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ORNL/TM-13409

AN UPDATE OF THE DOSIMETRY CROSS-SECTION DATA BASE FOR THE LSL-M2 ADJUSTMENT CODE

October 2004

I. Remec

UT-BATTELLE

ORNL-27 (2-03)

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Oak Ridge National Laboratory
Manuscript Completed: April 1997
Date Published: October 2004

Prepared for Division of Engineering Technology Office of Nuclear Regulatory Research
U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001
NRC Job Code W6164

Prepared by
OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37831-6283
managed by
UT-BATTELLE, LLC
for the
U.S. DEPARTMENT OF ENERGY
under contract DE-AC05-00OR22725

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ABBREVIATIONS

ASTM	American Society for Testing and Materials
DLC	Data Library Collection
dpa	displacements per atom
ENDF	Evaluated Nuclear Data File
ENSDF	Evaluated Nuclear Structure Data File
IAEA	International Atomic Energy Agency
IRDF	International Reactor Dosimetry File
JEF	Joint Evaluated File
JENDL	Japanese Evaluated Nuclear Data Library
LB	Flag in the NI-type sub-subsection whose numerical value indicates the nature and structure of the subsection
LSL	Least-Squares Logarithmic
MAT	ENDF Numeric Material Identifier (four digits)
MT	ENDF Numeric Reaction Identifier (three digits)
NC	NC-type sub-subsection of the ENDF neutron cross-section covariance file (file 33)
NI	NI-type sub-subsection of the ENDF neutron cross-section covariance file (file 33)
ORNL	Oak Ridge National Laboratory
RDF	Russian Dosimetry File
RSIC	Radiation Shielding Information Center
SNLRML	Sandia National Laboratories Radiation Metrology Laboratory

ACKNOWLEDGMENTS

The author wishes to express his appreciation to the reviewers, J. V. Pace III of the Oak Ridge National Laboratory and Mark Williams of the Louisiana State University Nuclear Science Center, for their comments and suggestions. Special thanks goes to Carolyn Moser for providing the editorial review. Finally, the author gratefully acknowledges the programmatic support from Carolyn Fairbanks and Mike Mayfield of the Nuclear Regulatory Commission. This support was essential for accomplishing his objectives.

AN UPDATE OF THE DOSIMETRY CROSS-SECTION DATA BASE FOR THE LSL-M2 ADJUSTMENT CODE

ABSTRACT

This report describes the updated dosimetry cross-section data base for the LSL-M2 computer code package. The data base consists of a cross-section file, a nuclear-data file, cross-section files for cover materials, and a damage response functions file. The cross-section file includes cross sections from the IRDF-90/G-V2 and SNLRML-V1 libraries, as well as few reactions obtained by private communications. In total, 77 response functions are given. For 65 of the response functions, uncertainty information is provided; for six other reactions, the covariance matrixes are not given because they were not available; and for the remaining six reactions, covariance information is not given because they are not intended to be used as dosimeter response functions but may be needed to determine corrections for dosimeter impurities.

Differences in the cross sections and covariances obtained from different libraries are accessed through the comparison of averages over four theoretical spectra. In general, good consistency of cross-section data was found between the IRDF-90/G-V2 and SNLRML-V1 libraries. The cross sections for the few reactions, for which considerable differences were found, were based on different cross-section evaluations. For several reactions considerable differences were found in the cross-section covariances from the above two libraries. The impact of the differences in covariances on the results of adjustment calculations is expected to be minor, but an investigation into the causes of the differences is necessary. It was also found that the covariances of several reaction cross sections are not positive definite, which is a required property of covariance matrices, and are consequently inconsistent. Improvements in the evaluations of the uncertainties are therefore needed.

The numerical and graphical data are presented in several appendices. A comparison of cross sections and their standard deviations averaged over selected spectra is given in Appendix A. Appendix B summarizes the properties of correlation and covariance matrices of the cross sections. Appendix C contains plots of cross sections vs energy. Appendix D displays plots of cross-section correlation matrices and relative standard deviations vs energy. Appendix E gives details on the contributions from the LB = 0 and LB = 8 covariance subsections to the standard deviations of the spectrum-averaged cross sections.

1. INTRODUCTION

The first version of the LSL-M2 computer program for least-squares logarithmic adjustment of neutron spectra was released at the end of 1985. It is available from the Radiation Shielding Information Center (RSIC) (as PSR-233 micro) and documented in NUREG/CR4349 (Ref. 1). An updated version (version 2.0) was released in the fall of 1994, and a letter report describing the modifications and updating the manual was prepared. As parts of the original version of the code, the dosimetry cross sections and cross-section variances (file CROSS.XSO), nuclear data for the reactions (file LSLNUC.DAT), and the damage response functions (file DAMAGE.XSO) were provided. The cross sections and covariance data originated from ENDF-IV and IRDF-80 and were converted to a form suitable for LSL-M2. The nuclear data file was generated from Ref. 2. The damage response functions were given for the neutron flux with energy above 1 MeV ($\phi > 1$MeV), flux above 0.1 MeV ($\phi > 0.1$MeV) and displacements per atom (dpa) of iron as defined in ASTM E 693 (Ref. 3). The cross sections and response functions were given in the 640-energy-group structure defined first in SAND II (Ref. 4). For the cross-section covariances, the original group structures given in the ENDF and IRDF data were retained.

Some cross sections were later added to the original file, and some were updated, but the old data were still retained in the file. These modifications resulted in the cross-section file CROSSNEW.XSO, currently being used at ORNL for the LSL-M2 calculations.

During 1994, the dosimetry cross-section libraries IRDF-90/G-V2 (RSIC DLC-161, last update 01/12/94, Ref. 5) and SNLRML-V1 (RSIC DLC-178, Version 1.0, packaged at RSIC at 07/11/94, Ref. 6) became available.

The SNLRML-V1 library prepared by Griffin and coworkers is the Sandia National Laboratories Radiation Metrology Laboratory recommended dosimetry cross-section compendium. Thirteen libraries of evaluated cross sections were used as sources of data for the SNLRML library. Most of the data in SNLRML version 1 are drawn from the ENDF/B-VI (Ref. 7) and IRDF-90 Version 1 (IRDF-90/V1, Ref. 8). A few reactions are taken from the "GLUCS Library" (Ref. 9), JENDL-3 (Ref. 10), DOSCROS84 (Ref. 11), and ENDF/B-V libraries; a few others are obtained by private communications. The NJOY (Ref. 12) code was used to process ENDF/B-VI cross sections into the 640-energy-group SAND-II structure. Covariance matrices were also processed by NJOY and are given in a reaction-specific energy group structure as specified in ENDF/B-VI. The SNLRML provides 78 energy-dependent cross sections and three response functions. The response functions given are the dpa of iron (defined in ASTM E 693, Ref. 3), silicon displacement kerma factors (defined in ASTM E722, Ref. 3) and GaAs 1-MeV equivalent damage function (defined in ASTM E 722, Ref. 3). Out of the 78 cross section sets, 6 sets are absorption cross sections for materials usually used as dosimeter covers. The cover materials given are B_4C, natural Cd, B, Au, and B-10 and B-11 isotopes. The remaining 72 sets provide the cross sections for 57 different dosimetry reactions, 6 fission dosimeter contaminants, 6 alternative evaluations of the reaction cross sections, and 3 sets of cross sections for the specific fission foils used in SNLRML. The fission cross sections are given for the six fission dosimeter contaminants: ^{234}U, ^{236}U, ^{238}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu. The six alternative cross-section sets provided are JENDL-3D evaluation for ^{23}Na(n,γ), ^{45}Sc(n,γ), and ^{58}Fe(n,γ); ENDF/B-V for ^{32}S(n,p); ENDF/B-VI for ^{55}Mn(n,γ); and JEF2.2 for ^{58}Fe(n,γ). The three SNLRML fission foils for which detailed composition was taken into account in the cross-section and fission yields preparation are enriched uranium foil (93.00% of ^{235}U), depleted uranium foil (99.79% of ^{238}U), and plutonium foil (86.9965% ^{239}Pu). Covariance information is not provided for cover materials, fission dosimeter contaminants, or the following dosimetry reactions: Ti$(n,x)^{46}$Sc, Ti$(n,x)^{47}$Sc, ^{98}Mo(n,γ), ^{109}Ag(n,γ), ^{197}Au$(n,3n)$, ^{197}Au(n,p), and ^{232}Th$(n,2n)$.

The IRDF-90/G-V2 library was assembled by Kocherov and McLaughlin at the International Atomic Energy Agency (IAEA). It contains 58 neutron cross sections and 4 response functions. Thirty-nine cross sections are from the ENDF/B-VI, 14 are evaluations by Vonach and coworkers (Ref. 13) and 5 evaluations were prepared specially for IRDF-90/G-V2 by the Chinese Nuclear Data Center in Beijing (Ref. 14). Data from the ENDF/B-VI were processed into the 640 SAND-II format in Nuclear Data Section of the IAEA using the processing codes LINEAR, RECENT and GROUPIE by D. E. Cullen (Ref. 15). The covariance information from the ENDF/B-VI was not processed and is given in the IRDF-90/G-V2 in the original ENDF/B-VI form. Out of 58 cross-section sets, 53 are dosimetry reactions with covariance information and five are cross sections for cover materials Gd [total and (n,γ) cross section], Cd [total and (n,γ) cross section], and B (total cross section only).

The IRDF-90/G-V2 contains several most-recent cross-section evaluations that are not given in the SNLRML, Version 1. Therefore, in the preparation of the updated cross-section data base for the LSL-M2 code, the two libraries were combined.

2. METHODOLOGY

The Evaluated Nuclear Data File (ENDF) specifications were designed to accommodate a large variety of nuclear data at a time of limited computing power and are, for that reason, rather complex. On the other hand, all the specialized information needed in the dosimetry and spectrum adjustment codes can be described in a simple and straightforward format. In the LSL-M2 version 2 code package the conversion utility is provided to process the cross sections and uncertainty information from the ENDF/IRDF format to the form suitable for the use in the LSL-M2. The utility consists of two codes: CVXS and CVCT. CVXS extracts the cross sections and converts them into the SAND-II format. CVCT processes the uncertainty information. For each material a separate output file is created with the ENDF four-digit "MAT" material identifier as its name and extensions "XSS" for the cross-section file and "XSV" for the covariance file. New information is appended to existing files with the same MAT identifier. Each particular type of reaction is characterized in ENDF by a three-digit "MT" identifier which, together with MAT, occupies the first line of each data set in the output files. These two numbers must be replaced with the LSL-M2 reaction identifier and, of course, cross-section values and covariances must be combined into one data set on the same file.

The ENDF format for cross-section values is fairly straightforward, and many published files are already given in the 640-group format. ENDF coding may specify interpolations other than linear interpolation, which is ignored in the CVXS code. Also ignored are resonance parameters. ENDF allows a large number of different formats for cross-section covariance description. The CVCT code from the LSL-M2 version 2 package processes only the most important and most widely used covariance specifications. Ignored are the NC sections, which may be used to indicate that some or all of the contributions to the covariance matrix described in the subsection are to be found in a different subsection of the ENDF/B tape. The "NI" subsections are used to describe explicitly various components of the covariance matrix given in the subsection. The CVCT code processes only those with LB = 1, 2, 4, and 5. The LB = 6 subsections, which describe covariance matrices interrelating the cross sections for two different reaction types or materials, are not processed. Also ignored are the LB = 0 subsections and the LB = 8 subsections. The energy grids for all LB subsections of the same reaction are combined, and the sum of all covariances is calculated and written to the output file. This procedure is more economical than adopting a specific energy grid and does not lose any information. Conversion to a specific energy grid can be done with the FLXPRO code from the LSL-M2 code package (Ref. 1).

For the conversion of the IRDF-90/G-V2 library to the LSL-M2 format, the modified CVCT95 code was used, which also processes the LB = 0 and LB = 8 contributions to the covariance matrix. The LB = 0 and LB = 8 subsections define the absolute contributions to the covariance matrix rather than relative contributions as other LB subsections do. Therefore, cross sections in the group structure of the LB = 0 or LB = 8 subsections are needed to convert the contributions to the relative values before they can be combined with contributions from other LB subsections in the complete reaction covariance matrix. Therefore, the following modifications were incorporated into the CVCT95 code. The CVCT95 code merges the energy groups of all LB sections (LB = 0 and LB = 8 included) into one structure (CVCT ignores LB = 0 and LB = 8), and it does not stop if it encounters LB = 0. If subsection LB = 0 or LB = 8 is found for the MAT and MT being processed, the code tries to read the MAT-MT.XSG file, which is expected to contain the MAT-MT cross sections in the merged covariance group structure. If MAT-MT.XSG is available, the code reads the cross sections and processes the LB = 8 and/or LB = 0 information, together with all other subsections, and writes the reaction covariance matrix on the output file. If MAT-MT.XSG is not available, the code creates the MAT-MT.CVD file, which contains the merged covariance group structure. The covariance matrix is generated, but it does not include contributions from LB = 0 and/or LB = 8; a warning is written on the CVCT.LOG file. After the first CVCT run, FLXPRO code is used to generate the reaction cross sections in the merged covariance group structure. These cross sections must be placed on the MAT-MT.XSG file, and then the CVCT95 is run for

the second time. The second run generates the covariance matrices with LB = 8 and LB = 0 taken into account. In this procedure, a reference spectrum is used in FLXPRO to collapse the cross sections to the merged covariance group structure. For a few reactions on the IRDF-90/G-V2 library, the choice of the reference spectrum has some effect on the final reaction covariances (see Appendix E). Detailed discussion of the impact of the LB = 8 and LB = 0 subsections on the standard deviations of the spectrum-averaged cross sections is given in Appendix E.

The SNLRML library was prepared for the SAND-II, an iterative spectrum unfolding code. It is very similar in structure to the LSL-M2 cross-section library. The conversion utility code SANDLSL was written for conversion of the SNLRML library to the LSL-M2 format. The utility converts cross sections from barns to cm^2, converts the standard deviations from percentages to the standard deviations specified as fractions, and normalizes the correlation coefficients to 1000 and converts to integers, as required in LSL-M2. Each reaction is written on a separate file with a file name equal to the SNLRML reaction identifier and extension .XSO. These files can be incorporated in the LSL-M2 library. It may only be necessary to change the reaction identifier on the first card to make it consistent with LSL-M2.

For a few reactions, the data from the Russian Dosimetry File (RDF-94, version 2, evaluations by K. Zolotarev, S. Badikov, and A. Pashchenko) were obtained by private communication.[1] These data were in the ENDF-6 format and were converted to the LSL-M2 format, as described above for the IRDF-90 library.

[1] Private communication from T. Seren (VTT Chemical Technology, Findland) to F. W. Stallmann (forwarded to I. Remec), May 1994.

3. CONTENT OF THE UPDATED CROSS-SECTION DATA BASE

The updated cross-section data base for the LSL-M2 code consists of the following:

1. cross sections and covariances on the CROSS-95.XSO file,
2. response functions for the irradiation damage parameters on the DAMAGE95.XSO file,
3. the LSLNUC95.DAT nuclear data file, and
4. cross sections for materials often used as covers for dosimeters.

The CROSS-95.XSO file contains data for 77 reactions. The covariances are not given for a few of the reactions because they were not provided in the original libraries. These reactions are Ti$(n,x)^{46}$Sc, Ti$(n,x)^{47}$Sc, ^{98}Mo(n,γ), ^{197}Au$(n,3n)$, ^{197}Au(n,p), and ^{232}Th$(n,2n)$. Because of the lack of covariances, these reactions cannot be used directly in adjustment calculation but were retained in the library for the sake of completeness and possible future improvements. Only the cross sections are given for the fission foil "contaminants" ^{234}U, ^{236}U, ^{238}Pu, ^{240}Pu, ^{241}Pu, and ^{242}Pu. These reactions are not intended to be used as dosimetry reactions in spectrum adjustments. However, when fission foils are used, it may be necessary to take into account the presence of contaminants. Having the contaminant cross sections available in the standard cross-section library greatly simplifies the task of determining the necessary corrections. For most of the reactions in CROSS-95.XSO, two or more cross-section sets are given. If, for the reaction considered, the IRDF-90/G-V2 data set is available, it is given first, followed by the set from SNLRML and alternative sets. The first record of each data set begins with the reaction identifier. In CROSS-95.XSO, the information about the data set is added, consisting of the name of the source library from which the data were obtained (e.g., IRDF-90/G-V2), the material number MAT (e.g., 525 for B-10), and the reaction number MT as used in the source library (e.g., 107 for (n,α) reaction in IRDF-90/G-V2), and origin of data as specified in the source library (e.g., ENDF/B-VI). This information is not used in the cross-section processing but allows a user to distinguish between different cross-section sets provided for the same reaction (and therefore having the same reaction identifier). In the second record, following the required data, some additional information is given for the cross-section sets from the SNLRML library. Additional information in the second record is the atomic weight of target element, decay constant in s^{-1} or, if preceded by minus signs, half-life in hours, isotopic abundance, alphanumeric reaction ID (as used in the SNLRML), source library of data, and temperature (in degrees K) for which the cross sections were prepared. These data were transferred from the SNLRML library and are not used in the LSL-M2 calculations. The FLXPRO code for cross-section and covariance processing allows the user to process only the first set or all the sets of cross sections for the selected reaction in a single run. The second option allows, for example, easy intercomparisons of the different cross-section sets averaged over selected spectra. Then, once the user decides on the cross-section sets he or she prefers to use, he or she can reorder the cross sections in the library file and use the "process only first set for each reaction" option.

DAMAGE95.XSO file contains the response functions for the ϕ > 1MeV; ϕ> 0.1MeV; and dpa cross sections for Fe, Cr, and Ni. For iron, the ASTM and EURATOM dpa cross sections are given. The silicon displacement kerma factors (as used in ASTM E722-93, Ref. 3) and GaAs 1-MeV equivalent damage function (defined in ASTM E 722, Ref. 3) are given also.

LSLNUC95.DAT is a nuclear data file that provides, for each reaction, the target element atomic weight, decay constant (in s^{-1}), target isotope atom abundance, and, for fission products, the fission yields. Decay constants were assembled primarily from SNLRML data, which were in turn assembled from the Evaluated Nuclear Structure Data File (ENSDF, Ref. 16) and Nuclear Wallet Cards, an on-line data base at the National Nuclear Data Center at Brookhaven National Laboratory, and in a few cases from the Lemmel's detector calibration standards (Ref. 17). The atomic weights and isotopic abundances were taken from the Nuclear Wallet Cards (Ref. 18). Fission yields were obtained from the ENDF/B-VI files,

for the incident neutron energy of 0.5 MeV. These yields are appropriate for fast fissions. If fissions induced by thermal or 14-MeV neutrons are of importance, the corresponding fission yields need to be obtained from the ENDF files.

The cover material cross sections are given on the files with extension COV. They are in a format appropriate to be used by the COVER code from the LSL-M2 code package. The total cross sections for B, Cd, and Gd are given on the B-TOT, CD-TOT, and GD-TOT files, respectively. The (n,γ) cross sections for Cd and Gd are given in the CD-GAM and GD-GAM files. The neutron absorption cross sections for B, B-10, B-11, Cd, B_4C, and Au are given in files B-ABS, B10-ABS, B11-ABS, CD-ABS, B4C-ABS, and AU-ABS, respectively. The second record of each *.COV file gives the density of the material (in grams cm^{-3}) and atomic weight, which is used in the COVER code. The B_4C cross sections were taken from Ref. 6. They were prepared for the fission foils cover material used in the SNLRMR. The B_4C cross sections are given for B enriched to 91.67% in B-10 and are normalized per atom in B_4C. Therefore, to maintain consistency with COVER code, only one-fifth of the molecular weight is listed instead of the full weight. A user needs to verify that the particular B-10 content in B_4C is adequate for his application. In the B10-ABS.COV and B11-ABS.COV, only the densities and atomic weights for natural boron are given. The user needs to change these values as needed before using these data.

4. DISCUSSION

For comparison, the cross sections from CROSS-95.XSO were averaged over the three theoretical spectra specified in Ref. 19 and an additional combined spectrum often used for cross-section collapsing purposes at ORNL. These spectra are defined as follows:

1. Maxwell spectrum at a neutron temperature of 293.58 K (peak at $E = 0.025$ eV)
   $f(E) = 1.562395 \times 10^{+15} \times E \times e^{-(39527140 \times E)}$

2. $1/E$ spectrum between 0.536 eV and 1.05 MeV, and zero outside this interval
   $f(E) = 1/E$ for 0.536 eV $\leq E \leq$ 1.05 MeV
   $f(E) = 0$ outside this interval

3. Watt fission spectrum (with maximum at $E = 0.72$ MeV)
   $f(\mathrm{E}) = 0.484 \times \sinh (2E)^{0.5} \times e^{-E}$

4. "Combined" spectrum consists of Watt fission spectrum matched at 0.5 MeV to $1/E$ and at 0.14 eV to Maxwellian spectrum with neutron temperature of 323.16 K.

In the above expression the neutron energy E is given in MeV.

Cross sections averaged over the spectra specified above are given in Table A.1 in Appendix A. For the 1/E region the resonance integral is given rather than the average cross section. The resonance integral is calculated as the integral of the cross section multiplied by the 1/E spectrum from 0.536 eV to 1.05 MeV.

For each reaction, the spectrum-averaged cross section is given (in cm^2) for the first cross-section set in the library. If more than one set of cross sections is present the cross-section averages for the consecutive sets are divided by the value for the first set, and ratios are given. The last record gives data for the cross-section library CROSSNEW.XSO, which is currently being used at ORNL. The average cross sections were calculated with the FLXPRO code.

For the cross sections averaged over the combined spectrum, good agreement is found between the values calculated from the IRDF-90/G-V2 and SNLRML-V1 libraries. In general, differences do not exceed a few tenths of a percent. However, for a few reactions the differences are considerable:

Reaction	^{55}Mn(n,γ)	^{93}Nb$(n,2n)$	^{109}Ag(n,γ)	^{127}I$(n,2n)$	^{237}Np(n,f)
Difference (%)	1.8	8.3	14.2	10.1	3.2

For all of these reactions, the two cross-section libraries used different evaluated cross-section sets. The IRDF-90/G-V2 provides cross sections from more recent evaluations than SNLRML-V2 in all the cases except for ^{237}Np, which was taken from ENDF/B-IV for the IRDF-90/G-V2 and from ENDF/B-VI for SNLRML-V1.

When resonance integrals and averages over the Maxwell and Watt fission spectrum are compared, generally a good agreement between the IRDF-90/G-V2 and SNLRML-V1 is found, except for the reactions discussed above. However, for a few other reactions big differences are observed between the averages over the energy regions that are usually not important for the integral dosimeter response. Such examples are ^{45}Sc(n,γ) and ^{59}Co(n,γ) for which the averages over the fission spectrum, calculated for the IRDF-90/G-V2 and SNLRML-V1 differ by 9.6% and 20.7%, respectively. The average values of the alternative cross-section sets generally differ from the IRDF-90/G-V2 and SNLRML-V1 values by more

than the differences between these two libraries. The CROSSNEW.XSO library is generally in good agreement with the IRDF-90/G-V2 library, except for the following:

Reaction	^{58}Fe(n,γ)	^{59}Co(n,γ)	^{63}Cu$(n,2n)$	^{93}Nb$(n,2n)$	^{127}I$(n,2n)$	^{237}Np(n,f)
Difference (%)	9.7	2.5	1.7	8.3	10.1	3.2

Table A.1 also compares the cross-section standard deviations. For each reaction, the standard deviation for the first cross-section set in the library is given in percentage. If more than one set of the cross sections is given for the reaction, the standard deviations for the consecutive sets are divided by the standard deviation for the first set and the quotients are listed. The average standard deviations were calculated with the FLXPRO code. The locations in the Table A.1 denoted with "-" indicate the energy ranges with zero cross sections, where the standard deviations are not defined. For the majority of the reactions, the standard deviations calculated for the IRDF-90/G-V2 and SNLRML-V1 libraries are in good agreement; however, for many reactions the relative differences between the standard deviations are considerably larger than differences between the average cross sections. For example, even though the same value for the ^{27}Al(n,α) cross section averaged over the combined spectrum is obtained for the IRDF-90/G-V2 and SNLRML-V1 library, the corresponding standard deviation from the SNLRML-V1 library is 1.58 times larger than the standard deviation from the IRDF-90/G-V2 library. For this reaction the two libraries have covariance information drawn from different sources. However, by comparing the data from the two libraries for the reaction ^{58}Fe(n,γ), one observes that, while both libraries claim the same origin (i.e., ENDF/B-VI) and the two average cross sections agree within 0.5%, the standard deviation is 22% larger for the SNLRML-V1 library than for the IRDF-90/G-V2 library. The largest difference is observed for the ^{127}I$(n,2n)$ reaction, for which the standard deviation averaged over the combined spectrum is 2.2% for the IRDF-90/G-V2 (source is a new evaluation) and 17.2%; almost eight times higher for the SNLRML-V1 (source is ENDF/B-V). Similar observations can be drawn from a comparison of IRDF-90/G-V2 and CROSSNEW.XSO libraries. The largest change in the standard deviation was observed for the ^{32}S(n,p) reaction, from 40.5% for the CROSSNEW.XSO to 3.2% for the IRDF-90/G-V2. Generally, the standard deviations calculated from the IRDF-90/G-V2 file tend to be smaller than the standard deviations from older evaluations, which is, of course, the expected and favorable trend.

For the spectrum adjustment calculations, the covariances of the calculated reaction rates are important parameters. They consist of the dosimetry cross-section covariances and the covariances of the calculated neutron flux. The flux covariances are usually the dominant contributors. Therefore, even though the differences in the cross-section covariances between different libraries are considerable, the impact of these differences on the results of the neutron spectrum adjustment calculations is expected to be minor.

The mathematical properties of the covariance and correlation matrices are listed in Table B.1 in Appendix B. Out of 65 reactions with uncertainty information, 21 have positive definite covariance and correlation matrices; their uncertainty information is therefore consistent. These reactions are denoted with the flag "OK" in the column K of Table B.1. Twenty-six threshold reactions have positive definite correlation matrices, but have zero standard deviation specified for one energy group covering the region below threshold, and non-negative covariance matrices (with one eigenvalue equal to zero). These covariance matrices can be considered acceptable because they can easily be made positive definite by truncating the first energy group (below threshold), which is not needed. In Table B.1 a label "A" in the column K denotes these reactions. Six reactions have consistent correlation matrices but have an energy region with nonzero cross section and zero standard deviation; therefore the covariance matrices have one eigenvalue equal to zero. These reactions are denoted with label "B" in Table B.1. Two reactions have consistent correlation matrices, but a region with nonzero cross section with zero standard deviation and covariance matrices with negative eigenvalues. Label "C" in the column K marks these reactions. Finally, for ten reactions both the correlation and covariance matrices have negative eigenvalues and are therefore

not consistent. In Table B.1 label "D" in column K denotes these reactions. Uncertainty information of the eighteen reactions (labeled by "B," "C," and "D" in Table B.1) should probably be reevaluated.

5. CONCLUSION

The updated dosimetry cross-section library, named CROSS-95.XSO, for the LSL-M2 computer code package includes cross sections from IRDF-90/G-V2 and SNLRML-V1 libraries as well as a few reactions obtained from private communications. In general, good consistency of the cross-section data was found between the IRDF-90/G-V2 and SNLRML-V1 dosimetry libraries. However, for the reactions ^{55}Mn(n,γ), ^{93}Nb$(n,2n)$, ^{109}Ag(n,γ), ^{127}I$(n,2n)$, and ^{237}Np(n,f), considerable differences were found. These variations appear to be caused by the disparities in the original data since the two libraries used different cross section evaluations for the above listed reactions.

For some reactions the uncertainty information is not consistent and should be reevaluated. The inconsistencies found are energy regions with nonzero cross sections but zero standard deviations, and correlation and covariance matrices that are not positive definite. For several reactions considerable differences were also found in the cross-section covariances from the IRDF-90/G-V2 and SNLRML-V1 libraries. The impact of differences in covariances on results of adjustment calculations is expected to be minor, but it is nevertheless felt that an investigation into the causes of the differences is necessary.

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APPENDIX A

COMPARISON OF SPECTRUM-AVERAGED CROSS SECTIONS AND STANDARD DEVIATIONS

Table A.1. Summary of cross sections and standard deviations by reaction

Reaction	Averaged cross sections (cm^2)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Li6(N,T)H4													
	8.340E-22	3.231E-25	4.004E-22	2.131E-22	0.14	0.57	0.14	0.14	IRDF-90/G-V2	325	105	ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				
Li6(N,X)He4													
	8.349E-22	4.678E-25	4.008E-22	2.134E-22	0.14	2.39	0.14	0.14	SNLRML-V1	325	851	ENDF/B-VI	IRDF-90
B10(N,A)Li7													
	3.403E-21	4.56E-25	1.624E-21	8.689E-22	0.16	6.05	0.16	0.16	IRDF-90/G-V2	525	107	ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				
B10(N,X)He4													
	3.407E-21	5.419E-25	1.626E-21	8.701E-22	0.16	6.60	0.16	0.16	SNLRML-V1	525	851	ENDF/B-VI	IRDF-90
F19(N,2N)F18													
	0	6.204E-30	0	1.387E-30	-	2.50	-	2.50	IRDF-90/G-V2	925	16	Original	Original
	0	1.006	0	1.006	-	1.22	-	1.22	SNLRML-V1	925		IRDF-90	IRDF-90
	0	1.006	0	1.006	-	1.22	-	1.22	CROSSNEW.XSO				
Na23(N,G) Na24													
	4.680E-25	2.743E-28	3.022E-25	1.226E-25	2.00	12.89	3.02	1.95	IRDF-90/G-V2	1123	102	Priv comm	Priv comm
	1.001	1.014	1.001	1.001	1.00	1.06	1.53	1.02	SNLRML-V1	1125		ENDF/B-VI	ENDF/B-VI
	1.009	0.830	0.986	1.007	1.00	0.99	0.96	1.01	SNLRML-V1	1131		JENDL-3D	JENDL-3D
	1.000	1.013	1,000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				ENDF/B-V
Mg24(N,P)Na24													
	0	1.474E-27	0	3.294E-28	-	2.35	-	2.35	IRDF-90/G-V2	1225	103	Original	Original
	0	1.001	0	1.001	-	1.00	-	1.00	SNLRML-V1	1225		IRDF-90	IRDF-90
	0	1.001	0	1.001	-	1.00	-	1.00	CROSSNEW.XSO				
Al27(N,P)Mg27													
	0	3.826E-27	0	8.552E-28	-	5.53	-	5.53	IRDF-90/G-V2	1325	103	Priv comm	Priv comm
	0	1.000	0	1.000	-	1.08	-	1.08	SNLRML-V1	1325		IRDF-90	GLUCS
	0	1.000	0	1.000	-	1.40	-	1.4	CROSSNEW.XSO				
Al27(N,A)Na24													
	0	6.863E-28	0	1.534E-28	-	1.40	-	1.4	IRDF-90/G-V2	1325	107	Original	Original
	0	1.000	0	1.000	-	1.58	-	1.58	SNLRML-V1	1325		IRDF-90	GLUCS
	0	1.000	0	1.000	-	0.99	-	0.99	CROSSNEW.XSO				
P31(N,P)Si31													
	0	2.783E-26	0	6.221E-27	-	3.56	-	3.56	IRDF-90/G-V2	1525	103	Original	Original
	0	1.000	0	1.000	-	1.00	-	1.00	SNLRML-V1	1525		IRDF-90	IRDF-90
	0	1.000	0	1.000	-	1.00	-	1.00	CROSSNEW.XSO				
S32(N,P)P32													
	0	6.346E-26	1.225E-30	1.419E-26	-	3.22	38.82	3.22	IRDF-90/G-V2	1625	103	Priv comm	Priv comm
	0	0.999	0.999	0.999	-	1.00	0.87	1.00	SNLRML-V1	1625		IRDF-90	GLUCS
	0	1.081	0.767	1.08	-	-	-	-	SNLRML-V1	6439		ENDF/B-VD	-
	0	0.999	1.000	0.999	-	12.58	1.15	12.58	CROSSNEW.XSO				
Sc45(N,G)Sc46													
	2.411E-23	5.065E-27	1.118E-23	6.144E-24	0.75	8.94	0.78	0.75	IRDF-90/G-V2	2126	102	Priv comm	Priv comm
	1.000	1.096	1.016	1.001	1.09	0.36	1.45	1.17	SNLRML-V1	2125		ENDF/B-VI	ENDF/B-VI
	1.001	1.422	1.000	1.001	1.09	0.37	1.45	1.17	SNLRML-V1	2131		JENDL-3D	JENDL-3D
	0.998	1.095	1.014	0.999	1.09	0.36	1.46	1.17	CROSSNEW.XSO				
Ti46(N,2N)Ti45													
	0	3.220E-30	0	7.199E-31	-	2.21	-	2.21	RDF-94	2212	16	Priv comm	Priv comm
Ti46(N,P)Sc46													
	0	1.002E-26	0	2.240E-27	-	2.48	-	2.48	IRDF-90/G-V2	2225	103	Priv comm	Priv comm
	0	1.000	0	1.000	-	1.00	-	1.00	SNLRML-V1	2225		IRDF-90	GLUCS
	0	1.044	0	1.043	-	0.99	-	0.99	RDF-94	2212	103	Priv comm	Priv comm
	0	1.000	0	1.000	-	1.60	-	1.60	CROSSNEW.XSO				

Table A.1. (continued)

Reaction	Averaged cross sections (cm²)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Ti47(N,NP)Sc46													
	0	7.961E-30	0	1.779E-30	-	30.00	-	30.00	IRDF-90/G-V2	2228	28	ENDF/B-VI	ENDF/B-VI
	0	1.001	0	1.002	-	1.00	-	1.00	SNLRML-V1	2228		ENDF/B-VI	ENDF/B-VI
Ti47(N,P)Sc47													
	0	1.760E-26	3.231E-28	3.934E-27	-	3.77	5.00	3.78	IRDF-90/G-V2	2228	103	ENDF/B-VI	ENDF/B-VI
	0	1.000	1.000	1.000	-	1.00	1.00	1.00	SNLRML-V1	2228		IRDF-90	GLUCS
	0	1.000	1.000	1.000	-	1.14	1.57	1.13	CROSSNEW.XSO				
Ti47(N,X)Sc46													
	0	4.739E-30	0	1.059E-30	-	7.59	-	7.59	RDF-94	2222	28	Priv comm	Priv comm
Ti48(N,NP)Sc47													
	0	1.303E-30	0	2.912E-31	-	30.00	-	30.00	IRDF-90/G-V2	2231	28	ENDF/B-VI	ENDF/B-VI
	0	1.001	0	1.001	-	1.00	-	1.00	SNLRML-V1	2231		ENDF/B-VI	ENDF/B-VI
Ti48(N,P)Sc48													
	0	2.597E-28	0	5.804E-29	-	2.56	-	2.56	IRDF-90/G-V2	2231	103	Priv comm	Priv comm
	0	1.000	0	1.000	-	1.00	-	1.00	SNLRML-V1	2231		IRDF-90	GLUCS
	0	1.000	0	1.000	-	1.56	-	1.56	CROSSNEW.XSO				
Ti(N,X)Sc46													
	0	1.003E-26	0	2.241E-27	-	-	-	-	SNLRML-V1	2200	851	Priv comm	
Ti(N,X)Sc47													
	0	1.762E-26	3.231E-28	3.938E-27	-	-	-	-	SNLRML-V1	2200	852	Priv comm	-
V(N,A)Sc													
	0	2.250E-29	0	5.029E-30	-	3.38	-	3.38	IRDF-90/G-V2	2300	107	Priv comm	Priv comm
Cr52(N,2N)Cr51													
	0	3.195E-29	0	7.141E-30	-	2.62	-	2.62	IRDF-90/G-V2	2431	16	Original	Original
	0	1.001	0	1.001	-	1.00	-	1.00	CROSSNEW.XSO				
Mn55(N,2N)Mn54													
	0	2.007E-28	0	4.486E-29	-	9.23	-	9.23	IRDF-90/G-V2	2525	16	ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.001	-	1.50	-	1.50	SNLRML-V1	2525		ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	1.00	-	1.00	CROSSNEW.XSO				
Mn55(N,G)Mn56													
	1.192E-23	2.937E-27	1.144E-23	3.267E-24	4.3	7.46	3.93	4.24	IRDF-90/G-V2	2525	102	ENDF/B-VI	ENDF/B-VI
	0.987	0.991	1.219	1.018	1.05	1.19	1.09	1.05	SNLRML-V1	7255		ENDF/B-VA	ENDF/B-VI
	0.999	1.000	0.998	0.998	-	-	-	-	SNLRML-V1	2525		ENDF/B-VI	-
Fe54(N,2N)Fe53													
	0	9.093E-31	0	2.033E-31	-	4.67	-	4.67	RDF-94	2612	16	Priv comm	Priv comm
Fe54(N,P)Mn54													
	0	7.881E-26	3.369E-29	1.762E-26	-	2.12	11.72	2.12	IRDF-90/G-V2	2625	103	Priv comm	Priv comm
	0	1.000	0.995	0.999	-	1.00	1.01	1.00	SNLRML-V1	2625		ENDF/B-VI	GLUCS
	0	1.000	1.000	1.000	-	1.12	1.03	1.12	CROSSNEW.XSO				
Fe54(N,A)Cr51													
	0	8.653E-28	0	1.934E-28	-	8.39	-	8.39	SNLRML-V1	2625	107	ENDF/B-VI	ENDF/B-VI
Fe56(N,P)Mn56													
	0	9.862E-28	0	2.204E-28	-	2.26	-	2.26	IRDF-90/G-V2	2631	103	ENDF/B-VI	ENDF/B-VI
	0	0.999	0	1.000	-	1.00	-	1.00	SNLRML-V1	2631		ENDF/B-VI	GLUCS
	0	1.000	0	1.000	-	1.21	-	1.21	CROSSNEW.XSO				
Fe58(N,G)Fe59													
	1.023E-24	2.548E-27	1.479E-24	2.998E-25	5.81	27.44	5.36	5.57	IRDF-90/G-V2	2637	102	ENDF/B-VI	ENDF/B-VI
	0.998	1.000	0.988	0.996	1.21	1.00	1.07	1.22	SNLRML-V1	2637		ENDF/B-VI	ENDF/B-VI
	1.104	0.739	1.037	1.09	0.28	0.87	0.78	0.27	SNLRML-V1	3264		JENDL-3D	JENDL-3D
	1.105	0.806	0.926	1.07	-	-	-	-	SNLRML-V1	2637		JEF 2.2	-
	1.111	0.993	1.041	1.097	0.28	1.14	0.82	0.28	CROSSNEW.XSO				ENDF/B-V

Table A.1. (continued)

Reaction	Averaged cross sections (cm^2)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Co59(N,2N)Co58													
	0	1.720E-28	0	3.844E-29	-	3.07	-	3.07	IRDF-90/G-V2	2725	16	Original	Original
	0	0.999	0	1.000	-	0.98	-	0.98	SNLRML-V1	2725		IRDF-90	IRDF-90
	0	1.000	0	1.000	-	0.98	-	0.98	CROSSNEW.XSO				
Co59(N,G)Co60													
	3.311E-23	6.236E-27	7.354E-23	1.068E-23	0.65	4.60	0.77	0.69	IRDF-90/G-V2	2725	102	ENDF/B-VI	ENDF/B-VI
	0.997	0.793	1.013	1.002	1.00	0.88	1.00	1.00	SNLRML-V1	2725		ENDF/B-VI	ENDF/B-V
	0.999	0.793	1.015	1.004	1.00	0.88	1.00	1.00	CROSSNEW.XSO			ENDF/B-V	ENDF/B-V
	0.998	0.784	1.099	1.025	3.08	2.98	12.35	4.46	CROSSNEW.XSO			CND	CND
Co59(N,P)Fe59													
	0	1.380E-27	0	3.084E-28	-	4.11	-	4.11	SNLRML-V1	2725	103	ENDF/B-VI	ENDF/B-VI
Co59(N,A)Mn56													
	0	1.467E-28	0	3.280E-29	-	2.81	-	2.81	IRDF-90/G-V2	2725	107	ENDF/B-VI	ENDF/B-VI
	0	0.999	0	0.999	-	1.00	-	1.00	SNLRML-V1	2725		ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	1.00	-	1.00	CROSSNEW.XSO				
Ni58(N,2N)Ni57													
	0	2.948E-30	0	6.589E-31	-	3.10	-	3.10	IRDF-90/G-V2	2825	16	Original	Original
	0	1.001	0	1.001	-	0.76	-	0.76	SNLRML-V1	2825		ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	0.94	-	0.94	CROSSNEW.XSO				
Ni58(N,P)Co58													
	0	1.038E-25	6.275E-28	2.321E-26	-	2.22	9.76	2.22	IRDF-90/G-V2	2825	103	ENDF/B-VI	ENDF/B-VI
	0	1.000	0.998	1.000	-	1.10	1.36	1.10	SNLRML-V1	2825		ENDF/B-VI	GLUCS
	0	1.000	1.000	1.000	-	1.28	1.91	1.28	CROSSNEW.XSO				
Ni60(N,P)Co60													
	0	1.868E-27	0	4.175E-28	-	10.65	-	10.65	IRDF-90/G-V2	2831	103	ENDF/B-VI	ENDF/B-VI
	0	0.999	0	1.000	-	0.98	-	0.98	SNLRML-V1	2831		ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	0.99	-	0.99	CROSSNEW.XSO				
Cu63(N,2N)Cu62													
	0	7.737E-29	0	1.730E-29	-	1.85	-	1.85	IRDF-90/G-V2	2925	16	Original	Original
	0	1.002	0	1.002	-	2.48	-	2.49	SNLRML-V1	2925		IRDF-90	GLUCS
	0	1.002	0	1.002	-	0.99	-	0.99	CROSSNEW.XSO				
	0	0.984	0	0.983	-	3.43	-	3.43	CROSSNEW.XSO			FU-Hetrick	FU-Hetrick
Cu63(N,G)Cu64													
	3.974E-24	1.071E-26	4.840E-24	1.129E-24	4.12	8.61	4.04	3.96	IRDF-90/G-V2	2925	102	ENDF/B-VI	ENDF/B-VI
	0.998	1.000	1.004	0.999	1.04	1.02	1.02	1.05	SNLRML-V1	2925		ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.000	1.000	1.00	1.00	0.97	1.00	CROSSNEW.XSO				
Cu63(N,A)Co60													
	0	5.019E-28	0	1.122E-28	-	2.37	-	2.37	IRDF-90/G-V2	2925	107	ENDF/B-VI	ENDF/B-VI
	0	0.999	0	0.999	-	1.00	-	1.00	SNLRML-V1	2925		ENDF/B-VI	GLUCS
	0	1.000	0	1.000	-	1.37	-	1.37	CROSSNEW.XSO				
Cu65(N,2N)Cu64													
	0	2.895E-28	0	6.471E-29	-	1.94	-	1.94	IRDF-90/G-V2	2931	16	ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	1.00	-	1.00	SNLRML-V1	2931		ENDF/B-VI	GLUCS
	0	1.000	0	1.000	-	1.75	-	1.75	CROSSNEW.XSO				
Zn64(N,P)Cu64													
	0	3.774E-26	4.086E-29	8.437E-27	-	4.88	30.00	4.88	IRDF-90/G-V2	3025	103	Original	Original
	0	1.000	0.997	1.000	-	1.00	0.14	1.00	SNLRML-V1	3025		IRDF-90	IRDF-90
	0	1.000	1.000	1.000	-	1.00	0.14	1.00	CROSSNEW.XSO				
Y89(N,2N)Y88													
	0	1.252E-28	0	2.800E-29	-	4.59	-	4.59	IRDF-90/G-V2	3925	16	ENDF/B-VI	ENDF/B-VI
	0	1.000	0	1.000	-	1.00	-	1.00	CROSSNEW.XSO				

Table A.1. (continued)

Reaction	Averaged cross sections (cm^2)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Zr90(N,2N)Zr89													
	0	7.539E-29	0	1.685E-29	-	1.66	-	1.66	IRDF-90/G-V2	4025	16	Original	Original
	0	0.999	0	0.999	-	1.00	-	1.00	SNLRML-V1	4025		IRDF-90	IRDF-90
	0	1.000	0	1.000	-	1.00	-	1.00	CROSSNEW.XSO				
Nb93(N,2N)Nb92													
	0	3.880E-28	0	8.672E-29	-	2.79	-	2.79	IRDF-90/G-V2	4125	16	Original	Original
	0	0.917	0	0.917	-	2.19	-	2.19	SNLRML-V1	4125		IRDF-90	IRDF-90
	0	0.951	0	0.951	-	0.39	-	0.39	RDF-94	4112	16	Priv comm	Priv comm
	0	0.917	0	0.917	-	2.19	-	2.19	CROSSNEW.XSO				
Nb93(N,N')Nb93													
	0	1.376E-25	3.624E-26	3.087E-26	-	3.01	7.03	3.00	IRDF-90/G-V2	4125	51	Original	Original
	0	1.000	1.000	1.000	-	1.00	1.00	1.00	SNLRML-V1	4125		IRDF-90	IRDF-90
	0	1.004	0.83	1.003	-	0.87	0.63	0.87	RDF-94	4112	51	Priv comm	Priv comm
	0	1.000	1.000	1.000	-	1.00	1.02	1.00	CROSSNEW.XSO				
Nb93(N,G)Nb94													
	1.029E-24	2.754E-26	9.888E-24	6.281E-25	10.00	7.55	9.84	9.87	IRDF-90/G-V2	4125	102	ENDF/B-VI	ENDF/B-VI
	0.998	1.000	0.998	0.998	1.00	1.00	1.00	1.00	SNLRML-V1	4125		ENDF/B-VI	ENDF/B-VI
Mo98(N,G)Mo99													
	1.154E-25	2.683E-26	6.521E-24	2.822E-25	-	-	-	-	SNLRML-V1	3426	102	JENDL-3	-
Rh103(N,N')Rh103													
	0	6.967E-25	4.170E-25	1.577E-25	-	3.10	6.76	3.28	IRDF-90/G-V2	4525	51	Original	Original
	0	1.000	1.000	1.000	-	1.00	0.93	0.99	SNLRML-V1	4525		IRDF-90	IRDF-90
	0	1.000	1.000	1.000	-	1.00	0.93	0.99	CROSSNEW.XSO				
Ag109(N,G)Ag110													
	4.181E-24	9.756E-27	6.539E-23	3.522E-24	5.10	7.47	6.93	6.07	IRDF-90/G-V2	4731	102	Priv comm	Priv comm
	1.036	0.591	1.184	1.142	-	-	-	-	SNLRML-V1	7479		DOSCROS84	-
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO			CND	
In115(N,2N)In114													
	0	7.538E-28	0	1.685E-28	-	3.99	-	3.99	IRDF-90/G-V2	4931	16	Priv comm	Priv comm
In115(N,N')In115													
	0	1.828E-25	2.047E-26	4.087E-26	-	2.16	2.53	2.16	IRDF-90/G-V2	4931	51	ENDF/B-VI	ENDF/B-VI
	0	1.000	1.000	1.000	-	1.00	1.00	1.00	SNLRML-V1	4931		IRDF-90	IRDF-90
	0	1.000	1.000	1.000	-	1.00	1.00	1.00	CROSSNEW.XSO				
In115(N,G)In116													
	1.908E-22	1.594E-25	3.267E-21	1.7300E-22	6.00	4.30	5.99	5.99	IRDF-90/G-V2	4931	102	Priv comm	Priv comm
	0.998	1.001	1.000	0.999	1.00	1.00	1.00	1.00	SNLRML-V1	4931		ENDF/B-VI	ENDF/B-VI
	0.996	1.000`	0.999	0.999	1.00	1.00	1.00	1.00	CROSSNEW.XSO			ENDF/B-V	ENDF/B-V
I127(N,2N)I126													
	0	1.045E-27	0	2.336E-28	-	2.20	-	2.20	IRDF-90/G-V2	5325	16	Priv comm	Priv comm
	0	1.101	0	1.101	-	7.84	-	7.84	SNLRML-V1	5325		ENDF/B-VI	ENDF/B-V
	0	1.101	0	1.101	-	7.84	-	7.84	CROSSNEW.XSO			ENDF/B-V	ENDF/B-V
Pr141(N,2N)Pr140													
	0	8.963E-28	0	2.003E-28	-	6.49	-	6.49	RDF-94	5912	16	Priv comm	Priv comm
Au197(N,2N)Au196													
	0	3.113E-27	0	6.958E-28	-	4.41	-	4.42	IRDF-90/G-V2	7925	16	Original	Original
	0	0.999	0	0.999	-	1.00	-	1.00	SNLRML-V1	7925		IRDF-90	IRDF-90
	0	1.000	0	1.000	-	1.00	-	0.99	CROSSNEW.XSO				
Au197(N,3N)Au195													
	0	2.405E-30	0	5.376E-31	-	-	-	-	SNLRML-V1	7925	17	ENDF/B-VI	-

Table A.1. (continued)

Reaction	Averaged cross sections (cm^2)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Au197(N,G)Au198													
	8.823E-23	7.738E-26	1.562E-21	8.137E-23	0.14	1.15	0.17	0.16	IRDF-90/G-V2	7925	102	ENDF/B-VI	ENDF/B-VI
	0.998	1.000	0.998	0.998	1.00	0.67	1.00	1.00	SNLRML-V1	7925		ENDF/B-VI	IRDF-90
	1.000	1.000	1.000	1.000	1.00	0.67	1.00	1.00	CROSSNEW.XSO				
Au197(N,P)Pt197													
	0	1.241E-30	0	2.774E-31	-	-	-	-	SNLRML-V1	7925	103	ENDF/B-VI	
Th232(N,2N)Th231													
	0	1.484E-26	0	3.316E-27	-	-	-	-	SNLRML-V1	9040	16	ENDF/B-VI	-
Th232(N,F)													
	0	7.373E-26	2.456E-28	1.648E-26	-	5.10	7.19	5.10	IRDF-90/G-V2	9040	18	ENDF/B-VI	ENDF/B-VI
	0	1.000	0.998	0.999	-	1.00	0.99	1.00	SNLRML-V1	9040		ENDF/B-VI	ENDF/B-VI
	0	1.000	1.000	1.000	-	1.00	0.99	1.00	CROSSNEW.XSO				
Th232(N,G)Th233													
	6.546E-24	9.315E-26	8.556E-23	4.855E-24	4.03	11.69	10.93	7.60	IRDF-90/G-V2	9040	102	ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.002	1.001	1.00	1.00	1.00	1.00	SNLRML-V1	9040		ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				
U235(N,F)													
	5.090E-22	1.217E-24	2.655E-22	1.297E-22	0.19	0.32	0.28	0.19	IRDF-90/G-V2	9228	18	ENDF/B-VI	ENDF/B-VI
	0.998	1.000	1.000	0.998	1.00	1.00	1.00	1.00	SNLRML-V1	9228		ENDF/B-VI	IRDF-90
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				
U238(N,F)													
	4.209E-29	2.997E-25	6.327E-27	6.708E-26	25.01	0.54	4.43	0.53	IRDF-90/G-V2	9237	18	ENDF/B-VI	ENDF/B-VI
	1.001	1.000	1.003	1.000	-	1.00	0.06	1.00	SNLRML-V1	9237		ENDF/B-VI	IRDF-90
	1.000	1.000	1.000	1.000	0.03	1.00	0.15	1.02	CROSSNEW.XSO				
U238(N,G)U239													
	2.405E-24	7.078E-26	2.774E-22	1.128E-23	0.35	1.06	0.37	0.36	IRDF-90/G-V2	9237	102	ENDF/B-VI	ENDF/B-VI
	1.000	1.000	1.000	1.000	1.00	1.16	1.00	1.00	CROSSNEW.XSO				
Np237(N,F)													
	1.572E-26	1.314E-24	1.110E-24	3.073E-25	364.70	9.35	11.30	10.53	SNLRML-V1	9346	18	ENDF/B-VI	ENDF/B-VD
	1.025	1.019	1.14	1.033	-	1.00	0.91	0.85	IRDF-90/G-V2	9337		ENDF/B-IV	ENDF/B-IV
	0.999	0.999	0.993	0.999	0	1.00	0.85	0.86	CROSSNEW.XSO				
Pu239(N,F)													
	6.992E-22	1.770E-24	2.739E-22	2.248E-22	0.25	0.41	0.26	0.25	IRDF-90/G-V2	9437	18	ENDF/B-VI	ENDF/B-VI
	0.999	1.008	0.993	0.999	2.64	4.98	16.19	2.76	SNLRML-V1	9237		ENDF/B-VI	JENDL-3D
	1.000	1.000	1.000	1.000	1.00	1.00	1.00	1.00	CROSSNEW.XSO				
Am241(N,F)													
	2.822E-24	1.394E-24	8.190E-24	1.571E-24	2.00	2.79	1.97	1.70	SNLRML-V1	9543	18	ENDF/B-VI	JENDL-3D
EnrU(N,F)SNL foil													
	4.724E-22	1.163E-24	2.469E-22	1.204E-22	0.19	0.32	0.28	0.19	SNLRML-V1	9228	851	ENDF/B-VI	IRDF-90
DepU(N,F)SNL foil													
	1.041E-24	3.015E-25	5.508E-25	3.323E-25	-	0.53	-	0.11	SNLRML-V1	9237	851	ENDF/B-VI	IRDF-90
Pu(N,F)SNL foil													
	6.181E-22	1.730E-24	2.431E-22	1.983E-22	0.66	2.04	4.22	0.70	SNLRML-V1	9437	851	ENDF/B-VI	JENDL-3D
U234(N,F)													
	4.072E-25	1.219E-24	1.613E-24	3.939E-25	-	-	-	-	SNLRML-V1	9225	18	ENDF/B-VI	-
U236(N,F)													
	4.195E-26	5.746E-25	4.566E-24	3.106E-25	-	-	-	-	SNLRML-V1	9231	18	ENDF/B-VI	-
Pu240(N,F)													
	5.819E-26	1.348E-24	3.679E-24	4.235E-25	-	-	-	-	SNLRML-V1	9440	18	ENDF/B-VI	-
Pu241(N,F)													
	9.438E-22	1.600E-24	5.690E-22	2.672E-22	-	-	-	-	SNLRML-V1	9443	18	ENDF/B-VI	-

Table A.1.(continued)

Reaction	Averaged cross sections (cm^2)[a]				Averaged standard deviations (%)[b]				Source library	MAT	MT	Origin of data	
	Maxwell	Watt fission	1/E	Combined spectrum	Maxwell	Watt fission	1/E	Combined spectrum				Cross section	Standard deviation
Pu238(N,F)	1.445E-23	1.969E-24	2.368E-23	4.527E-24	-	-	-	-	SNLRML-V1	9434	18	ENDF/B-VI	-
Pu242(N,F)	9.253E-28	1.118E-24	9.039E-25	2.604E-25	-	-	-	-	SNLRML-V1	9446	18	ENDF/B-VI	-

[a]Cross-section averages were calculated for the thermal Maxwell spectrum (at 293.58 K), Watt fission spectrum, and the "combined spectrum." The "combined spectrum" was Watt fission spectrum matched to 1/E at 0.5 MeV and to Maxwellian (T= 323.16 K) at 1.4 10^{-7} MeV. This "reference" spectrum is often used in LSL-M2 cross-section processing procedures. The resonance integral is calculated as the integral from 0.563 eV to 1.05 MeV of the cross section multiplied by the 1/E spectrum. For each reaction and for the first cross-section set in the library the average cross section is given in cm^2 (10^{-4} m^2). If more than one set of cross-sections is present the cross-section averages for the consecutive sets are divided by the value for the first set and the ratios are listed.

[b]Standard deviations are calculated for the cross sections averaged over Maxwell spectrum (at 293.58 K), 1/E spectrum between 0.563 eV and 1.05 MeV, Watt fission spectrum, and the combined spectrum. For each reaction and for the first cross-section set in the library the standard deviation is given in percents. If more than one set of cross-sections is present the standard deviations for the consecutive sets are divided by the standard deviation for the first set and the ratios are listed.

APPENDIX B

PROPERTIES OF CORRELATION AND COVARIANCE MATRICES OF THE DOSIMETRY CROSS SECTIONS

The properties of correlation and covariance matrices of the dosimetry cross sections are listed in Table B.1. The first and second columns from the left list the reaction and the source library, respectively. The dimension and trace of the correlation matrix are given in columns A and B, respectively. Column C lists the number of correlation coefficients greater than 1.00 (which indicates, of course, an inconsistent correlation matrix). Columns D and E give the numbers of negative and zero eigenvalues of the correlation matrix, respectively. Column F gives the number of standard deviations equal to zero. Columns G and H give the number of negative and zero eigenvalues of the covariance matrix, respectively. Columns I and J list the energies in eV, below which the standard deviations and cross sections of the reaction are equal to zero.

In column K a flag, indicating the status of the uncertainty information, is given. Flag "OK" indicates that the correlation and covariance matrices are positive definite and that nonzero standard deviations are given for all energy regions with nonzero cross section. Label "A" in the column K denotes the threshold reactions that have positive definite correlation matrices but have zero standard deviation specified for one energy group covering the region below threshold and nonnegative covariance matrices (with one eigenvalue equal to zero). These covariance matrices can be considered acceptable because they can easily be made positive definite by truncating the first energy group (below threshold), which is not needed. The reactions denoted with label "B" in column K have consistent correlation matrices but have an energy region with nonzero cross sections and zero standard deviation; therefore the covariance matrices have one eigenvalue equal to zero. Label "C" in the column K marks the reactions that have consistent correlation matrices but a region with nonzero cross section with zero standard deviation and covariance matrices with negative eigenvalues. Finally, the reactions which have both the correlation and covariance matrices with negative eigenvalues are labeled with "D" in column K. Uncertainty information of the eighteen reactions labeled by "B," "C," and "D" in column K is inconsistent and should be reevaluated.

Table B.1 Properties of the correlation and covariance matrices[a]

Reaction	Source library	Correlation matrix					Covariance matrix			I (eV)	J (eV)	K
		A	B	C	D	E	F	G	H			
Li6(N,T)He4												
	IRDF-90/G-V2	21	21.00	0	2	0	0	2	0	1.000E-05	1.000E-04	D
Li6(N,X)He4												
	SNLRML V1 (IRDF-90)	21	21.00	0	2	0	0	2	0	1.000E-05	1.000E-04	D
B10(N,A)Li7												
	IRDF-90/G-V2	18	18.00	0	2	0	0	2	0	1.000E-05	1.000E-04	D
B10(N,X)He4												
	SNLRML V1 (IRDF-90)	18	18.00	0	2	0	0	2	0	1.000E-05	1.000E-04	D
F19(N,2N)F18												
	IRDF-90/G-V2	25	25.00	0	0	0	1	0	1	1.090E+07	1.100E+07	A
	SNLRML V1 (IRDF-90)	21	21.00	0	1	1	1	2	1	1.100E+07	1.090E+07	
Na23(N,G)Na24												
	IRDF-90/G-V2	5	5.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	5	5.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
	SNLRML V1 JENDL-3D	5	5.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Mg24(N,P)Na24												
	IRDF-90/G-V2	46	46.00	0	0	0	1	0	1	4.600E+06	5.000E+06	A
	SNLRML V1 (IRDF-90)	45	45.00	0	0	0	1	0	1	4.600E+06	3.200E+06	
Al27(N,P)Mg27												
	IRDF-90/G-V2	61	61.00	0	0	0	3	5	0	2.000E+06	1.800E+06	C
	SNLRML V1 (GLUCS)	60	60.00	0	0	0	1	10	0	2.000E+06	2.100E+06	
Al27(N,A)Na24												
	IRDF-90/G-V2	50	50.00	0	0	0	1	0	1	5.000E+06	5.000E+06	A
	SNLRML V1 (GLUCS)	48	48.00	0	0	0	1	0	1	4.500E+06	4.900E+06	
P31(N,P)Si31												
	IRDF-90/G-V2	34	34.00	0	0	0	1	0	1	1.200E+06	1.300E+06	A
	SNLRML V1 (IRDF-90)	34	34.00	0	4	0	1	4	1	1.500E+06	1.200E+06	
S32(N,P)P32												
	IRDF-90/G-V2	98	98.00	0	0	0	1	29	0	9.570E+05	9.200E+05	C
	SNLRML V1 (GLUCS)	98	98.00	0	0	0	2	26	0	1.000E+06	9.200E+05	
Sc45(N,G)Sc46												
	IRDF-90/G-V2	19	19.00	0	1	4	0	2	3	1.000E-05	1.000E-04	D
	SNLRML V1 (ENDF/B-VI)	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
	SNLRML V1 (JENDL-3D)	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Ti46(N,2N)Ti45												
	RDF-94	18	18.00	0	2	0	1	2	1	1.360E+07	1.350E+07	D
Ti46(N,P)Sc46												
	IRDF-90/G-V2	24	24.00	0	0	0	1	0	1	2.400E+06	1.600E+06	B
	SNLRML V1 (GLUCS)	24	24.00	0	0	0	1	0	1	2.400E+06	1.700E+06	
	RDF-94	33	33.00	0	10	0	1	10	1	2.500E+06	1.700E+06	
Ti47(N,NP)Sc46												
	IRDF-90/G-V2	2	2.00	0	0	0	1	0	1	1.068E+07	1.070E+07	A
	SNLRML V1 (ENDF/B-VI)	2	2.00	0	0	0	1	0	1	1.068E+07	1.060E+07	
Ti47(N,P)Sc47												
	IRDF-90/G-V2	43	43.00	0	0	0	1	0	1	8.000E+05	6.900E+05	B
	SNLRML V1 (GLUCS)	43	43.00	0	0	0	1	0	1	8.000E+05	6.900E+05	
Ti47(N,X)Sc46												
	RDF-94	15	15.00	0	2	0	1	2	1	1.150E+07	1.070E+07	D
Ti48(N,NP)Sc47												
	IRDF-90/G-V2	2	2.00	0	0	0	1	0	1	1.169E+07	1.170E+07	A
	SNLRML V1 (ENDF/B-VI)	2	2.00	0	0	0	1	0	1	1.169E+07	1.160E+07	
Ti48(N,P)Sc48												
	IRDF-90/G-V2	26	26.00	0	0	0	1	0	1	4.000E+06	3.200E+06	B
	SNLRML V1 (GLUCS)	26	26.00	0	0	0	1	0	1	4.000E+06	3.300E+06	
V(N,A)Sc												
	IRDF-90/G-V2	13	13.00	0	0	0	1	0	1	2.095E+06	2.100E+06	A

Table B.1 (continued)[a]

Reaction	Source library	Correlation matrix					Covariance matrix			I (eV)	J (eV)	K
		A	B	C	D	E	F	G	H			
Cr52(N,2N)Cr51												
	IRDF-90/G-V2	15	15.00	0	0	0	1	0	1	1.220E+07	1.230E+07	A
Mn55(N,2N)Mn54												
	IRDF-90/G-V2	3	3.00	0	0	0	1	0	1	1.040E+07	1.040E+07	A
	SNLRML V1 (ENDF/B-VI)	7	7.00	0	0	0	1	0	1	1.042E+07	1.040E+07	
Mn55(N,G)Mn56												
	IRDF-90/G-V2	13	13.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B- VI)	13	13.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Fe54(N,2N)Fe53												
	RDF-94	17	17.00	0	0	0	1	0	1	1.370E+07	1.370E+07	A
Fe54(N,P)Mn54												
	IRDF-90/G-V2	45	45.00	0	0	0	1	0	1	5.000E+05	6.900E+05	A
	SNLRML V1 (GLUCS)	45	45.00	0	0	0	1	0	1	5.000E+05	6.900E+05	
Fe54(N,A)Cr51												
	SNLRML V1 (ENDF/B-V1)	9	9.00	0	0	0	1	0	1	3.000E+06	3.000E+06	A
Fe56(N,P)Mn56												
	IRDF-90/G-V2	47	47.00	0	1	0	2	1	2	3.000E+06	3.000E+06	D
	SNLRML V1 (ENDF/B-VI)	47	47.00	0	1	0	2	1	2	3.000E+06	2.900E+06	
Fe58(N,G)Fe59												
	IRDF-90/G-V2	15	15.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	15	15.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
	SNLRML V1 (JENDL-3D)	8	8.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Co59(N,2N)Co58												
	IRDF-90/G-V2	20	20.00	0	0	0	1	0	1	1.060E+07	1.060E+07	A
	SNLRML V1 (IRDF-90)	17	17.00	0	0	0	1	0	1	1.080E+07	1.060E+07	
Co59(N,G)Co60												
	IRDF-90/G-V2	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-V)	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Co59(N,P)Fe59												
	SNLRML V1 (ENDF/B-VI)	28	28.00	0	0	0	1	0	1	7.960E+05	1.500E+06	A
Co59(N,A)Mn56												
	IRDF-90/G-V2	34	34.00	0	1	0	0	1	0	1.000E-05	3.900E+06	D
	SNLRML V1 (ENDF/B-VI)	34	34.00	0	1	0	0	1	0	1.000E-05	3.900E+06	
Ni58(N,2N)Ni57												
	IRDF-90/G-V2	17	17.00	0	0	0	1	0	1	1.240E+07	1.240E+07	A
	SNLRML V1 (ENDF/B-VI)	17	17.00	0	0	0	2	0	2	1.300E+07	1.240E+07	
Ni58(N,P)Co58												
	IRDF-90/G-V2	41	41.00	0	0	0	1	0	1	5.000E+05	5.000E+05	A
	SNLRML V1 (GLUCS)	40	40.00	0	1	0	1	1	1	5.000E+05	5.000E+05	
Ni60(N,P)Co60												
	IRDF-90/G-V2	7	7.00	0	0	0	1	0	1	2.000E+06	2.000E+06	A
	SNLRML V1 (ENDF/B-VI)	7	7.00	0	0	0	1	0	1	2.076E+06	2.100E+06	
Cu63(N,2N)Cu62												
	IRDF-90/G-V2	18	18.00	0	0	0	1	0	1	1.090E+07	1.100E+07	A
	SNLRML V1 (GLUCS)	19	19.00	0	0	0	2	0	2	1.130E+07	1.060E+07	
Cu63(N,G)Cu64												
	IRDF-90/G-V2	23	23.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Cu63(N,A)Co60												
	IRDF-90/G-V2	47	47.00	0	0	0	1	0	1	2.000E+06	1.700E+06	B
	SNLRML V1 (GLUCS)	47	47.00	0	0	0	1	0	1	2.000E+06	1.700E+06	
Cu65(N,2N)Cu64												
	IRDF-90/G-V2	23	23.00	0	0	0	1	0	1	1.010E+07	1.000E+07	B
	SNLRML V1 (GLUCS)	23	23.00	0	0	0	1	0	1	1.010E+07	1.000E+07	
Zn64(N,P)Cu64												
	IRDF-90/G-V2	33	33.00	0	0	0	1	0	1	5.000E+05	5.000E+05	A
	SNLRML V1 (IRDF-90)	33	33.00	0	0	0	1	0	1	9.900E+05	5.000E+05	

Table B.1 (continued)[a]

Reaction	Source library	Correlation matrix					Covariance matrix			I (eV)	J (eV)	K
		A	B	C	D	E	F	G	H			
Y89(N,2N)Y88												
	IRDF-90/G-V2	11	11.00	0	0	0	1	0	1	1.160E+07	1.160E+07	A
Zr90(N,2N)Zr89												
	IRDF-90/G-V2	21	21.00	0	0	0	1	0	1	1.210E+07	1.210E+07	A
	SNLRML V1 (IRDF-90)	21	21.00	0	0	0	1	0	1	1.220E+07	1.210E+07	
Nb93(N,2N)Nb92												
	IRDF-90/G-V2	19	19.00	0	0	0	1	0	1	8.900E+06	9.000E+06	A
	SNLRML V1 (IRDF-90)	17	17.00	0	0	0	1	0	1	9.000E+06	8.800E+06	
	RDF-94	19	19.00	0	0	0	1	0	1	9.000E+06	9.000E+06	
Nb93(N,N')Nb93												
	IRDF-90/G-V2	33	33.00	0	0	0	1	0	1	3.000E+04	3.000E+04	A
	SNLRML V1 (IRDF-90)	32	32.00	12	8	0	1	8	1	1.200E+05	3.000E+04	
	RDF-94	33	33.00	0	4	0	1	4	1	1.200E+05	3.200E+04	
Nb93(N,G)Nb94												
	IRDF-90/G-V2	16	16.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	16	16.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Rh103(N,N')Rh103												
	IRDF-90/G-V2	33	33.00	0	0	0	1	0	1	4.000E+04	4.000E+04	A
	SNLRML V1 (IRDF-90)	33	33.00	0	5	0	1	5	1	1.000E+05	4.000E+04	
Ag109(N,G)Ag110												
	IRDF-90/G-V2	10	10.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
In115(N,2N)In114												
	IRDF-90/G-V2	14	14.00	0	0	0	0	0	0	1.000E-05	9.300E+06	OK
In115(N,N')In115												
	IRDF-90/G-V2	30	30.00	0	0	0	0	0	0	1.000E-05	3.200E+05	OK
	SNLRML V1 (IRDF-90)	30	30.00	0	0	0	0	0	0	1.000E-05	3.400E+05	
In115(N,G)In116												
	IRDF-90/G-V2	6	6.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	6	6.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
I127(N,2N)I126												
	IRDF-90/G-V2	12	12.00	0	0	0	0	0	0	1.000E-05	9.000E+06	OK
	SNLRML V1 (ENDF/B-V)	6	6.00	0	0	0	1	0	1	9.284E+06	9.200E+06	
Pr141(N,2N)Pr140												
	RDF-94	20	20.00	0	7	0	1	7	1	9.500E+06	9.500E+06	D
Au197(N,2N)Au196												
	IRDF-90/G-V2	27	27.00	0	0	0	1	0	1	8.100E+06	8.100E+06	A
	SNLRML V1 (IRDF-90)	23	23.00	0	0	0	1	0	1	8.250E+06	8.100E+06	
Au197(N,G)Au198												
	IRDF-90/G-V2	16	16.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (IRDF-90)	17	17.00	0	0	0	1	0	1	4.000E+06	1.000E-04	
Th232(N,F)												
	IRDF-90/G-V2	11	11.00	0	0	0	1	0	1	5.000E+05	5.000E+05	A
	SNLRML V1 (ENDF/B-VI)	11	11.00	0	0	0	1	0	1	6.000E+05	5.000E+05	
Th232(N,G)Th233												
	IRDF-90/G-V2	4	4.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (ENDF/B-VI)	4	4.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
U235(N,F)												
	IRDF-90/G-V2	23	23.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (IRDF-90)	23	23.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
U238(N,F)												
	IRDF-90/G-V2	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (IRDF-90)	10	10.00	0	0	0	1	0	1	9.500E+05	1.000E-04	
U238(N,G)U239												
	IRDF-90/G-V2	14	14.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK

Table B.1 (continued)[a]

Reaction	Source library	Correlation matrix					Covariance matrix			I (eV)	J (eV)	K
		A	B	C	D	E	F	G	H			
Np237(N,F)												
	SNLRML V1 (ENDF/B-VD)	4	4.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	IRDF-90/G-V2	4	4.00	0	0	0	1	0	1	1.000E+01	1.000E-04	
Pu239(N,F)												
	IRDF-90/G-V2	21	21.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
	SNLRML V1 (JENDL-3D)	15	15.00	0	0	0	0	0	0	1.000E-05	1.000E-04	
Am241(N,F)												
	SNLRML V1 (JENDL-3D)	16	16.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
EnrU(N,F) SNL foil												
	SNLRML V1 (IRDF-90)	23	23.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK
DepU(N,F) SNL foil												
	SNLRML V1 (IRDF-90)	10	10.00	0	0	0	1	0	1	9.500E+05	1.000E-04	B
Pu(N,F) SNL foil												
	SNLRML V1 (JENDL-3D)	15	15.00	0	0	0	0	0	0	1.000E-05	1.000E-04	OK

[a]**Column key**

A = dimension of the correlation matrix
B = trace of the correlation matrix
C = number of correlation coefficients greater than 1.00 (indicates an inconsistent correlation matrix)
D = number of negative eigenvalues of the correlation matrix
E = number of zero eigenvalues of the correlation matrix
F = number of standard deviations equal to zero
G = number of negative eigenvalues of the covariance matrix
H = number of zero eigenvalues of the covariance matrix
I = energy below which the standard deviations of the reaction cross sections equal zero
J = energy below which the cross sections of the reaction equal zero
K = flags indicate the status of the uncertainty information:

OK: the correlation and covariance matrices are positive definite and nonzero standard deviations are given for all energy regions with nonzero cross section.

A: the threshold reactions that have positive definite correlation matrices but have zero standard deviation specified for one energy group covering the region below threshold and nonnegative covariance matrices (with one eigenvalue equal to zero).

B: reactions have consistent correlation matrices but have an energy region with nonzero cross section and zero standard deviation; therefore, the covariance matrices have one eigenvalue equal to zero.

C: reactions have consistent correlation matrices but a region with nonzero cross section with zero standard deviation and covariance matrices with negative eigenvalues.

D: reactions have both the correlation and covariance matrices with negative eigenvalues.

APPENDIX C

PLOTS OF CROSS SECTIONS VS ENERGY

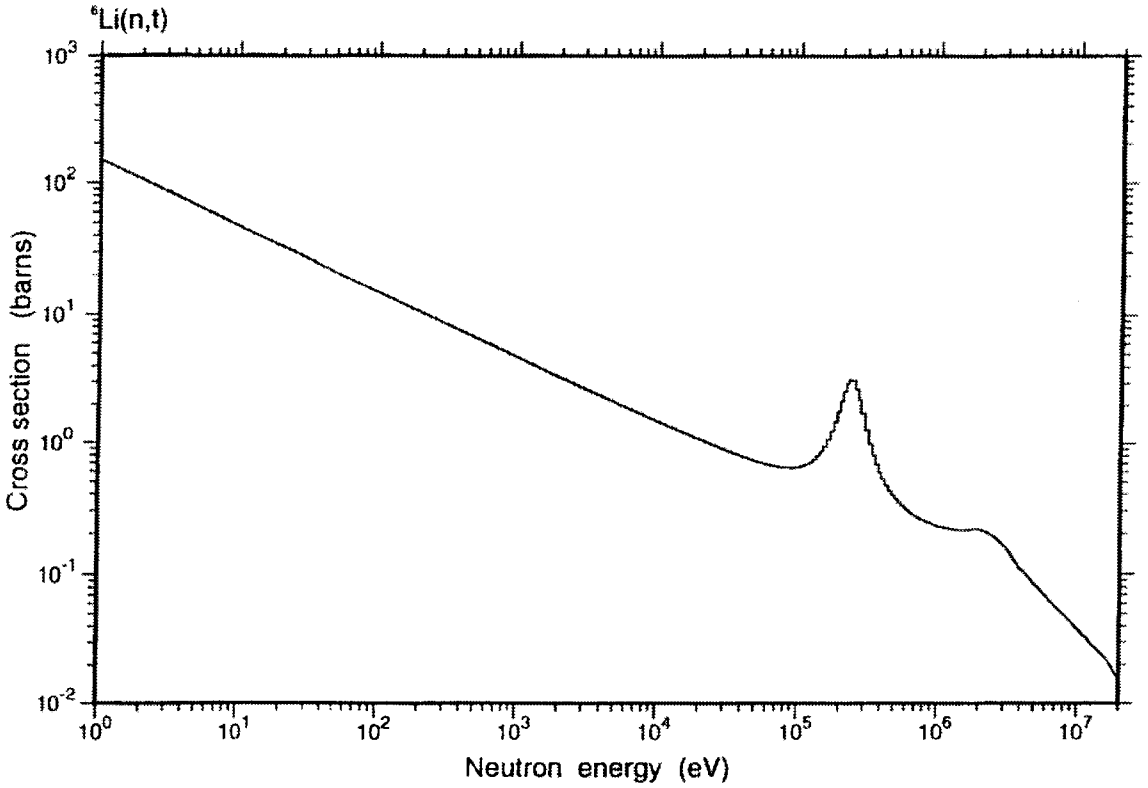

Fig. C.1. ^{6}Li(n,t)^{4}He (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

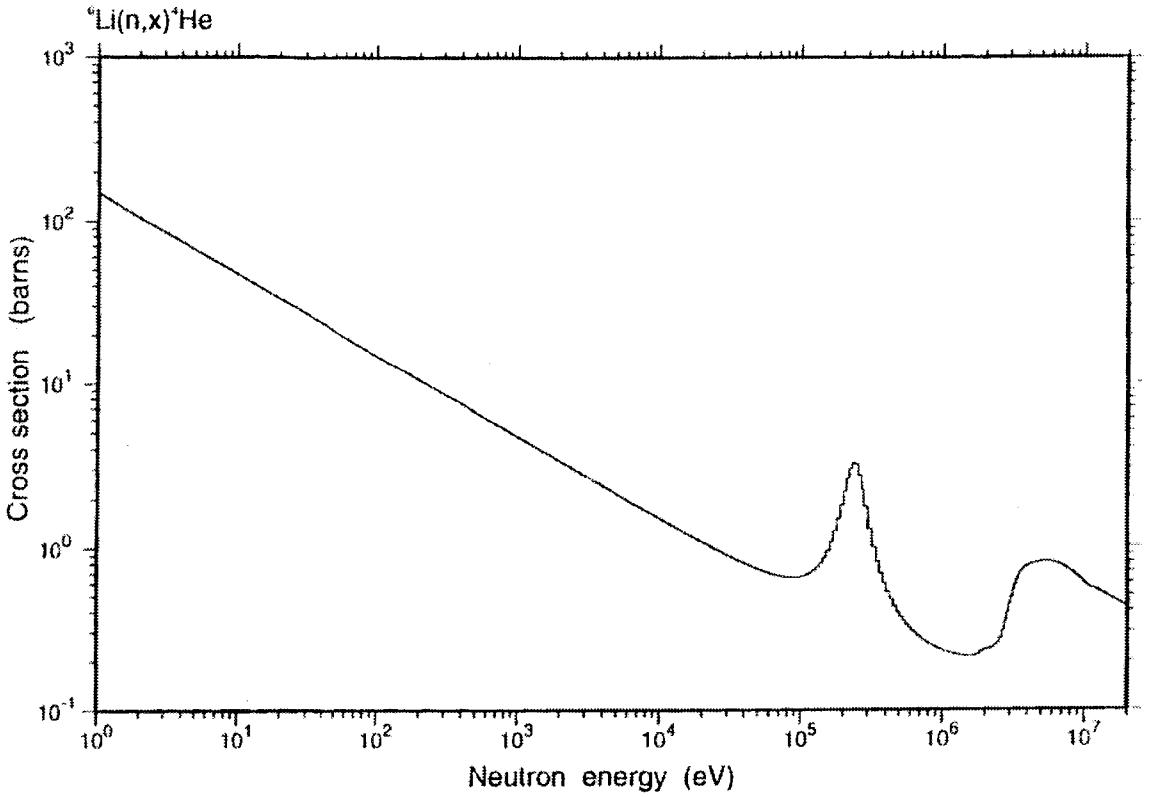

Fig. C.2. ^{6}Li(n,x)^{4}He (SNLRML-V1, ENDF/B-VI) cross section vs energy.

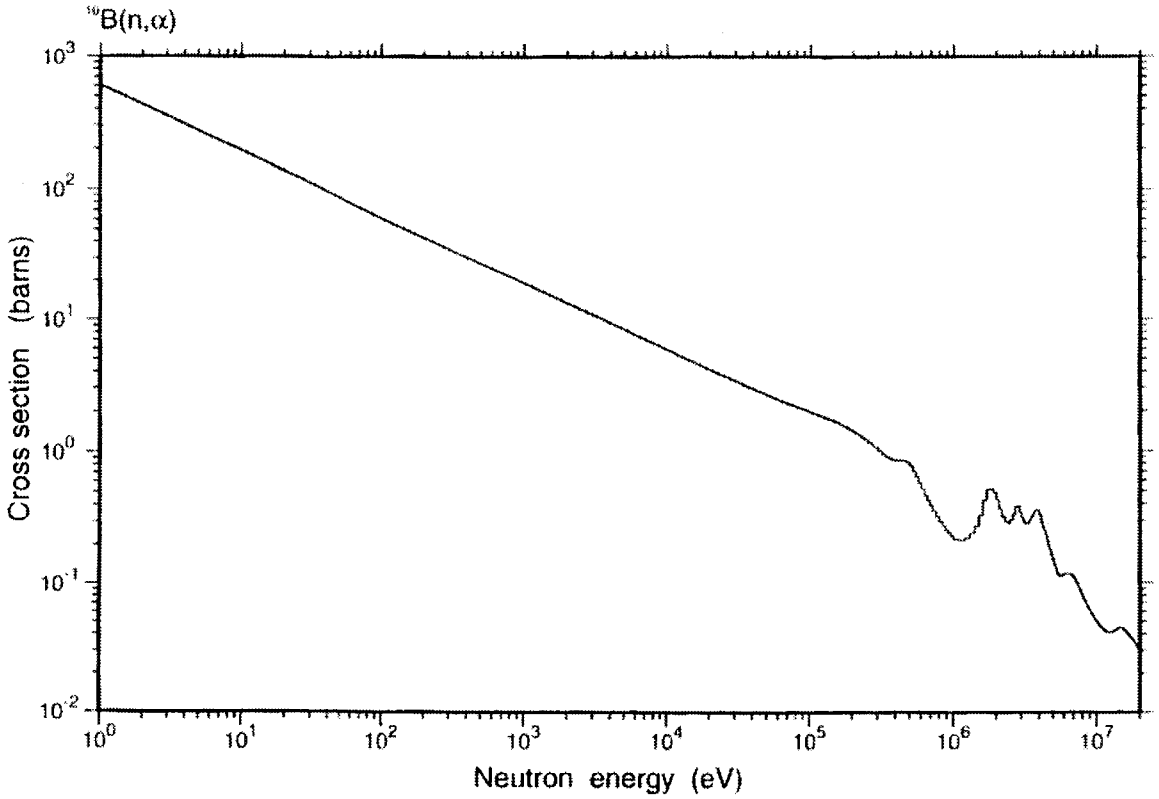

Fig. C.3 . ^{10}B(n,α)^{7}Li (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

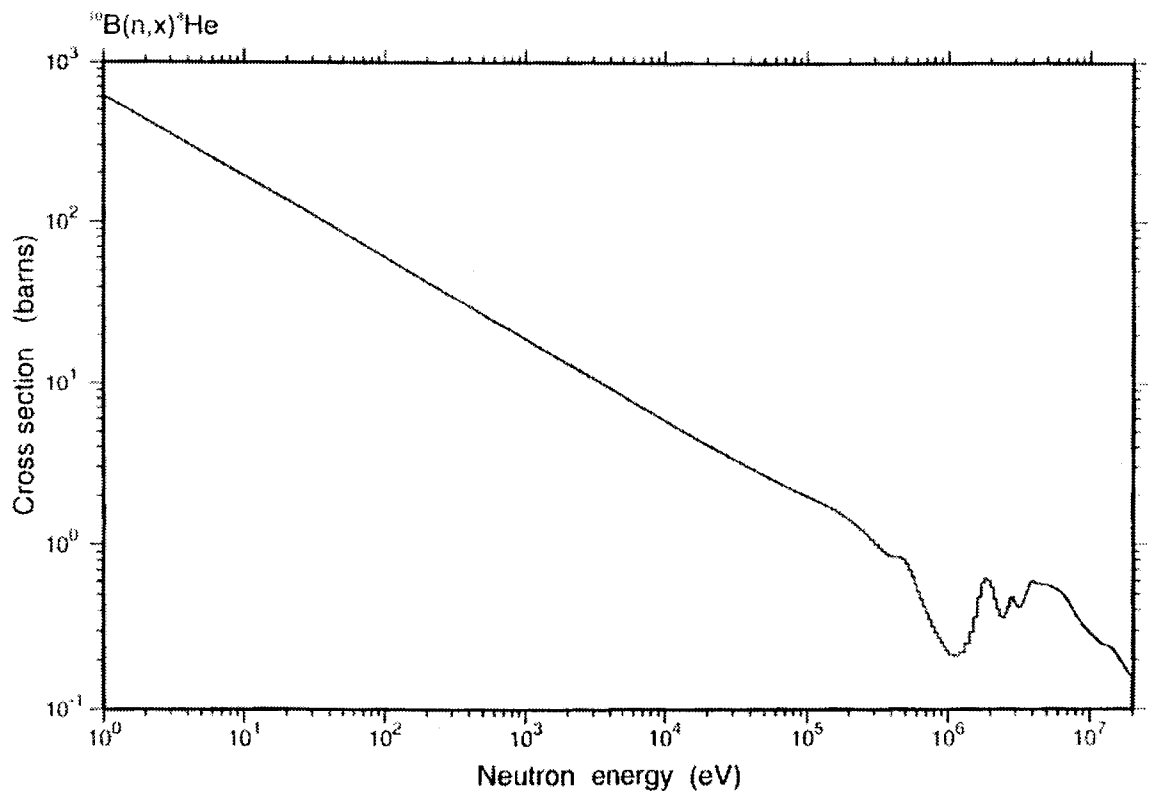

Fig. C.4. ^{10}B(n,x)^{4}He (SNLRML-V1, ENDF/B-VI) cross section vs energy.

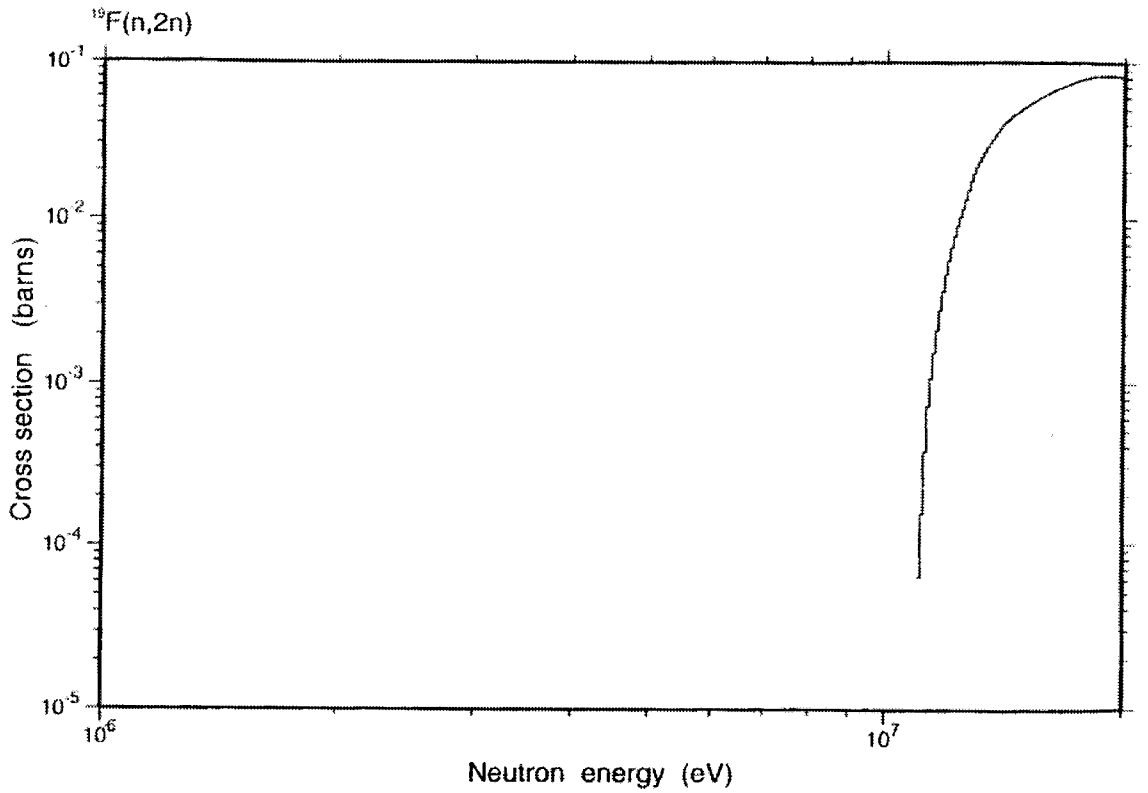

Fig. C.5. ^{19}F(n,2n)^{18}F (IRDF-90/G-V2, original evaluation) cross section vs energy.

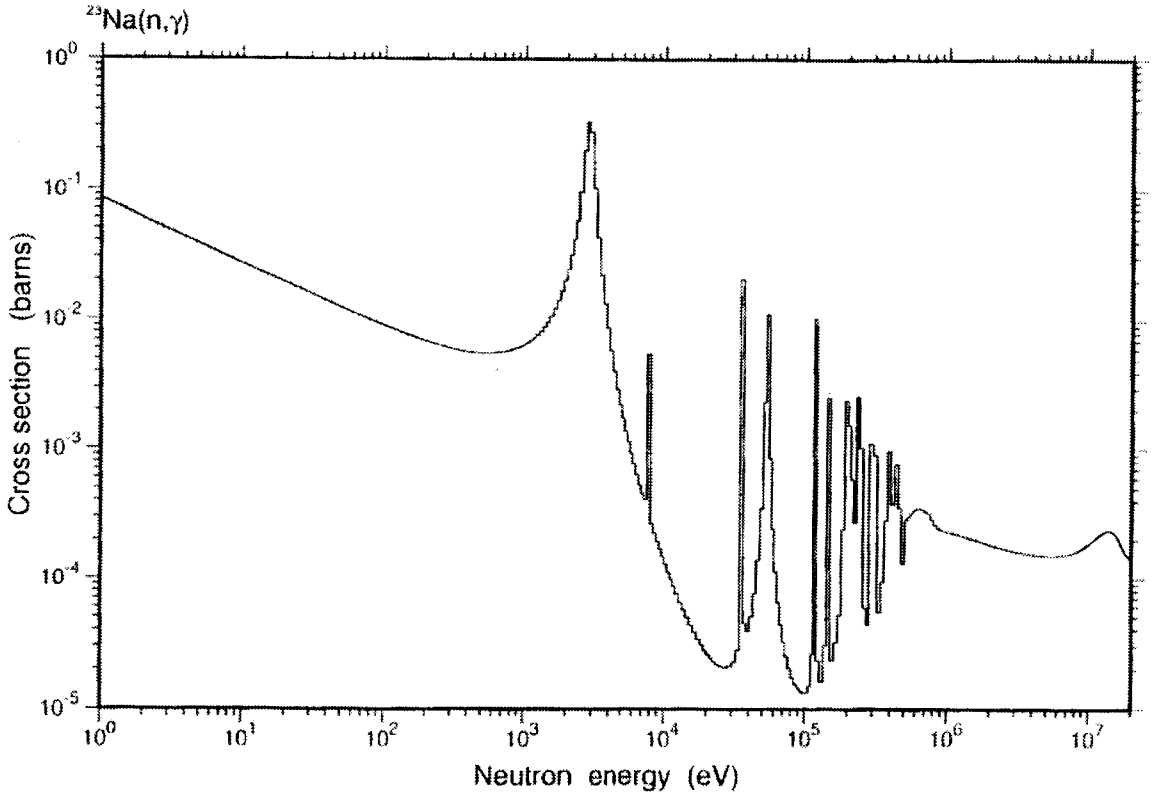

Fig. C.6. ^{23}Na(n,γ)^{24}Na (IRDF-90/G-V2, private communication) cross section vs energy.

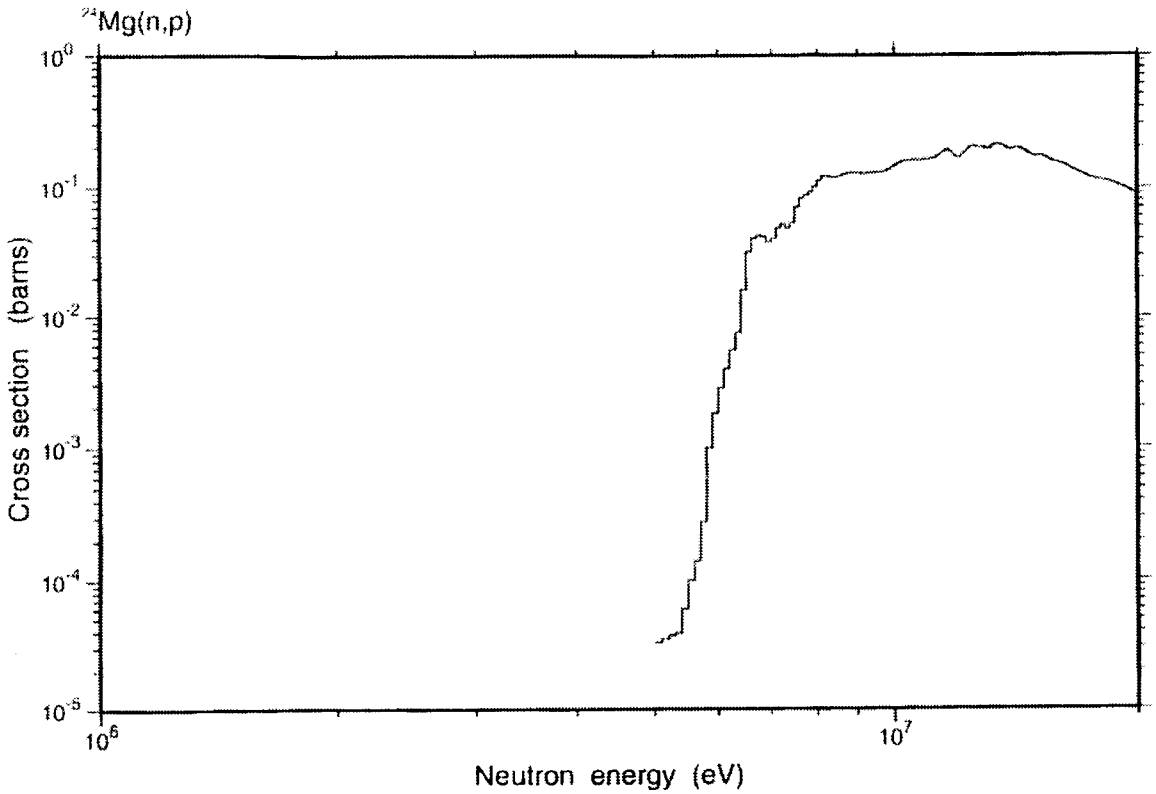

Fig. C.7. ^{24}Mg(n,p)^{24}Na (IRDF-90/G-V2, original evaluation) cross section vs energy.

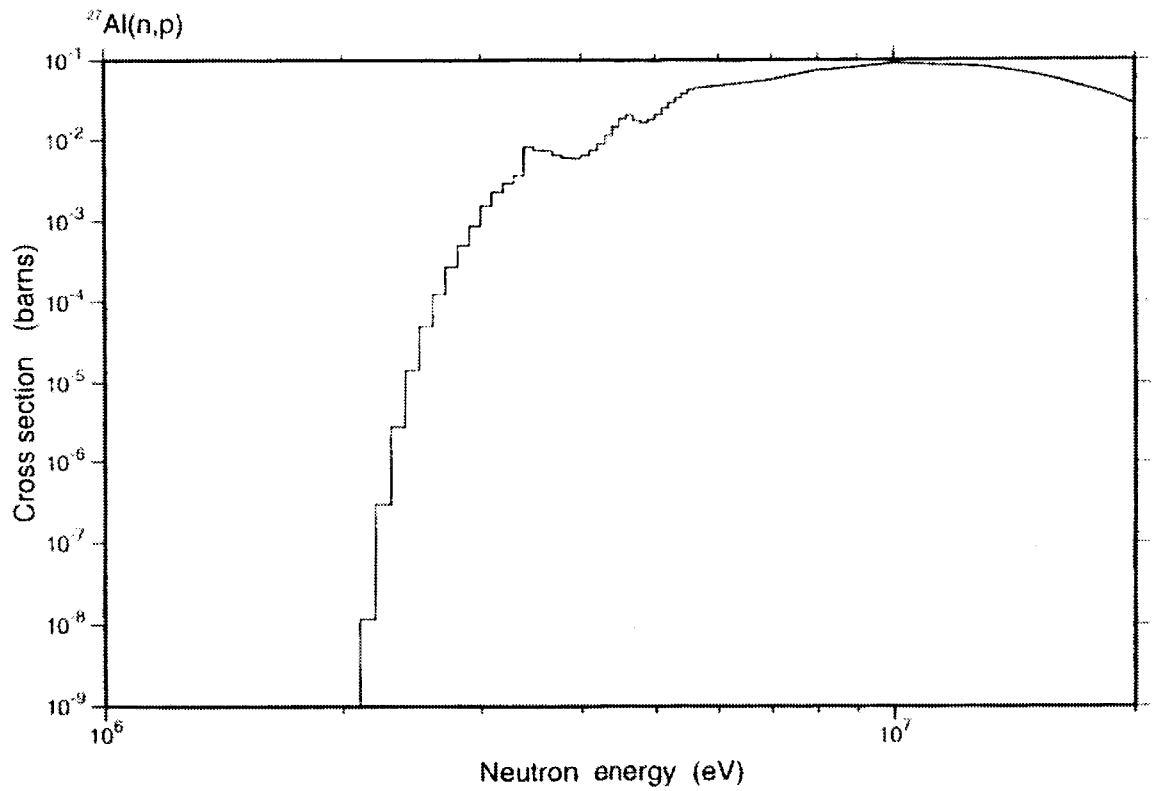

Fig. C.8. ^{27}Al(n,p)^{27}Mg (IRDF-90/G-V2, private communication) cross section vs energy.

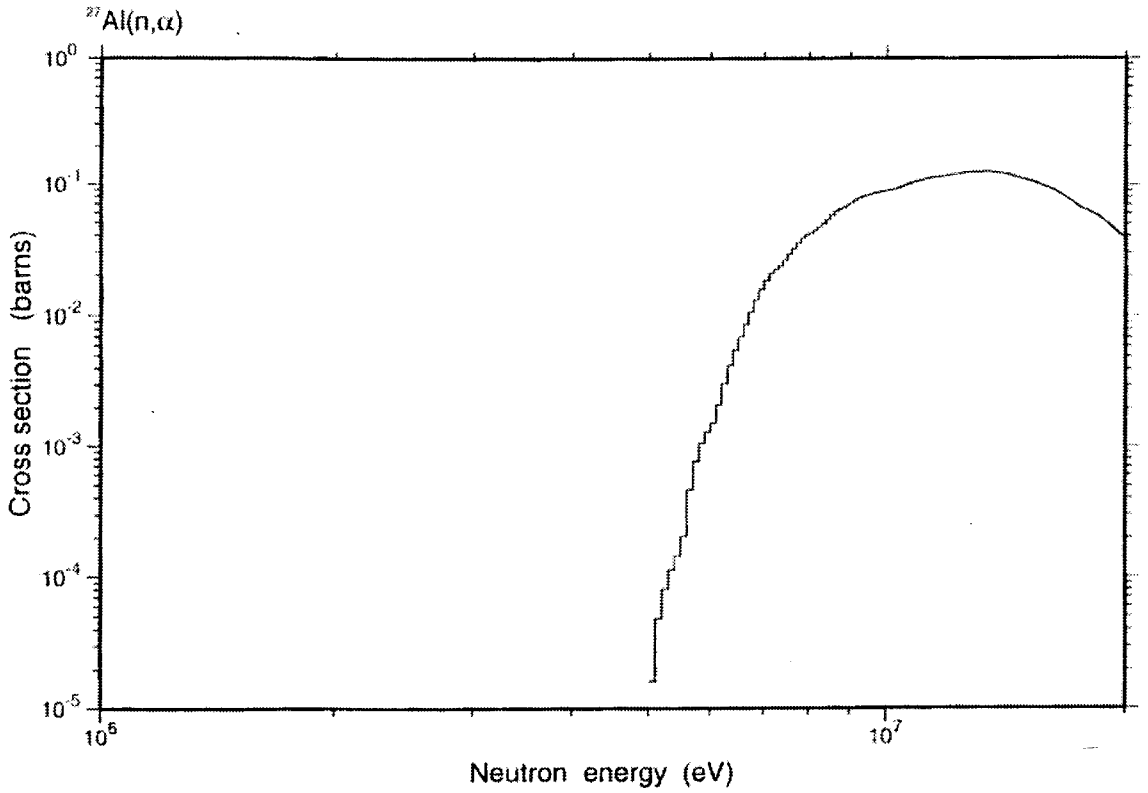

Fig. C.9. $^{27}Al(n,\alpha)^{24}Na$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

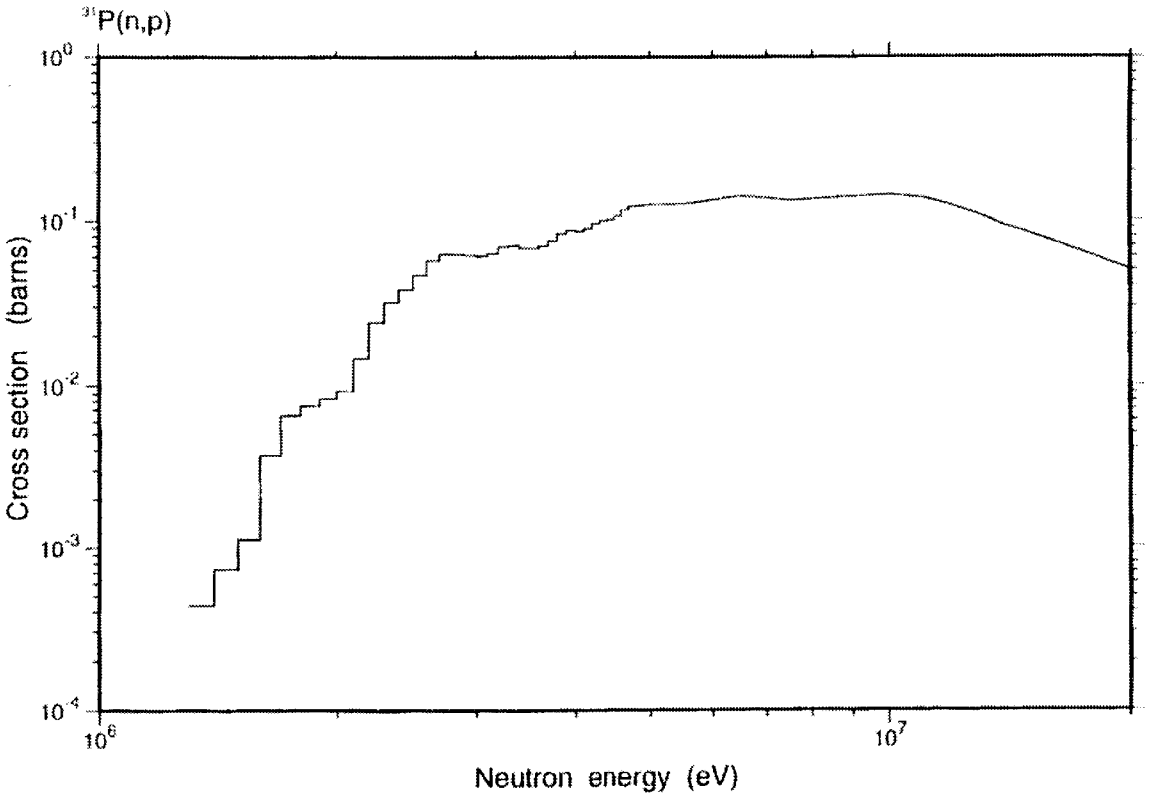

Fig. C.10. $^{31}P(n,p)^{31}Si$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

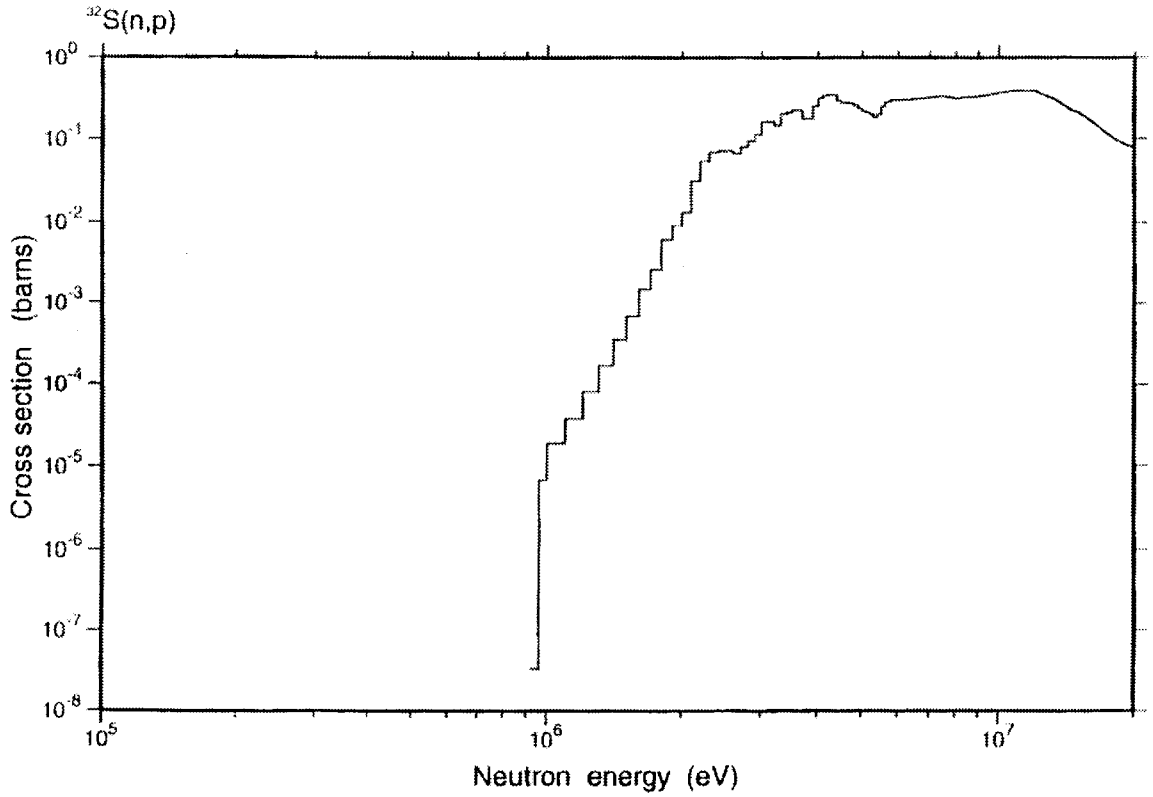

Fig. C.11. $^{32}S(n,p)^{32}P$ (IRDF-90/G-V2, private communication) cross section vs energy.

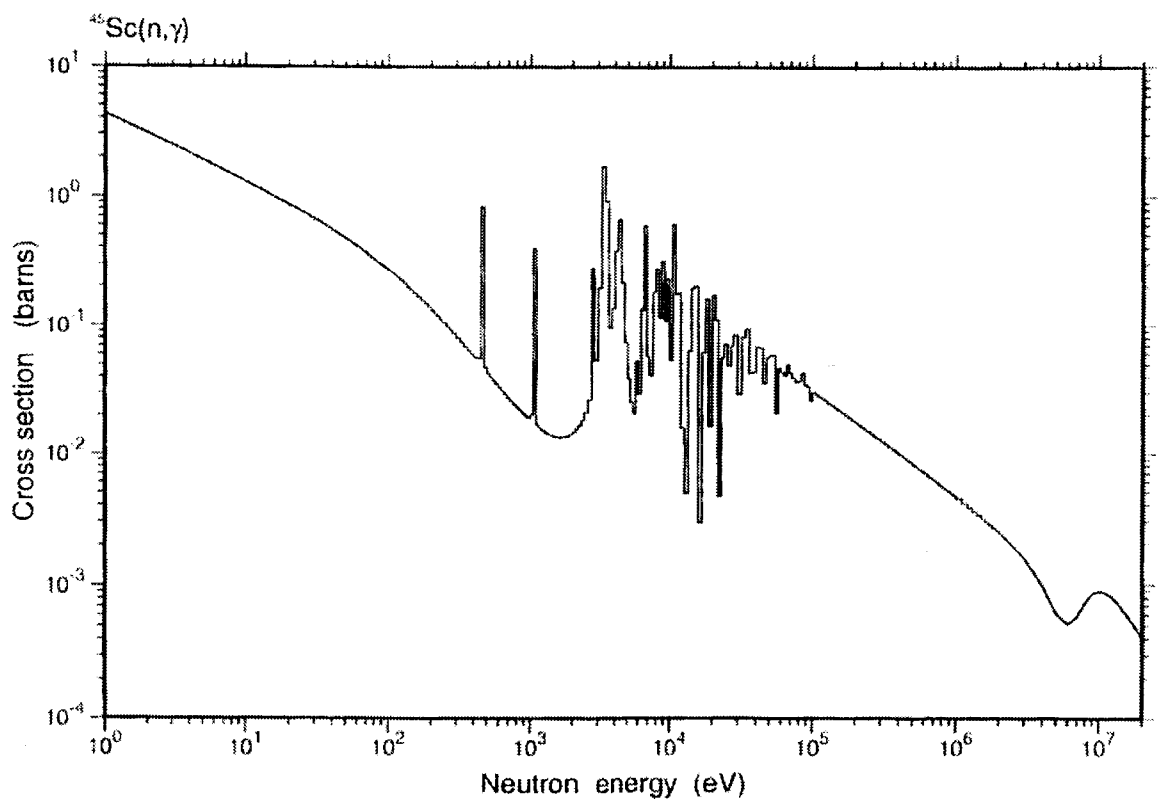

Fig. C.12. $^{45}Sc(n,\gamma)^{46}Sc$ (IRDF-90/G-V2, private communication) cross section vs energy.

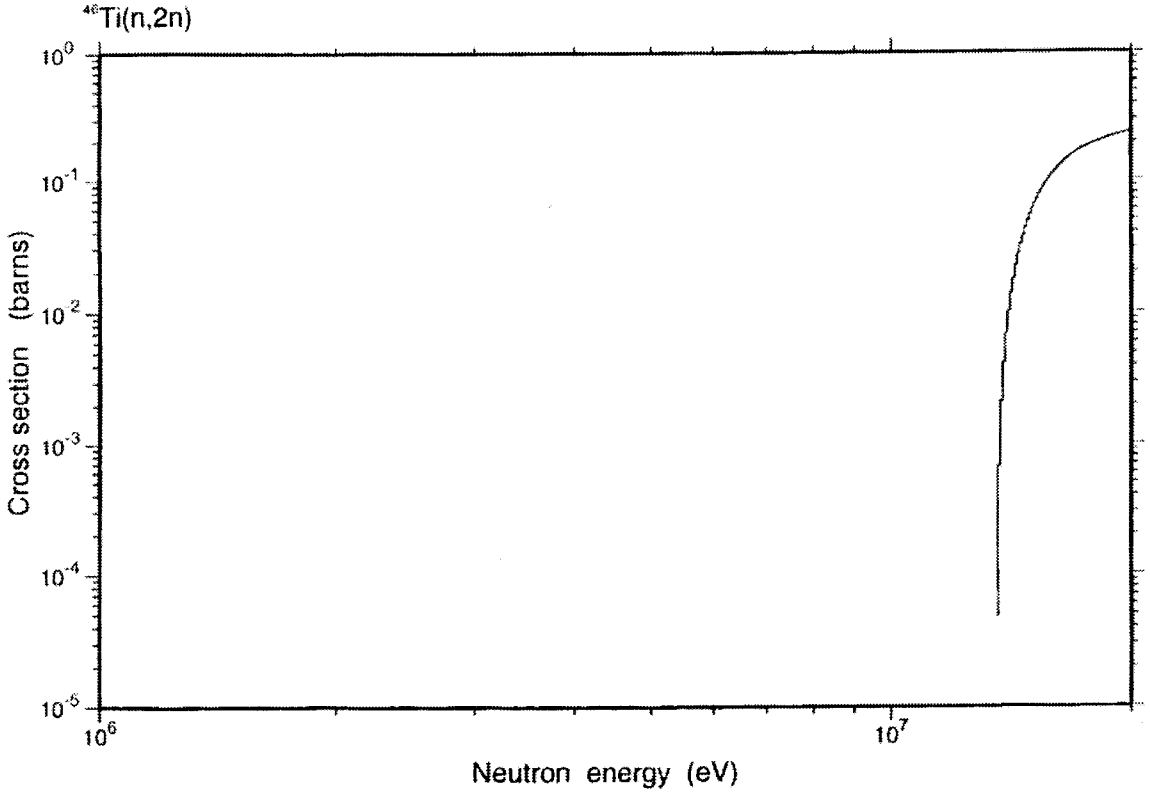

Fig. C.13. ^{46}Ti(n,2n)^{45}Ti (RDF-94, private communication) cross section vs energy.

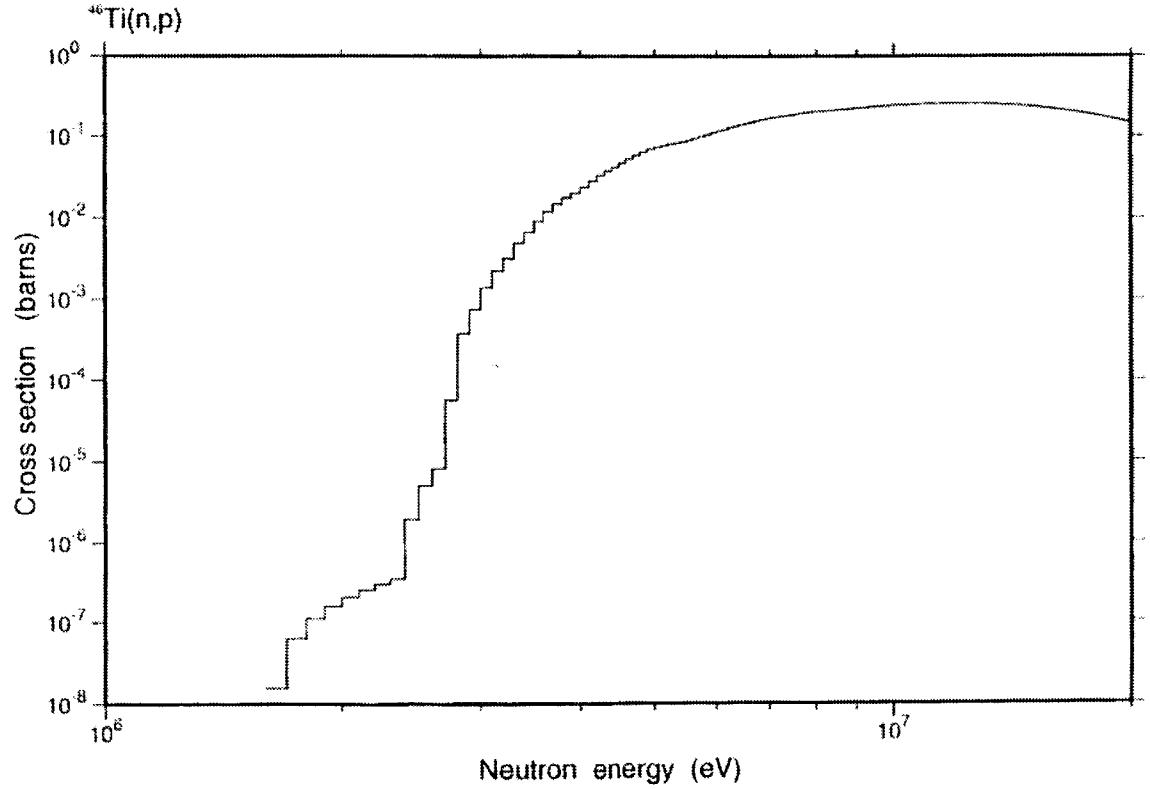

Fig. C.14. ^{46}Ti(n,p)^{46}Sc (IRDF-90/G-V2, private communication) cross section vs energy.

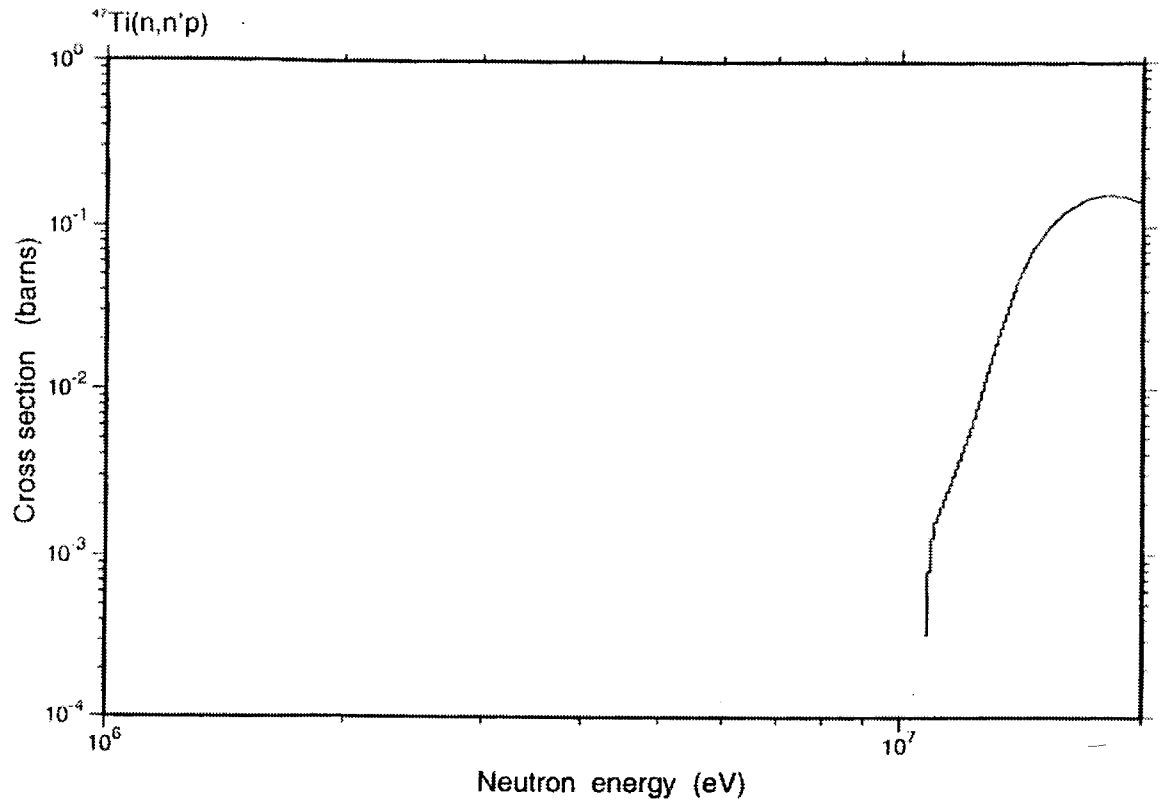

Fig. C.15. $^{47}Ti(n,n'p)^{46}Sc$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

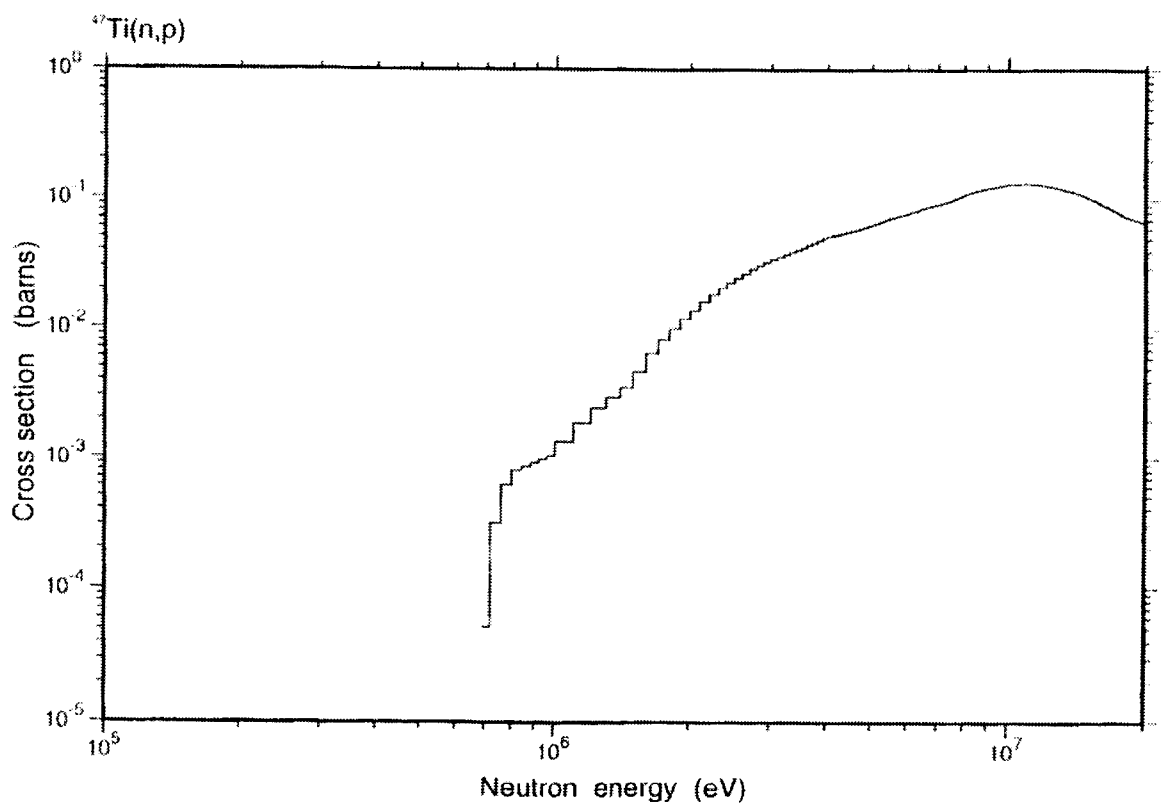

Fig. C.16. $^{47}Ti(n,p)^{47}Sc$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

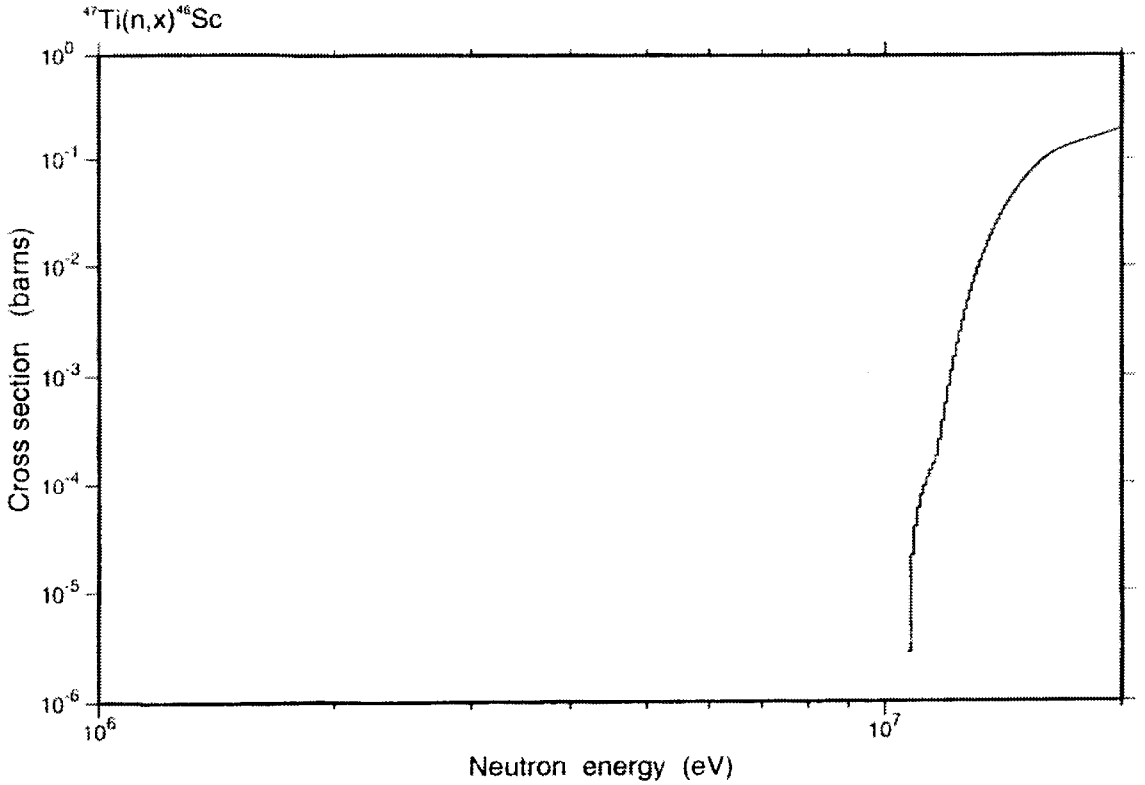

Fig. C.17. ^{47}Ti(n,x)^{46}Sc (RDF-94, private communication) cross section vs energy.

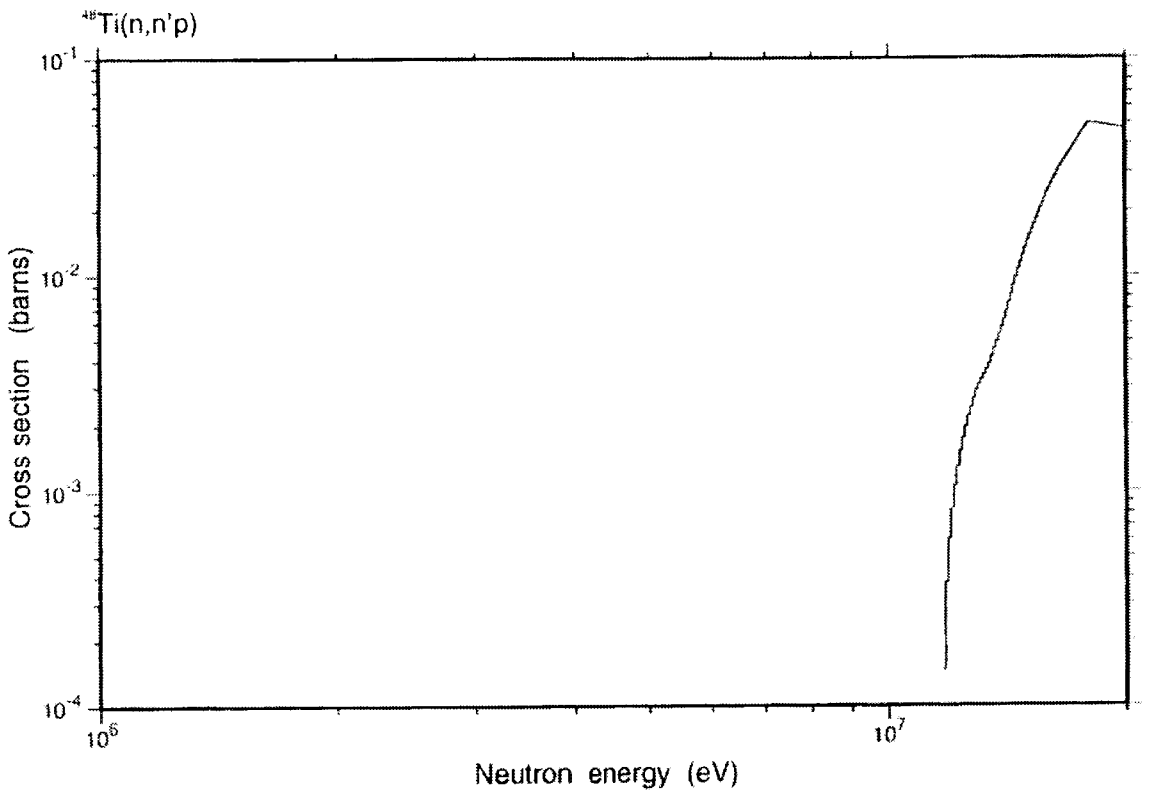

Fig. C.18. ^{48}Ti(n,$n'p$)^{47}Sc (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

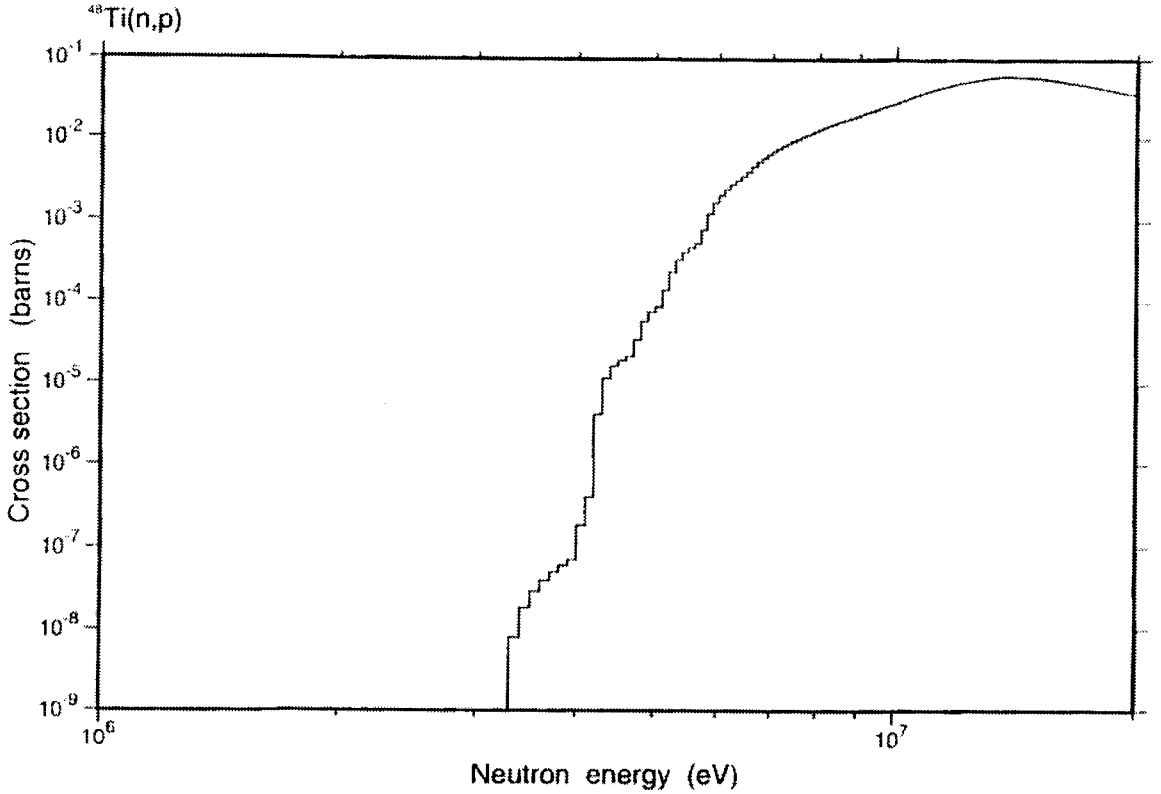

Fig. C.19. ^{48}Ti(n,p)^{48}Sc (IRDF-90/G-V2, private communication) cross section vs energy.

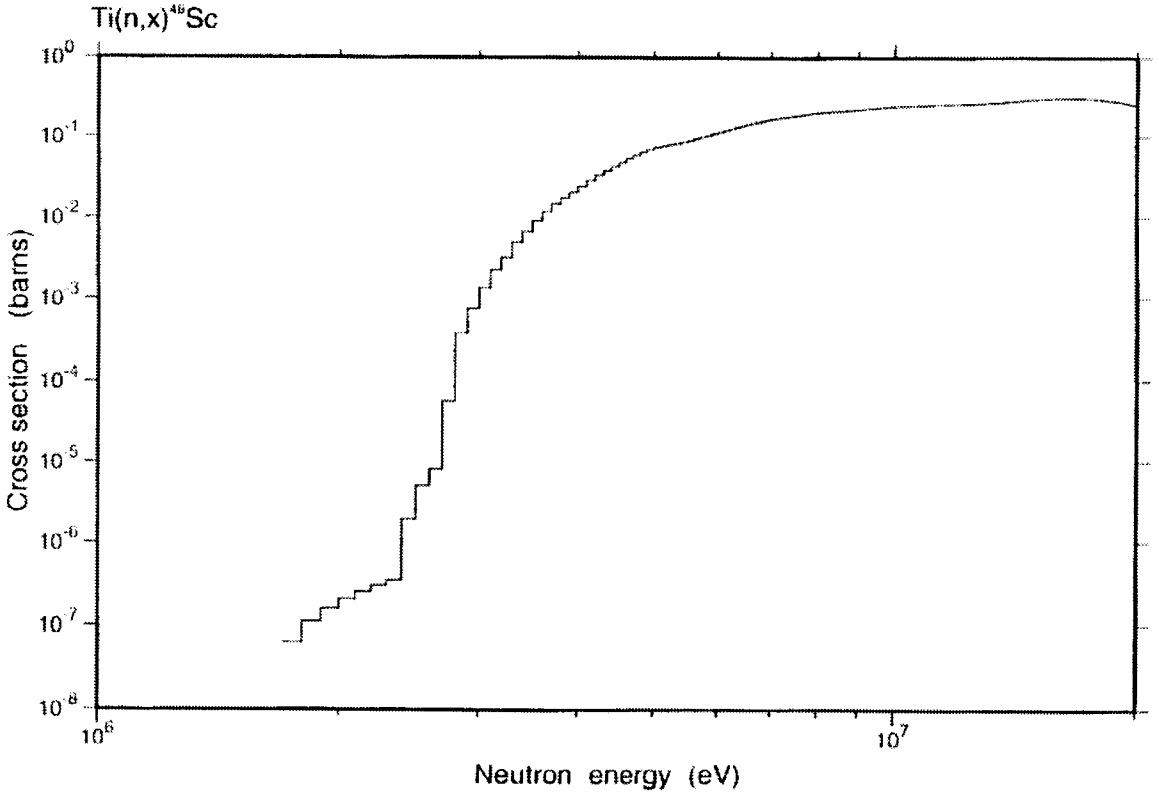

Fig. C.20. Ti(n,x)^{46}Sc (SNLRML-V1, private communication) cross section vs energy.

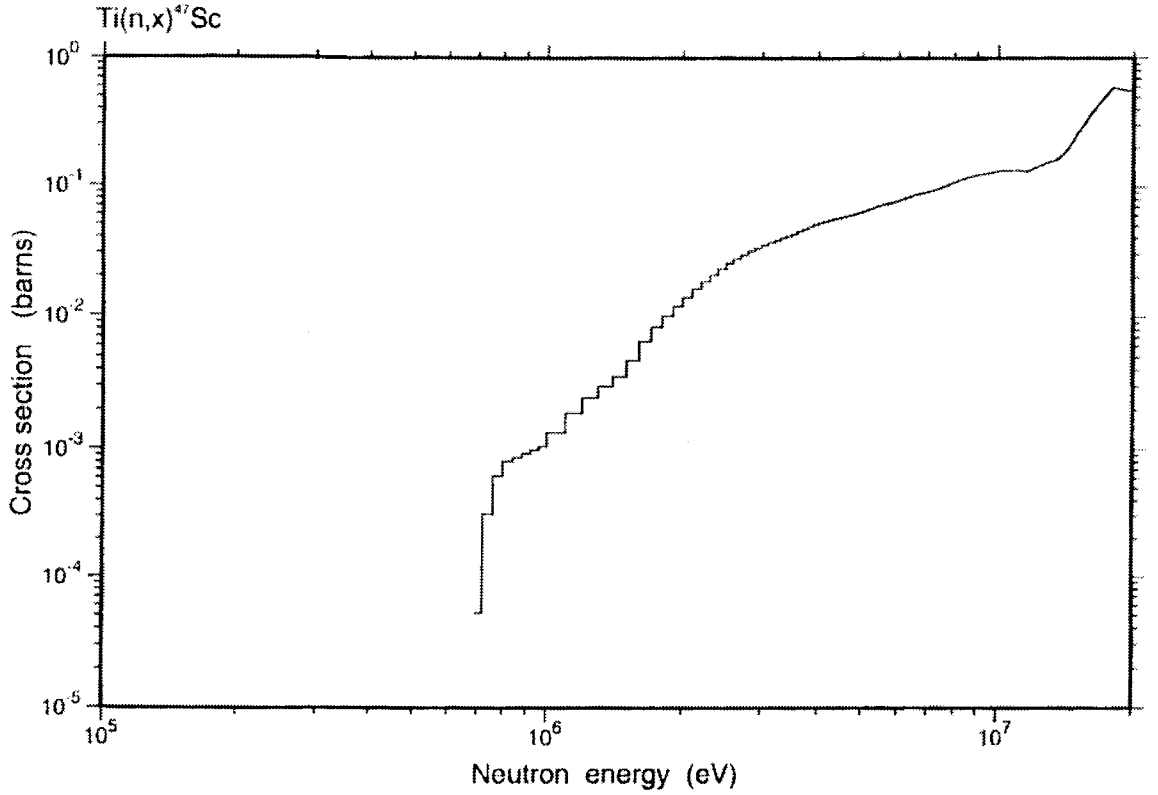

Fig. C.21. Ti(n,x)^{47}Sc (SNLRML-V1, private communication) cross section vs energy.

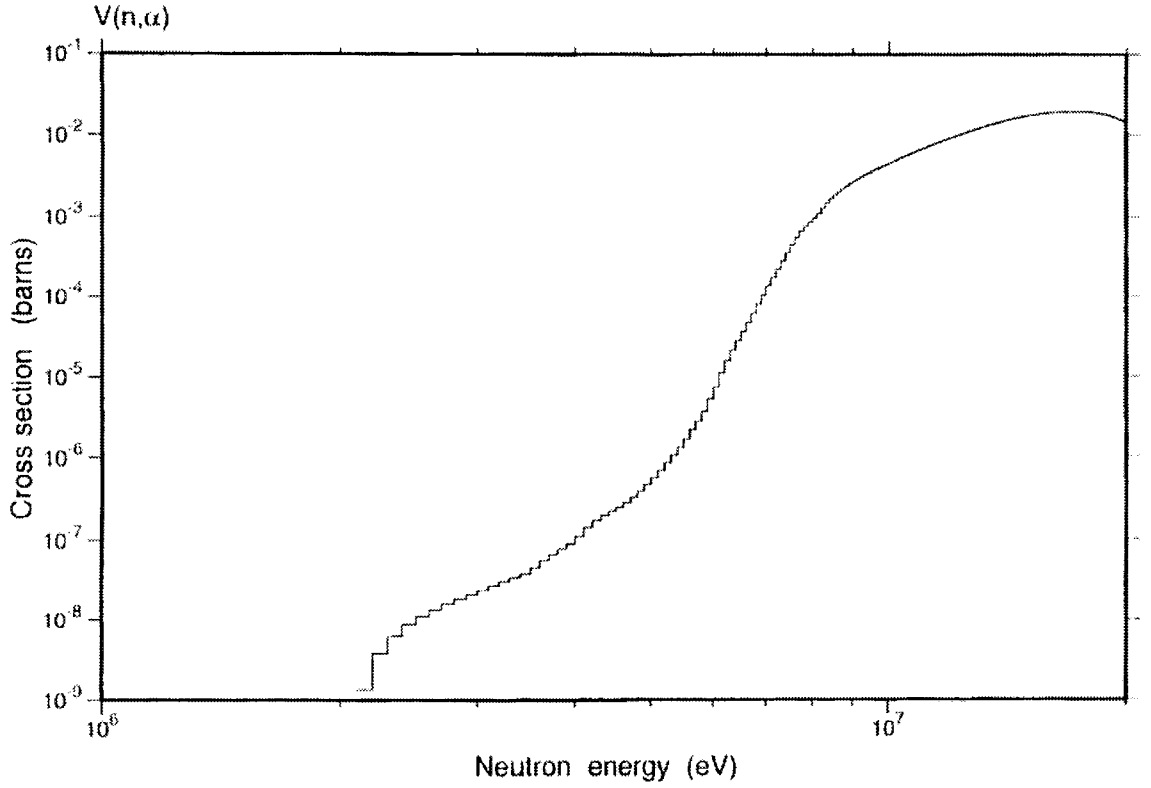

Fig. C.22. V(n,α)Sc (IRDF-90/G-V2, private communication) cross section vs energy.

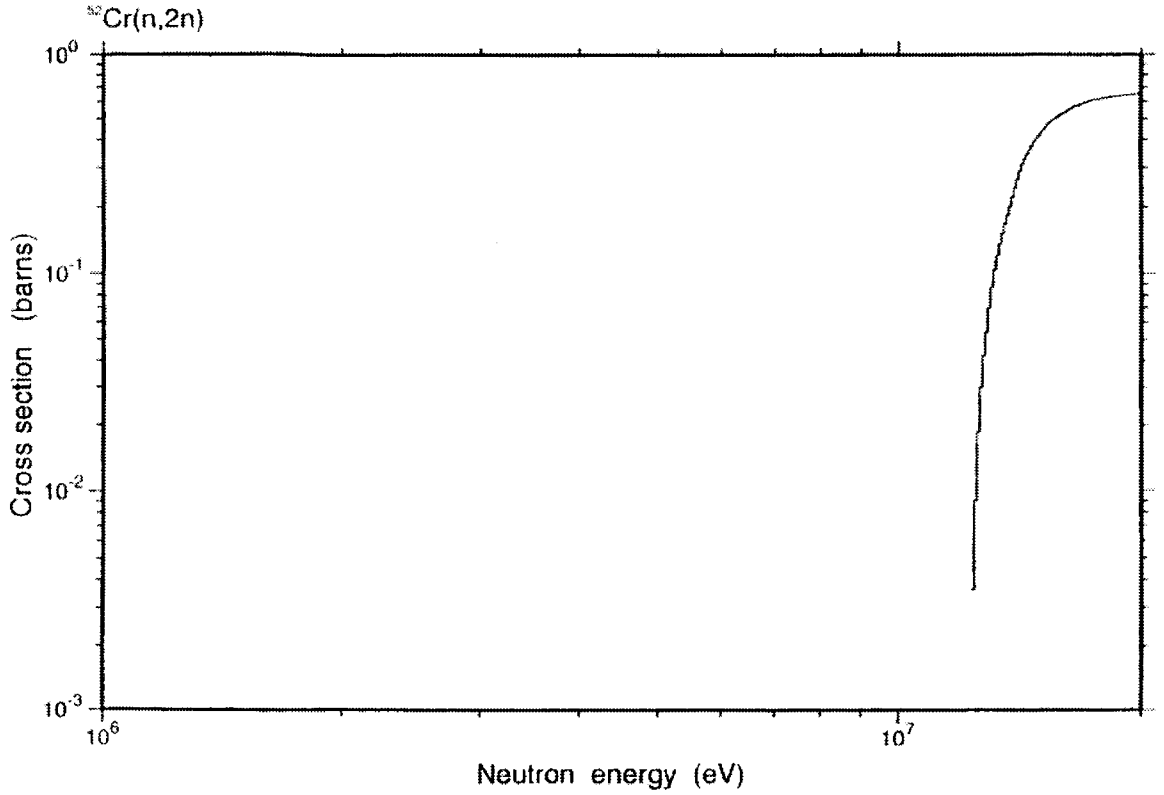

Fig. C.23. $^{52}Cr(n,2n)^{51}Cr$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

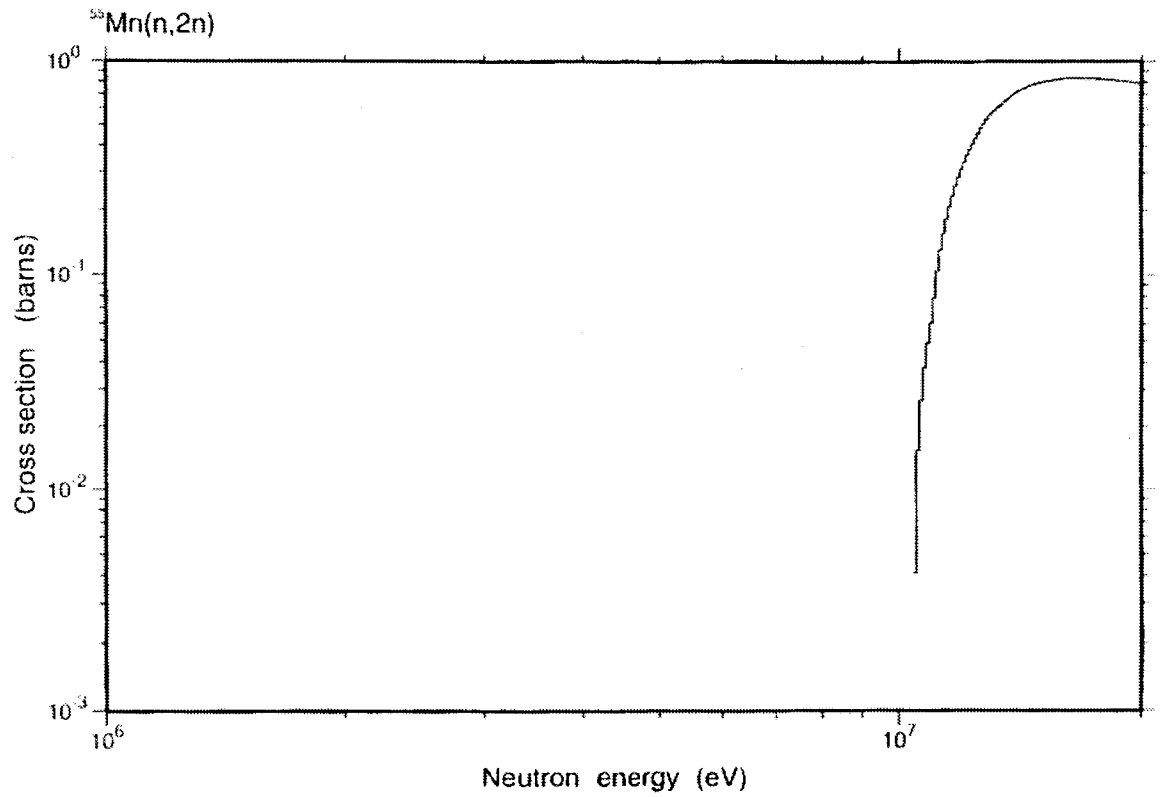

Fig. C.24. $^{55}Mn(n,2n)^{54}Mn$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

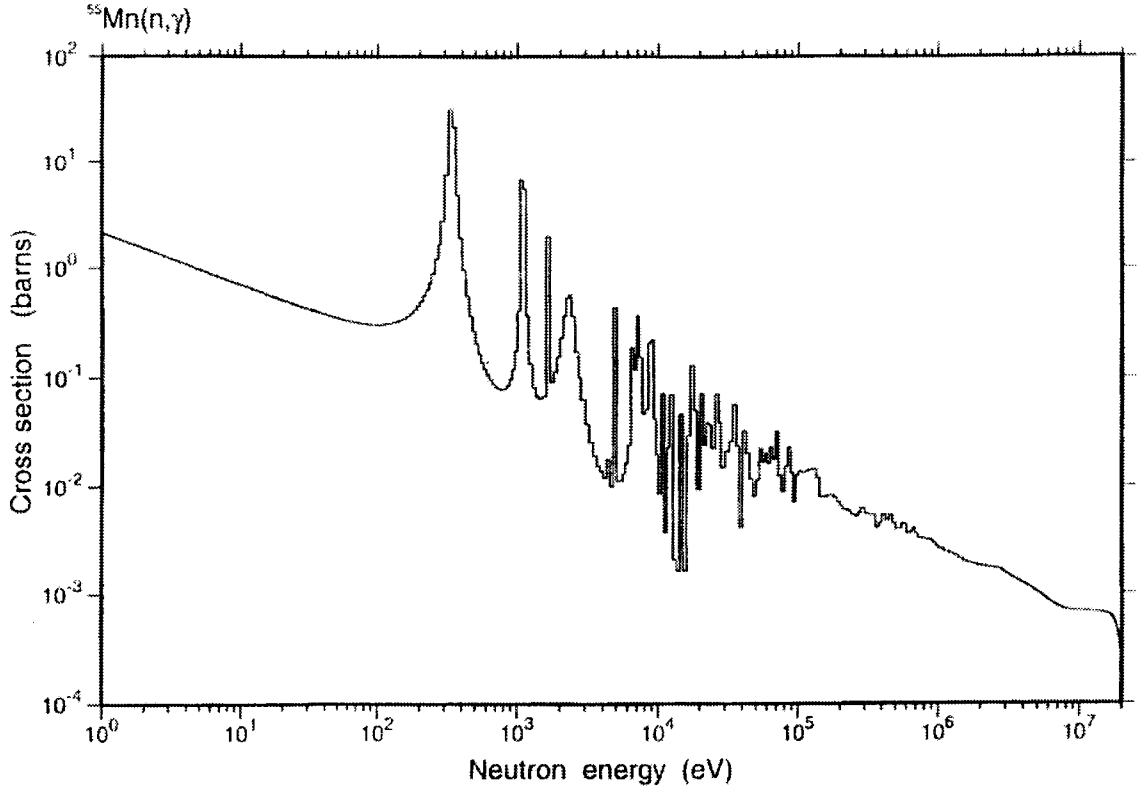

Fig. C.25. $^{55}Mn(n,\gamma)^{56}Mn$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

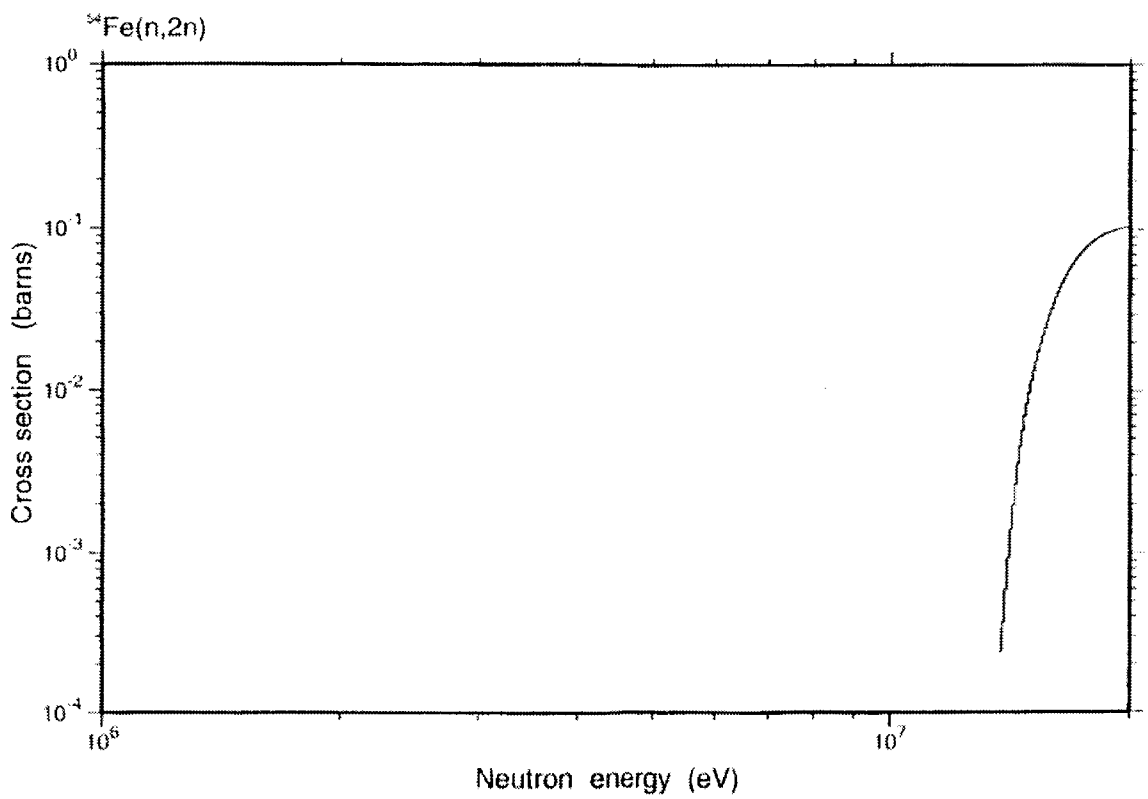

Fig. C.26. $^{54}Fe(n,2n)^{53}Fe$ (RDF-94, private communication) cross section vs energy.

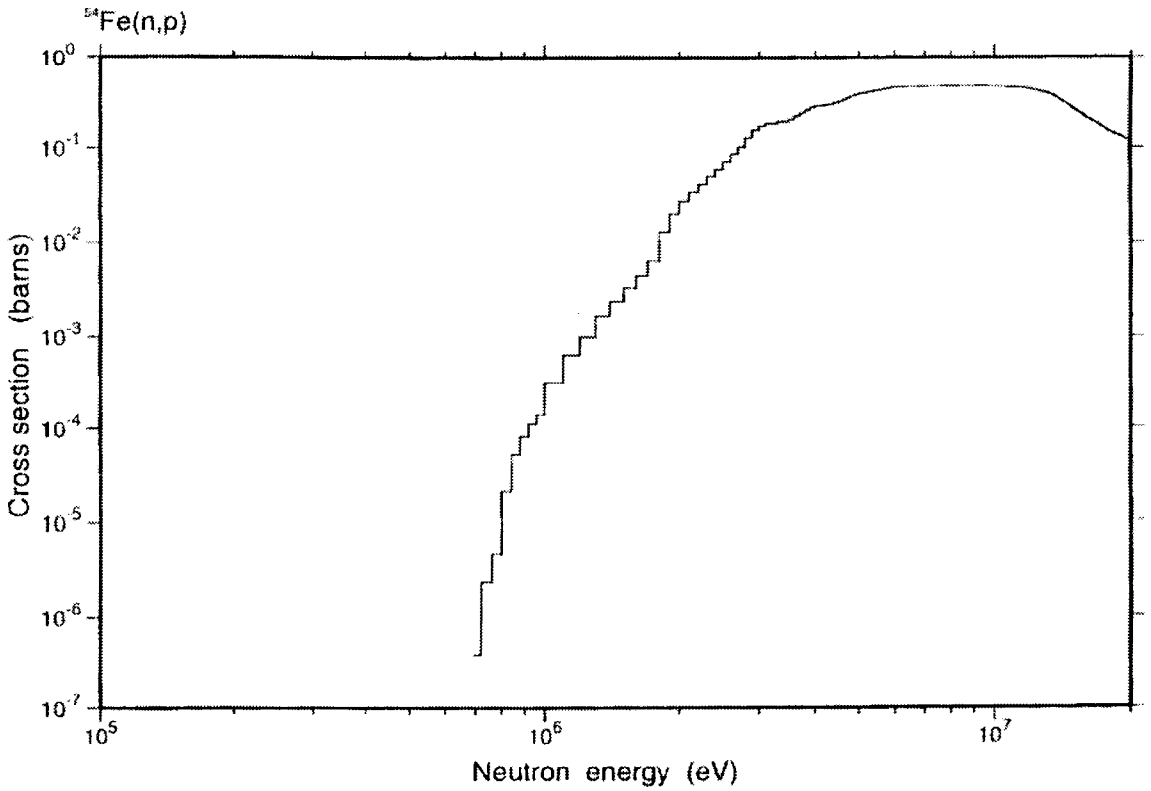

Fig. C.27. ^{54}Fe(n,p)^{54}Mn (IRDF-90/G-V2, private communication) cross section vs energy.

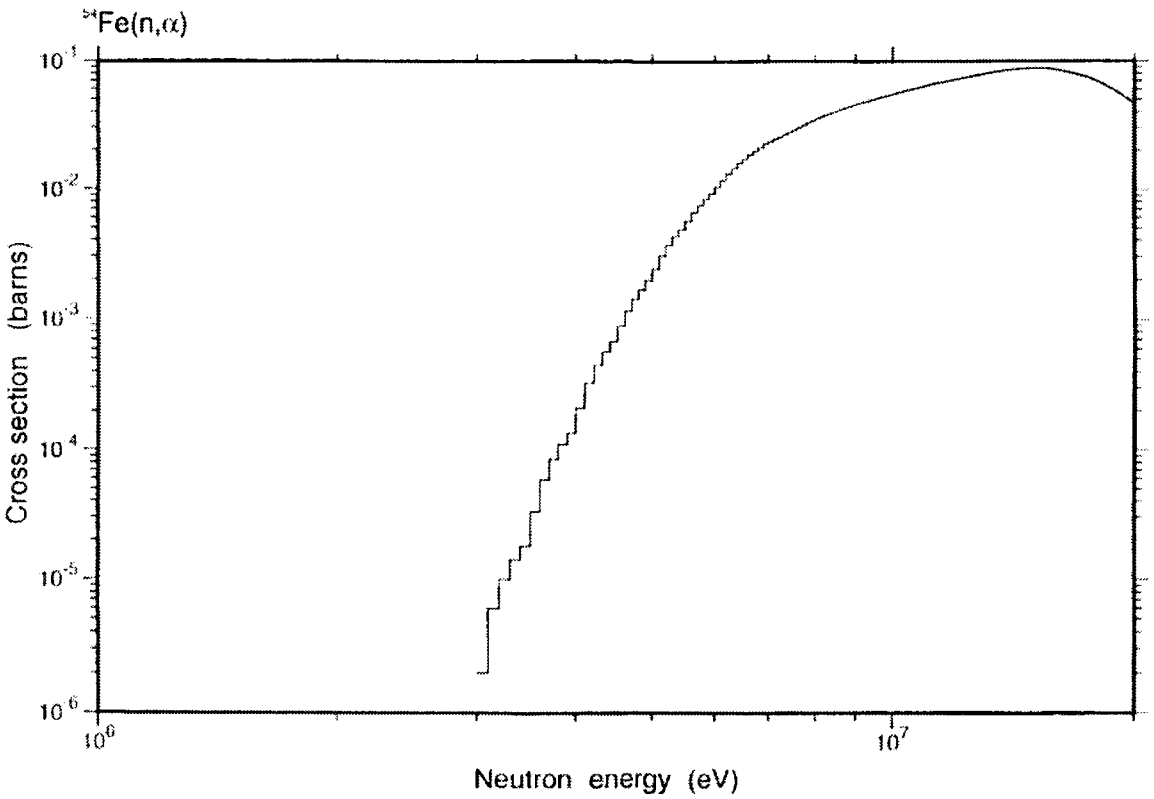

Fig. C.28. ^{54}Fe(n,α)^{51}Cr (SNLRML-V1, ENDF/B-VI), cross section vs energy.

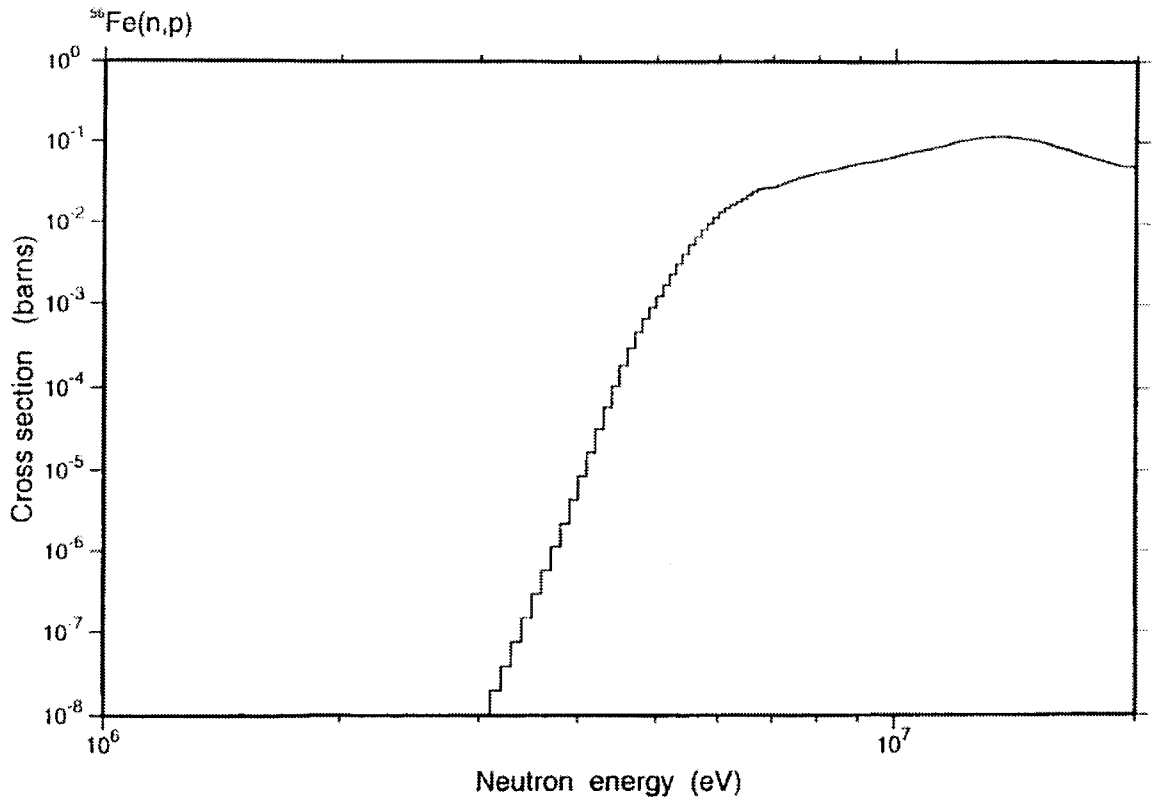

Fig. C.29. $^{56}Fe(n,p)^{56}Mn$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

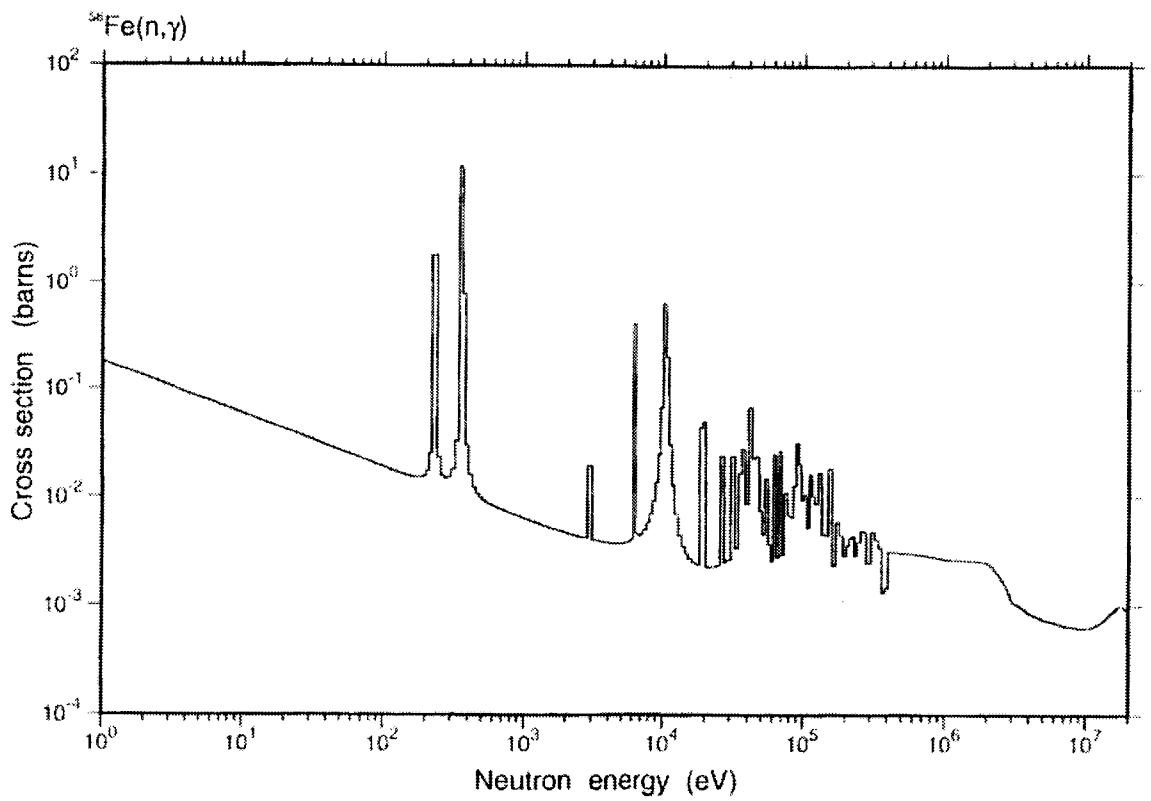

Fig. C.30. $^{58}Fe(n,\gamma)^{59}Fe$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

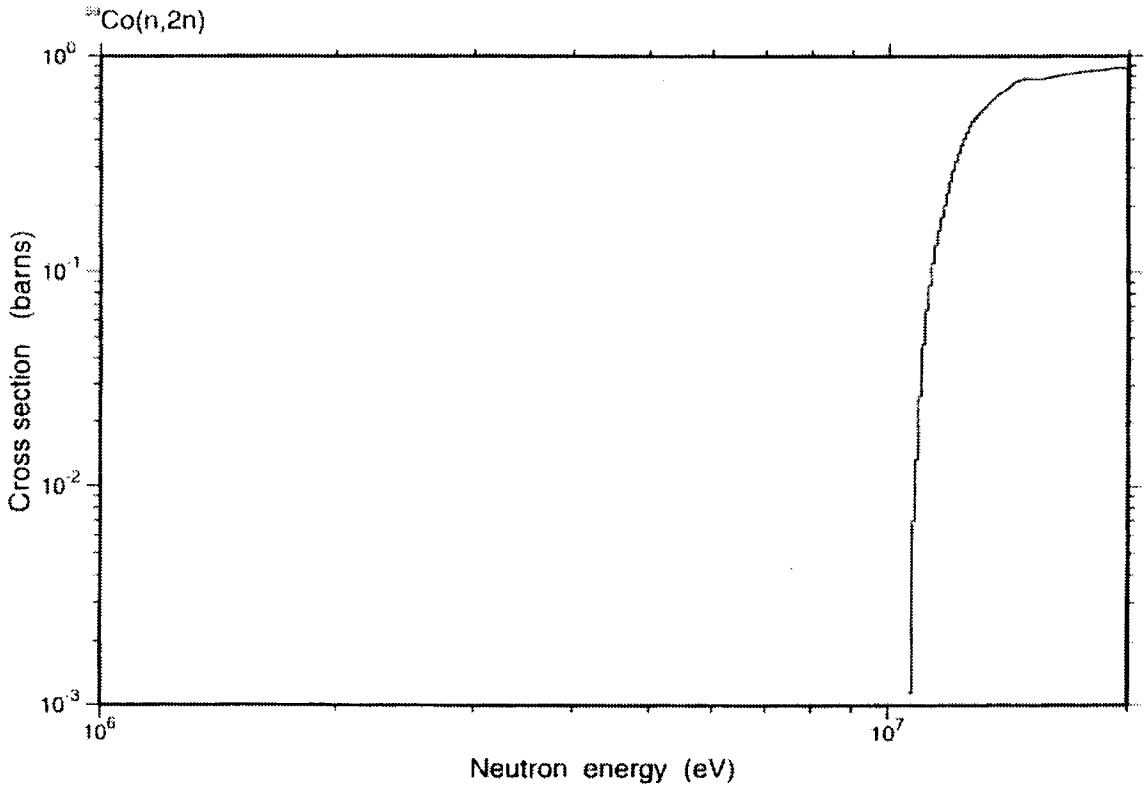

Fig. C.31. $^{59}Co(n,2n)^{58}Co$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

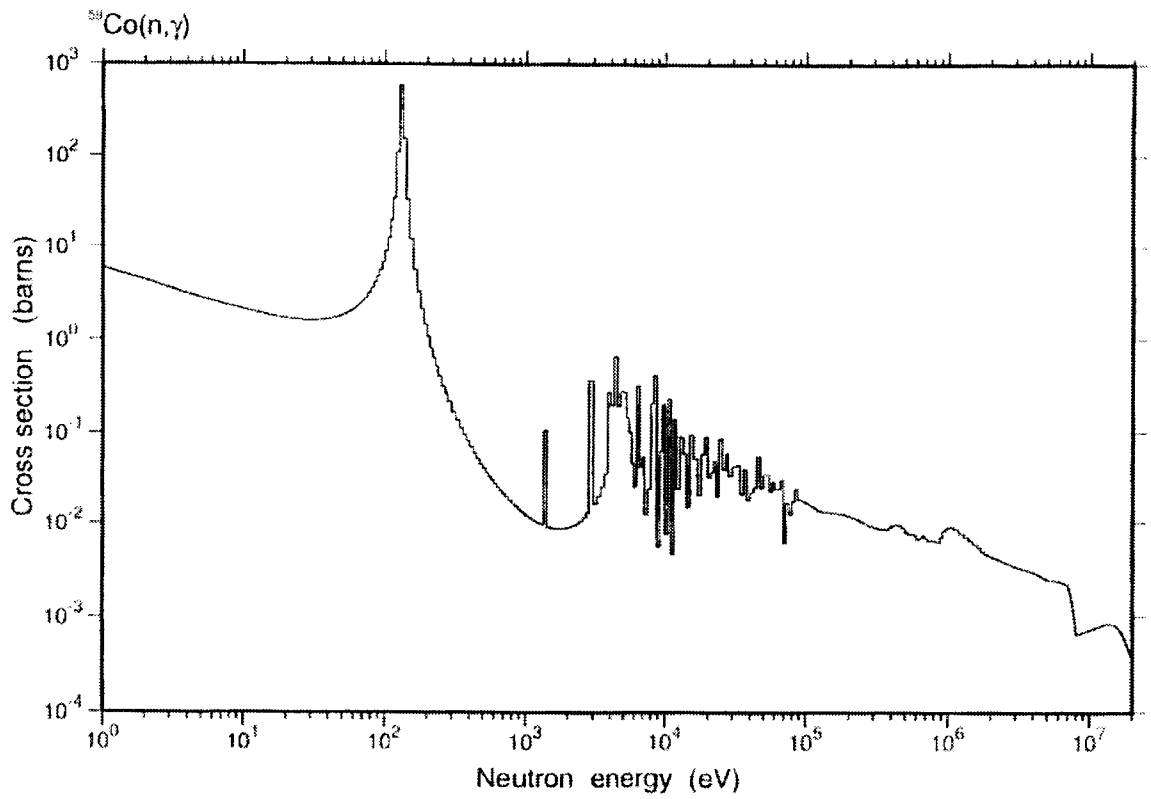

Fig. C.32. $^{59}Co(n,\gamma)^{60}Co$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

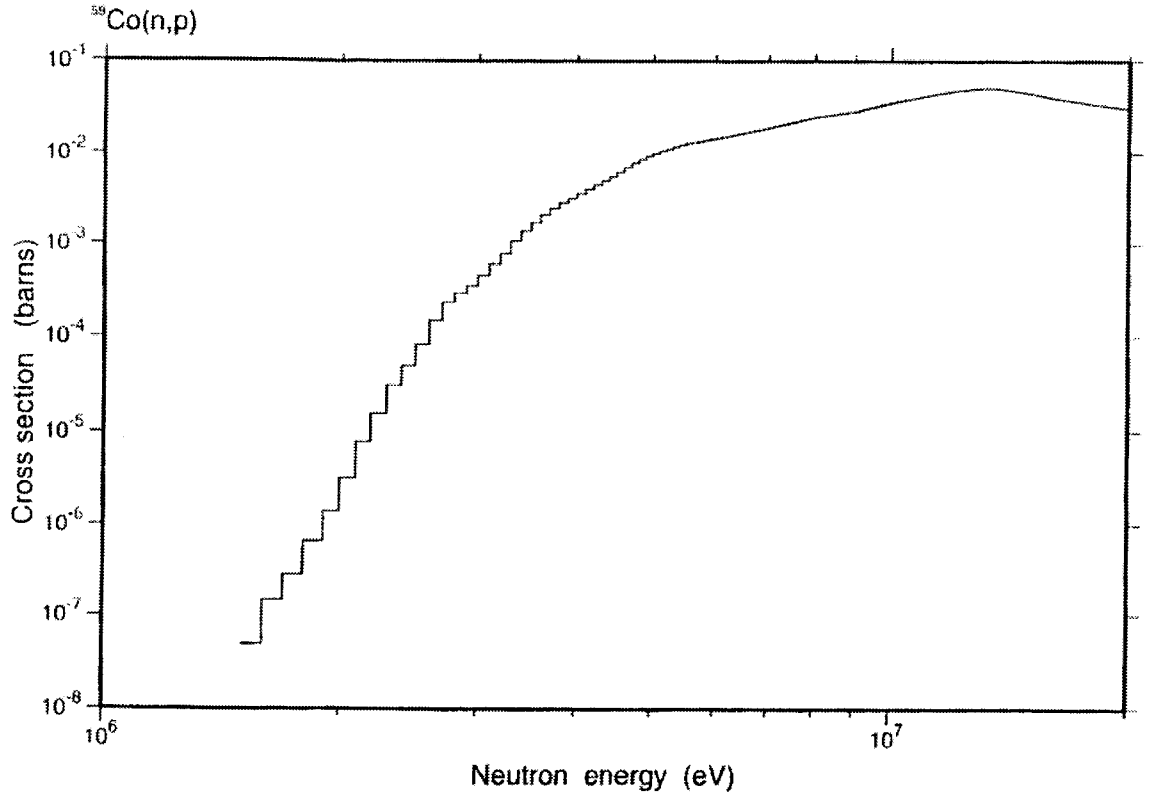

Fig. C.33. $^{59}Co(n,p)^{59}Fe$ (SNLRML-V1, ENDF/B-VI), cross section vs energy.

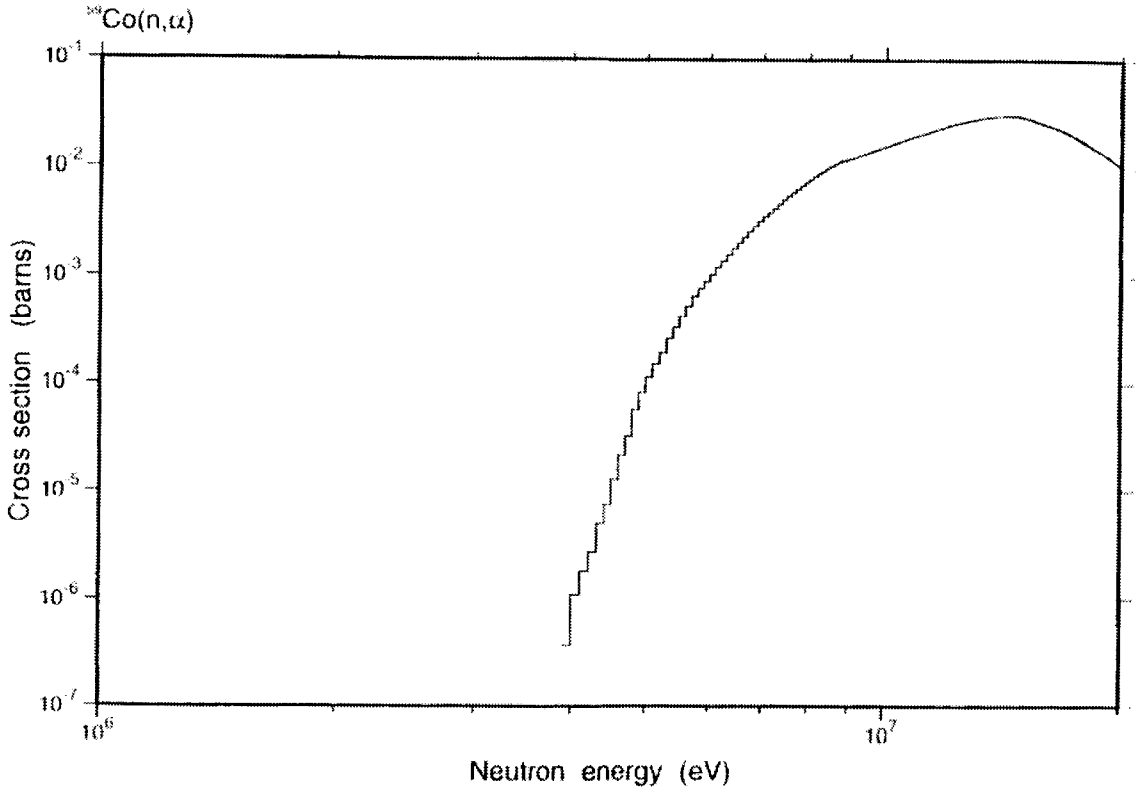

Fig. C.34. $^{59}Co(n,\alpha)^{56}Mn$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

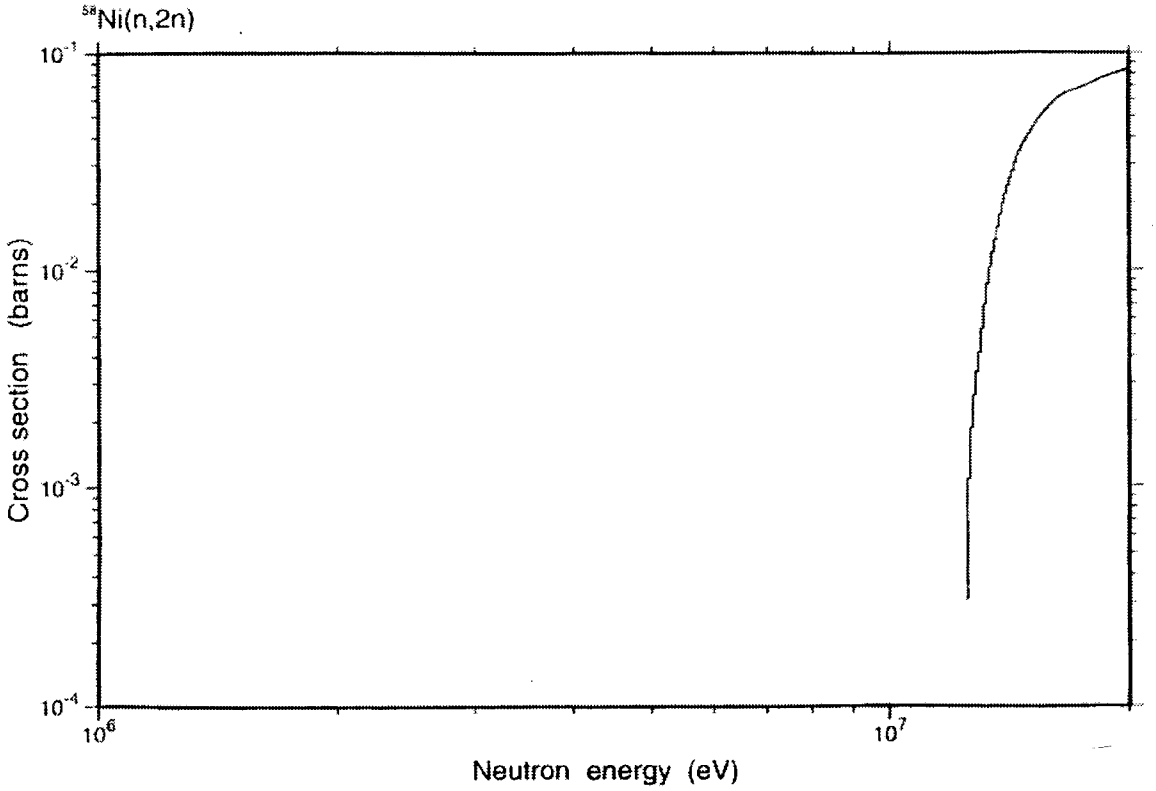

Fig. C.35. ^{58}Ni(n,2n)^{57}Ni (IRDF-90/G-V2, original evaluation) cross section vs energy.

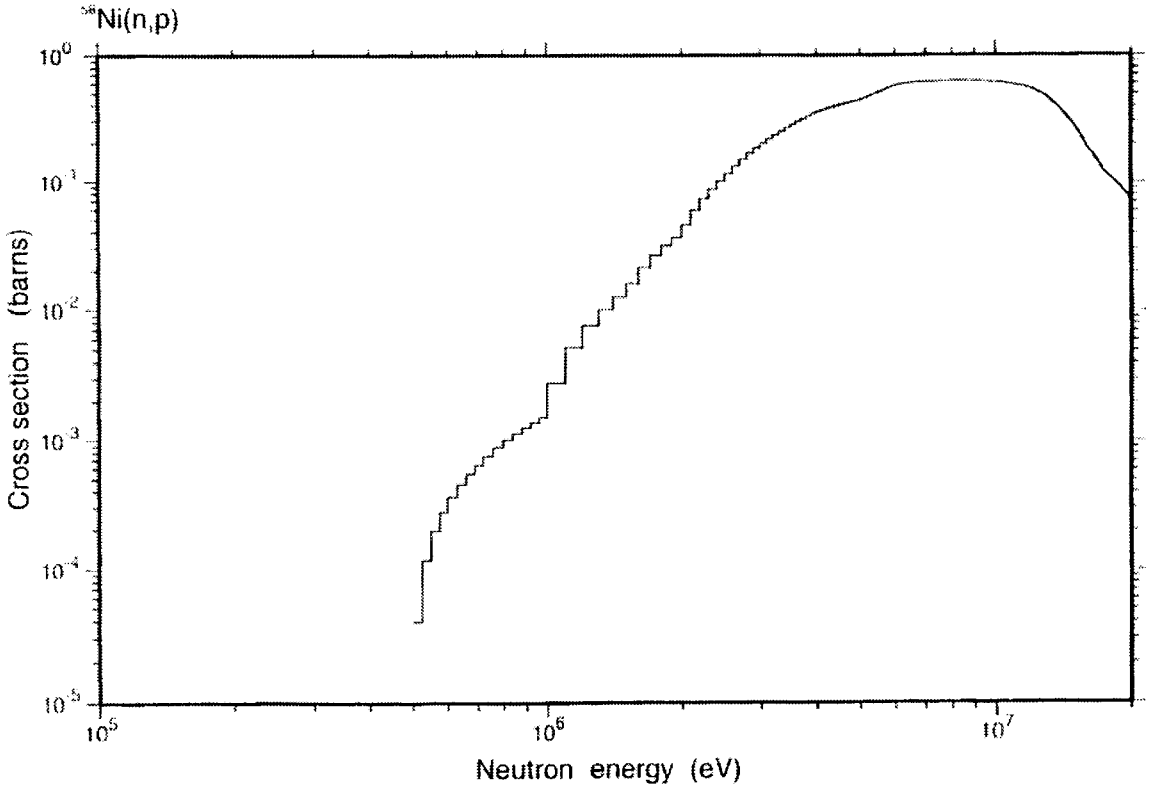

Fig. C.36. ^{58}Ni(n,p)^{58}Co (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

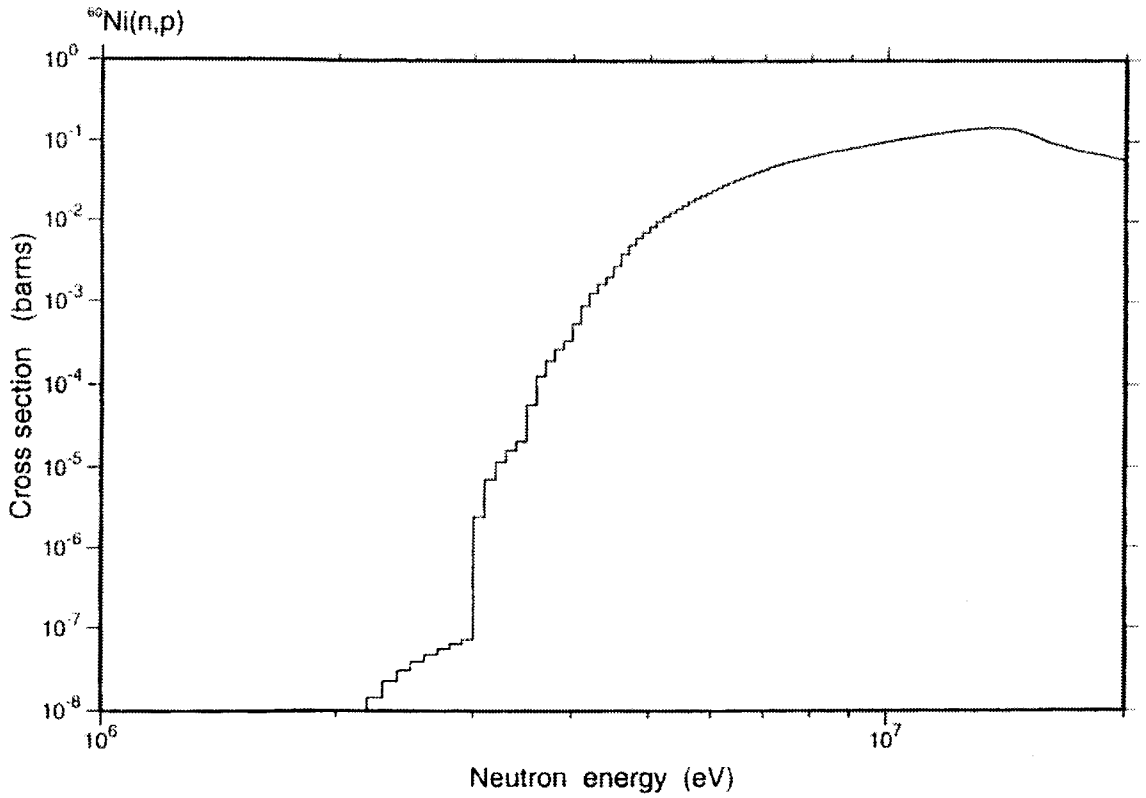

Fig. C.37. ^{60}Ni(n,p)^{60}Co (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

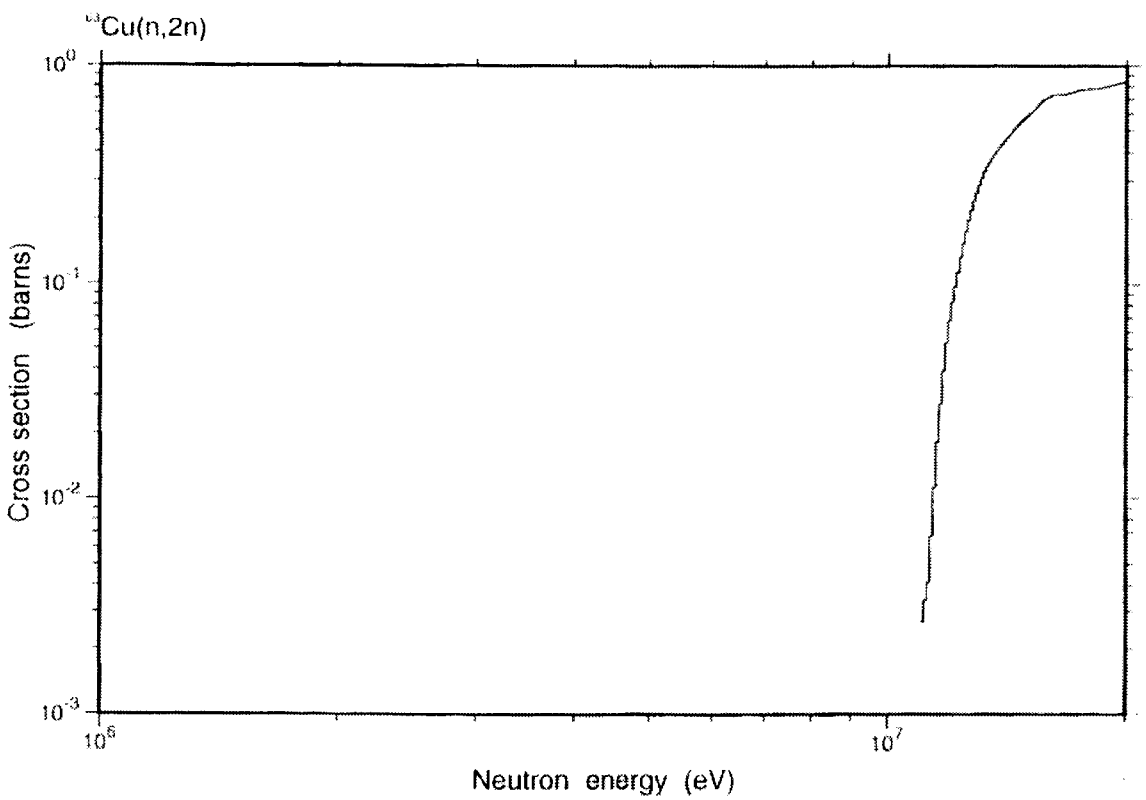

Fig. C.38. ^{63}Cu($n,2n$)^{62}Cu (IRDF-90/G-V2, original evaluation) cross section vs energy.

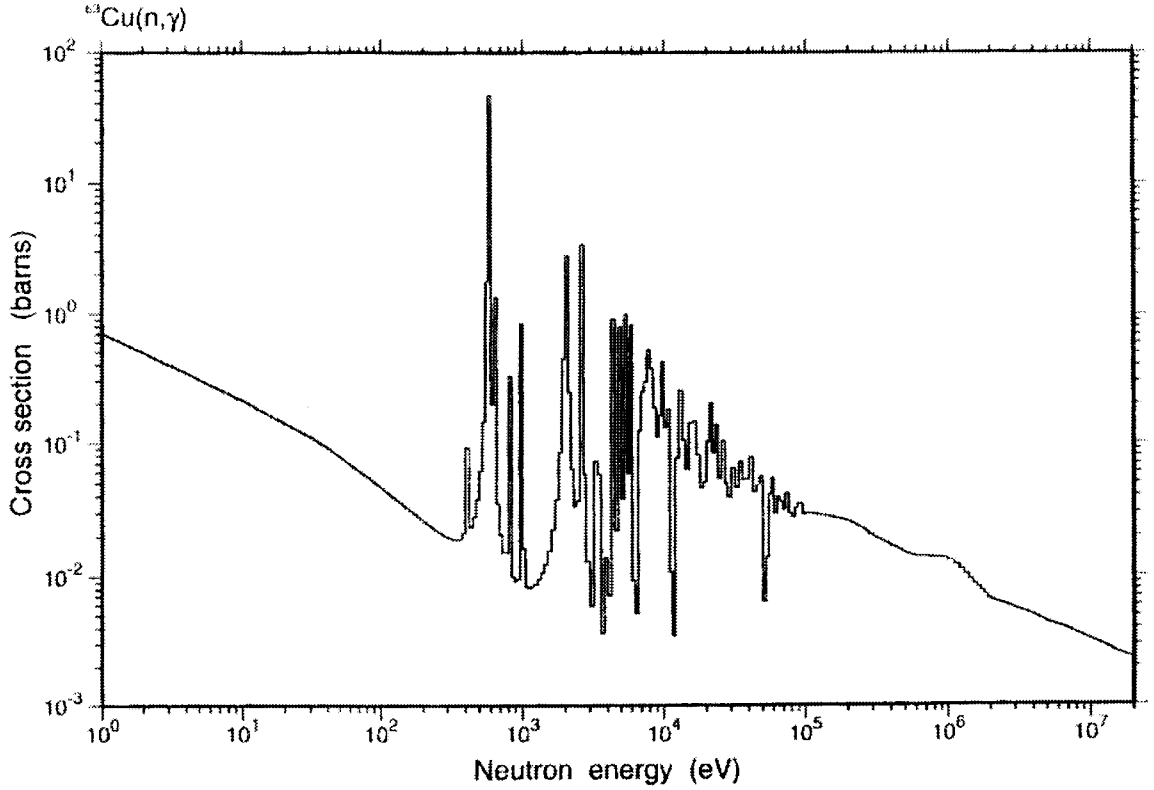

Fig. C.39. ^{63}Cu(n,γ)^{64}Cu (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

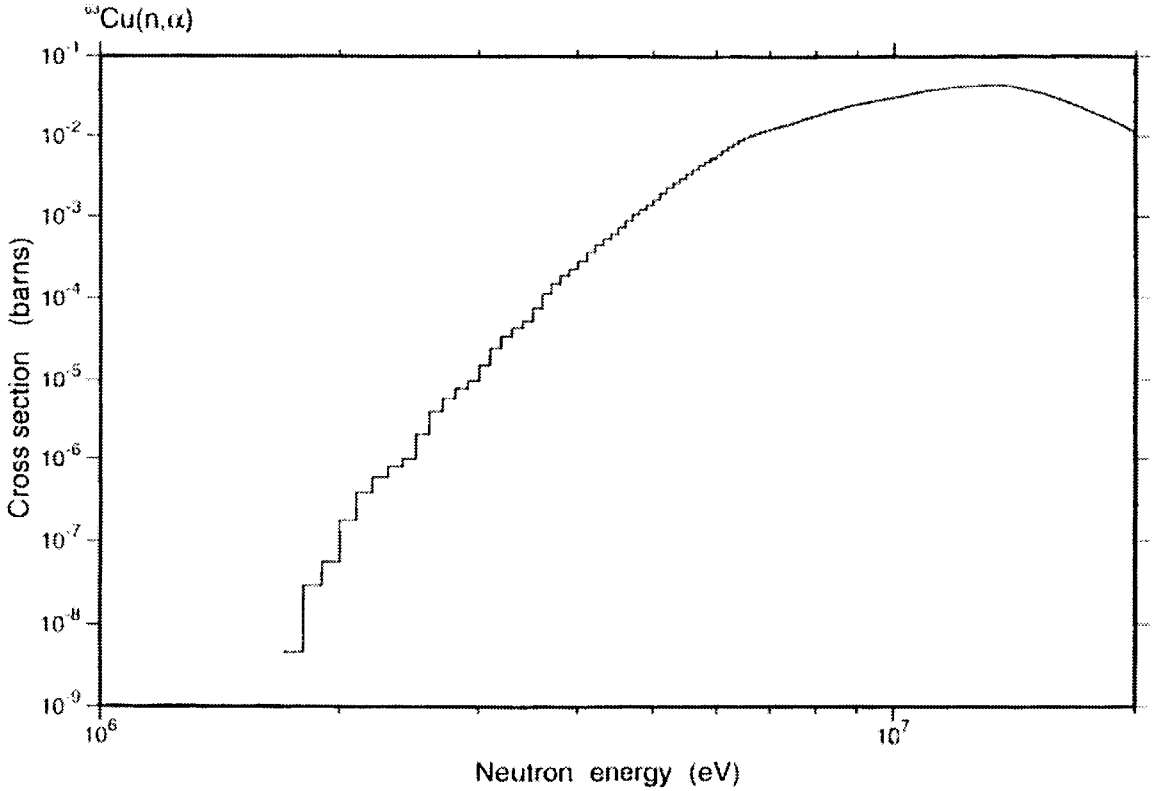

Fig. C.40. ^{63}Cu(n,α)^{60}Co (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

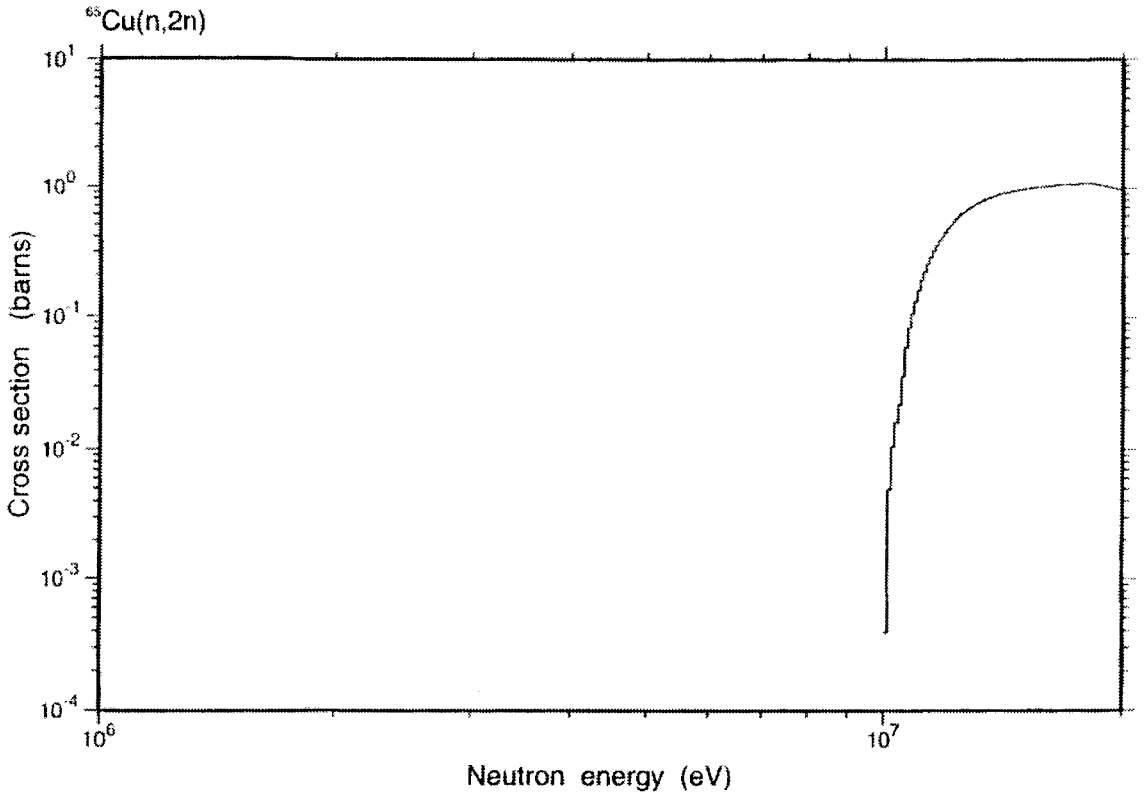

Fig. C.41. $^{65}Cu(n,2n)^{64}Cu$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

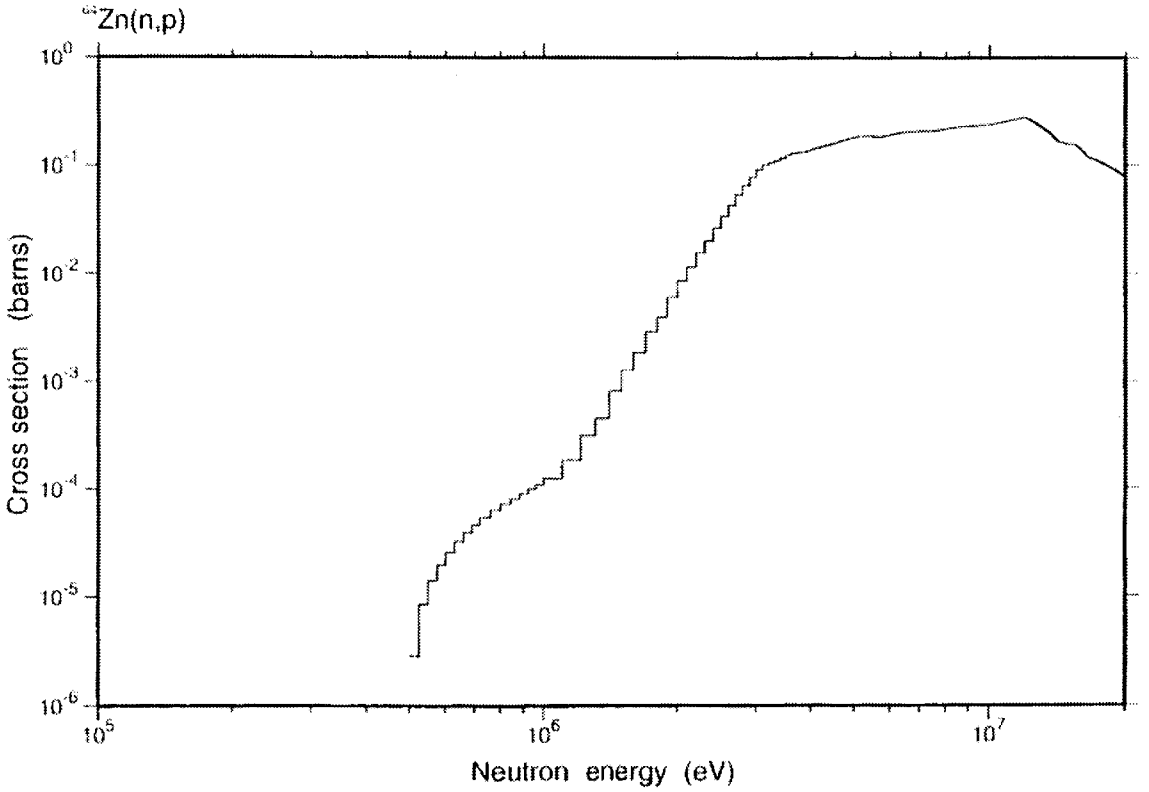

Fig. C.42. $^{64}Zn(n,p)^{64}Cu$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

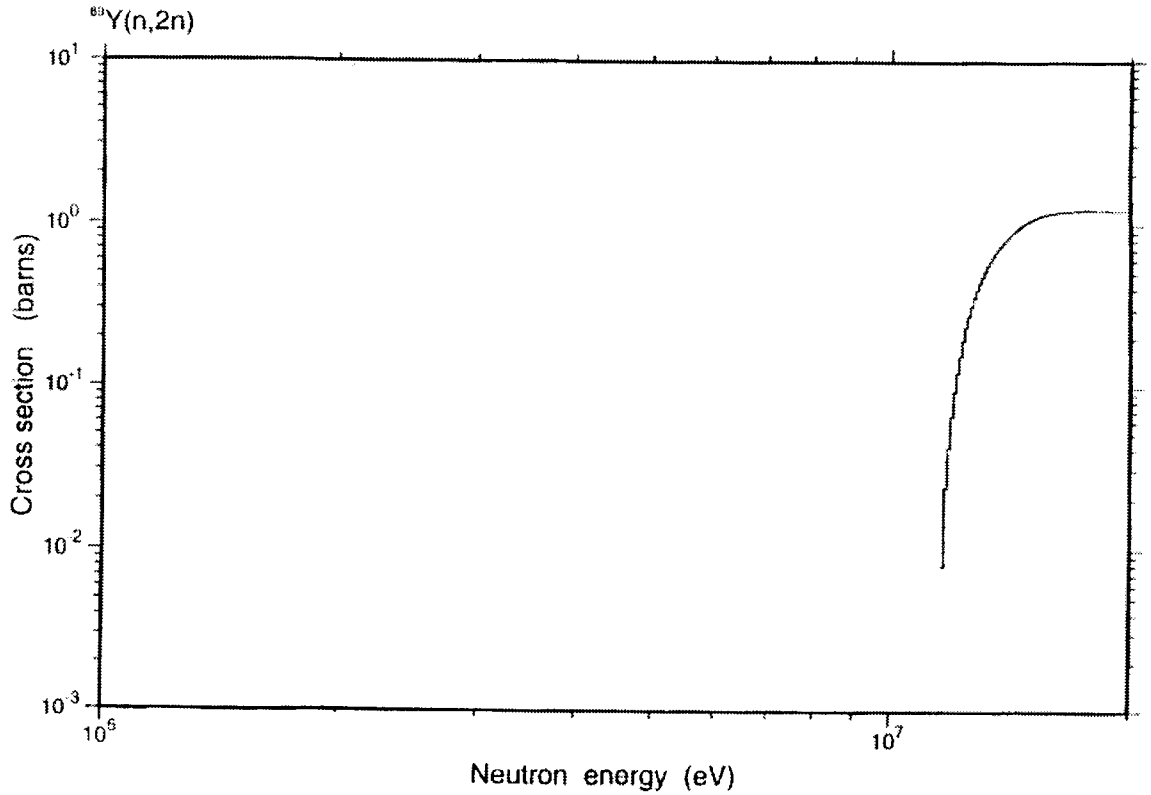

Fig. C.43. $^{89}Y(n,2n)^{88}Y$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

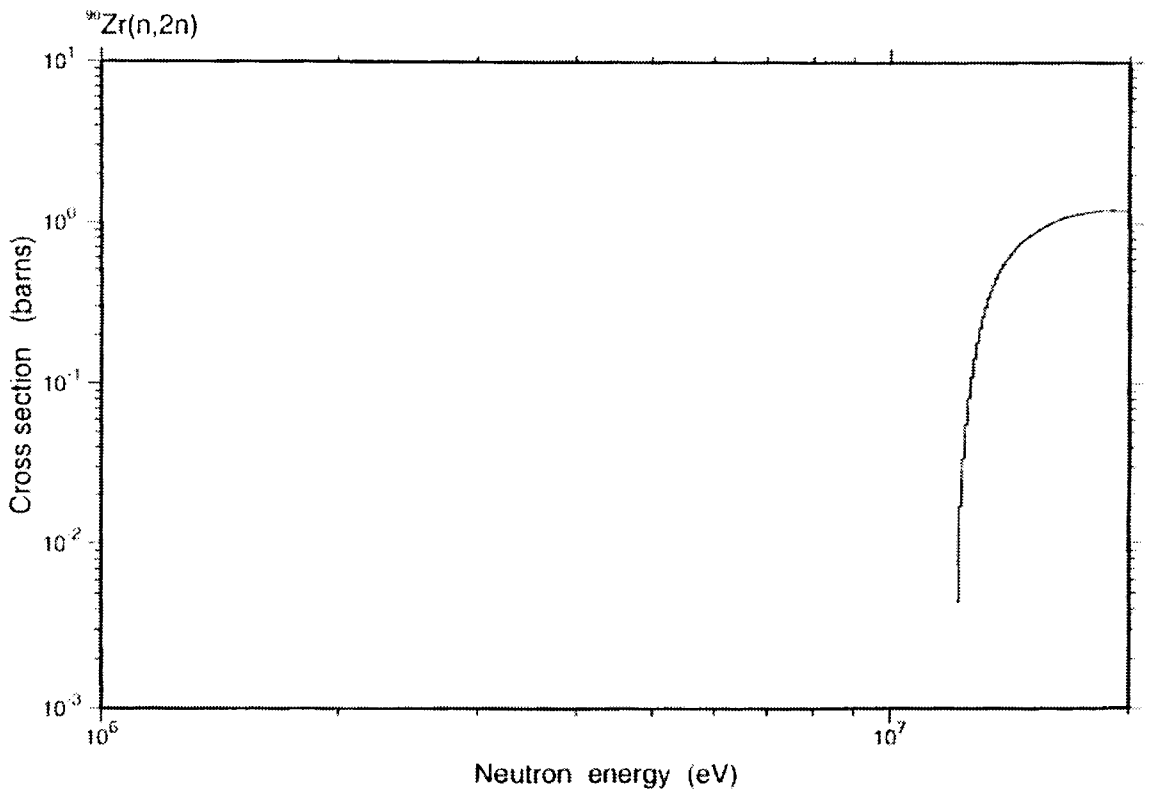

Fig. C.44. $^{90}Zr(n,2n)^{89}Zr$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

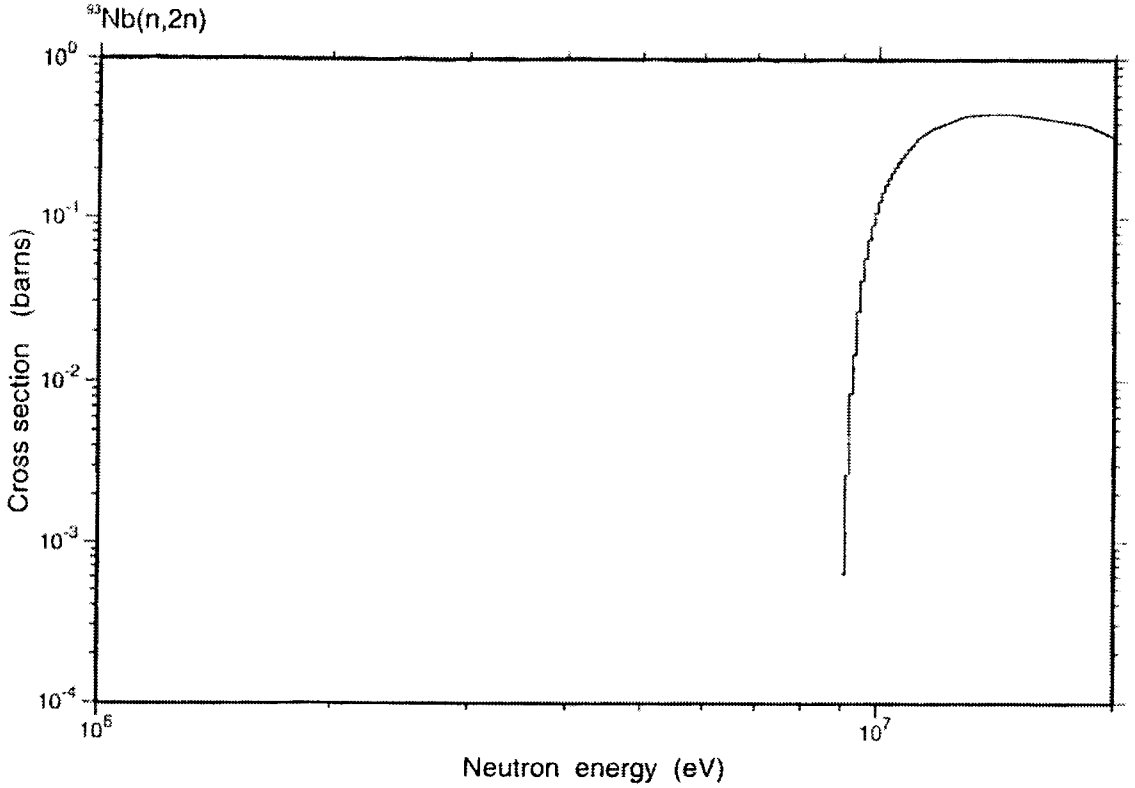

Fig. C.45. ^{93}Nb(n,2n)^{92}Nb (IRDF-90/G-V2, original evaluation) cross section vs energy.

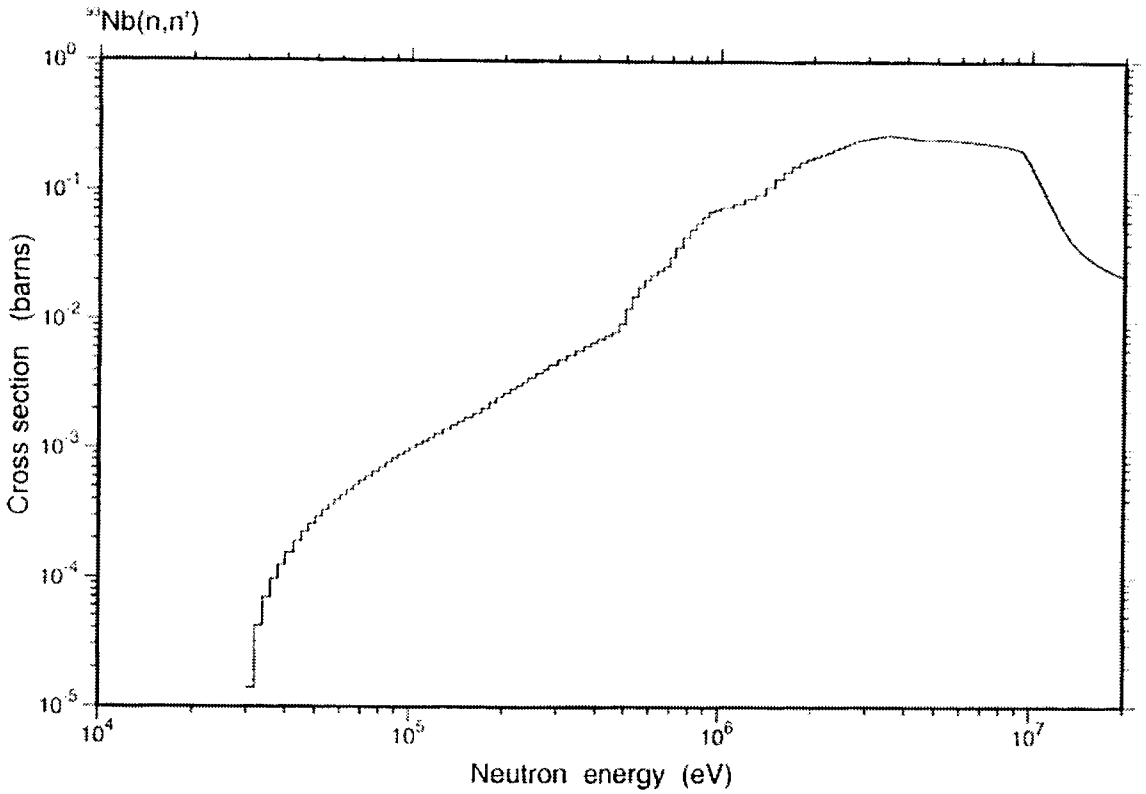

Fig. C.46. ^{93}Nb(n,n')^{93}Nb (IRDF-90/G-V2, original evaluation) cross section vs energy.

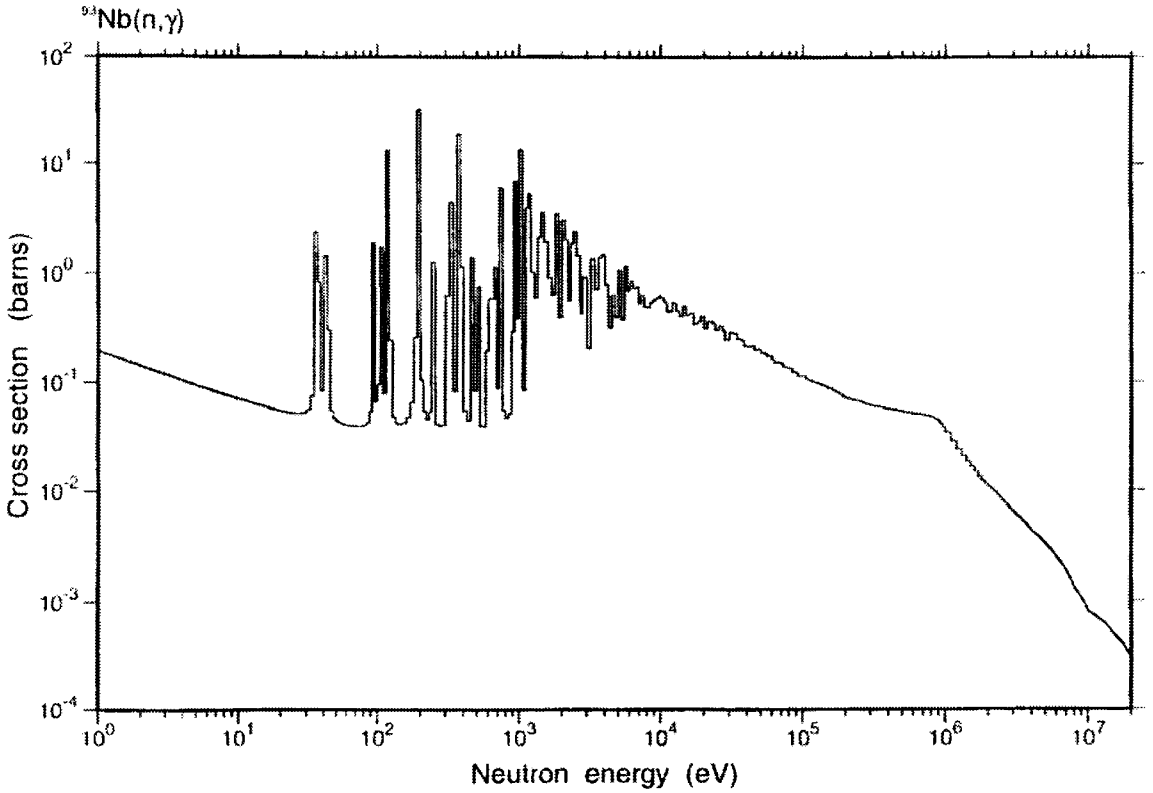

Fig. C.47. $^{93}Nb(n,\gamma)^{94}Nb$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

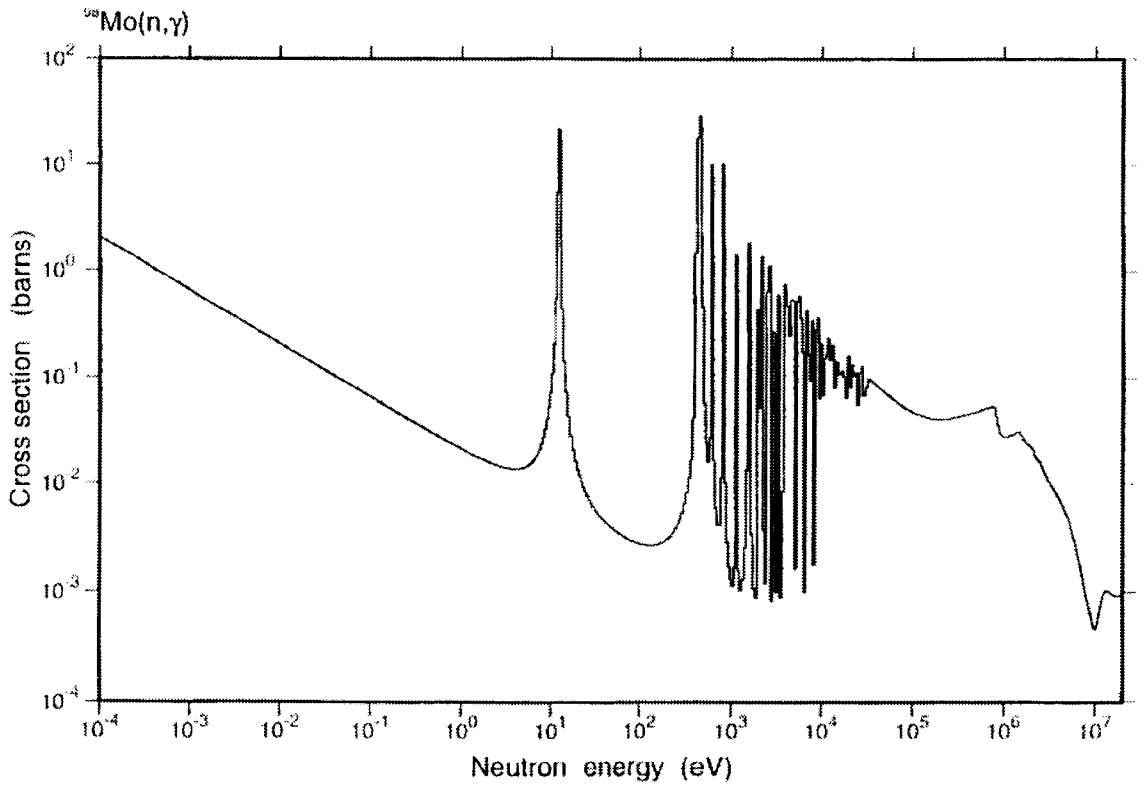

Fig. C.48. $^{98}Mo(n,\gamma)^{99}Mo$ (SNLRML-V1, JENDL-3) cross section vs energy.

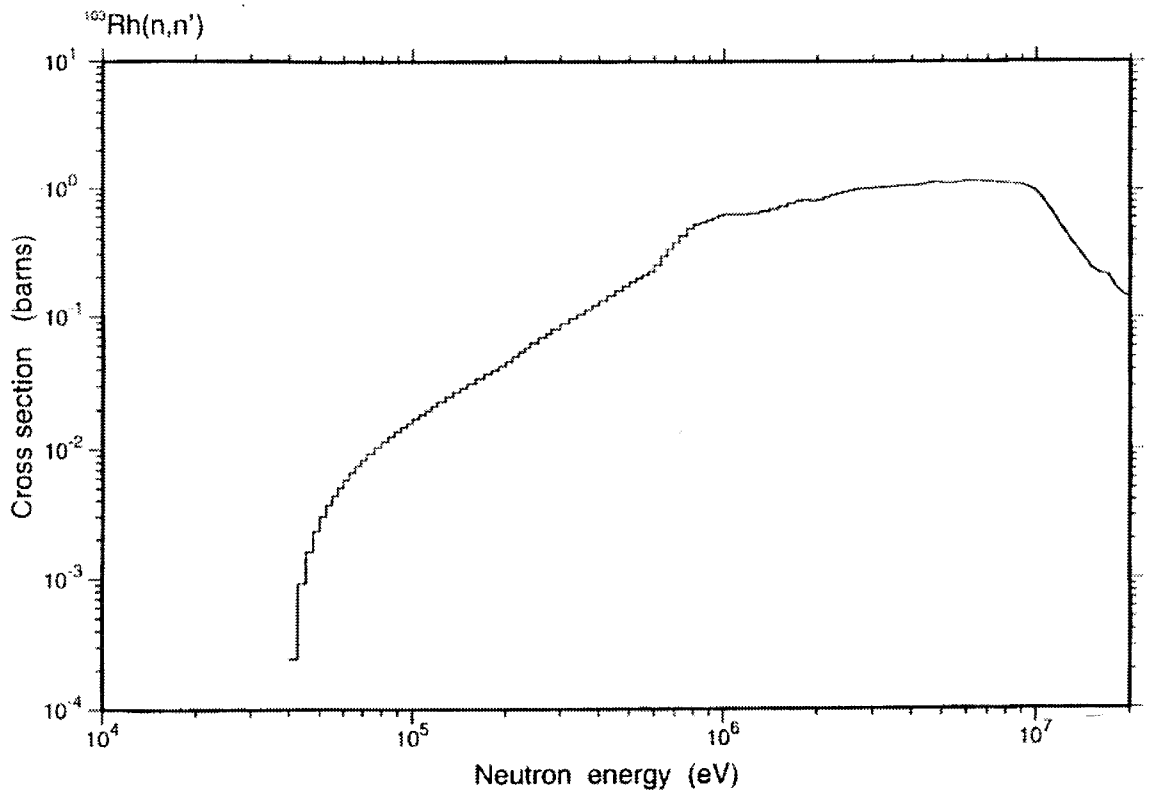

Fig. C.49. ^{103}Rh(n,n')^{103}Rh (IRDF-90/G-V2, original evaluation) cross section vs energy.

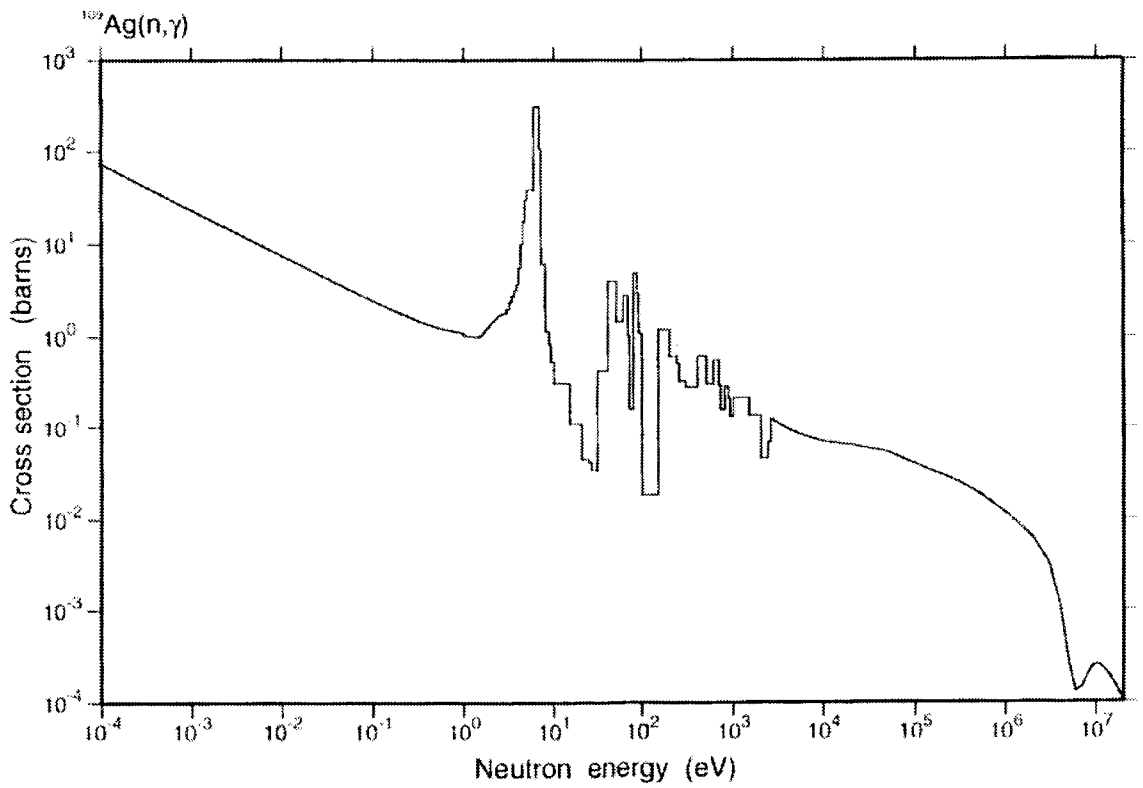

Fig. C.50. ^{109}Ag(n,γ)^{110}Ag (IRDF-90/G-V2, private communication) cross section vs energy.

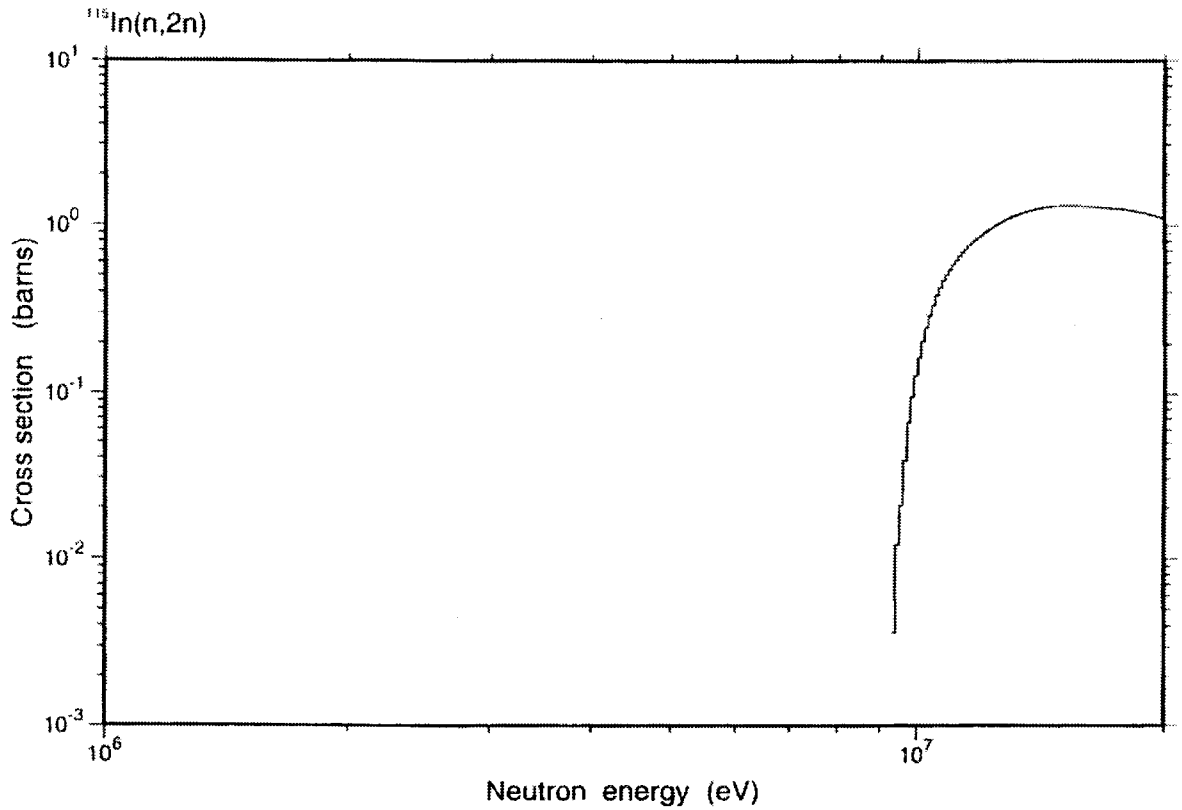

Fig. C.51. $^{115}In(n,2n)^{114}In$ (IRDF-90/G-V2, private communication) cross section vs energy.

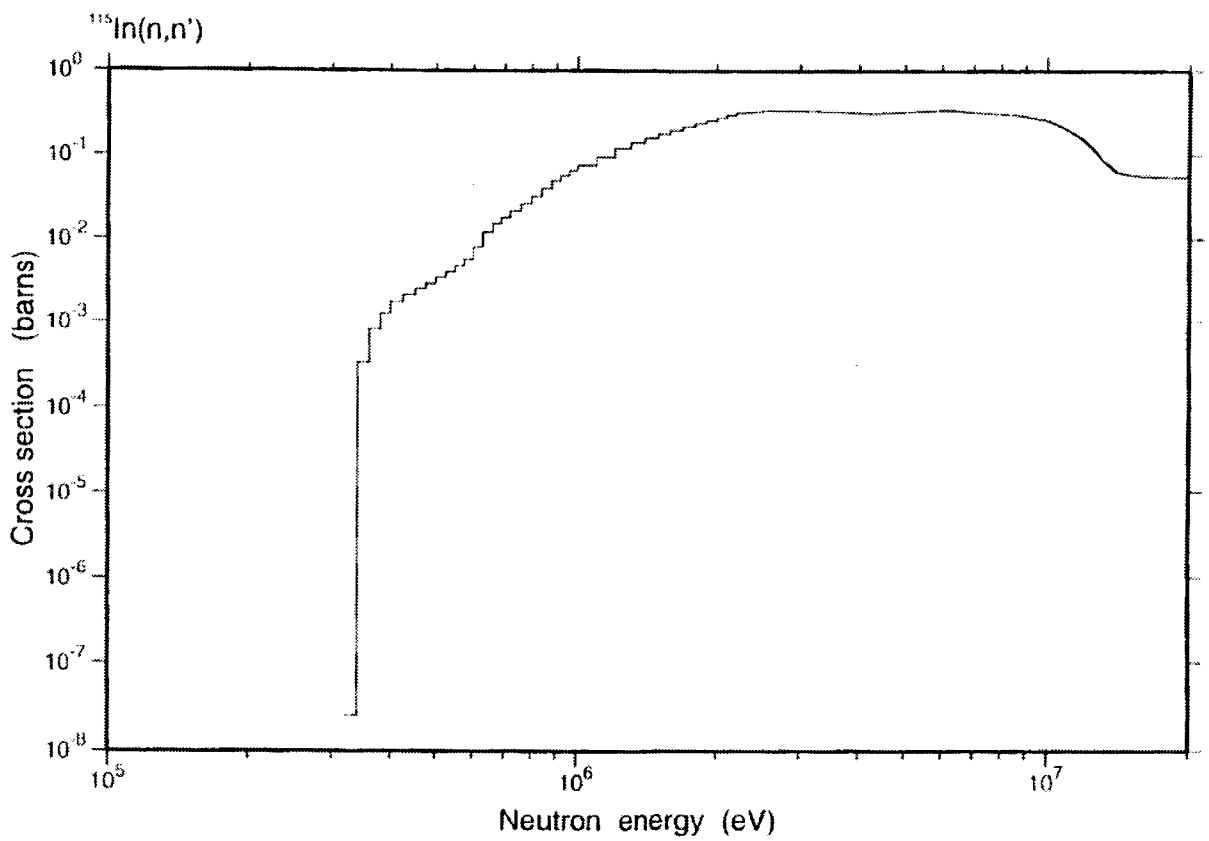

Fig. C.52. $^{115}In(n,n')^{115}In$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

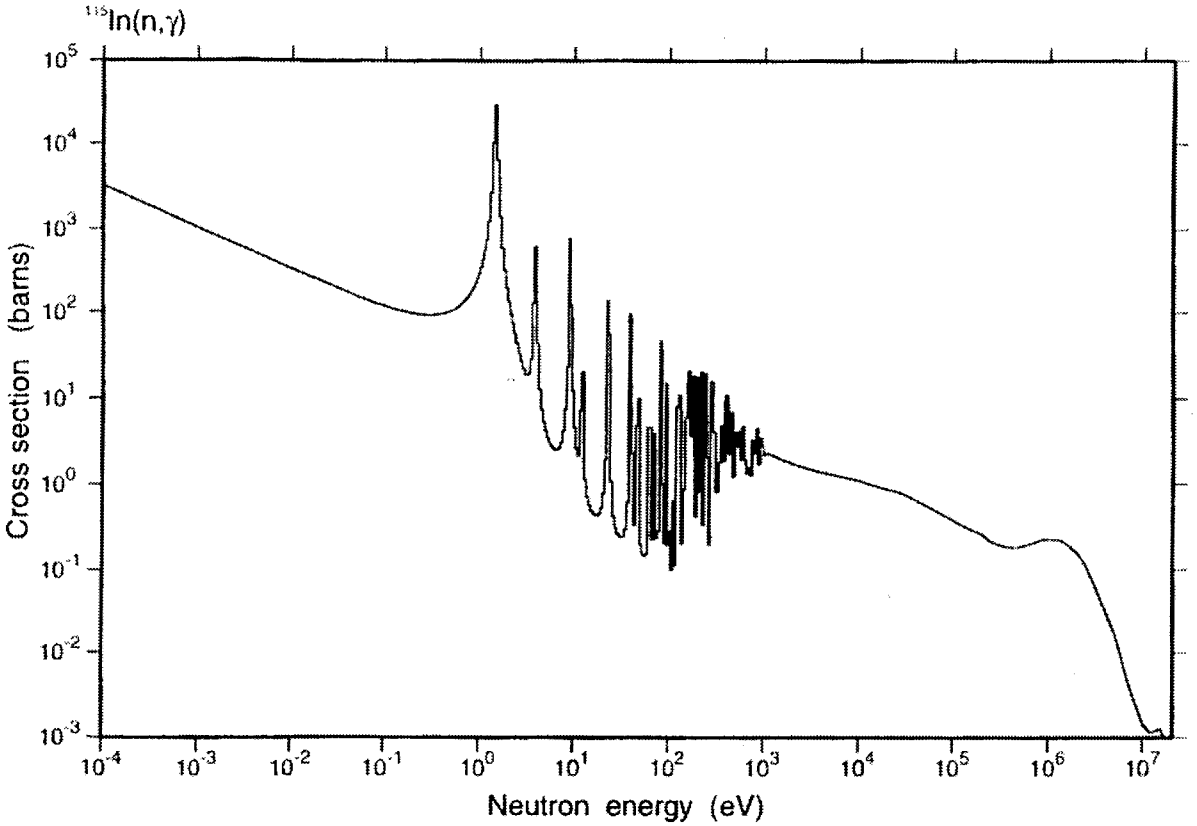

Fig. C.53. $^{115}In(n,\gamma)^{116}In$ (IRDF-90/G-V2, private communication) cross section vs energy.

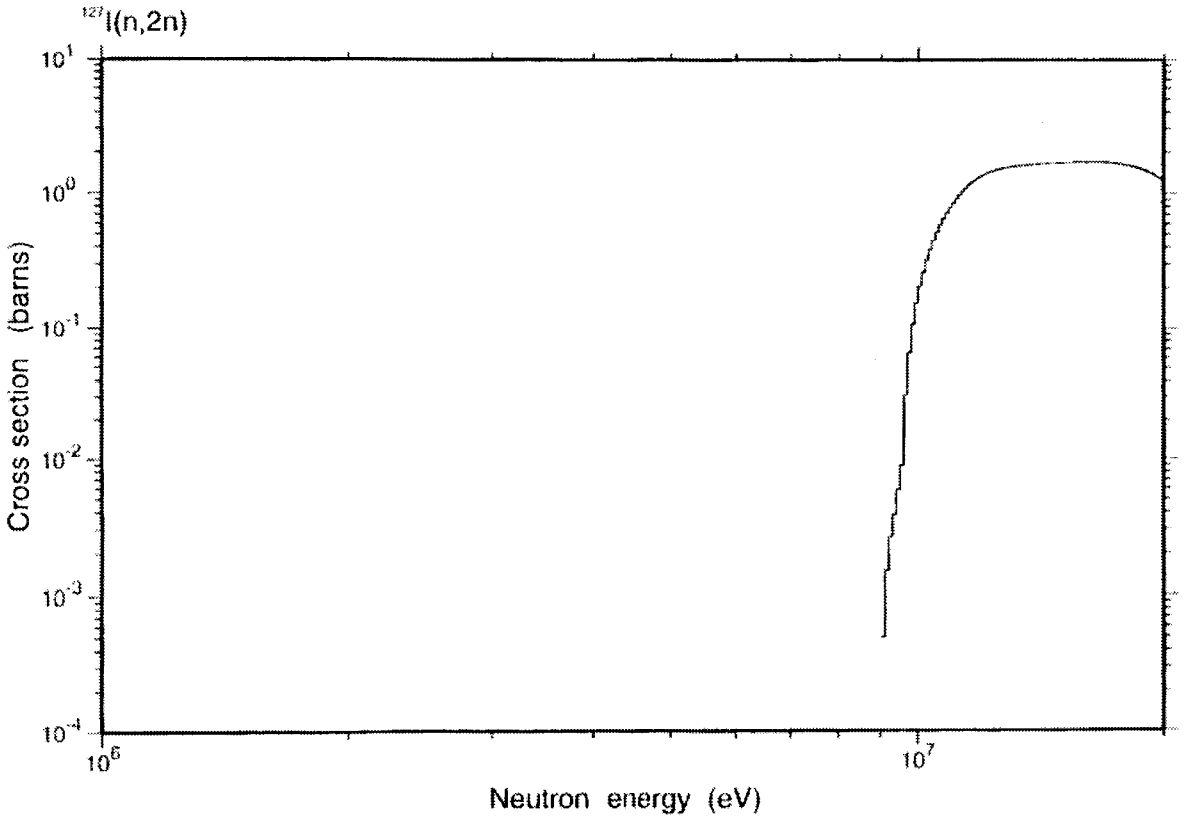

Fig. C.54. $^{127}I(n,2n)^{126}I$ (IRDF-90/G-V2, private communication) cross section vs energy.

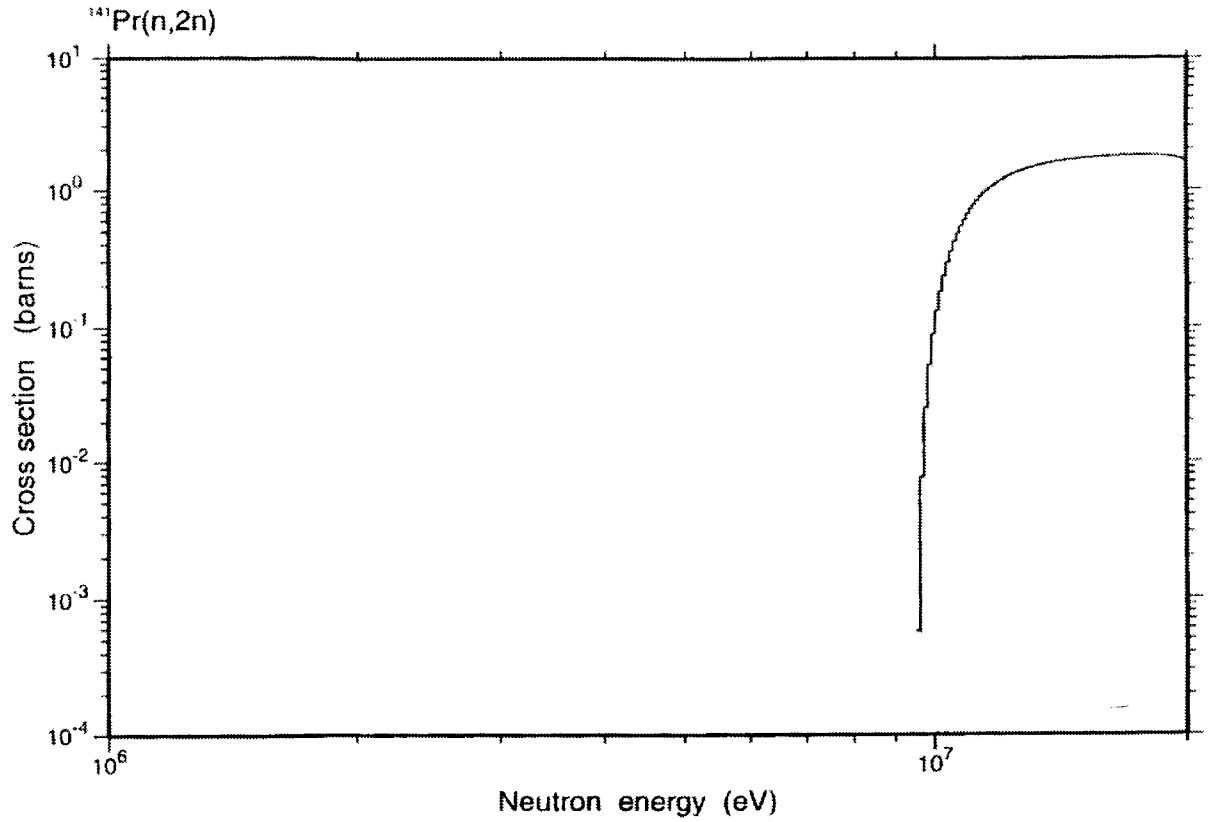

Fig. C.55. $^{141}Pr(n,2n)^{140}Pr$ (RDF-94, private communication) cross section vs energy.

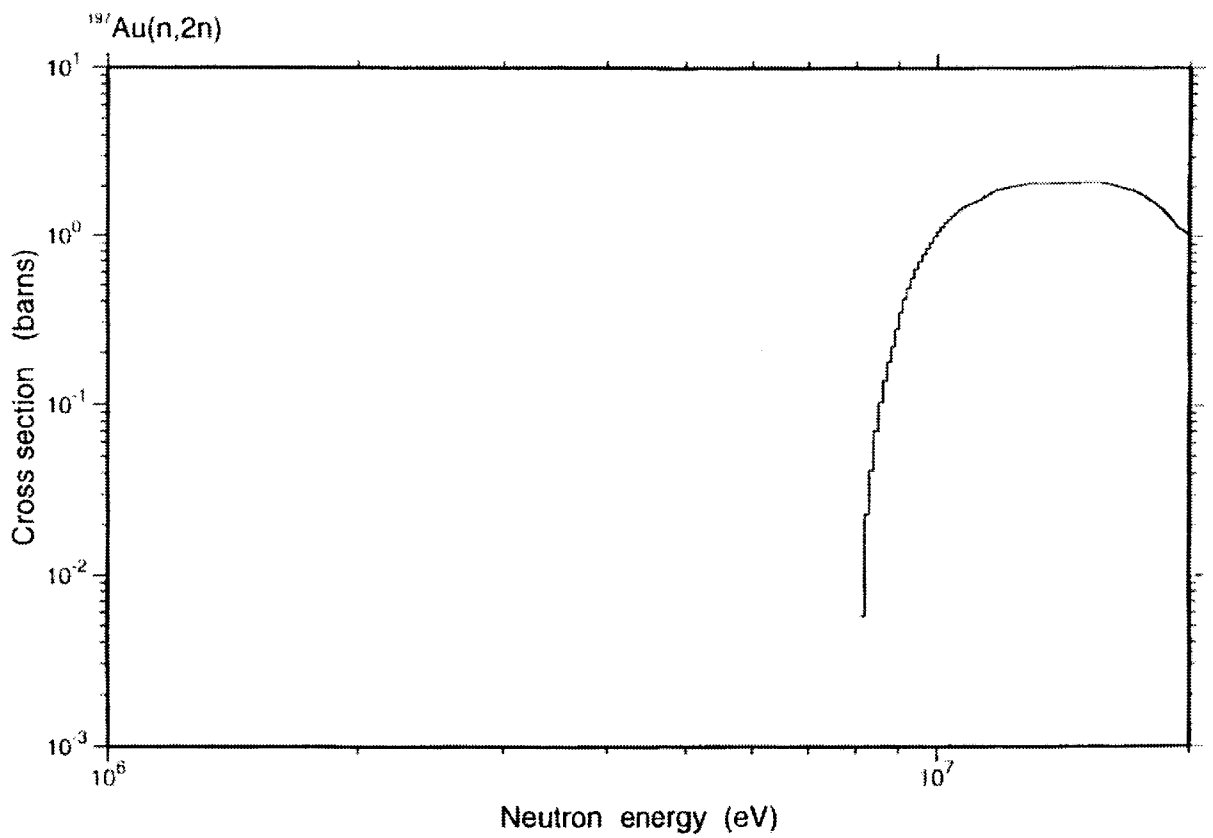

Fig. C.56. $^{197}Au(n,2n)^{196}Au$ (IRDF-90/G-V2, original evaluation) cross section vs energy.

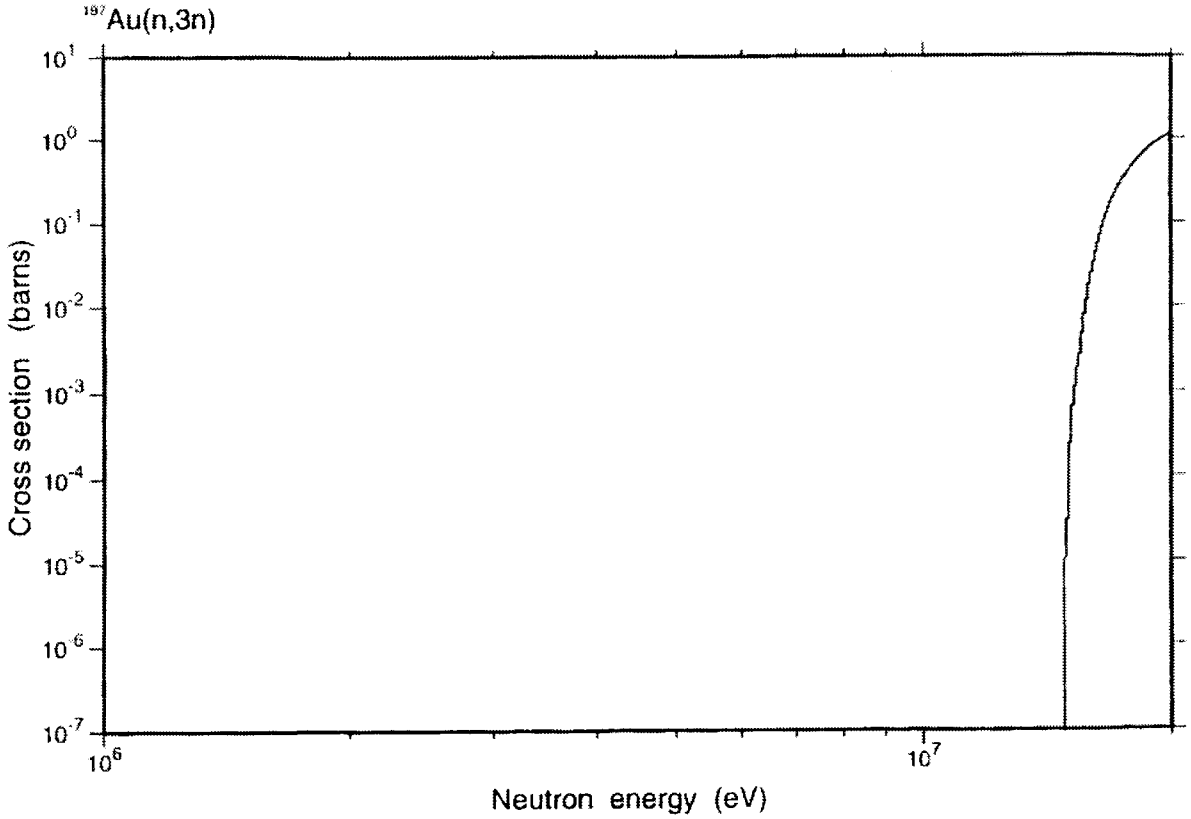

Fig. C.57. ^{197}Au(n,3n)^{195}Au (SNLRML-V1, ENDF/B-VI), cross section vs energy.

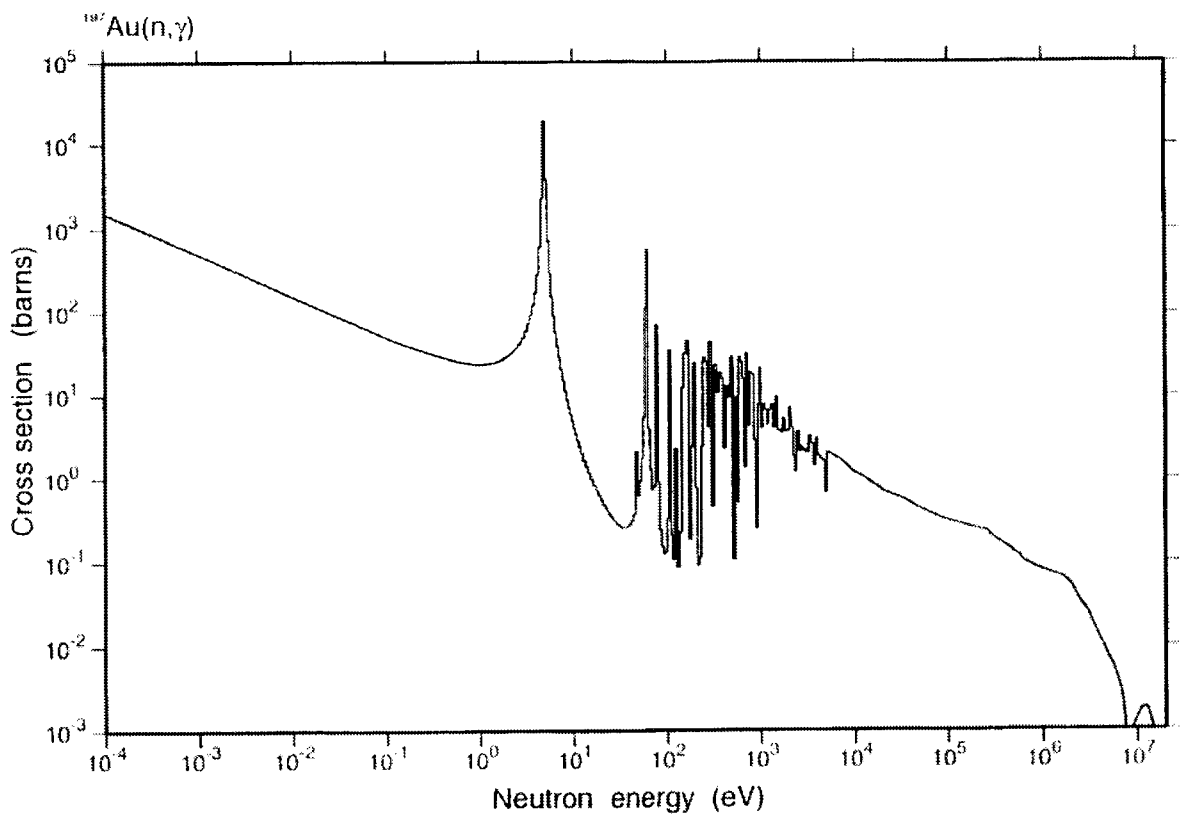

Fig. C.58. ^{197}Au(n,γ)^{198}Au (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

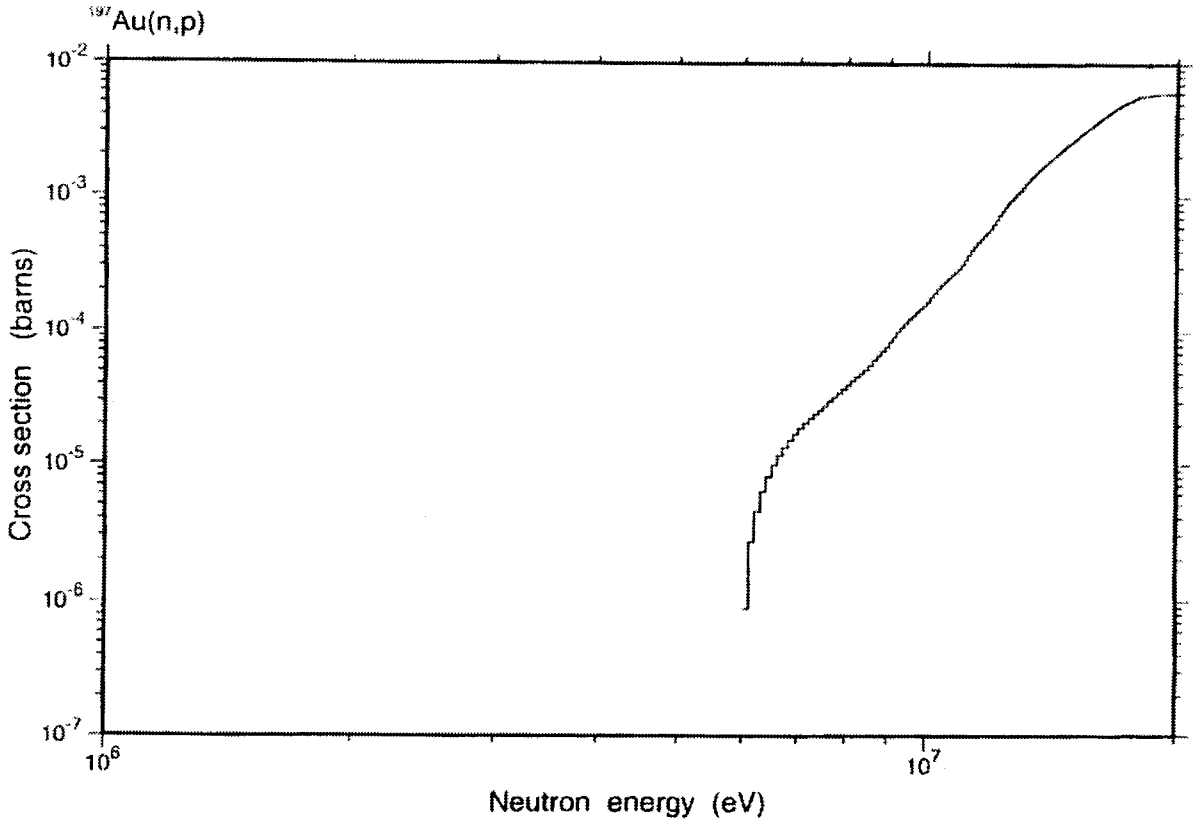

Fig. C.59. $^{197}Au(n,p)^{197}Pt$ (SNLRML-V1, ENDF/B-VI), cross section vs energy.

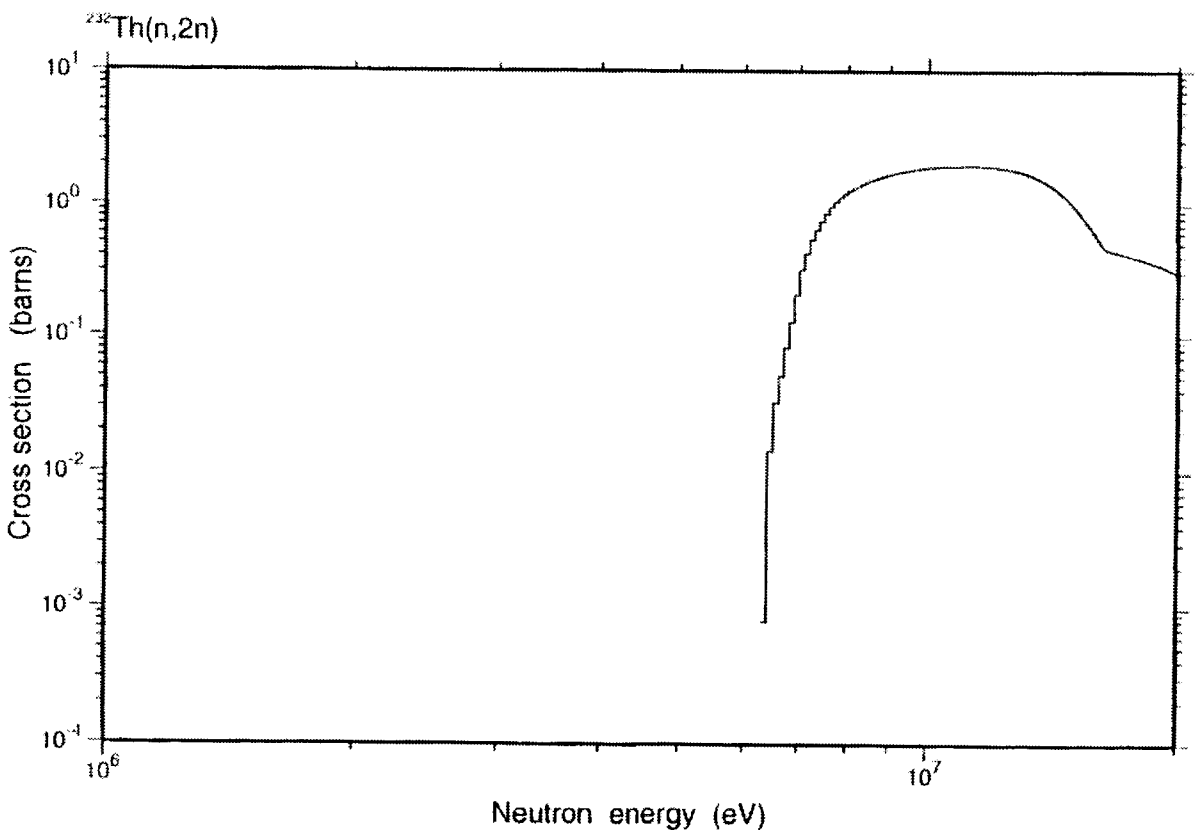

Fig. C.60. $^{233}Th(n,2n)^{231}Th$ (SNLRML-V1, ENDF/B-VI), cross section vs energy.

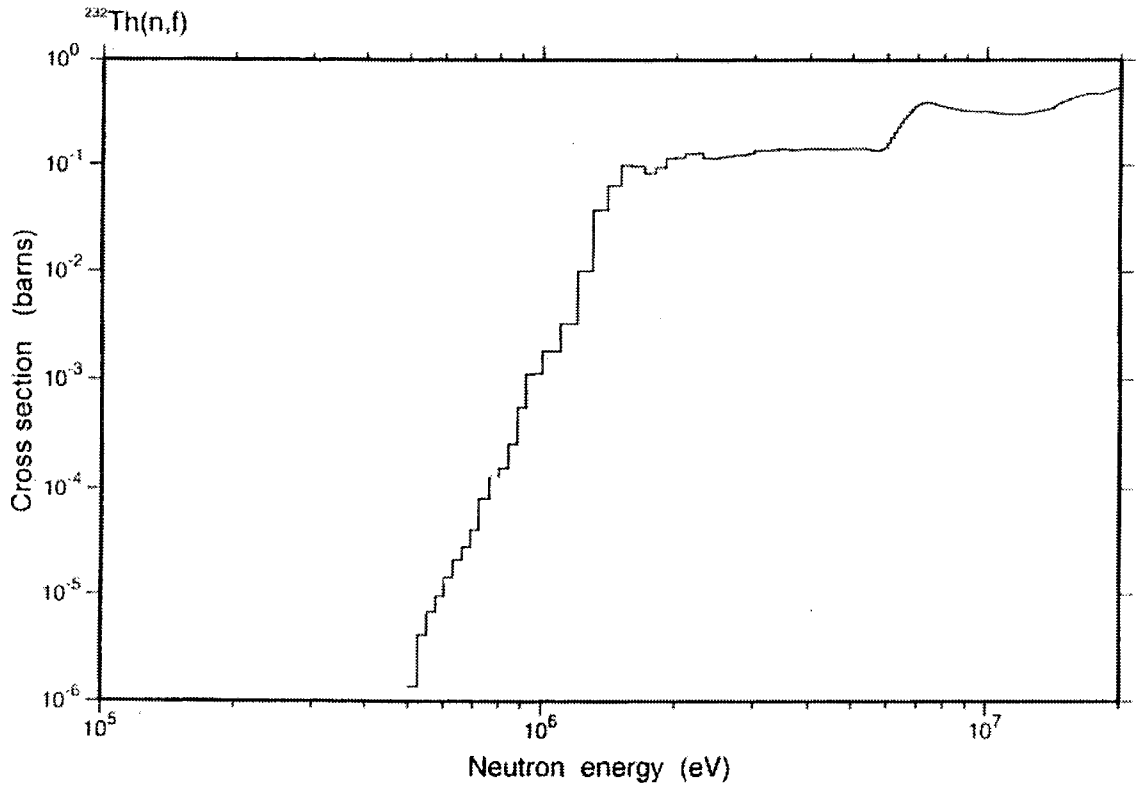

Fig. C.61. ^{232}Th(n,f) (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

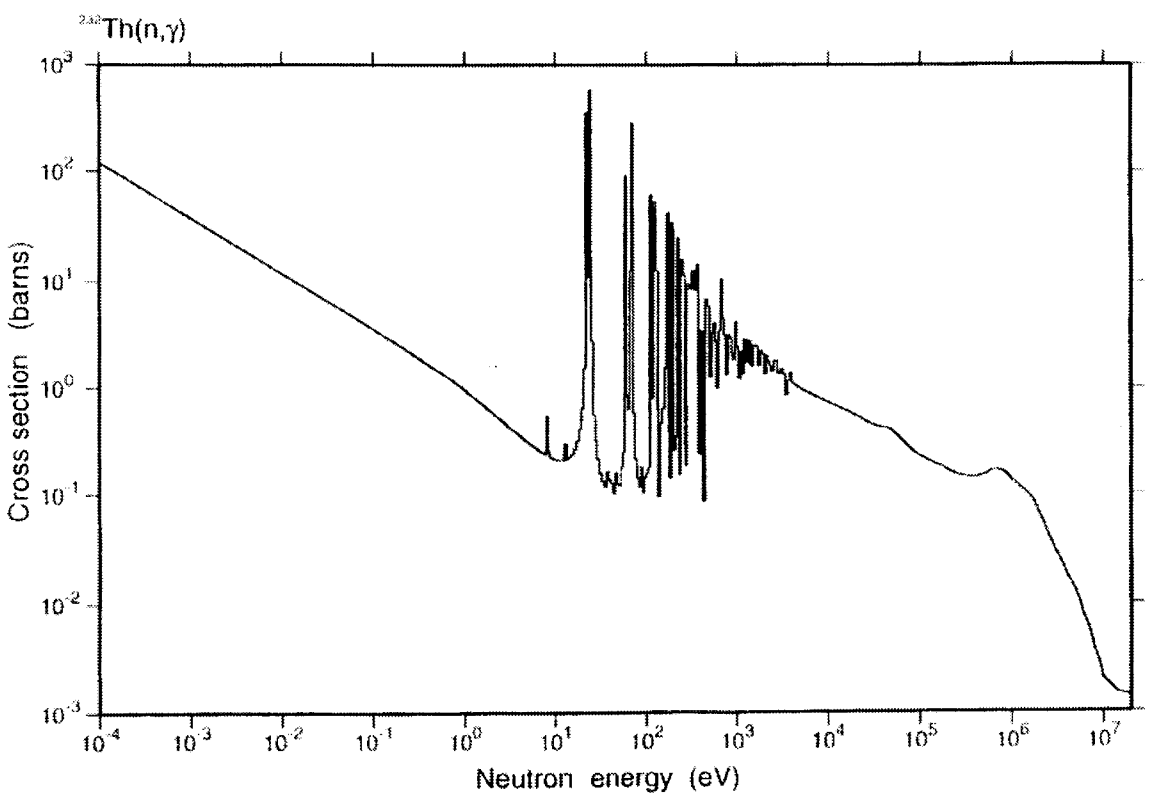

Fig. C.62. ^{232}Th(n,γ)^{233}Th (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

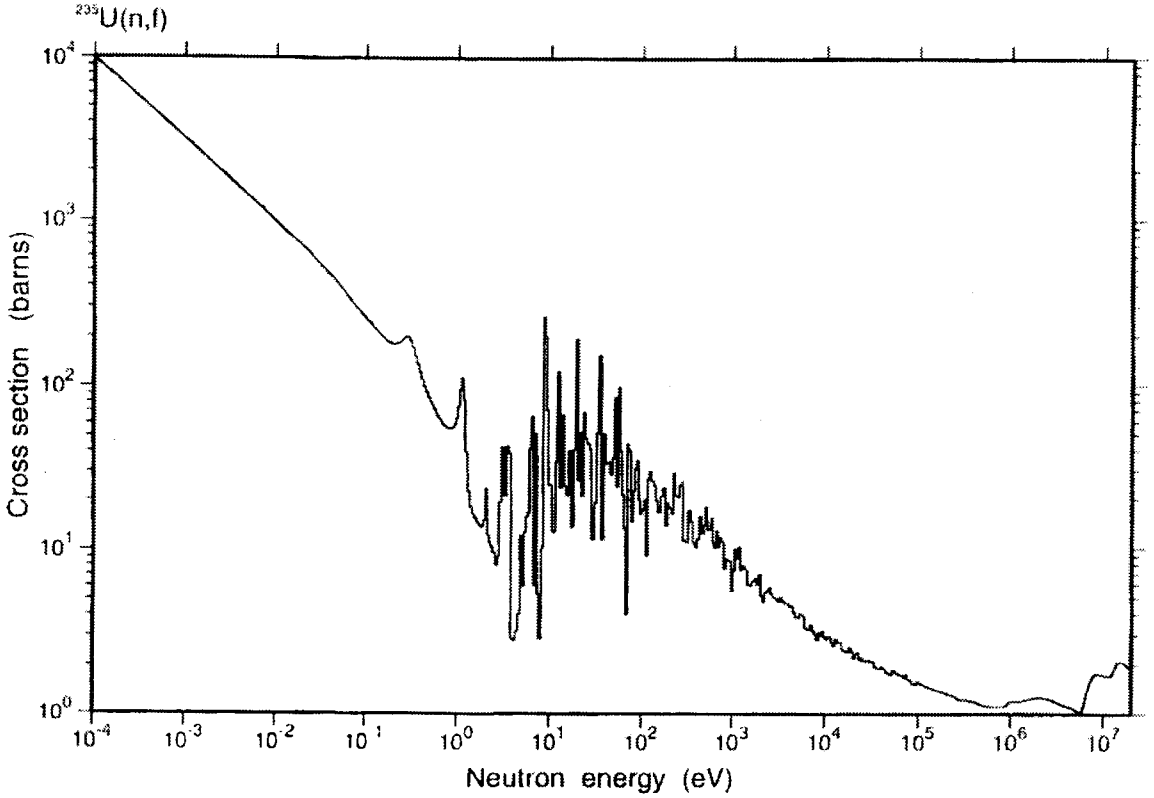

Fig. C.63. $^{235}U(n,f)$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

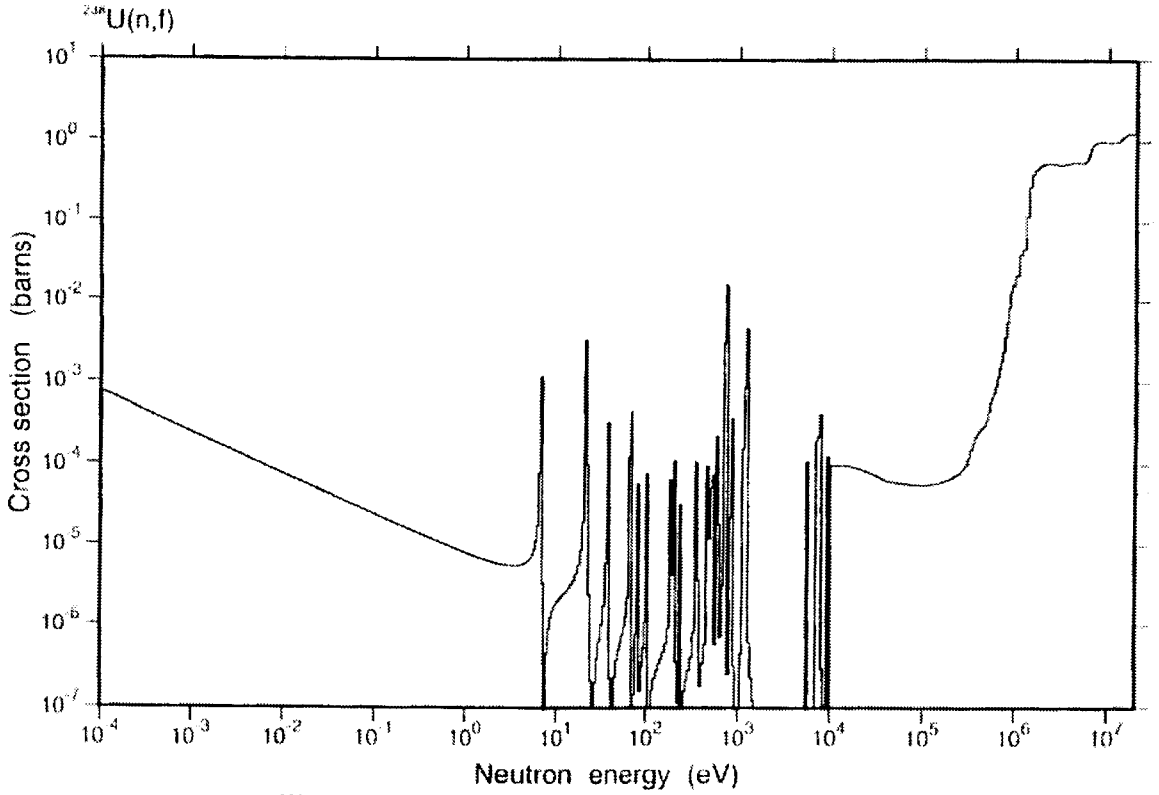

Fig. C.64. $^{238}U(n,f)$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

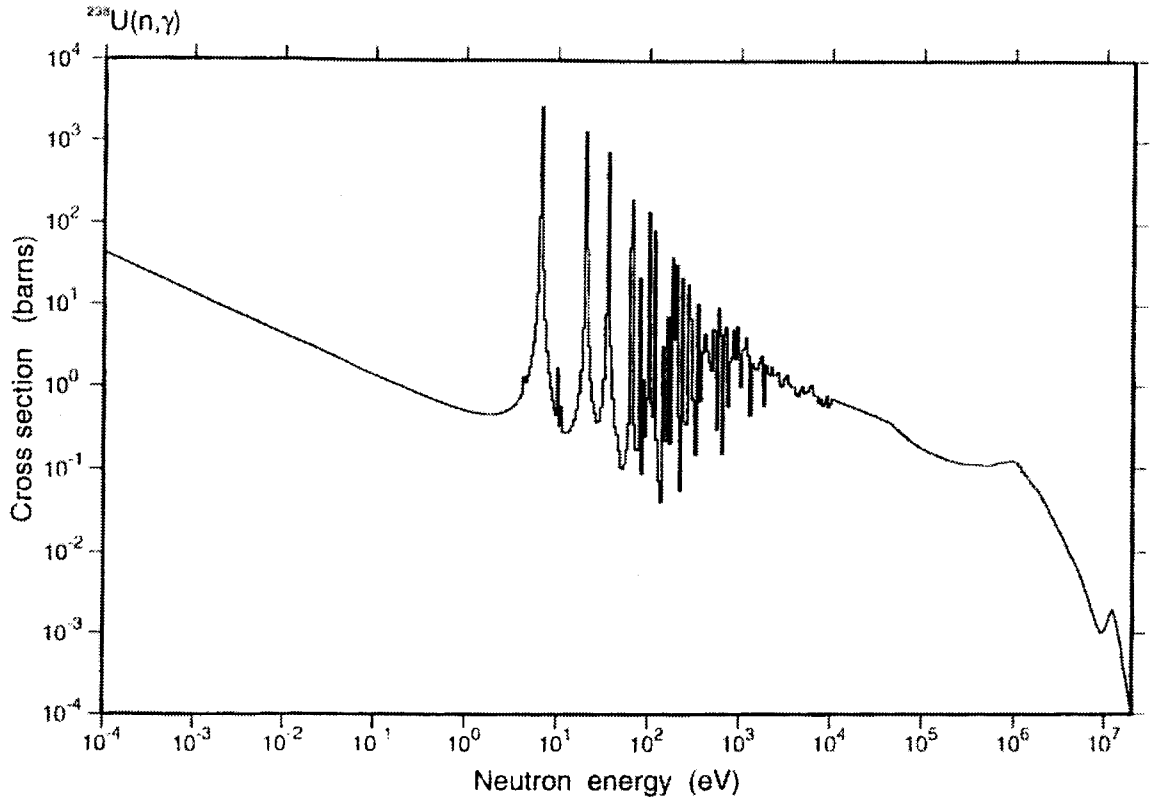

Fig. C.65. $^{238}U(n,\gamma)^{239}U$ (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

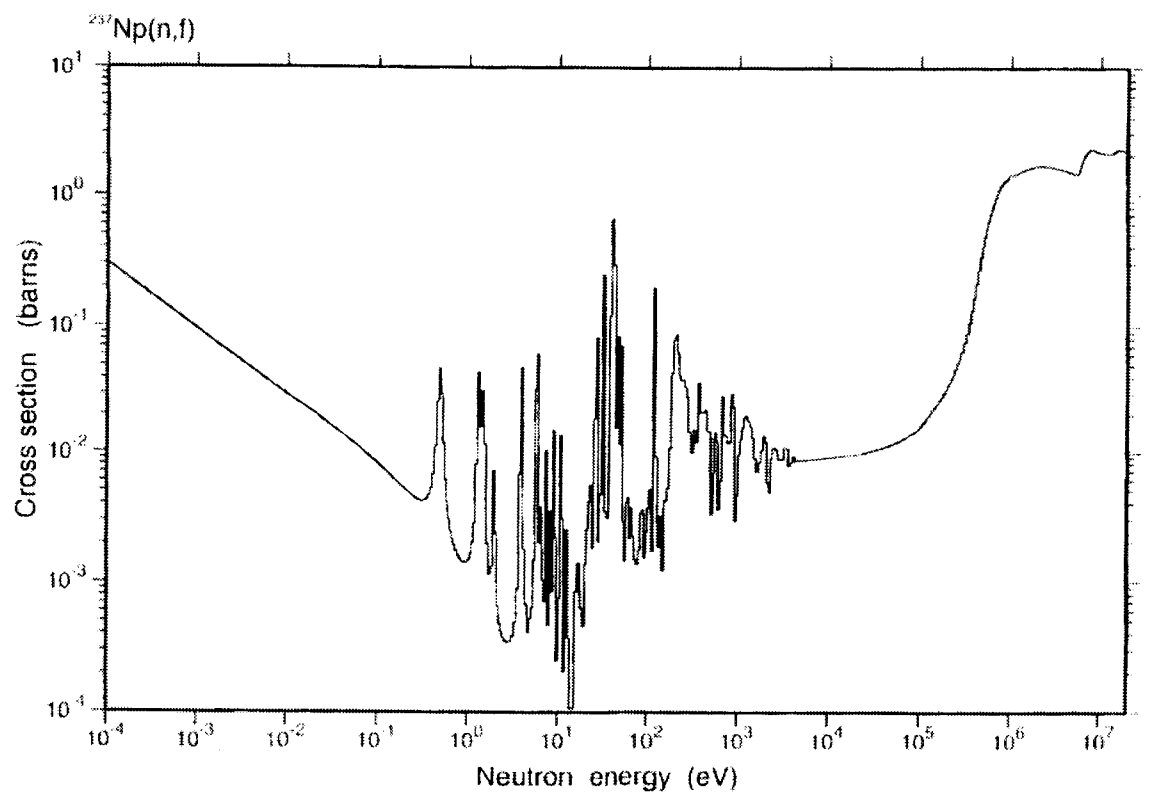

Fig. C.66. $^{237}Np(n,f)$ (SNLRML-V1, ENDF/B-VI) cross section vs energy.

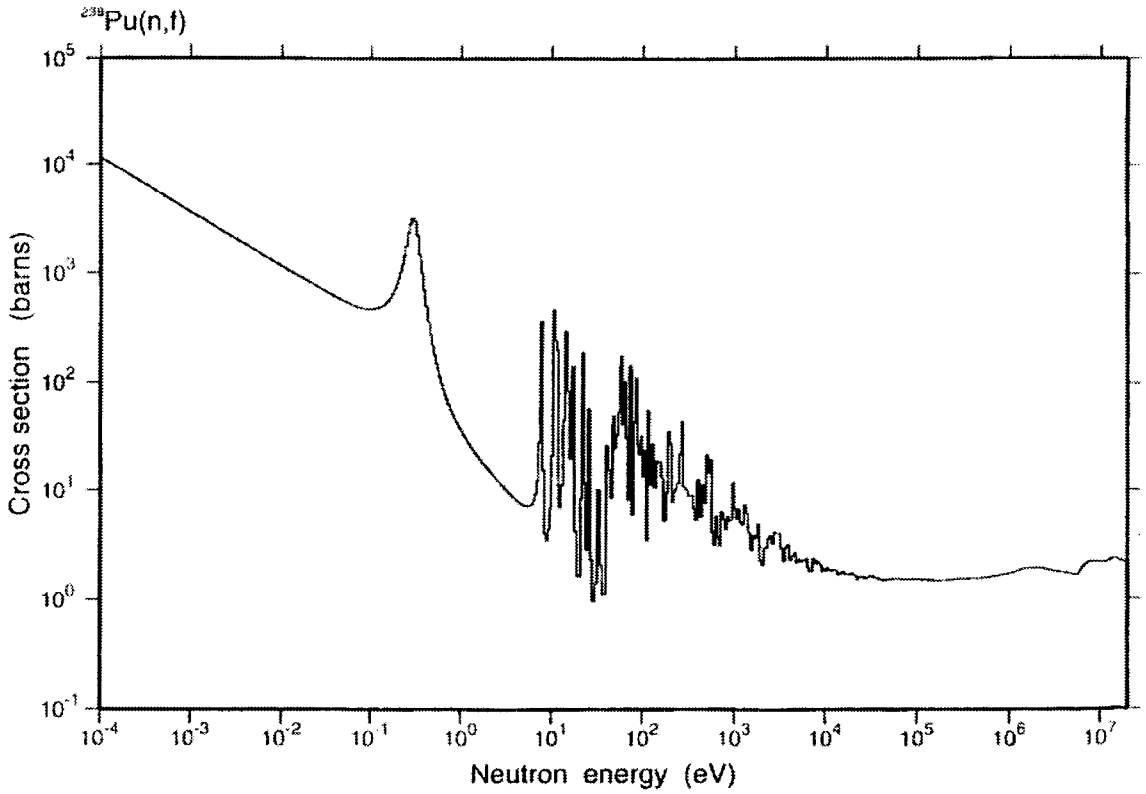

Fig. C.67. ^{239}Pu(*n,f*) (IRDF-90/G-V2, ENDF/B-VI) cross section vs energy.

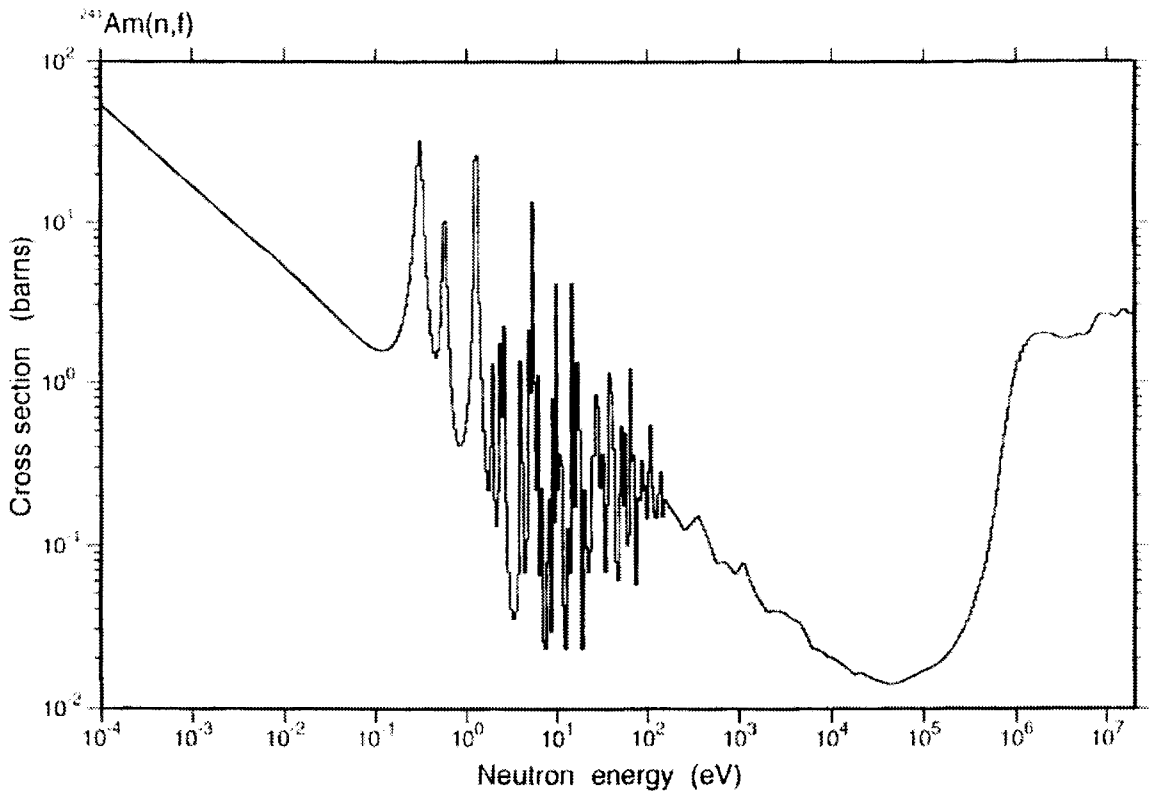

Fig. C.68. ^{241}Am(*n,f*) (SNLRML-V1, ENDF/B-VI) cross section vs energy.

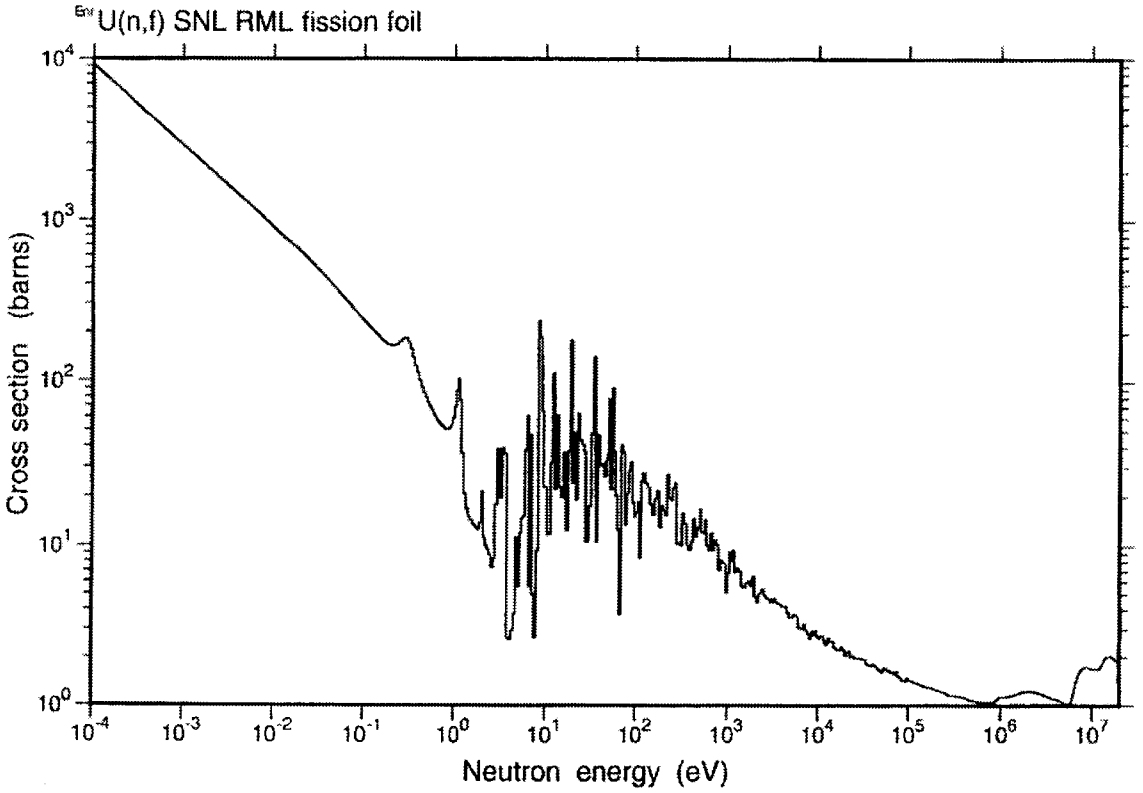

Fig. C.69. Enriched uranium fission foil (SNLRML-V1, ENDF/B-VI) cross section vs energy. (Atom fractions 0.9300, 0.00981, 0.00359, and 0.0566 for ^{235}U, ^{234}U, ^{236}U, and ^{238}U, respectively.)

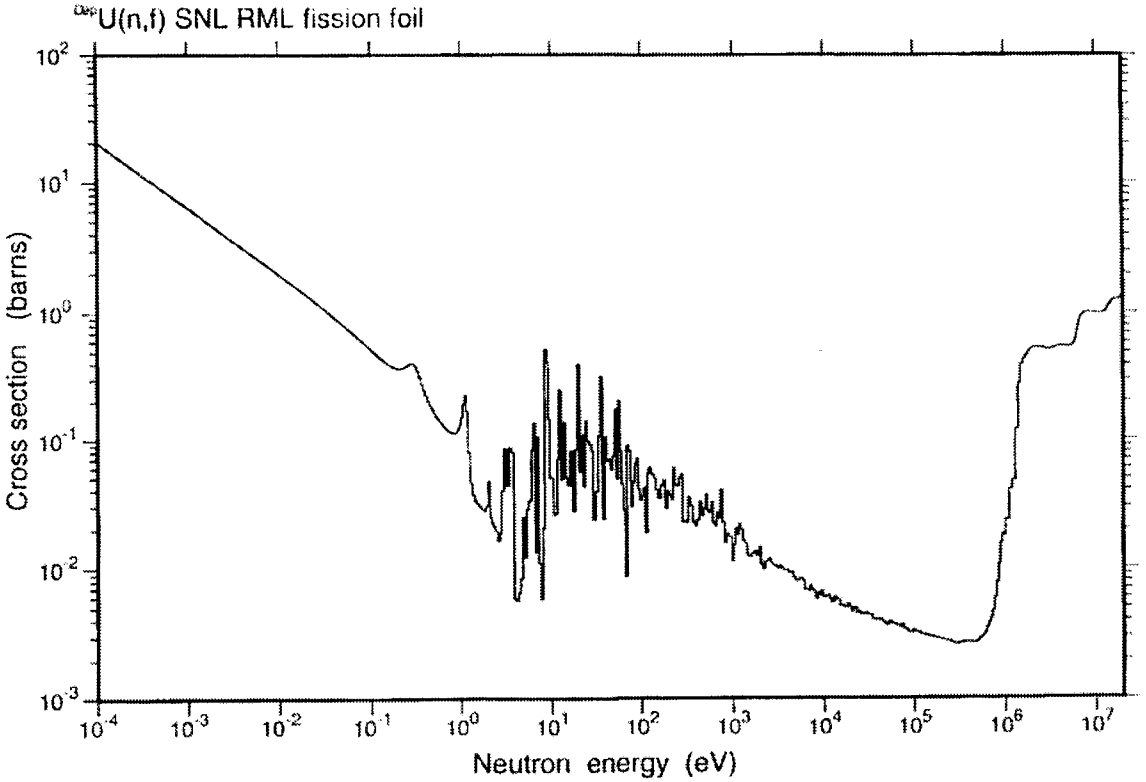

Fig. C.70. Depleted uranium fission foil (SNLRML-V1, ENDF/B-VI) cross section vs energy. (Atom fractions 0.9979, 0.00001, 0.00205, and 0.00004 for ^{238}U, ^{234}U, ^{235}U, and ^{236}U, respectively.)

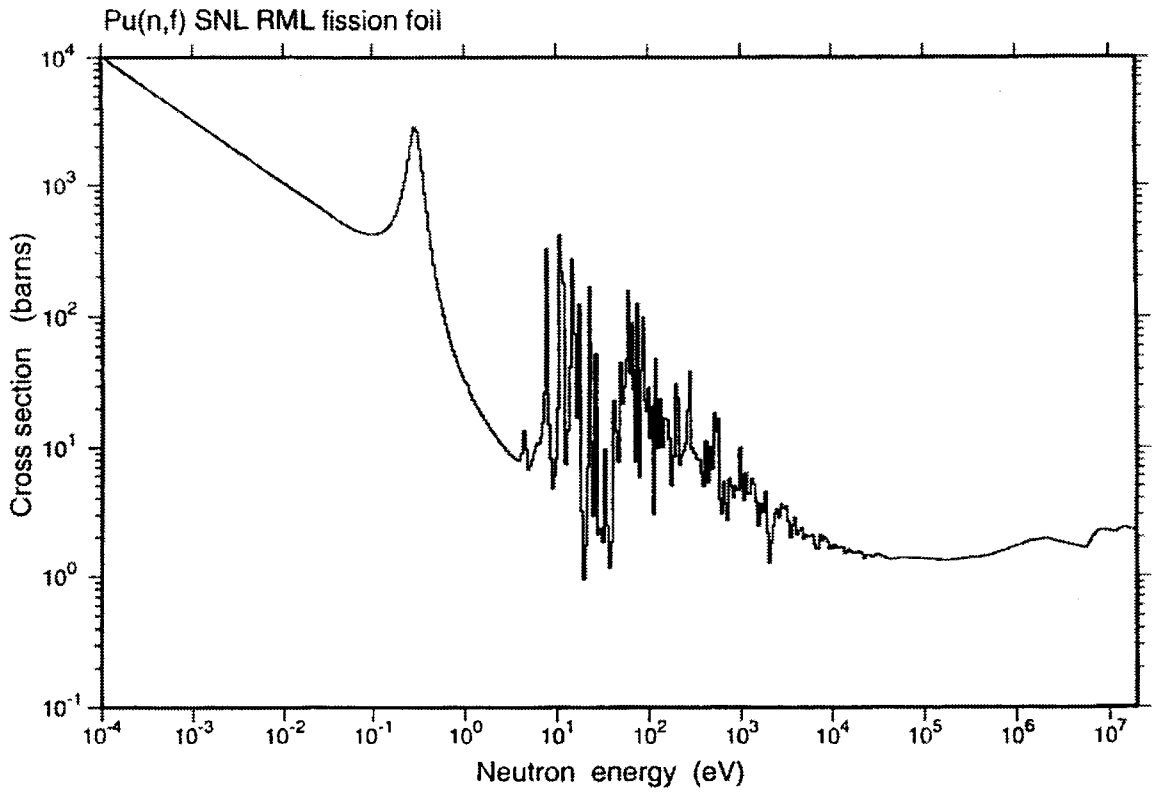

Fig. C.71. Plutonium fission foil (SNLRML-V1, ENDF/B-VI) cross section vs energy. (Atom fractions 0.869965, 0.0006798, 0.115968688, 0.010797, 0.00235936, 0.000199946, and 0.00002999 for ^{239}Pu, ^{238}Pu, ^{240}Pu, ^{241}Pu, ^{242}Pu, ^{235}U, and ^{237}Np, respectively.)

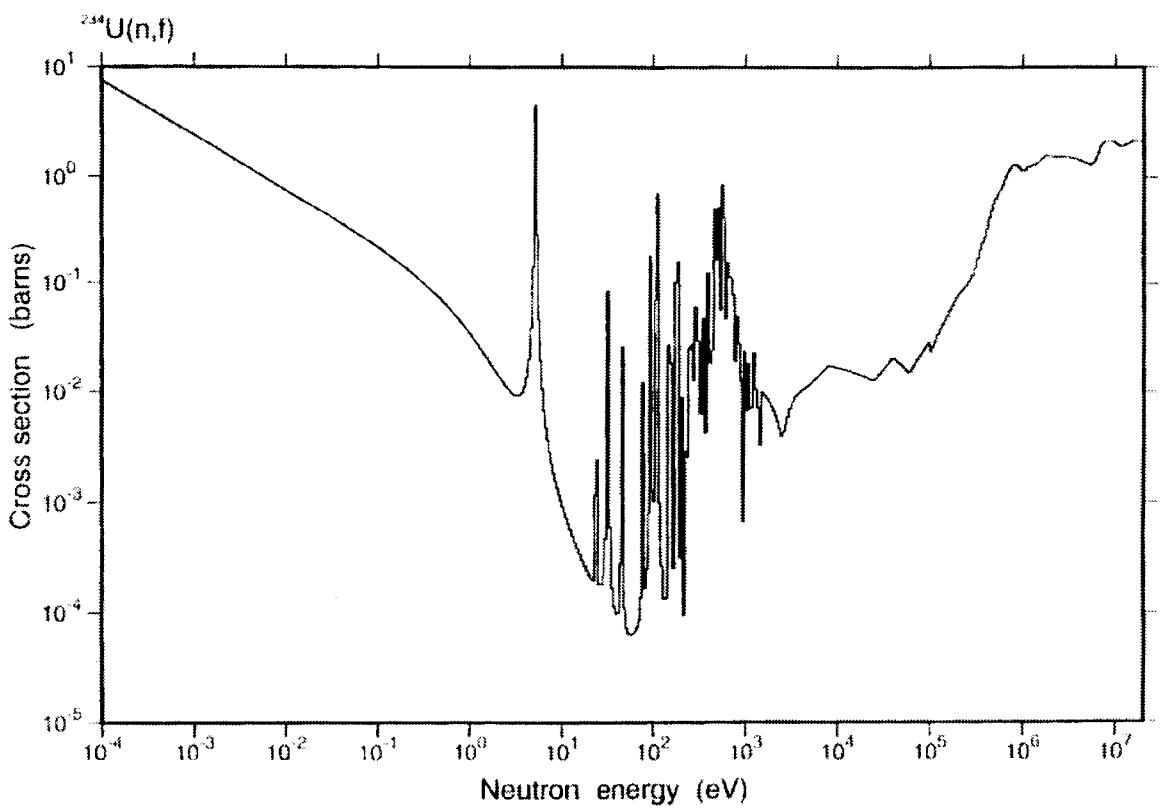

Fig. C.72. ^{234}U(*n,f*) (SNLRML-V1, ENDF/B-VI) cross section vs energy.

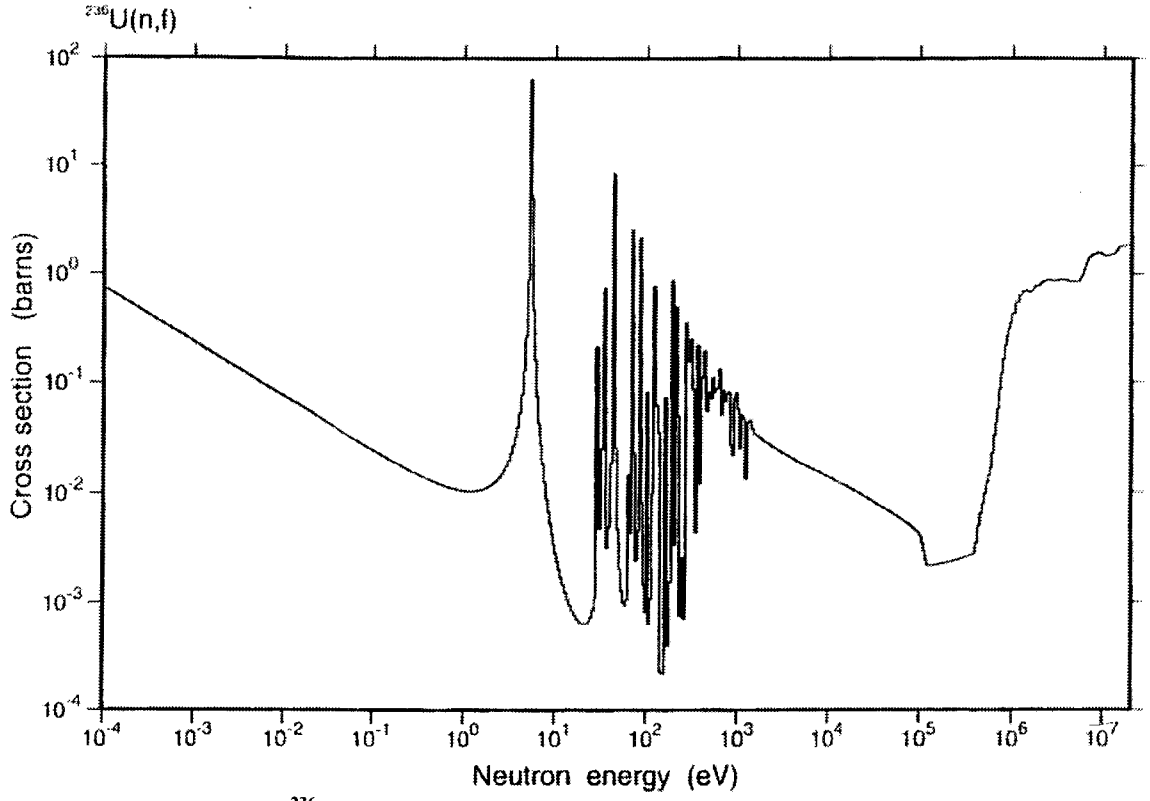

Fig. C.73. $^{236}U(n,f)$ (SNLRML-V1, ENDF/B-VI) cross section vs energy.

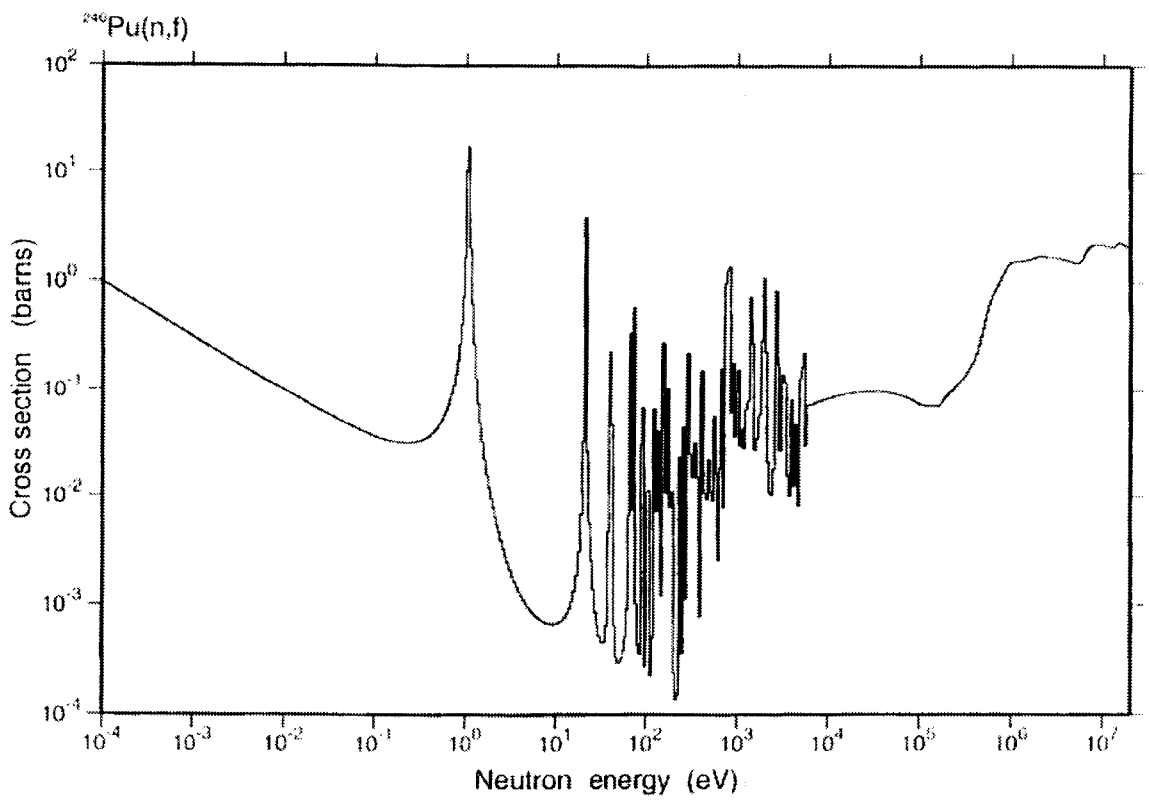

Fig. C.74. $^{240}Pu(n,f)$ (SNLRML-V1, ENDF/B-VI) cross section vs energy.

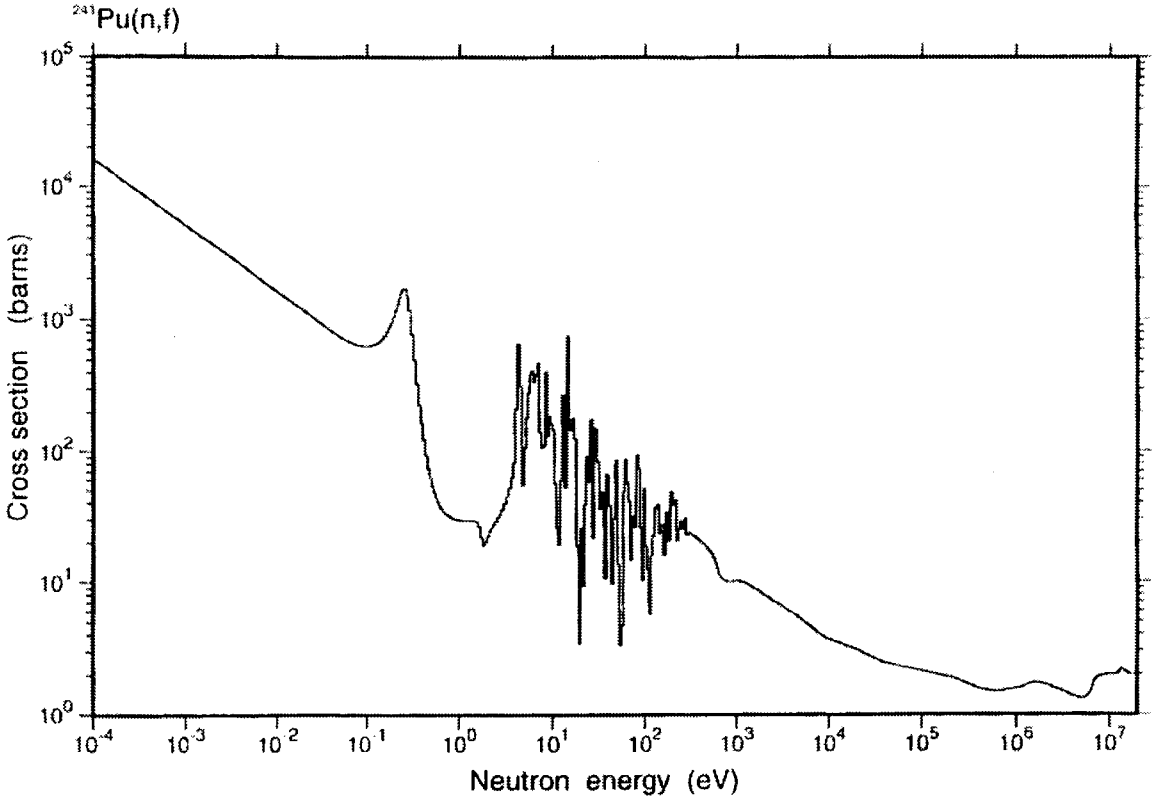

Fig. C.75. ^{241}Pu(n,f) (SNLRML-V1, ENDF/B-VI) cross section vs energy.

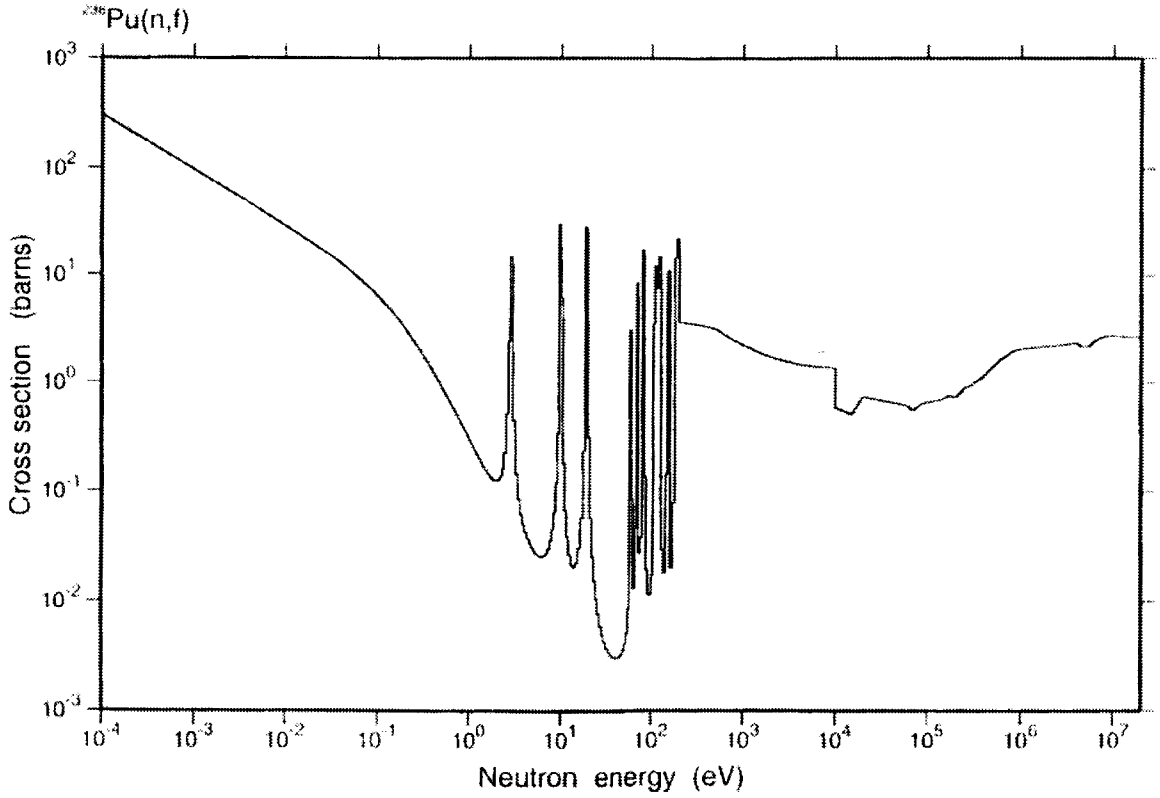

Fig. C.76. ^{238}Pu(n,f) (SNLRML-V1, ENDF/B-VI) cross section vs energy.

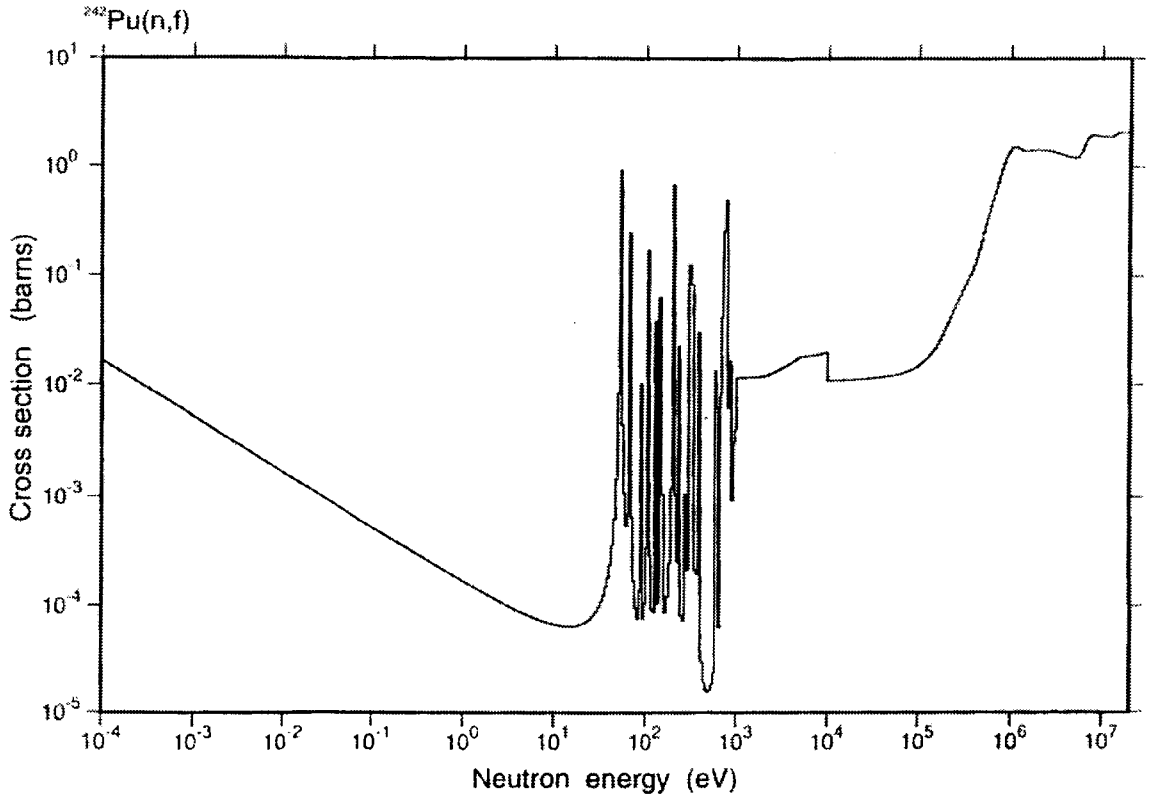

Fig. C.77. ^{242}Pu(n,f) (SNLRML-V1, ENDF/B-VI) cross section vs energy.

APPENDIX D

PLOTS OF CROSS-SECTION CORRELATION MATRICES AND RELATIVE STANDARD DEVIATIONS VS ENERGY

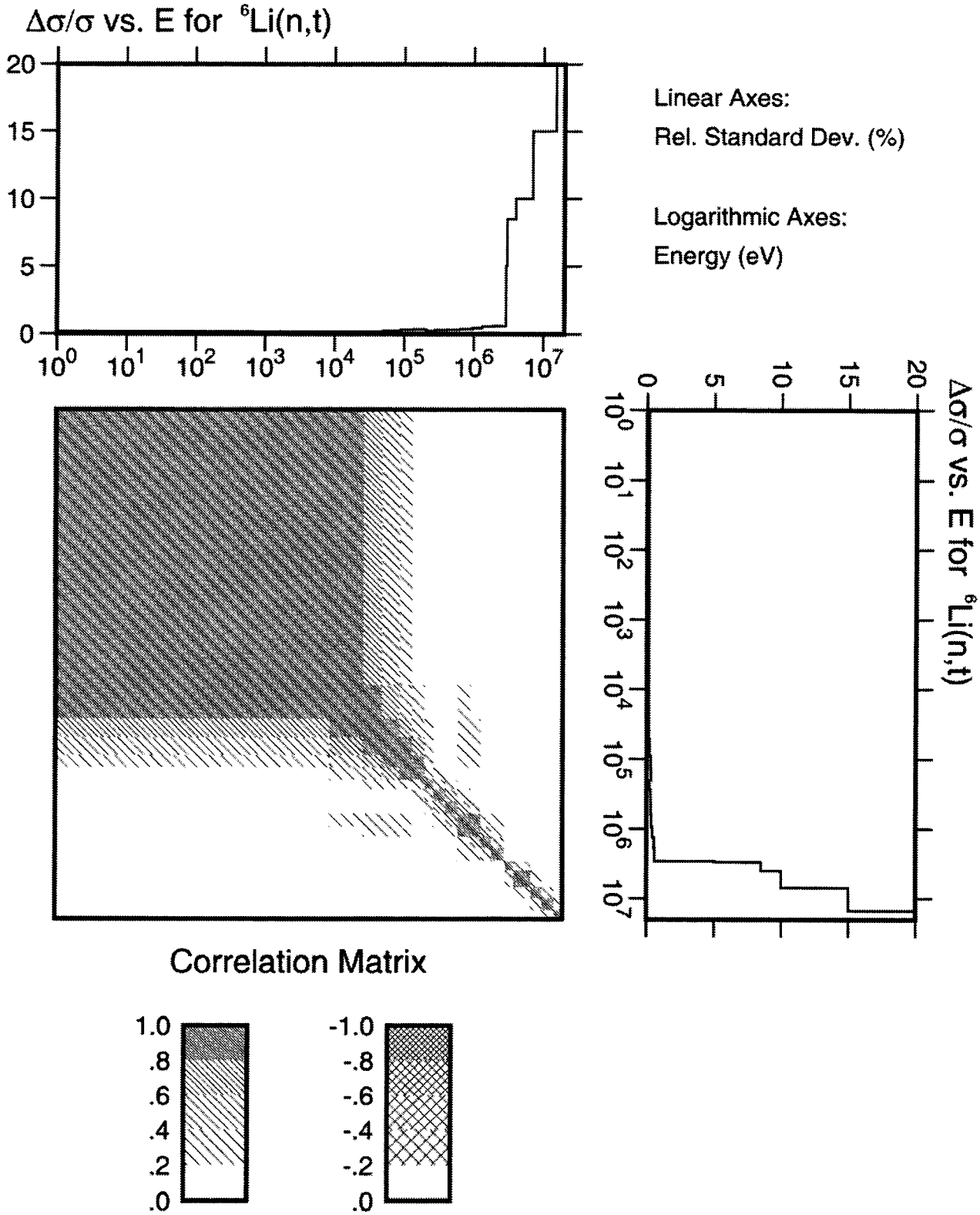

Fig. D.1. ^{6}Li(n,t)^{4}He (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

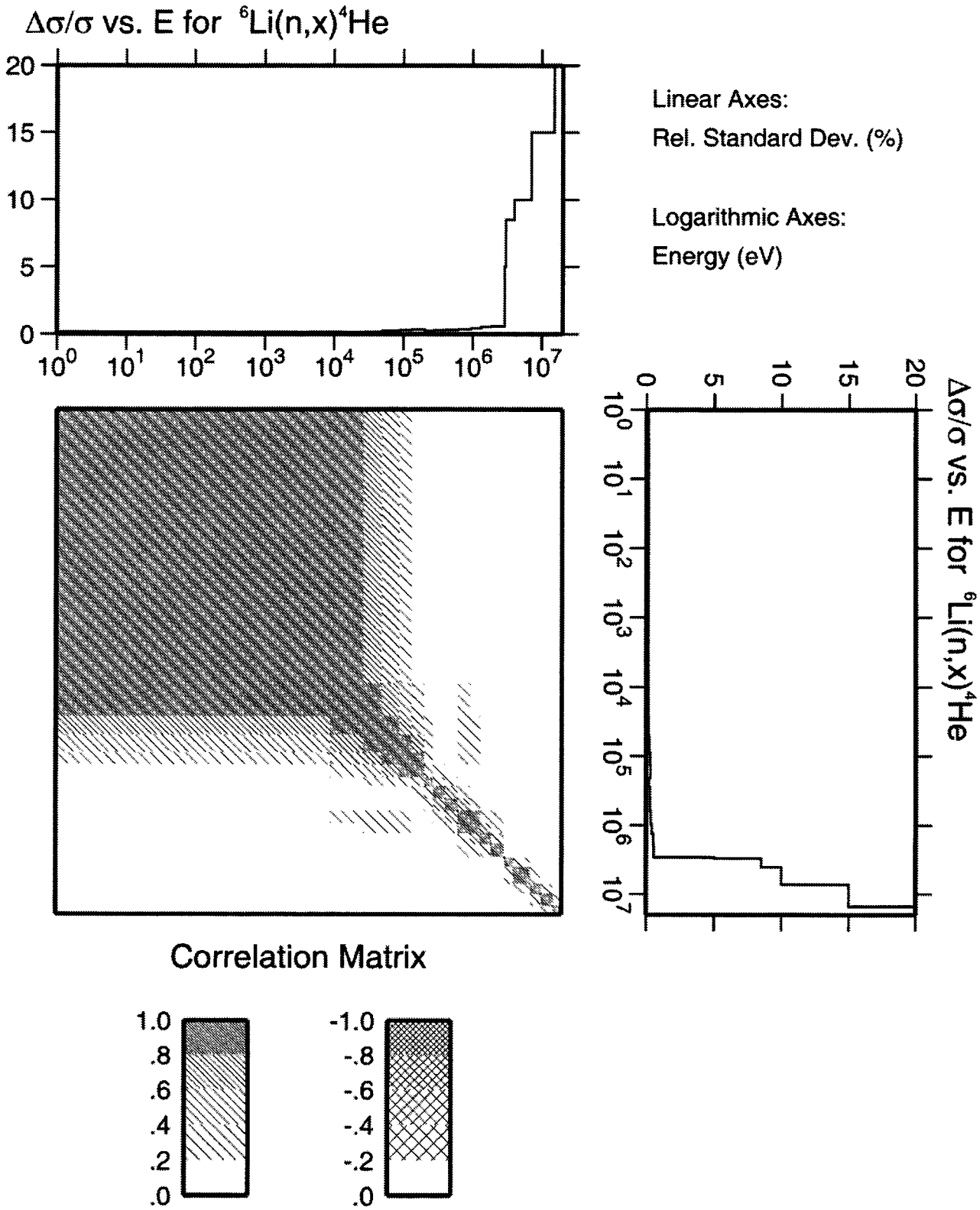

Fig. D.2. ^{6}Li(n,x)^{4}He [SNLRML-V1, IRDF-90, covariance for ^{6}Li(n,t)^{4}He] correlation matrix and standard deviation.

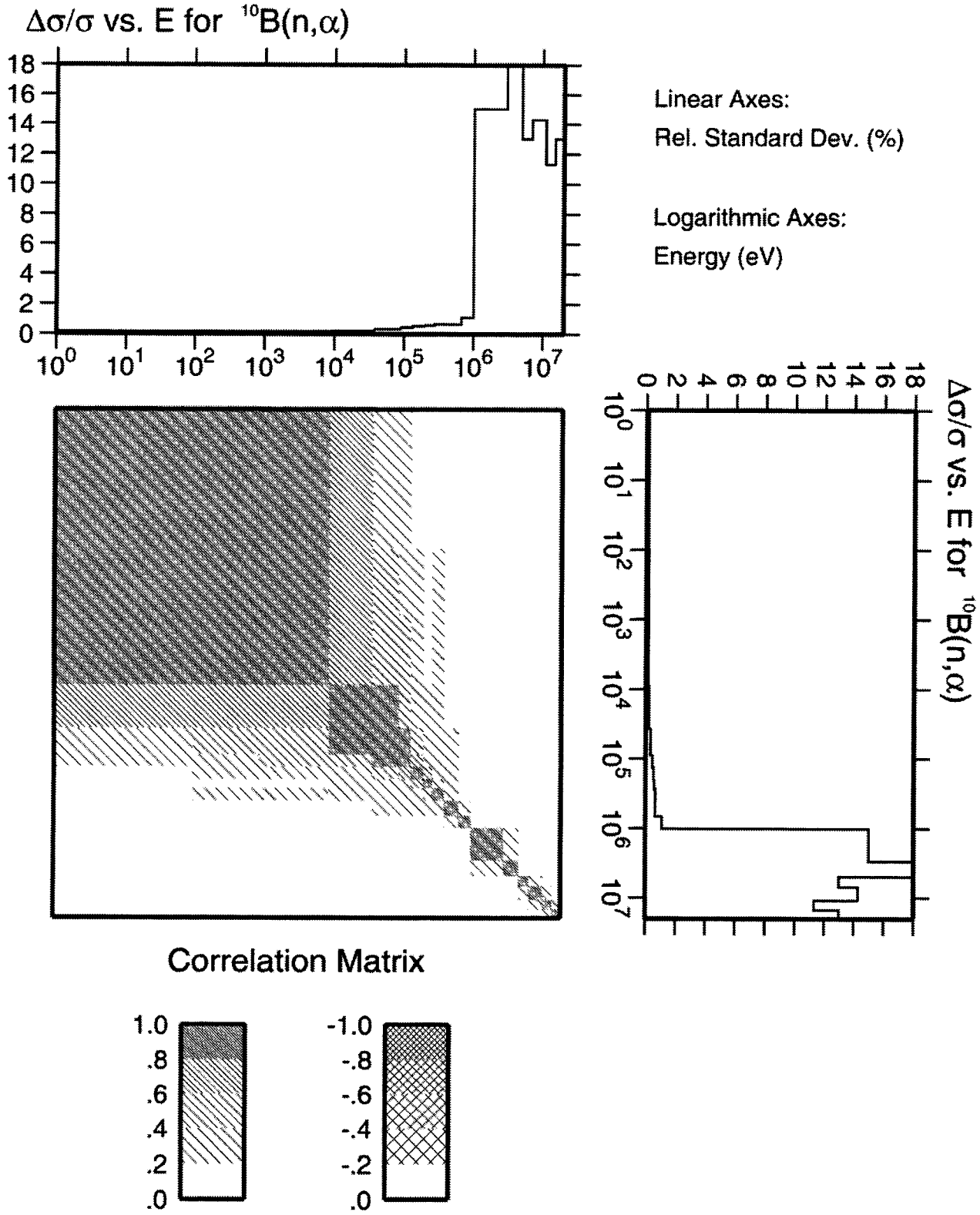

Fig. D.3. $^{10}B(n,\alpha)^{7}Li$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

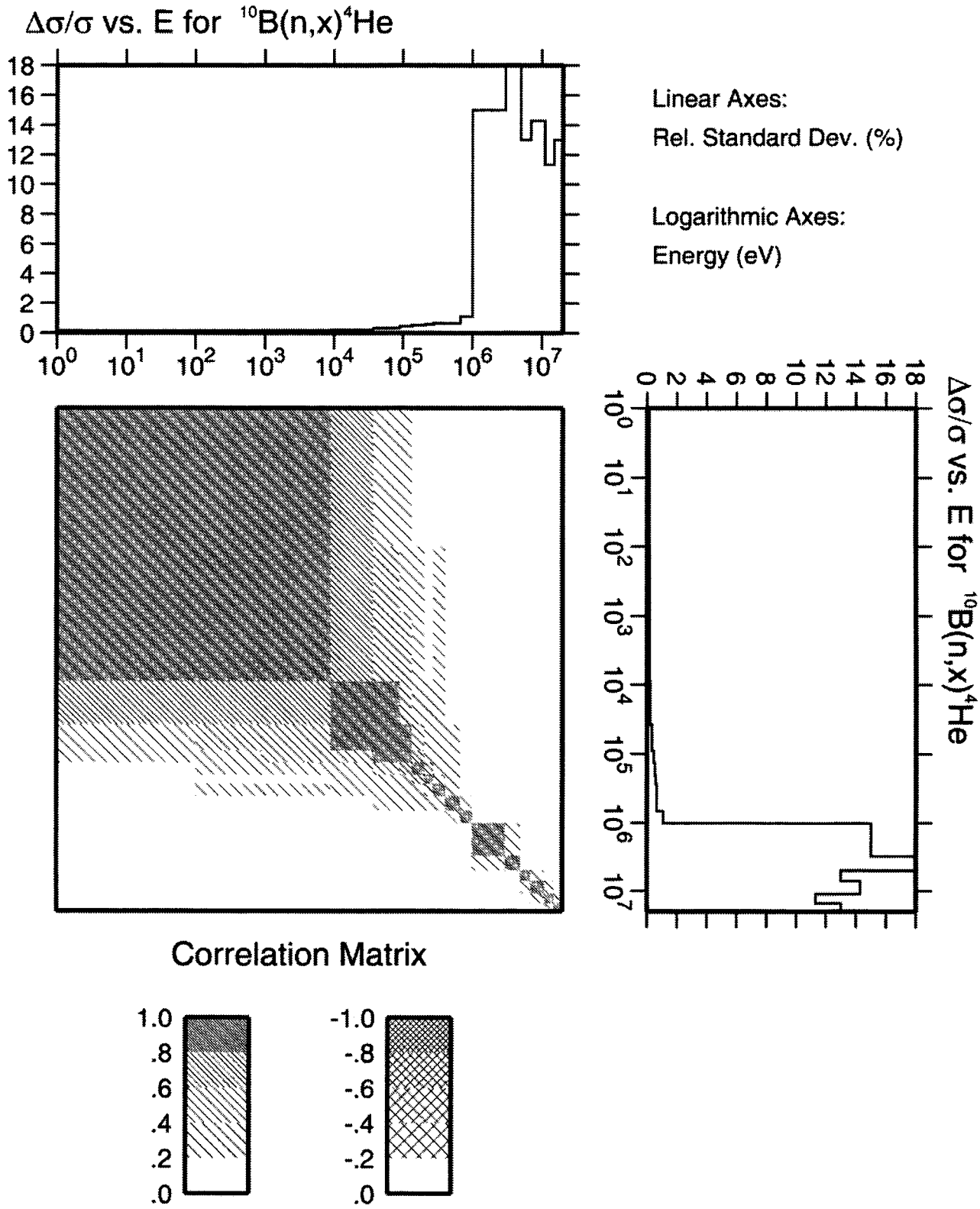

Fig. D.4. ^{10}B(n,x)^{4}He [SNLRML-V1, IRDF-90, covariance for ^{10}B(n,α)^{7}Li] correlation matrix and standard deviation.

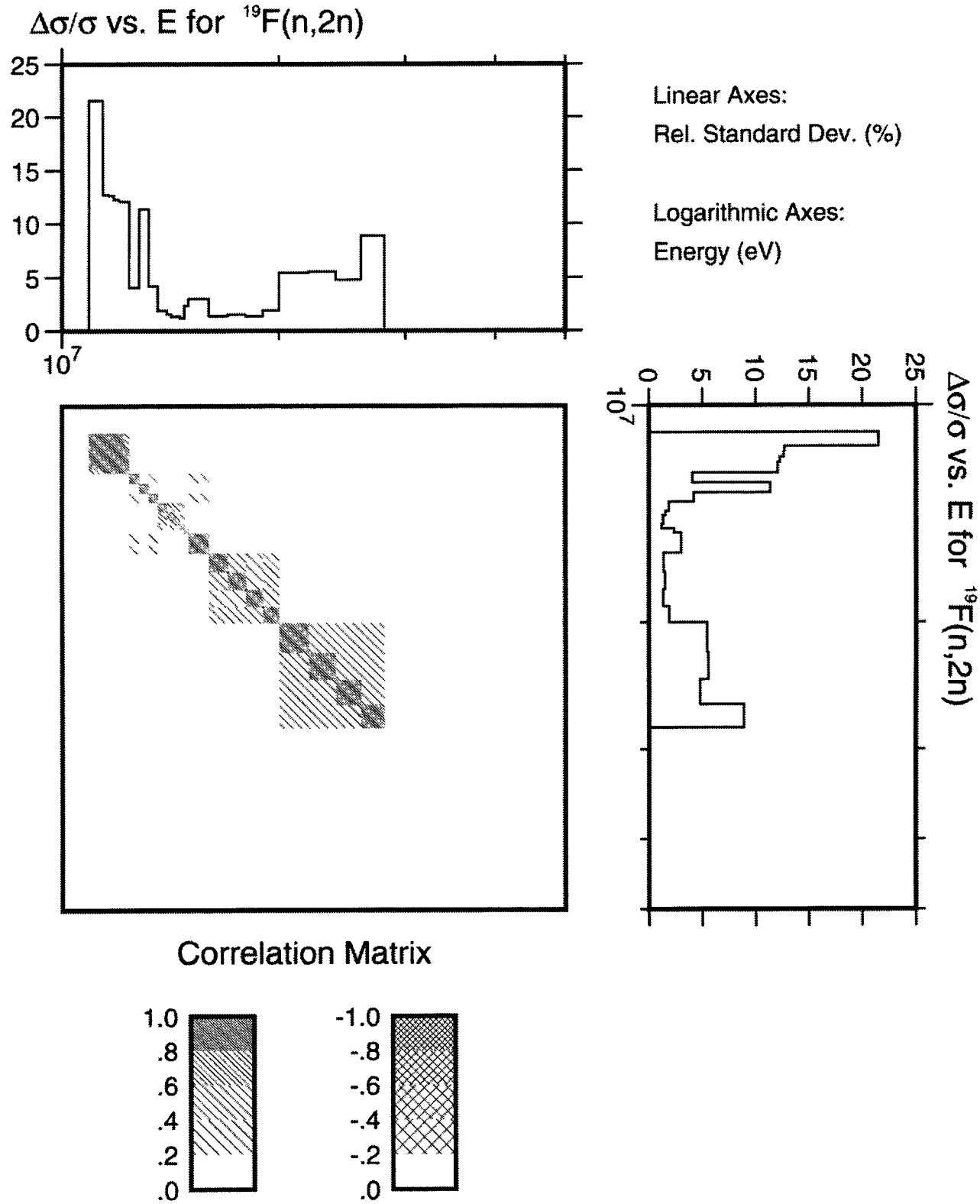

Fig. D.5. ^{19}F(n,2n)^{18}F (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

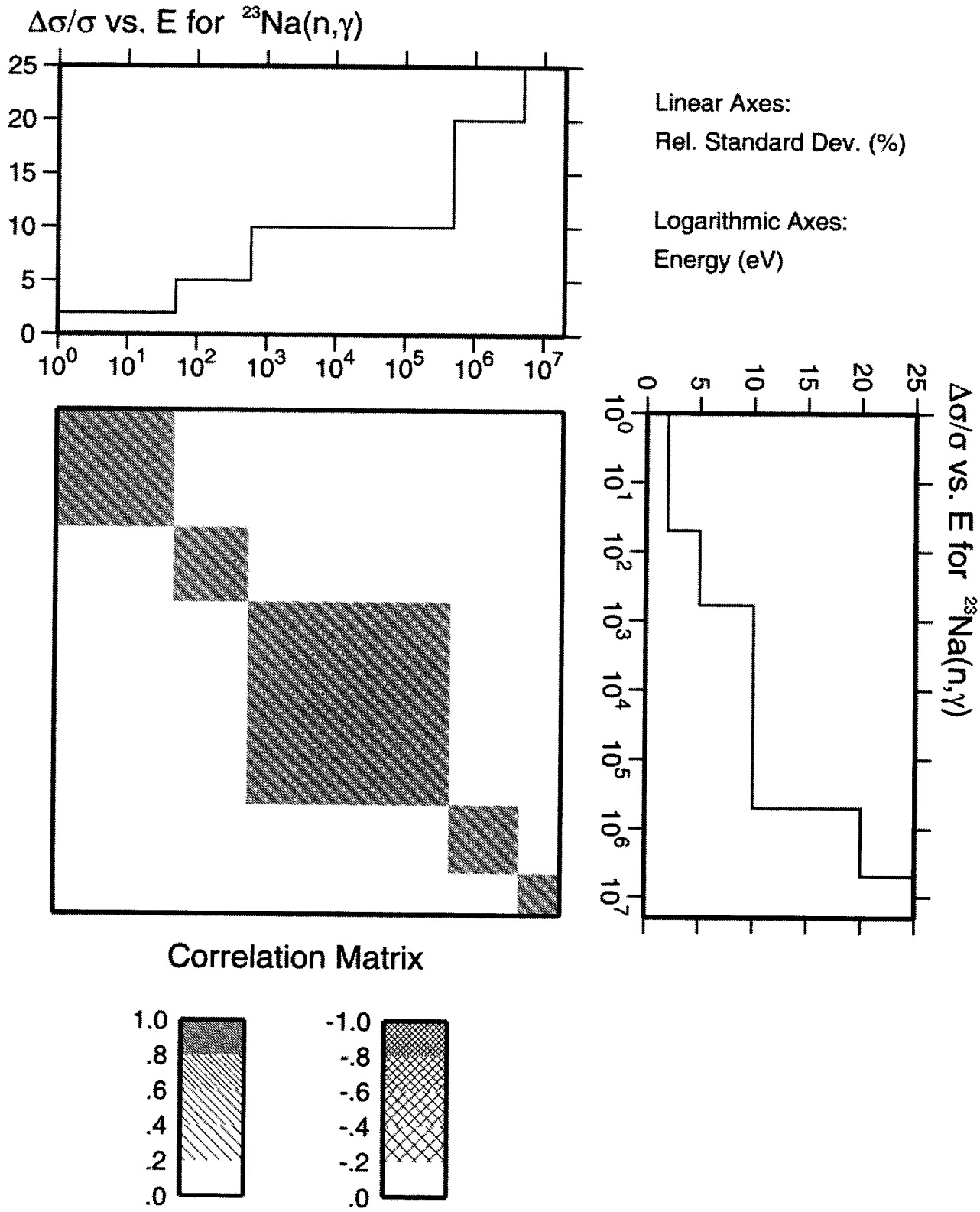

Fig. D.6. $^{23}Na(n,\gamma)^{24}Na$ (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

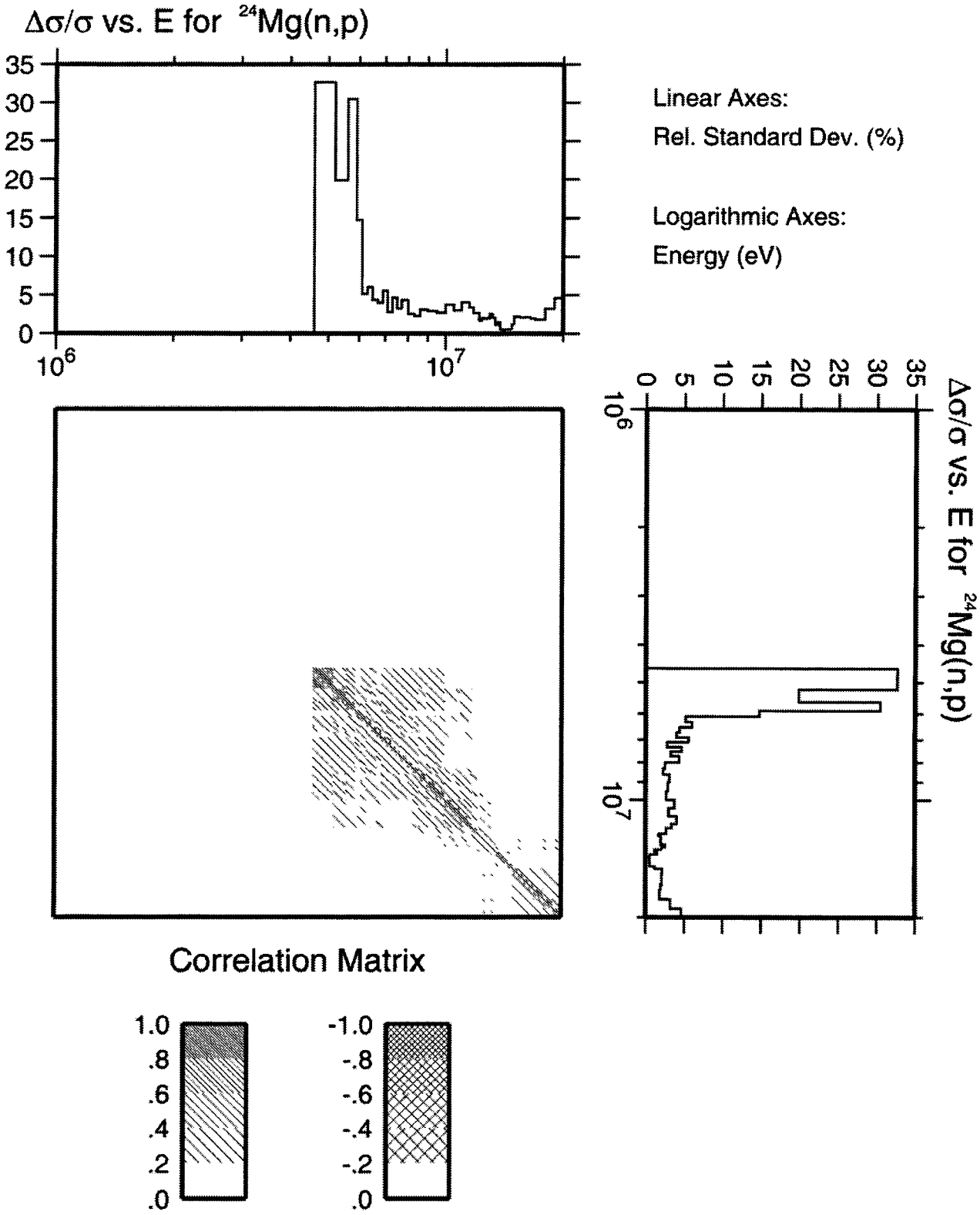

Fig. D.7. $^{24}Mg(n,p)^{24}Na$ (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

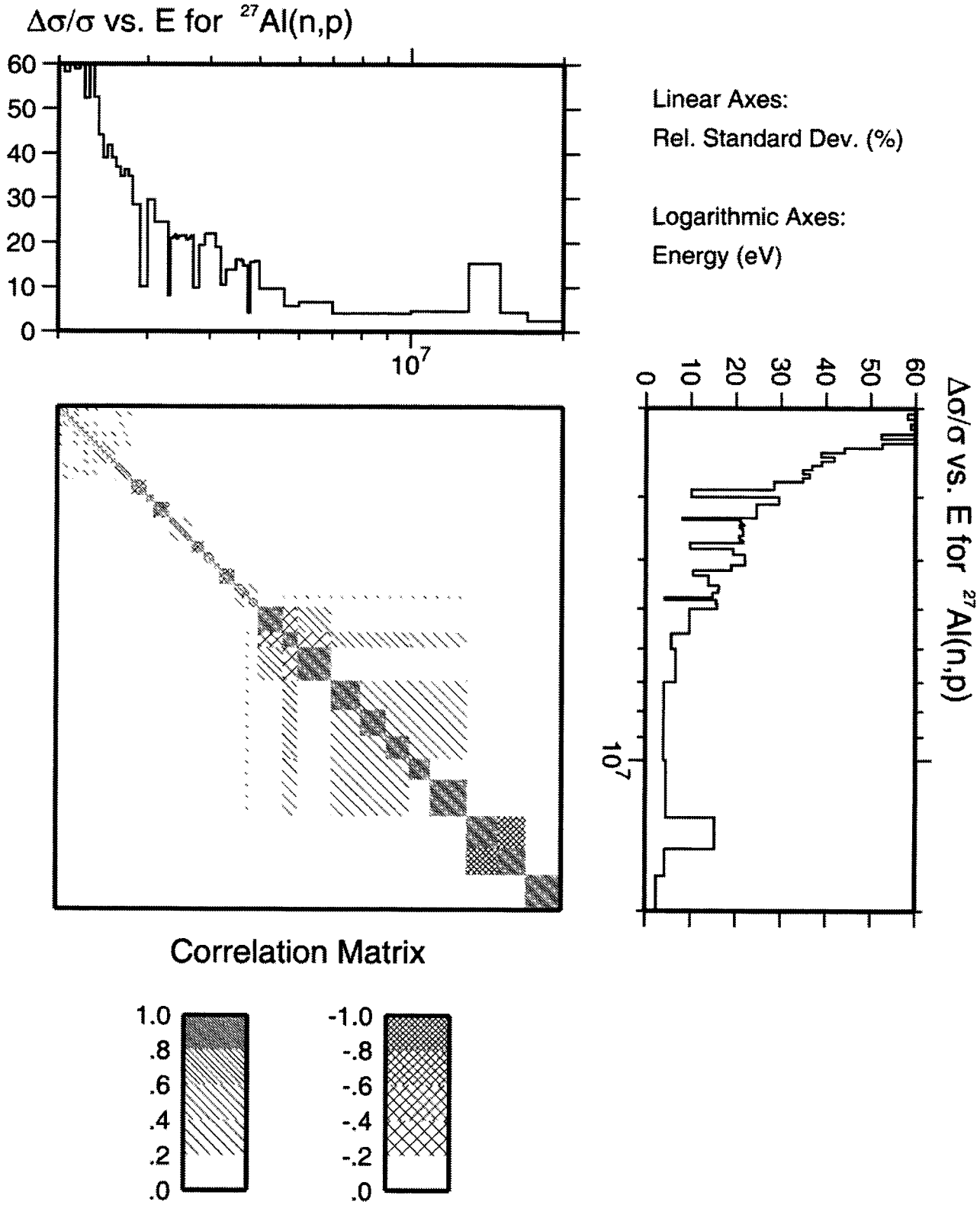

Fig. D.8. $^{27}Al(n,p)^{27}Mg$ (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

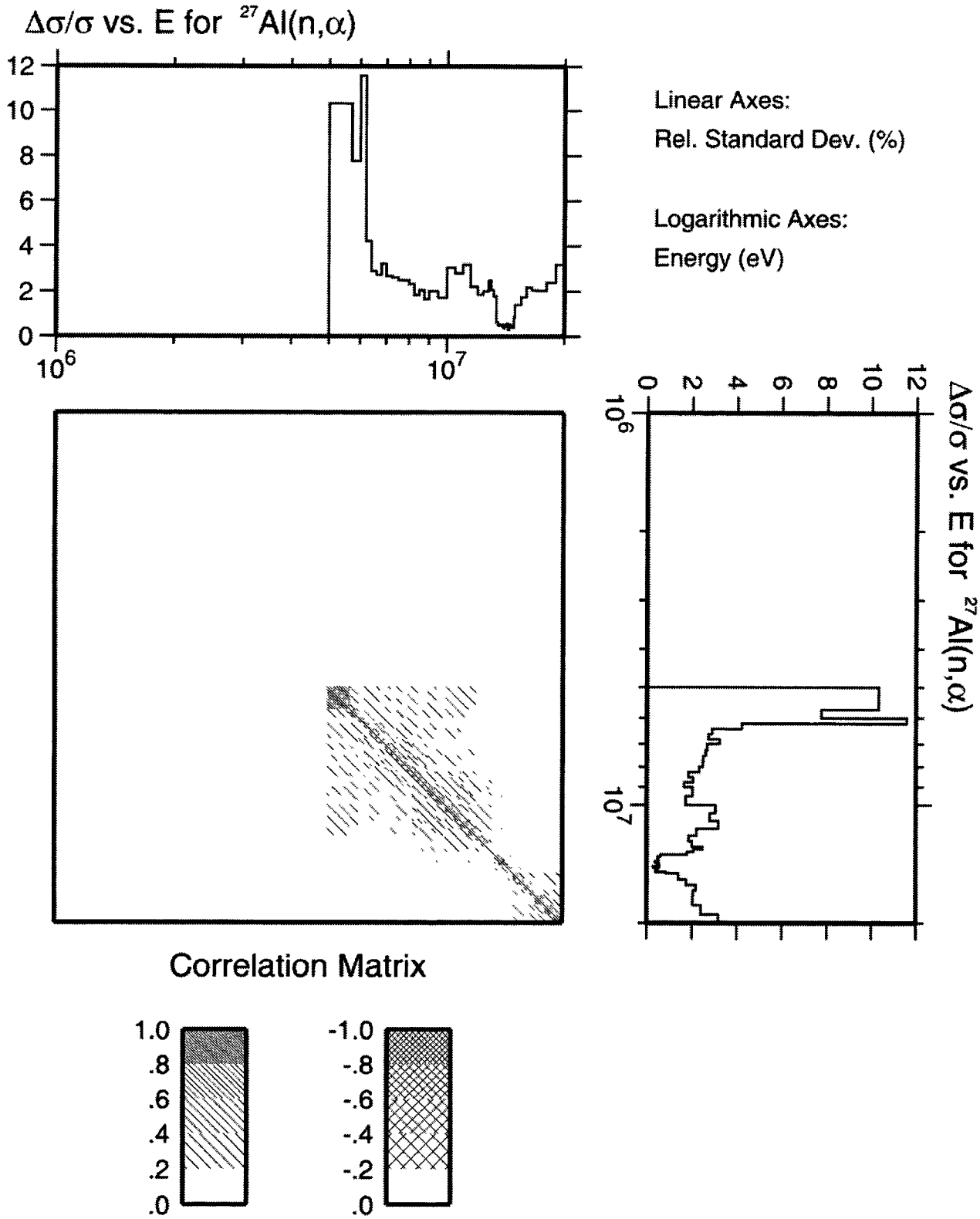

Fig. D.9. ^{27}Al(n,α)^{24}Na (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

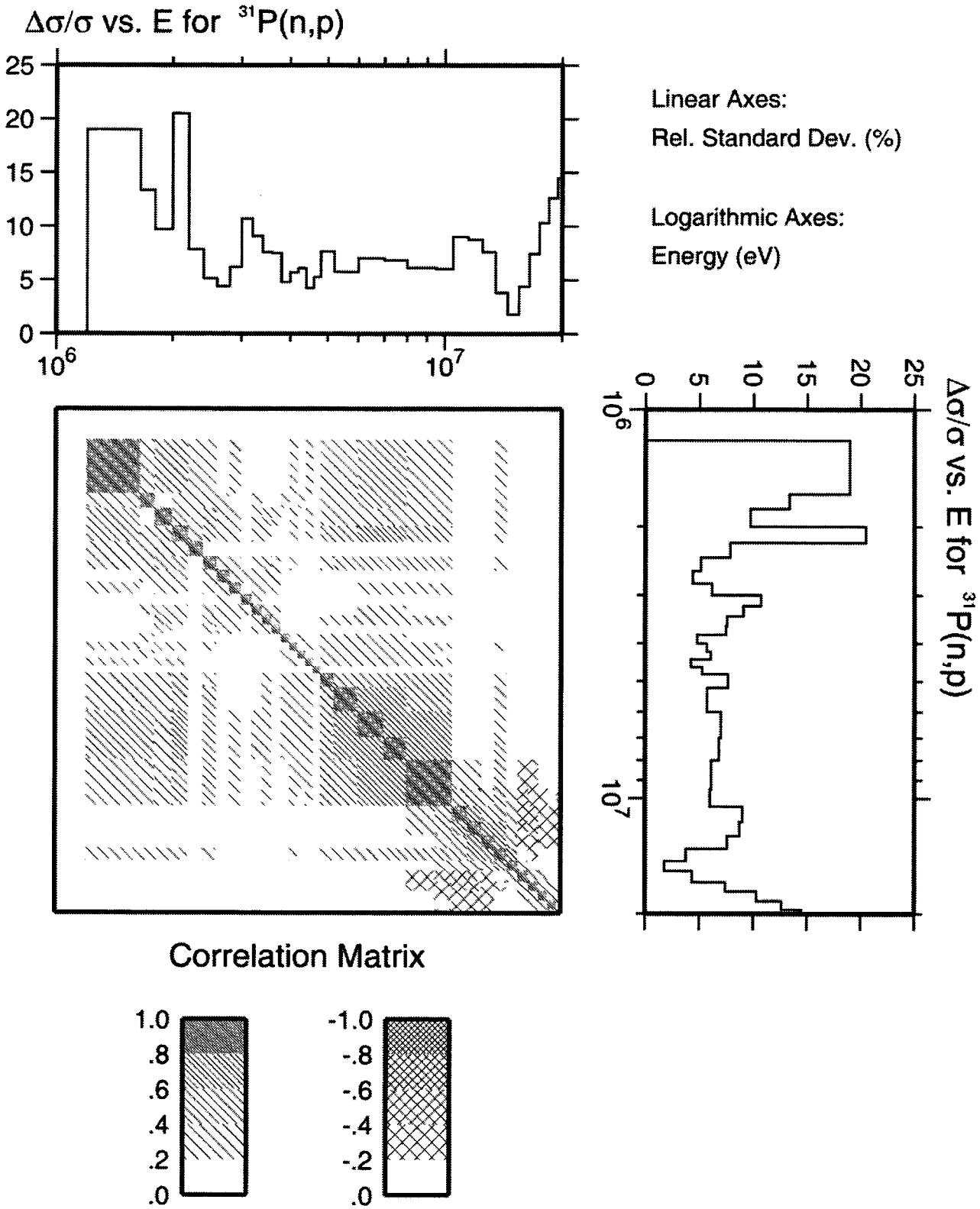

Fig. D.10. $^{31}P(n,p)^{31}Si$ (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

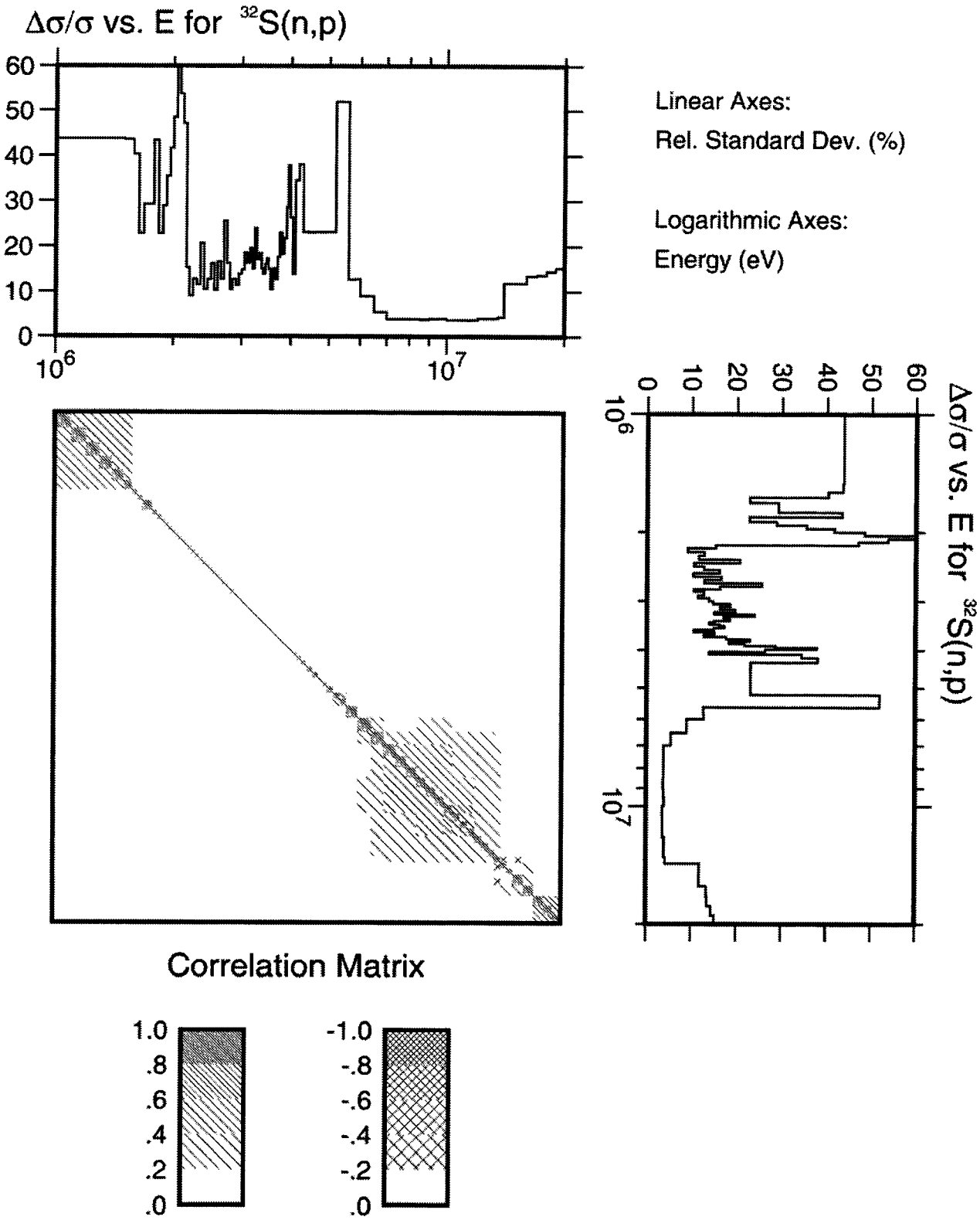

Fig. D.11. $^{32}S(n,p)^{32}P$ (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

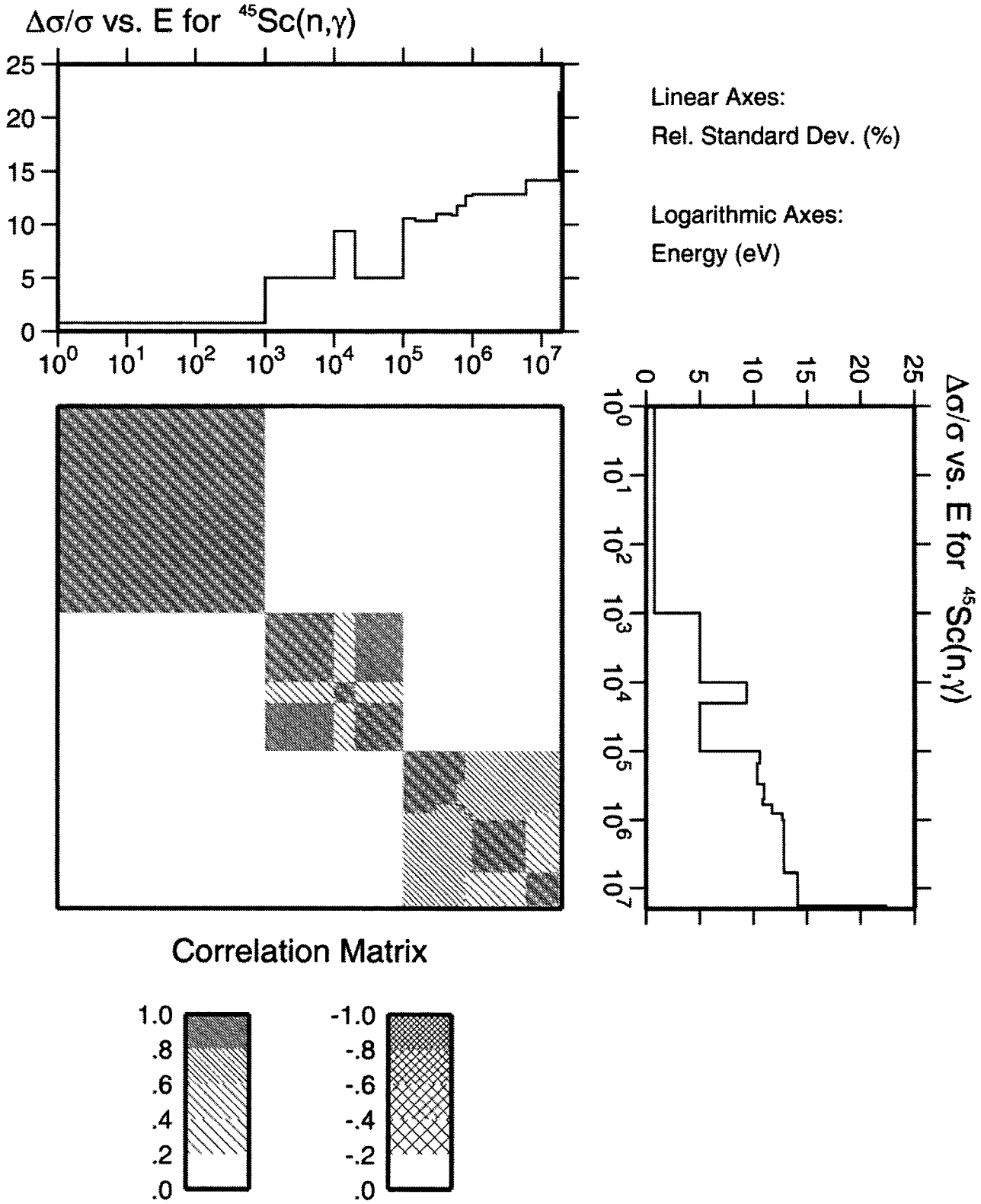

Fig. D.12. $^{45}Sc(n,\gamma)^{46}Sc$ **(IRDF-90/G-V2, private communication) correlation matrix and standard deviation.**

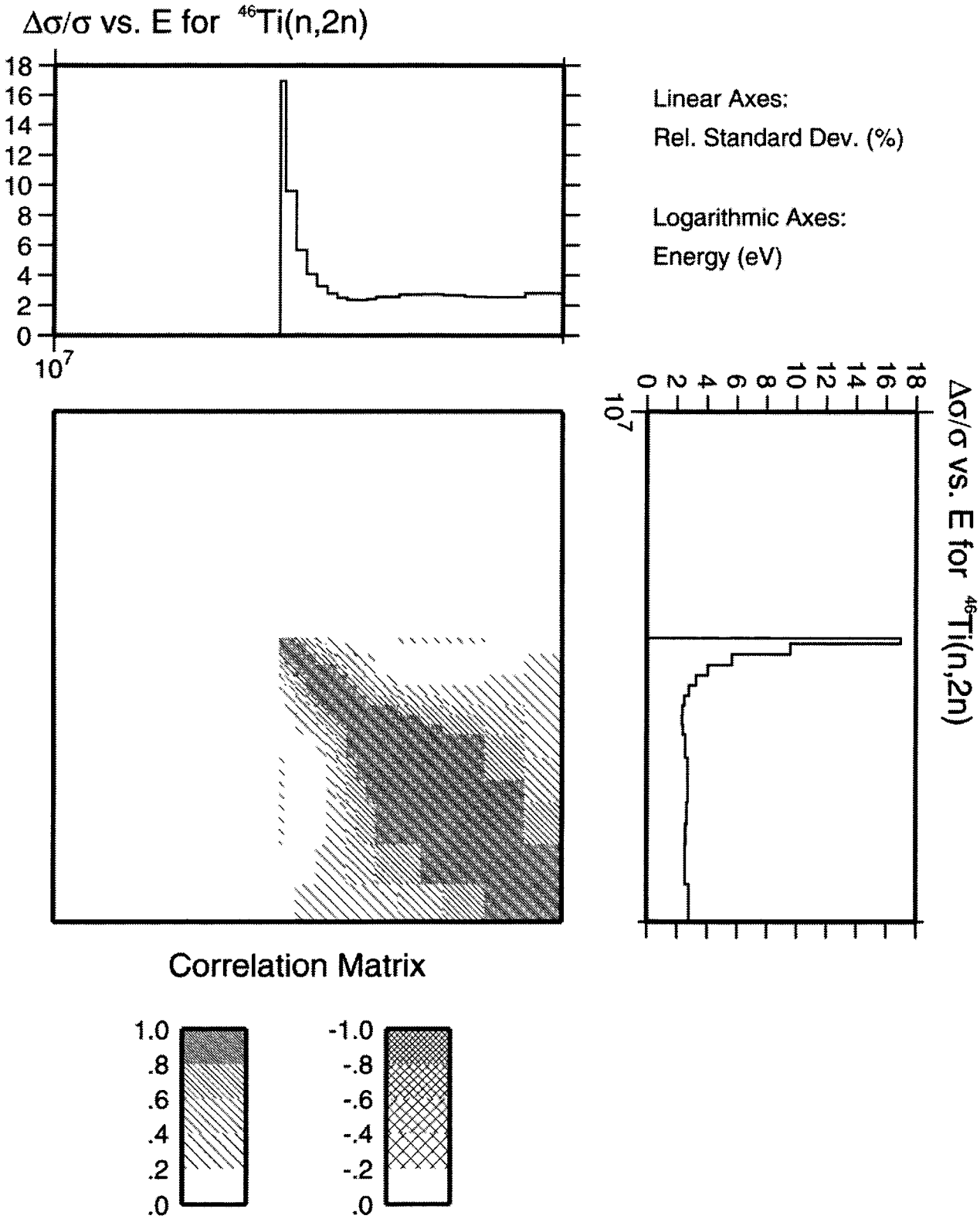

Fig. D.13. ^{46}Ti(n,2n)^{45}Ti (RDF-94, private communication) correlation matrix and standard deviation.

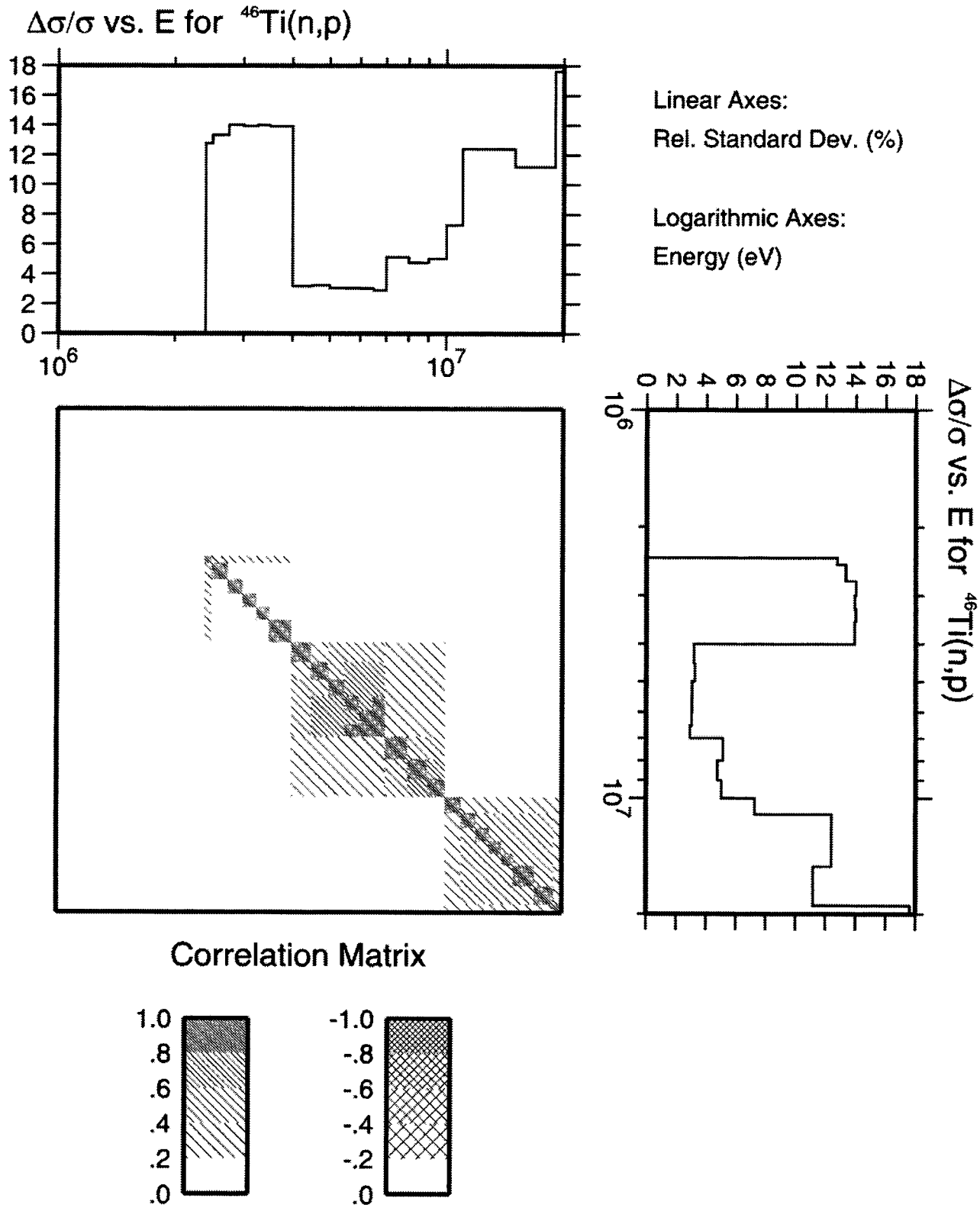

Fig. D.14. ^{46}Ti(n,p)^{46}Sc (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

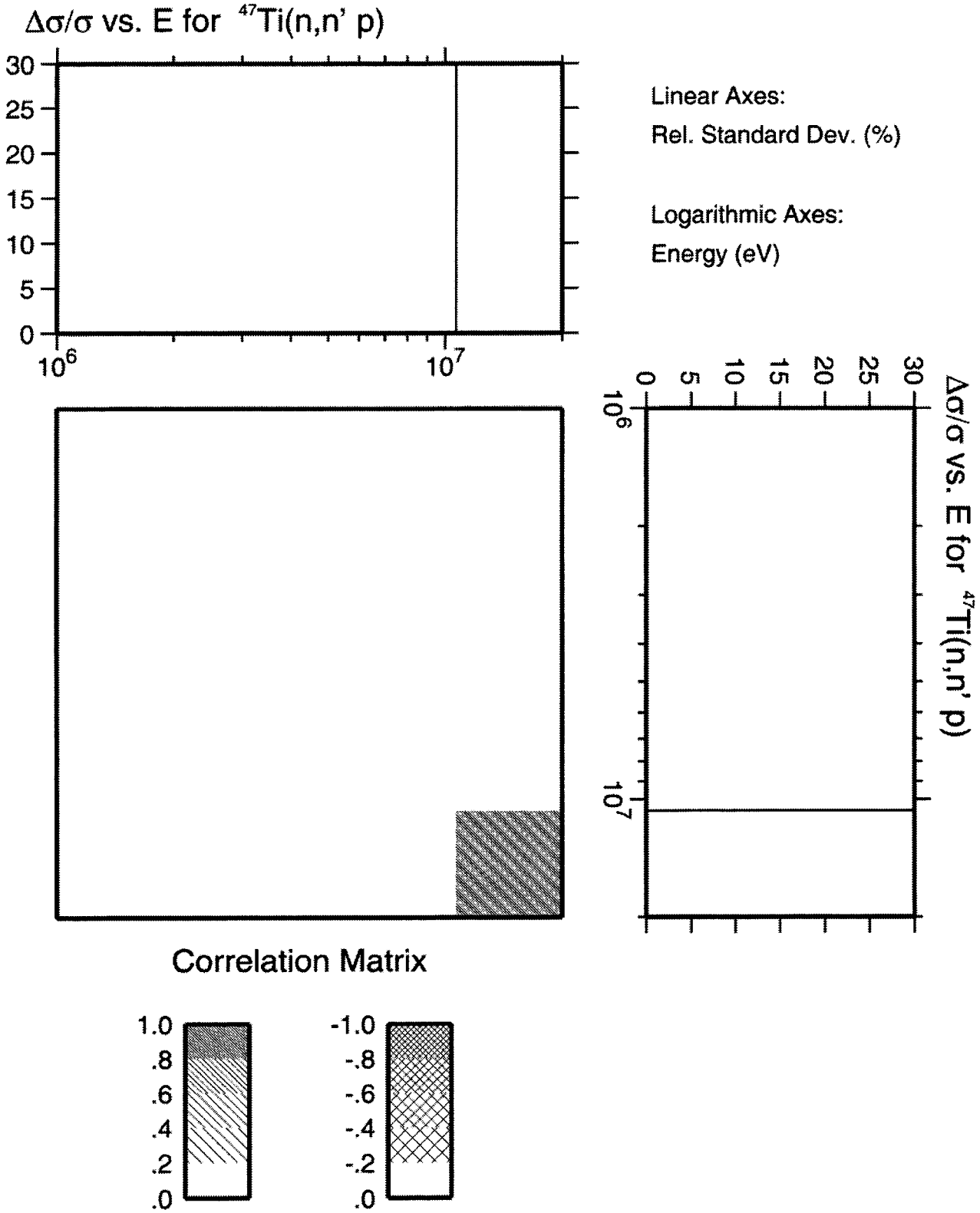

Fig. D.15. ^{47}Ti($n,n'p$)^{46}Sc (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

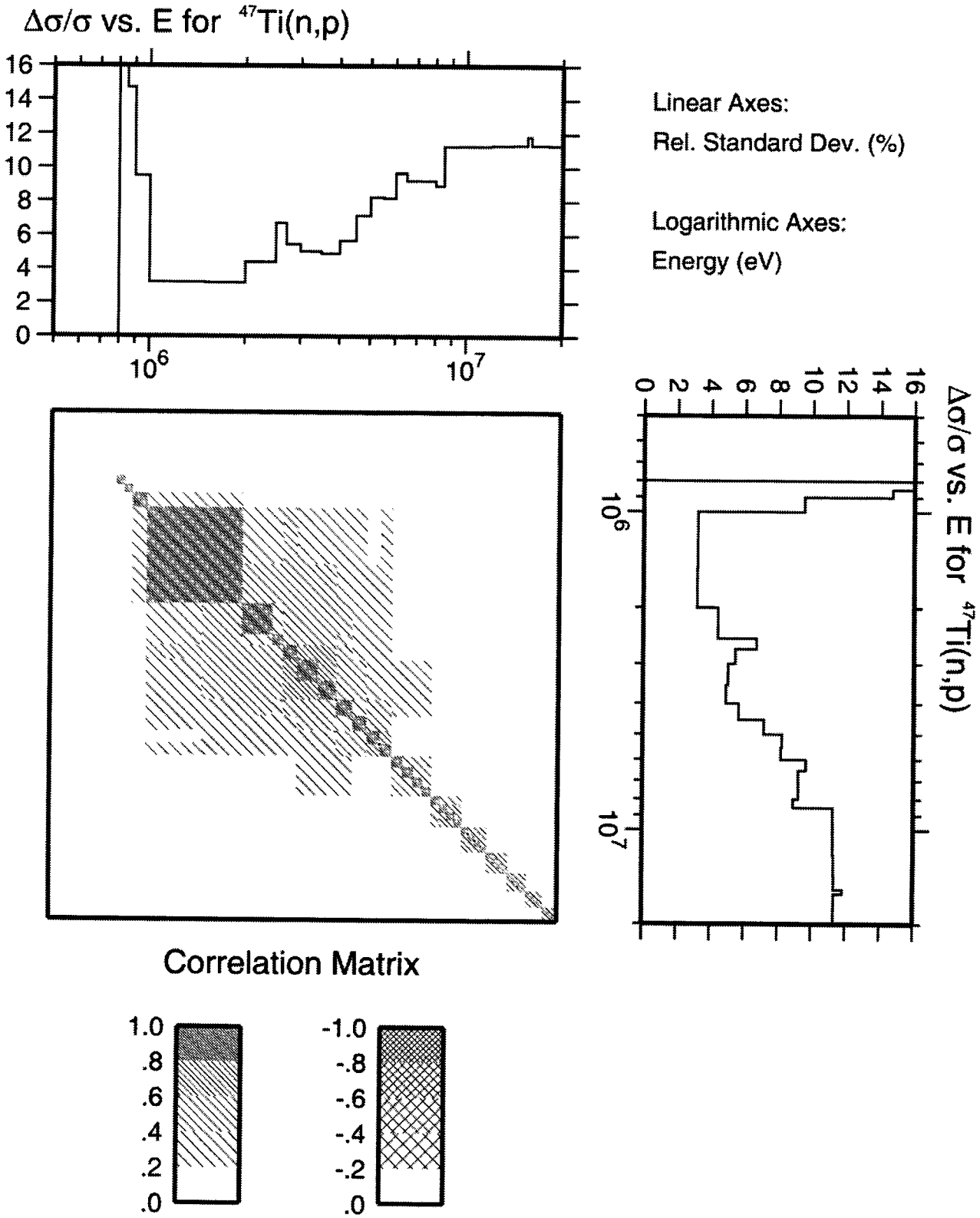

Fig. D.16. $^{47}Ti(n,p)^{47}Sc$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

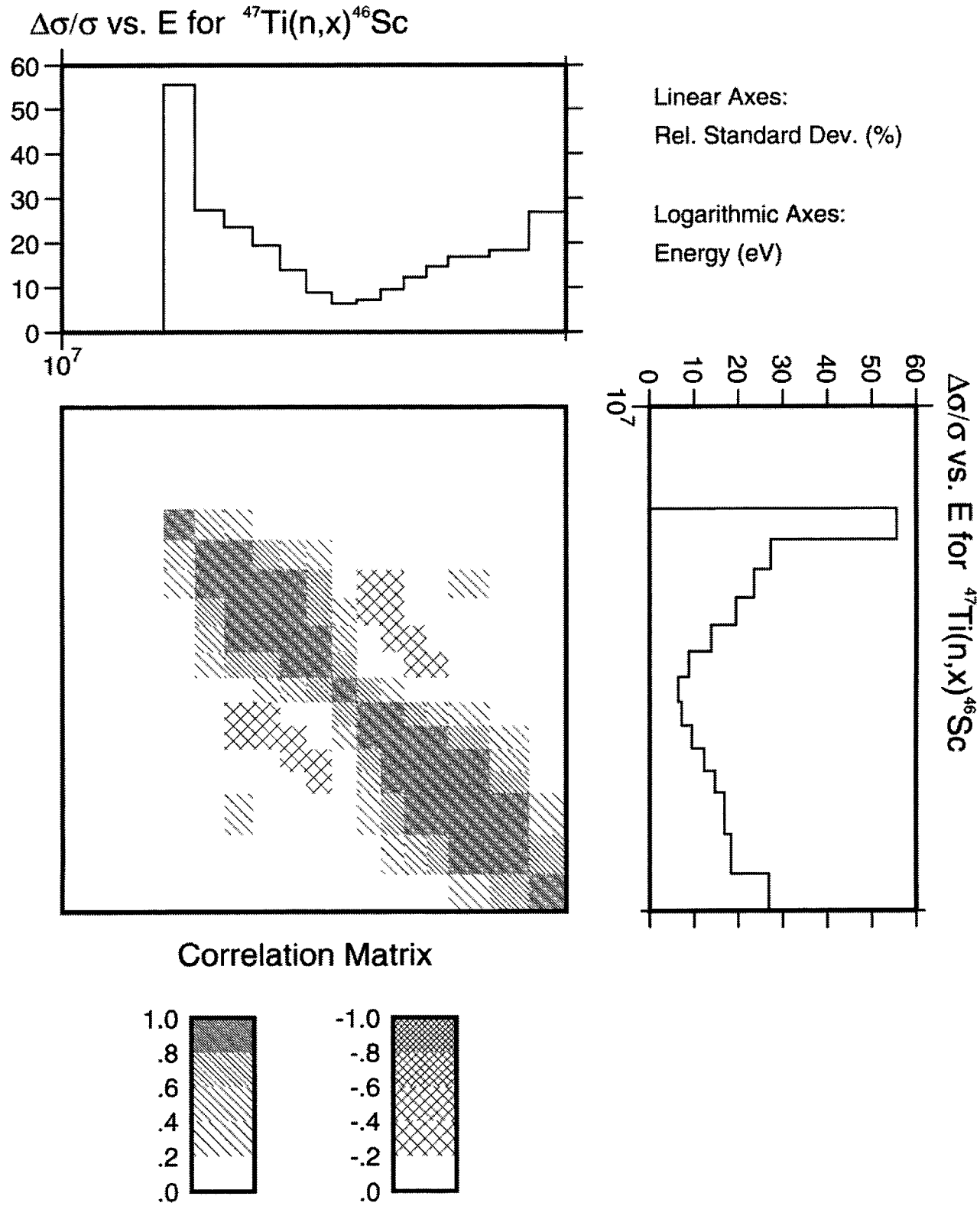

Fig. D.17. ^{47}Ti(n,x)^{46}Sc **(RDF-94, private communication) correlation matrix and standard deviation.**

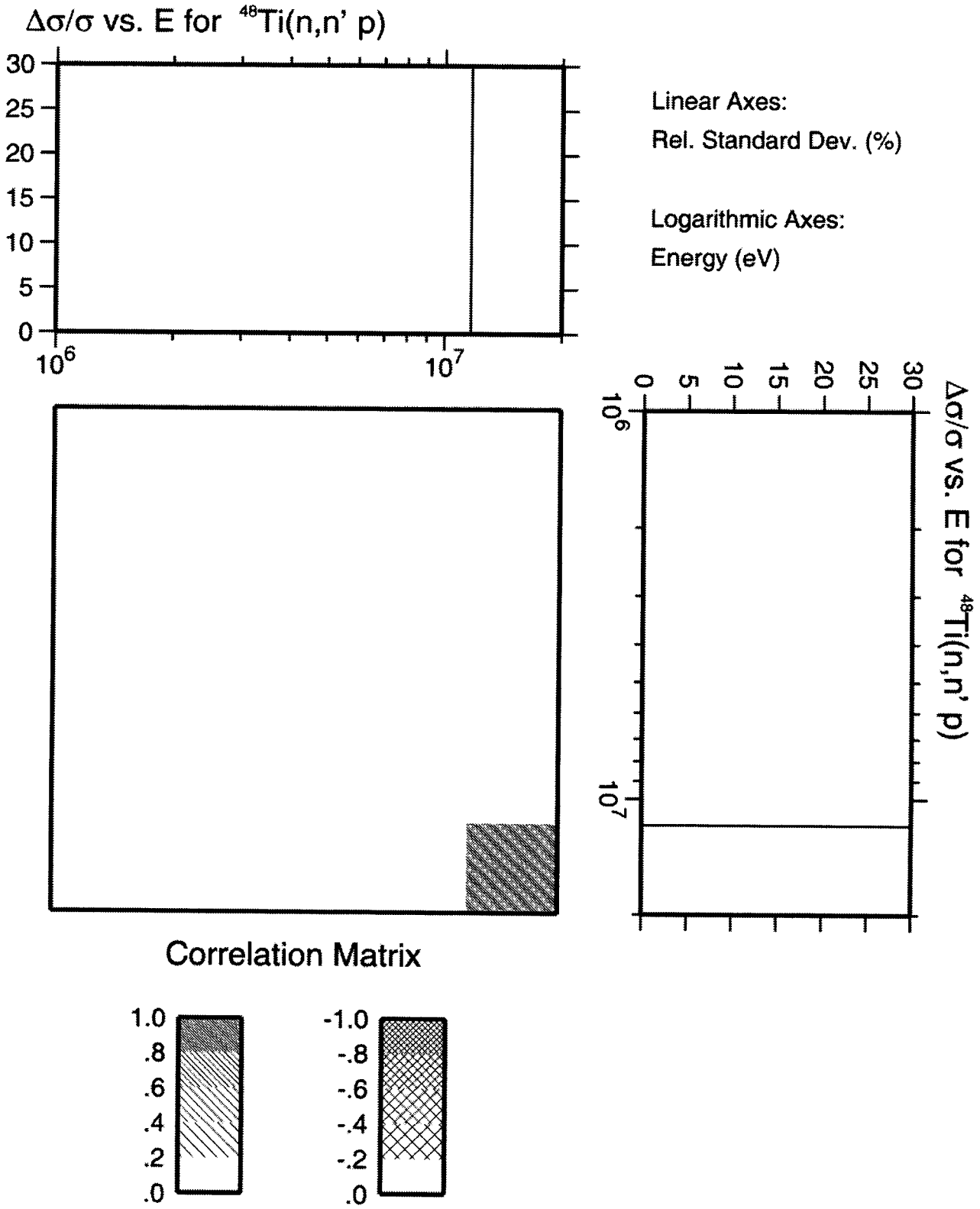

Fig. D.18. ^{48}Ti($n,n'p$)^{47}Sc (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

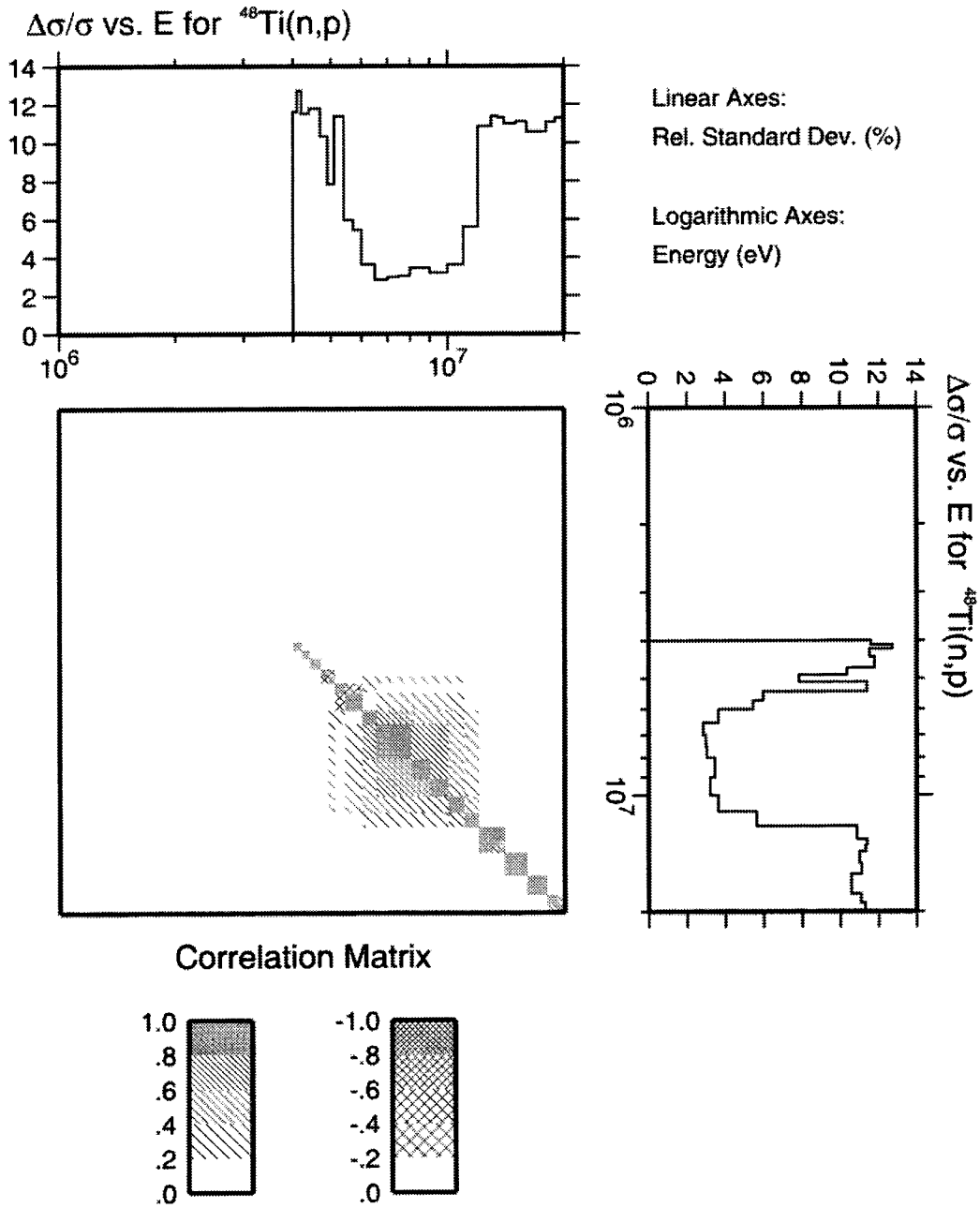

Fig. D.19. ^{48}Ti(n,p)^{48}Sc (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

No covariance data available for this reaction

Fig. D.20. Ti(n,x)^{46}Sc (SNLRML-V1, private communication) correlation matrix and standard deviation. (No covariance data available.)

No covariance data available for this reaction

Fig. D.21. Ti(n,x)^{47}Sc (SNLRML-V1, private communication) correlation matrix and standard deviation. (No covariance data available.)

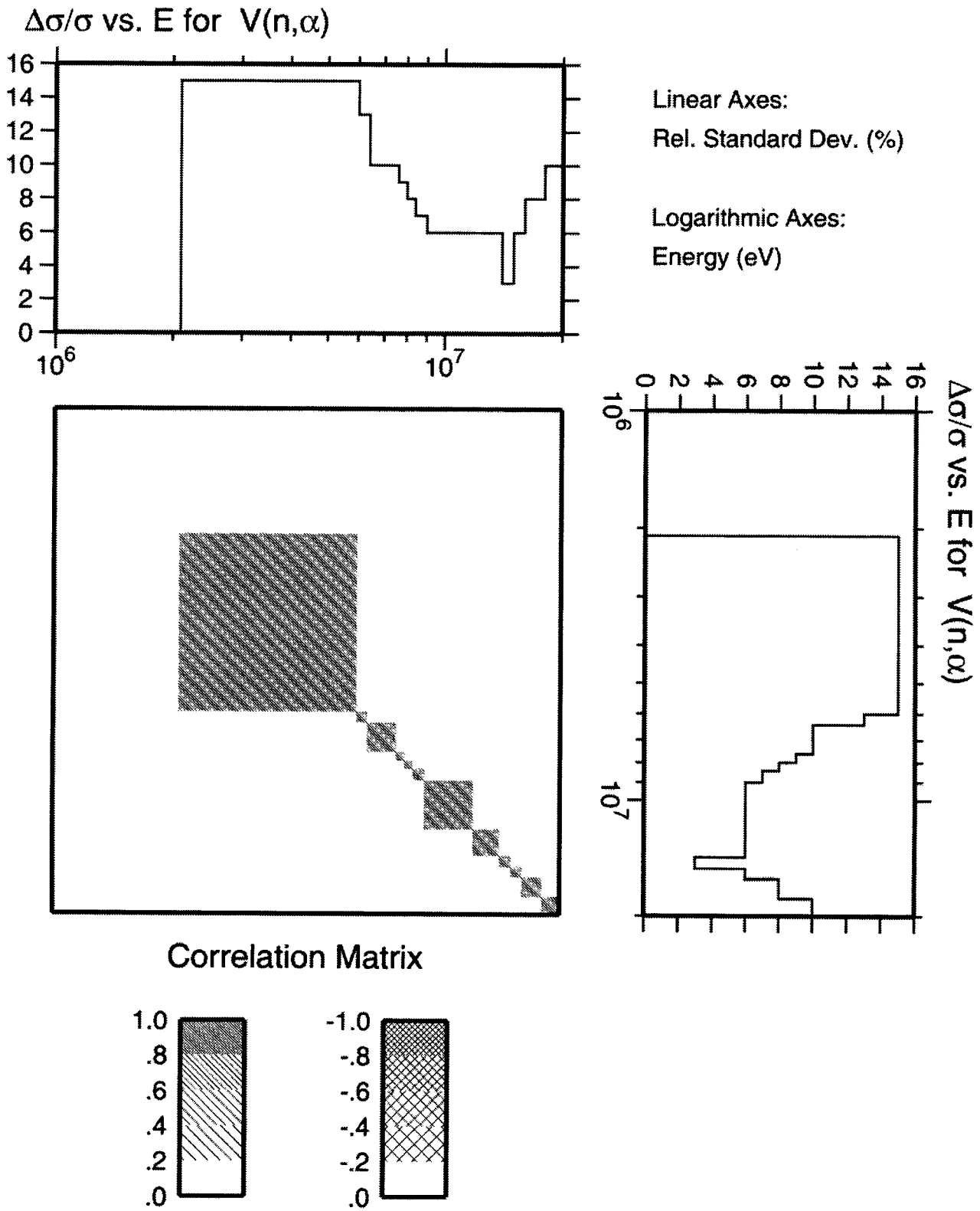

Fig. D.22. V(*n,α*)Sc (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

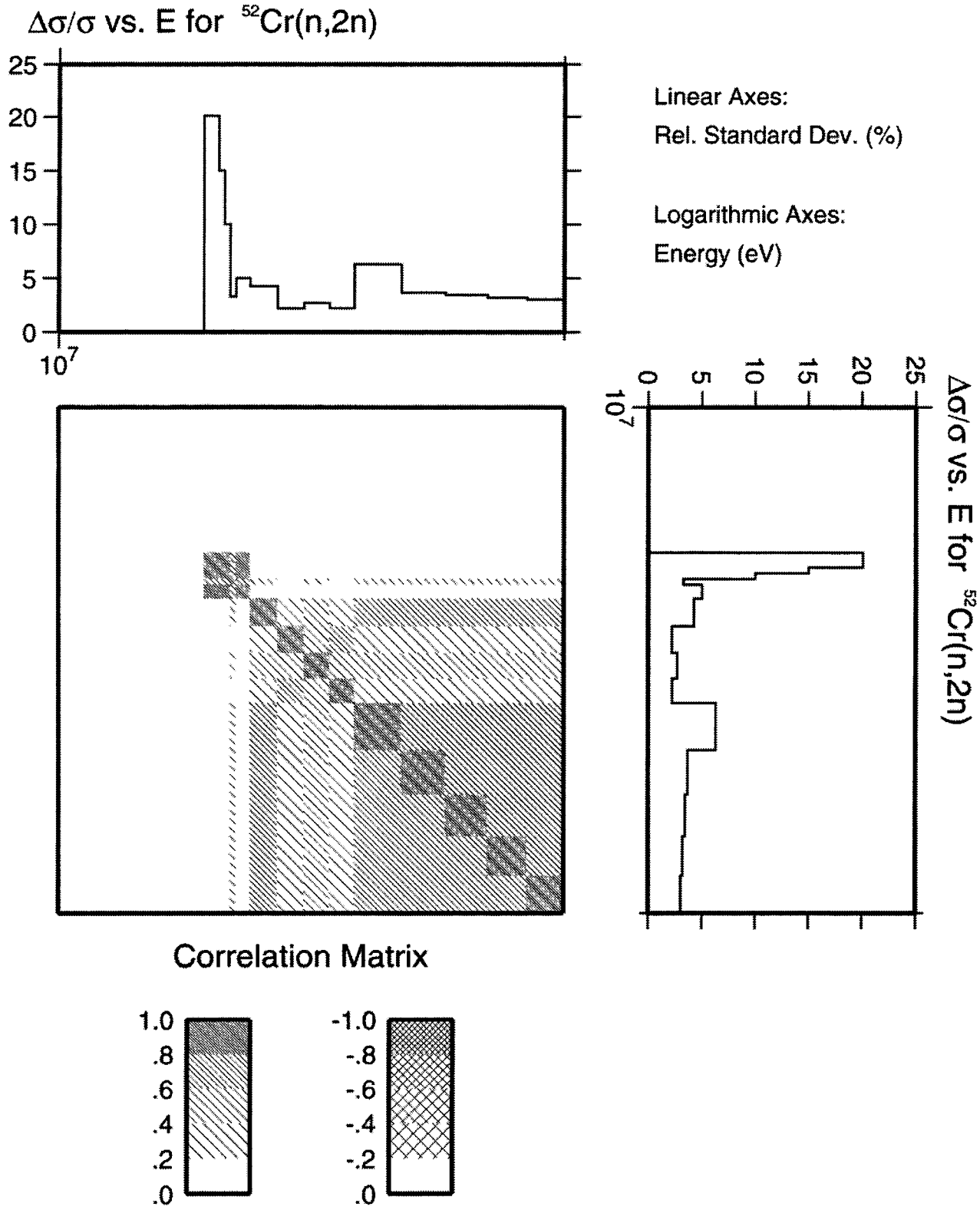

Fig. D.23. $^{52}Cr(n,2n)^{51}Cr$ (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

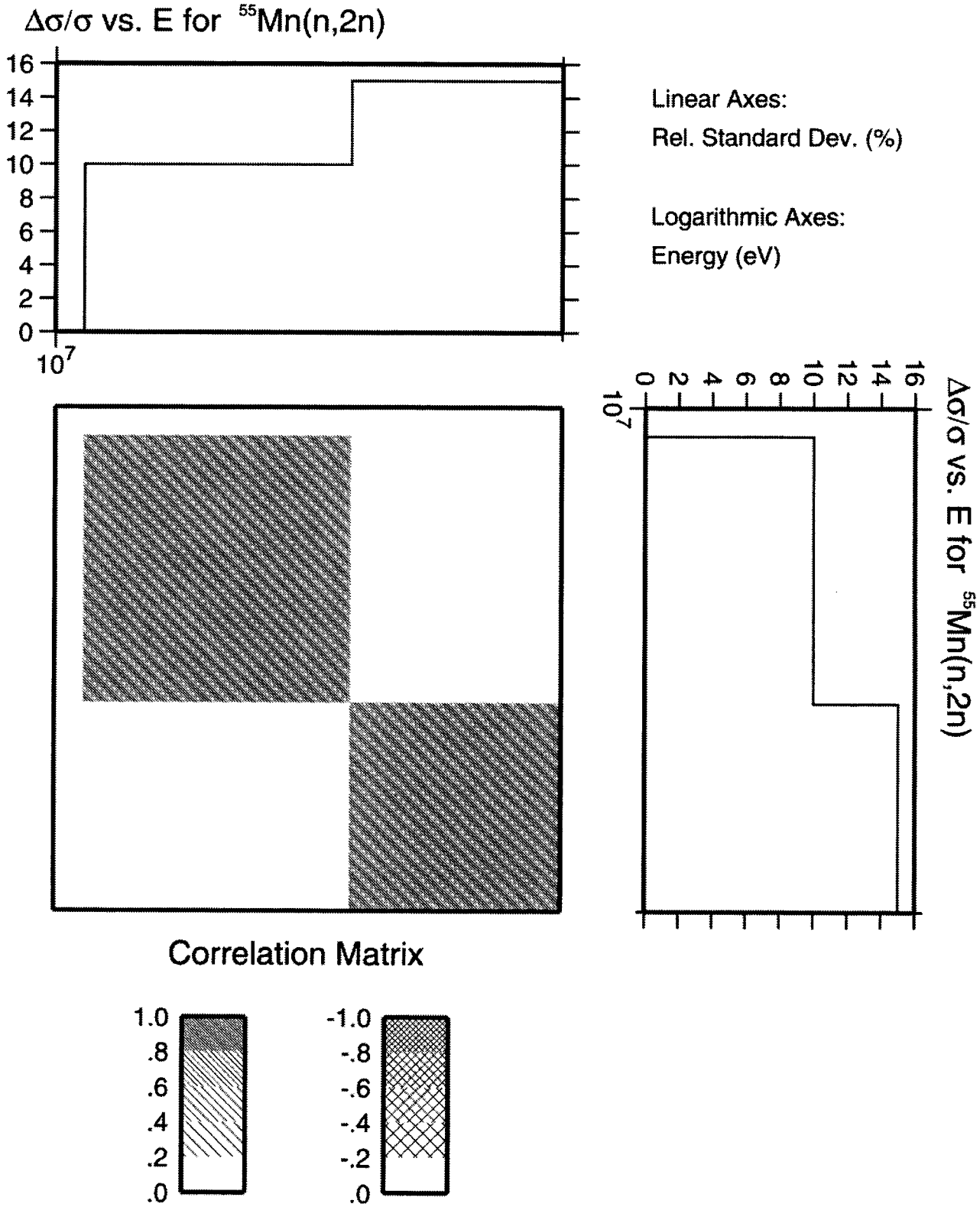

Fig. D.24. ^{55}Mn(n,2n)^{54}Mn (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

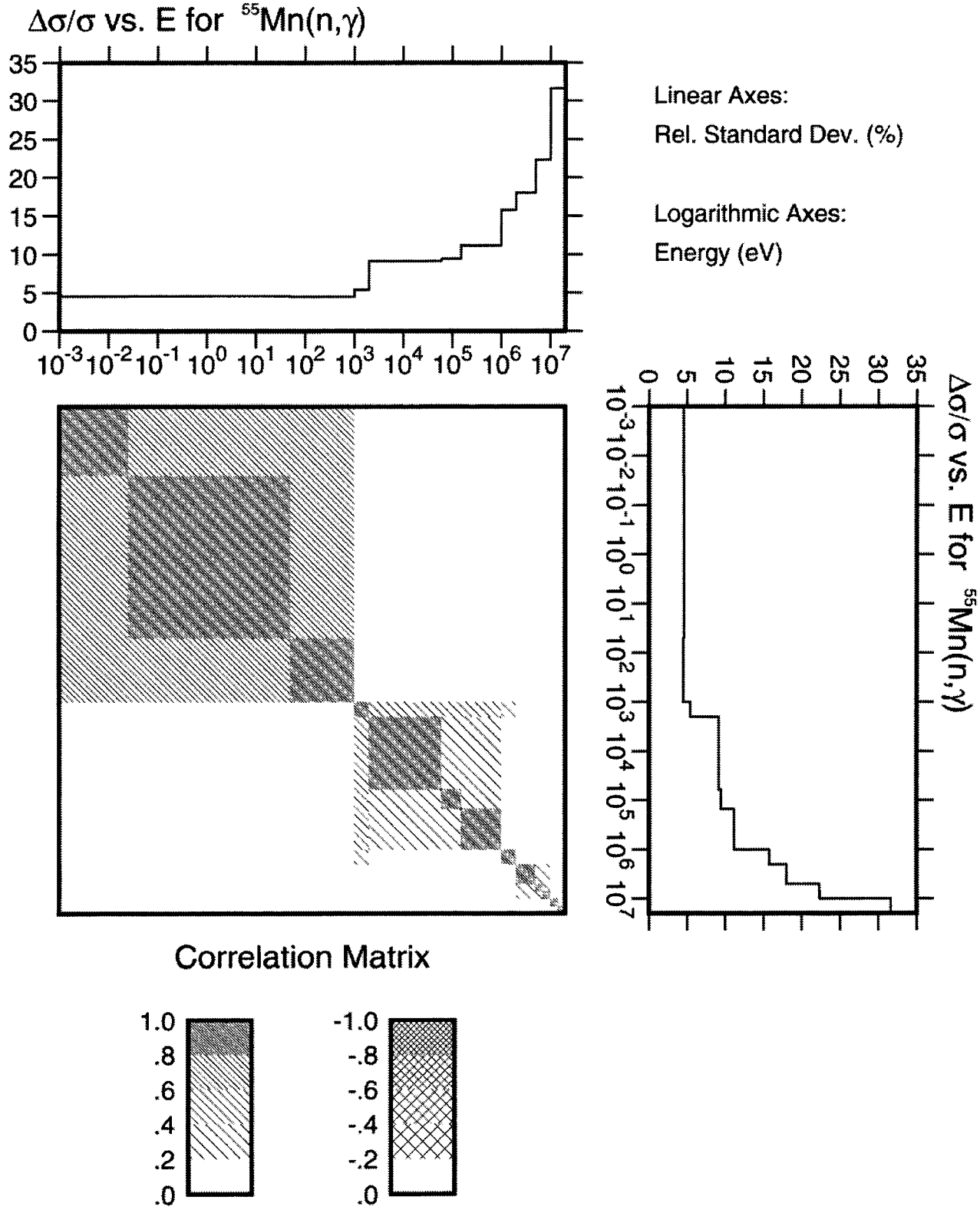

Fig. D.25. $^{55}Mn(n,\gamma)^{56}Mn$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

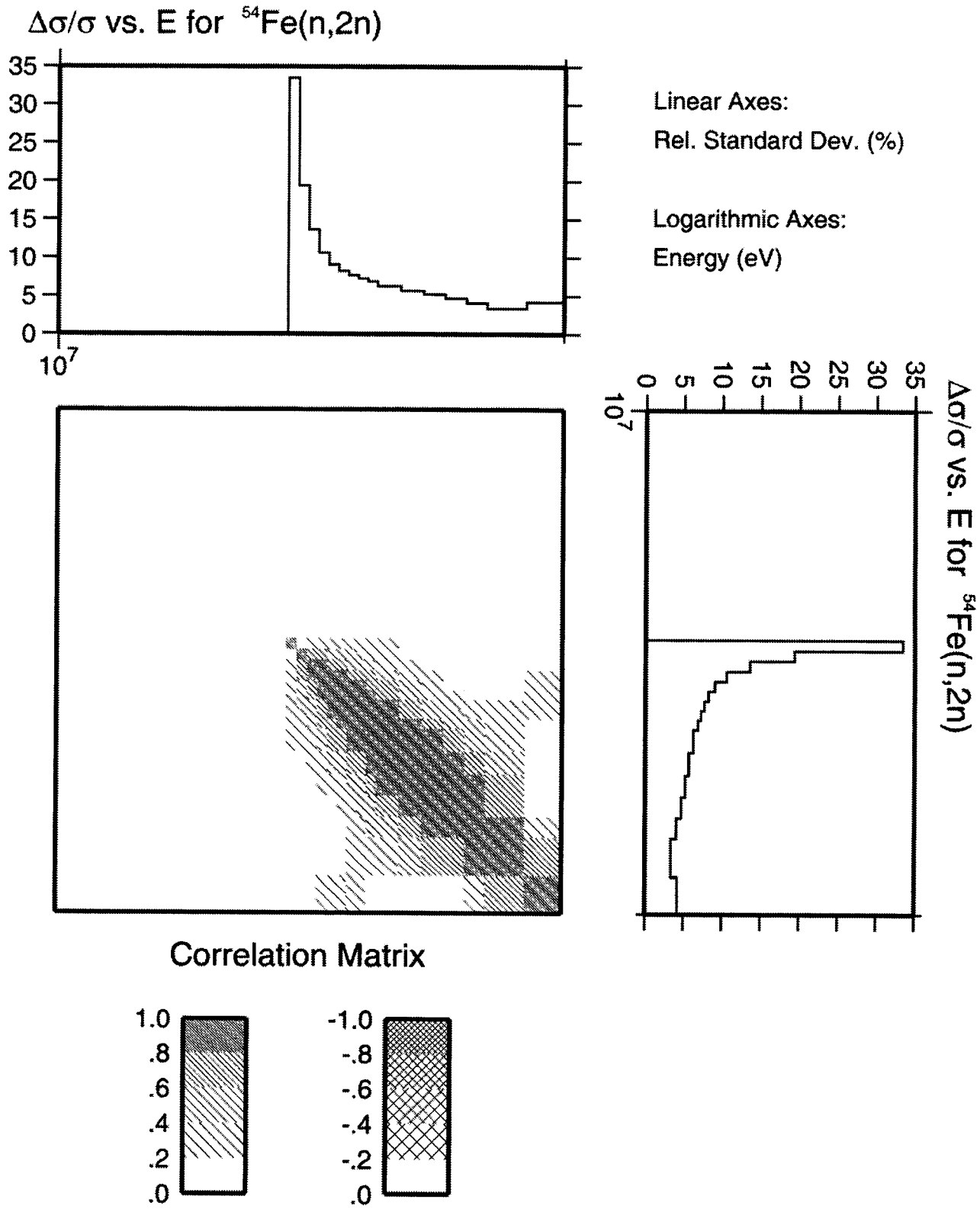

Fig. D.26. $^{54}Fe(n,2n)^{53}Fe$ (RDF-94, private communication) correlation matrix and standard deviation.

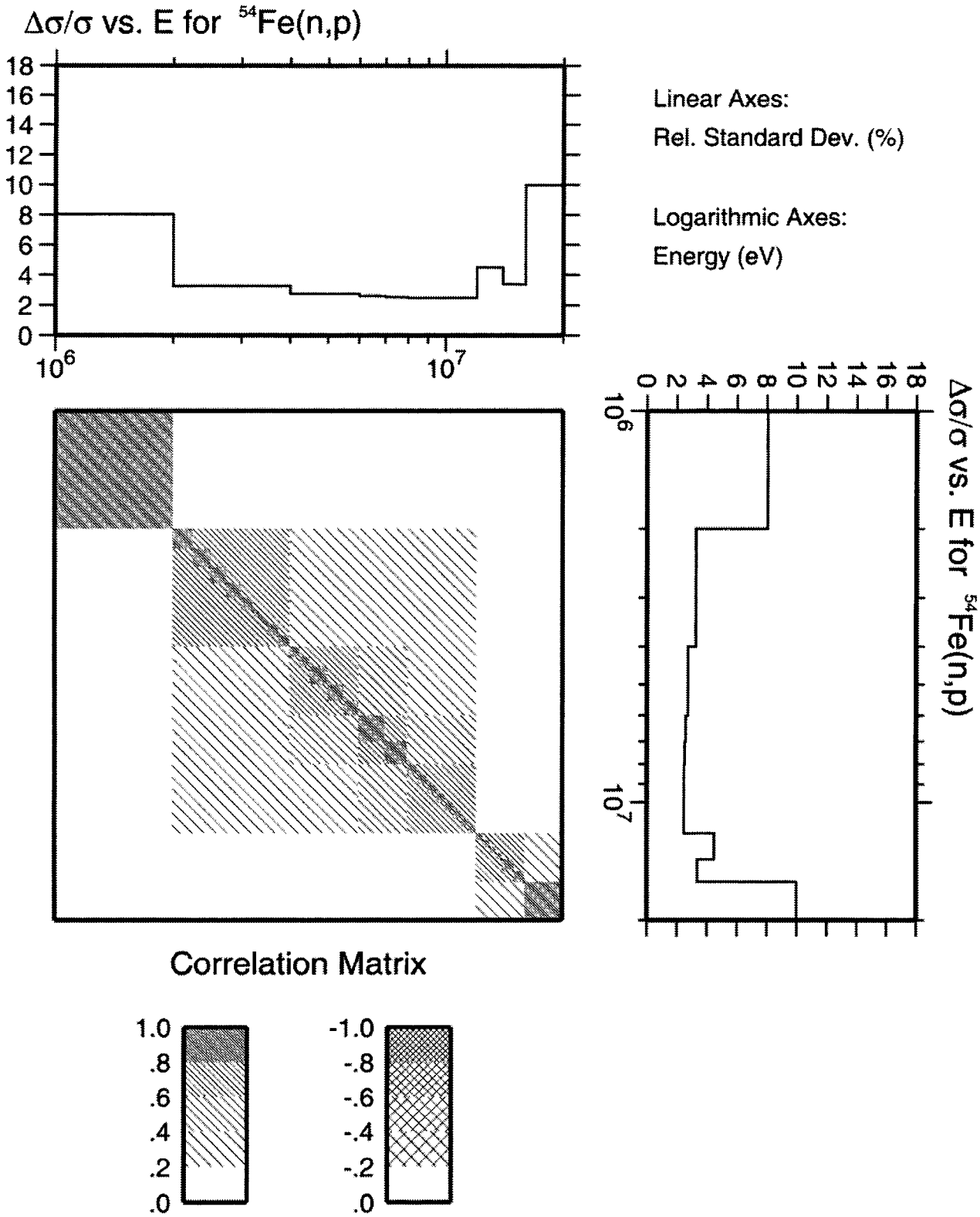

Fig. D.27. $^{54}Fe(n,p)^{54}Mn$ (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

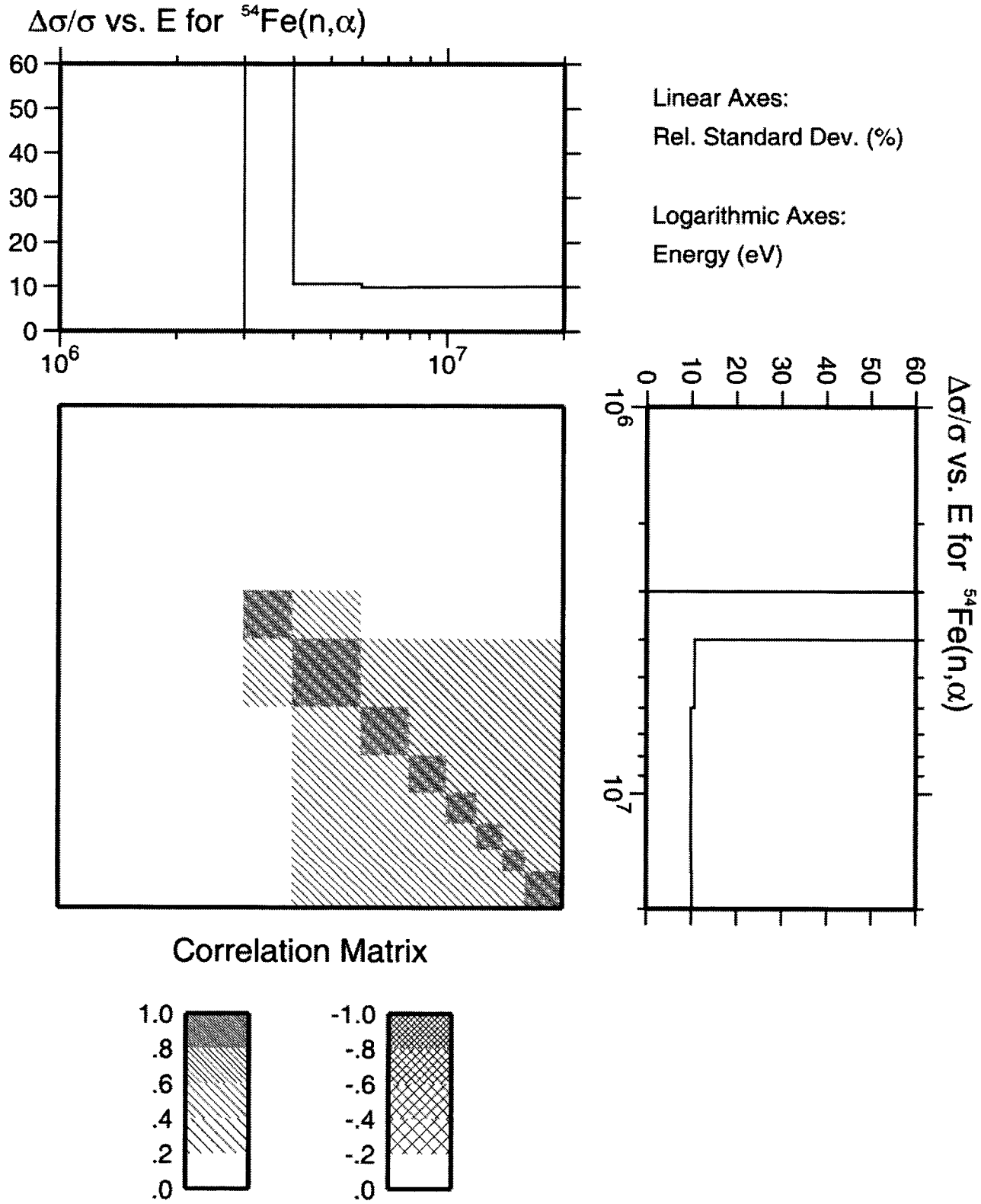

Fig. D.28. ^{54}Fe(n,α)^{51}Cr (SNLRML-V1, ENDF/B-VI), correlation matrix and standard deviation.

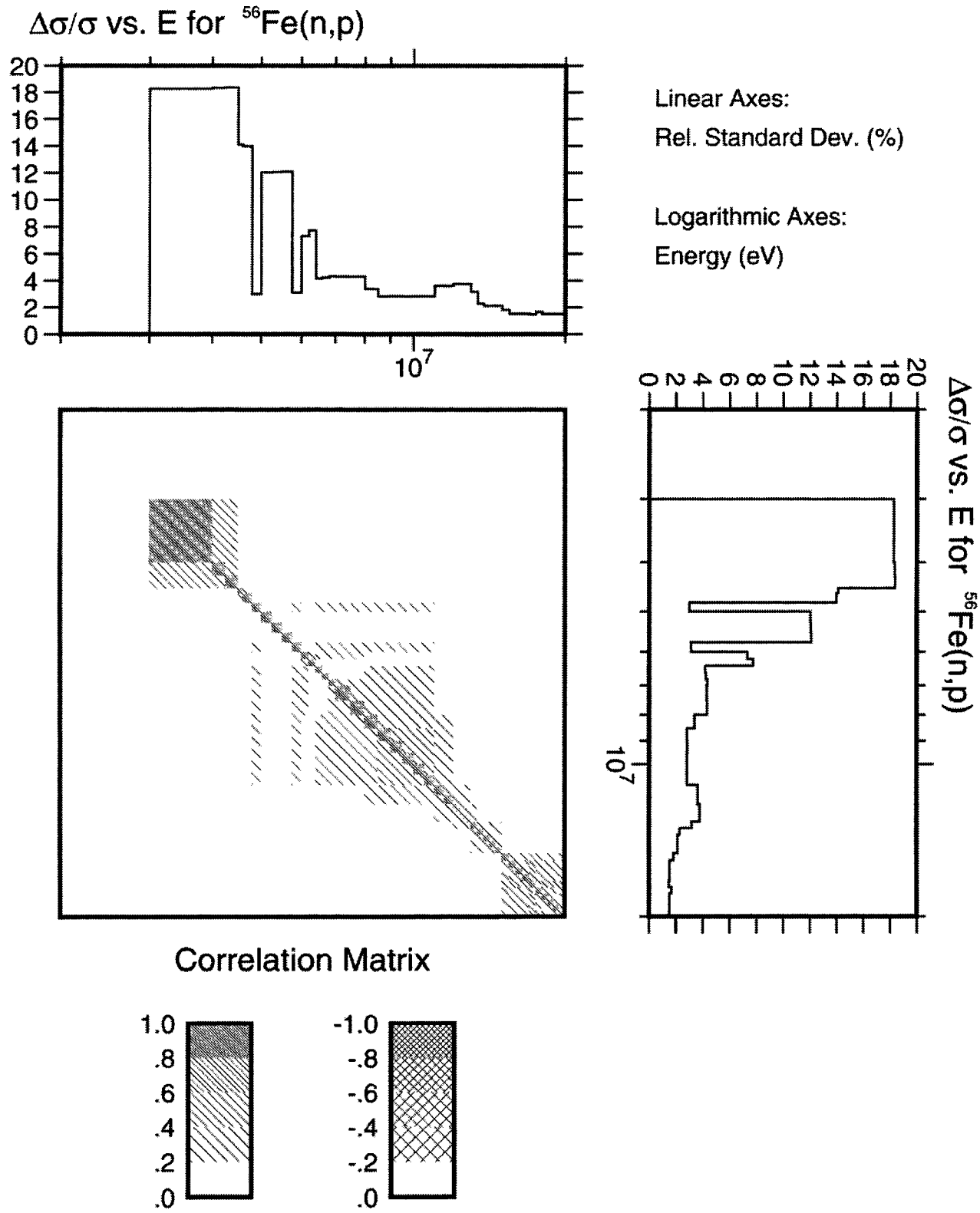

Fig. D.29. $^{56}Fe(n,p)^{56}Mn$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

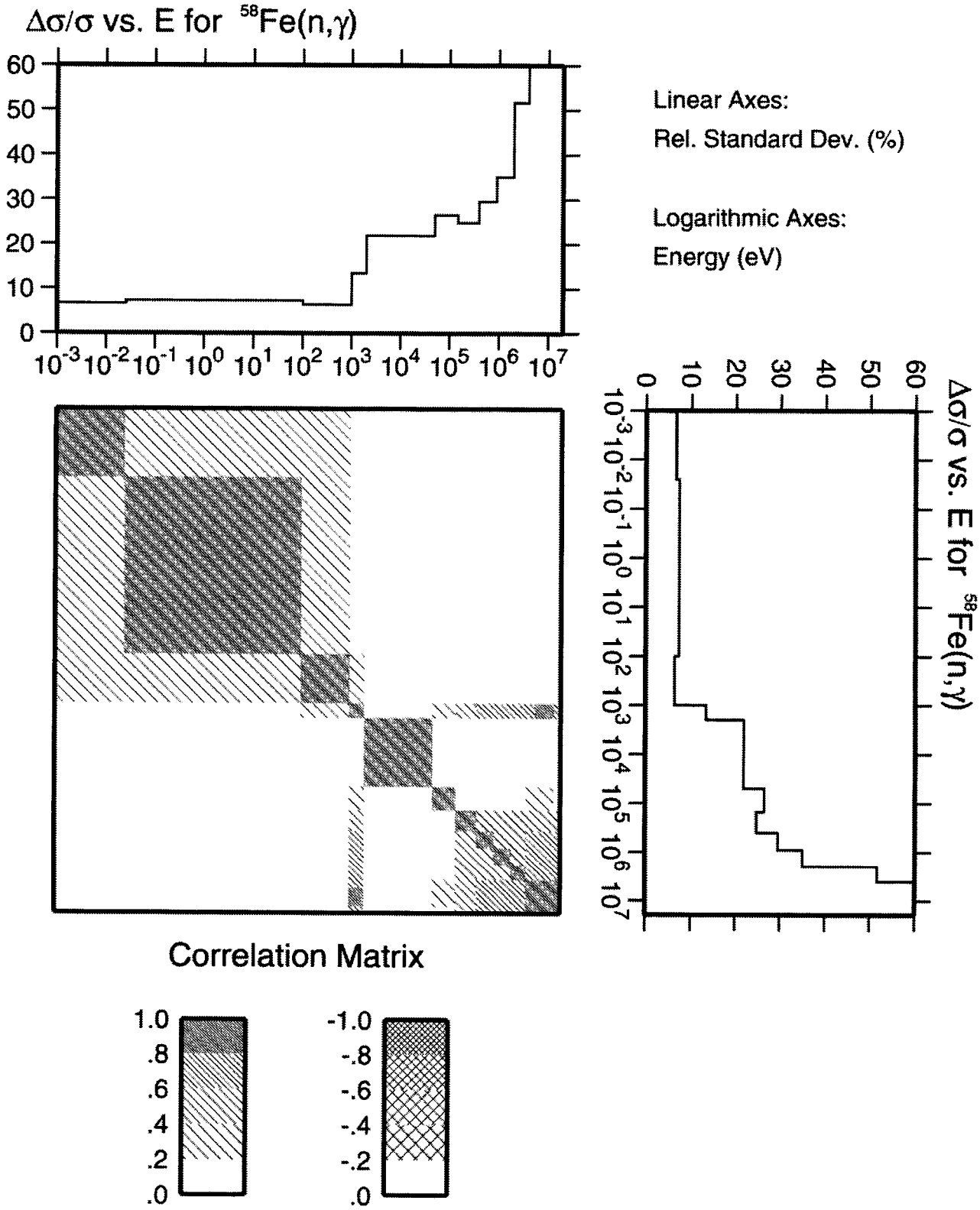

Fig. D.30. $^{58}Fe(n,\gamma)^{59}Fe$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

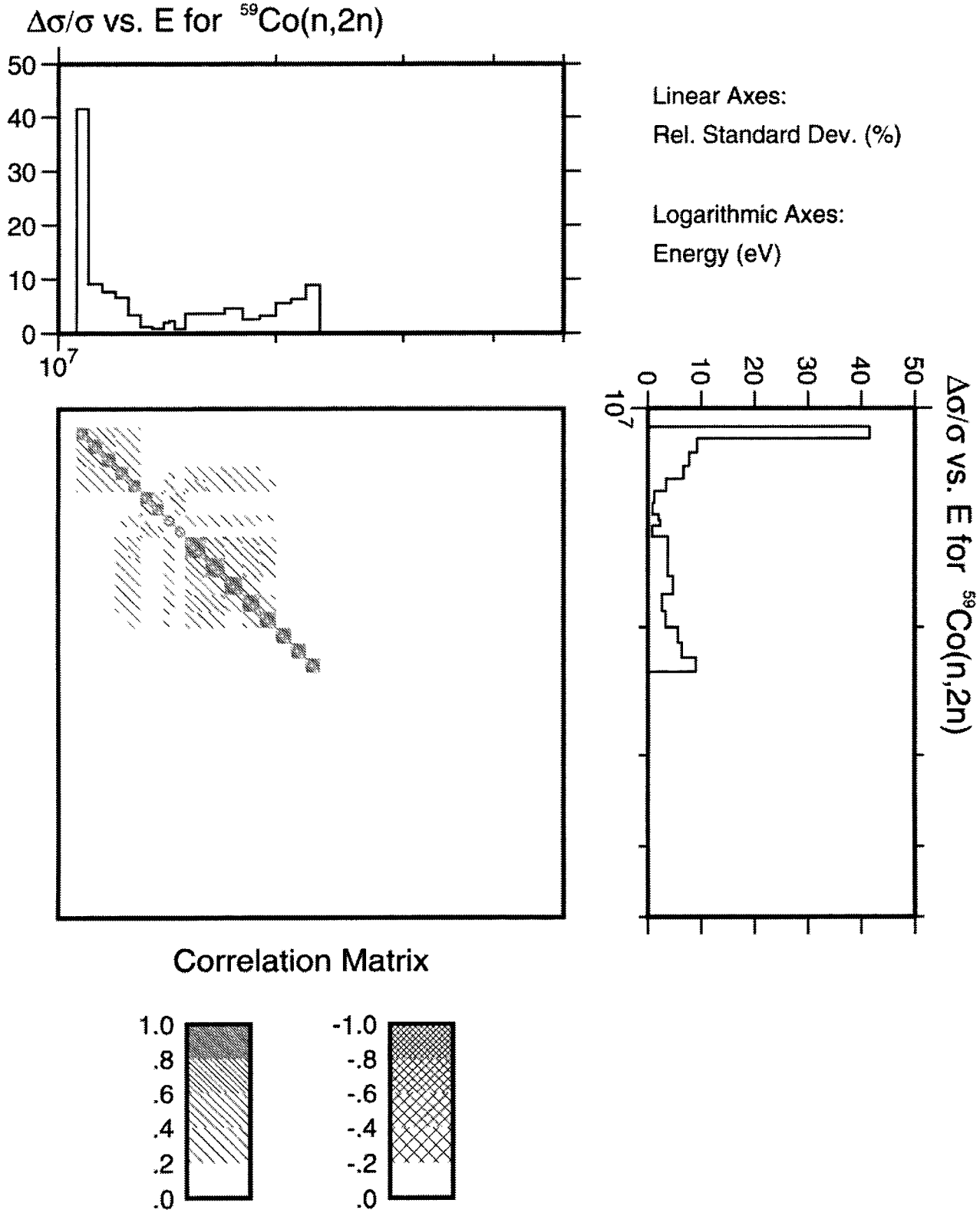

Fig. D.31. ^{59}Co(n,2n)^{58}Co (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

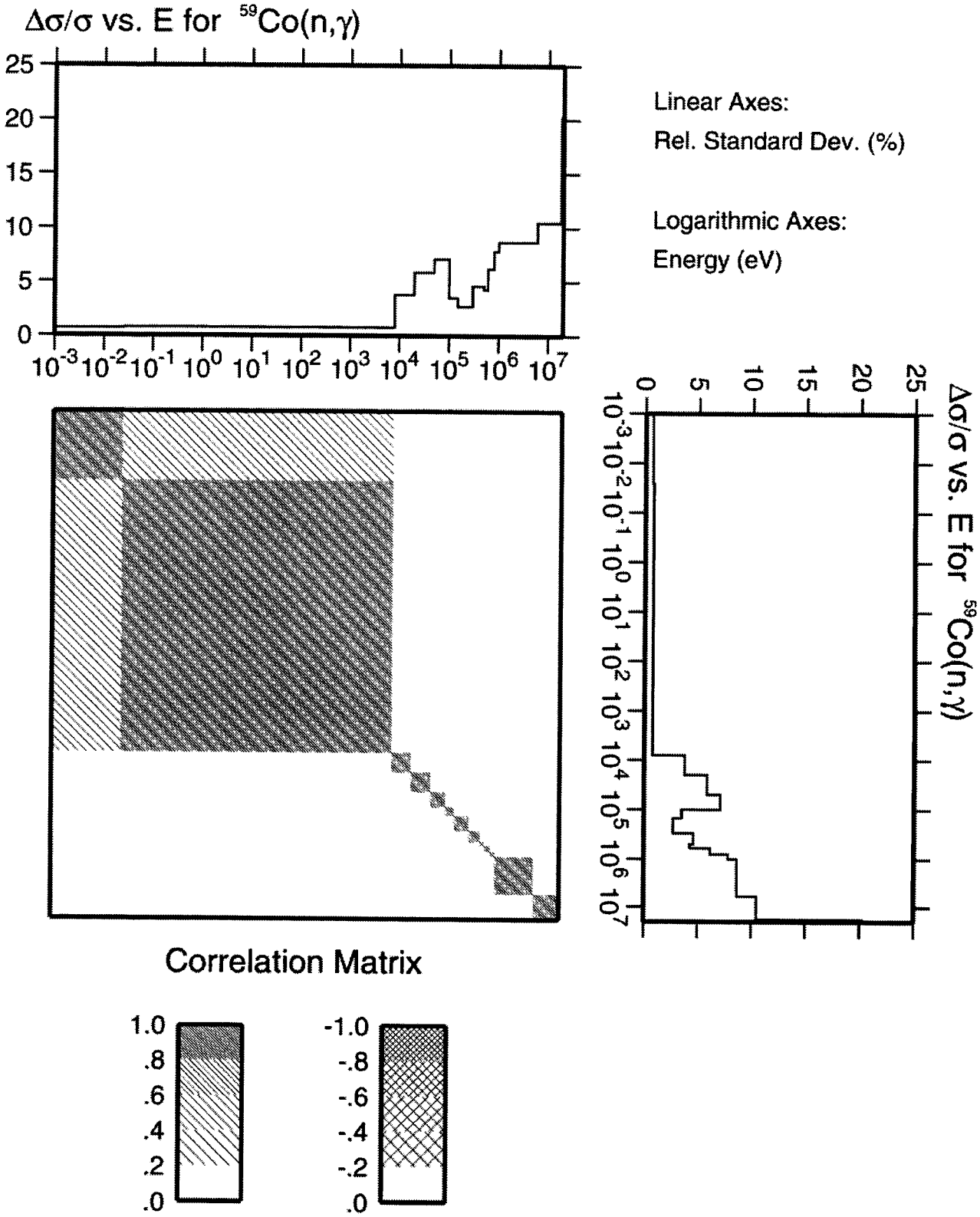

Fig. D.32. $^{59}Co(n,\gamma)^{60}Co$ **(IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.**

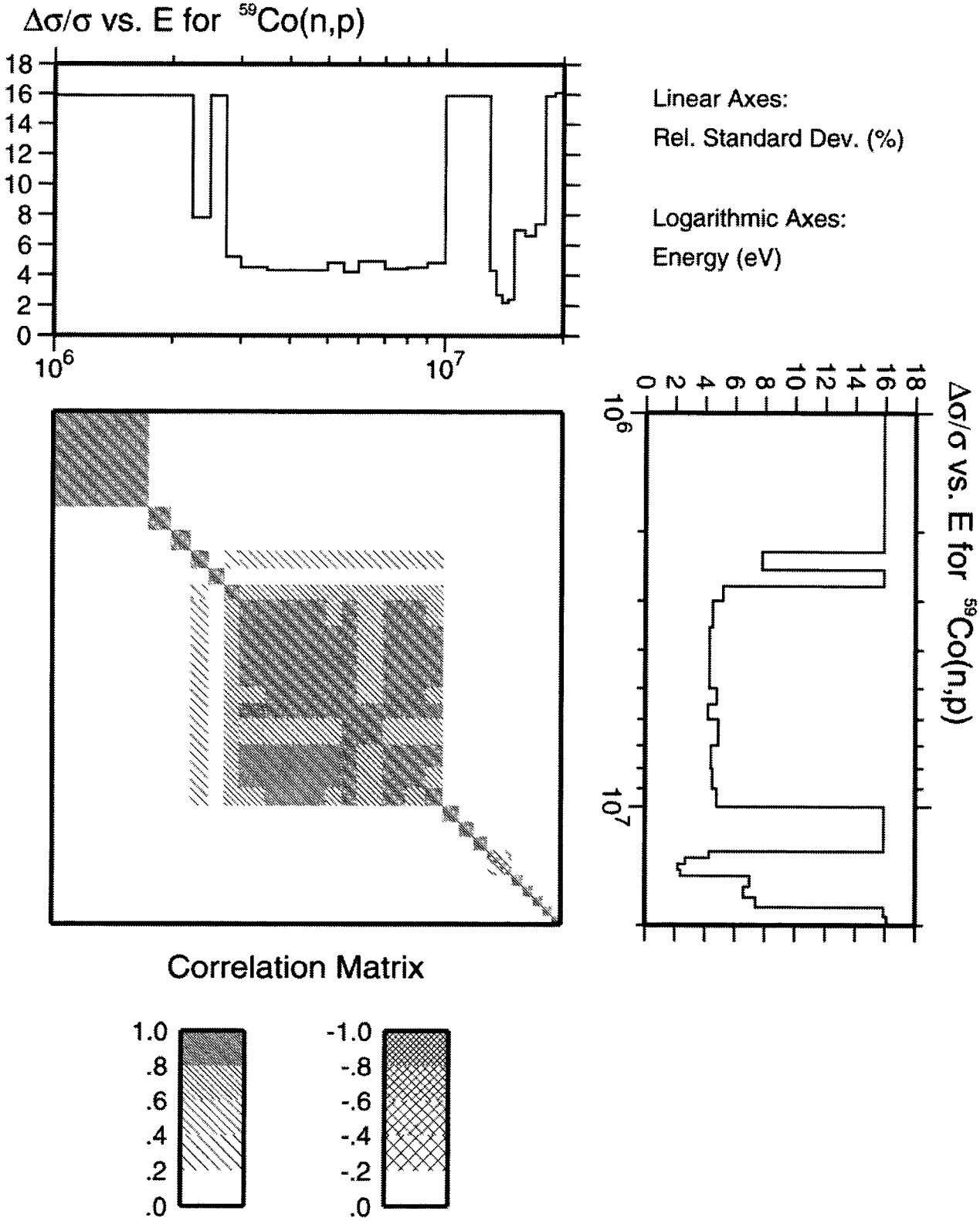

Fig. D.33. $^{59}Co(n,p)^{59}Fe$ (SNLRML-V1, ENDF/B-VI), correlation matrix and standard deviation.

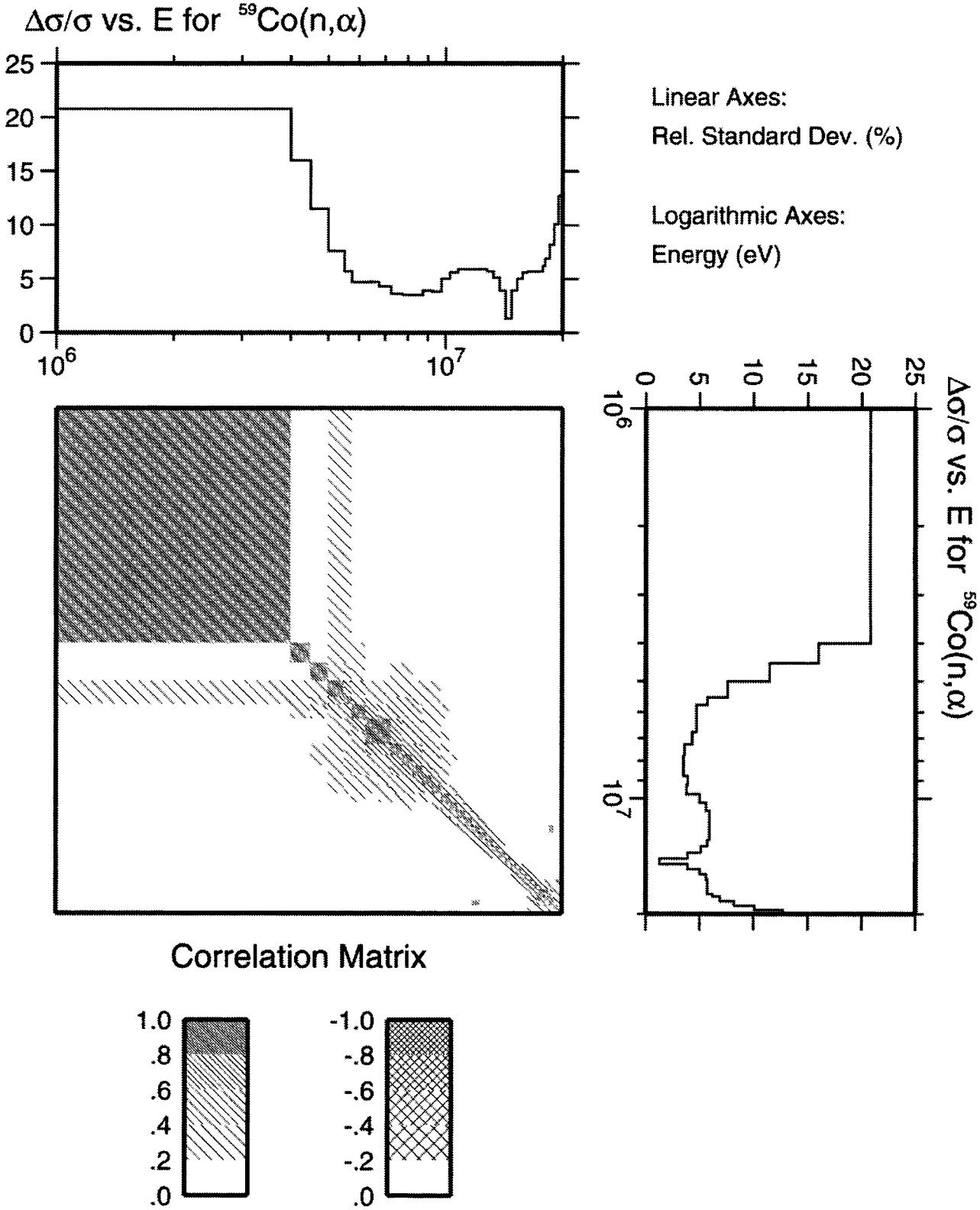

Fig. D.34. ^{59}Co(n,α)^{56}Mn (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

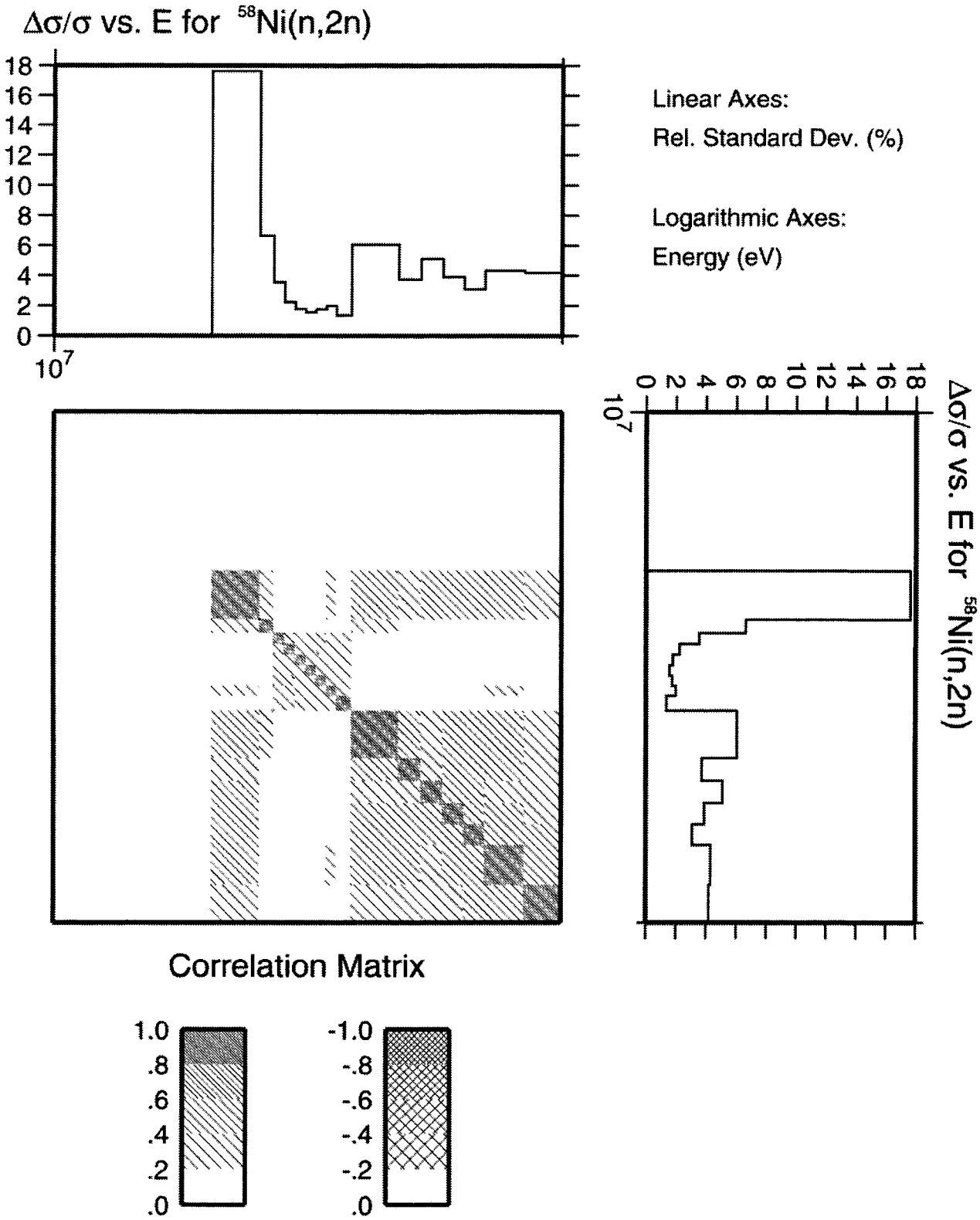

Fig. D.35. ^{58}Ni($n,2n$)^{57}Ni (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

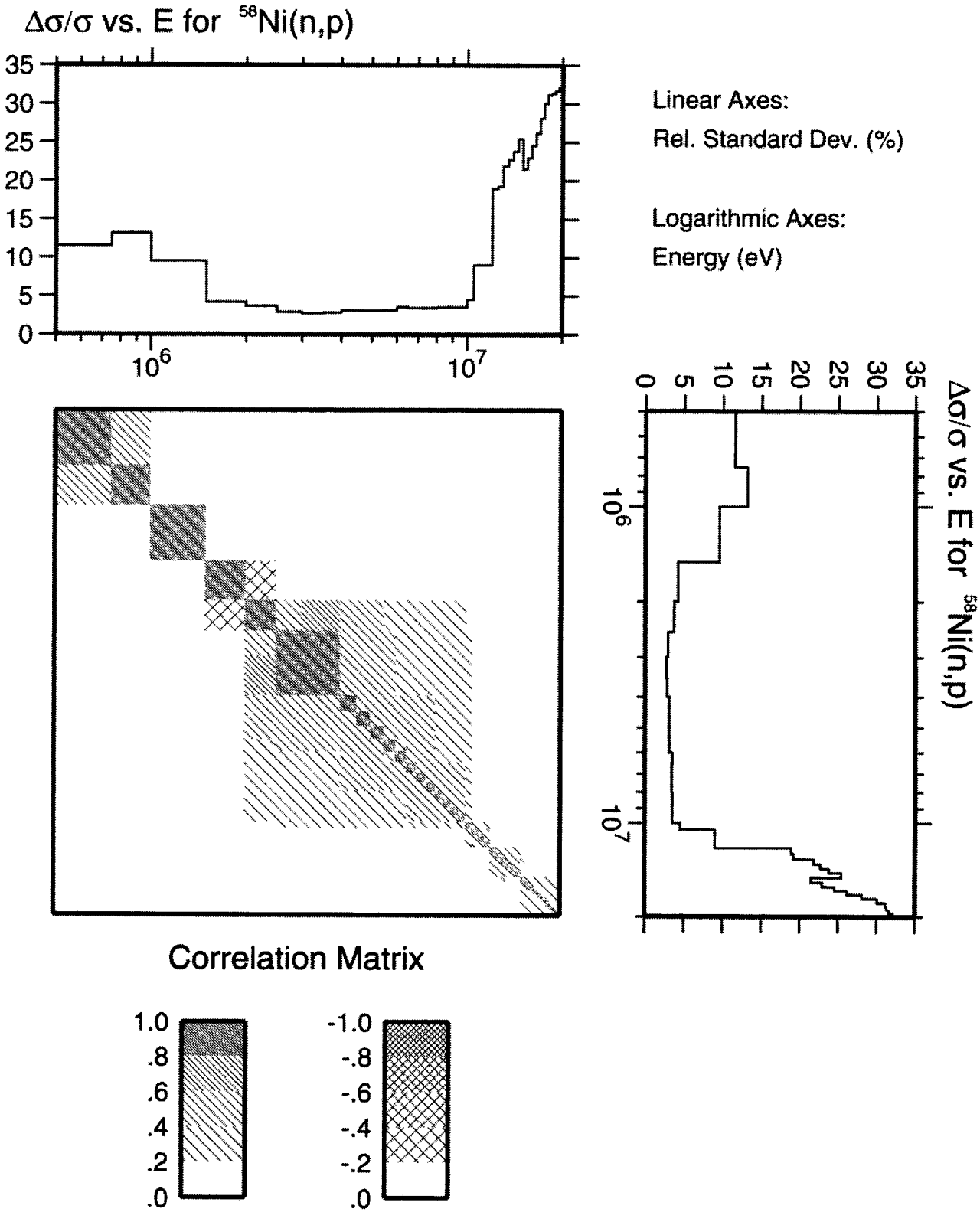

Fig. D.36. ^{58}Ni(n,p)^{58}Co (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

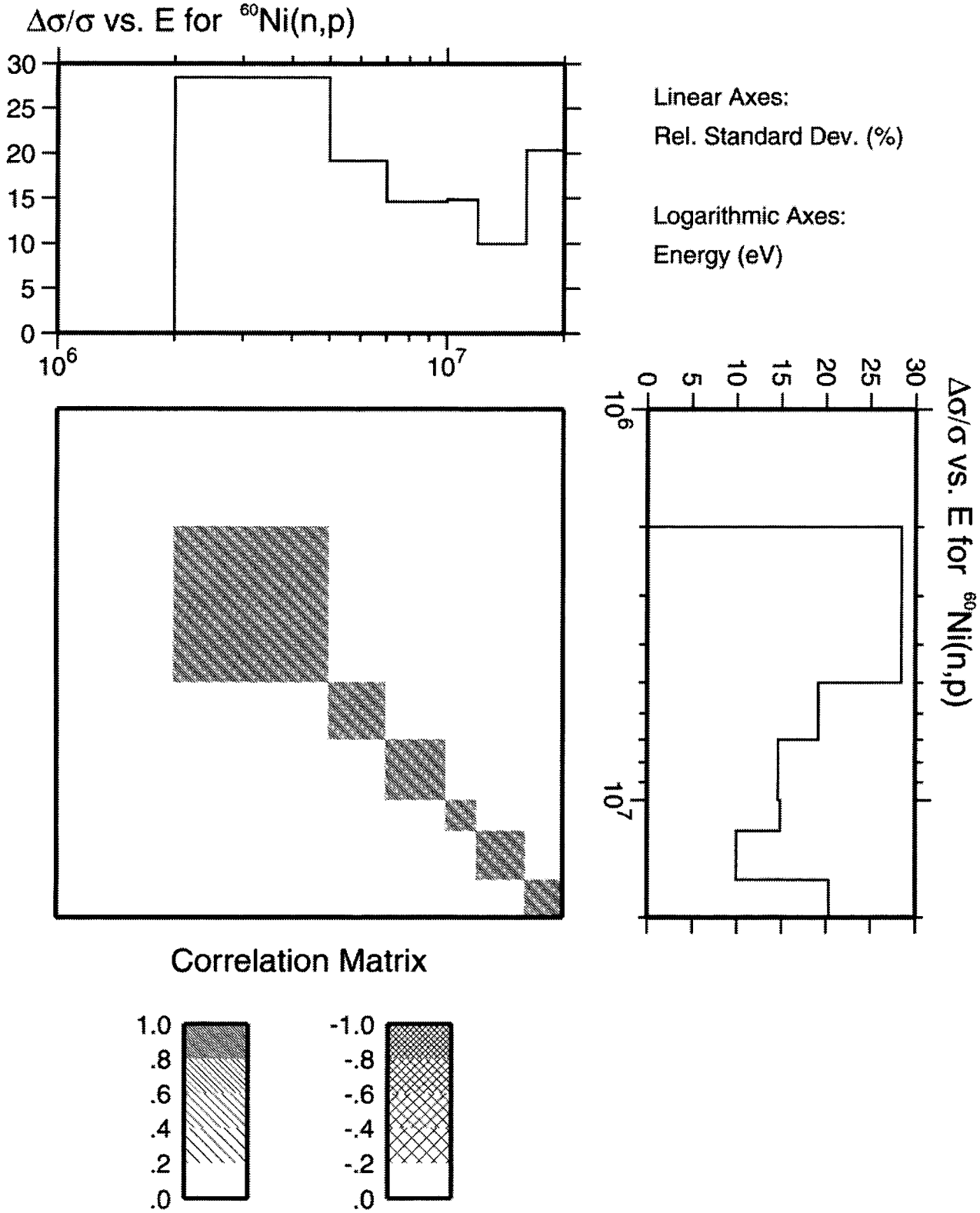

Fig. D.37. ^{60}Ni(n,p)^{60}Co (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

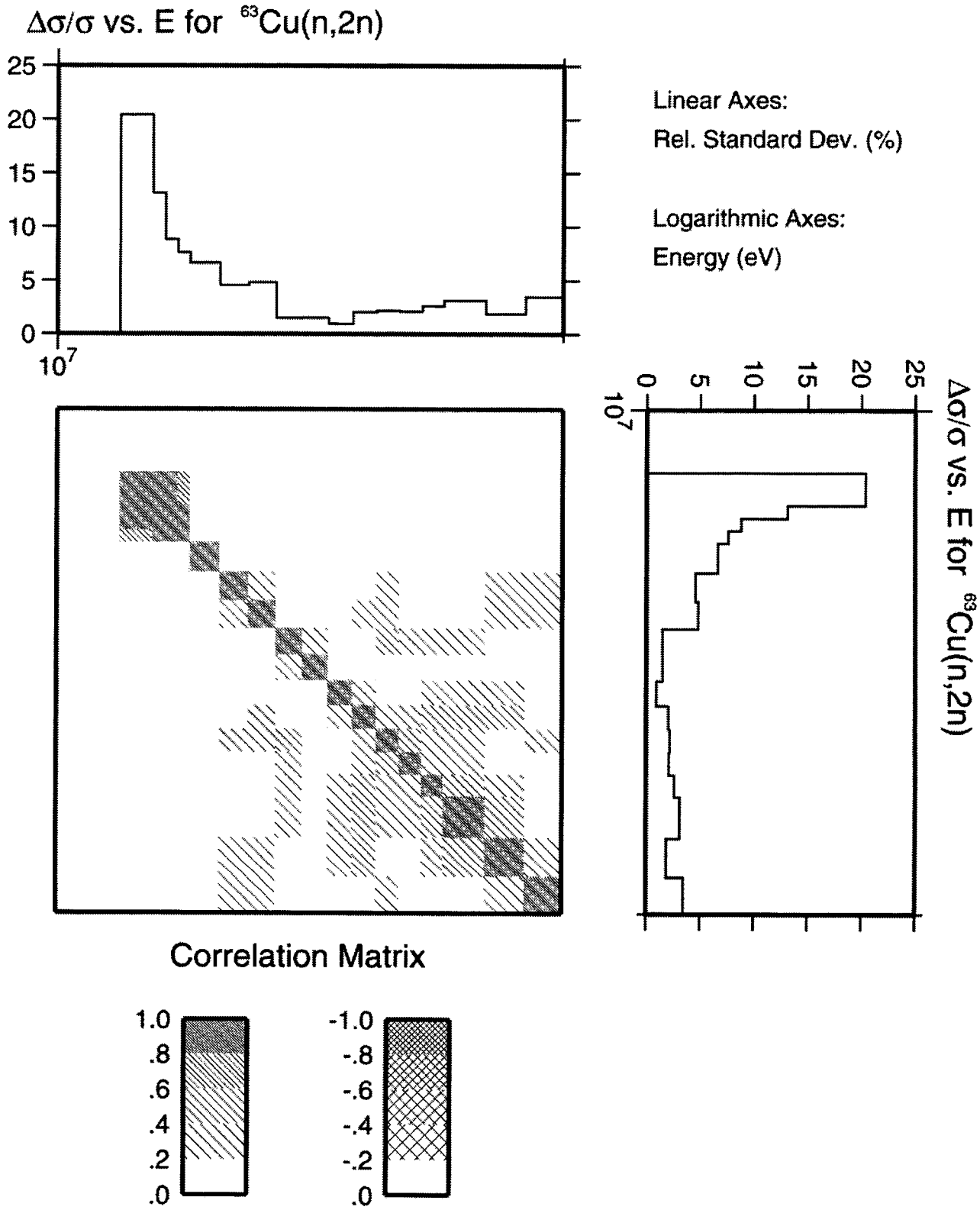

Fig. D.38. $^{63}Cu(n,2n)^{62}Cu$ (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

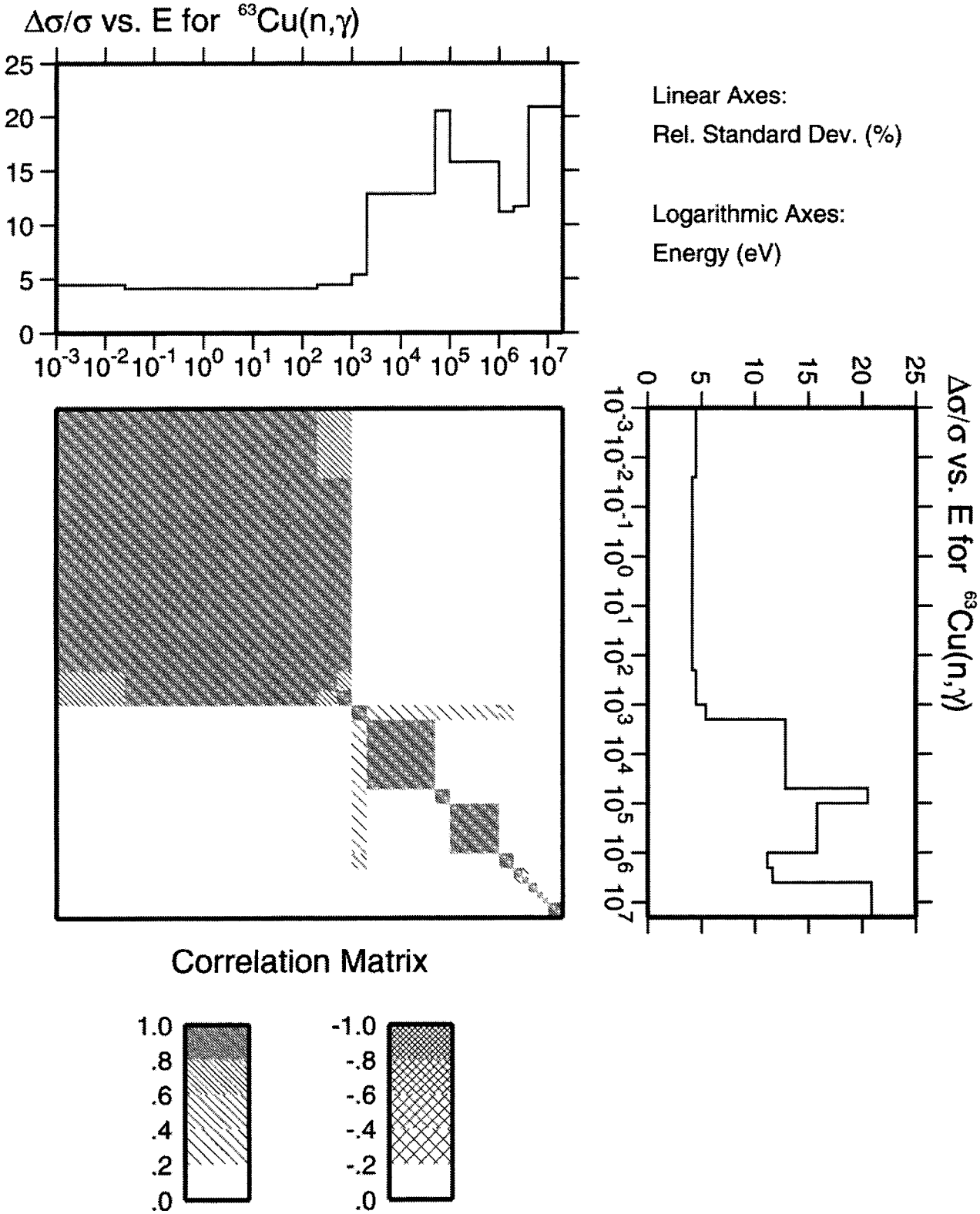

Fig. D.39. ^{63}Cu(n,γ)^{64}Cu (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

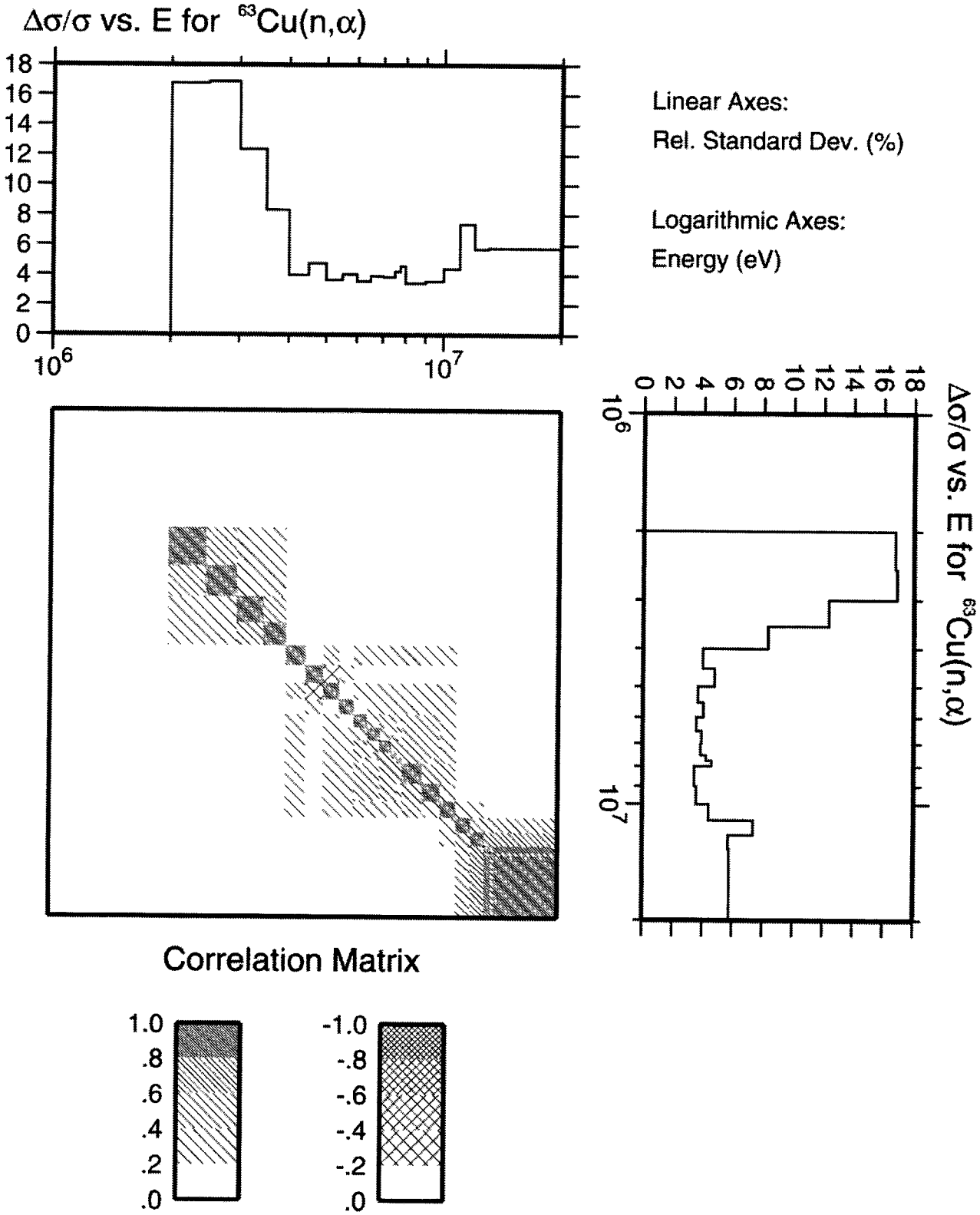

Fig. D.40. ^{63}Cu(n,α)^{60}Co (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

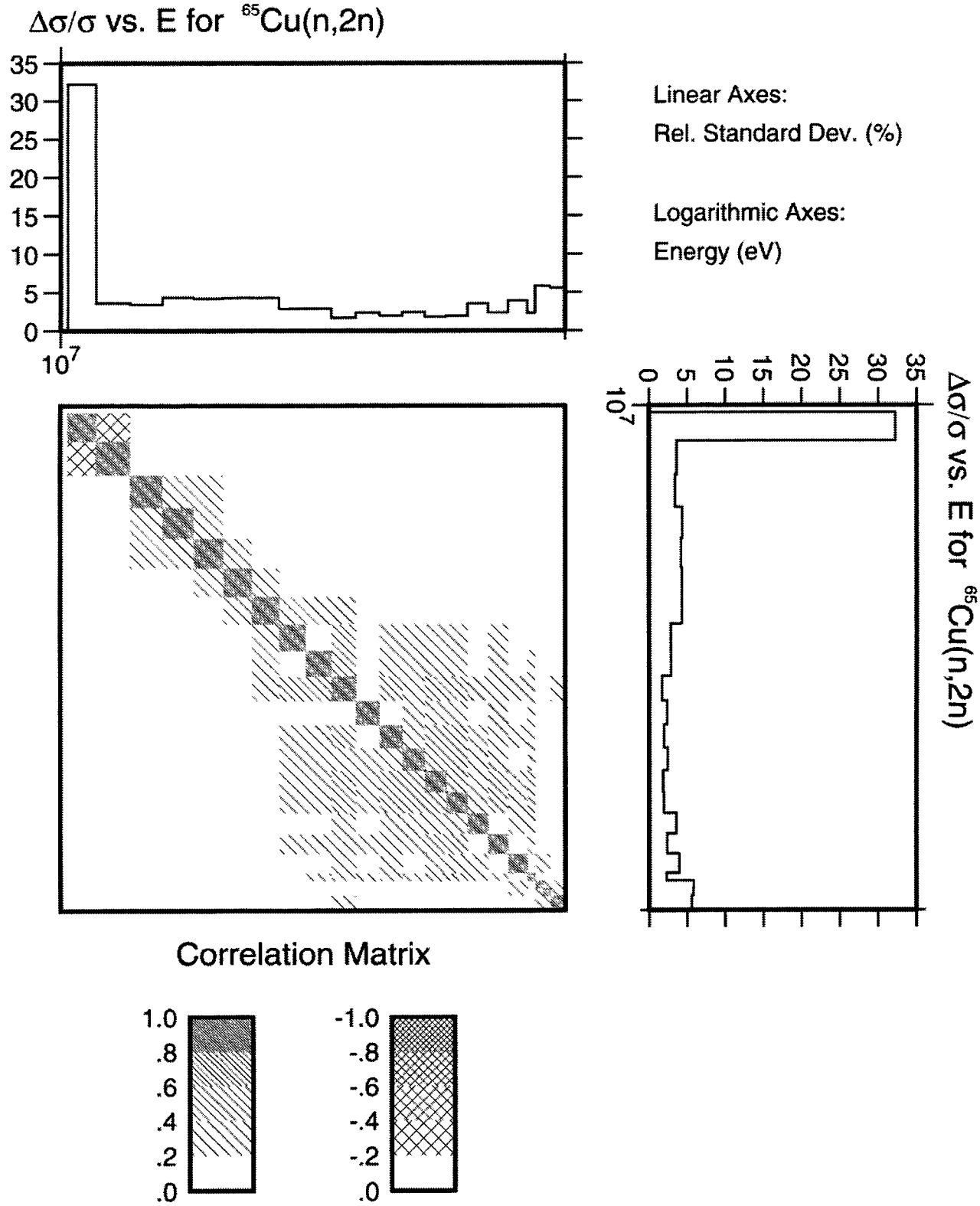

Fig. D.41. $^{65}Cu(n,2n)^{64}Cu$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

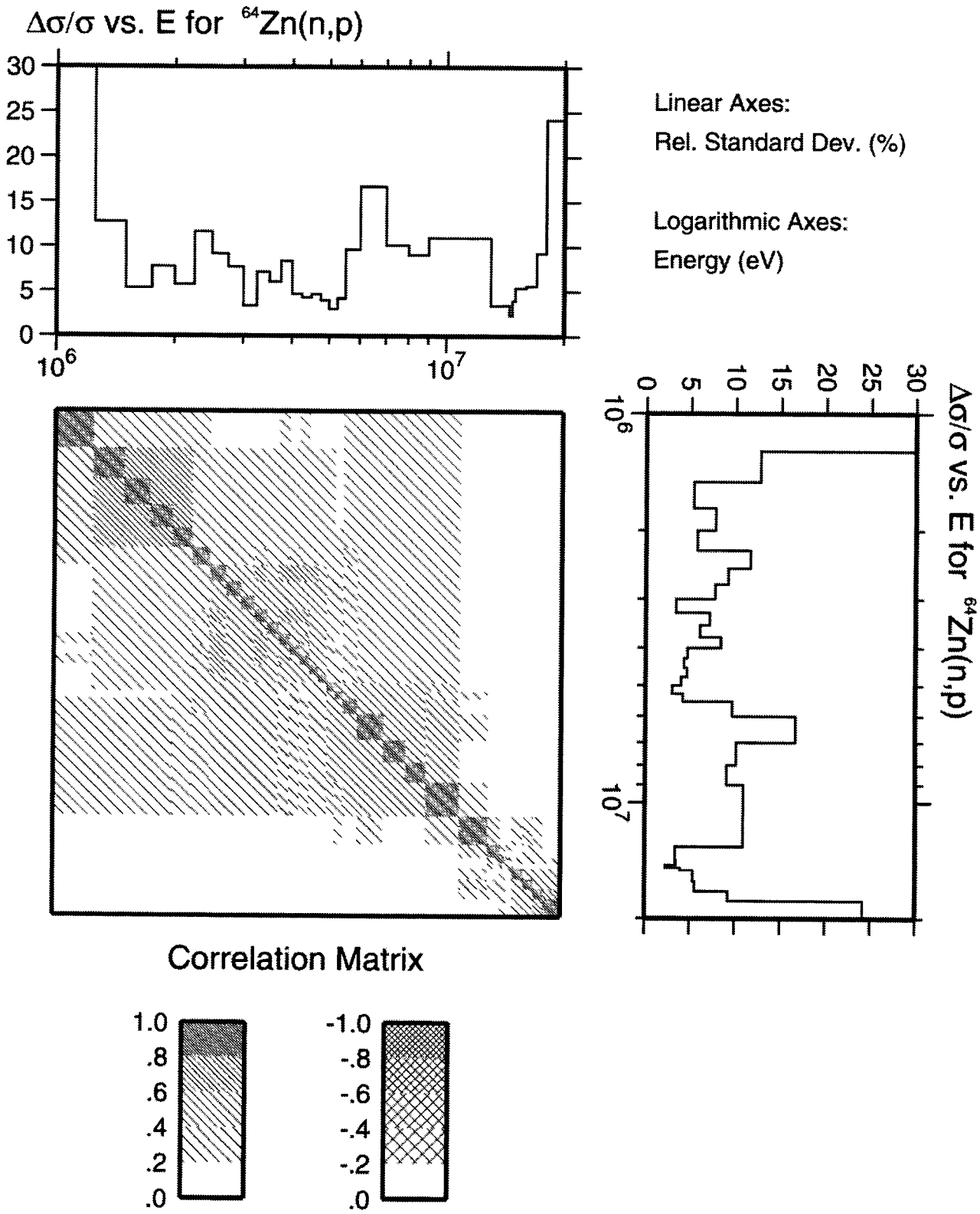

Fig. D.42. $^{64}Zn(n,p)^{64}Cu$ (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

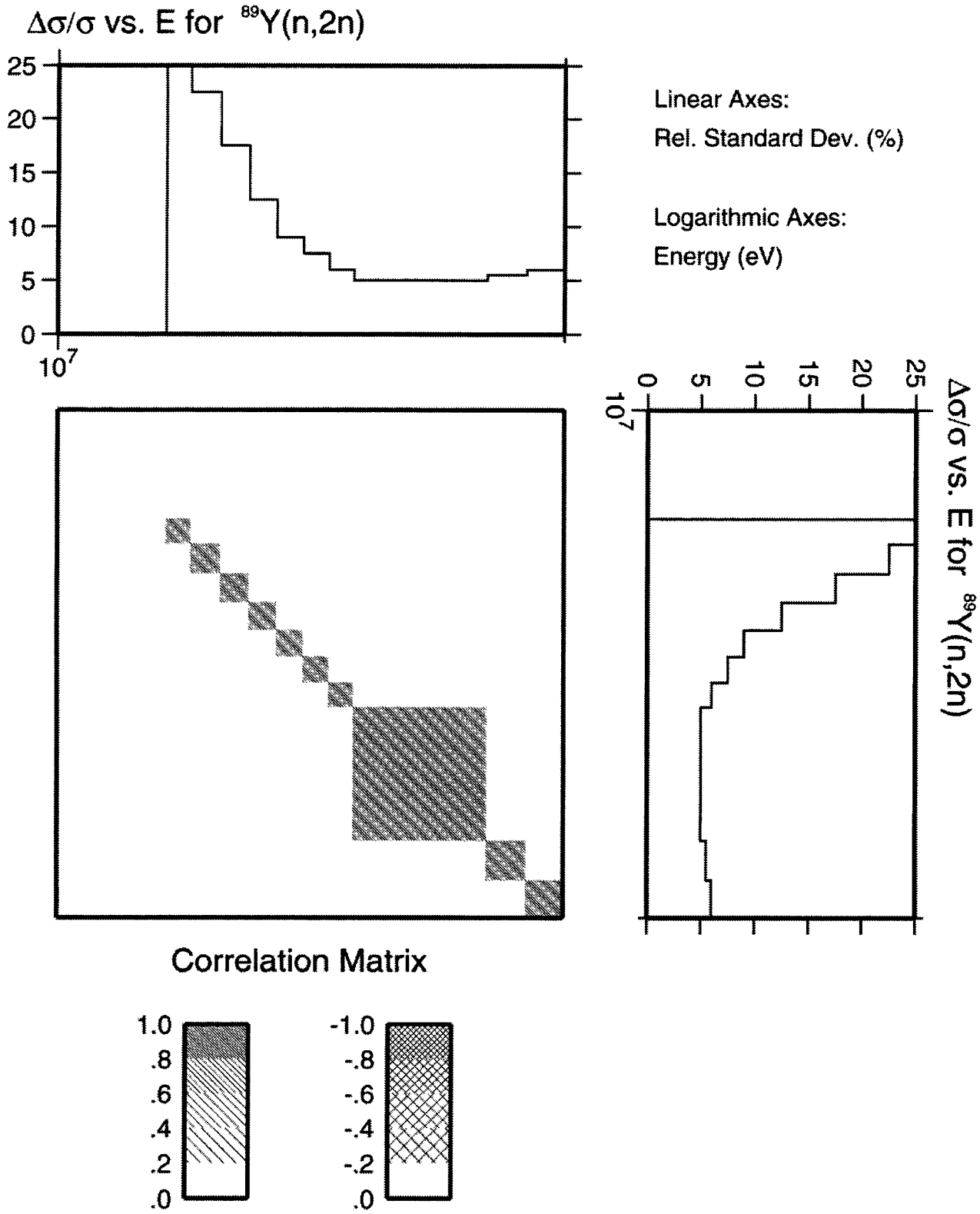

Fig. D.43. $^{89}Y(n,2n)^{88}Y$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

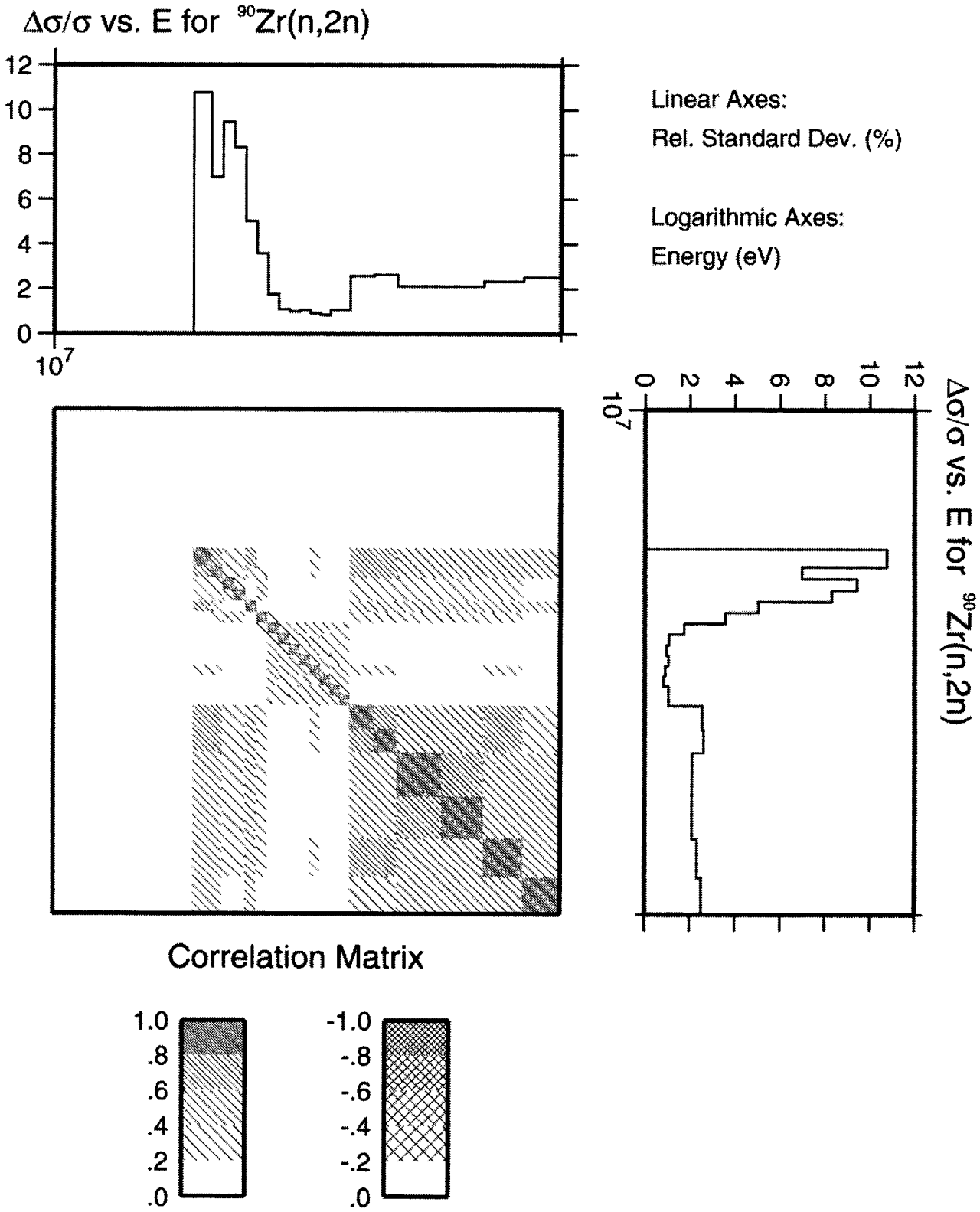

Fig. D.44. ^{90}Zr(n,2n)^{89}Zr (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

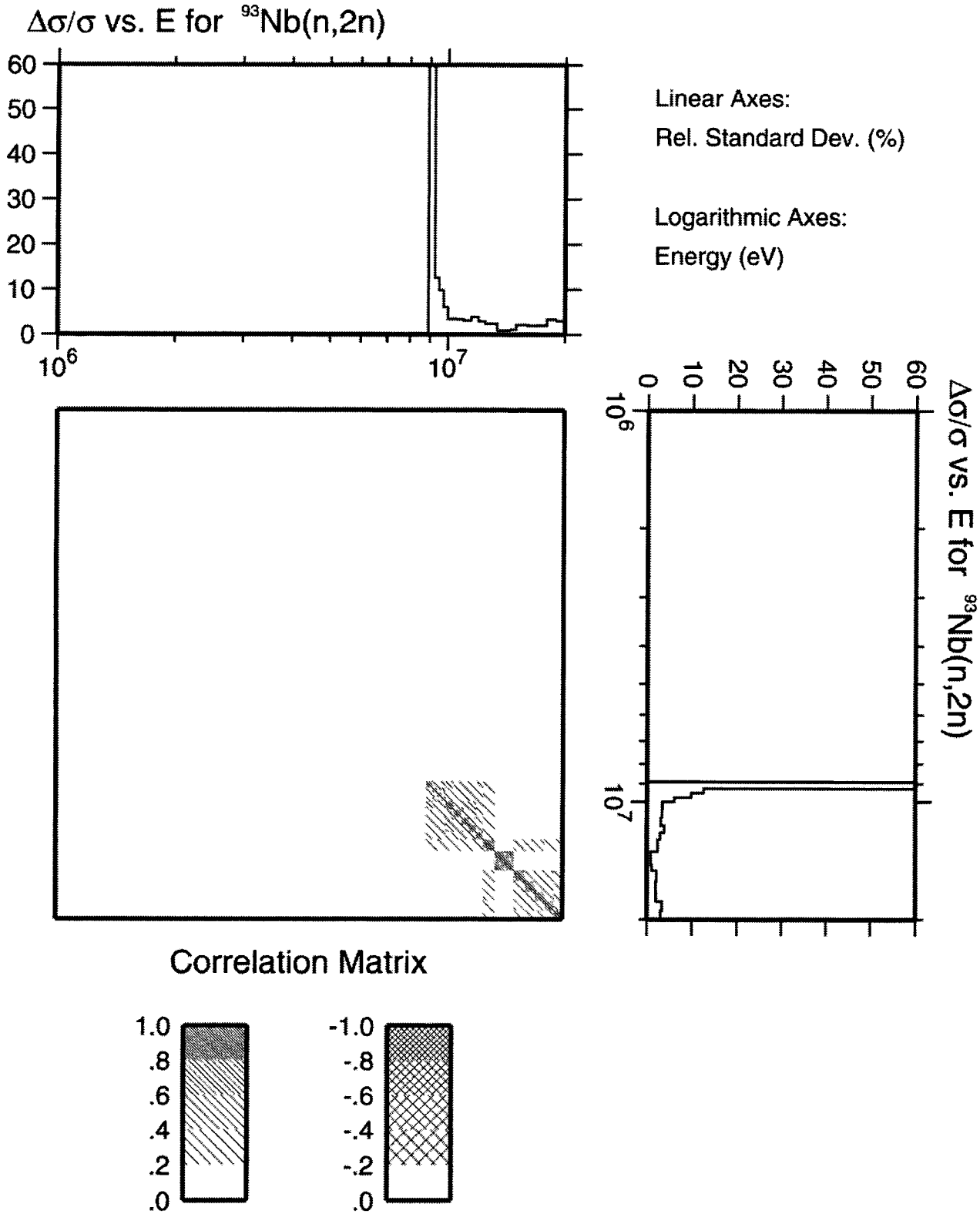

Fig. D.45. ^{93}Nb(n,2n)^{92}Nb (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

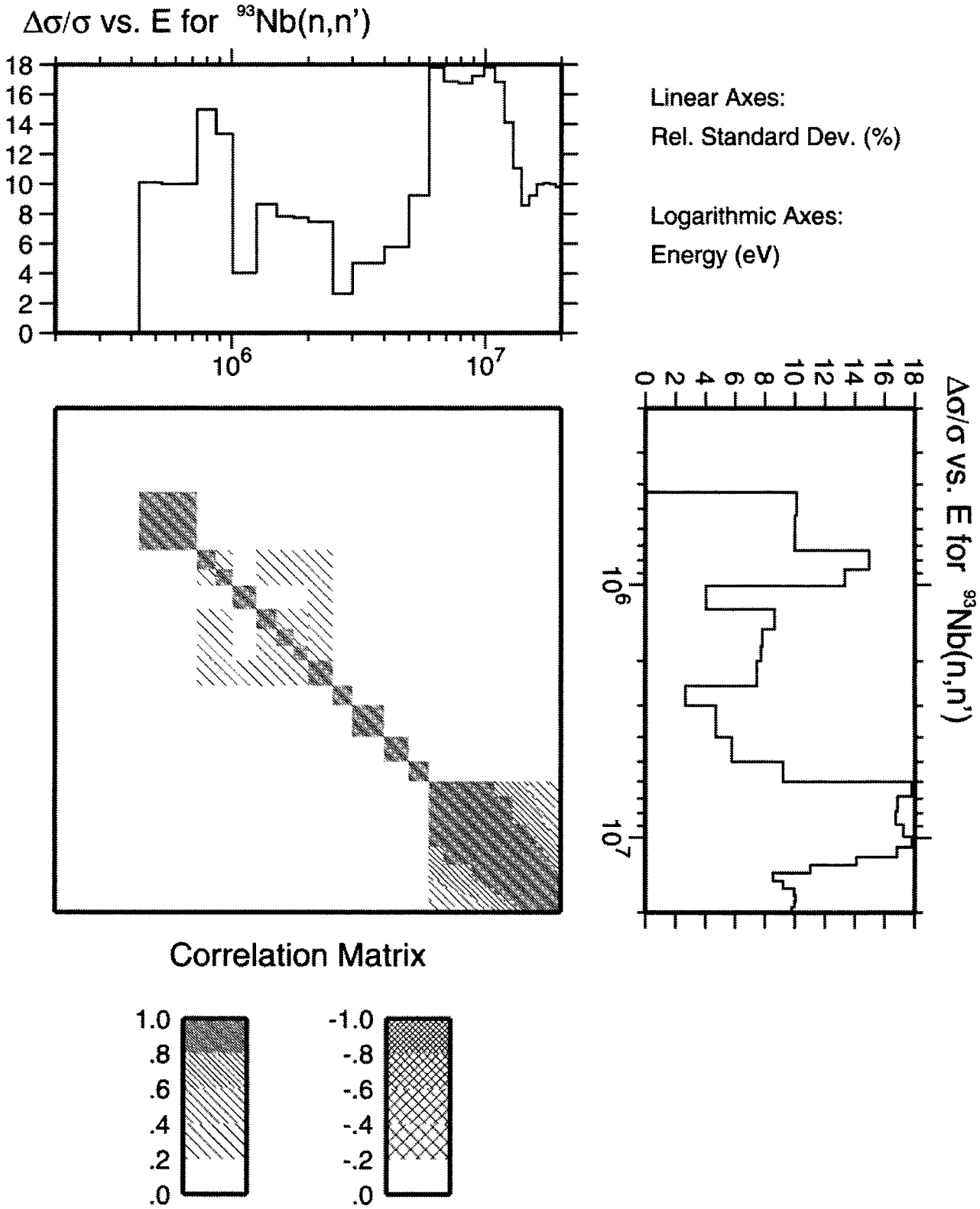

Fig. D.46. ^{93}Nb(n,n')^{93}Nb (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

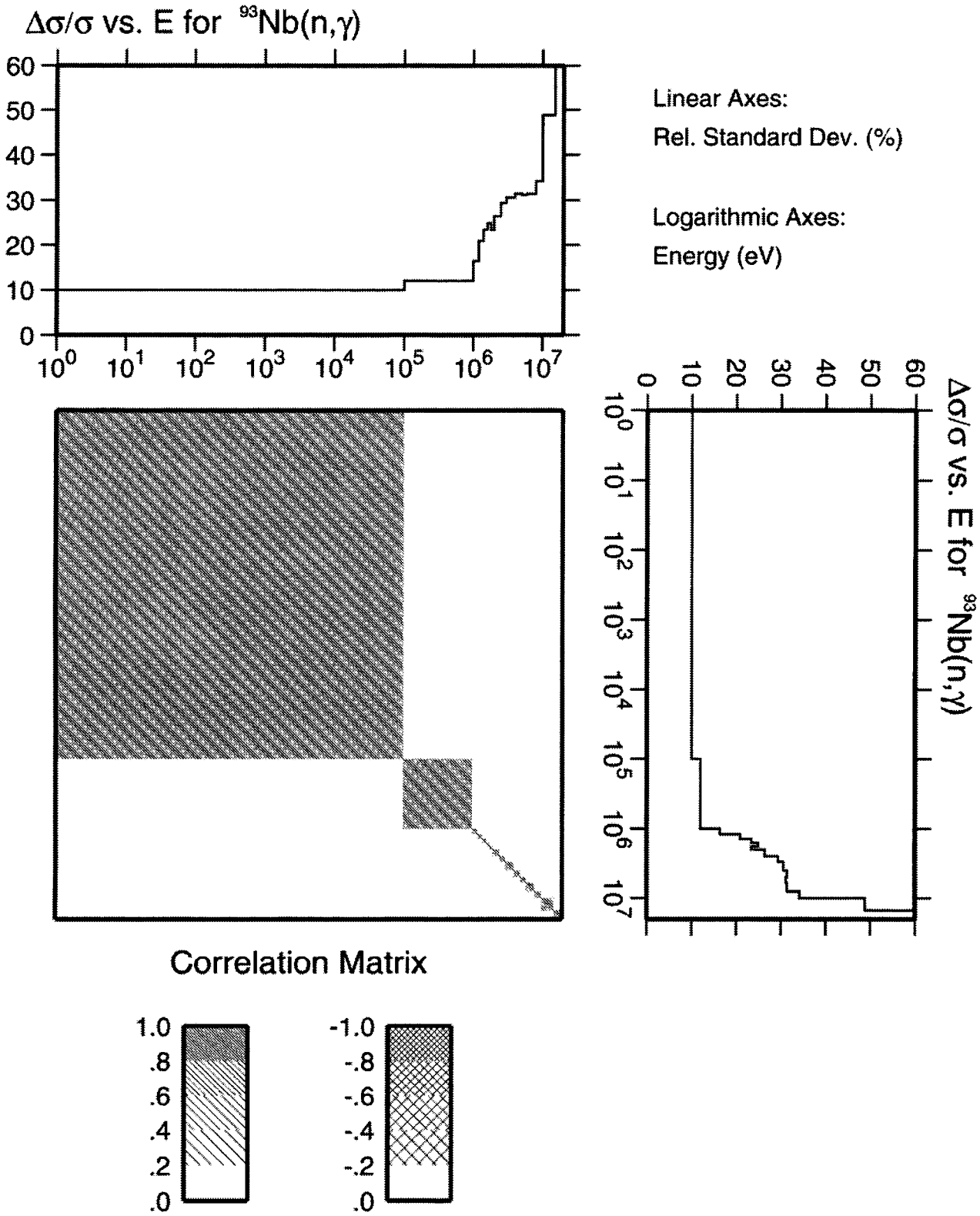

Fig. D.47. ^{93}Nb(n,γ)^{94}Nb (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

No covariance data available for this reaction

Fig. D.48. $^{98}Mo(n,\gamma)^{99}Mo$ (SNLRML-V1, JENDL-3) correlation matrix and standard deviation. (No covariance data available.)

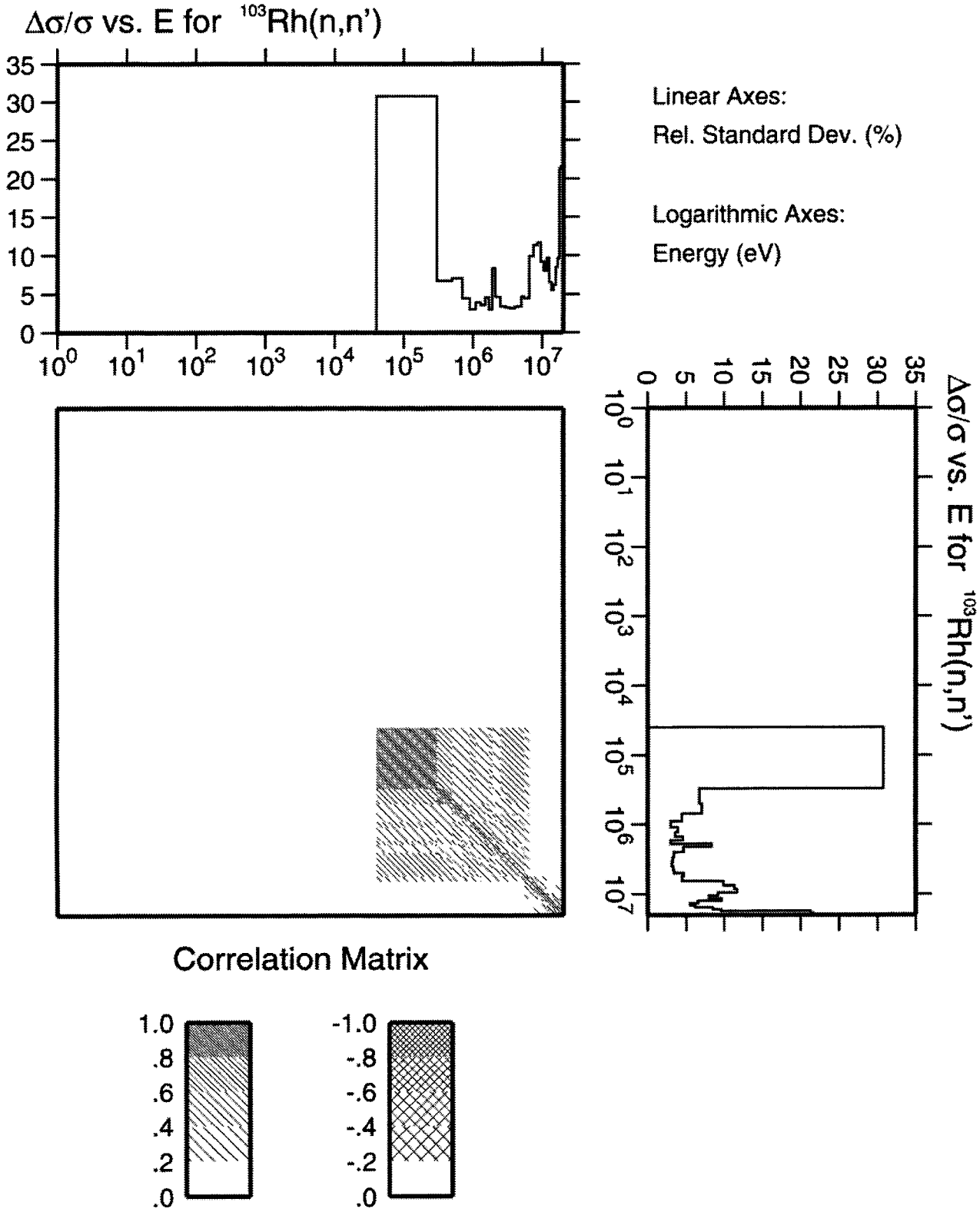

Fig. D.49. ^{103}Rh(n,n')^{103}Rh (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

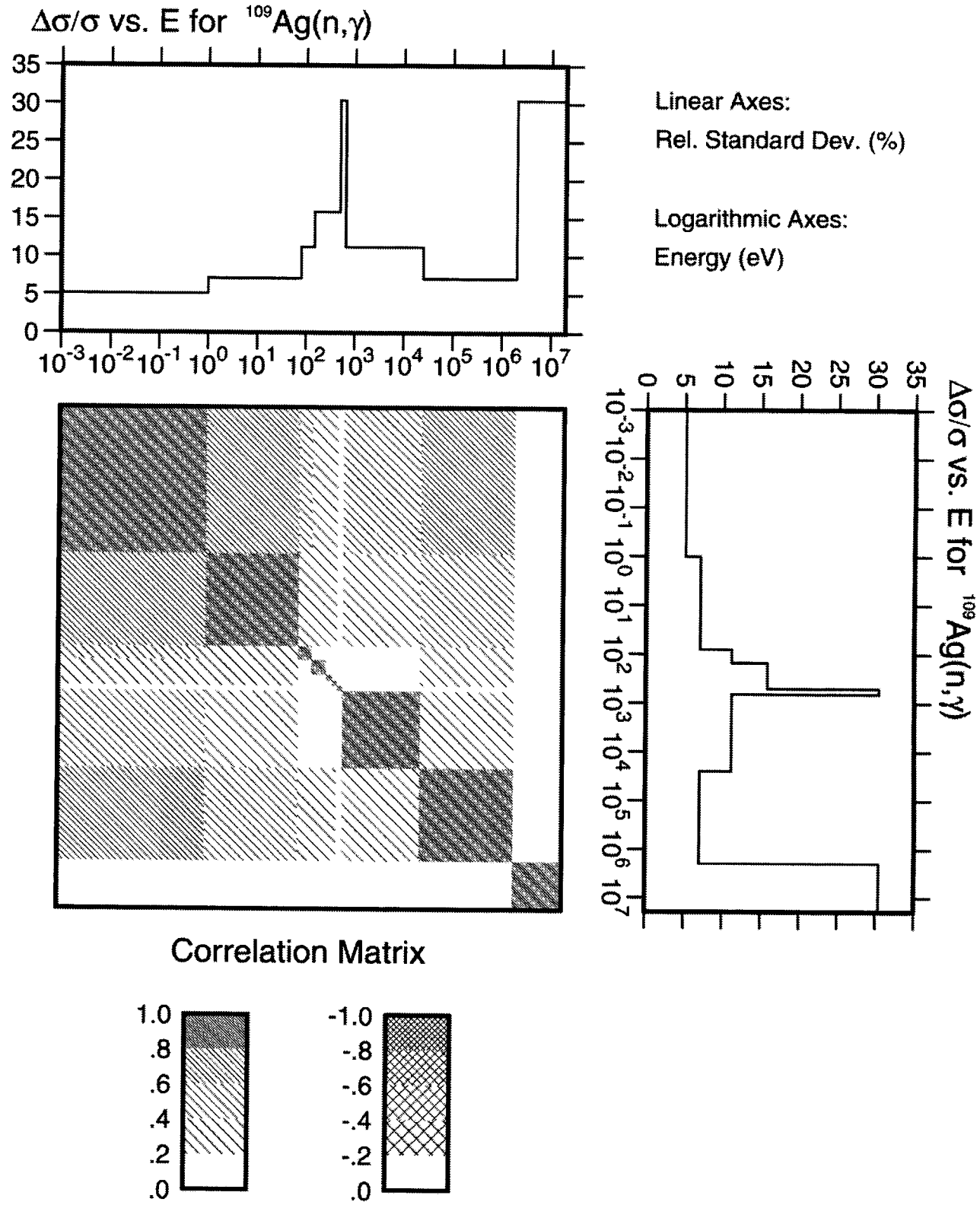

Fig. D.50. $^{109}Ag(n,\gamma)^{110}Ag$ **(IRDF-90/G-V2, private communication) correlation matrix and standard deviation.**

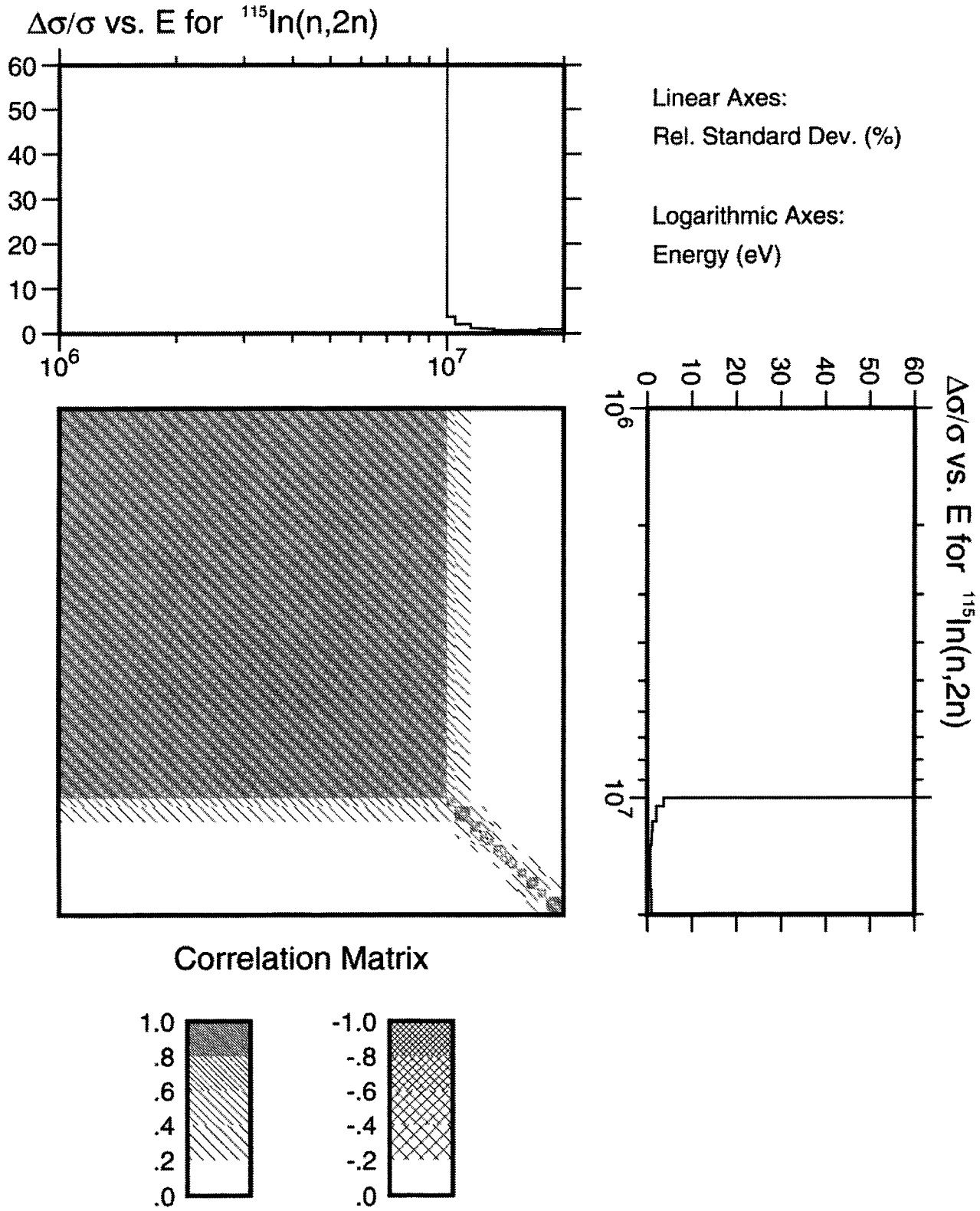

Fig. D.51. ^{115}In(n,2n)^{114}In (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

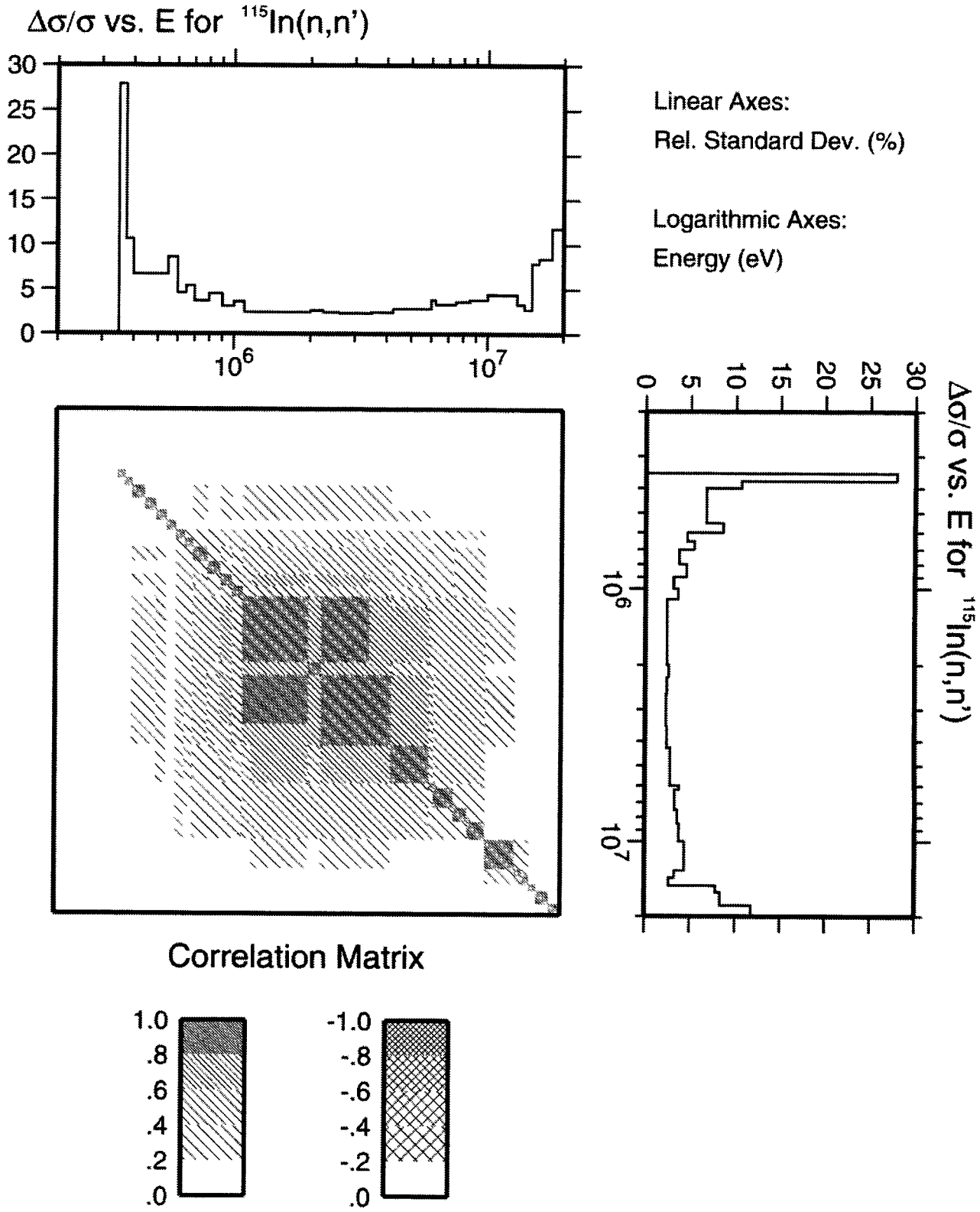

Fig. D.52. ^{115}In(n,n')^{115}In (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

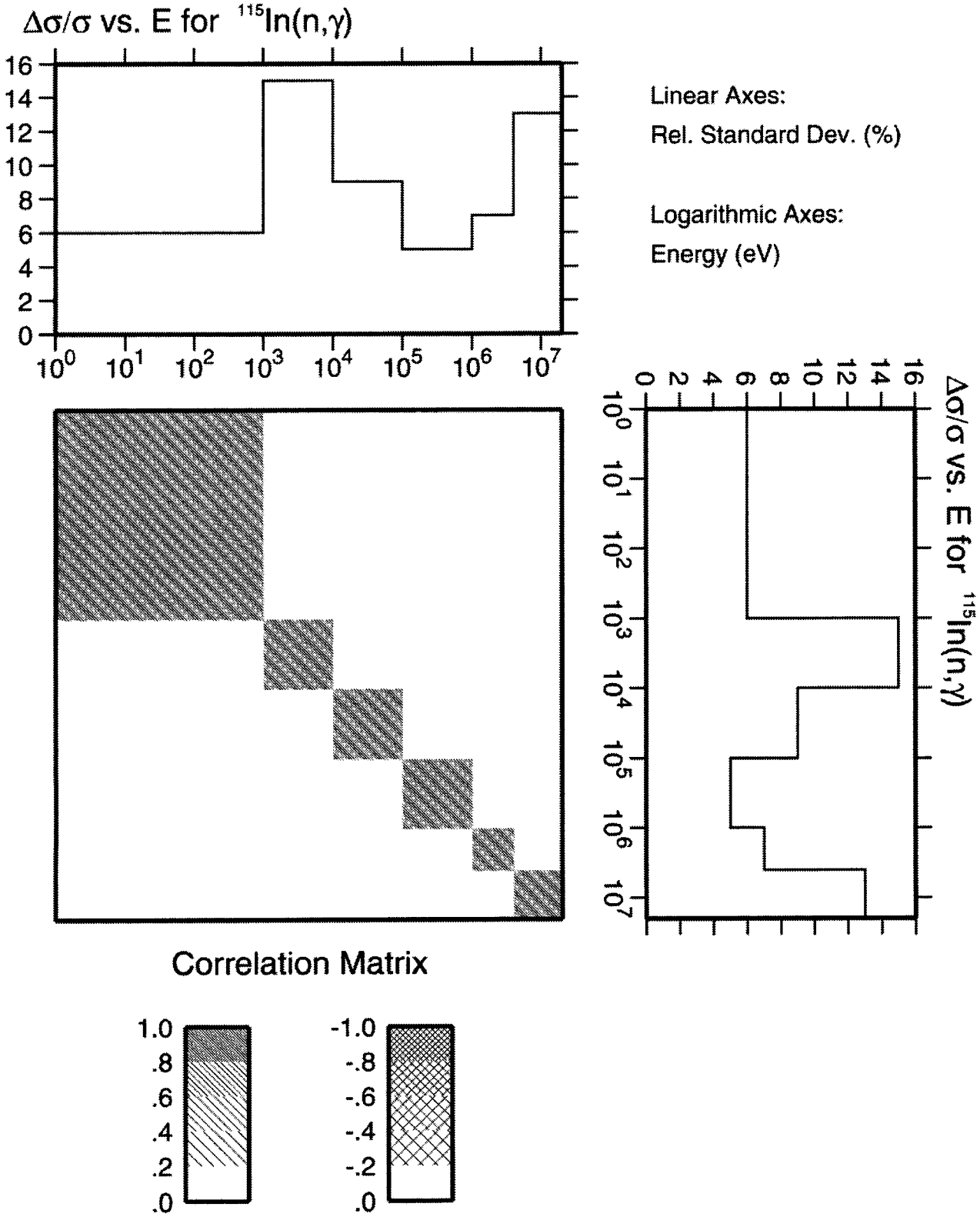

Fig. D.53. ^{115}In(n,γ)^{116}In (IRDF-90/G-V2, private communication) correlation matrix and standard deviation.

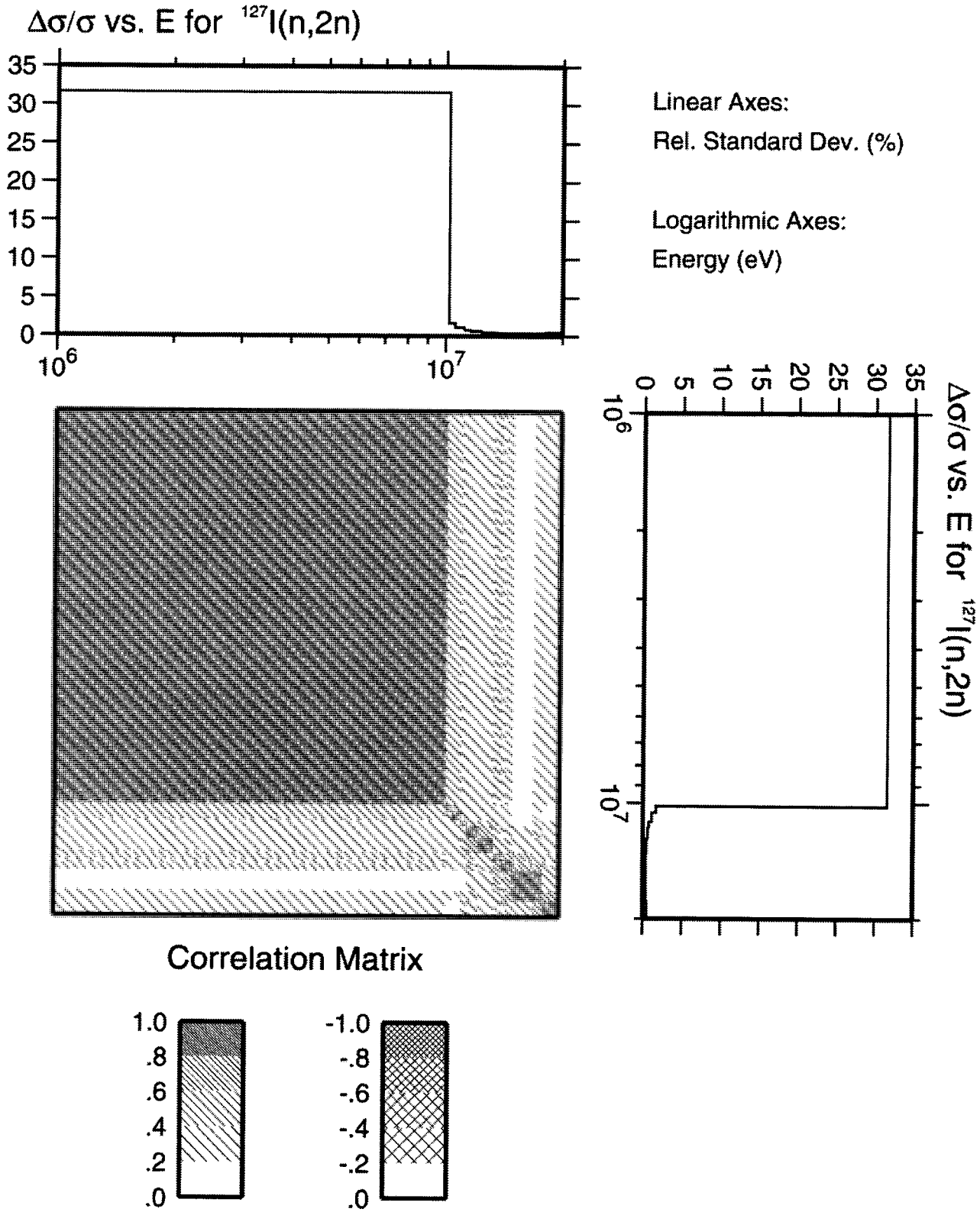

Fig. D.54. $^{127}I(n,2n)^{126}I$ **(IRDF-90/G-V2, private communication) correlation matrix and standard deviation.**

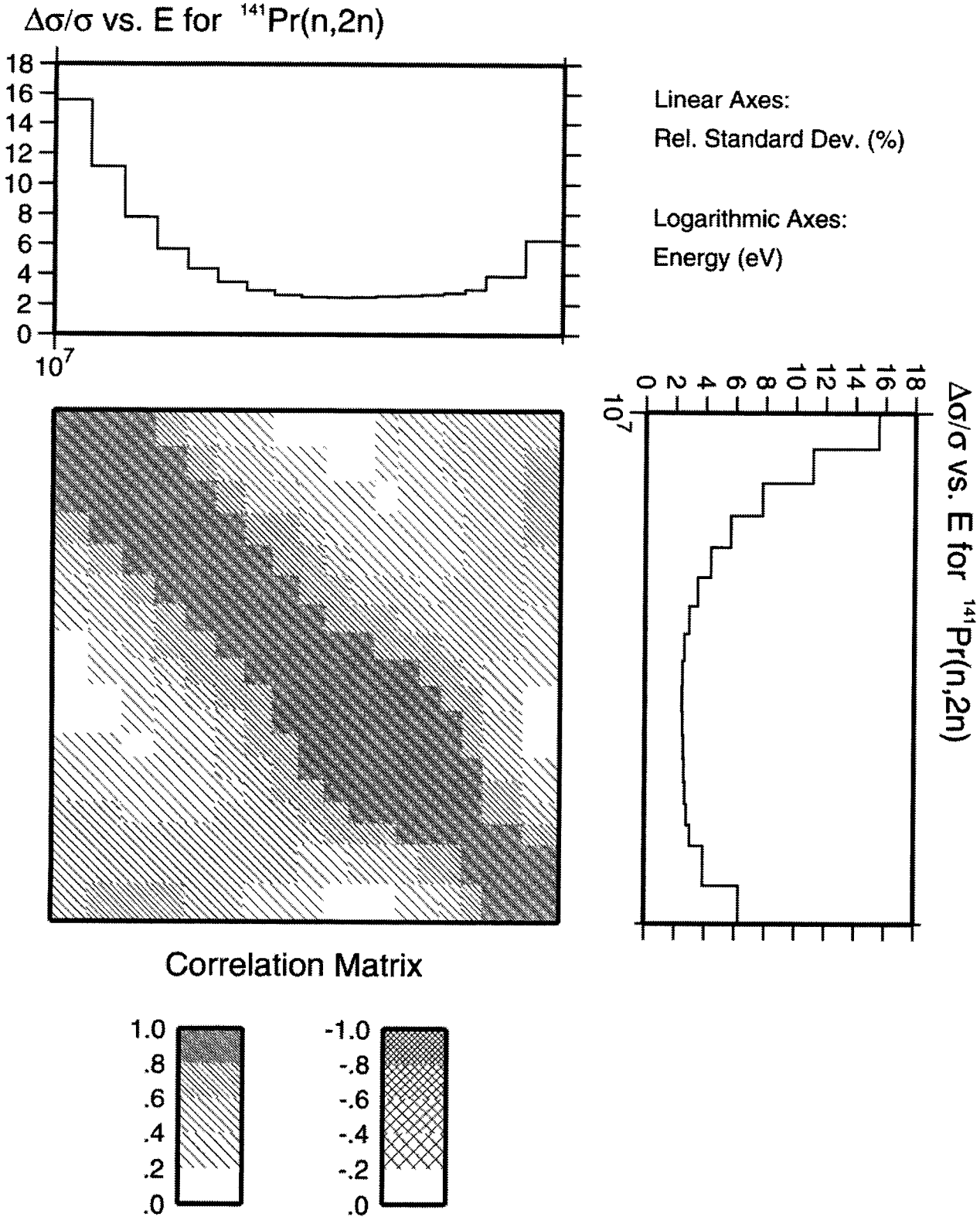

Fig. D.55. ^{141}Pr(n,2n)^{140}Pr (RDF-94, private communication) correlation matrix and standard deviation.

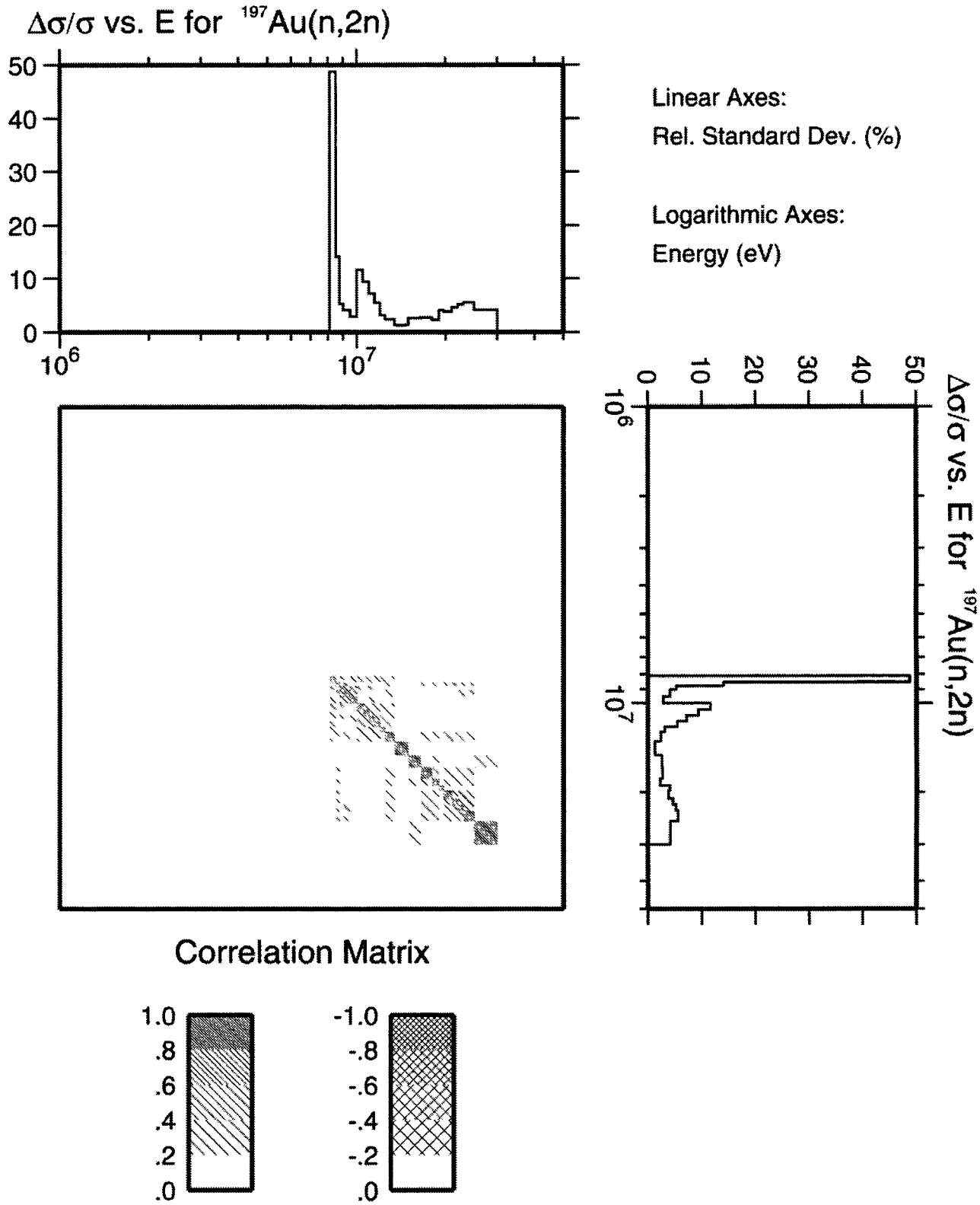

Fig. D.56. ^{197}Au(n,2n)^{196}Au (IRDF-90/G-V2, original evaluation) correlation matrix and standard deviation.

No covariance data available for this reaction

Fig. D.57. ^{197}Au(n,3n)^{195}Au (SNLRML-V1, ENDF/B-VI), correlation matrix and standard deviation. (No covariance data available.)

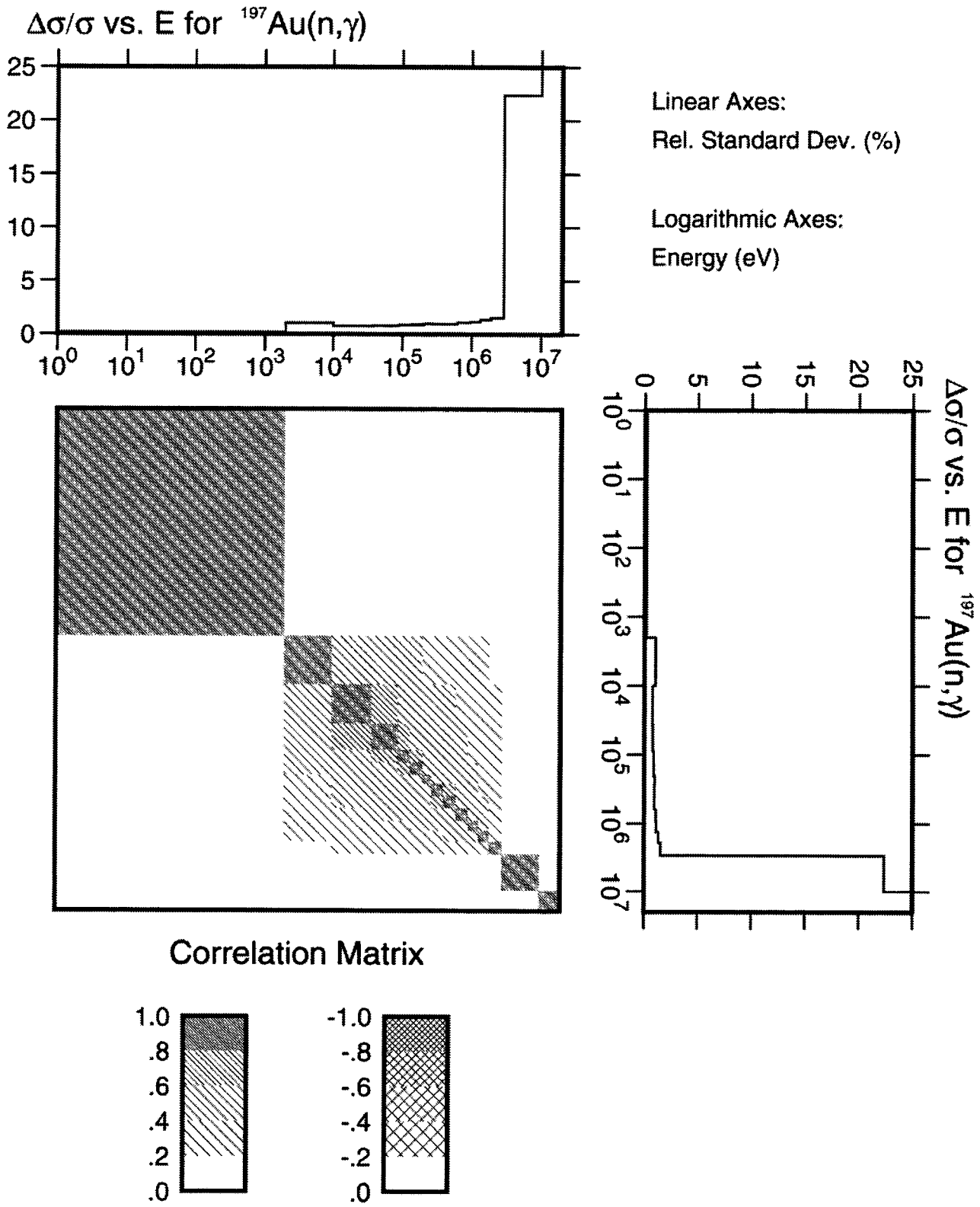

Fig. D.58. $^{197}Au(n,\gamma)^{198}Au$ **(IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.**

No covariance data available for this reaction

Fig. D.59. $^{197}Au(n,p)^{197}Pt$ (SNLRML-V1, ENDF/B-VI), correlation matrix and standard deviation. (No covariance data available.)

No covariance data available for this reaction

Fig. D.60. $^{232}Th(n,2n)^{231}Th$ (SNLRML-V1, ENDF/B-VI), correlation matrix and standard deviation. (No covariance data available.)

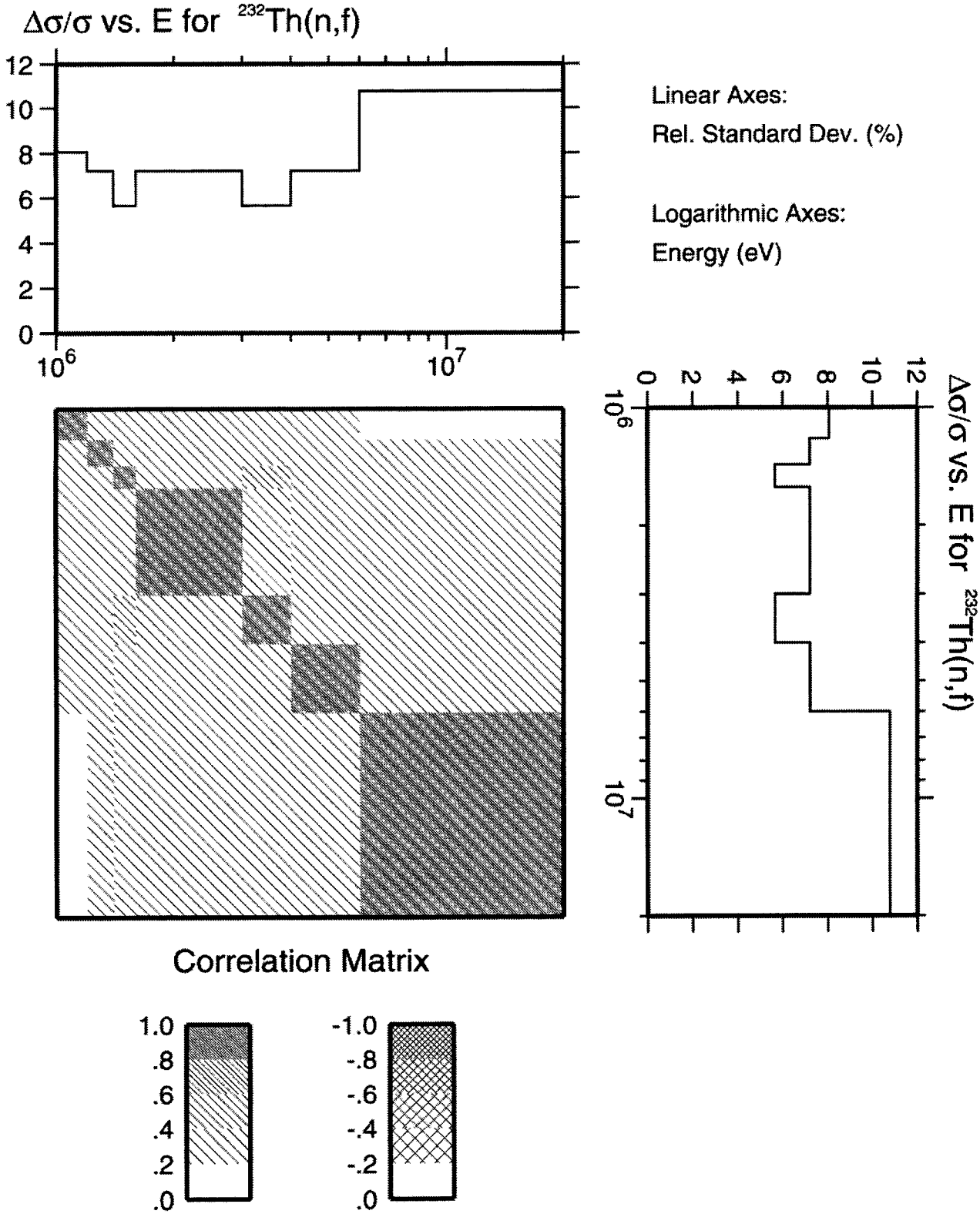

Fig. D.61. ^{232}**Th(*n,f*) (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.**

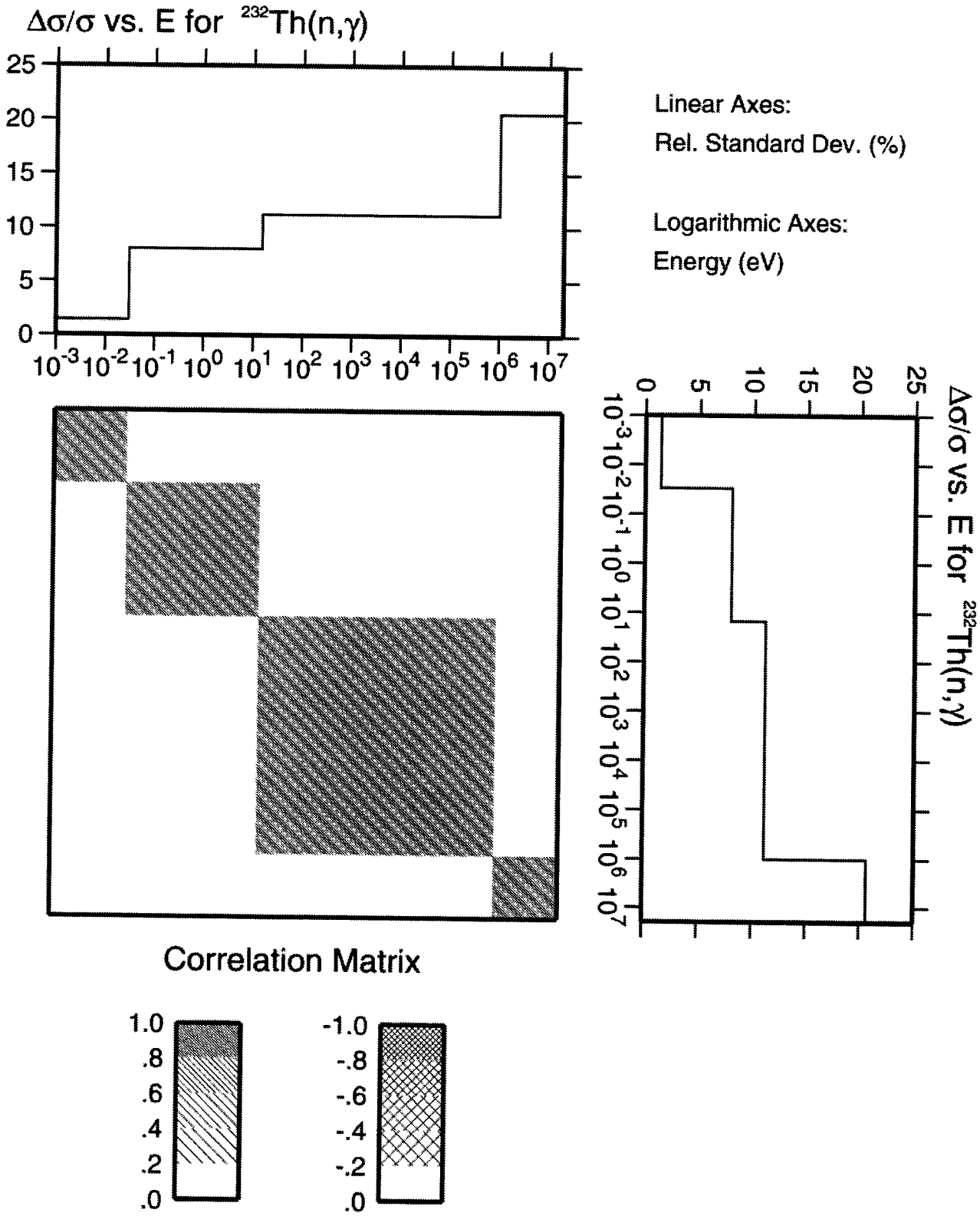

Fig. D.62. ^{232}Th$(n,\gamma)^{233}$Th (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

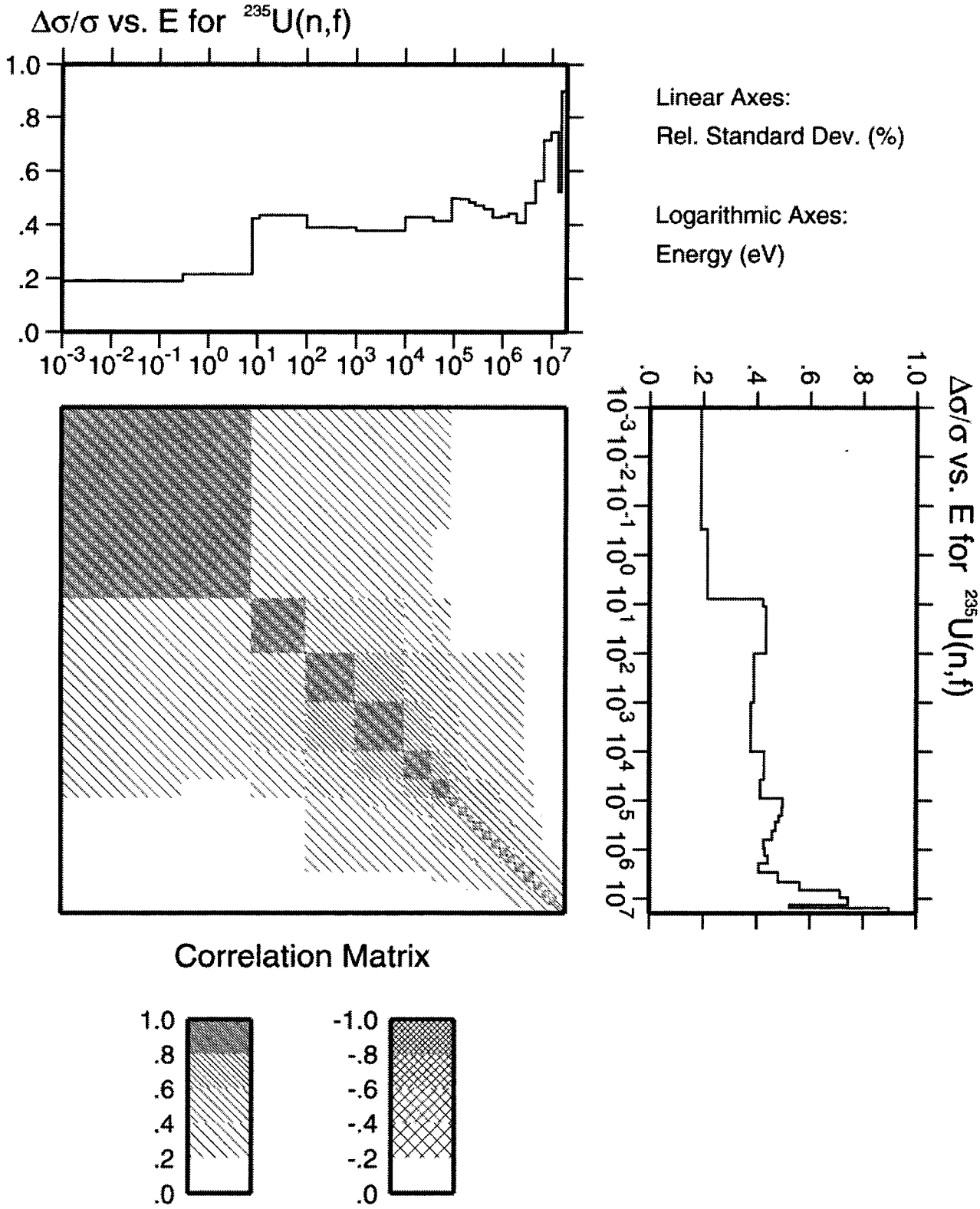

Fig. D.63. $^{235}U(n,f)$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

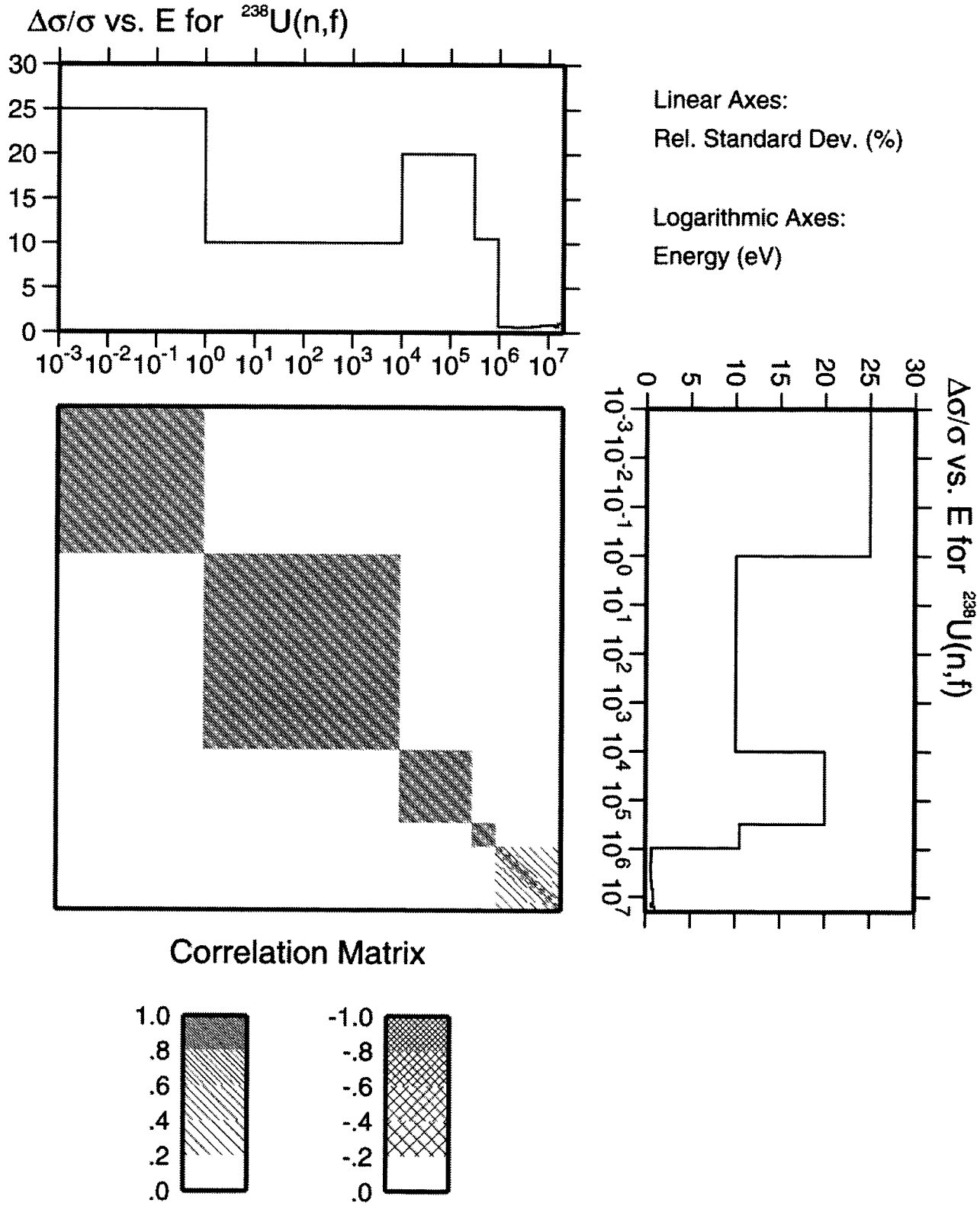

Fig. D.64. $^{238}U(n,f)$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

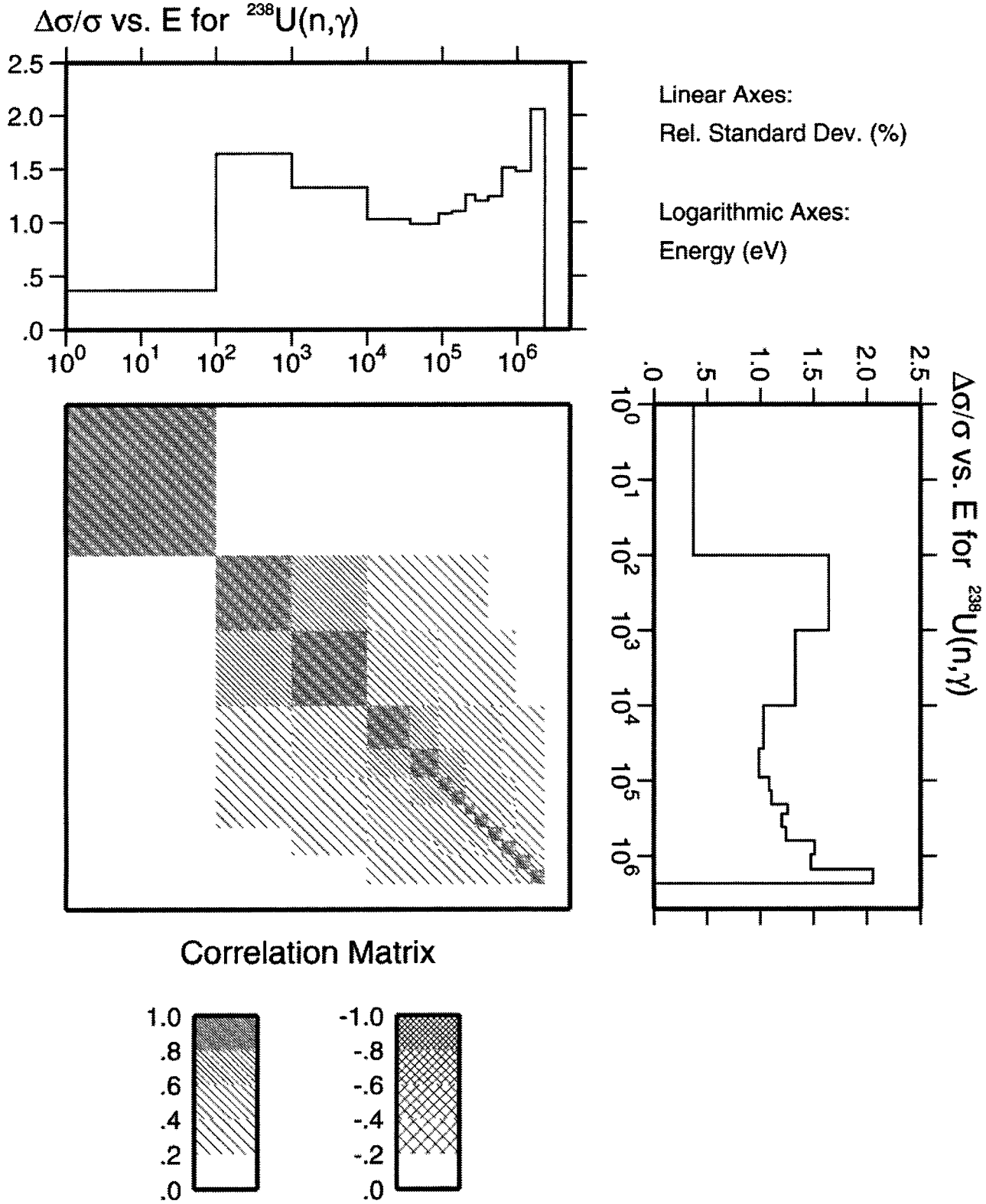

Fig. D.65. $^{232}U(n,\gamma)^{239}U$ (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

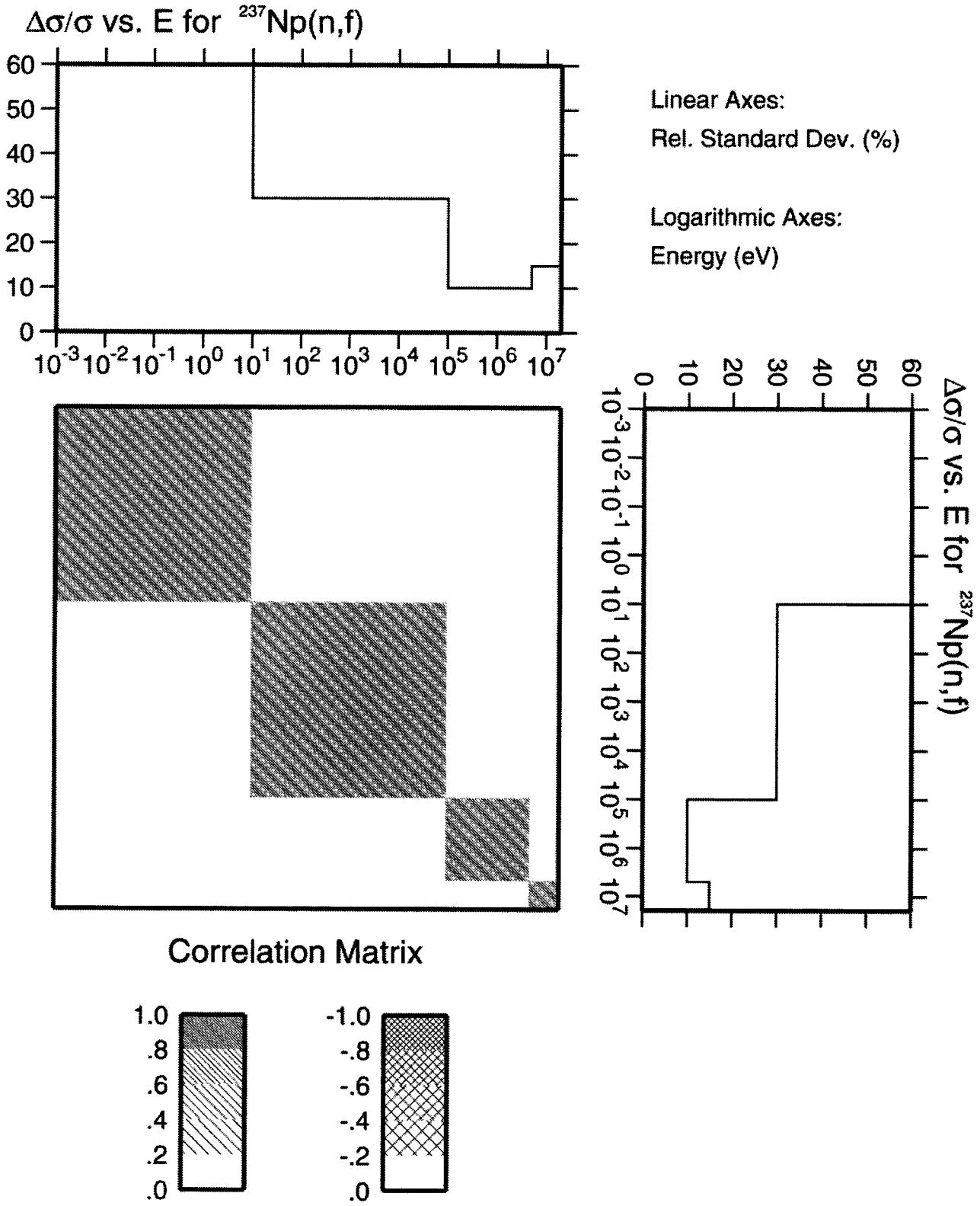

Fig. D.66. ^{237}Np(*n,f*) (SNLRML-V1, ENDF/B-VI) correlation matrix and standard deviation.

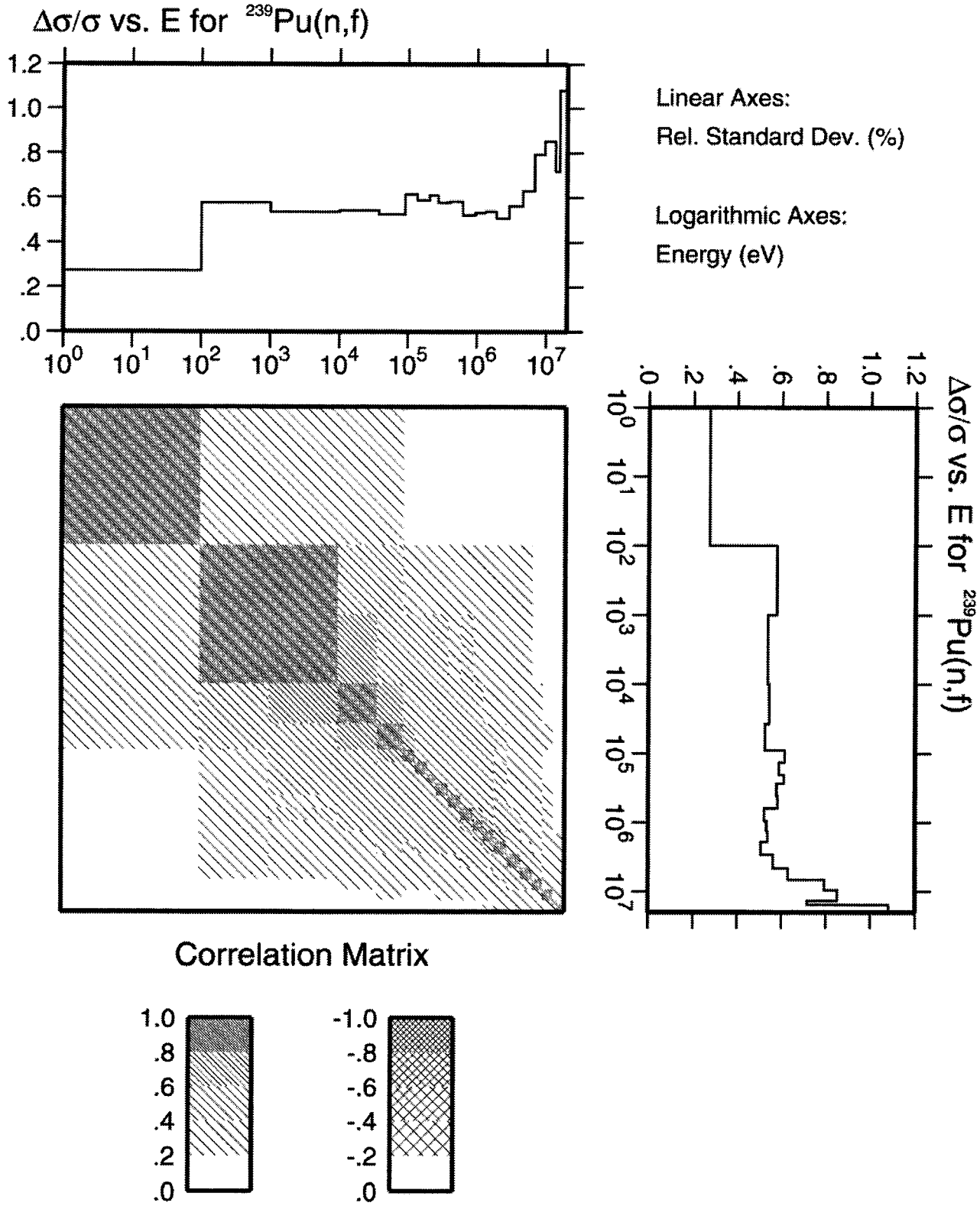

Fig. D.67. ^{239}Pu(*n,f*) (IRDF-90/G-V2, ENDF/B-VI) correlation matrix and standard deviation.

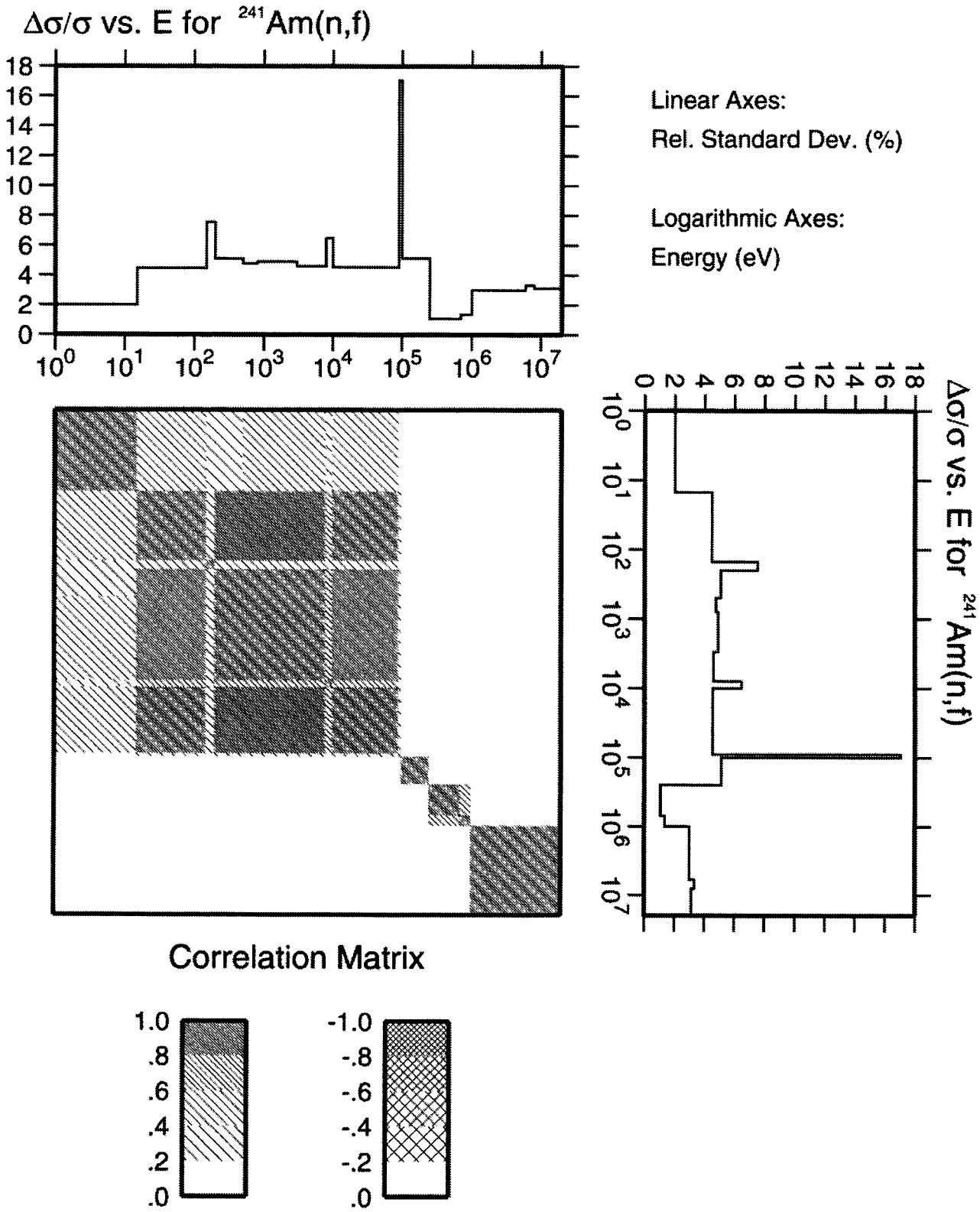

Fig. D.68. ^{241}Am(*n,f*) (SNLRML-V1, JENDL-3D) correlation matrix and standard deviation.

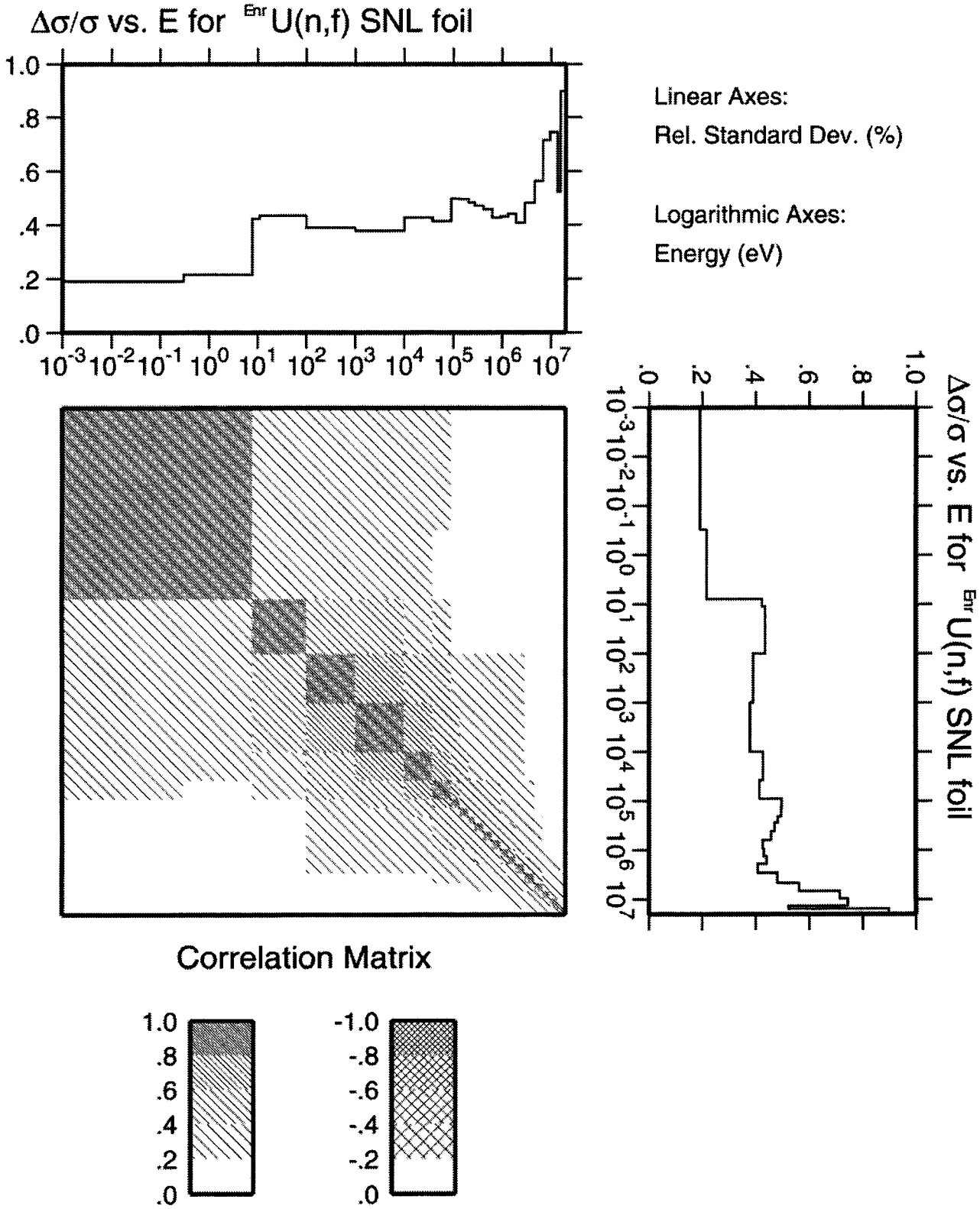

Fig. D.69. Enriched uranium fission foil [SNLRML foil, SNLRML-V1, IRDF-90, covariance for $^{235}U(n,f)$] correlation matrix and standard deviation.

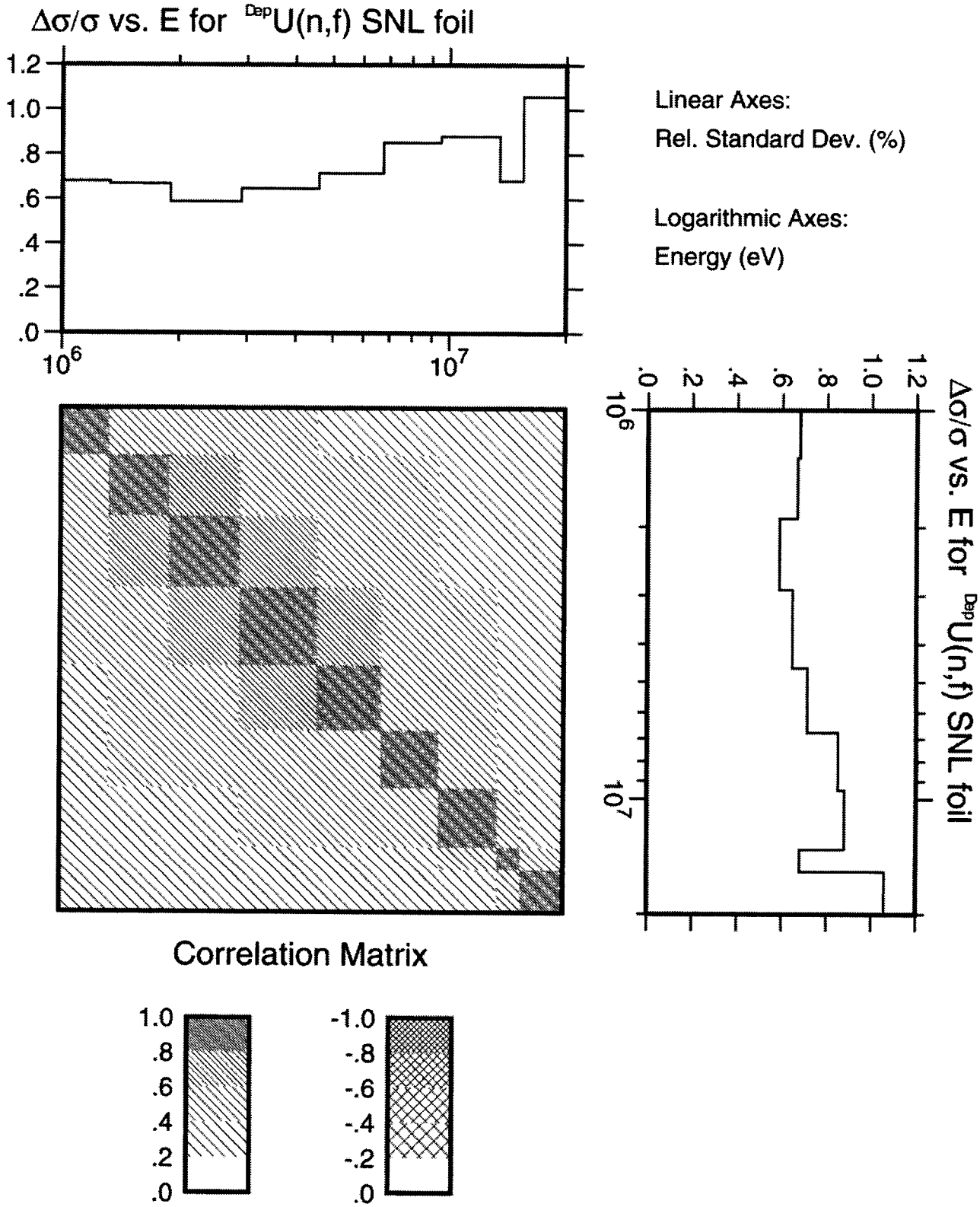

Fig. D.70. Depleted uranium fission foil [SNLRML foil, SNLRML-V1, IRDF-90, covariance for $^{238}U(n,f)$] correlation matrix and standard deviation.

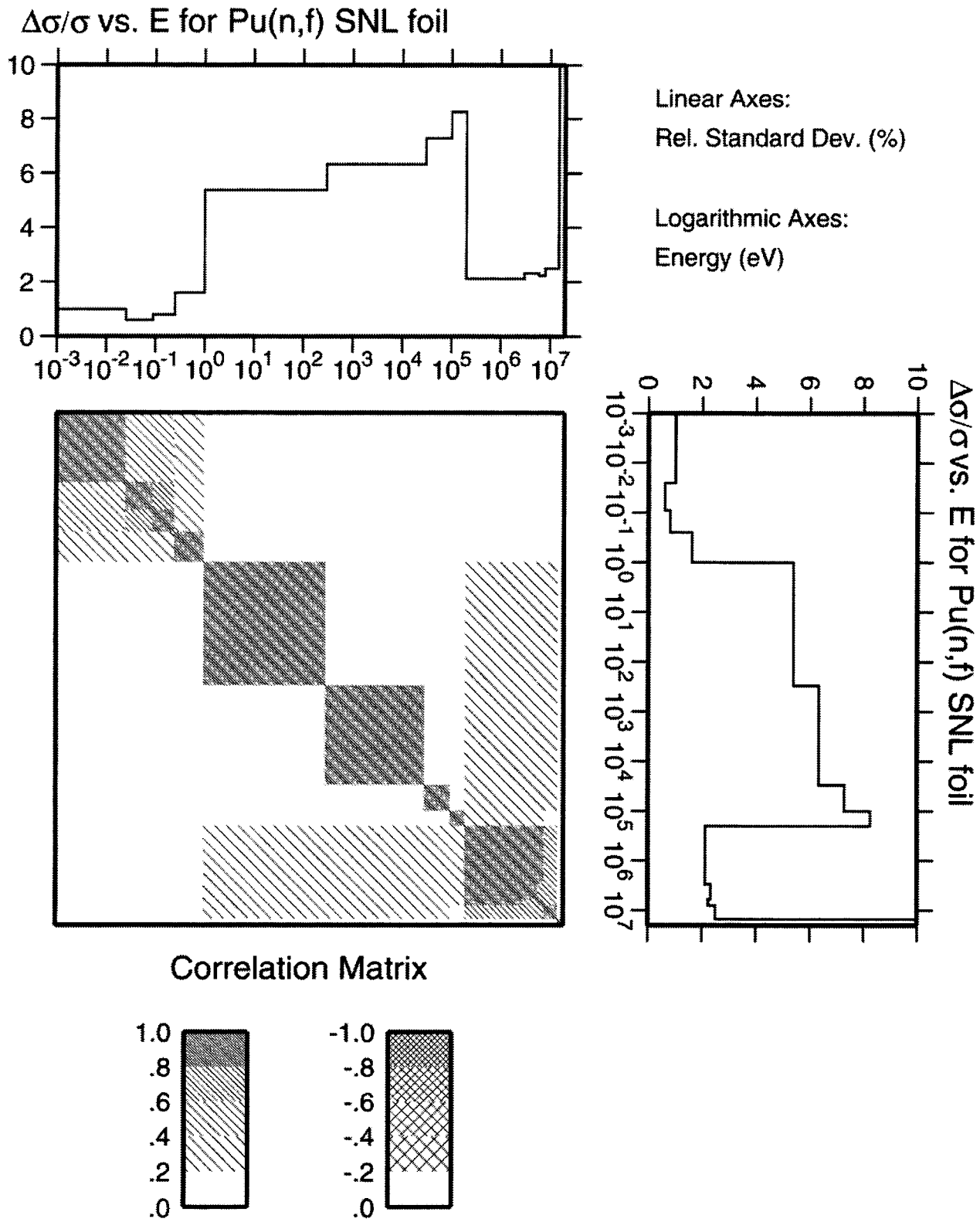

Fig. D.71. Plutonium fission foil [SNLRML foil, SNLRML-V1, JENDL-3D, covariance for ^{239}Pu(*n,f*)] correlation matrix and standard deviation.

No covariance data available for the following reactions:

^{234}U(n,f) (SNLRML-V1, ENDF/B-VI)

^{236}U(n,f) (SNLRML-V1, ENDF/B-VI)

^{240}Pu(n,f) (SNLRML-V1, ENDF/B-VI)

^{241}Pu(n,f) (SNLRML-V1, ENDF/B-VI)

^{238}Pu(n,f) (SNLRML-V1, ENDF/B-VI)

^{242}Pu(n,f) (SNLRML-V1, ENDF/B-VI)

Fig. D.72. Uranium and plutonium reactions without covariance data.

APPENDIX E

CONTRIBUTIONS FROM THE LB=0 AND LB=8 COVARIANCE SUBSECTIONS TO THE STANDARD DEVIATIONS OF THE SPECTRUM-AVERAGED CROSS SECTIONS

The effect of taking into account the LB=0 and LB=8 contributions to the covariance matrix is shown in Table E.1 for all the reactions on the IRDF-90/G-V2 file that contain the LB=8 or LB=0 information. The standard deviations of the cross sections averaged over the "combined" spectrum, once calculated without the contributions from LB=0 and LB=8 subsections, and once calculated with these contributions are compared. As expected, the changes are very small. However, when calculated without the LB=8 contribution the Ti-48 (n,p) reaction covariance matrix includes some correlation coefficients greater than 1.0 and is therefore incorrect. The errors are eliminated when LB=8 is taken into account. Also, the only uncertainty contribution given for the Fe-58 (n,γ) reaction is described with the LB=0 subsection. A corresponding covariance matrix can be obtained only when this component is processed.

The choice of the 640-group reference spectrum for processing the LB=0 and LB=8 subsections has very little or only a negligible effect on the resulting standard deviations, except for a few reactions listed in Table E.2.

Table E.1. Comparison of the standard deviations of the average cross sections

Reaction	Standard deviation[a]	
	Without (%)[b]	With (%)[c]
Al27(N,P)Mg27	5.47	5.53
S32(N,P)P32	3.07	3.22
Sc45(N,G)Sc46	0.73	0.75
Ti46(N,P)Sc46	2.44	2.48
Ti47(N,P)Sc47	3.74	3.78
Ti48(N,P)Sc48	Error[d]	2.56
Mn55(N,G)Mn56	4.11	4.24
Fe54(N,P)Mn54	2.12	2.12
Fe56(N,P)Mn56	2.24	2.26
Fe58(N,G)Fe59	Error[e]	5.57
Ni58(N,P)Co58	2.21	2.22
Ni60(N,P)Co60	10.51	10.65
Cu63(N,G)Cu64	3.95	3.96
Cu63(N,A)Co60	2.33	2.37
Cu65(N,2N)Cu64	1.89	1.94

[a]Relative standard deviation (in %) of the cross section averaged over the "combined" spectrum.
[b]Calculated without the contributions from the LB=8 and LB=0 subsections of covariance information. The LB=0 and LB=8 subsections define the absolute contributions to the covariance matrix rather then relative contributions as other LB subsections do.
[c]Calculated with the contributions from the LB=8 and LB=0 subsections of covariance information.
[d]Correlation coefficients greater than 1.0 present.
[e]All uncertainty information is in the LB=0 subsection.

Table E.2. Effect of the reference spectrum used for processing LB=0 and LB=8[a]

Reaction	Spectrum[b]	Standard deviation for the spectrum-averaged cross section (%)				Source library	MAT	MT	Origin of data
		Maxwell	1/E	Watt fission	Combined spectrum				
Sc45(N,G)Sc46									
	REF640B.FLX	0.75	0.78	8.94	0.75	IRDF-90/G-V2	2126	102	Priv. comm
	REFFLAT.FLX	8.08	12.82	8.94	9.17	IRDF-90/G-V2	2126	102	Priv. comm.
Mn55(N,G)Mn56									
	REF640B.FLX	4.30	3.93	7.46	4.24	IRDF-90/G-V2	2525	102	ENDF/B-VI
	REFFLAT.FLX	8.88	7.11	7.46	9.23	IRDF-90/G-V2	2525	102	ENDF/B-VI
Fe58(N,G)Fe59									
	REF640B.FLX	5.81	5.36	27.44	5.57	IRDF-90/G-V2	2637	102	ENDF/B-VI
	REFFLAT.FLX	30.86	16.62	28.44	30.49	IRDF-90/G-V2	2637	102	ENDF/B-VI
Ni60(N,P)Co60									
	REF640B.FLX	0.0	0.0	10.65	10.65	IRDF-90/G-V2	2831	103	ENDF/B-VI
	REFFLAT.FLX	0.0	0.0	10.61	10.61	IRDF-90/G-V2	2831	103	ENDF/B-VI
Cu63(N,G)Cu64									
	REF640B.FLX	4.12	4.04	8.61	3.96	IRDF-90/G-V2	2925	102	ENDF/B-VI
	REFFLAT.FLX	4.12	3.98	8.61	3.96	IRDF-90/G-V2	2925	102	ENDF/B-VI

[a]Listed above are the reactions for which the reference spectrum selected in the processing of covariance subsections LB=8 and LB=0 affects the values of the standard deviations. Out of 15 reactions with LB=8 or LB=0, only the listed reactions were affected. (Standard deviations are given in percents.) The LB=0 and LB=8 subsections define the absolute contributions to the covariance matrix, rather than the relative contributions as other LB subsections do.

[b]Spectrum REFFLAT.FLX is constant (energy independent), while the REF640B.FLX is the "combined spectrum" described in the footnote to Table A.1.

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