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EVALUATION AND PROCESSING OF NUCLEAR DATA*

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ABSTRACT

The role a nuclear data evaluator plays in obtaining evaluated nuclear data, needed for applications, from measured nuclear data is surveyed. Specific evaluation objectives, problems, and procedures are discussed. The use of nuclear systematics to complement nuclear experiment and theory is described. Using the Evaluated Nuclear Data File (ENDF) as an example the formatting, checking, and processing of nuclear data is discussed as well as the testing of evaluated nuclear data in the calculation of integral benchmark experiments. Other important topics such as the Probability Table Method and interrelation between differential and integral data are also discussed.

1.0 Introduction

1.1 The Psychology of Evaluation

The process of evaluation involves decision making. The objectives of a nuclear data evaluator are to recommend values for nuclear data and also indicate the degree of confidence that can be placed in those recommendations. Often the experimental data being examined by the evaluator has quoted errors that are not realistic. Nevertheless, the evaluator is expected to estimate the

most probably correct values of nuclear data. The evaluator is like a juror. From the evidence, no matter how contradictory it may be, the evaluator is supposed to get at the truth. As a juror's decision must be within the court of law, the evaluator's recommendation must be consistent with the best laws of physics. The evaluator need not be an expert in all phases of nuclear physics but where his knowledge is deficient he must be capable of incorporating the recommendations of other experts into his evaluation.

There is no prescribed college course for nuclear data evaluation as there is for nuclear physics, nuclear engineering, reactor physics, nuclear chemistry, etc. Evaluation is a combination of art and science. Evaluation used to be more art than science, since there was little data and the evaluator depended on nuclear systematics or just plain guesswork in order to recommend data for use in applications. Today, evaluation is more science than art. There is more data that can be considered and the evaluation is expected to be consistent with all of the observable facts. Sometimes observable facts and evaluations take on political importance. Good evaluations that can calculate observable facts from first principles are taken seriously by reactor designers and even play a role in international data exchange agreements. The evaluator through his recommendation can have an impact on nuclear power programs.

The evaluator must have the finest moral character. He must be uncorruptible. His recommendations must be supported by experimental and theoretical considerations and not be strongly influenced by values favored by particular nuclear applications.

The nuclear data base provided by the evaluator is important to both basic science and applied science. The basic scientist examines the nuclear data base and wants to know why the nuclear data are what they are. He seeks to explain the systematics of nuclear data through an understanding of fundamental nuclear

forces. For the basic scientist the nuclear data base must consist of hard facts confirmed by experiment. On the other hand, the applied scientist accepts nuclear data as they are and proceeds to apply them. However, gaps in the hard facts in the data base often prevent the applied scientist from finishing his work and he favors a data base complete in what he requires even if the gaps must be filled by approximate methods.

2.0 The Observable Facts

2.1 Deductive Conclusions

A configuration for measuring nuclear cross sections (probabilities) is a beam of projectiles incident on a sample of target nuclei with detectors to measure the reactions products or radioactivity of the residual nuclei. Conclusions are always inferred and never directly observed. For example, a neutron can be observed through the n-alpha reaction, capture gamma rays, proton recoil measurements, and induced radioactivity but not through the direct observation of a neutron. Because of the indirect methods that are used, the evaluator should closely examine the basis upon which the measurer drew his conclusions.

2.2 Inherent Averaging

All measured quantities are averaged to some degree. Neutron sources, even nominally monoenergetic ones, produce neutrons over a range of energy. As a result, the measured cross section may not apply to a single valued energy. If the measurement is made at a certain angle with respect to the incident beam, the dimensions of the source and detector will cause an averaging over angle. For nonelastic processes, energy and angular averaging of the cross sections for the particles emitted will take place for the same reason. If the energy and angular averaging is small, the measurements are called differential. For some nuclear data, energy averaged cross sections are deliberately obtained such as

cross sections averaged over a $1/E$ (resonance integral), Maxwellian, polonium-beryllium, and Cf-252 spontaneous fission spectra. These are called integral measurements. More complex integral measurements are those that occur in reactor physics. The most important of these is criticality of a reactor assembly. When the neutron sources due to neutron producing reactions is exactly balanced by the neutron losses due to absorption and leakage the multiplication factor is unity to a high degree of precision, i.e., the multiplication factor is typically 1.000 plus or minus 0.002. Other measurements are sub and supercriticality, reactivity coefficients, activations, and ratios of activations in the reactor spectrum. These integral measurements are difficult to interpret because there are a large number of nuclides and reactions as well as averaging over energy and angle but the measurements can be very precise when compared to differential data measurements.

2.3 Experimental Difficulties

When measuring nuclear cross sections, a number of experimental difficulties must be overcome. A common difficulty is shown in Figure 2.1. Measurements of neutron cross sections can be uniquely determined only if the sample consists entirely of the original nucleus or neighboring isotopes whose abundances and cross sections are well known and therefore can be used to correct the measurement.

Thick targets can give rise to erroneously large cross sections due to multiple scattering of the incident beam instead of a simple once-through the target reaction probability. Scattering of the incident beam can greatly increase the particle mean free path in the target, thus increasing the reaction probability.

The loss of energy of incident particles in the sample through elastic or inelastic scattering can also affect the cross section measurement if the cross section is strongly energy-dependent. If the energy of the incident particle is sharply lowered the measured cross section will include some reactions taking place at energies where the cross section may be very different than the cross section at the energy of the incident beam.

A well-known case of the effects of multiple scattering and energy degradation arises in the measurement of the high energy capture cross section, which is quite small. Neutrons with their energy degraded due to scattering are captured at much lower energies where the cross section is high. This problem can be avoided only by using very thin samples for the target or by performing extensive correction analysis.

Source characteristics are also important. The energies of particles from white sources (distributed over energy) are usually separated by time-of-flight measurements. The major uncertainties in the particle energy is caused by the finite width of the time channels in which the particles are produced and detected. Incident beams produced by charged particle reactions can be nominally monoenergetic but also have an energy spread due to accelerator and target properties. Furthermore, as the energy of the charged particle producing source reactions is increased, additional source reactions may be created that must be corrected for in the cross section measurements.

Ideally, the geometry of the measurement should be simple in order to facilitate analysis. Slab, cylindrical, and spherical geometries are the easiest to analyze. In practice, however, the experimental arrangement makes many compromises. Point or plain sources are only approximately realized and the detecting geometry is seldom ideal.

Hopefully, a cross section measurement will have a reasonable count rate for the events taking place. If the count rate is small, it will be difficult to obtain good statistical accuracy and maintain stable conditions for long periods of time. Conversely, large count rate experiments can be plagued by count rate losses due to successive events arriving at the detector before it has recovered.

The above discussion contains examples of the corrections that can be made in the analysis of any experiment. If a large correction is necessary, it does not necessarily mean that it is a poor experiment. It is the accuracy of the correction that is important. A large correction that can be accurately made can be preferable to smaller, less precise corrections.

2.4 Experimental Uncertainties

Every recommended number should have an uncertainty associated with that number. These uncertainties are called data covariances. Numbers without an assigned uncertainty have little scientific significance. They cannot be assumed to have zero error but can be taken only as an order of magnitude even if several digits are given. If the number is known better than to an order of magnitude, its degree of precision should be given. Data covariances can consist of random and systematic errors. The random errors are statistical in nature and can be reduced only by more, longer, or more efficient measurements. The systematic errors are usually inherent to measurements of a given type. They are difficult to identify and generally are apparent only when comparing measurements of one type with another. Correlations are also present and often important. Shape measurements with normalization at a single energy point is an example where uncertainties at one energy are correlated with uncertainties at another energy. Normalization of a measurement to a given standard will cause the quoted uncertainties of a measurement to be correlated with the

uncertainties in the measurement standard. The correlated uncertainties are important because the resultant uncertainty can be significantly different from the case where the uncertainties are considered to be statistically independent.

Many quoted errors are unreliable as evidenced by the fact that often the error bars from different measurements of the same quantity don't overlap. This means that the errors were incorrectly calculated or that all possible errors were not included. The evaluator must be prepared to reassign errors before determining a weighted result. When the evaluator has made his recommendation for nuclear data, he must also take responsibility for assigning the final covariances.

3.0 Evaluation Difficulties

3.1 Data Required

The types of data needed are determined by the application. Generally the objective is to solve for the transport of radiation through matter. This involves a vector characterization of the scattering, disappearance and appearance of particles. Mathematically, this is expressed by the Boltzmann equation where the terms have their usual meaning

$$\frac{dn}{dt}(R, \Omega, E) = \frac{1}{v} \frac{d\phi}{dt}(R, \Omega, E) \quad (3.1.1)$$

$$= -\Omega \cdot \text{grad}\phi(R, \Omega, E) - \sum_t(E) \phi(R, \Omega, E) + \iint \sum_s(\Omega' \rightarrow \Omega, E' \rightarrow E) \phi(R, \Omega', E') d\Omega' dE' + \int \sum_f(E' \rightarrow E) \phi(R, \Omega', E') d\Omega' dE' \quad (3.1.2)$$

The application also determines what data are emphasized. With regard to the energy range of the data, an evaluation that is to be used for thermal reactor, fast reactor, and fusion reactor design will require data from very low energies up to 20 MeV. But an evaluation used only for thermal reactor design needs accurate data in the energy range below a few MeV. For criticality calculations the scattering, absorption, fission, and inelastic scattering cross sections are necessary to determine neutron balance. For radiation damage calculations, gas production cross sections such as those arising from the (n,p) or the (n,alpha) reactions are important. Neutron capture cross sections are important for activation analysis. For safeguards the delayed neutron yields and spectra are important to determine the signature for fissionable materials. For reactor emergency core cooling calculations the decay properties of radioactive nuclei are needed. For shielding calculations, total cross sections and the angular distribution of secondary particles are especially needed.

3.2 Gaps in Information

Sometimes the data needed are not directly observed or measured but must be derived. In other cases the measurements are incomplete and must be supplemented by theory. In Figure 3.1, the measurement and the analysis of elastic and inelastic neutron scattering from U-238 is shown. The energy resolution of the experiment is not capable of separating elastic scattering from inelastic scattering to the low-lying level at 44 keV. But an optical model calculation of the total scattering cross section in agreement with the measurements can provide the needed data for partial cross sections. There are also blind spots in the data caused by the difficulty in finding adequate neutron sources at certain energies. Examples of this are shown in Figures 3.2 and 3.3 where no data are given in the energy range 8-12 MeV for the krypton total cross section or the Ni-58 (n,2n) cross section, respectively. A frequent

measurement difficulty is low signal strength, such as in the case of the measurement of the U-235 fission spectrum at very low and very high energies where the data are not of sufficient statistical accuracy to precisely define the shape at very low or high energies.

The evaluator does not have an easy task. He is expected to provide a complete information base from incomplete and uncertain information. He is presented with differential and integral data without a sharp definition where one ends and the other begins. Both types are in the realm of observables that the evaluator must consider. Evaluated nuclear data and calculations using the data must be consistent with the best differential and integral data.

4.0 Nuclear Systematics

About 2000 nuclides are known. In detail cross sections are often complicated, yet many cross sections for a wide range of nuclides can be fit by models using relatively few parameters. Therefore, a simple unified understanding of nuclear reactions may yet be found. This section reviews some systematics of nuclear reactions that have been observed.

At low energies, the scattering cross section shows resonances. Between them it is flat and has the value that is given by potential scattering: $4\pi R^2$. Figure 4.1 shows that the nuclear radius required for optical model analysis as a function of mass number scatter around the curve $R=1.35A^{1/3}$ Fermis. At higher energies, the radius is very predictable: e.g., the experimental nonelastic cross section versus mass number are well approximated by the line $(1.2A^{1/3} + 2.1)$ Fermis (Fig. 4.2). For the description of resonances multilevel and single-level Breit-Wigner formulas and other approximations are used, as is shown in Figure 4.3 for the total and the capture cross section of Ti-48. The envelope containing the resonant cross sections is determined by the neutron wavelength. The systematic behavior of s- and p-wave neutron strength functions

and total average s- and p-wave neutron radiation width versus A can be seen from Figure 4.4. The structure in the shape of these curves are associated with the closing of shells. Figure 4.5 shows experimental total, elastic, nonelastic, $(n,2n)$, (n,p) , and (n,α) cross sections for Co-59 as a function of incident neutron energy. As this element is monoisotopic and easy to handle as a sample and, moreover, has radioactive residual nuclei resulting from particle emission, the measurement record is more complete than it is for other nuclides. It can be seen that in the MeV region, the elastic cross section approximately equals the nonelastic cross section, which is generally true. In this case the charged particle cross sections are smaller than the $(n,2n)$ cross section, the (n,α) cross section being less than that for (n,p) .

In general, whenever a new channel opens, it rides on the coat tails and competes with the channels already open, and the envelope of the excitation function curves, i.e., the nonelastic cross section is almost constant. The sums of channels having the same origin, such as all $(n,n'x)$ cross sections, will be part of the same envelope.

If one considers $(n,2n)$ cross sections for various nuclei as a function of neutron energy (Fig. 4.6), one observes that the displacement of the excitation function curves is small (threshold between 12 and 14 MeV) and that the cross section increases as one goes from light to heavy nuclei. The (n,p) cross section for various nuclei (Fig. 4.7), on the other hand, decreases as one goes from light to heavy nuclei as the Coulomb barrier increases. The same tendency can be observed with (n,α) excitation functions (Fig. 4.8) where also the displacement of the curves for different masses increases as there is a larger increase in the Coulomb barrier. These general trends are even more impressively demonstrated by a plot (Fig. 4.9) of the ratio of peak charged particle cross sections to nonelastic cross sections at 14 MeV versus the

asymmetry parameter $\frac{N-Z}{N+Z}$. An exponential decrease with increasing $\frac{N-Z}{N+Z}$ is observed. The charged particle cross section peaks are shifted to higher energy with increasing asymmetry parameter as can be seen from Figure 4.10 so that 14 MeV cross sections can correspond to the rise, peak, or tail of the cross-section excitation curve.

With increasing Z and A (increasing asymmetry parameter) the magnitude of the $(n,2n)$ cross section increases as neutron emission becomes more probable than charged particle emission because of the large proportion of neutrons in the nucleus and the Coulomb barrier for charged particles. The charged particle emission becomes very small and the $(n,2n)$ cross section approaches the nonelastic cross section. For heavy nuclides elastic and inelastic scattering $(n,2n)$, and fission are the principal cross sections.

In Figure 4.11, the same experimental data for Co-59 as in Figure 4.5 are displayed, but now they are compared with nuclear model calculations. By means of this comparison, a decision between two strongly differing sets of data for the (n,p) cross section (having the same symbol) can be made as well as predictions for cross sections for which no experimental data exist.

Figure 4.12 displays diffraction of monochromatic light through a circular aperture; an analogy by which the optical model has been developed to describe the diffraction of neutrons. In diffraction through an aperture, if the aperture size is increased while the frequency is kept constant the pattern begins to show more and more fringes. In an analogous manner, the number of diffraction peaks of the differential elastic cross section for 14 MeV neutrons increases with increasing mass number of the target nucleus, as is demonstrated in Figure 4.13. On the other hand, the same effect can be achieved by keeping the size of the aperture or mass of the target nucleus constant and decreasing the wavelength of light which corresponds to increasing the neutron energy.

This is shown for the case of Pb-208 in Figure 4.14. For the angular distribution of elastically scattered neutrons, an approximation based on Fraunhofer diffraction is displayed in Figure 4.15. It is a "universal" curve and is relatively accurate for predicting cross sections and location of diffraction peaks at forward angles. For the extrapolation of differential elastic cross sections to small angles, Wick's limit may be helpful. It is derived from the optical theorem and says

$$\frac{d\sigma_e(0^\circ)}{d\Omega} \geq \frac{\sigma_e^2}{4\lambda^2} \quad (4.1.1)$$

where σ_e is the total elastic cross section. How Wick's limit agrees with experimental differential elastic cross sections is displayed in Figure 4.16 for the case of 14 MeV neutron scattering from natural chromium.

5.0 Evaluation Objectives

5.1 Consistency with Best Information Available

Experimental differential data is the starting point of an evaluation. The evaluation should be tested for consistency with good integral data experiments. Sometimes integral data can be used to support a particular differential data experiment from among a discrepant set of experiments. Each source of information should be carefully regarded. If necessary, one must rely on nuclear systematics or nuclear theory must be used to fill gaps in experimental data. The evaluated data should not be biased toward a particular application.

5.2 Good Documentation

Good documentation can take even longer than the evaluation procedure. The evaluator should aim at producing a document he would like to read. Each step of the evaluation procedure should be defined, e.g., the assignment of uncertainties for weighting data, normalization of cross sections, etc. The

documentation should be useful not only for physicists understanding the basis for data selection and programmers manipulating the data, but also for reactor licensing procedures.

5.3 Complete Data Base

The data base must contain all nuclear reaction data over the entire energy range expected by the cross section processing codes that are used. The detail to which it must be provided is, of course, determined by the applications considered.

5.4 Covariances

The data covariances, or degree of confidence that can be placed in the recommended data, should be clearly stated in the documentation and/or computerized file.

5.5 Convenient To Use

In order for evaluated data to be convenient to use, elaborate representations of data should not be employed. Also, many processing alternatives, e.g., interpolation schemes, should not be used or changed frequently unless necessary.

6.0 Evaluation Procedures

6.1 Adopt Measurement Standards

The ENDF/B (see Section 11) standards can be adopted. Frequently used standards are the hydrogen total and differential scattering, He-3(n,p), Li-6(n,alpha), B-10(n,alpha), C-12 differential elastic scattering, An-197(n,gamma), and U-235(n,f) cross sections. When reviewing measurements, normalization to the same standards should be performed before comparing data.

6.2 Critically Evaluate Each Piece of Promising Data and Recommend Data

Disqualification of data because of its age is not justified unless the method or apparatus used has become inferior to newer ones. Data must be compared on a consistent basis. Data should be evaluated using objective criteria with any personal bias eliminated. The nominal values given by the experiments should be averaged using the given uncertainty as weighting. If no consistency is achieved it may be necessary to revise the quoted errors. In the case of energy dependent data, one set should be chosen as a reference and the ratios of the others to the reference set should be plotted to determine which measurements have similar shapes.

The consistency between internal error

$$(\Delta\sigma)^2 = \frac{1}{N} \sum_{n=1}^N (\Delta\sigma_n)^2 \quad (6.2.1)$$

and external error

$$(\Delta\sigma)^2 = \frac{1}{N-1} \sum_{n=1}^N (\sigma_n - \bar{\sigma})^2 \quad (6.2.2)$$

should be checked.

When situations do not follow standard procedures the evaluator must use his best judgement (common sense).

6.3 Fill Gaps

Gaps can be filled with the aid of nuclear models, nuclear systematics or free hand, if necessary.

6.4 Test Data

The data can be tested by use in calculations of integral benchmark experiments, calculation of resonance integrals, etc. Observe if the disagreement between calculations and experiment can be correlated with particular cross sections or other features.

6.6 Review and Revise, If Necessary

Iteration of the whole procedure may be useful, especially when the calculation of integral benchmark experiments lead to disagreements between calculations and experiment. The choice of experimental differential data and assignment of errors should be reviewed. Finally, if the evaluation could be helped by better measurements, their specifications should be inserted in the international IAEA measurement request list WRENDA.

6.7 Evaluation Tools and Examples

The assembly of relevant experimental data is aided by performing a literature survey. For neutron data the IAEA index to neutron bibliography, CINDA (Fig. 6.1) is useful. By blocking of references to the same experiment, time can be saved because the first reference supercedes the others. One of the sources of information referred to in CINDA is the EXFOR library (Fig. 6.2). This is formatted information containing experimental data with a brief summary of the documentation of the article from which the data were taken. These data are available from the world data centers each serving a particular geographical area. However, assembling the pertinent data is only the first step.

A common problem is energy scales that do not agree for different measurements. Figure 6.3 shows the ENDF/B-IV and V curves for the U-235(n,f) cross sections together with experimental data. There is a knee in the curve at about 6 MeV, and the energy at which it appears strongly differs in the various experiments (Fig. 6.4).

Ratio measurements are generally more accurate than absolute magnitude measurements. Figure 6.5 displays measured fission cross section ratios and the relatively small scatter in the experimental data.

Nuclear model code calculations are useful for filling gaps in the data base but only after extensive comparisons with reference data and other codes. It is interesting to see that various statistical model codes did not a priori give the same results for the Co-59 (n,d) cross section when tested in a code intercomparison in Spring 1977. As a result of additional code intercomparison, the calculated results 6 months later agreed much better with each other as well as with the experimental data. A number of optical model codes are in use that calculate elastic scattering and inelastic scattering from discrete levels.

Examples of widely used codes are the following:

ABACUS II	E. Auerbach, BNL-6592 (1969). Spherical Potential
JUPITOR	T. Tamura, ORNL-4152 (1967). Deformed potential

When it is necessary to calculate nuclear reaction cross sections a number of statistical model codes are available. Examples of widely used codes are the following:

GNASH	P.G. Young, E.D. Arthur, LA-6947, includes precompound.
HAUSER V	F. Mann, HEDL-TME 78-83
THRES2	S. Pearlstein, JNE <u>27</u> (1973), limited applicability

The above codes and others are available from Nuclear Energy Agency Data Bank or the International Atomic Energy Agency Nuclear Data Section (see References).

7.0 Combining Differential and Integral Data

The uncertainties assigned to evaluated differential data should be included in the calculation of integral quantities in order to determine the confidence level that can be placed in the calculated results, e.g., effective multiplication factor, reaction rates, etc. However, the measured integral parameters may also have experimental uncertainties of their own. As stated earlier, differential and integral data are observables that contain important

information for use in predicting the behavior of nuclear systems. Revisions in integral as well as differential data may be required in order to achieve consistency. However, changes must be realistic and not violate in any way the physics of the measurements. Only integral benchmark experiments that are highly precise, free from systematic errors, and thoroughly documented should be used to test the consistency of evaluated nuclear data.

7.1 Least Square Fitting

Consider a set of parameters T_i , $i=1,2,\dots,I$, which can correspond to differential data, e.g., group cross sections, and a set of results R_n , $n=1,2,\dots,N$ which can correspond to integral data, e.g., criticality experiments.

In the method of least square fitting, it is required that the quantity

$$\psi^2 = \sum_{n=1}^N W_n (R_c - R_E)_n^2 \quad (7.1.1)$$

is minimum. Here the subscripts C and E refer to calculated and experimental integral data. The quantity W is the weight assigned to each term and is usually taken as inversely proportional to the square of the assigned uncertainty which in the case of uncorrelated uncertainties is

$$W_n = \frac{1}{(\Delta R_n)^2} \quad (7.1.2)$$

Minimizing ψ^2 requires that

$$\frac{d\psi^2}{dT_k} = 2 \sum_{n=1}^N W_n (R_c - R_E)_n \left(\frac{dR_c}{dT_k} \right)_n \quad (7.1.3)$$

If we assume that a change in the differential quantity affects the calculated integral parameters in a linear way then we can expand R as

$$R_c = R_c(T_0) + \sum_{k=1}^I \frac{dR_c(T_0)}{dT_k} dT_k \quad (7.1.4)$$

where T_0 refers to an initial choice of parameters. Substitution in the previous equation produces the normal equations

$$\sum_{n=1}^N W_n \sum_{j=1}^I \frac{dR}{dT_j} \frac{dR}{dT_k} dT_j = \sum_{n=1}^N W_n \left[R_c(T_0) - R_E \right]_n \frac{dR_c}{dT_k} \quad (7.1.5)$$

These are equivalent to a matrix equation of the form

$$A \cdot dT = R \quad (7.1.6)$$

where A is an MxM square matrix and dT and R are Mx1 column vectors.

$$dT = A^{-1} \cdot R \quad (7.1.7)$$

In cases where R is a linear equation in T, only one iteration is necessary to obtain the final answer. But in the nonlinear case, the initial parameter guesses are altered by dT and used as guesses for the next iteration until a convergence T is achieved. The covariances in the differential quantities are calculated by the equation

$$\langle dT_j \cdot dT_k \rangle = \left[A^{-1} \right]_{jk} \frac{\psi^2}{N-1} \quad (7.1.8)$$

The covariances calculated for T apply to calculations of the integral quantities R and may be different from the covariances assigned to T before least squares fitting. Nonzero off-diagonal elements ($j \neq k$) of the covariance matrix indicate correlations exist between the parameter uncertainties. The correlation coefficient ≤ 1 is

$$e_{jk} = \frac{\langle dT_j \cdot dT_k \rangle}{dT_j \cdot dT_k} \quad (7.1.9)$$

where dT_j is the square root of the j row and j column diagonal element of the covariance matrix.

The assignment of uncertainties to integral measurements is very important.

To facilitate the discussion matrix symbols are used that are consistent with the primary reference (Dragt) used for this Section. It is useful to define the "sensitivity coefficients" to account for calculated integral quantities on differential data in the following way

$$G_{jk} = \frac{dR_j}{dT_k} \quad (7.1.10)$$

where the subscripts j and k stand for particular integral data and differential data, respectively. The elements of G form the sensitivity matrix of order $(N \times I)$.

The accuracy of the differential data T can be expressed in a covariance matrix M of order $(I \times I)$ as

$$M = \left[\left\langle dT_j \cdot dT_k \right\rangle \right] \quad (7.1.11)$$

If $\langle \Delta R \cdot \Delta R \rangle$ is the covariance of the two integral parameters, a covariance matrix V of order $(N \times N)$ similar to M may be defined to express the uncertainties in the integral data.

The uncertainty in the integral parameter R will depend upon the covariance of the differential parameter in the following way

$$dR^2 = \sum_{j=1}^I \sum_{k=1}^I \frac{dR}{dT_j} \frac{dR}{dT_k} \left\langle dT_j \cdot dT_k \right\rangle \quad (7.1.12)$$

Then Eq. 7.1.5 can be written as

$$\left[\hat{G} \cdot V^{-1} \hat{G} \right] \cdot dT = \hat{G} \cdot R \quad (7.1.13)$$

or

$$dT = \left[\hat{G} \cdot V^{-1} \hat{G} \right]^{-1} \hat{G} \cdot R \quad (7.1.14)$$

The covariance matrix M' for differential data, using Eq. 7.1.13 can be written as

$$M' = \left[\hat{G} \cdot V^{-1} \hat{G} \right]^{-1} \frac{\psi^2}{N-1} \quad (7.1.15)$$

which gives an estimate of the uncertainty in the differential data. The uncertainty in the integral data would then be given by

$$dR^2 = G \cdot M' \cdot G \quad (7.1.16)$$

7.2 Dragt's Method

Dragt worked out a similar set of equations by assuming that all variables are normally distributed and that a linear relationship exists between the variations in the integral and differential quantities. If R' are the new calculated quantities and T' are the adjusted differential parameters, then this assumption implies

$$R'_c = R_c + G \cdot (T' - T) \quad (7.2.1)$$

The vector T' and the final covariance matrix M' for differential data then are the solutions of the equations:

$$\left(M^{-1} + \hat{G} \cdot V^{-1} \hat{G} \right) \cdot (T' - T) = \hat{G} \cdot V^{-1} \cdot (R'_E - R_c) \quad (7.2.2)$$

$$M' = \left(M^{-1} + \hat{G} \cdot V^{-1} \hat{G} \right)^{-1} \quad (7.2.3)$$

The solution of these equations can be simplified by substituting

$$N = G \cdot M \cdot \hat{G} \quad (7.2.4)$$

which is the covariance matrix of the calculated integral parameters R . The Equations 7.2.2 and 7.2.3 may then be written as

$$(T' - T) = M \cdot \hat{G} \cdot (N+V)^{-1} (R_E - R_C) \quad (7.2.5)$$

$$M' = M - M \cdot \hat{G} \cdot (N+V)^{-1} G \cdot M \quad (7.2.6)$$

The inversion of a large matrix of order $(M \times M)$ in Equations 7.2.2 and 7.2.3 has been reduced to the inversion of a small matrix of order $(N \times N)$ in Equations 7.2.5 and 7.2.6. As the number of integral data N is much less than the number of differential parameters M this simplification is quite important from a computational point of view.

If all the integral experiments are uncorrelated, i.e., V is a diagonal matrix, the adjustment calculation can be performed without any matrix inversion at all. The fitted parameters should lie within the experimental limits of uncertainties. The adjustment of a particular parameter does not necessarily mean that the adjusted value of that parameter is more accurate. It may only mean that the whole set of adjusted parameters can predict the behavior of the system in a more consistent fashion and can be considered a parameterization of the system.

8.0 Probability Table Method

Tabulated cross sections versus energy in the resolved resonance region can require thousands of data points to describe resonance shapes. This requires large computer memory and a great deal of computer processing time. Therefore

it is desirable to describe resonance region cross sections without having large tables in the data library. In ENDF the resonance parameters are given in the resolved resonance region. For one resonance one card contains all the necessary information. The cross section at a particular energy point can be calculated using suitable formulas. The U-235 fission cross section in the unresolved resonance region is shown in Fig. 8.1.

In the unresolved resonance region only the statistical distribution of resonance parameters is given. Sometimes it is possible to extrapolate the resolved resonance region parameters into the unresolved resonance region, provided they give the same average cross sections as observed in measurements.

In a reactor calculation detailed cross-sectional data are required in order to calculate energy self-shielding and Doppler broadening effects. Approximate methods are used. One such method consists in generating a ladder of resonance parameters using statistical models. More than one such ladder must be generated since it is not known what specific ladder corresponds to the actual data.

Another method is the use of the Probability Table Method, which consists in compiling a probability distribution of cross-sections against the cross section. For example in Fig. 8.2, the probability distribution of the fission cross section of U-235 has been plotted against the U-235 fission cross section in the range of 80-200 eV. An alternate approach could be to plot the integrated probability against the cross section rather than the probability. By choosing a random number between 0 and 1 and entering the probability table an estimate of the cross section can be made. The probability tables can be plotted for each of the ladder of resonances. The same can be done for other energy ranges, temperature, etc. The main advantage of this method is that the computer has to handle fewer and smaller tables.

This method appears to be successful even if resonance cross sections are highly energy correlated. A particular example would be the U-238 resonance region from 200 to 500 eV which shows some strong auto-correlations. However, calculations of capture rates using detailed cross section profiles are not statistically distinguishable from calculations using the Probability Table Method.

9.0 Multigroup Constants

9.1 Multigroup Solutions

The solution of Boltzmann's equation requires a description of the nuclear cross sections, material compositions, and spatial configuration. Cross sections describe nuclear reaction probabilities as a function of energy and direction of the incident particle. Although solutions that are continuous in energy and angle are possible using statistical sampling Monte Carlo methods, results from averaging over discrete ranges of energy and angle are commonly used. Cross sections averaged over discrete ranges are called multigroup constants.

The derivation of multigroup theory from Boltzmann's equation can be found in standard texts. Basically, the rate of change in particle density within each group is determined by the net difference between the rate of particle production and loss. A set of simultaneous equations results from writing multigroup balances at each spatial node of the configuration. The equations are solved by matrix inversion or relaxation methods.

9.2 Bondarenko Method

Ordinarily multigroup cross sections are problem dependent since multigroup cross sections are averaged over particle fluxes which in turn are determined by material composition and geometry. To avoid numerous recalculations of average cross sections, the Bondarenko method calculates energy averaged cross sections

for an isotope over a range of cross sections representing that of the remaining isotopes in a mixture. A table of these values are stored and cross sections for individual cases are obtained by interpolation.

10.0 Benchmarks

10.1 Benchmark Analysis

A benchmark experiment is an integral experiment (see 2.2) that is carefully performed and documented sufficiently for detailed calculation. Cross section data, material compositions, and the geometry are used as input to reactor physics codes to calculate the parameters obtained by the integral experiment. If the methods of calculation are accurate and the systematic uncertainties of the integral experiment are negligible, the comparison of calculation with experiment can be used to determine the degree of confidence that can be placed in current nuclear technology for the applications considered.

A list of documented benchmark experiments is maintained by the Cross Section Evaluation Working Group (see References).

10.2 Reactor Physics Codes

Several computer codes used for cross section preparation and solution of reactor physics are in wide use and distributed by data centers. Unless a user has special energy group or angular approximation requirements, the prepared group libraries that are available can save time and money. A list of representative cross section processing and reactor physics codes is given in Appendix I. Generally, these codes are available from the IAEA or with their assistance from cooperating centers.

10.3 Benchmark Specifications and Results

Experimental integral benchmarks for fast reactor, thermal reactor, shielding, and dosimetry applications have been compiled by the U.S. Cross Section Evaluation Working Group. The specifications are reported in ENDF-202 and the comparison between calculation and experiment are described in ENDF-230. For comparisons between calculation and experiment for benchmarks and other integral data, refer to reports appearing in Nuclear Science and Engineering and Transactions of the American Nuclear Society.

11.0 Evaluated Nuclear Data Systems

11.1 Formats and Procedures

Several widely used formats for evaluated data include UKNDL (United Kingdom), KEDAK (Karlsruhe), SOKRATOR (USSR), and ENDF (Brookhaven).

A well designed evaluated data system includes codes for format checking and standardization, physics checking, data correction, generation of infinite dilute cross sections, cross section integration over energy intervals, data plotting, data retrieval, data merging, and data listing. The best documented system is ENDF. Documents of general and specific interest and also reference guidelines for ENDF are given in Appendix II.

11.2 U.S. Evaluation System

In 1966, the U.S. formed the Cross Section Evaluation Working Group (CSEWG) consisting of representatives from over 20 federal, industrial, and university laboratories. The CSEWG structure consists of Evaluations, Data Testing and Applications, and Methods and Formats Committees reporting to an Executive Committee.

The Executive Committee consists of the CSEWG chairman, funding agency representatives, committee chairmen, three additional members appointed for limited terms by the CSEWG chairman in consultation with sponsors, and one member-at-large elected by CSEWG. The Executive Committee sets policy and gives

final approval to recommendations by the other Committees.

The Evaluations committee would 1) recommend evaluation responsibilities, 2) schedule and oversee completion of individual evaluations, 3) selection of reviewers, 4) review physics contents, 5) recommend suitability of evaluations 6) maintain a discrepancy list, 7) review requests for nuclear data, 8) recommend new nuclear data measurements, and 9) organize seminars, workshops, etc., to solve specific evaluation problems.

The Data Testing and Applications Committee would 1) recommend data testing responsibilities, 2) schedule and oversee completion of individual data testing, 3) review integral data experiments, 4) analyze integral data calculations, 5) select integral data benchmarks, maintain an integral data discrepancy list, 7) recommend suitability of evaluations, 8) collect needs of applied users, 9) recommend priorities for measurements based on discrepancies between calculation and integral experiments, and 10) organize seminars, workshops, etc., to solve specific data testing problems.

The Methods and Formats Committee would 1) develop ENDF formats for data and covariances, 2) develop ENDF utility codes, 3) develop ENDF processing codes, 4) recommend standard interfaces, 5) investigate analysis methods, and 6) organize seminars, workshops, etc., to solve specific methods and formats problems.

12.0 Acknowledgements

Although not individually identified, some material used in illustrating cross section analysis were provided by Mulki Ehat, Mundrathi Divadeenam, Said Mughabghab, and Gus Prince of Brookhaven National Laboratory. The list of example cross section processing and reactor physics codes was prepared by Phillip Rose of Brookhaven National Laboratory. Their help is very much appreciated.

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Finally, I wish to acknowledge the keen attention given to the Triest Winter College on Nuclear Physics and Reactors by the course directors, M.K. Mehta and J.J. Schmidt.

13.0 References

13.1 Neutron Physics

1. D.J. Hughes, *Pile Neutron Research*, Addison-Wesley, Reading, Mass. 1953.
2. K.H. Beckurts and K. Wirtz, *Neutron Physics*-Springer Verlag, 1964.
3. S.F. Mughabghab, et al. *Neutron Cross Sections, Vol. 1, Resonance Parameters, Third Edition (Introduction)*, BNL-325 (1978).

13.2 Specific Neutron Data

(n,2n)

1. M. Caner, M. Segev, and S. Yiftah, *Nucl. Sci. Eng.* 59 (1976) 395.

fission ratios

1. J.W. Behrens, R.S. Newbury, and J.W. Morgana, *Nucl. Sci. Eng.* 66 (1978) 433.

13.3 Nuclear Systematics

1. S. Pearlstein, *Nucl. Sci. Eng.* 49 (1972) 162.
2. S. Pearlstein, *J. Nucl. Energy* 27 (1973) 81.
3. S. Pearlstein, *Washington Conference, March 1975*, p. 331.
4. S. Pearlstein, *Nucl. Sci. Eng.* 68 (1978) 55.
5. I. Angeli and J. Csikai, *Nucl. Phys.* A158 (1970) 389.

13.4 Covariances

1. F.G. Perey, Neutron Physics and Nuclear Data, Harwell, September 1976, p. 104.
2. C.Y. Fu, D.M. Hetrick, and F.G. Perey, Knoxville Conference, October 1979.

13.5 Data Adjustment

1. P. Greebler and B.A. Hutchins, Physics of Fast and Intermediate Reactors, IAEA, Vienna 1962, Vol. 3, p. 121.
2. J.B. Dragt, J.W.M. Dekker, H. Gruppelaar, and A.J. Hansen, Nucl. Sci. Eng. 62 (1977) 117.
3. W.P. Peonitz, Neutron Standards and Flux Normalization, AEC Symposium Series 23, 1970, p. 331.
4. Y. Chao, Nucl. Sci. Eng. 72 (1979) 1.

13.6 Group Constants

1. I.I. Bondarenko (Editor), Group Constants for Nuclear Reactor Calculations, Consultants Bureau, N.Y. (1964).

13.7 Probability Table Method

1. L.B. Levitt, Nucl. Sci. Eng. 49 (1972) 450.
2. J.M. Otter, R.C. Lewis, and L.B. Levitt, U3R, "A Code to Calculate Unresolved Resonance Cross Section Probability Tables, " AI-AEC-13024, Atomics International (1972).
3. S. Pearlstein, Nucl. Sci. Eng. 58 (1975) 354.

4. S. Pearlstein, Nucl. Sci. Eng. 68 (1978) 10.

13.8 Bibliography

1. CINDA, an Index to the Literature on Microscopic Neutron Data, International Atomic Energy Agency, Vienna.
2. Burrows, T.W., et al. The Bibliography of Integral Charged Particle Nuclear Data, Third Edition. BNL-NCS-50640, April 1979.
3. Burrows, T.W. and Holden, N.E. A Source List of Nuclear Data Bibliography, Compilations, and Evaluations. BNL-NCS-50702, Second Edition, October 1978.
4. S. Pearlstein, Adv. Nucl. Science and Tech. 8 (1975) 115.
5. The Evaluation of Neutron Nuclear Data, Proceedings of a Panel, Vienna, 1971, IAEA-153.

13.9 Nuclear Reaction Data Centers

1. National Nuclear Data Center, Brookhaven National Laboratory, Bldg. 197D, Upton, N.Y. 11973
2. European Nuclear Energy Agency Data Bank, Neutron Data Compilation Center (Nuclear Energy Agency) P.O. No. 9, 91190 Gif-sur-Yvette, France.
3. Nuclear Data Section (International Atomic Energy Agency) Karntnerring 11/13, A-1010 Vienna, Austria.
4. Center Po Jadernum Dannym (USSR), Institute of Physics and Energy, Obninsk, USSR.
5. Karlsruhe Charged Particle Group, Kernforschungszentrum, D-7500 Karlsruhe, Germany.
6. Centre Po Atomn. i Jadernum Dannym, Kurchatov Inst. Moscow, USSR.

APPENDIX I

EXAMPLE CROSS-SECTION PROCESSING AND REACTOR PHYSICS CODES

February 1980

Spectrum Calculations by Contents, etc.

FORM-Thermal (Fast x-sects) Fortran version of MUFT systems
HAMMER-Thermal lattice contains THERMOS
LASER-Thermal lattice EPRI-CELL
MC**2-LMFBR (ANL) calculated spectrum (P_1, B_1 and consistent P_1+B_1) used for weighting and constructing gp x-sects
NJOY(MINX)-General but geared for LMFBR (LASL)
RAHAB-Thermal lattice (SRL) good for D_2O systems
SUPERTOG-Fast group constants. General systems.
TEMPEST-(Thermal-x-sects) Thermal systems.
THERMOS-Thermal lattice Integral Transport Solution (In Hammer)
WIMS-Thermal lattice
IDX-Benchmark calculation (will process X-sections to ANISN format)
AMPX-General (ORNL)
GGTC-ENEL-Thermal multigroup x-sects
ETOG-Thermal analysis (Fast x-sects) Produces HAMMER Library etc.also ETOM (ETOE)
FLANGE-Thermal scattering matrices
LITHE-Thermal (Processes thermal x-sects for HAMMER uses FLANGE output)
SPHINK-Similar to IDX uses 'CCCC' format transport calculation
PUPX-Handles data for IDX code
ETOX-Fast cross sections similar to ETOG but geared for LMFBR systems

Static Design

ANISN-General ID transport ORNL
DOT-2+3D transport ORNL
DTF-Similar to ANISN
PERT-Perturbation code uses ANISN output etc.
VENTURE-3D system ORNL-LMFBR
TWOTRAN-2D Transport LASL
2DB(2DF)-2D diffusion LMFBR cores
PDQ-7(TRITON,SQUID)-2-3D diffusion Thermal cores
EXTERMWATOR-2D diffusion x-y R-z R- θ searches
CITATION-3D system ORNL-LMFBR
TRIDENT-2D Transport X-Y, R-Z general anistropic scattering
ONETRAN-General ID transport LASL

Dynamic Design

RAUM-ZEIT-(old)
SYNTH-3D-3D transient space-time synthesis
FRAP-T3-Light Water Reactor transient response
RECAP series-Dynamic response for light water systems
3OIL-1-Meltdown sequence
NATRANSIENT-Pressure transients in LMFBR's

Monte Carlo Codes (Static & Transient Design)

VIM-Fast and thermal analysis
RECAP-Thermal lattices
SAM-CE-Fast and thermal analysis--shielding applications FUSION
KENO-Criticality hazards

Shielding

SAM-CE-(see above) DOT (see before)
MORSE- γ Shielding Monte Carlo
ISOSHLD-3-General purpose shielding analysis
GAMLEG- γ cross sections
GAMSOURCE- γ ray source from neutron capture
QAD-Point kernel shielding calculations (LASL)
RADHEAT-Coupled n+ γ calculates transport and energy deposition
LAPHANO-Calculates γ source from n

Depletion & Fuel Management

ORIGEN-Fission product decay heat
REBUS-Fuel management
CINDER-Fission product decay heat
2DB-In depletion mode
ORSIM-Fuel management
HYACINTH-Heavy isotope inventory with depletion
RICE-CEGB-Actinide and fusion product inventory of irradiated fuel
LEOPARD-A spectrum-dependent nonspatial depletion code (thermal analysis)
DTF-BURN-Depletion using DTF code
NUCY-Depletion package can be adopted to ANISN, etc.

Fusion

MACK-IV-Nuclear response functions important to the neutronics analysis of nuclear and fusion systems. Mostly nuclear heating from kernal factors.
HETC-High energy nucleon-meson transport code package
See other packages for static design.

Some References

DTF	LA-3373, LA-3267, NAA-SR-10951
OGRE	ORNL-3805
QAD	LA-3573
ISOSHLD	BNWL-236, HW-83784
ANISN	K-1693 (ORNL)
DOT	ORNL-TM-4280
SAM-C	UNC-5157, MAGI-6701, EPRINP-1042
MORSE	ORNL-4585
2DB	BNWL
HETC	ORNL-4744

Some References cont.

TWØTRAN LA-4600, LA-4848, LA-4774, LA-4432, LA-4058
 ORIGEN ORNL-4628
 TRIPLET (2D Triangular mesh discrete ordinates) LA-5428-MS
 IDX BNWL-954
 ETOX BNWL-1002
 NJOY LA-7584-M
 MINK LA-6486-MS
 FLANGE DP-1278, ENDF-152
 ETØG WCAP-3845-1
 HAMMER DP-1064 (SRL)
 THERMOS BNL-5826
 MC² ANL-8144
 VENTURE ORNL
 CITATION ORNL

NEA-DATA BANK

<u>Library</u>	<u>Comments</u>	<u>Application</u>
5.1	LIB-IV "CCCC" Format	LMFBR
5.2	26 gp ARAMCO	General
5.3	Benjamin - SRL library	Actinide depletion
5.4	460 gp ENDF/B-IV Thermal	Thermal Reactors
5.5	SAND-II	Dosimetry
5.6	-	-
5.7	-	-
5.8	-	-
5.9	-	-
5.10	ENDL 175 gp library	Fusion
5.11	-	-

APPENDIX II

ENDF DOCUMENTS of GENERAL INTEREST

ENDF-102	BNL-NCS-50496	GARBER, D.	DATA FORMATS AND PROCEDURES FOR THE EVAL. NUCLEAR DATA FILE
ENDF-110	BNL-50300	OZER, O.	DESCRIPTION OF THE ENDF/B PROCSSING CODES AND RETRIEVAL SUBROUTINES
ENDF-201	BNL-17541	GARBER, D.	ENDF/B SUMMARY DOCUMENTATION
ENDF-202	BNL-19302	ALTER, H.	CROSS SECTION EVAL. WORKING GROUP BENCHMARK SPECIFICATIONS
ENDF-210	ANCR-1157	REICH, C.W.	RADIOACTIVE-NUCLIDE DECAY DATA FOR ENDF/B
ENDF-216	BNL-NCS-50446	MAGURNO, B.A.	ENDF/B-IV DOSIMETRY FILE
ENDF-223	LA-6116-MS	ENGLAND, T.R.	ENDF/B-IV FISSION-PRODUCT FILES: SUMMARY OF MAJOR NUCLIDE DATA
ENDF-225	BNL-NCS-50464	MAGURNO, B.A.	ENDF/B-IV CROSS SECTION MEASUREMENT STANDARDS
ENDF-230	BNL-NCS-21118	BOHN, E.	BENCHMARK TESTING OF ENDF/B-IV
ENDF-243	BNL-NCS-50545	ROSE, P.F.	ENDF/B FISSION PRODUCT DECAY DATA
ENDF-244	LA-6518-MS	HALE, G.M.	LIGHT ELEMENT STANDARD CROSS SECTIONS FOR ENDF/B-IV
ENDF-249	ORNL-TM-5938	PEREY, F.G.	DATA COVARIANCE FILES FOR ENDF/B-V
ENDF-265	BNL-NCS-24853	WEISBIN, C.R.	SENSITIVITY COEFFICIENT COMPILATION FOR CSEWG DATA TESTING BENCHMARKS

ENDF DOCUMENTS OF SPECIFIC TOPIC

ENDF-152	DP-1278	HONECK, H.C.	FLANGE II (VERSION 71-1) A CODE TO PROCESS THERMAL NEUTRON DATA FROM AN ENDF/B TAPE
ENDF-218	ORNL-TM-4847	WEISBIN, C.	CROSS SECTION AND METHOD UNCERTAINTIES: THE APPLICATION OF SENSITIVITY ANALYSIS TO STUDY THEIR RELATIONSHIP IN RADIATION TRANSPORT BENCHMARK PROBLEMS
ENDF-237	LA-6486-MS	WEISBIN, C.	MINX: A MULTIGROUP INTERPRETATION OF NUCLEAR CROSS SECTIONS FROM ENDF/B
ENDF-238	ANCR-1322	GRIMESEY, R.A.	ETOP 14: A FORTRAN CODE TO PROCESS ENDF/B DATA INTO THE 68-GROUP PHROG LIBRARY FORMAT
ENDF-239	ANL-8144	HENRYSON, H.	MC ² -2: A CODE TO CALCULATE FAST NEUTRON SPECTRA AND MULTIGROUP CROSS SECTIONS
ENDF-251	HEDL-TME-77-54	MANN, F.	HEDL EVALUATION OF ACTINIDE CROSS SECTION FOR ENDF/B-V
ENDF-266	TREE-1259	HARKER, Y.D.	FISSION PRODUCT AND REACTOR DOSIMETRY STUDIES AT COUPLED FAST REACTIVITY MEASUREMENTS FACILITY
ENDF-269	GA-8774	KOPPEL, J.U.	REFERENCE MANUAL FOR ENDF THERMAL NEUTRON SCATTERING DATA
ENDF-272	LA-7584-M	MACFARLANE, R.	THE NJOY NUCLEAR DATA PROCESSING SYSTEM: USER'S MANUAL

REFERENCE GUIDELINES FOR ENDF/B

Case I: Use of ENDF/B evaluations in a secondary manner, where many elements are used together, or other cases where NO CONCLUSIONS ARE DRAWN CONCERNING QUALITY OF EVALUATIONS. In this case we propose the following form for ENDF/B-V.

"ENDF/B Summary Documentation, BNL-NCS-17541 (ENDF-201), 3rd Edition (ENDF/B-V), edited by R. Kinsey, available from the National Nuclear Data Center, Brookhaven National Laboratory, Upton, N.Y. (July 1979)."

Case II: Use of ENDF/B evaluations in a direct manner, for example comparing measured results with evaluated results, or ANY CASE WHERE CONCLUSIONS ARE DRAWN ABOUT AN EVALUATION FOR A PARTICULAR MATERIAL. We propose, for ^{12}C from ENDF/B-V as an example:

"ENDF/B data file for ^{12}C (MAT 1306,MOD 1), evaluation by C.Y. Fu and F.G. Perey (ORNL), BNL-NCS-17541 (ENDF-201), 3rd Edition (ENDF/B-V), edited by R. Kinsey, available from the Brookhaven National Laboratory, Upton, N.Y. (July 1979)."

Case III: Use of ENDF/B evaluations to generate a multigroup library. In this case we propose that the report describing the library contain a table which includes the following information for each evaluation:

<u>Material</u>	<u>MAT,MOD</u>	<u>Authors</u>	<u>Institution</u>
-----------------	----------------	----------------	--------------------

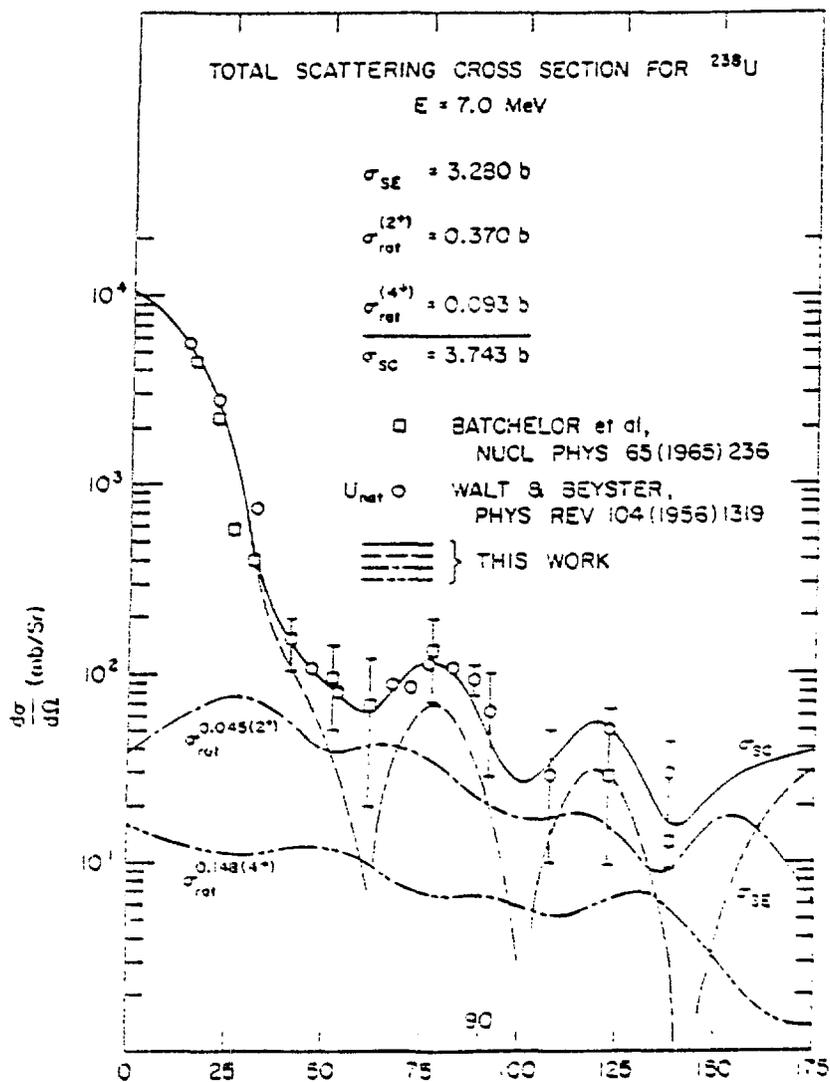
This table may contain in addition other useful information concerning the multigroup library. Finally, a general reference should be given of the type described in Case I.

As shown in Cases II and III, a correct reference would contain the material name, MAT number, author list and institution(s), along with a reference to the Summary Documentation. In addition, for ENDF/B-Version V, updates will be allowed to the evaluations prior to the release of ENDF/B-VI. Thus, references to ENDF/B-V evaluations should also contain the appropriate MOD number, which serves to define the current status of an evaluation. All of this information is readily available in File 1 of each evaluation. The only exception to the above cases would be where a published document, prepared by the authors of the evaluation, is available. This document should then be referenced directly.

Fig. 2.1

	N-2	N-1	N	N+1
Z	n, 3n	n, 2n	Original Nucleus n, n	n, γ
Z-1	n, nt n, tn	n, t n, nd n, dn	n, d n, np n, pn	n, p
Z-2	n, α n n, n α	n, α n, He ³ n n, n He ³	n, He ³ n, pd n, dp	

Fig. 3.1



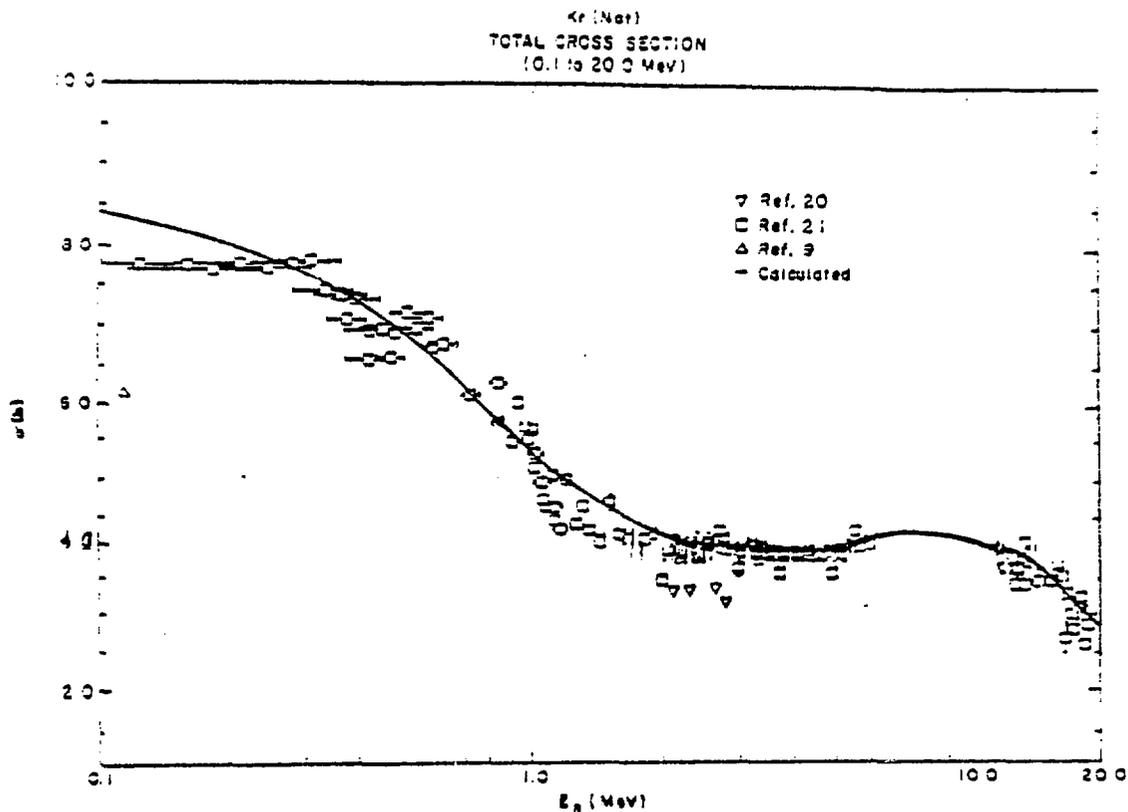


Fig. 3.2

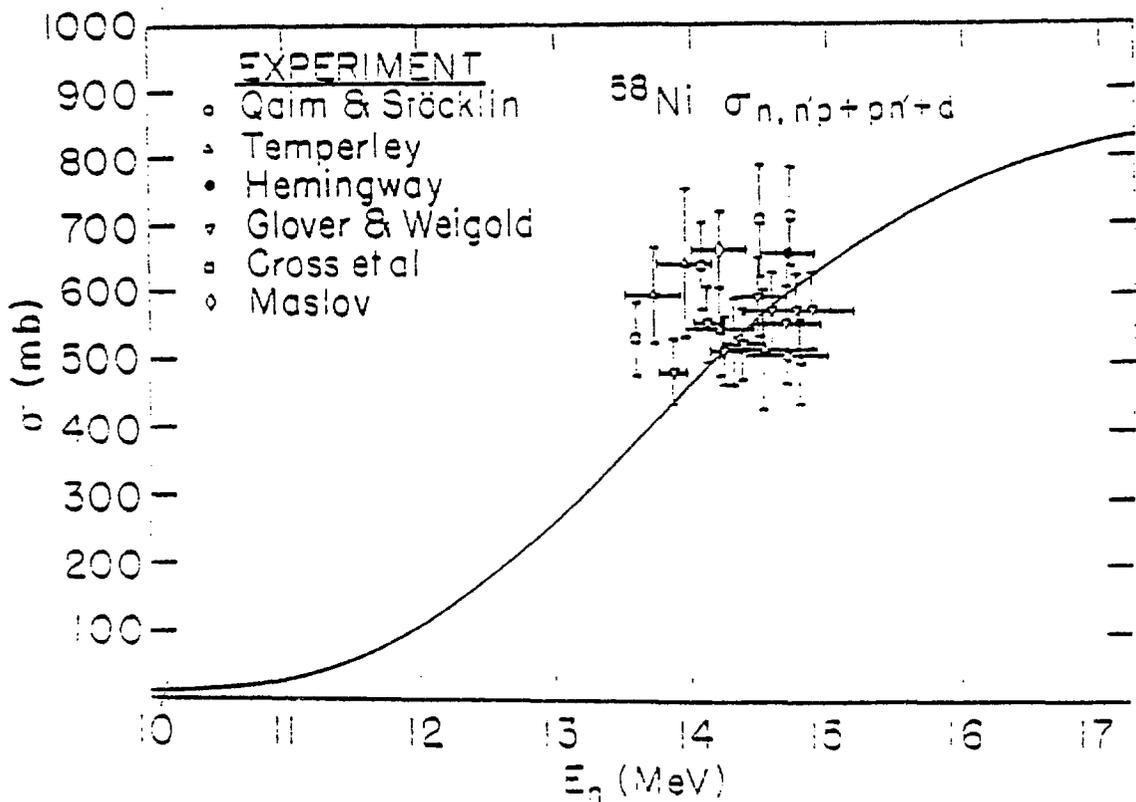
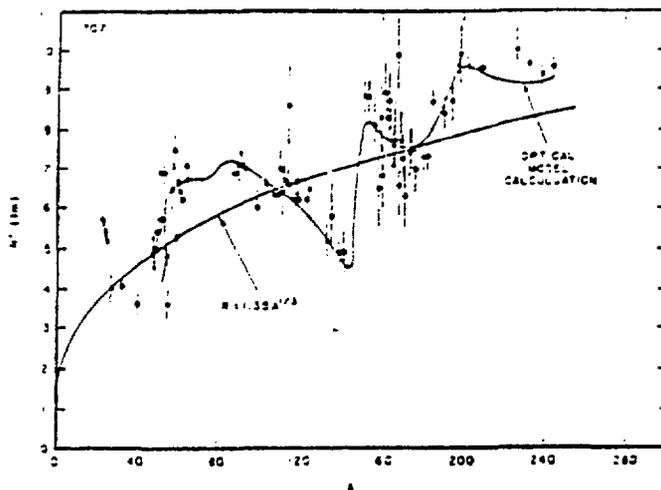


Fig. 3.3



Variation of R' with A . The solid curve is based on the vibrational rotational optical model with parameters $V_0 = 43.5$ MeV, $r_{10} = 1.35$ fm, $V_{10} = 8$ MeV and surface absorption $W_D = 5.4$ MeV.

Fig. 4.1

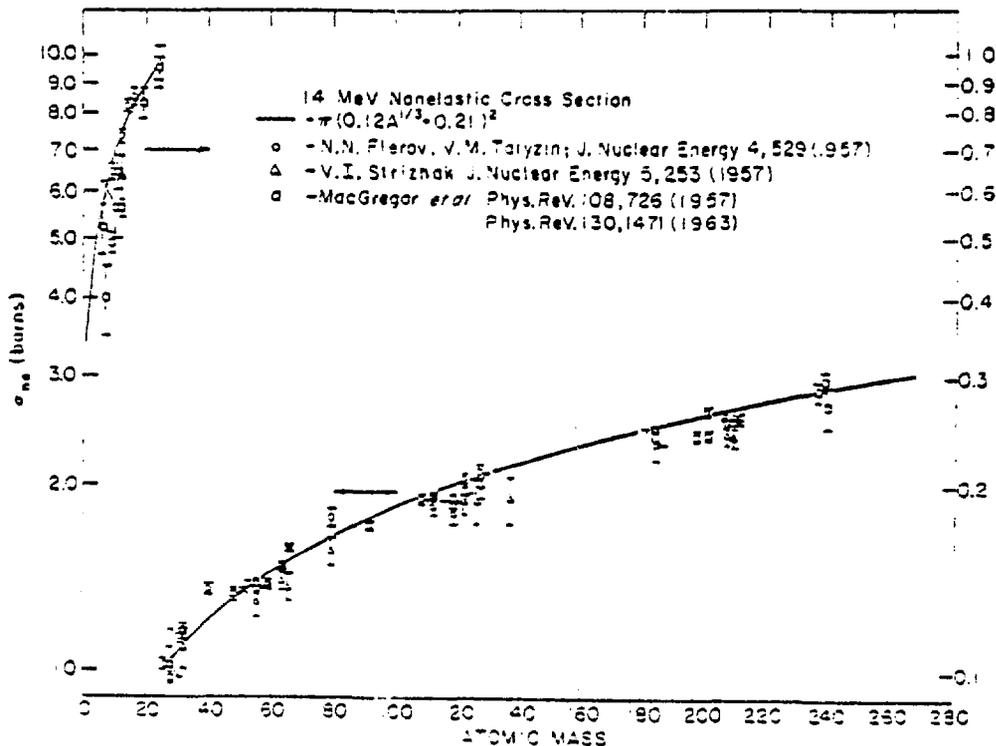


Fig. 4.2

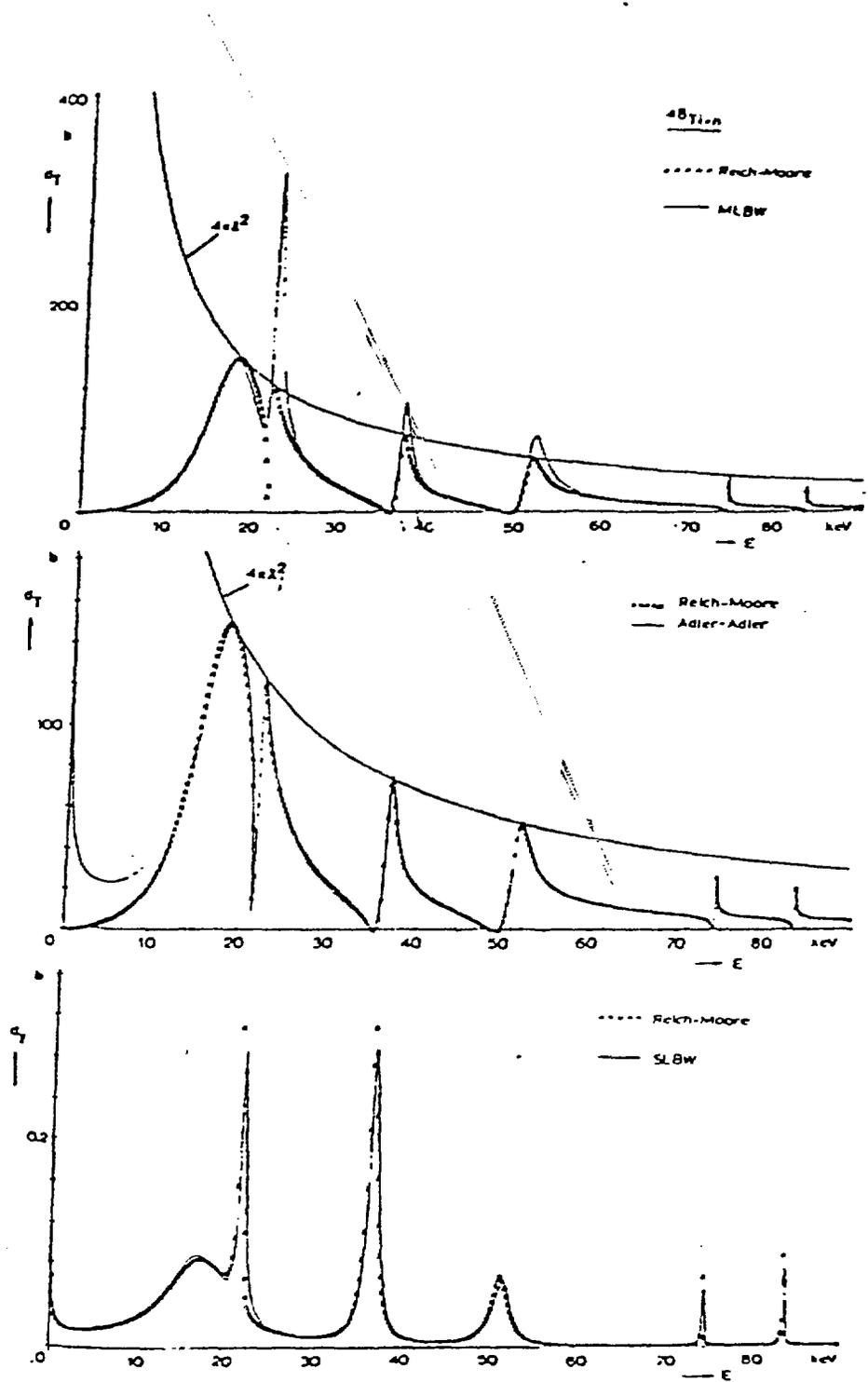
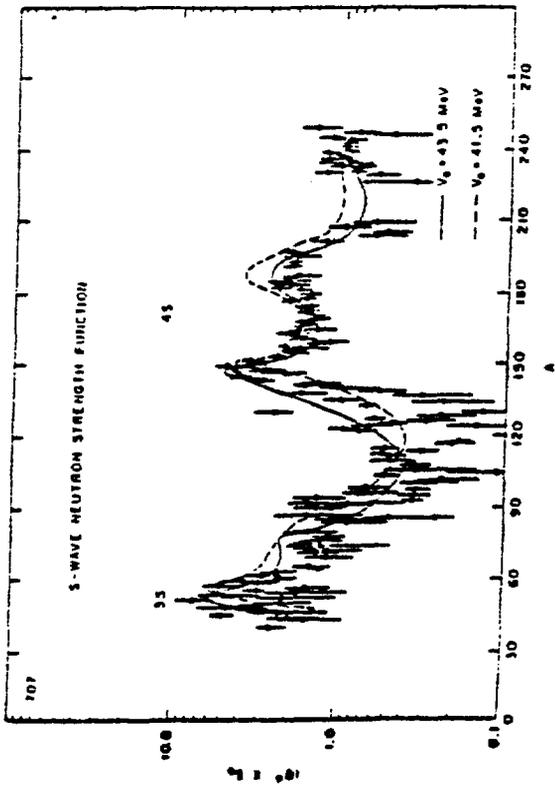
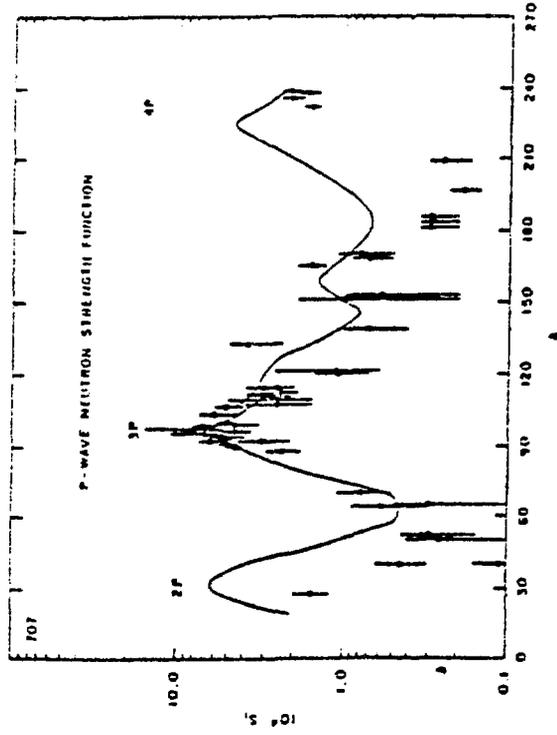
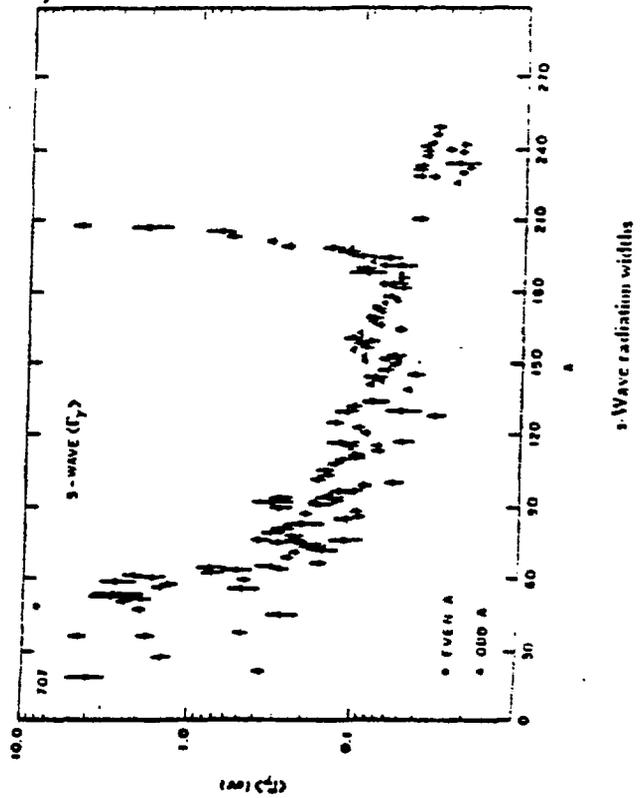


Illustration of inadequacy of approximations admitted by FNDF conventions for a medium-mass nucleus, ^{48}Ti . The Reich-Moore values are exact (parameters from /18/). The unitarity limit $4\pi^2$ is seriously violated by MLBW, less by Adler-Adler. Both these approximations are useless for capture cross section calculation here.

Fig. 4.3



s-Wave neutron strength function. The parameters of the optical potential are the same as in Figure 1.



Experimental values of p-wave strength functions compared with predictions of the optical model.

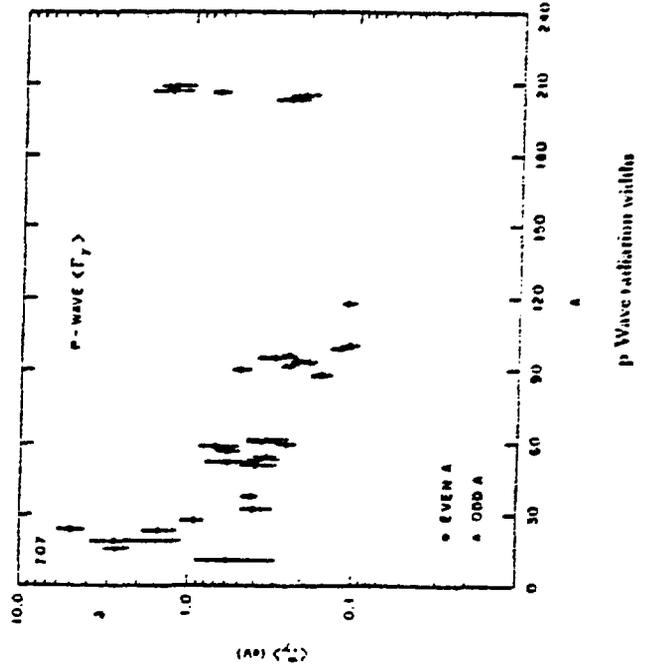


Fig. 4.4

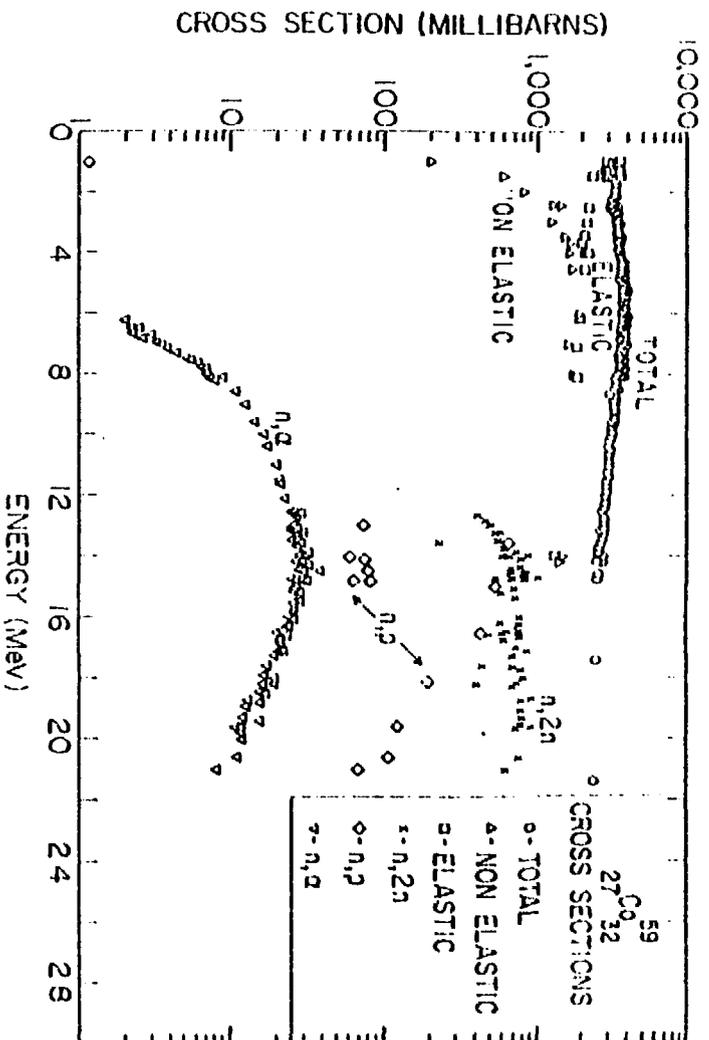


Fig. 4.5

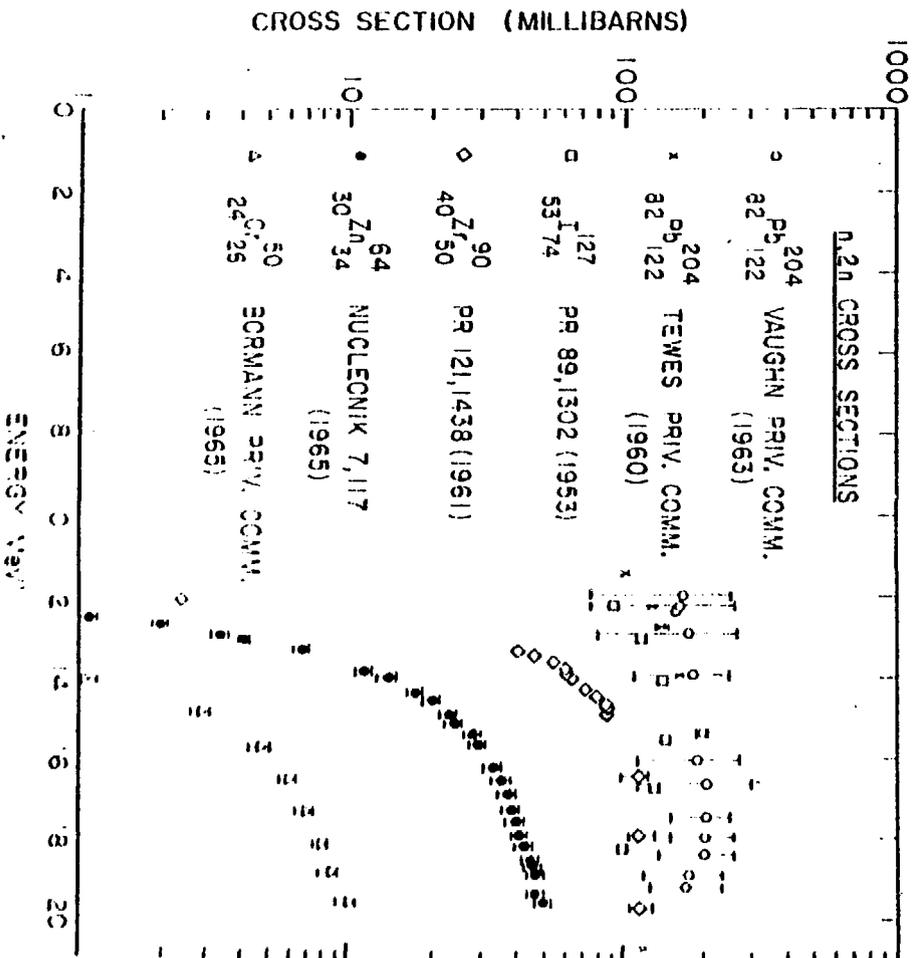


Fig. 4.6

Fig. 4.7

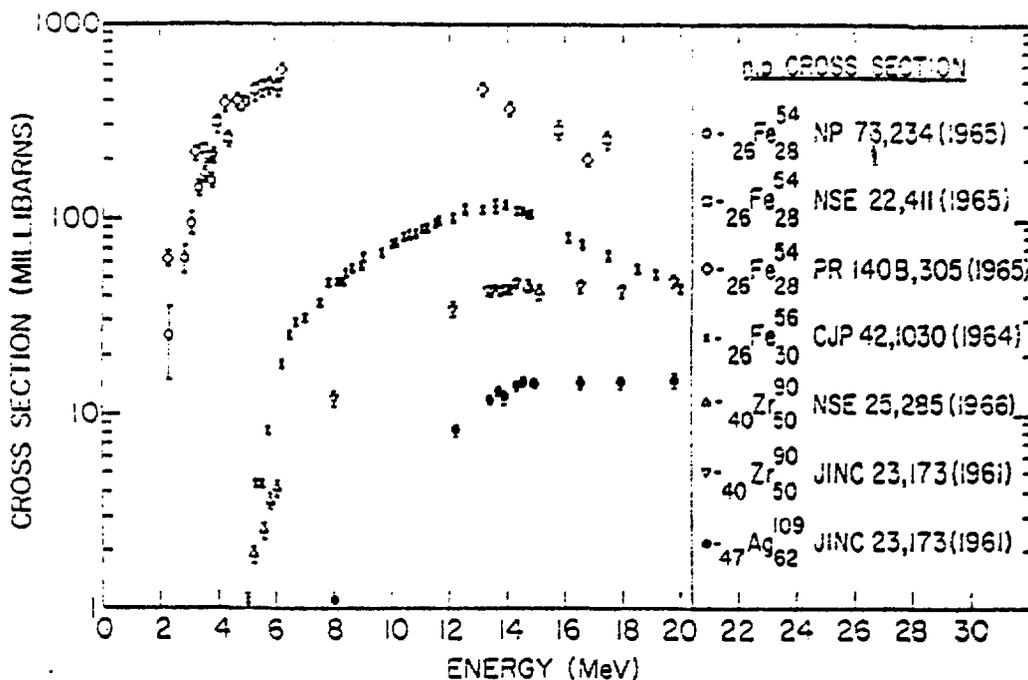
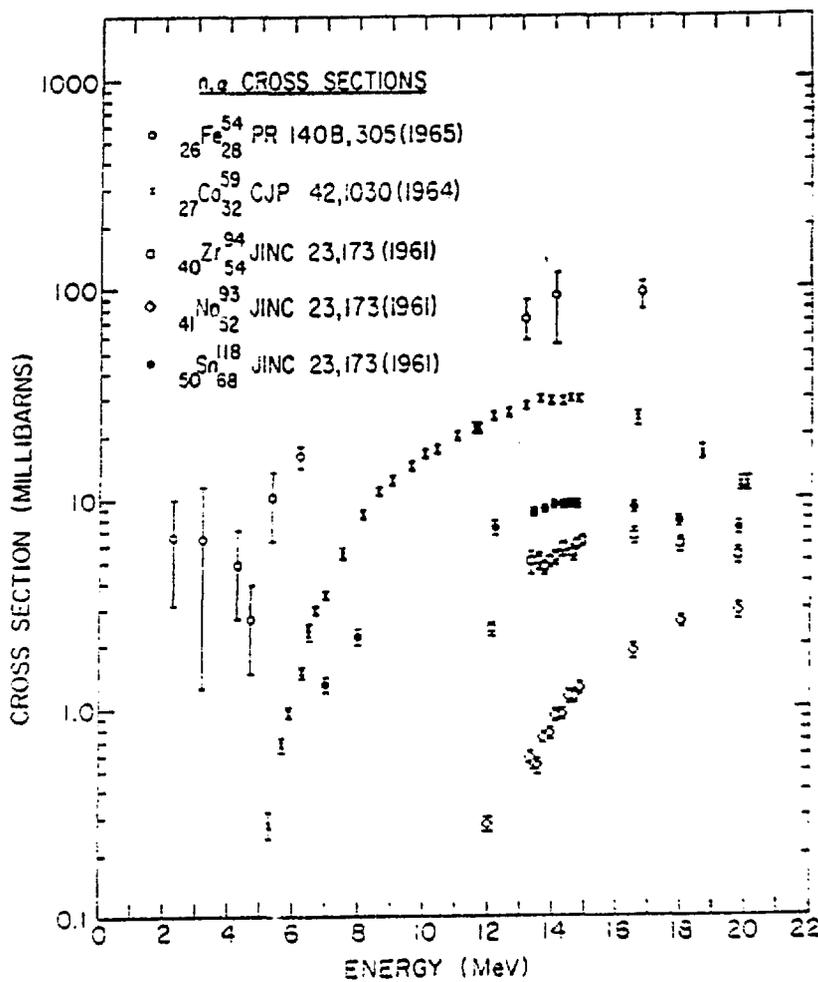


Fig. 4.8



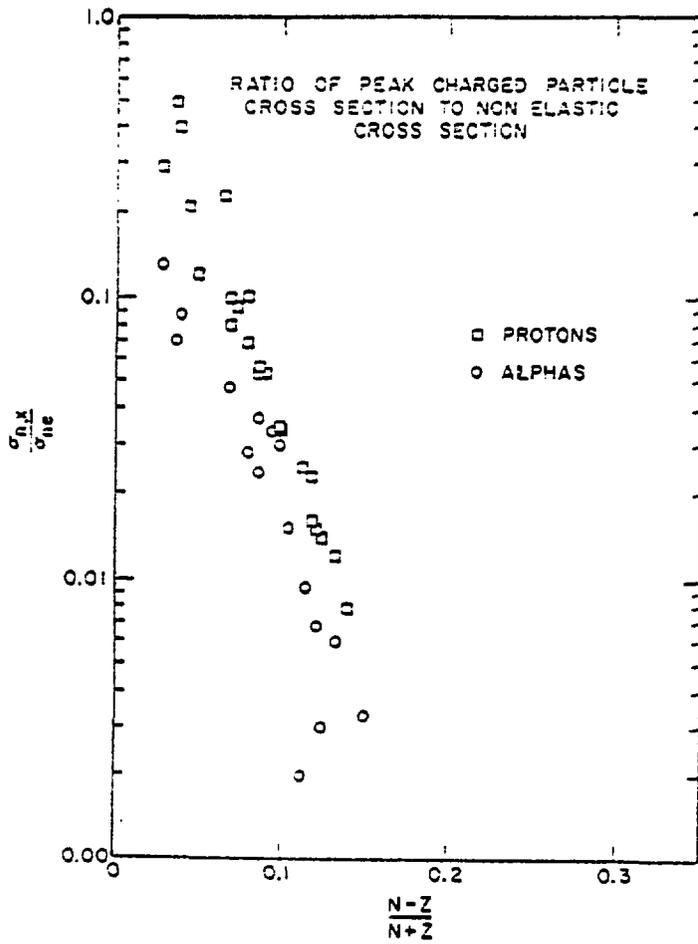


Fig. 4.9

(n,p) CROSS SECTION

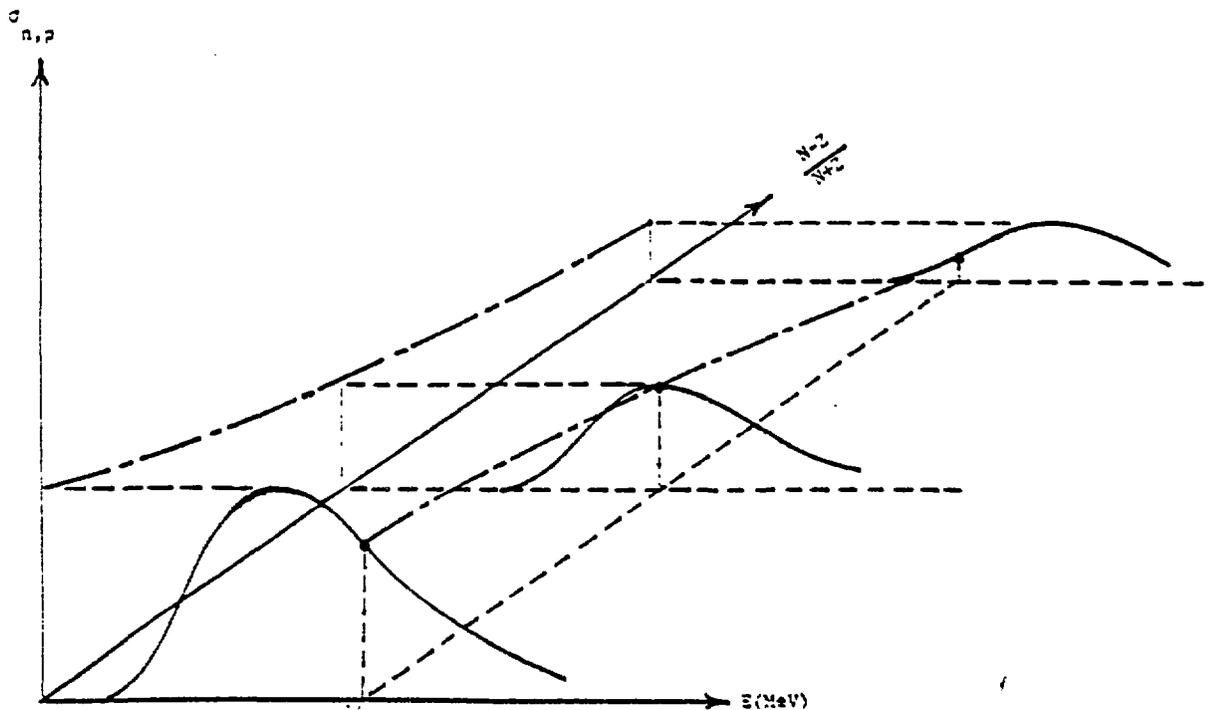


Fig. 4.10

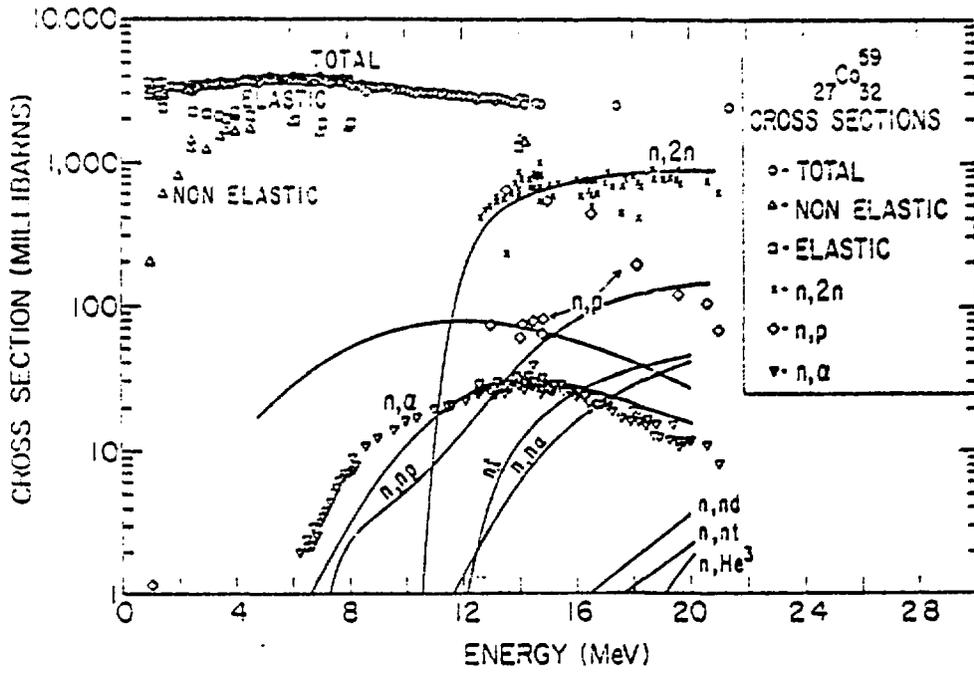


Fig. 4.11

DIFFRACTION THROUGH CIRCULAR APERTURE

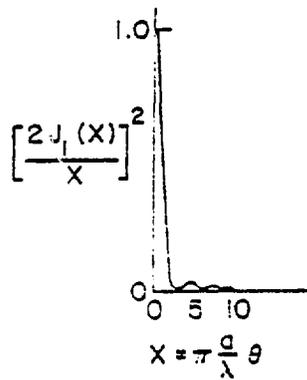


Fig. 4.12

Fig. 4.13

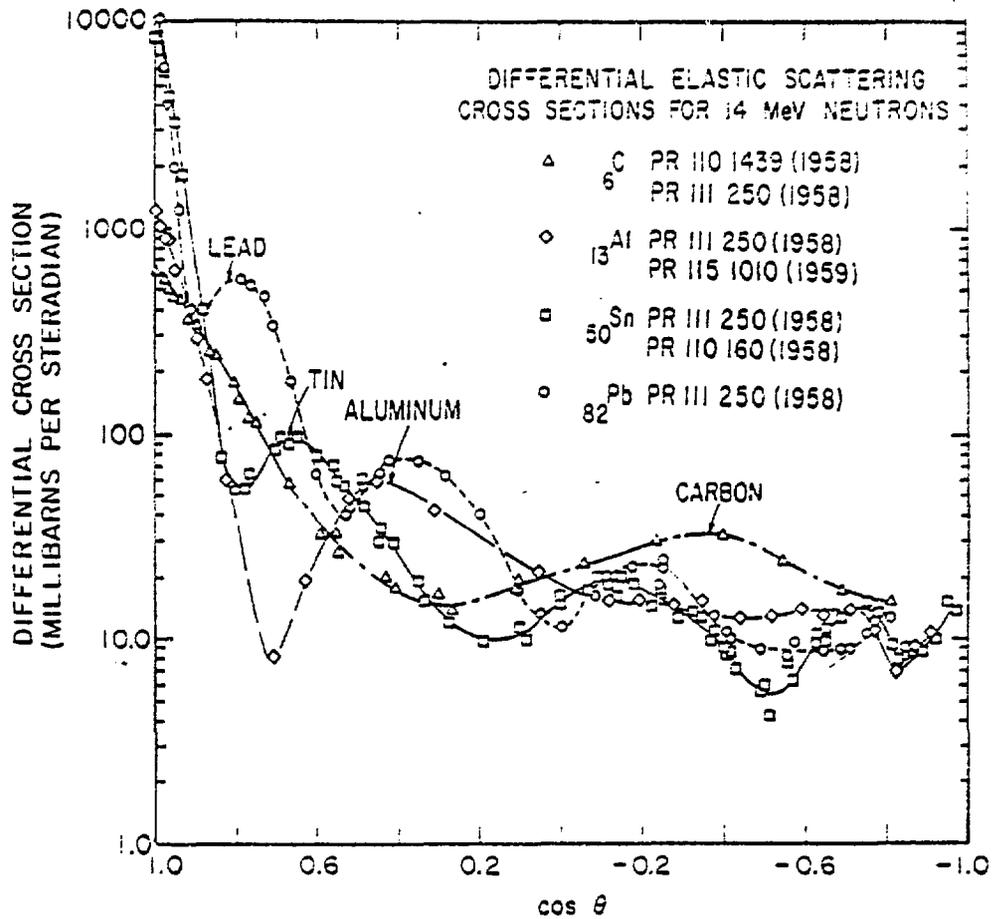


Fig. 4.14

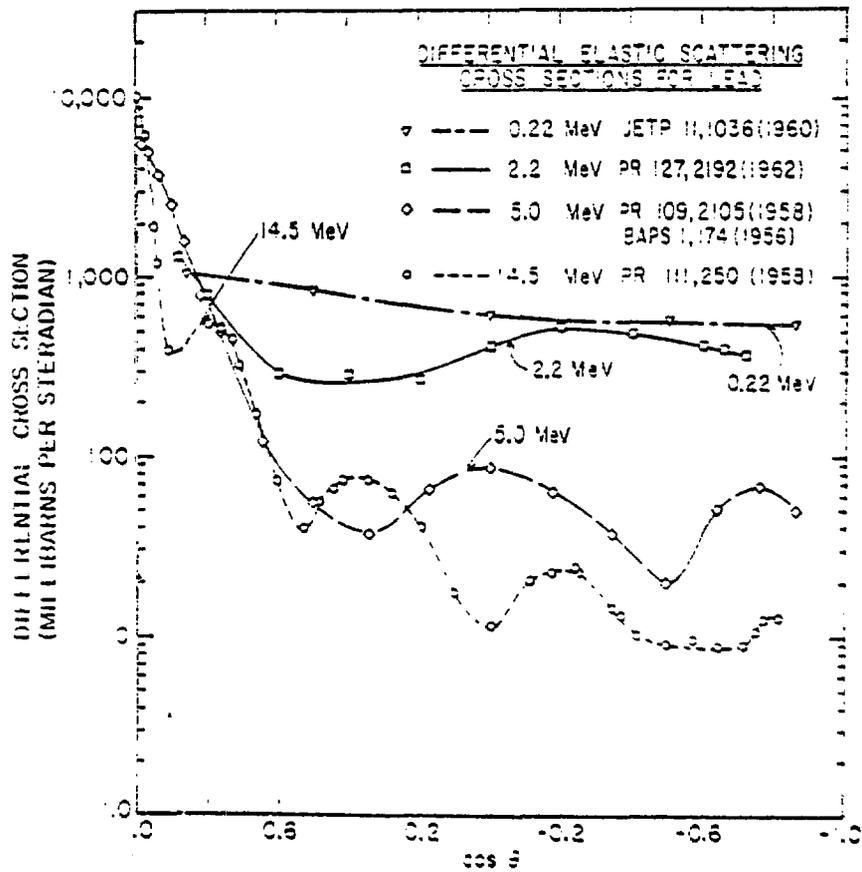
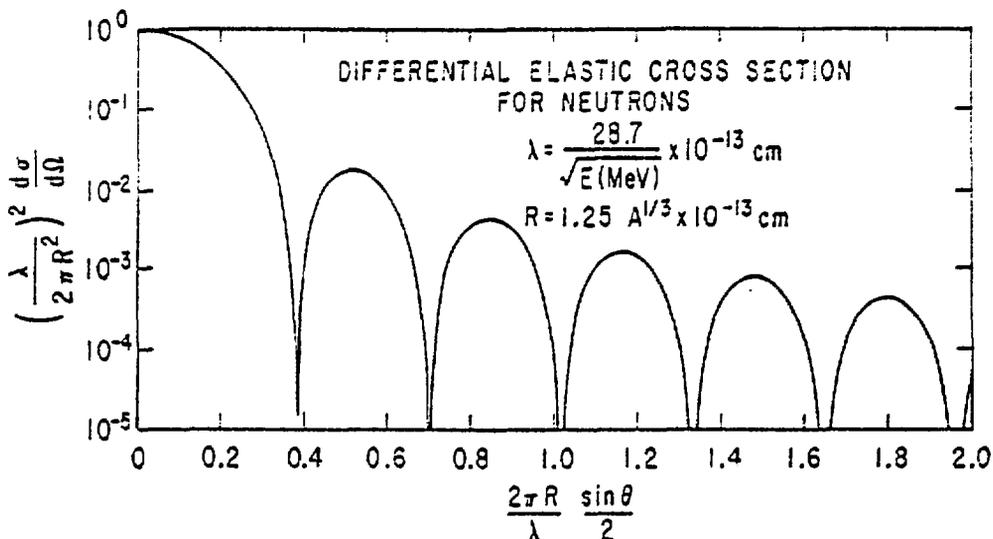


Fig. 4.15



Cr (NATURAL) DIFFERENTIAL ELASTIC SCATTERING

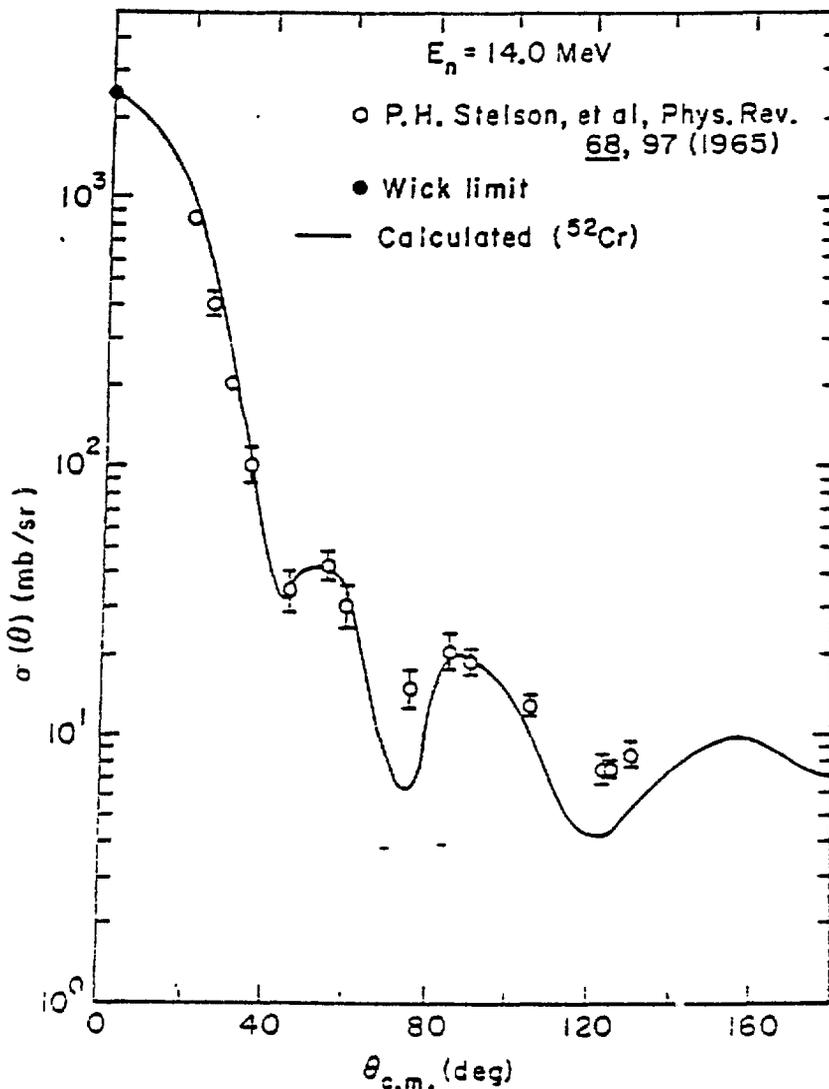


Fig. 4.16

92 Uranium 235

Quantity	Energy (ev)		Lab	Type	Documentation		Date	Author, Comments
	Min	Max			Ref	Vol Page		
Fission	2.7+5		MHG	Expt	Abst	ANS 21 503	Jun 75	Robertson+CS=1.32B +-2PC.REL NBS-2
Fission	Fast		DOU	Expt	Prog	UKNDC(75)P71	Jul 75	Davies+ IRRADIATION PFR SMALL SAMPLE
Fission	Maxwl		ILL	Expt	Jour	NP/A 247 74	Jul 75	Clerc+ MEASURED Z-DIST OF FP.TBL
Fission	Maxwl		SRC	Expt	Prog	UKNDC(75)P71	Jul 75	Alam+ BETA ENERGY EMITTED AFTER FISS
Fission	Pile		WIN	Expt	Prog	UKNDC(75)P71	Jul 75	Taylor+BETA ENERGY AFTER FISS
Fission	9.6+5		MHG	Expt	Jour	ANE 2 637	Sep 75	Gilliam+ CS(964KEV) = 1.21 +-0.025 B
				Conf		75Wash. 634	Mar 75	- +CS=1.21B+-2.1PCT.NA-3E NEUTS
				Abst		ANS 15 946	Nov 72	- + SUPERSEDED
	9.6+5			Data		EXFOR10314.002	Feb 76	1 PT. 964 KEV
Fission	5.3+6	1.0+7	ANL	Expt	Jour	NSE 58 255	Oct 75	Meadows.TBL,GRPH 238/235U NF RATIO
Fission	1.5+7		BRC	Expt	Abst	ANS 22 664	Nov 75	Cance+ 14.6MEV.CFD OTHER EXPT,ENDF
Fission	9.6+5		MHG	Expt	Jour	ANE 2 637	Nov 75	Gilliam+ ABSOL EXPT NA-3E SOURCE
Fission	4.0+1	2.0+2	BNL	Theo	Jour	NSE 58 354	Dec 75	Pearlstein.PROB TABLE METHOD.GRPHS
Fission	Maxwl		IBJ	Expt	Jour	NP/A 255 387	Dec 75	Piasecki+P.D.T,ALF EMISSION MEASURED
Fission	8.0+5	4.0+6	LRL	Expt	Jour	NSE 58 371	Dec 75	Czirr+ TBL,GRPHS.REL NP SCATTERING
	8.0+5	2.0+7		Prog		ERDA-NDC-2 70	May 75	- +RELTO N-P SCAT.NDG.
				Conf		75Wash. 615	Mar 75	Sidhu+TABLE,GRAPH.REL TO N-P SCATT.
	7.5+5	2.0+7		Data		EXFOR10428.002	May 75	38PTS,SIGMA
Fission	0.0+0	1.0+7	CAI	Theo	Jour	AKE 27 1 47	76	Ei Nadi+ QUASI-MOLECULAR MODEL
Fission	2.0-2	2.0+5	ORL	Expt	Jour	NSE 59 79	Feb 76	Gwin+ TBL,GRPHS AVG CS

Fig. 6.1

ENTRY	10900	790830	5	1090000000001
SUBENT	10900001	790830		10900001 1
BIB	14	44		10900001 2
INSTITUTE	(1USAANL)			10900001 3
REFERENCE	(J,BAP,24,631,7904)			10900001 4
	(S,SMITH,7904)			10900001 5
AUTHOR	(D.L.SMITH,J.W.MEADOWS)			10900001 6
TITLE	CROSS SECTION MEASUREMENT FOR THE CR-52(N,PROTON)V-52			10900001 7
	REACTION NEAR THRESHOLD			10900001 8
FACILITY	ARGONNE FAST NEUTRON GENERATOR			10900001 9
N-SOURCE	(D-0) DEUTERONS ON DEUTERIUM GAS			10900001 10
SAMPLE	SOLID CHROMIUM CYLINDER, MASS OF 86.76-GRAMS APPROX			10900001 11
	10-CM FROM NEUTRON SOURCE.			10900001 12
DETECTOR	(GELI) TRUE COAXIAL GERMANIUM-LITHIUM DETECTOR IN			10900001 13
	MASSIVE SHIELD. VOLUME IS APPROX 100-CC. LOCATED			10900001 14
	APPROX 100-CM FROM SAMPLE, 90-DEG FROM BEAM.			10900001 15
	(FISCH) U-238(93.90-PERCENT),U-235(6.09-PERCENT),			10900001 16
	U-235(0.01-PERCENT) FISSION CHAMBER FOR NEUTRON FLUX			10900001 17
	MEASUREMENT.			10900001 18
DECAY-DATA	(23-V-52,3.75MIN,0G,1434.)			10900001 19
MONITOR	(92-U-235(N,F),,SIG) FROM ENDF/B-IV			10900001 20
METHOD	(ACTIV). ENERGY SCALE OF ACCELERATOR CALIBRATED USING			10900001 21
	PROTON BEAM AND OBSERVING REACTION THRESHOLDS FOR			10900001 22
	PROTONS ON LI-7, B-11, AND AL-27. CRYSTAL CONTROLLED			10900001 23
	DELAY CIRCUIT USED TO INTRODUCE A DELAY TO ALLOW			10900001 24
	SCATTERED NEUTRON BACKGROUND TO DIE AWAY.			10900001 25
CORRECTION	CORRECTED FOR GEOMETRY, NEUTRON SOURCE CHARACTERISTICS,			10900001 25
	ABSORPTION AND MULTIPLE SCATTERING OF NEUTRONS,			10900001 27
	ABSORPTION OF GAMMAS, TARGET THICKNESS, AND DECAY			10900001 29
	HALF-LIFE.			10900001 29
ERR-ANALYS	SOURCE OF ERRORS IN MEASURED RATIO INCLUDE-			10900001 30
	-DETERMINATION DETECTOR COUNTS, 2-16 PERCENT			10900001 31
	-GEOMETRIC EFFECTS, 3-PERCENT			10900001 32
	-NEUTRON SOURCE CHARACTERISTICS, 3-PERCENT			10900001 33
	-GAMMA-RAY DETECTOR EFFICIENCY, 4-PERCENT			10900001 34
	-MASS OF URANIUM DEPOSIT, 1-PERCENT			10900001 35
	-GAMMA RAY ABSORPTION IN CHROMIUM, 2-PERCENT			10900001 36
	-NEUTRON ABSORPTION IN CHROMIUM, 2-PERCENT			10900001 37
	-NEUTRON MULTIPLE SCATTERING IN CHROMIUM, 2-PERCENT			10900001 38
	-HALF LIFE AND DUTY CYCLE, 2-PERCENT			10900001 39
	-BACKGROUND EFFECTS, 1-4 PERCENT			10900001 40
	-GAMMA DECAY BRANCHING FACTOR, 1-PERCENT			10900001 41
	TOTAL RMS ERROR, 3-18 PERCENT			10900001 42
	SYSTEMATIC ERROR, APPROX 8-PERCENT			10900001 43
	ERROR IN U-238 FISSION CROSS SECTIONS ASSUMED TO BE			10900001 44
	APPROX 5-PERCENT.			10900001 45
HISTORY	(790412C)			10900001 45
END8BIB	44			10900001 47
COMMON	1	3		10900001 48
Q-VAL				10900001 49
MEV				10900001 50
	-3.195			10900001 51
ENDCOMMON	3			10900001 52
ENDSURENT	51			1090000199999
SUBENT	10900002	790830		10900002 1
BIB	2	5		10900002 2
REACTION	1((24-CR-52(N,P)23-V-52,,SIG)/(92-U-238(N,F),,SIG))			10900002 3
	2(24-CR-52(N,P)23-V-52,,SIG)			10900002 4

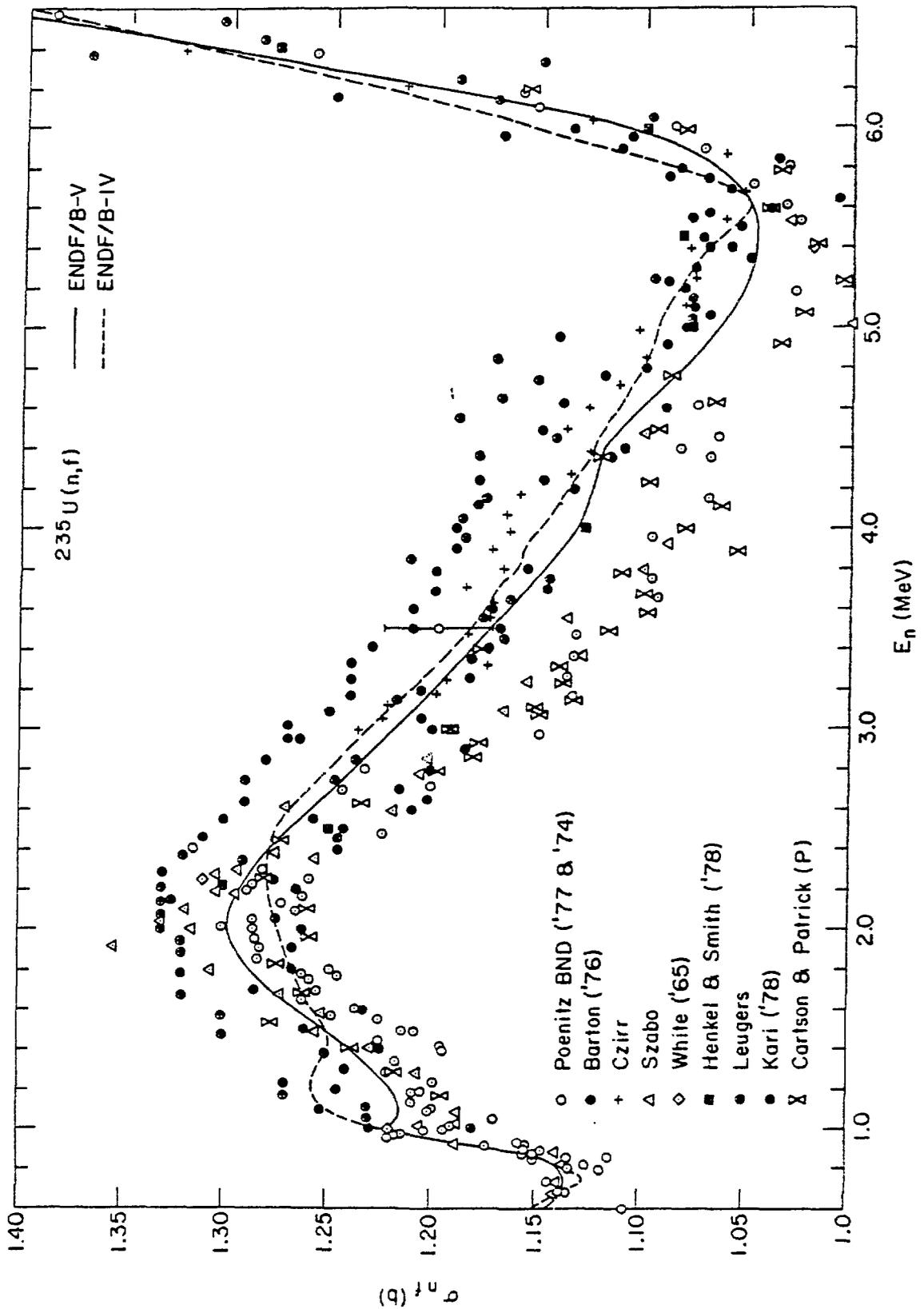


Fig. 6.3

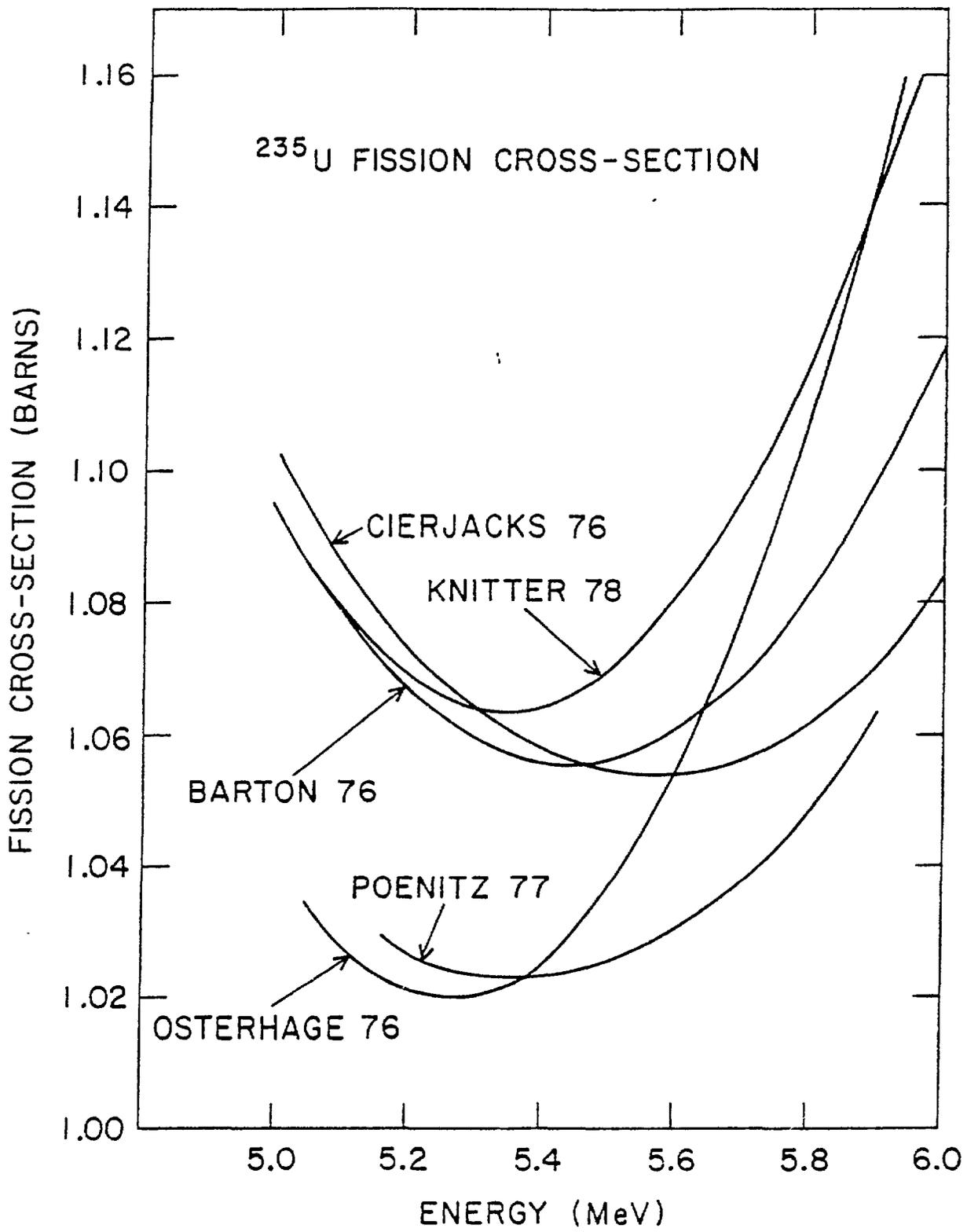
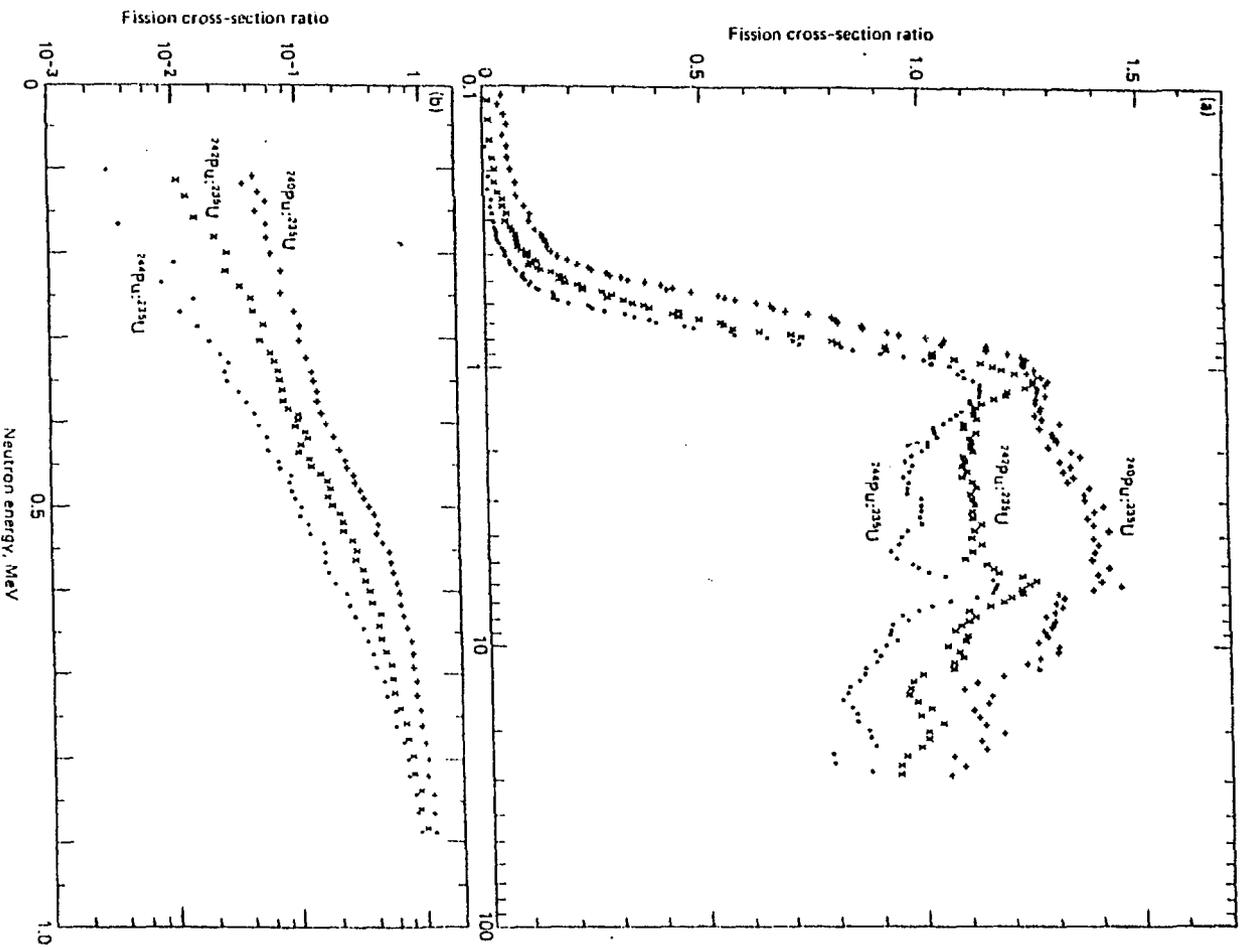


Fig. 6.4



Present measurements of the ^{240}Pu : ^{235}U , ^{242}Pu : ^{235}U , and ^{244}Pu : ^{235}U fission cross-section ratios (a) over the neutron-energy range from 0.1 to 30 MeV, and (b) from 0.1 to 0.9 MeV.

Fig. 6.5

^{235}U FISSION CROSS SECTION
 80-200 eV
 R. B. PEREZ et al
 Nuc. Sci. and Eng. 52 (1973) 46-72

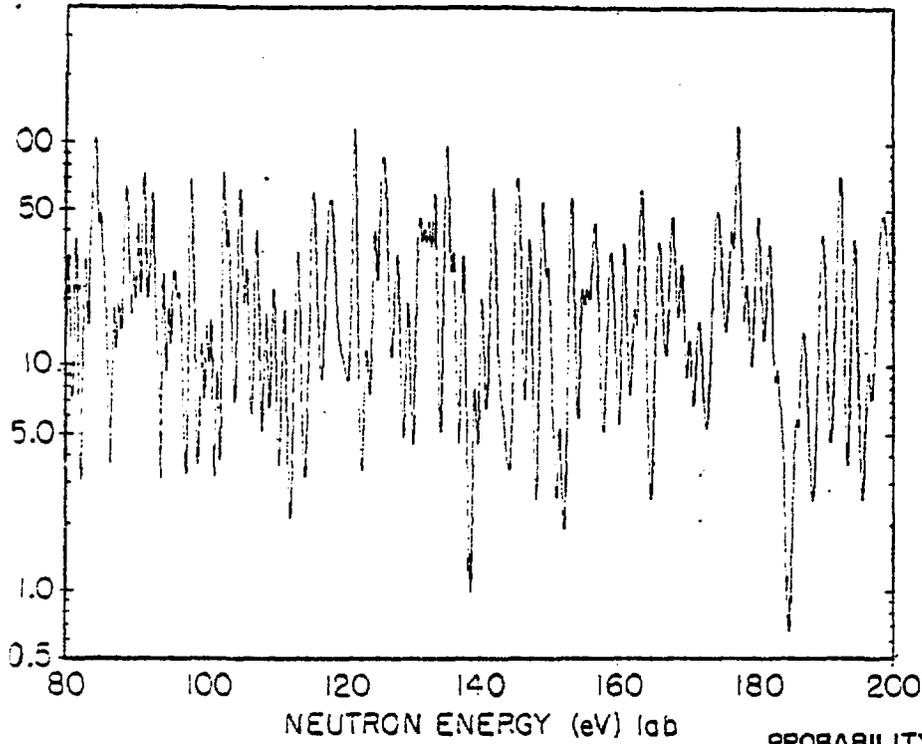


Fig. 8.1

PROBABILITY TABLE
 ^{235}U FISSION CROSS SECTION
 80-200 eV

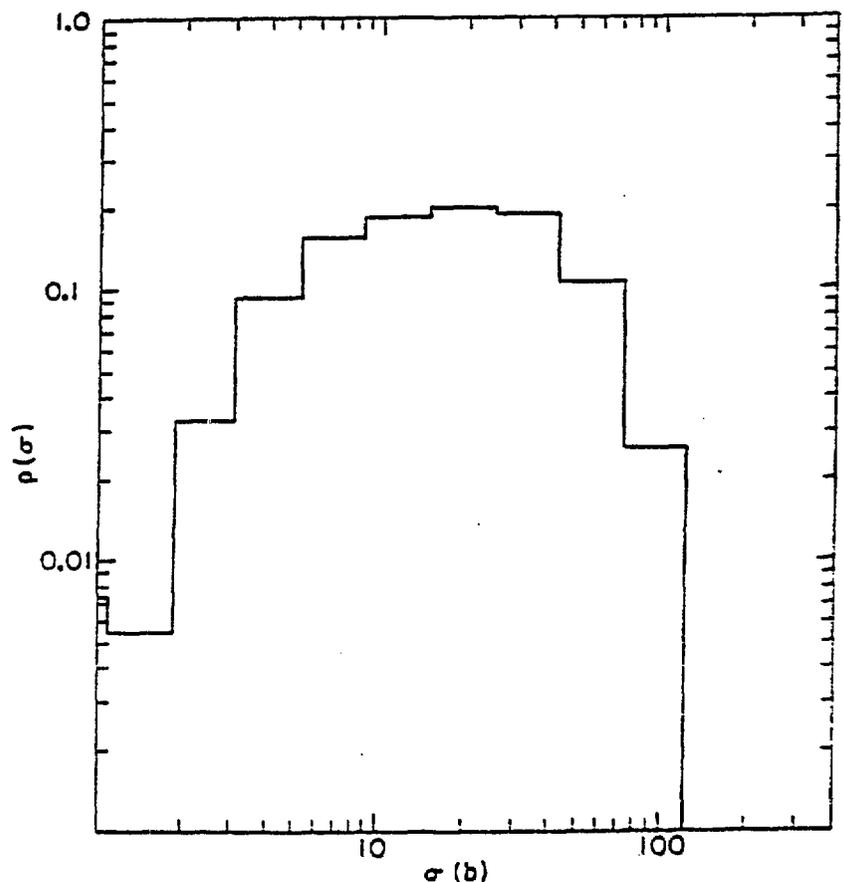


Fig. 8.2

EVALUATION AND PROCESSING OF NUCLEAR DATA

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ABSTRACT

The role a nuclear data evaluator plays in obtaining evaluated nuclear data, needed for applications, from measured nuclear data is surveyed. Specific evaluation objectives, problems, and procedures are discussed. The use of nuclear systematics to complement nuclear experiment and theory is described. ^{with} ~~Using~~ the Evaluated Nuclear Data File (ENDF) as an example, the formatting, checking, and processing of nuclear data ^{are} ~~is~~ discussed as well as the testing of evaluated nuclear data in the calculation of integral benchmark experiments. Other important topics such as the Probability Table Method and interrelation between differential and integral data are also discussed.

25 figures

1.0 Introduction

1.1 The Psychology of Evaluation

The process of evaluation involves decision making. The objectives of a nuclear data evaluator are to recommend values for nuclear data and also indicate the degree of confidence that can be placed in those recommendations. Often the experimental data being examined by the evaluator has quoted errors that are not realistic. Nevertheless, the evaluator is expected to estimate the

most probably correct values of nuclear data. The evaluator is like a juror. From the evidence, no matter how contradictory it may be, the evaluator is supposed to get at the truth. As a juror's decision must be within the court of law, the evaluator's recommendation must be consistent with the best laws of physics. The evaluator need not be an expert in all phases of nuclear physics but where his knowledge is deficient he must be capable of incorporating the recommendations of other experts into his evaluation.

There is no prescribed college course for nuclear data evaluation as there is for nuclear physics, nuclear engineering, reactor physics, nuclear chemistry, etc. Evaluation is a combination of art and science. Evaluation used to be more art than science, since there was little data and the evaluator depended on nuclear systematics or just plain guesswork in order to recommend data for use in applications. Today, evaluation is more science than art. There is more data that can be considered and the evaluation is expected to be consistent with all of the observable facts. Sometimes observable facts and evaluations take on political importance. Good evaluations that can calculate observable facts from first principles are taken seriously by reactor designers and even play a role in international data exchange agreements. The evaluator through his recommendation can have an impact on nuclear power programs.

The evaluator must have the finest moral character. He must be uncorruptible. His recommendations must be supported by experimental and theoretical considerations and not be strongly influenced by values favored by particular nuclear applications.

The nuclear data base provided by the evaluator is important to both basic science and applied science. The basic scientist examines the nuclear data base and wants to know why the nuclear data are what they are. He seeks to explain the systematics of nuclear data through an understanding of fundamental nuclear