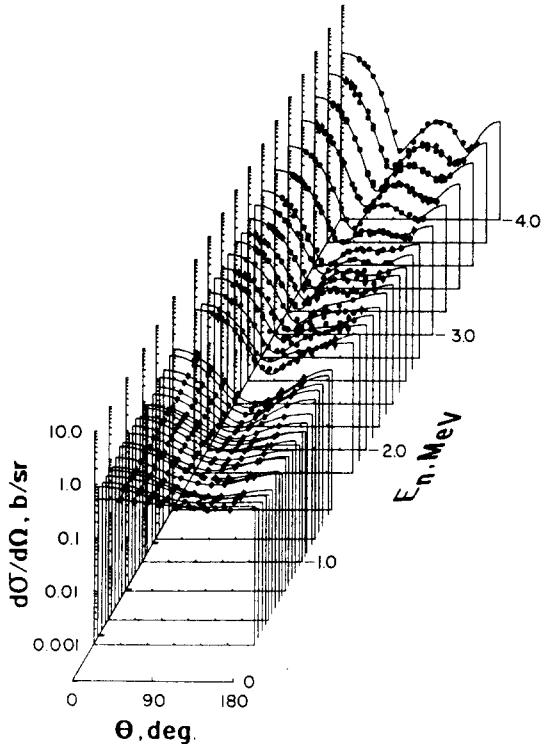

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In January 1975, the research and development functions of the former U.S. Atomic Energy Commission were incorporated into those of the U.S. Energy Research and Development Administration.

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NUCLEAR DATA AND MEASUREMENTS SERIES

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GRAPHICAL REPRESENTATION OF NEUTRON DIFFERENTIAL CROSS SECTION DATA FOR REACTOR DOSIMETRY APPLICATIONS*

by

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ABSTRACT

The need for improved understanding of the relationships between available differential and integral data for neutron reactions used in reactor dosimetry has prompted investigation of a method for graphically representing experimental differential data in a form which appears to be quite useful for dosimetry applications. The method involves weighting the differential cross sections by spectral functions and plotting these values. Graphs of this form clearly indicate which differential data are important for spectrum unfolding applications. Simultaneous plots of experimental and evaluated differential cross sections--weighted by spectral functions--provide a means for comparing evaluations from the point of view of their impact on specific dosimetry applications.

*This work was sponsored by the U.S. Energy Research and Development Administration.

I. INTRODUCTION

An important goal of current research in the field of reactor dosimetry is the elimination of disparities in the results of differential and integral measurements for neutron reactions commonly used in dosimetry by the activation method. Good agreement between measured spectrum-average cross sections and those computed from evaluated differential reaction cross sections and spectral information for various well-established benchmark fields is a prerequisite to application of this method in dosimetry for less-well-known fields (e.g. those in large power reactors).

Considerable progress has been made in this area during the past few years. Several well-characterized benchmark fields have been developed and an international effort has been undertaken to intercompare the results of measurements at these facilities (1). Similarly, there has been a great deal of progress in improving the knowledge of the differential cross sections. This has come about as a result of recent experimental activity and improved evaluation procedures in this area.

Consistency of integral and differential results to within 2-5% is sought (1). It appears that this goal is realistic (1,2), but it has not yet been achieved (3). The relevant question now is: "Where do we go from here?" It is clear that improvements are required in measurements of microscopic integral and differential cross sections as well as in the characterization of the benchmark fields utilized for cross-section data validation. However, because of the considerable cost involved in funding such an effort, careful planning is needed to define the scope and accuracy of the requisite experimental work so that resources will not be wasted on unnecessary programs.

Recently, an attempt was made to examine relationships between microscopic integral and differential cross sections with the objective of

developing a simple procedure for determining where emphasis should be directed in the measurement of differential cross sections (4). The present report deals with results from a continuing investigation of this topic. In particular, a method is described for examining the quality of the experimental differential data base and existing evaluations of these data for neutron activation reactions from the point of view of specific dosimetry applications. The method involves plotting both experimental and evaluated results after weighting the cross sections by functions representing neutron spectra of interest for applications. This graphical approach has been used before for representation of evaluated data, but it does not appear to have been used for the examination of experimental results. Several examples are presented in this report to illustrate the method.

II. CONCEPT AND EXAMPLES

Traditionally, the compilation and evaluation of differential experimental data has been approached almost entirely from the point of view of pure microscopic cross sections; as an example, Fig. 1 shows results from a recent analysis of the $^{115}In(n;n')^{115m}In$ reaction (5). This type of graphical representation emphasizes the relative importance of various energy regions from the point of view of a uniform neutron field (equal numbers of neutrons per unit energy interval at all energies, $\phi(E)$ = constant) and it is useful for monoenergetic-neutron considerations.

However, plots of (E_i,σ_i) pairs and $\sigma(E)$ curves are not useful for examining the differential data from the point of view of realistic spectra ($\phi(E) \neq$ constant). It is suggested that a superior method for presentation of differential data for reactor applications is to plot pairs $(E_i, \sigma_i\phi\ (E_i))$ and curves $\sigma(E)\phi(E)$ over the range where $\sigma\phi$ is not essentially zero. As an example, Fig. 2 presents the same differential data for $^{115}In(n;n')^{115m}In$ as

appeared in Fig. 1 except that the cross sections are weighted by two benchmark spectral functions: i) thermal-neutron fission of ^{235}U (6) and, ii) th CFRMF spectrum (7). From the point of view of these two benchmark fields, Fig. 2 provides a great deal more useful information than Fig. 1 about the quality of the available experimental data and on how adequately the evaluat represent these data in the regions of importance. While it is fission drive the CFRMF spectrum is considerably softer than a pure ^{235}U fission spectrum and, therefore, tests the differential data for the $^{115}In(n;n')^{115m}In$ reactic below $\sim$ 1 MeV to a greater extent. Since the spectrum-average cross section is the integral of $\sigma\phi$ for a normalized ϕ, it is clear that the existing differential data define the shape of the response functions with relatively little ambiguity for both ^{235}U fission and CFRMF except possibly in the range E_n = 1 - 2.5 MeV where the differences are large enough to introduce noticeable uncertainty. The recent evaluation of Smith (5) provides a significant improvement in integral-differential comparisons over ENDF/B-IV (8). Natural it must be kept in mind that integral-differential comparisons depend on the accuracy of the neutron spectrum representation for the fields in questio as well as on the quality of the microscopic integral and differential cross section data.

It should be pointed out that plots such as those in Fig. 2 will not indicate problems in overall normalization which might result from improper neutron fluence measurement or use of incorrect decay data--unless these errors apply to isolated data sets while most of the experimental data are properly normalized. It is imperative that integral and differential measurements must utilize consistent decay data and, if possible, consistent fluence normalization techniques. Otherwise, intercomparison is meaningless and shape effects become hopelessly inseparable from overall normalization considerations.

Since the pure fission spectra are always harder than the corresponding fission driven reactor spectra, it is clear from Fig. 2 that there is no real need for additional differential measurements on the $^{115}In(n;n')^{115}In$ reaction above $\sim$ 6 MeV for fission reactor applications.

The (n;p) reactions for $^{46,47,48}Ti$ were recently evaluated by Philis et al. (9). These reactions have higher effective thresholds than the $^{115}In(n;n')^{115m}In$ reaction and respond to the high-energy regions of fission-driven neutron spectra. Therefore, plots of $\sigma\phi$ for various such systems differ little from each other in shape. Consideration of results for a ^{235}U fission spectrum is sufficient to investigate shape effects.

The current status for the $^{46}Ti(n;p)^{46}Sc$ reaction is summarized in Figs. 3 and 4. Clearly the situation is unacceptable. There are several differential data sets covering the region of largest response (E_n = 4 - 10 MeV), but there are large discrepancies. Approximately half of the response region (E_n = 3.7 - 5 MeV and 7 - 10 MeV) is covered by only one data set. A small- but not negligible-portion of the response comes from below $\sim$ 3.7 MeV and no experimental data cover this region. There is a need for more differential measurements in the region from 3 - 10 MeV for this important dosimetry reaction.

If one apparently discrepant data set is ignored, it can be seen from Figs. 6 and 7 that the $^{47}Ti(n;p)^{47}Sc$ reaction is reasonably well defined by differential data over the important response region of 1 - 10 MeV for fission neutron systems. Additional measurements above 6 MeV would be desirable since this region is covered by only one differential data set.

The situation for the $^{48}Ti(n;p)^{48}Sc$ reaction is quite poor. The large differences in the ENDF/B-IV (8) and Philis et al. (9) evaluations are apparent in Fig. 7, but are truly accentuated in Fig. 8. Since there is essentially only one differential data set covering the most important response

region for $^{48}Ti(n;p)^{48}Sc$, new differential measurements are certainly needed for this reaction. The region of emphasis for fission reactor applications is 5 - 13 MeV.

III. SUMMARY

The proposed method for plotting experimental and evaluated differential cross sections appears to offer a convenient means for comparison of experimental and evaluated results from the point of view of reactor dosimetry. Furthermore, weaknesses in the existing experimental data base and the evaluations are emphasized by this approach. The present report presents examples which include only partially-moderated fission neutron spectra and threshold reactions. The method is also applicable for reactions with no threshold and for soft neutron spectra characteristic of heavily moderated systems or hard neutron spectra characteristic of proposed fusion devices or medical irradiation facilities. The need for optimal use of available research funds in the development of nuclear data for technological applications provides a strong justification for the use of the present approach and other analytical techniques which can help to define specific nuclear data needs.

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FIGURE CAPTIONS

Fig. 1. Plot of experimental data (Ref. 5) and the ENDF/B-IV (Ref. 8) and Smith (Ref. 5) evaluations for the differential cross section of the $^{115}In(n;n')^{115m}In$ reaction.
(ANL Neg. No. 116-77-392)

Fig. 2. Plots of $\sigma\phi$ for the $^{115}In(n;n')^{115m}In$ reaction computed using the spectrum corresponding to thermal-neutron fission of ^{235}U (Ref. 6) and the CFRMF spectrum (Ref. 7). Experimental differential data obtained from Ref. 5. Differential evaluations are those of ENDF/B-IV (Ref. 8) and Smith (Ref. 5).
(ANL Neg. No. 116-77-387)

Fig. 3. Plot of experimental data (Ref. 9) and the ENDF/B-IV (Ref. 8) and Philis et al. (Ref. 9) evaluations for the differential cross section of the $^{46}Ti(n;p)^{46}Sc$ reaction.
(ANL Neg. No. 116-77-386)

Fig. 4. Plot of $\sigma\phi$ values for the $^{46}Ti(n;p)^{46}Sc$ reaction computed using the spectrum corresponding to thermal-neutron fission of ^{235}U (Ref. 6). Experimental differential data obtained from Ref. 9. Differential evaluations are those of ENDF/B-IV (Ref. 8) and Philis et al. (Ref. 9).
(ANL Neg. No. 116-77-388)

Fig. 5. Similar to Fig. 3 except that the reaction considered is $^{47}Ti(n;p)^{47}Sc$.
(ANL Neg. No. 116-77-393)

Fig. 6. Similar to Fig. 4 except that the reaction considered is $^{47}Ti(n;p)^{47}Sc$.
(ANL Neg. No. 116-77-390)

Fig. 7. Similar to Fig. 3 except that the reaction considered is $^{48}Ti(n;p)^{48}Sc$.
(ANL Neg. No. 116-77-391)

Fig. 8. Similar to Fig. 4 except that the reaction considered is $^{48}Ti(n;p)$ (ANL Neg. No. 116-77-389)

Fig. 1

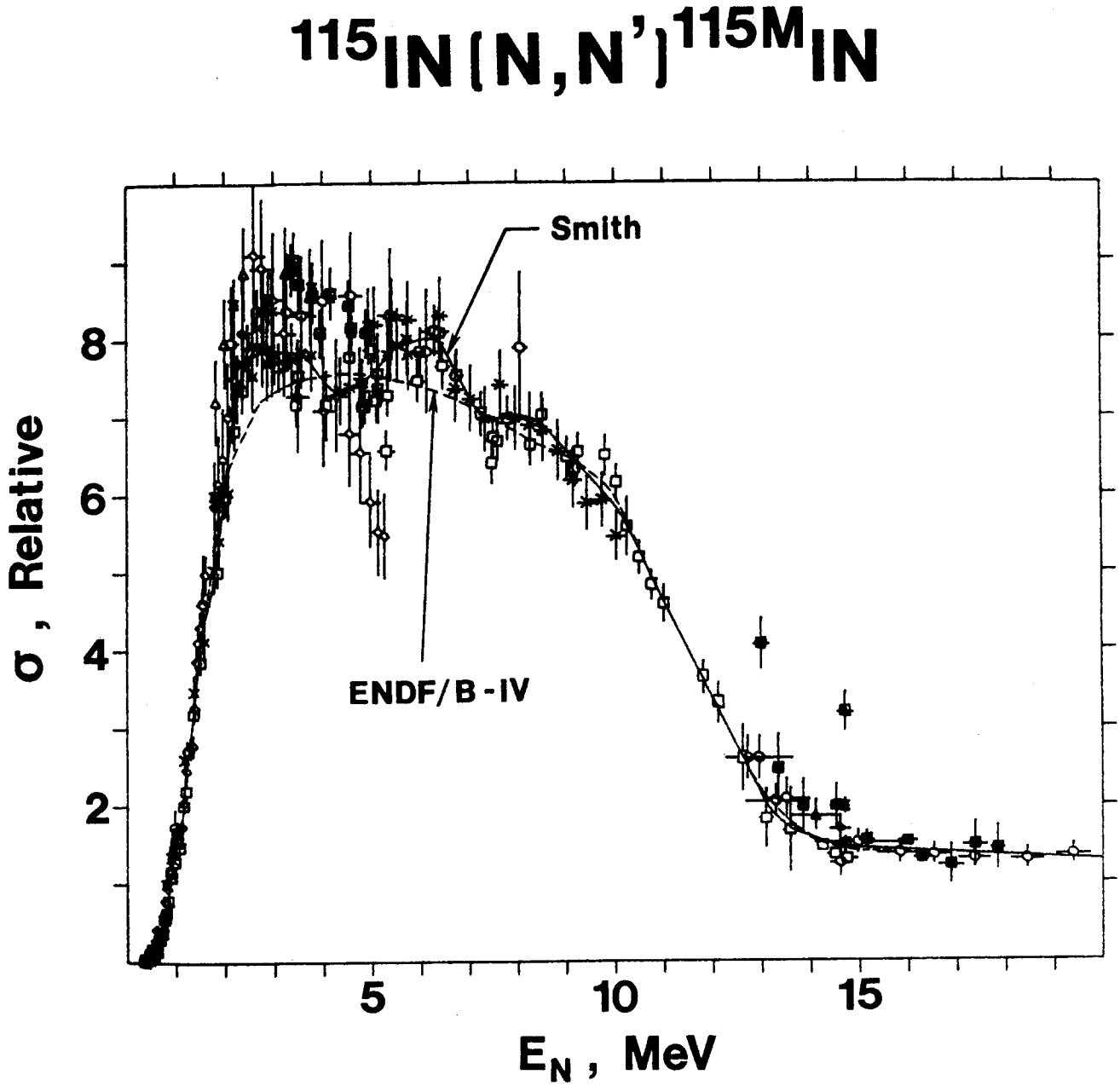


Fig. 2

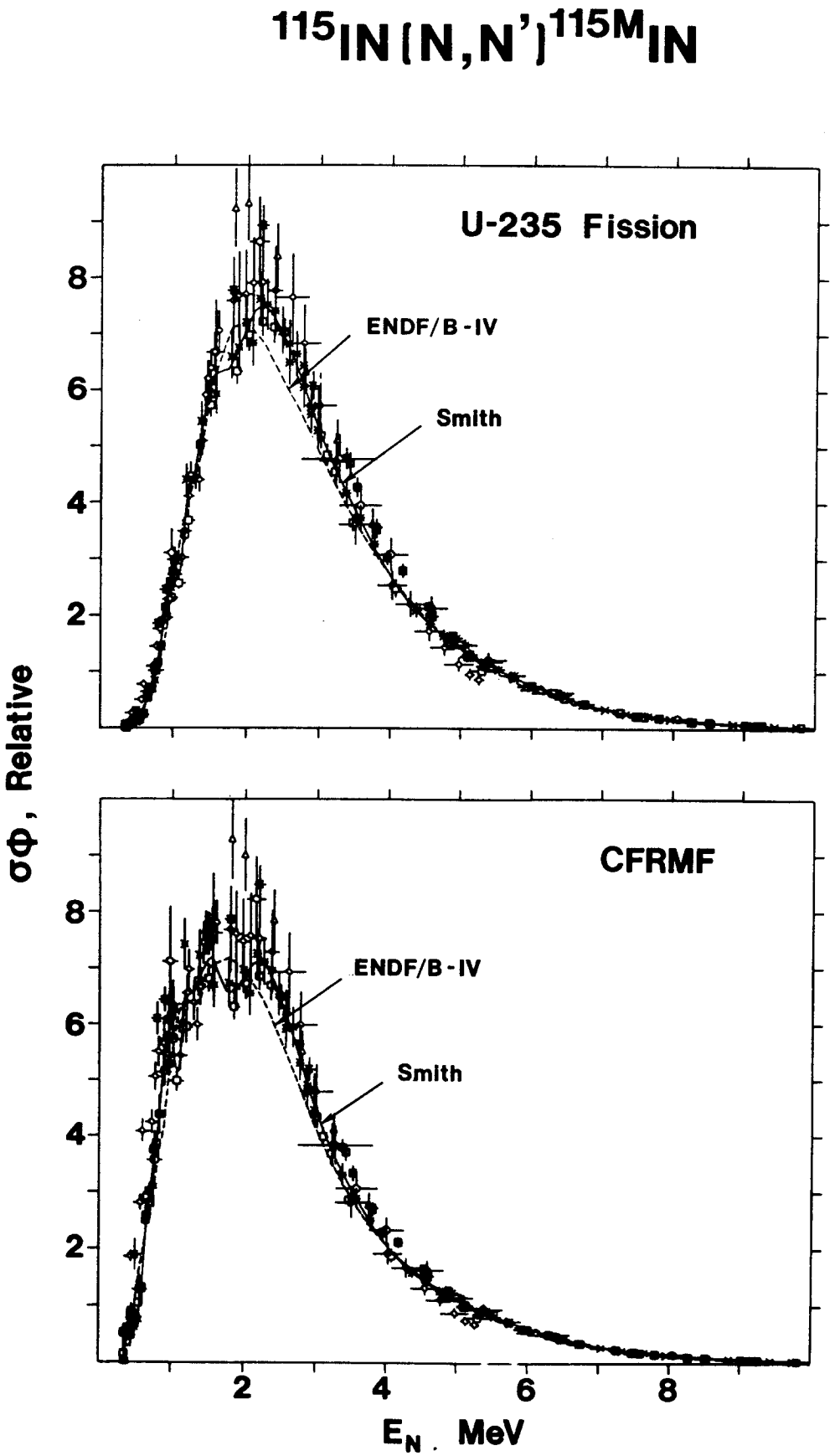

^{115}IN(N,N')^{115M}IN
U-235 Fission
ENDF/B-IV
Smith
CFRMF
ENDF/B-IV
Smith
σΦ, Relative
2
4
6
8
2
4
6
8
E_N, MeV

Fig. 3

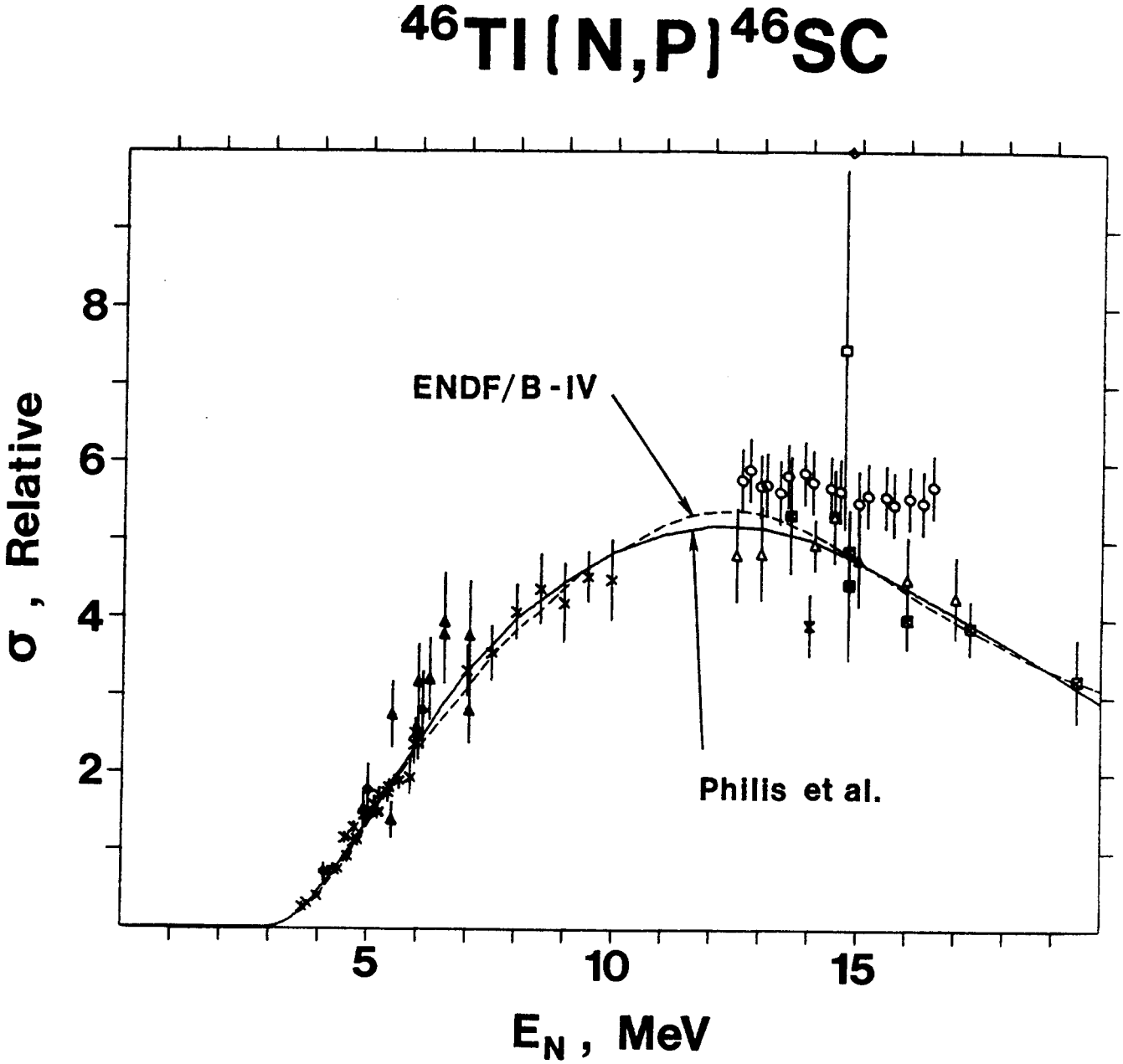
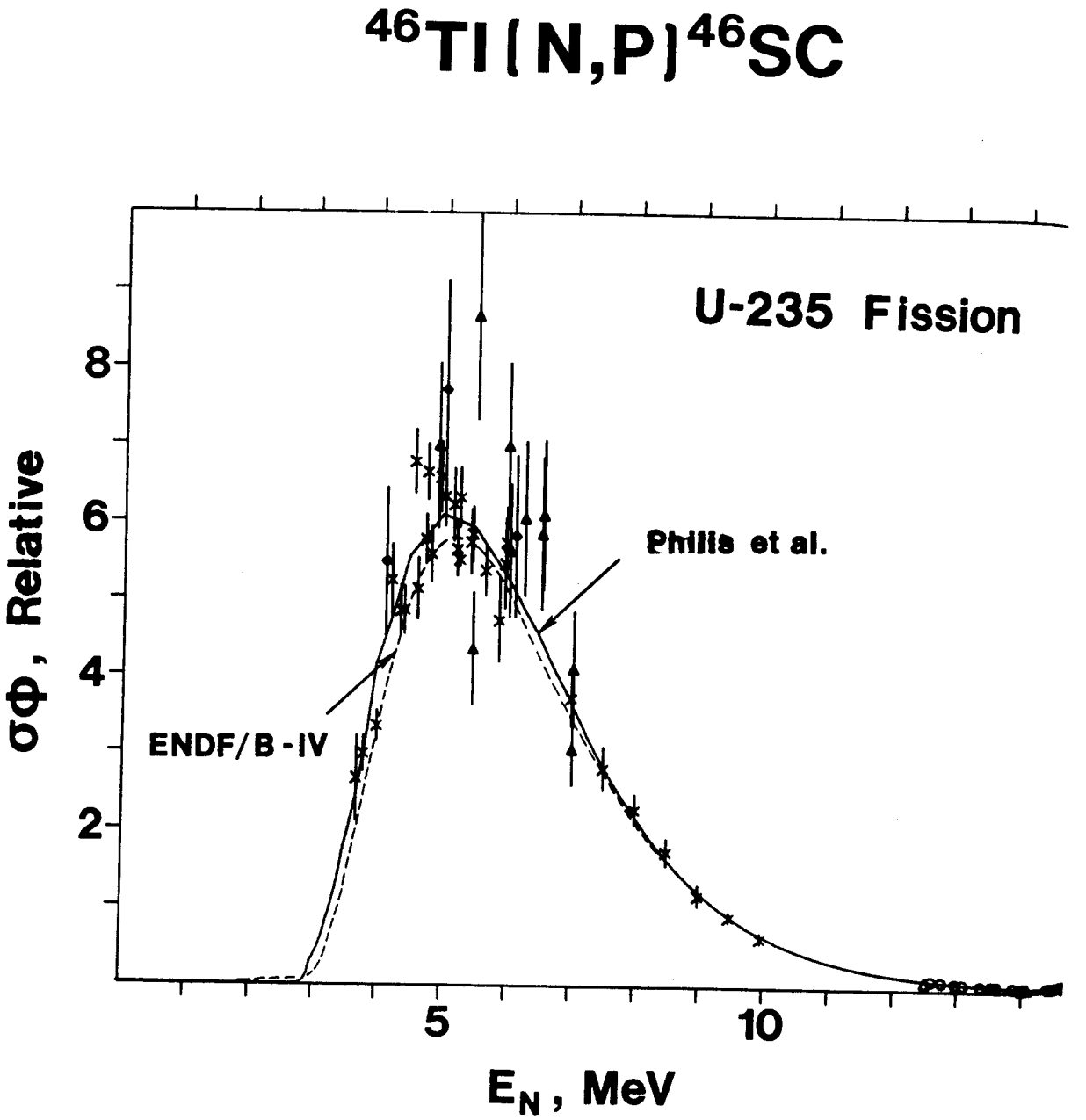

46TI(N,P)46SC
ENDF/B-IV
Phillis et al.
σ, Relative
8
6
4
2
5
10
15
EN, MeV

Fig. 4


46TI(N,P)46SC
U-235 Fission
Philis et al.
ENDF/B-IV
σφ, Relative
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EN, MeV

Fig. 5

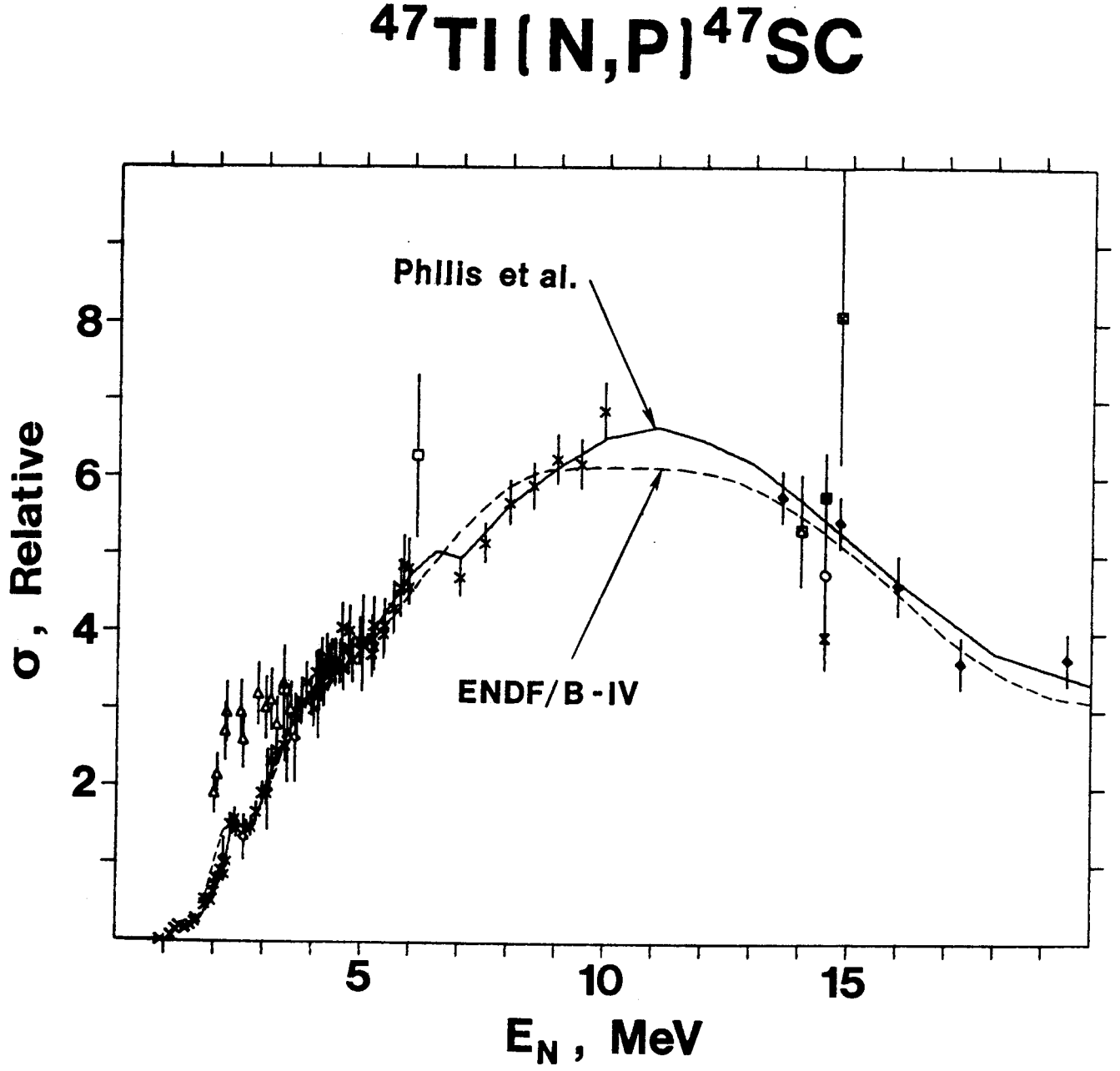
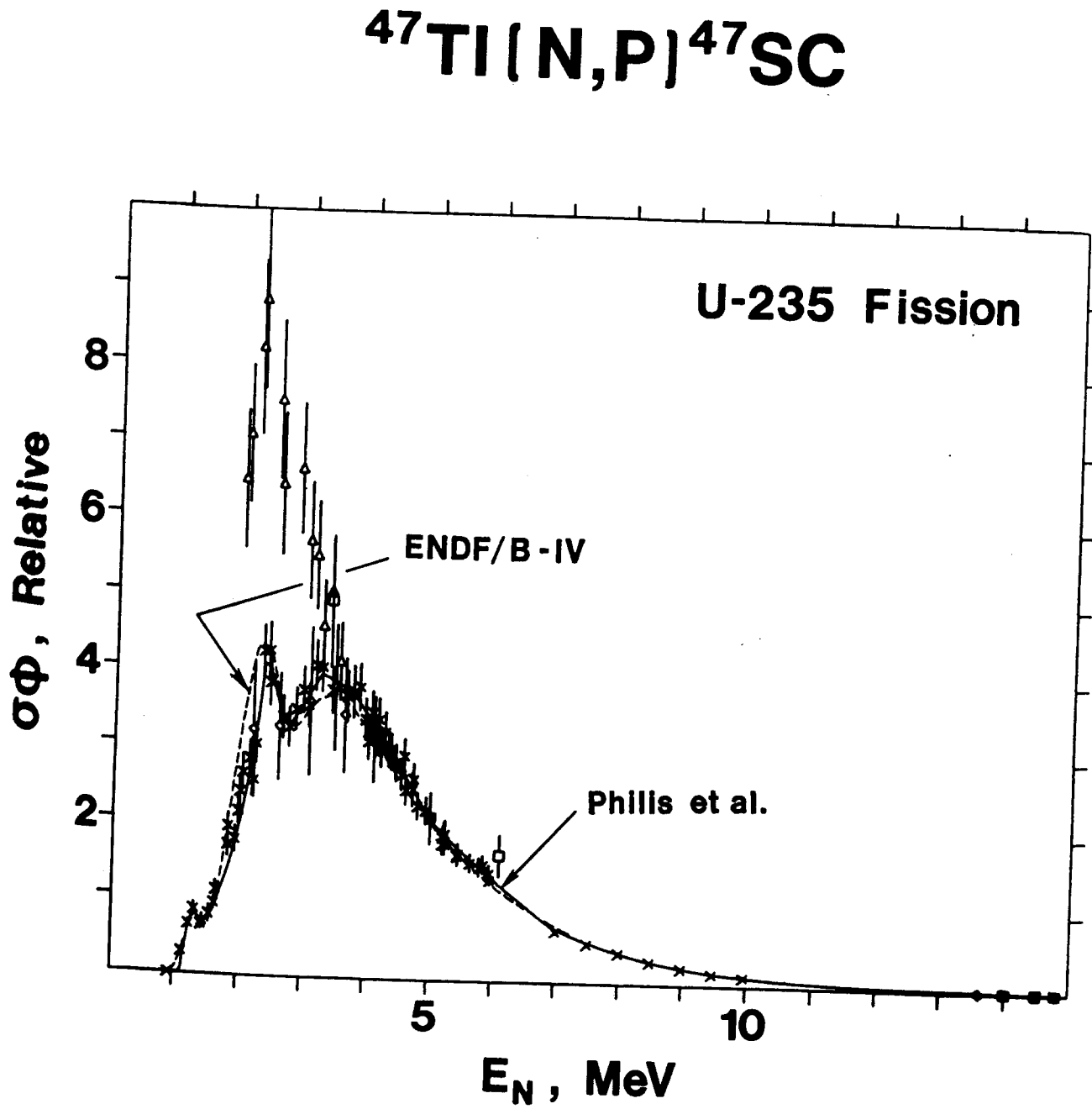


Fig. 6


^{47}TI(N,P)^{47}SC
U-235 Fission
ENDF/B-IV
Phillis et al.
σΦ, Relative
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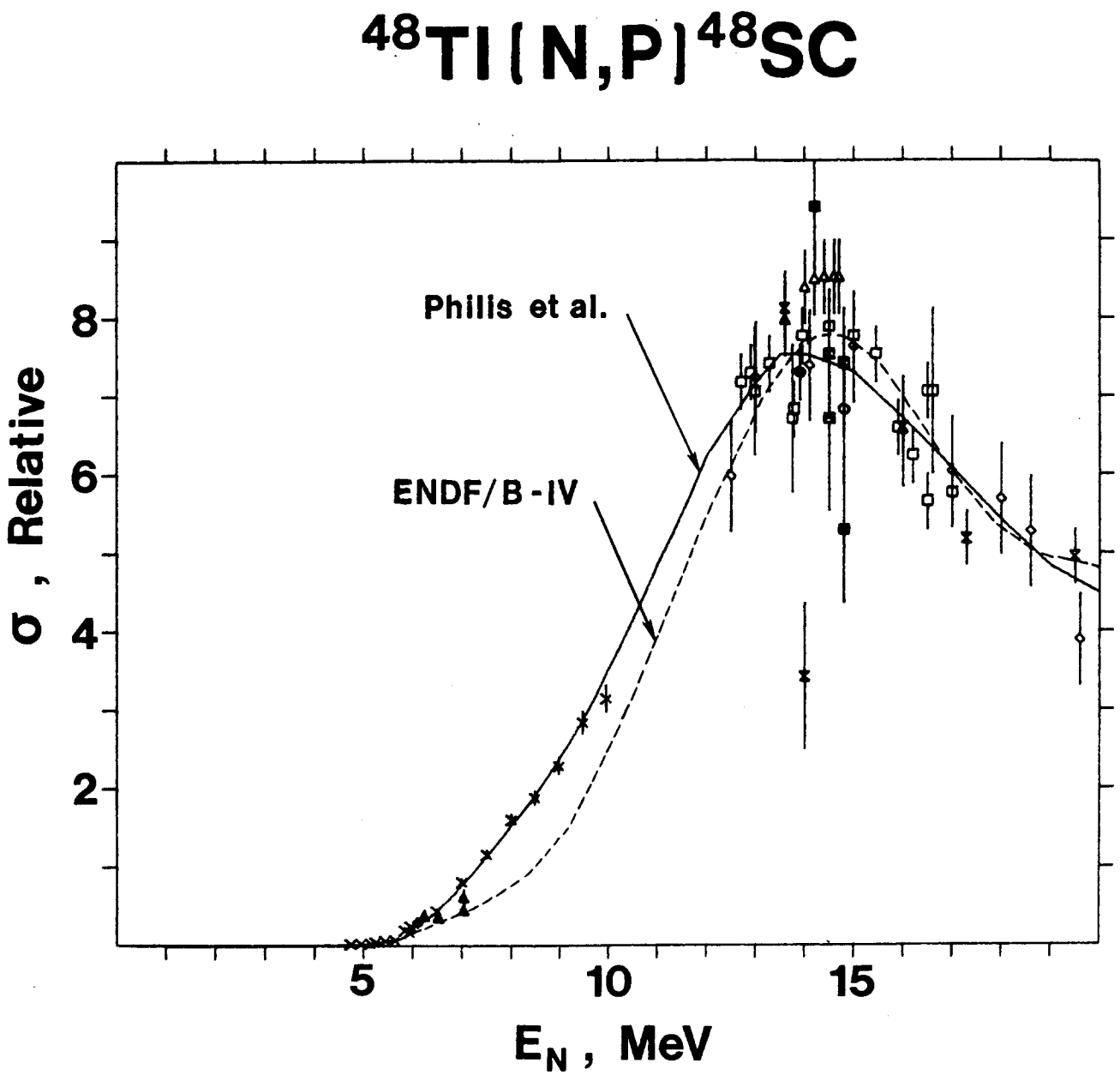
E_N, MeV

Fig. 7


^{48}TI(N,P)^{48}SC
Phillis et al.
ENDF/B-IV
σ , Relative
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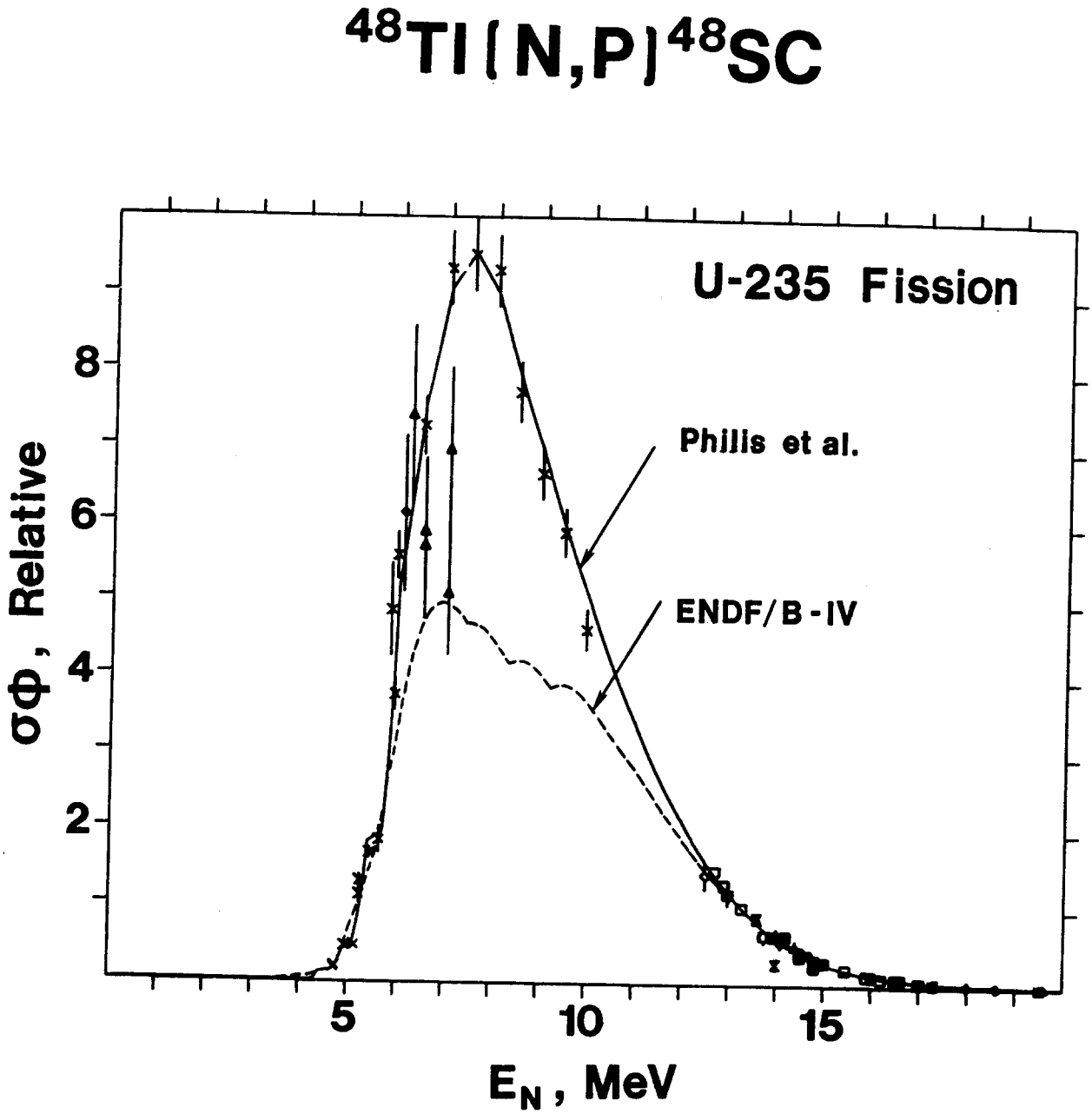
E_N , MeV

Fig. 8


^{48}TI(N,P)^{48}SC
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