NUCLEAR DATA AND MEASUREMENTS SERIES

ANL/NDM-24

Fast Neutron Cross Sections of Vanadium and an Evaluated Neutronic File

by

P. Guenther, D. Havel, R. Howerton, F. Mann, D. Smith, A. Smith, and J. Whalen

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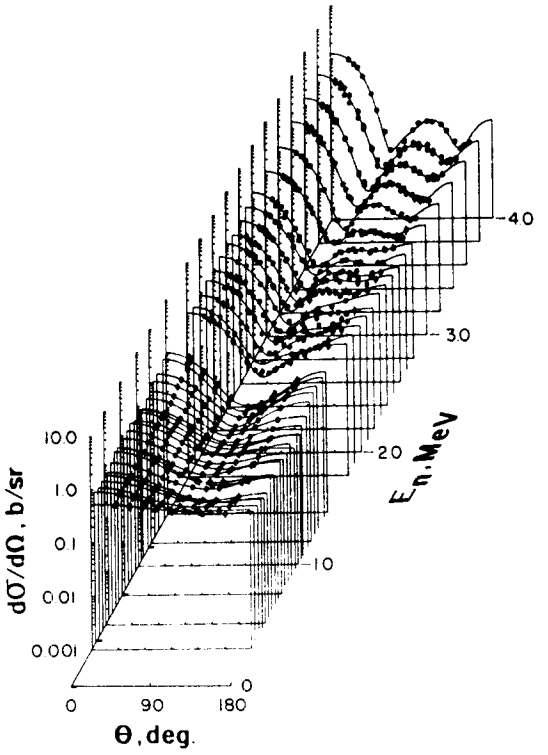

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May 1977

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 MEASUREMENT SERIES

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FAST NEUTRON CROSS SECTIONS OF VANADIUM AND AN EVALUATED NEUTRONIC FILE*

by

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ABSTRACT

Energy-averaged total cross sections of elemental vanadium were measured from 1.5 to 5.5 MeV. Differential elastic and inelastic neutron scattering cross sections were measured from 1.8 to 4.0 MeV. Neutrons corresponding to the excitation of states in vanadium at 321±10, 938±15, 1603±19, 1811±21, 2409±27, $\approx$ 2500, 2706±30 and 2773±30 keV were observed. From these experimental results an energy-average model was deduced suitable for extrapolating and interpolating the measured values. These results and those reported elsewhere were used to construct a comprehensive Evaluated Neutronic File in the ENDF format with particular attention to higher-energy processes having an impact on FBR, CTR, dosimetry and gas production applications.

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I. INTRODUCTION

Vanadium is a promising structural metal for use in high-temperature neutronic systems particularly where tritium containment is a concern (e.g., in fusion power systems). The element is essentially monoisotopic (> 99% ^{51}V), "magic" in neutron number, an odd nucleus near the peak of the s-wave strength function and the spectroscopic structure is relatively well known to excitations of several MeV (1). These unusual characteristics make the interaction of fast neutrons with vanadium of physical interest and, in some ways, simplify experimental studies of the associated fast-neutron interactions and their interpretation. At incident energies of $\approx$ 1.0 MeV the respective neutron cross sections are known to fluctuate with energy and to have some aspects characteristic of intermediate structure (2). Between 1.0 and 5.0 MeV there is a transitional region where the character of the neutron interactions change from a discrete-resonance to an energy-averaged behavior. Compound-nucleus processes in this mass-energy region remain a matter of physical interest (3). Moreover, experimental information in the few-MeV region is useful in establishing nuclear models which subsequently can be applied to extrapolate and interpolate difficult to measure cross sections. This is particularly true for vanadium as a number of important (n;X) cross sections are not well measured; and, as a consequence, model-based theory must be used to extrapolate the measured values over wide energy ranges in order to obtain

cross section values important to many applied calculations.

The primary objective of the present work is the provision of a good contemporary data base for applied use. It was an intent of the measurements to better define the fast neutron interaction with vanadium in the few-MeV range and thus provide a basis for the development of a realistic energy-average model for subsequent extrapolation and interpolation. The present results and those reproted elsewhere in the literature were used to develope a comprehensive Evaluated Neutronic File in the ENDF format (4). Particular attention was given to the higher-energy processes of importance in FBR and CTR applications and to the dosimetry and gas-production aspects of the data file. Subsequent portions of this paper outline the present experimental and calculational results and the construction of the evaluated file. The complete numerical contents of the file are given in the Appendix.

II. EXPERIMENTAL METHODS

All samples used in the measurements consisted of natural metallic vanadium cylinders with 99+ percent chemical purity. The samples used in the neutron-total-cross-section measurements were cylinders approximately 2.5 cm in diameter with lengths adjusted to provide neutron transmissions of 50 percent or greater in the axial direction. Sample-thickness perturbations were assumed small. The scattering samples were solid cylinders 2 cm. in diameter and 2 cm. long; arranged so that neutrons were incident upon their lateral surfaces.

Both the neutron total and scattering cross section measurements employed the $^{7}Li(p,n)^{7}Be$ source reaction (5). The incident neutron energy scale was determined relative to the threshold of this reaction. The incident neutron-energy resolution was determined by the thickness of the lithium-metal film used as the neutron producing target. Throughout the measurements the source was pulsed for a duration of approximately 1 nsec at a repetition rate of 2 MHz using the pulsing capabilities of the Argonne National Laboratory Tandem Dynamitron (6).

The neutron total cross sections were determined from the observed transmissions of approximately monoenergetic neutrons through the vanadium samples. A large collimator was placed about the neutron source so arranged as to provide an approximately monoenergetic neutron beam of about 1 cm diameter at a source-reaction angle of zero degrees. The collimator was approximately a meter thick and care was taken to assure that the aperture did not perturb the neutron-energy distribution. The samples were placed approximately 30 cm beyond the collimator with the beam passing through them in an axial direction. The samples were mounted on a wheel which changed the samples $\sim$ 180 times a minute. In addition to the vanadium samples, the wheel contained a void and a carbon reference standard. A scintillation detector was placed upon the beam axis about 5 m from the neutron source. The time of flight of the neutrons from the source to the detector was determined using conventional electronic systems. The experimental time resolution was sufficient to control experimental perturbations such as back-

ground and to separate the second and minority neutron groups from the primary $^{7}Li(p,n)^{7}Be$ source neutrons. Several BF_3 neutron detectors were arranged in moderating media to provide a monitoring of neutron-source intensity. The data acquisition and reduction to cross sections was done on-line using a digital computing system which automatically correlated sample position and detector count. The measurements were carried to predetermined statistical-cross-section accuracies, typically 1-3 percent. The details of the experimental apparatus and its application have been extensively described elsewhere (7).

The neutron scattering cross sections were determined using the pulsed-beam time-of-flight technique (8) and a ten-angle detection system. The incident neutron energy spread varied from 20 to 50 keV. Scattered-neutron flight paths were nominally five meters and scattered neutron velocity resolutions $\lesssim$ 0.4 nsec/m. The detection apparatus consisted of a massive shield with 10 well-collimated flight paths distributed over an arc of approximately 140 deg. The placement of individual detectors could be varied to yield scattering angles from +160 to -160 deg. The relative angular position of the detectors was determined to within ± 1.0 deg. using a precision optical transit. The reference zero of this angular system was determined to ± 1.0 deg. by observing the energy shift of neutrons scattered from hydrogen at a number of angles both left and right of the zero-deg. reaction angle. Measurements were made concurrently at 10 scattering

angles. Additional angular placements of the collimator system were used to obtain 20 or more differential scattering cross sections at a given incident neutron energy.

The scattered-neutron detectors were proton-recoil liquid scintillators. Electronic pulse-shape-selection circuitry was employed to reject most gamma-ray induced events. Data acquisition was by means of a digital computing system. These were programmed to select the dector sensitivities for optimum signal-to-background ratios and/or time resolutions. Generally, the detectors were sensitive to scattered neutron energies of $\gtrsim$ 500 keV.

The neutron source intensity was monitored using four long counters (9) and a special-purpose time-of-flight system. The monitors were arranged so as to be insensitive to the angular placement of the collimator system and to the presence of the scattering samples.

The relative energy-dependent sensitivities of each of the scattered-neutron detectors were experimentally determined by the observation of neutrons scattered from the hydrogen in a polyethylene sample. Subsequently, the normalization of these relative sensitivities was established by the observation of scattering from hydrogen at each incident energy (8). Thus all scattering cross sections were determined relative to the well known H(n,n) cross sections (10). These detector calibration procedures were not sensitive to the energy dependence of the monitors nor did they require an absolute flux determination or an exact knowledge of the

source reaction. The calibration procedures were repeated at each of approximately ten measurement periods. There was considerable redundancy in the measured cross section values. Thus the consistency of the results was a measure of the reliability of the techniques.

All of the experimental data were corrected for multiple scattering, incident beam attenuation and angular resolution effects using a combination of Monte-Carlo and analytical procedures (11). These correction procedures were applied to the vanadium samples and the polyethylene standard samples. At each incident energy, in addition to the polyethylene-standard and vanadium measurements, a carbon-reference measurement was made. The carbon cross sections are reasonably well known at the energies of the present experiments and their determination provided a verification of the experimental procedures (12).

The details of the scattering apparatus and the associated procedures have been described elsewhere (8,13).

III. EXPERIMENTAL RESULTS

A. Neutron Total Cross Sections

The total cross sections measurements were made in two separate sets, extending in energy from less than 1.0 MeV to 5.5 MeV. The energy resolution of one set was approximately 5 keV and that of the other approximately 25 keV. The primary objective was the accurate determination of the energy-averaged magnitudes of the total cross sections so as to: verify the results of previously reported higher-resolution measurements, provide a basis for the

development of energy-averaged models (e.g. optical model), and assure the consistency of the present scattering cross section results with the total cross section values. Considerable care was taken to avoid experimental distortions due to backgrounds, "in-scattering" and dead times. The resolution of details of the resonance structure was not emphasized. The individual data values fluctuated with energy depending upon the resolutions employed in the particular measurements and the exact incident neutron energies. The energy-averaged values were derived by averaging the individual measurements over energy intervals of 100 keV. These averaged results were relatively smooth functions of energy and reasonably comparable with the angle-integrated scattering values deduced from the present measurements. The statistical errors of the energy-averaged values were 1 to 4 percent and the systematic uncertainties were believed to be appreciably smaller. A possible exception is the sample size effect. It was assumed small (< 2 percent) due to the relatively large average transmissions of the samples. The energy-averaged results are illustrated in Fig. 1. At the lower energies (e.g., below 1.5 MeV) the present results compare favorably with those reported previously from this laboratory (2). More generally, the present results support the higher-resolution values of several authors as summarized in the evaluation described in Sec. V (e.g., Cierjacks et al. Ref. 14).

B. Elastic Neutron Scattering Cross Sections

The differential elastic neutron scattering cross sections

were measured at 20 or more scattering angles distributed in the range of 20 and 160 deg. and at incident neutron energy intervals of $\leq$ 0.2 MeV from 1.8 to 4.0 MeV. The experimental resolution was sufficient to resolve the elastically-scattered component from all known inelastically-scattered contributions. The accuracies of the individual differential data points were $\gtrsim$ 5 percent or 2 to 3 mb/sr, whichever was larger. These uncertainty estimates were inclusive of experimental counting statistics ($\gtrsim$ 1 percent), uncertainties in the reference H(n;n) cross sections ($\gtrsim$ 1 percent), uncertainties in the correction procedures (1 to 3 percent) and estimated systematic uncertainties associated with the determination of detector sensitivities and scattering angles (3 to 5 percent). The data were obtained from a number of measurement periods each with essentially independent calibration procedures. The incident neutron energy resolutions varied from 20 to 50 keV and the estimated uncertainty in the mean incident energy was $\approx$ 10 keV. Angular distributions obtained at different measurement periods were not always consistent, with discrepancies as large as several times the estimated uncertainties particularly at the lower extremity of the measured energy range (i.e., 1.8 to 2.0 MeV). These discrepancies were attributed to the large fluctuations in the cross sections evident in the results of high-resolution total cross section measurements (See Fig. 1). Small variations in the mean incident neutron energy and/or resolution employed in scattering measurements at nominally the same energy could easily lead to considerably different results, particularly since the fluctuations observed in the total cross are expected to be

accentuated in the individual elastic channel. These fluctuations make it very difficult to compare the results obtained at different times (or at different laboratories). In order to obtain a better representation of the energy-average of the differential cross sections, the present results obtained at nominally the same incident energies (though with somewhat varying incident resolutions and mean energies) were averaged using a Legendre Polynomial fit to the data to interpolate the scattering angles so as to provide a common angular scale. The resulting energy-averaged differential elastic distributions are shown in Fig. 2. Even with this averaging procedure it is evident from Fig. 2 that some fluctuation persists, particularly at lower (e.g., near 2.0 MeV) energies.

The differential elastic scattering cross sections were fitted by least-squares with Legendre Polynomial expansions. These were used to deduce the angle-integrated elastic scattering cross sections. There were no constraints on this fitting procedure other than the introduction of a few 180 deg. cross section values, deduced from the model described in Sec. IV, in order to assure a well-behaved distribution at scattering angles beyond the measured angular range. The zero degree cross sections derived from this fitting procedure were consistent with Wick's Limit (15). These angle-integrated elastic scattering cross sections are compared with the measured neutron total cross sections in Fig. 1. The estimated uncertainty in the angle-integrated elastic scattering values is $\lesssim$ 5 percent and this is largely attributed to uncertainty in detector normaliza-

tion. The angle-integrated elastic scattering cross sections are consistent with the total and inelastic scattering cross sections as outlined in Sec. V.

The present elastic scattering results are in good agreement with those reported by Towle et al. (16) at 2135 MeV as illustrated in Fig. 3. There is similar good agreement with the 4.0 MeV results of Holmquist et al. (17). The 3.5-MeV values of Holmqvist et al. are reasonably consistent with the present results but at 2.47 and 3.0 MeV their values are systematically smaller than those of the present work. In addition, the values reported by Becker et al. (18) at 3.2 MeV are smaller than those from the present work. Whether these discrepancies are real or the consequence of cross section fluctuations is not clear. However, if the lower elastic scattering values are accepted, there will tend to be an inconsistency with the total and inelastic scattering cross sections.

C. Inelastic Neutron Scattering Cross Sections

The observed energies of states excited by inelastic neutron scattering were determined from the measured transit times of the scattered neutrons over the known flight paths. The method was verified by the observation of well-known inelastically-scattered neutron groups, particularly that due to the excitation of the 846.7 keV state in ^{56}Fe. Simple averages of the measured excitation energies were obtained from a minimum of five and as many as fifty individual measurements distributed over a four-year period. The corresponding uncertainties were expressed as RMS

deviations from the average values. Cross sections were determined for the excitation of the six "states" at 321±10, 938±15, 1603±19, 1811±21, 2409±27, and 2706±30 keV. In addition, neutrons corresponding to "states" at 2773±30 and approximately 2500 keV were observed in a few (less than 5) measurements. The observed values are compared with the evaluated quantities given in the Nuclear Data Sheets in Table 1 (1). The present results are consistent with those determined from a diversity of other measurements using a variety of techniques. However, it should be emphasized that the objective of the present energy determinations was limited to the correlation of the observed scattered neutron groups with known structures in ^{51}V which have been determined more precisely using other methods.

In addition to the above "primary states", neutrons were observed corresponding to excitation energies of 751±20, 1370±30, 2060±30, 2260±30, and 2862±50 keV. These excitation energies and the corresponding cross sections were very consistent with the excitation of the primary states by the secondary neutron group from the $^{7}Li(p,n)^{7}Be^{*}$ source reaction. Therefore, these "states" were attributed to an artifact of the source reaction. Any true scattered neutron component underlying these satellite neutron groups would have been of low intensity and not in agreement with previous knowledge of excited states in ^{51}V.

Intensities of scattered neutron groups were observed over the incident neutron range 1.8 to 4.0 MeV at intervals of $\lesssim$ 0.2 MeV. Minimum-energy scattered neutron sensitivities were $\gtrsim$ 0.5

MeV. Prominent inelastic-neutron groups were observed at up to forty scattering angles at each incident energy. Less intense groups, particularly at higher excitation energies, were observed at fewer angles, but a minimum of four at a given incident energy was required for a cross section determination. Generally, the inelastic-neutron groups were nearly isotropically distributed in angle, and there was no significant deviation from symmetry about ninety degrees as illustrated in Fig. 4. Each observed angular distribution was fitted with a low-order ($\leq$ 2) Legendre Polynomial series, using the method of least-squares, thereby providing angel-integrated values. The sharp resonance structure clearly evident in the high-resolution total cross section measurements, whould be even more evident in isolated exit channels. Moreover, systematic inelastic-neutron gamma ray studies (e.g., those of Smith (19)) clearly display fluctuations that are generally consistent with the present experimental results. In view of these fluctuations, cross sections obtained at energies identical within the experimental incident-energy resolution were averaged to obtain a more representative sample of the true value. Even with this procedure, considerable variations over small incident energy changes (e.g., 50 keV) can be expected.

The accuracies of the differential inelastic cross sections varied depending upon the particular experimental conditions and inelastic component involved. Under optimum conditions (e.g., the excitation of the prominent 1603-keV state by $\approx$ 3.0 MeV neutrons) the cross-section uncertainties were $\approx$ 5 percent, ap-

proximately equally divided between contributions from the counting statistics, the relative detector response and the normalization to the H(n,n) reference-standard reaction. These uncertainties were inclusive of contributions from data correction procedures such as multiple scattering corrections. The angle-integrated cross section values benefited from improved statistical accuracies and, in the best cases, were believed known to $\lesssim$ 5 percent accuracies. Uncertainties associated with minor inelastic components, particularly those with relatively high excitation energies, were much larger than the above (e.g. in the worst cases, 30-50 percent uncertainties). Moreover, no appreciable effort was made to extend sensitivities below a level of $\approx$ 1 mb/sr and this limit was a large fraction of some of the minor inelastic components. The cross section uncertainties increased as scattered neutron energies decreased below $\approx$ 1.5 MeV due primarily to detector efficiencies that changed rapidly with energy and fewer detectors having a reliable response to neutrons with energies of $\lesssim$ 0.5-1.0 MeV.

The present cross sections for the neutron excitation of the above states are summarized and compared with previously reported values in Fig. 5.

The cross sections for the 321 keV state were large throughout the measured incident-neutron energy range. However, at incident energies above 3.4 MeV, the measurements were complicated by the presence of the nearby elastic component due to the second

neutron-source group from the $^{7}Li(p,n)^{7}Be^{*}$ reaction. At these energies the secondary elastic and the primary inelastic groups could not be clearly resolved at all angles thus inelastic contributions were deduced by correcting the observed composite cross sections for the elastic contribution using the measured elastic scattering cross sections and the known relative intensities of the two source components (5). This correction procedure increased the uncertainties. The present results for the 321 keV state are in a very good agreement with those reported by Towle et al. (16) and generally consistent to within the respective uncertainties with those of Alman-Ramstrom (20) and of Holmqvist et al. (21). The lower-energy values extrapolate reasonably well to the work of Smith et al. (2). A number of cross section sets for the production of 321 keV gamma-rays by inelastic neutron scattering have been reported. These are not always easy to correlate with directly observed neutron values due to uncertain branching ratios. However, up to incident neutron energies of $\sim$ 1.6 MeV the only restriction of that type is due to the excitation of the 929 keV state, and the reported gamma-ray data were corrected for that contribution before making comparisons with the present results. Here, and generally throughout the comparisons with gamma-ray production cross sections, attention was given to possible effects of gamma-ray anisotropy. Unfortunately, it was not always possible to interpret reported experimental values in the context of such angular-distribution effects. The corrected gamma-ray results of Porter et al. (22), Barrows

(23), and Smith (19) were reasonably consistent with the present and other reported neutron cross sections results. As expected the gamma-ray results are larger than the neutron values at higher energies (above $\sim$ 1.6 MeV) due to uncorrected and uncertain branching-ratio affects. Some of the gamma-ray results provide more energy detail than the neutron measurements tending to confirm the presence of fluctuating structure. This is particularly so of the results of Smith (19).

The present cross sections for the excitation of the 938 keV state are in good agreement with those reported by Towle et al. (16) and by Cranberg and Levin (24). They are somewhat larger than the values given by Alman-Ramstrom (20) and Holmqvist et al. (21). The cross sections for the production of 938 keV gamma rays should reasonably correspond to the directly observed neutron cross sections due to the spin configurations of the first few excited states in ^{51}V. Indeed, the gamma-ray production cross sections reported by Barrows (23), and Smith (19) are consistent with the present values for the neutron cross sections. The gamma-ray cross section values of Porter et al. (22) tend to be somewhat lower.

Neutron cross sections for the excitation of the 1603- and the 1811-keV states obtained in the present work were in good agreement with those reported by Towle et al. (16), Alman-Ramstrom (20), Cranberg and Levin (24) and Holmqvist et al. (21). The present neutron cross sections were further supported by the re-

sults of the inelastic neutron gamma-ray studies of Porter et al. (22) and Barrows (23). Any difference between neutron and gamma-ray results could easily be attributed to uncertainties in gamma-ray branches and/or the angular distribution of emitted gamma-rays.

Neutrons corresponding to the excitation of a 2409 keV state were clearly evident in the present work. In addition, a neutron component corresponding to the excitation of a state at $\sim$ 2500 keV was observed in a few measurements. The cross sections for the latter state were very small (of the order of a mb/sr) and qualitatively consistent with the small values reported by Alman-Ramstrom (20). However, since the present work was primarily oriented toward applied needs the cross sections for the excitation of the 2409 and 2500 keV were combined into a single value. Even these collective cross sections, were relatively small. They were in good agreement with the similar composite quantity derived from the work of Alman-Ramstrom (20) and reasonably extrapolated to the results of Perey and Kinney (25) at 4.19 MeV and above.

Neutrons corresponding to the excitation of a state at 2706 keV were attributed to the combined contributions from previously reported states at 2675 and 2699 keV. There were only a limited number of results at a given incident energy, and as a consequence, the uncertainties associated with these values were relatively large. However, the present results are consistent with those of Alman -Ramstrom (20) and with an extrapolation of the values of Perey and Kinney (25). At incident energies of 4.0 MeV there were

two occasions when a neutron group corresponding to a state of 2773 keV was observed. This energy was not well determined, but it was consistent with the state reported at 2790 keV. The corresponding cross sections were very small, similar to the few mb/sr values reported by Alman-Ramstrom (20), and quantitative values were not deduced due to their uncertainty and their dependence upon rapidly changing scattered-neutron detector sensitivities.

IV. CALCULATIONAL MODELS

Here the objective was the development of an energy-averaged calculational model suitable for extrapolating and interpolating measured values particularly as required for the construction of the evaluated data file. The interpretation was based upon the conventional spherical-optical and statistical models (26,27). Initial estimates of the optical-model parameters were obtained from χi-square fitting of the differential elastic distributions determined in the present experiments. The fitting procedures were applied to all measured distributions from incident energies of 1.8 to 4.0 MeV. Compound-nucleus processes were calculated using the Hauser-Feshbach formula with width fluctuation corrections (28), and the previously reported spectroscopic properties of states to excitations of approximately 3.2 MeV as given in the Nuclear Data Sheets (1). At incident energies above approximately 3.0 MeV, the observed fluctuations in the elastic distributions became small and the energy-averaged model was appropriate. However, at these same

energies the knowledge of the contributing compound-nucleus channels was uncertain and as a consequence it was difficult to calculate quantitatively the still appreciable compound-elastic component. Therefore the model parameters were selected from fitting procedures at energies of 3.2 MeV or less where the calculations were more definitive. As the incident energy decreases from 3.0 MeV the fluctuations in the observed distributions increase with a consequent impact upon the deduced parameters. The fit to any isolated distribution can be good, as illustrated in Fig. 6, but the resulting potential parameters may differ from those obtained from a neighboring distribution by relatively large amounts. Parameters obtained from a single distribution may not be representative of the average, and even be deceptive. Thus the selected parameter set was obtained from a simple average of the parameters deduced from the fits to the individual distributions to incident energies of 3.2 MeV. The resulting selected parameter set is given in Table 2. The uncertainties associated with the respective parameters were derived from the RMS deviation of the individual values from the mean. In all, 15 individual distributions and fits were involved spanning nearly 1.5 MeV and thus the average parameters should be reasonably representative of the energy-averaged optical potential in the region 2.0 to 3.0 MeV.

Neutron total cross sections calculated with the above potential were reasonably descriptive of the measured values to 20.0 MeV as illustrated in Fig. 7. The largest discrepancy between

calculation and measurement was near 5.0 MeV, amounting to about 6 percent. The use of energy dependent real and imaginary potentials as indicated by wider-scope analysis, did not appreciably alter this discrepancy (29). The introduction of volume absorption with increasing energy as noted, for example, by Engelbrecht and Fiedeldey (29) may be an additional factor but could not be verified by the present measurements. Calculated elastic scattering distributions at approximately 8.0 MeV give a good description of the measured values reported by Holmqvist and Wiedling (17) and by Perey and Kinney (25). Moreover, the calculated ℓ_0 strength function was 4.5×10^{-4}, in good agreement with that reported from the analysis of low energy neutron resonances (30). Generally, the potential of Table 2 is very similar to that of Holmqvist and Wiedling (17), the main difference being a larger radius in the absorption term of the present potential. The present potential is different from that deduced by Moldauer (31) from lower energy phenomena. The latter has smaller real-potential strengths and larger radii than the potential of Table 2. There tends to be a characteristic difference between potentials in this mass region at energies of 1.0 to 2.0 MeV and, indeed, on occasion the present fitting procedures at the lower-energy extreme of the experimental range resulted in potential parameters very similar to those proposed by Moldauer and quite different from those obtained from the body of the measurements at somewhat higher energies.

The above potential and the Hauser-Feshbach formula, with and without width fluctuation corrections, were used to calculate the inelastic excitation cross sections (28,32). The potential was not adjusted to improve the description of the inelastic scattering processes. The spectroscopic character of states to excitations of 2.75 MeV is reasonably known with the additional identification of some states to excitations of 3.5 MeV and above (1). Thus the calculations should be reasonably reliable to about 3.5 MeV but less definitive at higher energies as the uncertainties associated with contributing channels increases. These calculational results are compared with the measured values in Fig. 5. From this figure it is clear that comparisons with the calculated results must be made in an energy-averaged context and not at isolated energy points. Both measured and calculated neutron inelastic scattering distributions are very nearly isotropic as illustrated in Fig. 4.

An average of the cross sections for excitation of the 321 keV state tends to lie approximately midway between results calculated with and without width fluctuation corrections. This suggests that there is a channel-channel correlation enhancement of this cross section and use of the "Q-parameter" of Moldauer resulted in better agreement with experiment. However, this parameter adjustment is a simple approximation of the complexity of effects contained in the "M-matrix" which are not readily calculable (28,3). Therefore the Q-parameter adjustment was not pursued. In view of these uncertainties, the well

established 5/2- character of this state is consistent with the present measurements and calculations.

The calculated excitation of the 938 keV state tends to be 10 to 20 percent lower than the present experimental results. The 3/2- character of this state appears to be well established thus this small discrepancy is probably due to the inappropriateness of the model and/or calculational methods. The width-fluctuation-corrected calculated result is in good agreement with the measured excitation of the 1603 keV state and reasonably descriptive of the measured values for the 1811 keV state. These results tend to support the tentative J-π values of 11/2- and 9/2- given in Ref. 1. Alternate choices consistent with (n;n',γ) measurements proved less suitable. The observed cross sections for the composite excitation of the 2409- and 2545 keV states were relatively small (less than 80 mb). Calculations based upon the tentative J-π values of 3/2- and 1/2+, respectively, were consistent with the measured values, but due to the small size of the measured quantities and the experimental uncertainties, a quantitative test of spectroscopic assignments was judged unreliable. Cross sections for the observed 2706 keV "state" were attributed to the composite contributions from the 2675(3/2+) and 2699(1/2+) states. The calculated results were consistent with experimental values but the latter were small and uncertain and thus precluded detailed analysis of spectroscopic properties. The remaining state at 2773 keV was observed in

the present work at only one incident energy, 4.0 MeV. The corresponding measured cross section was only qualitative. For completeness, the calculations assumed that this state was 3/2- but this assumption was not justifiable from the present measurements. Alternate choices had little impact upon the results calculated for the excitation of lower-lying states.

V. EVALUATION

Experimental and calculational results were used to construct a comprehensive evaluated nuclear data file in the ENDF format (33). Primary emphasis was given to higher-energy cross sections extending to 20 MeV. The following data types were considered:

A. Resonance region.

B. Neutron total cross sections.

C. Neutron elastic scattering.

D. Neutron inelastic scattering.

E. Radiative neutron capture.

F. The (n;2n') process.

G. The (n;p) process.

H. The (n;α) process.

I. The (n;n',p) process.

J. The (n;n',α) process.

K. The (n;d), (n;t), (n;^{3}He) and (n;2p) processes.

L. Gamma-ray production processes.

The evaluation is based upon the present measured and calculated

results (above) and measured values reported in the literature, interpolated with theoretical estimates where necessary (34). Results were accepted for the evaluation process only when they could be reasonably documented. The literature search was inclusive of the major journals, the NNCSC files and reference materials cited in CINDA 76/77 (35). The file is elemental and primary attention is given to the predominant isotope ^{51}V (99.76% abundant). However, in some of the reaction-cross-sections, contributions from the minor isotope ^{50}V were a factor and were considered. Throughout, attention was given to conciseness and physical validity. Subsequent paragraphs of this Section address the various file components and the remarks are the reference documentation for the file.

A. Resonance Region

Some new measurements give better resolution of resonance structure in the $\lesssim$ 100 keV region. A preliminary evaluation was undertaken employing the Reich-Moore multi-level formalism. This approach was promising but in midstream the ENDF formats were changed deleting the Reich-Moore formalism. The Adler-Adler representation is an alternative but apparently no generalized conversion of Reich-Moore to Adler-Adler descriptions has been worked out. Other format options are not particularly suitable and have resulted in physically curious results when applied in other similar evaluations; e.g. large and fluctuating backgrounds. In view of this physical problem with ENDF formats and the

present emphasis on higher-energy processes the present evaluation describes the resonance region below 100 keV with a pointwise representation. This is considered an interim step until such time as the above format issues are resolved. The corresponding pointwise data as given in ENDF/B-IV (MAT-1196) at energies below 100 keV was explicitly used throughout. Some comparisons with newer data indicated some changes from that previous evaluation but not of sufficient impact to warrant a revision of the file in this region until the above format problems can be solved.

B. Neutron Total Cross Sections

Above 100 keV the evaluation represents the total cross sections with point values. The evaluation employs the data of Rohr et al. (36) over the energy range 100 to 220 keV. Considering the varying resolutions, the Rohr et al. cross section magnitudes are in reasonable agreement with the results of Rainwater et al. (37), of Whalen (38), and of Blair et al. (39). From 220 to 360 keV the evaluation is based upon the results of Whalen (38). That particular set of measurements did not have the resolution of the Rohr et al. (36) data (at the lower energy extreme) or of the data of Cierjacks et al. (40) (at the upper energy limit), but it apparently is the only set of reasonably good resolution data available in this energy region. The magnitudes of the Whalen cross sections are consistent with the broader resolution results of Blair et al. (39). However, the energy scale of the

Whalen results appears a few keV different from those of both Rohr et al. or Cierjacks et al. This discrepancy probably is not significant in most applications. The evaluation employs the results of Cierjacks et al. (40) throughout the energy range 0.36 to 20.0 MeV. Below 2.0 MeV their magnitudes are reasonably consistent with the lesser resolution results of Whalen (38), of Cabe (41), of Smith et al. (2) and of Blair et al. (39). Generally, the resolution of the Cierjacks et al. results appears superior. There are some differences in energy scales between the various results amounting to as much as 5 keV. These energy discrepancies appear to be rather random. From 2.0 to 4.0 MeV the Cierjacks et al. results are consistent with those of the present work, of Foster and Glasgow (42), of Kent et al. (43), of Manero (44), and of Sorriaux (45). There is similar agreement with the results of Galloway et al. (46) above 2.5 MeV, although the Galloway et al. values tend to be lower than the Cierjacks results in the range 2.0 to 2.5 MeV. The results of Bennett et al. (47) and of Thibault et al. (48) appear anomalous and were abandoned. From 4.0 to 20.0 MeV, the values of Cierjacks et al. are in good agreement with those of Galloway et al., of Foster and Glasgow, of Manero, and of the present work. An exception is the region from 8.0 to 10.0 MeV where the results of Refs. 42 and 46 tend to lie on alternate sides of the Cierjacks values. The differences are only a few percent. In addition, the 14.0 MeV values of Goodman (49) and Lasday (50) are consistent with the Cierjacks et al. results.

The evaluation was constructed from the above data base in two manners. Below 6.0 MeV the selected data sets were plotted on a large scale. From these graphical displays measured values were selected in sufficient detail to well describe the fluctuating structure. Above 6.0 MeV the data base is a relatively smooth function of energy. Therefore, energy-averages of the measured values were constructed starting with an increment of approximately 20 keV at 6.0 MeV, increasing to approximately 100 keV at 20.0 MeV. The final result is compared with that of ENDF/B, MAT-1196 in Fig. 8. The two evaluations do not appreciably differ. This is not surprising as both place considerable reliance upon the data of Ref. 40. The energy-average magnitude uncertainties in the present evaluation were estimated to be less than 5 percent. However, it is clear that the structure is not fully resolved and that, particularly at energies below approximately 360 keV, appreciable increases in the structure may be expected as the result of new and higher resolution measurements at some future date. In addition, it is not clear to what extent sample-size effects may have perturbed the various experimental values. These will be most prominent below $\sim$ 2.0 MeV and in some cases it is suspected that rather thick samples have been used. If so, the magnitude of the average cross sections may have been underestimated.

C. Elastic Neutron Scattering

The relative elastic scattering distributions were derived from experimental results, interpolated using the above model

where necessary. Below 300 keV the energy-averaged model was used. This should give an average trend of the distributions although it obviously will not follow the resonance structure of the cross section. Experimental knowledge of these fluctuating distributions appears unknown. From 0.3 to 1.0 MeV the experimental results of Smith et al. (2) were used, averaged over $\approx$ 100 keV increments. Such an average should reasonably outline the intermediate resonance phenomena while at the same time avoiding the pronounced fluctuations due to one or several isolated distributions that may not be representative of the energy increment as a whole. From 1.0 to 1.5 MeV, a 200 keV average of the Smith et al. results was used. Between $\approx$ 1.5 and 1.8 MeV the 1.610 MeV value of Towle et al. was used (16). Over the range 1.8 to 4.0 MeV the present measurements were employed using a 200 keV average. Considering the fluctuations involved, they are in agreement with previous values and are the most comprehensive set of data in this region. From 4.5 through 8.5 MeV the measured values of Perey and Kinney and Holmqvist and Wiedling are used at $\approx$ 1-MeV intervals (e.g., 4.5, 5.5, 6.5, 7.5 and 8.5 MeV) (25,17). In interpreting these experimental values "Wick's Limit" was used at zero degrees and the model at 180 deg. as necessary to assure a good Legendre polynomial fit to the data at back angles. Above 8.5 MeV, the evaluation relies upon the above model. The results are consistent with the values measured at 14.7 by Western et al. (51). Throughout, consistency with

"Wick's Limit" was assured up to energies of 3.0 MeV. The magnitudes of the elastic scattering cross sections were set by the total cross section and non-elastic-channel contributions. Above 3.0 MeV the continuum inelastic scattering cross section increases and is uncertain. Thus the elastic-scattering cross section was set by the above distributions and the non-elastic cross section determined from this and the total cross section. The various particle-emission cross sections are not always certain (see below) but they can be reasonably determined or are relatively small. The continuum inelastic scattering was treated as a free parameter to assure consistency between the partial cross sections and the total cross section.

The above procedures introduced various energy-dependent-elastic-scattering-cross-section uncertainties. A "guideline" is $\approx$ 5 percent uncertainty in the energy-average elastic scattering magnitude to $\approx$ 8 MeV; increasing to $\approx$ 10 percent at 20 MeV.

D. Inelastic Neutron Scattering

1. Discrete Inelastic Neutron Scattering Cross Sections

The evaluation considers the inelastic neutron excitation of the seven "states" defined in Table 3. The excitation energies are taken from Ref. 1. The first four and the seventh inelastic neutron groups are uniquely associated with individual excited states. The fifth and sixth groups are composites of the excitations of several states as noted in the Table.

These composites are consistent with the resolutions generally available in the determination of the respective scattering cross sections. The excitation of states above 3.08 MeV is included in the continuum component as the respective cross sections are poorly known.

The evaluated excitation cross sections are based upon the present and previously reported experimental measurements interpolated, particularly toward thresholds, using model calculations. The majority of the data base is illustrated in Fig. 5 and extends upward to 8.5 MeV primarily via the work of Perey and Kinney (25). The evaluated cross sections were subjectively constructed from the experimental data base attempting to portray the more qualitative features of the fluctuating structure where possible. The correlation between measured values and evaluation is indicated in Fig. 5. Above 8.5 MeV there is no experimental information and thus the evaluation is smoothly extrapolated to a 20 MeV value. This extrapolation qualitatively represents some of the relatively small direct-excitation cross sections that must be present. The emitted-neutron angular distributions are taken to be isotropic. This is a good assumption up to several MeV where the cross sections are large as illustrated in Fig. 4. Experimental values at 8.5 MeV remain reasonably isotropic but at higher energies the distributions doubtless become forward peaked. The omission of this latter effect will have little impact on many applications of the file.

The ENDF/B-IV file contains cross sections for the excitation of only the first four states. These are compared with the present evaluation in Fig. 9. The energy-averages of the two evaluations for these states are very similar. It is estimated that the present discrete-cross-section evaluation is known to within about 10 percent for the more prominent cross sections. This estimate is supported by the agreement between the energy-averaged results of the present work and those of ENDF/B-IV though the two evaluations were derived quite independently.

2. Continuum Inelastic Scattering Distributions

As noted above, this cross section is constructed from the non-elastic cross section and the sum of the other reaction cross sections in such a manner as to assure internal consistency of the file. The primary competition is from the (n;2n) reaction which is an appreciable portion of the non-elastic cross section at higher energies. Competition from other channels is relatively minor until 20 MeV is approached.

The evaluation below 10 MeV is primarily based upon the measured values of Perey and Kinney (25). Spectra were observed for incident energies between 5.5 and 8.5 MeV. They are isotropically distributed and can generally be described by a Maxwellian temperature distribution with T= $\sim$ 1.5 MeV. The energy dependence of T is consistent with a $\sqrt{E}$ behavior although this is not well defined by the limited incident-energy range of the measurements. Superimposed upon the general temperature dependence is consider-

able structure. The evaluation generally describes the measured values with the temperature distribution upon which is qualitatively superimposed the gross features of the observed structure. Extrapolation from 5.5 MeV to threshold and from 8.5 to 10.0 MeV is via a simple-temperature distribution.

Above 10.0 MeV the evaluation is deduced from the experimental results of Hermsdorf et al. (52). These authors report detailed emission-spectrum measurements at an incident energy of approximately 14.5 MeV at a number of reaction angles extending from 40 to 150 deg. The spectra are characterized by a soft component that can be primarily attributed to the (n;2n') process, an extensive intermediate component and a high energy very-hard component associated with pre-compound processes. The latter is very strongly angular dependent with anisotropies of more than an order of magnitude at some energies. The lower energy portions of the distribution become nearly isotropic. This type of energy-angle correlation of the spectrum appears to require file-6 representation (33). Unfortunately, that format is not recommended and compromises must be made in order to describe the information in the more acceptable file-4 and -5 formats. The compromise consisted of averaging the Hermsdorf et al. data over angle and assuming general isotropic emission. This is a reasonable approximation for emission energies of less than approximately 6 MeV but is very inappropriate at energies above 10 MeV. The consequences of these compromises on the calculation of deep-material penetration of high energy neutrons

may be very serious in some cases and the user is warned to beware. The problem seems unavoidable until such time as file-6 or equivalent formats come into general use. Using the above angle average of the 14-MeV results of Hermsdorf et al. the general distribution is reasonably represented by a Maxwellian temperature of T= ∿ 2.5 MeV with the addition of the hard pre-compound component. The softer temperature component was ignored as being due to the (n;2n') process. The evaluation smoothly extrapolates this distribution to the 10 MeV value based upon the Perey and Kinney measurements (above) and upward to 20 MeV. The result is given as a set of pointwise distributions. In view of the above approximations, it is not clear that uncertainty estimates are warranted.

E. Radiative Neutron Capture

Despite the fact that measurements have been made at incident neutron energies above 0.1 MeV for nearly forty years these cross sections are not well known. The uncertainties may be attributed to the relatively small size of the cross sections. The present evaluation is based on the data of Refs. 53-64. Most of the measured values were obtained with activation techniques with a few additional sets of scintillation-tank results. Some of the results were available only from small journal figures. These were read with reasonable care but the corresponding numerical values remain somewhat qualitative. The data base is summarized in Fig. 10. One value, from Ref. 60, was widely discrepant and chronologically very early therefore it was omitted. There

are obvious variations in the measured values at lower energies. However, this is a region of sharply fluctuating cross sections where measured results can be very sensitive to the exact energy scale and experimental resolution, thus none of the lower energy values were rejected.

The evaluation was obtained subjectively via a graphical estimate of the energy-averaged behavior of the cross section with the results shown in Fig. 10. The data suggests cusps in the regions corresponding to the onset of prominent inelastic-scattering cross sections, and this feature was retained in the evaluation. The cross section calculated from the general reaction model was qualitatively consistent with the experimentally-based evaluation, thereby supporting the general validity of the calculational model. At lower energies, the present evaluation is pronouncedly higher than that of ENDF/B-IV. The latter is not particularly consistent with the energy-averaged trends of the experimental results in this region nor with the theoretical estimates. Above several MeV, the present evaluation is larger than that of ENDF/B-IV and more consistent with reported experimental values. In view of Fig. 10, there is considerable uncertainty associated with the present evaluation. Some energy-average guidelines are: 15-25 percent from 0.1-0.3 MeV, 10-20 percent from 0.3-3.0 MeV and more than 15 percent above 3.0 MeV.

F. The V(n;2n') Reaction, Q= -11.052 MeV

The product nucleus is stable prohibiting activation measurements. A scintillator measurement at 14.1 MeV by Ashby et al. (65) gives a result of 660 mb. The same authors give a Cu(n;2n) value considerably larger than that given in the ENDF/B-IV dosimetry file (66). Therefore the vanadium value of Ref. 65 was renormalized to the ENDF/B copper value to give a cross section of 523 mb. Recently Gaussig et al. (67) have reported measured values in the range 11.4 to 14.8 MeV. The results are relative to the ^{238}U fission cross section as given in AERE-R-7273 which is not greatly different from that of ENDF/B-IV. This limited data is the basis for the evaluation to 15 MeV as shown in Fig. 11. The relative energy dependence of the cross section is consistent with the results of calculations and the latter were used to guide the extrapolation to 20 MeV together with the analogous $^{59}Co(n;2n)$ and $^{55}Mn(n;2n)$ cross sections. The evaluation is qualitatively consistent with other theoretical estimates (68,69). The uncertainty associated with the evaluation is estimated to be $\approx$ 10 percent below 16 MeV and larger at higher energies. The emission spectrum was assumed to be a temperature distribution with the temperature consistent with the results of 14 MeV neutron emission measurements (see Section V-D, above). The present evaluation is considerably smaller in cross section magnitude than that of ENDF/B-IV and the difference is probably significantly beyond the respective uncertainties.

G. The V(n;p) Reaction, $Q(^{51}V)=-1.675$ and $(^{50}V)=+3.000$ MeV

The ^{51}V reaction is suitable for study by activation techniques although the combination of relatively small cross section and short product half-life (5.8 min.) makes the measurements difficult. Most experimental values were obtained near an incident energy of $\approx$ 14 MeV as outlined in Refs. 70-85. For the evaluation these measured results were referenced to a consistent set of standards as defined in Ref. 66. All of the measured values are for the $^{51}V(n;p)$ reaction. Apparently, no experimental values are available for energies appreciably below 14 MeV except in-pile results which were not used. The measured values of Mitra and Ghose (75), Borman et al.(73), Crumpton (82) and Robertson et al. (85) were averaged to obtain the cross section at 14.5 MeV. The data of Paul and Clarke (70), Poularikas and Fink (71), Bramlitt and Fink (72), Allan (74), Dressler et al. (76), Levkovskii et al. (77), Khurana and Govil (79), Strain and Ross (80), Pai and Clarke (83) and Prasad and Sarkar (84) were also considered but were not used in the evaluation because of various problems or inconsistencies in the results. Above 14.5 MeV the evaluation follows the results of Borman et al. (73) as they are reasonably consistent with other values at 14.5 MeV and give a definition of shape to 20 MeV. Near 14 MeV the evaluation is qualitatively consistent with the calculational estimates of Brown and Muirhead (78), Gardner and Rosenblum (81) and Pai and Clarke (83). The evaluation below approximately 13 MeV is based

entirely upon the present theoretical calculations. These included contributions from the $^{50}V(n;p)$ process. The latter, although of small magnitude, extend to zero energy with a relatively large contribution to the"sub-threshold"cross section region. The calculated results also qualitatively show the effect of the competition from the (n;n'p) process. The present evaluation is compared with that of ENDF/B-IV and with the data base in Fig. 12. The present evaluation is similar to that of ENDF/B-IV above approximately 14 MeV but at lower energies it is considerably smaller. The uncertainties in this low energy region are large (25-50 percent) but probably not as big as the differences between the two evaluations. At the experimentally-studied energies (above $\sim$ 13 MeV) the uncertainty in the present evaluation is estimated to be 10-20 percent.

H. <u>The V(n;α) Reaction, Q= -2.055 (^{51}V) and +0.759 (^{50}V) MeV</u>

The primary residual reaction product has a well known decay thus this process is amendable to study via activation techniques. In spite of this the experimental data is not particularly comprehensive or consistent. The present evaluation gives consideration to the data and calculations of Refs. 86 to 103. Most of this data is concentrated near 14 MeV and there appears to be no experimental information below 12 MeV. Several isolated energy measurements appeared to be anomalous and were not accepted. The data of Ref. 88 is of wide energy scope but appears to be high by approximately a factor of two. The data of Ref.

103 is in good agreement with other results near 14 MeV but is much lower at higher energies. In view of this divergent experimental information reliance was placed upon theoretical calculations inclusive of the contributions from the ^{50}V isotope. The calculated result was then subjectively adjusted to agree with the experimental data near 14 MeV where the measured values are most consistent. The normalization factor was 0.8. The evaluation is compared with the experimental data base in Fig.13. The evaluation is qualitatively consistent with the theoretical results of Savelev et al. (101), Gardner and Yu-Wen Lu (98) and Facchini et al. (97). It is also very similar to that given in ENDF/B-IV though the uncertainties in both evaluations are probably $\approx$ 20 percent.

I. <u>The V(n;n',p) Reaction, Q= -8.052 MeV</u>

There appears to be no experimental information on this process therefore the evaluation relies entirely upon theoretical calculations. The emitted neutron spectrum is assumed to be an isotropic temperature distribution. This assumption is a qualitative approximation in view of the pre-compound components that must be a contributing factor at higher energies. The present evaluation is qualitatively consistent with the theoretical predictions of Brown and Muirhead (78). This evaluation is compared with that of ENDF/B-IV in Fig. 14. There are obviously large differences in both shape and magnitude well beyond a reasonable uncertainty in the present calculation.

J. The V(n;n',α) Reaction, Q= -10.293 MeV

There are several energy-isolated measurements of this cross section near 14 MeV (90,104,105) and a more comprehensive set of experimental results reported by Borman et al. (106). The measurements are relatively consistent with one another. The evaluation is subjectively constructed through these measured values with the results shown in Fig. 15. The shape of the evaluation is different than predicted by theoretical calculation, being considerably larger near threshold. Both the calculations and the evaluation are larger than the values given in ENDF/B-IV. The uncertainty is approximately 20 percent providing the data of Ref. 106 is reliable as assumed. For many applications the uncertainties are not a concern as the cross section is small at 14 MeV and below. The emitted neutron spectrum is assumed to be an isotropic temperature distribution.

K. The V(n;d) Reaction, Q= -5.827 MeV

The primary product is the stable ^{50}Ti and the deuterons must be observed by direct particle detection. This is complicated by the angular distribution of the emitted particles and by difficulties associated with the detection of deuterons from transitions to states at higher excitations. Colli et al. (107, 108) have measured the 15 deg. cross section for the ground-state transition with the result of 0.48 mb/sr. Slaus et al. (109) have measured the angular distribution of the ground state transition. Combining these two measured results in a ground-state cross section of about 9 mb at 14 MeV. Ilakovac et al. (110)

have measured the cross sections for the ground and first three excited states with a 6 mb result at 14 MeV. These measured values suggest the total (n;d) reaction at 14 MeV is about 5-10 mb. The relative calculated excitation function was normalized by a factor of 6.6 to agree with this estimate and used for the evaluation. The present evaluation is much smaller than the comparable ENDF/B-IV cross sections below 18 MeV. There are large uncertainties in this evaluation but they will have little impact on most applications as the cross sections are very small below 14 MeV.

L. The V(n;t) Reaction, Q= -10.518 MeV

The primary product nucleus (^{49}Ti) is stable and there apparently have been no attempts to directly observe the emitted triton. A general survey by Khurana and Govil (111) suggests cross sections in the order of one milli barn at 14 MeV. The evaluated cross section relies upon a theoretically calculated shape renormalized by the same factor employed in the (n;d) reaction, above. The results are consistent with the Khurana and Govil estimate but considerably smaller than those of ENDF/B-IV. Both evaluations are only qualitative estimates and the cross sections in both cases are so small as to be of little concern in most applications.

M. The V(n;^{3}He) Reaction, Q= -12.505 MeV

Kunabe et al. (112) and Bramlitt et al. (113) report measured cross sections at approximately 14 MeV of less than 1 mb. Theoretical calculations were consistent with these measured

values. These small cross sections combined with the large negative Q-value makes the reaction of little applied interest. Therefore, it was ignored in the present evaluation.

N. The V(n;2p) Reaction, Q= -14.153 MeV

Measurements near 14 MeV by Bramlitt and Fink (90) and Lukic et al. (114) indicate cross sections of less than 30 and 60 micobarns, respectively. These small values are qualitatively supported by theoretical estimates. The very small cross sections combined with the high threshold makes this reaction of little applied importance. Thus, this evaluation ignores this component.

O. Photon Production Cross Sections

1. Photon Production from the Neutron Capture Process

The multiplicity and photon spectra to be applied to the (n,γ) reaction are taken from Rasmussen et al. (115) for thermal neutron energy. It is assumed that the spectrum is energy indepenent for energies less than 3.0 MeV. The multiplicity is adjusted to conserve photon energy. For incident neutron energies greater than 3.0 MeV, photon production from the neutron capture process is subsumed into the total photon production process.

2. Photon Production by Inelastic Scattering of Neutrons from Specific Levels

Branching ratios for photon decay from the levels V^{51} are not known for several of the levels for which level excitation functions have been given in this evaluation. It was therefore necessary to assume branching ratios for V^{51} and this was done as follows:

Level Energy	Transition	Probability	Photon Energy
.3201	0	1.	.3201
.9290	.3201	.16	.609
.9290	0	.84	.929
1.609	0	1.0	1.609
1.813	.32	.22	1.493
1.813	0	.78	1.813
2.409	.32	1.0	2.089
2.675	0	1.0	2.675
3.080	0	1.0	3.080

To facilitate comparison with experimental data the cross sections for producing these photons were entered into the evaluation explicitly rather than using multiplicities that would be applied to the level excitation functions. Only a rough check can be made against the experimental data reported by Newman and Morgan (116) because of the relatively broad neutron and photon resolution of the experiment. The check that could be made shows reasonable agreement between evaluation and experiment.

3. Photon Production from Continuum Neutron Inelastic Scattering and Other Reactions

For incident neutron energies greater than or equal to 3.0 MeV the R-Parameter method of Perkins, Haight and Howerton (117) was used with the R-Parameter values determined from the experimental data of Newman and Morgan (115). This method provides a means for extrapolation to lesser photon energies than the 0.26 MeV experimental cut-off and also insures conservation of energy between the neutron

interaction and photon production portions of the evaluation. Comparison between evaluated and experimental values shows good agreement.

VI. CONCLUDING REMARKS

The present measurements improve the knowledge of neutron total and elastic and inelastic scattering cross sections of vanadium to incident energies of 4.0 MeV. The scope of the measurements is sufficient to assure a good representation of the energy-average behavior of the cross sections over a region of strong fluctuations. The measurements do not define that structure in detail. An energy-averaged model, based upon the present measured values, was descriptive of the general behavior of measured cross section results to 20.0 MeV. Primary uncertainties associated with the model are attributed to an inability to calculate channel-correlation properties of the compound-nucleus component from basic theory. This shortcoming has a detrimental impact on the quantitative calculational estimate of the compound-nucleus contributions to both elastic and inelastic scattering cross sections. Within these calculation uncertainties, the correlation of the present measured and calculated results is consistent with previously reported characteristics of excited states in vanadium to $E_x \approx 3.0$ MeV.

The present measured values, those reported elsewhere and calculations were used to construct a comprehensive evaluated nuclear data file in the ENDF format. The primary uncertainties in this evaluated file are associated with: 1) the resonance

parameters where experimental results and evaluated file formats have shortcomings, 2) high energy neutron emission spectra where both the data base and file formats are deficient and 3) some of the poorly known (n;x) reactions where primary reliance must be placed upon calculational estimates. Despite these shortcomings, it is believed that the present evaluated file provides a considerably improved data base for the applied user.

ACKNOWLEDGEMENTS

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APPENDIX, **Numerical Evaluated Data File**

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TAPE FØR NATURAL VANADIUM
2.30000+ 4 5.05040+ 1          0          0          0          1  23 0  0      0
0.00000+ 0 0.00000+ 0          0          0          0          0  23 1451     1
0.00000+ 0 0.00000+ 0          0          0          4         48  23 1451     2
23-V -    ANLLLLHEDL EVAL-JAN77 A.SMITH+,H.HØWERTØN,F.MANN.        23 1451     3
ANL/NDM-24,1977       DIST-JAN77                                   23 1451     4
SEE REFERENCE FØR DØCUMENTATIØN                                    23 1451     5
                                                                   23 1451     6
                                                                   23 1451     7
                                1        451         49          1  23 1451     8
                                2        151          4          0  23 1451     9
                                3          1        583          1  23 1451    10
                                3          2        583          1  23 1451    11
                                3          4         87          1  23 1451    12
                                3         16          8          1  23 1451    13
                                3         22          6          1  23 1451    14
                                3         28         11          1  23 1451    15
                                3         51         14          1  23 1451    16
                                3         52         13          1  23 1451    17
                                3         53         10          1  23 1451    18
                                3         54          9          1  23 1451    19
                                3         55          7          1  23 1451    20
                                3         56          7          1  23 1451    21
                                3         57          7          1  23 1451    22
                                3         91         76          1  23 1451    23
                                3        102         49          1  23 1451    24
                                3        103         24          1  23 1451    25
                                3        104         15          1  23 1451    26
                                3        105          7          1  23 1451    27
                                3        107         19          1  23 1451    28
                                3        203         27          1  23 1451    29
                                3        207         19          1  23 1451    30
                                3        251         13          1  23 1451    31
                                3        252         13          1  23 1451    32
                                3        253         13          1  23 1451    33
                                4          2        135          1  23 1451    34
                                4         16          2          1  23 1451    35
                                4         22          2          1  23 1451    36
                                4         28          2          1  23 1451    37
                                4         51          2          1  23 1451    38
                                4         52          2          1  23 1451    39
                                4         53          2          1  23 1451    40
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                                4         55          2          1  23 1451    42
                                4         56          2          1  23 1451    43
                                4         57          2          1  23 1451    44
                                4         91          2          1  23 1451    45
                                5         16          7          1  23 1451    46
                                5         22          7          1  23 1451    47
                                5         28          7          1  23 1451    48
                                5         91        117          1  23 1451    49
                               12        102          4          1  23 1451    50
                               13          3        105          1  23 1451    51
                               14          3          1          1  23 1451    52
                               14        102          1          1  23 1451    53
                               15          3        163          1  23 1451    54
                               15        102         42          1  23 1451    55
                                                                   23 1  0     56
                                                                   23 0  0     57
2.30000+ 4 5.05040+ 1          0          0          1          0  23 2151    58
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2.30510+ 4	1.00000+ 0	0	0	1	0	23	2151		59
1.00000- 5	1.00000+ 5	0	0	0	0	23	2151		60
3.50000+ 0	6.30800- 1	0	0	0	0	23	2151		61
						23	2	0	62
						23	0	0	63
.23000E 05	.50504E 02	0	99	0	0	23	3	1	64
.00000E 00	.00000E 00	0	0	2	1719	23	3	1	65
225	5	1719	2			23	3	1	66
.10000E-04	.25951E 03	.10000E-03	.85480E 02	.50000E-03	.40993E 02	23	3	1	67
.10000E-02	.30451E 02	.20000E-02	.22997E 02	.40000E-02	.17726E 02	23	3	1	68
.60000E-02	.15390E 02	.80000E-02	.13998E 02	.10000E-01	.13048E 02	23	3	1	69
.25300E-01	.10060E 02	.40000E-01	.90242E 01	.60000E-01	.82857E 01	23	3	1	70
.80000E-01	.78455E 01	.10000E 00	.75450E 01	.20000E 00	.67997E 01	23	3	1	71
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.10000E 02	.52545E 01	.40000E 02	.51283E 01	.10000E 03	.56000E 01	23	3	1	74
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.16700E 03	.12800E 02	.16800E 03	.11967E 02	.16900E 03	.10964E 02	23	3	1	78
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.34000E 04	.30000E 02	.36000E 04	.50000E 02	.38000E 04	.10693E 03	23	3	1	88
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.23260E	06	.37400E	01	.23360E	06	.70100E	01	.23560E	06	.65700E	01	23	3	1	177
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.24460E	06	.73800E	01	.24660E	06	.70700E	01	.24760E	06	.69700E	01	23	3	1	179
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.93890E 06	.27900E 01	.94020E 06	.31700E 01	.94240E 06	.31300E 01	23	3	1	393
.94460E 06	.34700E 01	.94680E 06	.29700E 01	.94720E 06	.34700E 01	23	3	1	394
.94860E 06	.24700E 01	.94990E 06	.24189E 01	.95170E 06	.22400E 01	23	3	1	395
.95260E 06	.26200E 01	.95350E 06	.34000E 01	.95660E 06	.36200E 01	23	3	1	396
.95840E 06	.35300E 01	.96030E 06	.27600E 01	.96250E 06	.23600E 01	23	3	1	397
.96480E 06	.24000E 01	.96620E 06	.28500E 01	.96750E 06	.35300E 01	23	3	1	398
.96890E 06	.38900E 01	.96980E 06	.40700E 01	.97000E 06	.44700E 01	23	3	1	399
.97120E 06	.44000E 01	.97260E 06	.42486E 01	.97310E 06	.33400E 01	23	3	1	400
.97350E 06	.25100E 01	.97490E 06	.23900E 01	.97580E 06	.25100E 01	23	3	1	401
.97720E 06	.36299E 01	.97860E 06	.38699E 01	.98100E 06	.39200E 01	23	3	1	402
.98280E 06	.36000E 01	.98380E 06	.31300E 01	.98610E 06	.28600E 01	23	3	1	403
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.99610E 06	.26799E 01	.99660E 06	.28000E 01	.99800E 06	.31499E 01	23	3	1	406
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.10030E 07	.25767E 01	.10050E 07	.27700E 01	.10090E 07	.38600E 01	23	3	1	408
.10120E 07	.28599E 01	.10150E 07	.28999E 01	.10180E 07	.37399E 01	23	3	1	409
.10250E 07	.40299E 01	.10300E 07	.29100E 01	.10360E 07	.21699E 01	23	3	1	410
.10390E 07	.23000E 01	.10420E 07	.21299E 01	.10440E 07	.28599E 01	23	3	1	411
.10460E 07	.27199E 01	.10490E 07	.29300E 01	.10520E 07	.21199E 01	23	3	1	412
.10570E 07	.23799E 01	.10590E 07	.20699E 01	.10650E 07	.30700E 01	23	3	1	413
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.10950E 07	.38300E 01	.10960E 07	.31500E 01	.10980E 07	.35600E 01	23	3	1	417
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.11200E	07	.38899E	01	.11230E	07	.30699E	01	.11250E	07	.32500E	01	23	3	1	421
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.11330E	07	.34599E	01	.11340E	07	.33299E	01	.11370E	07	.36699E	01	23	3	1	423
.11400E	07	.36699E	01	.11420E	07	.32900E	01	.11440E	07	.28100E	01	23	3	1	424
.11460E	07	.31700E	01	.11470E	07	.43999E	01	.11500E	07	.45400E	01	23	3	1	425
.11510E	07	.42000E	01	.11520E	07	.26200E	01	.11540E	07	.23300E	01	23	3	1	426
.11560E	07	.23400E	01	.11590E	07	.28099E	01	.11610E	07	.28000E	01	23	3	1	427
.11640E	07	.30399E	01	.11660E	07	.32300E	01	.11690E	07	.38499E	01	23	3	1	428
.11700E	07	.33500E	01	.11720E	07	.26099E	01	.11730E	07	.25400E	01	23	3	1	429
.11760E	07	.30733E	01	.11780E	07	.33399E	01	.11800E	07	.38100E	01	23	3	1	430
.11820E	07	.37099E	01	.11850E	07	.35300E	01	.11870E	07	.36300E	01	23	3	1	431
.11900E	07	.41400E	01	.11940E	07	.37800E	01	.11960E	07	.33700E	01	23	3	1	432
.11990E	07	.29299E	01	.12000E	07	.27800E	01	.12020E	07	.33899E	01	23	3	1	433
.12050E	07	.34833E	01	.12090E	07	.36700E	01	.12120E	07	.32600E	01	23	3	1	434
.12150E	07	.30599E	01	.12180E	07	.35000E	01	.12220E	07	.30500E	01	23	3	1	435
.12240E	07	.27099E	01	.12260E	07	.25300E	01	.12290E	07	.28399E	01	23	3	1	436
.12310E	07	.34100E	01	.12330E	07	.34000E	01	.12350E	07	.31299E	01	23	3	1	437
.12370E	07	.32200E	01	.12400E	07	.36399E	01	.12440E	07	.38300E	01	23	3	1	438
.12460E	07	.36200E	01	.12480E	07	.26200E	01	.12490E	07	.28599E	01	23	3	1	439
.12500E	07	.32000E	01	.12520E	07	.36899E	01	.12550E	07	.37999E	01	23	3	1	440
.12580E	07	.37000E	01	.12620E	07	.35300E	01	.12650E	07	.39600E	01	23	3	1	441
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.12750E	07	.26099E	01	.12770E	07	.24400E	01	.12790E	07	.22799E	01	23	3	1	443
.12810E	07	.22900E	01	.12840E	07	.30200E	01	.12890E	07	.34199E	01	23	3	1	444
.12930E	07	.33600E	01	.12960E	07	.28999E	01	.12970E	07	.24799E	01	23	3	1	445
.12980E	07	.31199E	01	.13000E	07	.39300E	01	.13030E	07	.41300E	01	23	3	1	446
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.13100E	07	.25100E	01	.13120E	07	.25400E	01	.13160E	07	.31099E	01	23	3	1	448
.13180E	07	.35100E	01	.13190E	07	.37700E	01	.13220E	07	.36099E	01	23	3	1	449
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.13310E	07	.33199E	01	.13350E	07	.32300E	01	.13370E	07	.31900E	01	23	3	1	451
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.13710E	07	.37000E	01	.13740E	07	.35300E	01	.13770E	07	.31500E	01	23	3	1	455
.13810E	07	.27700E	01	.13860E	07	.27700E	01	.13900E	07	.29500E	01	23	3	1	456
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.14010E	07	.31100E	01	.14030E	07	.29200E	01	.14040E	07	.26300E	01	23	3	1	458
.14060E	07	.24300E	01	.14100E	07	.24300E	01	.14120E	07	.25699E	01	23	3	1	459
.14140E	07	.32499E	01	.14180E	07	.33900E	01	.14190E	07	.31800E	01	23	3	1	460
.14210E	07	.27800E	01	.14220E	07	.27600E	01	.14250E	07	.30500E	01	23	3	1	461
.14260E	07	.40000E	01	.14280E	07	.38899E	01	.14300E	07	.33199E	01	23	3	1	462
.14320E	07	.30900E	01	.14340E	07	.33499E	01	.14370E	07	.33900E	01	23	3	1	463
.14400E	07	.30400E	01	.14420E	07	.30000E	01	.14440E	07	.29900E	01	23	3	1	464
.14460E	07	.31200E	01	.14470E	07	.28199E	01	.14500E	07	.27800E	01	23	3	1	465
.14540E	07	.34200E	01	.14560E	07	.38600E	01	.14590E	07	.37200E	01	23	3	1	466
.14620E	07	.36699E	01	.14650E	07	.41100E	01	.14690E	07	.41100E	01	23	3	1	467
.14730E	07	.41100E	01	.14760E	07	.42900E	01	.14780E	07	.40200E	01	23	3	1	468
.14810E	07	.37400E	01	.14840E	07	.38700E	01	.14860E	07	.39599E	01	23	3	1	469
.14880E	07	.38500E	01	.14900E	07	.34800E	01	.14950E	07	.33700E	01	23	3	1	470
.15000E	07	.34500E	01	.15030E	07	.30200E	01	.15070E	07	.30200E	01	23	3	1	471
.15120E	07	.30000E	01	.15130E	07	.34299E	01	.15150E	07	.34400E	01	23	3	1	472
.15180E	07	.33700E	01	.15200E	07	.32999E	01	.15240E	07	.28900E	01	23	3	1	473
.15280E	07	.31300E	01	.15310E	07	.35499E	01	.15330E	07	.36000E	01	23	3	1	474
.15370E	07	.34300E	01	.15400E	07	.35399E	01	.15440E	07	.34500E	01	23	3	1	475
.15490E	07	.32700E	01	.15510E	07	.30700E	01	.15540E	07	.31199E	01	23	3	1	476
.15580E	07	.33900E	01	.15620E	07	.31599E	01	.15660E	07	.31200E	01	23	3	1	477
.15700E	07	.31700E	01	.15720E	07	.35000E	01	.15740E	07	.34099E	01	23	3	1	478

.15780E	07	.29200E	01	.15800E	07	.28700E	01	.15850E	07	.31599E	01	23	3	1	479
.15890E	07	.28299E	01	.15910E	07	.34700E	01	.15950E	07	.31799E	01	23	3	1	480
.15990E	07	.32300E	01	.16000E	07	.41000E	01	.16010E	07	.41700E	01	23	3	1	481
.16050E	07	.35000E	01	.16080E	07	.41300E	01	.16100E	07	.41699E	01	23	3	1	482
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.16410E	07	.39800E	01	.16430E	07	.39800E	01	.16470E	07	.42300E	01	23	3	1	487
.16520E	07	.41801E	01	.16560E	07	.39500E	01	.16590E	07	.35700E	01	23	3	1	488
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.16950E	07	.37100E	01	.16980E	07	.35600E	01	.17000E	07	.39700E	01	23	3	1	492
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.17310E	07	.28700E	01	.17340E	07	.29400E	01	.17360E	07	.36100E	01	23	3	1	497
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.19470E	07	.36700E	01	.19510E	07	.36001E	01	.19550E	07	.36200E	01	23	3	1	519
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.24600E 07	.41699E 01	.24690E 07	.39899E 01	.24810E 07	.36100E 01	23	3	1	547
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.15000E	06	.86720E	00	.15060E	06	.46782E	00	.15160E	06	.36979E	01	23	3	2	734
.15290E	06	.90670E	01	.15350E	06	.92600E	01	.15480E	06	.64160E	01	23	3	2	735
.15650E	06	.40461E	01	.15820E	06	.23141E	01	.15920E	06	.49322E	01	23	3	2	736
.16030E	06	.11323E	02	.16100E	06	.11453E	02	.16240E	06	.13213E	02	23	3	2	737
.16340E	06	.10403E	02	.16380E	06	.10323E	02	.16450E	06	.11303E	02	23	3	2	738
.16560E	06	.99684E	01	.16780E	06	.93844E	01	.16860E	06	.89925E	01	23	3	2	739
.17010E	06	.29445E	01	.17050E	06	.29065E	01	.17160E	06	.55386E	01	23	3	2	740
.17360E	06	.70506E	01	.17440E	06	.68056E	01	.17560E	06	.45757E	01	23	3	2	741
.17640E	06	.73237E	01	.17720E	06	.84117E	01	.17840E	06	.69758E	01	23	3	2	742
.17920E	06	.70358E	01	.18000E	06	.85408E	01	.18170E	06	.93679E	01	23	3	2	743
.18340E	06	.12204E	02	.18430E	06	.12304E	02	.18510E	06	.99370E	01	23	3	2	744
.18640E	06	.43550E	01	.18690E	06	.44270E	01	.18860E	06	.13714E	02	23	3	2	745
.18950E	06	.13624E	02	.19090E	06	.10494E	02	.19270E	06	.77192E	01	23	3	2	746
.19450E	06	.63663E	01	.19550E	06	.49863E	01	.19640E	06	.56093E	01	23	3	2	747
.19830E	06	.24534E	01	.19980E	06	.73194E	01	.20000E	06	.75073E	01	23	3	2	748
.20120E	06	.75073E	01	.20220E	06	.86344E	01	.20370E	06	.10194E	02	23	3	2	749
.20520E	06	.11024E	02	.20670E	06	.10354E	02	.20880E	06	.83825E	01	23	3	2	750
.21040E	06	.62665E	01	.21190E	06	.47056E	01	.21250E	06	.81246E	01	23	3	2	751
.21410E	06	.80756E	01	.21460E	06	.56896E	01	.21960E	06	.47996E	01	23	3	2	752
.22460E	06	.23677E	01	.22860E	06	.25248E	01	.22960E	06	.36948E	01	23	3	2	753
.23260E	06	.37348E	01	.23360E	06	.70049E	01	.23560E	06	.65649E	01	23	3	2	754
.23660E	06	.75449E	01	.23860E	06	.72749E	01	.24160E	06	.79349E	01	23	3	2	755
.24460E	06	.73750E	01	.24660E	06	.70650E	01	.24760E	06	.69651E	01	23	3	2	756
.25000E	06	.55951E	01	.25060E	06	.55471E	01	.25170E	06	.55351E	01	23	3	2	757
.25370E	06	.51751E	01	.25470E	06	.57251E	01	.25670E	06	.53251E	01	23	3	2	758
.25970E	06	.58751E	01	.26270E	06	.47152E	01	.26470E	06	.30552E	01	23	3	2	759
.26870E	06	.43052E	01	.27070E	06	.42952E	01	.27270E	06	.28952E	01	23	3	2	760
.27470E	06	.25352E	01	.27570E	06	.25752E	01	.27670E	06	.42953E	01	23	3	2	761
.28370E	06	.51253E	01	.28770E	06	.64853E	01	.28970E	06	.53453E	01	23	3	2	762
.29270E	06	.58453E	01	.29470E	06	.56454E	01	.29570E	06	.59254E	01	23	3	2	763
.29870E	06	.52554E	01	.29970E	06	.54554E	01	.30000E	06	.59654E	01	23	3	2	764
.30270E	06	.59724E	01	.30390E	06	.60354E	01	.30490E	06	.75954E	01	23	3	2	765
.30690E	06	.64754E	01	.30890E	06	.73554E	01	.31190E	06	.73254E	01	23	3	2	766
.31390E	06	.45754E	01	.31690E	06	.51854E	01	.31890E	06	.46254E	01	23	3	2	767
.32000E	06	.63954E	01	.32100E	06	.61911E	01	.32400E	06	.60054E	01	23	3	2	768
.32600E	06	.44455E	01	.32640E	06	.39955E	01	.32800E	06	.42115E	01	23	3	2	769
.32900E	06	.50703E	01	.33100E	06	.49971E	01	.33150E	06	.39506E	01	23	3	2	770
.33250E	06	.51990E	01	.33350E	06	.59157E	01	.33650E	06	.60125E	01	23	3	2	771
.33750E	06	.51728E	01	.34150E	06	.55895E	01	.34350E	06	.45966E	01	23	3	2	772
.34650E	06	.46301E	01	.35050E	06	.36803E	01	.35150E	06	.21774E	01	23	3	2	773
.35450E	06	.37841E	01	.35750E	06	.52144E	01	.35950E	06	.64747E	01	23	3	2	774
.36010E	06	.51282E	01	.36080E	06	.35862E	01	.36180E	06	.32739E	01	23	3	2	775
.36260E	06	.40707E	01	.36330E	06	.29681E	01	.36400E	06	.29358E	01	23	3	2	776
.36450E	06	.35036E	01	.36530E	06	.33419E	01	.36620E	06	.53094E	01	23	3	2	777
.36670E	06	.48464E	01	.36780E	06	.51848E	01	.36830E	06	.45012E	01	23	3	2	778

.36880E	06	.46296E	01	.36940E	06	.54980E	01	.37040E	06	.41961E	01	23	3	2	779
.37090E	06	.37528E	01	.37150E	06	.48412E	01	.37200E	06	.37992E	01	23	3	2	780
.37260E	06	.37176E	01	.37380E	06	.46957E	01	.37470E	06	.26718E	01	23	3	2	781
.37500E	06	.44989E	01	.37600E	06	.41517E	01	.37770E	06	.29946E	01	23	3	2	782
.37810E	06	.14191E	01	.37860E	06	.15878E	01	.37910E	06	.36956E	01	23	3	2	783
.38000E	06	.36946E	01	.38150E	06	.25217E	01	.38250E	06	.20968E	01	23	3	2	784
.38310E	06	.31636E	01	.38410E	06	.18816E	01	.38490E	06	.11184E	01	23	3	2	785
.38530E	06	.14958E	01	.38600E	06	.10745E	01	.38660E	06	.20022E	01	23	3	2	786
.38770E	06	.13403E	01	.38900E	06	.28867E	01	.39000E	06	.11025E	01	23	3	2	787
.39050E	06	.28392E	01	.39330E	06	.24876E	01	.39380E	06	.48285E	01	23	3	2	788
.39500E	06	.43569E	01	.39620E	06	.50130E	01	.39670E	06	.36191E	01	23	3	2	789
.39920E	06	.46175E	01	.40000E	06	.36994E	01	.40140E	06	.37041E	01	23	3	2	790
.40240E	06	.37154E	01	.40440E	06	.39644E	01	.40550E	06	.44924E	01	23	3	2	791
.40600E	06	.29914E	01	.40720E	06	.30709E	01	.40830E	06	.41697E	01	23	3	2	792
.40960E	06	.32586E	01	.41180E	06	.43973E	01	.41260E	06	.32951E	01	23	3	2	793
.41330E	06	.41343E	01	.41510E	06	.34136E	01	.41640E	06	.39519E	01	23	3	2	794
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.42140E	06	.13366E	01	.42240E	06	.18056E	01	.42370E	06	.14246E	01	23	3	2	796
.42470E	06	.20033E	01	.42590E	06	.28623E	01	.42770E	06	.27112E	01	23	3	2	797
.42900E	06	.33394E	01	.43000E	06	.28681E	01	.43110E	06	.28871E	01	23	3	2	798
.43200E	06	.32160E	01	.43360E	06	.28951E	01	.43390E	06	.49435E	01	23	3	2	799
.43440E	06	.49132E	01	.43510E	06	.42027E	01	.43700E	06	.45021E	01	23	3	2	800
.43800E	06	.40102E	01	.43900E	06	.24892E	01	.44050E	06	.15282E	01	23	3	2	801
.44150E	06	.14767E	01	.44250E	06	.18957E	01	.44290E	06	.49347E	01	23	3	2	802
.44390E	06	.59143E	01	.44490E	06	.36033E	01	.44650E	06	.27923E	01	23	3	2	803
.44770E	06	.26808E	01	.44820E	06	.35896E	01	.44870E	06	.48891E	01	23	3	2	804
.45000E	06	.57286E	01	.45060E	06	.51973E	01	.45070E	06	.51973E	01	23	3	2	805
.45160E	06	.48978E	01	.45220E	06	.49779E	01	.45560E	06	.43886E	01	23	3	2	806
.45670E	06	.49591E	01	.45790E	06	.39818E	01	.45970E	06	.46427E	01	23	3	2	807
.46080E	06	.36637E	01	.46210E	06	.63152E	01	.46410E	06	.47260E	01	23	3	2	808
.46640E	06	.39171E	01	.46730E	06	.34387E	01	.46790E	06	.29306E	01	23	3	2	809
.46870E	06	.28013E	01	.46980E	06	.29818E	01	.47130E	06	.23124E	01	23	3	2	810
.47200E	06	.21133E	01	.47210E	06	.18246E	01	.47290E	06	.15714E	01	23	3	2	811
.47370E	06	.15352E	01	.47490E	06	.20458E	01	.47590E	06	.14965E	01	23	3	2	812
.47670E	06	.23975E	01	.47760E	06	.39683E	01	.47860E	06	.28889E	01	23	3	2	813
.47920E	06	.22097E	01	.48000E	06	.29505E	01	.48080E	06	.36710E	01	23	3	2	814
.48200E	06	.37316E	01	.48310E	06	.39922E	01	.48420E	06	.36832E	01	23	3	2	815
.48540E	06	.36841E	01	.48620E	06	.31250E	01	.48700E	06	.36860E	01	23	3	2	816
.48760E	06	.31366E	01	.48850E	06	.30073E	01	.49040E	06	.24278E	01	23	3	2	817
.49210E	06	.27985E	01	.49240E	06	.32700E	01	.49310E	06	.27514E	01	23	3	2	818
.49390E	06	.28416E	01	.49490E	06	.25122E	01	.49590E	06	.31029E	01	23	3	2	819
.49680E	06	.22937E	01	.49750E	06	.31645E	01	.49830E	06	.27552E	01	23	3	2	820
.49910E	06	.27758E	01	.50000E	06	.32464E	01	.50020E	06	.21771E	01	23	3	2	821
.50170E	06	.15278E	01	.50380E	06	.14476E	01	.50460E	06	.80628E	00	23	3	2	822
.50550E	06	.62139E	00	.50720E	06	.14137E	01	.50900E	06	.31729E	01	23	3	2	823
.50970E	06	.43313E	01	.51040E	06	.42097E	01	.51090E	06	.47791E	01	23	3	2	824
.51120E	06	.39585E	01	.51250E	06	.35480E	01	.51300E	06	.35478E	01	23	3	2	825
.51370E	06	.36366E	01	.51500E	06	.33161E	01	.51600E	06	.24655E	01	23	3	2	826
.51660E	06	.20743E	01	.51710E	06	.26734E	01	.51820E	06	.35129E	01	23	3	2	827
.51940E	06	.38825E	01	.52050E	06	.35915E	01	.52110E	06	.30204E	01	23	3	2	828
.52230E	06	.29994E	01	.52320E	06	.32289E	01	.52430E	06	.27078E	01	23	3	2	829
.52540E	06	.28970E	01	.52650E	06	.19860E	01	.52890E	06	.15550E	01	23	3	2	830
.52930E	06	.12440E	01	.53020E	06	.11619E	01	.53080E	06	.13915E	01	23	3	2	831
.53170E	06	.22007E	01	.53250E	06	.19902E	01	.53320E	06	.27494E	01	23	3	2	832
.53380E	06	.31056E	01	.53470E	06	.28080E	01	.53570E	06	.26475E	01	23	3	2	833
.53600E	06	.29567E	01	.53680E	06	.25258E	01	.53750E	06	.19855E	01	23	3	2	834
.53830E	06	.15948E	01	.53870E	06	.19442E	01	.53940E	06	.30334E	01	23	3	2	835
.53980E	06	.29301E	01	.54040E	06	.16725E	01	.54120E	06	.11721E	01	23	3	2	836
.54190E	06	.83156E	00	.54290E	06	.10308E	01	.54350E	06	.19002E	01	23	3	2	837
.54420E	06	.24093E	01	.54480E	06	.22608E	01	.54520E	06	.31982E	01	23	3	2	838

.54600E	06	.34276E	01	.54620E	06	.31473E	01	.54700E	06	.24365E	01	23	3	2	839
.54760E	06	.23364E	01	.54810E	06	.23656E	01	.54870E	06	.21151E	01	23	3	2	840
.54950E	06	.14947E	01	.55010E	06	.13041E	01	.55190E	06	.15234E	01	23	3	2	841
.55230E	06	.12429E	01	.55340E	06	.12112E	01	.55420E	06	.14609E	01	23	3	2	842
.55480E	06	.11899E	01	.55540E	06	.16592E	01	.55700E	06	.22586E	01	23	3	2	843
.55760E	06	.25081E	01	.55820E	06	.17867E	01	.55840E	06	.17661E	01	23	3	2	844
.55860E	06	.20956E	01	.55880E	06	.20254E	01	.55900E	06	.19252E	01	23	3	2	845
.56020E	06	.16951E	01	.56150E	06	.14649E	01	.56200E	06	.77380E	00	23	3	2	846
.56290E	06	.12026E	01	.56410E	06	.10379E	01	.56470E	06	.74138E	00	23	3	2	847
.56570E	06	.14203E	01	.56700E	06	.14198E	01	.56780E	06	.12689E	01	23	3	2	848
.56860E	06	.17677E	01	.56880E	06	.25370E	01	.57010E	06	.41363E	01	23	3	2	849
.57130E	06	.43661E	01	.57260E	06	.37549E	01	.57420E	06	.44038E	01	23	3	2	850
.57460E	06	.44227E	01	.57530E	06	.41612E	01	.57670E	06	.36409E	01	23	3	2	851
.57760E	06	.28203E	01	.57990E	06	.15790E	01	.58160E	06	.14882E	01	23	3	2	852
.58310E	06	.24561E	01	.58360E	06	.30746E	01	.58530E	06	.36933E	01	23	3	2	853
.58590E	06	.35126E	01	.58680E	06	.15613E	01	.58770E	06	.16807E	01	23	3	2	854
.58980E	06	.28699E	01	.59070E	06	.35891E	01	.59180E	06	.42672E	01	23	3	2	855
.59290E	06	.39264E	01	.59380E	06	.31955E	01	.59470E	06	.28445E	01	23	3	2	856
.59580E	06	.32337E	01	.59640E	06	.26529E	01	.59710E	06	.23319E	01	23	3	2	857
.59750E	06	.25713E	01	.59840E	06	.31107E	01	.59950E	06	.30903E	01	23	3	2	858
.60000E	06	.26495E	01	.60040E	06	.25485E	01	.60090E	06	.23648E	01	23	3	2	859
.60180E	06	.22184E	01	.60250E	06	.22387E	01	.60340E	06	.31993E	01	23	3	2	860
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.60740E	06	.27110E	01	.60900E	06	.27127E	01	.60990E	06	.26429E	01	23	3	2	862
.61130E	06	.30440E	01	.61220E	06	.27446E	01	.61290E	06	.35255E	01	23	3	2	863
.61430E	06	.28461E	01	.61530E	06	.23665E	01	.61620E	06	.21674E	01	23	3	2	864
.61710E	06	.23681E	01	.61780E	06	.28087E	01	.61850E	06	.33392E	01	23	3	2	865
.62020E	06	.29297E	01	.62110E	06	.23002E	01	.62180E	06	.15813E	01	23	3	2	866
.62280E	06	.19719E	01	.62390E	06	.21323E	01	.62560E	06	.18430E	01	23	3	2	867
.62610E	06	.78368E	00	.62750E	06	.44879E	00	.62870E	06	.55912E	00	23	3	2	868
.62970E	06	.12160E	01	.63090E	06	.22268E	01	.63180E	06	.25575E	01	23	3	2	869
.63330E	06	.35182E	01	.63520E	06	.33088E	01	.63750E	06	.42298E	01	23	3	2	870
.63790E	06	.38711E	01	.63890E	06	.43925E	01	.64010E	06	.42028E	01	23	3	2	871
.64160E	06	.28935E	01	.64360E	06	.20642E	01	.64460E	06	.18152E	01	23	3	2	872
.64510E	06	.17665E	01	.64590E	06	.24172E	01	.64660E	06	.26875E	01	23	3	2	873
.64710E	06	.34680E	01	.64790E	06	.37385E	01	.64910E	06	.36488E	01	23	3	2	874
.64960E	06	.33493E	01	.65010E	06	.36101E	01	.65040E	06	.42904E	01	23	3	2	875
.65160E	06	.49608E	01	.65240E	06	.51910E	01	.65290E	06	.32817E	01	23	3	2	876
.65390E	06	.24523E	01	.65490E	06	.23226E	01	.65570E	06	.27432E	01	23	3	2	877
.65700E	06	.33539E	01	.65850E	06	.33844E	01	.65930E	06	.40653E	01	23	3	2	878
.66030E	06	.47162E	01	.66110E	06	.42468E	01	.66160E	06	.37574E	01	23	3	2	879
.66240E	06	.38879E	01	.66350E	06	.36683E	01	.66480E	06	.27288E	01	23	3	2	880
.66580E	06	.21495E	01	.66760E	06	.18003E	01	.66790E	06	.21510E	01	23	3	2	881
.66890E	06	.17722E	01	.66920E	06	.18224E	01	.67030E	06	.27330E	01	23	3	2	882
.67130E	06	.29932E	01	.67200E	06	.30439E	01	.67260E	06	.24546E	01	23	3	2	883
.67420E	06	.22235E	01	.67660E	06	.20254E	01	.67820E	06	.18465E	01	23	3	2	884
.67930E	06	.17780E	01	.67990E	06	.22391E	01	.68040E	06	.31798E	01	23	3	2	885
.68150E	06	.33002E	01	.68200E	06	.32605E	01	.68260E	06	.37612E	01	23	3	2	886
.68450E	06	.38416E	01	.68580E	06	.37020E	01	.68690E	06	.37832E	01	23	3	2	887
.68780E	06	.43740E	01	.68860E	06	.38648E	01	.69050E	06	.39153E	01	23	3	2	888
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                                                       1           2315   3 2127
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                                                       1          10 2315   3 2131
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9.6718E+06 1.8431E-08 1.0226E+07 1.4714E-08 1.0246E+07 0.          2315   3 2145
0.         5.0000E+06          0           0           1          41 2315   3 2146
        41          2                                               2315   3 2147
1.0000E+04 0.         1.9807E+04 7.7950E-09 4.9228E+04 1.9092E-08  2315   3 2148
8.8457E+04 3.3618E-08 1.6691E+05 6.0932E-08 2.7479E+05 9.4905E-08  2315   3 2149
3.9248E+05 1.2760E-07 5.0035E+05 1.5391E-07 6.0823E+05 1.7702E-07  2315   3 2150
8.4360E+05 2.1762E-07 1.0888E+06 2.4773E-07 1.3536E+06 2.6896E-07  2315   3 2151
1.6478E+06 2.8171E-07 1.9616E+06 2.8569E-07 2.3245E+06 2.8127E-07  2315   3 2152
2.7462E+06 2.6773E-07 3.9623E+06 2.0749E-07 4.1976E+06 1.4632E-07  2315   3 2153
4.4330E+06 9.1471E-08 4.5507E+06 6.6389E-08 4.6684E+06 4.2838E-08  2315   3 2154
4.7861E+06 2.0788E-08 4.8449E+06 1.0315E-08 4.8841E+06 3.5328E-09  2315   3 2155
4.9037E+06 2.0180E-10 5.8248E+06 1.4220E-10 6.2573E+06 1.1965E-10  2315   3 2156
6.6898E+06 1.0019E-10 7.1463E+06 8.2704E-11 7.6269E+06 6.7284E-11  2315   3 2157
8.1315E+06 5.3946E-11 8.6841E+06 4.2166E-11 9.0205E+06 3.6220E-11  2315   3 2158
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9.3809E+06 3.0729E-11 9.7654E+06 2.5744E-11 1.0150E+07 2.1535E-11   2315   3 2159
1.0558E+07 1.7787E-11 1.1015E+07 1.4338E-11 1.1471E+07 1.1539E-11   2315   3 2160
1.1976E+07 9.0587E-12 1.2000E+07 0.                                 2315   3 2161
0.         7.0000E+06          0          0          1         38   2315   3 2162
                                                                    2315   3 2163
        38         2                                                2315   3 2164
1.0000E+04 0.         1.2061E+04 7.4292E-10 2.3738E+04 4.8647E-09   2315   3 2165
6.4952E+04 1.3100E-08 1.4738E+05 2.8784E-08 2.4355E+05 4.5852E-08   2315   3 2166
3.9466E+05 7.0224E-08 5.5952E+05 9.3696E-08 7.1064E+05 1.1261E-07   2315   3 2167
8.7549E+05 1.3066E-07 1.2052E+06 1.5965E-07 1.5624E+06 1.8191E-07   2315   3 2168
1.9333E+06 1.9692E-07 2.3455E+06 2.0593E-07 2.7988E+06 2.0872E-07   2315   3 2169
3.2522E+06 2.0602E-07 3.7742E+06 1.9811E-07 5.1892E+06 1.6312E-07   2315   3 2170
5.6426E+06 1.1063E-07 6.0410E+06 6.9253E-08 6.2470E+06 4.9908E-08   2315   3 2171
6.4531E+06 3.1948E-08 6.6592E+06 1.5345E-08 6.7553E+06 8.0512E-09   2315   3 2172
6.8103E+06 4.0114E-09 6.8652E+06 6.3429E-11 7.4587E+06 4.8106E-11   2315   3 2173
8.0113E+06 3.7089E-11 8.5880E+06 2.8130E-11 8.9244E+06 2.3888E-11   2315   3 2174
9.2848E+06 2.0021E-11 9.6693E+06 1.6554E-11 1.0078E+07 1.3503E-11   2315   3 2175
1.0510E+07 1.0864E-11 1.0967E+07 8.6198E-12 1.1447E+07 6.7437E-12   2315   3 2176
1.1976E+07 5.1375E-12 1.2000E+07 0.                                 2315   3 2177
0.         9.0000E+06          0          0          1         36   2315   3 2178
        36         2                                                2315   3 2179
1.0000E+04 0.         2.7669E+04 5.2493E-09 4.5338E+04 8.5395E-09   2315   3 2180
1.3368E+05 2.4277E-08 2.2203E+05 3.8881E-08 3.6338E+05 6.0066E-08   2315   3 2181
5.2240E+05 8.0985E-08 6.6375E+05 9.7233E-08 8.0510E+05 1.1148E-07   2315   3 2182
1.1231E+06 1.3718E-07 1.4412E+06 1.5544E-07 1.7945E+06 1.6868E-07   2315   3 2183
2.1833E+06 1.7650E-07 2.6073E+06 1.7894E-07 3.1727E+06 1.7514E-07   2315   3 2184
3.8441E+06 1.6392E-07 6.6005E+06 9.6779E-08 7.1305E+06 6.4899E-08   2315   3 2185
7.3956E+06 5.1290E-08 7.6606E+06 3.9104E-08 7.9433E+06 2.7563E-08   2315   3 2186
8.2260E+06 1.7420E-08 8.3673E+06 1.2840E-08 8.5264E+06 8.0554E-09   2315   3 2187
8.6677E+06 4.1166E-09 8.7384E+06 2.2535E-09 8.7914E+06 9.0140E-10   2315   3 2188
8.8267E+06 2.1142E-11 9.2846E+06 1.6851E-11 9.6693E+06 1.3933E-11   2315   3 2189
1.0078E+07 1.1366E-11 1.0510E+07 9.1439E-12 1.0967E+07 7.2550E-12   2315   3 2190
1.1447E+07 5.6760E-12 1.1976E+07 4.3241E-12 1.2000E+07 0.           2315   3 2191
0.         1.1000E+07          0          0          1         37   2315   3 2192
        37         2                                                2315   3 2193
1.0000E+04 0.         3.1600E+04 6.7812E-09 5.3199E+04 1.1303E-08   2315   3 2194
1.1800E+05 2.4337E-08 2.0440E+05 4.0508E-08 3.3399E+05 6.2381E-08   2315   3 2195
4.6359E+05 8.1622E-08 5.9319E+05 9.8456E-08 7.2279E+05 1.1312E-07   2315   3 2196
9.6198E+05 1.3667E-07 1.2628E+06 1.5491E-07 1.5652E+06 1.6767E-07   2315   3 2197
1.8892E+06 1.7513E-07 2.2564E+06 1.7759E-07 2.7316E+06 1.7406E-07   2315   3 2198
3.3147E+06 1.6310E-07 6.0579E+06 8.8866E-08 7.0515E+06 6.6411E-08   2315   3 2199
8.0451E+06 4.8384E-08 8.5203E+06 3.4335E-08 8.7579E+06 2.8375E-08   2315   3 2200
9.0171E+06 2.2644E-08 9.4059E+06 1.5500E-08 9.6218E+06 1.2163E-08   2315   3 2201
9.8378E+06 9.2130E-09 1.0054E+07 6.6155E-09 1.0291E+07 4.1274E-09   2315   3 2202
1.0399E+07 3.1138E-09 1.0529E+07 1.9874E-09 1.0659E+07 9.5378E-10   2315   3 2203
1.0723E+07 4.6995E-10 1.0767E+07 1.5916E-10 1.0788E+07 7.2248E-12   2315   3 2204
1.0967E+07 6.5571E-12 1.1447E+07 5.1300E-12 1.1976E+07 3.9082E-12   2315   3 2205
1.2000E+07 0.                                                       2315   3 2206
0.         1.3000E+07          0          0          1         55   2315   3 2207
        55         2                                                2315   3 2208
1.0000E+04 0.         1.3382E+04 1.2324E-06 1.6764E+04 1.4271E-06   2315   3 2209
2.0147E+04 1.5852E-06 2.6911E+04 1.8088E-06 3.3676E+04 1.9334E-06   2315   3 2210
4.3822E+04 1.9858E-06 5.0587E+04 1.9580E-06 6.0733E+04 1.8560E-06   2315   3 2211
7.4262E+04 1.6571E-06 1.0470E+05 1.1555E-06 1.1823E+05 9.5636E-07   2315   3 2212
1.3852E+05 7.0611E-07 1.4867E+05 6.0318E-07 1.5882E+05 5.1405E-07   2315   3 2213
1.6896E+05 4.3757E-07 1.8249E+05 3.5296E-07 1.9602E+05 2.8524E-07   2315   3 2214
2.0955E+05 2.3008E-07 2.2308E+05 1.8734E-07 2.3661E+05 1.5453E-07   2315   3 2215
2.5352E+05 1.2438E-07 2.7043E+05 1.0336E-07 2.8396E+05 9.1445E-08   2315   3 2216
3.0425E+05 7.9414E-08 3.2116E+05 7.3115E-08 3.4146E+05 6.8778E-08   2315   3 2217
3.6175E+05 6.6852E-08 3.7866E+05 6.6490E-08 4.4631E+05 7.0400E-08   2315   3 2218
6.5876E+05 9.1677E-08 8.9904E+05 1.1092E-07 1.1633E+06 1.2593E-07
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1.4277E+06 1.3568E-07 1.7400E+06 1.4188E-07 2.0764E+06 1.4362E-07  2315  3 2219
2.4849E+06 1.4082E-07 2.9895E+06 1.3245E-07 5.5365E+06 7.1239E-08  2315  3 2220
6.3534E+06 5.5024E-08 7.1704E+06 4.1828E-08 7.6990E+06 3.4802E-08  2315  3 2221
8.2516E+06 2.8430E-08 8.8523E+06 2.2627E-08 9.4771E+06 1.7731E-08  2315  3 2222
9.7894E+06 1.4211E-08 1.0126E+07 1.1035E-08 1.0534E+07 7.9080E-09  2315  3 2223
1.0751E+07 6.5319E-09 1.0991E+07 5.1992E-09 1.1231E+07 4.0728E-09  2315  3 2224
1.1471E+07 3.1178E-09 1.1712E+07 2.3016E-09 1.1976E+07 1.5444E-09  2315  3 2225
1.2000E+07 0.                                                      2315  3 2226
0.         1.4000E+07          0          0          1         45  2315  3 2227
        45         2                                               2315  3 2228
1.0000E+04 0.         1.5348E+04 1.5949E-07 2.0695E+04 2.0927E-07  2315  3 2229
3.1591E+04 2.9941E-07 4.2086E+04 3.7731E-07 5.2781E+04 4.4451E-07  2315  3 2230
6.8824E+04 5.2783E-07 9.0215E+04 6.1096E-07 1.1695E+05 6.7826E-07  2315  3 2231
1.4369E+05 7.1398E-07 1.7043E+05 7.2617E-07 2.0252E+05 7.1815E-07  2315  3 2232
2.4530E+05 6.8249E-07 4.4316E+05 4.2222E-07 5.0198E+05 3.5643E-07  2315  3 2233
5.6081E+05 3.0145E-07 6.3567E+05 2.4287E-07 6.6776E+05 2.2218E-07  2315  3 2234
7.3728E+05 1.8742E-07 7.7471E+05 1.7352E-07 8.1215E+05 1.6238E-07  2315  3 2235
8.9236E+05 1.4506E-07 9.3514E+05 1.3863E-07 1.0314E+06 1.2998E-07  2315  3 2236
1.2400E+06 1.2511E-07 1.9803E+06 1.3088E-07 2.3888E+06 1.2818E-07  2315  3 2237
2.6453E+06 1.2107E-07 5.3923E+06 6.3132E-08 6.2092E+06 4.8044E-08  2315  3 2238
7.0502E+06 3.5622E-08 7.4587E+06 3.0634E-08 7.8912E+06 2.6028E-08  2315  3 2239
8.3477E+06 2.1885E-08 8.8043E+06 1.8278E-08 9.4050E+06 1.4281E-08  2315  3 2240
9.7894E+06 1.2165E-08 1.0174E+07 1.0343E-08 1.0510E+07 8.1614E-09  2315  3 2241
1.0847E+07 6.3409E-09 1.1111E+07 5.1441E-09 1.1375E+07 4.1238E-09  2315  3 2242
1.1664E+07 3.1862E-09 1.1976E+07 2.3478E-09 1.2000E+07 0.          2315  3 2243
0.         1.5000E+07          0          0          1         41  2315  3 2244
        41         2                                               2315  3 2245
1.0000E+04 0.         1.7313E+04 6.3007E-08 3.1939E+04 1.1331E-07  2315  3 2246
5.3878E+04 1.7922E-07 6.8505E+04 2.1818E-07 9.0444E+04 2.6989E-07  2315  3 2247
1.1238E+05 3.1421E-07 1.3432E+05 3.5200E-07 1.7820E+05 4.1043E-07  2315  3 2248
2.2208E+05 4.4994E-07 2.7327E+05 4.7730E-07 3.2446E+05 4.8918E-07  2315  3 2249
3.9028E+05 4.8835E-07 4.7072E+05 4.7080E-07 9.2413E+05 2.9430E-07  2315  3 2250
1.0850E+06 2.4101E-07 1.1728E+06 2.1685E-07 1.2678E+06 1.9623E-07  2315  3 2251
1.3629E+06 1.8008E-07 1.4726E+06 1.6534E-07 1.6774E+06 1.4726E-07  2315  3 2252
1.8310E+06 1.3869E-07 2.1600E+06 1.2738E-07 4.6234E+06 7.1590E-08  2315  3 2253
5.2481E+06 5.8584E-08 6.0891E+06 4.3596E-08 6.4976E+06 3.7480E-08  2315  3 2254
6.9301E+06 3.1810E-08 7.3626E+06 2.6872E-08 7.6509E+06 2.4040E-08  2315  3 2255
8.2997E+06 1.8428E-08 8.7802E+06 1.5097E-08 9.3809E+06 1.1745E-08  2315  3 2256
1.0006E+07 8.9125E-09 1.0438E+07 7.3436E-09 1.0895E+07 5.9701E-09  2315  3 2257
1.1135E+07 5.0284E-09 1.1399E+07 4.1371E-09 1.1688E+07 3.3149E-09  2315  3 2258
1.1976E+07 2.6277E-09 1.2000E+07 0.                                2315  3 2259
0.         1.7500E+07          0          0          1         42  2315  3 2260
        42         2                                               2315  3 2261
1.0000E+04 0.         2.2227E+04 1.2996E-08 3.4028E+04 2.1363E-08  2315  3 2262
8.2084E+04 4.9381E-08 1.5417E+05 8.7217E-08 2.4710E+05 1.2941E-07  2315  3 2263
3.5653E+05 1.6979E-07 4.6239E+05 2.0208E-07 5.1459E+05 2.1447E-07  2315  3 2264
6.8483E+05 2.4741E-07 8.1250E+05 2.6432E-07 9.7112E+05 2.7761E-07  2315  3 2265
1.2114E+06 2.8436E-07 1.3672E+06 2.8361E-07 1.6199E+06 2.7401E-07  2315  3 2266
2.0284E+06 2.4917E-07 2.6020E+06 2.0570E-07 2.9077E+06 1.7591E-07  2315  3 2267
3.2134E+06 1.4973E-07 3.5190E+06 1.2728E-07 3.8492E+06 1.0675E-07  2315  3 2268
4.2389E+06 8.6686E-08 4.5461E+06 7.3864E-08 4.8877E+06 6.1723E-08  2315  3 2269
5.2308E+06 5.1856E-08 5.7287E+06 4.0483E-08 6.1111E+06 3.3652E-08  2315  3 2270
6.5937E+06 2.7550E-08 7.0742E+06 2.2416E-08 7.5548E+06 1.8116E-08  2315  3 2271
8.0834E+06 1.4255E-08 8.6361E+06 1.1034E-08 8.9725E+06 9.4160E-09  2315  3 2272
9.3329E+06 7.9244E-09 9.7173E+06 6.5841E-09 9.9515E+06 5.8877E-09  2315  3 2273
1.0510E+07 4.4468E-09 1.0967E+07 3.5313E-09 1.1336E+07 2.9136E-09  2315  3 2274
1.1447E+07 2.7393E-09 1.1976E+07 2.0862E-09 1.2000E+07 0.          2315  3 2275
0.         2.0000E+07          0          0          1         36  2315  3 2276
        36         2                                               2315  3 2277
1.0000E+04 0.         2.7140E+04 7.1144E-09 3.4028E+04 9.6239E-09  2315  3 2278
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8.2084E+04 2.2602E-08 1.4712E+05 3.9090E-08 2.3282E+05 5.9231E-08   2315  3 2279
3.8708E+05 9.0764E-08 5.1459E+05 1.1294E-07 6.1196E+05 1.2711E-07   2315  3 2280
7.0681E+05 1.4028E-07 8.4348E+05 1.5544E-07 1.1393E+06 1.8064E-07   2315  3 2281
1.3796E+06 1.9301E-07 1.6679E+06 2.0130E-07 2.0284E+06 2.0379E-07   2315  3 2282
2.4609E+06 1.9879E-07 3.1467E+06 1.8053E-07 4.9807E+06 1.1694E-07   2315  3 2283
5.7046E+06 8.4384E-08 6.2833E+06 6.2756E-08 6.7118E+06 4.9317E-08   2315  3 2284
6.9541E+06 4.2636E-08 7.3803E+06 3.2140E-08 7.6031E+06 2.7416E-08   2315  3 2285
8.0438E+06 1.9188E-08 8.3059E+06 1.5073E-08 8.5640E+06 1.1325E-08   2315  3 2286
8.9244E+06 9.4168E-09 9.2848E+06 7.8217E-09 9.6693E+06 6.4005E-09   2315  3 2287
1.0078E+07 5.1478E-09 1.0510E+07 4.0756E-09 1.0967E+07 3.1676E-09   2315  3 2288
1.1447E+07 2.4178E-09 1.1976E+07 1.8080E-09 1.2000E+07 0.           2315  3 2289
                                                                    2315  0 2290
2.3000E+04 5.0505E+01                                 1             2315102 2291
0.                          0           1             1           2 2315102 2292
         2           2                                              2315102 2293
1.0000E-05 1.0000E+00 2.0000E+07 1.0000E+00                         2315102 2294
                                                      1           3 2315102 2295
         3           2                                              2315102 2296
0.         1.0000E-05           0           0           1          30 2315102 2297
        30           1                                              2315102 2298
2.5000E+05 3.0852E-07 5.0000E+05 3.0061E-07 7.5000E+05 2.3425E-07   2315102 2299
1.0000E+06 2.9226E-08 1.2500E+06 1.0899E-07 1.5000E+06 2.2853E-07   2315102 2300
1.7500E+06 1.7953E-07 2.0000E+06 1.8041E-07 2.2500E+06 4.0432E-08   2315102 2301
2.5000E+06 1.4942E-08 2.7500E+06 5.7572E-08 3.0000E+06 1.0768E-08   2315102 2302
3.2500E+06 1.6041E-08 3.5000E+06 8.2184E-08 3.7500E+06 2.0876E-08   2315102 2303
4.0000E+06 5.8012E-08 4.2500E+06 4.3068E-08 4.5000E+06 2.4831E-08   2315102 2304
4.7500E+06 6.6364E-08 5.0000E+06 2.1930E-07 5.2500E+06 1.0108E-08   2315102 2305
5.5000E+06 2.2084E-07 5.7500E+06 2.4348E-07 6.0000E+06 4.6148E-09   2315102 2306
6.2500E+06 2.1447E-07 6.5000E+06 4.0960E-07 6.7500E+06 2.3908E-07   2315102 2307
7.0000E+06 2.9424E-07 7.2500E+06 6.4397E-07 7.3040E+06 0.           2315102 2308
0.         3.0000E+06           0           0           1          30 2315102 2309
        30           1                                              2315102 2310
2.5000E+05 3.0852E-07 5.0000E+05 3.2393E-07 7.3200E+05 2.1852E-07   2315102 2311
1.0000E+06 2.9226E-08 1.2500E+06 1.0899E-07 1.5000E+06 2.2853E-07   2315102 2312
1.7500E+06 1.7953E-07 2.0000E+06 1.8041E-07 2.2500E+06 4.0432E-08   2315102 2313
2.5000E+06 1.4942E-08 2.7500E+06 5.7572E-08 3.0000E+06 1.0768E-08   2315102 2314
3.2500E+06 1.6041E-08 3.5000E+06 8.2184E-08 3.7500E+06 2.0876E-08   2315102 2315
4.0000E+06 5.8012E-08 4.2500E+06 4.3068E-08 4.5000E+06 2.4831E-08   2315102 2316
4.7500E+06 6.6364E-08 5.0000E+06 2.1930E-07 5.2500E+06 1.0108E-08   2315102 2317
5.5000E+06 2.2084E-07 5.7500E+06 2.4348E-07 6.0000E+06 4.6148E-09   2315102 2318
6.2500E+06 2.1447E-07 6.5000E+06 4.0960E-07 6.7500E+06 2.3908E-07   2315102 2319
7.0000E+06 2.9424E-07 7.2500E+06 9.8232E-08 7.6040E+06 0.           2315102 2320
0.         2.0000E+07           0           0           1          30 2315102 2321
        30           1                                              2315102 2322
2.5000E+05 3.0852E-07 5.0000E+05 3.2393E-07 7.3200E+05 2.1852E-07   2315102 2323
1.0000E+06 2.9226E-08 1.2500E+06 1.0899E-07 1.5000E+06 2.2853E-07   2315102 2324
1.7500E+06 1.7953E-07 2.0000E+06 1.8041E-07 2.2500E+06 4.0432E-08   2315102 2325
2.5000E+06 1.4942E-08 2.7500E+06 5.7572E-08 3.0000E+06 1.0768E-08   2315102 2326
3.2500E+06 1.6041E-08 3.5000E+06 8.2184E-08 3.7500E+06 2.0876E-08   2315102 2327
4.0000E+06 5.8012E-08 4.2500E+06 4.3068E-08 4.5000E+06 2.4831E-08   2315102 2328
4.7500E+06 6.6364E-08 5.0000E+06 2.1930E-07 5.2500E+06 1.0108E-08   2315102 2329
5.5000E+06 2.2084E-07 5.7500E+06 2.4348E-07 6.0000E+06 4.6148E-09   2315102 2330
6.2500E+06 2.1447E-07 6.5000E+06 4.0960E-07 6.7500E+06 2.3908E-07   2315102 2331
7.0000E+06 2.9424E-07 7.2500E+06 9.8232E-08 7.6040E+06 0.           2315102 2332
                                                                    2315  0 2333
                                                                    23 0  0 2334
                                                                     0 0  0 2335
                                                                    -1 0  0 2336
@@@@                                                                -1 0  0 2337
@@@@                                                                -1 0  0 2338
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Table 1. Comparison of Present-Measured and Previously-Reported Vanadium Excitation Energies.

No. of State	Excitation Energies (keV)	
	Exp.	NDS(1)
1	321±10	320.1(5/2-)
2	938±15	929.0(3/2-)
3	1603±19	1609.0(11/2-)
4	1811±21	1813.0(9/2-)
5	2409±27	2409.0(3/2-)
6	≈ 2500± ?	2545.0(1/2+)
7	2706±30	2675.0(3/2+) 2699.0(----)
8	2773±30	2790.0(----)

Table 2. Optical Model Parameters[a]

Real Potential[b]		
$V=50.76 \pm 1.21$ MeV,	$R_o = 1.188 \pm 0.016$ F,	$A=0.663 \pm 0.047$ F.
Imaginary Potential[c]		
$W=8.95 \pm 0.62$ MeV,	$R_o = 1.270 \pm 0.045$ F,	$A=0.467 \pm 0.064$ F.

a. All uncertainties are RMS deviations from the average of the parameter values deduced from χi-square fitting procedures as described in the text. A spin-orbit potential of the Thomas form with a 7.0 MeV magnitude was used throughout. All radii are expressed in the form $R=R_oA^{1/3}$.

b. Saxon form.

c. Saxon-derivative form.

Table 3. Effective Inelastic Neutron Groups Used in this Evaluation

No.	E_x (MeV)[a]	Thres. (MeV)
1	0.3201	0.3264
2	0.929	0.9472
3	1.609	1.641
4	1.813	1.849
5	2.409 2.545	2.456
6	2.675 2.699 2.790	2.727
7	3.080	3.140

a) All E_x values taken from the Nuclear Data Sheets (1).

FIGURE CAPTIONS

Fig. 1. Neutron total and elastic scattering cross sections of vanadium. 100 keV energy averages of the present total cross section results are represented by square data points. Circular data points indicate the angle-integrated elastic scattering cross sections deduced from the present measurements. The evaluated total neutron cross section, described in Sec. V of the text, is indicated by a curve.

(ANL Negative No. 116-76-54)

Fig. 2. Differential elastic scattering cross sections of vanadium. The values determined in the present measurements are indicated by the circular data points. The curves are the result of Legendre polynomial fits to the data as described in Sec. IV of the text.

(ANL Negative No. 116-76-52)

Fig. 3. Some comparisons of the present differential elastic scattering cross sections of vanadium with those previously reported in the literature. The results of the present work are indicated by circular data points, those of Ref. 16 by ◁, of Ref. 17 by + and of Ref. 18 by X. Curves are the results of model calculations as discussed in the text.

(ANL Negative No. 116-76-53)

Fig. 4. Angular distributions of elastic- and inelastic-scattered neutrons at an incident energy of 3.0 MeV. Presently-measured values are noted by data points. Curves denote the results of calculation using the potential of Table 2 as described in the text. Results obtained with (WFC) and without (H-F) width fluctuation corrections to the compound-nucleus component are shown.

(ANL Negative No. 116-76-358)

Fig. 5. Comparison of measured and calculated inelastic neutron scattering excitation cross sections. The present results are noted by circular data points. Solid curves indicate calculated results as described in the text; lower indicates results with width fluctuation correction, the upper without. Measured values taken from the literature are noted by various symbols as per the following: + = Ref. 2, △ = Ref. 25, ◇ = Ref. 20, X = Ref. 16, ↑ = Ref. 17, ||| = Ref. 19, Z = Ref. 23, Y = Ref. 22, and ⌘ = Ref. 24. The dashed curve is the present evaluation.

(ANL Negative No. 116-76-354)

Fig. 6. Measured and calculated elastic scattering distributions. Presently-measured values are indicated by data points. Xi-square optical model fits to the individual distributions are noted by curves.

(ANL Negative No. 116-77-17)

Fig. 7. Measured and calculated total neutron cross sections of vanadium.

Fig. 8. Comparison of the present evaluated total nuclear cross sections with those of ENDF/B-IV.

(ANL Negative No. 116-75-55)

Fig. 9. Comparison of the present evaluated discrete inelastic excitation cross sections with those of ENDF/B-IV.

(ANL Negative No. 116-76-178)

Fig. 10. Measured and evaluated radiative capture cross sections of vanadium. Experimental data values referenced as follows:

○ = Ref. 53, ◇ = Ref. 57, ⌖ = Ref. 58, ⧖ = Ref. 59, Z = Ref. 60, Y = Ref. 61, ⌑ = Ref. 62, ⧖ = Ref. 64, △ = Ref. 54, × = Ref. 56, + = Ref. 55 and ✳ = Ref. 63.

(ANL Negative No. 116-76-181)

Fig.11. Measured and evaluated (n;2n') cross sections of vanadium. The solid notes the present evaluation and the dotted curve that of ENDF/B-IV. Experimental values are referenced as follows:

□ = Ref. 65 and 0 = Ref. 67. (ANL Negative No. 116-76-179)

Fig.12. Measured and evaluated (n;p) cross sections of vanadium. Experimental values are referenced as follows: ○ = Ref. 70, △ = Ref. 71, + = Ref. 72, × = Ref. 73, ◇ = Ref. 74, ↑ = Ref. 75, ⧖ = Ref. 76, Z = Ref. 77, Y = Ref. 79, ⌑ = Ref. 80, ✳ = Ref. 82, ⧖ = Ref. 84 and ✡ = Ref. 85. Solid curve indicates present evaluation, dashed curve that of ENDF/B-IV.

(ANL-Negative No. 116-77-29)

Fig.13. Measured and evaluated (n;α) cross sections of vanadium. Experimental values are referenced as follows: ○ = Ref. 103, ↑ = Ref. 86, × = Ref. 88, ◇ = Ref. 89, ↑ = Ref. 90, ⧖ = Ref. 91, Z = Ref. 92, Y = Ref. 93, ⌑ = Ref. 94, ✳ = Ref. 95, ⧖ = Ref. 96, ✡ = Ref. 99, ν = Ref. 100 and % = Ref. 102. The solid curve denotes the present evaluation, the dashed that of ENDF/B-IV.

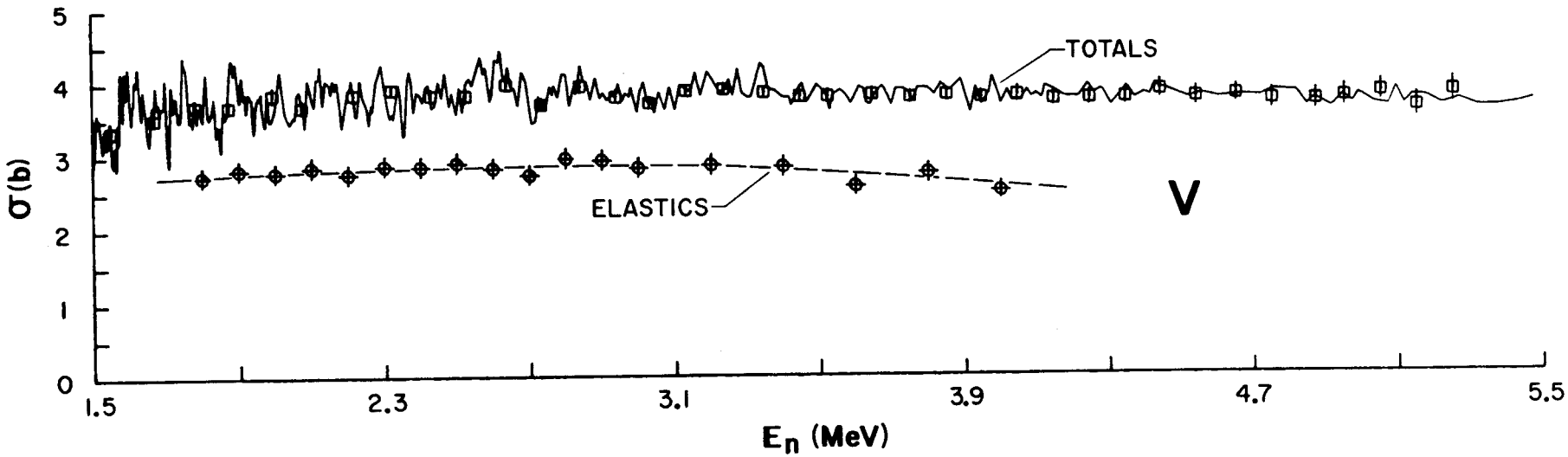
(ANL-Negative No. 116-77-30)

Fig.14. Comparison of evaluated (n;n'p) cross sections. The solid curve indicates the present evaluation, the dashed that of ENDF/B-IV.

(ANL-Negative No. 116-77-32)

Fig.15. Comparison of measured and evaluated (n;n',α) cross sections. The experimental values are referenced as follows: × = Ref. 106, ○ = Ref. 104, △ = Ref. 105 and + = Ref. 90. The solid curve denotes the present evaluation, the dashed that of ENDF/B-IV.

(ANL-Negative No. 116-77-31)


5
4
3
2
1
0
σ(b)
TOTALS
ELASTICS
V
1.5
2.3
3.1
3.9
4.7
5.5
En (MeV)

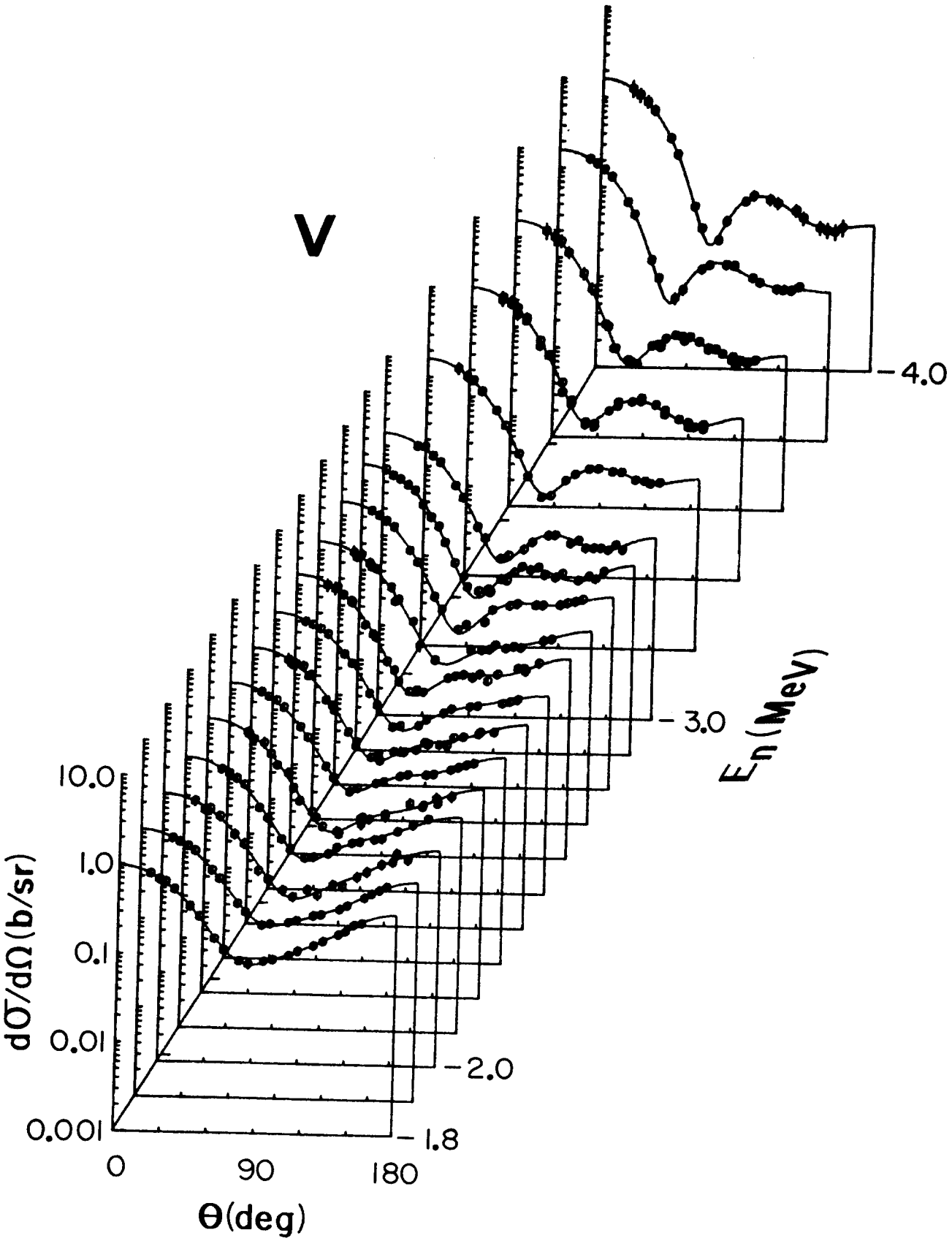


Fig. 2

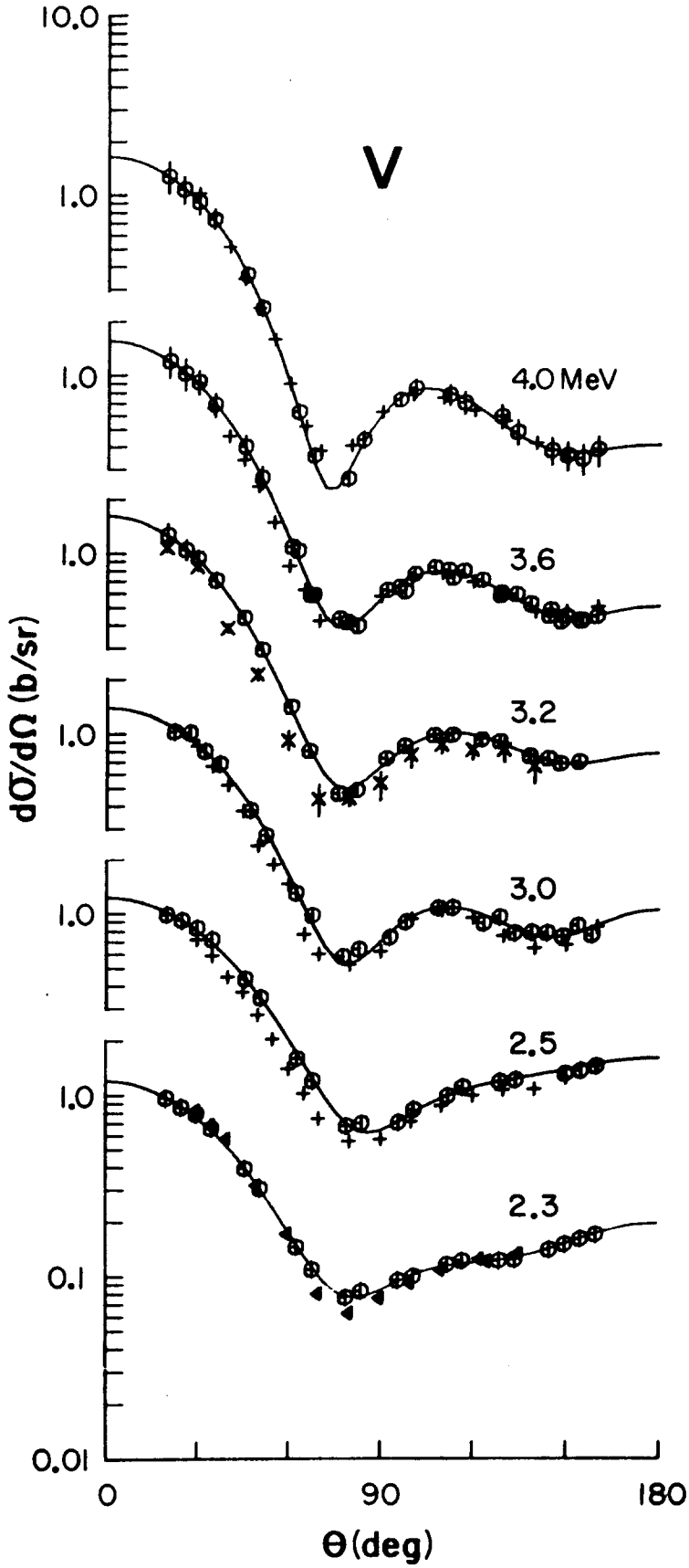

V
10.0
1.0
1.0
1.0
1.0
1.0
1.0
0.1
0.01
4.0 MeV
3.6
3.2
3.0
2.5
2.3
dσ/dΩ (b/sr)
0
90
180
θ(deg)

Fig. 3

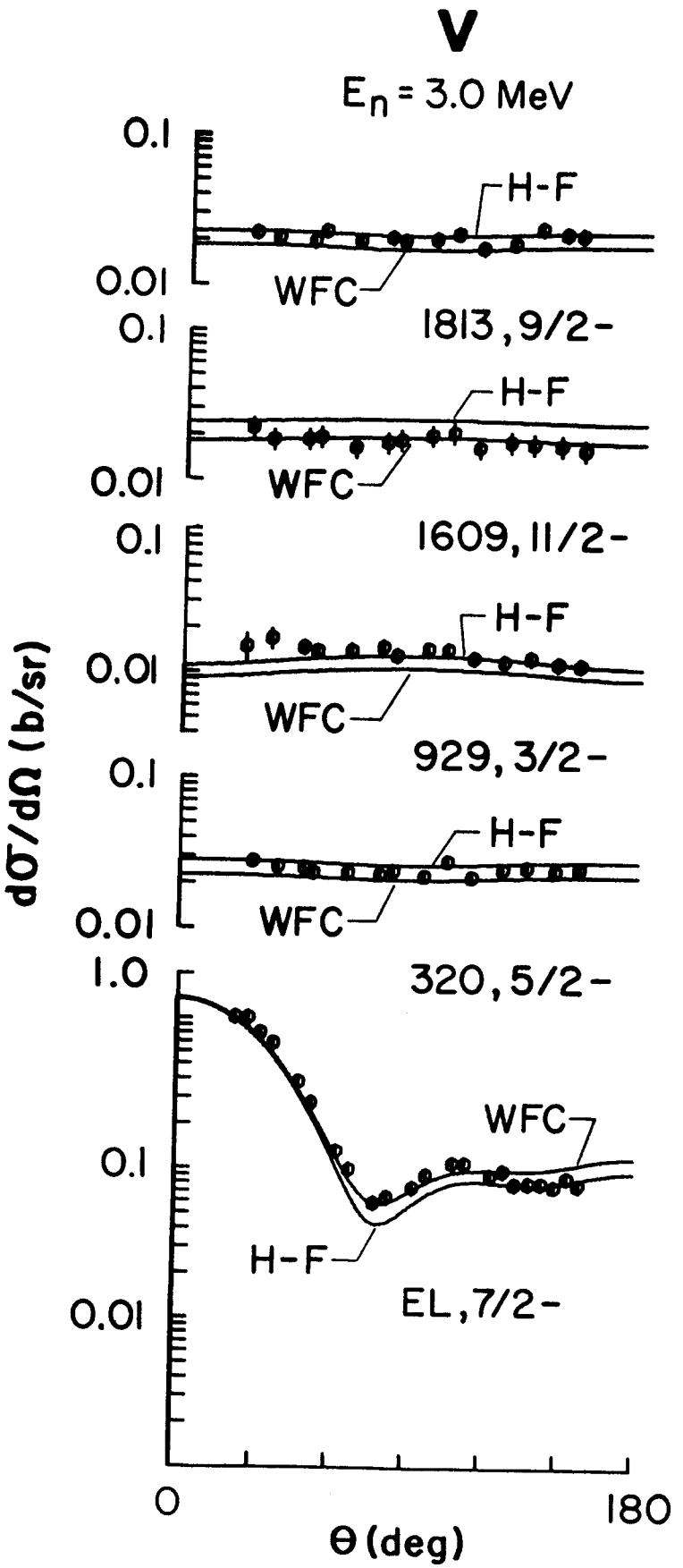

V
E_n = 3.0 MeV
0.1
0.01
H-F
WFC
1813, 9/2-
1609, 11/2-
929, 3/2-
320, 5/2-
EL, 7/2-
1.0
dσ/dΩ (b/sr)
0
180
θ (deg)

Fig. 4

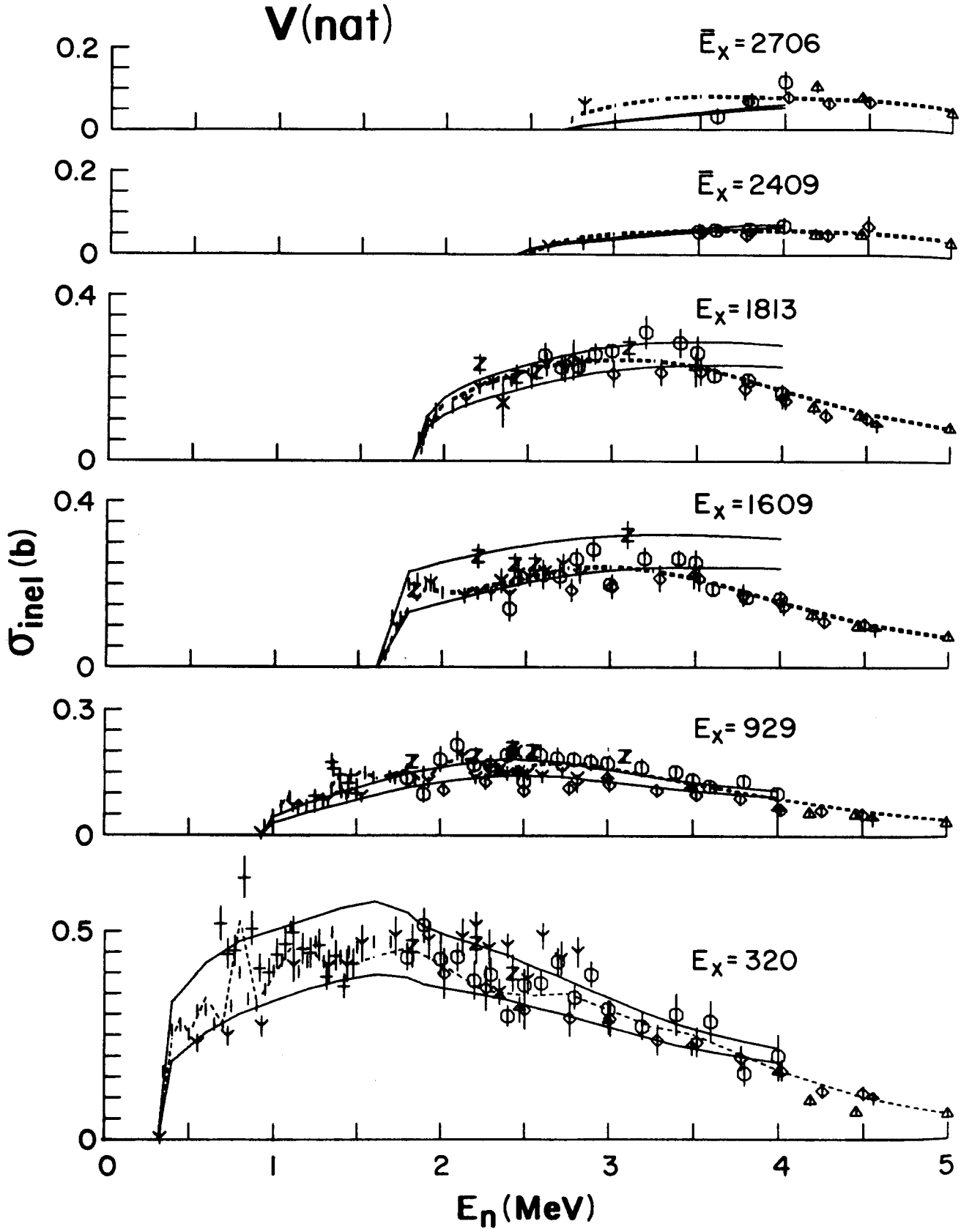

V(nat)
$\bar{E}_x$ = 2706
$\bar{E}_x$ = 2409
E_x = 1813
E_x = 1609
E_x = 929
E_x = 320
σ_{inel} (b)
E_n (MeV)
0.2
0
0.2
0
0.4
0
0.4
0
0.3
0
0.5
0
0
1
2
3
4
5

Fig. 5

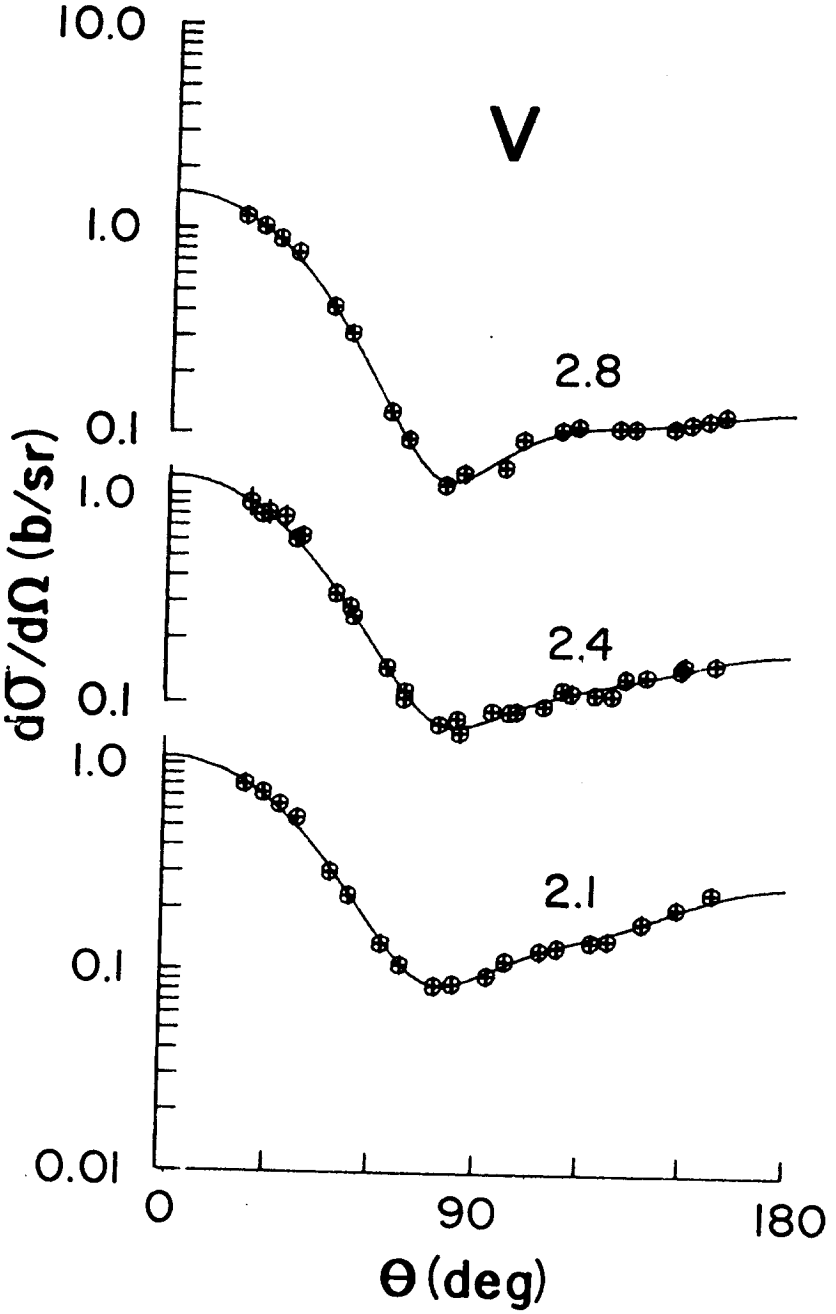
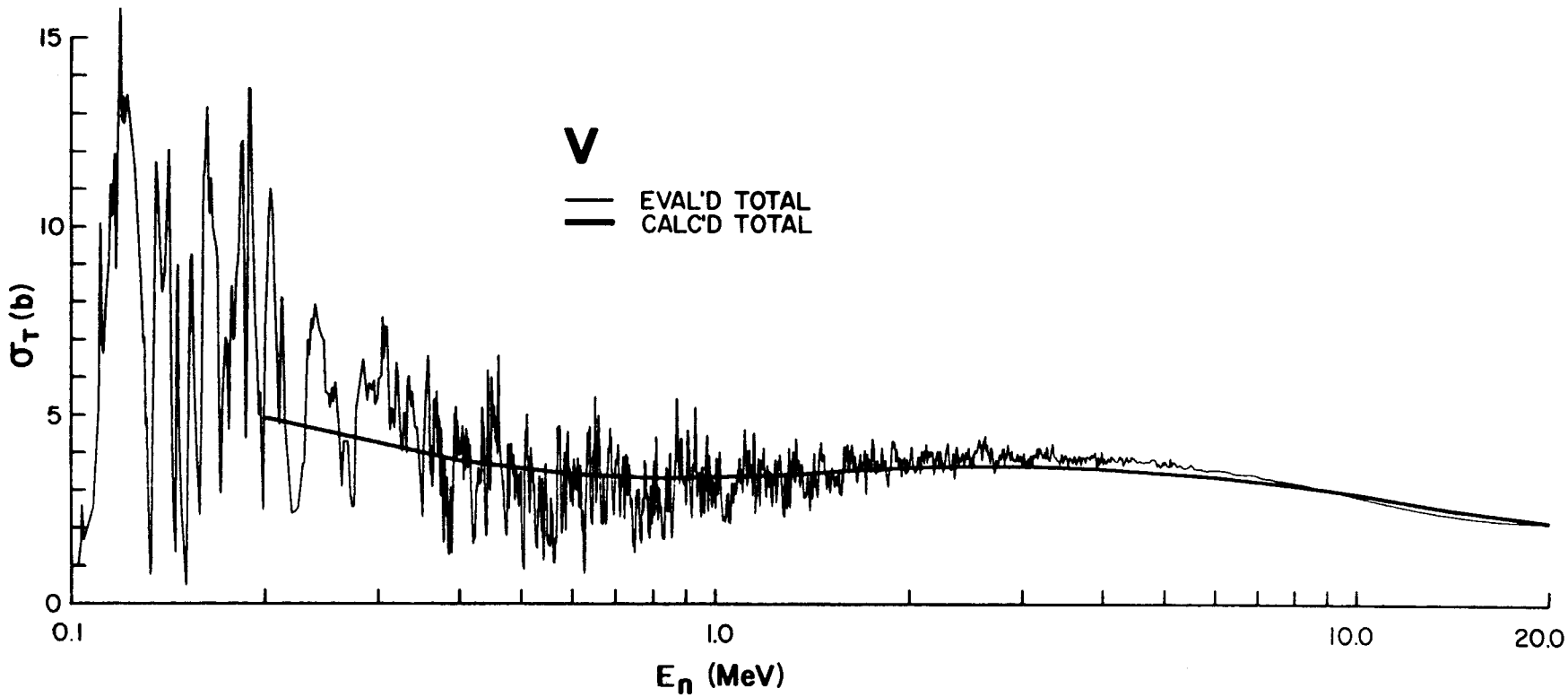


Fig. 6


V
EVAL'D TOTAL
CALC'D TOTAL
15
10
5
0
σ_T (b)
0.1
1.0
10.0
20.0
E_n (MeV)

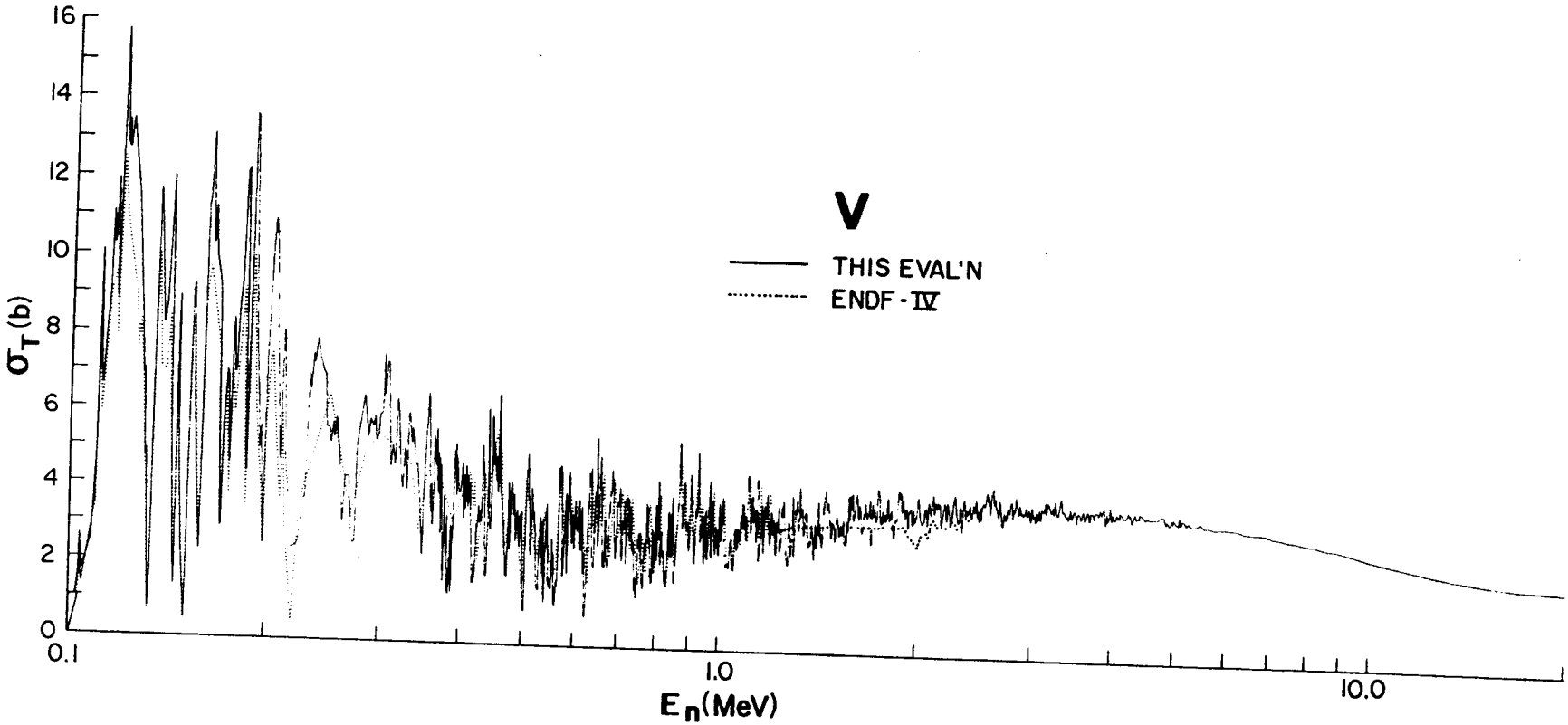
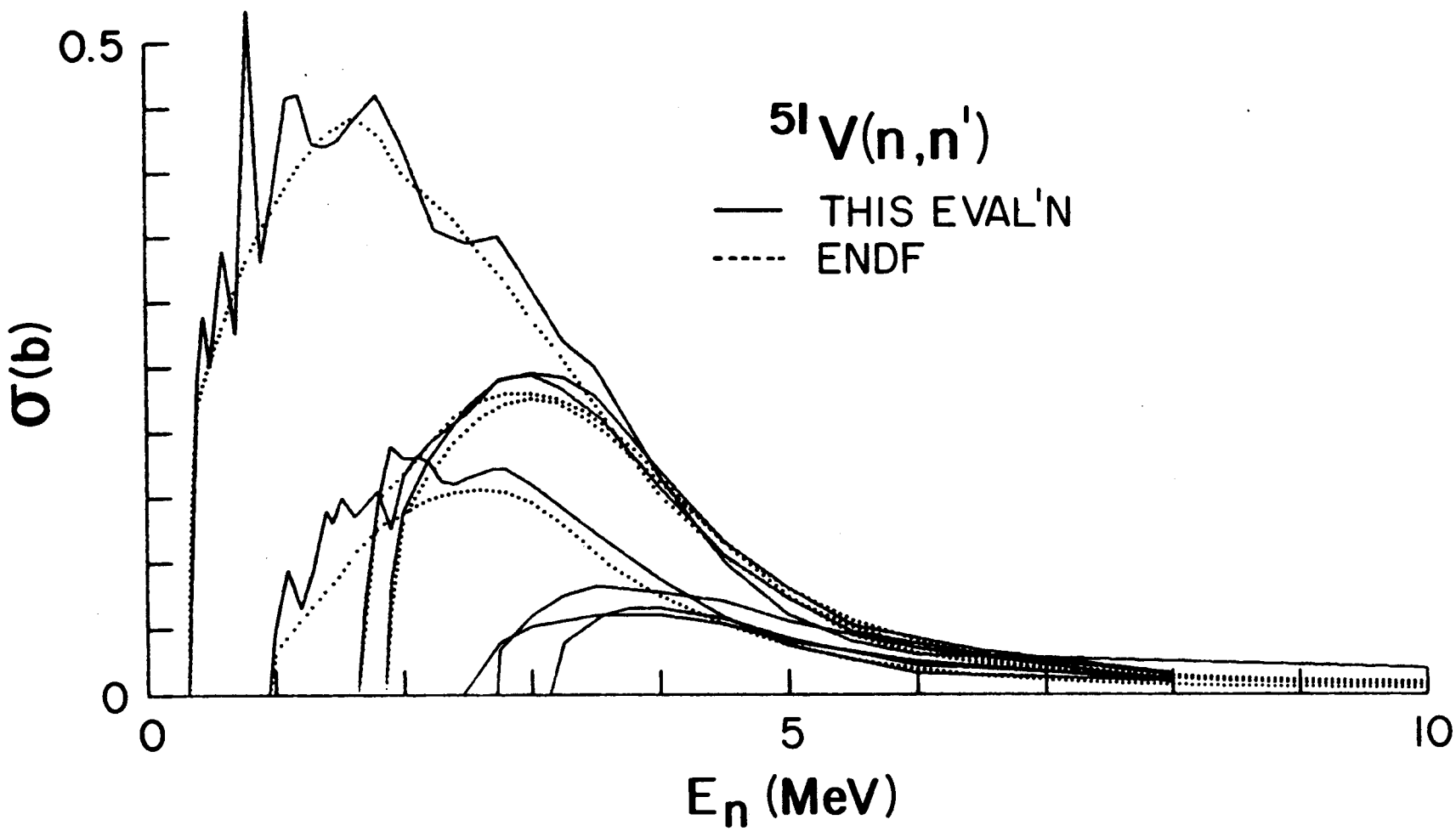


Fig. 8


0.5
0
σ(b)
0
5
10
E_n (MeV)
$^{51}V(n,n')$
THIS EVAL'N
ENDF

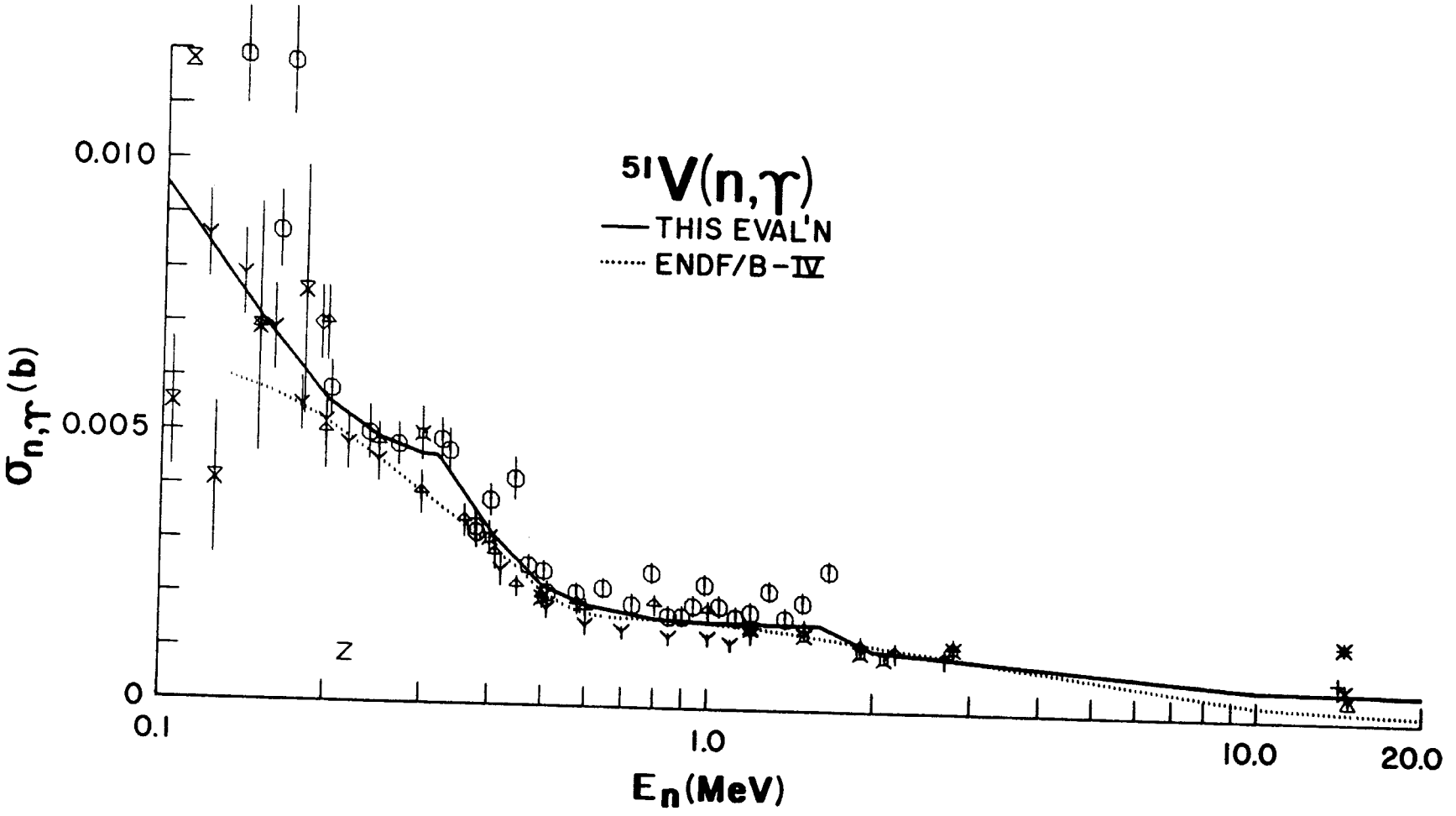
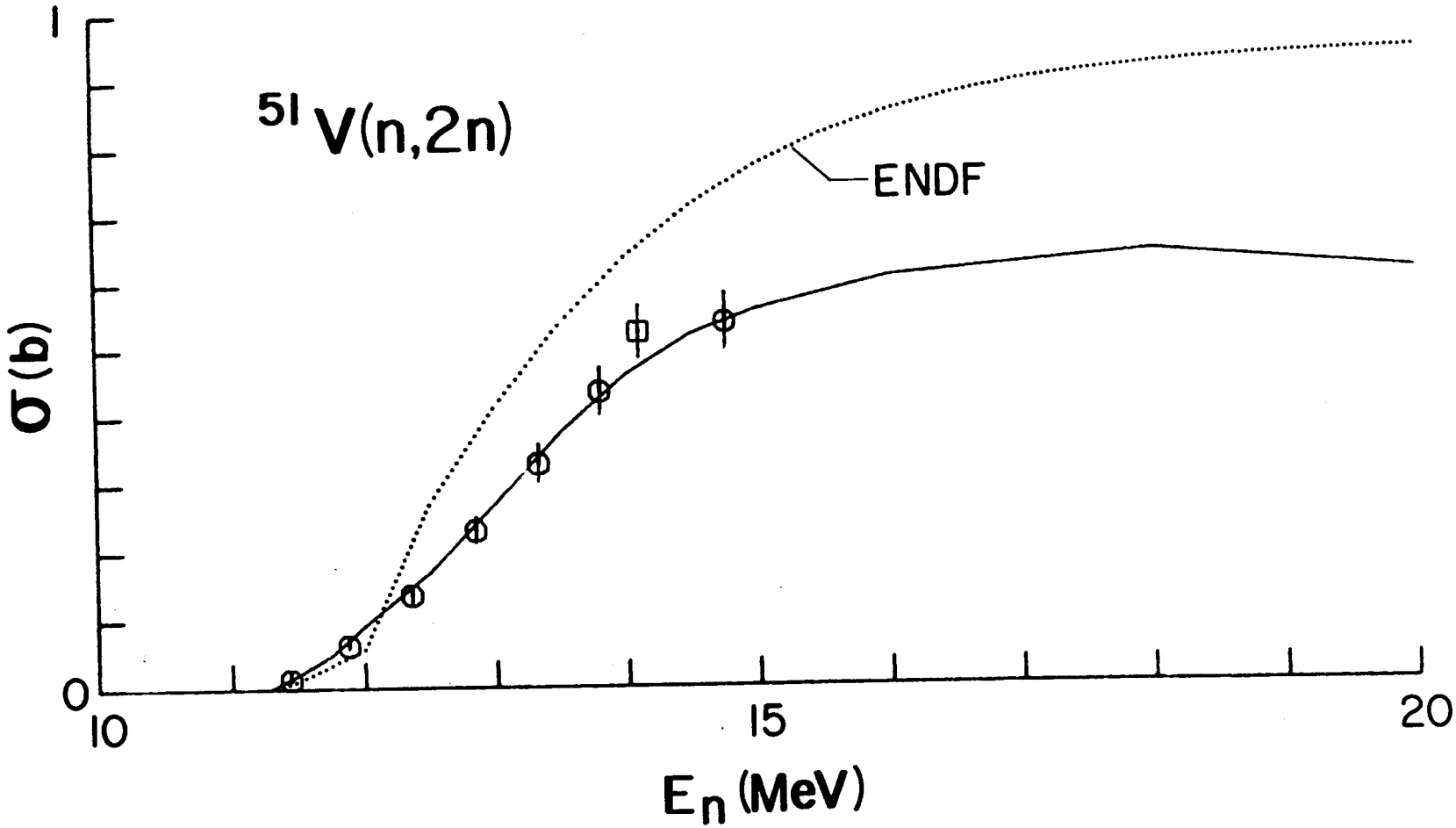

$^{51}V(n,\gamma)$
THIS EVAL'N
ENDF/B-IV
$\sigma_{n,\gamma}$ (b)
0.010
0.005
0
0.1
1.0
10.0
20.0
E_n(MeV)

Fig.10


^{51}V(n,2n)
ENDF
σ (b)
1
0
10
15
20
E_n (MeV)

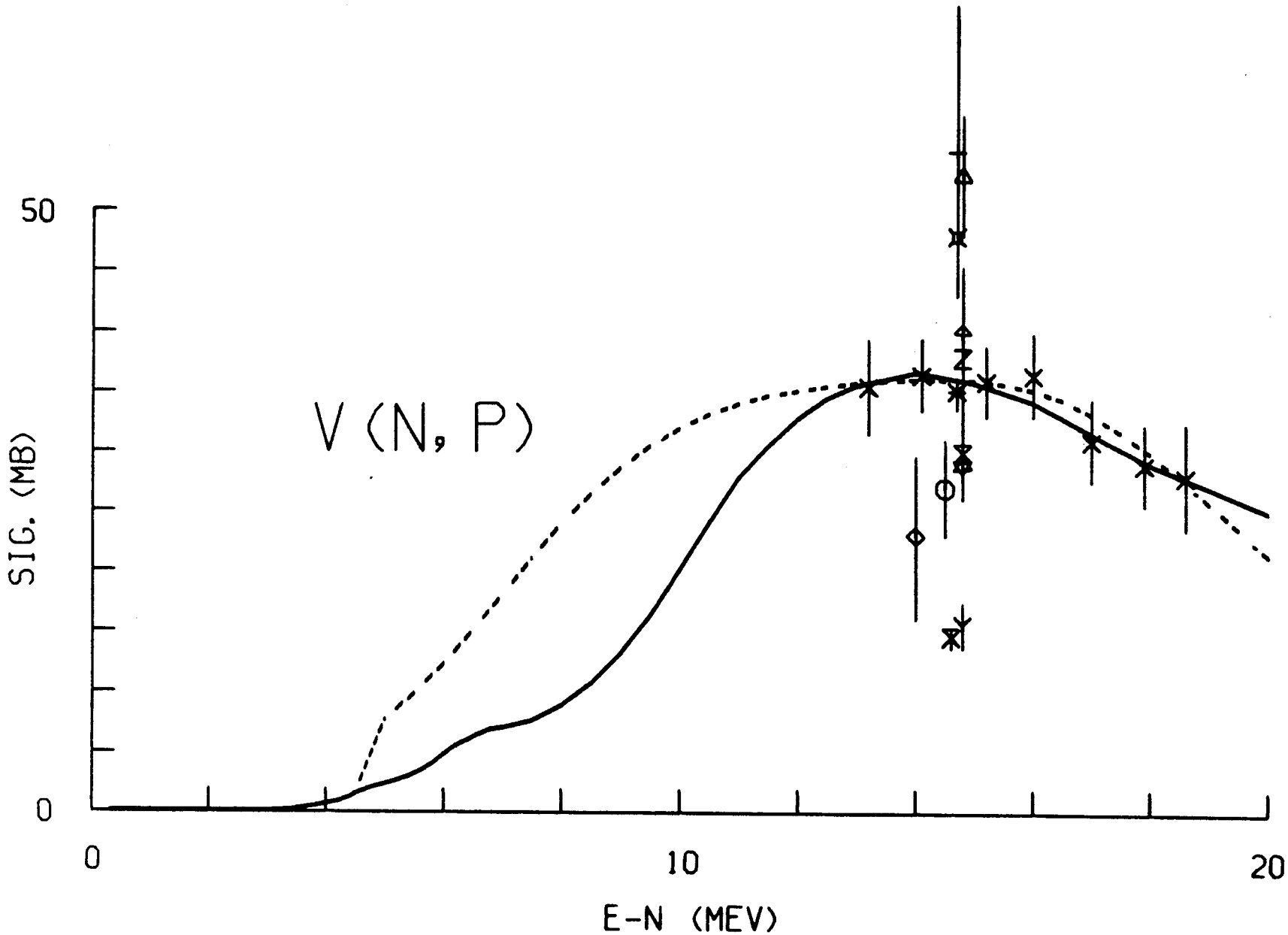


Fig.12

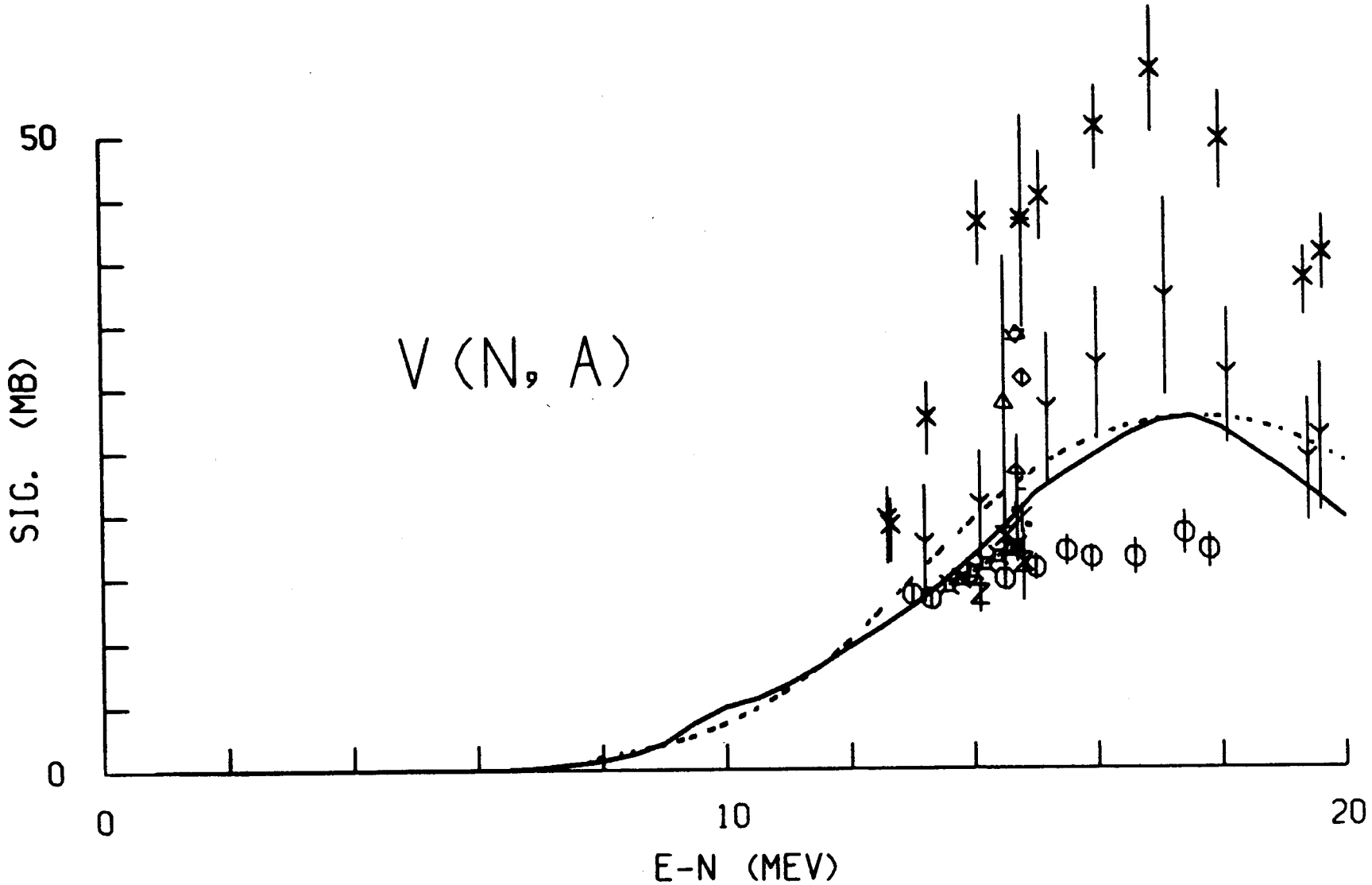


Fig.13

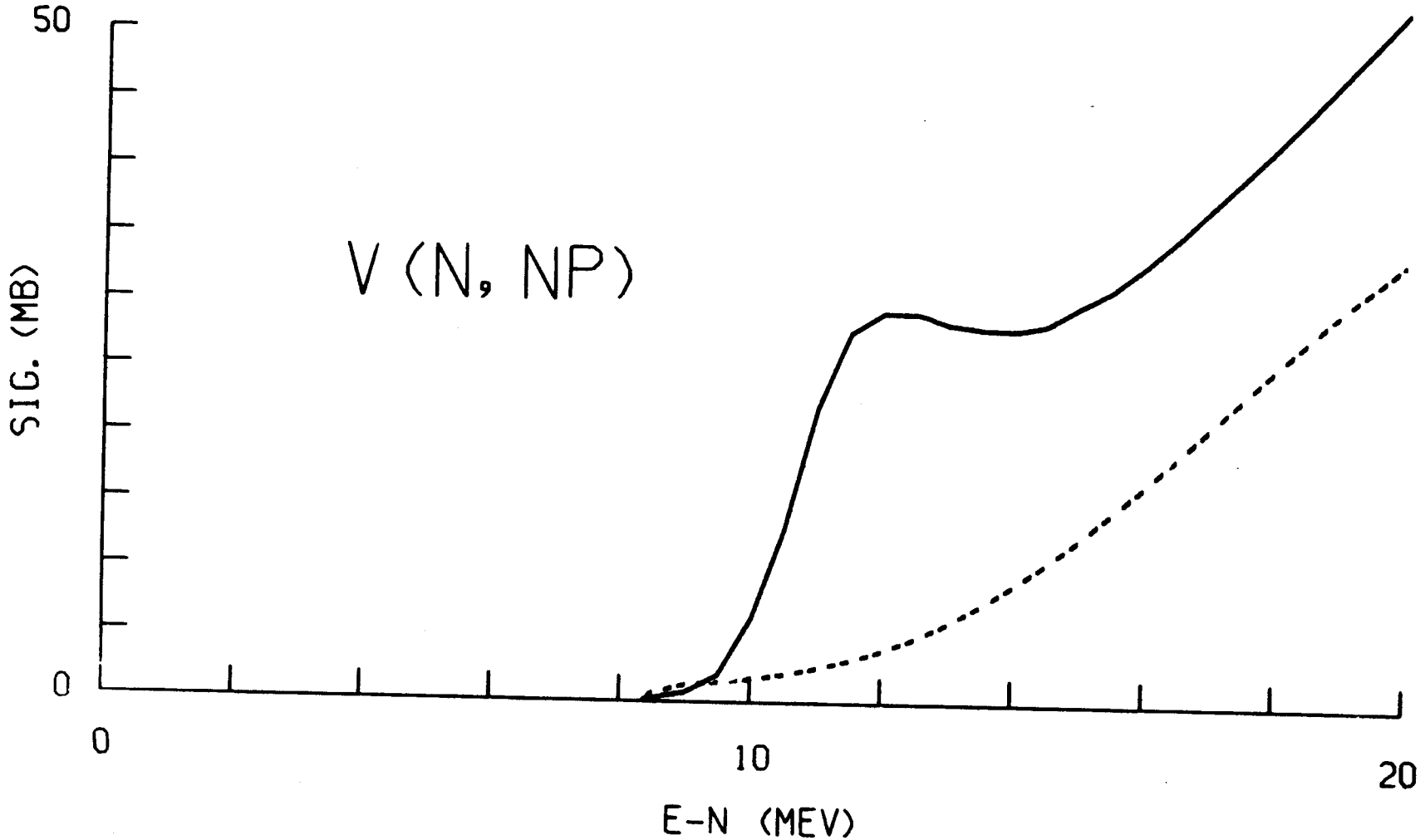


Fig.14

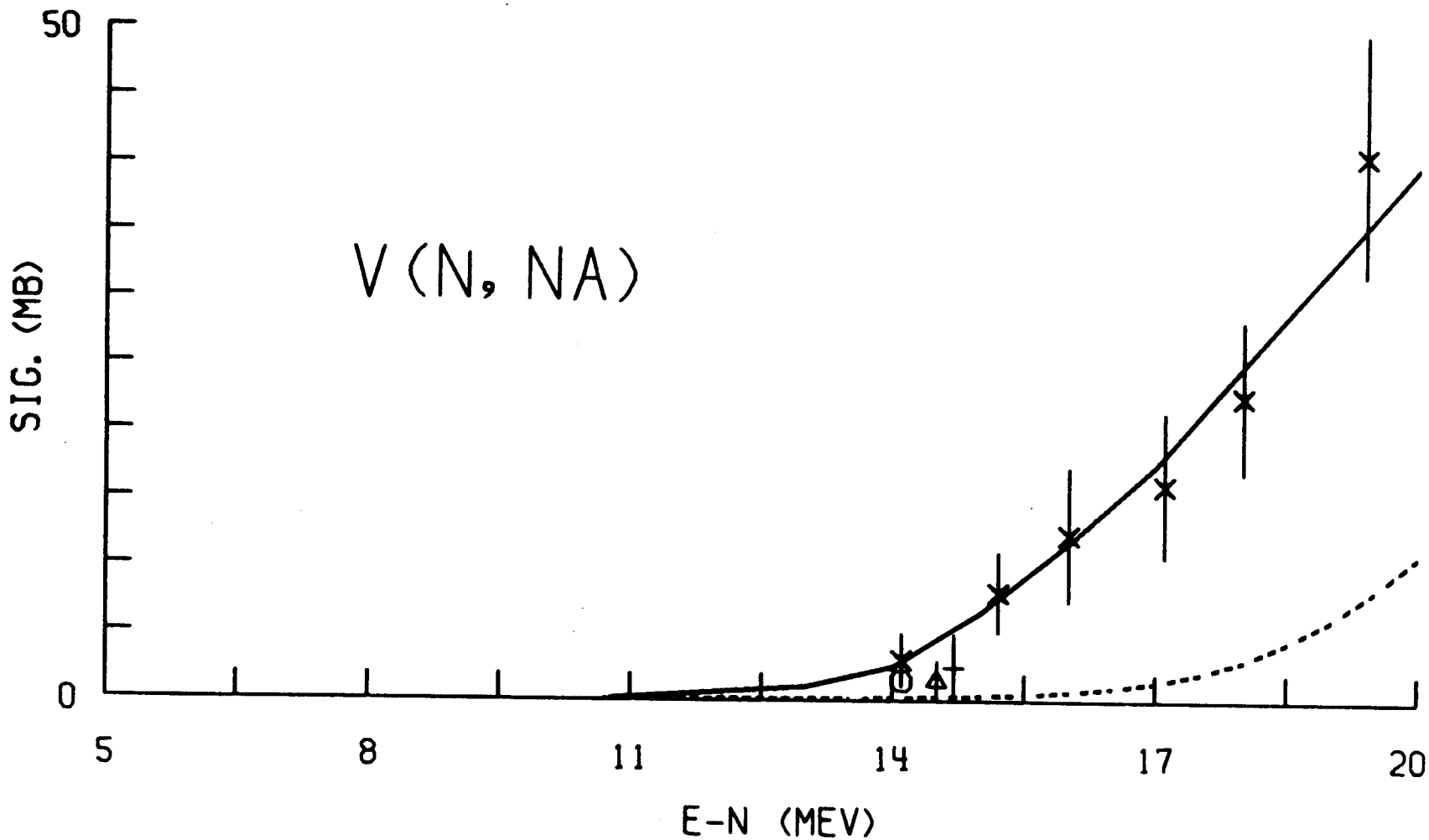


Fig.15