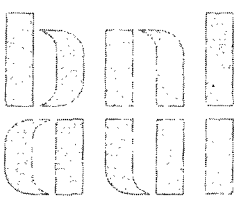
BNL-NCS-50496
(ENDF-102)
2nd Edition (ENDF/B-V) Revised
UC-80
(General Reactor Technology - TID-4500)

ENDF-102

DATA FORMATS AND PROCEDURES FOR THE EVALUATED NUCLEAR DATA FILE, ENDF/B-V

Edited by R. Kinsey
OCTOBER 1979
Revised by B.A. Magurno
NOVEMBER 1983



NATIONAL NUCLEAR DATA CENTER

BROOKHAVEN NATIONAL LABORATORY
ASSOCIATED UNIVERSITIES, INC.
UPTON, LONG ISLAND, NEW YORK 11973

UNDER CONTRACT NO. DE-AC02-76CH00016 WITH THE
UNITED STATES DEPARTMENT OF ENERGY

This publication is based on previous reports:

BNL 8381, ENDF — Evaluated Nuclear Data File Description and Specifications, June 1964, H.C. Honeck.

BNL 50066, ENDF 102, Vol. 1 — ENDF/B — Specifications for an Evaluated Nuclear Data File for Reactor Applications, May 1966, H.C. Honeck; Rev. July 1967, S. Pearlstein; Rev. Oct. 1970, M.K. Drake, Editor.

LA 4549, ENDF 102, Vol. II — ENDF Formats and Procedures for Photon Production and Interaction Data, Oct. 1970, D.J. Dudziak.

BNL-NCS-50496, ENDF 102, Data Formats and Procedures for the Evaluated Nuclear Data File, ENDF, Oct. 1975, D. Garber, C. Dunford, and S. Pearlstein, Editors.

ORNL/TM-5938, ENDF-249, The Data Covariance Files for ENDF/B-V, July 1977, F. Perey.

As in the previous edition, it brings under one cover both the neutron and photon formats and in addition includes the new data covariance formats.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161

DATA FORMATS AND PROCEDURES

FOR THE ENDF NEUTRON CROSS SECTION LIBRARY

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

0. ENDF/B - V PREFACE

These revisions to Data Formats and Procedures for the ENDF Neutron Cross Section Library, ENDF-102, pertain to the latest version of ENDF/B-V. The descriptions of the formats have been brought up to date and important procedural matters have been explained. Users of this manual who note deficiencies or have suggestions are encouraged to contact the National Nuclear Data Center at Brookhaven National Laboratory, Upton, New York 11973.

0.1. INTRODUCTION

0.1.1. ENDF System

This report describes the philosophy, of the Evaluated Nuclear Data File (ENDF)* and the data formats and procedures that have been developed for it. The ENDF system was designed for the storage and retrieval of the evaluated nuclear data that are required for neutronics, photonics and decay heat calculations. This system is composed of several parts that include neutron cross section, photon cross section, and nuclear structure data libraries with associated processing codes.

The ENDF system was developed to provide a unified format that could be used to store and retrieve evaluated sets of neutron cross sections. It was designed to allow easy exchange of cross section information between various laboratories. The initial system contained format specifications for neutron cross sections and other related nuclear constants. During the later stages of development the formats were expanded to include photon interaction cross sections, photon production data (photons produced by neutron interactions) and nuclear structure data.

The basic data formats developed for the library are versatile enough to allow accurate description of the cross sections considered for a wide range of incident neutron energies (10^{-5} eV to 20 MeV). The ENDF formats are flexible in the sense that almost any type of neutron interaction mechanism can be accurately described. They are restrictive in that only a limited number of different representations are allowed for any given neutron reaction mechanism.

*This report supersedes the descriptions of the ENDF/B library given in BNL 8381 and in BNL 50066 (ENDF 102).

0.1.2. ENDF Documentation

The purpose of this report is to describe the data formats and the procedures to be used for entering data into the ENDF library. In addition, this report describes the relationship between the ENDF evaluated data libraries and the experimental data library CSISRS (Cross Section Information Storage and Retrieval System). The relationship between the ENDF libraries and the processing codes that are used to generate secondary data libraries (for example, fine group-averaged cross section libraries) is also described. The processing codes connected with the ENDF libraries are summarized in Appendix I, but the codes themselves are described in separate documents.

This report is organized in the following manner. The first section describes the general features of the ENDF libraries, the relationship between ENDF and CSISRS, and the relationship between ENDF and its secondary libraries. Section 0.5 describes the standard formats used in all record types. An understanding of Section 0.5 will facilitate understanding the data formats given in Sections 1-33. BCD card-image formats are given in Appendix N.

0.1.3. A and B Libraries

Two different evaluated data libraries are maintained at the National Nuclear Data Section Center (NNDC). The ENDF/A library contains either complete or partial data sets (partial in the sense that the data set may be, for example, an evaluation of the fission cross section for ^{235}U in the energy range 100 keV to 15 MeV only). This library may also contain several different evaluations of the cross sections for a particular nuclide. The ENDF/B library, on the other hand, contains only one evaluation of the cross sections for each material in the library, but each material contains cross sections for all significant reactions. The data set selected for the ENDF/B library is the set

recommended by the Cross Section Evaluation Working Group (CSEWG)*. The ENDF/B library contains reference data sets with which other information may be compared, as opposed to data sets that are revised often on the basis of new information so as to constitute current standard data sets. After an extensive review period of 1 or more years, CSEWG may from time to time replace an older set with a new data set. The ENDF/A and ENDF/B libraries are described in more detail in Section 0.2.

0.2. GENERAL FEATURES OF THE EVALUATED NUCLEAR DATA FILE

0.2.1. Evaluated Data

The process of analyzing experimentally measured cross section data, combining it with the predictions from nuclear model calculations, and attempting to extract the true value of a cross section is referred to as evaluation. Parametrization and reduction of the data to tabular form produces an evaluated data set. If the written description of the preparation of a unique data set from the data sources is available, it is referred to as a documented evaluation. The ENDF format was developed to store the numerical results of an evaluation in a form suitable for automated retrieval for further processing.

0.2.2. A and B Libraries

The demands on an evaluated data file vary according to the user's applications. Whether the user is interested in performing a reactor physics calculation or in doing a shielding analysis, he wants evaluated data for all neutron-induced reactions, covering the full range of incident neutron energies, for each material in the system that he is analyzing. Also, the user expects

*A cooperative effort of many laboratories coordinated by the National Nuclear Data Center.

that the data file will contain information such as the angular and energy distributions for secondary neutrons. The ENDF/B library will contain only one representation (or interpretation) of the cross sections for a particular material at any given time. ENDF/A may contain several alternative sets of evaluated data for the materials on the ENDF/B library. The data sets that are contained on the ENDF/A library may or may not be complete in the sense that ENDF/B is a complete evaluated file. The ENDF/A library is, in effect, a system for compiling evaluated data sets.

The formats used for these two libraries are basically identical; i.e., the codes that are used to read and process data from the ENDF/B library may be used for the ENDF/A library. The data formats for these libraries are given in the following sections. The differences in the formats for the ENDF/A and ENDF/B libraries are given in Appendix H.

0.2.3. Choices of Data

The data sets contained on the ENDF/B library are those chosen by the CSEWG from evaluations submitted to it. The data set that represents the cross sections for a particular material may change from time to time upon the recommendation of CSEWG. Such a recommendation generally is made when (1) new and significant experimental results become available, (2) integral data testing shows that the data set gives erroneous results, or (3) users' requirements indicate a need for more accurate and/or a better representation of the cross sections for a particular material.

0.2.4. Library Modes

The neutron cross section libraries comprise the central part of the ENDF system. The libraries are contained on magnetic tapes or disks. Two different

modes of the data tapes are maintained: a binary form and a BCD card-image form. The formats for these two modes are very similar. The data formats for a binary tape are defined in Sections 1-15. Basically there are only four different types of binary records (see Section 0.5.2), each with a specific format. Control numbers and flags always appear in the same position within a record of a particular type. Understanding the definitions of all record types will facilitate understanding the particular formats described later on.

Since binary tapes generated with use of a particular computer may not be easily read on another type of computer, a BCD card-image format was developed. The card-image formats are described in Section 0.5.3 and are similar to those used for binary records. Certain key data words (for example, material and cross section type identifiers), which are given only at the beginning of each binary record, are given at the end of each BCD card-image record. BCD card-image formats are described in Appendix N along with examples of data sets.

0.2.5. Systematization of Data

The ordering of nuclear data for a particular material is described in Section 0.4.2. Integral cross section data (for example, the total cross section) may be represented by giving tabulated values of σT vs. neutron energy. An interpolation scheme is also specified to define the cross section at intermediate energy values. Also, resolved and/or unresolved resonance parameters may be given. Note that if resonance parameters are given, then contributions to a particular cross section from the resonance parameters must be added to the integral cross sections to obtain the complete cross section. In other words, the integral cross sections and the resonance parameters are not redundant.

Note that the angular and energy distribution differential data are expressed as probabilities. Therefore these data must be combined with integral (integrated) data to obtain absolute differential cross sections.

0.3. RELATIONSHIP OF THE ENDF TO OTHER DATA SYSTEMS

0.3.1. Experimental Data Libraries

NNDC maintains a library for experimentally measured neutron cross section data, known as CSISRS.

The CSISRS library is quite flexible, with many types of data stored. Bibliographic information (a succinct abstract to a reference) is stored with many details about each experiment (standard, renormalizations, corrections, etc.).

At the beginning of the evaluation process the evaluator generally requests the available experimental cross sections that are stored in the CSISRS data library for a particular material. The retrieved information may be in the following forms:

(a) Listings of all or selected data sets.

(b) Magnetic tapes containing the requested data.

(c) Graphical displays containing selected data.

The experimental cross section data are supplemented by other nuclear data, such as spins, energies, and parities of excited states. The experimental data are then analyzed, and in some cases the results are combined with predictions from model calculations to obtain recommended cross sections. The recommended cross sections are then encoded in the ENDF formats for subsequent incorporation into either the ENDF/A or the ENDF/B library.

A number of systems have been developed[1,2] for automating much of the time-consuming parts of the evaluation process. These systems, by permitting man/computer interaction through computer graphics, shorten the time required for the evaluation process. Since the evaluator can make more detailed analyses of the cross sections, the quality of the evaluation process should be improved.

0.3.2. Processing Codes and Neutronics Calculations

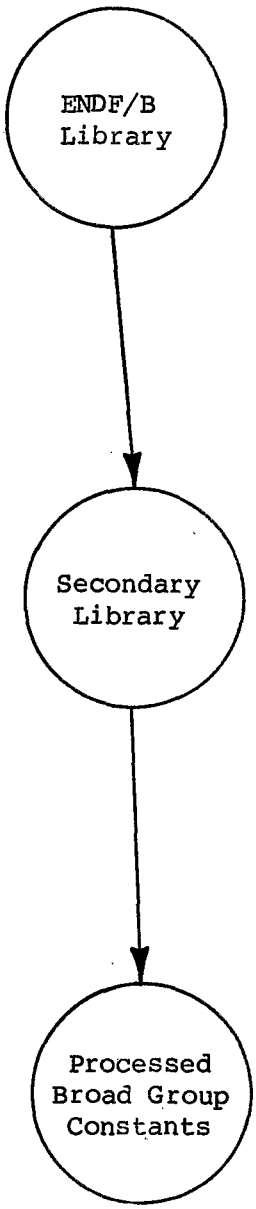
The purpose of the ENDF library is to provide evaluated cross section data sets in a form that can be used in various neutronics and photonics calculations. The existing codes that perform these calculations require data libraries that are quite different from one another and from the ENDF library. Therefore a series of processing codes have been written which read the ENDF library as input and generate a secondary cross section library. The secondary libraries, in turn, are read as input to a spectrum-generating code, and generally broad group-average cross sections are obtained for use in the neutronics calculations. The available processing codes are summarized in Appendix I. Figure 0.3.2.1 shows the flow of data for a particular set of codes.

The basic data formats for the ENDF library have been developed in such a manner that few constraints are placed on using the data as input to the codes that generate any of the secondary libraries.

---

(1) C. L. Dunford et al., "SCORE II, An Interactive Neutron Evaluation System," USAEC Report AI-AEC-12757 (ENDF-126), March 1, 1969.

(2) R. R. Kinsey, C. Rindfleish, D. Garber, "TIGER," The Interactive Graphics Evaluation Routine, NNDC, 1973.



Processing Code: MINX, ETOX, ETOG, etc.

Spectrum Generating Code: 1OX, SPHINX, etc.

Neutronic Code: ANISN, DTF, etc.

Figure 0.3.2.1: Schematic of the flow of nuclear data from compilation to reactor calculation.

0.4. GENERAL DESCRIPTION OF THE ENDF LIBRARY

The ENDF library is a collection of documented data evaluations stored in a defined computer readable format that can be used as the main input into cross section processing programs. As such it has been designed with the processing programs in mind and requires some familiarity with the FORTRAN programming language. The ordering of data on the tape allows the use of segmented as well as ordinary programs.

Two formats are provided, one for binary data and the other for BCD card images. Magnetic tapes containing BCD card-image data generally can be exchanged between laboratories. Also, it is much easier to use the BCD card-image formats when translating evaluated cross sections into the ENDF library.

0.4.1. Definitions and Conventions

It has been traditional to define a version of the ENDF/B library, for example ENDF/B-V. The version number (V) indicates a reissue of the ENDF/B library, superceding all previous versions. In general, the issue of a version also corresponds to the implementation of formats and procedures modifications.

Within the ENDF system, it is possible to define different sub-libraries such as ENDF/B, ENDF/A, CTR, etc. Each material in the ENDF library must be identified as to the appropriate sub-library by use of the NLIB number. See Section 1.1 for the format and definition of NLIB.

A material is defined as either an isotope or a collection of isotopes. It may be a single nuclide, a natural element containing several isotopes, a molecule containing several elements, or a standard mixture of elements (such as 304 type stainless steel). Each evaluated set of cross sections for a material in the ENDF library is assigned a unique identification number. These numbers are designated by the symbol MAT and they range from 1 to 9999. Two different

evaluated sets of cross sections for ^{235}U would have different MAT numbers even though they describe the cross sections for the same nuclide. A program that processes data from the ENDF library generally refers to the materials by their MAT number, but a (Z,A) designation is also given for each material and this value may be used.

When an evaluated set of cross sections for a material (in the ENDF format) is sent to the NNDC, the Center assigns a MAT number to this material. This number will never be assigned to another set of evaluated cross sections.

In the past no new materials evaluations have been released between versions of the ENDF/B library. In the future these restrictions will be relaxed so that new or revised evaluations can be added to the ENDF/B library between release of two versions. This is the modification (MOD) concept. Each material has a NMOD number associated with it.

As explained in Section 1.1, NMOD=0 means that the evaluation was translated into the ENDF/B-V format from a previous version of the library with no changes to the evaluation, NMOD=1 means that the evaluation was either revised or is a new evaluation for ENDF/B-V, and NMOD=2 or greater means that the evaluation has been revised since its initial release as a part of the ENDF/B-V library.

In addition, each data section has a MOD number associated with it and given in the dictionary for the material described in Section 1.1. The meaning of the numerical values given in the MOD field are identical to those in the NMOD field except that they apply to the given data section only. Of course, the MOD number for any section must always be less than or equal to NMOD. Whenever a material is re-released as an NMOD=2 or greater, those sections which where corrected or revised will have MOD=NMOD.

As an example, consider the following sequence of events. User X evaluates a set of data for ^{235}U and assigns the material number 278 to this set. Within his installation the data set is always referred to as material 278. After checking and testing, the user feels that the data set is satisfactory and transmits it to the NNDC. The Center adds the data set to its files and assigns it a MAT number of 1395 subject to CSWEG's approval of the evaluation. This evaluation has NMOD=1, and MOD=1 for all new or revised data sections. The Center then issues a newsletter describing data received and available for distribution. User Y reads the newsletter and requests material 1395 from the Center's files. Upon receipt of the data he adds it to his ENDF tape as material 1395 and refers to it in later processing programs by this number. Should user Y subsequently alter the data, he could send the revised sections to NNDC and with CSWEG's approval NMOD=2 would be released with MOD=2 for all the revised sections.

File Number (MF)	Class of Data
1	General information
2	Resonance parameter data
3	Neutron cross sections
4	Angular distributions of secondary neutrons
5	Energy distributions of secondary neutrons
6	Energy-angular distributions of secondary neutrons
7	Thermal neutron scattering law data
8	Radioactive decay and fission product yield data
9	Multiplicities for production of radioactive nuclides
10	Cross sections for production of radioactive nuclides
12	Multiplicities for photons (from neutron reactions)
13	Cross sections for photons (from neutron reactions)
14	Angular distributions of photons (from neutron reactions)
15	Energy distributions of photons (from neutron reactions)
16	Energy-angular distributions of photons (from reactions)
17	Time dependent discrete photon production data
18	Time dependent continuum photon production data
19	Electron multiplicities and transition probability arrays
20	Electron production cross sections
21	Electron angular distributions
22	Continuous electron energy spectra
23	Photon interaction cross sections
24	Angular distributions of photons (from photon reactions)
25	Energy distributions of photons (from photon reactions
26	Energy-angular distributions of photons (from photon reactions)
27	Atomic form factors (for photon interactions)
31	Data covariance matrices for $\bar{\upsilon}$
32	Data covariance matrices for resonance parameters
33	Data covariance matrices for neutron cross sections

Each file is divided into sections, each containing the data for a particular reaction type. The various reaction types are identified by the symbol MT. The definitions for allowed reaction types (MT numbers) are given in Appendix B.

The first record of each section contains a ZA number that identifies the specific material. ZA is the (Z,A) designation (charge, mass). ZA for a specific material is constructed by

$$ZA = (1000.0*Z) + A,$$

where Z is the atomic number and A is the mass number for the material. For example, ZA = 92238.0 for ^{238}U. If the material is an element containing two or more naturally occurring isotopes, A, in the above equation, is taken to be 0.0. The ZA designators for materials that are molecules or common mixtures have been assigned certain values. These designators are given in Appendix C.

The first record of each section also contains a quantity that is proportional to the nuclear mass of the material. This quantity symbol AWR, is defined as the ratio of the nuclear mass of the material (isotope, element, molecule, or mixture) to that of the neutron. The mass of a neutron is taken to be 1.008665 (in the carbon-12 system).

The data given in all sections always use the same set of units. These are summarized below.

Parameters	Units
energies	electron volts
angles	dimensionless cosines of the angle
cross sections	barns
temperatures	°Kelvin
mass	in units of the neutron mass
angular distributions	probability per unit cosine

energy distributions	probability per electron volt
energy-angle distributions	probability per unit cosine per electron volt
half life	seconds

0.4.2. Structure of an ENDF Data Tape

The structure of an ENDF data tape is illustrated schematically in Figure 0.4.2-1. The structure of the tape is the same whether it is a BCD card image tape or binary tape.

The tape contains a single record at the beginning that identifies the tape (TPID), and a single record at the end (TEND) that signals the end of the tape. The major subdivision between these records is by material. The data for a material is divided into files, and each file (MF number) contains the data for a certain class of information. A file is subdivided into sections, each containing data for a particular reaction type (MT number). Finally a section is divided into records. The content of each record is different and depends on whether a binary tape format or a BCD card-image format is used. Every record on a tape contains three identification numbers: MAT, MF, and MT. For a binary record, these numbers are given at the beginning of each record. For BCD card-image records, they are given in the last three fields of each record. These numbers are always in increasing numerical order, and the hierarchy is MAT, MF, MT. The end of a section, file, or material is signaled by special records.

The single record at the beginning of any ENDF tape, (TPID), consists of a 66 character free text field and a tape number. For a binary tape this record is 20 words. The first word is the integer value of the tape number. The second and third word are integer zeros for consistency with the other ENDF records which all have three integer values (MAT,MF,MT) to label the record

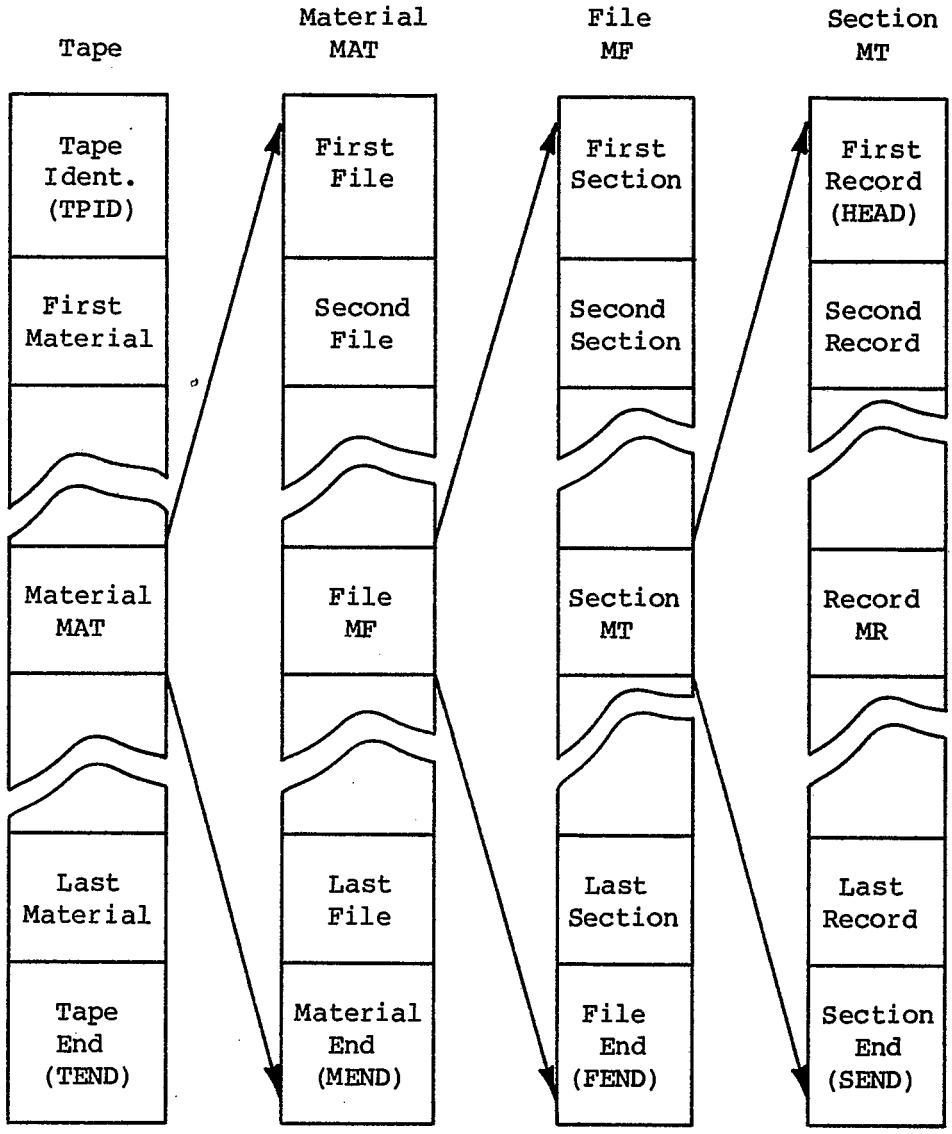


Figure 0.4.2.1: Arrangement of an ENDF Tape

type. The final 17 words are Hollerith data which is assumed to be in (16A4,A2) format. For a BCD tape the 80 character record has the following format:

Cols.	Contents
1-66	Alphanumeric tape label
67-70	NTAPE, integer tape number
71-72	Ø
73-75	Ø
76-80	Sequence number, Ø for tape label

Following the tape end card (TEND) on any BCD library tape will be a file giving the current contents of the library. This feature will allow the user to compare the contents of his library with the master library. The file will consist of one 80 character record per material in the following format:

Cols.	Contents
1-11	IZA = 1000* Z + A
12-17	MAT = material number
18-22	MOD = modification number
23-33	Tape number on which material appears
34-44	Distribution data (MMDDYY)
45-66	Laboratory and evaluator
67-70	ENDF version number
71-80	Sequence number

The last card will have a -1 in the first field.

0.4.3. Representation of Data

The data in the ENDF library are given by providing parameters to known analytic functions (such as resonance formulae or secondary energy distribution

laws), or are presented by tabulating the data in one $(y(x))$ or two $(y(x,z))$ dimensional arrays.

Consider how a simple function, y(x), which might be a cross section, $\sigma(E)$, is represented. y(x) is represented by a series of tabulated values, pairs of x and y(x), plus a method for interpolating between input values. The pairs are ordered by increasing values of x. There will be NP values of the pair, x and y(x), given. The complete region over which x is ds defined is broken into NR interpolation ranges. An interpolation range is defined as a range of the independent variable x in which a specified interpolation scheme can be used; i.e., the same scheme can be used to obtain interpolated values of y(x) for any value of x that is within this range. To illustrate this, see Fig. 0.4.3-1 and the definitions below:

X(n) is the n^{th} value of x.

Y(n) is the n^{th} value of y.

NP is the number of pairs (X and Y) given.

INT(m) is the interpolation scheme identification number used in the m^{th} range.

NBT(m) is the value of N separating the m^{th} and $(m+1)^{th}$ interpolation ranges.

The allowed interpolation schemes are

INT	Description
1	y is constant in x (constant)
2	y is linear in x (linear-linear)
3	y is linear in ℓnx (linear-log)
4	ℓn y is linear in x (log-linear)
5	ℓn y is linear in ℓn x (log-log)

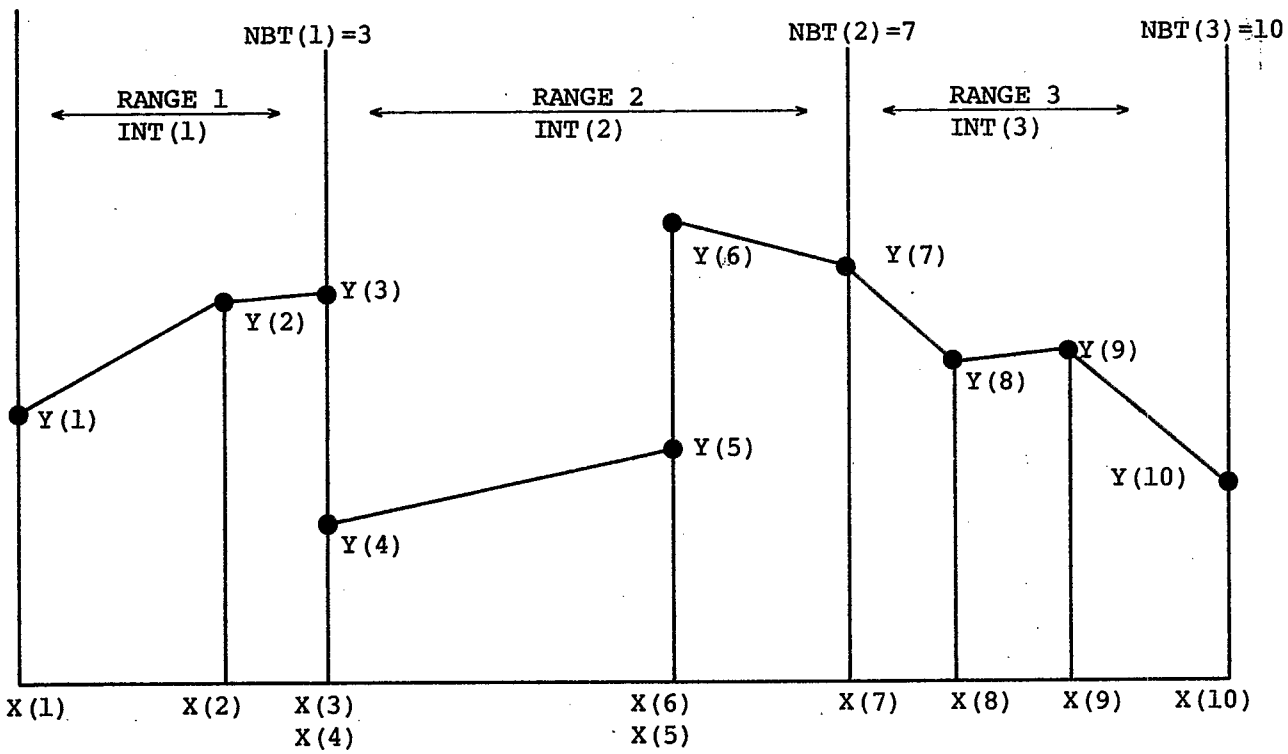


Figure 0.4.3.1: Tabulated one dimensional function illustrated for the case NP=10, NR=3.

Interpolation code, INT = 1 (constant), implies that the function is constant and equal to the value given at the lower limit of the interval.

Note that where a function is discontinuous (for example, when resonance parameters are used to specify the cross section in one range), the value of X is repeated and a pair (X,Y) given for each of the two values at the discontinuity (see Fig. 0.4.3-1).

Next consider a two-dimensional function of x and z, y(x,z). Again, the function is represented by a series of tabulated values of $y(x,z_i)$ plus rules for interpolating the function between values of z. The function is thus considered to be a sequence of one-dimensional functions, $y_i(x)$, each evaluated at a particular z_i. The individual $y_i(x)$ can be represented as illustrated above. The only additional information that need be given is a breakpoint and interpolation table for interpolation between values of z.

0.5. GENERAL DESCRIPTION OF THE DATA FORMATS

0.5.1. Nomenclature

An attempt has been made to use an internally consistent notation. We list here some of the rules used.

a) Symbols starting with letters I, J, K, L, M, or N are integers. All other symbols refer to floating point numbers.

b) The letter I or a symbol starting with I refers to an interpolation code (see Appendix E).

c) Letters J, K, L, M, or N, when used alone, are indices.

d) A symbol starting with M is a control number. Examples are MAT, MT, MF.

e) A symbol starting with L is a test number. Examples are LFI, LCT, LTT.

f) A symbol starting with N is a count of items. Examples are N1, NR, NP, NFP.

g) Brackets () denote one record on a binary tape.

h) Brackets <> denote a group of records.

Several frequently used symbols are defined below.

MAT - Material number

MF - File number

MT - Reaction type number

ZA - The (Z,A) designation for a material (see Appendix C)

AWR - The ratio of the mass of an atom (or molecule) to that of the neutron

NP - The number of points in a tabulation of y(x) that are contained in the same record

NR - The number of different interpolation intervals in a tabulation of y(x) that are contained in the same record

T - Temperature

E - Energy

μ - Cosine of an angle

LT - Temperature dependence (see Appendix F).

0.5.2. Types of Records

All records on an ENDF tape are one of four possible types, denoted by CONT, LIST, TAB1, and TAB2. A record always consists of nine numbers followed (depending on the record type) by none, one, or two arrays of numbers. A gen-

eral description of these nine numbers is given below, but the actual definition of each number will depend on its usage.

MAT is the material number (integer).

MF is the file number (integer).

MT is the reaction type number (integer).

C1 is a constant (floating point).

C2 is a constant (floating point).

L1 is an integer generally used as a test.

L2 is an integer generally used as a test.

N1 is a count of items in a list to follow.

N2 is generally a count of items in a second list to follow.

0.5.2.1. CØNT Records

The smallest possible record is a control (CØNT) record consisting of the nine numbers given above and no associated array. For convenience, a CØNT record is denoted by

(MAT, MF, MT/C1, C2; L1, L2; N1, N2)CØNT

The numbers within the brackets are symbolic of the numbers in a CØNT record. The semicolon is used to mark the separation between floating point numbers, test numbers, and counts. The slash is a reminder that the numbers MAT, MF, and MT appear in a different position on BCD card-image records then they do on the binary records. The BCD card-image format is described in Section 0.5.3 in more detail.

There are five special cases of a CØNT record, denoted by HEAD, SEND, FEND, MEND, and TEND. The HEAD record is the first in a section and has the same form as a CØNT record.

The structure of a typical HEAD record is

(MAT, MF, MT/ ZA, AWR, L1, L2, N1, N2)HEAD

where ZA is the (Z,A) designation for a material (see Appendix C),

AWR is the ratio of the mass of the atom (or molecule) to that of the neutron (carbon-12 system),

L1 is an integer to be used as a flag or a test,

L2 is an integer to be used as a flag or a test,

N1 is an integer to be used as a count of items in a list to follow except for MT=451, and

N2 is an integer to be used as a count of items in a second list to follow except for MT=451.

The symbolism used above to represent the HEAD record and to be used in the following format descriptions can be interpreted in either the binary format or the BCD-card image format. In the BCD-card image format, the six items after the slash are the contents of the first six fields on the card. Each field is eleven characters wide. After the first 66 characters, there are four characters for the MAT, two for the MF, and three for the MT. In the binary format, the symbolism just indicates the list of items to be read.

The SEND, FEND, MEND, and TEND records use only the first three numbers in the CØNT record, and they are used to signal the end of a section, file, material, and tape, respectively:

(MAT, MF, 0/0.0, 0.0; 0, 0; 0, 0)SEND

(MAT, 0 , 0/0.0, 0.0; 0, 0; 0, 0)FEND

(0 , 0 , 0/0.0, 0.0; 0, 0; 0, 0)MEND

(-1 , 0 , 0/0.0, 0.0; 0, 0; 0, 0)TEND

A FORTRAN statement to read any CØNT record from a binary tape on unit LIB would be

```
READ (LIB) MAT, MF, MT, C1, C2, L1, L2, N1, N2
```

0.5.2.2. LIST Records

The second type of record is the LIST record, used to list a string of floating point numbers, B_1, B_2, B_3, etc. These numbers are given in an array, B(N), and there are N1 of them. A FORTRAN statement to read a LIST record from Tape LIB would be

```
READ (LIB) MAT, MF, MT, C1, C2, L1, L2, N1, N2, (B(N), N=1, N1)
```

For convenience, this record is denoted by

(MAT, MF, MT/C1, C2; L1, L2; N1, N2/ B_n)LIST

For example, to enumerate the particular items in a list (A, B, C, D, E), the record would be

(MAT, MF, MT/C1, C2; L1, L2; 5, N2/ A, B, C, D, E)LIST

where the 5 indicates that there are five items in the list.

0.5.2.3. TAB1 Records

The third type of record is the TAB1 record used for one-dimensional tabulated functions such as y(x). The data needed to specify a one-dimensional tabulated function are the interpolation tables NBT(N) and INT(N) for each of the NR ranges, and the NP tabulated pairs of X(N) and Y(N). The FORTRAN statement to read a TAB1 record is

```
READ (LIB) MAT, MF, MT, C1, C2, L1, L2, NR, NP,
    (NBT(N), INT(N), N=1, NR), (X(N), Y(N), N=1, NP)
```

For convenience, the TAB1 record is denoted by

(MAT, MF, MT/ C1, C2; L1, L2; NR, NP/x_{int}/y(x))TAB1

The term x_{int} means the interpolation table for interpolating between successive values of the variable x. y(x) means pairs of x and y(x). x is generally used as the incident neutron energy E, and y(x) is generally a parameter such as the cross section $\sigma(E)$.

0.5.2.4. TAB2 Records

The last record type is the TAB2 record, which is used to control the tabulation of a two-dimensional function, y(x,z). It specifies how many values of z are to be given and how to interpolate between successive value of z. Tabulated values of $y_i(x)$ at each value of z_i are given in TAB1 or LIST records following the TAB2 record, with the appropriate value of z in the field designated as C2. The FORTRAN statement to read a TAB2 record is

```
READ (LIB) MAT, MF, MT, C1, C2, L1, L2, NR, NZ, (NBT(N),
    INT(N), N=1, NR)
```

where NZ in the number of values of z. For convenience, a TAB2 record is denoted by

$$(\text{MAT, MF, MT/C1, C2; L1, L2; NR, NZ/z}_{int})\text{TAB2}$$

For example, a TAB2 record is used in specifying angular distribution data. NZ in the TAB2 record specifies the number of incident neutron energies at which angular distributions are given. Each distribution is given in a TAB1 record, and there will be NZ such records.

0.5.3. Card-Image (BCD) Formats

An alternative format is used when data are contained on BCD card-image tapes. Basically the data are stored in the same order for this format as in the binary tape format. The major difference is the position of the three numbers MAT, MF, and MT. Also a card sequence number has been added to the card-

image format. In general, more than one BCD card-image record will be required to contain the data in a binary record.

BCD card-image formats for each type of data section are also given in Appendix N. In the following discussion of record types, note that when arrays of numbers are punched, the first element is always in Field 1. The last element may fall in any field, depending on the array size. Thus, the fact that X(NP) in Section 0.5.3.3 is shown in Field 6 should not be taken literally.

A standard 80-column card is divided into the following ten fields

Field	Columns	Description
1	1-11	Datum
2	12-22	Datum
3	23-33	Datum
4	34-44	Datum
5	45-55	Datum
6	56-66	Datum
7	67-70	MAT
8	71-72	MF
9	73-75	MT
10	76-80	Sequence number, starting with 1 for the first card of a material

0.5.3.1. CONT Records

As in section 0.5.2.1, consider a CONT record denoted by

(MAT, MF, MT/C1, C2; L1, L2; N1, N2)CONT

This record would be punched on cards in the following way

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	N1	N2	MAT	MF	MT

A FORTRAN statement to read any CONT record from a card image tape on unit INP would be

```
      READ (INP,1) C1, C2, L1, L2, N1, N2, MAT, MF, MT, NS
    1 FORMAT (2E11.4, 4I11, I4, I2, I3, I5)
```

0.5.3.2. LIST Records

The LIST record denoted by

(MAT, MF, MT/ C1, C2; L1, L2; N1, N2/ B_n)LIST

is punched in the following way:

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	N1	N2	MAT	MF	MT
B(1)	B(2)	B(3)	B(4)	B(5)	B(6)	MAT	MF	MT
B(7)	B(8)	B(9)	----	----	----	MAT	MF	MT
----	----	----	----	----	B(NI)	MAT	MF	MT

The FORTRAN statements to read a LIST record from input tape INP would be

```
      READ (INP, 30) C1, C2, L1, L2, N1, N2, MAT, MF, MT, (B(N), N=1, N1)
   30 FORMAT (2E11.4, 4I11, I4, I2, I3/6E11.4))
```

An exception occurs when the LIST record contains Hollerith information (see File 1):

(MAT, MF, MT/ C1, C2; L1, L2; NWD, N2/ H_n) LIST

In this case the FORTRAN READ statements depend on the type of computer being used, but the cards should be machine independent. Define NWD as the number of cards containing Hollerith information punched in Cols. 1-66. The READ statements would be

```
      READ (INP, 40), C1, C2, L1, L2, NWD, N2, MAT, MF, MT
   40 FORMAT (2E11.4, 4I11, I4, I2, I3)
      NH = 17*NWD
      READ (INP, 50) (H(N), N=1, NH)
   50 FORMAT (16A4, A2)
```

0.5.3.3. TAB1 Records

As in Section 0.5.2.3, consider a TAB1 record that was denoted by

(MAT, MF, MT/C1, C2; L1, L2; NR, NP/x_{int} /y(x))TAB1

This record would be punched on cards in the following way

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	NR	NP	MAT	MF	MT
NBT(1)	INT(1)	NBT(2)	INT(2)	NBT(3)	INT(3)	MAT	MF	MT
NBT(4)	INT(4)	NBT(5)	INT(5)	-----	------	MAT	MF	MT
-----	-----	-----	-----	NBT(NR)	INT(NR)	MAT	MF	MT
X(1)	Y(1)	X(2)	Y(2)	X(3)	Y(3)	MAT	MF	MT
X(4)	Y(4)	X(5)	Y(5)	-----	-----	MAT	MF	MT
-----	-----	-----	-----	X(NP)	Y(NP)	MAT	MF	MT

The FORTRAN statements to read a TAB1 record from input tape INP would be

```
      READ(INP, 10)C1, C2, L1, L2, NR, NP, MAT, MF, MT, (NBT(N), INT(N),
          N = 1, NR)
   10 FORMAT (2E11.4, 4I11, I4, I2, I3/(6I11))
      READ (INP, 20) (X(N), Y(N), N=1, NP)
   20 FORMAT (6E11.4)
```

0.5.3.4. TAB2 Records

As in Section 0.5.2.4, consider a TAB2 record that was denoted by

(MAT, MF, MT / C1, C2; L1, L2; NR, NZ / z_{int})TAB2

This record would be punched on cards in the following way

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	NR	NZ	MAT	MF	MT
NBT(1)	INT(1)	NBT(2)	INT(2)	NBT(3)	INT(3)	MAT	MF	MT
------	------	------	------	NBT(NR)	INT(NR)	MAT	MF	MT

The FORTRAN statement to read a TAB2 record from input tape INP would be

```
      READ(INP,10) C1, C2, L1, L2, NR, NZ, MAT, MT,
          (NBT(N), INT(N), N=1, NR)
   10 FORMAT(2E11.4, 4I11, I4, I2, I3/(6I11))
```

1. FILE 1, GENERAL INFORMATION

File 1 is the first part of any set of evaluated cross section data for a material. Each material must have a File 1, which consists of one or more sections that contain neutron cross section information and other related nuclear data. File 1 provides a brief documentation of how the data were evaluated and a dictionary that summarizes the data files and cross section types given in Files 2, 3, 4, 5, etc. File 1 may also contain such basic nuclear data as the number of neutrons per fission (for fissile materials).

File 1 consists of at least one section and may contain as many as four other sections for fissile materials. Each section has been assigned an MT number (see below), and the sections are arranged in order of increasing MT numbers. Each section always starts with a HEAD record and ends with a SEND record. The end of File 1 (and all other files) is indicated by a FEND record. These record types are defined in detail in Section 0.5.2.1.

1.1. Descriptive Data and Dictionary (MT = 451)

This section is always the first section of any material, and has two main parts: (1) a brief documentation of the cross section data, and (2) a dictionary.

In the first part, a brief description of the evaluated data sets is given. This information should include the significant experimental results used to obtain the evaluated data as well as other important features about the evaluated data set. The descriptive information is given as a series of Hollerith characters. The information is contained in an array H(N), N = 1, 2, ... NWD. Each element of the array, H(N), contains 66 Hollerith characters. On cards the information is punched in Cols. 1-66, and NWD such cards are prepared.

The first two cards of the Hollerith information should contain titling information for the material. This information is used to provide titles for listings and plots and, while part of the Hollerith, has been organized and formatted. The following quantities are defined within the Hollerith and organized as shown in section 1.1.1.

(First BCD Card Image Record)

ZSYMA is a Hollerith representation of the material Z-chemical symbol-A and metastable state flag with

Z right justified in col. 1 to 3

\- hyphen in col. 4

chemical symbol left justified in col. 5 and 6

\- hyphen in col. 7

A right justified in 8 - 10 or blank

M for the indication of a metastable state or blank for the ground state in col. 11

ALAB Mnemonic of originating laboratory(s) (left adjusted) in cols. 12-22.

EDATE date of evaluation, EVAL- in cols. 23-27, three character month in 28-30, followed by two character year 31-32 (i.e. EVAL-DEC74)

AUTH author(s) of evaluation (left adjusted) cols. 34-66

(Second BCD-Card Image Record)

REF reference 2-22

DDATE original distribution date (left adjusted DIST- followed by month-year as in EDATE) in cols. 23-33.

RDATE date and number of last revision (REV1- followed by month-year as in EDATE) in cols. 34-44.

ENDATE Master file entry date (right adjusted YYMMDD) cols. 56-61.

The following quantities are defined.

LRP is a flag that indicates that resolved and/or unresolved resonance parameters are given in File 2.

LRP = 0, no resonance parameter data given;

LRP = 1, resolved and/or unresolved resonance parameter data given in File 2.

LFI is a flag that indicates whether this material is fissionable:

LFI = 0, this is not a fissionable material;

LFI = 1, this material is fissionable.

NLIB is the library identifier:

NLIB = 0, ENDF/B.

NLIB = 1, ENDF/A.

NLIB = 2, ENDF/A (translated from UK library).

NLIB = 3, ENDF/A (translated from KEDAK).

NMOD is the modification number:

NMOD = 0, evaluation carried over from previous library version.

NMOD = 1, new or revised evaluation for the current library version.

NMOD = 2 etc., successive modification indicators.

ELIS is the excitation energy of the target nucleus relative to 0.0 for the ground state.

STA is the target stability flag. If the target is unstable radioactive decay data should be given in MF=8, MT 457:

STA = 0.0, stable nucleus,

STA = 1.0, unstable nucleus.

LIS is the state number of the target nucleus. The ground state is indicated by LIS=0.

LISØ is the isomeric state number. The ground state is indicated by LISØ=0, etc. LIS need not be the same as LISØ.

NWD is the count of the number of elements in the Hollerith section. For BCD card image tapes, NWD is the number of card images used to describe the data set for this material (NWD $\leq$ 294). For binary tapes, NWD is the number of words containing the Hollerith information, and it is understood that 17 words are required for each card image (66 characters) and the format is (16A4, A2). (NWD $\leq$ 5000.)

NXC is an integer count of all the sections to be found in the dictionary. Each section of this material is represented by a single card image that contains MF, MT, (reaction number), and NC (a count of the number of cards in the section). NXC is the total number of sections for the complete material; i.e., it is equal to the sum of all the sections in the different files.

H(N) is the array containing the Hollerith information that describes the particular evaluated data set. For a BCD card-image tape, each element of the array is contained on one card image.

MF_n, MT_n, NC_n and $MØD_n$ are included in each of the NXC items in the dictionary.

MF_n is the MF of the n^{th} section.

MT_n is the MT of the n^{th} section.

NC_n is the number of BCD card images in a given section (the n^{th} section). This card count does not include the SEND card.

(Note that NC_1 = NXC + NWD + 3.)

MOD_n is the modification indicator for section MF_n and MT_n (see NMOD for definitions).

1.1.1. Formats

This section always begins with a HEAD record and ends with a SEND record. Its structure is

(MAT, 1, 451/ZA , AWR, LRP, LFI, NLIB, NMOD)HEAD

(MAT, 1, 451/ELIS, STA, LIS, LISØ, 0, 0)CØNT

(MAT, 1, 451/0.0, 0.0, 0, 0, NWD, NXC/

ZSYMA, ALAB, EDATE, AUTH(33 characters)/

REF(22 characters), DDATE, RDATE, b, ENDATE/

H(N))LIST*

(MAT, 1, 451/0.0, 0.0, MF_1, MT_1, NC_1, MOD_1)CØNT

(MAT, 1, 451/0.0, 0.0, MF_2, MT_2, NC_2, MOD_2)CØNT

(MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}, NC_{NXC}, MOD_{NXC})CØNT

(MAT, 1, 0 /0.0, 0.0, 0 , 0 , 0 , 0)SEND

1.1.2. Procedures

The flag LRP indicates whether resolved and/or unresolved resonance parameter data are to be found in File 2 (Resonance Parameters). Every material will have a File 2 unless only file 1 is present, but not every File 2 will contain resonance parameter data. File 2 for certain materials will contain a scattering length (see sections 2.1. and 3.2.2.). For cases in which File 2 contains information on the scattering length only, LRP will be set at zero.

*Note: ZSYMA to ENDATE are part of H(N)

The flag LFI = 1 indicates that this material is fissionable. In this case, a section specifying the total number of neutrons per fission, $\nu(E)$, must be given, i.e., MF = 1, MT = 452. Sections may also be given that specify fission product yields (MF = 8, MT = 454 and 459), the number of delayed neutrons per fission (MF =1, MT = 455), and the number of prompt neutrons per fission (MF = 1, MT = 456).

The descriptive data in the Hollerith section must be given for every material. The first card image should be a self-contained title for the material. (This title should contain a material identification, name of the person and laboratory preparing the evaluation, and a date). The remaining card images should give a verbal description of the evaluated data sets for the material. This should include mention of the important experimental results upon which the recommended cross sections were based, the evaluation procedures, brief history and origin of evaluation, resonance integrals and thermal values, and references. Also, any limitations on the use of the particular data set should be clearly pointed out, along with other remarks that will assist the user in understanding the data. The 2200-m/sec cross sections contained in the data set should be given. This information is not always easy to find, since there may be contributions from resolved resonance parameters. The infinite dilution resonance integrals should be given for the radiative capture cross sections and the fission cross section (if applicable).

If the material is an element containing more than one naturally occurring isotope, the basis for establishing the reaction Q-values (given in File 3) should be explained.

Use of MOD in the dictionary (1-451).

1. MOD for any MF and MT must be $\leq$ NMOD, the material MOD number found in field 6 on the first card in MT = 451.

2. Any new section or revision of an old section requires that MOD for that MF, MT in the dictionary should be set equal to NMOD. Any section that is a carry-over from a previous version of ENDF/B should have MOD=0.

1.2. Number of Neutrons per Fission, $\bar{\nu}$, (MT = 452)

If the material is fissionable (LFI = 1), then a section specifying the average total number of neutrons per fission, $\bar{\nu}$ (MT = 452) must be given. $\bar{\nu}$ is given as a function of incident neutron energy. The energy dependence of $\bar{\nu}$ may be found by tabulating $\bar{\nu}$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}(E)$,

$$\bar{\nu}(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}(E)$ is the average total (prompt plus delayed) number of neutrons per fissions produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

1.2.1. Formats

The structure of this section depends on whether values of $\bar{\nu}(E)$ are tabulated as a function of incident neutron energy or whether $\bar{\nu}$ is represented by a polynomial. The following quantities are defined:

LNU is a test that indicates what representation of $\bar{\nu}(E)$ has been used:
LNU = 1, polynomial representation has been used;
LNU = 2, tabulated representation.

NC is a count of the number of terms used in the polynomial expansion. ($NC \leq 4$)

C_n are the coefficients of the polynomial. There are NC coefficients given.

NR is the number of interpolation ranges used to tabulate values of $\bar{\nu}(E)$. (See Appendix E.)

NP is the total number of energy points used to tabulate $\bar{\nu}(E)$.

E_{int} is the interpolation scheme (see Appendix E for details.)

$\bar{\nu}(E)$ is the average number of neutrons per fission.

If LNU = 1, the structure of the section is

(MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 452/ 0.0, 0.0, 0, 0 , NC, 0/C_1, C_2, ... C_{NC})LIST

(MAT, 1, 0 / 0.0, 0.0, 0, 0 , 0, 0)SEND

If LNU = 2, the structure of the section is

(MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 452/ 0.0, 0.0, 0, 0, NR, NP/E_{int}/$\bar{\nu}(E)$)TAB1

(MAT, 1, 0/ 0.0, 0.0, 0, 0, 0, 0)SEND

1.2.2. Procedures

If a polynomial representation (LNU = 1) has been used to specify $\bar{\nu}(E)$, this representation is valid over any range in which the fission cross section is specified (as given in Files 2 and 3). When using a polynomial to fit $\bar{\nu}(E)$, the fit shall be limited to a third-degree polynomial (NC = 4). If such a fit does not reproduce the recommend values of $\bar{\nu}(E)$, a tabulated form (LNU = 2) should be used.

If tabulated values of $\bar{\nu}$ are specified (LNU = 2), then pairs of energy-$\bar{\nu}$ values are given. Values of $\bar{\nu}(E)$ should be given that cover any energy range in which the fission cross section is given in File 2 and/or File 3.

The values of $\bar{\nu}(E)$ given in this section are for the average total number of neutrons produced per fission event. Even though another section (MT = 455) that specifies the delayed neutron from fission may be given, $\bar{\nu}_d$, the average number of delayed neutrons per fission must be included in the values of $\bar{\nu}(E)$ given in this section (MT = 452).

1.3. Delayed Neutron Data, $\bar{\nu}_d$, (MT = 455)

This section describes the delayed neutrons resulting from fission events. The average total number of delayed neutron precursors emitted per fission, $\bar{\nu}_d$ is given, along with the decay constants, λ_i, for each precursor family. The fraction of $\bar{\nu}_d$ generated for each family is given in File 5 (section 5. of this report). The energy distributions of the secondary neutrons associated with each precursor family are also given in File 5.

The total number of delayed neutron precursors is given as a function of incident neutron energy. Two representations are provided to specify the energy dependence. They are the same as those used in this file, (MT = 452), to describe the average total number of neutrons produced per fission event (see section 1.2.). The incident energy dependence may be specified by tabulating $\bar{\nu}_d(E)$ at a series of incident neutron energies or by providing the coefficients of a polynomial expansion in energy.

The total number of delayed neutron precursors emitted per fission event, at incident energy E, is given in this file and is defined as the sum of the number of precursors emitted for each of the precursor families,

$$\bar{\nu}_d(E) = \sum_{i=1}^{NNF} \bar{\nu}_i(E) ,$$

where NNF is the number of precursor families. The fraction of the total, $P_i(E)$, emitted for each family is given in File 5 (see section 5) and is defined as

$$P_i(E) = \frac{\bar{\nu}_i(E)}{\bar{\nu}_d(E)}$$

1.3.1. Formats

The structure of a section depends on whether $\bar{\nu}_d(E)$ is tabulated as a function of incident energy or given as coefficients of a polynomial expansion in energy. If a polynomial is used, $\bar{\nu}_d(E)$ is defined as

$$\bar{\nu}_d(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}(E)$ is the average number of delayed neutrons per fission produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

The following quantities are defined:

LNU is a test that indicates which representation is used:
LNU = 1 means that a polynomial expansion is used;
LNU = 2 means that a tabulated representation is used.

NC is the number of terms in the polynomial expansion. (NC $\leq$ 4)

C_n are the coefficients for the polynomial.

NR is the number of interpolation ranges used. (NR $\leq$ 20)

NP is the total number of incident energy points used to represent $\bar{\nu}_d(E)$ when a tabulation is used.

E_{int} is the interpolation scheme (see Appendix E).

$\bar{\nu}_d(E)$ is the total average number of delayed neutron precursors formed per fission event.

NNF is the number of precursor families considered.

λ_i is the decay constant (sec^{-1}) for the i^{th} precursor.

The structure of a section when a polynomial representation has been used (LNU = 1) is

(MAT, 1, 455/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NNF, 0/λ_1, λ_2,...λ_{NNF})LIST

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NC, 0/C_1, C_2,...C_{NC})LIST

(MAT, 1, 0 / 0.0, 0.0, 0, 0, 0, 0)SEND

The structure when values of $\bar{\nu}_d$ are tabulated (LNU = 2) is

(MAT, 1, 455/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NNF, 0/λ_1, λ_2,...λ_{NNF})LIST

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NR, NP/E_{int}/$\bar{\nu}_d$(E))TAB1

(MAT, 1, 0 / 0.0, 0.0, 0, 0, 0 0)SEND

1.3.2. Procedures

When the polynomial representation is used, the calculated values of $\bar{\nu}_d$(E) may be used over any range in which the fission cross section has been given in Files 2 and/or 3. When tabulated values of $\bar{\nu}_d$(E) are specified, they should be given for the same energy range as that used to specify the fission cross section.

The probability of producing the precursors for each family and the energy distributions of neutrons produced by each precursor family are given in File 5 (section 5 of this report). It is extremely important that the same precursor families be given in File 5 as are given in File 1 (MT = 455), and the ordering of the families should be the same in both files. It is recommended that the families be ordered by decreasing half-lives ($\lambda_1 < \lambda_2 < \ldots < \lambda_{NNF}$).

1.4. Number of Prompt Neutrons per Fission, $\bar{\nu}_p$, (MT = 456)

If the material is fissionable (LFI = 1), a section specifying the average number of prompt neutrons per fission, $\bar{\nu}_p$, (MT = 456) can be given using formats identical to MT = 452. $\bar{\nu}_p$ is given as a function of incident neutron energy. The energy dependence of $\bar{\nu}_p$ may be given by tabulating $\bar{\nu}_p$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}_p(E)$.

$$\bar{\nu}_p(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}_p(E)$ is the average number of prompt neutrons per fission produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

1.4.1. Formats

The structure of this section depends on whether values of $\bar{\nu}(E)$ are tabulated as a function of incident neutron energy or whether $\bar{\nu}$ is represented by a polynomial. The following quantities are defined:

LNU is a test that indicates what representation of $\bar{\nu}(E)$ has been used;

LNU = 1, polynomial representation has been used;

LNU = 2, tabulated representation.

NC is a count of the number of terms used in the polynomial expansion. (NC $\leq$ 4)

C_n are the coefficients of the polynomial. There are NC coefficients given.

NR is the number of interpolation ranges used to tabulate values of $\bar{\nu}_p(E)$. (See Appendix E.)

NP is the total number of energy points used to tabulate $\bar{\nu}(E)$.

E_{int} is the interpolation scheme (see Appendix E.)

$\bar{\nu}_p(E)$ is the average number of prompt neutrons per fission.

If LNU = 1 (polynomial representation used), the structure of the section is

(MAT, 1, 456/ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 456/0.0, 0.0, 0, 0, NC, 0/C_1, C_2, ... C_{NC})LIST

(MAT, 1, 0/ 0.0, 0.0, 0, 0, 0, 0)SEND

If LNU = 2 (tabulated values of $\bar{\nu}$), the structure of the section is

(MAT, 1, 456/ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 456/0.0, 0.0, 0, 0, NR, NP/E_{int}/$\bar{\nu}_p(E)$)TAB1

(MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0)SEND

1.4.2. Procedures

If a polynomial representation (LNP = 1) has been used to specify $\bar{\nu}_p(E)$, this representation is valid over any range in which the fission cross section is specified (as given in Files 2 and 3). When using a polynomial to fit $\bar{\nu}_p(E)$, the fit shall be limited to a third-degree polynomial (NCP = 4). If such a fit does not reproduce the recommended values of $\bar{\nu}_p(E)$ a tabulated form (LNP = 2) should be used.

If tabulated values of $\bar{\nu}_p(E)$ are specified (LNP = 2), then pairs of energy-$\bar{\nu}$ values are given. Values of $\bar{\nu}_p(E)$ should be given that cover any energy range in which the fission cross section is given in File 2 and/or File 3.

The values of $\bar{\nu}_p(E)$ given in this section are for the average number of prompt neutrons produced per fission event. Even though another section (MT = 455) that specifies the delayed neutron from fission may be given; $\bar{\nu}_d$, the number of delayed neutrons per fission, and $\bar{\nu}_p$, the number of prompt neutrons per fission, must be included in the values of $\bar{\nu}(E)$ given in the section (MT = 452); i.e., $\bar{\nu}(MT = 452) = \bar{\nu}_d(MT = 455) + \bar{\nu}_p(MT = 456)$.

1.5 Number of Neutrons from the Spontaneous Fissioning of the Target Nucleus (MT = 465, 466)

MT = 465 for $\bar{\nu}_{SD}$ (delayed component of spontaneous fission $\bar{\nu}$).

MT = 466 for $\bar{\nu}_{SP}$ (prompt component of spontaneous $\bar{\nu}$).

1.5.1. Formats

(MAT, 1, 465/	ZA	AWR	0	2	0	0)	HEAD*
(MAT, 1, 465/	0.0	0.0	0	0	1	2)	TAB 1
(MAT, 1, 465/	2	1	0	0	0	0)	
(MAT, 1, 465/	1.0E-5	$\bar{\nu}_{SD}$	2.0E+7	$\bar{\nu}_{SD}$	0	0)	
(MAT, 1, 0/	0.0	0.0	0	0	0	0)	SEND

and for MT = 466, the same format would be followed.

1.5.2. Procedures

Files 3 and 4 are not required for MT = 465 and 466.

*LNU = 1 (polynomial expansion) is not allowed as input.

1.6. Components of the Energy Release Due to Fission (MT = 458)

The energy released in fission is carried by fission fragments, neutrons, gammas, betas (+ and -), and (anti-) neutrinos. The term fragments includes all charged particles that are emitted promptly, since for energy-deposition calculations, all such particles have short ranges and are usually considered to lose their energy locally. Neutrons and gammas transport their energy elsewhere and need to be considered separately. In addition, some gammas and neutrons are delayed, and in a shut-down assembly one needs to know the amount of energy tied up in these particles and the rate at which it is released from the metastable nuclides or precursors. The neutrino energy is lost completely in most applications, but is part of the Q-value. As far as the betas are concerned, any prompt ones would deposit their energy localy with the fragments, being charged, and their prompt energy is correctly included with the fragment energy.

ET is the sum of all the partial energies which follow. This sum is the total energy release per fission and equals the Q value.

EFR is the kinetic energy of the fragments

ENP is the kinetic energy of the "prompt" fission neutrons

END is the kinetic energy of the delayed fission neutrons

EGP is the total energy released by the emission of "prompt" γ rays

EGD is the total energy released by the emission of delayed γ rays

EB is the total energy released by delayed β's

ENU is the energy carried away by the neutrinos

ER is ET - ENU (the total energy less the energy of the neutrinos). This ER is equal to the pseudo-Q in File 3 for MT = 18.

$E_i(0) = E_i(EINC) + \delta E_i$

where E_i is any of the energy release components

$E_i(0)$ is the value at EINC = 0

$E_i(EINC)$ is the value at incident energy EINC

EINC = 0 is ficticious and represents an artifice by which it is possible to recover the values at any EINC.

The δE_i's are given by the following:

$\delta QG = EINC - (1.057\ EINC - 8.07(\nu(EINC) - \nu(0)))$

$= - 0.057\ EINC + 8.07(\nu(EINC) - \nu(0))$.

$\delta EB = 0.075\ EINC$.

$\delta EGD = 0.075\ EINC$.

$\delta ENU = 0.100\ EINC$.

$\delta EFR = 0$.

$\delta ENP = - (1.307\ EINC - 8.07\ (\nu(EINC) - \nu(0))$.

$\delta EGP = 0$

1.6.1. Formats

The structure of this section always starts with a HEAD record and end with a SEND record. The section contains no subsections and only one LIST record.

The structure of a section is:

(MAT, 1, 458/ ZA, AWR, 0, 0, 0, 0) HEAD

(MAT, 1, 458/ 0.0, 0.0, 0, 0, 18, 9/

EFR, ΔEFR, ENP, ΔENP, END, ΔEND

EGP, ΔEGP, EGD, ΔEGD, EB, ΔEB

ENU, ΔENU, ER, ΔER, ET, ΔET) LIST

(MAT, 1, 0/ 0.0, 0.0, 0, 0, 0, 0) SEND,

where the Δ's allow the error estimates on the quantities listed above.

1.6.2. Procedures

This section should be used for fertile and fissile isotopes only. ($Z \geq 88$) for ENDF/B-V.

Consistency should be maintained between the Q values in File 3, the energies calculated from Files 5 and 15 and the energies listed in File 1. Note that ER = the pseudo-Q for fission (MT = 18) in File 3.

Other components are not so readily determined or checked. The procedure should be that File 5 and File 15 data take precedent, whenever available. That is, the "prompt" fission neutron energy calculated from File 5 spectra from MT = 18 should be used in File 1; the same holds true for the delayed neutron spectra given in File 5, MT = 455. The "prompt" gamma energy calculated from File 15 (MT = 18 for fission) should be input into File 1, that is the prompt gammas due to the fission process.

These quantities should be calculated at the lowest energy given in the Files for MT = 18 except for fissile isotopes for which the thermal spectra should be used. For fertile materials, the spectrum given at threshold would be appropriate. Note that the File 5 spectra for MT = 18 should be used with $\bar{\nu}$ prompt (not $\bar{\nu}$ total) for the fission neutrons. MT = 455 in File 5 contains the delayed fission neutron spectra.

In many reactor applications, time dependent energy deposition rates are required rather than the components of the total energy per fission which are the values given in this MT. Time-dependent energy deposition parameters can be obtained from the six-group spectra in File 5 (MT = 455) for delayed neutrons. Codes such as CINDER, RIBID, and ORIGEN must be used, however, to obtain more detailed information on the delayed neutrons and <u>all</u> time-dependent parameters for the betas and the gammas due to the fission process.

The time-integrated energies for delayed neutrons, delayed gammas, and delayed betas as calculated from the codes listed above may not always agree with the energy components given in File 1. The File 1 components must sum to ET (the total energy released per fission).

In heating calculations, the energy released in all nuclear reactions besides fission, principally the gamma-energy released in neutron radiative capture, enters analogously to the various fission energy components. Thus the (n,γ) energy-release would be equal to the Q-value in file 3, MT=102, of the capturing nuclide. The capture gammas can be prompt or delayed, if branching to isomeric states is involved, and this is relevant to various fission- and burnup-product calculations. The "sensible energy" in a heating calculation is the sum of ER, defined previously, and the energy released in these other reactions.

2. FILE 2, RESONANCE PARAMETERS

2.1. General Description

Every material must contain a File 2. It has only one section, which has been assigned the reaction type number MT = 151. If no resonance parameters are given for the material, the purpose of the File 2 section is to specify the scattering radius, AP. If resonance parameters are given for some isotopes of an elemental evaluation, an isotope without resonance parameters may be included. If it is included, the subsection for the first and only energy range of that isotope will consist of a single CONT record giving the spin, SPI, and scattering radius, AP. In these two cases, the inclusion of a scattering radius is solely for the convenience of users who wish an estimate of the potential scattering. It is not used to calculate a contribution to the scattering cross section, which in these two cases, is represented entirely in File 3.

The primary function of File 2 is to contain data for both resolved and unresolved resonance parameters. When resonance parameters are given the total (MT = 1), elastic scattering (MT = 2), fission (MT = 18), and radiative capture (MT = 102) cross sections given in File 3 must be added to corresponding contributions calculated from the resolved and/or unresolved parameters given in File 2 in order to obtain the correct reaction cross sections.

All other reactions, if any, are grouped together into a single "competitive width," Γ_x. This width is given explicitly in the unresolved resonance region, and implicitly in the resolved region. In the latter region, it is permissible for the total width to exceed the sum of the neutron, radiative capture, and fission widths. The difference is interpreted as the competitive width:

$$\Gamma_x = \Gamma - (\Gamma_n + \Gamma_\gamma + \Gamma_f) \quad .$$

By convention, the individual competitive reactions (if any) given in File 3 represent the entire cross section and the contribution from File 2 is not to be added to them. (See Appendix D. Section D.3.3.)

For those isotopes with resonance parameters, the potential scattering term is given by

$$\sigma_{nn}^{\ell} = \frac{4\pi}{k^2} (2\ell + 1) \sin^2 \phi_\ell, \quad \ell = 0, 1, 2,$$

where

$$k = 2.196771 \frac{AWRI}{AWRI + 1.0} \times 10^{-3} \sqrt{E}, \quad E \text{ in eV},$$

$$\phi_0 = \hat{\rho}, \ \phi_1 = \hat{\rho} - \tan^{-1} \hat{\rho}, \ \phi_2 = \hat{\rho} - \tan^{-1} \frac{3\hat{\rho}}{3 - \hat{\rho}^2},$$

and

$$\hat{\rho} = k(AP).$$

In order to calculate the penetrability and shift factors when resonances are present, the current ENDF convention is to employ the channel radius, a, also known as the "hard-sphere radius" or the "nuclear radius,"

$$a = .123 \ AWRI^{1/3} + .08 \quad \text{in units of } 10^{-12} \text{ cm.}$$

The resonance parameter data for a material are obtained by specifying the parameters for each isotope in the material. The data for the various isotopes

are ordered by increasing ZAI values (charge-isotopic mass number). The data for each isotope may be divided into several incident neutron energy ranges, and the data for the energy ranges are ordered by increasing energy. The energy ranges should not overlap; each will contain a different representation of the resonance parameters. Normally two energy ranges will be specified for each isotope. The first will contain resolved parameters, and the second, unresolved resonance parameters. The problems connected with these restrictions are discussed in the Procedures Sections, 2.2.2 and 2.4.

Several representations are allowed for specifying the resolved resonance parameters. The particular representation used for a particular energy range is indicated by a flag, LRF.

The allowed representations for the resolved resonance parameters are

LRF = 1, single-level Breit-Wigner;

LRF = 2, multilevel Breit-Wigner (resonance-resonance interference effects are included in the elastic scattering cross section and the total cross section):

LRF = 4, Adler-Adler multilevel (resonance-resonance interference effects are included in all cross sections).

For fissile nuclides, the preferred formalism for evaluation purposes is the Reich-Moore multilevel scheme,* because it has less flexibility than the Adler method and is therefore better able to distinguish between various grades of experimental data. However, for purposes of presentation in ENDFB, Reich-Moore evaluations must be converted to Adler format, since the latter permits the use of ψ- and χ-functions for Doppler-broadening, whereas the Reich-Moore

*Previously allowed as LRF = 3; now restricted to ENDF/A.

scheme requires kernel methods. Further discussion of the above three formalisms is contained in the Procedures Sections, 2.2.2 and 2.4.

Each energy range contains a flag, LRU, that indicates whether it contains resolved or unresolved resonance parameters. LRU = 1 means resolved, LRU = 2 means unresolved.

Only one representation is allowed for the unresolved resonance parameters, namely average single-level Breit-Wigner. However, several options are permitted, designated by the Flag LRF. With the first option, LRF = 1, only the average fission width is allowed to vary as a function of incident neutron energy. The second option, LRF = 2, allows the following average parameters to vary: level spacing, fission width, reduced neutron width, radiation width, and a width for an unspecified competitive reaction.

The data formats for the various resonance parameter representations are given in Sections 2.2.1 (resolved) and 2.3.1 (unresolved). The formulae for calculating cross sections for the various resonance region theories are given in Appendix D.

Several quantities used in File 2 have definitions that are the same for all resonance parameter representations:

NIS is the number of isotopes in this material (NIS $\leq$ 10).

ZAI is the (Z,A) designation for an isotope.

ABN is the abundance of an isotope in the material. This is a number fracton, not a weight fraction, nor a percent.

LFW is a flag indicating whether average fission widths are given in the unresolved resonance region for this isotope:

LFW = 0, average fission widths are not given;

LFW = 1, average fission widths are given.

NER is the number of energy ranges for this isotope (NER $\leq$ 2).

EL is the lower limit for an energy range.*

EH is the upper limit for an energy range.*

LRU is a flag indicating whether this energy range contains data for resolved or unresolved resonance parameters:

LRU = 0, only the scattering radius is given (LRF = 0, NLS = 0, LFW = 0 is required with this option);

LRU = 1, resolved resonance parameters are given;

LRU = 2, unresolved resonance parameters are given.

LRF is a flag indicating which representation has been used for the energy range. The definition of LRF depends on the value of LRU:

If LRU = 1 (resolved parameters), then

LRF = 1, single-level Breit-Wigner (SLBW) parameters;

LRF = 2, multilevel Breit-Wigner (MLBW) parameters;

LRF = 4, Adler-Adler parameters;

If LRU = 2 (unresolved parameters), then

LRF = 1, only average fission widths are energy-dependent;

LRF = 2, average level spacing, competitive reaction widths, reduced neutron widths, radiation widths, and fission widths are energy-dependent.

*These energies are the limits to be used in calculating cross sections from the parameters. Some resolved resonance levels, e.g., bound levels, will have resonance energies outside the limits.

The general structure of a section is as follows:

(MAT, 2, 151/ ZA, AWR, 0, 0, NIS, 0)HEAD

(MAT, 2, 151/ ZAI, ABN, 0, LFW, NER, 0)CONT (isotope)

(MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0)CONT (range)

<Subsection for the first energy range for the first isotope (depends on LRU and LRF)>

(MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0)CONT (range)

<Subsection for the second energy range for the first isotope (depends on LRU and LRF)>

(MAT, 2, 151/ EL, EH, LRU, LRF, 0 0)CONT (range)

<Subsection for the last energy range for the last isotope for this material>

(MAT, 2, 0 / 0.0, 0.0, 0, 0, 0, 0)SEND

The data are given for all ranges for a given isotope, and then for all isotopes. The data for each range start with a CONT (range) record; those for each isotope, with a CONT (isotope) record. The specifications for the subsections which include resonance parameters are given in Sections 2.2.1 and 2.3.1, below.

The structure of File 2 for the special case, in which just the scattering radius is specified, is given below (no resolved or unresolved parameters are given for this material):

(MAT, 2, 151/ ZA, AWR, 0, 0, NIS, 0)HEAD (NIS = 1)

(MAT, 2, 151/ ZAI, ABN, 0, LFW, NER, 0)CONT (LFW = 0, NER = 1)

(MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0)CONT (LRU = 0, LRF = 0)

(MAT, 2, 151/ SPI, AP, 0, 0, NLS, 0)CONT (NLS = 0)

(MAT, 2, 0 / 0.0, 0.0, 0, 0, 0, 0)SEND

(MAT, 0, 0 / 0.0, 0.0, 0, 0, 0, 0)FEND

The structure of a subsection for an isotope for which no resonance parameters are given is:

(MAT, 2, 151/ SPI, AP, 0, 0, NLS, 0)CONT (NLS = 0)

and as above LFW = 0, NER = 1, LRU = 0, and LRF = 0 for this isotope.

2.2. Resolved Resonance Parameters (LRU = 1)

2.2.1. Formats

Three different resonance formulations are allowed to represent the resolved resonance parameters. The pertinent formulae associated with these representations are given in detail in Appendix D. The flag LRU = 1, given in the CØNT (range) record, indicates that resolved resonance parameters are given for a particular energy range. Another flag, LRF, in the same record specifies which resonance formulation has been used.

The structure of a subsection is the same for LRF = 1 (single-level Breit-Wigner parameters) as it is for LRF = 2 (multi-level Breit-Wigner parameters). The following quantities are defined for use when LRF = 1 and 2 (see Appendix D for formulae):

Resolved Resonance Parameters if LRF = 1 (SLBW) and LRF = 2 (MLBW)

SPI is the nuclear spin of the target nucleus, I (positive number).

AP is the scattering radius in units of 10^{-12} cm.

NLS is the number of sets of resonance parameters given in this energy region. A set of parameters is given for each ℓ-value (neutron orbital angular momentum). (NLS $\leq$ 3.)

AWRI is the ratio of the mass of a particular isotope to that of a neutron.

QX is an effective Q-value to be added to the incident particle's center-of-mass energy to determine the channel energy for use in the penetrability factor of the competitive width. The conversion to a laboratory system energy depends on the reduced mass in the exit channel. For inelastic scattering to a discrete level, the Q-value is minus the level excitation energy. QX = 0.0 if LRX = 0.

L is the value of ℓ.

LRX is a flag indicating whether this energy range contains a competitive width:*

LRX = 0, no competitive width is given, and $\Gamma = \Gamma_n + \Gamma_\gamma + \Gamma_f$ in the resolved resonance region, while $\langle\Gamma_x\rangle = 0$ in the unresolved resonance region;

LRX = 1, a competitive width is given, and is effectively a single inelastic process to the first excited state. In the resolved region, it is determined by subtraction, $\Gamma_x = \Gamma - (\Gamma_n + \Gamma_\gamma + \Gamma_f)$;

LRX = 2, like LRX = 1, but more than one competitive reaction is energetically possible. (not yet implemented)

NRS is the number of resolved resonances for a given ℓ-value. (NRS $\leq$ 500.)

ER is the resonance energy (in the laboratory system).

AJ is the floating point value of J (the spin, or total angular momentum of the resonance).

GT is the resonance total width Γ evaluated at the resonance energy ER.

GN is the neutron width Γ_n evaluated at the resonance energy ER.

GG is the radiation width Γ_γ evaluated at the resonance energy ER.

GF is the fission width Γ_f evaluated at the resonance energy ER.

GX is the competitive width Γ_x evaluated at the resonance energy ER. It is not given explicitly, but is to be obtained by subtraction, GX = GT - (GN + GG + GF), if LRX $\neq$ 0;

The structure of a subsection containing data for (LRU = 1 and LRF = 1) or (LRU = 1 and LRF = 2) is

*A discussion of the penetrability factor to be associated with the competitive width is contained in Appendix D.

(MAT, 2, 151/ SPI, AP, 0, 0, NLS, 0)CØNT

(MAT, 2, 151/ AWRI, QX, L, LRX, 6*NRS, NRS/

ER_1, AJ_1, GT_1, GN_1, GG_1, GF_1,

ER_2, AJ_2, GT_2, GN_2, GG_2, GF_2,

ER_{NRS}, AJ_{NRS}, GT_{NRS}, GN_{NRS}, GG_{NRS}, GF_{NRS})LIST

The LIST record is repeated until each NLS ℓ-values has been specified (in order of increasing value of ℓ). The values of ER for each ℓ-state shall be ordered by increasing neutron energy.

Resolved Resonance Parameters if LRF = 4 (Adler-Adler multilevel parameters)

LI is a flag to indicate the kind of parameters given:

If LI = 1, total widths only*

= 2, fission widths only*

= 3, total and fission widths*

= 4, radiative capture widths only*

= 5, total and capture widths

= 6, fission and capture widths*

= 7, total, fission, and capture widths.

NX is the count of the number of sets of background constants to be given. There are six constants per set. Each set refers to a particular cross section type. The background correction for the total cross section is calculated by using the six constants in the following manner:

$$\sigma_T \text{ (background)} = \frac{C}{\sqrt{E}} (AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1*E + BT_2*E^2)$$

*Reserved for use in ENDF/A only. Formulas and formats for the Reich-Moore formalism, also restricted to ENDF/A, are given in Appendix D.

where $C = \pi\lambda^2 = \pi/k^2$ and $k = 2.196771 \times 10^{-3} \left(\frac{AWRI}{AWRI + 1.0}\right) \sqrt{E(eV)}$ is the center-of-mass wave number in terms of the laboratory energy. The background terms for the fission and radiative capture cross sections are calculated in a similar manner.

If NX = 2, background constants are given for the total and capture cross sections.

= 3, background constants are given for the total, capture, and fission cross sections.

AJ is the floating-point value of J (the spin of the resonance).

L is the value of ℓ.

NLS is the number of ℓ-values for which parameters will be given (NLS $\leq$ 3).

NJS is the number of sets of resolved resonance parameters (each set having its own J-value) for a specified ℓ.

NLJ is the number of resonances for which parameters are given, for a specified AJ and L.

SPI is the spin of the target nucleus, I.

AWRI is the ratio of the mass of a particular isotope to that of the neutron.

AP is the scattering radius in units of 10^{-12} cm. It is assumed to be independent of channel spin.

AT_1, AT_2, AT_3, AT_4, BT_1, BT_2 are the background constants for the total cross section.

AF_1, AF_2, AF_3, AF_4, BF_1, BF_2 are the background constants for the fission cross section.

$\underline{AC_1}$, $\underline{AC_2}$, $\underline{AC_3}$, $\underline{AC_4}$, $\underline{BC_1}$, $\underline{BC_2}$ are the background constants for the radiative capture cross section.

$\underline{DET_r}$ is the resonance energy, (μ), for the total cross section. Here and below, the subscript r denotes the r^{th} resonance.

$\underline{DEF_r}$ is the resonance energy, (μ), for the fission cross section.

$\underline{DEC_r}$ is the resonance energy, (μ), for the radiative capture cross section.

$\underline{DWT_r}$ is the value of $\Gamma/2$, (ν), used for the total cross section.

$\underline{DWF_r}$ is the value of $\Gamma/2$, (ν), used for the fission cross section.

$\underline{DWC_r}$ is the value of $\Gamma/2$, (ν), used for the radiative capture cross section.

Note: $DET_r = DEF_r = DEC_r$ and $DWT_r = DWF_r = DWC_r$. The redundancy is an historical carryover.

$\underline{GRT_r}$ is the symmetrical total cross section parameter, G_r^T.

$\underline{GIT_r}$ is the asymmetrical total cross section parameter, H_r^T.

$\underline{GRF_r}$ is the symmetrical fission parameter, G_r^f.

$\underline{GIF_r}$ is the asymmetrical fission parameter, H_r^f.

$\underline{GRC_r}$ is the symmetrical capture parameter, G_r^γ.

$\underline{GIC_r}$ is the asymmetrical capture parameter, H_r^γ.

The structure of a subsection containing data for (LRU = 1 and LRF = 4) depends on the value of NX (the number of sets of background constants). For the most general case (NX = 3) the structure is

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0)CØNT)

(MAT, 2, 151/AWRI, 0.0, LI, 0, 6*NX, NX/

AT_1, AT_2, AT_3, AT_4, BT_1, BT_2

AF_1, ----------, BF_2

AC_1, ----------, BC_2)LIST)

(MAT, 2, 151/0.0, 0.0, L, 0, NJS, 0)CØNT(ℓ)

(MAT, 2, 151/AJ, 0.0, 0, 0, 12*NLJ, NLJ/

DET_1, DWT_1, GRT_1, GIT_1, DEF_1, DWF_1,

GRF_1, GIF_1, DEC_1, DWC_1, GRC_1, GIC_1,

DET_2, DWT_2, ----------

----------, GIC_2,

DET_3, ----------

----------, GIC_{NLJ})LIST

The last LIST record is repeated for each J-state (there will be NJS such LIST records). A new CØNT (ℓ) record will be given which will be followed by NJS LIST records. Note that if NX = 2 then the quantities AF_1,----, BF_2 will not be given in the first LIST record. Also, if LI ≠ 7 then certain of the parameters for each level may be set at zero, i.e., the fields for parameters not given (depending on LI) will be set to zero.

Since the format has no provision for giving the Adler-Adler parameters for the scattering cross-section, this is obtained by subtracting the sum of capture and fission cross sections from the total cross section.

2.2.2. Procedures for the Resolved Resonance Region*

For certain resonances the value of ℓ is known but the resonance spin J is not. In such a case, the resonance spins J may be assigned to follow the level density law $\rho_{\ell J} = \frac{1}{D_{\ell J}} \cdot (2J+1)$ where $\rho_{\ell J}$ is the density of compound nucleus levels of spin J** and $D_{\ell J}$ their spacing. The statistical weight factors g_J corresponding to resonances of spin J are such that $\sum_{J,s} g_J = (2\ell+1)$ where the

*See also Section 2.4.

**And the parity which corresponds to the ℓ-value.

summation is over the different spin states J and the channel spins s, each taken as many times as it occurs.† The resonance spins, if not known, should not be set equal to the target nucleus spin, since this can result in negative scattering cross sections even when MLBW is specified.*

The upper (EH) and lower (EL) energy limits of an energy range indicate the energy range of validity for the given parameters for calculating cross sections. Outside this energy range the cross sections must be obtained from the parameters given in another energy range and/or from data in File 3.

It is sometimes necessary to give parameters whose energies lie outside a specified energy range in order to compute the cross section for neutron energies that are within the energy range. (For example, the inclusion of bound levels may be required to predict the cross section at low energies, and resonances will usually be needed above EH to compensate the opposite, positive, bias at the high energy end.

For materials that contain more than one isotope, it is recommended that the lower energy limit of the resolved resonance region be the same for all isotopes. It is also recommended that the upper energy limit for the unresolved resonance range be same for all isotopes. If resolved and/or unresolved resonance parameters are given for only some of the naturally occurring isotopes, then AP should be given for the others, unless they have been included by increasing the abundance of a neighboring isotope and the fact has been noted in the summary documentation.

If more than one energy range is used, they must be contiguous and not overlap. It is a current restriction that the data for each isotope be divided

†See Appendix D. Section D.2.2.2.

*An exception to this rule can be made if none of the spins are known, and all J-values are set equal to I.

into no more than two energy ranges, one for resolved and the other for unresolved resonance parameters.

In the single-level Breit-Wigner formalism it is common for negative cross sections to occur, and this formalism ought not to be used for actual line-shape calculations. The problem can be avoided by the following procedures:

1. Use of the MLBW formalism, together with the addition of negative-energy levels to compensate the end-effect bias. High-energy resonances can likewise be added.
2. Insertion of File 3 "background" to produce a File 2 + File 3 cross section which is positive.

Further discussion of "Procedures" will be found in Section 2.4.

2.3. Unresolved Resonance Parameters (LRU = 2)

2.3.1. Formats

Only the SLBW formalism for unresolved resonance parameters is allowed (see Appendix D for pertinent formulae). However, several options are available for specifying the energy-dependence of the parameters, designated by the flag LRF. Since unresolved resonance parameters are averages of resolved resonance parameters over energy, they are constant with respect to energy throughout the energy-averaging interval. However, they are allowed to vary from interval to interval, and it is this energy-dependence which is referred to above and in the following paragraphs.

The parameters depend on both ℓ (neutron orbital angular momentum) and J (total angular momentum). Each width is distributed according to a chi-squared distribution with a certain number of degrees of freedom. This number may be different for neutron and fission widths and for different (ℓ,J) channels.

The following quantities are defined for use in specifying unresolved resonance parameters (LRU =2):

SPI is the spin of the target nucleus, I.

AP is the scattering radius in units of 10^{-12} cm. No channel quantum number dependence is currently permitted by the format.

NE is the number of energy points at which energy-dependent widths are tabulated. (NE $\leq$ 250.)

NLS is the number of ℓ-values (NLS $\leq$ 3.)

ES_i is the energy of the i^{th} point used to tabulate energy-dependent widths.

L is the value of ℓ.

AWRI is the ratio of the mass of a particular isotope to that of the neutron.

NJS is the number of J-states for a particular ℓ-state. (NJS $\leq$ 6.)

AJ is the floating-point value of the J (the spin, or total angular momentum of the set of parameters).

D is the average level spacing for resonances with spin J. (It may be energy dependent if LRF = 2.)

AMUX is the number of degrees of freedom used in the competitive width distribution. (Assuming it is inelastic, 1.0 $\leq$ AMUX $\leq$ 2.0, determined by whether the spin of the first excited state is zero or not.)*

AMUN is the number of degrees of freedom in the neutron width distribution. (1.0 $\leq$ AMUN $\leq$ 2.0.)

*See Appendix D. Section D.2.2.6.

AMUG is the number of degrees of freedom in the radiation width distribution. (At present AMUG = 0.0. This implies a constant value of Γ_γ.)

AMUF is the number of degrees of freedom in the fission width distribution. ($1.0 \leq$ AMUF ≤ 4.0.)

MUF is the integer value of the number of degrees of freedom for fission widths. ($1 \leq$ MUF ≤ 4)

INT defines the interpolation scheme to be used for interpolating between the cross-sections obtained from average resonance parameters. Parameter interpolation is discussed in the Procedures Sections 2.3.2 and 2.4.

GNO is the average reduced neutron width. It may be energy-dependent if LRF = 2.

GG is the average radiation width. It may be energy-dependent if LRF = 2.

GF is the average fission width. It may be energy-dependent if LRF = 1 or 2.

GX is the average competitive reaction width, given only when LRF = 2, in which case it may be energy-dependent.

The structure of a subsection* depends on whether LRF = 1 or LRF = 2. If LRF = 1, only the fission width is given as a function of energy. If LRF = 1 and the fission width is not given (indicated by LFW = 0), then the simplest form of a subsection results. If LRF =2, energy-dependent values may be given for the level density, competitive width, reduced neutron width, radiation width, and fission width. Three sample formats are shown below (all LRU = 2).

*The structure of a section was defined previously, and covers both resolved resonance and unresolved resonance subsections.

A. LFW = 0 (fission widths not given),

LRF = 1 (all parameters are energy-independent).

The structure of a subsection is:

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0)CØNT

(MAT, 2, 151/AWRI, 0.0, L, 0, 6*NJS, NJS/

D_1, AJ_1, $AMUN_1$, GNO_1, GG_1, 0.0

D_2, AJ_2, $AMUN_2$, GNO_2, GG_2, 0.0

D_{NJS}, AJ_{NJS}, $AMUN_{NJS}$, GNO_{NJS}, GG_{NJS}, 0.0)LIST

The LIST record is repeated until data for all ℓ-values have been specified.

B. LFW = 1 (fission widths given),

LRF = 1 (only fission widths are energy-dependent; the rest are energy-independent).

The structure of a subsection is:

(MAT, 2, 151/SPI, AP, 0, 0, NE, NLS/

ES_1, ES_2, ES_3, .., .., ..

.., .., .., .., ES_{NE})LIST

(MAT, 2, 151/AWRI, 0.0, L, 0, NJS, 0)CØNT(ℓ)

(MAT, 2, 151/0.0, 0.0, L, MUF, NE+6, 0/

D, AJ, AMUN, GNO, GG, 0.0,

GF_1, GF_2, GF , .., .., ..

.., .., .., GF_{NE})LIST

The last LIST record is repeated for each J-state (there will be NJS such LIST records). A new CØNT(ℓ) record will then be given which will be followed by its NJS LIST records until data for all ℓ-values have been specified (there will be NLS sets of data).

In the above section, no provision was made for INT, and interpolation is assumed to be lin-lin.

C. LFW = 0 or 1 (does not depend on LFW).

LRF = 2 (all energy-dependent parameters).

The structure of a subsection is:

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0)CØNT

(MAT, 2, 151/AWRI, 0.0, L, 0, NJS, 0)CØNT

(MAT, 2, 151/AJ, 0.0, INT, 0, (6*NE)+6, NE /

0.0, 0.0, AMUX, AMUN, AMUG, AMUF,

ES_1, D_1, GX_1, GNO_1, GG_1, GF_1,

ES_2, D_2, GX_2, GNO_2, GG_2, GF_2,

ES_{NE}, D_{NE}, GX_{NE}, GNO_{NE}, GG_{NE}, GF_{NE})LIST

The LIST record is repeated until all the NJS J-states have been specified for a given ℓ-value. A new CØNT (ℓ) record is then given, and all data for each J-state for that ℓ-value are given. The structure is repeated until all ℓ-states have been specified.

2.3.2. Procedures for the Unresolved Resonance Region*

Recommended values for the various degrees of freedom are discussed in Section 2.4.21 and summarized in Appendix D, Section D.2.2.6. Examples A and B above make no provision for AMUG, AMUF, or AMUX, although example B specifies MUF. It is assumed that AMUG = 0.0 in both, that AMUF = MUF in B, and that example C (LRF = 2) will be used if a competitive reaction is given. The degrees of freedom are constant throughout the unresolved resonance region.

*See also Section 2.4.

Up to 250 energy points are permitted for specifying energy-dependent average parameters. This number is presumed to be sufficient to reproduce the gross structure in the unresolved cross sections. Within a given isotope the same energy grid must be used for all J-states and ℓ-values. The grids may be different for different isotopes. Unresolved resonance parameters should be provided for neutron energy regions where temperature-broadening or self-shielding effects are important. It is recommended that the unresolved resonance region extend up to at least 20 keV.

A point of confusion in the past has been "parameter interpolation" versus "cross section interpolation." The latter is the correct procedure, but since parameter interpolation is built into many codes, evaluators are urged to use an energy mesh dense enough to make differences between the two methods small. A fuller discussion will be found in the Procedures Section, 2.4.

The channel radius,

$$a = 0.123\ \mathrm{AWRI}^{1/3} + 0.08\ ,$$

is used in the penetration and shift factors, while the scattering radius, AP, is used in the phase shifts which fix the potential scattering cross section.

2.4 Additional Procedures for the Resolved and Unresolved Resonance Regions

CONTENTS OF THIS SECTION

2.4.1. Abbreviations and Nomenclature

ABBREVIATIONS

UR(R) - unresolved resonance (region)

RR(R) - resolved resonance (region)

RRP - resolved resonance parameter(s)

URP - unresolved resonance parameter(s)

SLBW - single-level Breit-Wigner

MLBW - multi-level Breit-Wigner

MLAG - multi-level Adler-Gauss

NOMENCLATURE

To avoid confusion due to definitions, the following will be adhered to:

1. The resolved resonance region is one in which resonance parameters for individual resonances are given. Usually this implies that experimental resolution is good enough to "see" the resonances, and to determine their parameters by area or shape analysis, but an evaluator may choose to supply fictitious resolved parameters if he so desires. A file 3 background may be given. The essential point is that resonance self-shielding can be accounted for by the user for each resonance individually.

2. The unresolved resonance region is that region in which the resonances are still not overlapping, and hence selfshielding is important, but experimental resolution is inadequate to determine the parameters of individual resonances.

In this situation, self-shielding must be handled on a statistical basis, and the formulas in Appendix D, and their generalizations to mixtures of materials ("heterogeneity") and Doppler-broadened cross sections, are done on the basis of the single-level Breit-Wigner formalism.

3. Still higher in energy, the resonances overlap and the cross section smooths out, subject only to Ericson fluctuations. Even though the resonances are "unresolved" here, this is not the meaning of the word unresolved in these notes, since parameterization of Ericson fluctuations for selfshielding purposes would be entirely different from the usual unresolved resonance treatment, which is a simple average over resolved resonances, subject to known statistical distributions.

2.4.2. Interpolation in the Unresolved Resonance Region (URR)

The recommended procedure is to interpolate on the cross sections derived from the unresolved resonance parameters (URP). This is a change from the previous procedure which was to interpolate on the parameters. The BNL RR codes AVRAGE3, AVRAGE4, and RESEND* follow the old procedure in this respect, also in using the Greebler-Hutchins scheme rather than MC^2-II (Henryson), and also ignore threshold and penetrability effects related to the competitive width.

In the common case that insufficient information is available about the URR, evaluators may provide a single set of URP, based on the distributions of the resolved resonance parameters (RRP), or on "systematics". Such a set implies a definite energy-dependence of the unresolved cross sections (UCS's), due to the slowly-varying penetrability-and λ^2-factors in the SLBW formula. This is the correct result, and not a set of constant UCS's. It is unfortunately true that in order to get this energy-dependence you must evaluate the UCS's at some mesh over the URR, and this requires "interpolating" on the (single) set of parameters. It is important that the mesh be dense enough to provide accurate cross section interpolation. Evaluating the cross section only

*A new version of RESEND is available which interpolates on cross sections.

at the end points could lead to errors, since no ENDF/B interpolation scheme can reproduce a combination of s-, p-, and possibly d-wave energy-dependences. R.H. MacFarlane (LASL) has a computer code which determines a mesh that provides any desired percent accuracy in the interpolated cross sections and has agreed to make it available.

It is recommended that evaluators provide the URP's on a mesh dense enough that the difference in results of interpolating on either the parameters or the cross sections be small, since it is expected that parameter interpolation will be around for a long time to come. A 1% maximum difference would be ideal, but 5% is probably quite acceptable.

It would also be desirable for NNDC to replace single sets of URP's in existing files with the same set on a dense mesh, using MacFarlane's procedure.

In both cases, the flag LRF should be set equal to 2, all parameters energy-dependent, to reduce the number of options the user must consider, and to hopefully remind evaluators and users that even though the URP's are constant, they should treat them as energy-dependent and give them on a dense mesh. LRF=1 should not be used. (Ref: May '75 RRS/C Minutes)

This procedure also provides a solution to the problems of threshold and penetrability of the competitive width, even though processing codes regard $\langle\Gamma_x\rangle$ as a constant. The evaluator can "turn on" $\langle\Gamma_x\rangle$ at its threshold, and build up its value according to its penetrability.

To re-iterate: A single set of unresolved resonance parameters implies an energy-dependent average cross section. If INT=1 (parameters specified only at the lower end of the URR) or INT=2 (linear-linear) with a sparse mesh, is given by the evaluator, then the user who calculates the cross section only at the end-points, and then interpolates on the cross section will get the wrong answer, even though in a sense he is following the recommended procedure. He

needs to evaluate $\langle\sigma\rangle$ at a dense mesh and this requires "interpolating" on the parameters. This semantic bind can be avoided by treating a single set of parameters like a multiple set, and providing it on a dense mesh in FILE2.

Finally, even if a dense mesh is provided by the evaluator, in the case that there is genuine structure in the cross section, the user who has a set of multigroup breakpoints different from the one the evaluator used will wind up with numbers different from what the evaluator "intended". There is no solution to this problem, but the dense mesh procedure minimizes the importance of the discrepancy.

In order to permit the user to determine what "error" he is incurring, it is recommended that evaluators state in the documentation what average cross sections they intended to represent by the parameters in FILE2.

2.4.3. Overlapping of Resolved and Unresolved Resonance Regions

This is not permitted, because of constraints in the processing codes. As a result, the accuracy of the representation may suffer unless the evaluator is careful in his choice of parameters for the resolved and unresolved resonance regions.

Problem 1: At the upper end of the resolved range, the smaller resonances will begin to be missed. An equivalent contribution needs to be added to the background in FILE3. This contribution will not be self-shielded by the processing codes, so it cannot be allowed to become "significant". A better procedure is to supply fictitious resolved parameters, based on the statistics of the measured ones, checking that the average cross section agrees with whatever poor-resolution data are availahle. If both procedures are employed, care needs to be taken not to distort the statistics of the underlying parameter distributions.

Problem 2: Because d-wave resonances are narrower than p-waves, which are narrower than s-waves, everything else being equal, the point at which p-waves will be instrumentally unresolved can be expected to be lower in energy than for s-waves, and lower still for d-waves. Thus the unresolved region for p-waves will usually overlap the resolved region for s-waves, and similarly for d-waves. Current procedure does not permit representing this effect explicitly - one cutpoint must serve for all ℓ-values.

The remedies are the same as above, either putting known or estimated resonances into the background in the URR, or putting estimated ("fictitious") resonances into the RRR. The latter is preferred because narrow resonances tend to self-shield more than broad ones, hence the error incurred by treating them as unshielded FILE3 background contributions is potentially significant.

A desirable expansion of current procedures would be to maintain the restriction to one resolved and one unresolved region, but allow the cutpoint to vary with ℓ. The evaluator would still have to take care that in choosing his parameters he does not distort the statistics of the distributions, but at least he could keep the s-, p-, and d-waves disentangled from each other more simply than at present.

In any event, the evaluator should be aware of when self-shielding effects are significant for each ℓ-value, and govern his choice of cutpoints and procedures accordingly.

2.4.4. Energy Range Boundary Problems

There may be as many as three boundaries under current procedures:

1 - between a low-energy FILE3 representation (range 1) and EL for the RRR (range 2),

2 - between the RRR (range 2) and the URR (range 3),

3 - between EH for the URR (range 3) and the high-energy FILE3 representation (range 4).

Discontinuities can be expected at each boundary. At 1, it will occur if range 1 and range 2 are not consistently Doppler-broadened. In general, only an identical kernel-treatment will produce continuity, and this only if the range 1 cross sections are suitably broadened, i.e., from the temperature at which they were measured, if that is appropriate, and not necessarily absolute zero, the way the range 2 cross sections would be. A kernel treatment of range 1, or no broadening at all, will be discontinuous with a ψ-χ treatment of range 2. This effect is not expected to be serious at normal reactor temperatures, and presumably the CTR and weapons communities are cognizant of the Doppler problem. In view of these problems, a double energy point will not usually produce continuity in the complete cross section, (FILE2 + FILE3) unless evaluator and user employ identical methods throughout.

Discontinuity at 2 is unavoidable, since the basic representation has changed, and should present no problem. On the other hand, if the RRR cross sections are group-averaged or otherwise smoothed, the disconinuity* should be reasonably small. If it exceeds 10 or 15% after the averaging interval has been suitably chosen, it indicates that the evaluator did not properly consider his choice of parameters relative to the poor-resolution data, experimental or calcu-

*"This refers to the discontinuity between the average cross section in the RRR, and the pointwise cross section in the URR, which has been generated from the URR parameters."

lated. A double energy point will normally occur at this boundary, but will not eliminate the discontinuity.*

Discontinuity at #3 should be small, since both the URR and range 4 represent rather smooth cross sections, and the opportunity for evaluator-error ought to be small. Anything over 5% or so should be viewed with suspicion.

2.4.5. Numerical Integration Procedures in the URR

The evaluation of effective cross sections in the URR can involve Doppler effects, flux-depression, and resonance-overlap as well as the statistical distributions of the underlying resonance parameters for a mixture of materials.

The previous ENDF/B recommendation for doing the complicated multi-dimensional integrations was the Greebler-Hutchins[1] scheme, basically a trapezoidal integration. For essentially the same computing effort, a more sophisticated weighted-ordinate method can be used and it has been shown that the scheme in MC^2-II[2] produces results differing by up to several percent from G-H. H. Henryson, II (ANL) has made the MC^2-II[2] subroutine available through NNDC and it is recommended that it be used.

The M. Beer[3] analytical method has also been suggested, and is quite elegant, but unfortunately will not treat the general heterogeneous case.

*"The RESEND-SIGAR code package generates pointwise cross sections in the RRR and URR and then kernel (Doppler) broadens through discontinuity #2, as though the cross sections were one continuous representation. Large peaks or valleys in the RRR near boundary #2 will thus raise or lower the doppler-broadened cross sections over an energy region of several Doppler widths. This is one case where the use of ψ- and χ-functions would probably produce more physically reasonable results. Some versions of SIGAR handle discontinuity 2 incorrectly, assigning the same (low-energy) value to both of the repeated values of the double-energy point. This will cause errors in cross sections and resonance integrals."

1. P. Greebler and B. Hutchins. Physics of Fast and Intermediate Reactors. III, 121 (1962) International Atomic Energy Agency, Vienna, 1962.
2. H. Henryson, II, B. J. Toppel, and C. G. Stenberg, ANL-8144, June, 1976.
3. M. Beer, NSE 50 (1973) 171.

2.4.6. Doppler-Broadening of File 3 Background Cross Sections

In principle, the contribution to each cross section from FILE3 should be Doppler-broadened, but in practice, many codes ignore it. It is therefore recommended that the evaluator keep file 3 contributions in the RRR and URR small enough and/or smooth enough so that omission of Doppler-broadening does not "significantly" alter combined FILE2 plus FILE3 results up to 3000°K.

Unfortunately, the diversity of applications of the data in ENDF/B make the work "significantly" impossible to define.

NOTE 1. A common source of structured FILE3 data is the representation of multilevel or MLBW cross sections in the SLBW format, the difference being put into FILE3. This difference is a series of residual interference blips and dips, which unfortunately affect the between-resonance valleys and hence the transmission in thick regions and in turn the absorption rates in lumped poisons, shields, blankets, etc. Users who are tied to the SLBW scheme exclusively should be aware of the need to estimate these valley-distortion effects for significant regions. However, a neat remedy is now available in the Multilevel Adler-Gauss form of MLBW.* If the resonance-resonance interference term in MLBW is expanded in partial fractions, it becomes a single sum of symmetric and asymmetric SLBW-type terms. Although two coefficients occur which require a single sum over all resonances for each resonance, these new sums are weakly energy dependent and lend themselves to approximations that could greatly facilitate the use of ψ- and χ-functions with MLBW.

NOTE 2. The "in-principle" correct method for constructing resonance cross sections is:

*To be published. Formulas given in Section 2.4.14.

a) Use a Sollbrig kernel to broaden FILE2 to the temperature of FILE3, since the latter may be based on room-temperature or other non-zero degree K data.

b) Add FILE2 and FILE3.

c) Sollbrig-broaden the result to operating temperature.

Using a Gaussian kernel incurs a small error at low energies, unless it is misused, in which case the error can be large. Using ψ- and χ-functions introduces further errors. In fact, the Sollbrig kernel already approximates the true motion of the target molecules by a free-gas law, but anything more accurate is quite difficult to handle.

2.4.7. Use of Fictitious J-Values

If a target nucleus has non-zero spin I, then s-wave neutrons can excite resonances with J equal to $I \pm 1/2$, and in all multilevel resonance formalisms except Adler-Adler, the J-value determines which resonances interfere with each other. In A-A, the J-values are not used and the pseudo-resonance interference is determined by adjusting parameters to fit the observed cross section. Usually, J is known only for a few resonances, and measurers report $2g\Gamma_n$ for the others. This number is then assumed to be Γ_n by processing codes, thereby incurring an error of uncertain magnitude, depending on how different

$$g = \frac{2J+1}{2(2I+1)}$$

is from 1/2, how large Γ_n is relative to the other partial widths, and how important resonance-resonance interference is.

The ENDF/B format uses J, not g, and g=1/2 is represented by the fictitious value J=I.

The mixing of the three families J=I, I ± 1/2 can result in negative scattering cross sections, or even worse distortions of the potential scattering term, depending on what formalism is used and how it is evaluated.

It is therefore recommended that evaluators assign J-values to each resonance, in proportion to the level density factor 2J+1. It is recognized that this forces interference between resonances on a random basis, but this is no worse than the spurious neglect of interference implied by J=I. To reduce the amount of interference, the J-values of strong neighboring resonances, which would produce the largest interference effects, can be chosen from different families.

An exception to this rule is the case where no J-values are known, since if all resonances are assigned J=I, the scattering cross section is non-negative, if MLBW is specified.

Users who employ the "amplitude-squared" formula,*

$$\sigma_{nn} = \sum_{J\ell s} g_J \left| A^{J\ell s}_{pot} + A^{J\ell s}_{resonance} \right|^2$$

should be especially careful with fictitious J-values, since the equivalence between this form and a "squared" form like SLBW, or the ENDF/B (Gregson) formulas for MLBW will be destroyed if the sum on Jℓs does not go over physically-correct values. Since this form is now in fairly common use, fictitious J-values should probably be avoided altogether.

*Section 2.4.19

2.4.8. Equivalent Single-Level Representations

In response to users who are able to process only SLBW resonance parameters, various schemes have been proposed over the years to re-cast multi-level cross sections into SLBW + background format. Since Reich-Moore parameters are now to be given in equivalent Adler format, and the Multilevel Adler-Gauss formula of Section 2.4.14 provides an SLBW equivalent to MLBW, the need for such procedures is much less. The Adler parameters can be handled by SLBW codes, the only new feature being that asymmetric terms (Doppler Voight profiles χ) occur in capture and fission as well as scattering. Since these integrate approximately to zero, their neglect in constructing multigroup cross sections is presumably in a class with the many other approximations which need to be ultimately corrected for in multigroup analyses.

The MLAG formalism uses SLBW capture and fission, and hence introduces no new requirements on these cross sections.

2.4.9. Use of the Reich-Moore Formalism

The computer code POLLA (ORNL-TM-2599, June 12, 1969, G. DeSaussure and R.B. Perez), as well as some others, will convert a set of Reich-Moore multilevel resonance parameters to Adler format. Although R-M has some advantages in actually evaluating data, mainly that it uses resonance spins and is more closely tied to familiar resonance parameters, the Adler format is more convenient since it permits the use of ψ- and χ-functions for Doppler broadening.

All R-M evaluations will therefore be converted to Adler format for presentation in ENDF/B, and the R-M option is deleted, although ENDF/A will continue to accept it.

If the conversion causes differences between the Adler and R-M cross sections which exceed 0.1%, these should be put into FILE3, since it is not the intent of the procedure to in any way alter the original cross sections. Such differences can presumably be minimized by feeding the POLLA output parameters to a least-squares search code based on the Adler formalism, and "fitting" the original R-M values.

According to the discussion in BNL 50296 (ENDF 148) "ENDF/B Processing Codes for the Resonance Region", M.R. Bhat, June, 1971, the Reich-Moore code RAMPl, now incorporated in RESEND, sets the shift factor Δ_λ equal to zero. Since $\Delta_\lambda = -\gamma^2_{\lambda\ell} S_\ell$ this is equivalent to setting $S_\ell = 0$. On the other hand, the Reich-Moore formalism and RAMPl set the quantity C equal to zero. Since $C = -(S_\ell + \ell)/ P_\ell$, this is equivalent to setting $S_\ell = -\ell$, another common choice in R-matrix theory. However, these two choices for S_ℓ are consistent only for s-waves, and RESEND should be applied to Reich-Moore p- and d-waves with caution, if at all.

2.4.10. Competitive Width in the Resonance Region

2.4.10.1. Resolved Region

Procedures for the Resolved Resonance Region are contained in Section D.3.1 of Appendix D.

2.4.10.2. Unresolved Region

Procedures for the Unresolved Resonance Region are contained in Section D.3.2 of Appendix D. Users are directed to the discussion of the total cross section in Appendix D, Section D.3.3, since, as pointed out by H. Henryson, II, in connection with MC^2 procedures, a possibility for erroneous calculations exists.

2.4.11. Negative Cross Sections in the Resolved Resonance Region

2.4.11.1. In the SLBW Formalism

Capture and fission use the positive symmetric Breit-Wigner shape and should never be negative. A possible exception might occur if an Adler-format cross section were transformed to an "equivalent" set of SLBW parameters. This might produce unphysical negative widths and hence negative cross sections. Scattering involves the asymmetric term which goes negative for $E < E_\lambda$ and can cause negative cross sections. A single resonance, or a series of well-separated resonances, will usually not produce negative cross sections, but when two or more resonances "cooperate", their negative tails can combine to produce negative values. In nature, the negative tails are compensated by either the positive tails of lower-lying resonances or multilevel interference effects. However, in evaluated data files the resonances are usually given only down to "E=0", a quite arbitrary point from the standpoint of the compound nucleus, so that "negative-energy" resonances are needed to compensate the negativity bias. Although the negative scattering cross sections themselves can usually be classed as an inconvenience, their effect in distorting the total cross section, which governs neutron penetration, can be a serious source of error. Perhaps even more important is the fact that even when the cross section remains positive, it is still often too low due to the same effect and this bias again affects the total cross section and calculated absorption rates. To compensate the end-effect bias, the evaluator should put in either a series of negative energy resonances with reasonable size and spacing ("picket fence", or reflect the positive-energy ones around E=0) or a few large fictitious ones ("barber poles"), or a compensating background in FILE 3.*

*E.g., G. DeSaussure, et al., NSE 61, 496, (1976).

To compensate interior-region negativity requires a multilevel treatment of which MLBW is the simplest. Although there is no guarantee that MLBW cross sections will be more accurate than SLBW, they are guaranteed to be non-negative (but see next section) and are generally to be preferred over SLBW.

The same bias effect occurs at the upper end of the resolved resonance range, where it is less noticeable because it is a positive bias, and most calculations are not as sensitive to this region as they are to the low-energy end. The remedy is the same - extra resonances above the RRR, or compensation in FILE 3. The latter remedy requires a negative FILE 3 contribution, which is physically acceptable, but produces undesirable side-effects in some processing codes, hence the extra-resonance remedy is preferred. It is probably safe to say that there is rarely a compelling reason to use the SLBW formula for the calculation of pointwise scattering cross sections. If one is doing a calculation that is sophisticated enough to warrant the use of pointwise cross sections, then a multilevel formalism is certainly justifiable. If one is merely deriving multigroup cross sections, then the other approximations involved justify the use of any reasonable "fix" for the negative scattering, such as simply setting $\sigma_s = 0$ when it goes negative.

2.4.11.2. In the MLBW Formalism

Capture and fission use the SLBW formulas and are positive, subject to the same proviso concerning Adler-equivalence as in the previous section. Scattering uses a formula which can be written as an absolute square and as such is non-negative. The use of fictitious J-values (which see) can destroy the correspondence between the absolute-square form and the expanded form given in Appendix D, and result in negative scattering cross sections. Despite its non-negativity, MLBW still produces biassed cross sections at both ends of the RRR

unless compensating extra resonances or FILE3 contributions are included above and below it, and the evaluator should generally correct for this effect.

2.4.11.3. In the Reich-Moore Formalism

This is again based on an absolute square and cannot be negative. However, it can be biassed and extra resonances or FILE3 contributions should always be provided. If conversion to Adler format produces negative cross sections, the parameters should be adjusted to eliminate them.

2.4.11.4. In the Adler Formalism

Although the formulas are derived from an absolute square and are in principle non-negative, in practice the parameters are chosen to fit measured data, so that the physical and mathematical constraints among the parameters, which prevent negative cross sections, are lost, and any of the cross sections can be negative. If the Adler formalism is used for evaluations, negativity should be checked for. The end-effect bias exists in this formalism also and should be checked for in the scattering and total cross sections by comparing with experiment.

2.4.12. Negative Cross Sections in the Unresolved Resonance Region

R. Prael (ANL) has reported a difficulty with SLBW resonance ladders created by VIM from the unresolved resonance parameters in Mo (MAT. NO. 1287), namely that the negative FILE3 capture background sometimes causes negative capture cross sections in the resonance valleys.

The evaluator evidently intended the background to compensate for an excess of capture in the average unresolved capture cross section, but did not anticipate the problem which would arise when the parameters were used in a different context. One remedy is to drop out the negative FILE3 background and

adjust $\langle\Gamma_\gamma\rangle$ on whatever energy mesh is needed to produce agreement with the dilute poor-resolution data.

The creation of SLBW ladders from average parameters can be expected to produce the same kind of end-effect bias and frequent negative scattering cross sections that are seen in the resolved resonance region. Again, the scattering cross section per se may not be important, but the biassed total cross section will adversely affect calculated reaction rates.

As mentioned before, for a sophisticated application like the VIM Monte Carlo code, the use of MLBW is certainly indicated, and the Multilevel Adler Gauss* formula makes its use quite simple.

2.4.13. Use of Two Nuclear Radii

Two different nuclear radii are defined by the current ENDF/B formats:

a) the scattering radius, AP, and

b) the channel radius, a.

The scattering radius is also referred to as "the effective scattering radius" and "the potential scattering radius". The channel radius is also referred to as "the hard-sphere radius", or "the nuclear radius". The former is the quantity defined as AP (for a_+, or $\hat{a}$) in FILE 2, which must be given even if no resonance parameters are given, while the latter is defined by

$$a = (0.123\ \mathrm{AWRI}^{1/3} + 0.08) \text{ in units of } 10^{-12} \text{ cm.}$$

(Strictly speaking, this formula should use the target mass, $A^{1/3}$ but AWRI is currently specified.)

*Section 2.4.14

The channel radius is a basic quantity in R-matrix theory, where the internal and external wave-functions are joined and leads to the appearance of hard-sphere phase shifts defined in terms of it. The necessity to relax the definition and permit two radii can be thought of as a "distant-level effect", not usually explicit in R-matrix discussions.

The currently recommended ENDF/B procedure is:

Use the channel radius to calculate the penetrabilities P_ℓ (ka) and the shift factors S_ℓ(ka), but the scattering radius to calculate the hard-sphere phase shifts ϕ_ℓ(ka).

Since the phase-shifts define the potential scattering cross section, the evaluator has the freedom to fit AP = â to a measured cross section while still leaving undisturbed those codes that use the $A^{1/3}$ formula to calculate the channel radius.

Note that the scattering radius also appears in the phaseshift dependent terms $\cos 2\phi_\ell$, $\sin 2\phi_\ell$, etc., that appear in the resonance terms of the various formalisms.

The original ENDF/B formats made provision for an AM, or "A-minus", although it was always required that evaluators put AM=0, to signify that it was equal in value to AP. In the current formats, AM is eliminated, but one can anticipate that more sophisticated evaluation techniques may eventually force the re-instatement of not only AM, but a more general dependence of the scattering radius on the channel quantum numbers, especially as higher energies become important.

In theory, the scattering radius depends on all the channel quantum numbers, and in practice it is common to find that different optical model parameters are required for different ℓ-values (s, p, d,...) and for different j-

values ($p^{1/2}$, $p^{3/2}$, ,,,). This implies that one would require a different scattering radius for each of these states.

For the special case of s-waves, only two J-values are possible, namely I ± 1/2, commonly denoted J_+ and J_-. This is the origin of the terminology A_+ and A_-.

2.4.14. The Multilevel Adler-Gauss Formula for MLBW

Appendix D of ENDF 102 gives (implicitly) for the MLBW formalism the equations:

$$\sigma_{n,\gamma}^{MLBW}(E) \equiv \sigma_{n,\gamma}^{SLBW}(E) \ ,$$

$$\sigma_{n,F}^{MLBW}(E) \equiv \sigma_{n,f}^{SLBW}(E) \ ,$$

$$\sigma_{n,n}^{MLBW}(E) = \sigma_{n,n}^{SLBW}(E) + \sum_{\ell=0}^{NLS-1} \sigma_{n,n,RRI}^{\ell}(E) \ ,$$

where RRI labels the resonance-resonance-interference term for a given ℓ-value:

$$\frac{\hbar}{k^2} \sum_J g_J \sum_{r=2}^{NR_J} \sum_{s=1}^{r-1} \frac{2\Gamma_{nr}\Gamma_{ns}\left((E-E_r')(E-E_s')+\Gamma_r\Gamma_s/4\right)}{\left((E-E')^2+(\Gamma/2)^2\right)\left((E-E')^2+(\Gamma/2)^2\right)} \ . \qquad (1)$$

As most users are aware, this double sum over resonances can eat prodigious amounts of computer time unless handled very tactfully. Thus, for a 200-resonance material, there are ∿40000 cross terms, of which only 20000 need to be evaluated because the expression is symmetric in r and s.

It has been noted many times in the past that partial fractions can reduce Equation (1) to a form with only a single Breit-Wigner denominator. Most recently, DeSaussure, Olsen, and Perez (NSE 61, 496 (1976)) have written it compactly as

$$\frac{\pi}{k^2} \sum_J g_J \sum_{r=1}^{NR_J} \frac{G_r\Gamma_r + 2H_r(E-E_r')}{(E-E_r')^2 + (\Gamma_r/2)^2}$$

where

$$G_r = \frac{1}{2} \sum_{\substack{s=1 \\ (s\neq r)}}^{NR_J} \frac{\Gamma_{nr}\Gamma_{ns}(\Gamma_r+\Gamma_s)}{(E_r'-E_s')^2 + \frac{1}{4}(\Gamma_r+\Gamma_s)^2}$$

$$H_r = \sum_{\substack{s=1 \\ (s\neq r)}}^{NR_J} \frac{\Gamma_{nr}\Gamma_{ns}(E_r'-E_s')}{(E_r'-E_s')^2 + \frac{1}{4}(\Gamma_r+\Gamma_s)^2}$$

D-O-P give the special case for $I=\ell=0$, but it is valid for any set of quantum numbers. Thus an existing SLBW code can be converted to MLBW by adding $G_r\Gamma_r$ to the symmetric part of the SLBW formula,

$$\Gamma_{nr}^2 \cos 2\phi_\ell - 2\Gamma_{nr}(\Gamma_{\gamma r} + \Gamma_{fr}) \sin^2\phi_\ell$$

and $2H_r$ to the coefficient of $(E-E_r')$ in the asymmetric part,

$$2\Gamma_{nr} \sin 2\phi_\ell .$$

Since G_r and H_r are weakly energy-dependent, via the penetrabilities and shift factors, they lend themselves to approximations that can sharply reduce computing time compared to the Gregson form with the "double" Breit-Wigner denominator. In fact, if the resonances are all given s-wave shifts of zero, s-wave penetrabilities of $\sqrt{E}$, and the total widths are taken constant, G_r/k^2 and H_r/k^2 become independent of the neutrons energy and consume a negligible amount of computing time so that MLBW and SLBW become equivalent in this respect.

The amplitude-squared form of MLBW also reduces computing time, and is given in section 2.4.19.

2.4.15. Notes on the Adler Formalism

The following is a condensation and updating of the Appendix in the June, 1974 Minutes of the Resonance Region Subcommittee:

Questions concerning the ENDF/B treatment of the Adler formalism are enumerated below, together with recommended procedures for handling them:

1. The resonance energy μ and total half-width ν are the same for each reaction for a given resonance in the Adler formalism, but the formulas of the October 1970 version of ENDF-102, page D-7, and the format descriptions of pages 7.9 and N-12 permit different values for the total, fission, and capture cross sections.

This is a misreading of the formalism and the remedy is to constrain the equalities $DET_n = DEF_n = DEC_n$ and $DWT_n = DWF_n = DWC_n$. The formulas for capture and fission should also have the phases eliminated in Appendix D.

2. The Adler formalism, as applied by the Adlers, breaks the resolved resonance region up into subregions, and each is analyzed separately. This avoids problems with contributions from distant resonances, but requires that the polynomial background be tailored to each subregion. However, the ENDF/B formats allow only one resolved resonance energy region, so this procedure cannot be used.

If a single set of polynomial background constants is insufficient, additional background can be put into FILE 3, point-by-point.

3. The ENDF/B formats formerly permitted incomplete specification of the cross sections. The allowed values of LI were 5 (total and capture widths); 6 (fission and capture); and 7 (total, fission, and capture). LI = 6 leaves the scattering (and total) undefined and LI = 5 is deficient for fissile elements. LI=6 is now restricted to ENDF/A, and LI=5 should be used only for non-fissile elements.

4. The nomenclature for the G's and H's is confusing. There is an unfortunate 3-way nomenclature problem. The Adlers use for the total cross section the definitions:

$$G_t = \alpha \cos 2ka + \beta \sin 2ka$$

$$H_t = \beta \cos 2ka - \alpha \sin 2ka$$

and then the combination

$$\nu G_t + (\mu - E)\, H_t.$$

For the reaction cross sections there are no phases, and they write

$$\nu G_c + (\mu - E)\, H_c \qquad \text{(capture)}$$

$$\nu G_f + (\mu - E)\, H_f \qquad \text{(fission)}$$

These are properly designated as "symmetrical" and "asymmetrical" parameters. Nomenclature switch No. 1 occurs in the ENDF-102 formula, which changes α to G_t and β to H_t, viz:

$$\nu \left(G_t \cos 2ka + H_t \sin 2ka\right) + (\mu - E) \left(H_t \cos 2ka - G_t \sin 2ka\right)$$

These G_t's and H_t's are no longer symmetrical and asymmetrical, but the manual refers to them that way. The precedent for this altered nomenclature is probably BNL 50045 (T-455), D.B. Adler, March 1967, page 7.

Nomenclature switch No. 2 is due to DeSaussure and Perez, who in their published tables of G and H, incorporate the Adler's constant c into their definition, but otherwise leave the formalism unchanged.

Users and evaluators should adhere to the ENDF-102 definitions.

5. The flag NX, which tells what reactions have polynomial background coefficients given, should be tied to LI, so that the widths and backgrounds are given for the same reactions, i.e., use NX = 2 with LI = 5 (total and capture), and NX = 3 with LI = 7 (total, capture, and fission). Since no NX is defined for LI = 6 (fission and capture), one is forced to use NX = 3 with the background total coefficients set equal to zero, but this now occurs only in ENDF/A, if at all.

2.4.16. Multilevel Versus Single-Level Formalisms in the Resolved and Unresolved Resonance Regions

2.4.16.1. In the Resolved Resonance Region

The SLBW formalism is adequate for resonance treatments that do not require actual pointwise scattering cross section, as, e.g., multigroup slowing-down codes. Because of the frequent occurrence of negative scattering cross sections, when two or more resonance-potential interference terms overlap, SLBW should never be used to compute pointwise scattering cross sections. Instead, the MLBW formalism should be used, although MLBW is not a true multilevel formalism, but the limit of one which is valid if Γ/D is small. There is no simple way to test the quality of MLBW scattering cross sections but they are probably almost always superior to SLBW.

The Reich-Moore reduced R-matrix formalism is a true multilevel formalism, and is recommended for fissile element evaluations. All of its cross sections are non-negative, and its only significant drawbacks, apart from the effort required for its application, are the difficulty of determining a suitable R^{∞} to represent distant-level effects, and of determining the parameters of negative-energy resonances.

The Adler form of the Kapur-Peierls formalism is also a true multilevel treatment, but in actual applications the parameters are determined by fitting data and the theoretical constraints among them are lost, so that any Adler cross section can be negative.

The simplest true multilevel formalism is the reduced R-function, in which all channels except scattering have been eliminated. It makes a very adequate evaluation tool for nonfissile elements up to the threshold for inelastic scattering, since below that the eliminated channels are (usually) simply radiative capture. It can be corrected for distant-level effects by

substituting optical-model phase shifts for the hard-sphere ones which occur in the formalism, and by introducing an appropriate R^{∞}.

Multi-channel multilevel fitting is also feasible in many cases, and permits the simultaneous use of non-neutron data leading to the same compound nucleus. The conversion of R-function and R-matrix analyses to ENDF/B-acceptable formats is discussed in the next section.

2.4.16.2 In the Unresolved Resonance Region

In principle, if the statistical distributions of the resolved resonance parameters are known, any formalism can be used to construct fictitious cross sections in the unresolved region. At the present time, only the SLBW formalism is allowed in ENDFB, for the reason that no significant multilevel effect can be demonstrated, when the SLBW is properly handled.

If resolved region statistics are used without adjustment to poor resolution data, then large multilevel/single-level differences can result, but there is no simple way to determine which is better. If both are adjusted to yield the same average cross sections, and for fissiles, the same capture-to-fission ratio, then the remaining differences are within the statistical and measurement errors inherent in the method.

As noted in Section 2.4.12, the use of SLBW to construct resonance profiles in the unresolved region will result in the defects associated with this formalism elsewhere, and is not recommended. This application calls for MLBW or better, and the SLBW scheme should be used only for constructing average cross sections where the negative scattering effects will combine with the other approximations and presumably be "normalized out" someplace along the line.

The above comments on multilevel effects in the unresolved resonance region are based on the work of G. DeSaussure and R.B. Perez "Multilevel Effects in the Unresolved Resonance Region, etc.", NSE 52, 382 (1973).

2.4.17. Conversion of R-Function and R-Matrix Evaluations to Allowed Formats

R-function and R-matrix evaluations have been made for various materials and the use of these genuine multilevel formalisms is to be encouraged, but with the recent decision to convert Reich-Moore parameters to Adler format, the R-parameters cannot be used directly.

Several alternatives exist. The simplest, applicable to light nuclei, is to simply give a pure FILE3, point-by-point, representation. Doppler-broadening will be negligible at reactor temperatures, and ultra-high temperature requirements can be met by kernel-broadening.

If enough parameters have been given to make parameterization desirable, then one can calculate the cross section from the R-parameters using the MLBW formalism, do a point-by-point subtraction from the R-cross section, and put the difference into FILE3. Since the materials which have been evaluated this way have usually small (or no) Doppler effect, no problems should arise from FILE3 "structure".

Unfortunately, this procedure encounters problems connected with the use of single-level shift factors, $S_\ell(E) - S_\ell(|E_r|)$, in the MLBW formalism, which puts the resonances at different places than does the R-formalism. MLBW also breaks down for overlapping resonances, (large Γ/D values), and loses the detailed angular distributions inherent in the R-formalism phase shifts.

More complicated, but potentially capable of producing a smaller residual FILE3, is to run the parameters through POLLA, or one of its equivalents, and use the Adler-equivalent parameters in FILE2. The following points need to be considered:

1. POLLA is written for s-waves, hence the conversion of higher ℓ-waves will produce a residual difference due to the penetrabilities and shift factors. If sufficient interest existed, POLLA could probably

be fixed up, using $\sqrt{E}$ expansions, to do a reasonably good job on any ℓ-wave, at the cost of introducing several additional constants per resonance.

2. For light nuclei, with keV or MeV widths, the neglect of the energy-dependence of Γ_n in the total width in the Adler denominator may be more serious than it is for eV-wide resonances.

G. DeSaussure is of the opinion that POLLA could not be made to work at all for p- and d-wave resonances, due to the resultant string energy-dependence of the Adler parameters (Ref: Minutes of the May, 1977, Resonance Region Subcommittee, letter from G. DeSaussure to C.R. Lubitz).

Because of the above difficulties, this topic is currently an open one. It is under investigation, with a view toward including an R-formalism, probably less complicated than Reich-Moore, into the set of allowed formats.

In BNL 50296, "ENDF/B Processing Codes for the Resonance Region", M.R. Bhat, June 1971, it is stated the shift is set equal to zero in the Code RAMP1. This implies that RESEND cannot be used as an R-function code for p- and d-waves. See discussion in Section 2.4.9.

2.4.18. Computer Time for Generating MLBW Cross Sections

Previous solutions to the problem of evaluating the Gregson form of the MLBW resonance-resonance interference term in a reasonable amount of time have been to use the amplitude-squared form from which it was derived, and kernel-broadening, or to optimize the calculation of inner and outer loop quantities and ψ/χ -broadening.

A third solution is now available, namely to use the Multilevel Adler-Gauss* formulas and possibly approximate the energy-dependence of the G_r - and H_r - coefficients.

For reference, the amplitude-squared form of MLBW is given in Section 2.4.19.

2.4.19. Amplitude-Squared Form of the MLBW Formulas

The Gregson form of the MLBW (scattering cross section), given in Appendix D and in Section 2.4.14, is mathematically identical to the more fundamental "amplitude-squared" form given in Appendix D, as equations (5) - (7) of Section D.1.2.

Those equations can be coded in complex Fortran, or broken up into their real and imaginary parts before coding. The essential point is that they sum the resonances before squaring. This avoids turning two "linear" sums into one "quadratic" one. If an isotope has 200 resonances, the above formulas have two sums with 200 terms each, whereas the Gregson/ENDF-102 form has a sum with 40000 cross terms. A discussion of points to consider in coding the above equations is given in the June 1974 Minutes of the Resonance Region Subcommittee.

In addition to those notes, we remind the reader that $U_{nn}^{J}(E)$ in equations (6) and (7), and Γ_{nr} and Γ_r in equation (7), should properly be written $U_{nn}^{J\ell}(E)$, $\Gamma_{nr}^{J\ell}$, and $\Gamma_r^{J\ell}$, respectively, to show explicitly the channel quantum number dependence which is only implicit in the ENDF 102 notation, and which has occasioned some misunderstanding in the past.

The main drawback to the above equations is that they do not admit Doppler-broadening with ψ- and χ-functions, but require kernel methods instead.

*Section 2.4.14

2.4.20. Degrees of Freedom for Unresolved Resonance Parameters

A resonance in the system (neutron plus a target of mass A) corresponds to a quasi-stationary state in the compound nucleus A + 1. Such a resonance can decay in one or more ways, each described as a channel. These are labelled by the identity of the emitted particle (two-body decay), the spins I and i of the residual nucleus and the emitted particle, and the orbital angular momentum ℓ of the pair. To uniquely specify the channel, two more quantum numbers are needed, since the magnetic quantum numbers can be eliminated by "geometrical" considerations, for unpolarized particles.

It is common to give the channel spin, S, which is the vector sum of I and i, plus $\vec{J} = \vec{S} + \vec{\ell}$, since this facilitates the isolation of the ℓ-dependence of all channel quantities. The important point is that the same set of three ingredient angular momenta, I, i, and ℓ, will give rise to a number of different channels, according to the rules for coupling angular momenta. The resonance will decay into each of these channels, with a probability that is governed by a real number $\gamma_{\alpha I i J \ell S}$, the reduced width amplitude, where α gives the identity of the emitted particle, the state of excitation of the daughter nucleus, etc. The partial width for the channel is

$$\Gamma_{\alpha I i J \ell S} = 2P_{\alpha I i J \ell S}\gamma^2_{\alpha I i J \ell S}.$$

The penetrabilities depend only on ℓ, and are given in Appendix D for uncharged particles. For charged particles, their Coulomb analogs can be found in texts on the subject, and for gamma rays one uses $\sqrt{\Gamma}$ rather than γ and P.

If the collection of channel quantum numbers (αIiJℓS) is denoted by c, then the total width for the level is $\Gamma = \Sigma_c \Gamma_c$. ($\Sigma_c$ means a sum over all channels). The argument from statistical compound nucleus theory is that the

γ_c's are random variables, normally distriubted with zero mean and equal variance. The population referred to is the set of γ_c's for a given channel and all the levels (or resonances). It follows that the total width is distributed as a chi-squared distribution with N degrees of freedom, since this is the statistical consequence of squaring and adding N normal variates. For N=1, this is the Porter-Thomas distribution. In determining the behavior of any quantity that is going to be averaged over resonances, it is necessary to know the way in which the widths are distributed, hence the inclusion of these degrees of freedom in ENDFB.

1. The neutron width is governed by AMUN, which is specified for a particular ℓ-value. Usually, only the lowest allowed ℓ-value will be significant in any decay, although the formats would allow giving both s- and d-wave widths for the same resonance. Since there is only one J-value for a given resonance, and we label the widths by one ℓ-value, there can be at most 2 channels for neutrons (i = 1/2), labelled by the channel spin values S = I ± 1/2. If I = 0, there is only one channel, S = i = 1/2. Hence the restriction, $1.0 \leq \text{AMUN} \leq 2.0$.

Although there is no evidence to support it, it is assumed that the average partial widths for each channel spin, if I > 0, are equal, and that $\langle\Gamma_n\rangle$ is the sum of two equal average partial widths. In Appendix D; this factor of 2 is absorbed into the definition of $\langle\Gamma_n\rangle$, through the use of a multiplicity which is the number of channel spins, 1. or 2.

2. The competitive width is currently restricted to inelastic scattering, which has the same behavior as elastic scattering, measured from a different "zero channel energy," hence

$$1.0 \leq \text{AMUX} \leq 2.0$$

Note that one should not set AMUX = 0 out of ignorance of its true value, as previously suggested in ENDF 102. This implies a constant from resonance to resonance, since the chi-squared distribution approaches a delta function as $N \to \infty$. An inelastic reaction can be expected to proceed through a small number of channels and hence to fluctuate strongly from level to level.

3. For the radiative process, AMUG should be set equal to zero. Radiative capture proceeds through many channels and it is not worthwhile deciding if AMUG is 30° or 40°. (If some nucleus has selection rules which restrict radiative decay to a few channels, then a different value of AMUG might be appropriate.)

4. The fission value should be stated as $1.0 \leq \text{AMUF} \leq 4.0$ and the value zero would be incorrect. These small values violate the previous discus sion of (Wigner-type) channels and instead obey statistics governed by fission barrier tunneling (Bohr-channels).

3. FILE 3, NEUTRON CROSS SECTIONS

3.1. General Description

Neutron cross sections, such as the total cross section, elastic scattering cross section, and radiative capture cross section, are given in File 3. Certain derived quantities are also given. These data are given as a function of energy, E, where E is the incident neutron energy (in eV) in the laboratory system. They are given as energy-cross section (or derived quantity) pairs. An interpolation scheme is given that specifies the energy variation of the data for neutron energies between a given energy point and the next higher energy point.

File 3 is divided into sections, each containing the data for a particular reaction type (MT number). The sections are ordered by increasing MT number. A complete list of MT's and their definitions can be found in Appendix B.

3.2. Formats

File 3 is made up of sections where each section gives the neutron cross sections (or derived quantities) for a particular reaction type (MT number). Each section always starts with a HEAD record and ends with a SEND record.

The common variables used in this other file are defined in Section 0.5.1 and in the Glossary (Appendix A). For File 3 the following quantities are defined:

LIS is an indicator that specifies the initial state of the target nucleus (for materials that represent nuclides).

LIS = 0, the initial state is the ground state.

= 1, the initial state is the first excited state (generally the first metastable state).

= 2, the initial state is the second excited state.

etc.

LFS is an indicator that specifies the final excited state of the residual nucleus produced by a particular reaction.

LFS = 0, the final state is the ground state.

= 1, the final state is the first excited state.

= 2, the final state is the second excited state.

. .

. .

. .

= 98, an unspecified range of final states.

= 99, all final states.

Q is defined for the reaction a + A → b + B as:

$Q(eV) = \left[(m_a + m_A) - (m_b + m_B + E_B(eV)/(9.315016+08))\right](9.315016+08)$,

where the masses are given in terms of the ^{12}C atomic mass units (amu) and E_B represents the excitation energy of the final state (ground or i^{th} excited state) of the residual nucleus B. Note that for inelastic scattering to a discrete level, $Q = -E_B$.

S is the temperature (°K). NOTE: If the LR flag is used, S becomes the breakup energy for the final reaction products corresponding to that particular LR value. (Exceptions are LR = 39 and LR = 40, which are set equal to 0.0).

LT is a flag to specify whether temperature-dependent data are given. S and LT are normally zero. Details on temperature-dependent data are given in Appendix F.

LR is a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x). (See Section 3.2.4.4.)

NR is the number of energy ranges that have been given. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data. (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range. (For details, see Section 0.4.3.).

$\sigma_{(E)}$ is the cross section (barns) for a particular reaction type at incident energy point, E, in (eV). Data are given in energy-cross section pairs.

The structure of a section is

(MAT, 3, MT/ZA , AWR, LIS, LFS, 0 , 0)HEAD

(MAT, 3, MT/S , Q , LT, LR , NR, NP/E_{int}/σ(E))TABl

(MAT, 3, 0 /0.0, 0.0, 0, 0, 0 , 0)SEND

3.3 Procedures

3.2.1. Reaction Types to be Included

A complete list of possible reaction types and their definitions can be found in Appendix B. Cross sections for all reaction types that are not zero or negligibly small should be given in File 3. As a minimum, data for the reactions listed below should be given, if applicable.

MT	Reaction
1	Total cross section
2	Elastic scattering cross section
4	Inelastic scattering cross section (total)
16	(n,2n) cross section
17	(n,3n) cross section
18	Fission cross section

MT	Reaction
51	Inelastic excitation cross section for the 1st level
52	" " " " " " 2nd level
.	
.	
.	
90	Inelastic excitation cross section for the 40th level
	(continued on the next page)
91	Inelastic excitation cross section for the 40th continuum
102	(n, γ) radiative capture cross section
103	(n,p) cross section
104	(n,d) " "
105	(n,t) " "
106	(n,He^3) " "
107	(n,α) " "
108	(n,2α) " "
251	$\bar{\mu}_{Lab}$
252	ξ
253	γ

3.2.2. General Procedures

1. All significant cross sections must be given in this file using the reaction types (MT numbers) that have been defined. Appendix B summarizes all currently defined reaction types. If new MT numbers are needed, the National Nuclear Data Center at Brookhaven National Laboratory should be contacted.

Select the most appropriate MT number to represent the reactions. In many cases different MT numbers may be used to represent the same reaction mechanism, e.g., Li-6(n,t) and Li-6(n,α). This situation arises when the

reaction produces multiple secondary particles or when the secondary particle and the residual nucleus are interchangeable. Many reactions of neutrons on light targets fall into this category. It is not possible to establish rigid ground rules, but in general, the MT chosen includes the lightest nucleus in the exit channel.

2. Reaction Q-values are important, and attempts should be made to obtain a value for each section in File 3. Even when the material represents a natural element containing two or more isotopes, reaction Q-values should be given. In these cases for which there is no unique Q-value, the value given should be the Q-value that produces the lowest threshold energy.

3.2.3. Initial and Final States

The formats have been generalized to specify data for excited states of the target (initial) and residual (final) nucleus using the LIS and LFS flags. If the initial state is isomeric and has a half-life > 1 sec, current ENDF procedures require data to be given as a separate ENDF Material. Reaction data producing known final states are given within the Material associated with the initial nucleus.

Where several final states are produced by a reaction, the summation, discrete level, and continuum cross sections can be specified. Specification of summation reaction cross sections to all states, discrete and continuum, is given within an MT number by LFS = 99. Data for an unspecified range of final states is given by using the same MT number with LFS = 98.

For the (n,p), (n,d), (n,t), (n,^{3}He), and (n,α) reactions, the summation cross sections must be given in the MT = 103, 104, 105, 106, and 107 sections, respectively. Use of LFS = 99 in these cases is redundant. The cross sections to the ground and the first 17 discrete excited levels of the final nucleus must

be given in the MT = 700 series. Use of LFS is redundant for a number of levels $\leq$17. For the (n,2n) reaction the summation cross section must be given in the MT = 16 section. Use of LFS = 99 is redundant. The (n,2n) cross section to isomeric states should be given in the MT = 26 section, using the LFS flag to indicate the isomeric states (counting all states) designated.

3.2.4. Procedures for Specific Reactions

3.2.4.1. Index for Section 3.2.4.

3.2.4. Subsection	Relevant MT Nos. (See Appendix B)
1	1
2	2
3	3
4	4
5	6-9, 16, 17, 26, 46-49
6	18, 19, 20, 21, 38
7	27
8	51-91
9	101
10	120
11	102-114
12	700-799

3.2.4.1 Total Cross Sections

1. If resolved or unresolved resonance parameters are given in File 2, the contribution to the total cross section in the resonance region is the sum calculated from File 2 and MT = 1 in File 3. (see Section 3.3.)

2. The total cross section is generally the most important cross section in a shielding material. Considerable care should be exercised in evaluating this cross section and in deciding how to represent it.

3. Cross section minima (potential windows) and cross section structure should be carefully examined. Sufficient energy points must be used in describing the structure and minima to reproduce the experimental data to the measured degree of accuracy.

4. The total cross section as well as any partial cross section must be represented by 5000 incident energy points or less. The set of points or energy mesh used for the total cross section must be a union of all energy meshes used for the partial cross sections. Within the above constraints, every attempt should be made to minimize the number of points used. The total cross section must be the sum of MT = 2 (elastic) and MT = 3 (nonelastic). If MT = 3 is not given, then the elastic cross section plus all nonelastic components must sum to the total cross section.

3.2.4.2. Elastic Scattering Cross Section

1. If resolved or unresolved resonance parameters are given in File 2, the contribution to the elastic scattering cross section in the resonance region is the sum calculated from File 2 and MT = 2 in File 3 (see Section 3.3).

2. The elastic scattering cross section is generally not known to the same accuracy as the total cross section. Frequently the elastic scattering cross section is obtained by subtracting the non-elastic cross section from the total cross section. This procedure can cause problems. The result is an elastic scattering cross section that contains unreal structure. There may be several causes. First, the nonelastic cross section, or any part thereof, is not generally measured with the same energy resolution that the total cross

section has been measured. When the somewhat poorer resolution nonelastic cross section is subtracted from the total cross section, much of the structure (at times very unrealistic) is placed in the elastic scattering cross section. Second, if the observed structure in the nonelastic cross section is improperly correlated with the structure in the total cross section, an unrealistic structure is generated in the elastic scattering cross section.

3. Frequently the experimentally measured elastic scattering cross section is obtained by integrating angular distribution data. These data may contain contributions from neutrons producing nonelastic reactions. This contamination is generally due to contributions from inelastic scattering to low lying levels that are not resolved in the experiment. Care must be taken in using such results to obtain integrated cross sections. Such angular distribution data can also cause similar problems when they are used to prepare File 4 data.

3.2.4.3. Nonelastic Cross Section (MT = 3)

The nonelastic cross section is not required unless any part of the photon production cross section data given in File 12 uses MT = 3 to represent these data. In this case MT = 3 is required in File 3. If MT = 3 is given, then the set of points used to specify this cross section should be a union of the sets used for all its partials.

3.2.4.4. Inelastic Scattering Cross Sections

1. A total inelastic scattering cross section must be given if any of the partials are given, i.e., discrete level excitation cross section, MT = 51, 52, 53,....., 90, or continuum inelastic scattering, MT = 91.

2. The set of incident energy points used for the total inelastic cross section (MT = 4) must be a union of all the sets used for the partials.

3. Values should be assigned to the level excitation cross sections for the first few levels for the entire energy range (up to 20 MeV). Frequently the inelastic level cross section for the first few levels can be obtained from experimental measurements. At other times, deformed nucleus model calculations must be made. Direct interaction contributions are important in neutron inelastic scattering particularly for deformed nuclei with 0^+ ground states. The secondary energy distribution for these neutrons resembles elastic scattering more than an evaporation spectrum.

4. The recommended procedure for specifying inelastic scattering cross sections is to give level excitation cross sections for as many levels as possible and up to an incident energy for which level energies, spins, and parities are known. Above this point and up to 20 MeV, estimates should be made for those levels that have significant direct interaction contributions. Any remaining inelastic scattering should be treated as continuum.

5. Level excitation cross sections must start with zero cross section at the threshold energy. If the cross section for a particular level does not extend to 20 MeV, it must be <u>double-valued at the highest</u> energy point, for which the cross section is non-zero. The second cross section value at that point should be zero and it should be followed by another zero value at 20 MeV. This will positively show that the cross section has been truncated.

6. An LR flag specifies inelastic scattering to levels that de-excite by particle emission or pair production rather than by γ emission. Use the LR flag to completely define a reaction like (n,n'x). The LR flag is to be used in the reactions MT = 51,52,53,..., 90, and 91 to define x in (n,n'x). If x is unspecified then LR = 0. If x is a particle then LR becomes the MT number (see Appendix B) that defines the reaction; e.g., if the reaction is (n,n'p), then LR = 28. When LR > 0, then S is the breakup energy for the combined reaction prod-

ucts specified by the LR value. Even when S ⊟ Q, both values must be entered in the file. When LR = 39 and LR = 40, S ≡ 0.0 must be used.

When the LR flags are used, the following reactions take on slightly different meanings. MT = 4 means the reaction is (n,n' everything). MT = 51-91 means the reaction is (n,n' something), where "something" is defined by the LR flag. When MT = 91 is a composite of several de-excitation modes, then LR = 4.

This system has been established to facilitate accurate descriptions of the energy and angular distribution of these neutrons (the angular distributions are given in File 4).

If a particular level, which has been left in an excited state, decays by emission of particles of more than one type, then several sections must be given in File 3. Consider the case in which an excited state decays by emission of a proton and an α particle. That part of the reaction that represents (n,n'α) would use LR = 22, and the other part would be given in the next section (next higher MT number) and would use LR = 28 (n,n'p). The angular distribution for the neutron would have to be given in two different MT numbers in File 4, even though they represent the same neutron. Competition for the de-excitation of a level should be considered only if it is at least 10% of the total de-excitation cross section for that level. The section must be ordered by increasing Q-values. See Appendix B, page B.3 for specific LR Flag procedures.

3.2.4.5. <u>(n,2n), (n,3n), (n,4n) Cross Sections</u>

1. If any of the these reactions takes place, it must be given in File 3.

2. If the (n,2n) cross section reaction produces an isomeric state, then in addition to the total (n,2n) cross section (given in MT = 16), the isomeric state production cross section can be given in MT = 26. Processing codes con-

cerned only with the neutron cross sections can ignore the data given in MT = 26.

3. It is possible to represent the (n,2n) cross section either totally as direct (n,2n) (MT = 16) or as a combination of this and a time sequential reaction.

In the time-sequential (n,2n) reaction $A(n,n_1)A^*(n_2)(A-1)^*$, the "first" neutron (n_1) is essentially an inelastic scattering event that may leave the nucleus A* in one of several excited states. The "second" neutron (n_2) is subsequently emitted by the decay of the recoiling nucleus A*. Conservation of energy and momentum ensure a correlation between first and second neutrons for this time-sequential (n,2n) reaction for each level. The second neutron lab system angular and energy distributions can be drastically different for each level and must be described separately. In addition to the time-sequential (n,2n) reaction there may be a direct (n,2n) reaction, which proceeds without going through any intermediate states. The total (n,2n) reaction must therefore be considered as a composite of time-sequential (n,2n) plus a direct (n,2n).

The (n,2n) level events are described by treating the first neutron as coming from an inelastic level (energy ordered in MT = 51-90) and the second neutrons from levels represented by MT = 46-49, but there is no correlation between energy spectra of the first and second neutrons. Reaction types MT = 6-9, are used to represent the first neutron from the first few individual levels, and reaction types MT = 46-49 would be used to represent the second neutron from individual levels. Reaction type MT = 16 is to be used for the representation of both neutrons when time-sequential (n,2n) reactions do not apply or when detailed data are not available. The total (n,2n) cross section is the sum of

*This format is not restricted to ^{9}Be.

reaction types MT = 6-9 and 16 and does not include reaction types MT = 46-49. This procedure removes the necessity for representing the first neutron from an (n,2n) reaction by an inelastic level (MT = 51-90).

3.3.4.6. Fission Cross Sections

1. The total fission cross section must be given in MT = 18 for fissionable materials. Every attempt must be made to break this cross section into its various parts: first chance fission (n,f), MT = 19; second chance fission (n,n'f), MT = 20; third chance fission (n,2nf), MT = 21, and fourth chance fission (n,3nf), MT = 38.

2. The data in MT = 18 is to be the sum of data in MT = 19, 20, 21 and 38. The set of energy points used for MT = 18 should be the union of all sets for the partials.

3. If resolved or unresolved resonance parameters are given in File 2, the contributions to the total fission cross section in the resonance region are the sum calculated from File 2 and MT = 18 from File 3 (see Section 3.3). If data are given in File 3 for MT = 19-21 or 38, they must sum to data in File 3 for MT = 18. Since only the total fission cross section can be calculated from the resonance parameters to be added to File 3 values for MT = 18, the resonance region should not extend above the threshold for second chance fission data (MT = 20) given in File 3, to prevent inconsistency between the total fission cross section and its partials. If MT = 20 data are present, MT = 19 must exist and cover the full range of MT = 18 data.

4. The energy release per fission (Q minus neutrino energy) is entered as the Q value for MT's = 18, 19, 20, 21, and 38.

3.3.4.7. Absorption Cross Sections (MT = 27)

The absorption cross section is not required. It is defined as the sum of MT = 18 (total fission) plus MT = 101 (total neutron disappearance).

3.3.4.8. (n,n'x) Reaction Cross Sections

The cross sections for those reactions in which the secondary neutron leaves the target nucleus in an excited state should be given in File 3 as sections using the MT's in the series MT = 51, 52,..., 90, 91. In cases in which there are several reactions like (n,n'x), it is better to enter the reactions separately in File 3 under their regular MT numbers.

3.3.4.9. Neutron Disappearance Cross Section (MT - 101)

The neutron disappearance cross section is the sum of all cross sections in which a neutron is not in the exit channel. It is the sum of MT = 102 - 109 and 111 - 114.

3.3.4.10. Target Destruction Cross Section (MT = 120)

The target destruction cross section will depend on the various reaction mechanisms present. In general, it is the nonelastic cross section minus the total (n,n'γ) cross sections.

3.3.4.11. (n,x) Reaction Cross Sections (MT = 102, ..., 114)

1. If resolved or unresolved resonance parameters are given in File 2, the contribution to the radiative capture cross section in the resonance region is the sum calculated from File 2 and MT = 102 in File 3 (see Section 3.3.).

2. If both (n,p) and (n,2p) are given, they are not redundant. Both should be given, if present.

3. Partial cross sections such as n,p_0; n,p_1; ..., etc., should be given using the MT = 700 series for materials in which particle heating is important. The (n,p) cross section MT = 103 is equal to the sum of MT = 700 through MT = 718.

3.3.4.12. Reaction Cross Sections to Discrete and Continuum Levels (MT = 700 series)

For studies of radiation damage and/or long-lived activation in fission and fusion reactors, additional information about exit particles is natural. Exit protons will be used for illustration, although the same arguments hold for deuterons, tritons, and ^{3}He and ^{4}He particles. The MT = 700 series allows the cross sections and the energy and angular distributions for protons leaving the final nucleus in the ground through the 17th excited state plus a continuum level to be described by using MT = 700 to MT = 718.

Data in sections MT = 700 to MT = 718 for (n,p_o) through (n,p_c) must add up to MT = 103. In some cases cross section information about the exit proton is needed that has already been included in the ENDF/B files. For example, the (n,n'p) cross section is usually found under MT = 51-91. It may be, however, that this proton energy distribution is more important for radiation damage studies than the energy of the neutron. In this case, a so-called redundant cross section that is already part of σ_{tot} may be included in Section MT = 719. If used, this section gives the production cross section of the exit protons excluding protons from the discrete (n,p) reactions (MT = 700 to 717).

Similar procedures are used in Sections MT = 720 through MT = 799 to describe the exit deuteron (720+), triton (740+), and ^{3}He (760+) and ^{4}He (780+) particles.

3.4. Relationship Between File 3 and Other Files

If File 2 (Resonance Data) contains resolved and/or unresolved resonance parameters, then in order to obtain the total cross section (MT=1) the radiative capture cross section (MT=102), fission cross section (MT=18), and elastic scattering cross section (MT=2), the cross sections calculated from these parameters must be added to the appropriate data given in File 3. Any contribution from File 2 to radiative capture or fission must also be included in the File 3 nonelastic cross section (MT=3). The contributions from Files 2 and 3 must be summed to obtain the correct cross sections for neutron energies within the energy ranges specified for the resolved and/or unresolved resonance parameters. For this case, the cross sections given in File 3 may contain, for example, corrections (background cross sections) to take into account multilevel interference effects that were apparent in the experimental data where it was not possible to construct a set of resonance parameters that adequately fitted the measured data. Cross sections in File 3 to be added to File 2 are specified at 0° Kelvin and are intended to be combined with File 2 data calculated at 0° Kelvin.

Some materials will not have resonance parameters. However, they will have a scattering length, given in File 2, that can be used to calculate the potential elastic scattering cross section, which is then used to calculate resonance self-shielding effects in other materials. For these materials the elastic scattering cross section in File 3 must not be added to this potential scattering cross section, since the File 3 data for these materials comprise the entire scattering cross section.

Double-valued points (discontinuities in the cross sections) are allowed anywhere in File 3. They must always be given at the upper and lower energy limits of the resolved and unresolved resonance regions.

To obtain absolute values for differential (in angle) scattering cross sections, the data in File 4 have to be combined with the cross sections for the corresponding MT number given in File 3. The File 4 data (see Section 4) may be given as either tabulated normalized probability distributions, $p(\mu,E)$, or Legendre polynomial expansion coefficients, $f_\ell(E)$.

Note that the derived quantities $\bar{\mu}_{Lab}$, ξ, and γ are entirely from File 4 angular distribution data for elastic scattering. These data are included in File 3 for convenience and must be consistent with File 4 data.

Secondary energy distributions are expressed as normalized probability distributions and are given in File 5. The differential (in secondary energy) cross sections for a reaction of a particular type are obtained by multiplying the normalized probability distribution by the corresponding (same MT number) cross section, $\sigma(E)$, given in File 3. An exception is the data for inelastic scattering to various levels and the continuum; only the secondary energy distribution for the continuum is to be found in File 5. The excitation cross sections for discrete levels are given in File 3, and the angular distributions for these secondary neutrons are given in File 4; therefore, the secondary neutron energies are uniquely defined.

Absolute values for the double differential (in secondary energy and angle) scattering cross sections may be obtained by combining the data in File 6 and the cross sections in File 3.

3.5. General Suggestions for Preparing Data for File 3

The limit on the number of energy points (NP) to be used to represent a particular cross section is 5000. The evaluator should not use more points than are necessary to represent the cross section accurately.

Cross section data for nonthreshold reaction types should cover the energy range from 10^{-5} eV to 20 MeV for all materials. For other reactions the cross section data should start at the reaction threshold energy (with a value of 0.0) and continue up to 20 MeV. For nonthreshold reactions a cross section value must be given at 0.0253 eV.

The reaction Q-value is defined as the kinetic energy (eV) released by a reaction (positive) or required for a reaction (negative). Except for fission, for a reaction having a threshold, the threshold energy E_{th} is given by

$$E_{th} = \frac{AWR + 1}{AWR} |Q|$$

where AWR is the atomic mass ratio given on the HEAD card of each section.

For a material that is a mixture of several isotopes, the Q-value is not uniquely defined. The threshold energy generally should pertain to the particular isotope that contributes to the cross section at the lowest energy, but see discussion in Section 3.2.2.2.

The total cross section should, as a minimum, be given at every energy point at which at least one partial cross section is given. This will allow the partial cross sections to be added together and checked against the total cross section for any possible errors. In certain cases more points may be necessary in the total cross section over a given energy range than are required to specify the corresponding partial cross sections. For example, a constant elastic scattering cross section and a 1/v (n,γ) cross section could be exactly specified over a given energy range by linear interpolation on a log-log scale (INT = 5), but the sum of the two cross sections would not be exactly linear on a log-log scale.

The inelastic scattering cross section (MT = 4) should be given and should be exactly equal to the sum of the cross sections for inelastic scattering to the various discrete levels (MT = 51, 52, 53,..., 90) and the continuum (MT = 91).

The total inelastic scattering cross section and the contributing partial cross sections should be specified on the same energy mesh above the respective thresholds. Linear-linear interpolation (INT=2) or linear-log (INT=3) should be used for these cross sections.

In general, care must be used in specifying cross sections and the interpolation scheme to be used to determine the cross sections between input energy points. For example, if a cross section has a value of zero at the threshold energy and a non-zero value at the next higher energy point, a problem will be created if a log-linear or a log-log interpolation scheme is used.

4. FILE 4, ANGULAR DISTRIBUTIONS OF SECONDARY NEUTRONS

4.1. General Description

File 4 contains representations of angular distributions of secondary neutrons. Normally, these distributions will be given for elastically scattered neutrons and for the neutrons resulting from discrete level excitation due to inelastic scattering. However, angular distributions must also be given for neutrons resulting from (n,n' continuum) and (n,2n) reactions. In these cases the angular distributions will be integrated over all final neutron energies.

Angular distributions for a specific reaction type (MT number) are given for a series of incident neutron energies, in order of increasing energy. The energy range covered should be the same as that for the same reaction type in File 3. Angular distributions for several different reaction types (MT's) may be given in File 4 for each material, in ascending order of MT number.

The angular distributions are expressed as normalized probability distributions, i.e.,

$$\int_{-1}^{1} p(\mu,E)\,d\mu = 1,$$

where $p(\mu,E)\,d\mu$ is the probability that a neutron of incident energy E will be scattered into the interval $d\mu$ about an angle whose cosine is μ. The units of $p(\mu,E)$ are (unit cosine)$^{-1}$. Since the angular distribution of scattered neutrons is generally assumed to have azimuthal symmetry, the distribution may be represented as a Legendre polynomial series,

$$p(\mu,E) = \frac{2\pi}{\sigma_s(E)} \frac{d\sigma}{d\Omega}(\Omega,E) = \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell(E) P_\ell(\mu)$$

where μ = cosine of the scattered angle in either the laboratory or the center-of-mass system;

E = energy of the incident neutron in the laboratory system;

$\sigma_s(E)$ = the scattering cross section, e.g., elastic scattering at energy E as given in File 3 for the particular reaction type (MT);

ℓ = order of the Legendre polynomial;

$\frac{d\sigma}{d\Omega}(\Omega,E)$ = differential scattering cross section in units of barns per steradian;

a_ℓ = the ℓ^{th} Legendre polynomial coefficient and it is understood that $a_o = 1.0$.

The angular distributions may be given in one of two representations, and in either the CM or LAB systems. In the first method the distributions are given by tabulating the normalized probability distribution, $p(\mu,E)$, as a function of incident neutron energy. Using the second method, the Legendre polynomial expansion coefficients, $a_\ell(E)$, are tabulated as a function of incident neutron energy.

Absolute diffential cross sections are obtained by combining data from Files 3 and 4. If tabulated distributions are given, the absolute differential cross section (in barns per steradian) is obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi} p(\mu,E)$$

where $\sigma_s(E)$ is given in File 3 (for the same MT number) and $p(\mu,E)$ is given in File 4. If the angular distributions are represented as Legendre polynomial coefficients, the absolute differential cross sections are obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi} \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell(E) P_\ell(\mu)$$

where $\sigma_s(E)$ is given in File 3 (for the same MT number) and the coefficients $a_\ell(E)$ are given in File 4.

Also, a transformation matrix may be given in File 4 that can be used to transform a set of Legendre expansion coefficients, which are given to describe elastic scattering angular distributions, from one frame of reference to the other. The Legendre expansion coefficients $a_\ell(E)$ in the two systems are related through an energy-independent transformation matrix, $U_{\ell m}$, and its inverse, $U^{-1}_{\ell m}$:

$$a_\ell^{Lab}(E) = \sum_{m=0}^{NM} U_{\ell m}\, a_m^{CM}(E)$$

and

$$a_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell m}\, a_m^{Lab}(E)$$

Expressions for the matrix elements of U and U^{-1} may be found in papers by Zweifel and Hurwitz[1] and Amster[2]. Transformation matrices for nonelastic reactions are not incident energy independent and are not given in File 4.

The transformation matrices should be square, with the number of rows equal to NM + 1 where NM is the maximum order of the Legendre polynomial series used to describe any elastic angular distribution in this file. The transformation matrix is given as an array of numbers, V_K, where K = 1, .,.,., NK, and NK = $(NM + 1)^2$, and where $K = 1 + \ell + m\,(NM + 1)$. The values of K indicates how the $(\ell,m)^{th}$ element of the matrix may be found in array V_K. This means that the elements of the matrix $U_{\ell,m}$ or $U^{-1}_{\ell,m}$ are given column-wise in the array V_K:

$$
\begin{array}{cccc}
U_{0,0} & U_{0,1} & \cdots\cdots\cdots & U_{0,NM} \\
U_{1,0} & U_{1,1} & \cdots\cdots\cdots & U_{1,NM} \\
\vdots & \vdots & \cdots\cdots\cdots & \vdots \\
\vdots & \vdots & \cdots\cdots\cdots & \vdots \\
\vdots & \vdots & \cdots\cdots\cdots & \vdots \\
U_{NM,0} & U_{NM,1} & \cdots\cdots\cdots & U_{NM,NM}
\end{array}
$$

4.2. Formats

File 4 is divided into sections, each containing data for a particular reaction type (MT number) and ordered by increasing MT number. Each section always starts with a HEAD record and ends with a SEND record. If the section contains a description of the angular distributions for elastic scattering, the transformation matrix is given first (if present) and this is followed by the representation of the angular distributions.

---

1. P.F. Zweifel and H. Hurwitz, Jr., J. Appl. Phys. 25, 1241 (1954).
2. H. Amster, J. Appl. Phys. 29, 623 (1958).

The following quantities are defined.

LTT is a flag to specify the representation used and it may have the following values:

LTT = 0, all angular distributions are isotropic,

LTT = 1, the data are given as Legendre expansion coefficients, $a_\ell(E)$;

LTT = 2, the data are given as normalized probability distributions, $p(\mu,E)$.

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 0, not all isotropic,

LI = 1, all isotropic.

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NL is the highest order Legendre polynomial that is given at each energy ($NL \leq 20$).

NK is the number of elements in the transformation matrix ($NK \leq 441$). $NK = (NM + 1)^2$.

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering

in either the center-of-mass or the laboratory system. NM should be an even number.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy (NP $\leq$ 101).

Other commonly used variables are given in the Glossary (Appendix A).

The structure of a section depends on the values of LTT (representation used, $a_\ell(E)$ or $p(\mu,E)$), and LVT (transformation matrix given?), but it always starts with a HEAD record of the form

(ZA, AWR, LVT, LTT, 0, 0)HEAD.

4.2.1. Legendre Polynomial Coefficients and Transformation Matrix Given:

LTT = 1, LVT = 1, and LI = 0

When LTT = 1 (angular distributions given in terms of Legendre polynomial coefficients) and LVT = 1, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0)HEAD LTT = 1, LVT = 1

(MAT, 4, MT/0.0, AWR, LI, LCT, NK, NM/V_K)LIST LI = 0

(MAT, 4, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 4, MT/T , E_1 , LT, 0 , NL, 0/$a_\ell(E_1)$)LIST

(MAT, 4, MT/T , E_2 , LT, 0 , NL, 0/$a_\ell(E_2)$)LIST

(MAT, 4, MT/T , E_{NE}, LT, 0 , NL, 0/$a_\ell(E_{NE})$)LIST

(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0)SEND

Note that T and LT refer to temperature (in °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified.

4.2.2. Legendre Polynomial Coefficients Given and the Transformation Matrix Not Given: LTT = 1, LVT = 0, and LI = 0

If LTT = 1 and LVT = 0, the structure of a section is the same as above, except that the second record (a LIST record) is replaced by

(MAT, 4, MT/ 0.0, AWR, LI, LCT, 0, 0) CONT. LI = 0

This form is always used for angular distributions of nonelastically scattered neutrons when Legendre polynomial expansion coefficients are used.

4.2.3. Tabulated Probability Distributions and Transformation Matrix Given: LTT = 2, LVT = 1, and LI = 0

If the angular distributions are given as tabulated probability distributions, LTT = 2, and a transformation matrix is given for elastic scattering, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0)HEAD LVT = 1, LTT = 2

(MAT, 4, MT/0.0, AWR, LI, LCT, NK, NM/V_K)LIST LI = 0

(MAT, 4, MT/0.0, 0.0, 0 , 0 , NR, NE/E_{int})TAB2

(MAT, 4, MT/T , E_1 , LT, 0, NR, NP/μ_{int}/$p(\mu,E_1)$)TAB1

(MAT, 4, MT/T , E_2 , LT, 0, NR, NP/μ_{int}/$p(\mu,E_2)$)TAB1

(MAT, 4, MT/T , E_{NE}, LT, 0 , NR, NP/μ_{int}/$p(\mu,E_{NE})$)TAB1

(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0, 0)SEND

T and LT are normally zero. See Appendix F for details on temperature dependence.

4.2.4. Tabulated Probability Distributions Given and Transformation Matrix Not Given: LTT = 2, LVT = 0, and LI = 0

The structure of a section is the same as above, except that the second record (a LIST record) is replaced by

(MAT, 4, MT/ 0.0, AWR, LI, LCT, 0, 0)CONT. LI = 0

This form is always used for angular distribution of nonelastically scattered neutrons when tabulated angular distributions are given.

4.2.5. All Angular Distributions Isotropic:

LTT=0, LVT=0, LI=1

When all angular distributions for a given MT are assumed to be isotropic then the section structure is:

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0)HEAD LVT=0, LTT=0

(MAT, 4, MT/0.0, AWR, LI, LCT, 0,0)CONT LI=1

(MAT, 4, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

4.3. Procedures

The angular distributions for elastic scattering should be given as Legendre polynomial coefficients, $a_\ell(E)$'s (LTT = 1), and they should be given in the CM system (LCT = 2). It is recommended that the angular distributions of neutrons from nonelastic reactions (such as continuum inelastic, fission, etc.) be given as tabulated distributions, $p(\mu,E)$'s, and that they be in the Lab system. All angular distribution data should be given at the minimum number of

incident energy points that will accurately describe the energy variation of the distributions.

When the angular distributions are represented as Legendre polynomial coefficients, certain procedures should be followed. Enough Legendre coefficients should be used to accurately represent the recommended angular distribution at a particular energy point and ensure that the interpolated distribution is everywhere positive. The number of coefficients (NL) may vary from energy point to energy point; in general, NL will increase with increasing incident energy. A linear-linear interpolation scheme (INT = 2) must be used to obtain coefficients at intermediate energies. This is required to ensure that the interpolated distribution is positive over the cosine interval from -1.0 to + 1.0; it is also required because some coefficients may be negative. In no case should NL exceed a value of 20. If more than 20 coefficients appear to be required to obtain a non-negative distribution, a constrained Legendre polynomial fit to the data should be given. NL = 1 is allowed at low energies to specify an isotropic angular distribution.

When angular distributions are represented as tabulated data, certain procedures should be followed. Sufficient angular points (cosine values) should be given to accurately represent the recommended distribution. The number of angular points may vary from distribution to distribution. The cosine interval must be from -1.0 to +1.0. The interpolation scheme for $p(\mu,E)$ vs. μ should be log-linear (INT = 4), and that for $p(\mu,E)$ vs. E should be linear-linear (INT = 2).

Representation of angular distributions of neutrons for the thermal energy range presents a problem. Either free-atom or bound-atom scattering data may be given in File 4 for a material, but not both.

The formats given above do not allow an energy-dependent transformation matrix to be given, so transformation matrices may not be given for nonelastic scattering reaction types. When a processing code wishes to transfer inelastic level angular distributions expressed as Legendre polynomial coefficients from the Lab to the CM system, or CM to LAB, a distribution should be generated and transformed point-wise to the desired frame of reference. The pointwise angular distributions can then be converted to Legendre polynomial coefficients in the new frame of reference.

The formats given above do not allow both Legendre polynomial coefficients and tabulated data to represent angular distributions for a given reaction type (MT number). If tabulated data are required to describe highly structured elastic scattering angular distributions at high energies, tabulated data must also be used to describe elastic scattering angular distributions at low energies.

4.4. Procedures for Specific Reactions

4.4.1. Elastic Scattering (MT = 2)

1. A transformation matrix should be given in File 4 for elastic scattering. If the angular distributions are given for the CM system, the matrix should be for CM to LAB conversion. The parameter NM should be even, and it must be equal to or greater than ℓ_{max} used in any of the angular distributions (if Legendre coefficients are given). The parameter NK is equal to $(NM + 1)^2$.

2. Legendre polynomial representations should be used for elastic scattering angular distributions and discrete channel scattering and must be given in the CM system. When this representation is given, the maximum order of the polynomial for each incident energy should be even and ℓ_{max} must be ≤ 20.

3. Care must be exercised in selecting an incident energy mesh for certain light-to-medium mass materials. Here it is important to relate any known structure in the elastic scattering cross section to the energy dependent variations in the angular distributions. These two features of the cross sections cannot be analyzed independent of one another. Remember, processing codes operate on MT = 2 data given in Files 3 and 4. (Structure in the total cross section is not considered when generating energy transfer arrays.) It is better to maintain consistency in any structure effects between File 3 and File 4 data than to introduce structure in one file and ignore it in the other.

4. Consistency must be maintained between angular distribution data given for elastic and inelastic scattering. This applies not only to structural effects but also to how the distributions were obtained. Frequently, the evaluated elastic scattering angular distributions are based on experimental results that at times contain contributions from inelastic scattering to low-lying levels (which in turn may contain direct interaction effects). If inelastic contributions have been subtracted from the experimental angular distributions, this process must be done in a consistent manner. The same contributions must be subtracted from both the integrated elastic scattering and the angular distribution. Be sure that these contributions are included in the inelastic scattering cross section (both integrated data and angular distributions). This is particularly important when the inelastic contributions are due to direct interaction, since the angular distributions are not isotropic or symmetric about 90° but they are generally forward peaked.

5. Do not use an excessive number of incident energy points for the angular distributions. The number used should be determined by the amount of variation in the angular distributions.

6. An incident energy point must be given at 10^{-5} eV. It is helpful, but not required, to include a point at 0.0253 eV. A point must be given at the highest energy point for which the angular distribution is isotropic. The highest incident energy point must be 20 MeV.

7. A relationship exists between the total cross section and the differential cross section at forward angles (Wick's limit or optical theorem).

$$\frac{d\sigma}{d\Omega}(0) \geq \sigma_w = (3.0560 \times 10^{-8} E_o) \frac{AWR^2}{(1 + AWR)^2} (\sigma_T)^2 \frac{\text{barns}}{\text{steradian}} ,$$

where E_o is in eV and σ_T in barns. Care should be taken to observe this inequality, especially at high energies.

4.4.2. Inelastic Scattering Cross Sections

1. Do not give angular distribution data for MT = 4.

2. Always give angular distribution data for any of the following if they are given in File 3: MT = 51, 52, 53, ..., 91.

3. Discrete level data must be given in the CM system. All continuum reactions, such as MT = 91, must be given in the LAB system.

4. Discrete channel (two body) angular distributions (e.g., MT = 2, 51 - 90, 701...) should be given as Legendre coefficients in the CM system. Continuum angular distributions should be tabular in the LAB system.

5. Isotropic angular distributions should be used unless the degree of anisotropy exceeds 5%. If any level excitation cross sections contain significant direct interaction contributions, angular distributions are very important.

6. Use the precautions outlined above when dealing with level excitation cross sections that contain a large amount of structure.

7. Do not overcomplicate the data files. Restrict the number of distributions to the minimum required to accurately represent the data.

8. Angular distributions for exit protons, etc., are given in the MT = 700 series, and for photons in File 14.

4.4.3. All Other Neutron Producing Reactions

Angular distribution data must be given for all neutron producing reactions. Make sure, giving these data, that they are realistic. Tabulated angular distributions are preferred.

5. FILE 5, ENERGY DISTRIBUTION OF SECONDARY NEUTRONS

5.1. General Description

File 5 contains data for the energy distributions of secondary neutrons, expressed as normalized probability distributions. Each section of the file gives the data for a particular reaction type (MT number). The sections are then ordered by increasing MT number.

Data will be given in File 5 for all reaction types that produce secondary neutrons, unless the secondary neutron energy distributions can be implicitly determined from data given in Files 3 and/or 4. No data will be given in File 5 for elastic scattering (MT = 2), since the secondary energy distributions can be obtained from the angular distributions in File 4. No data will be given for neutrons that result from excitation of discrete inelastic levels when data for these reactions are given in both File 3 and File 4 (MT = 51, 52, ..., 90). Data should be given in File 5 for MT = 91 (inelastic scattering to a continuum of levels), MT = 18 (fission), MT = 16 (n,2n), MT = 17 (n,3n), MT = 455 (delayed neutrons from fission), and certain other nonelastic reactions that produce secondary neutrons.

The energy distributions, $p(E \rightarrow E')$, are normalized so that

$$\int_0^{E'_{max}} p(E \rightarrow E')\, dE' = 1 ,$$

where E'_{max} is the maximum possible secondary neutron energy and its value depends on the incoming neutron energy E and the analytic representation of $p(E \rightarrow E')$. The secondary neutron energy E' is always expressed in the laboratory system.

The differential cross section is obtained from

$$\frac{d\sigma\ (E \to E')}{dE'} = m\ \sigma(E)\ p\ (E \to E'),$$

where $\sigma(E)$ is the cross section as given in File 3 for the same reaction type number (MT) and m is the neutron multiplicity for this reaction (m is implicit; e.g., m = 2 for n,2n reactions).

The energy distributions $p(E \to E')$ can be broken down into partial energy distributions, $f_k(E \to E')$, where each of the partial distributions can be described by different analytic representations;

$$p(E \to E') = \sum_{k=1}^{NK} p_k(E)\ f_k(E \to E'),$$

and at a particular incident neutron energy E,

$$\sum_{k=1}^{NK} p_k(E) = 1,$$

where $p_k(E)$ is the fractional probability that the distribution $f_k(E \to E')$ can be used at E.

The partial energy distributions $f_k(E \to E')$ are represented by various analytical formulations. Each formulation is called an energy distribution law

and has an identification number associated with it (LF number). The allowed energy distribution laws are given below.

Secondary Energy Distribution Laws

LF = 1, Arbitrary tabulated function:

$f(E \to E') = g(E \to E')$.

A set of incident energy points is given, E_i and $g(E_i \to E')$ is tabulated as a function of E'.

LF = 5, General evaporation spectrum:

$f(E \to E') = g\left(E'/\theta(E)\right)$.

$\theta(E)$ is tabulated as a function of incident neutron energy, E;

$g(x)$ is tabulated as a function of x, $x = E'/\theta(E)$.

LF = 7, Simple fission spectrum (Maxwellian):

$$f(E \to E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)} .$$

I is the normalization constant,

$$I = \theta^{3/2}\left[\frac{\sqrt{\pi}}{2}\,\mathrm{erf}\left(\sqrt{(E-U)/\theta}\right) - \sqrt{(E-U)/\theta}\; e^{-(E-U)/\theta}\right] ;$$

θ is tabulated as a function of energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

LF = 9, Evaporation spectrum:

$$f\ (E \to E') = \frac{E'}{I} e^{-E'/\theta} .$$

I is the normalization constant,

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta} \left(1 + \frac{E-U}{\theta}\right)\right];$$

θ is tabulated as a function of incident neutron energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

LF = 11, Energy dependent Watt spectrum:

$$f(E \rightarrow E') = \frac{e^{-E'/a}}{I} \sinh\left(\sqrt{bE'}\right).$$

I is the normalization constant,

$$I = \frac{1}{2}\sqrt{\frac{\pi a^3 b}{4}} \exp\left(\frac{ab}{4}\right) \left[\operatorname{erf}\left(\sqrt{\frac{E-U}{a}} - \sqrt{\frac{ab}{4}}\right) + \operatorname{erf}\left(\sqrt{\frac{E-U}{a}} + \sqrt{\frac{ab}{4}}\right)\right] - a \exp\left(-\left(\frac{E-U}{a}\right) \sinh\left(\sqrt{b(E-U)}\right)\right)$$

a and b are energy dependent;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

NOTE: Distribution laws are not presented for LF = 2, 3, 4, 6, 8, or 10. These laws are no longer used.

The data are given in each section by specifying the number of partial energy distributions that will be used. The same energy mesh should be used for each one. The partial energy distributions may all use the same energy distribution law (LF number) or they may use different laws.

5.2. Formats

Each section of File 5 contains the data for a particular reaction type (MT number), starts with a HEAD record, and ends with a SEND record. Each subsection contains the data for one partial energy distribution. The structure of a subsection depends on the value of LF (the energy distribution law).

The following quantities are defined

NK — is the number of partial energy distributions. There will be one subsection for each partial distribution.

U — is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ — is a parameter used to describe the secondary energy distribution.

If LF = 5, 7, or 9, θ is an effective nuclear temperature in eV.

LF — is a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution). (The definitions for LF are given in Section 5.1.).

$p_k(E_N)$ — is the fractional part of the particular cross section that can be described by the k^{th} partial energy distribution at the N^{th} incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ — is the k^{th} partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

NP is the number of incident energy points at which $p_k(E)$ is given.

a,b are parameters used in the energy dependent Watt spectrum. (LF = 11.)

NE is the number of incident energy points at which a tabulated distribution is given.

(NE $\leq$ 200.)

NF is the number of secondary energy points in a tabulation.

(NF $\leq$ 1000.)

The structure of a section has the following form:

(MAT, 5, MT/ZA, AWR, 0, 0, NK, 0)HEAD

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

(MAT, 5, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

The structure of a subsection depends on the value of LF. Subsections should be ordered by increasing values of LF. For cases in which more than one subsection contains data using the same LF, these subsections should be ordered by increasing values of θ. The formats for the various values of LF are given below.

LF = 1, Arbitrary tabulated function

(MAT, 5, MT/T , 0.0 , LT , LF=1 , NR , NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int})TAB2

(MAT, 5, MT/T , E_1 , LT , 0 , NR , NF/E'_{int}/

E'_1, $g(E_1 \rightarrow E'_1)$, E'_2 , $g(E_1 \rightarrow E'_2)$, E'_3 , $g(E_1 \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_1 \rightarrow E'_{NF})$)TAB1

(MAT, 5, MT/T , E_2, , LT , 0 , NR , NF/E'_{int}/

E'_1 , $g(E_2 \rightarrow E'_1)$, E'_2 , $g(E_2 \rightarrow E'_2)$, E'_3 , $g(E_2 \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_2 \rightarrow E'_{NF})$)TAB1

(MAT, 5, MT/T , E_{NE} , LT , 0 , NR , NF/E'_{int}/

E'_1 , $g(E_{NE} \rightarrow E'_1)$, E'_2 , $g(E_{NE} \rightarrow E'_2)$, E'_3 , $g(E_{NE} \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_{NE} \rightarrow E'_{NF})$)TAB1

Note that the incident energy mesh for $p_k(E)$ does not have to be the same as the E mesh used to specify the energy distributions. The interpolation scheme used between incident energy points, E, and between secondary energy points, E', should be linear-linear. T and LT refer to possible temperature (physical) dependence.

LF = 5, General evaporation spectrum

(MAT, 5, MT/ U , 0.0 , 0 , LF=5, NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR, NE/E_{int}/θ(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR, NF/x_{int}/g(x))TAB1 $x = \frac{E'}{\theta(E)}$

LF = 7, Simple fission spectrum (Maxwellian)

(MAT, 5, MT/ U , 0.0 , 0 , LF=7, NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int}/θ(E))TAB1

LF = 9, Evaporation spectrum

(MAT, 5, MT/ U , 0.0, 0, LF=9,NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/θ(E))TAB1

LF = 11, Energy dependent Watt spectrum

(MAT, 5, MT/ U , 0.0, 0, LF=11,NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/a(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/b(E))TAB1

Note that no formats have been described for LF = 2, 3, 4, 6, 8, or 10. These laws are no longer defined.

5.3. Procedures

As many as three different energy meshes may be required to describe the data in a subsection (one partial distribution). These are the incident energy mesh for $p_k(E)$, the incident energy mesh at which secondary distributions are given, $f_k(E \rightarrow E)$, and the secondary energy mesh for $f_k(E \rightarrow E')$. It is recommended that a linear-linear or a linear-log interpolation scheme be used for the first two energy meshes, and a linear-linear interpolation for the last energy mesh.

Double energy points must be given in the incident energy mesh whenever there is a discontinuity in any of the $p_k(E)$'s (this situation occurs fairly frequently). This energy mesh must also include threshold energy values for all reactions being described by the $p_k(E)$'s. Zero values for p_k must be given for energies below the threshold (if applicable).

Two nuclear temperatures may be given for the (n,2n) reaction. Each temperature, θ, may be given as a function of incident neutron energy. In this case $p_1(E) = p_2(E) = 0.5$. A similar procedure may be followed for the (n,3n) and other reactions.

A constant, U, is given for certain distribution laws (LF = 5, 7, 9, or 11). The constant, U, is provided to define the proper upper limit for the secondary energy distribution so that $0 \leq E' \leq E - U$. The value of U depends on how the data are represented for a particular reaction type. Consider U for inelastic scattering:

Case A The total inelastic scattering cross section is described as a continuum. U is the threshold energy for exciting the lowest level in the residual nucleus.

Case B For the energy range considered, the first three levels are described explicitly (either in File 3, MT = 51, 52, and 53, or in File 5, and the rest of the inelastic cross section is treated as a continuum. U is the threshold energy (known or estimated) for the fourth level in the residual nucleus.

If the reaction being described is fission, then U should be a large negative value ($U \approx -20.0 \times 10^6$ eV to $-30. \times 10^6$ eV). In this case neutrons can be born with energies much larger than the incident neutron energy. It is common practice to describe the inelastic cross section as the sum of excitation cross sections (for discrete levels) for neutron energies up to the point where level positions are no longer known. At this energy point, the total inelastic cross section is treated as a continuum. This practice can lead to erroneous secondary energy distributions for incident neutron energies just above the cutoff energy. It is recommended that the level excitation cross sections for the first several levels (e.g., 4 or 5 levels) be estimated for several MeV above the cutoff energy. The continuum portion of the inelastic cross section will be zero at the cutoff energy, and it will not become the total inelastic cross section until several MeV above the cutoff energy.

It is recommended that the cross sections for excitation of discrete inelastic levels be described in File 3 (MT = 51, 52, ..., etc.). The angular distributions for the neutrons resulting from these levels should be given in File 4 (same MT numbers). The secondary energy distributions for these neutrons can be obtained analytically from the data in Files 3 and 4. This procedure is the only way in which the energy distributions can be given for these neutrons. For inelastic scatter, the only data required in File 5 are for MT = 91 (continuum part).

5.4. Additional Procedures

5.4.1. General Comments

1. Do not give File 5 data for the discrete level excitation data given in File 3 as MT = 51, 52, ...,90. If MT = 91 is given in File 3, a section for MT = 91 must be given in File 5. A section must also be given in File 5 for all other neutron producing reactions. Energy distributions for exit protons, etc., are given in the MT = 700 series, and for photons, in File 15.

2. Care must be used in selecting the distribution law number (LF) to be used to represent the data. As a rule, use the simplest law that will accurately represent the data. Use only tabulated distributions (LF = 1) when the data cannot be represented by an evaporation spectrum (LF = 9) or a Maxwellian (LF = 7).

3. A section in File 5 must cover the same incident energy range as was used for the same MT number in File 3. The sum of the probabilities for all laws used must be equal to unity for all incident energy points.

4. If the incident neutron energy exceeds several MeV, pre-equilibrium particle emission can be important, as illustrated from high resolution neutron and proton spectra measurements and analysis of pulsed sphere experiments. In

these cases either tabulated spectra or "mocked-up" levels can be constructed to supplement or replace simple evaporation spectra.

5. Note that prompt fission spectra are given under MT=18 with LF=11 preferred. The delayed fission spectra are given under MT=455 and 465.

5.4.2. LF = 1 (Tabulated Distributions)

Use only tabulated distributions to represent complicated energy distributions. Use the minimum number of incident energy points and secondary neutron energy points to accurately represent the data. The integral over secondary neutron energies for each incident energy point must be unity to within four significant figures. All interpolation schemes must be with linear-linear or log-linear (INT = 1, 2, or 3) to preserve probabilities upon interpolation. All secondary energy distributions must start and end with zero values for the distribution function $g(E \rightarrow E')$.

5.4.3. LF = 5 (General Evaporation Spectrum)

This law may be used for MT = 455 and 456, otherwise, LF = 1 should be used.

5.4.4. LF = 7 (Maxwellian Spectrum)

A linear-linear interpolation scheme is preferred for specifying the nuclear temperature as a function of energy.

5.4.5. LF = 9 (Evaporation Spectrum)

An evaporation spectrum is preferred for most reactions. Care must be taken in describing the nuclear temperature near the threshold of a reaction. Nuclear temperatures that are too large can violate conservation of energy.

5.4.6. LF = 11 (Watt Spectrum)

This is the preferred law to use for the prompt fission spectrum. A linear-linear interpolation scheme is preferred for specifying the parameter as a function of energy.

5.4.7. Selection of the Integration Constant, U

1. When LF = 5, 7, 9, or 11 is used, an integration constant, U is required. This constant is used in defining the upper energy limit of secondary neutrons; i.e., $E'_{max} = E_n - U$, where E_n is the incident neutron energy. U is a constant for the complete energy range covered by a subsection in File 5 and is given in the LAB system.

2. U is negative for fission reactions. The preferred value is -20 MeV.

3. In practice, U can be taken to be the absolute value of Q for the lowest level (known or estimated) that can be excited by the particular reaction within the incident energy range covered by the subsection. U is actually a function of the incident neutron energy, but it can be shown that it is always greater than the absolute value of Q and less than the threshold energy of the reaction. At large AWR, since E_{th} and $|Q|$ are approximately equal, either could be used but the absolute value of Q is preferred. At small AWR, using $|Q|$ for U is the best approximation and must be used.

4. The following three cases commonly occur in data files; procedures are given for obtaining U values.

Case A: The complete reaction is treated as a continuum.

$U = -Q$

where Q is the reaction Q-values.

Case B: The reaction is described by excitation of three levels (in File 3 as MT = 51, 52, 53) and a continuum part where Q_4 is the known or estimated Q-value for the fourth level.

$$U = -Q_4$$

Case C: The reaction is described by excitation of the first three levels (in File 3 as MT = 51, 52, 53) for neutron energies from the level thresholds up to 20 MeV, excitation of the next five levels (in File 3 as MT = 54, ..., 58) from their thresholds up to 8 MeV, and by a continuum part that starts at 5 MeV.

In this case two subsections should be used, one to describe the energy range from 5 to 8 MeV and another to describe the energy region from 8 to 20 MeV. In the first subsection (5 - 8 MeV),

$$U = -Q_9,$$

and the second (8 - 20 MeV),

$$U = -Q_4.$$

5.4.8. Multiple Nuclear Temperatures

Certain reactions, such as (n,2n), may require specification of more than one nuclear temperature. $\theta(E)$ should be given for each neutron in the exit channels; this is done by using more than one subsection for a reaction. The U value is the same for all subsections. The upper energy limit is determined by the threshold energy and not by level densities in the residual nuclei.

5.4.9. Average Energy for a Distribution

The average energy of a secondary neutron distribution must be less than the available energy for the reaction:

$$E_{avail} = E + \frac{1 + AWR}{AWR} Q .$$

The mean energy should be calculated from the distribution at each value of E. This mean is analytic in the three cases given below.

LF	$\bar{E}$
7	$\frac{3}{2}\theta - \frac{\theta^{5/2}}{I}\left(\frac{E-U}{\theta}\right)^{3/2} e^{-(E-U)/\theta}$
9	$2\theta - \frac{\theta^3}{I}\left(\frac{E-U}{\theta}\right)^{2} e^{-(E-U)/\theta}$
11	$\frac{1}{2I}\left\{a \exp\left(\frac{ab}{4}\right)\sqrt{\pi \frac{a^3 b}{4}}\left(3+\frac{ab}{2}\right)\left[\frac{\operatorname{erf}\left(\sqrt{\frac{E-U}{a}} - \sqrt{\frac{ab}{4}}\right) + \operatorname{erf}\left(\sqrt{\frac{E-U}{a}} + \sqrt{\frac{ab}{4}}\right)}{2}\right]\right.$

$$- 3a^2\sqrt{ab}\exp\left(-\left(\frac{E-U}{a}\right)\right)\left[\sqrt{\frac{E-U}{a}}\cosh\sqrt{b(E-U)} - \sqrt{\frac{ab}{4}}\sinh\sqrt{b(E-U)}\right]$$

$$- 2a^2\exp\left(-\left(\frac{E-U}{a}\right)\right)\left[\left(\left(\frac{E-U}{a}\right)+\frac{ab}{4}\right)\sinh\sqrt{b(E-U)} - \sqrt{b(E-U)}\cosh\sqrt{b(E-U)}\right]$$

$$\left. - 2a^2\left(1+\frac{3}{4}ab\right)\exp\left(-\left(\frac{E-U}{a}\right)\right)\sinh\sqrt{b(E-U)}\right\}$$

$$I = \int_0^{E'_{max}} f\,(E \to E')\,dE'$$

= the normalizing dominator (see 5.3). Thus $E_{avail}(E)$ > multiplicity * $\bar{E}$.

6. FILE 6, ENERGY-ANGULAR DISTRIBUTIONS FOR SECONDARY NEUTRONS

6.1. General Description

The use of File 6 is discouraged but the formats and procedures are presented here in case the File should be activated.

This file is provided to represent energy-angular distributions of secondary neutrons. Data are given here when it is not possible to provide accurate representation by using Files 4 and/or 5. This situation frequently arises when trying to provide a description of the secondary neutrons resulting from certain neutron reactions with fairly light nuclei.

Each section of the file contains the data for a particular reaction type (MT number) and the sections are ordered by increasing MT number. If data are given in File 6 for a particular reaction, no data will be given in Files 4 or 5 for the same reaction. The secondary neutron energy-angle distribution are expressed as normalized probability distributions, $p(E \rightarrow E', \mu)$.

$$\int_0^{E'_{max}} dE' \int_{-1}^{1} p(E \rightarrow E', \mu)\, d\mu = 1.$$

The differential cross section (in barns per steradian per eV) is obtained from

$$\frac{d^2\sigma\ (E \rightarrow E', \mu)}{d\Omega\ dE'} = \frac{\sigma\ (E)}{2\pi}\ m\ p(E \rightarrow E', \mu)\ ,$$

where $\sigma(E)$ is the cross section for a particular reaction as given in File 3 and/or File 2 for the same reaction type and m is the implied neutron multiplicity.

The angular part of the distribution may be specified in one of two ways (1), secondary energy distributions may be tabulated at a set of secondary

angles; (2), the probability distributions may be expressed as a Legendre polynomial expansion.

$$p(E \rightarrow E', \mu) = \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} \; a_\ell \; (E \rightarrow E',) \; P_\ell(\mu) \quad .$$

In this case, the zeroth coefficient, $a_o(E \rightarrow E')$, is not unity (as in File 4), but for a particular incident neutron energy, E,

$$\int a_o \; (E \rightarrow E') \; dE' = 1 \; .$$

The incident neutron energies and secondary angles and energies must be given in the laboratory system.

If distributions $p(E \rightarrow E', \mu)$ are tabulated at a series of angles, a set of secondary angles (cosines of the scattered angles) is selected. This set is the same for all incident energy points and the data are ordered by increasing values of the cosine (-1.0 to +1.0). At each angular point, the probability distributions $p(E \rightarrow E', \mu)$ are given for a set of incident neutron energies; i.e., a subsection of data is given for each angle and the format of a subsection resembles the format of a section in File 5. The secondary energy distribution laws (LF numbers) defined in Section 5.2 are used in this file.

When the distributions are represented by Legendre polynomial expansion coefficients, then a subsection is given for each coefficient, $a_\ell(E \rightarrow E')$. Its format is similar to that for a section in File 5 (Section 5.2). The first subsection contains data for the zeroth coefficient, $a_o(E \rightarrow E')$. The subsections are then ordered by increasing ℓ-value of the coefficients.

The following quantities are defined.

LCT is a flag indicating which reference frame is used for both secondary angles and energies.

LCT = 1: The data are given in the LAB system.

LCT = 2: (CM) Do not use.

LTT is a flag indicating which representation is used.

LTT = 1: The data are given as Legendre expansion coefficients.

LTT = 2: The data are given as a tabulation.

NL is the order of the expansion (when Legendre polynomial coefficients are given, LTT = 1).

LA is the value of ℓ (for the ℓ^{th} coefficient).

NA is the number of angles (cosines) at which the secondary distributions are given (NA $\leq$ 101.).

μ is the cosine of the scattered angle.

NK is the number of partial probability distributions used for this reaction type (used as in File 5, Section 5.2.)

LF is a flag that indicates which secondary energy distribution law is used for a particular partial probability distribution. (See Section 5.2).

6.2. Formats

Each section of the file gives the data for a particular reaction type. The structure of a section depends on the value of LTT (representation used).

The structure of a section for LTT = 1 (Legendre polynomial expansion) is

(MAT, 6, MT/ZA , AWR, 0, LTT, 0, 0)HEAD LTT = 1

(MAT, 6, MT/0.0, 0.0, 0, LCT, NL, 0) CØNT

<Subsection for $a_o(E \rightarrow E')$>

<Subsection for $a_{NL}(E \rightarrow E')$>

(MAT, 6, 0/0.0, 0.0, 0, 0, 0, 0)SEND

The structure of a subsection is identical to that of a section for secondary energy distributions in File 5 (Section 5.2) except that the SEND record is deleted (since the section in File 5 is used here as a subsection), and, the HEAD record is changed to read

(MAT, 6, MT/0.0, 0.0, LA, 0, NK, 0)CØNT

The following is the structure for a typical section, where LTT = 1 (Legendre expansion coefficients given), NK = 1 (one partial probability distribution, and LF = 1 (an arbitrary tabulated distribution).

(MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0)HEAD LTT = 1

(MAT, 6, MT/0.0, 0.0, 0, LCT, NL, 0)CØNT

(MAT, 6, MT/0.0, 0.0, LA, 0, NK, 0)CØNT LA = 0, NK = 1

(MAT, 6, MT/T, 0.0, LT, LF, NR, NP/$E_{int}/a_o(E)$)TAB1 LF = 1 Subsection for $a_o(E \rightarrow E')$

(MAT, 6, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 6, MT/T, E_1, LT, 0, NR, NF/$E'_{int}/a_o(E_1 \rightarrow E')$)TAB1

(MAT, 6, MT/T, E_2, LT, 0, NR, NF/$E'_{int}/a_o(E_2 \rightarrow E')$)TAB1

(MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/$E'_{int}/a_o(E_{NE} \rightarrow E')$)TAB1

(MAT, 6, MT/0.0, 0.0, LA, 0, NK, 0)CØNT LA = 1, NK = 1

(MAT, 6, MT/T, 0.0, LT, LF, NR, NP/$E_{int}/a_1(E)$)TAB1 LF = 1 Subsection for $a_1(E \rightarrow E')$

(MAT, 6, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 6, MT/T, E_1, LT, 0, NR, NF/$E'_{int}/a_1(E_1 \rightarrow E')$)TAB1

(MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/$E'_{int}/a_1(E_{NE} \rightarrow E')$)TAB2

<Subsection for $a_2(E \rightarrow E')$>

<Subsection for $a_{NL}(E \rightarrow E')$>

(MAT, 6, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

T and LT refer to possible temperature dependence (see Appendix F for format).

The structure of a section for LTT = 2 (tabulated distributions at a series of scattering angles) is

(MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0)HEAD LTT=2

(MAT, 6, MT/0.0, 0.0, 0, LCT, NR, NA/μ_{int})TAB2

<Subsection for $p(E \rightarrow E', \mu_1)$>

<Subsection for $p(E \rightarrow E', \mu_2)$>

<Subsection for $p(E \rightarrow E', \mu_{NA})$>

(MAT, 6, MT/0.0, 0.0, 0. 0. 0. 0)SEND

Again the structure of a subsection is identical to that of a section for secondary energy distributions in File 5 except that the SEND record is deleted (since the section in File 5 is used here as a subsection), and the HEAD record is changed to read

(MAT, 6, MT/ 0.0, μ, 0, 0, NK 0)CØNT

The structure of a typical section with LTT = 2 (tabulated distributions at a series of μ's), NK = 1 (one partial probability distribution) and LF = 1 (an arbitrary tabulated distributions) is

(MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0)HEAD LTT = 2

(MAT, 6, MT/0.0, 0.0, 0, LCT, NR, NA/μ_{int})TAB2

(MAT, 6, MT/0.0, μ_1 , 0, 0, NK, 0)CØNT NK = 1

(MAT, 6, MT/T, 0.0, LT, LF, NR, NP/E_{int}/$p(E, \mu_1)$)TAB1 LF = 1

(MAT, 6, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 6, MT/T, E_1 ,LT, 0, NR, NF/E'_{int}/$p(E_1 \rightarrow E', \mu_1)$)TAB1

(MAT, 6, MT/T, E_2 ,LT, 0, NR, NF/E'_{int}/$p(E_2 \rightarrow E', \mu_1)$)TAB1

(MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p(E_{NE} \rightarrow E', \mu_1)$)TAB1

(MAT, 6, MT/0.0, μ_2, 0, 0, NK, 0)CØNT NK = 1

(MAT, 6, MT/T, 0.0, LT, LF, NR, NP/E_{int}/$p(E, \mu_2)$)TAB1 LF = 1

(MAT, 6, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 6, MT/T, E_1, LT, 0, NR, NF/E'_{int}/$p(E_1 \rightarrow E', \mu_2)$)TAB1

(MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p(E_{NE} \rightarrow E', \mu_2)$)TAB1

<Subsection for $p(E \rightarrow E', \mu_3)$>

<Subsection for $p(E \rightarrow E', \mu_{NA})$>

(MAT, 6, MT/0.0, 0.0, 0, 0, 0, 0)SEND

Again T and LT refer to possible temperature dependence.

6.3. Procedures

All interpolation schemes used in this section should be linear-linear to ensure that the probability distributions will have the proper normalization everywhere. It is strongly recommended that an arbitrary tabulated distribution law (LF = 1) be used for secondary energy distribution for both LTT = 1 and 2.

7. FILE 7, THERMAL NEUTRON SCATTERING LAW DATA

7.1. General Description

File 7 contains inelastic neutron scattering (MT = 4) data for the thermal neutron energy range for moderating materials (E < 5 eV). The data in this file must be combined with that in Files 2 and 4 (MT = 2) to obtain the total scattering cross sections for certain materials.

Inelastic scattering is represented by the thermal neutron scattering law, $S(\alpha,\beta,T)$, and is defined (for a moderating molecule) by

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \to E',\mu,T) = \sum_{n=0}^{NS} \frac{M_n\, \sigma_{bn}}{4\pi\, T} \sqrt{\frac{E'}{E}}\; e^{-\beta/2}\; S_n(\alpha,\beta,T)\ ,$$

where there are (NS + 1) types of atoms in the molecule (i.e., for H_2O, NS = 1) and

$\underline{M_n}$ is the number of atoms of the n^{th} type in the molecule,

$\underline{T}$ is the moderator temperature (°K),

$\underline{E}$ is the incident neutron energy (eV),

$\underline{E'}$ is the secondary neutron energy (eV),

$\underline{\beta}$ is the energy transfer, $\beta = (E' - E)/kT$,

$\underline{\alpha}$ is the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$,

$\underline{A_n}$ is the mass of the n^{th} type atom, A_o is the mass of the principal scattering atom in the molecule,

$\underline{\sigma_{bn}}$ is the bound atom scattering cross section of the n^{th} type atom,

$$\sigma_{bn} = \sigma_{fn}\left(\frac{A_n + 1}{A_n}\right)^2$$

σ_{fn} is the free atom scattering cross section of the n^{th} type atom,

k is Boltzmann's constant, and

μ is the cosine of the scattering angle (in the lab system).

The data in File 7 for any particular material contain only the scattering law for the principal scatterer, $S_0(\alpha,\beta,T)$, i.e., the 0^{th} atom in the molecule. These data are given as an arbitrary tabulated function. The scattering properties for the other atom types (n = 1, 2, ..., NS) are represented by analytic functions. Note that the scattering properties of all atoms in the molecule may be represented by analytic functions. In this case there is no principal scattering atom.

The constants required for the scattering law data and the analytic representations for the nonprincipal scattering atoms are given in an array, B(N), N = 1, 2..., NI, where NI = 6*(NS + 1). Six constants are required for each atom type (one BCD card-image record). The first six elements pertain to the principal scattering atom, n = 0. The elements of the array B(N) are defined as

B(1) = $M_o\sigma_{fo}$, the total free atom cross section for the principal scattering atom. If B(1) = 0.0, there is no principal scattering atom and the scattering properties for this material are completely described by analytic functions for each atom type in this material.

B(2) = ε, the value of E/kT above which the static model of elastic scattering is adequate (total scattering properties may be obtained from MT = 2 as given in Files 2 and 4 of the appropriate materials).

B(3) = A_o, the ratio of the mass of the atom to that of the neutron that was used to compute α ($\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o\,kT$).

B(4) = E_{max}, the upper energy limit for the constant σ_{f0} (upper energy limit in which $S_0(\alpha,\beta,T)$ may be used).

B(5), not used.

B(6), not used.

The next six constants specify the analytic functions to be used in describing the scattering properties of the first non-principal scattering atom, (n = 1); i.e., for H_2O, this atom would be oxygen if the principal atom was hydrogen.

B(7) = a_1, a test indicating the type of analytic function used for this atom type.

a_1 = 1.0, use a free gas scattering law.

a_1 = 2.0, use a diffusive motion scattering law.

B(8) = $M_1\sigma_{f1}$, the total free atom cross section for this atom type.

B(9) = A_1, effective mass for this atom type.

B(10)= 0.0, B(10) is not used.

B(11)= 0.0, B(11) is not used.

B(12)= 0.0, B(12) is not used.

The next six constants, B(13) through B(18), are used to describe the second nonprincipal scattering atom (n = 2), if required. The constants are defined in the same way as for n = 1; e.g., B(13) is the same type of constant as B(7).

The scattering law is given by tabulating $S(\alpha, \beta)$ at a specific temperature (°K) or at a series of temperatures. Since scattering law data are generally given at more than one temperature, it is extremely important to understand the data formats for specifying temperature-dependent data (see Appendix F for details). The data are presented at given values of β. The β's are ordered by increasing values. For each value of β, pairs of α vs $S(\alpha,\beta)$ are given. (The data are given in this form only for the first temperature; see Appendix F for

the formats for temperature dependent data.) Two interpolation schemes are given to interpolate between values of β and α.

In certain cases a more accurate temperature representation may be obtained by replacing the value of the actual temperature, T, that is used in the definition of α and β with a constant, T_o (T_o = 0.0253 eV or the equivalent depending on the units of Boltzmann's constant). A flag (LAT) is given for each material to indicate which temperature has been used in generating the $S(\alpha,\beta)$ data.

7.2. Formats

There is only one section in File 7, but the format varies slightly, depending on whether temperature-dependent data are given. The following quantities are defined:

LAT is a flag indicating which temperature has been used to compute α and β.

LAT = 0, the actual temperature was used.

LAT = 1, the constant T_o = 0.0253 eV has been used.

NS is the number of non-principal scattering atom types. For most moderating materials there will be (NS + 1) types of atoms in the molecule ($NS \leq 3$).

NI is the total number of items in the B(N) list. NI = 6*(NS + 1).

B(N) is the list of constants. Definitions are given above (Section 7.1).

NR is the number of interpolation ranges for a particular parameter, either β or α.

NB is the total number of β values given.

NP is the number of α values given for each value of β for the first temperature described, NP is the number of pairs, α and $S(\alpha,\beta)$, given.

β_{int} and α_{int} are the interpolation schemes used (see Appendix E for interpolation formats).

The structure of a section is

(MAT, 7, MT/ZA, AWR, 0, LAT, 0, 0)HEAD

(MAT, 7, MT/0.0, 0.0, 0, 0, NI, NS/B(N))LIST

(MAT, 7, MT/0.0, 0.0, 0, 0, NR, NB/β_{int})TAB2

(MAT, 7, MT/T, β_1, LT, 0, NR, NP/α_{int}/S(α,β_1))TAB1

(MAT, 7, MT/T, β_2, LT, 0, NR, NP/α_{int}/S(α,β_2))TAB1

(MAT, 7, MT/T, β_{NB}, LT, 0, NR, NP/α_{int}/S(α,β_{NB}))TAB1

(MAT, 7, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

T and LT refer to possible temperature dependence. If the scattering law data are completely specified by analytic functions (no principal scattering atom type, as indicated by B(1) = 0), tabulated values of $S_0(\alpha,\beta)$ are omitted and the TAB2 and TAB1 records are not given.

7.3. Procedures

Any material may contain a File 7 to describe inelastic scattering cross sections for the thermal neutron energy range. Except for moderating materials, a free gas scattering law is generally adequate.

File 7 is the most important part of the cross section data for moderator type materials. Moderator materials should also contain a File 3, and, as a minimum, the radiative capture cross section (MT = 102) should be given (as well as any other type of absorptive cross sections). If there are elastic scattering (i.e., coherent scattering) contributions to the total scattering cross section,

then MT = 2 must be given in File 3. The data in File 3 shall at least cover the same energy range (constant, B(4)) as the scattering law data, $S_0(\alpha,\beta)$. The scattering law data should cover the energy range in which thermal inelastic effects are important. The recommended energy range is 10^{-5} eV to 3.0 eV; however, it may not be possible to obtain scattering law data for every moderating material for this energy range. The β mesh for $S(\alpha,\beta)$ should be selected in such a manner as to accurately represent the scattering properties of the material with a minimum of β points. The α mesh at which $S(\alpha,\beta)$ is given should be the same for each value of β and for each temperature.

Note that the differential scattering cross section, as given in the equation in Section 7.1, represents the cross section for the complete molecule. The differential scattering cross section for a single atom of any component can be obtained by replacing $N_n\alpha_{bn}$ by σ_{bn}.

8. FILE 8, RADIOACTIVE DECAY AND FISSION PRODUCT YIELD DATA

Information concerning the decay of the reaction products (any MT) is given in this file. In addition, fission product yield data (MT=454 and 459) for fissionable materials (see Section 8.2) and spontaneous radioactive decay data (MT=457) for the nucleus (see Section 8.3) are also included. See descriptions of File 9 and File 10 for information on isomeric state production from the various reactions. Since a reaction may result in more than one unstable end product, data for the most important unstable end product should be entered.

8.1. Radioactive Nuclide Production

For any isotope, sections may be given which specify that the end product (from a neutron interaction) is radioactive.

Information is given for any MT by specifying the end products in the reaction identified by their ZAP (ZA for the product), and how these end products decay. A section will contain only minimal information about the chain which follows that neutron reaction. One or more isomeric states of the target or the radioactive end product isotope will be described.

The following quantities are defined:

ZA is the designation of the original nuclide (ZA = 1000.0 * Z + A).

ZAP is the designation of the nuclide produced in the reaction (ZAP = 1000.0 * Z + A).

MATP is the material number for the reaction product (ZAP).

NS is the total number of states (LFS) of the radioactive reaction product for which decay are given.

LMF is the file number in which the multiplicity or cross section for this MT number will be found.

LIS is the state number (including ground and all levels) of the target (ZA).

LISØ is the isomeric state number of the target.

LFS is the level number (including ground and all levels) of the state of ZAP formed by the neutron interaction (to be given in ascending order).

ELFS is the excitation energy of the state of ZAP produced in the interaction (in eV above ground state).

NØ is the flag to allow the identification of a reaction product which is radioactive in its ground state and for which the cross section will not be found in File 9 or 10.

NØ = 0, cross section in File 9 or 10.
NØ = 1, cross section not in File 9 or 10.

ND is the number of branches into which the nuclide ZAP decays.

HL is the half-life of the nuclide ZAP in seconds.

ZAN is the Z and mass identifier of the next nuclide produced along the chain.

BR is the branching ratio for the production of that particular ZAN and level.

ER is the end-point energy of the particle or quantum emitted (this does not include the gamma energy, following beta decay, for example).

CT is the chain terminator which gives minimal information about the formation and decay of ZAN. The hundreths digits of CT designate the excited level in which ZAN is formed. The condition $1.0 \leq CT < 2.0$ indicates that the chain terminates with ZAN, possibly after one or more gamma decays. The condition $CT \geq 2.0$ indicates that

ZAN is unstable and decays further to other nuclides. For example, consider the nuclide (ZAP) formed via a neutron reaction (MT number) in a final state (LFS number); ZAP then decays to a level in ZAN; the level number is part of the CT indicator and includes nonisomeric states in the count. The following examples may help explain the use of CT:

CT = 1.00 ZAN was formed in the ground state which is stable.

CT = 1.06 ZAN was formed in the sixth excited state; the sixth state decayed to the ground state which is stable.

CT = 2.00 ZAN was formed in the ground state which is unstable. (No delayed gammas are associated with the formation and decay of this particular ZAN.) The next decay in the chain is specified under the RTYP.

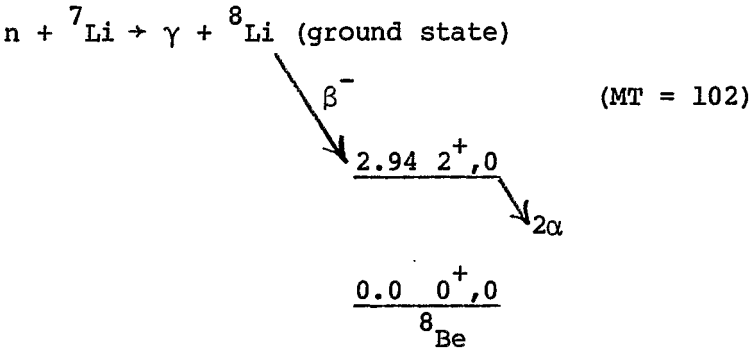
CT = 2.11 ZAN was formed in the 11th excited state but the chain does not terminate with that ZAN. The next decay in the chain is specified under the RTYP.

It is readily apparent from the above that CT = "1." indicates that the chain terminates with that particular ZAN and CT = "2." means that one or more decays are involved before stability is reached. Note, however, that stability can be reached instantaneously upon occasion with the emission of one or more light particles. The hundreths digits of the CT (06) indicate the state (sixth) in which ZAN is formed.

RTYP is the mode of decay using the same definitions specified in MT = 457 (see Section 8.3 of this report). As an example, consider MT = 102. Then RTYP = 1.44 would be interpreted as follows: The first two columns of the RTYP (1.) indicates β^- decay of ZAP; the third and fourth columns (44) indicate that the nucleus ZAN

(formed in the β^- decay) then immediately emits two α particles.

This example is represented by the following reaction:



For this example:

$ZA(^7Li) = 3.007 + 03$ $\quad LIS(^7Li) = 0$ $\quad LIS\emptyset(^7Li) = 0$

$ZAP(^8Li) = 3.008 + 03$ $\quad LFS(^8Li) = 0$

$ZAN(^8Be) = 4.008 + 03$ $\quad CT(^8Be) = 2.01$

Since ^8Be has a half-life of the order of compound-nucleus formation times, decay data for MT = 457 are not required, and the complete chain can easily be represented and read from the information given here.

8.1.1. Formats

The structure of each section always starts with a HEAD record and ends with a SEND record. If NØ=0, the section is divided into subsections, which contain data for a particular final state of the reaction product (LFS).

The number of subsections NS is given on the HEAD record for the section. Each subsection consists of a single LIST record. The subsections are ordered by increasing value of LFS.

The structure of a section is:

(MAT, 8, MT/ZA, AWR, LIS, LISØ, NS, NØ)HEAD

<subsection for LFS_1>

<subsection for LFS_2>

.
.
.

<subsection for LFS_{NS}>

(MAT, 8, 0/0.0, 0.0; 0, 0, 0, 0)SEND

The structure of a subsection is:

(MAT, 8, MT/ZAP, ELFS; LMF, LFS ; 6*ND, MATP/

HL(1), RTYP(1), ZAN(1), BR(1), ER(1), CT(1)

HL(2), RTYP(2), ZAN(2), BR(2), ER(2), CT(2)

.
.
.

HL(ND), RTYP(ND), ZAN(ND), BR(ND), ER(ND), CT(ND)) LIST

If NØ=1, then the reaction gives rise to a significant product which is radioactive in its ground state, and the evaluator wishes only to identify the radioactive product. The evaluator must supply MF=8, MT=457 data elsewhere to describe the decay of the product. It is understood that the cross section for producing the radioactive product is to be found in MF=3 for the given MT.

For NØ=1, the structure of a section is:

(MAT, 8, MT /ZA, AWR, LIS, LISO, NS, NØ=1)HEAD

(MAT, 8, MT /ZAP, ELFS=0., LMF=3, LFS=0, 6*ND=0, MATP)CØNT

(MAT, 8, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

8.1.2. Procedures

1. Data should be given for all unstable states of the reaction product nucleus for which cross sections are given in File 10 or multiplicities in File 9. No information of this type is allowed in evaluations for mixtures of elements, molecules, or elements with more than one naturally occurring isotope.

2. In order to provide more general usefulness as these files are being constructed, the following procedures are mandatory. For each reaction type (MT), File 9 (multiplicities) or File 10 (cross sections) must be constructed.

3. If the ENDF/B file also contains a complete evaluation of the neutron cross sections for the reaction product nucleus (ZAP, LIS), then the radioactive decay data for the evaluation of (ZAP, LIS) found in MF=8, MT=457 must be consistent with the decay data in this section.

8.2. Fission Product Yield Data (MT = 454 and MT = 459)

MT numbers 454 and 459 specify the incident neutron energy-dependent fission product yield data. A complete set of fission product yield data is given for a particular incident neutron energy. Data sets should be given at sufficient incident energies to completely specify yield data for the energy range given for the fission cross section (as determined from Files 2 or 3). These data are given by specifying fission product identifiers and fission product yields.

MT=454 is used for independent yields (YI) and MT=459 is used for cumulative yields (YC). The formats for MT=454 and MT=459 are identical. Independent yields (YI) are direct yields per fission prior to delayed neutron, beta, etc. decay. The sum of all independent yields is $\sim$2.0 for any particular incident neutron energy. Cumulative yields (YC) are specified for the same set of fission products. These account for all decay branches, including delayed neutrons.

The fission products are specified by giving an excited state designation (FPS) and a (charge, mass) identifier (ZAFP). Thus, fission product nuclides are given, not mass chains. More than one (Z,A) may be used to represent the yields for a particular mass chain.

The following quantities are defined

NFP is the number of fission product nuclide states to be specified at each incident energy point (this is actually the number of sets of fission product identifiers - fission product yields). (NFP $\leq$ 1250.)

ZAFP is the (Z,A) identifier for a particular fission product. (ZAFP = (1000.0* Z) + A).

FPS is the state designator (floating-point number) for the fission product nuclide (FPS = 0.0 means the ground state, FPS = 1.0 means the first excited state, etc.)

YI (MT=454) is the fractional (independent) yield for a particular fission product prior to particle decay.

DYI (MT=454) is the 1σ uncertainty in YI.

YC (MT=459) is the cumulative yield.

DYC (MT=459) is the 1σ uncertainty in YC.

$C_n(E_i)$ is the array of yield data for the i^{th} energy point. This array contains NFP sets of four parameters in the order ZAFP, FPS, YI, and DYI in MT=454 and ZAFP, FPS, YC, and DYC in MT=459.

NN is equal to 4*NFP, the number of items in the $C_n(E_i)$ array.

E_i is the incident neutron energy of the i^{th} point (eV).

LE is a test to determine whether energy-dependent fission product yields are given:

LE = 0 implies no energy-dependence (only one set of fission product yield data given);

LE > 0 means that (LE + 1) sets of fission product yield data are given at (LE + 1) incident neutron energies.

I_i is the interpolation scheme (see Appendix E) to be used between the E_{i-1} and E_i energy points.

8.2.1. Formats

The structure of a section always starts with a HEAD record and ends with a SEND record. Sets of fission product yield data are given for one or more incident neutron energies. The sets are ordered by increasing neutron energy. For a particular neutron energy the data are presented by giving four

parameters (ZAFP, FPS, YI, and DYI in MT=454 and ZAPF, FPS, YC, and DYC in MT 459) for each fission product state. The data are first ordered by increasing values of ZAFP. If more than one yield is given for the same (Z,A) the data are ordered by increasing value of the state designator (FPS).

The structure for a section is

(MAT, 8, MT/ZA, AWR, LE + 1, 0, 0, 0)HEAD

(MAT, 8, MT/E_1, 0.0, LE, 0, NN, NFP/$C_n(E_1)$)LIST

(MAT, 8, MT/E_2, 0.0, I , 0, NN, NFP/$C_n(E_2)$)LIST

(MAT, 8, MT/E_3, 0.0, I , 0, NN, NFP/$C_n(E_3)$)LIST

(MAT, 8, 0/0.0, 0.0, 0, 0, 0, 0)SEND,

where MT = 454 for independent yield data and MT = 459 for cumulative yield data. There are (LE + 1) LIST records.

8.2.2. Procedures

The data sets for fission product yields should be given over the same energy range as that in Files 2 and/or File 3 for the fission cross section. The yields are given as fractional values at each energy, and normally the independent yields will sum to $\sim$ 2.0.

This format provides for the yields (YI or YC) to each excited state (FPS) of the nuclide designated by ZAFP, and hence accommodates the many metastable fission products having direct fission yields. Data may be given for one or more fission product nuclide states to represent the yield for a particular mass chain. If independent yield data are given for more than one nuclide, the yield for the lowest Z (charge) nuclide state for a particular mass chain should be

the same as the cumulative fractional yield in MT=459, and all other independent yields for this same chain should be direct fractional yields.

The direct fission product yields are those prior to delayed neutron emission; for this reason, the summation of independent yields over the nuclides in each mass chain does not necessarily equal the isobaric chain yield. The cumulative yield for each nuclide (ZAFP, FPS) can be determined by use of the branching fractions in MT=457 or directly from MT=459.

Yields for the same fission product nuclides should be given at each energy point. This will facilitate interpolation of yield data between incident energy points. Also, a linear-linear interpolation scheme should be used.

8.3. Radioactive Decay Data (MT=457)

The spontaneous radioactive decay data are given in Section 457. This section is given for materials that are single nuclides in their ground state or an isomeric state. (An isomeric state is a "long lived" excited state of the nucleus.) The main purpose of MT=457 is to describe absolutely the energy spectra resulting from radioactive decay and give average parameters useful for applications such as decay heat, waste disposal, depletion and buildup studies, shielding, and fuel integrity. The information in this section can be divided into three parts:

a. General information about the material

ZA is the designation of the original (radioactive) nuclide (=1000* Z+A).

AWR is the ratio of the LIS state nuclide mass to that of neutron.

LIS is the state of the original nuclide (LIS=0, ground state, LIS=1, first excited state, etc.).

LISØ is the isomeric state number for original nuclide (LISØ=0, ground state; LISØ=1, first isomeric state; etc.).

$T_{\frac{1}{2}}$ is the half-life of the original nuclide (seconds).

$E_{"x"}$ is the average decay energy (eV) of "x" radiation for decay heat applications. The "β," "γ," and "α" energies are given in that order with space reserved for zero "β" or "γ" entries. See procedures for precise definitions of "β", "γ" and "α."

SPI is the spin of the nuclide in its LIS state. (SPI = - 77.777 = SPIN UNKNOWN)

PAR is the parity of the nuclide in its LIS state (±1.0).

b. Decay mode information - for each mode of decay:

NDK is the total number of decay modes given (cannot be zero).

RTYP is the mode of decay of the nuclide in its LIS state.

Decay modes defined:

RTYP	Mode of decay	
0.	γ	γ-ray (not used in 457)
1.	β^-	Beta decay
2.	e.c., (β^+)	Electron capture and/or positron emission
3.	IT	Isomeric transition (will in general be present only when the state being considered is an isomeric state)
4.	α	Alpha decay
5.	n	Neutron emission (not "delayed neutron decay" considered below)
6.	SF	Spontaneous fission
7.	p	Proton emission.
10.	-	Unknown origin.

Multiple particle decay is also allowed using any combination of the above RTYP variables as illustrated in the following examples:

RTYP	Mode of decay	
1.5	β^-,n	Beta decay followed by neutron emission ("delayed neutron decay")
1.4	β^-,α	Beta decay followed by alpha emission (N-16 decay)
2.4	β^+,α	Positron decay followed by alpha emission.

8.3. Radioactive Decay Data (MT=457)

The spontaneous radioactive decay data are given in Section 457. This section is given for materials that are single nuclides in their ground state or an isomeric state. (An isomeric state is a "long lived" excited state of the nucleus.) The main purpose of MT=457 is to describe absolutely the energy spectra resulting from radioactive decay and give average parameters useful for applications such as decay heat, waste disposal, depletion and buildup studies, shielding, and fuel integrity. The information in this section can be divided into three parts:

a. General information about the material

ZA is the designation of the original (radioactive) nuclide (=1000* Z+A).

AWR is the ratio of the LIS state nuclide mass to that of neutron.

LIS is the state of the original nuclide (LIS=0, ground state, LIS=1, first excited state, etc.).

LISØ is the isomeric state number for original nuclide (LISØ=0, ground state; LISØ=1, first isomeric state; etc.).

$T_{\frac{1}{2}}$ is the half-life of the original nuclide (seconds).

$E_{"x"}$ is the average decay energy (eV) of "x" radiation for decay heat applications. The "β," "γ," and "α" energies are given in that order with space reserved for zero "β" or "γ" entries. See procedures for precise definitions of "β", "γ" and "α."

SPI is the spin of the nuclide in its LIS state. (SPI = - 77.777 = SPIN UNKNOWN)

PAR is the parity of the nuclide in its LIS state (±1.0).

b. Decay mode information - for each mode of decay:

NDK is the total number of decay modes given (cannot be zero).

RTYP is the mode of decay of the nuclide in its LIS state.

Decay modes defined:

RTYP	Mode of decay	
0.	γ	γ-ray (not used in 457)
1.	β^-	Beta decay
2.	e.c., (β^+)	Electron capture and/or positron emission
3.	IT	Isomeric transition (will in general be present only when the state being considered is an isomeric state)
4.	α	Alpha decay
5.	n	Neutron emission (not "delayed neutron decay" considered below)
6.	SF	Spontaneous fission
7.	p	Proton emission.
10.	-	Unknown origin.

Multiple particle decay is also allowed using any combination of the above RTYP variables as illustrated in the following examples:

RTYP	Mode of decay	
1.5	β^-,n	Beta decay followed by neutron emission ("delayed neutron decay")
1.4	β^-,α	Beta decay followed by alpha emission (N-16 decay)
2.4	β^+,α	Positron decay followed by alpha emission.

RFS is the isomeric state flag for daughter nuclide. (RFS=0.0, ground state; RFS=1.0, first isomeric state; etc.)

Q is the total decay energy (eV) available in the corresponding decay process. (This is not necessarily the same as the maximum energy of the emitted radiation. In the case of an isomeric transition Q will be the energy of the isomeric state. For both β^+ and β^-, Q equals the energy corresponding to the mass difference between the initial and final atoms.)

BR is the fraction of the decay of the nuclide in its LIS state which proceeds by the corresponding decay mode. (e.g. If only β^- occurs and no isomeric states in the daughter nucleus are excited then BR = 1.0 for β^-decay).

c. Resulting radiation spectra

NSP is the total number of radiation types (STYP) for which spectral information is given (NSP may be zero).

STYP is the decay radiation type

Decay radiations defined:

STYP	Radiation type	
0.	γ	Gamma rays
1.	β^-	Beta rays
2.	e.c., (β^+)	Electron capture and/or positron emission
4.	α	Alpha particles
5.	n	Neutrons
6.	SF	Spontaneous fission fragments
7.	p	Protons
8.	e^-	"Discrete electrons"

Decay Radiations Defined (Cont'd)

STYP	Radiation type	
9.	x	X-rays and annihilation radiation (photons not arising as transitions between nuclear states)

ER is the energy (eV) of radiation produced (E_{γ}, $E_{\beta-}$, $E_{e.c.}$, etc.)

RI is the intensity of radiation produced (relative units)

RP is the spectrum of the continuum component of the radiation in units of probability/eV such that $\int RP(ER)dER = 1$

TYPE is the type of transition for beta and electron capture

Types Defined:

TYPE	Spectrum Definition
0.0	not required for STYP
1.0	allowed, nonunique
2.0	First-forbidden unique
3.0	Second-forbidden unique

RICC is the total internal conversion coefficient (STYP=0.0 only)

RICK is the K-shell internal conversion coefficient (STYP=0.0 only)

RICL is the L-shell internal conversion coefficient (STYP=0.0 only)

RIS is the internal pair formation coefficient (STYP=0.0)

positron intensity (STYP=2.0)

0.0 (Otherwise)

LCON* is the continuum spectrum flag

LCON = 0, no continuous spectrum given

LCON = 1, only continuous spectrum given

*Spontaneous $\bar{\nu}$: For RTYP = 6. and STYP = 5. LCON = 1 and FC = $\bar{\nu}$ and FD = 0.0

LCON = 2, both discrete and continuous spectra

LCON = 5, spectral information in file 5 format

NT is the number of entries given for each discrete energy (ER)

FC* is the continuum spectrum normalization factor (absolute intensity/relative intensity)

FD is the discrete spectrum normalization factor (absolute intensity/relative intensity)

NER is the total number of tabulated discrete energies for a given spectral type (STYP)

$\overline{ER}$** is the average decay energy of radiation produced

NR is the number of interpolation ranges for the continuum spectrum

NP is the number of points at which the distribution will be given

X_{int} is the interpolation scheme for the continuum spectrum

NK is the number of partial energy distributions when LCON = 5 is used.

Δ is the uncertainty in any quantity.

8.3.1. Formats

The structure of this section always starts with a HEAD record and ends with a SEND record. This section is divided into subsections as follows:

(MAT, 8,457/ ZA AWR LIS LISØ b NSP) HEAD

(MAT, 8,457/ $T_{½}$ $\Delta T_{½}$ b b 6 b /

$\overline{E}_{,\beta,}$ $\Delta\overline{E}_{,\beta,}$ $\overline{E}_{,\gamma,}$ $\Delta\overline{E}_{,\gamma,}$ $\overline{E}_{,\alpha,}$ $\overline{\Delta E}_{,\alpha,}$) LIST

*Spontaneous $\bar{\nu}$: For RTYP = 6. and STYP = 5. LCON = 5 or LCON = 1 and FC = $\bar{\nu}$ and FD = 0.0

**For STYP = 2, this is the average positron energy; for STYP = 4, this includes energy of recoil nucleus.

(MAT,8,457/ SPI PAR b b 6*NDK NDK /

$RTYP_1$ RFS_1 Q_1 ΔQ_1 BR_1 ΔBR_1

.
.
.

$RTYP_{NDK}$ RFS_{NDK} Q_{NDK} ΔQ_{NDK} BR_{NDK} ΔBR_{NDK}) LIST

Repeat NSP times

(MAT,8,457/ b STYP LCON b 6 NER / (omit if NSP=0)

FD ΔFD $\overline{ER}$ $\overline{\Delta ER}$ FC ΔFC) LIST

(MAT,8,457/ ER_1 ΔER_1 b b NT b / (omit if LCON=1or5)

$RTYP_1$ $TYPE_1$ RI_1 ΔRI_1 RIS_1 ΔRIS_1

$RICC_1$ $\Delta RICC_1$ $RICK_1$ $\Delta RICK_1$ $RICL_1$ $\Delta RICL_1$) LIST

ER_{NER} ΔER_{NER} b b NT b / (omit if LCON=5)

$RTYP_{NER}$ $TYPE_{NER}$ RI_{NER} ΔRI_{NER} ----) LIST

(MAT,8,457/ RTYP 0.0 b b NR NP / (omit if LCON=0or5)

X_{int} / ER_K RP_K) TAB1

(MAT,8,457/ ZA AWR ER ΔER NK 0) HEAD (omit if LCON ≠5)

<SUBSECTION for K = 1>

<SUBSECTION for K = NK>

where subsections are described in Chapter 5, "File 5, Energy Distributions of Secondary Neutrons" page 5.1.

(MAT,8,0/ b b b b b b) SEND.

8.3.2. Procedures

I. The initial state of the parent nucleus is designated by LISØ which equals 0 for the ground state and equals n for the nth isomeric state. Only isomeric states are included in the count of LISØ. (In other files isomeric and non-isomeric states may be included in the count of levels.)

II. The average decay energy $\overline{E}_{"x"}$ for decay heat application is given for three general radiation types, $\overline{E}_{"\beta"}$, $\overline{E}_{"\gamma"}$, and $\overline{E}_{"\alpha"}$. The sum of these three quantities is the total average (neutrino energies excluded) energy available per decay to the decay heat problem. The three quantities are more precisely defined as

$$\overline{E}_{"\beta"} = \overline{E}_{\text{elect.}} = \overline{E}_{\beta-} + \overline{E}_{\beta+} + \overline{E}_{e-} + \ldots$$

$$\overline{E}_{"\gamma"} = \overline{E}_{\text{phot.}} = \overline{E}_{\gamma} + \overline{E}_{\text{x-ray}} + \overline{E}_{\text{ann. rad.}} + \ldots$$

$$\overline{E}_{"\alpha"} = \overline{E}_{\text{h.p.}} = \overline{E}_{\alpha} + \overline{E}_{SF} + \overline{E}_{p} + \overline{E}_{n},$$

where $\overline{E}_{"\beta"}$ ($\overline{E}_{\text{elect.}}$) means the average energy of <u>all</u> "electron-related" radiation such as β^-, β^+, conversion-electrons, Auger, etc. The quantity $\overline{E}_{"\gamma"}$ ($\overline{E}_{\text{phot.}}$) means the average energy of <u>all</u> "electromagnetic" radiations such as gamma rays, x-rays, and annihilation radiation. The quantity $\overline{E}_{"\alpha"}$ is the average energy of <u>all</u> heavy charged particles and delayed neutrons.

$\overline{E}_{"\beta"}$, $\overline{E}_{"\gamma"}$, and $\overline{E}_{"\alpha"}$ must be specified in that order with space reserved for zero or unknown information. The average alpha energy, $\overline{E}_{\alpha}$, also includes the recoil energy but the alpha energy alone can be separated out by the usual $\frac{M_R}{M_R+M_\alpha}$ factor where M_R and M_α are the recoil nucleus and alpha masses respectively.

III. The symbol RTYP indicates the mode of decay as determined by the initial event. A nucleus undergoing beta decay to an excited state of the daughter

nucleus which subsequently decays by gamma emission is in the beta decay mode. RTYP = 0.0 is not allowed in MT = 457 (although used under 8.1).

An isomeric state of the daughter nuclide resulting from the decay of parent nuclides is designated by RFS following the procedures used for LISØ. Q represents the total energy available in the decay process and is equal to the energy difference available between the initial and final states (both of which may be isomeric). The branching ratio BR for each decay mode is given as a fraction and the sum over all decay modes must equal unity. Multiple particle emission is also allowed by using any combination of the RTYP variables. This will account for particle emission from nuclear states excited in the decay of the parent ("delayed-particle" emission) whose half-lives are too short to warrant separate entry in the file. It will also allow users and processing codes to identify the various intermediate states, without having to examine all the spectrum listings to determine radiation types. The multiple-particle RTYP should be constructed in the order in which the particles are emitted. (e.g. RTYP = 1.5 indicates β decay followed by neutron emission).

IV. The source-of-radiation should be specified for each spectral line or continuous spectra. The source of radiation is a floating point integer corresponding to the RTYP definitions. If the source-of-radiation is not known RTYP = 10. should be used.

V. The energy spectra should be specified if they are known and identified by STYP. Gamma spectra are described using STYP = 0.0. Relative intensities and errors in the relative intensity should be specified. Absolute normalization is made through multiplication by FC and FD. If absolute discrete spectra are given FD must equal unity. The radiation intensity should total the contributions from all decays leading to radiation within a particular decay type, STYP, having an energy $E_r \pm \Delta E_r$.

a. For gamma ray emission (STYP = 0.0), no other information is required if X, Auger electron, conversion electron, and pair formation intensities have not been calculated for these transitions. In this case NT = 6.

The amount of additional information depends upon the detail in which quantities were obtained for inclusion in STYP = 8. or 9. spectra, and the number of decay modes. (This detail will also be reflected in the uncertainties assigned in STYP = 8. or 9. spectra.) If only the total conversion electron emission is calculated, RICC and ΔRICC should be included and NT is specified as 8. If contributors from the individual K, L, and M shells are calculated the K and L shell conversion coefficients should be included and NT = 12. In the rare case (i.e. $^{16}_{7}N$), where internal pair formation is included the internal pair formation coefficient should be included along with the conversion coefficients as the quantity RIS and ΔRIS.

b. For electron capture (STYP = 2.) the quantity RIS is 0.0 provided $E_{e.c.} \leq 1.022$ MeV. If positron emission is energetically possible, RIS and ΔRIS must be specified (as $I_{\beta+}$ and $\Delta I_{\beta+}$).

c. The spectra should be ordered in increasing values of STYP, and discrete spectral data should be specified before continuous spectra.

d. For STYP = 6. (spontaneous fission neutrons) LCON and NER must be zero and $\overline{E}_{SF}$ and $\Delta\overline{E}_{SF}$ should be given.

VI. The specification of data uncertainties is an important quantity which is difficult to represent in a simple way. Although a one sigma variance is desired, a number should be entered that at least indicates qualitatively how well the parameter is known.

For STYP = 8. and 9., ΔE will reflect the detail in which these values were derived. For example, if only the total conversion electron emission has been calculated, ΔE would be the spread between K-conversion and M-conversion electron energies. If a very detailed calculation has been made, ΔE would reflect the uncertainties in the electron binding energy and the transition energy.

VII. LCON = 5 should be used whenever data is given in MF = 5 as well as MF = 8, MT = 457 so that consistency between the two files is ensured.

VIII. Every effort should be made to determine the spin and parity of the original nucleus, either by experimental evidence or by strong theoretical arguments. If the spin cannot be determined, it should be reported as -777.77; if the parity cannot be determined it should be reported as zero.

9. FILE 9, MULTIPLICITIES FOR PRODUCTION OF RADIOACTIVE NUCLIDES

9.1. General Description

Neutron cross sections can be obtained by the use of multiplicities in File 9. The multiplicity represents that fraction of the cross section which produces the LFS state in the daughter nucleus. File 9 must be employed for those reactions which are represented by resonance parameters in File 2.

The multiplicities are given as a function of energy, E, where E is the incident neutron energy (in eV) in the laboratory system. They are given as energy-multiplicity pairs. An interpolation scheme must be given to specify the energy variation of the data for neutron energies between a given energy point and the next higher energy point.

File 9 is divided into sections, each section containing data for a particular reaction type (MT number). The sections are ordered by increasing MT number. Within a section for a given MT are subsections for different final states of the daughter product (LFS). File 9 is only allowed for evaluations which represent data for single nuclides.

9.2. Formats

File 9 is made up of sections where each section gives the neutron multiplicity for a particular reaction type (MT number). Each section always starts with a HEAD record and ends with a SEND record. For File 9, the following quantities are defined:

LIS is an indicator to specify the level number of the target.

LFS is an indicator to specify the level number of the nuclide (ZAP) (as defined in MF=8) produced in the neutron reaction (MT number).

LFS = 0 the final state is the ground state.

LFS = 1 the final state is the first excited state.

LFS = 2 the final state is the second excited state.

.
.
.

LFS = 98 an unspecified range of final states.

Q is the reaction Q-value (eV).

NS is the number of final states for each MT for which multiplicities are to be given.

NR is the number of energy ranges. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range. (For details, see Section 0.4.3.)

Y(E) is the multiplicity for a particular reaction type at incident energy E(eV). Data are given for energy-multiplicity pairs.

The structure of a section is:

(MAT, 9, MT/ZA, AWR, LIS, 0, NS, 0) HEAD

<NS subsections, one for each value of LFS>

(MAT, 9, 0/0.0, 0,0, 0, 0, 0, 0) SEND

The structure of a subsection is:

(MAT, 9, MT/0.0, Q, 0, LFS, NR, NP/E_{int}/Y(E))TAB1

9.3. Procedures

Multiplicities must be given in File 9 for those reactions described in MF=8 which have LMF=9 in the LIST record of the subsection for that particular MT number and value of LFS. The multiplicities in File 9 describe the fraction of the cross section which produces the LFS state in the daughter nucleus. For

a reaction represented by resonance parameters in File 2, File 10 cannot be used; only multiplicities in File 9 are allowed.

The data in File 9 must cover the entire energy range for each reaction in File 3 from threshold to 20 MeV. That is, multiplicities cannot be used over a portion of the incident neutron energy range with cross sections covering another portion. For negative Q reactions, the first energy point should be at threshold given in File 3. If a subsection Q is not equal to the Q in File 3, the multiplicity should be given as zero up to the energy point corresponding to the threshold of the subsection.

The set of points or energy mesh used for the total cross section in File 3 must include the union of all energy meshes in File 9 for each MT number. Although 5000 incident energy points are allowed for the total cross section, every attempt should be made to minimize the number of points in File 9.

The multiplicities in File 9 should be equal to or less than unity since the cross sections to be generated must be equal to or less than the cross sections in File 3 for each MT number.

In summary, the proper procedure would be to not enter data in MF=8 and given MT <u>until</u> the File 9 multiplicities (or File 10 cross sections) are added to the evaluations. That is, every MT number (except MT=454, 457, or 459) in MF=8 with LMF=9 as an indicator in the LIST record of the subsection for that particular MT and value of LFS must have the corresponding multiplicities in File 9.

10. FILE 10, CROSS SECTIONS FOR PRODUCTION OF RADIOACTIVE NUCLIDES

10.1. General Description

Neutron activation cross sections (such as the (n,p) and (n,2n) cross sections) and cross sections for a particular state of a radioactive target are given in File 10. These cross sections are given as a function of energy, E, where E is the incident neutron energy (in eV) in the laboratory system. They are given as energy--cross-section pairs. An interpolation scheme must specify the energy variation of the data for neutron energies between a given energy point and the next higher energy point.

File 10 is divided into sections, each section containing the data for a particular reaction type (MT number). The sections are ordered by increasing MT number. Within a section for a given MT are subsections for different final states (LFS) of the daughter product nucleus. File 10 is allowed only for evaluations which represent the data for single isotopes.

10.2. Formats

File 10 is made up of sections where each section gives the neutron cross section for a particular reaction type (MT number). Each section always starts with a HEAD record and ends with a SEND record. For File 10, the following quantities are defined:

LIS is an indicator to specify the level number of the target.

LFS is an indicator to specify the level number of the nuclide (ZAP) produced in the neutron reaction (MT) number.

LFS = 0 the final state is the ground state.

LFS = 1 the final state is the first excited state.

LFS = 2 the final state is the second excited state.

.
.
.

LFS = 98 an unspecified range of final states.

Q is the reaction Q-value (eV).

NS is the number of final states for each MT for which cross sections are to be given.

NR is the number of energy ranges. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range. For details, see Section 0.4.3.).

$\sigma(E)$ is the cross section in barns for a particular reaction type at incident energy E (eV). Data are given for energy, cross-section pairs.

The structure of a section is:

(MAT, 10, MT/ZA, AWR, LIS, 0, NS, 0) HEAD

<NS subsections one for each LFS>

(MAT, 10, 0/0.0, 0.0, 0, 0, 0, 0) SEND

The structure of a subsection is:

(MAT, 10, MT/0.0, Q, 0, LFS, NR, NP/E_{int}/$\sigma(E)$) TAB1

10.3. Procedures

Isomer production cross sections must be given in File 10 for those reactions described in MF=8 which have LMF=10 in the LIST record of the subsection for that particular MT number and value of LFS. The data in File 10

are the cross sections for the production of a final state (LFS) of the daughter product nucleus. For a reaction represented by resonance parameters in File 2, File 10 cannot be used; only multiplicities in File 9 are allowed.

The data in File 10 must cover the entire energy range for each reaction from the threshold of the subsection in File 10 up to 20 MeV. That is, cross sections cannot be used over a portion of the incident neutron energy range with multiplicities covering another portion. For negative Q reactions, the first energy point should be at the threshold of the subsection in File 10 and the cross section at this point must be zero.

The set of points or energy mesh used for the total cross section in File 3 must be the union of all energy meshes in File 10 for each MT number. Although 5000 incident energy points are allowed for the total cross section, every attempt should be made to minimize the number of points in File 10.

Using the ^{93}Nb(n,2n) ^{92}Nb cross section as an example, only the cross section for the production of the 10.16 day isomer in ^{92}Nb would appear under MT=16 with LIS=0 and LFS=1 in File 10. The sum of all partial cross sections for the (n,2n) reaction would still be found in File 3 under MT=16 (note that this is the only (n,2n) cross section required for neutron transport calculations). It should be noted, however, in this particular case, that the evaluator would have the choice of using energy-dependent multiplicities in File 9 instead of cross sections in File 10.

The cross sections which appear in File 10 are redundant; that is, they should not be included in the check sum for the total cross section. The cross sections in File 10 must be equal to or less than the cross sections for that MT number which appear in File 3.

In summary, the proper procedure would be to not enter data in MF=8 and given MT until the File 10 cross sections (or File 9 multiplicities) are added

to the evaluations. That is, every MT number (except MT=454, 457, or 459) with LMF=10 as an indicator in the LIST record of the subsection for that particular MT and value of LFS must have the corresponding cross sections in File 10.

11. GENERAL COMMENTS OF PHOTON PRODUCTION

Photon production data are divided into five distinct files.

File	Description
12	Multiplicities and transition probability arrays
13	Photon production cross sections
14	Photon angular distributions
15	Continuous photon energy spectra
16	Photon energy-angle distributions

With the exception of File 12, all the files are closely analogous to the corresponding neutron data files with the same number (modulo 10). The purpose of File 12 is to provide additional methods for representing the energy dependence of photon production cross sections. The allowed reaction type (MT) numbers are the same as those assigned for neutron reactions, Files 1 through 7. However, they may have somewhat different meanings for photon production that require additional explanation in some cases:

(1) MT = 3 should be used in Files 12 through 16 to represent composite cross sections, that is, photon production cross sections from more than one reaction type that have been lumped together.

(2) There is no apparent reason to have redundant or derived data for the photon production files, as is the case for the neutron files, i.e., MT = 3, 4, etc. Therefore, to avoid confusion, the join of all sections of Files 12 and 13 should represent the photon production, with each section being disjoint from all others.

(3) Let us consider how one might represent the inelastic γ-ray production data. The differential cross section for producing a γ-ray of energy E_γ resulting from the excitation of the m_o^{th} level of the residual nucleus and the subsequent transition between two definite lev-

els ($j \rightarrow i$), including the effects of cascading from the $m_o - j$ levels higher than j, is

$$\frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) = \delta(E_\gamma - \varepsilon_j + \varepsilon_i) A_{j,i} \sigma_{m_o}(E) \prod_{\ell=j}^{m_o - j} \sum_{m_\ell = j}^{m_{\ell-1}-1} TP_{m_{\ell-1}, m_\ell} \qquad (1)$$

where

$\sigma_{m_o}(E)$ = neutron cross sections for exciting the m_o^{th} level with neutron energy E,

$\delta(E_\gamma - \varepsilon_j + \varepsilon_i)$ = delta function with ε_j, ε_i being energy levels of the residual nucleus,

$TP_{k,\ell}$ = probability of the residual nucleus having a transition to the ℓ^{th} level given that it was initially in the excited state corresponding to the k^{th} level, and

$A_{k,\ell}$ = probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the k^{th} to the ℓ^{th} level.

We are at once beset by the problem that no clear choice of ENDF representation in terms of section number is possible. The data may naturally be identified with both the m_o^{th} level and the j^{th} level. To avoid this problem, we can sum Eq. (1) over m_o:

$$\frac{d\sigma_1}{dE_\gamma}(E_\gamma, E, i, j) = \sum_{m_o=j}^{N} \frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) , \qquad (2)$$

where N is the highest level that can be excited by a neutron of incident energy E (i.e., $\varepsilon_N \leq \frac{AWR}{AWR + 1} E$). This gives a <u>de-excitation</u> cross section that can single out a definite γ-ray transition and has the advantage when experimental data are to be represented. The de-excitation cross section is identified with the j^{th} level. Alternatively, we can sum Eq. (1) over i and j:

$$\frac{d\sigma_2}{dE_\gamma}(E_\gamma,E,m_o) = \sum_{j=1}^{m_o} \sum_{i=0}^{j-1} \frac{d\sigma}{dE_\gamma}(E_\gamma,E,m_o,i,j) \ . \tag{3}$$

This gives an <u>excitation</u> cross section that can single out a definite excited state and has the advantage when calculated data are to be represented. The excitation cross section is identified with the m_o^{th} level. If Eq. (2) is summed over i and j, or if Eq. (3) is summed over m_o, then

$$\frac{d\sigma}{dE_\gamma}(E_\gamma,E) = \sum_{m_o=1}^{N} \frac{d\sigma_2}{dE_\gamma}(E_\gamma,E,m_o) \equiv \tag{4}$$

$$\sum_{j=1}^{N} \sum_{i=0}^{j-1} \frac{d\sigma_1}{dE_\gamma}(E_\gamma,E,i,j) \ .$$

This gives a cross section for all possible excitations and transitions and thus corresponds to the total inelastic neutron cross section for discrete levels.

It is recommended that MT = 4 be used for the data represented by Eq. (4), as well as for the continuum. If, however, it is expedient or useful to use MT = 51 through 91, then one must use either the de-excitation cross sections of Eq. (2) or the excitation cross sections of Eq. (3), but not both. A restriction is imposed if the transition probability array option is used and if the entire neutron energy range is not covered by the known transition probabilities. Then, for MT = 51 through 90 in File 12 to be used for the remaining neutron energy range, a representation by excitation multiplicities must be used.

The integrated cross sections of File 13 are obtained by integrating Eqs. (1) through (4) over E_γ.

(4) The remarks in Item (3) apply for discrete rays from (n,pγ), (n,dγ), (n,tγ), (n,^{3}Heγ), and (n,αγ) reactions, and the use of MT = 103, 104, 105, 106, and 107 is recommended for these cases.

12. FILE 12: PHOTON PRODUCTION MULTIPLICITIES AND TRANSITION PROBABILITY ARRAYS

File 12 can be used to represent the neutron energy dependence of photon production cross sections by means of either multiplicities or transition probability arrays. Both methods rely upon processing codes that use neutron cross sections from File 2 and/or File 3 to generate absolute photon production cross sections.

Multiplicities can be used to represent the cross sections of discrete photons and/or the integrated cross sections of continuous photon spectra. The MT numbers in File 12 designate the particular neutron cross sections (File 2 and/or File 3) to which the multiplicities are referred. The use of multiplicites is the recommended method of presenting (n,γ) capture γ-ray cross sections, provided, of course, that the (n,γ) cross section is adequately represented in File 2 and/or File 3.

For well-established level decay schemes, the use of transition probability arrays offers a concise method for presenting $(n,x\gamma)$ information. With this method, the actual decay scheme of the residual nucleus for a particular reaction (defined by MT number) is entered in File 12. This information can then be used by a processing code together with discrete level excitation cross sections from File 3 to calculate discrete γ-ray production cross sections. This option cannot be used to represent the integrals of continuous photon spectra.

12.1. Formats

Each section of File 12 gives information for a particular reaction type (MT number), either as multiplicities (LØ = 1) or as transition probability

arrays (LØ = 2). Each section always starts with a HEAD record and ends with a SEND record.

12.1.1. Option 1 (LØ = 1): Multiplicities

The neutron energy dependence of photon production cross sections is represented by tabulating a set of neutron energy and multiplicity pairs $(E, y_k(E))$ for each discrete photon and for the photon energy continuum.* The subscript k designates a particular discrete photon or a photon continuum, and the total number of such sets is represented by NK.

The multiplicity or yield $y_k(E)$ is defined by

$$y_k(E) = \frac{\sigma_k^\gamma(E)}{\sigma(E)} \quad \text{(photons)}$$

where E designates neutron energy and $\sigma(E)$ is the neutron cross section in File 2 and/or File 3 to which the multiplicity is referred (by the MT number). For discrete photons, $\sigma_k^\gamma(E)$ is the photon production cross section for the discrete photon designated by k. For photon continua, $\sigma_k^\gamma(E)$ is the cross section for the photon continuum integrated over photon energy. In the continuum case,

$$y_k(E) = \frac{\sigma_k^\gamma(E)}{\sigma(E)} = \frac{\int_0^\infty \frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)\, dE_\gamma}{\sigma(E)}$$

$$= \frac{\int \sigma(E)\ y_k(E_\gamma \leftarrow E)\ dE_\gamma}{\sigma(E)} = \int_0^{E_\gamma^{max}} y_k(E_\gamma \leftarrow E)\ dE_\gamma$$

*There should be no more than one energy continuum for each MT number used. If the decomposition of a continuum into several parts is desired, this can be accomplished in File 15.

where E_γ designates photon energy (eV),

$$\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)$$

is the absolute photon energy distribution in barns/eV, and $y_k(E_\gamma \leftarrow E)$ is the relative energy distribution in photons/eV. The quantity $y_k(E_\gamma \leftarrow E)$ can be broken down further as

$$y_k(E_\gamma \leftarrow E) = y_k(E)\ f_k(E_\gamma \leftarrow E)\ ,$$

which results in the requirement that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\, dE_\gamma = 1.$$

Any time a continuum representation is used for a given MT number in either File 12 or 13, then the normalized energy distribution $f_k(E_\gamma \leftarrow E)$ must be given in File 15 under the same MT number.

As a check quantity, the total yield

$$Y(E) = \sum_{k=1}^{NK} y_k(E) \quad \text{(photons)}$$

is also tabulated for each MT number if NK > 1.

The structure of a section for LØ = 1 is

(MAT, 12, MT/ZA, AWR; LØ=1, b; NK, b)HEAD

(MAT, 12, MT/ b, b; b, b; NR, NP/E_{int}/Y(E))TAB1*

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

(MAT, 12, 0/ b, b; b, b; b, b)SEND ,

and the structure of each subsection is

(MAT, 12, MT/EG_k, ES_k; LP, LF; NR, NP/E_{int}/y_k(E))TAB1 ,

where

NK the number of discrete photons including the continuum.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

EG_k the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k \equiv 0.0$ should be used.

LP indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

LP = 2, for primary photons where the photon energy EG'_k is given by

$$EG'_k = EG_k + \frac{AWR}{AWR+1} E_n .$$

LF the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

= 2, a discrete photon energy.

12.1.2. Option 2 (LØ = 2): Transition Probability Arrays

With this option, the only data required are the level energies, de-excitation transition probabilities, and (where necessary) conditional photon emission probabilities. Given this information, the photon energies and their multiplicities can readily be calculated. Photon production cross sections can then be computed for any given level from the excitation cross section in File 3, along with the transition probability array. Similarly, multiplicities and photon production cross sections can be constructed for the total cascade. For any given level, the transition and photon emission probability data given in the section are for photons originating at that level only; any further cascading is determined from the data for the lower levels.

Now define the following variables.

LG = 1, simple case (all transitions are γ emission).

= 2, complex case (internal conversion or other competing processes occur).

NS number of levels below the present one, including the ground state. (The present level is also uniquely defined by the MT number and by its energy level.)

NT number of transitions for which data are given in a list to follow (i.e., number of nonzero transition probabilities), NT $\leq$ NS.

ES_i energy of the ith level, i = 0,1,2... NS. ($ES_0 \equiv 0.0$, the ground state.)

TP_i $TP_{NS,i}$, the probability of a direct transition from level NS to level i, i = 0,1,2... (NS-1).

GP_i $GP_{NS,i}$, the probability that, given a transition from level NS to level i, the transition is a photon transition (i.e., the conditional probability of photon emission).

B_i an array of NT doublets or triplets depending on LG value.

Note that each level can be identified by its NS number. Then the energy of a photon from a transition to level i is given by $E_\gamma = ES_{NS} - ES_i$, and its multiplicity is given by $y(E_\gamma \leftarrow E) = (TP_i)\ (GP_i)$. It is implicitly assumed that the transition probability array is independent of incident neutron energy.

The structure of a section for LØ = 2 is

(MAT, 12, MT/ ZA, AWR; LØ=2, LG; NS, b)HEAD.

(MAT, 12, MT/ES_{NS}, b; LP, b; (LG+1)*NT, NT/B_i)LIST.

(MAT, 12 0/ b, b; b, b; b, b)SEND.

If LG = 1, the array B_i consists of NT doublets (ES_i,TP_i); if LG = 2, it consists of NT triplets (ES_i,TP_i,GP_i). Here the subscript i is a running index over the levels below the level for which the transition probability array is being given (i.e., below level NS). The doublets or triplets are given in decreasing magnitude of energy ES_i.

12.2. File 12 Procedures

1. Under Option 1, the subsections are given in decreasing magnitude of EG_k.

2. Under Option 1, the convention is that the subsection for the continuum photons, if present, is last. In this case, the last value of EG_k (EG_{NK}) is set equal to 0.0, and logical consistency with Procedure 1 is maintained.

3. Under Option 1, the values of EG_k should be consistent to within four significant figures with the corresponding EG_k values for the File 14 photon angular distributions. This allows processing and "physics" checking codes to match photon yields with the corresponding angular distributions.

4. Under Option 1, ES_k is the energy of the level from which the photon originates. If ES_k is unknown or not meaningful (as for the continuous photon spectrum), the value 0.0 should be entered.

5. If capture and fission resonance parameters are given in File 2, photon production for these reactions should be given by using Option 1 of File 12, instead of using photon production cross sections in File 13. This is due to the voluminous data required to represent the resonance structure in File 13 and the difficulty of calculating multigroup photon production matrices from such data.

6. Under Option 1, the total yield table, Y(E), should exactly span the same energy range as the combined energy range of all the $y_k(E)$. Within that range,

$$Y(E) = \sum_{k=1}^{NK} y_k(E)$$

should hold within four significant figures.

7. The excitation cross sections for all the levels appearing in the transition probability arrays must, of course, be given in File 3.

8. The join of all sections, regardless of the option used, should represent the photon production data, with no redundancy. For example, MT = 4 cannot include any photons given elsewhere under MT = 51 through 91. Likewise, there can be no redundancy between Files 12 and 13.

9. If only one energy distribution is given under Option 1 (NK = 1), the TAB1 record for the Y(E) table is deleted to avoid repetitive entries.

10. Data should not be given in File 12 for reaction types that do not appear in Files 2 and/or 3.

11. Under Option 2, the level energies, ES_i, in the transition probability arrays are given in decreasing magnitude.

12. The MT numbers for which transition probability data are given should be for consecutive levels, beginning at the first level, with no embedded levels omitted.

13. The energies of photons arising from level transitions should be consistent within four significant figures with the corresponding EG_k values in File 14. Therefore, care must be taken to specify level energies to the appropriate number of significant figures.

14. Under Option 2, the sum of the transition probabilities (TP_i) over i should equal 1.0000 (that is, should be unity to within five significant figures).

15. The limit on the number of energy points in any tabulations of Y(E) or $y_k(E)$ is 1000. This is an upper limit that will rarely be approached in practice because yields are normally smoothly varying functions of incident neutron energy.

16. The limit on the number of interpolation regions is 10.

17. Tabulations of nonthreshold data should normally cover at least the energy range 10^{-5} eV $\leq$ E $\leq$ 2 x 10^{7} eV, where practical. Threshold data should be given from threshold energy up to 2 x 10^{7} eV, where practical.

18. Transition Probability Arrays for (n,n'γ) photons.

a. The use of transition probability arrays (File 12, LØ = 2) is a convenient way to represent a portion of the γ-rays produced by de-excitation of discrete levels populated by (n,n') and other reactions.

b. Several conditions must be met before this representation can be used. Level excitation cross sections (given in File 3 as MT = 51,...) must be given from threshold energies up to the same maximum energy (no exceptions). Decay properties of all n levels must be known. The information given in File 12 must be consistent with data given in File 3.

c. Usually, not all the conditions can be met. Part of the problem is the recommendation that level excitation cross sections for the first few levels be given for neutron energies up to 20 MeV. It is seldom that all level data can be given for neutron energies up to 15 MeV.

13. FILE 13: PHOTON PRODUCTION CROSS SECTIONS

The purpose of File 13 is the same as that of File 12, namely, it can be used to represent the neutron and photon energy dependence of photon production cross sections. In File 13, however, absolute cross sections in barns are tabulated, and there is no need to refer to the neutron files.

13.1. Formats

As in File 12, each section in File 13 gives information for a particular reaction type (MT number). Each section always starts with a HEAD record and ends with a SEND record.

The representation of the energy dependence of the cross sections is accomplished by tabulating a set of neutron energy-cross section pairs $\left(E,\sigma_k^\gamma(E)\right)$ for each discrete photon and for the photon energy continuum. The subscript k designates a particular discrete photon or the photon continuum, and the total number of such sets is NK. For discrete photons, $\sigma_k^\gamma(E)$ is the photon production cross section (b) for the photon designated by k. For the photon continuum, $\sigma_k^\gamma(E)$ is the integrated (over photon energy) cross section for the photon continuum* designated by k. In the continuum case,

$$\sigma_k^\gamma(E) = \int_0^{E_\gamma^{max}} \frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)\, dE_\gamma \qquad \text{(b)}$$

*There should be no more than one energy continuum for each MT number used. If the decomposition of a continuum into several parts is desired, this can be accomplished in File 15.

where E_γ designates photon energy (eV), and $\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)$ is the absolute photon energy distribution in b/eV. The energy distribution can be further broken down as

$$\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E) = \sigma_k^\gamma(E)\ f_k(E_\gamma \leftarrow E) ,$$

which obviously requires that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\ dE_\gamma = 1 .$$

Any time a continuum representation is used for a given MT number in File 13, the normalized energy distribution, $f_k(E_\gamma \leftarrow E)$, must be given in File 15 under the same MT number.

As a check quantity, the total photon production cross section,

$$\sigma_{TOT}^\gamma(E) = \sum_{k=1}^{NK} \sigma_k^\gamma(E) \qquad \text{(barns)} ,$$

is also tabulated for each MT number, unless only one subsection is present (i.e., NK = 1).

The structure of a section in File 13 is

(MAT, 13, MT/ZA, AWR; b, b; NK, b)HEAD

(MAT, 13, MT/ b, b; b, b; NR, NP/$E_{int}/\sigma^{\gamma}_{TOT}(E)$)TAB1*

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

(MAT, 13, 0/ b, b; b, b; b, b)SEND

and the structure of each subsection is

(MAT, 13, MT/EG_k, ES_k; LP, LF; NR, NP/$E_{int}/\sigma^{\gamma}_k(E)$)TAB1 ,

where

NK the number of discrete photons including the continuum.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

EG_k the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k = 0.0$ should be used.

LP Indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons, where the photon energy is given by

$$EG_k + \frac{AWR}{AWR + 1} E_n \cdot$$

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

LF the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

LF = 2, a discrete photon energy.

13.2. File 13 Procedures

1. The subsections are given in decreasing magnitude of EG_k.
2. The convention is that the subsection for the continuum photons, if present, is last. In this case, $EG_{NK} \equiv 0.0$.
3. The values of EG_k should be consistent to within four significant figures with the corresponding EG_k values in File 14.
4. ES_k is the energy of the level from which the photon originates, if known. Otherwise $ES_k \equiv 0.0$.
5. If capture and fission resonance parameters are given in File 2, the corresponding photon production should be given by using Option 1 of File 12, instead of using photon production cross sections.
6. The total photon production cross section table, $\sigma^{\gamma}_{TOT}(E)$, should exactly span the same energy range as the combined energy range of all the $\sigma^{\gamma}_{k}(E)$. Within that range,

$$\sigma^{\gamma}_{TOT}(E) = \sum_{k=1}^{NK} \sigma^{\gamma}_{k}(E)$$

should hold within four significant figures. If only one energy distribution is given, either discrete or continuous (NK = 1), the TAB1 record for the $\sigma^{\gamma}_{TOT}(E)$ is deleted.

7. The join of all sections in Files 12 and 13 combined should represent the photon production data with no redundancy. For example, MT = 4 cannot include any photons given elsewhere under MT = 51 through 91.

8. The limit on the number of energy points in a tabulation for any photon production subsection is 1000. This is an upper limit; in practice, the minimum number of points possible should be used. If there is extensive structure, the use of File 12 should be seriously considered, because yields are normally much smoother functions of incident neutron energy than cross sections.

9. The limit on the number of interpolation regions is 10.

10. Tabulations of nonthreshold data should normally cover at least the energy range 10^{-5} eV $\leq E \leq 2 \times 10^{7}$ eV, where practical. Threshold data should be given from threshold energy up to 2×10^{7} eV, where practical.

13.3. File 13 Preferred Representations

1. The recommended representation for (n,n'γ) reactions is photon production cross sections (File 13) using MT = 4. All discrete and continuum γ rays are given in a series of subsections.

2. Photon production cross sections resemble the frequently measured or reported results.

3. The use of MT = 4 eliminates confusion about whether the data represent an excitation or de-excitation cross section.

4. If for any reason MT = 51, 52 ... is used, it is understood that these data represent de-excitation and not excitation cross sections (see 3 above). MT = 51, 52, ... in File 3, of course, means excitation cross sections.

5. Combined use of MT = 4 and MT = 51, 52, ... is not allowed.

6. Above a certain energy point it probably will not be possible to separate the various components of the total γ production cross section. When this happens, it is preferred that the data be given as MT = 3.

7. All other reactions. Data for other reactions should be given as photon production cross sections (File 13) using the appropriate MT numbers. The same general rules outlined above should be used.

14. FILE 14: PHOTON ANGULAR DISTRIBUTIONS

The purpose of File 14 is to provide a means for representing the angular distributions of secondary photons produced in neutron interactions. Angular distributions should be given for each discrete photon and photon continuum appearing in Files 12 and 13, even if the distributions are isotropic.

The structure of File 14 is, with the exception of isotropic flag (LI), closely analogous to that of File 4. Angular distributions for a specific reaction type (MT number) are given for a series of incident neutron energies in order of increasing neutron energy. The energy range covered should be the same as that for the data given under the corresponding reaction type in File 12 or File 13. The data are given in ascending order of MT number.

The angular distributions are expressed as normalized probability distributions, that is,

$$\int_{-1}^{1} p_k(\mu,E)\,d\mu = 1 \ ,$$

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as a Legendre series expansion,

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} \quad (\underline{\Omega},E)$$

$$= \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} a_\ell^k(E) \; P_\ell(\mu) \; ,$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^\gamma(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

ℓ = order of the Legendre polynomial.

$\frac{d\sigma_k^\gamma}{d\Omega}$ = differential photon production cross section in barns/steradian.

$a_\ell^k(E)$ = the ℓth Legendre coefficient associated with the discrete photon or photon continuum specified by k. (It is understood that $a_o^k(E) \equiv 1.0$.)

$$a_\ell^k(E) = \int_{-1}^{1} P_k(\mu,E) \; P_\ell(\mu) d\mu \; .$$

Angular distributions may be given in File 14 by tabulating as a function of incident neutron energy either the normalized probability distribution function, $p_k(\mu,E)$, or the Legendre polynomial expansion coefficients, $a_\ell^k(E)$. Provision is made in the format for simple flags to denote isotropic angular distributions, either for a block of individual photons within a reaction type or for all photons within a reaction type taken as a group.

Note that File 14 assumes separability of the photon energy and angular distributions for the continuum spectrum. If this is not the case, File 16 (analogous to File 6) must be used instead of Files 14 and 15. (Since File 14 implicitly specifies an energy-angle distribution for discrete photons, File 16 is required only for the continuum spectrum.

14.1. Formats

As usual, sections are ordered by increasing reaction type (MT) numbers. The following definitions are required.

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

= 1, distribution is isotropic for all photons from this reaction type.

LTT = 1, data are given as Legendre coefficients, where $a_o^k(E) \equiv 1.0$ is understood.

= 2, data are given as a tabulation.

NK = the number of discrete photons including the continuum (must equal the value given in File 12 or 13).

NI number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.

NE number of neutron energy points given in a TAB2 record.

NL_i highest value of ℓ required at each neutron energy E_i.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

EG_k the photon energy as given in File 12 or 13. For a continuous photon energy distribution, EG_k, = 0.0 should be used.

a. LI = 1: Isotropic Distribution

If LI = 1, then all photons for the reaction type (MT) in question are assumed to be isotropic. This is a flag that the processing code can sense, and thus needless isotropic distribution data are not entered in the file. In this case, the section is composed of a HEAD card and a SEND card, as follows:

(MAT, 14, MT/ZA, AWR; LI=1, b; NK, b)HEAD

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

b. LI = 0: Anisotropic Distribution

If LI = 0, there are two possible structures for a section, depending upon the value of LTT.

i. LTT = 1: Legendre Coefficient Representation

The structure of a section with LI = 0 and LTT = 1 is

(MAT, 14, MT/ZA, AWR; LI=0, LTT=1; NK, NI)HEAD

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

The structure of each record in the first block of NI subsections, which is for the NI isotropic photons, is

(MAT, 14, MT/EG_k, ES_k; b, b; b, b)CØNT .

There is just one CØNT record for each isotropic photon. (The set of CØNT records is empty if NI = 0.) The subsections are ordered in decreasing magnitude of EG_k (photon energy), and the continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons in decreasing magnitude of EG_k. The continuum, if present and anisotropic, appears last, with $EG_k \equiv 0.0$. The structure for the last NK-NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int})TAB2

(MAT, 14, MT/ b, E_1; b, b; NL_1, b/$a_\ell^k(E_1)$)LIST

(MAT, 14, MT/ b, E_2; b, b; NL_2, b/$a_\ell^k(E_2)$)LIST

.

.

.

(MAT, 14, MT/ b, E_{NE}; b, b; NL_{NE}, b/$a_\ell^k(E_{NE})$)LIST .

Note that lists of the $a_\ell^k(E)$ start at $\ell = 1$ because $a_o^k(E) \equiv 1.0$ is always understood.

ii. LTT = 2: Tabulated Angular Distributions

The structure of a section for LI = 0 and LTT = 2 is

(MAT, 14, MT/ZA, AWR; LI=0, LTT=2; NK, NI)HEAD

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

The structure of the first block of NI subsections (where NI may be zero) is the same as for the case of a Legendre representation; i.e., it consists of one CONT record for each of the NI isotropic photons in decreasing mag-

nitude of EG_k. The continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

The structure of the first NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; b, b)CONT .

This block of NI subsections is then followed by a block of NK-NI sub-sections for the anisotropic photons, again in decreasing magnitude of EG_k, with the continuum, if present and anisotropic, appearing last, with $EG_k \equiv 0.0$. The structure of the last NK-NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int})TAB2

(MAT, 14, MT/ b, E_1; b, b; NR, NP/μ_{int}/$p_k(\mu,E_1)$)TAB1

(MAT, 14, MT/ b, E_2; b, b; NR, NP/μ_{int}/$p_k(\mu,E_2)$)TAB1

.

.

.

(MAT, 14, MT/ b, E_{NE}; b, b; NR, NP/μ_{int}/$p_k(\mu,E_{NE})$)TAB1 .

14.2. Procedures

1. The subsections are given in decreasing magnitude of EG_k within each of the isotropic and anisotropic blocks.

2. The convention is that the subsection for the continuous photon spectrum, if present, appears last in its block. In this case, $EG_{NK} \equiv 0.0$.

3. The values of EG_k should be consistent within four significant figures with the corresponding EG_k values in File 12 or 13. File 12, Option 2 (transition probability arrays), the values of EG_k are implicitly determined by the level energies.

4. ES_k is the energy of the level from which the photon originates, if known. Otherwise, $ES_k \equiv 0.0$ (as is always the case for the continuum).

5. Data should not appear in File 14 for photons that do not have production data given in File 12 or 13. Conversely, for every photon appearing in File 12 or 13 an angular distribution must be given in File 14. The neutron energy range for which the angular distributions are given should be the same as that for which the photon production data are given in File 12 or 13.

6. For LTT = 1 (Legendre coefficients), the value of NL should be the minimum number of coefficients that will reproduce the angular distribution with sufficient accuracy and be positive everywhere. In all cases, NL should be an even number, ≤ 20.

7. The TAB1 records for the $p_k(\mu,E_i)$ within a subsection are given in increasing order of neutron energy, E_i.

8. The tabulated probability functions, $p_k(\mu,E_i)$, should be normalized within four significant figures (to unity).

9. The interpolation scheme for $p_k(\mu,E)$ with respect to E must be linear-linear or log-linear (INT = 2 or 3) to preserve normality of the interpolated distributions. It is recommended that the interpolation in μ be linear-linear (INT = 2).

10. For LI = 1 (isotropic distribution), the parameter NK is the number of photons in that section and should be consistent with the NK values in Files 12 and 13.

11. The minimum amount of data should be used that will accurately represent the angular distribution as a function of both μ and E.

12. If all photons for a reaction type (MT number) are isotropic, the LI = 1 flag should be used. The use of LI = 0 and NI = NK is strongly discouraged. Likewise, isotropic distributions should not be entered explicitly as a tabulation or as a Legendre expansion with $a_\ell^k(E) \equiv 0$, $\ell \geq 1$.

13. Angular distributions for photons must be given for all discrete and continuum photons. This can be done by specifying the data explicitly (by giving distributions) or implicitly by using a flag meaning that all photons for a particular reaction (MT number) are isotropic. Isotropic angular distributions should be specified unless the anisotropy is > 20%.

15. FILE 15: CONTINUOUS PHOTON ENERGY SPECTRA

File 15 provides a means for representing continuous energy distributions of secondary photons, expressed as normalized probability distributions. The energy distribution of each photon continuum occurring in Files 12 and 13 should be specified in File 15 over the same neutron energy range used in Files 12 and 13. Each section of File 15 gives the data for a particular reaction type (MT number) and the sections are ordered by increasing MT number.

The energy distributions, $f(E_\gamma \leftarrow E)$, are in units of eV^{-1} and are normalized so that

$$\int_0^{E_\gamma^{max}} f(E_\gamma \leftarrow E)\, dE_\gamma = 1 \quad ,$$

where E_γ^{max} is the maximum possible secondary photon energy and its value depends on the incoming neutron energy as well as the particular nuclei involved.* The energy distributions $f(E_\gamma \leftarrow E)$ can be broken down into the weighted sum of several different normalized distributions in the following manner:

$$f(E_\gamma \leftarrow E) = \sum_{j=1}^{NC} p_j(E)\, g_j(E_\gamma \leftarrow E) \qquad (eV)^{-1} \quad ,$$

*Note that the subscript k used in describing Files 12 and 13 has been dropped from $f(E_\gamma \leftarrow E)$. This is done because only one energy continuum is allowed for each MT number, and the subscript k has no meaning in File 15. It is, in fact, the NK^{th} subsection in File 12 or 13 that contains the production data for the continuum.

where

NC $\equiv$ the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$,

$g_j(E_\gamma \leftarrow E)$ $\equiv$ the j^{th} normalized partial distribution in the units eV^{-1}, and

$p_j(E)$ $\equiv$ the probability or weight given to the j^{th} partial distribution, $g_j(E_\gamma \leftarrow E)$.

The following normalization condition is imposed.

$$\int_0^{E_\gamma^{max}} g_j(E_\gamma \leftarrow E)\, dE_\gamma = 1 .$$

Thus,

$$\sum_{j=1}^{NC} p_j(E) = 1 .$$

The absolute energy distribution cross section, $\sigma^\gamma(E_\gamma \leftarrow E)$, can be constructed from the expression

$$\sigma^\gamma(E_\gamma \leftarrow E) = \sigma^\gamma(E)\ f(E_\gamma \leftarrow E) \qquad \text{(b/eV)},$$

where $\sigma^\gamma(E)$ is the integrated cross section for the continuum given either directly in File 13 or through the combination of Files 2, 3, and 12.

The system used to represent continuous photon energy distributions in File 15 is similar to that used in File 5. At present, however, there is only one continuous distribution law activated for File 15, i.e.,

$$g_j(E_\gamma \leftarrow E) = g(E_\gamma \leftarrow E),$$

where $g(E_\gamma \leftarrow E)$ represents an arbitrary tabulated function. In the future, new laws (for example, the fission gamma-ray spectrum) may be added.

15.1 File 15 Format

The structure of a section is

(MAT, 15, MT/ZA, AWR; b, b; NC, b)HEAD

<subsection for j = 1>

<subsection for j = 2>

.

.

.

<subsection for j = NC>

(MAT, 15, 0/ b, b; b, b; b, b)SEND .

For LF = 1, the structure of a subsection is

(MAT, 15, MT/b, b; b, LF=1; NR, NP/E_{int}/$p_j(E)$)TAB1

(MAT, 15, MT/b, b; b, b; NR, NE/E_{int})TAB2

(MAT, 15, MT/b, E_1; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_1)$)TAB1

(MAT, 15, MT/b, E_2; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_2)$)TAB1

.

.

.

(MAT, 15, MT/b, E_{NE}; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_{NE})$)TAB1 .

Only one distribution law is presently available (tabulated secondary photon energy distribution). Therefore, formats for other laws remain to be defined, but their structures will probably closely parallel those in File 5 for LF =

5, 7, 9, and 11. When histogram representations are used (interpolation scheme, INT = 1), 0.25 to 0.5-MeV photon energy bands should be used. The incident energy ranges must agree with data given in Files 12 and/or 13. Other procedures are the same as those recommended for File 5 data (tabulated distribution).

15.2. File 15 Procedures

1. Photon energies, E_γ, within a subsection are given in order of increasing magnitude.

2. The TAB1 records for the $g(E_\gamma \leftarrow E_i)$ within a subsection are given in increasing order of neutron energy, E_i.

3. The tabulated functions, $g(E_\gamma \leftarrow E_i)$, should be normalized to unity within four significant figures.

4. The interpolation scheme for $p_j(E)$ must be either linear-linear or log-linear (INT = 1, 2, or 3) to preserve probabilities upon interpolation. Like-wise, the interpolation scheme for $g(E_\gamma \leftarrow E)$ must be linear-linear or log-linear with respect to E.

5. The neutron energy mesh should be a subset of that used for the $y_{NK}(E)$ tabulation in File 12 or for the $\sigma^{\gamma}_{NK}(E)$ tabulation in File 13, and the energy ranges must be identical. However, the neutron energy mesh for $p_j(E)$ need not be the same as that for $g(E_\gamma \leftarrow E)$, as long as they span the same range.

6. For an MT number appearing in both File 12 and File 13, a continuous photon energy distribution (LF = 1) can appear in only one of those files. Otherwise the distribution as given in File 15 could not in general be uniquely associated with a corresponding multiplicity or production cross section.

7. Use the minimum amount of data that will accurately represent the energy distribution as a function of both E_γ and E. However, do not use too

course a mesh for E, even if the distributions are slowly varying functions of E, since the interpolated distribution will always have a nonzero component up to the maximum energy at which either of the original distributions has a nonzero component.

8. The limit on the number of neutron energy points for either $p_j(E)$ or $g(E_\gamma \leftarrow E)$ is 200. The limit on the number of photon energy points for $g(E_\gamma \leftarrow E)$ is 1000.

16. FILE 16: PHOTON ENERGY-ANGLE DISTRIBUTIONS*

If the photon-energy and angular dependences of the cross section for production of the continuous spectrum are not separable, then a file analogous to File 6 is required instead of Files 14 and 15. The energy-angle distributions for discrete photons are completely determined in File 14 and should never appear in File 16. A knowledge of the formats of Files 14 and 15 is assumed in this discussion.

Consider a nonnormalized energy-angle distribution function $F(E_\gamma \leftarrow E,\mu)$, where the angular dependence is normalized so that

$$\int_{-1}^{1} F(E_\gamma \leftarrow E,\mu)\,d\mu = y(E_\gamma \leftarrow E) \quad .$$

Then the multiplicity (yield) can be separated out, leaving a function, $h(E_\gamma \leftarrow E,\mu)$, normalized in both E_γ and μ:

$$F(E_\gamma \leftarrow E,\mu) = y(E)\ h(E_\gamma \leftarrow E,\mu) \quad .$$

The differential photon production cross section is then obtained from

$$\frac{\partial^2 \sigma(E_\gamma \leftarrow E,\mu)}{\partial E_\gamma\ \partial\mu} = \sigma(E)\ y(E) h(E_\gamma \leftarrow E,\mu) \qquad \text{(b-photons/eV)},$$

where $\sigma(E)$ is the cross section for the reaction type being considered, as determined in Files 2 and 3.

---

*The use of File 16 is discouraged but the formats and procedures are presented here in case the file should be activated.

As in File 14, the angular part of the distribution may be specified either in tabular form or as Legendre coefficients, $a_\ell(E_\gamma \leftarrow E)$. The Legendre expansion is

$$h(E_\gamma \leftarrow E,\mu) = \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} a_\ell(E_\gamma \leftarrow E)\ P_\ell(\mu) \ .$$

16.1. File 16 Format

Two options are allowed, corresponding to the options in File 14: the angular distribution can be represented by either Legendre coefficients (LTT = 1) or by tabulated angular distributions (LTT = 2).

LTT = 1: Legendre Coefficient Representation. In this option, the Legendre coefficients are tabulated as functions of both incident neutron energy and photon energy. The structure of a section for LTT = 1 is

(MAT, 16, MT/ZA, AWR; b, LTT=1; b, b)HEAD

(MAT, 16, MT/ b, b; b, b; NL, b)CONT

<subsection for ℓ = 0>

<subsection for ℓ = 1>

.

.

.

<subsection for ℓ = NL>

(MAT, 16, 0/ b, b; b, b; b, b)SEND .

The subsections contain the energy distributions, and are identical in structure to a section for a continuous energy distribution (File 15), with the following exceptions:

a. The SEND record is deleted.

b. The HEAD record is changed to read

(MAT, 16, MT/b, b; b, b; NC, b)CØNT.

c. $g(E_\gamma \leftarrow E)$ is replaced by $a_\ell(E_\gamma \leftarrow E)$.

LTT = 2: Tabulated Angular Distribution. In the option, the subsections consist of tabulations for $h(E_\gamma \leftarrow E,\mu_m)$, m = 1,2...NA. The structure of a section for LTT = 2 is

(MAT, 16, MT/ZA, AWR; b, LTT=2; b, b)HEAD

(MAT, 16, MT/ b, b; b, b; NR, NA/μ_{int})TAB2

<subsection for m = 1>

<subsection for m = 2>

.

.

.

<subsection for m = NA>

(MAT, 16, 0/ b, b; b, b; b, b)SEND .

As with the Legendre coefficient representation, this section for a tabulation contains subsections, identical in structure to a section for File 15, with the following exceptions:

a. The SEND record is deleted.

b. The HEAD record is changed to read

(MAT, 16, MT/b, μ_m; b, b; NC, b)CØNT .

c. $g(E_\gamma \leftarrow E)$ is replaced by $g(E_\gamma \leftarrow E,\mu_m)$, where each subsection is for a particular value of μ_m, m = 1,2...NA.

16.2. File 16 Procedures

The procedures for this file are the same as those for Files 14 and 15, where applicable.

17. FILE 17, DISCRETE DELAYED GAMMA RAYS

17.1. General Remarks

The gamma rays given in Files 12-16 are emitted in a negligibly short time following a neutron interaction. Many interactions, however, of which fission is a prominent example, produce residual nuclides which emit gamma rays with measurable half-lives, or which decay into other products which subsequently emit gamma rays. Since the emitting nuclide may be unknown, and in many practical instances its photon emission is the only property of interest, information concerning these nuclides may be obtained from Files 17 and 18 of the MAT for the target nuclide instead of being collected from all of the separate MATs for the intermediate products. Clearly Files 17 and 18 are redundant with any evaluation for such an intermediate nuclide.

File 17 is used to represent data for resolved discrete delayed photons; File 18 is used for various types of unresolved data on delayed emission. Within a particular MAT, Files 17 and 18 are completely disjoint with each other and with Files 12-16, so that the total delayed photon emission is the sum of Files 17 and 18 and is not redundant with any of the prompt emission in Files 12-16 of the same MAT.

All of the information in Files 17 and 18 is given in the form of multiplicities (photons/fission, photons/capture, etc.). Consequently, it is always necessary to refer to Files 2 and/or 3 to obtain cross sections for the production of the delayed photons.

17.2. Formats

File 17 contains information concerning individual delayed gamma rays. At present the only option recognized is LF = 1, yield and half-life of discrete

gamma rays of known energy. The data are entered as triplets of information (photon energy, half-life, yield) using the following structure:

(MAT, 17, MT/ZA , AWR, b, b, b, b) HEAD

(MAT, 17, MT/ b, b, LF=1, b, NR, NE/E_{int}) TAB2

[
(MAT, 17, MT/ b, E_1, b, b, NZ, b) CONT

(MAT, 17, MT/ZA_1, b, b, b, NN_1, $NG_1/C_1(E_1)$) LIST

(MAT, 17, MT/ZA_2, b, b, b, NN_2, $NG_2/C_2(E_2)$) LIST

.
.
.

(MAT, 17, MT/ZA_{NZ}, b, b, b, NN_{NZ},$NG_{NZ}/C_{NZ}(E_1)$) LIST
]

(Repeat above bracket for E_2)

(Repeat above bracket for E_3)

.
.
.

(Repeat above bracket for E_{NE})

(MAT, 17, 0/ b, b, b, b, b, b) SEND

where

LF = 1 Indicates that half-life and yield data are given. No other options exist at present.

E = Incident neutron energy (eV).

NZ = Number of residual nuclei for which discrete photon data are given.

ZA_i = ZA-number (1000*Z+A) of the i^{th} residual nucleus for which discrete photon data are given. NOTE: ZA = 0.0 is used to indicate photons of unknown or undesignated origin.

NG_i = Number of discrete photons attributed to the nucleus ZA_i.

NN_i = $3*NG_i$.

$\underline{C_i(E_j)}$ = The array of yield and half-life data for the i^{th} residual nucleus at neutron energy E_j. The array contains NG_i triplets of data in the order EG, $T_{\frac{1}{2}}$, Y (in increasing order of EG).

EG = Photon energy (eV).

$T_{\frac{1}{2}}$ = Half-life (seconds).

Y = Yield (e.g., photons/fission, photons/capture, etc.).

17.3. Procedures

1. One section is included for each reaction type MT which gives rise to delayed emission. These are arranged in order of increasing MT.

2. Since File 17 contains multiplicities, the corresponding production cross sections over the same range of neutron energies must be available in Files 2 and/or 3.

3. File 17 is not redundant with any other files in the same MAT. Therefore, since all entries in File 17 imply gamma intensities at zero time, these intensities must be excluded from the prompt spectrum that appears in Files 12-16 and from any spectra given in File 18.

4. File 17 is intended to summarize, in the MAT for the target nuclide, information which in principle could be found by searching MATs containing data on each of the product nuclides which give rise to delayed photon emission. Accordingly, ZA $\neq$ 0.0 (and hence NZ > 1), should be used only when there is compelling reason for identifying the emitting nuclides (in order to distinguish volatile from nonvolatile fission products, or example). In fact, it is strongly recommended that evaluators of data for product nuclides with half-lives longer than 1 msec make a practice of routinely forwarding summaries of photon-emmision data to the evaluators assigned to the target nuclides.

ZAs which are known to be covered in other MATs should be mentioned in File 1 to prevent ambiguity in determining redundancy. If more than one ZA is used, ZA = 0.0 is given first, followed by the others in increasing order of ZA.

5. In contrast to the convention in File 12, discrete gamma rays are given in increasing order of photon energy. Continuum gamma rays are given in File 18 (not in File 17 with $E_\gamma = 0.0$).

6. No mechanism is provided for distinguishing photons which are part of a known cascade. Although members of a cascade are tabulated separately, they should be given identical half-lives and consistent yields.

7. In the case of delayed photons from fission, many of the photons have been observed following fission of more than one nuclide. Wherever the identification is reasonably certain, the half-life and energy should be the same in File 17 for each of the target MATs involved.

18. FILE 18, CONTINUOUS SPECTRA OF DELAYED-PHOTON EMISSION

18.1. General Remarks

Resolved photons are catalogued by energy and half-life in File 17. File 18 provides a means of representing the spectrum of delayed unresolved secondary photons as a function of time after interaction.

Three options are provided for representing these continuous distributions:

LFT = 1 The photon energy distribution $P(E,T,E_\gamma)$ is given as a function of time T after the primary reaction for a given neutron energy E.

LFT = 2 Same as LFT = 1 except that only average photon energies are given instead of the complete energy distributions.

LFT = 3 The photon yield $P'(E,T)$ is given as a function of E and T with no information provided on the photon energy spectrum except a lower-energy cutoff.

The distribution for LFT = 1 is normalized so that the integral with respect to E_γ is equal to unity. To date, no provision has been made for the use of parameterized analytical distribution laws.

18.2. Formats

Time-dependent data for energy distributions of delayed unresolved secondary photons are given in File 18 using only one LFT option for a given reaction type MT.

18.2.1. LFT = 1 for complete time-yield-energy array

The structure of a section with LFT = 1 is as follows:

(MAT,18,MT/ZA, AWR, b, b, b, b) HEAD

(MAT,18,MT/ b, b, LFT = 1, b, NR_E, NE/E_{int}) TAB2

[

(MAT,18,MT/ b, E_1, b, b, NR_T, $NT/T_{int}/Y(E_1,T)$) TAB1

(MAT,18,MT/ b, b, b, b, NR_T, NT/T_{int}) TAB2

(MAT,18,MT/ b, T_1, b, b, NN_1, $NB_1/EG_{int}/P(E_1,T_1,EG)$) TAB1

(MAT,18,MT/ b, T_2, b, b, NN_2, $NB_2/EG_{int}/P(E_1,T_2,EG)$) TAB1

.
.
.

(MAT,18,MT/ b, T_{NT}, b, b, NN_{NT}, $NB_{NT}/EG_{int}/P(E_1,T_{NT},EG)$) TAB1

]

(Repeat above bracket for E_2)

(Repeat above bracket for E_3)

.
.
.

(Repeat above bracket for E_{NE})

(MAT,18,0/ b, b, b, b, b, b) SEND

where

E = Incident neutron energy (eV).

T = Time after primary reaction (seconds).

Y(E,T) = Yield per second per interaction of photons at time T and neutron energy E. Units are, for example, photons/fission/sec, photons/capture/sec, etc. .

NT = Number of times at which yields or spectra are given.

NB_i = Number of photon energies or energy bins given at time T_i.

NN_i = $3*NB_i$.

EG = Photon energy (eV).

P(E,T,EG) = Probability at neutron energy E for emission of a photon of energy EG at time T. Units are (eV) and the normalization is $\int P(E,T,EG)\,d(EG) = 1$.

18.2.2. LFT = 2 for yield-time array with average photon energies given

The structure of a section with LFT = 2 is as follows

(MAT,18,MT/ZA, AWR, b, b, b, b)HEAD

(MAT,18,MT/ b, b, LFT = 2, b, NR_E, NE/E_{int})TAB2

(MAT,18,MT/ b, E_1, I_1, J_1, NN_1, $NT_1/R(E_1)$)LIST

(MAT,18,MT/ b, E_2, I_2, J_2, NN_2, $NT_2/R(E_2)$)LIST

.
.
.

(MAT,18,MT/ b, E_{NE}, I_{NE}, J_{NE}, NN_{NE}, $NT_{NE}/R(E_{NE})$)LIST

(MAT,18,0/ b, b, b, b, b, b)SEND

where the quantities have the same meanings as for LFT = 1, and

NT_i = The number of times at which data are given for the i^{th} neutron energy, E_i.

NN_i = $3*NT_i$.

$R(E_i)$ = The array of time, yield, and average photon energy given for the i^{th} neutron energy. The array contains NT_i triplets of data T, Y, $\overline{EG}$, in increasing order of T.

$\overline{EG}$ = The average energy of photons emitted at time T and neutron energy E (eV).

I_i = The interpolation scheme to be used for Y in the LIST record for the i^{th} neutron energy.

J_i = The interpolation scheme to be used for $\overline{EG}$ in the LIST record for the i^{th} neutron energy.

18.2.3. LFT = 3 for yield-time array with no information given on the photon energy spectrum except a lower-energy cutoff

The format in this case is the same as for LFT = 2 except that $\overline{EG}$ is replaced by EC, the lower energy cutoff (in eV) for photons emitted at time T and neutron energy E.

18.3. Procedures

1. There is one section for each reaction type MT which produces a delayed continuum. These are arranged in order of increasing MT.

2. Since File 18 contains multiplicities (yields), the corresponding production cross sections for the primary reactions must be available over the same range of neutron energy for Files 2 and/or 3.

3. File 18 is not redundant with the "prompt" data in Files 12-16. Therefore, either T_1 must be a time after which no more of the "prompt" yield occurs, or else $T_1 = 0.0$ and the zerotime yield has been subtracted from the "prompt" yield. In either case, none of the resolved lines in File 17 is included in either the yield or spectrum in File 18.

4. ZA = 0.0 is used for the yield and spectrum of the sum of all unresolved delayed photons whose emitting nuclides are unknown or are unspecified. If NZ > 1, ZA = 0.0 is given first, followed by the others in increasing order of ZA. Refer to 17.3.3. for considerations on whether or not to use ZA = 0.0.

5. The range of T should be great enough to include essentially all of the yield from each delayed process. It need not be the same for different reactions types MT or different emitters ZA. The energies used for representing the yield and spectrum should be subsets of a single mesh. The primary neutron

energies should preferably be a subset of the mesh used for representing the cross section in File 3.

6. Note that for LFT = 2 or 3, the interpolation scheme for E is given in the usual way in the TAB2 record, but separate interpolation methods for Y and $\overline{EG}$ (or EC) versus T are given at the beginning of each LIST record. Although these methods need not be the same and can change from one incident-neutron energy to the next, the same method must be used for the entire range of T.

19. FILE 19, ELECTRON MULTIPLICITIES AND TRANSITION PROBABILITY ARRAYS

The formats and procedures for encoding evaluated data or electron production from neutron induced reactions are dentical to those for photon production data. (See Chapter 12.)

20. FILE 20, ELECTRON PRODUCTION CROSS SECTIONS

The formats and procedures for encoding evaluated data for electron production from neutron induced reactions are identical to those for photon production data. (See Chapter 13,)

21. FILE 21, ELECTRON ANGULAR DISTRIBUTIONS

The formats and procedures for encoding evaluated data for electron production from neutron induced reactions are identical to those for photon production data. (See Chapter 14.)

22. FILE 22, CONTINUOUS ELECTRON ENERGY SPECTRA

The formats and procedures for encoding evaluated data for electron production from neutron induced reactions are identical to those for photon production data. (See Chapter 15.)

23. FILE 23: "SMOOTH" PHOTON INTERACTION CROSS SECTIONS

23.1. General Comments on Photon Interaction

Photon interaction data are divided into five files, the first four analogous to Files 3 through 6.

File	Description
23	"Smooth" cross sections
24	Secondary angular distributions
25	Secondary energy distributions
26	Secondary energy-angle distributions
27	Coherent scattering form factors and incoherent scattering functions

As with the photon production data files, the photon interaction data formats parallel as closely as possible those for the neutron data files of the same number (modulo 20). This facilitates the use of existing retrieval routines in processing codes for photon interaction data (as in CHECKER). For Compton scattering at higher energies ($\gtrsim$1 MeV), the energy and angular distribution files would not normally be used because a simple analytical representation of these distributions is available. Also, provision is made for the entry of coherent scattering form factors as well as incoherent scattering functions. The secondary energy and angular distribution files can be used for both photon secondaries or particulate secondaries (e.g., photoneutrons).

23.2. General Description

This file is for the integrated photon interaction cross sections, including those usually called microscopic attenuation or energy-deposition coefficients, as well as photonuclear reaction cross sections. The reaction

type (MT) numbers for photon interaction are in the 500 and 600 series. Several common photon interactions have been assigned MT numbers:

MT	Reaction Description
501	Total
502	Coherent scattering
504	Incoherent scattering
515	Pair production, electron field
516	Pair production, nuclear and electron field (i.e., pair plus triplet production)
517	Pair production, nuclear field
518	Photofission (γ,f)
532	Photoneutron (γ,n)
533	Total photonuclear
602	Photoelectric

Photon cross sections, such as the total cross section, coherent elastic scattering cross section, and incoherent (Compton) cross section, are given in File 23, which has essentially the same structure as File 3. These data are given as a function of energy, E_γ, where E_γ is the energy of the incident photon (in eV). The data are given as energy-cross-section pairs. An interpolation scheme is given that specifies the energy variation of the cross section for photon energies between a given energy point and the next higher energy point. The photon cross sections are given in one or more energy ranges. Within any one energy range, the interpolation scheme is unchanged. The interpolation scheme may change from one to another energy range.

Each section in File 23 contains the data for a particular reaction type (MT number). The sections are ordered by increasing MT number.

23.3. Formats

The format is almost identical to that of File 3, as follows.

(MAT, 23, MT/ZA, AWR; b, b; b, b)HEAD

(MAT, 23, MT/ b, b; b, b; NR, NP/$E_{\gamma\ int}$/$\sigma(E_\gamma)$)TABl

(MAT, 23, 0/ b, b; b, b; b, b)SEND .

23.2. Procedures

1. Values are usually for elements; hence, except for monoisotopic elements, ZA = Z x 1000; also, AWR should be for the naturally occurring element.

2. Photoelectric edges will not be multivalued. The edge will be defined by two energies differing in the fourth or fifth significant figure.

3. The total pair production values are given for reaction type MT = 516. Reaction type 517 is reserved for the portion of the pair production cross section due to the nuclear field, i.e., excluding triplet production.

4. Interpolation is normally log-log (INT = 5).

5. Kerma factor (energy deposition coefficients) libraries will normally be local because there is no universal definition. The application will determine whether annihilation or other radiation fractions are subtracted.

24. FILE 24: SECONDARY ANGULAR DISTRIBUTIONS

The structure of File 24 is identical to that for File 4, so the pertinent discussion from Section 4 of this report is reviewed here for convenience.

Secondary angular distributions are expressed as probability density functions, $p(\mu,E_\gamma)$. These functions can be represented either as a tabulation or as the Legendre coefficients, $a_\ell(E_\gamma)$, in

$$\frac{d\sigma(E_\gamma,\mu)}{d\mu} = \sigma(E_\gamma) \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell(E_\gamma) P_\ell(\mu) ,$$

$$a_o(E_\gamma) \equiv 1.0.$$

Here, $\mu = \cos\theta$, where θ is the polar angle of scattering in either the center-of-mass or the laboratory system. The secondary may be either a photon (coherently scattered) or a particle (e.g., photoneutrons). When the secondary distribution is for a photon, the laboratory system is always used.

24.1. Formats

The format is identical to that for File 4 and will not be reproduced here (see section 4). However, for the case in which the secondary distribution is for a photon, the LCT flag is not relevant, and the following arbitrary convention is adopted:

a. LCT = 1, data are given in the laboratory system.

b. LVT = 0, transformation matrix is not given.

25. FILE 25: SECONDARY ENERGY DISTRIBUTIONS

The structure of the analogous File 5 appears to be entirely adequate (see Section 5). Thus, the format will not be reproduced here, but will be adopted by reference to File 5.

26. FILE 26: SECONDARY ENERGY-ANGLE DISTRIBUTIONS

The structure of the analogous File 6 appears to be entirely adequate (see Section 6). Thus, the format will not be reproduced here but adopted by reference to File 6. The inclusion of File 26 (as well as Files 6 and 16) is, at the present stage of development of cross section data, strictly pro forma.

27. FILE 27: ATOMIC FORM FACTORS OR SCATTERING FUNCTIONS

The ENDF system for neutron and photon production data allows two alternatives for storing angular distribution data. One is by probability per unit $\cos\theta$ vs $\cos\theta$, and the other is by Legendre coefficients. Actually, neither of these is a "natural" method for photons. The natural method would be atomic form factors or incoherent scattering functions. These are discussed briefly below.

a. Incoherent Scattering. The cross section for incoherent scattering is given by

$$\frac{d\sigma_i}{d\mu} = S(q;Z) \quad \frac{d\sigma_c}{d\mu} \ ,$$

where $d\sigma_c/d\mu$ is the Klein-Nishina cross section, which can be written in closed form. The factor $S(q;Z)$ is the incoherent scattering function. At high momentum transfer (q), S approaches Z. In the other limit $S(0,Z) = 0$. The quantity q is the momentum of the recoil electron (in inverse angstroms*).

$$q = \alpha \left[1 + \left(\frac{\alpha'}{\alpha}\right)^2 - 2\mu \left(\frac{\alpha'}{\alpha}\right) \right]^{1/2}$$

where

$$\alpha = E_\gamma / m_o c^2 \ ,$$

E'_γ = scattered photon energy, and

$\mu = \cos\theta$.

---

*In ENDF, q is given in inverse angstroms as customarily reported in the literature. The above equations show q in "natural" $m_o C$ units. Inverse angstroms, $\sin(\theta/2)/\gamma$, can be converted to $m_o C$ units by the factor 2 x 12398.1/511006. = 0.0048524.

The angular distribution can then easily be calculated, given a table of S(q;Z). Because S is a smoothly varying function of q, it can be represented by a reasonably small array of numbers. The quantities S(q;Z) are tabulated as a function of q in File 27. The user presumably will have subroutines available for calculating q for energies and angles of interest and for calculating Klein-Nishina cross sections. He will then generate his cross sections for the appropriate cases by calculating q's, looking up the appropriate values of S, and substituting them in the above formula.

b. Coherent Scattering. The coherent scattering cross section is given by

$$\frac{d\sigma_{coh}}{d\mu} = \pi r_o^2 \quad (1 + \mu^2) \; (F(q;Z))^2,$$

where

$q = \alpha\left(2(1 - \mu)\right)^{1/2}$, the recoil momentum of the atom (in inverse angstroms, see note on previous page), and

$r_o = e^2/m_o c^2$, the classical radius of the electron.

The quantity F(q;Z) is a form factor. This quantity is also easily tabulated. At high momentum transfer (q), F approaches zero. In the other limit F(0;Z) = Z.

An alternative way of presenting the photon scattering data, then, would be to tabulate incoherent scattering functions and form factors. Users could then provide processing codes to generate the cross sections from this information. The calculation is quite straightforward and allows the user to generate all his scattering data from a relatively small table of numbers. The incoherent and coherent scattering data should always be presented as scattering functions and form factors, respectively, whether or not data are included in Files 24, 25, or 26.

27.1. File 27 Format

The structure of a section is very similar to that of File 3 (and 23) and is

(MAT, 27, MT/ZA, AWR; b, b; b, b)HEAD

(MAT, 27, MT/ b, Z; b, b; NR, NP/q_{int}/H(q;Z))TABl

(MAT, 27, 0/ b, b; b, b; b, b)SEND.

The general symbol H(q;Z) is used for either F(q;Z) or S(q;Z) for coherent and incoherent scattering, respectively.

27.2. File 27 Procedures

1. Values of H(q;Z) should be entered in each case for the entire energy range for which integrated coherent and incoherent cross sections are given in File 23. This is true even though the respective values may be 0.0 or Z over most of the (higher) energy range,
2. The value of Z is entered in floating-point format.

30. INTRODUCTION TO DATA COVARIANCE FILES

Until ENDF/B-IV, the only means available to evaluators for communicating the estimated uncertainties in the data was through publication of the documentation of the evaluations. During the preparation of ENDF/B-IV, a Data Covariance Subcommittee of CSEWG was formed to coordinate the efforts at standardizing statements made about the data uncertainties and their correlations. One of the important aspects of nuclear data and of cross sections in particular is that the uncertainties in the various data tend to be highly correlated through the measurement processes and the different corrections made to the observable quantities to obtain the microscopic cross sections. In many applications when one is interested in estimating the uncertainties in calculated results based on the cross sections, the correlations in the uncertainties of the different data play a crucial role. In principle the uncertainties in the results of a calculation due to the data uncertainties can be calculated provided one is given all of the covariances of the data. In practice, in addition to the uncertainties due to the basic data, the results of calculations have uncertainties due to the calculational models used which may not represent perfectly the actual situation for which the calculations are intended. In some situations "modeling uncertainties" may dominate the uncertainties in computed results; in others they are negligible compared to the microscopic data uncertainties. In principle "modeling uncertainties" may be reduced by improving the models, although sometimes at large cost. The data uncertainties may also be reduced, often at large cost, by performing better measurements, new kinds of measurements or more refined analysis of existing data.

One of the requirements of the uncertainty information is that it be easily processed to yield the covariances in the "data" used in the calculations

themselves. For ENDF/B-IV the principle of having the uncertainty information on the data tape was adopted and a trial formalism was developed. This formalism has the virtue that the information is in such a form that it can be easily processed with minor modification to existing processing codes. Only a few evaluations of ENDF/B-IV were issued with data covariance information in this format as a trial measure. Since then, considerably more work has been done in trying to quantify data covariances within the ENDF/B formalism and using the information for purposes of sensitivity studies. These sensitivity studies have been made in three different areas where the data covariances play a crucial role: propagation of uncertainties to final calculated results, adjustment of data sets incorporating information from some integral measurements and, determination of data accuracies needed to meet targeted uncertainties in results taking into account the correlations in uncertainties in the present data sets. Although most of the work done to date within the ENDF/B structure can only be termed exploratory in nature, some significant results have already been obtained. It is therefore with much greater confidence in the ultimate usefulness of this effort that the formalism and formats for representing data covariances in ENDF/B-V have been extended to cover all of the neutron cross section data in the files.

Currently, the formats and procedures exist for representing the data covariances in ENDF/B-V for the neutron cross sections. It should be noted here that two important classes of data, energy distributions (File 4) and angular distributions (File 5), have not yet received formats for representing the data uncertainties and their correlations. It is realized that in some applications this will limit the usefulness of the present data covariance file for neutron transport applications. Hopefully, these deficiencies will be removed in the near future; however, it should be realized that very possibly an enormous pro-

cessing task lies ahead for these types of data since one must now generate covariance matrices for group-to-group transfer cross sections as well as their Legendre moments! For many applications the lack of treatment of covariances in continuous energy distributions is most serious. However, within the current formats of ENDF/B, it is possible to bypass the problem, at least for inelastic scattering, by using the subterfuge of pseudodiscrete levels to treat a continuum. In this case the current formats may be used since we have then effectively eliminated the continuum from the files. Unfortunately this expedient is of little use for some important "continuum types of data" such as (n,2n) spectra.

31. FILE 31, COVARIANCES OF THE AVERAGE NUMBER OF NEUTRONS PER FISSION

31.1. General Description

For fissionable materials, in File 1, MT=452 must be used to specify $\bar{\nu}$, the average total number of neutrons per fission. MT=455 may be used to specify the average total number of delayed neutrons per fission, $\bar{\nu}_d$, and MT=456 may also be used to specify the average number of prompt neutrons per fission, $\bar{\nu}_p$. The average number of neutrons per fission is given as a function of incident neutron energy. This energy dependence may be given by tabulating the values as a function of incident neutron energy or by providing the coefficients for a polynomial expansion as a function of incident neutron energy. Whichever method is used, the result is that the quantities are specified as a function of incident neutron energy and in this sense are similar to the data given in File 3. Therefore, the problems associated with representing the covariances of the average number of neutrons per fission are identical to those in File 33.

31.2. Formats

The formats for File 31, MT=452, 455 and 456 are the same as those for File 33 given in section 33.2.

31.3. Procedures

All procedures given in 33.3 concerning the ordering and completeness of sections of File 33 apply to sections of File 31: (MAT,31,452), (MAT,31,455) and (MAT,31,456).

We note that in File 1 $\bar{\nu}$ (MT=452), $\bar{\nu}_d$ (MT=455) and $\bar{\nu}_p$ (MT=456) satisfy the relation:

$$\bar{\nu}\ (E) = \bar{\nu}_d\ (E) + \bar{\nu}_p\ (E).$$

Therefore if one of these quantities is "derived" in terms of the other two, it is permissible to use "NC-type" sub-subsections with LTY=0 to indicate that it is a "derived redundant cross section." See section 33.2.I.a. for an explanation of this format.

When a section of File 31 for either MT=452, 455 or 456 is used, there must be a section in File 33 for the fission cross sections, i.e., section (MAT,33,18).

Note: 1. Since $\bar{\nu}_d$ is much smaller than $\bar{\nu}_p$, it should never be evaluated by subtracting $\bar{\nu}_p$ from $\bar{\nu}$.

2. When a polynomial representation is used to describe the data in File 1 MT=452, 455 and 456, the covariance file applies to the tabular reconstruction of the data as a function of energy and not to the polynomial coefficients.

32. FILE 32, COVARIANCES OF RESONANCE PARAMETERS

32.1. General Description

In File 32, MT=151, the covariances of the resonance parameters of the resolved resonances in File 2, MT=151, may be given. The resonance parameters in File 2, MT=151, used with the appropriate resonance formalism, provide an efficient way to represent the important correlations in the magnitudes of the different partial cross sections over the resonances, compared to the use of File 3 only. Similarly with File 32, the use of the covariances of the resonance parameters of individual resonances provides an efficient way of representing the rapid variation of the covariances of the partial cross sections over the individual resonances. In the resonance region the covariances of the partial cross sections are often characterized by a) "long-range" components which affect the covariances over many resonances, which should be given in File 33, and b) "short-range" components affecting the covariances of the different partial cross sections over the individual resonances, which should be given in File 32. When the cross sections are averaged over many resonances, as is often the case for some applications in the higher energy range of the resolved resonance energy region and for the unresolved resonance energy region, the effects of the "short-range" components tend to "average out" and the covariances of the averaged cross sections are dominated by the long-range components given in File 33. Therefore, the covariances of the cross sections in the unresolved resonance energy region should be given by means of File 33 only. In the resolved resonance energy region the covariances of the partial cross sections may be given using only File 33, in which case the "short-range" variations which occur over individual resonances are ignored. For some applications such as: a) the calculation of the uncertainties in Doppler effects and self-shielding and b) the calculation of group cross section covariances where the groups are narrow

compared to the resonance width (or when only very few resonances are within a group), a knowledge of the "short-range" correlations of uncertainties within the resonance is needed. It is for these purposes that File 32 should be used, with File 33, to describe the covariances of the cross sections. Because this situation may only be important in the lower energy range of the resolved resonance region, File 32 may only contain data for the lowest energy resonances and need not cover the whole range of resonances given in File 2. It should be noted that it is the sum of the covariances in Files 32 and 33 which gives the total covariances of the partial cross sections and that in the resolved and unresolved resonance energy region one should not identify the components of the covariance matrix given in File 33 with the covariances of the "residual cross sections" which may or may not be present in File 3.

For ENDF/B-V the use of File 32 is limited to the Breit-Wigner representations (LRF=1 or 2). For the Adler-Adler representation one is limited to the description of the "long-range" components given in File 33.

For the Breit-Wigner representations (LRF=1, single level; or LRF=2, multi-level formalism) for each resonance given in File 2 we have:

ER the resonance energy (in the laboratory system);

AJ the floating-point value of J (the spin, or total angular momentum of the resonance);

GT the resonance total width, Γ, evaluated at the resonance energy ER;

GN the neutron width, Γ_n, evaluated at the resonance energy ER;

GG the radiation width, Γ_γ, evaluated at the resonance energy ER;

GF the fission width, Γ_f, evaluated at the resonance energy ER.

GT is no longer redundant since a "competitive width" GX is allowed in File 2. However, for purposes of File 32 we ignore GX and therefore only concern ourselves with the covariance matrix of the parameters ER, AJ, GN, GG and

GF. Because the covariance of ER and the other resonance parameters is very small in practice in File 32 we only concern ourselves with the variance of ER. The uncertainties on ER in practical applications are expected to be significant only for some calculations involving the thermal region for a few resonances. The value of the total angular momentum AJ of the resonance may only take on some discrete values. When the value of AJ can be determined from the experimental data, then its variance is zero as well as the covariance of AJ and of the other resonance parameters. However, frequently the exact value of AJ cannot be obtained from the experiments; it is then permissible (see 2.2.2 Procedures for File 2) to assign to it a value in File 2 such that the statistical weight factor g_L is given properly, on the average, for each value of L, the neutron angular momentum. In such cases it may not be possible within the resonance formalism used, to represent all of the available data and their uncertainties without assigning a variance to AJ and a covariance of AJ and of the other resonance parameters. In the above situation AJ is considered as a constrained parameter of the formalism and the covariances of AJ and of the other parameters a device for representing accurately the uncertainties in various quantities related to the resonance.

32.2. Formats

The format for File 32, MT=151, parallels the format for File 2, MT=151, with the restrictions that only resolved resonance parameters of the single-level and multi-level Breit-Wigner formalisms are allowed. The major difference is that in File 32 we require more information per resonance, the covariances of the resonance parameters, than in File 2. In the description of the format for File 32, MT=151, which follows, we use the same symbolism for naming the quantities as given in File 2, section 2, of the manual:

NIS is the number of isotopes in this material (NIS < 10).

ZAI is the (Z,A) designation for an isotope.

ABN is the abundance (weight fraction) of an isotope in this material.

EL is the lower energy limit of the energy range.

EH is the upper energy limit of the energy range.

LRF is a flag indicating which representation is used:

LRF=1, single-level B-W parameters.

LRF=2, multi-level B-W parameters.

The general structure of File 32 is as follows:

(MAT, 32, 151/ ZA, AWR; 0, 0; NIS, 0) HEAD

(MAT, 32, 151/ ZAI, ABN; 0, 0; 1, 0) CONT (isotope)

(MAT, 32, 151/ EL, EH; 1, LRF; 0, 0) CONT (range)

<subsection for the first isotope>

.

.

.

(MAT, 32, 151/ ZAI, ABN; 0, 0; 1, 0) CONT (isotope)

(MAT, 32, 151/ EL, EH; 1, LRF; 0, 0) CONT (range)

<subsection for the last isotope>

.

.

.

(MAT, 32, 0/ 0.0, 0.0; 0, 0; 0, 0) SEND

(MAT, 0, 0/ 0.0, 0.0; 0, 0; 0, 0) FEND

The structure of a subsection is the same for LRF=1 (single-level B-W parameters) as it is for LRF=2 (multi-level B-W parameters). The following quantities are defined:

SPI is the nuclear spin of the target nucleus, I (positive number).

AP is the scattering radius in units of 10^{-12} cm.

QX is an effective Q-value.

NLS is the number of sets of resonance parameters given.

A set of parameters is given for each ℓ-value (neutron orbital angular momentum (NLS $\leq$ 3).

L is the value of ℓ.

AWRI is the ratio of the mass of a particular isotope to that of a neutron.

NRS is the number of resonances for a given ℓ-value (NRS $\leq$ 500).

The symbols for the resonance parameters and the covariances of the resonance parameters are:

ER the resonance energy (in the laboratory system);

AJ the floating-point value of J (the spin, or total angular momentum of the resonance);

GT the resonance total width, Γ, evaluated at the resonance energy ER;

GN the neutron width, Γ_n, evaluated at the resonance energy ER;

GG the radiation width, Γ_γ, evaluated at the resonance energy ER;

GF the fission width, Γ_f, evaluated at the resonance energy ER;

DE^2 the variance of ER in units of eV squared;

DJ^2 the variance of AJ;

DJDN the covariance of AJ and of GN in units of eV;

DN^2 the variance of GN in units of eV squared;

DJDG the covariance of AJ and of GG in units of eV;

DNDG the covariance of GN and of GG in units of eV squared;

DG^2 the variance of GG in units of eV squared;

DJDF the covariance of AJ and of GF in uits of eV;

DNDF the covariance of GN and of GF in units of eV squared;

DGDF the covariance of GG and of GF in units of eV squared;

DF^2 the variance of GF in units of eV squared.

The structure of a subsection is as follows:

(MAT, 32, 151/ SPI, AP; 0, 0; NLS, 0) CONT

(MAT, 32, 151/ AWRI, QX; L, 0; 18*NRS, NRS/

ER_1, AJ_1, GT_1, GN_1, GG_1, GF_1,
DE_1^2, DN_1^2, $DNDG_1$, DG_1^2, $DNDF_1$, $DGDF_1$,
DF_1^2, $DJDN_1$, $DJDG_1$, $DJDF_1$, DJ_1^2, 0.0,
ER_2, AJ_2, GT_2, GN_2, GG_2, GF_2,
DE_2^2, DN_2^2, $DNDG_2$, DG_2^2, $DNDF_2$, $DGDF_2$,
DF_2^2, $DJDN_2$, $DJDG_2$, $DJDF_2$, DJ_2^2, 0.0,

.
.
.

ER_{NRS}, AJ_{NRS}, GT_{NRS}, GN_{NRS}, GG_{NRS}, GF_{NRS},
DE_{NRS}^2, DN_{NRS}^2, $DNDG_{NRS}$, DG_{NRS}^2, $DNDF_{NRS}$, $DGDG_{NRS}$.
DF_{NRS}^2, $DJDN_{NRS}$, $DJDG_{NRS}$, $DJDF_{NRS}$, DJ_{NRS}^2, 0.0)LIST.

32.3. Procedures

The data in File 32 for each resonance are only intended to provide information concerning the rapid variations of the covariance matrices of the different partial cross sections over the resonance, the long-range components of the covariance matrices being given in File 33. Since the long-range components of the covariance matrices may be the dominant ones, in particular in the higher energy range of the resolved resonance region for some material, it is not neces-

sary to provide in File 32 the covariances of the resonance parameters for all the resonances given in File 2, although it might be desirable to do so.

I. Correspondence between File 32 and File 2

The following procedures indicate the relationships which may or must exist between corresponding quantities in File 2 and File 32. We shall indicate quantities in File 2 with the index 2 and the corresponding quantities in File 32 with the index 32.

1. In File 2 for each subsection EL_2 and EH_2 indicate the range of energies where the resonance formalism must be used to generate the partial cross sections even though some of the resonances may have a resonance energy outside the range of EL_2 to EH_2. In File 32, since the resonances with resonance energies outside EL_{32} and EH_{32} may only contribute long-range components to the covariance matrices of the partial cross sections in the range EL_{32} to EH_{32}, it is not necessary to include them in File 32 if their contributions to the covariance matrices have been given in File 33.

2. In the corresponding subsections of File 2 and File 32 the value of EL must be the same, i.e. $EL_{32}=EL_2$. However, since the long-range components of the covariance matrices given in File 33 may dominate completely the covariance matrices in the higher energy region of the resolved resonance range EH_{32} may be lower than EH_2, i.e. $EH_{32} \leq EH_2$.

3. Within the energy range EL_{32} to EH_{32} all of the resonances given in the subsections of File 2 must also be given in File 32.

II. Completeness of File 32

Procedure I-3 above requires that all the resonances of File 2 within the energy range EL_{32} to EH_{32} be given in File 32. All of the covariances of the parameters of each resonance need not be non-zero in File 32. In particular, the variance of AJ, DJ^2, and the covariances of AJ and of the other parame-

ters will be zero when the value of AJ is known or when it is not used as a constrained free parameter of the formalism. However, when the covariance of two different resonance parameters is non-zero, the variances of each of these parameters must be non-zero. We note that because of self-shielding effects the analysis of most experimental data results in a large covariance of GN and of GG as well as a large covariance of GN and of GF.

32.4. Example

In order to illustrate the use of the formats and procedures for File 32, we give in Tables 32.1 and 32.2 Files 2 and 32 for an hypothetical evaluation of Fe-56. The resonance parameters were evaluated using: transmission data which yielded values of $g\Gamma_n$ and their variances, angular distribution data which yielded J, and capture resonance areas ($g\Gamma_n\Gamma_\gamma/\Gamma_t$) with their variances after correction for selfshielding effects in the samples. Without going into the details of the evaluation of the various resonances most of the covariances of Γ_n and Γ_γ came from the variance in the resonance areas. When $\Gamma_n >> \Gamma_\gamma$ the capture area is essentially a measure of Γ_γ and since Γ_t was known the covariance of Γ_n and Γ_γ was small and neglected in the File 32. When $\Gamma_\gamma >> \Gamma_n$ the capture area is essentially a measure of Γ_n. Since the self-shielding is small, the covariance of Γ_n and Γ_γ was small and it was neglected in the File 32.

In the example of Table 32.2 we do not give the File 33 which would accompany this File 32. This File 33 would contain a section for MT=1, the total cross sections, with long-range components; it would also contain a section for MT=102, the capture cross sections with a long-range component of about 10% to account for the normalization of all of the capture data.

To illustrate the procedures, in File 2 we indicate the range of the resolved resonance energy region as 10^{-5} eV to 100 keV and give a few resonances

outside the range. In File 3 we indicate the range from 10^{-5} eV to 85 keV with variances and some covariances for all the resonances in File 2 within this range. Presumably the File 33, not shown for this material, would represent entirely the covariances of the cross sections above 85 keV.

Table 32.1

```
 2.60560+ 4 5.54540+ 1          0          0          1          01180 2151 HEAD
 2.60560+ 4 9.16600- 1          0          0          1          01180 2151 CONT
 1.00000- 5 1.00000+ 5          1          2          0          01180 2151 CONT
 0.00000+ 0 5.00000- 1          0          0          3          01180 2151 CONT
 5.54540+ 1 0.00000+ 0          0          0         36          61180 2151 LIST
-2.00000+ 3 5.00000- 1 1.80640+ 2 1.80000+ 2 6.40000- 1 0.00000+ 01180 2151
 2.76700+ 4 5.00000- 1 1.52140+ 3 1.52000+ 3 1.40000+ 0 0.00000+ 01180 2151
 7.39800+ 4 5.00000- 1 5.35730+ 2 5.35000+ 2 7.30000- 1 0.00000+ 01180 2151
 8.36500+ 4 5.00000- 1 1.25130+ 3 1.25000+ 3 1.30000+ 0 0.00000+ 01180 2151
 1.29800+ 5 5.00000- 1 5.01100+ 2 5.00000+ 2 1.10000+ 0 0.00000+ 01180 2151
 1.40400+ 5 5.00000- 1 2.70220+ 3 2.70000+ 3 2.20000+ 0 0.00000+ 01180 2151
 5.54540+ 1 0.00000+ 0          1          0        114         191180 2151 LIST
 1.14900+ 3 5.00000- 1 6.60000- 1 6.00000- 2 6.00000- 1 0.00000+ 01180 2151
 1.24500+ 4 5.00000- 1 5.42300- 1 2.30000- 3 5.40000- 1 0.00000+ 01180 2151
 1.77500+ 4 5.00000- 1 5.59000- 1 1.90000- 2 5.40000- 1 0.00000+ 01180 2151
 2.27900+ 4 5.00000- 1 8.10000- 1 2.70000- 1 5.40000- 1 0.00000+ 01180 2151
 3.42000+ 4 1.50000+ 0 1.33000+ 0 7.90000- 1 5.40000- 1 0.00000+ 01180 2151
 3.84000+ 4 1.50000+ 0 8.60000- 1 3.20000- 1 5.40000- 1 0.00000+ 01180 2151
 4.60400+ 4 5.00000- 1 1.05300+ 1 1.00000+ 1 5.30000- 1 0.00000+ 01180 2151
 5.21200+ 4 1.50000+ 0 1.24200+ 1 1.20000+ 1 4.20000- 1 0.00000+ 01180 2151
 5.35400+ 4 5.00000- 1 1.67000+ 0 1.00000+ 0 6.70000- 1 0.00000+ 01180 2151
 5.53700+ 4 5.00000- 1 2.02000+ 0 1.90000+ 0 1.20000- 1 0.00000+ 01180 2151
 5.92000+ 4 1.50000+ 0 4.49000+ 0 4.00000+ 0 4.90000- 1 0.00000+ 01180 2151
 6.34400+ 4 1.50000+ 0 1.35000+ 0 8.00000- 1 5.50000- 1 0.00000+ 01180 2151
 7.29800+ 4 5.00000- 1 2.07200+ 1 2.00000+ 1 7.20000- 1 0.00000+ 01180 2151
 7.70400+ 4 5.00000- 1 3.93000+ 0 3.60000+ 0 3.30000- 1 0.00000+ 01180 2151
 9.02900+ 4 1.50000+ 0 1.44600+ 1 1.40000+ 1 4.60000- 1 0.00000+ 01180 2151
 9.27800+ 4 1.50000+ 0 1.06000+ 0 5.20000- 1 5.40000- 1 0.00000+ 01180 2151
 9.62900+ 4 5.00000- 1 1.70000+ 0 1.30000+ 0 4.00000- 1 0.00000+ 01180 2151
 9.65700+ 4 1.50000+ 0 2.90000+ 0 2.50000+ 0 4.00000- 1 0.00000+ 01180 2151
 1.02630+ 5 1.50000+ 0 2.13600+ 1 2.10000+ 1 3.60000- 1 0.00000+ 01180 2151
 5.54540+ 1 0.00000+ 0          2          0         36          61180 2151 LIST
 2.35000+ 3 1.50000+ 0 8.40200- 1 2.00000- 4 8.40000- 1 0.00000+ 01180 2151
 2.01700+ 4 1.50000+ 0 8.44700- 1 4.70000- 3 8.40000- 1 0.00000+ 01180 2151
 3.67000+ 4 2.50000+ 0 9.50000- 1 1.10000- 1 8.40000- 1 0.00000+ 01180 2151
 8.08000+ 4 2.50000+ 0 7.74000+ 0 7.00000+ 0 7.40000- 1 0.00000+ 01180 2151
 9.26500+ 4 1.50000+ 0 2.25000+ 0 1.60000+ 0 6.50000- 1 0.00000+ 01180 2151
 9.61400+ 4 2.50000+ 0 1.77000+ 0 6.70000- 1 1.10000+ 0 0.00000+ 01180 2151
 0.00000+ 0 0.00000+ 0          0          0          0          01180 2  0 SEND
 0.00000+ 0 0.00000+ 0          0          0          0          01180 0  0 FEND
```

Table 32.2

```
2.60560+ 4 5.54540+ 1          0          0          1          0118032151 HEAD
2.60560+ 4 9.16600- 1          0          0          1          0118032151 CONT
1.00000- 5 1.00000+ 5          1          2          0          0118032151 CONT
0.00000+ 0 5.00000- 1          0          0          3          0118032151 CONT
5.54540+ 1 0.00000+ 0          0          0         36          2118032151 LIST
2.76700+ 4 5.00000- 1 1.52140+ 3 1.52000+ 3 1.40000+ 0 0.00000+ 0118032151
9.00000+ 2 9.00000+ 2 0.00000+ 0 1.00000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
7.39800+ 4 5.00000- 1 5.35730+ 2 5.35000+ 2 7.30000- 1 0.00000+ 0118032151
2.50000+ 3 1.00000+ 2 0.00000+ 0 4.90000- 3 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
5.54540+ 1 0.00000+ 0          1          0        180         10118032151 LIST
1.14900+ 3 5.00000- 1 6.60000- 1 6.00000- 2 6.00000- 1 0.00000+ 0118032151
4.00000+ 0 1.60000- 5 1.40000- 4 3.60000- 3 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
1.24500+ 4 5.00000- 1 5.42300- 1 2.30000- 3 5.40000- 1 0.00000+ 0118032151
1.60000+ 3 9.00000- 8 0.00000+ 0 2.50000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
1.77500+ 4 5.00000- 1 5.59000- 1 1.90000- 2 5.40000- 1 0.00000+ 0118032151
2.50000+ 3 4.00000- 6 0.00000+ 0 2.50000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
2.27900+ 4 5.00000- 1 8.10000- 1 2.70000- 1 5.40000- 1 0.00000+ 0118032151
2.50000+ 3 3.60000- 3-5.80000- 3 2.50000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
3.42000+ 4 1.50000+ 0 1.33000+ 0 7.90000- 1 5.40000- 1 0.00000+ 0118032151
2.50000+ 3 9.00000- 2-4.30000- 2 2.50000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
3.84000+ 4 1.50000+ 0 8.60000- 1 3.20000- 1 5.40000- 1 0.00000+ 0118032151
2.50000+ 3 6.40000- 3-9.30000- 3 2.50000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
4.60400+ 4 5.00000- 1 1.05300+ 1 1.00000+ 1 5.30000- 1 0.00000+ 0118032151
2.50000+ 3 9.00000+ 0 0.00000+ 0 2.50000- 3 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
5.21200+ 4 1.50000+ 0 1.24200+ 1 1.20000+ 1 4.20000- 1 0.00000+ 0118032151
4.00000+ 2 1.00000+ 0 0.00000+ 0 1.60000- 3 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
5.35400+ 4 5.00000- 1 1.67000+ 0 1.00000+ 0 6.70000- 1 0.00000+ 0118032151
4.00000+ 2 1.60000- 1-5.50000- 2 2.90000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
5.53700+ 4 5.00000- 1 2.02000+ 0 1.90000+ 0 1.20000- 1 0.00000+ 0118032151
4.00000+ 2 9.00000- 2 0.00000+ 0 2.50000- 3 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
5.54540+ 1 0.00000+ 0          2          0         54          3118032151 LIST
2.35000+ 3 1.50000+ 0 8.40200- 1 2.00000- 4 8.40000- 1 0.00000+ 0118032151
2.50000+ 3 3.60000- 9 0.00000+ 0 6.20000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
2.01700+ 4 1.50000+ 0 8.44700- 1 4.70000- 3 8.40000- 1 0.00000+ 0118032151
2.50000+ 3 2.50000- 7 0.00000+ 0 6.20000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
3.67000+ 4 2.50000+ 0 9.50000- 1 1.10000- 1 8.40000- 1 0.00000+ 0118032151
2.50000+ 3 1.00000- 4 0.00000+ 0 6.20000- 2 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0118032151
0.00000+ 0 0.00000+ 0          0          0          0          0118032  0 SEND
0.00000+ 0 0.00000+ 0          0          0          0          01180 0  0 FEND
```

33. FILE 33, COVARIANCES OF NEUTRON CROSS SECTIONS

33.1 General Description

The covariances of neutron cross section information appearing in File 3 are given in File 33. File 33 is intended to provide a measure of the "accuracies and their correlations" of the data in File 33 and does not indicate the precision with which the data are entered in the File 33. Since ENDF/B represents our knowledge of the microscopic data, the File 33 is used to give the covariances of these microscopic data. However, it should be stressed that for most practical applications to which the files are intended the data will be processed into group cross sections. While generating File 33 it should be remembered that one of their major aims is to represent adequately:

i. The variances of the group cross sections,

ii. The correlations of the uncertainties between the several adjacent groups, and

iii. The long-range correlations of the uncertainties over many groups.

Table 33.1 illustrates the relation of these three covariances with experimental uncertainties.

These primary considerations and the inherent difficulties associated with quantifying uncertainties should dictate the details given in Files 33.

In the resolved resonance region, some of the covariances of the cross sections, within each resonance, may be given through the covariances of the resonance parameters in File 32. In this case, the long-range components of the covariance matrix of the cross sections which span several resonances are given in File 33. It is permissible, in the resolved resonance region, to represent the covariance matrix in File 33, since often the major components of the matrix are

Table 33.1

Analogies Between File 33 Covariances Within One Section and Experimental Uncertainties*

File 33	Experimental	Energy Dependence
short-range	statistical	Rapid variation
medium-range	Detector Efficiency Multiple Scattering In/Out Scattering	Slowly varying
long-range	Geometry Flux Background Normalization	More or less constant

Example: If we had a 2% uncertainty due to statistics (short-range), 2% due to multiple scattering (medium-range) and, 1% due to background (long-range), we would cite a 3% uncertainty for a discrete measurement (one group covering a small ($\lesssim$ 200 keV) energy range); $\lesssim$ 2.5% over an energy range encompassing several measurements (several groups which together cover a 1- to 2-MeV range); and $\lesssim$ 1% over the entire energy range.

*As with all analogies, this should be used with care. It is designed to show in a familiar way of thinking how the covariance within a section are related.

long-range. In the unresolved resonance region the covariances of the cross sections must be given entirely in File 33.

33.2 Formats

File 33 is divided into sections identified by the value of MT. Within a section, (MAT,33,MT), several subsections may appear. Each section of File 33 starts with a HEAD record, ends with a SEND record, and has the following structure:

(MAT, 33, MT/ZA, AWR; b, MTL; b, NL) HEAD

<subsection for L = 1>

<subsection for L = 2>

.

.

.

<subsection for L = NL>

(MAT, 33, 0/ b, b; b, b; b, b) SEND

NL in the HEAD record denotes the number of subsections within a section. A non zero value of MTL is used as a flag to indicate that reaction MT is one component of the evaluator defined lumped reaction MTL, as discussed below; in this case, no covariance information is given for reaction MT (other than the HEAD record) and NL = 0.

Subsections

Each subsection of the section (MAT,33,MT) is used to describe a single covariance matrix. It is the covariance matrix of the energy-dependent cross sections given in section (MAT,3,MT) and energy-dependent cross sections given in section (MAT1,3,MT1) of the ENDF/B tape. The values of MAT1 and MT1 are given in the CONT record which begins every subsection. Each subsection is

therefore identified with a unique combination of values (MAT,MT) and (MAT1,MT1), and we may use the notation (MAT,MT;MAT1,MT1) to specify a subsection.

Each subsection may contain several sub-subsections. Two different types of sub-subsections may be used; they are referred to as "NC-type" and "NI-type" sub-subsections. Each sub-subsection describes an independent contribution, called component, to the covariance matrix given in the subsection. The total covariance matrix in the subsection is made up of the sum of the contributions of the individual sub-subsection.

The structure of a subsection describing the covariance matrix of the cross sections given in the ENDF/B type (MAT,3,MT) and (MAT1,3,MT1) is:

(MAT, 33, MT/ b, b; MAT1, MT1; NC, NI) CONT

<sub-subsection for n_c = 1>

<sub-subsection for n_c = 2>

.

.

.

<sub-subsection for n_c = NC>

<sub-subsection for n_i = 1>

<sub-subsection for n_i = 2>

.

.

.

<sub-subsection for n_i = NI>

NC is the number for "NC-type" sub-subsections which follow the CONT record.

NI is the number of "NI-type" sub-subsections which follow the "NC-type" sub-subsections.

Sub-subsections

There are two different types of sub-subsections which have a different structure, the "NC-type" and "NI-type" sub-subsections.

The "NC-type" sub-sections 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 major purpose of the "NC-type" subsections is to eliminate from the ENDF/B tape a large fraction of the mostly redundant information which would otherwise be needed if only "NI-type" sub-subsections were used.

The "NI-type" sub-subsections are used to describe explicitly the various components of the covariance matrix of the subsection.

I. "NC-type" Sub-subsections

The "NC-type" sub-subsections may be used to describe the covariance matrices in energy ranges where the cross sections in (MAT,3,MT) can be "derived" in terms of other "evaluated" cross sections in the same energy range. In the context of File 33, and for purposes of discussing "NC-type" sub-subsections, we define an "evaluated" cross section, in a given energy range, as one for which the covariance matrix in that energy range is given entirely in terms of "NI-type" sub-subsections. The covariance matrices involving the "derived" cross sections may be obtained in terms of the covariance matrices of the "evaluated" cross sections already given in File 33 and therefore need not be given explicitly again.

a. LTY=0, "Derived Redundant Cross Sections"

In File 33 the evaluator may indicate by means of an LTY=0 sub-subsection that in a given energy range the cross sections in (MAT,3,MT) were strictly obtained, in the general sense of evaluated, as a linear combination of other "evaluated" cross sections having the same MAT number but different MT values. We recall that we use the definition of "evaluated" cross sections in the sense that the covariances of these cross sections are given in File 33 only in terms of "NI-type" sub-subsections. In general the linear relationship given in an LTY=0 sub-subsection applies not only to the range of energy specified, but also over the whole range of the file; however, it may not be the method whereby the cross sections were obtained, in the sense of evaluated, over the whole energy range of the file.

The structure of an "NC-type" sub-subsection with LTY=0 is:

(MAT, 33, MT/ b, b; b, LTY=0; b, b) CONT

(MAT, 33, MT/ E1, E2: b, b; 2*NCI, NCI/ {CI, XMTI}) LIST

In the LIST record, E1 and E2 define an energy range where the cross sections given in the section (MAT,3,MT) were "derived" in terms of other "evaluated" cross sections given in the sections (MAT,3,MTI)s.

NCI is the number of pairs of values in the array {CI, XMTI}.*

{CI, XMTI} are pairs of numbers. The coefficient CI is associated in the pair with a value of MTI, given as a floating point number and indicated as XMTI. The pairs of numbers indicate, in the energy range E1 to E2, that the cross sections in file (MAT,3,MT), written as ${}^{MAT}\sigma_{MT}(E)$, were obtained in terms of the cross sections in files (MAT,3,MTI), written as ${}^{MAT}\sigma_{MTI}(E)$, as follows:

*The notation {AI,BI} stands for A_1, B_1; A_2, B_2; ...; A_i, B_i in a LIST record.

$$^{MAT}\sigma_{MT}(E) = \sum_{i=1}^{NCI} C_i * {}^{MAT}\sigma_{MT_i}(E).$$

In this expression we have written the CI's as C_i, and XMTI's as MT_i. The numbers CI are constant numbers over the whole range of energy E1 to E2, usually $\pm$ 1.

Note: In general each subsection describes a single covariance matrix. However, when an "NC-type" sub-subsection with LTY=0 is used in a subsection, several covariance matrices may be implied and these are not explicitly given as subsections in the File 33 (see procedure II-a-3). Therefore, in such cases the subsection may be thought of as describing several covariance matrices.

b. LTY=1, 2 and 3, "Covariances of Cross Sections Derived via Ratio Measurements"

Many important cross sections of ENDF/B are determined through "ratio" measurements. Evaluation of cross sections by means of "ratio" measurements is one of the main sources of information on covariances of cross sections having different MAT values. These covariances play an important role in many applications where the results depend on the relative magnitude of different cross sections. In order to represent efficiently these important covariance matrices in the Files 33, evaluators may use "NC-type" sub-subsections with LTY=1, 2 and 3 in appropriate subsections of the Files 33.

Let the cross sections in (MAT,33,MT) be strictly "derived", in the general sense of evaluated, in the energy range E1 to E2, through the evaluation of ratio measurements to other "evaluated" cross sections given in (MATS,3,MTS), referred to also as the "standard" cross sections for this "ratio evaluation".

Then in the subsection (MAT,MT;MAT,MT) of the File 33 for the material MAT, an LTY=1 sub-subsection must be used to describe in part the covariance matrix in the energy range E1 to E2. This part, or component, of the covariance matrix, given by the LTY=1 sub-subsection, is derived from the covariance matrix of the "standard" cross sections in the subsection (MATS,MTS;MATS,MTS) of the File 33 of the "standard" material MATS. The other part, or component, of the covariance matrix comes from the evaluation of the "ratios" and is given explicitly, over the range E1 to E2, by means of "NI-type" sub-subsections in the subsection (MAT,MT;MAT,MT) of the File 33. In addition, since this method of evaluation introduces a covariance of the "derived" cross sections in (MAT,3,MT) over the energy range E1 to E2 and the "standard" cross sections in (MATS,3,MTS) over their complete energy range, in the File 33 of the material MAT, in subsection (MAT,MT;MATS,MTS), there must be an LTY=2 sub-subsection to describe this covariance matrix. This LTY=2 sub-subsection (which contains the same information as the previously given LTY=1 sub-subsection in the subsection (MAT,MT;MAT,MT)) refers to a different covariance matrix than the LTY=1 sub-subsection previously mentioned, but it can also be derived from the covariance matrix of the "standard" cross sections in the subsection (MATS,MTS;MATS,MTS)of the File 33 of the "standard" material MATS. Finally, as a consequence of the evaluation of the cross sections in (MAT,3,MT) in the energy range E1 to E2, as a "ratio" to the "standard" cross sections in (MATS,3,MTS), there must be in the subsection (MATS,MTS;MAT,MT) of the File 33 of the "standard" material MATS an LTY=3 sub-subsection (which also contains the same information as the previously given LTY=1 sub-subsection in the subsection (MAT,MT;MAT,MT)) serves in the material MATS the same role as the LTY=2 sub-subsection in the material MAT since they describe the same covariance matrix. But, in addition, the LTY value of 3

serves as a "flag" to the user, and the processing codes, to indicate that there are additional covariances of cross sections using the same "standard" cross sections (MATS,3,MTS) not explicitly given in the Files 33. These additional covariance matrices can be derived from the appropriate LTY=3 sub-subsections and the covariance matrix of the "standard" cross sections in the subsection (MATS,MTS;MATS,MTS) of the File 33 of the "standard" material MATS.

The structure of an "NC-type" sub-subsection with LTY=1,2 and 3 is:

(MAT, 33, MT/ b, b; b, LTY; b, b) CONT

(MAT, 33, MT/ E1, E2; MATS, MTS; 2*NEI, NEI/ {EI, WEI}) LIST

In the LIST record, E1 and E2 define an energy range where the cross sections given in the section (MAT,3,MT) were "derived" in terms of ratio measurements to "evaluated" cross sections given in section (MATS,3,MTS).

For ENDF/B-V the only value of NEI allowed is 2 and the list EI, WEI must be: {E1, 1.; E2, 0.}.

Note A: The above structure for LTY=1, 2 and 3 is dictated by two considerations:

1. Compatibility with the LTY=0 sub-subsection structure,

2. The possible extension of the use of the format LTY=1, 2 and 3 when the cross sections given in (MAT,3,MT) are only partially determined from ratio measurements to the cross sections given in (MATS,3,MTS). In such cases the list {EI, WEI} will indicate the relative weights of the ratio measurements in the evaluation of the cross sections in (MAT,3, MT).

Note B: LTY=1, 2 and 3 sub-subsections are all used as flags in subsections to describe relative covariance matrix components obtained from the relative covariance matrix of the "standard" cross sections already given in a File 33. There is, however, a major difference between covariance matrices ob-

tained with LTY=1 sub-subsections and those obtained from LTY=2 and 3 sub-subsections. This difference results from the definition of their use given above. LTY=2 and 3 sub-subsections are always used in subsections where one of the cross sections involved is the "standard" cross section used. The LTY=2 subsection appears in the File 33 of the material whose cross sections are "derived," whereas the LTY=3 sub-subsection appears in the File 33 of the material whose cross sections are the "standard"; LTY=1 sub-subsections always appear in subsections describing covariance matrices of cross sections "derived" from a "standard" and no LTY=2 or 3 sub-subsections may appear in such subsections. An LTY=1 sub-subsection describes a covariance matrix which in principle is a "square matrix" of dimension E1 to E2. An LTY=2 or 3 sub-subsection describes in principle a "rectangular matrix": the covariance matrix of the "derived" cross sections over the energy range E1 to E2 and of the "standard" cross sections over their complete energy range.

In general, if cross sections in (MAT,3,MT) are "derived," over an energy range E1 to E2, by "ratios" to "standard" cross sections in (MATS,3,MTS), there will be three "NC-type" sub-subsections with LTY=1, 2 and 3 generated in File 33. The LTY=1 sub-subsection is given in the subsection (MAT,MT;MAT,MT); the LTY=2 sub-subsection is given in the subsection (MAT,MT;MATS,MTS). Both of these subsections are given in the File 33 of the material MAT of the "derived" cross sections (MAT,3,MT). The LTY=3 sub-subsection is given in the subsection (MATS,MTS;MAT,MT) which is in the File 33 of the material MATS of the "standard" cross sections (MATS,3,MTS). There are, however, some instances, such as the one taken in example 33.4A, where "still another cross section" such as those in (MAT,3,MT1) are "indirectly derived" from the cross sections in (MATS,3,MTS) through evaluation of ratios of the cross sections in (MAT,3,MT1) to those in

(MAT,3,MT). In such cases an LTY=1 sub-subsection will also be used in the subsections (MAT,MT1;MAT,MT1) and (MAT,MT;MAT,MT1) and LTY=2 sub-subsection will also be used in the subsection (MAT,MT1;MATS,MTS). All three of these subsections are in the File 33 of the material MAT. Corresponding to the LTY=2 sub-subsection in the subsection (MAT,MT1;MATS,MTS) of the File 33 of the material MAT, there will also be an LTY=3 sub-subsection in the subsection (MATS,MTS;MAT,MT1) of the File 33 of the material MATS.

Note C: For purposes of discussing the covariance matrices of cross sections "derived" through evaluation of ratio measurements, the label "standard" cross sections was used for the cross sections relative to which the ratio measurements were made and the symbol (MATS,3,MTS) was used for these cross sections. The cross sections for which the label "standard" was used may be any "evaluated" cross sections of ENDF/B and are not restricted to the special set of "standard cross sections" maintained in the ENDF/B library. The "standard cross sections of ENDF/B" are the preferred ones to use for ratio measurements in order to minimize the magnitude of the covariance matrix elements obtained from LTY=1, 2 and 3 sub-subsections. However, they may not always be the ones which were used in the data available to evaluators to perform evaluations.

II. "NI-type" Sub-subsections

The "NI-type" sub-subsections are used to describe explicitly the various components of the covariance matrix given in the subsection. In each "NI-type" sub-subsection there is a flag, the LB flag, whose numerical value indicates whether the components are "relative" or "absolute" and the kinds of correlations as a function of energy represented by the components in the sub-subsection.

For values of the LB flag from 0 to 4, the "NI-type" sub-subsection has the following structure:

(MAT,33,MT/b,b; LT, LB; 2*NP, NP/{E_k , F_k}{E_ℓ , F_ℓ}) LIST

LB is a flag whose numerical value determines the meaning of the numbers given in the arrays {E_k , F_k}{E_ℓ , F_ℓ}.

NP is the total number of pairs of numbers in the arrays {E_k , F_k}{E_ℓ , F_ℓ}.

LT is the number of pairs of numbers in the second array, {E_ℓ , F_ℓ}.

LT may be zero, in which case we have a single array {E_k , F_k}. When LT≠0, we have two arrays and the first one, {E_k , F_k}, has (NP-LT) pairs of numbers in it.

{E_k , F_k}{E_ℓ , F_ℓ} are two arrays of pairs of numbers. Each array is referred to as an E table, the E_k table and the E_ℓ table. In each E table the first member of a pair is an energy, E_n; the second member of the pair, F_n, is a number associated with the energy interval between the two entries E_n and E_{n+1}.

The E_k table, and the E_ℓ table when present, must cover the complete energy range of the file. The first energy entry in an E table must therefore be 10^{-5} eV and the last one $2 \times 10^{+7}$ eV. Some of the F_k's, or F_ℓ's, may be zero, as must be the case below threshold for a threshold reaction, and the last value of F in an E table must be zero since it is not defined.

We now define the meaning of the F values entered in the E tables for different values of LB. Let X_i refer to the cross section in (MAT,3,MT) at energy E_i and Y_j refer to the cross section in (MAT1,3,MT1) at energy E_j. The contribution of the sub-subsection to the covariance matrix COV(X_i,Y_j), having the units

of "barns squared," described in the subsection, is defined as follows for the different values of LB:

LB=0 Absolute components only correlated within each E_k interval

$$COV(X_i,Y_j) = \sum_k P^{i;k}_{j;k} F_{xy,k}$$

LB=1 Fractional components only correlated within each E_k interval

$$COV(X_i,Y_j) = \sum_k P^{i;k}_{j;k} F_{xy,k} X_i Y_j$$

LB=2 Fractional components correlated over all E_k intervals

$$COV(X_i,Y_j) = \sum_{k,k'} P^{i;k}_{j;k'} F_{xy,k} F_{xy,k'} X_i Y_j$$

LB=3 Fractional components correlated over E_k and E_ℓ intervals

$$COV(X_i,Y_j) = \sum_{k,\ell} P^{i;k}_{j;\ell} F_{x,k} F_{yk\ell} X_i Y_j$$

LB=4 Fractional components correlated over all E_ℓ intervals within each E_k interval

$$COV(X_i,Y_j) = \sum_{K,\ell,\ell'} P^{i;k,\ell}_{j;k,\ell'} F_k F_{xy,\ell} F_{xy,\ell'} X_i Y_j.$$

For LB=0, 1 and 2 we have LT=0, i.e., only one E_k table. For LB=3 and LB=4 we have LT≠0, i.e., two E tables, the E_k and the E_ℓ tables.

The dimensionless operators P in the above definitions are defined in terms of the operator S as follows:

$$P^{i;k,\ell,\ldots}_{j;m,n,\ldots} \equiv S^k_i S^\ell_i \cdots S^m_j S^n_j \cdots ,$$

where

$S^k_i \equiv 1$ when the energy E_i is in the interval E_k to E_{k+1} of an E_k table,

$S^k_i \equiv 0$ when the energy E_i is outside the range of E_k to E_{k+1} of an E_k table.

It is often possible during the evaluation process to generate the relative covariance matrix of some cross sections averaged over some energy intervals. Such relative covariance matrices may be suitable for use in File 33. Although the use of LB=3 sub-subsections allows the representation of such matrices, one row (or one column) at a time this method of representation is very inefficient since one sub-subsection must be used for every row (or column) and the same energy mesh is repeated in the E_k table (or E_ℓ table) of every sub-subsection. Often, in addition, such relative covariance matrices are symmetric about their diagonal and there is no way to avoid repeating almost half of the entries with LB=3 sub-subsections. In order to allow such relative covariance matrices to be entered efficiently in the files directly LB=5 sub-subsections may be used. The following definition applies for LB=5 sub-subsections:

LB=5 Relative covariance matrix components

$$COV(X_i,Y_j) = \sum_{k,k'} P^{i;k}_{j;k'} F_{xy;k,k'} X_i Y_j$$

A single list of energies $\{E_k\}$ is required to specify the energy intervals labeled by the indices k and k'. The numbers $F_{xy;k,k'}$ represent fractional components correlated over the energy intervals E_k and $E_{k'}$.

Since we no longer have the need for the E_k tables with pairs of numbers (E_k,F_k) found in sub-subsections with LB<5 we need a new structure for LB=5 sub-subsections. The structure of an LB=5 sub-subsection is:

(MAT, 33, MT/ b, b; LS, LB=5; NT, NE/ $\{E_k\}\{F_{k,k'}\}$) LIST.

NT is the total number of entries in the two arrays $\{E_k\}$ and $\{F_{k,k'}\}$.

NE is the number of entries in the array $\{E_k\}$ defining (NE-1) energy intervals.

LS is a flag indicating whether the $F_{k,k'}$ matrix is symmetric or not:

LS=0 Asymmetric matrix

The matrix elements $F_{k,k'}$ are ordered by rows in the array $\{F_{k,k'}\}$:

$$\{F_{k,k'}\} \equiv F_{1,1} ; F_{1,2} ;\cdots; F_{1,NE-1} ; F_{2,2} ;\cdots; F_{NE-1,N}$$

There are $(NE-1)^2$ numbers in the array $\{F_{k,k'}\}$ and NT=NE + $(NE-1)^2$

LS=1 Symmetric matrix

The matrix elements $F_{k,k'}$ are ordered by rows starting from the diagonal term in the array $\{F_{k,k'}\}$:

$$\{F_{k,k'}\} \equiv F_{1,1} ; F_{1,2} ;\cdots; F_{1,NE-1} ; F_{2,2} ; F_{2,3} ;\cdots; F_{NE-1,NE-1}$$

There are NE*(NE-1)/2 numbers in the array $\{F_{k,k'}\}$ and

$$NT = NE + NE*(NE-1)/2$$

A lumped reaction is an evaluator-defined "redundant' cross section, defined in File 33 for the purpose of specifying the uncertainty in the sum of a set of cross sections, such as those for a set of neighboring discrete inelastic levels. The uncertainty in a lumped-reaction cross section, as well as its correlations with other reactions, are given in the usual way using the formats described above. On the other hand, the uncertainties and correlations of the individual parts or components of a lumped reaction are not given.

The File-33 section for one component of a lumped reaction consists of a single HEAD record that contains, in the second integer field, the section number MTL of the lumped reaction to which the component contributes. The value of MTL must lie in the range 851-870, which has been reserved specifically for lumped reactions. These MT-numbers may not be used in Files 3, 4 or 5, so the

net cross section and net scattering matrix for a lumped reaction must be constructed at the processing stage by summing over the reaction components.

A list of the components of a given lumped reaction is given only indirectly, namely, on the above-mentioned HEAD records. These special HEAD records, with MTL $\neq$ 0 and NL = 0, form a kind of index that can be scanned easily by the processing program in order to control the summing operation.

Except for the need to sum the cross-section components during uncertainty processing, lumped reactions are "normal" reactions, in that all covariance formats can be used to describe their uncertainties. For example, one expects in general that the covariances of a lumped reaction with other reactions, including other lumped reactions, will be given by the evaluator. Also, a lumped reaction may be represented, using an "NC-type" sub-subsection with LTY = 0, as being "derived" from other reactions, including other lumped reactions. (However, since uncertainties are not provided for the separate component reactions, a lumped reaction may not be represented as being "derived" from its components.)

33.3 Procedures

Although it is not necessary to have a section in File 33 for every section in File 3, the most important values of MT for the applications to which the evaluation was intended should have a section in File 33.

I. Ordering of Sections, Subsections and Sub-subsections

a. Sections

The sections in File 33 are ordered by increasing value of MT.

b. Subsections

Within a section, (MAT,33,MT), the subsections are ordered in a rigid manner. A subsection of File 33 is uniquely identified by the quartet of num-

bers: (MAT,MT;MAT1,MT1); the first pair of numbers indicate the section and the second pair of numbers appear in the appropriate field, MAT1 and MT1, of the CONT record which begins every subsection.

1. The subsections within a section are ordered by increasing values of MAT1.

2. In order to have the covariance matrices of the cross sections for which MAT1=MAT appear first in a section, and follow procedure I-b-1, the value MAT1=0 shall be used to mean MAT1=MAT in the CONT record which begins the subsection.

3. When there are several subsections with the same value of MAT1 in a section, these subsections shall be ordered by increasing values of MT1 given in the CONT record which begins the subsections.

4. When MAT1=0, which according to procedure I-b-2 means that MAT1=MAT, only subsections for MT1 $\geq$ MT shall be given.

c. Sub-subsections

When both "NC-type" and "NI-type" sub-subsections are present in a subsection, the format requires that the "NC-type" sub-subsections be given first.

1. "NC-type" sub-sections. Several "NC-type" sub-subsections may be given in a subsection. When more than one is given, these must be ordered according to the value of the energy range E1 to E2 given in the LIST record. We note that by definition, if several "NC-type" sub-subsections are given in a subsection, the energy ranges E1 to E2 of these different sub-subsections cannot overlap. The value of the LTY flag of "NC-type" sub-subsections does not affect the ordering of the sub-subsections within a subsection.

2. "NI-type" sub-subsections. There is no special ordering requirement of a "NI-type" sub-subsection within a subsection. However, it often happens that the full energy range of the file is covered by different sub-subsections, the F-values being set to zero in the E-tables outside the different ranges. It would improve the readability of the files if these different sub-subsections were grouped together by the energy range effectively covered in the sub-subsections.

II. Completeness

As previously stated, there is presently no minimum requirement on the number of sections and subsections in File 33. However, the presence of some subsections in a File 33, as well as the presence of some sub-subsections in a subsection, implies the presence of other subsections either in the same File 33 or the File 33 of another material. In what follows we shall identify the subsections by their value of the quartet:

(MAT,MT;MAT1,MT1)

a. Subsections for which MAT1=0

By subsections for which MAT1=0, we mean the subsection having the quartet: (MAT,MT;0,MT1), which according to procedure I-b-2 means MAT1=MAT.

1. If there is a subsection (MAT,MT;0,MT1) with MT1≠MT, there must be within the same File 33 the two subsections: (MAT,MT;0,MT) and (MAT,MT1;0,MT1). Note that the converse is not necessarily true since the two cross sections (MAT,3,MT) and (MAT,3,MT1) may have zero covariances, which are not required to be explicitly stated in the files. This procedure and procedure I-b-4 guarantee that every section of File 33, (MAT,33,MT), starts with the subsection (MAT,MT;0,MT).

2. In a subsection (MAT,MT;0,MT), if there is an "NC-type" sub-subsection with LTY=0, it contains a list of MTI values. There must be a subsection (MAT,MT;0,MTI) for every value of MTI given in the "NC-type" sub-subsection.

3. "NC-type" sub-subsections with LTY=0 must be given only in sub-sections of the type (MAT,MT;0,MT), i.e. with MT1=MT. "NC-type" sub-subsections with LTY=0, for "derived redundant cross sections," imply many covariance matrices of the "derived" cross sections and of the "evaluated" cross sections. It is the task of the processing code to generate these covariance matrices from the information given in the File 33.

4. In a subsection (MAT,MT;0,MT) if there is an "NC-type" sub-subsection with LTY=1, this sub-subsection contains values of MATS, MTS. There must be another material MATS with a File 33 containing the subsection (MATS,MTS;0,MTS). However, in the same File 33, there must be a sub-subsection (MAT,MT;MATS,MTS). Note that according to procedure III-a, given below, MATS must be different from MAT in an "NC-type" sub-subsection with LTY=1.

5. In a subsection (MAT,MT;0,MT), if there is an "NC-type" sub-subsection with LTY=1 which covers the energy range E1 to E2, in the same subsection there must be some "NI-type" sub-subsections with F-values different from zero in this energy range E1 to E2. These "NI-type" sub-subsections represent the relative covariance matrix of the evaluated ratio measurements.

b. Subsection for MAT1≠0

If there is a subsection (MAT,MT;MAT1,MT1) with MAT1≠0, similar to procedure II-a-1, there must also be a subsection (MAT,MT;0,MT) in the same File 33, but there must also be the two sub-subsections: (MAT1,MT1;0,MT1) and (MAT1,MT1;MAT,MT) in the File 33 for material MAT1.

III. Other Procedures

a. "NC-type" sub-subsections with LTY=1 shall only be used with MATS≠MAT. The use of LTY=1 sub-subsections is reserved for covariance matrix components arising out of ratio measurements of cross sections of different nuclides, i.e. different values of MAT.

b. If a single "NC-type" sub-subsection with LTY=0 is used in a subsection and there are no "NI-type" sub-subsections, the value of E1 must be 10^{-5} eV and the value of E2 must be $2 \times 10^{+7}$ eV.

c. As a consequence of the definition of "NC-type" sub-subsections with LTY=0, if there are any "NI-type" sub subsections in the same sub-section, the F-values in their E-tables must be zero within the range E1 to E2 of these "NC-type" sub-subsections.

d. "NI-type" sub-subsections with LB=0 shall in general be avoided and forbidden in the case of cross sections involved in ratio measurements. Therefore the "standard cross sections of ENDF/B" shall not have LB=0, "NI-type" sub-subsections. The use of LB=0 "NI-type" sub-subsections should be reserved for the description of covariance matrices of cross sections which fluctuate rapidly and for which details of the uncertainties in the "deep valleys" of the cross sections are important.

e. The formats of File 33 allow for the possibility of great details to be entered in the files if needed. The number of "NI-type" sub-subsections and the number of energy entries in their E_k and E_ℓ tables will be a function of the details of the covariance matrices available and the need to represent them within their estimated accuracies. However, good judgement should be used to minimize as much as possible the number of different entries in the E_k and E_ℓ tables. The important quantity to remember is the union of all of the E values

of the E_k and E_ℓ tables of a File 33. A reasonable upper limit of the order of 100 different E values for the union of all energy entries in all of the E_k and E_ℓ tables in a File 33 should be considered.

f. The lumping of reactions for uncertainty purposes will be useful mainly in connection with discrete-level inelastic scattering cross sections. However, other reactions, such as (n,n'p), (n,n'α), and (n,n' continuum), may also be treated in this way.

g. In order not to lose useful uncertainty information, reactions lumped together should have similar characteristics. Ordinarily, the level energies of discrete inelastic levels lumped together should not span a range greater than 30-40%, and the angular distributions should be similar.

h. The components of a lumped reaction need not have adjacent MT-numbers.

i. Lumped-reaction MT-numbers must be assigned sequentially, beginning at 851. The sequence is determined by ordering the lumped reactions according to the lowest MT-number of their respective components. Thus, the first value of MTL encountered on any component-reaction HEAD record will be 851. The next new value of MTL encountered will be 852, and so on.

j. Lumped reactions with only a single component are permitted. This is recommended practice when, for example, an important discrete inelastic level is treated individually, while all of its neighbors are lumped. Covariances for both the individual level and the nearby lumped levels can then be placed together in sections 851-870.

33.4 Example

We illustrate here the use of File 33 by means of two concrete examples.

A. Use of LTY=1 and LTY=2 "NC-type" subsections

Let us consider the hypothetical evaluation of Pu-239, MAT=1264. The decision is made that in File 33 only the fission cross sections and the capture cross sections shall have covariances represented. The following methods were used in performing the evaluation:

1. Fission cross sections, MT=18

Let X_i stand for the fission cross section of Pu-239 at the energy E_i.

a. From 10^{-5} eV to an energy ES, X_i was evaluated in terms of "direct" or "absolute" measurements, A_i. By this we mean that in this energy range, X_i and its uncertainties are independent of any other cross sections. In this energy range $X_i \equiv A_i$.

b. From ES to 20 MeV, X_i was evaluated by means of ratio measurements to Y_i the fission cross section of U-235, to which we assign the MAT number 1261. In this energy range $X_i = R_i Y_i$, where R_i is the evaluated ratio at energy E_i.

2. Capture cross sections, MT=102

Let Z_i stand for the capture cross section of Pu-239 at the energy E_i. In this evaluation, Z_i was obtained by the evaluation of a_i over the complete range of the file. Therefore we have $Z_i = a_i X_i$.

In this evaluation then, only 3 quantities were evaluated: A_i from 10^{-5} eV to ES, R_i from ES to 20 MeV, and a_i from 10^{-5} eV to 20 MeV. The evaluation of these quantities resulted in the evaluation of three covariance matrices: $COV(A_i,A_j)$, $COV(R_i,R_j)$ and $COV(a_i,a_j)$. Let us now assume that in addition it has been determined that the uncertainties in these three different

quantities are uncorrelated, i.e. covariances such as $COV(A_i,a_j)$ are essentially zero.

Let us denote relative covariance matrices such as $\frac{COV(A_i,A_j)}{A_i\ A_j}$ as $\langle dA_i \cdot dA_j \rangle$, and similarly for the other quantities.

From 10^{-5} eV to ES, since $X_i = A_i$ and $Z_i = a_i X_i$, we have:

$$\langle dX_i \cdot dX_j \rangle = \langle dA_i \cdot dA_j \rangle$$

$$\langle dX_i \cdot dZ_j \rangle = \langle dA_j \cdot dA_j \rangle$$

$$\langle dZ_i \cdot dZ_j \rangle = \langle da_i \cdot da_j \rangle + \langle dA_i \cdot dA_j \rangle$$

From ES to 20 MeV, since $X_i = R_i Y_i$ and $Z_i = a_i X_i$, we have:

$$\langle dX_i \cdot dX_j \rangle = \langle dR_i \cdot dR_j \rangle + \langle dY_i \cdot dY_j \rangle$$

$$\langle dX_i \cdot dZ_j \rangle = \langle dR_i \cdot dR_j \rangle + \langle dY_i \cdot dY_j \rangle$$

$$\langle dX_i \cdot dY_j \rangle = \langle dY_i \cdot dY_j \rangle$$

$$\langle dZ_i \cdot dZ_j \rangle = \langle da_i \cdot da_j \rangle + \langle dR_i \cdot dR_j \rangle + \langle dY_i \cdot dY_j \rangle$$

$$\langle dZ_i \cdot dY_j \rangle = \langle dY_i \cdot dY_j \rangle$$

We note that in the above we have expressed all of the covariance matrices of the cross sections only in terms of the covariance matrices of the evaluated quantities and the covariance matrix of the U-235 fission.

For purposes of illustrating the use of the formats we need not know the details of how the covariance matrices $\langle dA_i \cdot dA_j \rangle$, $\langle dR_i \cdot dR_j \rangle$ and $\langle da_i \cdot da_j \rangle$ are represented. They must be represented by one or more "NI-type" sub-subsections having an E_k table, or could be so represented. For our purposes, we symbolically represent each one of them in terms of a single "NI-type" sub-subsection with a single E_k table:

$$\langle dA_i \cdot dA_j \rangle \rightarrow \{E_k^A, F_k^A\}$$

$$\langle dR_i \cdot dR_j \rangle \rightarrow \{E_k^R, F_k^R\}$$

$$\langle da_i \cdot da_j \rangle \rightarrow \{E_k^a, F_k^a\}$$

Whether one or more "NI-type" sub-subsection is used, each one of the E tables used in the sub-subsections can be written as:

$$\{E_k^A, F_k^A\} = \{1.0E\text{-}5,\ F_1^A; \ldots;\ E_k^A, F_k^A; \ldots;\ ES,\ 0.0;\ 2.0E{+}7, 0.0\},$$

$$\{E_k^R, F_k^R\} = \{1.0E\text{-}5,\ 0.0;\ ES,\ F_1^R; \ldots;\ E_k^R,\ F_k^R; \ldots;\ 2.0E{+}7, 0.0\},$$

$$\{E_k^a, F_k^a\} = \{1.0E\text{-}5,\ F_1^a;\ \ldots;\ E_k^a, F_k^a;\ \ldots;\ 2.0E{+}7,\ 0.0\},$$

the E and F values explicitly shown must have the values indicated above for this example.

In the listing given in Table 33.2 for the File 33 of MAT=1264, corresponding to our example, we have shown with only one sub-subsection each of the matrices $\langle dA_i \cdot dA_j \rangle$, $\langle dR_i \cdot dR_j \rangle$ and $\langle da_i, da_j \rangle$ with the E tables indicated symbolically as:

(EAK, FAK) for $\{E_k^A, F_k^A\}$, etc.

Note: In the File 33 of MAT-1261 in the subsections (1261, 18;1264,18) and (1261,18;1264,102) and LTY=3 "NC-type" sub-subsection corresponding to the LTY=2 sub-subsections of Table I-33 must be inserted.

B. Use of LTY=0, "NC-type" sub-subsections

Let us consider a hypothetical evaluation of C-12, MAT=1274. The decision is made that in File 33 the MT values 1,2,4,102 and 107 shall have covariances represented. We shall use the notation developed in the previous example. The following method was used in this evaluation:

Table 33.2

```
                                    (1264,18;0,18)
9.42390+ 4 2.36999+ 2          0           0           0          3126433 18 HEAD
0.00000+ 0 0.00000+ 0          0          18           1          2126433 18 CONT
0.00000+ 0 0.00000+ 0          0           1           0          0126433 18 CONT
2.00000+ 5 2.00000+ 7       1261          18           4          2126433 18 LIST
2.00000+ 5 1.00000+ 0 2.00000+ 7 0.00000+ 0                        126433 18<dYi*dYj>
0.00000+ 0 0.00000+ 0          0           1          14          7126433 18 LIST
1.00000- 5 0.00000+ 0 1.00000+ 0 2.50000- 3 3.00000+ 2 3.60000- 3126433 18<dAi*dAj>
3.00000+ 4 4.90000- 3 1.00000+ 5 6.40000- 3 2.00000+ 5 0.00000+ 0126433 18
2.00000+ 7 0.00000+ 0                                              126433 18
0.00000+ 0 0.00000+ 0          0           1           6          3126433 18 LIST
1.00000- 5 0.00000+ 0 2.00000+ 5 4.00000- 4 2.00000+ 7 0.00000+ 0126433 18<dRi*dRj>
                                    (1264,18;0,102)
0.00000+ 0 0.00000+ 0          0         102           1          2126433 18 CONT
0.00000+ 0 0.00000+ 0          0           1           0          0126433 18 CONT
2.00000+ 5 2.00000+ 7       1261          18           4          2126433 18 LIST
2.00000+ 5 1.00000+ 0 2.00000+ 7 0.00000+ 0                        126433 18<dYi*dYj>
0.00000+ 0 0.00000+ 0          0           1          14          7126433 18 LIST
1.00000- 5 0.00000+ 0 1.00000+ 0 2.50000- 3 3.00000+ 2 3.60000- 3126433 18<dAi*dAj>
3.00000+ 4 4.90000- 3 1.00000+ 5 6.40000- 3 2.00000+ 5 0.00000+ 0126433 18
2.00000+ 7 0.00000+ 0                                              126433 18
0.00000+ 0 0.00000+ 0          0           1           6          3126433 18 LIST
1.00000- 5 0.00000+ 0 2.00000+ 5 4.00000- 4 2.00000+ 7 0.00000+ 0126433 18<dRi*dRj>
                                  (1264,18;1261,18)
0.00000+ 0 0.00000+ 0       1261          18           1          0126433 18 CONT
0.00000+ 0 0.00000+ 0          0           2           0          0126433 18 CONT
2.00000+ 5 2.00000+ 7       1261          18           4          2126433 18 LIST
2.00000+ 5 1.00000+ 0 2.00000+ 7 0.00000+ 0                        126433 18<dYi*dYj>
0.00000+ 0 0.00000+ 0          0           0           0          0126433  0 SEND
                                  (1264,102;0,102)
9.42390+ 4 2.36999+ 2          0           0           0         2126433102 HEAD
0.00000+ 0 0.00000+ 0          0         102           1         3126433102 CONT
0.00000+ 0 0.00000+ 0          0           1           0         0126433102 CONT
2.00000+ 5 2.00000+ 7       1261          18           4         2126433102 LIST
2.00000+ 5 1.00000+ 0 2.00000+ 7 0.00000+ 0                       126433102<dYi*dYj>
0.00000+ 0 0.00000+ 0          0           1          14         7126433102 LIST
1.00000- 5 0.00000+ 0 1.00000+ 0 2.50000- 3 3.00000+ 2 3.60000- 3126433102<dAi*dAj>
3.00000+ 4 4.90000- 3 1.00000+ 5 6.40000- 3 2.00000+ 5 0.00000+ 0126433102
2.00000+ 7 0.00000+ 0                                             126433102
0.00000+ 0 0.00000+ 0          0           1           6         3126433102 LIST
1.00000- 5 0.00000+ 0 2.00000+ 5 4.00000- 4 2.00000+ 7 0.00000+ 0126433102<dRi*dRj>
0.00000+ 0 0.00000+ 0          1           5          21         6126433102 LIST
1.00000- 5 2.53000- 2 9.00000- 2 2.50000- 1 1.00000+ 0 2.00000+ 7126433102<dαi*dαj>
2.21000- 3 4.84000- 4 3.62000- 4 3.56000- 4 0.00000+ 0 4.84000- 4126433102
3.10000- 4 3.04000- 4 0.00000+ 0 6.25000- 4 2.30000- 4 0.00000+ 0126433102
2.21000- 3 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0126433102
                                  (1264,102;1261,18)
0.00000+ 0 0.00000+ 0       1261          18           1         0126433102 CONT
0.00000+ 0 0.00000+ 0          0           2           0         0126433102 CONT
2.00000+ 5 2.00000+ 7       1261          18           4         2126433102 LIST
2.00000+ 5 1.00000+ 0 2.00000+ 7 0.00000+ 0                       126433102<dYi*dYj>
0.00000+ 0 0.00000+ 0          0           0           0         0126433  0 SEND
```

1. Total cross sections, MT=1

The total cross sections, σ_i^T, were evaluated over the complete energy range, with the covariance matrix obtained, and:

$$<d\sigma_i^T \cdot d\sigma_j^T> \rightarrow \{E_k^T, F_k^T\} ,$$

with

$$\{E_k^T, F_k^T\} = \{1.0E\text{-}5,\ F_1^T; \ldots;\ E_k^T, F_k^T; \ldots;\ 2.0E+7,\ 0.0\}$$

2. Elastic cross sections, MT=2

The elastic cross sections, σ_i^E, were "derived" up to 8.5 MeV from the "evaluated" cross sections:

$$\sigma_i^E = \sigma_i^T - \sigma_i^I - \sigma_i^C - \sigma_i^\alpha .$$

Above 8.5 MeV the elastic cross sections were evaluated and:

$$<d\sigma_i^E \cdot d\sigma_j^E> \rightarrow \{E_k^E, F_k^E\} ,$$

with

$$\{E_k^E, F_k^E\} = \{1.0E\text{-}5,\ 0.0;\ 8.5E+6,\ F_1^E;\ \ldots;\ E_k^E, F_k^E; \ldots; 2.0E+7,\ 0.0\}.$$

3. Inelastic cross sections, MT=4

The inelastic cross sections, σ_i^I, were evaluated from threshold, 4.8 MeV, to 8.5 MeV and:

$$<d\sigma_i^I, d\sigma_i^I> \rightarrow \{E_k^I, F_k^I\} ,$$

with

$$\{E_k^I, F_k^I\} = \{1.0E\text{-}5,\ 0.0;\ 4.8E{+}6,\ F_1^I; \ldots;\ E_k^I, F_k^I; \ldots;\ 8.5E{+}6,\ 0.0;\ 2.0E{+}7,\ 0.0\}\ .$$

Above 8.5 MeV the inelastic cross sections were "derived" and:

$$\sigma_i^I = \sigma_i^T - \sigma_i^E - \sigma_i^C - \sigma_i^\alpha\ .$$

4. Capture cross sections, MT=102

The capture cross sections, σ_i^C, were evaluated over the complete energy range and:

$$\langle d\sigma_i^C \bullet d\sigma_j^C \rangle \rightarrow \{E_k^C, F_k^C\}\ ,$$

with

$$\{E_k^C, F_k^C\} = \{1.0E\text{-}5,\ F_1^C; \ldots;\ E_k^C,\ F_k^C; \ldots;\ 2.0E{+}7,\ 0.0\}\ .$$

5. The (n,α) cross sections, MT=107

The (n,α) cross sections, σ_i^α, were evaluated from threshold, 6.18 MeV to 20 MeV and:

$$\langle d\sigma_i^\alpha \bullet d\sigma_j^\alpha \rangle \rightarrow \{E_k^\alpha, F_k^\alpha\}$$

with

$$\{E_k^\alpha, F_k^\alpha\} = \{1.0E\text{-}5,\ 0.0;\ 6.18E{+}6,\ F_1^\alpha; \ldots;\ E_k^\alpha,\ F_k^\alpha; \ldots;\ 2.0E{+}7,\ 0.0\}\ .$$

In the listing given in Table 33.3 for File 33 of MAT=1274, corresponding to our example, we have shown only one "NI-type" sub-subsection for each evaluated covariance matrix with the E tables indicated symbolically as:

$$(\text{ETK, FTK}) \text{ for } \{E_k^T, F_k^T\} \text{ etc...}$$

The above example has great similarity to the way the evaluation of C-12 was made, the major difference being that instead of MT=4 being evaluated, the evaluation was made for MT=51 and MT=91. Since it will illustrate some of the procedures of File 33, let us now consider adding to the above File 33 for MAT=1274 the covariance matrices for MT=51 and MT=91.

a. MT=51

The inelastic scattering to the first excited state, σ_i^{51}, up to 8.5 MeV is identical to σ_i^I. Therefore we may consider up to 8.5 MeV that σ_i^{51} is a "derived" cross section with: $\sigma_i^{51} = \sigma_i^I$. This is permissible because MT=4 has only "NI-type" sub-subsections in this energy range.

From 8.5 MeV to 20 MeV, MT=51 was evaluated and:

$$\langle d\sigma_i^{51} \cdot d\sigma_j^{51} \rangle \rightarrow \{E_k^{51}, F_k^{51}\} .$$

with

$$\{E_k^{51}, F_k^{51}\} = \{1.0E\text{-}5,\ 0.0;\ 8.5E+6,\ F_1^{51};\ldots;\ E_k^{51},\ F_k^{51};\ldots;\ 2.0E+7,0.0\}.$$

b. MT=91

From 8.5 to 20 MeV, the continuum inelastic, σ_i^{91}, was "derived" as: $\sigma_i^{91} = \sigma_i^I - \sigma_i^{51}$. However, we cannot use this relationship for

purposes of File 33 because σ_i^I in this energy range is indicated in the file as being already "derived."

Therefore, for purposes of File 33, we must write:

$$\sigma_i^{91} = \sigma_i^T - \sigma_i^E - \sigma_i^{51} - \sigma_i^C - \sigma_i^\alpha$$

which now only refers to cross sections having exclusively "NI-type" sub-subsections. Therefore we may now add the sections to the File 33, MAT=1274, shown in Table 33.4, to have a more complete File 33.

Table 33.3

```
                                    (1274,1;0,1)
 6.01200+ 3 1.18969+ 1           0           0           0          1112733  1 HEAD
 0.00000+ 0 0.00000+ 0           0           1           0          1112733  1 CONT
 0.00000+ 0 0.00000+ 0           0           1           6          3112733  1 LIST
 1.00000- 5 0.00000+ 0 2.00000+ 6 2.50000- 5 2.00000+ 7 0.00000+ 0112733  1
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  0 SEND
                                    (1274,2;0,2)
 6.01200+ 3 1.18969+ 1           0           0           0          1112733  2 HEAD
 0.00000+ 0 0.00000+ 0           0           2           1          1112733  2 CONT
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  2 CONT
 1.00000- 5 8.50000+ 6           0           0           8          4112733  2 LIST
 1.00000+ 0 1.00000+ 0-1.00000+ 0 4.00000+ 0-1.00000+ 0 1.02000+ 2112733  2
-1.00000+ 0 1.07000+ 0                                              112733  2
 0.00000+ 0 0.00000+ 0           0           1           8          4112733  2 LIST
 1.00000- 5 4.00000- 6 1.00000+ 3 8.00000- 6 2.00000+ 6 2.50000- 5112733  2
 2.00000+ 7 0.00000+ 0                                              112733  2
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  0 SEND
                                    (1274,4;0,4)
 6.01200+ 3 1.18970+ 1           0           0           0          1112733  4 HEAD
 0.00000+ 0 0.00000+ 0           0           4           1          1112733  4 CONT
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  4 CONT
 8.50000+ 6 2.00000+ 7           0           0           8          4112733  4 LIST
 1.00000+ 0 1.00000+ 0-1.00000+ 0 2.00000+ 0-1.00000+ 0 1.02000+ 2112733  4
-1.00000+ 0 1.07000+ 2                                              112733  4
 0.00000+ 0 0.00000+ 0           0           1          20         10112733  4 LIST
 1.00000- 5 0.00000+ 0 8.29600+ 6 1.00000- 2 8.45000+ 6 2.50000- 3112733  4
 8.95000+ 6 1.00000- 2 1.10000+ 7 4.00000- 2 1.20000+ 7 4.00000- 2112733  4
 1.30000+ 7 2.25000- 2 1.40000+ 7 1.00000- 2 1.50000+ 7 4.00000- 2112733  4
 2.00000+ 7 0.00000+ 0                                              112733  4
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  0 SEND
                                  (1274,102;0,102)
 6.01200+ 3 1.18969+ 1           0           0           0          1112733102 HEAD
 0.00000+ 0 0.00000+ 0           0         102           0          1112733102 CONT
 0.00000+ 0 0.00000+ 0           0           1           6          3112733102 LIST
 1.00000- 5 3.60000- 3 1.00000+ 3 4.00000- 2 2.00000+ 7 0.00000+ 0112733102
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  0 SEND
                                  (1274,107;0,107)
 6.01200+ 3 1.18969+ 1           0           0           0          1112733107 HEAD
 0.00000+ 0 0.00000+ 0           0         107           0          1112733107 CONT
 0.00000+ 0 0.00000+ 0           0           0           8          4112733107 LIST
 1.00000- 5 0.00000+ 0 6.32000+ 6 1.00000- 4 7.36000+ 6 0.00000+ 0112733107
 2.00000+ 7 0.00000+ 0                                              112733107
 0.00000+ 0 0.00000+ 0           0           0           0          0112733  0 SEND
```

Table 33.4

```
                                        (1274,51;0,51)
 6.01200+ 3 1.18969+ 1           0           0           0           1127433 51 HEAD
 0.00000+ 0 0.00000+ 0           0          51           1           1127433 51 CONT
 0.00000+ 0 0.00000+ 0           0           0           0           0127433 51 CONT
 1.00000- 5 8.50000+ 6           0           0           2           1127433 51 LIST
 1.00000+ 0 4.00000+ 0                                                127433 51
 0.00000+ 0 0.00000+ 0           0           1           6           3127433 51 LIST
 1.00000- 5 0.00000+ 0 8.29600+ 6 2.50000- 3 2.00000+ 7 0.00000+     0127433 51
 0.00000+ 0 0.00000+ 0           0           0           0           0127433  0 SEND
                                        (1274,91;0,91)
 6.01200+ 3 1.18969+ 1           0           0           0           1127433 91 HEAD
 0.00000+ 0 0.00000+ 0           0          91           1           0127433 91 CONT
 0.00000+ 0 0.00000+ 0           0           0           0           0127433 51 CONT
 8.50000+ 6 2.00000+ 7           0           0          10           5127433 51 LIST
 1.00000+ 0 1.00000+ 0-1.00000+ 0 2.00000+ 0-1.00000+ 0 5.10000+     1127433 51
-1.00000+ 0 1.02000+ 2-1.00000+ 0 1.07000+ 2                         127433 51
 0.00000+ 0 0.00000+ 0           0           0           0           0127433  0 SEND
```

APPENDIX A

Glossary

		Section
A_n	the mass of the n^{th} type atom, A_o is the mass of the principal scattering atom in the molecule.	7
$A_{k,\ell}$	probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the k^{th} to the ℓ^{th} level.	11
a_ℓ	ℓ^{th} Legendre coefficient.	4,6,14,6
a,b	parameters used in the Watt spectrum.	5
ABN	the abundance (weight fraction) of an isotope in this material.	2,32
AC_1, AC_2, AC_3, AC_4, BC_1, BC_2	the background constants for the Adler-Adler radiative capture cross section.	2
AF_1, AF_2, AF_3, AF_4, BF_1, BF_2	the background constants for the Adler-Adler fission cross section.	2
ALAB	Mnemonic of laboratory originating evaluation.	1.1
AJ	the compound nucleus spin, J (the spin of the resonance)(floating point).	2,32
AMUF	the number of degrees of freedom used in the fission width distribution.	2
AMUG	the number of degrees of freedom used in the radiation width distribution.	2
AMUN	the number of degrees of freedom used in the neutron width distribution.	2

		Section
AMUX	the number of degrees of freedom used in the competitive width distribution.	2
AT_1, AT_2, AT_3, AT_4, BT_1, BT_2	the background constants for the Adler-Adler total cross section.	2
AP	the scattering radius	2,32
AUTH	author of evaluation.	1.1
AWR	the ratio of the mass of the atom (or molecule) to that of the neutron.	All
AWRI	the ratio of the mass of the particular isotope to that of the neutron.	2,32
$B_i(N)$	the list of constants.	7,12
BR	the branching ratio for the production of a particular nuclide and level	8
CI	the coefficient of the reaction contributing to the sum (usually $\pm$ 1.)	31,33
$C_i(E_j)$	the array of triplets (EG, $T_{1/2}$, Y) in increasing order of EG of the photon energy half-life and yield data for the i^{th} residual nucleus at neutron energy E_j.	17
C_n	the coefficients of a polynomial. There are NC coefficients given.	1
$C_n(E_i)$	the array of yield data for the i^{th} energy point. There are 4 numbers per fission product.	8.2
CT	the chain indicator.	8.1
D	the mean level spacing for a particular J-state.	2

		Section
DDATE	original distribution date of the evaluation.	1.1
DET_r	the Adler-Adler resonance energy for the total cross section. Here and below, the subscript r denotes the r^{th} resonance.	2
DE^2	the variance of the resonance energy ER.	32
DEF_r	the resonance energy for the fission cross section.	2
DEC_r	the resonance energy for the radiative capture cross section.	2
DF^2	the variance of GF.	32
DG^2	the variance of GG.	32
DGDF	the covariance of GG and GF.	32
DJ^2	the variance of AJ.	32
DJDF	the covariance of AJ and GF.	32
DJDG	the covariance of AJ and GG.	32
DJDN	the covariance of AJ and GN.	32
DN^2	the variance of GN.	32
DNDF	the covariance of GN and GF.	32
DNDG	the covariance of GN and GG.	32
DWT_r	the value of $\Gamma/2$, (ν), used for the total cross section.	2
DWF_r	the value of $\Gamma/2$, (ν), used for the fission cross section.	2
DWC_r	the value of $\Gamma/2$, (ν), used for the radiative capture cross section.	2
$\frac{d\sigma(\Omega,E)}{d\Omega}$	differential scattering cross section in units of barns per steradian.	4

		Section
$\frac{d\sigma_k^\gamma}{d\Omega}$	differential photon production cross section in barns per steradian.	14
E	energy of the incident neutron.	All
E'	the secondary neutron energy (eV).	5,6,7
E_{avail}	available energy.	5
E_{int}	the interpolation scheme for each energy range. (Appendix E).	All
E_{th}	the threshold energy.	3
$E_{"x"}$	average decay energy (eV) of "x" radiation for decay heat applications.	8.3
EB	the total energy released by delayed β's.	1.5
EC	the lower energy cutoff (eV) for photons emitted at time T and neutron energy E.	18
EDATE	date of evaluation.	1.1
EFR	the kinetic energy of the fragments.	1.5
EG	the photon energy (eV).	17,18
$\bar{EG}$	the average energy of photons emitted at time T and neutron energy E	18
EG_k	the photon energy or Binding Energy.	12,13,14
EGD	the total energy released by the emission of delayed γ rays.	1.5
EGP	the total energy released by the emission of "prompt" γ-rays.	1.5
EI	the energy points where the weighting of the standard cross section is given.	31,33

		Section
EL,EH	the lower and upper limits for a resonance region energy range.	2,32
ELIS	excitation energy of the target nucleus.	1.1
ELFS	the excitation energy of the reaction product.	8
END	the kinetic energy of the delayed fission neutrons.	1.5
ENDATE	the master file entry date (YYMMDD)	1.1
ENP	the kinetic energy of the "prompt" fission neutrons.	1.5
ENU	the energy carried away by neutrinos.	1.5
ER	the total energy release due to fission minus the neutrino energy.	1.5
ER	the resonance energy (in the laboratory system).	2,32
ER	the energy (eV) of radiation produced.	8
$\overline{ER}$	the average decay energy of radiation produced.	8.3
ES_i	energy of the i^{th} level.	12
ES_k	the energy of the level from which the photon originates.	12,13,14
ES(N)	the energy of the N^{th} point used to tabulate energy-dependent widths.	2
ET	the total energy release due to fission.	1.5
E1,E2	a range of neutron energies.	31,33
F(q;Z)	The form factor for coherent photon scattering.	27
$f_k(E \rightarrow E')$	the k^{th} partial energy distribution. The definition depends on the value of LF.	5

		Section
$f_k(E_\gamma \leftarrow E)$	A normalized (to unity) photon energy distribution (or probability density) function at incident neutron energy E for the kth subsection within a reaction type (eV^{-1}).	15
FC	the continuum spectrum normalization factor.	8.3
FD	the discrete spectrum normalization factor.	8.3
FPS	the state designator (floating-point number) for a fission product nuclide.	8.2
GRT_r	related to the symmetrical total cross section parameter.	2
GIT_r	related to the asymmetrical total cross section parameter.	2
GRF_r	the symmetrical fission parameter.	2
GIF_r	the asymmetrical fission parameter.	2
GRC_r	the symmetrical capture parameter.	2
GIC_r	the asymmetrical capture parameter.	2
GG	the average radiation width. It is energy dependent if LRU = 2.	2
GF	the average fission width. This value may be energy dependent.	2
GX	the average competitive reaction width.	2
GNO	the average reduced neutron width. It is energy dependent.	2
$GP_{j,i} \equiv GP_i$	The conditional probability of photon emission in a direct transition from level j to level i, i < j.	12

		Section
GT	the resonance total width Γ evaluated at the resonance energy ER.	2,32
GN	the neutron width Γ_n evaluated at the resonance energy ER.	2,32
GG	the radiation width Γ_γ pvaluated at the resonance energy ER.	2,32
GF	the fission width Γ_f evaluated at the resonance energy ER.	2,32
GX	the competitive width Γ_x evaluated at the resonance energy ER	2
$g(E_\gamma \leftarrow E)$	A particular class of the functions $q_j(E_\gamma \leftarrow E)$ in File 15; those which are tabulated (eV^{-1}).	15
H(q;Z)	A general symbology for a form factor or incoherent scattering function; either F(q;Z) or S(q;Z), respectively.	27
H(N)	the array containing the Hollerith information that describes the particular evaluated data set.	1.1
$h(E_\gamma \leftarrow E,\mu)$	A normalized (to unity) energy-angle distribution function for photon production (eV^{-1}).	16
HL	the half life of the reaction product.	8
I	the normalizing denominator (see 5.3).	5
I_i	the interpolation scheme (see Appendix E) to be used between the E_{i-1} and E_i energy points.	8.2
I_i	the interpolation scheme for the yields in the $R(E_i)$ array at the i^{th} neutron energy.	18

		Section
INT	the interpolation scheme to be used for interpolating between the cross-sections obtained from average resonance parameters.	2
INT(m)	the interpolation scheme identification number used in the m^{th} range.	0
J_i	the interpolation scheme for the average photon energies in the $R(E_i)$ array of the i^{th} neutron energy.	18
k	Boltzmann's constant.	7
L	the value of the ℓ-state (neutron angular momentum quantum number).	2,32
LA	the value of ℓ (for the ℓ^{th} coefficient).	6
LAT	a flag indicating which temperature has been used to compute α and β .	7
LCON	the continuum spectrum flag.	8.3
LCT	a flag indicating which reference frame is used for both secondary angles and energies.	4,6
LB	a flag which determines the meanings of the numbers in the arrays $\{E_k,F_k\}\{E_\ell,F_\ell\}$.	31,33
LE	a test to determine whether energy-dependent fission product yields are given.	8.2
LF	a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution).	5,6,12,13,15,16
LF	a flag to specify that half-life and yield data are given	17

		Section
LFI	a flag that indicates whether this material is fissionable.	1.1
LFS	an indicator that specifies the final excited state of the residual nucleus produced by a particular reaction.	3,8,9,10
LFT	a flag for specifying the option used to represent the continuous spectra of delayed photon emission.	18
LFW	a flag indicating whether average fission widths are given in the unresolved resonance region for this isotope.	2
LG	The transition probability array flag for distinguishing between doublet and triplet arrays in File 12.	12
LI	a flag to indicate the kind of Adler-Adler parameters given. The isotropy flag in File 4 and 14.	2,4,14
LIS	state number of the target nucleus (for materials that represent nuclides).	1,3,8,9,10
LISØ	the isomeric state number of the target nucleus.	1,8
LMF	the file number for this MT containing the multiplicity or cross section.	8
LNU	a test that indicates what representation of $\bar{\nu}(E)$ has been used.	1
LØ	The option flag to determine whether multiplicities or transition probability arrays are to be given in File 12.	12

		Section
LP	Indicator of whether or not the particular photon is a primary.	12,13
LR	a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x). (See Section 3.2.4.4.)	3
LRF	a flag indicating which resonance parameter representation has been used for this energy range. The definition of LRF depends on the value of LRU for this energy range.	2,32
LRP	a flag that indicates that resolved and/or unresolved resonance parameters are given in File 2.	1.1
LRU	a flag indicating whether an energy range contains data for resolved or unresolved resonance parameters.	2
LRX	a flag indicating whether a competitive width is given.	2
LS	a flag indicating whether the $F_{k,k'}$ matrix is asymmetric or symmetric. LB=5	31,33
LT	a flag to specify whether temperature-dependent data are given.	3,4,5,6,7
LT	Temperature dependence (see Appendix F).	0
LT	the number of pairs of numbers in the array $\{E_\ell, F_\ell\}$	31,33
LTT	a flag to specify whether Legendre or probability representation is used.	4,6,14,16
LTY	the flag used in "NC-type" sub-subsections to indicate the procedure used to obtain the covariance matrix.	31,33

		Section
LVT	a flag to specify whether a transformation matrix is given for elastic scattering.	4
L1	an integer to be used as a flag or a test.	1
L2	an integer to be used as a flag or a test.	1
M_n	the number of atoms of the nth type in the molecule.	7
MAT	Material number.	0
MATP	the material number for the reaction product.	8
MF	File number.	0
MT	Reaction type number.	0
MUF	the integer value of the number of degrees of freedom for fission widths.	2
MF_n	the MF of the n^{th} section.	1.1
MT_n	the MT of the n^{th} section.	1.1
MOD_n	the modification indicator for section MF_n and MT_n	1.1
NA	the number of angles (cosines) at which the secondary distributions are given.	6,16
NB	the total number of β values given.	7
NB	the number of photon energies or energy bins.	18
NBT(n)	the value of N separating the m^{th} and $(m+1)^{th}$ interpolation ranges.	0
NC	the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$.	15,16
NC	a count of the number of terms used in the polynomial expansion.	1
NC	the number of "NC-type" sub-subsections.	33

		Section
NC_n	the number of BCD card images in a given section (the n^{th} section).	1.1
NCI	the number of reactions summed to obtain the reaction of interest.	31,33
ND	the number of branches into which the reaction product decays.	8.1
NDK	total number of decay modes given.	8.3
NE	number of neutron energy points given in a TAB2 record.	14,15,17,18
NE	the number of energy points at which energy-dependent widths are tabulated.	2
NE	the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given.	4,5
NEI	the number of energy points in the {EI,WEI} list.	33
NER	the number of energy ranges given for this isotope.	2
NER	the total number of discrete energies for a given spectral type (STYP).	8.3
NF	the number of secondary energy points in a tabulation.	5
NFP	the number of fission product nuclide states to be specified at each incident energy point.	8.2
NG_i	the number of discrete photons for nucleus ZA_i.	17
NI	the total number of items in the B(N) list. NL = 6* (NS + 1)	7

		Section
NI	number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.	14
NI	the number of "NI-type" sub-subsections.	31,33
NIS	the number of isotopes in this material.	2,32
NJS	the number of sets of resolved resonance parameters (each having the same J state) for a specified ℓ-state.	2
NK	the number of partial energy distributions. There will be one subsection for each partial distribution.	5,6
NK	the number of elements in the transformation matrix $NK = (NM + 1)^2$.	4
NK	the number of discrete photons plus the photon continuum	12,13,14
NL	the highest order Legendre polynomial that is given at each energy.	4,6,14,16
NL	the number of subsections within a section.	33
NLIB	library identifier.	1.1
NLJ	the count of the number of levels for which parameters will be given.	2
NLS	the number of ℓ-values considered. A set of resonance parameters is given for each ℓ-value.	2,32
NM	the maximum order Legendre polynomial that will be required to describe the angular distributions.	4

		Section
NMOD	modification number.	1.1
NN	the number of elements in the LIST record	8.2,17,18
NNF	the number of precursor families considered.	1.4
NP	the number of points in a tabulation of y(x) that are contained in the same record.	All
NP	the total number of pairs of numbers in the arrays $\{E_k, F_k\}\{E_\ell, F_\ell\}$.	31,33
NR	the number of different interpolation intervals in a tabulation of y(x) that are contained in the same record.	All
NRS	the number of resolved resonances for a given ℓ-state.	2,32
NS	the number of non-principle scattering atom types	7
NS	number of levels below the present one, including the ground state.	12
NS	the number of states of the radioactive reaction product.	8,9,10
NSP	total number of spectra radiation types (STYP) given.	8.3
NT	the number of entries for each discrete energy (ER).	8.3
NT	number of transitions for which data are given in a list to follow.	12
NT	the number of times at which yield or spectra are given.	18
NT_i	the number of times at which data are given for the i^{th} neutron energy.	18

		Section
NWD	the count of the number of elements in the Hollerith section.	1.1
NX	the count of the number of sets of background constants to be given.	2
NXC	an integer count of all the sections to be found in the dictionary.	1.1
NZ	the number of residual nuclei for which discrete photon data are given.	17
NØ	the flag to identify a reaction product which is radioactive in its ground state.	8
N1	an integer to be used as a count of items in a list to follow except for MT 451.	1
N2	an integer to be used as a count of items in a second list to follow.	1
$P_j(E)$	the probability or weight given to the j^{th} partial distribution, $q_j(E_\gamma \leftarrow E)$.	15
$P_k(E_N)$	the fractional part of the particular cross section that can be described by the k^{th} partial energy distribution at the N^{th} incident energy point.	5.2
P(E,T,EG)	the probability at neutron energy E for the emission of a photon of energy EG at time T.	18
$p(\mu,E)$	$\frac{2\pi}{\sigma_s(E)} \frac{d\sigma}{d\Omega} (\Omega,E)$	4
PAR	the parity of the target nuclide.	8.3
Q	the reaction Q-value (eV). Q = (rest mass of initial state - rest mass of final state.)	3,9,10

		Section
Q	Total decay energy (eV) available in the corresponding decay process. (This is not necessarily the same as the maximum energy of the emitted radiation.)	8.3
QX	an effective Q value for the competitive width.	2,32
$q_j(E_\gamma \leftarrow E)$	the j^{th} normalized partial distribution in the units eV^{-1}.	15
$R(E_i)$	the array of triplets of time, yield, and average photon energy for the i^{th} neutron energy.	18
RDATE	date and number of last revision REV1 - followed by month-year as in EDATE	1.1
REF	reference to evaluation	1.1
RFS	Isomeric state flag for daughter nuclide.	8.3
RI	the intensity of radiation produced (relative units).	8.3
RICC	the total internal conversion coefficient.	8.3
RICK	the K-shell internal conversion coefficient.	8.3
RICL	the L-shell internal conversion coefficient.	8.3
RIS	the internal pair formation coefficient (STYP=0.0) or positron intensity (STYP=2.0).	8.3
RP	the spectrum of the continuum component of the radiation $\int RP(ER)\,dER = 1$.	8.3
RTYP	the mode of decay of the nuclide in its LISØ state.	8
S	the temperature (°K). NOTE: If the LR flag is used, S becomes Q_1 for the reaction corresponding to LR.	3

		Section
$S(\alpha,\beta,T)$	defined (for a moderating molecule) by the relation $$\frac{d^2\sigma}{d\Omega\, dE'}(E \leftarrow E', \mu, T) = \sum_{n=0}^{NS} \frac{M_n \sigma_{bn}}{4\ell T} \frac{E'}{E} e^{-\beta/2} - S_n(\alpha,\beta,T).$$	7
SPI	the nuclear spin of the target nucleus, I (positive number).	2,8,32
STA	the target stability flag.	1.1
STYP	the decay radiation type (defined in 8.3).	8.3
T	the temperature (°K) at which temperature dependent data is given.	4,5,6,7
T	the time after the primary interaction (secs.).	18
$T_{1/2}$	half-life of the original nuclide (seconds).	8.3
$T_{1/2}$	the half-life emission of a gamma ray from a given residual nucleus.	17
TP_i	the probability of a direct transition from level NS to level i, i=0,1,2,...(NS-1).	12
TYPE	a flag indicating the type of transition for beta and electron capture.	8.3
U	a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).	5
V_k	the matrix elements of the transformation matrices.	4
WEI	the weighting of the standard cross section at a given EI to the next given energy.	31,33
x	$E'/\theta(E)$	5
x(n)	the n^{th} value of x.	0.1

		Section
XMTI	the floating point equivalent of the MT number of the reaction contributing to the sum.	31,33
Y	the yield of photons per interaction.	17
YC,DYC	the cumulative yield and 1 σ uncertainty for a particular fission product.	8.2
Y(E)	the total multiplicity at energy E(eV). Data are given as energy-multiplicity pairs.	9,12
Y(E)	the partial multiplicity at energy E(eV).	12
Y(E,T)	the yield of photons per second per interaction at time T and neutron energy E.	18
YI,DYI	the fractional (independent) yield and 1 σ uncertainty for a particular fission product.	8.2
y(n)	the n^{th} value of y.	0.1
ZA	the designation of the original nuclide (ZA = (1000.0*Z) + A)	All
ZAI	the (Z,A) designation for an isotope.	2,32
ZA_i	the ZA-number of the i^{th} residual nucleus.	17
ZAFP	the (Z,A) identifier for a particular fission product. (ZAFP = (1000.0*Z) + A).	8.2
ZAN	the (Z,A) designation of the next nuclide in the chain.	8.1
ZAP	the (Z,A) designation of the product nuclide (ZAP = (1000.0*Z) + A).	8.1
ZSYMA	a Hollerith representation of the material Z-chemical symbol - A.	8.1
α	the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$.	7
β	the energy transfer, $\beta = (E' - E)/kT$.	7

		Section
Δ	the uncertainty in a quantity.	1.6,8.3
$\delta(E_\gamma - \varepsilon_j + \varepsilon_i)$	delta function with ε_j, ε_i being energy levels of the residual nucleus.	11
$\bar{\nu}(E)$	the total average number of neutrons formed per fission event.	1
λ_i	the decay constant (sec^{-1}) for the i^{th} precursor.	1.4
θ	a parameter used to describe the secondary energy distribution. The definition of θ depends on the energy distribution law (LF).	5
$\sigma(E)$	the cross section (barns) for a particular reaction type at incident energy point, E, in (eV).	3,10
σ_{bn}	the bound atom scattering cross section of the n^{th} type atom, $\sigma_{bn} = \sigma_{fn}\left(\frac{A_n + 1}{A_n}\right)^2$	7
σ_{fn}	the free atom scattering cross section of the n^{th} type atom.	7
$\sigma_k^\gamma(E)$	photon production cross section for the discrete photon or photon continuum specified by k.	13
$\sigma_{m_o}(E)$	neutron cross sections for exciting the m_o^{th} level with neutron energy E.	11
$\sigma_s(E)$	the scattering cross section, e.g., elastic scattering at energy E as given in File 3 for the particular reaction type (MT).	4
σ_T (background)	$\frac{C}{\sqrt{E}}(AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1*E + BT_2*E^2)$.	2

		Section
σ_w	Wick's limit cross section in units of barns per steradian.	4
μ	cosine of the scattered angle in either the laboratory or the center-of-mass system.	4,6,14,16

APPENDIX B

Definition of Reaction Types

Reaction types are identified by an integer, MT. The allowed reaction types are listed below. The reaction type number (MT) generally refers to a specific neutron-nucleus interaction mechanism, but occasionally it indicates that a particular type of information is given. The general rules for assignment of MT numbers are

MT (range)	Description of Class of Reactions
1-100	Reaction types in which secondary particles of the same type as the incident particles are emitted
101-150	Reaction types in which no secondary particles of the same type as the incident particles are emitted
151-200	Resonance region information
201-450	Quantities derived from the basic data
451-699	Miscellaneous quantities
700-799	Excitation cross sections for reactions that emit charged particles
800-999	(not assigned)

The specific MT assignments are given in the table below. For the most part, they are consistent with those used in the UKAEA Nuclear Data File.

MT	Description
1	Total cross section (redundant, equal to the sum of all partial cross sections)
2	Elastic scattering cross section
3	Nonelastic cross section (redundant, equal to the sum of all partial cross sections except elastic scattering)
4	Total inelastic cross section (redundant, equal to the sum of MT = 51, 52, 53, ..., 90, 91)

MT	Description
5	(to be assigned)
6	(n,2n) cross section for first excited state (describes first neutron)
7	(n,2n) cross section for second excited state (describes first neutron)
8	(n,2n) cross section for third excited state (describes first neutron)
9	(n,2n) cross section for fourth excited state (describes first neutron)
10-15	(to be assigned)
16	direct (n,2n) cross section (total (n,2n) cross section is sum of MT = 6, 7, 8, 9, and 16)
17	(n,3n) cross section
18	Total fission cross section (sum of MT = 19, 20, 21, 38)
19	(n,f) cross section (first chance fission)
20	(n,n'f) cross section (second chance fission)
21	(n,2nf) cross section (third chance fission)
22	(n,n'α) cross section
23	(n,n'3α) cross section
24	(n,2nα) cross section
25	(n,3nα) cross section
26	(n,2n) isomeric state cross section
27	Absorption cross section (sum of MT = 18 and 101) (includes particle reactions)
28	(n,n'p) cross section
29	(n,n'2α) cross section
30	(n,2n 2a) cross section
31	to be used as LR flag only*
32	(n,n'd) cross section

MT	Description
33	(n,n't) cross section
34	(n,n'^{3}He)
35	(n,n'd2α) cross section
36	(n,n't2α) cross section
37	(n,4n) cross section
38	(n,3nf) cross section (fourth chance fission)
39	To be used as LR flag only*
40	To be used as LR flag only*
41-45	(to be assigned)
46	cross section for describing the second neutron form (n,2n) reaction for first excited state
47	cross section for describing the second neutron from (n,2n) reaction for second excited state
48	cross section for describing the second neutron from (n,2n) reaction for third excited state

*The following MT numbers are used only as LR flags in order to indicate the mode of decay of the residual nucleus:

LR	Description
31	Indicates that γ-emission is the mode of decay of the residual nucleus formed in the primary reaction.
39†	Indicates that internal conversion is the mode of decay of the residual nucleus formed in the primary reaction.
40†	Indicates that electron-positron pair formation is the mode of decay of the residual nucleus formed in the primary reaction.

(The "primary" reaction could be, for example, an (n,n'), (n,p), (n,α), (n,np), etc. reaction.)

†For local heating calculations, S ≡ 0.0.

MT	Description
49	cross section for describing the second neutron from (n,2n) reaction for fourth excited state (Note: MT = 46, 47, 48 and 49 should not be included in the sum for the total (n,2n) cross section)
50	(to be assigned)
51	(n,n) to the first excited state
52	(n,n) to the second excited state
. . .	
90	(n,n) to the 40th excited state
91	(n,n') to the continuum
92-100	(to be assigned)
101	neutron disappearance (sum of all cross sections in which a neutron is not in the exit channel). $MT = 101 \text{ is } \sum_{i=2}^{14} (MT=100+i)$
102	(n,γ) radiative capture cross section
103	(n,p) cross section
104	(n,d) cross section
105	(n,t) cross section
106	(n,^{3}He) cross section
107	(n,α) cross section
108	(n,2α) cross section
109	(n,3α) cross section
110	(to be assigned)
111	(n,2p) cross section
112	(n,pα) cross section
113	(n,t2α) cross section

MT	Description
114	(n,d2α) cross section
115-119	(to be assigned)
120	Target destruction = nonelastic less total (n,n'γ)
121-150	(to be assigned)
151	General designation for resonance information
152-200	(to be assigned for specific resonance information)
201-202	(to be assigned)
203	Total hydrogen production
204	Total deuterium production
205	Total tritium production
206	Total ^{3}He production
207	Total ^{4}He production
208-250	(to be assigned)
251	$\bar{\mu}_L$, the average cosine of the scattering angle (laboratory system) for elastic scattering
252	ξ, the average logarithmic energy decrement for elastic scattering
253	γ, the average of the square of the logarithmic energy decrement for elastic scattering, divided by twice the average logarithmic decrement for elastic scattering
254-300	(to be assigned)
301-450	Energy release rate parameters, $\overline{E^*\sigma}$, for total and partial cross sections. Subtract 300 from this number to obtain the specific reaction type identification. For example, MT = 302 = (300 + 2) denotes elastic scattering
451	Heading or title information (given only in File 1)
452	$\bar{\nu}$, average total (prompt plus delayed) number of neutrons released per fission event
454	Independent fission product yield data
455	Delayed neutrons from fission

MT	Description
456	Prompt neutrons from fission
457	Radioactive decay data
458	Energy Release in fission
459	Cumulative fission product yield data
465	Delayed neutrons from spontaneous fission
466	Prompt neutrons from spontaneous fission
467-500	(to be assigned)
501	Total photon interaction cross section
502	Photon coherent scattering
503	(to be assigned)
504	Photon incoherent scattering
505-514	(to be assigned)
515	Pair production, electron field
516	Pair production, nuclear and electron field (i.e., pair plus triplet production)
517	Pair production, nuclear field
518	Photofission (γ,f)
519-531	(to be assigned)
532	Photoneutron (γ,n)
533	Total photonuclear
534-601	(to be assigned)
602	Photoelectric
603-699	(to be assigned)
700	(n,p_0) cross section (cross section for leaving the residual nucleus in the ground state)
701	(n,p_1) cross section for 1st excited state
702	(n,p_2) " " " 2nd "

MT	Description
703	(n,p_3) " " " 3rd "
704	(n,p_4) cross section for 4th excited state
.	
.	
.	
718	(n,p_c) " " " continuum excited state
719	(n,p_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing proton)
720	(n,d_0) cross section for ground state
721	(n,d_1) cross section for 1st excited state
722	(n,d_2) cross section for 2nd excited state
.	
.	
.	
738	(n,d_c) cross section for continuum excited state
739	(n,d_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing deuteron)
740	(n,t_0) cross section for ground state
741	(n,t_1) " " " 1st excited state
742	(n,t_2) " " " 2nd " "
.	
.	
.	
750	(n,t_c) " " " continuum excited state
759	(n,t_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing triton)
760	$(n,{}^3He_0)$ cross section for ground state
761	$(n,{}^3He_1)$ cross section for 1st excited state
.	
.	
.	
778	$(n,{}^3He_c)$ cross section for continuum
779	$(n,{}^3He_c)$ cross section for continuum specifically not included in σ total (redundant, used for describing outgoing ^3He)
780	(n,α_0) cross section for ground state

MT	Description
781	(n,α_1) cross section for 1st excited state
.	
.	
.	
798	(n,α_c) cross section for continuum
799	(n,α_c') cross section for continuum specifically not included in σ_T (redundant, used to describe outgoing α)
800-999	(to be assigned)

APPENDIX C

ZA Designation of Materials

A floating point number, ZA, is used to identify materials. If Z is the charge number and A the mass number, then ZA is computed from

$$ZA = (1000.0*Z) + A$$

For example, ZA for ^{238}U is 92238.0, and ZA for beryllium is 4009.0. For materials other than isotopes, the following rules apply.

(1) If the material is an element that has more than one naturally occurring isotope, then A is set to 0.0. For example, ZA for the element tungsten is 74000.0.

(2) For all other types of material, Z is set to zero, and the appropriate ZA is given in the following table. For example, ZA for H_2O is given as 100.0. The following classifications apply.

ZA (range)	Class of Materials
1-99	Hypothetical materials
100-199	Liquid moderators and coolants
200-299	Solid moderators
300-399	Metal alloys, cladding, and structural materials
400-499	Lumped fission products

Table of Appropriate ZA Designations

ZA	Material
1	Pure 1/v absorber. σ_{abs} (2200 m/sec) = 1.0
2	Pure scatterer. $\sigma_s(E) = 1.0$
3-99	(to be assigned)
100	Water, H_2O
101	Heavy water, D_2O

ZA	Material
102	Biphenyl, $C_{12}H_{10}$
103	Sodium hydroxide, NaOH
104	Santowax R, $C_{18}H_{14}$
105	Dowtherm A
106	Benzene
107-199	(to be assigned)
200	Beryllium oxide, BeO
201	Beryllium carbide, Be_2C
202	Beryllium fluoride, BeF_2
203	Zirconium hydride, ZrH_x
204	Polystyrene, $(CH)_n$
205	Polyethylene $(CH_2)_n$
206-300	(to be assigned)
301	Zircalloy 1
302	Zircalloy 2
303	Zircalloy 4
304	304-type stainless steel
305-309	(to be assigned)
310	Uranium dioxide, UO_2
311-314	(to be assigned)
315	Uranium carbide, UC
316	316-type stainless steel
317-327	(to be assigned)
328	Inconel 718
329-399	(to be assigned)
400	^{233}U fission products (rapidly saturating) for thermal reactors

ZA	Material
401	^{235}U fission products (rapidly saturating) for thermal reactors
402	^{239}U " " " " " " "
403	^{241}Pu " " " " " " "
404	^{232}Th fission products (rapidly saturating) for thermal reactors
405	^{238}U " " " " " " "
406	^{240}Pu " " " " " " "
407-409	(to be assigned)
410	^{233}U fission products (slowly saturating) for thermal reactors
411	^{235}U " " " " " " "
412	^{239}Pu " " " " " " "
413	^{241}Pu " " " " " " "
414	^{232}Th " " " " " " "
415	^{238}U " " " " " " "
416	^{240}Pu " " " " " " "
417-419	(to be assigned)
420	^{233}U fission products (nonsaturating) for thermal reactors
421	^{235}U " " " " " "
422	^{239}Pu " " " " " "
423	^{241}Pu " " " " " "
424	^{232}Th " " " " " "
425	^{238}U " " " " " "
426	^{240}Pu " " " " " "
427-429	(to be assigned)
430	^{233}U fission products (rapidly saturating) for fast reactors
431	^{235}U " " " " " " "
432	^{239}Pu " " " " " " "

ZA	Material
433	^{241}U fission products (rapidly saturating) for fast reactors
434	^{232}Th " " " " " " "
435	^{238}U " " " " " " "
436	^{240}Pu fission products (rapidly saturating) for fast reactors
437-439	(to be assigned)
440	^{233}U fission products (slowly saturating) for fast reactors
441	^{235}U " " " " " " "
442	^{239}Pu " " " " " " "
443	^{241}Pu " " " " " " "
444	^{232}Th " " " " " " "
445	^{238}U " " " " " " "
446	^{240}Pu " " " " " " "
447-449	(to be assigned)
450	^{233}U fission products (non-saturating) for fast reactors
451	^{235}U " " " " " "
452	^{239}Pu " " " " " "
453	^{241}Pu " " " " " "
454	^{232}Th " " " " " "
455	^{238}U " " " " " "
456	^{240}Pu " " " " " "
457-499	(to be assigned)

APPENDIX D

Resonance Region Formulae*

D.1. THE RESOLVED RESONANCE REGION

D.1.1. Single-Level Breit-Wigner Formula: LRU=1, LRF=1

The formulae appearing in Gregson et al.,[1] which omit the resonance-resonance interference terms, are adopted. These formulae, written in the laboratory system for all ℓ-values and without Doppler broadening, are (for a particular isotope)

1. Elastic Scattering Cross Section

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS-1} \sigma^{\ell}_{n,n}(E) \quad ,$$

where

$$\sigma^{\ell}_{n,n}(E) = (2\ell + 1)\frac{4\pi}{k^2}\sin^2\phi_{\ell}$$

$$+ \frac{\pi}{k^2}\sum_{J} g_J \sum_{r=1}^{NR_J} \frac{\Gamma^2_{nr}\cos 2\phi_P - 2\Gamma_{nr}(\Gamma_{\gamma r}+\Gamma_{fr})\sin^2\phi_{\ell} + 2(E-E'_r)\Gamma_{nr}\sin 2\phi_{\ell}}{(E-E'_r)^2 + \frac{1}{4}\Gamma^2_r}$$

*Several processing codes have been developed to calculate cross sections with use of the formulae given here. These codes are given in Appendix I.

(1) K. Gregson, M.F. James, and D.S. Norton, "MLBW - A Multilevel Breit-Wigner Computer Programme", UKAEA Report AEEW-M-517, March 1965.

The hard-sphere phase shifts ϕ_ℓ, the wave number k, the primed resonance energy E_r', the neutron width Γ_{nr}, and through it the total width Γ_r, are all functions of energy, $\phi_\ell(E)$, $k(E)$, $E_r'(E)$, $\Gamma_{nr}(E)$, and $\Gamma_r(E)$, but this dependence is not shown explicitly. Also, each resonance parameter carries the implicit quantum numbers ℓ and J, determined by the appropriate flags in the ENDF/B file.

2. Radiative Capture Cross Section

$$\sigma_{n,\gamma}(E) = \sum_{\ell=0}^{NLS-1} \sigma^{\ell}_{n,\gamma}(E),$$

where

$$\sigma^{\ell}_{n,\gamma}(E) = \frac{\pi}{k^2} \sum_{J} g_J \sum_{r=1}^{NR_J} \frac{\Gamma_{nr}\Gamma_{\gamma r}}{(E-E_r')^2 + \frac{1}{4}\Gamma_r^2}$$

and $\Gamma_{\gamma r}$ is the radiative capture width.

3. Fission Cross Section

$$\sigma_{n,f}(E) = \sum_{\ell=0}^{NLS-1} \sigma^{\ell}_{n,f}(E) \quad ,$$

where

$$\sigma^{\ell}_{n,f}(E) = \frac{\pi}{k^2} \sum_{J} g_J \sum_{r=1}^{NR_J} \frac{\Gamma_{nr}\Gamma_{fr}}{(E-E'_r)^2 - \frac{1}{4}\Gamma^2_r}$$

and Γ_{fr} is the fission width.

4. The Competitive Reaction Cross Section The competitive reaction cross section, $\sigma_{n,x}(E)$, is given in terms of analogous formulas involving Γ_{xr} the competitive width. By convention, the cross section for the competitive reaction is given entirely in File 3, and is not to be computed from the resonance parameters. The reason for this is that the latter calculation can be done correctly only in the case of a single competitive channel, since the file can define only a single competitive width.

The statistical factor $g_J = (2J+1)/2(2I+1)$ is obtained from the target spin I and the resonance spin J given in File 2 as SPI and AJ, respectively.

The sum on ℓ extends over all ℓ-states described. There will be NLS terms in the sum. NLS is given in File 2 for each isotope. It is important for the evaluator to provide, and processing codes to include, contributions from ℓ-values for which there are no resonances, in order that the potential scattering be correctly calculated.

The sum on J extends over all possible J-states for a particular ℓ-state. NR is the number of resonances for a given pair of ℓ and J values.

$$NRS = \sum_{J} NR_J$$

NRS is given in File 2 for each ℓ-value.

$\Gamma_{nr}(|E_r|) \equiv GN_r$ is the neutron width, for the r^{th} resonance for a particular value of ℓ, evaluated at the resonance energy E_r. For bound levels, the absolute value $|E_r|$ is used.

$$\Gamma_{nr} = \frac{P_\ell(E)\Gamma_{nr}(|E_r|)}{P_\ell(|E_r|)}$$

$\Gamma_r = \Gamma_{nr}(E) + \Gamma_{\gamma r} + \Gamma_{fr} + \Gamma_{xr}$ is the total width, a function of energy through Γ_{nr} and (possibly) Γ_{xr} since $\Gamma_{\gamma r}$ and Γ_{fr} are assumed constant with respect to energy. The "competitive" width, Γ_{xr} is not entered explicitly in File 2. It is calculated implicitly from the equation:

$$\Gamma_{xr} = \Gamma_r - \Gamma_{nr} - \Gamma_{\gamma r} - \Gamma_{fr} \quad \text{at } E_r$$

The following quantities are given in File 2 for each resonance:

E_r = ER, the resonance energy

J = AJ, the spin of the resonance state

$\Gamma_{nr}(|E_r|)$ = GN, the neutron width

$\Gamma_{\gamma r}$ = GG, the radiation width

Γ_{fr} = GF, the fission width and

$\Gamma_r(|E_r|)$ = GT, the total width evaluated at the resonance energy.

Since the competitive width, Γ_{xr}, is not given, Γ_r should be obtained from File 2 directly, and <u>not</u> by summing partial widths. Γ_{xr}, if non-zero, should be obtained by subtraction.

For p-, d-, and higher ℓ-values, the primed resonance energy E_r' is energy-dependent:

$$E_r' = E_r + \frac{S_\ell(|E_r|) - S_\ell(E)}{2P_\ell(|E_r|)} \Gamma_{nr}(|E_r|)$$

although the fact that the shift is zero at <u>each</u> E_r is an artifact of the SLBW formalism, and implies a different R-matrix boundary condition for each resonance.

$$k = 2.196771 \frac{AWRI}{AWRI + 1.0} \times 10^{-3}\sqrt{E},$$

is the neutron wave number in the center-of-mass system, in terms of the laboratory energy, and AWRI is the ratio of the mass of a particular isotope to that of the neutron. E is the incident neutron energy (Laboratory system, eV); S_ℓ is the shift factor,

$$S_o = 0$$

$$S_1 = -\frac{1}{1 + \rho^2}$$

$$S_2 = -\frac{18 + 3\rho^2}{9 + 3\rho^2 + \rho^4}$$

P_ℓ is the penetration factor,

$$P_o = \rho$$

$$P_1 = \frac{\rho^3}{1 + p^2}$$

$$P_2 = \frac{\rho^5}{9 + 3\rho^2 + \rho^4}$$

where $\rho = ka$ and "a" is the channel radius (in units of 10^{-12} cm), defined as

$$a = .123\ AWRI^{1/3} + .08;\ AWRI = A/1.00865^*$$

ϕ_ℓ is the (negative of a) hard-sphere phase shift,

$$\phi_0 = \hat{\rho}$$

$$\phi_1 = \hat{\rho} - \tan^{-1}\hat{\rho}$$

$$\phi_2 = \hat{\rho} - \tan^{-1}\frac{3\hat{\rho}}{3-\hat{\rho}^2},$$

where $\hat{\rho} = k\,(AP)$ and AP is the scattering radius, given in File 2, which determines the low energy potential scattering cross section.

D.1.2. Multilevel Breit-Wigner Formula: LRU=1, LRF=2

The equations are exactly the same as above, except that a resonance-resonance interference term is included in the equation for elastic scattering of ℓ-wave neutrons, $\sigma^{\ell}_{n,n}(E)$:

$$\frac{\pi}{k^2}\sum_J g_J \sum_{r=2}^{NR_J}\sum_{s=1}^{r-1} \frac{2\Gamma_{nr}\Gamma_{ns}\left((E-E'_r)(E-E'_s) + \frac{1}{4}\Gamma_r\Gamma_s\right)}{\left((E-E'_r)^2 + \frac{1}{4}\Gamma_r^2\right)\left((E-E'_s)^2 + \frac{1}{4}\Gamma_s^2\right)} . \qquad (1)$$

This form, which as $\sim N^2/2$ energy-dependent terms and can involve a great deal of computer time, can be written in the following form with only N terms: (See Section 2.4.15)

*A is the target mass in amu. The channel radius, strictly speaking, involves $A^{1/3}$, and not $(AWRI)^{1/3}$, but as long as the mass of the incident particle is approximately unity, as it is for neutrons, the difference is not important.

$$\frac{\pi}{k^2} \sum_{J} g_J \sum_{r=1}^{NR_J} \frac{G_r \Gamma_r + 2H_r(E-E'_r)}{(E-E'_r)^2 + (\Gamma_r/2)^2} \tag{2}$$

where

$$G_r = \frac{1}{2} \sum_{\substack{s=1 \\ (s\neq r)}}^{NR_J} \frac{\Gamma_{nr}\Gamma_{ns}(\Gamma_r+\Gamma_s)}{(E'_r-E'_s)^2 + \frac{1}{4}(\Gamma_r+\Gamma_s)^2} , \tag{3}$$

$$H_r = \sum_{\substack{s=1 \\ (s\neq r)}}^{NR_J} \frac{\Gamma_{nr}\Gamma_{ns}(E'_r-E'_s)}{(E'_r-E'_s)^2 + \frac{1}{4}(\Gamma_r+\Gamma_s)^2} . \tag{4}$$

For the user who does not require ψ- and χ-broadening, the following equations, which are mathematically identical to the MLBW equations, require very little computing time: (See Section 2.4.20)

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS} \sigma^{\ell}_{n,n}(E) \tag{5}$$

$$\sigma^{\ell}_{nn}(E) = \frac{\pi}{k^2} \sum_{J} g_J \left|1 - U^J_{nn}(E)\right|^2 \tag{6}$$

$$U^J_{nn}(E) = \exp(2i\phi_\ell) - \sum_{r=1}^{NR_J} \frac{i\Gamma_{nr}}{E'_r \quad \Gamma_r/2} \tag{7}$$

D.1.3. Reich-Moore Formulae

A detailed derivation of these formulae is to be found in Reich and Moore.[2] Neutron cross sections with an exit channel c are given by*

$$\sigma_{nc} = \frac{\pi}{k^2} \sum_J g_J \left| \delta_{nc} - U^J_{nc} \right|^2 \quad , \tag{1}$$

where k and g_J are the same as in the previous section.

In the Reich-Moore formalism one has for the scattering matrix, labelled by channel subscripts (n in, c out)

$$U^J_{nc} = e^{-i(\phi_n + \phi_c)} \left\{ 2\left((I-K)^{-1} \right)_{nc} - \delta_{nc} \right\} \quad , \tag{2}$$

where

$$(I-K)_{cc'} = \delta_{cc'} - \frac{i}{2} \sum_r \frac{\Gamma^{1/2}_{cr} \Gamma^{1/2}_{c'r}}{E_r - E - \frac{i}{2}\Gamma_{\gamma r}} \tag{3}$$

*These formulae are to be used for the 0°K case (no Doppler broadening terms given).

(2) C.W. Reich and M. S. Moore, Phys. Rev. 111, 929 (1958).

and the summation is over the resonances r; E_r is the resonance energy; $\Gamma_{\gamma r}$ is the "eliminated" radiation width; and Γ_{cr} and $\Gamma_{c'r}$ are the partial widths for the r-th resonance in channels c and c', respectively.

If we define

$$\rho_{nc} = \delta_{nc} - \left((I-K)^{-1}\right)_{nc} = \delta_{nc} - \frac{m_{nc}}{\Delta} \quad ,$$

where $\Delta = |I-K|$ is the determinant of the matrix I-K and m_{nc} is the cofactor of the element $(I-K)_{nc}$ of the matrix I-K, we obtain

$$\sigma_T(E) = \sum_J \sigma_T^J = \frac{2\pi}{k^2} \sum_J g_J \, \mathrm{Re}(1-U^J_{mn}) \tag{4}$$

$$= \frac{2\pi}{k^2} \sum g_J \left\{(1-\cos 2\phi_\ell) + 2\mathrm{Re}(e^{-2i\phi_\ell}\rho_{nn})\right\}$$

$$\sigma_{n,n}(E) = \frac{\pi}{k^2} \sum_J g_J \, |1-U^J_{nn}|^2 \tag{5}$$

$$\sigma_{n,\mathrm{Abs}}(E) = \sigma_T(E) - \sigma_{n,n}(E) = \frac{4\pi}{k^2} \sum_J g_J \left(\mathrm{Re}(\rho_{nn}) - |\rho_{nn}|^2\right) \tag{6}$$

$$\sigma_{n,f}(E) = \frac{4\pi}{k^2} \sum_J g_J \left(\sum_c |\rho_{nc}|^2\right) \tag{7}$$

$$\sigma_{n,\gamma}(E) = \sigma_{n,Abs}(E) - \sigma_{n,f}(E). \quad (8)$$

The phase shifts are evaluated in terms of the scattering radius, the penetrability in terms of the channel radius (see previous sections). The shift factor has been set equal to zero in the above equations ($E_\gamma' \rightarrow E_r$) and hence they should be used only for s-wave resonances.

Because the section and subsection structure were deleted from the File 2 description, they will be given here for reference and for use in ENDF-A.

The structure of a subsection is similar to that for SLBW and MLBW, the major difference being that the total resonance widths are not given and two fission widths are allowed for each resolved resonance. The quantities for use when LRF = 3 are defined below.

Resolved Resonance Parameters

If LRF = 3 (Reich-Moore multilevel parameters) ENDF/A only

SPI is the spin of the target nucleus, I.

AP is the scattering radius in units of 10^{-12} cm.

NLS is the number of ℓ-states considered. A set of resolved resonance parameters is given for each ℓ-state. (NLS $\leq$ 3.)

L is the value of ℓ (neutron angular momentum quantum number). AWRI is the ratio of the mass of a particular isotope to that of the neutron.

NRS is the number of resolved resonances for a given ℓ-state. (NRS $\leq$ 500.)

ER is the resonance energy (in the laboratory system).

AJ is the floating point value of J (the spin of the resonance).

GN is the neutron width Γ_n evaluated at the resonance energy.

GG is the radiation width Γ_γ evaluated at the resonance energy.

GFA is the first partial fission width.

GFB is the second partial fission width. GFA and GFB are signed quantities, their signs being determined by the realtive phases of the width amplitudes in the two fission channels.

The structure of a subsection when LRU = 1 (resolved parameters) and LRF = 3 (Reich-Moore multilevel parameters) is

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0) CONT

(MAT, 2, 151/AWRI, 0., L, 0, 6*NRS, NRS/

ER_1, AJ_1, GN_1, GG_1, GFA_1, GFB_1,

ER_2, AJ_2, GN_2, GG_2, GFA_2, GFB_2,

- -

ER_{NRS}, AJ_{NRS}, GN_{NRS}, GG_{NRS}, GFA_{NRS}, GFB_{NRS}) LIST

The LIST record is repeated until each of the NLS ℓ-states has been specified in order of increasing value of ℓ. The values of ER for each ℓ-state are ordered by increasing value of ER.

D.1.4. Adler-Adler Multilevel Resonance Parameters: LRU=1, LRF=2

The formulae for obtaining cross sections, taken from Adler and Adler[3,4] are given for the total, radiative, capture, and fission cross sections (without Doppler broadening), They have been slightly re-cast to make them conform to the definitions used earlier in this Appendix. Furthermore,

(3) F.T. Adler and D.B. Adler, Conf. on Neutron Cross Section Technology, Vol. II, 873 (1967)

(4) D.B. Adler and F.T. Adler, ANL-6792, 695 (1963).

only the ℓ=0 terms are given, consistent with current usage of this formalism. Procedures are described in Section 2.4.16.

1. Total Cross Section

$$\sigma_T(E) = \frac{4\pi}{k^2}\sin^2\phi_o$$

$$+ \frac{\pi\sqrt{E}}{k^2}\left[\sum_{r=1}^{NRS} \frac{\nu_r \; G_r^T\cos 2\phi_o + H_r^T\sin 2\phi_o \quad + (\mu_r - E) \quad H_r^T\cos 2\phi_o - G_r^T\sin 2\phi_o}{(\mu_r - E)^2 + (\nu_r)^2}\right.$$

$$\left. + AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1 * E + BT_2 * E^2\right] .$$

2. Radiative Capture Cross Section

$$\sigma_{n,\gamma}(E) =$$

$$\frac{\pi\sqrt{E}}{k^2}\left[\sum_{r=1}^{NRS} \frac{\nu_r G_r^\gamma + (\mu_r - E) H_r^\gamma}{(\mu_r - E)^2 + (\nu_r)^2}\right.$$

$$\left. + AC_1 + AC_2/E + AC_3/E^2 + AC_4/E^3 + BC_1 * E + BC_2 * E^2\right] .$$

3. Fission Cross Section

$$\sigma_{n,f}(E) =$$

$$\frac{\pi\sqrt{E}}{k^2}\left[\sum_{r=1}^{NRS}\frac{\nu_r G_r^f + (\mu_r - E)H_r^f}{(\mu_r - E)^2 + (\nu_r)^2}\right.$$

$$\left. + AF_1 + AF_2/E + AF_3/E^2 + AF_4/E^3 + BF_1 * E + BF_2 * E^2\right] .$$

Although the format provides for different values of μ and ν for each reaction, these must be equal:

$$DET_r = DEF_r = DEC_r = \mu_r;$$

$$DWT_r = DWF_r = DWC_r = \nu_r.$$

D.2. The Unresolved Resonance Region: LRU=2, LRF=1 or 2

Average resonance parameters are provided in File 2 for the unresolved region. Parameters are given for possible ℓ- and J-states (up to d-wave, $\ell = 2$) and the following parameters may be energy dependent: $D_{\ell,J}$, $\Gamma^0_{n\ell,J}$, $\Gamma_{\gamma\ell,J}$, $\bar{\Gamma}_{f\ell,J}$ and $\bar{\Gamma}_{x\ell,J}$. The parameters are for a single-level Breit-Wigner formula with interface. Each width is distributed according to a chi-squared distribution with a designated number of degrees of freedom. The number of degrees of freedom may be different for neutron and fission widths and for different (ℓ, J) states. These formulae do not consider Doppler broadening.

D.2.1. Cross Sections in the Unresolved Region

Definitions and amplifying comments on the following are given in Section D.2.2.

a. Elastic Scattering Cross Section

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS} \sigma^{\ell}_{n,n}(E)$$

$$\sigma^{\ell}_{n,n}(E) = \frac{4\pi}{k^2}(2\ell+1)\sin^2\phi_{\ell}$$

$$+ \frac{2\pi^2}{k^2} \sum_{J}^{NJS_{\ell}} \frac{g_J}{\bar{D}_{\ell,J}} \left[\left\langle \frac{\Gamma_n \Gamma_n}{\Gamma} \right\rangle_{\ell,J} - 2\bar{\Gamma}_{n_{\ell,J}} \sin^2\phi_{\ell} \right]$$

The asymmetric term in $E-E'_r$ is assumed to average to zero under the energy-averaging denoted by $\langle\ \rangle$.

b. Radiative Captive Cross Section

$$\sigma_{n,\gamma}(E) = \sum_{\ell=0}^{NLS} \sigma^{\ell}_{n,\gamma}(E) \quad ,$$

$$\sigma^{\ell}_{n,\gamma}(E) = \frac{2\pi^2}{k^2} \sum_{J}^{NJS_{\ell}} \frac{g_J}{\bar{D}_{\ell,J}} \left\langle \frac{\Gamma_n \Gamma_\gamma}{\Gamma} \right\rangle_{\ell,J} .$$

c. Fission Cross Section

$$\sigma_{n,f}(E) = \sum_{\ell=0}^{NLS} \sigma^{\ell}_{n,f}(E) \quad ,$$

$$\sigma^{\ell}_{n,f}(E) = \frac{2\pi^2}{k^2} \sum_{J}^{NJS_{\ell}} \frac{g_J}{\bar{D}_{\ell,J}} \left\langle \frac{\Gamma_n \Gamma_f}{\Gamma} \right\rangle_{\ell,J} .$$

The sum over ℓ in the above equations extends up to $\ell = 2$ or to NLS (the number of ℓ-states for which data are given). For each value of ℓ, the sum over J has NJS_{ℓ} terms (the number of J-states for a particular ℓ-state). NLS and NJS are given in File 2.

The averages are re-written as

$$\left\langle \frac{\Gamma_n \Gamma_n}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \bar{\Gamma}_{n_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{n_{\ell,J}}$$

$$\left\langle \frac{\Gamma_n \Gamma_\gamma}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \bar{\Gamma}_{\gamma_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{\gamma_{\ell,J}}$$

$$\left\langle \frac{\Gamma_n \Gamma_f}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \bar{\Gamma}_{f_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{f_{\ell,J}}$$

where $R_{\gamma\ell,}$, $R_{f\ell,}$, and $R_{n\ell,J}$ are width-fluctuation factors for capture, fission, and elastic scattering, respectively. Associated with each factor is the number of degrees of freedom for each of the average widths, and the integrals are to be evaluated using the MC^2-II method.

Data given in File 2 for each (ℓ, J) state

$\mu_{n\ell,J}$ = AMUN, the number of degrees of freedom for neutron widths

$\mu_{f\ell,J}$ = AMUF, " " " " " " " fission widths

$\mu_{x\ell,J}$ = AMUX, " " " " " " " competitive

$\mu_{\gamma\ell,}$ = AMUG, " " " " " " " radiation widths

$\bar{\Gamma}_{x\ell,J}$ = GX, the average competitive reaction width

$\bar{\Gamma}^{o}_{n\ell,J}$ = GNO. the average reduced neutron width

$\bar{\Gamma}_{\gamma\ell,J}$ = GG, the average radiation width

$\bar{\Gamma}_{f\ell,J}$ = GF, the average fission width

$\bar{D}_{\ell,J}$ = D, the average level spacing

The average neutron widths are defined in Section D.2.2.2, Equation 10, where $\bar{\Gamma}_{n\ell,J} = \langle \Gamma_n(\ell,J) \rangle$.

The average total width, at energy E, is

$$\overline{\Gamma}_{\ell,J} = \overline{\Gamma}_{n_{\ell,J}} + \overline{\Gamma}_{\gamma_{\ell,J}} + \overline{\Gamma}_{f_{\ell,J}} + \overline{\Gamma}_{x_{\ell,J}} ,$$

and all widths are evaluated at energy E. J = AJ, I = SPI, and ℓ = L are all given in File 2.

The penetration factors P_ℓ are functions of ka; the phase shifts ϕ_ℓ, of k(AP), as in section D.1.1, a being the channel radius, and AP the scattering radius from File 2.

The correct procedure for determining the energy-dependence of an unresolved cross section is to evaluate it at the energy points at which the parameters are given, and then to interpolate on the cross section. In some cases, as when only one set of parameters is given, this appears to involve interpolating on the parameters, since a single set of parameters will yield a cross section that varies with energy as a combination of s-, p-, and possibly d-wave penetrabilities. In view of the still widespread use of parameter interpolation, it is recommended that the energy mesh for unresolved resonance parameters be dense enough to render the differences between the two methods small.

D.2.2 Definitions for the Unresolved Resonance Region

Previous editions of ENDF-102 have had substantive and typographical errors in the "Definitions" section of Appendix D (previously Section D.2.1). To clarify the points and facilitate parallel reading with Gynlasy and Perkins[5] their parenthesized indices will be used. Section D.2.3 contains a table of

(5) M. Gynlassy and S. T. Perkins, Nuclear Science Engineering 53, 482 (1974).

equivalences to the notation used in D.2.1. and Section D.2.4 compares the present discussion with those previously given.

D.2.2.1 Sums and Averages

In an energy interval $\Delta\varepsilon$, let the resonances be identified by a sequential subscript λ = 1, 2, --- which goes over all the resonances. The present discussion is concerned with the combinational aspects of level sequences, hence λ enumerates all the resonances, whether their widths are observably large or not. One purpose of this section is to permit estimation of missed resonances by comparing observed level densities or strength functions with the theoretically-expected relations. The latter are concerned with the set of all resonances, and not just those that are observable in a particular experiment.

Let x denote a set of quantum numbers that label a subset of resonances in the interval. If there are N (x) such resonances, their level density is

$$\rho(x) = N(x)/\Delta\varepsilon \tag{1}$$

and their level spacing is

$$D(x) = 1/\rho(x) \tag{2}$$

If y_λ is some quantity associated with each resonance, λ, the sum of the y-values over the subset x is

$$\sum_{\lambda}^{x} y\lambda \tag{3}$$

The summation index λ is written as a subscript, and the range of the summation is indicated by the superscript x. Equation (3) says "sum the quantity y over every resonance in the interval $\Delta\varepsilon$ which has the quantum numbers x." Usually, these resonances will possess other quantum numbers too, but it is the set x which determines whether they are included or not.

An <u>average</u> of the quantity y over the set x is

$$\langle y \rangle^{x} = \left(1/N(x)\right) \sum_{\lambda}^{x} y\lambda \tag{4}$$

D.2.2.2 <u>Reduced Widths</u>

Reduced widths follow the experimental notation. A partial width for the decay of a resonance into a particular channel carries many quantum numbers, but here we need at most three, the total and orbital angular momenta J and ℓ, and the channel spin s. For this case, the reduced neutron width is defined by:

$$\Gamma_{n\lambda}(\ell,J,s) = \Gamma_{n\lambda}^{\ell}(J,s)\ \sqrt{E}\ V_{\ell}(E) \tag{5}$$

where

$$\begin{aligned} V_o &= 1 \\ V_1 &= \rho^2/(1+\rho^2) \\ V_2 &= \rho^4/(g+3\rho^2+\rho^4) \end{aligned} \tag{6}$$

and $\rho = ka$, a being the channel radius.

Assuming additivity of partial widths,

$$\Gamma^{\ell}_{n\lambda}(J) = \Sigma_s \Gamma^{\ell}_{n\lambda}(J,s) \tag{7}$$

If we average over resonances, and assume that the average partial width is independent of channel spin,*

$$\begin{aligned} \langle\Gamma^{\ell}_{n}(J)\rangle &= \Sigma_s \langle\Gamma^{\ell}_{n}(J,s)\rangle \\ &= \mu_{\ell,J}\langle\Gamma^{\ell}_{n}(J,s)\rangle \end{aligned} \tag{8}$$

(8) introduces the multiplicity $\mu_{\ell,J}$, which for neutrons can have the value 1 or 2, depending on whether the channel spin has one or two values. For $\ell=0$, or $I=0$, or $J=0$, $\mu_{\ell,J}=1$. In other cases,s can take in the values $I \pm 1/2$ subject to the additional vector sum

$$\vec{s} = \vec{\ell} + \vec{J} \quad , \tag{8a}$$

which can again restrict $\mu_{\ell,J}$ to the value one.**

*This is not true for the individual resonances.

**E.g., if $I = 1/2$, $\mu_{1,2} = 1$

The other new notation is the line through the quantum number s, meaning that the quantity $\langle\Gamma_n(\ell,J,\not{s})\rangle$ does not depend on the value of s. This is not the same as omitting s from the parentheses, since that defines the left-hand side quantity. This is the primary source of confusion in previous discussions. Since V_ℓ only depends in ℓ,

$$\langle\Gamma_n(\ell,J)\rangle = \langle\Gamma_n^\ell(J)\ \sqrt{E}\ V_\ell\rangle \tag{9}$$

$$= \mu_{\ell,J}\ \langle\Gamma_n^\ell(J,\not{s})\ \sqrt{E}\ V_\ell\rangle$$

$$= \mu_{\ell,J}\ \langle\Gamma_n^\ell(J,\not{s})\rangle\ \overline{\sqrt{E}\ V_\ell}, \tag{10}$$

where the bar over $\sqrt{E}\ V_\ell$ denotes some average value appropriate to the interval.

D.2.2.3 Strength Functions

The pole-strength function was originally introduced as an average over the R-matrix reduced widths for a given channel, γ_c^2. Using the experimental convention,

$$S(\ell,J,s) = \langle\Gamma_n^\ell(J,s)\rangle\ {}^{\ell,J,s}/D(\ell,J,s) \tag{11}$$

Since the channel spin values are uniquely determined by J and ℓ, together with the target spin I which is common to all the resonances, s is superfluous in defining the subset over which the average is taken, and

$$S(\ell,J,s) = \langle\Gamma_n^\ell(J,s)\rangle\ {}^{\ell,J}/D(\ell,J) \tag{12}$$

If the parity π was used as an explicit quantum number, ℓ could be dropped,

$$S(\ell,J,s) = \langle \Gamma_n^{\ell}(J,s) \rangle^{J,\pi}/D(J,\pi) \tag{13}$$

because ℓ and π are equivalent for labelling resonances, that is, every resonance with a given J and π will have channels labelled by the same set of ℓ-values, whether their partial widths are observably large or not.

Some authors go one step further and drop π, so that J <u>means</u> J, π, but this is an invitation to confusion.

Expressing S (ℓ,J,s) as a sum over reduced widths gives

$$S(\ell,J,\not{s}) = \Sigma_{\lambda}^{\ell,J} \Gamma_{n\lambda}^{\ell}(J,s)/\Delta\varepsilon \tag{14}$$

where we use the assumed independence of $\langle \Gamma_n^{\ell}(J,\not{s}) \rangle$ on s to get the same result on the left-hand-side.

The strength function S (ℓ, J) is defined as

$$S(\ell,J) = \Sigma_s S(\ell,J,\not{s}) \tag{15}$$

$$= \mu_{\ell,J} S(\ell,J,\not{s}) \tag{16}$$

The corresponding sum and average forms are

$$S(\ell,J) = \Sigma_{\lambda}^{\ell,J} \Gamma_{n\lambda}^{\ell}(J)/\Delta\varepsilon \tag{17}$$

$$= \langle\Gamma_{n}^{\ell}(J)\rangle^{\ell,J}/D(\ell,J) \tag{18}$$

The next "natural" summation would be to collect the different ℓ-contributions to the total width, to form $S(J)$, but this is not what is observable. Instead one defines $S(\ell)$ as a weighted sum of the $S(\ell,J,s)$:

$$S(\ell) = \frac{\Sigma_{Js} g S(\ell,J,s)}{\Sigma_{Js}\, g} \tag{19}$$

This equation occurs in Lynn[6] as 6.126, with a confusing typographical error, namely the index S is missing from S (ℓ,J,s).

Actually, the s-wave strength function was introduced first in the form

$$S(0) = \Sigma_{\lambda}^{\ell=0} (g\Gamma_{n}^{0})_{\lambda}/D(\ell=0) \tag{20}$$

and later generalized by Saplakoglu, Bollinger, and Cote[7] to the p-wave form.

---

(6) J. E. Lynn, "The theory of Neutron Resonance Reactions," Clarendon Press, Oxford, 1968.

(7) Physics Review 109, 1258 (1958)

$$S(l) = (1/\Delta\varepsilon)(2\ell + 1)^{-1} \Sigma_{\lambda}^{\ell=l} (g\Gamma_n^l)_\lambda \tag{21}$$

For expository purposes, it is clearer to start from (19). The sum on J and s is for fixed ℓ:

$$\sum\nolimits_{Js} \equiv \sum_{s = |I - 1/2|}^{I + 1/2} \sum_{J = |\ell - s|}^{\ell + s} \tag{22}$$

It is important to note that the outer sum on channel spin is correct as written. It goes over the values $I \pm 1/2$ if $I \geq 1/2$, and over the single value 1/2, if $I = 0$. It is <u>not</u> further constrained by equation (8a) because now <u>it</u> is the "independent variable". The inner sum on J enumerates some J-values once, and some twice, the latter occurring when both s-values can produce that J-value. The number of times J occurs is the same $\mu_{\ell,J}$ that appeared previously. (The proof will be left to the reader).

If we are summing a quantity that is independent of s, then (22) can be rewritten:

$$\Sigma_{Js}\, y(\not{s}) = \tilde{\Sigma}_J\, \mu_{\ell,J}\, y(\not{s}) \tag{23}$$

the multiplicity $\mu_{\ell,J}$ takes care of the sum on s, and the tilde over the sum on J, as emphasized by Gyulassy and Perkins[5], reminds us that J goes over its full range, "once-only"

$$\sum_{J}^{f} = \sum_{J=|\ell-I-1/2|}^{\ell+I+1/2} \qquad \text{if } \ell \geq I \tag{24a}$$

$$= \sum_{J=|I-\ell-1/2|}^{\ell+I+1/2} \qquad \text{if } I \geq \ell \quad * \tag{24b}$$

The denominator in equation (19) can easily be shown to be

$$\Sigma_{Js}\, g = 2\ell + 1 \tag{25}$$

or, since g is independent of s

$$\sum_{J}^{f} \mu_{\ell,J}\, g = 2\ell + 1 \tag{26}$$

G-P assume, and later approximately justify by comparison to experiment, that $S(\ell, J, \not{s})$ is also independent of J. With this, (19) becomes

$$S(\ell) = (2\ell + 1)^{-1} \quad \Sigma_{Js}\, g\, S(\ell, \not{J}, \not{s})$$

$$= S(\ell, \not{J}, \not{s}) \tag{27}$$

$$= S(\ell, J)/\mu(\ell, J) \tag{28}$$

Using equation (16).

*Ref (5) has this written incorrectly

Note the peculiar fact that $S(\ell)$ and $S(\ell,\not{J},\not{s})$ are independent of J, but $S(\ell,J)$ is not. This is a consequence of the fact that more than one channel spin value can contribute to $S(\ell,J)$, inducing a "J-dependence" in the form of a possible factor of two.

As a sum over resonances,

$$S(\ell) = (2\ell + 1)^{-1} \quad \Sigma_{Js}\, gS(\ell,\not{J},\not{s}) \tag{29}$$

$$= (2\ell + 1)^{-1} \quad \Sigma_{Js}\, \Sigma_{\lambda}^{\ell,J}\, g\Gamma_{n\lambda}^{\ell}\ (J,s)/\Delta\varepsilon \tag{30}$$

$$= (2\ell + 1)^{-1}\, \textstyle\sum_{J}^{\mathcal{L}}\, \Sigma_{\lambda}^{\ell,J}\, g\Gamma_{\lambda n}^{\ell}\ (J)/\Delta\varepsilon \tag{31}$$

The r.h.s. of (31) says to sum $\Gamma_{n\lambda}^{\ell}(J)$ over all possible values of J, which is what is meant by equations (20) and (21). We can suppress the explicit J's and write

$$S(\ell) = (1/\Delta\varepsilon)\ (2\ell + 1)^{-1}\, \Sigma_{\lambda}^{\ell}\ (g\Gamma^{\ell})_{\lambda} \tag{32}$$

but we have to remember that Γ_{n}^{ℓ} is still $\Gamma_{n}^{\ell}\ (J)$, and not a new quantity.

As an average, using the same convention,

$$S(\ell) = (2\ell + 1)^{-1}\ \ <g\Gamma_{n}^{\ell}>^{\ell}\ /D(\ell) \tag{33}$$

otherwise, all the notation is correct: $D(\ell)$ is the spacing of ℓ-wave resonances without regard to their J-values, and the average $< >^{\ell}$ goes over all res-

onances possessing the quantum number ℓ, again without regard for their J-values. It is worth noting explicitly that although S (ℓ, J) is "almost" independent of J, this is not true of $\langle \Gamma_n^{\ell}(J) \rangle$. As equation (18) shows, its J-dependence is cancelled by the J-dependence of D (ℓ, J), up to the factor $\mu_{\ell,J}$. This property is what makes strength functions useful.

D.2.2.4 Level Spacings

G-P emphasize that

$$\rho(\ell) = \sum_J \rho(\ell,J) \tag{34}$$

which, together with the assumption

$$\rho(\ell, J) = K(\ell)(2J+1) \tag{35}$$

leads to

$$\rho(\ell) = \rho(\ell,J) \frac{(2\ell + 1)}{g} \omega_{I,\ell} \tag{36}$$

where

$$\omega_{I,\ell} = (\ell + 1)/(2\ell + 1) \text{ for } \ell \leq I \tag{37}$$

$$= (I + 1)/(2I + 1) \text{ for } \ell \geq I$$

and is unity if $\ell = 0$ or $I = 0$.

The reader is referred to their paper for a fuller discussion but here we can point out that, for a given parity, $\rho(\ell,J)$ is independent of ℓ, by definition. As noted, every resonance with a given J and π has the same set of associated ℓ-channels, whether they have observable widths or not. Hence

$$\rho(0, J) = \rho(2, J) = \rho(4, J) = \ldots \tag{38}$$

and $\rho(1, J) = \rho(3, J) = \rho(5, J) = \ldots$

The further assumption of parity-independence makes $\rho(\ell,J)$ totally independent of ℓ. As a result, G-P's $K(\ell)$ from (35) is independent of ℓ, and

$$\rho(\ell) = C(2\ell + 1)\, \omega_{I,\ell} \tag{39}$$

where C depends on the nucleus but not on any quantum numbers.

D.2.2.5 Gamma Widths

In the limited energy range of a few keV usually covered by the unresolved resonance region, the gamma width may be assumed to be constant and equal to that obtained from an analysis of the resolved resonances. If, however, the energy range is rather wide, an energy dependence as given by some of the well-known theoretical models[6] may be bullt in. Since the observed gamma width is a sum of a large number of primary gamma transitions, each assumed to have a chi-squared distribution of $\mu = 1$, it is found to have a $\mu \geq 20$. In

effect this implies that the gamma width is a constant, since a chi-squared distribution with a large number of degrees of freedom approximates a δ-function.

D.2.2.6 Degrees of Freedom

For the reasons enumerated in File 2, Section 2.4.21, the following values should be used:

1. Neutron width, 1. $\leq$ AMUN $\leq$ 2., and specifically, AMUN = $\mu_{\ell,J}$.
2. Radiation width, AMUG = 0.
3. Fission width, 1. $\leq$ AMUF $\leq$ 4., to be determined by comparison with experiment. Only integral values are permitted.
4. Competitive Widths, 1. $\leq$ AMUX $\leq$ 2., because only a single inelastic level excitation is permitted as a competitive reaction. Specifically, AMUX = $\mu^{\prime}_{\ell,J}$, where J is the spin of the resonance, and $\ell^{\prime}$ is the orbital angular momentum of the inelastically scattered neutron. Since the daughter "nucleus" may have a spin $I^{\prime}$ different from the target spin I, $\ell^{\prime}$ may be different from ℓ and the number of channel spin values $\mu^{\prime}_{\ell,J}$ may be different from $\mu_{\ell,J}$.

D.2.3 Equivalent Quantities in Sections D.1 and D.2

Symbol in D.1	Symbol in D.2.1	Symbol in D.2.2	Definition
r	-	-	This is a non-equivalence. λ enumerates all resonances.
-	-	λ	Renumerates those within a subset and hence implies a set of quantum numbers.
$\Gamma_{n\gamma}$		$\Gamma_{n\lambda}(\ell,J)$	The neutron width, summed over channel spin.
$\Gamma_{\gamma r},\Gamma_{fr},\ldots$		-	Not used in D.2.2, but the same implication of ℓ, J holds.
P_ℓ	ρV_ℓ	ρV_ℓ	Penetration factor.
	$D_{\ell,J}$	$D(\ell,J)$	Average level spacing for a subset of resonances with given ℓ and J.
-	$\bar{\Gamma}^o_{n_{\ell,J}}$	$\langle\Gamma^\ell_n(J)\rangle^{\ell,J}$	The ℓ-wave reduced width, averaged over all resonances with given ℓ and J.
-	$\bar{\Gamma}_{n_{\ell,J}}$	$\langle\Gamma_n(\ell,J)\rangle^{\ell,J}$	The average neutron width. In practice, the energy-dependence of this quantity is not average, but extracted before averaging.

D.2.4 Comparison with previous editions of ENDF-102

D.2.4.1 ENDF-102, October 1970 edition

1. Equation (1). D observed is D (ℓ)

2. Line 9. ℓ is not the angular momentum of "the incident neutron". The incident neutrons carry all angular momenta. ℓ is the orbital angular momentum of the resonance, or more precisely, of the channel or channels which are involved. The resonant ℓ^{th} phase shift will interfere with the non-resonant ones in angular distributions, but not in angle-integrated cross sections.

3. Equation (2).

ρ_J is $\rho(\ell,J)$; ρ_{obs} is $\rho(\ell)$; Σ is $\overset{\circ}{\Sigma}_J$

4. Equation (3). D_J is D (ℓ,J) and the right-hand-side should have a factor $\omega_{I,\ell}$.

5. Page D-11, line 1. "level-spacing" means D (ℓ,J). Line 8. The statement "If we assume the s-wave strength function is independent of J ..." presumably means assuming S (0,J) is independent of J, since the s-wave strength function itself; S (0), is a sum over J-states and is therefore "independent" of J by definition.

Equation 4 means

$$S(0) = S(0,\not{J},\not{s}) = S(0,J)/\mu_{0,J}$$

which because $\mu_{0,J}$=1 implies

$$S(0) = S(0,J) = \langle \Gamma_n^0(J) \rangle^{0,J} / D(0,J)$$

Equation 5 is trickier because $\mu_{l,J}$ is not indentically equal to 1, and the discussion appears to give the user the option of getting $D(\ell,J)$ from equation (3) and "the corresponding reduced neutron width" from equation (3), or of using the ENDFB convention, equation (6)'

The problem lies in the failure to distinguish $\Gamma_n^\ell(J)$ from $\Gamma_n^\ell(J,s)$.

Equation (6) states the "ENDFB convention":

$$\langle \Gamma_{nJ}^\ell \rangle = \langle \Gamma_{nJ}^{0\ell} \rangle \sqrt{E}\, V_\ell \mu_{\ell J}$$

We know that the correct relationship is

$$\langle \Gamma_n(\ell,J) \rangle = \langle \Gamma_n^\ell(J,s) \rangle \sqrt{E}\, V_\ell \mu_{\ell,J}$$

$$= \langle \Gamma_n^\ell(J) \rangle \sqrt{E}\, V_\ell$$

If we assume that $\langle \Gamma_{nf}^\ell \rangle$ in equation (6) is to agree with experiment; then $\langle \Gamma_{nJ}^{0\ell} \rangle$ must be

$$\langle \Gamma_n^\ell(J,s) \rangle = \langle \Gamma_n^\ell(J) \rangle / \mu_{\ell,J}$$

If we put a subscript E on to denote an ENDF/B-convention quantity,

$$\langle \Gamma_{nJ}^{0\ell} \rangle \rightarrow \langle \Gamma_{nJE}^{0\ell} \rangle = \langle \Gamma_{nJ}^{0\ell} \text{actual} \rangle / \mu_{\ell J}$$

and now

$$\frac{\langle \Gamma_n^0 \rangle^{\ell}_{J}}{D_{\ell J}} \text{ in (5) means } \frac{\langle \Gamma_{nE}^0 \rangle^{\ell}_{J}}{D_{\ell J}} = \frac{\langle \Gamma_n^0 \text{actual} \rangle^{\ell}_{J}}{D_{\ell,J}\mu_{\ell,J}}$$

which is correct.

Thus an ENDF/B reduced width $\langle \Gamma_{nJE}^{0\ell} \rangle$ will sometimes be half what an experimentalist would measure.

In the notation of D.2.2. equation (6) is

$$\langle \Gamma_n(\ell,J) \rangle^{\ell,J} = \langle \Gamma_n^{\ell}(J,\not{s}) \rangle^{\ell,J} \sqrt{E}\, v_{\ell}\mu_{\ell,J}$$

$$= \langle \Gamma_n^{\ell}(J) \rangle^{\ell,J} \sqrt{E}\, v_{\ell}$$

and $\Gamma_n^{\ell}(J)$ is the reduced width determined by experiment.

For p-waves,

$$S(1) = S(1,J)/\mu_{1,J}$$

$$= \langle \Gamma_n^{1}(J) \rangle^{1,J}/D(1,J)\mu_{1,J}$$

and equation (5) would read:

$$S_1 = \frac{\langle \Gamma_n^1(J_1) \rangle^{1,J_1}}{D(1,J)\mu_{1,J}} = \frac{\langle \Gamma_n^1(J_2) \rangle^{1,J_2}}{D(1,J_2)\mu_{1,J_2}} = \cdots$$

6. Three different symbols are used for the reduced width.

In a unified notation:

page D-9 $\bar{\Gamma}^{0}_{n_{\ell,J}} = \langle \Gamma^{\ell}_{n}(J) \rangle^{\ell,J}$

page D-11 $\langle \Gamma^{0}_{n} \rangle^{\ell=1}_{J} = \langle \Gamma^{\ell}_{n}(J) \rangle^{\ell,J}$

page D-12 $\langle \Gamma^{0\ell}_{nJ} \rangle = \langle \Gamma^{\ell}_{n}(J,\not{s}) \rangle^{\ell,J}$

D.2.4.2 Comments on ENDF-102, October 1975 edition, Section D.2.1

1. Equation (1). D_{ℓ}, observed is D (ℓ).

2. Same as comment 2 on the 1970 edition; above.

3. Equation (2). D_{obs} is D(ℓ); $\rho_{\ell,obs}$ is $\rho(\ell)$; Σ_{J} is $\mathfrak{L}_{J}$.

4. "All allowed ℓ-values label the same set of resonances" means $\rho(\ell,J)$ does not depend on ℓ (for given parity).

5. Equation (3). $(2J+1)^{-1}$ is missing from the right hand side.

6. Page D-12, second equation:

The quantity $\langle g\Gamma^{\ell}_{n} \rangle$ is $\langle g\Gamma^{\ell}_{n}(J) \rangle^{\ell}$. The bracket $\langle \rangle^{\ell}$ means that all J-values are summed over. The other two brackets are for particular J-values, i.e.,

$$\langle g\Gamma^{\ell}_{n} \rangle = \langle g\Gamma^{\ell}_{n}(J) \rangle^{\ell} = \frac{g\langle \Gamma^{\ell}_{n}(J) \rangle^{\ell,J}}{\omega_{I,\ell}\mu_{\ell,J}} = \frac{g\langle \Gamma^{\ell}_{n}(J,\not{s}) \rangle^{\ell,J}}{\omega_{I,\ell}}$$

7. "The strength functions for a given ℓ-value but different J-values" means $S(\ell,J)$. These are not all equal--it is the ratio $S(\ell,J)/\mu_{\ell,J}$ which is independent of J.

8. Equation (6) should read:

$$\langle\Gamma_n(\ell,J)\rangle = \langle\Gamma_n^{\ell}(J,s)\rangle \sqrt{E}\, V_\ell \mu_{\ell,J}$$

$$= \langle\Gamma_n^{\ell}(J)\rangle \sqrt{E}\, V_\ell$$

where $\Gamma_n^{\ell}(J)$ is the reduced width determined by experiment. That is, the relation involving $\mu_{\ell,J}$ is only valid for an <u>average</u> width, and hence $\sqrt{E}V_\ell$ must also be some appropriate average value. The quantum number s should be exhibited when μ is used.

D.3 The Competitive Width

D.3.1 Penetrability Factor For the Competitive Width in the Resolved Resonance Region

The only physical situation which can be handled without approximation is that in which a single competitive process is possible, because the format presently permits the definition of only one additional quantity. The most common case will occur when inelastic scattering to only the first excited state of the target nucleus is energetically possible. Ignoring, as in the case of elastic scattering, the possibility that the partial widths depend on channel spin, the penetrability is identical to that for elastic scattering, but the energy is reduced by the excitation energy of the first excited state, corrected for recoil, so that

$$\Gamma_{xr}(E) = \Gamma_{n'r}(E) = \frac{P_\ell(E-E_1^*)\Gamma_{xr}(|E_r|)}{P_\ell(|E_r| - E_1^*)}$$

if $E \geq E_1^*$, and $\Gamma_{xr}(E) = 0$ if $E < E_1^*$ where E^* is (AWRI + 1)/ AWRI times the excitation energy of the first excited state, E_1^{ex}. ($E_1^{ex} = -QX$ in File 2).

This definition involves two conventions, both taken over from the elastic case. One is the way in which an "experimental" reduced width $\Gamma_{n'}^{\ell}$, is defined in terms of the theoretical reduced width γ^2, and the other is the way in which negative energy levels are treated. Neither of these problems arises in the theory, where $\Gamma = 2P\gamma^2$ and all quantities are defined in terms of the channel energy.

It is conceivable that an (n,α) or (n,p) reaction to the ground state of the daughter nucleus could be open, without inelastic competition,

in which case the formula for Γ_{xr} would be the same, but the P_ℓ would be a Coulomb penetrability, and the excitation energy E_1^* would be replaced by the appropriate Q-value and reduced mass. There is presently no explicit provision for this option in ENDF/B.

If more than one competitive process is energetically possible, then the format is inadequate to give the correct energy dependence of the competitive width, since it supplies only one number, and a partial width is required for each process. E.g., when two inelastic levels can be reached,

$$\Gamma_{xr}(E) = \frac{P_{\ell_1}(E-E_1^*)\Gamma_{n_1'r}(|E_r|)}{P_{\ell_1}(|E_r| - E_1^*)} + \frac{P_{\ell_2}(E-E_2^*)\Gamma_{n_2'r}(|E_r|)}{P_{\ell_2}(|E_r| - E_2^*)}$$

with appropriate modification below each threshold.

The recommended procedure for processing codes which use an energy-dependent total width in the denominator of the single-and multi-level Breit-Wigner formulas is to use Γ_{xr} as though it were entirely a partial width to the first excited state.

For codes that presently approximate Γ as a constant in the denominator, the recommended procedure is to substitute a step function

$$\Gamma_{xr}(E) = 0 \qquad \text{if } E < E_1^*$$

$$= \Gamma_{xr}(|E_r|) \qquad \text{if } E \geq E_1^*$$

and then make some provision to handle the resultant discontinuity in the cross section.

Users who are unable to handle even this degree of c complexity, and would like to use GT_r from File 2 as the total width without regard for whether the competitive process is energetically possible or not should probably obtain pre-processed multigroup cross sections from some other organization.

Since the Reich-Moore and Adler-Adler formalisms are presently restricted to low-energy fissile materials, no recommendations concerning the treatment of Γ_{xr} need be given, and users can presume that it is zero.

D.3.2 Penetrability Factor for the Competitive Width in the Unresolved Resonance Region

Since many codes treat the average total width in the denominator of expressions like $\left\langle \frac{\Gamma_n \Gamma_\gamma}{\Gamma} \right\rangle$ as an energy-independent constant, the penetrability factor of the competitive width needs to be handled by specifying energy dependent unresolved resonance parameters.

The formalism, which is simple average over SLBW line shapes, takes account of the energy-dependence of the neutron width(s) in the numerator, by extracting their penetrability factors before the averaging is done. These then contribute to the energy-dependence of the average cross section. The energy-dependence of the neutron width in the denominator, i.e., in $\langle\Gamma\rangle$, is neglected. No such fix is readily available for the energy-dependence of the competitive width, whose penetrability factor will involve the threshold dependence of an inelastic cross section. The evaluator can circumvent this difficulty by specifying energy-dependent prarmeters and setting $\langle\Gamma_x\rangle = 0$ below its threshold; then allowing it to build up according to the formulas given in Section D.3.

The degrees of freedom AMUX should be 1.0 or 2.0. (See Section D.2.2.6).

D.3.3 Calculation of the Total Cross Section When a Competitive Reaction is Specified

When a competitive reaction is specified, and Γ exceeds $\Gamma_n + \Gamma_\gamma + \Gamma_f$, the ENDF convention is that the scattering, capture, and fission cross sections will be calculated from the sum of FILE 2 and FILE 3 contributions, but the competitive reaction will be contained entirely in FILE 3, and no FILE 2 contribution should be added to it. The reason for this is that users can avoid problems in coding up resonant competitive widths. In the above FILE 2 calculations, the correct total width Γ mist be used in order to get the correct line shape.

This leaves the total cross section in a special category. If it is calculated as the sum of σ_n, σ_γ, σ_f, and σ_x, then the above prescription works satisfactorily. However, if it is calculated directed from Γ, care must be taken to avoid counting the competitive reaction twice.

That is, since Γ includes Γ_x, the FILE 2 total cross section includes the competitive reaction, somewhat at variance with the idea that the competitive reaction will not be calculated from FILE 2. The evaluator must therefore insure that the FILE 3 total cross section does not include it.

In view of al the permutations and combinations of FILE 2/FILE 3 additions and subtractions and sum checks that are possible, this area should be approached with care.

APPENDIX E

Interpolation Schemes

Interpolation schemes are provided to obtain values of a function, y(x), from a tabulated series of X(N) and Y(N). The symbolism used to specify an interpolation scheme might be

(MAT, MF, MT/C1, C2; L1; L2; NR, NP/E_{int}/Y(E))TAB1

where E_{int} implies an interpolation scheme and Y(E) implies pairs of values for E(N) and Y(N). The binary record would actually contain the following numbers:

(MAT, MF, MT, C1, C2, L1, L2, NR, NP, NBT(1), INT(1), NBT(2), INT(2), NBT(3), INT(3), ..., NBT(NR), INT(NR), E(1), Y(1), E(2), E(3), Y(3), ..., E(NP), Y(NP))

NP is the number of pairs, E and Y, that are given. NR is the number of interpolation ranges given. NBT (1) is defined to mean that a particular interpolation scheme is to be used between point number one and the point number given by NBT(1). The interpolation scheme to be used in this range is specified by the value of INT(1). Likewise in the second interpolation region, between the point number given by NBT(1) and that given by NBT(2), the interpolation scheme is given by the value of INT(2). The procedure is followed until all interpolated regions have been specified. It should be obvious that the value of NBT(NR) is equal to the number NP. An illustration is shown in Figure E.1.

Interpolation schemes for a two-dimensional function y(E',E) are similar. The function is represented by a series of tabulated values and interpolation schemes. In this case two interpolation schemes must be given, one for E and and another for E'. This is specified by a TAB2 record followed by several TAB1 or LIST records. An example might be

(MAT, MF, MT/C1, C2; L1, L2; NR, NE/E_{int})TAB2

(MAT, MF, MT/C1, E(1); L1, L2; NR, NF/E'_{int}/g(E',E_1))TAB1

(MAT, MF, MT/C1, E(2); L1, L2; NR, NF/E'_{int}/g(E', E_2))TAB1

(MAT, MF, MT/C1, E(NE); L1, L2; NR, NF/E'_{int}/g(E', E_{NE}))TAB1

In this case NR, in the TAB2 record, indicates the number of interpolation ranges for (E). There will be NE TAB1 records, each will contain a value of E. E_{int} is the interpolation scheme used for the E mesh. NF in each TAB1 record indicates the number of pairs, E' and g(E',E) that will be given in the particular record. E_{int} is the interpolation scheme to be used. The allowed interpolation schemes are given below.

INT	Description
1	y is constant in x (constant)*
2	y is linear in x (linear-linear)
3	y is linear in ℓn x (linear-log)
4	ℓn y is linear in x (log-linear)
5	ℓn y is linear in ℓn x (log-log)

*Note: INT = 1 (constant) implies that the function is constant and equal to the value given at the lower limit of the interval.

APPENDIX F

Temperature Dependence

Any of the data given in Files 3, 4, 5, 6, or 7 may have a temperature dependence (where it is physically realistic). The temperature dependence is specified by repeating the data for each temperature given and indicating how to interpolate the data between temperatures. LT is a flag that indicates whether or not temperature-dependent data are given.

The following quantities are defined.

T_m is the m^{th} temperature (°K).

LT is a test for temperature dependence:

LT = 0 means no temperature dependence;

LT > 0 means that the function y(x,T) is given at (LT + 1) temperatures.

I_m is the interpolation scheme used between T_{m-1} and T_m. The values of I have the same definitions as INT given for other interpolation schemes (see Appendix E).

Since the data will always be given in a LIST or TAB1 record, consider a TAB1 record for a function y(x). In this case the functions must be y(x,T). The function at the first temperature $y(x,T_1)$ is given in a TAB1 record. The functions for the remaining temperatures are given in LIST records. The number of LIST records will be LT. An example might be

(MAT, MF, MT/ T_1, C2; LT, L2; NR, NP_1/ X_{int}/$Y(X,T_1)$)TAB1

(MAT, MF, MT/ T_2, C2; I_2, L2; NP_2, 0 / $Y_n(T_2)$)LIST

(MAT, MF, MT/ T_3, C2; I_3, L2; NP_3, 0 / $Y_n(T_3)$)LIST

(MAT, MF, MT/ T_{LT+1}, C2; I_{LT+1}, L2, NP_{LT+1}, 0/ $Y_n(T_{LT+1})$)LIST

The LIST records must be given in order of increasing value of the temperature T_m. Note that the interpolation scheme I_m is given in the same record position in the LIST record as LT in the TAB1 record. Also note that in the TAB1 record (for the first temperature) pairs of values are given, X(N) and $Y(X,T_1)$, while in the LIST record only values of $Y(X,T_2)$ are given. It is implied that $Y(X,T_2)$ given at the N^{th} point is for the same value of X(N) as is given for $Y(X,T_1)$. This means that the X mesh is given only once, for first temperature.

If a cross section exhibits a temperature dependence, it will generally occur only at low neutron energies, and the high energy data will be independent of temperature. Therefore, the LIST records for the second and higher temperatures may contain NP's that are less than the NP given on the TAB1 record. If the subscript n denotes the temperature, the following condition is defined:

$$NP_1 \geq NP_2 \geq \cdots\cdots\cdots \geq NP_{LT+1}.$$

For example, consider the fission cross section for a particular material where resonance parameters are not given. $\sigma_f(E)$ may be described by 1000 energy points (NP = 1000) that cover the energy range from 10^{-5} eV to 15.0×10^6 eV for a temperature of 293.0°K. These data would be given in a TAB1 record. If the fission cross section is given at 600.0°K and temperature effects are not important for neutron energies above 1.0×10^3 eV (described in the TAB1 record by the first 500 points), then a LIST record is given for 600°K and NP would be equal to 500. It is implied that the first 500 energy points for both sets of data are exactly the same.

If the temperature dependence refers to data already in a LIST record, all records are of the LIST type. The first LIST record contains the data for the first (lowest) temperature.

$$\left[\text{MAT, MF, MT/ } T_1 \text{ , C2; LT , } NP_1 \text{ , 0/ } B_n(T_1)\right]\text{LIST}$$

(MAT, MF, MT/ T_2 , C2; I_2 , L2; NP_2 , 0/ $B_n(T_2)$)LIST

(MAT, MF, MT/ T_3 , C_2; I_3 , L2; NP_3 , 0/ $B_n(T_2)$)LIST

(MAT, MF, MT/ T_{LT+1}, C2; I_{LT+1}, L2, NP_{LT+1}, 0/ $B_n(T_{LT+1})$)LIST

The same rules apply as for NP, i.e.,

$NP_1 \geq NP_2 \geq \ldots \geq NP_{LT+1}$.

The above mechanism is used in File 1 to describe the variation of fission product yields with incident neutron energy. In this special case, the neutron energy replaces the temperature in the above illustration, and the interpolation code I_m refers to neutron energy.

APPENDIX G

Alternative Structure for ENDF Data Tapes

The standard structure of an ENDF tape was described in Section 0.4.2 of this report. The standard structure is well suited for BCD (card image) and binary tapes. An alternative structure of the ENDF data tapes has been developed for use in certain cross section processing codes. This alternate arrangement, illustrated in Figure G.1, is simply an interchange of materials and files. The hierarchy is now MF, MAT, and MT.

Processing programs have been written that will convert an ENDF data tape (either BCD card image or binary) from the standard structure to the alternate structure (see Appendix I).

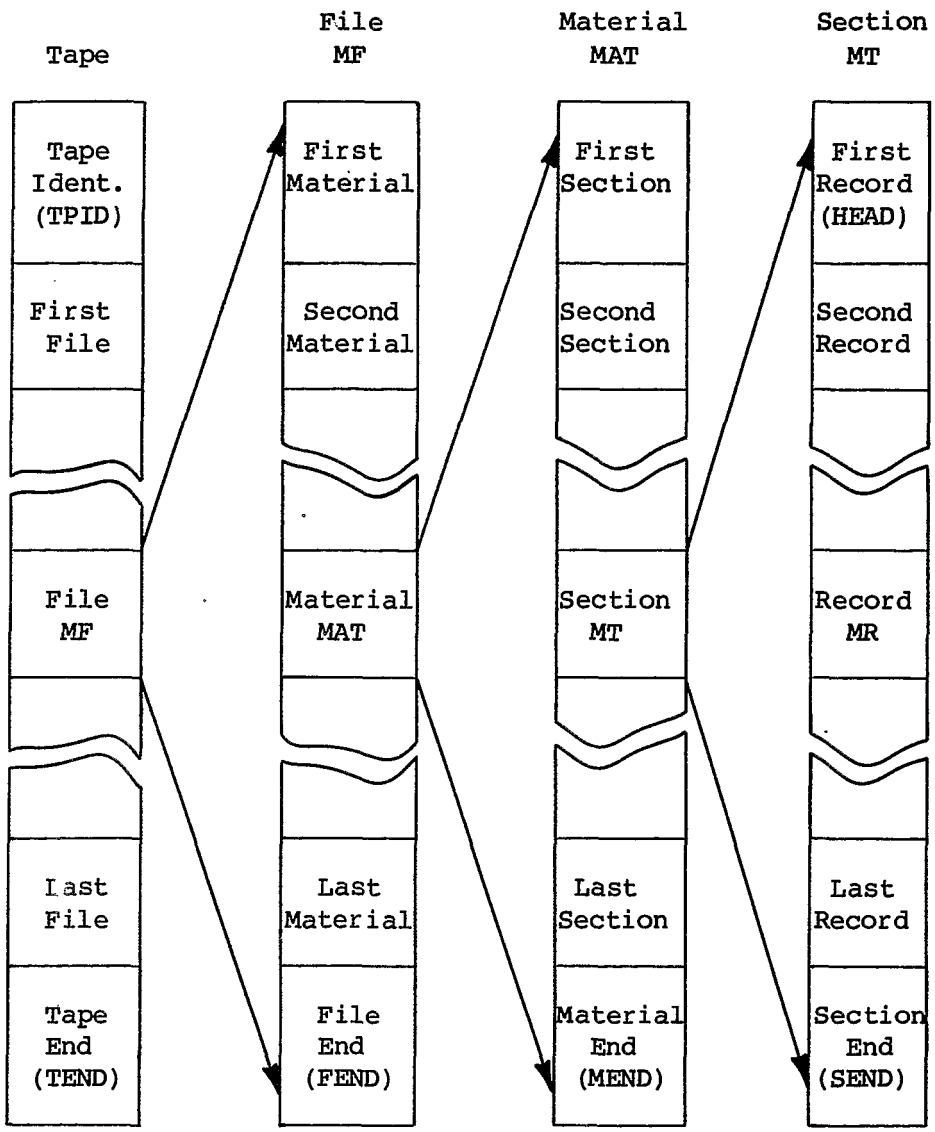


Figure G.1. Alternate Arrangement of an ENDF Tape

APPENDIX H

Data Formats for the ENDF/A Library

The data formats and procedures to be used for the ENDF/A library are essentially the same as those used for the ENDF/B. All processing codes, such as CHECKER, RIGEL, and ET/E, will be able to read the data tape, whether the tape is an ENDF/A or ENDF/B data type. The only difference between the two tapes is a flag in the HEAD record of the first section, MT = 451, in File 1. Also, the first part of the Hollerith information (first two BCD card-image records) will have an artificial structure. The modified structure for an ENDF/A tape is

(MAT, 1, 451/ZA, AWR; LRP, LFI; NLIB, NMOD)HEAD

(MAT, 1, 451/ELIS,STA,LIS,LISØ,0,0)CONT

(MAT, 1, 451/0.0, 0.0, 0, 0; NWD, NXC/

AID, ALAB, DATE1, AUTH /

REF, DATE2, DATE3, EMIN, EMAX/H (N))LIST

(MAT, 1, 451/0.0, 0.0; MF_1, MT_1, NC_1, MOD_1)CØNT

(MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}; NC_{NXC}, MOD_{NXC})CØNT

(MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0)SEND

where

NLIB is flag to indicate the type of data tape.

If NLIB = 0 - ENDF/B tape,

= 1 - ENDF/A tape,

= 2 - ENDF/A tape (translated from UKAEA library)

Code	Org.	Custodian	Comments
ETOE-2/MC^2-2	ANL	H. Henryson	Prepares broad group neutron cross sections for fast reactor calculations. Also used to prepare fine group neutron cross section libraries for use in the SDX code.
ETOG3	BNL	P. Rose	Special version of ETOG for use in Technion version of HAMMER.
ETOG-5	WNES	W. Henderson	Prepares neutron cross sections for use in the MUFT, GAM, ANISN, and LASER codes. WCAP-3845-1.
ETOP-15	EG&G	R. Grimesy	Multigroup processor for the PHROG fast spectrum code.
ETOMX	BAPL	J. Hardy	Prepares neutron cross sections for use in the MUFT and GAM codes.
ETOT-5	WNES	W. Henderson	Prepares pointwise or group thermal neutron cross sections from Files 2 and 3 of ENDF/B format data. WCAP-7363.
ETOX	HEDL	R. Schenter	Prepares neutron cross sections and shielding factor tables for use in the 1DX code.
ETOX	LASL	R. Kidman	Same as above.

APPENDIX H

Data Formats for the ENDF/A Library

The data formats and procedures to be used for the ENDF/A library are essentially the same as those used for the ENDF/B. All processing codes, such as CHECKER, RIGEL, and ET/E, will be able to read the data tape, whether the tape is an ENDF/A or ENDF/B data type. The only difference between the two tapes is a flag in the HEAD record of the first section, MT = 451, in File 1. Also, the first part of the Hollerith information (first two BCD card-image records) will have an artificial structure. The modified structure for an ENDF/A tape is

(MAT, 1, 451/ZA, AWR; LRP, LFI; NLIB, NMOD)HEAD

(MAT, 1, 451/ELIS,STA,LIS,LISØ,0,0)CONT

(MAT, 1, 451/0.0, 0.0, 0, 0; NWD, NXC/

AID, ALAB, DATE1, AUTH /

REF, DATE2, DATE3, EMIN, EMAX/H (N))LIST

(MAT, 1, 451/0.0, 0.0; MF_1, MT_1, NC_1, MOD_1)CØNT

(MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}; NC_{NXC}, MOD_{NXC})CØNT

(MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0)SEND

where

NLIB is flag to indicate the type of data tape.

If NLIB = 0 - ENDF/B tape,

= 1 - ENDF/A tape,

= 2 - ENDF/A tape (translated from UKAEA library),

Code	Org.	Custodian	Comments
ETOE-2/MC^2-2	ANL	H. Henryson	Prepares broad group neutron cross sections for fast reactor calculations. Also used to prepare fine group neutron cross section libraries for use in the SDX code.
ETOG3	BNL	P. Rose	Special version of ETOG for use in Technion version of HAMMER.
ETOG-5	WNES	W. Henderson	Prepares neutron cross sections for use in the MUFT, GAM, ANISN, and LASER codes. WCAP-3845-1.
ETOP-15	EG&G	R. Grimesy	Multigroup processor for the PHROG fast spectrum code.
ETOMX	BAPL	J. Hardy	Prepares neutron cross sections for use in the MUFT and GAM codes.
ETOT-5	WNES	W. Henderson	Prepares pointwise or group thermal neutron cross sections from Files 2 and 3 of ENDF/B format data. WCAP-7363.
ETOX	HEDL	R. Schenter	Prepares neutron cross sections and shielding factor tables for use in the 1DX code.
ETOX	LASL	R. Kidman	Same as above.

Code	Org.	Custodian	Comments
FIZCON	BNL	NNDC	Check ENDF data for physics consistency and check that recommended procedures are followed.
FLANGE II	SRL	D. Finch	Prepares thermal neutron cross sections from ENDF/B data including S(,) data in file 7. No Adler-Adler capability.
GFE4/GAND3	GA	D. Mathews	Prepares neutron cross sections for use in the GGC-4, GGC-5 and MICROX codes.
INTEND	BNL	NNDC	Computes a variety of integral quantities from a pointwise ENDF file.
INTER	BNL	NNDC	Same as above.
LISTFC	BNL	NNDC	Generates interpreted listings of ENDF files.
MACK	ANL	Y. Gohar	Calculates KERMA factors, activation, and gamma-ray production data.
MINX	LASL	R. MacFarlane	Prepares neutron cross section and shielding factor tables for the SPHINX code. CDC version.
MINX	ORNL	C. Weisbin	Same as above. IBM version.

Code	Org.	Custodian	Comments
NJOY	LASL	R. MacFarlane	General cross section processing code for fast neutrons, thermal neutrons, photon production, photo interaction, heat production, and covariances in multigroup and pointwise forms and several different library formats.
PLOTEF	BNL	NNDC	Plotting code for ENDF.
PSYCHE	BNL	NNDC	Physics tests on ENDF files.
PUFF	ORNL	C. Weisbin	Processes covariance data for use in sensitivity analysis.
RESEND	BNL	NNDC	Prepares infinitely dilute 0 K pointwise cross sections from File 2 + 3 information.
RIGEL	BNL	NNDC	ENDF file Editing code. Creates ENDF binary file formatted tapes.
SAMF	MAGI	H. Lichtenstein	A continuous energy Monte Carlo code for analysis of problems involving shielding and fast or thermal reactors.
SAMX	Magi	M. Beer	Prepares cross sections for the SAMF Monte-Carlo code.
SIGMA1	LLL	D. Cullen	Doppler broadens a linearized, pointwise ENDF file.

Code	Org.	Custodian	Comments
STNDRD	BNL	NNDC	To standardize data formats, update dictionary, and resequence an ENDF file.
SUMRIZ	BNL	NNDC	Summary of ENDF file.
SUPERTON	GE	C. Stuart	Prepares neutron cross sections for codes of the GAM/MUFT type.
UNCER	UW	C. Maynard	Processes covariance file data.
VIM	ANL	H. Henryson	A continuous energy Monte Carlo code for fast and thermal reactor analysis.

APPENDIX J

Materials in the ENDF/B-V Library

The following is a list of materials that constitute the ENDF/B-V Library. Those materials found on ENDF Tapes 501-517 are referred to as General Purpose Evaluations. Also included in the library are partial evaluations for the Special Purpose Actinide (521,522), Dosimetry (531), Activation (532), Gas Production (533), and Fission Product (541-546) files.

Other materials exist and are available in the ENDF format. For a list of materials in the ENDF/A Library please contact NNDC. Moderating Materials (i.e., Scattering Law Data) have been carried over from ENDF/B-III for H_2O, D_2O, Beryllium, BeO, Graphite, Polyethylene, Benzene, Zr in ZrH, and H in ZrH.

APPENDIX J

Materials in the ENDF/B-V Library

Z EL	A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
1-H -	1	1301	Neut. + gamma prod. + error	521	LASL	LA-4574(1971)	JAN 77	L. STEWART, R.J. LABAUVE, P.G. YOUNG	511
1-H -	2	1302	Neutron + gamma production	917	LASL	LA-3271(1968)	JAN 77	L. STEWART (LASL) A. HORSLEY (AWRE)	505
1-H -	3	1169	Neutron+decay data	856	LASL	LA-3270(65) UPDATE67	OCT 74	LEONA STEWART	501
1-H -	3	7013	Decay data only	30	INEL		APR 78	REICH	532
2-He-	3	1146	Neutron cross sections only	412	LASL	NO PUBLICATION	JUN 68	LEONA STEWART (LASL)	511
2-He-	4	1270	Neutron cross sections only	434	LASL	NO PUBLICATION	OCT 73	NISLEY, HALE, YOUNG (LASL)	501
3-Li-	6	1303	Neut. + gamma prod. +error	2664	LASL		SEP 77	G. HALE, L. STEWART, P.G. YOUNG	511
3-Li-	6	6424	Neutron+err.files	200	LASL		DEC 78	L. STEWART, G. HALE, P. YOUNG	531
3-Li-	6	7036	Neutron cross sections only	157	LASL		SEP 77	G. HALE, L. STEWART, P.G. YOUNG	532
3-Li-	7	1272	Neutron + gamma production	700	LASL		OCT 72	R.J. LABAUVE, L. STEWART, M. BATTAT	505
3-Li-	7	7037	Neutron cross sections only	101	LASL		OCT 72	R.J. LABAUVE, L. STEWART, M. BATTAT	532
3-Li-	8	7038	Decay data only	55	INEL		APR 78	REICH	532
4-Be-	9	1304	Neutron + gamma production	2792	LLL		OCT 76	HOWERTON, PERKINS	505
5-B -	10	1305	Neut. + gamma prod. +error	3635	LASL		JAN 77	G. HALE, L. STEWART, P. YOUNG	511
5-B -	10	6425	Neutron+err.files	180	LASL		JAN 79	L. STEWART, G. HALE, P. YOUNG	531
5-B -	10	7050	Neutron cross sections only	94	LASL		JAN 77	G. HALE, L. STEWART, P. YOUNG	532
5-B -	11	1160	Neutron cross sections only	1035	GE-BNL		NOV 74	C. COWAN	501
6-C -	0	1306	Neut. + gamma prod. +error	2713	ORNL	INDC(F/R)-7/L	JAN 77	C.Y. FU AND F.G. PEREY	511
6-C -	14	7064	Decay data only	29	INEL		APR 78	REICH	532
7-N -	14	1275	Neut. + gamma prod. +error	6223	LASL	LA-4725 (1972)	JUN 75	P. YOUNG, D. FOSTER, JR., G. HALE	505
7-N -	14	7074	Neutron cross sections only	167	LASL	LA-4725 (1972)	JUN 75	P. YOUNG, D. FOSTER, JR., G. HALE	532
7-N -	15	1307	Neutron + gamma production	3910	LASL		MAR 77	E. ARTHUR, P. YOUNG, G. HALE	505
7-N -	16	7076	Decay data only	127	INEL		APR 78	REICH	532
8-O -	16	1276	Neut. + gamma prod. +error	6151	LASL	LA-4780 (1972)	JUN 75	P. YOUNG, D. FOSTER, JR., G. HALE	505
8-O -	16	7086	Neutron cross sections only	86	LASL	LA-4780 (1972)	JUN 75	P. YOUNG, D. FOSTER, JR., G. HALE	532
8-O -	17	1317	Neutron cross sections only	874	BNL		JAN 78	B.A. MAGURNO	505
9-F -	18	7098	Decay data only	39	INEL		APR 78	REICH	532
9-F -	19	1309	Neut. + gamma prod. +error	4826	ORNL		DEC 76	C.Y. FU, D.C. LARSON, F.G. PEREY	503
9-F -	19	7099	Neutron+err.files	86	ORNL		NOV 79	C.Y. FU, D.C. LARSON, F.G. PEREY	532
10-Ne-	23	7103	Decay data only	48	INEL		APR 78	REICH	532
11-Na-	22	7112	Decay data only	48	INEL		APR 78	REICH	532
11-Na-	23	1311	Neut.(RP) + gamma prod. +error	4447	ORNL		DEC 77	D.C. LARSON	506
11-Na-	23	6311	Neut.(RP)+err.files	181	ORNL		DEC 77	D.C. LARSON	531
11-Na-	23	7113	Neut.(RP)+err.files	258	ORNL		NOV 79	D.C. LARSON	532
11-Na-	24	7114	Decay data only	48	INEL		APR 78	REICH	532
12-Mg-	0	1312	Neutron + gamma production	4704	ORNL		FEB 78	D.C. LARSON	506
12-Mg-	24	7124	Neutron cross sections only	71	HEDL,ORNL		NOV 79	MANN, LARSON	532
12-Mg-	27	7127	Decay data only	37	INEL		APR 78	REICH	532
13-Al-	27	1313	Neut. + gamma prod. +error	6206	LASL	LA-4726 (1973).	AUG 77	P.G. YOUNG, D.G. FOSTER, JR.	506
13-Al-	27	6313	Neutron+err.files	186	LASL	LA-4726 (1973).	AUG 77	P.G. YOUNG, D.G. FOSTER, JR.	531
13-Al-	27	7137	Neutron cross sections only	238	LASL	LA-4726 (1973).	AUG 77	P.G. YOUNG, D.G. FOSTER, JR.	532
13-Al-	28	7138	Decay data only	31	INEL		APR 78	REICH	532
14-Si-	0	1314	Neut. + gamma prod. +error	12882	ORNL		DEC 76	LARSON, PEREY, DRAKE, YOUNG	507

Z EL	A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
14-Si-	31	7141	Decay data only	53	INEL		APR 78	REICH	532
15-P -	31	1315	Neutron + gamma production	956	LLL	UCRL 50400 VOL15(E)	OCT 77	HOWERTON	503
15-P -	31	7151	Neutron cross sections only	48	LLL	UCRL 50400 VOL15(E)	OCT 77	HOWERTON	532
15-P -	32	7152	Decay data only	29	INEL		APR 78	REICH	532
16-S -	0	1347	Neutron(RP)+gamma production	4738	BNL		APR 79	DIVADEENAM	517
16-S -	32	1316	Neutron + gamma production	979	LLL	UCRL 50400 VOL15(E)	OCT 77	HOWERTON	503
16-S -	32	7162	Neutron cross sections only	51	LLL	UCRL 50400 VOL15(E)	OCT 77	HOWERTON	532
17-Cl-	0	1149	Neutron + gamma production	3783	GGA	GA-7829 VOL-4(1967)	FEB 67	M.S. ALLEN AND M.K. DRAKE	513
18-Ar-	40	7180	Neutron cross sections only(RP)	72	HEDL		JAN 79	MANN	532
18-Ar-	41	7181	Decay data only	65	INEL		APR 78	REICH	532
19-K -	0	1150	Neutron + gamma production	3946	GGA	GA-7829 VOL-5(1967)	FEB 67	M.K. DRAKE	513
19-K -	41	7191	Neutron cross sections only	56	HEDL		JAN 79	MANN	532
20-Ca-	0	1320	Neutron + gamma production	5999	ORNL	ADNDT 17(1976)127.	OCT 76	C.Y. FU AND F.G. PEREY	507
20-Ca-	45	7205	Decay data only	29	INEL		APR 78	REICH	532
20-Ca-	47	7207	Decay data only	70	INEL		APR 78	REICH	532
21-Sc-	44m	7213	Decay data only	75	INEL		APR 78	REICH	532
21-Sc-	44	7214	Decay data only	74	INEL		APR 78	REICH	532
21-Sc-	45	6426	Neut.(RP)+err.files	280	BNL		JUL 79	MAGURNO AND MUGHABGHAB	531
21-Sc-	45	7215	Neutron cross sections only(RP)	347	BNL		JUL 79	MAGURNO MUGHABGHAB, AND SCHWERER	532
21-Sc-	46	7216	Decay data only	63	INEL		APR 78	REICH	532
21-Sc-	47	7217	Decay data only	57	INEL		APR 78	REICH	532
21-Sc-	48	7218	Decay data only	75	INEL		APR 78	REICH	532
22-Ti-	0	1322	Neutron + gamma production	5436	BURANLLLL	ANL/NDM-28,1977	AUG 77	C. PHILIS, A. SMITH, R. HOWERTON	508
22-Ti-	46	6427	Neutron+err.files	61	ANL		JAN 77	C. PHILIS, O. BERSILLON, D. SMITH, ETC.	531
22-Ti-	46	7226	Neutron cross sections only	64	ANL		JAN 77	C. PHILIS, O. BERSILLON, D. SMITH, ETC.	532
22-Ti-	47	6428	Neutron+err.files	90	ANL		JAN 77	C. PHILIS, O. BERSILLON, D. SMITH, ETC.	531
22-Ti-	47	7227	Neutron cross sections only	100	ANL		JAN 77	C. PHILIS, O. BERSILLON, D. SMITH, ETC.	532
22-Ti-	48	6429	Neutron+err.files	89	ANL		JAN 77	C. PHILIS, O. BERSILLON, D. SMITH ETC.	531
22-Ti-	48	7228	Neutron cross sections only	127	ANL,LASL+		JAN 77	C. PHILIS, ARTHUR, FU+	532
22-Ti-	50	7220	Neutron cross sections only	44	LASL		MAY 79	E. ARTHUR	532
23-V -	0	1323	Neutron + gamma production	2375	ANLLLLHEDL	ANL/NDM-24,1977	JAN 77	A. SMITH+, H. HOWERTON, F. MANN	508
24-Cr-	0	1324	Neut.(RP) + gamma prod. +error	13517	BNL		DEC 77	A. PRINCE AND T.W. BURROWS	512
24-Cr-	49	7249	Decay data only	95	INEL		APR 78	REICH	532
24-Cr-	50	7240	Neutron cross sections only(RP)	180	BNL		AUG 79	A. PRINCE	532
24-Cr-	51	7241	Decay data only	57	INEL		APR 78	REICH	532
24-Cr-	52	7242	Neutron cross sections only	49	BNL		AUG 79	A.PRINCE	532
25-Mn-	54	7254	Decay data only	54	INEL		APR 78	REICH	532
25-Mn-	55	1325	Neut.(RP) + gamma prod. +error	2887	BNL		MAR 77	S.F. MUGHABGHAB	508
25-Mn-	55	6325	Neutron+err.files	64	BNL		MAR 77	S.F. MUGHABGHAB	531
25-Mn-	55	7255	Neutron cross sections only(RP)	306	BNL		MAR 77	S.F. MUGHABGHAB	532
25-Mn-	56	7256	Decay data only	93	INEL		APR 78	REICH	532
26-Fe-	0	1326	Neut.(RP) + gamma prod. +error	8982	ORNL	ORNL-4617(1970)	OCT 77	C.Y. FU AND F.G. PEREY	513
26-Fe-	54	6430	Neutron+err.files	80	HEDL		JUN 79	R. SCHENTER F. SCHMITTROTH F. MANN	531
26-Fe-	54	7264	Neutron cross sections only(RP)	295	HEDL,ORNL		JUN 79	SCHENTER ET AL. AND FU	532
26-Fe-	55	7265	Decay data only	59	INEL		APR 78	REICH	532
26-Fe-	56	6431	Neutron+err.files	109	ORNL		JUL 78	C.Y. FU	531
26-Fe-	56	7266	Neutron cross sections only	127	ORNL		JUL 78	C.Y. FU	532
26-Fe-	58	6432	Neut.(RP)+err.files	159	HEDL		JUN 79	R. SCHENTER F. SCHMITTROTH F. MANN	531
26-Fe-	58	7268	Neutron cross sections only(RP)	133	HEDL		JUN 79	R. SCHENTER F. SCHMITTROTH F. MANN	532
26-Fe-	59	7269	Decay data only	94	INEL		APR 78	REICH	532

Z EL	A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
27-Co-	57	7277	Decay data only	79	INEL		APR 78	REICH	532
27-Co-	58	7278	Decay data only	67	INEL		APR 78	REICH	532
27-Co-	59	1327	Neutron(RP)+gamma production	3326	BNL		JUN 77	S. MUGHABGHAB	501
27-Co-	59	6327	Neut.(RP)+err.files	368	BNL		JUN 77	S. MUGHABGHAB	531
27-Co-	59	7279	Neutron cross sections only(RP)	382	BNL		JUN 77	S. MUGHABGHAB	532
27-Co-	60	7270	Decay data only	69	INEL		APR 78	REICH	532
28-Ni-	0	1328	Neut.(RP) + gamma prod. +error	9287	BNL(NNDC)		MAR 77	M. DIVADEENAM	512
28-Ni-	57	7287	Decay data only	105	INEL		APR 78	REICH	532
28-Ni-	58	6433	Neutron+err.files	130	BNL		MAR 77	M. DIVADEENAM	531
28-Ni-	58	7288	Neutron cross sections only	1535	BNL		MAY 78	DI VADEENAM	532
28-Ni-	59	7289	Decay data only	66	INEL		APR 78	REICH	532
28-Ni-	60	6434	Neutron+err.files	71	BNL		MAR 77	M. DIVADEENAM	531
28-Ni-	60	7280	Neutron cross sections only	116	BNL		MAY 78	DIVADEENAM	532
28-Ni-	62	7282	Neutron cross sections only	1530	BNL		MAY 78	DIVADEENAM	532
28-Ni-	63	7283	Decay data only	26	INEL		APR 78	REICH	532
29-Cu-	0	1329	Neutron(RP)+gamma production	3731	ORNL,SAI	DNA-3356F(1974)	MAR 78	FU, DRAKE, FRICKE	508
29-Cu-	63	6435	Neut.(RP)+err.files	298	ORNL		JUL 78	C.Y. FU	531
29-Cu-	63	7293	Neutron cross sections only(RP)	250	ORNL		JUL 78	C.Y. FU	532
29-Cu-	64	7294	Decay data only	59	INEL		APR 78	REICH	532
29-Cu-	65	6436	Neutron+err.files	96	ORNL		JUL 78	C.Y. FU	531
29-Cu-	65	7295	Neutron cross sections only	79	ORNL		JUL 78	C.Y. FU	532
36-Kr	78	1330	Neutron cross sections only(RP)	1085	BNL		DEC 78	A. PRINCE	509
36-Kr-	80	1331	Neutron cross sections only(RP)	1037	BNL		DEC 78	A. PRINCE	509
36-Kr-	82	1332	Neutron cross sections only(RP)	1075	BNL		DEC 78	A. PRINCE	509
36-Kr-	83	1333	Neutron cross sections only(RP)	1097	BNL		DEC 78	A. PRINCE	509
36-Kr-	84	1334	Neutron cross sections only(RP)	910	BNL		APR 78	A. PRINCE	509
36-Kr-	86	1336	Neutron cross sections only(RP)	914	BNL		JUN 75	A. PRINCE	509
40-Zr-	0	1340	Neutron cross sections only(RP)	1859	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	508
40-Zr-	89m	7408	Decay data only	75	INEL		APR 78	REICH	532
40-Zr-	89	7409	Decay data only	73	INEL		APR 78	REICH	532
40-Zr-	90	1385	Neutron cross sections only(RP)	1127	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	501
40-Zr-	90	7400	Neutron cross sections only	74	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	532
40-Zr-	91	1386	Neutron cross sections only(RP)	1349	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	501
40-Zr-	92	1387	Neutron cross sections only(RP)	1183	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	501
40-Zr-	92	7402	Neutron cross sections only(RP)	160	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	532
40-Zr-	93	7403	Decay data only	28	INEL		APR 78	REICH	532
40-Zr-	94	1388	Neutron cross sections only(RP)	1258	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	501
40-Zr-	94	7404	Neutron cross sections only(RP)	170	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	532
40-Zr-	95	7405	Decay data only	73	INEL		APR 78	REICH	532
40-Zr-	96	1389	Neutron cross sections only(RP)	1068	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGIS, T. MAUNG	501
41-Nb-	92m	7411	Decay data only	65	INEL		APR 78	REICH	532
41-Nb-	92	7412	Decay data only	59	INEL		APR 78	REICH	532
41-Nb-	93	1189	Neutron(RP)+gamma production	2190	ANL,LLL		MAY 74	R. HOWERTON (LLL) AND A. SMITH	510
41-Nb-	93m	7410	Decay data only	53	INEL		APR 78	REICH	532
41-Nb-	93	7413	Neutron cross sections only(RP)	427	ANL,LLL		MAY 74	R. HOWERTON (LLL) AND A. SMITH	532
41-Nb-	94	7414	Decay data only	61	INEL		APR 78	REICH	532
42-Mo-	0	1321	Neutron(RP)+gamma production	1838	LLL,HEDL		FEB 79	HOWERTON, SCHMITTROTH, SCHENTER	513
42-Mo-	92	7422	Neutron cross sections only(RP)	89	HEDL		FEB 80	SCHENTER, SCHMITTROTH, ET AL	532
42-Mo-	93	7423	Decay data only	46	INEL		APR 78	REICH	532
42-Mo-	98	7428	Neutron cross sections only(RP)	96	HEDL		FEB 80	SCHENTER, SCHMITTROTH, ET AL	532

Z EL A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
42-Mo- 99	7429	Decay data only	141	INEL		APR 78	REICH	532
42-Mo-100	7420	Neutron cross sections only(RP)	120	HEDL		FEB 80	SCHENTER, SCHMITTROTH, ET AL	532
42-Mo-101	7421	Decay data only	471	INEL		APR 78	REICH	532
43-Tc- 99	1308	Neut.(RP)+decay data	857	HEDL,BAW		NOV 78	SCHENTER, LIVOLSI, SCHMITTROTH, ET AL	510
45-Rh-103	1310	Neutron cross sections only(RP)	995	HEDL,BAW		NOV 78	SCHENTER, LIVOLSI, SCHMITTROTH, ET AL	510
47-Ag-107	1371	Neutron cross sections only(RP)	902	HEDL,BNL		NOV 78	SCHENTER, BHAT, PRINCE, JOHNSON, ET AL	510
47-Ag-109	1373	Neutron cross sections only(RP)	831	HEDL,BNL		NOV 78	SCHENTER, BHAT, PRINCE, JOHNSON, ET AL	510
47-Ag-109	7479	Neutron cross sections only(RP)	185	HEDL,BNL		NOV 78	SCHENTER, BHAT, PRINCE, JOHNSON, ET AL	532
47-Ag-110	7470	Decay data only	104	INEL		APR 78	REICH	532
47-Ag-110m	7471	Decay data only	194	INEL		APR 78	REICH	532
48-Cd- 0	1281	Neutron cross sections only	2489	BNL		NOV 74	S. PEARLSTEIN (TRANS FROM U.K.)	501
48-Cd-113	1318	Neut.(RP)+decay data	582	BNL,HEDL		NOV 78	PEARLSTEIN, MANN, SCHENTER	510
49-In-115	6437	Neut.(RP)+err.files	305	HEDL/ANL		JAN 78	F. SCHMITTROTH/D.L. SMITH	531
49-In-115m	7494	Decay data only	67	INEL		APR 78	REICH	532
49-In-115	7495	Neutron cross sections only(RP)	313	HEDL/ANL		JAN 78	F. SCHMITTROTH/D.L. SMITH	532
49-In-116	7496	Decay data only	95	INEL		APR 78	REICH	532
50-Sn-120	7500	Neutron cross sections only(RP)	142	HEDL		OCT 74	R.E. SCHENTER AND F. SCHMITTROTH	532
50-Sn-121	7501	Decay data only	29	INEL		APR 78	REICH	532
50-Sn-121m	7506	Decay data only	58	INEL		APR 78	REICH	532
50-Sn-122	7502	Neutron cross sections only(RP)	98	HEDL		OCT 74	R.E. SCHENTER AND F. SCHMITTROTH	532
50-Sn-123	7503	Decay data only	83	INEL		APR 78	REICH	532
50-Sn-123m	7507	Decay data only	64	INEL		APR 78	REICH	532
50-Sn-124	7504	Neutron cross sections only(RP)	110	HEDL		OCT 74	R.E. SCHENTER AND F. SCHMITTROTH	532
50-Sn-125	7505	Decay data only	150	INEL		APR 78	REICH	532
50-Sn-125m	7508	Decay data only	98	INEL		APR 78	REICH	532
53-I -126	7536	Decay data only	117	INEL		APR 78	REICH	532
53-I -127	6438	Neutron+err.files	75	STANFORD		AUG 72	R. SHER	531
53-I -127	7537	Neutron cross sections only(RP)	223	HEDL		FEB 80	SCHENTER, SCHMITTROTH, ET AL	532
53-I -128	7538	Decay data only	101	INEL		APR 78	REICH	532
54-Xe-124	1335	Neutron cross sections only(RP)	742	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-126	1339	Neutron cross sections only(RP)	746	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-128	1348	Neutron cross sections only(RP)	710	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-129	1349	Neutron cross sections only(RP)	876	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-130	1350	Neutron cross sections only(RP)	815	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-131	1351	Neutron cross sections only(RP)	849	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-132	1352	Neutron cross sections only(RP)	721	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-134	1354	Neutron cross sections only(RP)	681	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
54-Xe-135	1294	Neutron+decay data	864	BNW	PRI.COM.JUNE,1967	MAY 75	B.R. LEONARD, JR. AND K.B. STEWART	509
54-Xe-136	1356	Neutron cross sections only	679	BNL		MAR 78	M.R. BHAT AND S.F. MUGHABGHAB	509
55-Cs-133	1355	Neutron cross sections only(RP)	885	HEDL,BNL		NOV 78	SCHENTER, BHAT, PRINCE, JOHNSON, ET AL	510
56-Ba-138	1353	Neutron + gamma production	1117	LLL	UCRL-50400 VOL. 15	AUG 78	HOWERTON	508
57-La-139	7579	Neutron cross sections only(RP)	150	HEDL		FEB 80	SCHENTER, SCHMITTROTH, ET AL	532
57-La-140	7570	Decay data only	193	INEL		APR 78	REICH	532
62-Sm-149	1319	Neut.(RP)+decay data	1643	HEDL,BNW		NOV 78	SCHENTER, LEONARD, STEWART, ET AL	510
63-Eu-151	1357	Neutron(RP)+gamma production	1917	BNL		DEC 77	S.F. MUGHABGHAB	509
63-Eu-152	1292	Neut.(RP)+decay data	3248	BNL		JUN 75	H. TAKAHASHI	509
63-Eu-153	1359	Neutron(RP)+gamma production	2110	BNL		FEB 78	S. MUGHABGHAB	509
63-Eu-154	1293	Neut.(RP)+decay data	2465	BNL		JUN 75	H. TAKAHASHI	509
64-Gd-152	1362	Neutron cross sections only(RP)	1453	BNL		JAN 77	B.A. MAGURNO	503
64-Gd-154	1364	Neutron cross sections only(RP)	1507	BNL		JAN 77	B.A. MAGURNO	503

Z EL A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
64-Gd-155	1365	Neutron cross sections only(RP)	1602	BNL		JAN 77	B.A. MAGURNO	503
64-Gd-156	1366	Neutron cross sections only(RP)	1495	BNL		JAN 77	B.A. MAGURNO	503
64-Gd-157	1367	Neutron cross sections only(RP)	1536	BNL		JAN 77	B.A. MAGURNO	503
64-Gd-158	1368	Neutron cross sections only(RP)	1524	BNL		JAN 77	B.A. MAGURNO	503
64-Gd-160	1370	Neutron cross sections only(RP)	1259	BNL		JAN 77	B.A. MAGURNO	503
66-Dy-164	1031	Neutron cross sections only(RP)	1312	BNW	PRI.COMM.JUNE,1967	JUN 67	B.R. LEONARD, JR. AND K.B. STEWART	501
66-Dy-164	7664	Neutron cross sections only(RP)	204	BNW	PRI.COMM.JUNE,1967	JUN 67	B.R. LEONARD, JR. AND K.B. STEWART	532
66-Dy-165	7665	Decay data only	220	INEL		APR 78	REICH	532
66-Dy-165m	7666	Decay data only	104	INEL		APR 78	REICH	532
71-Lu-175	1032	Neutron cross sections only(RP)	1102	BNW	PRI.COMM.JUNE,1967	JUN 67	B.R. LEONARD, JR. AND K.B. STEWART	502
71-Lu-176	1033	Neutron cross sections only(RP)	1172	BNW	PRI.COMM.JUNE,1967	JUN 67	B.R. LEONARD, JR. AND K.B. STEWART	502
72-Hf- 0	1372	Neutron cross sections only(RP)	1106	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-174	1374	Neutron cross sections only(RP)	695	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-176	1376	Neutron cross sections only(RP)	700	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-177	1377	Neutron cross sections only(RP)	898	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-178	1378	Neutron cross sections only(RP)	692	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-179	1383	Neutron cross sections only(RP)	735	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
72-Hf-180	1384	Neutron cross sections only(RP)	687	SAI	EPRI NP-250	APR 76	M. DRAKE, D. SARGES, T. MAUNG	501
73-Ta-181	1285	Neutron(RP)+gamma production	2852	LLL		NOV 74	HOWERTON, PERKINS, MACGREGOR	502
73-Ta-181	7731	Neutron cross sections only(RP)	183	LLL		NOV 74	HOWERTON, PERKINS, MACGREGOR	532
73-Ta-182	1127	Neutron cross sections only(RP)	794	AI	AI-AEC-12990 (1971)	APR 71	J. OTTER, C. DUNFORD, AND E. OTTEWITTE	502
73-Ta-182	7732	Decay data only	227	INEL		APR 78	REICH	532
74-W -182	1128	Neutron(RP)+gamma production	3240	AI,LASL		JUN 73	OTTER, OTTEWITTE, ROSE, YOUNG	502
74-W -183	1129	Neutron(RP)+gamma production	3576	AI,LASL		OCT 74	OTTER, OTTEWITTE, ROSE, YOUNG	502
74-W -184	1130	Neutron(RP)+gamma production	3132	AI,LASL		APR 74	OTTER, OTTEWITTE, ROSE, YOUNG	502
74-W -186	1131	Neutron(RP)+gamma production	3114	AI,LASL		JUN 73	OTTER, OTTEWITTE, ROSE, YOUNG	502
75-Re-185	1083	Neutron cross sections only(RP)	1396	GE(NMPO)	GEMP-587	JAN 68	W.B. HENDERSON AND J.W. ZWICK	503
75-Re-187	1084	Neut.(RP)+decay data	1473	GE(NMPO)	GEMP-587	JAN 68	W.B. HENDERSON AND J.W. ZWICK	513
79-Au-195	7795	Decay data only	88	INEL		APR 78	REICH	532
79-Au-196	7796	Decay data only	139	INEL		APR 78	REICH	532
79-Au-197	1379	Neut.(RP)+err.files	1976	BNL		FEB 77	S.F. MUGHABGHAB	511
79-Au-197	6379	Neut.(RP)+err.files	559	BNL		FEB 77	S.F. MUGHABGHAB	531
79-Au-197	7797	Neutron cross sections only(RP)	624	BNL		FEB 77	S.F. MUGHABGHAB	532
79-Au-198	7798	Decay data only	73	INEL		APR 78	REICH	532
81-Tl-208	8108	Decay data only	169	INEL		AUG 78	REICH	521
82-Pb- 0	1382	Neut. + gamma prod. +error	3992	ORNL	ADNDT,16(1975)409.	AUG 76	C.Y. FU AND F.G. PEREY	508
82-Pb-212	8212	Decay data only	84	INEL		AUG 78	REICH	521
83-Bi-212	8312	Decay data only	169	INEL		NOV 78	REICH	521
84-Po-216	8416	Decay data only	37	INEL		AUG 78	REICH	521
86-Rn-220	8620	Decay data only	33	INEL		AUG 78	REICH	521
88-Ra-224	8824	Decay data only	74	INEL		AUG 78	REICH	521
90-Th-228	8028	Decay data only	80	INEL		AUG 78	REICH	521
90-Th-230	8030	Neut.(RP)+decay data	802	HEDL		NOV 77	MANN	521
90-Th-231	7901	Decay data only	203	INEL		APR 78	REICH	532
90-Th-231	8031	Decay data only	201	INEL		AUG 78	REICH	521
90-Th-232	1390	Neut(RP)+decay+FPY+gam.+err.	7540	BNL		DEC 77	BHAT, SMITH, LEONARD, DESAUSSURE ET AL	516
90-Th-232	6390	Neut.(RP)+err.files	865	BNL		DEC 77	BHAT, SMITH, LEONARD, DESAUSSURE ET AL	531
90-Th-232	7902	Neut.(RP)+decay data	895	BNL		DEC 77	BHAT, SMITH, LEONARD, DESAUSSURE ET AL	532

Z EL A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
90-Th-233	7903	Decay data only	358	INEL		APR 78	REICH	532
90-Th-233	8033	Decay data only	358	INEL		AUG 78	REICH	521
91-Pa-231	7911	Neut.(RP)+decay data	406	HEDL		NOV 77	MANN	532
91-Pa-231	8131	Neut.(RP)+decay data	915	HEDL		NOV 77	MANN	521
91-Pa-232	7912	Decay data only	183	INEL		APR 78	REICH	532
91-Pa-232	8132	Decay data only	181	INEL		AUG 78	REICH	521
91-Pa-233	1391	Neut.(RP)+decay data	1171	HEDL,INEL		MAY 78	MANN, SCHENTER, REICH	514
91-Pa-233	7913	Neutron+decay data	226	HEDL,INEL		MAY 78	MANN, SCHENTER, REICH	532
92-U -232	7922	Decay data only	95	INEL		APR 78	REICH	532
92-U -232	8232	Neut.(RP)+decay data	726	HEDL		NOV 77	MANN	521
92-U -233	1393	Neut(RP)+decay+fiss.prod.yld	6741	LASL-ORNL	TANSAO 28,721,1978	DEC 78	STEWART ET AL, WESTON, MANN,HEDL	516
92-U -233	7923	Neutron+decay data	423	LASL-ORNL	TANSAO 28,721,1978	DEC 78	STEWART ET AL, WESTON, MANN,HEDL	532
92-U -234	1394	Neut.(RP)+decay data	1171	BNL,HEDL,+		JUL 78	DIVADEENAM, MANN, DRAKE, REICH, ET AL	514
92-U -235	1395	Neut(RP)+decay+FPY+gam.+err.	10238	BNL		NOV 77	M.R. BHAT	511
92-U -235	6395	Neut.(RP)+err.files	1440	BNL		APR 77	M.R. BHAT	531
92-U -236	1396	Neut(RP)+decay+fiss.prod.yld	2859	BNL,HEDL,+		JUL 78	DIVADEENAM, MANN, MCCROSSON, REICH	+514
92-U -237	8237	Neut.(RP)+decay+gamma prod.data	1778	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER (SEE COMMENTS)	521
92-U -238	1398	Neut(RP)+decay+FPY+gam.+err.	8461	ANL+	ANL/NDM-32	MAR 79	E. PENNINGTON, A. SMITH, W. POENITZ	516
92-U -238	6398	Neut.(RP)+err.files	861	ANL+	ANL/NDM-32	MAR 79	E. PENNINGTON, A. SMITH, W. POENITZ	531
92-U -238	7928	Neut.(RP)+decay data	850	ANL+	ANL/NDM-32	MAR 79	E. PENNINGTON, A. SMITH, W. POENITZ	532
92-U -239	7929	Decay data only	390	INEL		APR 78	REICH	532
92-U -239	8239	Decay data only	385	INEL		AUG 78	REICH	521
93-Np-236	8336	Decay data only	133	INEL		AUG 78	REICH	521
93-Np-236m	8346	Decay data only	106	INEL		AUG 78	REICH	521
93-Np-237	1337	Neut(RP)+decay+FPY+err.	4196	HEDL,SRL,+	HEDL TME 77-54	APR 78	MANN, BENJAMIN, SMITH, STEIN, REICH,	+514
93-Np-237	6337	Neut.(RP)+err.files	1235	HEDL,SRL,+	HEDL TME 77-54	APR 78	MANN, BENJAMIN, SMITH, STEIN, REICH,	+531
93-Np-238	8338	Neut.(RP)+decay data	602	SRL		AUG 75	BENJAMIN AND MCCROSSON	521
93-Np-239	8339	Decay data only	196	INEL		AUG 78	REICH	521
94-Pu-236	8436	Neut.(RP)+decay data	667	HEDL,SRL	HEDL TME 77-54	APR 78	MANN, SCHENTER, BENJAMIN, MCCROSSON	521
94-Pu-237	8437	Neutron+decay data	742	HEDL	HEDL TME 77-54	APR 78	MANN AND SCHENTER (FAST)	521
94-Pu-238	1338	Neut.(RP)+decay data	1179	HEDL,AI,+	HEDLTME 77-54	APR 78	MANN, SCHENTER, ALTER, DUNFORD,	+514
94-Pu-239	1399	Neut(RP)+decay+FPY+gam.+err.	9418	GE-FBRD		OCT 76	E. KUJAWSKI, L. STEWART (LASL)	515
94-Pu-239	1399	Neut.(RP)+err.files	917	GE-FBRD		OCT 76	E. KUJAWSKI, L. STEWART (LASL)	531
94-Pu-240	1380	Neut(RP)+decay+FPY+gam.+err.	4093	ORNL		APR77	L.W. WESTON	515
94-Pu-241	1381	Neut(RP)+decay+FPY+gam.+err.	6610	ORNL		OCT77	L.W. WESTON, R.Q. WRIGHT, HOWERTON	515
94-Pu-242	1342	Neut(RP)+decay+FPY+gam.+err.	4322	HEDL,SRL,+		OCT78	MANN, BENJAMIN, MADLAND, HOWERTON,	+514
94-Pu-243	8443	Neut.(RP)+decay+gamma prod.data	2100	BNL,SRL,LLL		JUL76	KINSEY-ASSEMBLER (SEE COMMENTS)	521
94-Pu-244	8444	Neut.(RP)+decay data	712	HEDL,SRL	HEDL TME 77-54	APR78	MANN, SCHENTER, BENJAMIN, MCCROSSON	521
95-Am-240	8540	Decay data only	176	INEL		AUG78	REICH	521
95-Am-241	1361	Neut.(RP)+decay+gamma prod.+err	2186	HEDL,ORNL		APR78	MANN, SCHENTER, AND WESTON	514
95-Am-241	7951	Neut.(RP)+decay data	529	HEDL,ORNL		APR78	MANN, SCHENTER, AND WESTON	532
95-Am-242m	1369	Neut.(RP)+decay+gamma prod.data	1294	HEDLSRLLLL	HEDL TME 77-54	APR78	MANN, BENJAMIN, HOWERTON, ET AL.	514
95-Am-242	7952	Decay data only	88	INEL		APR78	REICH	532
95-Am-242m	7953	Decay data only	104	INEL	HEDL TME 77-54	APR78	REICH	532
95-Am-242	8542	Neut.(RP)+decay data	481	SRL		AUG75	BENJAMIN AND MCCROSSON	521
95-Am-243	1363	Neut.(RP)+decay+gamma prod.data	2055	HEDLSRLLLL	HEDL TME 77-54	APR78	MANN, BENJAMIN, HOWERTON,ET AL.	514
95-Am-244	8544	Decay data only	108	INEL		AUG78	REICH	521
95-Am-244m	8554	Decay data only	79	INEL		AUG78	REICH	521
96-Cm-241	8641	Neutron+decay data	718	HEDL	HEDL TME 77-54	APR78	MANN AND SCHENTER	522
96-Cm-242	7962	Decay data only	98	INEL		APR78	REICH	532
96-Cm-242	8642	Neut.(RP)+decay+gamma prod.data	1224	HEDLSRLLLL	HEDL TME 77-54	APR78	MANN, BENJAMIN, HOWERTON,ET AL.	522

Z EL A	MAT NO.	SPECIFICATION	NO. CARDS	LABORATORY	REFERENCE	DATE	AUTHOR	TAPE NO.
96-Cm-243	1343	Neut.(RP)+decay+gamma prod.data	1620	HEDLSRLLLL	HEDL TME 77-54	APR 78	MANN,BENJAMIN,HOWERTON,ET AL.	514
96-Cm-244	1344	Neut.(RP)+decay+gamma prod.data	1554	HEDLSRLLLL	HEDL TME 77-54	APR 78	MANN,BENJAMIN,HOWERTON,ET AL.	514
96-Cm-245	1345	Neut.(RP)+decay+gamma prod.data	1826	SRL		SEP 75	BENJAMIN AND MCCROSSON	514
96-Cm-246	1346	Neut.(RP)+decay+gamma prod.data	2093	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	514
96-Cm-247	8647	Neut.(RP)+decay+gamma prod.data	2297	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
96-Cm-248	8648	Neut.(RP)+decay+gamma prod.data	1282	HEDLSRLLLL	HEDL TME 77-54	APR 78	MANN,BENJAMIN,HOWERTON,ET AL.	522
96-Cm-249	8649	Decay data only	130	INEL		AUG 78	REICH	522
97-Bk-249	8749	Neut.(RP)+decay+gamma prod.data	2080	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
97-Bk-250	8750	Decay data only	234	INEL		AUG 78	REICH	522
98-Cf-249	8849	Neut.(RP)+decay+gamma prod.data	2041	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
98-Cf-250	8850	Neut.(RP)+decay+gamma prod.data	1981	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
98-Cf-251	8851	Neut.(RP)+decay+gamma prod.data	2050	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
98-Cf-252	8852	Neut(RP)+decay+FPY+gam.	3315	BNL,SRL,LLL		JUL 76	KINSEY-ASSEMBLER(SEE COMMENTS)	522
98-Cf-253	8853	Neut.(RP)+decay data	838	SRL		DEC 75	BENJAMIN AND MCCROSSON	522
99-Es-253	8953	Neut.(RP)+decay data	1004	BNL,SRL		JUL 76	KINSEY,BENJAMIN, AND MCCROSSON	522

APPENDIX K

Sample Data Set

The following is a sample data set in the ENDF format. This sample was taken from an evaluation by Nisley, et. al. and contains neutron cross section data for Helium-4. For other examples of data in the ENDF format see Appendix N.

```
ENDF/B MATERIAL 1270, HE-4                                                  0 0  0    0
 2.00400+ 3 4.00150+ 0          0          0          0                 01270 1451    1
 0.00000+ 0 0.00000+ 0          0          0          0                 01270 1451    2
 0.00000+ 0 0.00000+ 0          0          0         70                 81270 1451    3
  2-HE-  4 LASL        EVAL-OCT73 NISLEY, HALE, YOUNG (LASL)             1270 1451    4
 NO PUBLICATION        DIST-MAY78                    771216              1270 1451    5
                                                                         1270 1451    6
* * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * * *   1270 1451    7
  LASL-OCT,1973    R.A. NISLEY, G.M. HALE, P.G. YOUNG.                   1270 1451    8
                                                                         1270 1451    9
 MF=3 ---------- SMOOTH CROSS SECTIONS ----------------------            1270 1451   10
                                                                         1270 1451   11
        THE 2200 M/S CROSS SECTIONS ARE AS FOLLOWS,                      1270 1451   12
            MT=1   SIGMA= 0.75916 BARNS                                  1270 1451   13
            MT=2   SIGMA= 0.75916 BARNS                                  1270 1451   14
                                                                         1270 1451   15
   MT=1       TOTAL CROSS SECTION                                        1270 1451   16
        SEE DISCUSSION UNDER MT=2 BELOW                                  1270 1451   17
                                                                         1270 1451   18
   MT=2       ELASTIC SCATTERING CROSS SECTION                           1270 1451   19
        ALTHOUGH THE ONLY REACTION POSSIBLE FOR NEUTRONS INCIDENT        1270 1451   20
      ON HE-4 BELOW 20 MEV IS ELASTIC SCATTERING, THE MAJORITY           1270 1451   21
      OF THE N-4HE DATA IS RATHER IMPRECISE.  IN ORDER TO OVER-          1270 1451   22
      COME THIS PROBLEM, AN R-MATRIX ANALYSIS WAS PERFORMED WITH         1270 1451   23
      A DATA SET WHICH INCLUDED NOT ONLY THE N-4HE DATA BUT ALSO         1270 1451   24
      VERY PRECISE P-4HE DATA.  ALL THE AVAILABLE N-4HE AND P-4HE        1270 1451   25
      DATA BELOW 20 MEV WERE CONSIDERED IN THE ANALYSIS.  SINCE          1270 1451   26
      THE PREVIOUS EVALUATION WAS COMPLETED IN 1968, SEVERAL             1270 1451   27
      N-4HE ELASTIC SCATTERING MEASUREMENTS HAVE BEEN DONE.  THE         1270 1451   28
      MOST SIGNIFICANT OF THESE ARE THE LOW ENERGY NEUTRON CROSS         1270 1451   29
      SECTION OF RORER (RO69),THE RPI TOTAL CROSS SECTION                1270 1451   30
      MEASUREMENT (GO73), WHICH COVER THE RANGE EN=0.7-30 MEV,           1270 1451   31
      AND THE RELATIVE ANGULAR DISTRIBUTIONS OF MORGAN (MO68).           1270 1451   32
      A COMPLETE LIST OF REFERENCES FOR THE N-4HE DATA USED IS           1270 1451   33
      GIVEN BELOW.  THE P-4HE DATA WAS SELECTED TO SATISFY VERY          1270 1451   34
      STRINGENT STATISTICAL CRITERIA AND WE BELIEVE THE POSSIBLE         1270 1451   35
      ERRORS OF THE PREDICTED VALUES FOR THE P-4HE SCATTERING            1270 1451   36
      TO BE LESS THAN 1.0 PER CENT.  A SIMPLE MODEL FOR THE              1270 1451   37
      CHARGE DIFFERENCES BETWEEN THE N-4HE AND P-4HE SYSTEMS             1270 1451   38
      WAS ASSUMED AND THE N-4HE AND P-4HE DATA SETS WERE                 1270 1451   39
      SIMULTANEOUSLY ANALYZED.  THE VALUES OF THE CROSS SECTIONS         1270 1451   40
      AND ANGULAR DISTRIBUTIONS CONTAINED IN FILES 3 AND 4 ARE           1270 1451   41
      PROBABLY ACCURATE TO WITHIN 2.0 PERCENT.                           1270 1451   42
                                                                         1270 1451   43
 MF=4 ---------- NEUTRON ANGULAR DISTRIBUTIONS ---------------           1270 1451   44
                                                                         1270 1451   45
   MT=2       ELASTIC SCATTERING ANGULAR DISTRIBUTIONS                   1270 1451   46
        OBTAINED FROM THE R-MATRIX ANALYSIS DESCRIBED ABOVE              1270 1451   47
      UNDER MF=3,MT=2.  LEGENDRE POLYNOMIAL REPRESENTATION USED.         1270 1451   48
                                                                         1270 1451   49
```

```
---------------------- REFERENCES---------------------------------  1270 1451  50
                                                                    1270 1451  51
AU62   S.M.AUSTIN ET AL., PHYS.REV. 126(1962) 1532.                 1270 1451  52
BR72   W.B.BROSTE ET AL., PHYS.REV. C5(1972) 761.                   1270 1451  53
BU66   F.W.BUSSER ET AL., NUCL.PHYS. 88(1966) 593.                  1270 1451  54
CR72   D.S.CRAMER + L.CRANBERG, NUCL.PHYS. A180(1972) 273.          1270 1451  55
FA63   U.FASOLI + G.ZAGO, NUOVO CIMENTO 30(1963) 1169.              1270 1451  56
GO73   C.A.GOULDING ET AL., BULL.AM.PHYS.SOC. 18(1973) 538.         1270 1451  57
HO66   B.HOOP,JR. + H.H.BARSCHALL, NUCL.PHYS. 83(1966) 65.          1270 1451  58
JE66   R.W.JEWELL ET AL., PHYS.REV. 142(1966) 687.                  1270 1451  59
MA63   T.H.MAY ET AL., NUCL.PHYS. 45(1963) 17.(REV. MO68 + SA68)    1270 1451  60
MO68   G.L.MORGAN + R.L.WALTER, PHYS.REV. 168(1968) 1114.           1270 1451  61
NI71   A.NIILER ET AL., PHYS.REV. C4(1971) 36. (REV. 9/72)         1270 1451  62
RO69   D.C.RORER ET AL., NUCL.PHYS. 133(1969) 410.                  1270 1451  63
SA68   J.R.SAWERS ET AL., PHYS.REV. 168(1968) 1102.                 1270 1451  64
SE53   J.D.SEAGRAVE, PHYS.REV. 92(1953) 1222.                      1270 1451  65
SH55   D.F.SHAW, PROC.PHYS.SOC.(LONDON) 68(1955) 43.                1270 1451  66
SH64   R.E.SHAMU + J.G.JENKIN, PHYS.REV. 135(1964) B99.            1270 1451  67
SM54   J.R.SMITH, PHYS.REV. 95(1954) 730.                          1270 1451  68
ST70   T.STAMMBACH ET AL., PHYS.REV. C2(1970) 434.                  1270 1451  69
WH57   R.E.WHITE + F.J.M.FARLEY, NUCL.PHYS. 3(1957) 476.           1270 1451  70
YO63   P.G.YOUNG ET AL., AUST.J.PHYS. 16(1963) 185.                 1270 1451  71
                                                                    1270 1451  72
                                                                    1270 1451  73
                                  1        451         81          01270 1451  74
                                  2        151          4          01270 1451  75
                                  3          1         89          01270 1451  76
                                  3          2         89          01270 1451  77
                                  3        251         19          01270 1451  78
                                  3        252         19          01270 1451  79
                                  3        253         19          01270 1451  80
                                  4          2         96          01270 1451  81
0.00000+ 0 0.00000+ 0             0          0          0          01270 1  0  82
0.00000+ 0 0.00000+ 0             0          0          0          01270 0  0  83
2.00400+ 3 4.00150+ 0             0          0          1          01270 2151  84
2.00400+ 3 1.00000+ 0             0          0          1          01270 2151  85
1.00000- 5 1.00000+ 5             0          0          0          01270 2151  86
0.00000+ 0 2.45790- 1             0          0          0          01270 2151  87
0.00000+ 0 0.00000+ 0             0          0          0          01270 2  0  88
0.00000+ 0 0.00000+ 0             0          0          0          01270 0  0  89
2.00400+ 3 4.00150+ 0             0         99          0          01270 3  1  90
0.00000+ 0 0.00000+ 0             0          0          1       2571270 3  1  91
       257         2                                               1270 3  1  92
1.00000- 5 7.59160- 1 2.53000- 2 7.59160- 1 1.00000+ 0 7.59160-   11270 3  1  93
1.00000+ 1 7.59160- 1 1.00000+ 2 7.59160- 1 1.00000+ 3 7.59110-   11270 3  1  94
5.00000+ 3 7.58910- 1 1.00000+ 4 7.58720- 1 2.00000+ 4 7.58490-   11270 3  1  95
4.00000+ 4 7.58750- 1 6.00000+ 4 7.60030- 1 8.00000+ 4 7.62430-   11270 3  1  96
1.00000+ 5 7.66090- 1 1.50000+ 5 7.81580- 1 2.00000+ 5 8.08170-   11270 3  1  97
2.50000+ 5 8.48990- 1 3.00000+ 5 9.08080- 1 3.50000+ 5 9.90590-   11270 3  1  98
4.00000+ 5 1.10300+ 0 4.50000+ 5 1.25360+ 0 5.00000+ 5 1.45250+   01270 3  1  99
5.50000+ 5 1.71140+ 0 6.00000+ 5 2.04380+ 0 6.25000+ 5 2.24180+   01270 3  1 100
6.50000+ 5 2.46300+ 0 6.75000+ 5 2.70870+ 0 7.00000+ 5 2.97950+   01270 3  1 101
7.25000+ 5 3.27550+ 0 7.50000+ 5 3.59580+ 0 7.75000+ 5 3.93860+   01270 3  1 102
```

```
8.00000+ 5 4.30060+ 0 8.25000+ 5 4.67700+ 0 8.50000+ 5 5.06160+  01270 3  1 103
8.75000+ 5 5.44690+ 0 9.00000+ 5 5.82410+ 0 9.25000+ 5 6.18420+  01270 3  1 104
9.50000+ 5 6.51800+ 0 9.75000+ 5 6.81730+ 0 1.00000+ 6 7.07520+  01270 3  1 105
1.02500+ 6 7.28670+ 0 1.05000+ 6 7.44940+ 0 1.07500+ 6 7.56270+  01270 3  1 106
1.10000+ 6 7.62850+ 0 1.12500+ 6 7.65010+ 0 1.15000+ 6 7.63210+  01270 3  1 107
1.17500+ 6 7.57980+ 0 1.20000+ 6 7.49870+ 0 1.22500+ 6 7.39450+  01270 3  1 108
1.25000+ 6 7.27210+ 0 1.27500+ 6 7.13620+ 0 1.30000+ 6 6.99090+  01270 3  1 109
1.32500+ 6 6.83950+ 0 1.35000+ 6 6.68500+ 0 1.37500+ 6 6.52950+  01270 3  1 110
1.40000+ 6 6.37480+ 0 1.45000+ 6 6.07350+ 0 1.50000+ 6 5.78820+  01270 3  1 111
1.55000+ 6 5.52260+ 0 1.60000+ 6 5.27790+ 0 1.65000+ 6 5.05390+  01270 3  1 112
1.70000+ 6 4.84960+ 0 1.75000+ 6 4.66360+ 0 1.80000+ 6 4.49430+  01270 3  1 113
1.85000+ 6 4.34010+ 0 1.90000+ 6 4.19960+ 0 1.95000+ 6 4.07130+  01270 3  1 114
2.00000+ 6 3.95410+ 0 2.05000+ 6 3.84670+ 0 2.10000+ 6 3.74820+  01270 3  1 115
2.15000+ 6 3.65760+ 0 2.20000+ 6 3.57410+ 0 2.25000+ 6 3.49710+  01270 3  1 116
2.30000+ 6 3.42590+ 0 2.35000+ 6 3.35980+ 0 2.40000+ 6 3.29860+  01270 3  1 117
2.45000+ 6 3.24160+ 0 2.50000+ 6 3.18850+ 0 2.55000+ 6 3.13890+  01270 3  1 118
2.60000+ 6 3.09250+ 0 2.65000+ 6 3.04910+ 0 2.70000+ 6 3.00840+  01270 3  1 119
2.75000+ 6 2.97010+ 0 2.80000+ 6 2.93400+ 0 2.85000+ 6 2.90000+  01270 3  1 120
2.90000+ 6 2.86790+ 0 2.95000+ 6 2.83750+ 0 3.00000+ 6 2.80870+  01270 3  1 121
3.10000+ 6 2.75540+ 0 3.20000+ 6 2.70700+ 0 3.30000+ 6 2.66270+  01270 3  1 122
3.40000+ 6 2.62210+ 0 3.50000+ 6 2.58440+ 0 3.60000+ 6 2.54940+  01270 3  1 123
3.70000+ 6 2.51650+ 0 3.80000+ 6 2.48550+ 0 3.90000+ 6 2.45610+  01270 3  1 124
4.00000+ 6 2.42800+ 0 4.10000+ 6 2.40110+ 0 4.20000+ 6 2.37530+  01270 3  1 125
4.30000+ 6 2.35030+ 0 4.40000+ 6 2.32600+ 0 4.50000+ 6 2.30240+  01270 3  1 126
4.60000+ 6 2.27940+ 0 4.70000+ 6 2.25690+ 0 4.80000+ 6 2.23490+  01270 3  1 127
4.90000+ 6 2.21320+ 0 5.00000+ 6 2.19190+ 0 5.10000+ 6 2.17100+  01270 3  1 128
5.20000+ 6 2.15040+ 0 5.30000+ 6 2.13010+ 0 5.40000+ 6 2.11000+  01270 3  1 129
5.50000+ 6 2.09030+ 0 5.60000+ 6 2.07070+ 0 5.70000+ 6 2.05150+  01270 3  1 130
5.80000+ 6 2.03240+ 0 5.90000+ 6 2.01360+ 0 6.00000+ 6 1.99500+  01270 3  1 131
6.10000+ 6 1.97670+ 0 6.20000+ 6 1.95860+ 0 6.30000+ 6 1.94070+  01270 3  1 132
6.40000+ 6 1.92300+ 0 6.50000+ 6 1.90550+ 0 6.60000+ 6 1.88830+  01270 3  1 133
6.70000+ 6 1.87120+ 0 6.80000+ 6 1.85440+ 0 6.90000+ 6 1.83780+  01270 3  1 134
7.00000+ 6 1.82130+ 0 7.10000+ 6 1.80510+ 0 7.20000+ 6 1.78910+  01270 3  1 135
7.30000+ 6 1.77330+ 0 7.40000+ 6 1.75770+ 0 7.50000+ 6 1.74230+  01270 3  1 136
7.60000+ 6 1.72710+ 0 7.70000+ 6 1.71200+ 0 7.80000+ 6 1.69720+  01270 3  1 137
7.90000+ 6 1.68250+ 0 8.00000+ 6 1.66810+ 0 8.10000+ 6 1.65380+  01270 3  1 138
8.20000+ 6 1.63970+ 0 8.30000+ 6 1.62580+ 0 8.40000+ 6 1.61200+  01270 3  1 139
8.50000+ 6 1.59850+ 0 8.60000+ 6 1.58510+ 0 8.70000+ 6 1.57180+  01270 3  1 140
8.80000+ 6 1.55880+ 0 8.90000+ 6 1.54590+ 0 9.00000+ 6 1.53320+  01270 3  1 141
9.10000+ 6 1.52060+ 0 9.20000+ 6 1.50820+ 0 9.30000+ 6 1.49590+  01270 3  1 142
9.40000+ 6 1.48380+ 0 9.50000+ 6 1.47180+ 0 9.60000+ 6 1.46000+  01270 3  1 143
9.70000+ 6 1.44840+ 0 9.80000+ 6 1.43690+ 0 9.90000+ 6 1.42550+  01270 3  1 144
1.00000+ 7 1.41420+ 0 1.01000+ 7 1.40320+ 0 1.02000+ 7 1.39220+  01270 3  1 145
1.03000+ 7 1.38140+ 0 1.04000+ 7 1.37070+ 0 1.05000+ 7 1.36010+  01270 3  1 146
1.06000+ 7 1.34970+ 0 1.07000+ 7 1.33930+ 0 1.08000+ 7 1.32920+  01270 3  1 147
1.09000+ 7 1.31910+ 0 1.10000+ 7 1.30910+ 0 1.11000+ 7 1.29930+  01270 3  1 148
1.12000+ 7 1.28960+ 0 1.13000+ 7 1.28000+ 0 1.14000+ 7 1.27050+  01270 3  1 149
1.15000+ 7 1.26120+ 0 1.16000+ 7 1.25190+ 0 1.17000+ 7 1.24280+  01270 3  1 150
1.18000+ 7 1.23370+ 0 1.19000+ 7 1.22480+ 0 1.20000+ 7 1.21600+  01270 3  1 151
1.21000+ 7 1.20720+ 0 1.22000+ 7 1.19860+ 0 1.23000+ 7 1.19010+  01270 3  1 152
1.24000+ 7 1.18170+ 0 1.25000+ 7 1.17330+ 0 1.26000+ 7 1.16510+  01270 3  1 153
1.27000+ 7 1.15700+ 0 1.28000+ 7 1.14890+ 0 1.29000+ 7 1.14100+  01270 3  1 154
1.30000+ 7 1.13310+ 0 1.31000+ 7 1.12540+ 0 1.32000+ 7 1.11770+  01270 3  1 155
```

```
 1.33000+ 7 1.11010+ 0 1.34000+ 7 1.10260+ 0 1.35000+ 7 1.09520+  01270 3  1 156
 1.36000+ 7 1.08780+ 0 1.37000+ 7 1.08060+ 0 1.38000+ 7 1.07340+  01270 3  1 157
 1.39000+ 7 1.06640+ 0 1.40000+ 7 1.05940+ 0 1.41000+ 7 1.05250+  01270 3  1 158
 1.42000+ 7 1.04560+ 0 1.43000+ 7 1.03890+ 0 1.44000+ 7 1.03220+  01270 3  1 159
 1.45000+ 7 1.02560+ 0 1.46000+ 7 1.01900+ 0 1.47000+ 7 1.01260+  01270 3  1 160
 1.48000+ 7 1.00620+ 0 1.49000+ 7 9.99900- 1 1.50000+ 7 9.93670-  11270 3  1 161
 1.51000+ 7 9.87510- 1 1.52000+ 7 9.81420- 1 1.53000+ 7 9.75400-  11270 3  1 162
 1.54000+ 7 9.69450- 1 1.55000+ 7 9.63570- 1 1.56000+ 7 9.57760-  11270 3  1 163
 1.57000+ 7 9.52010- 1 1.58000+ 7 9.46340- 1 1.59000+ 7 9.40720-  11270 3  1 164
 1.60000+ 7 9.35180- 1 1.61000+ 7 9.29700- 1 1.62000+ 7 9.24280-  11270 3  1 165
 1.63000+ 7 9.18930- 1 1.64000+ 7 9.13640- 1 1.65000+ 7 9.08410-  11270 3  1 166
 1.66000+ 7 9.03240- 1 1.67000+ 7 8.98140- 1 1.68000+ 7 8.93100-  11270 3  1 167
 1.69000+ 7 8.88110- 1 1.70000+ 7 8.83190- 1 1.71000+ 7 8.78320-  11270 3  1 168
 1.72000+ 7 8.73520- 1 1.73000+ 7 8.68770- 1 1.74000+ 7 8.64080-  11270 3  1 169
 1.75000+ 7 8.59440- 1 1.76000+ 7 8.54860- 1 1.77000+ 7 8.50340-  11270 3  1 170
 1.78000+ 7 8.45870- 1 1.79000+ 7 8.41460- 1 1.80000+ 7 8.37100-  11270 3  1 171
 1.81000+ 7 8.32800- 1 1.82000+ 7 8.28550- 1 1.83000+ 7 8.24350-  11270 3  1 172
 1.84000+ 7 8.20210- 1 1.85000+ 7 8.16110- 1 1.86000+ 7 8.12070-  11270 3  1 173
 1.87000+ 7 8.08080- 1 1.88000+ 7 8.04140- 1 1.89000+ 7 8.00250-  11270 3  1 174
 1.90000+ 7 7.96410- 1 1.91000+ 7 7.92620- 1 1.92000+ 7 7.88880-  11270 3  1 175
 1.93000+ 7 7.85180- 1 1.94000+ 7 7.81540- 1 1.95000+ 7 7.77940-  11270 3  1 176
 1.96000+ 7 7.74390- 1 1.97000+ 7 7.70880- 1 1.98000+ 7 7.67430-  11270 3  1 177
 1.99000+ 7 7.64010- 1 2.00000+ 7 7.60650- 1                      1270 3  1 178
 0.00000+ 0 0.00000+ 0          0          0          0          01270 3  0 179
 2.00400+ 3 4.00150+ 0          0         99          0          01270 3  2 180
 0.00000+ 0 0.00000+ 0          0          0          1       2571270 3  2 181
       257          2                                            1270 3  2 182
 1.00000- 5 7.59160- 1 2.53000- 2 7.59160- 1 1.00000+ 0 7.59160-  11270 3  2 183
 1.00000+ 1 7.59160- 1 1.00000+ 2 7.59160- 1 1.00000+ 3 7.59110-  11270 3  2 184
 5.00000+ 3 7.58910- 1 1.00000+ 4 7.58720- 1 2.00000+ 4 7.58490-  11270 3  2 185
 4.00000+ 4 7.58750- 1 6.00000+ 4 7.60030- 1 8.00000+ 4 7.62430-  11270 3  2 186
 1.00000+ 5 7.66090- 1 1.50000+ 5 7.81580- 1 2.00000+ 5 8.08170-  11270 3  2 187
 2.50000+ 5 8.48990- 1 3.00000+ 5 9.08080- 1 3.50000+ 5 9.90590-  11270 3  2 188
 4.00000+ 5 1.10300+ 0 4.50000+ 5 1.25360+ 0 5.00000+ 5 1.45250+  01270 3  2 189
 5.50000+ 5 1.71140+ 0 6.00000+ 5 2.04380+ 0 6.25000+ 5 2.24180+  01270 3  2 190
 6.50000+ 5 2.46300+ 0 6.75000+ 5 2.70870+ 0 7.00000+ 5 2.97950+  01270 3  2 191
 7.25000+ 5 3.27550+ 0 7.50000+ 5 3.59580+ 0 7.75000+ 5 3.93860+  01270 3  2 192
 8.00000+ 5 4.30060+ 0 8.25000+ 5 4.67700+ 0 8.50000+ 5 5.06160+  01270 3  2 193
 8.75000+ 5 5.44690+ 0 9.00000+ 5 5.82410+ 0 9.25000+ 5 6.18420+  01270 3  2 194
 9.50000+ 5 6.51800+ 0 9.75000+ 5 6.81730+ 0 1.00000+ 6 7.07520+  01270 3  2 195
 1.02500+ 6 7.28670+ 0 1.05000+ 6 7.44940+ 0 1.07500+ 6 7.56270+  01270 3  2 196
 1.10000+ 6 7.62850+ 0 1.12500+ 6 7.65010+ 0 1.15000+ 6 7.63210+  01270 3  2 197
 1.17500+ 6 7.57980+ 0 1.20000+ 6 7.49870+ 0 1.22500+ 6 7.39450+  01270 3  2 198
 1.25000+ 6 7.27210+ 0 1.27500+ 6 7.13620+ 0 1.30000+ 6 6.99090+  01270 3  2 199
 1.32500+ 6 6.83950+ 0 1.35000+ 6 6.68500+ 0 1.37500+ 6 6.52950+  01270 3  2 200
 1.40000+ 6 6.37480+ 0 1.45000+ 6 6.07350+ 0 1.50000+ 6 5.78820+  01270 3  2 201
 1.55000+ 6 5.52260+ 0 1.60000+ 6 5.27790+ 0 1.65000+ 6 5.05390+  01270 3  2 202
 1.70000+ 6 4.84960+ 0 1.75000+ 6 4.66360+ 0 1.80000+ 6 4.49430+  01270 3  2 203
 1.85000+ 6 4.34010+ 0 1.90000+ 6 4.19960+ 0 1.95000+ 6 4.07130+  01270 3  2 204
 2.00000+ 6 3.95410+ 0 2.05000+ 6 3.84670+ 0 2.10000+ 6 3.74820+  01270 3  2 205
 2.15000+ 6 3.65760+ 0 2.20000+ 6 3.57410+ 0 2.25000+ 6 3.49710+  01270 3  2 206
 2.30000+ 6 3.42590+ 0 2.35000+ 6 3.35980+ 0 2.40000+ 6 3.29860+  01270 3  2 207
 2.45000+ 6 3.24160+ 0 2.50000+ 6 3.18850+ 0 2.55000+ 6 3.13890+  01270 3  2 208
```

```
2.60000+ 6 3.09250+ 0 2.65000+ 6 3.04910+ 0 2.70000+ 6 3.00840+  01270 3  2 209
2.75000+ 6 2.97010+ 0 2.80000+ 6 2.93400+ 0 2.85000+ 6 2.90000+  01270 3  2 210
2.90000+ 6 2.86790+ 0 2.95000+ 6 2.83750+ 0 3.00000+ 6 2.80870+  01270 3  2 211
3.10000+ 6 2.75540+ 0 3.20000+ 6 2.70700+ 0 3.30000+ 6 2.66270+  01270 3  2 212
3.40000+ 6 2.62210+ 0 3.50000+ 6 2.58440+ 0 3.60000+ 6 2.54940+  01270 3  2 213
3.70000+ 6 2.51650+ 0 3.80000+ 6 2.48550+ 0 3.90000+ 6 2.45610+  01270 3  2 214
4.00000+ 6 2.42800+ 0 4.10000+ 6 2.40110+ 0 4.20000+ 6 2.37530+  01270 3  2 215
4.30000+ 6 2.35030+ 0 4.40000+ 6 2.32600+ 0 4.50000+ 6 2.30240+  01270 3  2 216
4.60000+ 6 2.27940+ 0 4.70000+ 6 2.25690+ 0 4.80000+ 6 2.23490+  01270 3  2 217
4.90000+ 6 2.21320+ 0 5.00000+ 6 2.19190+ 0 5.10000+ 6 2.17100+  01270 3  2 218
5.20000+ 6 2.15040+ 0 5.30000+ 6 2.13010+ 0 5.40000+ 6 2.11000+  01270 3  2 219
5.50000+ 6 2.09030+ 0 5.60000+ 6 2.07070+ 0 5.70000+ 6 2.05150+  01270 3  2 220
5.80000+ 6 2.03240+ 0 5.90000+ 6 2.01360+ 0 6.00000+ 6 1.99500+  01270 3  2 221
6.10000+ 6 1.97670+ 0 6.20000+ 6 1.95860+ 0 6.30000+ 6 1.94070+  01270 3  2 222
6.40000+ 6 1.92300+ 0 6.50000+ 6 1.90550+ 0 6.60000+ 6 1.88830+  01270 3  2 223
6.70000+ 6 1.87120+ 0 6.80000+ 6 1.85440+ 0 6.90000+ 6 1.83780+  01270 3  2 224
7.00000+ 6 1.82130+ 0 7.10000+ 6 1.80510+ 0 7.20000+ 6 1.78910+  01270 3  2 225
7.30000+ 6 1.77330+ 0 7.40000+ 6 1.75770+ 0 7.50000+ 6 1.74230+  01270 3  2 226
7.60000+ 6 1.72710+ 0 7.70000+ 6 1.71200+ 0 7.80000+ 6 1.69720+  01270 3  2 227
7.90000+ 6 1.68250+ 0 8.00000+ 6 1.66810+ 0 8.10000+ 6 1.65380+  01270 3  2 228
8.20000+ 6 1.63970+ 0 8.30000+ 6 1.62580+ 0 8.40000+ 6 1.61200+  01270 3  2 229
8.50000+ 6 1.59850+ 0 8.60000+ 6 1.58510+ 0 8.70000+ 6 1.57180+  01270 3  2 230
8.80000+ 6 1.55880+ 0 8.90000+ 6 1.54590+ 0 9.00000+ 6 1.53320+  01270 3  2 231
9.10000+ 6 1.52060+ 0 9.20000+ 6 1.50820+ 0 9.30000+ 6 1.49590+  01270 3  2 232
9.40000+ 6 1.48380+ 0 9.50000+ 6 1.47180+ 0 9.60000+ 6 1.46000+  01270 3  2 233
9.70000+ 6 1.44840+ 0 9.80000+ 6 1.43690+ 0 9.90000+ 6 1.42550+  01270 3  2 234
1.00000+ 7 1.41420+ 0 1.01000+ 7 1.40320+ 0 1.02000+ 7 1.39220+  01270 3  2 235
1.03000+ 7 1.38140+ 0 1.04000+ 7 1.37070+ 0 1.05000+ 7 1.36010+  01270 3  2 236
1.06000+ 7 1.34970+ 0 1.07000+ 7 1.33930+ 0 1.08000+ 7 1.32920+  01270 3  2 237
1.09000+ 7 1.31910+ 0 1.10000+ 7 1.30910+ 0 1.11000+ 7 1.29930+  01270 3  2 238
1.12000+ 7 1.28960+ 0 1.13000+ 7 1.28000+ 0 1.14000+ 7 1.27050+  01270 3  2 239
1.15000+ 7 1.26120+ 0 1.16000+ 7 1.25190+ 0 1.17000+ 7 1.24280+  01270 3  2 240
1.18000+ 7 1.23370+ 0 1.19000+ 7 1.22480+ 0 1.20000+ 7 1.21600+  01270 3  2 241
1.21000+ 7 1.20720+ 0 1.22000+ 7 1.19860+ 0 1.23000+ 7 1.19010+  01270 3  2 242
1.24000+ 7 1.18170+ 0 1.25000+ 7 1.17330+ 0 1.26000+ 7 1.16510+  01270 3  2 243
1.27000+ 7 1.15700+ 0 1.28000+ 7 1.14890+ 0 1.29000+ 7 1.14100+  01270 3  2 244
1.30000+ 7 1.13310+ 0 1.31000+ 7 1.12540+ 0 1.32000+ 7 1.11770+  01270 3  2 245
1.33000+ 7 1.11010+ 0 1.34000+ 7 1.10260+ 0 1.35000+ 7 1.09520+  01270 3  2 246
1.36000+ 7 1.08780+ 0 1.37000+ 7 1.08060+ 0 1.38000+ 7 1.07340+  01270 3  2 247
1.39000+ 7 1.06640+ 0 1.40000+ 7 1.05940+ 0 1.41000+ 7 1.05250+  01270 3  2 248
1.42000+ 7 1.04560+ 0 1.43000+ 7 1.03890+ 0 1.44000+ 7 1.03220+  01270 3  2 249
1.45000+ 7 1.02560+ 0 1.46000+ 7 1.01900+ 0 1.47000+ 7 1.01260+  01270 3  2 250
1.48000+ 7 1.00620+ 0 1.49000+ 7 9.99900- 1 1.50000+ 7 9.93670-  11270 3  2 251
1.51000+ 7 9.87510- 1 1.52000+ 7 9.81420- 1 1.53000+ 7 9.75400-  11270 3  2 252
1.54000+ 7 9.69450- 1 1.55000+ 7 9.63570- 1 1.56000+ 7 9.57760-  11270 3  2 253
1.57000+ 7 9.52010- 1 1.58000+ 7 9.46340- 1 1.59000+ 7 9.40720-  11270 3  2 254
1.60000+ 7 9.35180- 1 1.61000+ 7 9.29700- 1 1.62000+ 7 9.24280-  11270 3  2 255
1.63000+ 7 9.18930- 1 1.64000+ 7 9.13640- 1 1.65000+ 7 9.08410-  11270 3  2 256
1.66000+ 7 9.03240- 1 1.67000+ 7 8.98140- 1 1.68000+ 7 8.93100-  11270 3  2 257
1.69000+ 7 8.88110- 1 1.70000+ 7 8.83190- 1 1.71000+ 7 8.78320-  11270 3  2 258
1.72000+ 7 8.73520- 1 1.73000+ 7 8.68770- 1 1.74000+ 7 8.64080-  11270 3  2 259
1.75000+ 7 8.59440- 1 1.76000+ 7 8.54860- 1 1.77000+ 7 8.50340-  11270 3  2 260
1.78000+ 7 8.45870- 1 1.79000+ 7 8.41460- 1 1.80000+ 7 8.37100-  11270 3  2 261
```

```
1.81000+ 7 8.32800- 1 1.82000+ 7 8.28550- 1 1.83000+ 7 8.24350-  11270 3  2 262
1.84000+ 7 8.20210- 1 1.85000+ 7 8.16110- 1 1.86000+ 7 8.12070-  11270 3  2 263
1.87000+ 7 8.08080- 1 1.88000+ 7 8.04140- 1 1.89000+ 7 8.00250-  11270 3  2 264
1.90000+ 7 7.96410- 1 1.91000+ 7 7.92620- 1 1.92000+ 7 7.88880-  11270 3  2 265
1.93000+ 7 7.85180- 1 1.94000+ 7 7.81540- 1 1.95000+ 7 7.77940-  11270 3  2 266
1.96000+ 7 7.74390- 1 1.97000+ 7 7.70880- 1 1.98000+ 7 7.67430-  11270 3  2 267
1.99000+ 7 7.64010- 1 2.00000+ 7 7.60650- 1                       1270 3  2 268
0.00000+ 0 0.00000+ 0           0          0          0          01270 3  0 269
2.00400+ 3 4.00150+ 0           0          0          0          01270 3251 270
0.00000+ 0 0.00000+ 0           0          0          1         461270 3251 271
        46         2                                              1270 3251 272
1.00000- 5 1.69807- 1 1.00000+ 0 1.69807- 1 6.00000+ 4 9.66225-  21270 3251 273
1.00000+ 5 4.51875- 2 1.50000+ 5-2.04487- 2 2.00000+ 5-8.51365-  21270 3251 274
2.50000+ 5-1.45626- 1 3.00000+ 5-1.98263- 1 3.50000+ 5-2.39545-  11270 3251 275
4.00000+ 5-2.66678- 1 4.50000+ 5-2.78213- 1 5.00000+ 5-2.74235-  11270 3251 276
5.50000+ 5-2.56321- 1 6.00000+ 5-2.27057- 1 6.50000+ 5-1.89524-  11270 3251 277
7.00000+ 5-1.46778- 1 7.75000+ 5-7.85520- 2 8.50000+ 5-1.10802-  21270 3251 278
9.25000+ 5 5.17574- 2 1.00000+ 6 1.08145- 1 1.07500+ 6 1.57606-  11270 3251 279
1.17500+ 6 2.13288- 1 1.27500+ 6 2.58619- 1 1.40000+ 6 3.03341-  11270 3251 280
1.55000+ 6 3.43482- 1 1.70000+ 6 3.72766- 1 1.85000+ 6 3.94254-  11270 3251 281
2.05000+ 6 4.14458- 1 2.25000+ 6 4.28188- 1 2.50000+ 6 4.39742-  11270 3251 282
2.75000+ 6 4.47708- 1 3.00000+ 6 4.53851- 1 3.40000+ 6 4.62328-  11270 3251 283
4.10000+ 6 4.77341- 1 5.30000+ 6 5.04549- 1 6.10000+ 6 5.20935-  11270 3251 284
6.90000+ 6 5.34779- 1 7.70000+ 6 5.46099- 1 8.60000+ 6 5.56253-  11270 3251 285
9.60000+ 6 5.64991- 1 1.07000+ 7 5.72283- 1 1.20000+ 7 5.78728-  11270 3251 286
1.35000+ 7 5.84310- 1 1.54000+ 7 5.89935- 1 1.81000+ 7 5.97496-  11270 3251 287
2.00000+ 7 6.03470- 1                                             1270 3251 288
0.00000+ 0 0.00000+ 0           0          0          0          01270 3  0 289
2.00400+ 3 4.00150+ 0           0          0          0          01270 3252 290
0.00000+ 0 0.00000+ 0           0          0          1         461270 3252 291
        46         2                                              1270 3252 292
1.00000- 5 4.25185- 1 1.00000+ 0 4.25185- 1 6.00000+ 4 4.62683-  11270 3252 293
1.00000+ 5 4.89032- 1 1.50000+ 5 5.22649- 1 2.00000+ 5 5.55770-  11270 3252 294
2.50000+ 5 5.86731- 1 3.00000+ 5 6.13659- 1 3.50000+ 5 6.34761-  11270 3252 295
4.00000+ 5 6.48608- 1 4.50000+ 5 6.54461- 1 5.00000+ 5 6.52366-  11270 3252 296
5.50000+ 5 6.43133- 1 6.00000+ 5 6.28092- 1 6.50000+ 5 6.08821-  11270 3252 297
7.00000+ 5 5.86886- 1 7.75000+ 5 5.51894- 1 8.50000+ 5 5.17302-  11270 3252 298
9.25000+ 5 4.85097- 1 1.00000+ 6 4.56205- 1 1.07500+ 6 4.30867-  11270 3252 299
1.17500+ 6 4.02350- 1 1.27500+ 6 3.79139- 1 1.40000+ 6 3.56248-  11270 3252 300
1.55000+ 6 3.35708- 1 1.70000+ 6 3.20730- 1 1.85000+ 6 3.09744-  11270 3252 301
2.05000+ 6 2.99418- 1 2.25000+ 6 2.92402- 1 2.50000+ 6 2.86498-  11270 3252 302
2.75000+ 6 2.82423- 1 3.00000+ 6 2.79274- 1 3.40000+ 6 2.74915-  11270 3252 303
4.10000+ 6 2.67174- 1 5.30000+ 6 2.53139- 1 6.10000+ 6 2.44693-  11270 3252 304
6.90000+ 6 2.37562- 1 7.70000+ 6 2.31733- 1 8.60000+ 6 2.26506-  11270 3252 305
9.60000+ 6 2.22011- 1 1.07000+ 7 2.18260- 1 1.20000+ 7 2.14946-  11270 3252 306
1.35000+ 7 2.12077- 1 1.54000+ 7 2.09189- 1 1.81000+ 7 2.05309-  11270 3252 307
2.00000+ 7 2.02246- 1                                             1270 3252 308
0.00000+ 0 0.00000+ 0           0          0          0          01270 3  0 309
2.00400+ 3 4.00150+ 0           0          0          0          01270 3253 310
0.00000+ 0 0.00000+ 0           0          0          1         461270 3253 311
        46         2                                              1270 3253 312
1.00000- 5 3.09677- 1 1.00000+ 0 3.09677- 1 6.00000+ 4 3.22020-  11270 3253 313
1.00000+ 5 3.29881- 1 1.50000+ 5 3.39153- 1 2.00000+ 5 3.47671-  11270 3253 314
```

```
 2.50000+ 5 3.55304- 1 3.00000+ 5 3.61944- 1 3.50000+ 5 3.67515-  11270 3253 315
 4.00000+ 5 3.71966- 1 4.50000+ 5 3.75288- 1 5.00000+ 5 3.77494-  11270 3253 316
 5.50000+ 5 3.78636- 1 6.00000+ 5 3.78779- 1 6.50000+ 5 3.78013-  11270 3253 317
 7.00000+ 5 3.76438- 1 7.75000+ 5 3.72793- 1 8.50000+ 5 3.67934-  11270 3253 318
 9.25000+ 5 3.62204- 1 1.00000+ 6 3.55913- 1 1.07500+ 6 3.49320-  11270 3253 319
 1.17500+ 6 3.40416- 1 1.27500+ 6 3.31713- 1 1.40000+ 6 3.21487-  11270 3253 320
 1.55000+ 6 3.10505- 1 1.70000+ 6 3.01092- 1 1.85000+ 6 2.93223-  11270 3253 321
 2.05000+ 6 2.84922- 1 2.25000+ 6 2.78760- 1 2.50000+ 6 2.73422-  11270 3253 322
 2.75000+ 6 2.70045- 1 3.00000+ 6 2.68026- 1 3.40000+ 6 2.66528-  11270 3253 323
 4.10000+ 6 2.66068- 1 5.30000+ 6 2.65721- 1 6.10000+ 6 2.64767-  11270 3253 324
 6.90000+ 6 2.63436- 1 7.70000+ 6 2.61960- 1 8.60000+ 6 2.60321-  11270 3253 325
 9.60000+ 6 2.58651- 1 1.07000+ 7 2.57070- 1 1.20000+ 7 2.55526-  11270 3253 326
 1.35000+ 7 2.54112- 1 1.54000+ 7 2.52690- 1 1.81000+ 7 2.50961-  11270 3253 327
 2.00000+ 7 2.49761- 1                                            1270 3253 328
 0.00000+ 0 0.00000+ 0          0          0          0          01270 3  0 329
 0.00000+ 0 0.00000+ 0          0          0          0          01270 0  0 330
 2.00400+ 3 4.00150+ 0          0          1          0          01270 4  2 331
 0.00000+ 0 4.00150+ 0          0          2          0          01270 4  2 332
 0.00000+ 0 0.00000+ 0          0          0          1         461270 4  2 333
         46         2                                             1270 4  2 334
 0.00000+ 0 1.00000- 5          0          0          2          01270 4  2 335
 0.00000+ 0 0.00000+ 0                                            1270 4  2 336
 0.00000+ 0 1.00000+ 0          0          0          2          01270 4  2 337
 0.00000+ 0 0.00000+ 0                                            1270 4  2 338
 0.00000+ 0 6.00000+ 4          0          0          6          01270 4  2 339
-7.57300- 2 1.53160- 3-3.35670- 8 6.31490-11 4.93690-12-3.18450- 121270 4  2 340
 0.00000+ 0 1.00000+ 5          0          0          6          01270 4  2 341
-1.28630- 1 4.49460- 3-1.58070- 7 5.36690-10 9.00010-12-5.80170- 121270 4  2 342
 0.00000+ 0 1.50000+ 5          0          0          6          01270 4  2 343
-1.95710- 1 1.07550- 2-5.40090- 7 2.81240- 9 1.49430-11-9.62900- 121270 4  2 344
 0.00000+ 0 2.00000+ 5          0          0          6          01270 4  2 345
-2.61270- 1 2.01330- 2-1.28070- 6 8.99480- 9 2.16810-11-1.40450- 111270 4  2 346
 0.00000+ 0 2.50000+ 5          0          0          6          01270 4  2 347
-3.21920- 1 3.27200- 2-2.46670- 6 2.18560- 8 2.89110-11-1.89290- 111270 4  2 348
 0.00000+ 0 3.00000+ 5          0          0          6          01270 4  2 349
-3.73900- 1 4.83080- 2-4.13370- 6 4.43290- 8 3.61390-11-2.41000- 111270 4  2 350
 0.00000+ 0 3.50000+ 5          0          0          6          01270 4  2 351
-4.13670- 1 6.63400- 2-6.24850- 6 7.88260- 8 4.27970-11-2.93020- 111270 4  2 352
 0.00000+ 0 4.00000+ 5          0          0          6          01270 4  2 353
-4.38480- 1 8.59370- 2-8.70340- 6 1.26510- 7 4.82860-11-3.42510- 111270 4  2 354
 0.00000+ 0 4.50000+ 5          0          0          6          01270 4  2 355
-4.47010- 1 1.06020- 1-1.13290- 5 1.86780- 7 5.21520-11-3.86660- 111270 4  2 356
 0.00000+ 0 5.00000+ 5          0          0          6          01270 4  2 357
-4.39540- 1 1.25470- 1-1.39230- 5 2.57150- 7 5.41240-11-4.23310- 111270 4  2 358
 0.00000+ 0 5.50000+ 5          0          0          6          01270 4  2 359
-4.17870- 1 1.43340- 1-1.62860- 5 3.33610- 7 5.41840-11-4.51140- 111270 4  2 360
 0.00000+ 0 6.00000+ 5          0          0          6          01270 4  2 361
-3.84810- 1 1.58910- 1-1.82530- 5 4.11270- 7 5.24920-11-4.69790- 111270 4  2 362
 0.00000+ 0 6.50000+ 5          0          0          6          01270 4  2 363
-3.43620- 1 1.71820- 1-1.97090- 5 4.85150- 7 4.93270-11-4.79730- 111270 4  2 364
 0.00000+ 0 7.00000+ 5          0          0          6          01270 4  2 365
-2.97490- 1 1.81980- 1-2.05980- 5 5.50710- 7 4.49510-11-4.81990- 111270 4  2 366
 0.00000+ 0 7.75000+ 5          0          0          6          01270 4  2 367
```

```
-2.24860- 1 1.92380- 1-2.08570- 5 6.25550- 7 3.64860-11-4.73900- 111270 4  2 368
 0.00000+ 0 8.50000+ 5           0          0          6           01270 4  2 369
-1.53860- 1 1.97850- 1-1.99320- 5 6.64760- 7 2.56280-11-4.55770- 111270 4  2 370
 0.00000+ 0 9.25000+ 5           0          0          6           01270 4  2 371
-8.83210- 2 1.99540- 1-1.80140- 5 6.63510- 7 1.13750-11-4.31050- 111270 4  2 372
 0.00000+ 0 1.00000+ 6           0          0          6           01270 4  2 373
-2.99470- 2 1.98510- 1-1.53100- 5 6.19850- 7-8.25380-12-4.01820- 111270 4  2 374
 0.00000+ 0 1.07500+ 6           0          0          6           01270 4  2 375
 2.09180- 2 1.95630- 1-1.20140- 5 5.33560- 7-3.61300-11-3.68810- 111270 4  2 376
 0.00000+ 0 1.17500+ 6           0          0          6           01270 4  2 377
 7.77710- 2 1.90010- 1-6.97450- 6 3.53730- 7-9.26110-11-3.17990- 111270 4  2 378
 0.00000+ 0 1.27500+ 6           0          0          6           01270 4  2 379
 1.23690- 1 1.83300- 1-1.47360- 6 1.03160- 7-1.81190-10-2.55590- 111270 4  2 380
 0.00000+ 0 1.40000+ 6           0          0          6           01270 4  2 381
 1.68600- 1 1.74400- 1 5.71070- 6-3.02630- 7-3.56860-10-1.49900- 111270 4  2 382
 0.00000+ 0 1.55000+ 6           0          0          6           01270 4  2 383
 2.08480- 1 1.63910- 1 1.43750- 5-9.11740- 7-7.02210-10 4.64710- 121270 4  2 384
 0.00000+ 0 1.70000+ 6           0          0          6           01270 4  2 385
 2.37230- 1 1.54260- 1 2.27360- 5-1.63530- 6-1.24960- 9 3.69750- 111270 4  2 386
 0.00000+ 0 1.85000+ 6           0          0          6           01270 4  2 387
 2.58080- 1 1.45740- 1 3.05400- 5-2.45140- 6-2.06350- 9 8.94170- 111270 4  2 388
 0.00000+ 0 2.05000+ 6           0          0          6           01270 4  2 389
 2.77450- 1 1.36370- 1 3.98020- 5-3.64430- 6-3.68300- 9 2.08110- 101270 4  2 390
 0.00000+ 0 2.25000+ 6           0          0          6           01270 4  2 391
 2.90500- 1 1.29340- 1 4.75620- 5-4.90800- 6-6.06630- 9 4.12060- 101270 4  2 392
 0.00000+ 0 2.50000+ 6           0          0          6           01270 4  2 393
 3.01530- 1 1.23710- 1 5.50410- 5-6.51550- 6-1.03640- 8 8.52760- 101270 4  2 394
 0.00000+ 0 2.75000+ 6           0          0          6           01270 4  2 395
 3.09390- 1 1.21310- 1 6.01260- 5-8.09070- 6-1.63810- 8 1.60570-  91270 4  2 396
 0.00000+ 0 3.00000+ 6           0          0          6           01270 4  2 397
 3.15830- 1 1.21670- 1 6.30280- 5-9.59840- 6-2.43140- 8 2.82060-  91270 4  2 398
 0.00000+ 0 3.40000+ 6           0          0          6           01270 4  2 399
 3.25500- 1 1.26720- 1 6.37650- 5-1.18810- 5-4.11360- 8 6.23450-  91270 4  2 400
 0.00000+ 0 4.10000+ 6           0          0          6           01270 4  2 401
 3.43910- 1 1.43150- 1 5.55130- 5-1.59260- 5-8.01090- 8 1.99230-  81270 4  2 402
 0.00000+ 0 5.30000+ 6           0          0          6           01270 4  2 403
 3.77570- 1 1.74640- 1 1.46810- 5-2.51760- 5-1.13170- 7 9.60590-  81270 4  2 404
 0.00000+ 0 6.10000+ 6           0          0          6           01270 4  2 405
 3.97470- 1 1.91440- 1-3.19390- 5-3.27210- 5 2.41650- 8 2.26910-  71270 4  2 406
 0.00000+ 0 6.90000+ 6           0          0          6           01270 4  2 407
 4.14050- 1 2.04270- 1-9.02570- 5-3.90330- 5 5.31490- 7 4.81960-  71270 4  2 408
 0.00000+ 0 7.70000+ 6           0          0          6           01270 4  2 409
 4.27450- 1 2.13840- 1-1.51180- 4-4.06770- 5 1.79080- 6 9.41650-  71270 4  2 410
 0.00000+ 0 8.60000+ 6           0          0          6           01270 4  2 411
 4.39350- 1 2.21730- 1-2.05320- 4-3.14520- 5 4.86400- 6 1.84610-  61270 4  2 412
 0.00000+ 0 9.60000+ 6           0          0          6           01270 4  2 413
 4.49500- 1 2.28000- 1-2.16740- 4 1.41920- 6 1.18800- 5 3.59730-  61270 4  2 414
 0.00000+ 0 1.07000+ 7           0          0          6           01270 4  2 415
 4.57910- 1 2.32900- 1-1.19590- 4 7.92120- 5 2.70530- 5 6.91760-  61270 4  2 416
 0.00000+ 0 1.20000+ 7           0          0          6           01270 4  2 417
 4.65300- 1 2.37040- 1 2.32500- 4 2.52940- 4 6.18900- 5 1.37250-  51270 4  2 418
 0.00000+ 0 1.35000+ 7           0          0          6           01270 4  2 419
 4.71690- 1 2.40680- 1 1.10990- 3 6.10440- 4 1.39900- 4 2.74990-  51270 4  2 420
```

```
0.00000+ 0 1.54000+ 7          0          0          6          01270 4  2  421
4.78160- 1 2.44790- 1 3.23900- 3 1.40920- 3 3.35860- 4 5.89060- 51270 4  2  422
0.00000+ 0 1.81000+ 7          0          0          6          01270 4  2  423
4.86990- 1 2.51670- 1 8.89090- 3 3.51790- 3 9.31780- 4 1.45070- 41270 4  2  424
0.00000+ 0 2.00000+ 7          0          0          6          01270 4  2  425
4.94070- 1 2.58070- 1 1.50490- 2 5.92900- 3 1.68770- 3 2.46350- 41270 4  2  426
0.00000+ 0 0.00000+ 0          0          0          0          01270 4  0  427
0.00000+ 0 0.00000+ 0          0          0          0          01270 0  0  428
0.00000+ 0 0.00000+ 0          0          0          0          0   0 0  0  429
0.00000+ 0 0.00000+ 0          0          0          0          0  -1 0  0  430
```

APPENDIX L

Sample of Interpreted Data Set

The following is an interpreted listing of the ENDF data set for Helium-4. This listing was obtained using the LISTFC code (See Appendix I). Since this is an example, not all of the angular distributions have been listed.

2-HE- 4 LASL EVAL-OCT73 NISLEY, HALE, YOUNG (LASL)
NO PUBLICATION DIST-MAY78 771216

* *

LASL-OCT,1973 R.A. NISLEY, G.M. HALE, P.G. YOUNG.

MF=3 ---------- SMOOTH CROSS SECTIONS ----------

THE 2200 M/S CROSS SECTIONS ARE AS FOLLOWS,
MT=1 SIGMA= 0.75916 BARNS
MT=2 SIGMA= 0.75916 BARNS

MT=1 TOTAL CROSS SECTION
SEE DISCUSSION UNDER MT=2 BELOW

MT=2 ELASTIC SCATTERING CROSS SECTION

ALTHOUGH THE ONLY REACTION POSSIBLE FOR NEUTRONS INCIDENT ON HE-4 BELOW 20 MEV IS ELASTIC SCATTERING, THE MAJORITY OF THE N-4HE DATA IS RATHER IMPRECISE. IN ORDER TO OVERCOME THIS PROBLEM, AN R-MATRIX ANALYSIS WAS PERFORMED WITH A DATA SET WHICH INCLUDED NOT ONLY THE N-4HE DATA BUT ALSO VERY PRECISE P-4HE DATA. ALL THE AVAILABLE N-4HE AND P-4HE DATA BELOW 20 MEV WERE CONSIDERED IN THE ANALYSIS. SINCE THE PREVIOUS EVALUATION WAS COMPLETED IN 1968, SEVERAL N-4HE ELASTIC SCATTERING MEASUREMENTS HAVE BEEN DONE. THE MOST SIGNIFICANT OF THESE ARE THE LOW ENERGY NEUTRON CROSS SECTION OF RORER (RO69),THE RPI TOTAL CROSS SECTION MEASUREMENT (GO73), WHICH COVER THE RANGE EN=0.7-30 MEV, AND THE RELATIVE ANGULAR DISTRIBUTIONS OF MORGAN (MO68). A COMPLETE LIST OF REFERENCES FOR THE N-4HE DATA USED IS GIVEN BELOW. THE P-4HE DATA WAS SELECTED TO SATISFY VERY STRINGENT STATISTICAL CRITERIA AND WE BELIEVE THE POSSIBLE ERRORS OF THE PREDICTED VALUES FOR THE P-4HE SCATTERING TO BE LESS THAN 1.0 PER CENT. A SIMPLE MODEL FOR THE CHARGE DIFFERENCES BETWEEN THE N-4HE AND P-4HE SYSTEMS WAS ASSUMED AND THE N-4HE AND P-4HE DATA SETS WERE SIMULTANEOUSLY ANALYZED. THE VALUES OF THE CROSS SECTIONS AND ANGULAR DISTRIBUTIONS CONTAINED IN FILES 3 AND 4 ARE PROBABLY ACCURATE TO WITHIN 2.0 PERCENT.

MF=4 ---------- NEUTRON ANGULAR DISTRIBUTIONS ----------

MT=2 ELASTIC SCATTERING ANGULAR DISTRIBUTIONS

OBTAINED FROM THE R-MATRIX ANALYSIS DESCRIBED ABOVE UNDER MF=3,MT=2. LEGENDRE POLYNOMIAL REPRESENTATION USED.

---------- REFERENCES----------

AU62 S.M.AUSTIN ET AL., PHYS.REV. 126(1962) 1532.
BR72 W.B.BROSTE ET AL., PHYS.REV. C5(1972) 761.
CR72 D.S.CRAMER + L.CRANBERG, NUCL.PHYS. A180(1972) 273.
FA63 U.FASOLI + G.ZAGO, NUOVO CIMENTO 30(1963) 1169.
GO73 C.A.GOULDING ET AL., BULL.AM.PHYS.SOC. 18(1973) 538.
HO66 B.HOOP,JR. + H.H.BARSCHALL, NUCL.PHYS. 83(1966) 65.
JE66 R.W.JEWELL ET AL., PHYS.REV. 142(1966) 687.
MA63 T.H.MAY ET AL., NUCL.PHYS. 45(1963) 17.(REV. MO68 + SA68)
MO68 G.L.MORGAN + R.L.WALTER, PHYS.REV. 168(1968) 1114.
NI71 A.NIILER ET AL., PHYS.REV. C4(1971) 36. (REV. 9/72)
RO69 D.C.RORER ET AL., NUCL.PHYS. 133(1969) 410.

SA68 J.R.SAWERS ET AL., PHYS.REV. 168(1968) 1102.
SE53 J.D.SEAGRAVE, PHYS.REV. 92(1953) 1222.
SH55 D.F.SHAW, PROC.PHYS.SOC.(LONDON) 68(1955) 43.
SH64 R.E.SHAMU + J.G.JENKIN, PHYS.REV. 135(1964) B99.
SM54 J.R.SMITH, PHYS.REV. 95(1954) 730.
ST70 T.STAMMBACH ET AL., PHYS.REV. C2(1970) 434.
WH57 R.E.WHITE + F.J.M.FARLEY, NUCL.PHYS. 3(1957) 476.
YO63 P.G.YOUNG ET AL., AUST.J.PHYS. 16(1963) 185.

TABLE OF CONTENTS ENDF/B MATERIAL NO. 1270
GENERAL INFORMATION

RESONANCE DATA ENDF/B MATERIAL NO. 1270
RESONANCE PARAMETERS

ISOTOPE----------
FRACTIONAL ABUNDANCE--------- 1.0000E+00
NUCLEAR SPIN---------- 0.0000E+00
SPIN SCATTERING LENGTH (A+)-- 2.4579E-01

NO RESONANCE PARAMETERS ARE GIVEN FOR THIS MATERIAL

TOTAL ENDF/B MATERIAL NO. 1270
NEUTRON CROSS SECTION

INTERPOLATION LAW BETWEEN ENERGIES
RANGE DESCRIPTION
1 TO 257 Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
1	1.0000E-05	7.5916E-01	2.5300E-02	7.5916E-01	1.0000E+00	7.5916E-01	1.0000E+01	7.5916E-01	1.0000E+02	7.5916E-01
6	1.0000E+03	7.5911E-01	5.0000E+03	7.5891E-01	1.0000E+04	7.5872E-01	2.0000E+04	7.5849E-01	4.0000E+04	7.5875E-01
11	6.0000E+04	7.6003E-01	8.0000E+04	7.6243E-01	1.0000E+05	7.6609E-01	1.5000E+05	7.8158E-01	2.0000E+05	8.0817E-01
16	2.5000E+05	8.4899E-01	3.0000E+05	9.0808E-01	3.5000E+05	9.9059E-01	4.0000E+05	1.1030E+00	4.5000E+05	1.2536E+00
21	5.0000E+05	1.4525E+00	5.5000E+05	1.7114E+00	6.0000E+05	2.0438E+00	6.2500E+05	2.2418E+00	6.5000E+05	2.4630E+00
26	6.7500E+05	2.7087E+00	7.0000E+05	2.9795E+00	7.2500E+05	3.2755E+00	7.5000E+05	3.5958E+00	7.7500E+05	3.9386E+00
31	8.0000E+05	4.3006E+00	8.2500E+05	4.6770E+00	8.5000E+05	5.0616E+00	8.7500E+05	5.4469E+00	9.0000E+05	5.8241E+00
36	9.2500E+05	6.1842E+00	9.5000E+05	6.5180E+00	9.7500E+05	6.8173E+00	1.0000E+06	7.0752E+00	1.0250E+06	7.2867E+00
41	1.0500E+06	7.4494E+00	1.0750E+06	7.5627E+00	1.1000E+06	7.6285E+00	1.1250E+06	7.6501E+00	1.1500E+06	7.6321E+00
46	1.1750E+06	7.5798E+00	1.2000E+06	7.4987E+00	1.2250E+06	7.3945E+00	1.2500E+06	7.2721E+00	1.2750E+06	7.1362E+00
51	1.3000E+06	6.9909E+00	1.3250E+06	6.8395E+00	1.3500E+06	6.6850E+00	1.3750E+06	6.5295E+00	1.4000E+06	6.3748E+00
56	1.4500E+06	6.0735E+00	1.5000E+06	5.7882E+00	1.5500E+06	5.5226E+00	1.6000E+06	5.2779E+00	1.6500E+06	5.0539E+00
61	1.7000E+06	4.8496E+00	1.7500E+06	4.6636E+00	1.8000E+06	4.4943E+00	1.8500E+06	4.3401E+00	1.9000E+06	4.1996E+00
66	1.9500E+06	4.0713E+00	2.0000E+06	3.9541E+00	2.0500E+06	3.8467E+00	2.1000E+06	3.7482E+00	2.1500E+06	3.6576E+00
71	2.2000E+06	3.5741E+00	2.2500E+06	3.4971E+00	2.3000E+06	3.4259E+00	2.3500E+06	3.3598E+00	2.4000E+06	3.2986E+00
76	2.4500E+06	3.2416E+00	2.5000E+06	3.1885E+00	2.5500E+06	3.1389E+00	2.6000E+06	3.0925E+00	2.6500E+06	3.0491E+00
81	2.7000E+06	3.0084E+00	2.7500E+06	2.9701E+00	2.8000E+06	2.9340E+00	2.8500E+06	2.9000E+00	2.9000E+06	2.8679E+00
86	2.9500E+06	2.8375E+00	3.0000E+06	2.8087E+00	3.1000E+06	2.7554E+00	3.2000E+06	2.7070E+00	3.3000E+06	2.6627E+00
91	3.4000E+06	2.6221E+00	3.5000E+06	2.5844E+00	3.6000E+06	2.5494E+00	3.7000E+06	2.5165E+00	3.8000E+06	2.4855E+00
96	3.9000E+06	2.4561E+00	4.0000E+06	2.4280E+00	4.1000E+06	2.4011E+00	4.2000E+06	2.3753E+00	4.3000E+06	2.3503E+00
101	4.4000E+06	2.3260E+00	4.5000E+06	2.3024E+00	4.6000E+06	2.2794E+00	4.7000E+06	2.2569E+00	4.8000E+06	2.2349E+00
106	4.9000E+06	2.2132E+00	5.0000E+06	2.1919E+00	5.1000E+06	2.1710E+00	5.2000E+06	2.1504E+00	5.3000E+06	2.1301E+00
111	5.4000E+06	2.1100E+00	5.5000E+06	2.0903E+00	5.6000E+06	2.0707E+00	5.7000E+06	2.0515E+00	5.8000E+06	2.0324E+00
116	5.9000E+06	2.0136E+00	6.0000E+06	1.9950E+00	6.1000E+06	1.9767E+00	6.2000E+06	1.9586E+00	6.3000E+06	1.9407E+00
121	6.4000E+06	1.9230E+00	6.5000E+06	1.9055E+00	6.6000E+06	1.8883E+00	6.7000E+06	1.8712E+00	6.8000E+06	1.8544E+00
126	6.9000E+06	1.8378E+00	7.0000E+06	1.8213E+00	7.1000E+06	1.8051E+00	7.2000E+06	1.7891E+00	7.3000E+06	1.7733E+00
131	7.4000E+06	1.7577E+00	7.5000E+06	1.7423E+00	7.6000E+06	1.7271E+00	7.7000E+06	1.7120E+00	7.8000E+06	1.6972E+00
136	7.9000E+06	1.6825E+00	8.0000E+06	1.6681E+00	8.1000E+06	1.6538E+00	8.2000E+06	1.6397E+00	8.3000E+06	1.6258E+00
141	8.4000E+06	1.6120E+00	8.5000E+06	1.5985E+00	8.6000E+06	1.5851E+00	8.7000E+06	1.5718E+00	8.8000E+06	1.5588E+00
146	8.9000E+06	1.5459E+00	9.0000E+06	1.5332E+00	9.1000E+06	1.5206E+00	9.2000E+06	1.5082E+00	9.3000E+06	1.4959E+00
151	9.4000E+06	1.4838E+00	9.5000E+06	1.4718E+00	9.6000E+06	1.4600E+00	9.7000E+06	1.4484E+00	9.8000E+06	1.4369E+00
156	9.9000E+06	1.4255E+00	1.0000E+07	1.4142E+00	1.0100E+07	1.4032E+00	1.0200E+07	1.3922E+00	1.0300E+07	1.3814E+00
161	1.0400E+07	1.3707E+00	1.0500E+07	1.3601E+00	1.0600E+07	1.3497E+00	1.0700E+07	1.3393E+00	1.0800E+07	1.3292E+00
166	1.0900E+07	1.3191E+00	1.1000E+07	1.3091E+00	1.1100E+07	1.2993E+00	1.1200E+07	1.2896E+00	1.1300E+07	1.2800E+00
171	1.1400E+07	1.2705E+00	1.1500E+07	1.2612E+00	1.1600E+07	1.2519E+00	1.1700E+07	1.2428E+00	1.1800E+07	1.2337E+00
176	1.1900E+07	1.2248E+00	1.2000E+07	1.2160E+00	1.2100E+07	1.2072E+00	1.2200E+07	1.1986E+00	1.2300E+07	1.1901E+00
181	1.2400E+07	1.1817E+00	1.2500E+07	1.1733E+00	1.2600E+07	1.1651E+00	1.2700E+07	1.1570E+00	1.2800E+07	1.1489E+00
186	1.2900E+07	1.1410E+00	1.3000E+07	1.1331E+00	1.3100E+07	1.1254E+00	1.3200E+07	1.1177E+00	1.3300E+07	1.1101E+00
191	1.3400E+07	1.1026E+00	1.3500E+07	1.0952E+00	1.3600E+07	1.0878E+00	1.3700E+07	1.0806E+00	1.3800E+07	1.0734E+00
196	1.3900E+07	1.0664E+00	1.4000E+07	1.0594E+00	1.4100E+07	1.0525E+00	1.4200E+07	1.0456E+00	1.4300E+07	1.0389E+00
201	1.4400E+07	1.0322E+00	1.4500E+07	1.0256E+00	1.4600E+07	1.0190E+00	1.4700E+07	1.0126E+00	1.4800E+07	1.0062E+00
206	1.4900E+07	9.9990E-01	1.5000E+07	9.9367E-01	1.5100E+07	9.8751E-01	1.5200E+07	9.8142E-01	1.5300E+07	9.7540E-01
211	1.5400E+07	9.6945E-01	1.5500E+07	9.6357E-01	1.5600E+07	9.5776E-01	1.5700E+07	9.5201E-01	1.5800E+07	9.4634E-01
216	1.5900E+07	9.4072E-01	1.6000E+07	9.3518E-01	1.6100E+07	9.2970E-01	1.6200E+07	9.2428E-01	1.6300E+07	9.1893E-01

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
221	1.6400E+07	9.1364E-01	1.6500E+07	9.0841E-01	1.6600E+07	9.0324E-01	1.6700E+07	8.9814E-01	1.6800E+07	8.9310E-01
226	1.6900E+07	8.8811E-01	1.7000E+07	8.8319E-01	1.7100E+07	8.7832E-01	1.7200E+07	8.7352E-01	1.7300E+07	8.6877E-01
231	1.7400E+07	8.6408E-01	1.7500E+07	8.5944E-01	1.7600E+07	8.5486E-01	1.7700E+07	8.5034E-01	1.7800E+07	8.4587E-01
236	1.7900E+07	8.4146E-01	1.8000E+07	8.3710E-01	1.8100E+07	8.3280E-01	1.8200E+07	8.2855E-01	1.8300E+07	8.2435E-01
241	1.8400E+07	8.2021E-01	1.8500E+07	8.1611E-01	1.8600E+07	8.1207E-01	1.8700E+07	8.0808E-01	1.8800E+07	8.0414E-01
246	1.8900E+07	8.0025E-01	1.9000E+07	7.9641E-01	1.9100E+07	7.9262E-01	1.9200E+07	7.8888E-01	1.9300E+07	7.8518E-01
251	1.9400E+07	7.8154E-01	1.9500E+07	7.7794E-01	1.9600E+07	7.7439E-01	1.9700E+07	7.7088E-01	1.9800E+07	7.6743E-01
256	1.9900E+07	7.6401E-01	2.0000E+07	7.6065E-01						

ELASTIC
NEUTRON CROSS SECTION

ENDF/B MATERIAL NO. 1270

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 257	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
1	1.0000E-05	7.5916E-01	2.5300E-02	7.5916E-01	1.0000E+00	7.5916E-01	1.0000E+01	7.5916E-01	1.0000E+02	7.5916E-01
6	1.0000E+03	7.5911E-01	5.0000E+03	7.5891E-01	1.0000E+04	7.5872E-01	2.0000E+04	7.5849E-01	4.0000E+04	7.5875E-01
11	6.0000E+04	7.6003E-01	8.0000E+04	7.6243E-01	1.0000E+05	7.6609E-01	1.5000E+05	7.8158E-01	2.0000E+05	8.0817E-01
16	2.5000E+05	8.4899E-01	3.0000E+05	9.0808E-01	3.5000E+05	9.9059E-01	4.0000E+05	1.1030E+00	4.5000E+05	1.2536E+00
21	5.0000E+05	1.4525E+00	5.5000E+05	1.7114E+00	6.0000E+05	2.0438E+00	6.2500E+05	2.2418E+00	6.5000E+05	2.4630E+00
26	6.7500E+05	2.7087E+00	7.0000E+05	2.9795E+00	7.2500E+05	3.2755E+00	7.5000E+05	3.5958E+00	7.7500E+05	3.9386E+00
31	8.0000E+05	4.3006E+00	8.2500E+05	4.6770E+00	8.5000E+05	5.0616E+00	8.7500E+05	5.4469E+00	9.0000E+05	5.8241E+00
36	9.2500E+05	6.1842E+00	9.5000E+05	6.5180E+00	9.7500E+05	6.8173E+00	1.0000E+06	7.0752E+00	1.0250E+06	7.2867E+00
41	1.0500E+06	7.4494E+00	1.0750E+06	7.5627E+00	1.1000E+06	7.6285E+00	1.1250E+06	7.6501E+00	1.1500E+06	7.6321E+00
46	1.1750E+06	7.5798E+00	1.2000E+06	7.4987E+00	1.2250E+06	7.3945E+00	1.2500E+06	7.2721E+00	1.2750E+06	7.1362E+00
51	1.3000E+06	6.9909E+00	1.3250E+06	6.8395E+00	1.3500E+06	6.6850E+00	1.3750E+06	6.5295E+00	1.4000E+06	6.3748E+00
56	1.4500E+06	6.0735E+00	1.5000E+06	5.7882E+00	1.5500E+06	5.5226E+00	1.6000E+06	5.2779E+00	1.6500E+06	5.0539E+00
61	1.7000E+06	4.8496E+00	1.7500E+06	4.6636E+00	1.8000E+06	4.4943E+00	1.8500E+06	4.3401E+00	1.9000E+06	4.1996E+00
66	1.9500E+06	4.0713E+00	2.0000E+06	3.9541E+00	2.0500E+06	3.8467E+00	2.1000E+06	3.7482E+00	2.1500E+06	3.6576E+00
71	2.2000E+06	3.5741E+00	2.2500E+06	3.4971E+00	2.3000E+06	3.4259E+00	2.3500E+06	3.3598E+00	2.4000E+06	3.2986E+00
76	2.4500E+06	3.2416E+00	2.5000E+06	3.1885E+00	2.5500E+06	3.1389E+00	2.6000E+06	3.0925E+00	2.6500E+06	3.0491E+00
81	2.7000E+06	3.0084E+00	2.7500E+06	2.9701E+00	2.8000E+06	2.9340E+00	2.8500E+06	2.9000E+00	2.9000E+06	2.8679E+00
86	2.9500E+06	2.8375E+00	3.0000E+06	2.8087E+00	3.1000E+06	2.7554E+00	3.2000E+06	2.7070E+00	3.3000E+06	2.6627E+00
91	3.4000E+06	2.6221E+00	3.5000E+06	2.5844E+00	3.6000E+06	2.5494E+00	3.7000E+06	2.5165E+00	3.8000E+06	2.4855E+00
96	3.9000E+06	2.4561E+00	4.0000E+06	2.4280E+00	4.1000E+06	2.4011E+00	4.2000E+06	2.3753E+00	4.3000E+06	2.3503E+00
101	4.4000E+06	2.3260E+00	4.5000E+06	2.3024E+00	4.6000E+06	2.2794E+00	4.7000E+06	2.2569E+00	4.8000E+06	2.2349E+00
106	4.9000E+06	2.2132E+00	5.0000E+06	2.1919E+00	5.1000E+06	2.1710E+00	5.2000E+06	2.1504E+00	5.3000E+06	2.1301E+00
111	5.4000E+06	2.1100E+00	5.5000E+06	2.0903E+00	5.6000E+06	2.0707E+00	5.7000E+06	2.0515E+00	5.8000E+06	2.0324E+00
116	5.9000E+06	2.0136E+00	6.0000E+06	1.9950E+00	6.1000E+06	1.9767E+00	6.2000E+06	1.9586E+00	6.3000E+06	1.9407E+00
121	6.4000E+06	1.9230E+00	6.5000E+06	1.9055E+00	6.6000E+06	1.8883E+00	6.7000E+06	1.8712E+00	6.8000E+06	1.8544E+00
126	6.9000E+06	1.8378E+00	7.0000E+06	1.8213E+00	7.1000E+06	1.8051E+00	7.2000E+06	1.7891E+00	7.3000E+06	1.7733E+00

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
131	7.4000E+06	1.7577E+00	7.5000E+06	1.7423E+00	7.6000E+06	1.7271E+00	7.7000E+06	1.7120E+00	7.8000E+06	1.6972E+00
136	7.9000E+06	1.6825E+00	8.0000E+06	1.6681E+00	8.1000E+06	1.6538E+00	8.2000E+06	1.6397E+00	8.3000E+06	1.6258E+00
141	8.4000E+06	1.6120E+00	8.5000E+06	1.5985E+00	8.6000E+06	1.5851E+00	8.7000E+06	1.5718E+00	8.8000E+06	1.5588E+00
131	7.4000E+06	1.7577E+00	7.5000E+06	1.7423E+00	7.6000E+06	1.7271E+00	7.7000E+06	1.7120E+00	7.8000E+06	1.6972E+00
136	7.9000E+06	1.6825E+00	8.0000E+06	1.6681E+00	8.1000E+06	1.6538E+00	8.2000E+06	1.6397E+00	8.3000E+06	1.6258E+00
141	8.4000E+06	1.6120E+00	8.5000E+06	1.5985E+00	8.6000E+06	1.5851E+00	8.7000E+06	1.5718E+00	8.8000E+06	1.5588E+00
146	8.9000E+06	1.5459E+00	9.0000E+06	1.5332E+00	9.1000E+06	1.5206E+00	9.2000E+06	1.5082E+00	9.3000E+06	1.4959E+00
151	9.4000E+06	1.4838E+00	9.5000E+06	1.4718E+00	9.6000E+06	1.4600E+00	9.7000E+06	1.4484E+00	9.8000E+06	1.4369E+00
156	9.9000E+06	1.4255E+00	1.0000E+07	1.4142E+00	1.0100E+07	1.4032E+00	1.0200E+07	1.3922E+00	1.0300E+07	1.3814E+00
161	1.0400E+07	1.3707E+00	1.0500E+07	1.3601E+00	1.0600E+07	1.3497E+00	1.0700E+07	1.3393E+00	1.0800E+07	1.3292E+00
166	1.0900E+07	1.3191E+00	1.1000E+07	1.3091E+00	1.1100E+07	1.2993E+00	1.1200E+07	1.2896E+00	1.1300E+07	1.2800E+00
171	1.1400E+07	1.2705E+00	1.1500E+07	1.2612E+00	1.1600E+07	1.2519E+00	1.1700E+07	1.2428E+00	1.1800E+07	1.2337E+00
176	1.1900E+07	1.2248E+00	1.2000E+07	1.2160E+00	1.2100E+07	1.2072E+00	1.2200E+07	1.1986E+00	1.2300E+07	1.1901E+00
181	1.2400E+07	1.1817E+00	1.2500E+07	1.1733E+00	1.2600E+07	1.1651E+00	1.2700E+07	1.1570E+00	1.2800E+07	1.1489E+00
186	1.2900E+07	1.1410E+00	1.3000E+07	1.1331E+00	1.3100E+07	1.1254E+00	1.3200E+07	1.1177E+00	1.3300E+07	1.1101E+00
191	1.3400E+07	1.1026E+00	1.3500E+07	1.0952E+00	1.3600E+07	1.0878E+00	1.3700E+07	1.0806E+00	1.3800E+07	1.0734E+00
196	1.3900E+07	1.0664E+00	1.4000E+07	1.0594E+00	1.4100E+07	1.0525E+00	1.4200E+07	1.0456E+00	1.4300E+07	1.0389E+00
201	1.4400E+07	1.0322E+00	1.4500E+07	1.0256E+00	1.4600E+07	1.0190E+00	1.4700E+07	1.0126E+00	1.4800E+07	1.0062E+00
206	1.4900E+07	9.9990E-01	1.5000E+07	9.9367E-01	1.5100E+07	9.8751E-01	1.5200E+07	9.8142E-01	1.5300E+07	9.7540E-01
211	1.5400E+07	9.6945E-01	1.5500E+07	9.6357E-01	1.5600E+07	9.5776E-01	1.5700E+07	9.5201E-01	1.5800E+07	9.4634E-01
216	1.5900E+07	9.4072E-01	1.6000E+07	9.3518E-01	1.6100E+07	9.2970E-01	1.6200E+07	9.2428E-01	1.6300E+07	9.1893E-01
221	1.6400E+07	9.1364E-01	1.6500E+07	9.0841E-01	1.6600E+07	9.0324E-01	1.6700E+07	8.9814E-01	1.6800E+07	8.9310E-01
226	1.6900E+07	8.8811E-01	1.7000E+07	8.8319E-01	1.7100E+07	8.7832E-01	1.7200E+07	8.7352E-01	1.7300E+07	8.6877E-01
231	1.7400E+07	8.6408E-01	1.7500E+07	8.5944E-01	1.7600E+07	8.5486E-01	1.7700E+07	8.5034E-01	1.7800E+07	8.4587E-01
236	1.7900E+07	8.4146E-01	1.8000E+07	8.3710E-01	1.8100E+07	8.3280E-01	1.8200E+07	8.2855E-01	1.8300E+07	8.2435E-01
241	1.8400E+07	8.2021E-01	1.8500E+07	8.1611E-01	1.8600E+07	8.1207E-01	1.8700E+07	8.0808E-01	1.8800E+07	8.0414E-01
246	1.8900E+07	8.0025E-01	1.9000E+07	7.9641E-01	1.9100E+07	7.9262E-01	1.9200E+07	7.8888E-01	1.9300E+07	7.8518E-01
251	1.9400E+07	7.8154E-01	1.9500E+07	7.7794E-01	1.9600E+07	7.7439E-01	1.9700E+07	7.7088E-01	1.9800E+07	7.6743E-01
256	1.9900E+07	7.6401E-01	2.0000E+07	7.6065E-01						

MU BAR ENDF/B MATERIAL NO. 1270
NEUTRON CROSS SECTION

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1.0000E-05	1.6981E-01	1.0000E+00	1.6981E-01	6.0000E+04	9.6622E-02	1.0000E+05	4.5187E-02	1.5000E+05-	2.0449E-02
6	2.0000E+05-	8.5137E-02	2.5000E+05-	1.4563E-01	3.0000E+05-	1.9826E-01	3.5000E+05-	2.3955E-01	4.0000E+05-	2.6668E-01
11	4.5000E+05-	2.7821E-01	5.0000E+05-	2.7423E-01	5.5000E+05-	2.5632E-01	6.0000E+05-	2.2706E-01	6.5000E+05-	1.8952E-01
36	6.1000E+06	5.2093E-01	6.9000E+06	5.3478E-01	7.7000E+06	5.4610E-01	8.6000E+06	5.5625E-01	9.6000E+06	5.6499E-01
16	7.0000E+05-	1.4678E-01	7.7500E+05-	7.8552E-02	8.5000E+05-	1.1080E-02	9.2500E+05	5.1757E-02	1.0000E+06	1.0815E-01
21	1.0750E+06	1.5761E-01	1.1750E+06	2.1329E-01	1.2750E+06	2.5862E-01	1.4000E+06	3.0334E-01	1.5500E+06	3.4348E-01
26	1.7000E+06	3.7277E-01	1.8500E+06	3.9425E-01	2.0500E+06	4.1446E-01	2.2500E+06	4.2819E-01	2.5000E+06	4.3974E-01

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
31	2.7500E+06	4.4771E-01	3.0000E+06	4.5385E-01	3.4000E+06	4.6233E-01	4.1000E+06	4.7734E-01	5.3000E+06	5.0455E-01
36	6.1000E+06	5.2093E-01	6.9000E+06	5.3478E-01	7.7000E+06	5.4610E-01	8.6000E+06	5.5625E-01	9.6000E+06	5.6499E-01
41	1.0700E+07	5.7228E-01	1.2000E+07	5.7873E-01	1.3500E+07	5.8431E-01	1.5400E+07	5.8993E-01	1.8100E+07	5.9750E-01
46	2.0000E+07	6.0347E-01								

XI ENDF/B MATERIAL NO. 1270
NEUTRON CROSS SECTION

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1.0000E-05	4.2518E-01	1.0000E+00	4.2518E-01	6.0000E+04	4.6268E-01	1.0000E+05	4.8903E-01	1.5000E+05	5.2265E-01
6	2.0000E+05	5.5577E-01	2.5000E+05	5.8673E-01	3.0000E+05	6.1366E-01	3.5000E+05	6.3476E-01	4.0000E+05	6.4861E-01
11	4.5000E+05	6.5446E-01	5.0000E+05	6.5237E-01	5.5000E+05	6.4313E-01	6.0000E+05	6.2809E-01	6.5000E+05	6.0882E-01
16	7.0000E+05	5.8689E-01	7.7500E+05	5.5189E-01	8.5000E+05	5.1730E-01	9.2500E+05	4.8510E-01	1.0000E+06	4.5620E-01
21	1.0750E+06	4.3087E-01	1.1750E+06	4.0235E-01	1.2750E+06	3.7914E-01	1.4000E+06	3.5625E-01	1.5500E+06	3.3571E-01
26	1.7000E+06	3.2073E-01	1.8500E+06	3.0974E-01	2.0500E+06	2.9942E-01	2.2500E+06	2.9240E-01	2.5000E+06	2.8650E-01
31	2.7500E+06	2.8242E-01	3.0000E+06	2.7927E-01	3.4000E+06	2.7491E-01	4.1000E+06	2.6717E-01	5.3000E+06	2.5314E-01
36	6.1000E+06	2.4469E-01	6.9000E+06	2.3756E-01	7.7000E+06	2.3173E-01	8.6000E+06	2.2651E-01	9.6000E+06	2.2201E-01
41	1.0700E+07	2.1826E-01	1.2000E+07	2.1495E-01	1.3500E+07	2.1208E-01	1.5400E+07	2.0919E-01	1.8100E+07	2.0531E-01
46	2.0000E+07	2.0225E-01								

GAMMA ENDF/B MATERIAL NO. 1270
NEUTRON CROSS SECTION

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1.0000E-05	3.0968E-01	1.0000E+00	3.0968E-01	6.0000E+04	3.2202E-01	1.0000E+05	3.2988E-01	1.5000E+05	3.3915E-01
6	2.0000E+05	3.4767E-01	2.5000E+05	3.5530E-01	3.0000E+05	3.6194E-01	3.5000E+05	3.6752E-01	4.0000E+05	3.7197E-01
11	4.5000E+05	3.7529E-01	5.0000E+05	3.7749E-01	5.5000E+05	3.7864E-01	6.0000E+05	3.7878E-01	6.5000E+05	3.7801E-01
16	7.0000E+05	3.7644E-01	7.7500E+05	3.7279E-01	8.5000E+05	3.6793E-01	9.2500E+05	3.6220E-01	1.0000E+06	3.5591E-01
21	1.0750E+06	3.4932E-01	1.1750E+06	3.4042E-01	1.2750E+06	3.3171E-01	1.4000E+06	3.2149E-01	1.5500E+06	3.1050E-01
26	1.7000E+06	3.0109E-01	1.8500E+06	2.9322E-01	2.0500E+06	2.8492E-01	2.2500E+06	2.7876E-01	2.5000E+06	2.7342E-01
31	2.7500E+06	2.7005E-01	3.0000E+06	2.6803E-01	3.4000E+06	2.6653E-01	4.1000E+06	2.6607E-01	5.3000E+06	2.6572E-01
36	6.1000E+06	2.6477E-01	6.9000E+06	2.6344E-01	7.7000E+06	2.6196E-01	8.6000E+06	2.6032E-01	9.6000E+06	2.5865E-01

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
41	1.0700E+07	2.5707E-01	1.2000E+07	2.5553E-01	1.3500E+07	2.5411E-01	1.5400E+07	2.5269E-01	1.8100E+07	2.5096E-01
46	2.0000E+07	2.4976E-01								

ELASTIC ENDF/B MATERIAL NO. 1270
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

ELASTIC ENDF/B MATERIAL NO. 1270
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1.0000E-11 MEV IS ISOTROPIC
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 0.0000E+00 2 0.0000E+00

F(MU)=(SUM OVER L) (0.5*(2L+1)*F(L,E)*P(L,MU))

INDEX	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)
1	1.0000E+00	5.0000E-01	9.7500E-01	5.0000E-01	9.5000E-01	5.0000E-01	9.2500E-01	5.0000E-01	9.0000E-01	5.0000E-01
6	8.7500E-01	5.0000E-01	8.5000E-01	5.0000E-01	8.2500E-01	5.0000E-01	8.0000E-01	5.0000E-01	7.7500E-01	5.0000E-01
11	7.5000E-01	5.0000E-01	7.2500E-01	5.0000E-01	7.0000E-01	5.0000E-01	6.7500E-01	5.0000E-01	6.5000E-01	5.0000E-01
16	6.2500E-01	5.0000E-01	6.0000E-01	5.0000E-01	5.7500E-01	5.0000E-01	5.5000E-01	5.0000E-01	5.2500E-01	5.0000E-01
21	5.0000E-01	5.0000E-01	4.7500E-01	5.0000E-01	4.5000E-01	5.0000E-01	4.2500E-01	5.0000E-01	4.0000E-01	5.0000E-01
26	3.7500E-01	5.0000E-01	3.5000E-01	5.0000E-01	3.2500E-01	5.0000E-01	3.0000E-01	5.0000E-01	2.7500E-01	5.0000E-01
31	2.5000E-01	5.0000E-01	2.2500E-01	5.0000E-01	2.0000E-01	5.0000E-01	1.7500E-01	5.0000E-01	1.5000E-01	5.0000E-01
36	1.2500E-01	5.0000E-01	1.0000E-01	5.0000E-01	7.5000E-02	5.0000E-01	5.0000E-02	5.0000E-01	2.5000E-02	5.0000E-01
41	4.4238E-08	5.0000E-01	-2.5000E-02	5.0000E-01	-5.0000E-02	5.0000E-01	-7.5000E-02	5.0000E-01	-1.0000E-01	5.0000E-01
46	-1.2500E-01	5.0000E-01	-1.5000E-01	5.0000E-01	-1.7500E-01	5.0000E-01	-2.0000E-01	5.0000E-01	-2.2500E-01	5.0000E-01
51	-2.5000E-01	5.0000E-01	-2.7500E-01	5.0000E-01	-3.0000E-01	5.0000E-01	-3.2500E-01	5.0000E-01	-3.5000E-01	5.0000E-01
56	-3.7500E-01	5.0000E-01	-4.0000E-01	5.0000E-01	-4.2500E-01	5.0000E-01	-4.5000E-01	5.0000E-01	-4.7500E-01	5.0000E-01
61	-5.0000E-01	5.0000E-01	-5.2500E-01	5.0000E-01	-5.5000E-01	5.0000E-01	-5.7500E-01	5.0000E-01	-6.0000E-01	5.0000E-01
66	-6.2500E-01	5.0000E-01	-6.5000E-01	5.0000E-01	-6.7500E-01	5.0000E-01	-7.0000E-01	5.0000E-01	-7.2500E-01	5.0000E-01
71	-7.5000E-01	5.0000E-01	-7.7500E-01	5.0000E-01	-8.0000E-01	5.0000E-01	-8.2500E-01	5.0000E-01	-8.5000E-01	5.0000E-01
76	-8.7500E-01	5.0000E-01	-9.0000E-01	5.0000E-01	-9.2500E-01	5.0000E-01	-9.5000E-01	5.0000E-01	-9.7500E-01	5.0000E-01
81	-1.0000E+00	5.0000E-01								

ELASTIC ENDF/B MATERIAL NO. 1270

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1.0000E-06 MEV IS ISOTROPIC
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 0.0000E+00 2 0.0000E+00

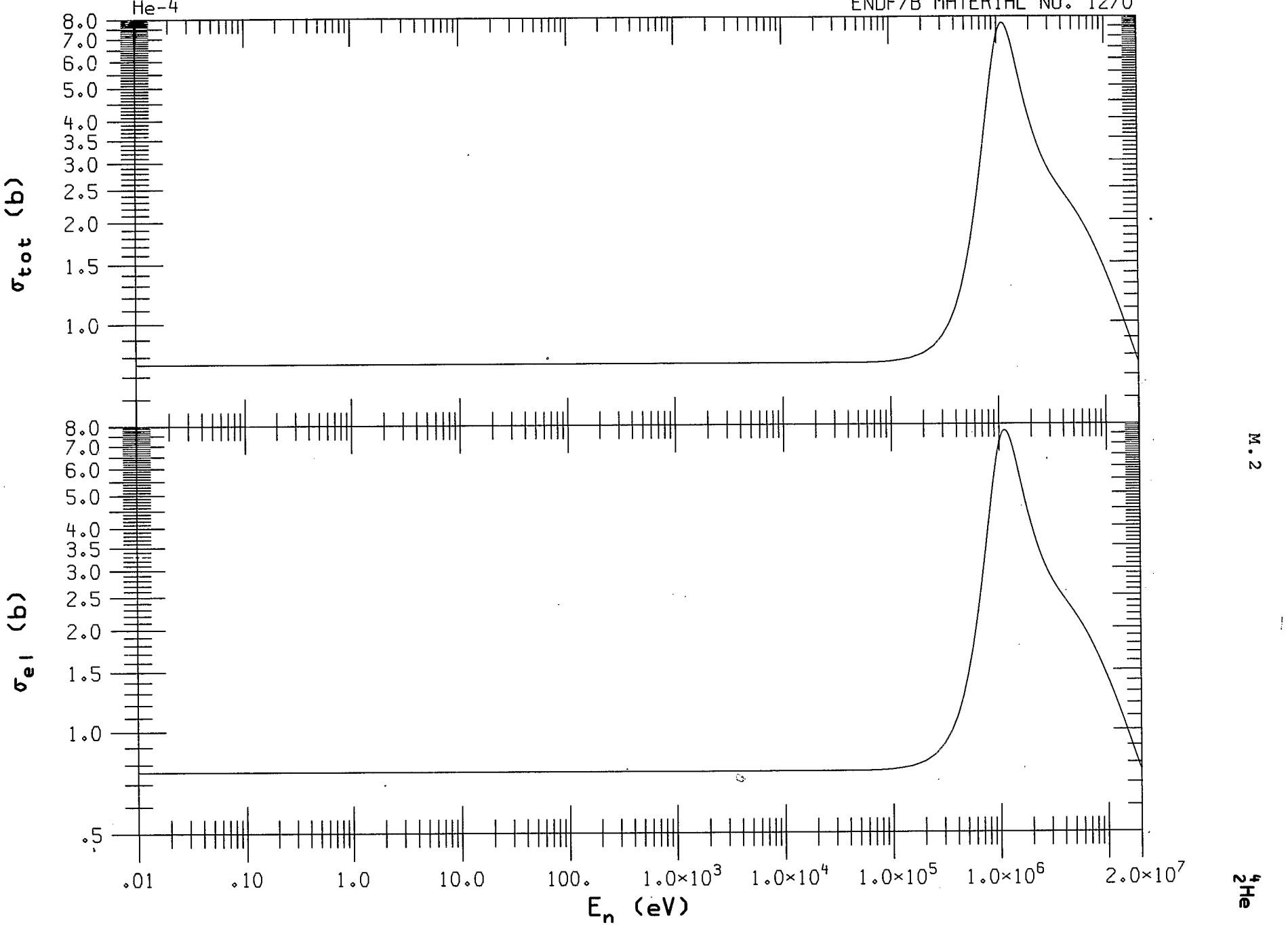
F(MU)=(SUM OVER L) (0.5*(2L+1)*F(L,E)*P(L,MU))

INDEX	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)
1	1.0000E+00	5.0000E-01	9.7500E-01	5.0000E-01	9.5000E-01	5.0000E-01	9.2500E-01	5.0000E-01	9.0000E-01	5.0000E-01
6	8.7500E-01	5.0000E-01	8.5000E-01	5.0000E-01	8.2500E-01	5.0000E-01	8.0000E-01	5.0000E-01	7.7500E-01	5.0000E-01
11	7.5000E-01	5.0000E-01	7.2500E-01	5.0000E-01	7.0000E-01	5.0000E-01	6.7500E-01	5.0000E-01	6.5000E-01	5.0000E-01
16	6.2500E-01	5.0000E-01	6.0000E-01	5.0000E-01	5.7500E-01	5.0000E-01	5.5000E-01	5.0000E-01	5.2500E-01	5.0000E-01
21	5.0000E-01	5.0000E-01	4.7500E-01	5.0000E-01	4.5000E-01	5.0000E-01	4.2500E-01	5.0000E-01	4.0000E-01	5.0000E-01
26	3.7500E-01	5.0000E-01	3.5000E-01	5.0000E-01	3.2500E-01	5.0000E-01	3.0000E-01	5.0000E-01	2.7500E-01	5.0000E-01
31	2.5000E-01	5.0000E-01	2.2500E-01	5.0000E-01	2.0000E-01	5.0000E-01	1.7500E-01	5.0000E-01	1.5000E-01	5.0000E-01
36	1.2500E-01	5.0000E-01	1.0000E-01	5.0000E-01	7.5000E-02	5.0000E-01	5.0000E-02	5.0000E-01	2.5000E-02	5.0000E-01
41	4.4238E-08	5.0000E-01	-2.5000E-02	5.0000E-01	-5.0000E-02	5.0000E-01	-7.5000E-02	5.0000E-01	-1.0000E-01	5.0000E-01
46	-1.2500E-01	5.0000E-01	-1.5000E-01	5.0000E-01	-1.7500E-01	5.0000E-01	-2.0000E-01	5.0000E-01	-2.2500E-01	5.0000E-01
51	-2.5000E-01	5.0000E-01	-2.7500E-01	5.0000E-01	-3.0000E-01	5.0000E-01	-3.2500E-01	5.0000E-01	-3.5000E-01	5.0000E-01
56	-3.7500E-01	5.0000E-01	-4.0000E-01	5.0000E-01	-4.2500E-01	5.0000E-01	-4.5000E-01	5.0000E-01	-4.7500E-01	5.0000E-01
61	-5.0000E-01	5.0000E-01	-5.2500E-01	5.0000E-01	-5.5000E-01	5.0000E-01	-5.7500E-01	5.0000E-01	-6.0000E-01	5.0000E-01
66	-6.2500E-01	5.0000E-01	-6.5000E-01	5.0000E-01	-6.7500E-01	5.0000E-01	-7.0000E-01	5.0000E-01	-7.2500E-01	5.0000E-01
71	-7.5000E-01	5.0000E-01	-7.7500E-01	5.0000E-01	-8.0000E-01	5.0000E-01	-8.2500E-01	5.0000E-01	-8.5000E-01	5.0000E-01
76	-8.7500E-01	5.0000E-01	-9.0000E-01	5.0000E-01	-9.2500E-01	5.0000E-01	-9.5000E-01	5.0000E-01	-9.7500E-01	5.0000E-01
81	-1.0000E+00	5.0000E-01								

APPENDIX M

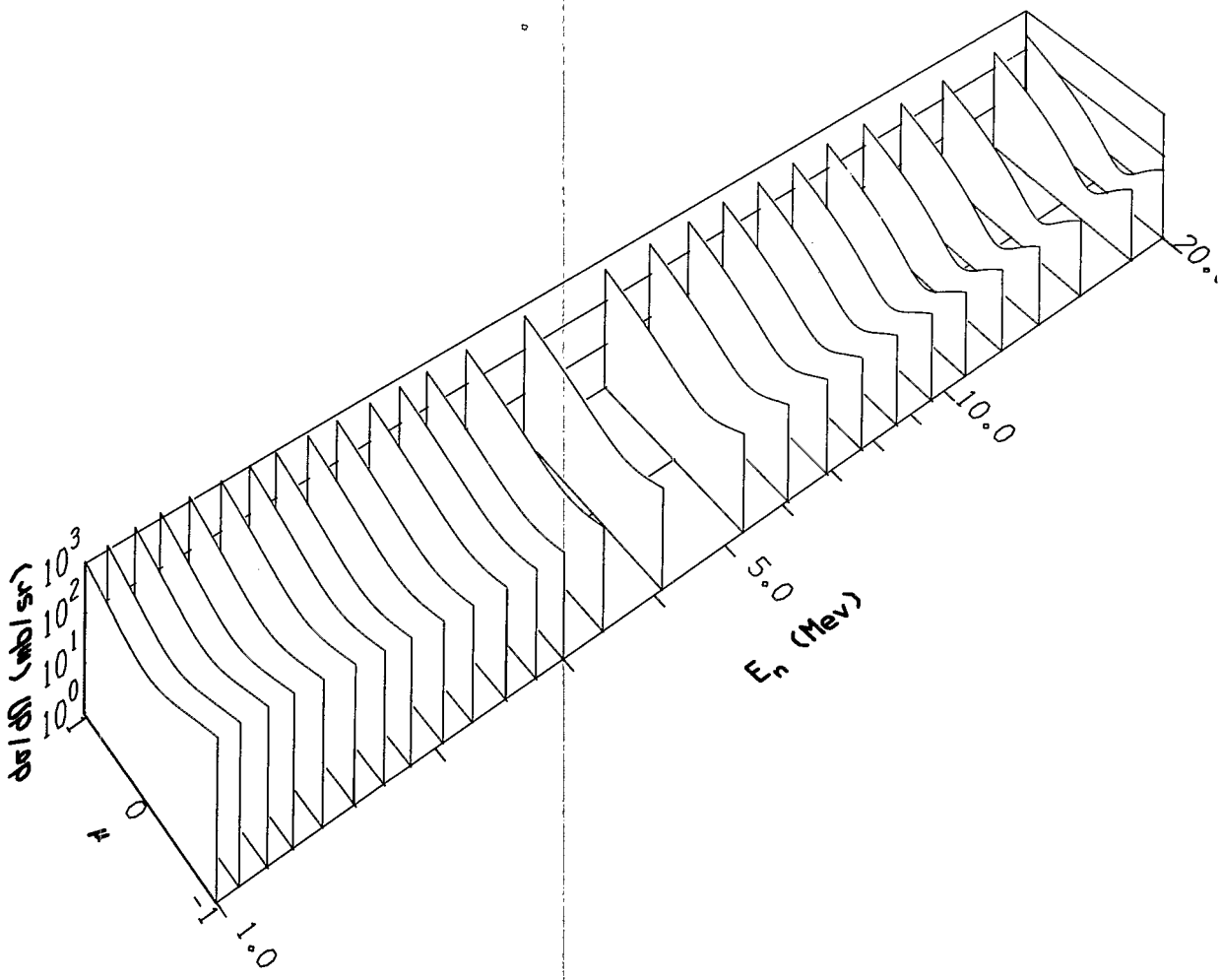
Sample Graphical Display

The following is a sample graphical display of the cross sections for Helium-4. A number of codes (see Appendix I) prepare graphical displays of materials in the ENDF format. The examples shown here are similar to those obtainable from the ENDF plotting code PLOTEF.


He-4
ENDF/B MATERIAL NO. 1270
σ_{tot} (b)
8.0
7.0
6.0
5.0
4.0
3.5
3.0
2.5
2.0
1.5
1.0
σ_{el} (b)
8.0
7.0
6.0
5.0
4.0
3.5
3.0
2.5
2.0
1.5
1.0
.5
.01
.10
1.0
10.0
100.
1.0×10^{3}
1.0×10^{4}
1.0×10^{5}
1.0×10^{6}
2.0×10^{7}
E_n (eV)
M.2
$^{4}_{2}He$

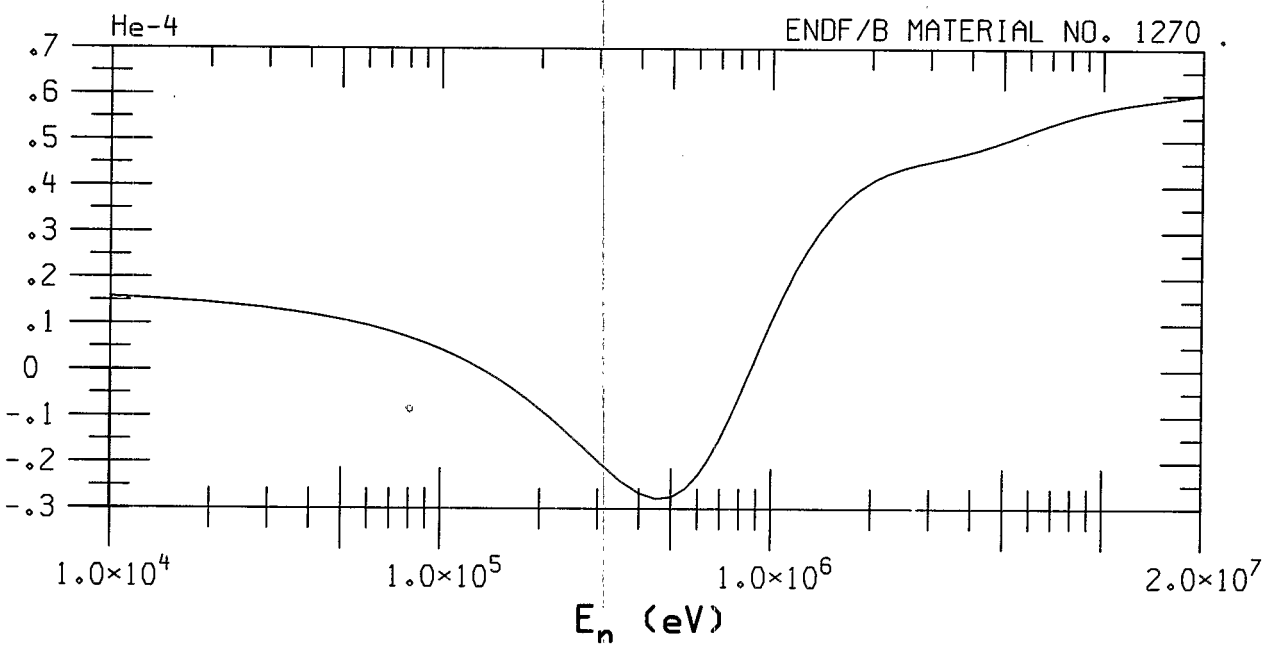
He-4

ENDF/B MATERIAL NO. 1270



He-4

ENDF/B MATERIAL NO. 1270



Appendix N

Examples of Card-Image Formats

This appendix describes examples of formats containing BCD card-image records.

The following appendix begins with a review as to how ENDF record types are organized in card-image format. The remaining pages contain the card-image examples and formats.

The top of each page indicates the particular file and data format described. Four pages (A, B, C, D) are used to describe each data type. When the appendix is opened to a particular data type, the left-hand page, A (the last page of the previous foldout sheet), contains a review of the variable names and their definitions. The right-hand page, D (before unfolding) contains the important formulae associated with the particular data type. When the right-hand page is unfolded, pages B and C are exposed. Page B gives the format. For explanation of the notation describing the ENDF record types refer to either the beginning of this appendix or Section 0.5.3. Page C gives an example of ENDF card-images described by the formats. Note that in some cases, the example would have contained too many cards to have been displayed on one page; therefore, cards have been omitted.

Examples of Record Format and Notation used in Appendix

All records on an ENDF binary tape are one of four possible types, denoted by CONT, LIST, TAB1, and TAB2. A record always consists of nine numbers followed (depending on the record type) by one or two arrays of numbers. A general description of these nine numbers is given below, but the actual definition of each number will depend on its usage.

MAT is the material number (integer).

MF is the file number (integer).

MT is the reaction type number (integer).

C1 is a constant (floating point).

C2 is a constant (floating point).

L1 is an integer generally used as a test.

L2 is an integer generally used as a test.

N1 is a count of items in a list to follow.

N2 is generally a count of items in a second list to follow.

Card-Image (BCD) Formats

A standard 80-column card is divided into the following ten fields:

Field	Columns	Description
1	1-11	Datum
2	12-22	"
3	23-33	"
4	34-44	"
5	45-55	"
6	56-66	"
7	67-70	MAT
8	71-72	MF
9	73-75	MT
10	76-80	Sequence number, starting with 1 for the first card of a material

CONT Records

The smallest possible record is a control (CONT) record consisting of the nine numbers given above. For convenience, a CONT record is denoted by

(MAT, MF, MT/C1, C2; L1, L2; N1, N2)CONT

There are five special cases of a CONT record, denoted by HEAD, SEND, FEND, MEND, and TEND. The HEAD record is the first in a section and has the same form as a CONT record. The numbers C1 and C2 are interpreted as ZA and AWR, respectively, on a HEAD record.

The SEND, FEND, MEND, and TEND records use only the first three numbers in the CONT record, and they are used to signal the end of a section, file, material, and tape, respectively:

(MAT, MF, 0/0.0, 0.0; 0, 0; 0, 0)SEND

(MAT, 0 , 0/0.0, 0.0; 0, 0; 0, 0)FEND

(0 , 0 , 0/0.0, 0.0; 0, 0; 0, 0)MEND

(-1 , 0 , 0/0.0, 0.0; 0, 0; 0, 0)TEND

The HEAD record consists of one card punched in Fields 1-9. The SEND, FEND, MEND, TEND, and TPID records each consist of one card punched in Fields 7-9 only. Note that a completely blank card (MEND record) signals the end of a material.

LIST Records

The second type of record is the LIST record, used to list a string of floating point numbers, B_1, B_2, B_3, etc. These numbers are given in an array, B(N), and there are N1 of them.

The LIST record denoted by

(MAT, MF, MT/ C1, C2; L1, L2; N1, N2/ B_n)LIST

is punched in the following way:

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	N1	N2	MAT	MF	MT
B(1)	B(2)	B(3)	B(4)	B(5)	B(6)	MAT	MF	MT
B(7)	B(8)	B(9)	----	----	----	MAT	MF	MT
----	----	----	----	----	B(N1)	MAT	MF	MT

TAB1 Records

The third type of record is the TAB1 record used for one-dimensional tabulated functions such as y(x). The data needed to specify a one-dimensional tabulated function are the interpolation tables NBT(N) and INT(N) for each of the NR ranges, and the NP tabulated pairs of X(N) and Y(N).

Consider a TAB1 binary record that was denoted by

(MAT, MF, MT/C1, C2, L1, L2; NR, NP/x_{int}/y(x))TAB1

This record would be punched on cards in the following way:

Field								
1	2	3	4	5	6	7	8	9
C1	C2	L1	L2	NR	NP	MAT	MF	MT
NBT(1)	INT(1)	NBT(2)	INT(2)	NBT(3)	INT(3)	MAT	MF	MT
NBT(4)	INT(4)	NBT(5)	INT(5)	------	------	MAT	MF	MT
------	------	------	------	NBT(NR)	INT(NR)	MAT	MF	MT
X(1)	Y(1)	X(2)	Y(2)	X(3)	Y(3)	MAT	MF	MT
X(4)	Y(4)	X(5)	Y(5)	------	------	MAT	MF	MT
------	------	------	------	X(NP)	Y(NP)	MAT	MF	MT

The term x_{int} means the interpolation table for interpolating between successive values of the variable x. y(x) means pairs of x and y(x). x is generally used as the incident neutron energy E, and y(x) is generally a parameter such as the cross section σ(E).

A TAB2 record is the same as the TAB1 record, except that the list of x and y values is omitted.

LRP is a flag that indicates that resolved and/or unresolved resonance parameters are given in File 2.

LRP = 0, no resonance parameter data given;

LRP = 1, resolved and/or unresolved resonance parameter data given in File 2.

LFI is a flag that indicates whether this material is fissionable:

LFI = 0, this is not a fissionable material;

LFI = 1, this material is fissionable.

NLIB is the library identifier:

NLIB = 0, ENDF/B.

NLIB = 1, ENDF/A.

NLIB = 2, ENDF/A (translated from UK library).

NLIB = 3, ENDF/A (translated from KEDAK).

NMOD is the modification number:

NMOD = 0, evaluation carried over from previous library version.

NMOD = 1, new or revised evaluation for the current library version.

NMOD = 2 etc., successive modification indicators.

ELIS is the excitation energy of the target nucleus relative to 0.0 for the ground state.

STA is the target stability flag. If the target is unstable radioactive decay data should be given in MF=8, MT 457:

STA = 0.0, stable nucleus,

STA = 1.0, unstable nucleus.

LIS is the state number of the target nucleus. The ground state is indicated by LIS=0.

LISØ is the isomeric state number. The ground state is indicated by LISØ=0, etc. LIS need not be the same as LISØ.

File 1 Descriptive Information and Index (MT=451)

This section always begins with a HEAD record and ends with a SEND record.

Its structure is

(MAT, 1, 451/ZA , AWR, LRP, LFI, NLIB, NMOD)HEAD

(MAT, 1, 451/ELIS, STA, LIS, LISØ, 0, 0)CØNT

(MAT, 1, 451/0.0, 0.0, 0, 0, NWD, NXC/

ZSYMA, ALAB, EDATE, AUTH(33 characters)/

REF(22 characters), DDATE, RDATE, b, ENDATE/

H(N))LIST*

(MAT, 1, 451/0.0, 0.0, MF_1, MT_1, NC_1, MOD_1)CØNT

(MAT, 1, 451/0.0, 0.0, MF_2, MT_2, NC_2, MOD_2)CØNT

(MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}, NC_{NXC}, MOD_{NXC})CØNT

(MAT, 1, 0 /0.0, 0.0, 0 , 0 , 0 , 0)SEND

*Note: ZSYMA to AUTH are part of H(N)

File 1 Descriptive Information and Index (MT = 451)

```
9.22350+ 4 2.33025+ 2          1          1          0          11395 1451   1
0.00000+ 0 1.00000+ 0          0          0          0          01395 1451   2
0.00000+ 0 0.00000+ 0          0          0        237          881395 1451   3
92-U -235 BNL        EVAL-APR77 M.R.BHAT                                1395 1451   4
                     DIST-MAY79 REV1-NOV77              790518         1395 1451   5
PRINCIPAL EVALUATOR - M.R.BHAT(BNL)                                     1395 1451   6
                                (cards deleted)
                                MF = 1                                  1395 1451  44
                                                                        1395 1451  45
MT=452(NU-BAR TOTAL) CONSISTENT WITH MT=455 AND 456                     1395 1451  46
                                                                        1395 1451  47
MT=455DELAYED NEUTRON YIELDS.KAISER AND CARPENTER(ANL) (REF 19)         1395 1451  48
                                                                        1395 1451  49
MT=456(PROMPT NEUTRONS)BASED ON CF-252NU-BAR(PROMPT)=3.757+-0.0151395 1451  50
     AND U-235 THERMALNU-BAR(PROMPT)=2.420+-0.012EVALUATION BASED1395 1451  51
     ON A LEAST-SQUARES FIT OVER A NUMBER OF ENERGY RANGES.DATA  1395 1451  52
     LISTED IN REF3 USED.                                               1395 1451  53
                                (cards deleted)
                                MF = 3                                  1395 1451  74
                                                                        1395 1451  75
THERMAL REGION(1.0E-05 TO 1.0EV) EVALUATION BY B.R.LEONARD JR ET.1395 1451  76
     AL DESCRIBED IN REF 2.   THIS FIT MODIFIED BETWEEN 0.85 TO  1395 1451  77
     1.0EV TO JOIN SMOOTHLY WITH THE RESONANCE REGION.                  1395 1451  78
     2.53E-02EV CROSS-SECTIONS   FISSION 583.54+-1.7B (REF 2)    1395 1451  79
                                 CAPTURE  98.38+-0.76B(REF 2)    1395 1451  80
     RESONANCE INTEGRALS         FISSION 281.7B                 1395 1451  81
          (0.5EV-2.0E+07EV)      CAPTURE 139.2B                 1395 1451  82
                                (cards deleted)
                                MF = 33                                 1395 1451 201
                                                                        1395 1451 202
COVARIANCE FILE BY R.W.PEELLE FROM REF18                                1395 1451 203
                                                                        1395 1451 204
                          PRINCIPAL REFERENCES                          1395 1451 205
                                                                        1395 1451 206
1 L.STEWART,H.ALTER AND R.HUNTER, ENDF-201 (1976)                       1395 1451 207
2 B.R.LEONARD JR,D.A.KOTTWITZ AND J.K.THOMPSON EPRI NP-167(1976) 1395 1451 208
3 M.R.BHAT U-235 EVALUATION FOR ENDF/B-V(ENDF-248)TO BE PUBLISHED1395 1451 209
                                (cards deleted)
31W.P.POENITZ,ANL/NDM-45(1978) TO BE PUBLISHED                          1395 1451 239
                                                                        1395 1451 240
                                1        451        328          11395 1451 241
                                1        452          7          11395 1451 242
                                1        455          8          11395 1451 243
                                1        456          6          11395 1451 244
                                1        458          5          11395 1451 245
                                2        151        973          11395 1451 246
                                3          1        470          11395 1451 247
                                3          2        470          11395 1451 248
                                (cards deleted)
                               33        102         42          11395 1451 328
0.00000+ 0 0.00000+ 0           0          0          0          01395 1  0 329
```

NWD is the count of the number of elements in the Hollerith section. For BCD card image tapes, NWD is the number of card images used to describe the data set for this material (NWD $\leq$ 294).

NXC is an integer count of all the sections to be found in the dictionary.

H(N) is the array containing the Hollerith information that describes the particular evaluated data set.

(First BCD Card Image Record of H(N))

ZSYMA is a Hollerith representation of the material Z-chemical symbol-A in cols. 1-11 (ZZZ-EL-AAAM, see Sec. 1.1).

ALAB Mnemonic of originating laboratory(s) (left adjusted) in cols. 12-22.

EDATE date of evaluation, EVAL- in cols. 23-27, three character month in 28-30, followed by two character year 31-32 (i.e. EVAL-DEC74)

AUTH author(s) of evaluation (left adjusted) cols. 34-66

(Second BCD-Card Image Record of H(N))

REF reference 2-22

DDATE original distribution date (left adjusted DIST- followed by month-year as in EDATE) in cols. 23-33.

RDATE date and number of last revision (REV1- followed by month-year as in EDATE) in cols. 34-44.

ENDATE Master file entry date (right adjusted YYMMDD) cols. 56-61.

MF_n, MT_n, NC_n and $MØD_n$ are included in each of the NXC items in the dictionary.

MF_n is the MF of the n^{th} section.

MT_n is the MT of the n^{th} section.

NC_n is the number of BCD card images in a given section (the n^{th} section). Count does not include the SEND card.

MOD_n is the modification indicator for section MF_n and MT_n.

LNU is a test that indicates what representation of $\bar{\nu}(E)$ has been used:

LNU = 1, polynomial representation has been used;

LNU = 2, tabulated representation.

NC is a count of the number of terms used in the polynomial expansion. (NC $\leq$ 4)

C_n are the coefficients of the polynomial. There are NC coefficients given.

NR is the number of interpolation ranges used to tabulate values of $\bar{\nu}(E)$. (See Appendix E.)

NP is the total number of energy points used to tabulate $\bar{\nu}(E)$.

E_{int} is the interpolation scheme (see Appendix E for details.)

$\bar{\nu}(E)$ is the average number of neutrons per fission.

The structure of this section depends on whether values of $\bar{\nu}(E)$ are tabulated as a function of incident neutron energy or whether $\bar{\nu}$ is represented by a polynomial.

If LNU = 1, the structure of the section is

(MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 452/ 0.0, 0.0, 0, 0 , NC, 0/C_1, C_2, ... C_{NC})LIST

(MAT, 1, 0 / 0.0, 0.0, 0, 0 , 0, 0)SEND

If LNU = 2, the structure of the section is

(MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 452/ 0.0, 0.0, 0, 0, NR, NP/E_{int}/$\bar{\nu}(E)$)TABl

(MAT, 1, 0/ 0.0, 0.0, 0, 0, 0, 0)SEND


```
9.22340+ 4 2.32030+ 2          0          1          0 01394 1452  1
0.00000+ 0 0.00000+ 0          0          0          2 01394 1452  2
2.35200+ 0 1.35000- 7                                  1394 1452  3
0.00000+ 0 0.00000+ 0          0          0          0 01394 1  0  4
```

```
9.22350+ 4 2.33025+ 2          0          2          0 01395 1452  1
0.00000+ 0 0.00000+ 0          0          0          1 121395 1452 2
        12         2                                   1395 1452  3
1.00000- 5 2.43670+ 0 2.53000- 2 2.43670+ 0 2.50000+ 4 2.43670+ 01395 1452  4
1.20000+ 6 2.55070+ 0 1.50000+ 6 2.57970+ 0 2.00000+ 6 2.62770+ 01395 1452  5
4.00000+ 6 2.91170+ 0 5.50000+ 6 3.12085+ 0 6.00000+ 6 3.26460+ 01395 1452  6
7.00000+ 6 3.40400+ 0 7.85000+ 6 3.52500+ 0 2.00000+ 7 5.25100+ 01395 1452  7
0.00000+ 0 0.00000+ 0          0          0          0          01395 1  0  8
```

The energy dependence of $\bar{\nu}$ may be found by tabulating $\bar{\nu}$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}(E)$,

$$\bar{\nu}(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}(E)$ is the average total (prompt plus delayed) number of neutrons per fissions produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

LNU is a test that indicates which representation is used:
LNU = 1 means that a polynomial expansion is used;
LNU = 2 means that a tabulated representation is used.

NC is the number of terms in the polynomial expansion. ($NC \leq 4$)

C_n are the coefficients for the polynomial.

NR is the number of interpolation ranges used. ($NR \leq 20$)

NP is the total number of incident energy points used to represent $\bar{\nu}_d(E)$ when a tabulation is used.

E_{int} is the interpolation scheme (see Appendix E).

$\bar{\nu}_d(E)$ is the total average number of delayed neutron precursors formed per fission event.

NNF is the number of precursor families considered.

λ_i is the decay constant (sec^{-1}) for the i^{th} precursor.

The structure of a section when a polynomial representation has been used (LNU = 1) is

(MAT, 1, 455/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NNF, 0/λ_1, λ_2,...λ_{NNF})LIST

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NC, 0/C_1, C_2,...C_{NC})LIST

(MAT, 1, 0 / 0.0, 0.0, 0, 0, 0, 0)SEND

The structure values of $\bar{\nu}_d$ are tabulated (LNU = 2) is

(MAT, 1, 455/ ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NNF, 0/λ_1, λ_2,...λ_{NNF})LIST

(MAT, 1, 455/ 0.0, 0.0, 0, 0, NR, NP/E_{int}/$\bar{\nu}_d$(E))TAB1

(MAT, 1, 0 / 0.0, 0.0, 0, 0, 0 0)SEND

NO EXAMPLE IN ENDF/B-V

The total number of delayed neutron precursors emitted per fission event, at incident energy E, is given in this file and is defined as the sum of the number of precursors emitted for each of the precursor families,

$$\bar{\nu}_d(E) = \sum_{i=1}^{NNF} \bar{\nu}_i(E) ,$$

where NNF is the number of precursor families. The fraction of the total, $P_i(E)$, emitted for each family is given in File 5 (see section 5) and is defined as

$$P_i(E) = \frac{\bar{\nu}_i(E)}{\bar{\nu}_d(E)}$$

The structure of a section depends on whether $\bar{\nu}_d(E)$ is tabulated as a function of incident energy or given as coefficients of a polynomial expansion in energy. If a polynomial is used, $\bar{\nu}_d(E)$ is defined as

$$\bar{\nu}_d(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}(E)$ is the average number of delayed neutrons per fission produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

LNU is a test that indicates what representation of $\bar{\nu}(E)$ has been used;

LNU = 1, polynomial representation has been used;

LNU = 2, tabulated representation.

NC is a count of the number of terms used in the polynomial expansion. (NC $\leq$ 4)

C_n are the coefficients of the polynomial. There are NC coefficients given.

NR is the number of interpolation ranges used to tabulate values of $\bar{\nu}_p(E)$. (See Appendix E.)

NP is the total number of energy points used to tabulate $\bar{\nu}(E)$.

E_{int} is the interpolation scheme (see Appendix E.)

$\bar{\nu}_p(E)$ is the average number of prompt neutrons per fission.

The structure of this section depends on whether values of $\bar{\nu}(E)$ are represented as a polynomial or tabulated as a function of incident neutron energy.

If LNU = 1, the structure of the section is

(MAT, 1, 456/ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 1

(MAT, 1, 456/0.0, 0.0, 0, 0, NC, 0/C_1, C_2, ... C_{NC})LIST

(MAT, 1 0/ 0.0, 0.0, 0, 0, 0, 0)SEND

If LNU = 2, the structure of the section is

(MAT, 1, 456/ZA, AWR, 0, LNU, 0, 0)HEAD LNU = 2

(MAT, 1, 456/0.0, 0.0, 0, 0, NR, NP/$E_{int}/\bar{\nu}_p(E)$)TAB1

(MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0)SEND

NO EXAMPLE IN ENDF/B-V

```
 9.22350+ 4 2.33025+ 2          0          2          0          01395 1456  1
 0.00000+ 0 0.00000+ 0          0          0          1          81395 1456  2
          8          2                                                1395 1456  3
 1.00000- 5 2.42000+ 0 2.53000- 2 2.42000+ 0 2.50000+ 4 2.42000+ 01395 1456  4
 1.50000+ 6 2.56300+ 0 2.00000+ 6 2.61100+ 0 5.50000+ 6 3.10800+ 01395 1456  5
 6.00000+ 6 3.25300+ 0 2.00000+ 7 5.24200+ 0                      1395 1456  6
 0.00000+ 0 0.00000+ 0          0          0          0          01395 1  0  7
```

The energy dependence of $\bar{\nu}_p$ may be given by tabulating $\bar{\nu}_p$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}_p(E)$.

$$\bar{\nu}_p(E) = \sum_{n=1}^{NC} C_n E^{(n-1)}$$

where $\bar{\nu}_p(E)$ is the average number of prompt neutrons per fission produced by neutrons of incident energy E(eV), C_n is the n^{th} coefficient, and NC is the number of terms in the polynomial.

ET is the sum of all the partial energies which follow: that is, the total energy release per fission and equals the Q value

EFR is the kinetic energy of the fragments

ENP is the kinetic energy of the "prompt" fission neutrons

END is the kinetic energy of the delayed fission neutrons

EGP is the total energy released by the emission of "prompt" γ rays

EGD is the total energy released by the emission of delayed γ rays

EB is the total energy released by delayed β's

ENU is the energy carried away by the neutrinos

ER is ET - ENU (the total energy less the energy of the neutrinos). This ER is equal to the pseudo-Q in File 3 for MT = 18.

The structure of a section is:

(MAT, 1, 458/ ZA, AWR, 0, 0, 0, 0) HEAD

(MAT, 1, 458/ 0.0, 0.0, 0, 0, 18, 9/

EFR, ΔEFR, ENP, ΔENP, END, ΔEND

EGP, ΔEGP, EGD, ΔEGD, EB, ΔEB

ENU, ΔENU, ER, ΔER, ET, ΔET) LIST

(MAT, 1, 0/ 0.0, 0.0, 0, 0, 0, 0) SEND,

where the Δ's allow the error estimates on the quantities listed above.


```
9.22350+ 4 2.33025+ 2          0          0          0          01395 1458  1
0.00000+ 0 0.00000+ 0          0          0         18          91395 1458  2
1.69580+ 8 6.80000+ 5 4.79000+ 6 7.00000+ 4 7.10000+ 3 7.10000+ 21395 1458  3
6.96000+ 6 7.00000+ 5 6.26000+ 6 5.00000+ 4 6.43000+ 6 5.00000+ 41395 1458  4
8.68000+ 6 6.00000+ 4 1.94020+ 8 1.20000+ 5 2.02700+ 8 1.00000+ 51395 1458  5
0.00000+ 0 0.00000+ 0          0          0          0          01395 1  0  6
```

File 1 Energy Release Due to Fission (MT=458)

File 2 Resonance Parameter Data (General Structure)

Several quantities used in File 2 have definitions that are the same for all resonance parameter representations:

NIS is the number of isotopes in this material (NIS $\leq$ 10).

ZAI is the (Z,A) designation for an isotope.

ABN is the abundance of an isotope in the material.

LFW is a flag indicating whether average fission widths are given in the unresolved resonance region for this isotope:

LFW = 0, average fission widths are not given;

LFW = 1, average fission widths are given.

NER is the number of energy ranges for this isotope (NER $\leq$ 2).

EL is the lower limit for an energy range.

EH is the upper limit for an energy range.

LRU is a flag indicating whether this energy range contains data for resolved or unresolved resonance parameters:

LRU = 0, only the scattering radius is given;

LRU = 1, resolved resonance parameters are given;

LRU = 2, unresolved resonance parameters are given.

LRF is a flag indicating which representation has been used for the energy range. The definition of LRF depends on the value of LRU:

If LRU = 1 (resolved parameters), then

LRF = 1, single-level Breit-Wigner (SLBW) parameters;

LRF = 2, multilevel Breit-Wigner (MLBW) parameters;

LRF = 4, Adler-Adler parameters;

If LRU = 2 (unresolved parameters), then

LRF = 1, only average fission widths are energy-dependent;

LRF = 2, average level spacing, competitive reaction widths, reduced neutron widths, radiation widths, and fission widths are energy-dependent.

The general structure of a section is as follows:

(MAT, 2, 151/ ZA, AWR, 0, 0, NIS, 0)HEAD

(MAT, 2, 151/ ZAI, ABN, 0, LFW, NER, 0)CONT (isotope)

(MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0)CONT (range)

<Subsection for the first energy range for the first isotope (depends on LRU and LRF)>

(MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0)CONT (range)

<Subsection for the second energy range for the first isotope (depends on LRU and LRF)>

(MAT, 2, 151/ EL, EH, LRU, LRF, 0 0)CONT (range)

<Subsection for the last energy range for the last isotope for this material>

(MAT, 2, 0 / 0.0, 0.0, 0, 0, 0, 0)SEND

The data are given for all ranges for a given isotope, and then for all isotopes. The data for each range start with a CONT (range) record; those for each isotope, with a CONT (isotope) record.


```
9.22350+ 4 2.33025+ 2          0          0          1          01395 2151   1
9.22350+ 4 1.00000+ 0          0          1          2          01395 2151   2
1.00000+ 0 8.20000+ 1          1          1          0          01395 2151   3
                              (subsection for the first energy range)
8.20000+ 1 2.50000+ 4          2          2          0          01395 2151 136
                              (subsection for the second energy range)
0.00000+ 0 0.00000+ 0          0          0          0          01395 2   0 974
```

FILE 2

Resonance Parameter Data

General Structure

$$\sum_{i=1}^{NIS} ABN_i = 1.000$$

The following quantities are defined:

SPI is the nuclear spin of the target nucleus, I (positive number).

AP is the scattering radius in units of 10^{-12} cm.

NLS is the number of sets of resonance parameters given in this energy region. NLS=0 for LRP=0.

The structure of File 2 for the special case, in which just the scattering radius is specified, is given below (no resolved or unresolved parameters are given for this material):

```
(MAT, 2, 151/ ZA,  AWR, 0,   0,   NIS, 0)HEAD    (NIS = 1)
(MAT, 2, 151/ ZAI, ABN, 0,   LFW, NER, 0)CØNT    (LFW = 0, NER = 1)
(MAT, 2, 151/ EL,  EH,  LRU, LRF, 0,   0)CØNT    (LRU = 0, LRF = 0)
(MAT, 2, 151/ SPI, AP,  0,   0,   NLS, 0)CØNT    (NLS = 0)
(MAT, 2,  0 / 0.0, 0.0, 0,   0,   0,   0)SEND
(MAT, 0,  0 / 0.0, 0.0, 0,   0,   0,   0)FEND
```



```
8.01600+ 3 1.58580+ 1   0   0   1   01276 2151  1
8.01600+ 3 1.00000+ 0   0   0   1   01276 2151  2
1.00000- 5 2.00000+ 7   0   0   0   01276 2151  3
0.00000+ 0 5.46140- 1   0   0   0   01276 2151  4
0.00000+ 0 0.00000+ 0   0   0   0   01276 2  0  5
```

File 2 Resonance Parameter Data (Special Case, LRP=0)

FILE 2

Resonance Parameter Data

Special Case
LRP = 0 (In File 1, MT = 451)

The only data given is the effective scattering radius. The s-wave potential scattering cross section is

$$\sigma_p = \frac{4\pi}{k^2} \sin^2 \theta$$

where

$$k = 2.196771 \frac{AWRI}{AWRI + 1.0} \times 10^{-3} \sqrt{E}$$

E in electron volts.

and

$$\theta = k*AP$$

File 2 Resolved Resonance Parameter Data (Single or Multilevel Breit-Wigner)

The following quantities are defined for use when LRF = 1 and 2 (see Appendix D for formulae):

Resolved Resonance Parameters if LRF = 1 (SLBW) and LRF = 2 (MLBW)

SPI is the nuclear spin of the target nucleus, I (positive number).

AP is the scattering radius in units of 10^{-12} cm.

NLS is the number of sets of resonance parameters given in this energy region. A set of parameters is given for each ℓ-value (neutron orbital angular momentum). (NLS $\leq$ 3.)

AWRI is the ratio of the mass of a particular isotope to that of a neutron.

QX is an effective Q-value to be added to the incident particle's center-of-mass energy to determine the channel energy for use in the penetrability factor of the competitive width. QX = 0.0 if LRX = 0.

L is the value of ℓ.

LRX is a flag indicating whether this energy range contains a competitive width:

LRX = 0, no competitive width is given;

LRX = 1, a competitive width is given;

NRS is the number of resolved resonances for a given ℓ-value. (NRS $\leq$ 500.)

ER is the resonance energy (in the laboratory system).

AJ is the floating point value of J (the spin, or total angular momentum of the resonance).

GT is the resonance total width Γ evaluated at the resonance energy ER.

GN is the neutron width Γ_n evaluated at the resonance energy ER.

GG is the radiation width Γ_γ evaluated at the resonance energy ER.

GF is the fission width Γ_f evaluated at the resonance energy ER.

GX is the competitive width Γ_x evaluated at the resonance energy ER. It is not given explicitly, but is to be obtained by subtraction, GX = GT - (GN + GG + GF), if LRX $\neq$ 0;

File 2 Resolved Resonance Parameter Data (Single or Multilevel Breit-Wigner)

The structure of a subsection containing data for (LRU = 1 and LRF = 1) or (LRU = 1 and LRF = 2) is

(MAT, 2, 151/ SPI, AP, 0, 0, NLS, 0)CONT

(MAT, 2, 151/ AWRI, QX, L, LRX, 6*NRS, NRS/

ER_1, AJ_1, GT_1, GN_1, GG_1, GF_1,

ER_2, AJ_2, GT_2, GN_2, GG_2, GF_2,

ER_{NRS}, AJ_{NRS}, GT_{NRS}, GN_{NRS}, GG_{NRS}, GF_{NRS})LIST

The LIST record is repeated until each NLS ℓ-values has been specified (in order of increasing value of ℓ). The values of ER for each ℓ-state shall be ordered by increasing neutron energy.

File 2 Resolved Resonance Parameter Data (Single or Multilevel Breit-Wigner)

```
 3.50000+ 0 9.56630- 1          0          0          1          01395 2151   4
 2.33025+ 2 0.00000+ 0          0          0        780        1301395 2151   5
-1.49000+ 0 3.50000+ 0 2.37680- 1 3.68200- 3 2.70000- 2 2.07000- 11395 2151   6
 2.90000- 1 3.50000+ 0 1.35000- 1 3.01570- 6 3.60000- 2 9.90000- 21395 2151   7
 1.14000+ 0 3.50000+ 0 1.50820- 1 1.51610- 5 3.46000- 2 1.16200- 11395 2151   8
 2.03500+ 0 3.50000+ 0 4.46960- 2 7.66050- 6 3.48740- 2 9.81400- 31395 2151   9
 2.92000+ 0 3.50000+ 0 2.20000- 1 4.85300- 6 2.00000- 2 2.00000- 11395 2151  10
 3.14700+ 0 3.50000+ 0 1.39610- 1 2.24050- 5 3.32100- 2 1.06370- 11395 2151  11
 3.60900+ 0 3.50000+ 0 8.43790- 2 4.55940- 5 3.36960- 2 5.06370- 21395 2151  12
 4.84800+ 0 3.50000+ 0 3.95920- 2 6.03520- 5 3.59450- 2 3.58700- 31395 2151  13
 5.44800+ 0 3.50000+ 0 9.01200- 2 3.36110- 6 6.00000- 2 3.01170- 21395 2151  14
 5.60000+ 0 3.50000+ 0 6.41920- 1 3.33190- 5 2.00000- 2 6.21890- 11395 2151  15
 6.21000+ 0 3.50000+ 0 2.30900- 1 6.37950- 5 4.34690- 2 1.87360- 11395 2151  16
 6.38200+ 0 3.50000+ 0 4.47880- 2 2.68340- 4 3.49720- 2 9.54800- 31395 2151  17
 7.07700+ 0 3.50000+ 0 6.39340- 2 1.26600- 4 3.55740- 2 2.82330- 21395 2151  18
 8.78100+ 0 3.50000+ 0 1.23290- 1 1.12340- 3 3.11700- 2 9.10000- 21395 2151  19
 9.28600+ 0 3.50000+ 0 1.10760- 1 1.63640- 4 3.56000- 2 7.50000- 21395 2151  20
                                (cards deleted)
 7.04040+ 1 3.50000+ 0 1.72720- 1 2.71560- 3 5.00000- 2 1.20000- 11395 2151 120
 7.07500+ 1 3.50000+ 0 2.37410- 1 2.40910- 3 3.50000- 2 2.00000- 11395 2151 121
 7.16100+ 1 3.50000+ 0 1.60290- 1 2.91360- 4 4.00000- 2 1.20000- 11395 2151 122
 7.23900+ 1 3.50000+ 0 1.38610- 1 2.61150- 3 3.10000- 2 1.05000- 11395 2151 123
 7.29100+ 1 3.50000+ 0 3.60370- 1 3.67170- 4 4.00000- 2 3.20000- 11395 2151 124
 7.45440+ 1 3.50000+ 0 1.01670- 1 2.72870- 3 3.80000- 2 6.09370- 21395 2151 125
 7.51700+ 1 3.50000+ 0 2.90890- 1 8.88330- 4 5.00000- 2 2.40000- 11395 2151 126
 7.55410+ 1 3.50000+ 0 2.33360- 1 1.36210- 3 3.20000- 2 2.00000- 11395 2151 127
 7.67500+ 1 3.50000+ 0 1.16110- 1 1.07320- 4 3.60000- 2 8.00000- 21395 2151 128
 7.74920+ 1 3.50000+ 0 1.12990- 1 9.86810- 4 4.00000- 2 7.20000- 21395 2151 129
 7.81170+ 1 3.50000+ 0 1.48220- 1 1.22450- 3 4.70000- 2 1.00000- 11395 2151 130
 7.96720+ 1 3.50000+ 0 1.29790- 1 7.85570- 4 4.40000- 2 8.50000- 21395 2151 131
 8.03570+ 1 3.50000+ 0 1.74840- 1 8.38510- 4 4.00000- 2 1.34000- 11395 2151 132
 8.14340+ 1 3.50000+ 0 1.32040- 1 1.04330- 3 4.10000- 2 9.00000- 21395 2151 133
 8.35900+ 1 3.50000+ 0 1.18270- 1 1.17030- 3 4.80000- 2 6.91000- 21395 2151 134
 8.68800+ 1 3.50000+ 0 8.01200- 2 7.19580- 4 5.20000- 2 2.74000- 21395 2151 135
```

File 2 Resolved Resonance Parameter Data (Single or Multilevel Breit-Wigner)

FILE 2

Resonance Parameter Data

LRU = 1, resolved parameters

LRF = 1 or 2, single or multilevel Breit-Wigner parameters

$$g_i = \frac{2*AJ_i + 1.0}{2(2*SPI + 1.0)}$$

$$GT_i = GN_i + GG_i + GF_i \text{ , if LRX=0}$$

File 2 Resolved Resonance Parameter Data (Adler-Adler Parameters)

Resolved Resonance Parameters if LRF = 4 (Adler-Adler multilevel parameters)

SPI is the spin of the target nucleus, I.
AP is the scattering radius in units of 10^{-12} cm.
NLS is the number of ℓ-values for which parameters will be given (NLS $\leq$ 3).
AWRI is the ratio of the mass of a particular isotope to that of the neutron.
LI is a flag to indicate the kind of parameters given:
LI = 5, total and capture widths
= 7, total, fission, and capture widths.
NX is the count of the number of sets of background constants to be given.
If NX = 2, background constants are given for the total and capture cross sections.
= 3, background constants are given for the total, capture, and fission cross sections.
$AT_1, AT_2, AT_3, AT_4, BT_1, BT_2$ are the background constants for the total cross section.
$AF_1, AF_2, AF_3, AF_4, BF_1, BF_2$ are the background constants for the fission cross section.
$AC_1, AC_2, AC_3, AC_4, BC_1, BC_2$ are the background constants for the radiative capture cross section.
AJ is the floating-point value of J (the spin of the resonance).
L is the value of ℓ.
NJS is the number of sets of resolved resonance parameters (each set having its own J-value) for a specified ℓ.
NLJ is the number of resonances for which parameters are given, for a specified AJ and L.
DET_r is the resonance energy, (μ), for the r^{th} total cross section. Here and below, the subscript r denotes the r^{th} resonance.
DEF_r is the resonance energy, (μ), for the fission cross section.
DEC_r is the resonance energy, (μ), for the radiative capture cross section.
DWT_r is the value of $\Gamma/2$, (ν), used for the total cross section.
DWF_r is the value of $\Gamma/2$, (ν), used for the fission cross section.
DWC_r is the value of $\Gamma/2$, (ν), used for the radiative capture cross section.
GRT_r is the symmetrical total cross section parameter, G_r^T.
GIT_r is the asymmetrical total cross section parameter, H_r^T.
GRF_r is the symmetrical fission parameter, G_r^f.
GIF_r is the asymmetrical fission parameter, H_r^f.
GRC_r is the symmetrical capture parameter, G_r^γ.
GIC_r is the asymmetrical capture parameter, H_r^γ.

File 2 Resolved Resonance Parameter Data (Adler-Adler Parameters)

The structure of a subsection containing data for (LRU = 1 and LRF = 4) depends on the value of NX (the number of sets of background constants). For the most general case (NX = 3) the structure is

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0)CONT)

(MAT, 2, 151/AWRI, 0.0, LI, 0, 6*NX, NX/

AT_1, AT_2, AT_3, AT_4, BT_1, BT_2

AF_1, ----------, BF_2

AC_1, ----------, BC_2)LIST)

(MAT, 2, 151/0.0, 0.0, L, 0, NJS, 0)CONT(ℓ)

(MAT, 2, 151/AJ, 0.0, 0, 0, 12*NLJ, NLJ/

DET_1, DWT_1, GRT_1, GIT_1, DEF_1, DWF_1,

GRF_1, GIF_1, DEC_1, DWC_1, GRC_1, GIC_1,

DET_2, DWT_2, ----------

----------, GIC_2,

DET_3, ----------

----------, GIC_{NLJ})LIST

The last LIST record is repeated for each J-state (there will be NJS such LIST records). A new CONT (ℓ) record will be given which will be followed by NJS LIST records. Note that if NX = 2 then the quantities AF_1,----, BF_2 will not be given in the first LIST record. Also, if LI ≠ 7 then certain of the parameters for each level may be set at zero, i.e., the fields for parameters not given (depending on LI) will be set to zero.

File 2 Resolved Resonance Parameter Data (Adler-Adler Parameters)

```
 2.50000+ 0 9.89300- 1          0          0          1          01393 2151   4
 2.31043+ 2 0.00000+ 0          7          0         18          31393 2151   5
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01393 2151   6
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01393 2151   7
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01393 2151   8
 0.00000+ 0 0.00000+ 0          0          0          1          01393 2151   9
 2.50000+ 0 0.00000+ 0          0          0        996          831393 2151  10
-1.34000+ 0 2.69400+ 0 2.11380- 4 2.94070- 5-1.34000+ 0 2.69400+ 01393 2151  11
 2.09850- 4 2.91180- 5-1.34000+ 0 2.69400+ 0 1.51940- 6 8.58300- 81393 2151  12
-1.16000+ 0 5.12000- 2 9.19060- 6-9.27590- 6-1.16000+ 0 5.12000- 21393 2151  13
 4.06920- 6-9.16170- 6-1.16000+ 0 5.12000- 2 5.11230- 6-1.24010- 71393 2151  14
-4.76100- 1 6.30000- 1 2.74420- 5 1.05920- 6-4.76100- 1 6.30000- 11393 2151  15
 1.96450- 5 2.05130- 6-4.76100- 1 6.30000- 1 7.79570- 6-1.03310- 61393 2151  16
-4.01800- 2 1.27600- 1-1.07010- 6-4.24730- 9-4.01800- 2 1.27600- 11393 2151  17
-7.11490- 7-5.66820- 7-4.01800- 2 1.27600- 1-3.58640- 7 5.64080- 71393 2151  18
 1.65900- 1 5.67600- 2-2.57060- 7-6.36420- 7 1.65900- 1 5.67600- 21393 2151  19
-4.63260- 7-6.34890- 7 1.65900- 1 5.67600- 2 2.05440- 7-1.11190- 91393 2151  20
                              (cards deleted)
 1.76200- 4-4.68140- 5 5.02900+ 1 6.97600- 1 4.58080- 6-4.46360- 71393 2151 150
 5.13000+ 1 1.23500- 1 7.09670- 6 8.30890- 6 5.13000+ 1 1.23500- 11393 2151 151
 5.24180- 6 8.17330- 6 5.13000+ 1 1.23500- 1 1.84570- 6 1.26830- 71393 2151 152
 5.20200+ 1 3.83500- 1 4.89300- 5 2.51580- 6 5.20200+ 1 3.83500- 11393 2151 153
 4.63410- 5 2.51580- 6 5.20200+ 1 3.83500- 1 2.55240- 6 1.60510- 91393 2151 154
 5.28100+ 1 1.22600- 1-1.45180- 5-4.59150- 5 5.28100+ 1 1.22600- 11393 2151 155
-2.14730- 5-3.95190- 5 5.28100+ 1 1.22600- 1 6.81950- 6-6.19210- 61393 2151 156
 5.28100+ 1 4.42500- 1 1.77570- 4-1.20810- 4 5.28100+ 1 4.42500- 11393 2151 157
 1.66910- 4-1.26140- 4 5.28100+ 1 4.42500- 1 9.97190- 6 5.27830- 61393 2151 158
 5.39800+ 1 1.30300- 1 1.12190- 4-3.24380- 5 5.39800+ 1 1.30300- 11393 2151 159
 9.35050- 5-3.22400- 5 5.39800+ 1 1.30300- 1 1.83200- 5-2.01480- 71393 2151 160
 5.47500+ 1 9.44700- 2 1.33980- 4 1.27710- 6 5.47500+ 1 9.44700- 21393 2151 161
 1.02630- 4 1.34620- 6 5.47500+ 1 9.44700- 2 3.01680- 5-1.92190- 71393 2151 162
 5.60100+ 1 2.28800- 1 1.40000- 4 1.20770- 4 5.60100+ 1 2.28800- 11393 2151 163
 1.23700- 4 1.19120- 4 5.60100+ 1 2.28800- 1 1.53660- 5 1.24790- 61393 2151 164
 5.63900+ 1 2.39900- 1 2.95440- 4 8.40940- 6 5.63900+ 1 2.39900- 11393 2151 165
 2.67270- 4 8.66520- 6 5.63900+ 1 2.39900- 1 2.69250- 5-1.34330- 71393 2151 166
 5.74100+ 1 5.66900- 1 5.73670- 4-2.29500- 5 5.74100+ 1 5.66900- 11393 2151 167
 5.50070- 4-2.10160- 5 5.74100+ 1 5.66900- 1 2.06200- 5-5.91490- 71393 2151 168
 5.85000+ 1 2.12800- 1 1.37180- 4 2.12290- 5 5.85000+ 1 2.12800- 11393 2151 169
 1.22380- 4 2.12750- 5 5.85000+ 1 2.12800- 1 1.44350- 5 1.33340- 71393 2151 170
 5.99000+ 1 3.29700- 1-2.63920- 7-1.07550- 5 5.99000+ 1 3.29700- 11393 2151 171
-7.15000- 7-1.07930- 5 5.99000+ 1 3.29700- 1 4.64630- 7 4.07980- 81393 2151 172
 6.13700+ 1 2.85300- 1 2.89810- 4 5.86310- 5 6.13700+ 1 2.85300- 11393 2151 173
 2.66660- 4 5.93620- 5 6.13700+ 1 2.85300- 1 2.14120- 5 8.51290- 81393 2151 174
 6.26900+ 1 7.99500- 2 2.22650- 4 1.22590- 5 6.26900+ 1 7.99500- 21393 2151 175
 1.64320- 4 1.25060- 5 6.26900+ 1 7.99500- 2 5.62400- 5 8.33020- 91393 2151 176
```

File 2 Resolved Resonance Parameter Data (Adler-Adler Parameters)

The background correction for the total cross section is calculated by using the six constants in the following manner:

$$\sigma_T \text{ (background)} = \frac{C}{\sqrt{E}} (AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1*E + BT_2*E^2)$$

where $C = \pi\lambda^2 = \pi/k^2$ and $k = 2.196771 \times 10^{-3} \left(\frac{AWRI}{AWRI + 1.0}\right)\sqrt{E(eV)}$

is the center-of-mass wave number in terms of the laboratory energy.

The background terms for the fission and radiative capture cross sections are calculated in a similar manner.

Since the format has no provision for giving the Adler-Adler parameters for the scattering cross-section, this is obtained by subtracting the sum of capture and fission cross sections from the total cross section.

File 2 Unresolved Resonance Parameter Data

(Parameters are Energy Independent)

(Fission Widths not given)

The following quantities are defined for use in specifying unresolved resonance parameters (LRU = 2, LRF = 1, LFW = 0):

SPI is the spin of the target nucleus, I.

AP is the scattering radius in units of 10^{-12} cm. No channel quantum number dependence is currently permitted by the format.

NLS is the number of ℓ-values (NLS $\leq$ 3.)

L is the value of ℓ.

AWRI is the ratio of the mass of a particular isotope to that of the neutron.

NJS is the number of J-states for a particular ℓ-state. (NJS $\leq$ 6.)

AJ is the floating-point value of the J (the spin, or total angular momentum of the set of parameters).

D is the average level spacing for resonances with spin J.

AMUN is the number of degrees of freedom in the neutron width distribution. (1.0 $\leq$ AMUN $\leq$ 2.0)

GNO is the average reduced neutron width.

GG is the average radiation width.

File 2 Unresolved Resonance Parameter Data

(Parameters are Energy Independent)

(Fission Widths not given)

If LFW = 0 (fission widths not given),

LRU = 2 (unresolved parameters),

LRF = 1 (all parameters are energy-independent).

The structure of a subsection is:

(MAT, 2, 151/SPI,	AP,	0,	0,	NLS,	0)CONT
(MAT, 2, 151/AWRI,	0.0,	L,	0,	6*NJS,	NJS/
D_1,	AJ_1,	$AMUN_1$,	GNO_1,	GG_1,	0.0
D_2,	AJ_2,	$AMUN_2$,	GNO_2,	GG_2,	0.0

D_{NJS},	AJ_{NJS},	$AMUN_{NJS}$,	GNO_{NJS},	GG_{NJS},	0.0)LIST

The LIST record is repeated until data for all ℓ-values have been specified.

File 2 Unresolved Resonance Parameter Data

(Parameters are Energy Independent)

(Fission Widths not given)

```
4.50000+ 0 7.02400- 1          0          0          3          01189 2151  226
9.21051+ 1 0.00000+ 0          0          0         12          21189 2151  227
1.55100+ 2 4.00000+ 0 1.00000+ 0 1.93400- 3 2.00000- 1 0.00000+ 01189 2151  228
1.26900+ 2 5.00000+ 0 1.00000+ 0 2.36400- 3 2.00000- 1 0.00000+ 01189 2151  229
9.21051+ 1 0.00000+ 0          1          0         24          41189 2151  230
3.49600+ 2 3.00000+ 0 1.00000+ 0 1.14400- 2 2.40000- 1 0.00000+ 01189 2151  231
2.71900+ 2 4.00000+ 0 2.00000+ 0 1.47100- 2 2.40000- 1 0.00000+ 01189 2151  232
2.22500+ 2 5.00000+ 0 2.00000+ 0 1.79800- 2 2.40000- 1 0.00000+ 01189 2151  233
1.88300+ 2 6.00000+ 0 1.00000+ 0 2.12400- 2 2.40000- 1 0.00000+ 01189 2151  234
9.21051+ 1 0.00000+ 0          2          0         36          61189 2151  235
4.89500+ 2 2.00000+ 0 1.00000+ 0 1.43000- 3 2.20000- 1 0.00000+ 01189 2151  236
3.49600+ 2 3.00000+ 0 2.00000+ 0 2.00200- 3 2.20000- 1 0.00000+ 01189 2151  237
2.71900+ 2 4.00000+ 0 2.00000+ 0 2.57400- 3 2.20000- 1 0.00000+ 01189 2151  238
2.22500+ 2 5.00000+ 0 2.00000+ 0 3.14600- 3 2.20000- 1 0.00000+ 01189 2151  239
1.88300+ 2 6.00000+ 0 2.00000+ 0 3.71700- 3 2.20000- 1 0.00000+ 01189 2151  240
1.63200+ 2 7.00000+ 0 1.00000+ 0 4.28900- 3 2.20000- 1 0.00000+ 01189 2151  241
```

File 2 Unresolved Resonance Parameter Data

(Parameters are Energy Independent)

(Fission Widths not given)

File 2 Unresolved Resonance Parameter Data

(Fission Widths given)

(Only Fission Widths are Energy Dependent)

The following quantities are defined for use in specifying unresolved resonance parameters (LRU = 2, LRF = 1, LFW = 1):

SPI is the spin of the target nucleus, I.

AP is the scattering radius in units of 10^{-12} cm. No channel quantum number dependence is currently permitted by the format.

NE is the number of energy points at which energy-dependent widths are tabulated. (NE $\leq$ 150.)

NLS is the number of ℓ-values (NLS $\leq$ 3.)

ES_i is the energy of the i^{th} point used to tabulate energy-dependent widths.

L is the value of ℓ.

AWRI is the ratio of the mass of a particular isotope to that of the neutron.

NJS is the number of J-states for a particular ℓ-state. (NJS $\leq$ 6.)

AJ is the floating-point value of the J (the spin, or total angular momentum of the set of parameters).

D is the average level spacing for resonances with spin J.

AMUN is the number of degrees of freedom in the neutron width distribution. (1.0 $\leq$ AMUN $\leq$ 2.0)

MUF is the integer value of the number of degrees of freedom for fission widths. (1 $\leq$ MUF $\leq$ 4)

GNO is the average reduced neutron width.

GG is the average radiation width.

GF is the average fission width.

File 2 Unresolved Resonance Parameter Data

(Fission Width are given)

(Only Fission Widths are Energy Dependent)

If LFW = 1 (fission widths given),

LRU = 2 (unresolved parameters),

LRF = 1 (only fission widths are energy-dependent; the rest are energy-independent).

The structure of a subsection is:

(MAT, 2, 151/SPI,	AP,	0,	0,	NE,	NLS/
ES_1,	ES_2,	ES_3,	.,	.,	.,
.,	.,	.,	.,	ES_{NE}	)LIST
(MAT, 2, 151/AWRI,	0.0,	L,	0,	NJS,	0)CONT(ℓ)
(MAT, 2, 151/0.0,	0.0,	L,	MUF,	NE+6,	0/
D,	AJ,	AMUN,	GNO,	GG,	0.0,
GF_1,	GF_2,	GF_3,	.,	.,	.,
.,	.,	.,	GF_{NE}		)LIST

The last LIST record is repeated for each J-state (there will be NJS such LIST records). A new CØNT(ℓ) record will then be given which will be followed by its NJS LIST records until data for all ℓ-values have been specified (there will be NLS sets of data).

In the above section, no provision was made for INT, and interpolation is assumed to be lin-lin.

File 2 Unresolved Resonance Parameter Data

(Fission Widths are given)

(Only Fission Widths are Energy Dependent)

```
0.00000+ 0 9.30900- 1          0          0          3          21338 2151  23
2.00000+ 2 5.00000+ 2 1.00000+ 4                                  1338 2151  24
2.36167+ 2 0.00000+ 0          0          0          1          01338 2151  25
0.00000+ 0 0.00000+ 0          0          3          9          01338 2151  26
1.37000+ 1 5.00000- 1 1.00000+ 0 2.55600- 3 3.48000- 2 0.00000+ 01338 2151  27
7.81000- 3 1.49800- 2 9.02500- 2                                  1338 2151  28
2.36167+ 2 0.00000+ 0          1          0          2          01338 2151  29
0.00000+ 0 0.00000+ 0          1          3          9          01338 2151  30
1.37000+ 1 5.00000- 1 1.00000+ 0 1.61000- 3 3.48000- 2 0.00000+ 01338 2151  31
7.81000- 3 1.49800- 2 9.02500- 2                                  1338 2151  32
0.00000+ 0 0.00000+ 0          1          3          9          01338 2151  33
6.85000+ 0 1.50000+ 0 1.00000+ 0 8.39000- 4 3.48000- 2 0.00000+ 01338 2151  34
3.91000- 3 7.49000- 3 4.51800- 2                                  1338 2151  35
```

File 2 Unresolved Resonance Parameter Data

(Fission Widths are given)

(Only Fission Widths are Energy Dependent)

File 2 Unresolved Resonance Parameter Data
(All Parameters are Energy Dependent)

The following quantities are defined for use in specifying unresolved resonance parameters (LRU = 2, LRF = 2):

SPI is the spin of the target nucleus, I.

AP is the scattering radius in units of 10^{-12} cm. No channel quantum number dependence is currently permitted by the format.

NE is the number of energy points at which energy-dependent widths are tabulated. (NE $\leq$ 150.)

NLS is the number of ℓ-values (NLS $\leq$ 3.)

ES_i is the energy of the i^{th} point used to tabulate energy-dependent widths.

L is the value of ℓ.

AWRI is the ratio of the mass of a particular isotope to that of the neutron.

NJS is the number of J-states for a particular ℓ-state. (NJS $\leq$ 6.)

AJ is the floating-point value of the J (the spin, or total angular momentum of the set of parameters).

D is the average level spacing for resonances with spin J.

AMUX is the number of degrees of freedom used in the competitive width distribution. (1.0 $\leq$ 2.0)

AMUN is the number of degrees of freedom in the neutron width distribution. (1.0 $\leq$ AMUN $\leq$ 2.0)

AMUG is the number of degrees of freedom in the radiation width distribution. (At present AMUG = 0.0. This implies a constant value of Γ_γ.)

AMUF is the number of degrees of freedom in the fission width distribution. (1.0 $\leq$ AMUF $\leq$ 4.0)

INT defines the interpolation scheme to be used for interpolating between the cross-sections obtained from average resonance parameters.

GNO is the average reduced neutron width.

GG is the average radiation width.

GF is the average fission width.

GX is the average competitive reaction width.

File 2 Unresolved Resonance Parameter Data

(All Parameters are Energy Dependent)

If LFW = 0 or 1 (does not depend on LFW).

LRU = 2 (unresolved parameters),

LRF = 2 (all energy-dependent parameters).

The structure of a subsection is:

(MAT, 2, 151/SPI, AP, 0, 0, NLS, 0)CONT

(MAT, 2, 151/AWRI, 0.0, L, 0, NJS, 0)CONT

(MAT, 2, 151/AJ, 0.0, INT, 0, (6*NE)+6, NE /

0.0, 0.0, AMUX, AMUN, AMUG, AMUF,

ES_1, D_1, GX_1, GNO_1, GG_1, GF_1,

ES_2, D_2, GX_2, GNO_2, GG_2, GF_2,

ES_{NE}, D_{NE}, GX_{NE}, GNO_{NE}, GG_{NE}, GF_{NE})LIST

The LIST record is repeated until all the NJS J-states have been specified for a given ℓ-value. A new CONT (ℓ) record is then given, and all data for each J-state for that ℓ-value are given. The structure is repeated until all ℓ-states have been specified.

File 2 Unresolved Resonance Parameter Data

(All Parameters are Energy Dependent)

```
3.50000+ 0 9.56630- 1          0          0          2          01395 2151  137
2.33025+ 2 0.00000+ 0          0          0          2          01395 2151  138
3.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  139
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 3.00000+ 01395 2151  140
8.20000+ 1 9.52600- 1 0.00000+ 0 1.03289- 4 3.50000- 2 4.43612- 21395 2151  141
8.65000+ 1 9.52600- 1 0.00000+ 0 8.85426- 5 3.50000- 2 9.71759- 21395 2151  142
9.10000+ 1 9.52600- 1 0.00000+ 0 1.15571- 4 3.50000- 2 7.29406- 21395 2151  143
                                (cards deleted)
2.46000+ 4 9.52600- 1 0.00000+ 0 9.19730- 5 3.50000- 2 1.61215- 11395 2151  276
2.50000+ 4 9.52600- 1 0.00000+ 0 8.93863- 5 3.50000- 2 1.24342- 11395 2151  277
4.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  278
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 2.00000+ 01395 2151  279
8.20000+ 1 8.09300- 1 0.00000+ 0 1.05873- 4 3.50000- 2 1.97216- 11395 2151  280
8.65000+ 1 8.09300- 1 0.00000+ 0 9.07582- 5 3.50000- 2 9.37354- 21395 2151  281
9.10000+ 1 8.09300- 1 0.00000+ 0 1.18463- 4 3.50000- 2 1.13654- 11395 2151  282
                                (cards deleted)
2.46000+ 4 8.09300- 1 0.00000+ 0 9.42745- 5 3.50000- 2 1.25252- 11395 2151  415
2.50000+ 4 8.09300- 1 0.00000+ 0 9.16230- 5 3.50000- 2 1.40220- 11395 2151  416
2.33025+ 2 0.00000+ 0          1          0          4          01395 2151  417
2.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  418
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 4.00000+ 01395 2151  419
8.20000+ 1 1.23830+ 0 0.00000+ 0 1.79430- 4 3.50000- 2 3.94000- 11395 2151  420
8.65000+ 1 1.23830+ 0 0.00000+ 0 1.79430- 4 3.50000- 2 3.94000- 11395 2151  421
9.10000+ 1 1.23830+ 0 0.00000+ 0 1.79430- 4 3.50000- 2 3.94000- 11395 2151  422
                                (cards deleted)
2.46000+ 4 1.23830+ 0 0.00000+ 0 1.79430- 4 3.50000- 2 3.94000- 11395 2151  555
2.50000+ 4 1.23830+ 0 0.00000+ 0 1.79430- 4 3.50000- 2 3.94000- 11395 2151  556
3.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  557
0.00000+ 0 0.00000+ 0 0.00000+ 0 2.00000+ 0 0.00000+ 0 3.00000+ 01395 2151  558
8.20000+ 1 9.52600- 1 0.00000+ 0 1.19170- 4 3.50000- 2 2.77000- 11395 2151  559
8.65000+ 1 9.52600- 1 0.00000+ 0 1.19170- 4 3.50000- 2 2.77000- 11395 2151  560
9.10000+ 1 9.52600- 1 0.00000+ 0 1.19170- 4 3.50000- 2 2.77000- 11395 2151  561
                                (cards deleted)
2.46000+ 4 9.52600- 1 0.00000+ 0 1.19170- 4 3.50000- 2 2.77000- 11395 2151  694
2.50000+ 4 9.52600- 1 0.00000+ 0 1.19170- 4 3.50000- 2 2.77000- 11395 2151  695
4.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  696
0.00000+ 0 0.00000+ 0 0.00000+ 0 2.00000+ 0 0.00000+ 0 4.00000+ 01395 2151  697
8.20000+ 1 8.09300- 1 0.00000+ 0 1.01240- 4 3.50000- 2 2.58000- 11395 2151  698
8.65000+ 1 8.09300- 1 0.00000+ 0 1.01240- 4 3.50000- 2 2.58000- 11395 2151  699
9.10000+ 1 8.09300- 1 0.00000+ 0 1.01240- 4 3.50000- 2 2.58000- 11395 2151  700
                                (cards deleted)
2.46000+ 4 8.09300- 1 0.00000+ 0 1.01240- 4 3.50000- 2 2.58000- 11395 2151  833
2.50000+ 4 8.09300- 1 0.00000+ 0 1.01240- 4 3.50000- 2 2.58000- 11395 2151  834
5.00000+ 0 0.00000+ 0          2          0        828        1371395 2151  835
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 3.00000+ 01395 2151  836
8.20000+ 1 7.47700- 1 0.00000+ 0 1.08340- 4 3.50000- 2 1.79000- 11395 2151  837
8.65000+ 1 7.47700- 1 0.00000+ 0 1.08340- 4 3.50000- 2 1.79000- 11395 2151  838
9.10000+ 1 7.47700- 1 0.00000+ 0 1.08340- 4 3.50000- 2 1.79000- 11395 2151  839
                                (cards deleted)
2.46000+ 4 7.47700- 1 0.00000+ 0 1.08340- 4 3.50000- 2 1.79000- 11395 2151  972
2.50000+ 4 7.47700- 1 0.00000+ 0 1.08340- 4 3.50000- 2 1.79000- 11395 2151  973
```

File 2 Unresolved Resonance Parameter Data

(All Parameters are Energy Dependent)

File 3 Neutron Cross Sections

For File 3 the following quantities are defined:

LIS is an indicator that specifies the initial state of the target nucleus (for materials that represent nuclides).

LIS = 0, the initial state is the ground state.

= 1, the initial state is the first excited state.

LFS is an indicator that specifies the final excited state of the residual nucleus produced by a particular reaction.

LFS = 0, the final state is the ground state.

= 1, the final state is the first excited state.

= 98, an unspecified range of final states.

= 99, all final states.

Q is defined for the reaction a + A → b + B as:

$$Q(eV) = \left((m_a + m_A) - (m_b + m_B + E_B(eV)/(9.315016+08)) \right)(9.315016+08),$$

S is the temperature (°K). NOTE: If the LR flag is used, S becomes the breakup energy for the final reaction products corresponding to that particular LR value. (Exceptions are LR = 39 and LR = 40, which are set equal to 0.0).

LT is a flag to specify whether temperature-dependent data are given. S and LT are normally zero.

LR is a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x).

NR is the number of energy ranges that have been given. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data. (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range.

$\sigma_{(E)}$ is the cross section (barns) for a particular reaction type at incident energy point, E, in (eV). Data are given in energy-cross section pairs.

The structure of a section is

(MAT, 3, MT/ZA , AWR, LIS, LFS, 0 , 0)HEAD

(MAT, 3, MT/S , Q , LT, LR , NR, NP/E_{int}/σ(E))TABl

(MAT, 3, 0 /0.0, 0.0, 0, 0, 0 , 0)SEND


```
3.00600+ 3 5.96340+ 0          0          99          0          01303 3103    1
0.00000+ 0-2.72730+ 6          0           0          1          771303 3103    2
        77          2                                             1303 3103    3
3.18460+ 6 0.00000+ 0 3.20000+ 6 2.00000- 4 3.30000+ 6 1.80000- 31303 3103    4
3.40000+ 6 6.00000- 3 3.50000+ 6 1.22000- 2 3.60000+ 6 1.90000- 21303 3103    5
3.70000+ 6 2.44000- 2 3.80000+ 6 2.85000- 2 3.90000+ 6 3.16000- 21303 3103    6
4.00000+ 6 3.40000- 2 4.05000+ 6 3.50000- 2 4.10000+ 6 3.53000- 21303 3103    7
4.15000+ 6 3.59000- 2 4.20000+ 6 3.58000- 2 4.25000+ 6 3.54000- 21303 3103    8
4.30000+ 6 3.46000- 2 4.35000+ 6 3.47000- 2 4.40000+ 6 3.22000- 21303 3103    9
4.45000+ 6 3.10000- 2 4.50000+ 6 3.00000- 2 4.55000+ 6 2.92000- 21303 3103   10
4.60000+ 6 2.87000- 2 4.65000+ 6 2.84000- 2 4.70000+ 6 2.85000- 21303 3103   11
4.75000+ 6 2.89000- 2 4.80000+ 6 2.90000- 2 4.85000+ 6 2.91000- 21303 3103   12
4.90000+ 6 2.90000- 2 5.00000+ 6 2.87000- 2 5.05000+ 6 2.85000- 21303 3103   13
5.10000+ 6 2.82000- 2 5.20000+ 6 2.75000- 2 5.30000+ 6 2.66000- 21303 3103   14
5.40000+ 6 2.57000- 2 5.50000+ 6 2.49000- 2 5.60000+ 6 2.40000- 21303 3103   15
5.80000+ 6 2.25000- 2 6.00000+ 6 2.11000- 2 6.20000+ 6 2.01000- 21303 3103   16
6.40000+ 6 1.92000- 2 6.60000+ 6 1.84000- 2 6.80000+ 6 1.77000- 21303 3103   17
7.00000+ 6 1.71000- 2 7.20000+ 6 1.65000- 2 7.40000+ 6 1.60000- 21303 3103   18
7.60000+ 6 1.55000- 2 8.00000+ 6 1.45000- 2 8.20000+ 6 1.41000- 21303 3103   19
8.40000+ 6 1.36000- 2 8.50000+ 6 1.35000- 2 8.60000+ 6 1.32000- 21303 3103   20
8.80000+ 6 1.29000- 2 9.00000+ 6 1.25000- 2 9.50000+ 6 1.15000- 21303 3103   21
1.00000+ 7 1.07000- 2 1.04000+ 7 1.00000- 2 1.05000+ 7 9.80000- 31303 3103   22
1.08000+ 7 9.40000- 3 1.10000+ 7 9.10000- 3 1.15000+ 7 8.40000- 31303 3103   23
1.20000+ 7 7.80000- 3 1.25000+ 7 7.30000- 3 1.30000+ 7 6.80000- 31303 3103   24
1.35000+ 7 6.40000- 3 1.40000+ 7 6.10000- 3 1.45000+ 7 5.80000- 31303 3103   25
1.50000+ 7 5.65000- 3 1.55000+ 7 5.50000- 3 1.60000+ 7 5.30000- 31303 3103   26
1.65000+ 7 5.20000- 3 1.70000+ 7 5.10000- 3 1.75000+ 7 5.00000- 31303 3103   27
1.80000+ 7 5.00000- 3 1.85000+ 7 5.00000- 3 1.90000+ 7 4.90000- 31303 3103   28
1.95000+ 7 4.90000- 3 2.00000+ 7 4.90000- 3                      1303 3103   29
0.00000+ 0 0.00000+ 0          0          0          0          01303 3  0   30
```

The reaction Q-value is defined as the kinetic energy (eV) released by a reaction (positive) or required for a reaction (negative). Except for fission, for a reaction having a threshold, the threshold energy E_{th} is given by

$$E_{th} = \frac{AWR + 1}{AWR} |Q|$$

where AWR is the atomic mass ratio given on the HEAD card of each section.

For a material that is a mixture of several isotopes, the Q-value is not uniquely defined. The threshold energy generally should pertain to the particular isotope that contributes to the cross section at the lowest energy, but see discussion in Section 3.2.2.2.

(Legendre Coefficients and Transformation)

The following quantities are defined.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 1, a transformation matrix is given.

LTT is a flag to specify the representation used.

LTT = 1, the data are given as Legendre expansion coefficients, $a_\ell(E)$;

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 0, not all isotropic,

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

NK is the number of elements in the transformation matrix ($NK \leq 441$). $NK = (NM + 1)^2$.

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system. NM should be an even number.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NL is the highest order Legendre polynomial that is given at each energy ($NL \leq 20$).

Other commonly used variables are given in the Glossary (Appendix A).

Legendre Polynomial Coefficients and Transformation Matrix Given:

LTT = 1, LVT = 1, and LI = 0

When LTT = 1 (angular distributions given in terms of Legendre polynomial coefficients) and LVT = 1, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0)HEAD LTT = 1, LVT = 1

(MAT, 4, MT/0.0, AWR, LI, LCT, NK, NM/V_K)LIST LI = 0

(MAT, 4, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2

(MAT, 4, MT/T , E_1 , LT, 0 , NL, 0/$a_\ell(E_1)$)LIST

(MAT, 4, MT/T , E_2 , LT, 0 , NL, 0/$a_\ell(E_2)$)LIST

(MAT, 4, MT/T , E_{NE}, LT, 0 , NL, 0/$a_\ell(E_{NE})$)LIST

(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0)SEND

Note that T and LT refer to temperature (in °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified.


```
6.00000+ 3 1.18969+ 1          1          1          0          01306 4 2    1
0.00000+ 0 1.18969+ 1          0          2        441          201306 4 2    2
1.00000+ 0 5.60370- 2 1.41449- 3-1.06648- 9 0.00000+ 0 0.00000+ 01306 4 2    3
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01306 4 2    4
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01306 4 2    5
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 9.95761- 1 1.00663- 11306 4 2    6
4.84477- 3 1.13334- 4 1.41574- 6-3.09538- 8 0.00000+ 0 0.00000+ 01306 4 2    7
                                  (cards deleted)
1.84395-13-4.23666-12 8.52332-11-1.50800- 9 2.34819- 8-3.21115- 71306 4 2   74
3.83673- 6-3.97165- 5 3.51714- 4-2.61541- 3 1.58843- 2-7.54255- 21306 4 2   75
2.59530- 1-5.51607- 1 4.33796- 1                                 1306 4 2   76
0.00000+ 0 0.00000+ 0          0          0          1         2171306 4 2   77
       217          2                                           1306 4 2   78
0.00000+ 0 1.00000- 5          0          0          1          01306 4 2   79
0.00000+ 0                                                       1306 4 2   80
0.00000+ 0 1.00000+ 3          0          0          1          01306 4 2   81
1.40110- 4                                                       1306 4 2   82
0.00000+ 0 5.00000+ 3          0          0          1          01306 4 2   83
6.98200- 4                                                       1306 4 2   84
0.00000+ 0 1.00000+ 4          0          0          1          01306 4 2   85
1.39060- 3                                                       1306 4 2   86
0.00000+ 0 5.00000+ 4          0          0          2          01306 4 2   87
6.72820- 3 7.49880- 5                                            1306 4 2   88
0.00000+ 0 1.00000+ 5          0          0          2          01306 4 2   89
1.29230- 2 2.79310- 4                                            1306 4 2   90
0.00000+ 0 2.00000+ 5          0          0          2          01306 4 2   91
2.38830- 2 9.73570- 4                                            1306 4 2   92
0.00000+ 0 3.00000+ 5          0          0          3          01306 4 2   93
3.31650- 2 1.91720- 3 6.33800- 5                                 1306 4 2   94
0.00000+ 0 4.00000+ 5          0          0          3          01306 4 2   95
4.09900- 2 2.99520- 3 1.28480- 4                                 1306 4 2   96
0.00000+ 0 5.00000+ 5          0          0          3          01306 4 2   97
4.75290- 2 4.12980- 3 2.24150- 4                                 1306 4 2   98
0.00000+ 0 6.00000+ 5          0          0          3          01306 4 2   99
5.29160- 2 5.27140- 3 3.49240- 4                                 1306 4 2  100
                                  (cards deleted)
0.00000+ 0 1.50000+ 7          0          0          6          01306 4 2  501
6.30000- 1 4.60000- 1 3.10000- 1 2.00000- 1 9.00000- 2 4.70000- 21306 4 2  502
0.00000+ 0 1.60000+ 7          0          0          6          01306 4 2  503
6.60000- 1 4.80000- 1 3.20000- 1 2.10000- 1 1.00000- 1 2.00000- 21306 4 2  504
0.00000+ 0 1.70000+ 7          0          0          6          01306 4 2  505
6.80000- 1 4.90000- 1 3.30000- 1 2.00000- 1 1.07000- 1 5.00000- 21306 4 2  506
0.00000+ 0 1.80000+ 7          0          0          6          01306 4 2  507
7.00000- 1 4.90000- 1 3.30000- 1 2.00000- 1 1.11000- 1 5.00000- 21306 4 2  508
0.00000+ 0 1.90000+ 7          0          0          6          01306 4 2  509
7.20000- 1 5.10000- 1 3.50000- 1 2.00000- 1 1.12000- 1 4.30000- 21306 4 2  510
0.00000+ 0 2.00000+ 7          0          0          6          01306 4 2  511
7.30000- 1 5.20000- 1 3.50000- 1 2.00000- 1 1.13000- 1 4.00000- 21306 4 2  512
0.00000+ 0 0.00000+ 0          0          0          0          01306 4 0  513
```

File 4 Angular Distributions of Secondary Neutrons

The absolute differential cross sections are obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi} \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell(E) P_\ell(\mu)$$

where μ = cosine of the scattered angle in either the laboratory or the center-of-mass system;

E = energy of the incident neutron in the laboratory system;

$\sigma_s(E)$ = the scattering cross section, e.g., elastic scattering at energy E as given in File 3 for the particular reaction type (MT);

ℓ = order of the Legendre polynomial;

$\frac{d\sigma}{d\Omega}(\Omega,E)$ = differential scattering cross section in units of barns per steradian;

a_ℓ = the ℓ^{th} Legendre polynomial coefficient and it is understood that $a_o = 1.0$.

The Legendre expansion coefficients $a_\ell(E)$ in the two systems are related through an energy-independent transformation matrix, $U_{\ell m}$, and its inverse, $U^{-1}_{\ell m}$:

$$a_\ell^{Lab}(E) = \sum_{m=0}^{NM} U_{\ell m}\, a_m^{CM}(E)$$

and

$$a_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell m}\, a_m^{Lab}(E) .$$

File 4 Angular Distributions of Secondary Neutrons

(Only Legendre Coefficients given)

The following quantities are defined.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given.

LTT is a flag to specify the representation used.

LTT = 1, the data are given as Legendre expansion coefficients, $a_\ell(E)$;

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 0, not all isotropic,

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

NE is the number of incident energy points at which angular distributions are given (NE $\leq$ 500).

NL is the highest order Legendre polynomial that is given at each energy (NL $\leq$ 20).

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons

(Only Legendre Coefficients Given)

Legendre Polynomial Coefficients Given and the Transformation Matrix

Not Given: LTT = 1, LVT = 0, and LI = 0

If LTT = 1 and LVT = 0, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0)HEAD	LTT = 1, LVT = 0
(MAT, 4 MT/0.0, AWR, LI, LCT, 0, 0) CONT.	LI = 0
(MAT, 4, MT/0.0, 0.0, 0, 0, NR, NE/E_{int})TAB2	
(MAT, 4, MT/T , E_1 , LT, 0 , NL, 0/$a_\ell(E_1)$)LIST	
(MAT, 4, MT/T , E_2 , LT, 0 , NL, 0/$a_\ell(E_2)$)LIST	

(MAT, 4, MT/T , E_{NE}, LT, 0 , NL, 0/$a_\ell(E_{NE})$LIST	
(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0)SEND	

Note that T and LT refer to temperature (in °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified. This form is always used for angular distributions of nonelastically scattered neutrons when Legendre polynomial expansion coefficients are used.

File 4 Angular Distributions of Secondary Neutrons

(Only Legendre Coefficients given)

```
 6.00000+ 3 1.18969+ 1          0          1          0 01306 4 51   1
 0.00000+ 0 1.18969+ 1          0          2          0 01306 4 51   2
 0.00000+ 0 0.00000+ 0          0          0          1 821306 4 51  3
        82          2                                   1306 4 51   4
 0.00000+ 0 4.81200+ 6          0          0          1 01306 4 51   5
 0.00000+ 0                                             1306 4 51   6
 0.00000+ 0 5.25000+ 6          0          0          2 01306 4 51   7
 1.50000- 2 2.00000- 3                                  1306 4 51   8
 0.00000+ 0 5.37000+ 6          0          0          2 01306 4 51   9
 3.35000- 2 1.20000- 2                                  1306 4 51  10
                                (cards deleted)
 0.00000+ 0 8.72810+ 6          0          0          5 01306 4 51 129
 1.03360- 1 1.82410- 1 2.97400- 2 2.83900- 2 2.06100- 2  1306 4 51 130
 0.00000+ 0 8.80000+ 6          0          0          5 01306 4 51 131
 5.49300- 2 1.78750- 1 2.39700- 2 2.31300- 2 1.90000- 2  1306 4 51 132
 0.00000+ 0 8.85000+ 6          0          0          4 01306 4 51 133
 2.12600- 2 1.76200- 1 1.99600- 2 1.94700- 2            1306 4 51 134
 0.00000+ 0 9.00000+ 6          0          0          4 01306 4 51 135
-1.00000- 2 1.90000- 1 2.00000- 2 2.00000- 2            1306 4 51 136
 0.00000+ 0 9.50000+ 6          0          0          4 01306 4 51 137
 9.00000- 2 2.40000- 1 2.00000- 2 3.00000- 2            1306 4 51 138
 0.00000+ 0 1.00000+ 7          0          0          4 01306 4 51 139
 3.50000- 1 1.60000- 1 3.00000- 2 2.00000- 2            1306 4 51 140
 0.00000+ 0 1.05000+ 7          0          0          4 01306 4 51 141
 9.00000- 2 1.90000- 1 4.00000- 2 4.00000- 2            1306 4 51 142
 0.00000+ 0 1.10000+ 7          0          0          4 01306 4 51 143
 1.90000- 1 2.20000- 1 5.00000- 2 2.00000- 2            1306 4 51 144
 0.00000+ 0 1.20000+ 7          0          0          4 01306 4 51 145
 2.90000- 1 2.40000- 1 5.00000- 2-5.00000- 3            1306 4 51 146
 0.00000+ 0 1.25000+ 7          0          0          4 01306 4 51 147
 1.80000- 1 2.60000- 1 4.00000- 2 2.50000- 2            1306 4 51 148
 0.00000+ 0 1.30000+ 7          0          0          4 01306 4 51 149
 1.50000- 1 3.10000- 1 4.00000- 2 2.50000- 2            1306 4 51 150
 0.00000+ 0 1.35000+ 7          0          0          5 01306 4 51 151
 1.30000- 1 2.60000- 1 2.00000- 2 1.60000- 2-3.00000- 2  1306 4 51 152
 0.00000+ 0 1.40000+ 7          0          0          5 01306 4 51 153
 1.60000- 2 1.70000- 1-3.00000- 3-3.00000- 2 1.00000- 2  1306 4 51 154
 0.00000+ 0 1.45000+ 7          0          0          5 01306 4 51 155
 2.10000- 1 2.20000- 1-4.00000- 2-2.00000- 2-3.00000- 2  1306 4 51 156
 0.00000+ 0 1.50000+ 7          0          0          5 01306 4 51 157
 2.50000- 1 2.30000- 1-3.00000- 2-1.00000- 2-4.00000- 2  1306 4 51 158
 0.00000+ 0 1.60000+ 7          0          0          5 01306 4 51 159
 3.10000- 1 2.50000- 1 1.00000- 2-5.00000- 3-2.00000- 2  1306 4 51 160
 0.00000+ 0 1.70000+ 7          0          0          4 01306 4 51 161
 3.70000- 1 2.60000- 1 6.00000- 2 5.00000- 3            1306 4 51 162
 0.00000+ 0 1.80000+ 7          0          0          4 01306 4 51 163
 4.30000- 1 2.70000- 1 1.10000- 1 1.00000- 2            1306 4 51 164
 0.00000+ 0 1.90000+ 7          0          0          5 01306 4 51 165
 4.60000- 1 2.85000- 1 1.60000- 1 4.00000- 2 2.00000- 2  1306 4 51 166
 0.00000+ 0 2.00000+ 7          0          0          5 01306 4 51 167
 4.90000- 1 3.00000- 1 2.10000- 1 1.00000- 1 3.00000- 2  1306 4 51 168
 0.00000+ 0 0.00000+ 0          0          0          0 01306 4  0 169
```

File 4 Angular Distributions of Secondary Neutrons

(Only Legendre Coefficients Given)

The absolute differential cross sections are obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi} \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell(E) P_\ell(\mu) ,$$

where μ = cosine of the scattered angle in either the laboratory or the center-of-mass system;

E = energy of the incident neutron in the laboratory system;

$\sigma_s(E)$ = the scattering cross section, e.g., elastic scattering at energy E as given in File 3 for the particular reaction type (MT);

ℓ = order of the Legendre polynomial;

$\frac{d\sigma}{d\Omega}(\Omega,E)$ = differential scattering cross section in units of barns per steradian;

a_ℓ = the ℓ^{th} Legendre polynomial coefficient and it is understood that $a_o = 1.0$.

File 4 Angular Distributions of Secondary Neutrons

(Tabulated Distribution and Transformation)

The following quantities are defined.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 1, a transformation matrix is given.

LTT is a flag to specify the representation used and it may have the following values:

LTT = 2, the data are given as normalized probability distributions, $p(\mu,E)$.

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 0, not all isotropic,

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

NK is the number of elements in the transformation matrix ($NK \leq 441$). $NK = (NM + 1)^2$.

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system. NM should be an even number.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy ($NP \leq 101$).

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons

(Tabulated Distribution and Transformation)

Tabulated Probability Distributions and Transformation Matrix Given:

LTT = 2, LVT = 1, and LI = 0

If the angular distributions are given as tabulated probability distributions, LTT = 2, and a transformation matrix is given for elastic scattering, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0)HEAD LVT = 1, LTT = 2

(MAT, 4, MT/0.0, AWR, LI, LCT, NK, NM/V_K)LIST LI = 0

(MAT, 4, MT/0.0, 0.0, 0 , 0 , NR, NE/E_{int})TAB2

(MAT, 4, MT/T , E_1 , LT, 0, NR, NP/μ_{int}/p(μ,E_1))TAB1

(MAT, 4, MT/T , E_2 , LT, 0, NR, NP/μ_{int}/p(μ,E_2))TAB1

(MAT, 4, MT/T , E_{NE}, LT, 0 , NR, NP/μ_{int}/p(μ,E_{NE}))TAB1

(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0, 0)SEND

T and LT are normally zero. See Appendix F for details on temperature dependence.

File 4 Angular Distributions of Secondary Neutrons

(Tabulated Distribution and Transformation)

```
 4.20000+ 4 9.51160+ 1          1          2          0          01321 4  2   1
 0.00000+ 0 9.51160+ 1          0          2        361         181321 4  2   2
 1.00000+ 0 7.00899- 3 2.21008- 5-1.38100-10 0.00000+ 0 0.00000+ 01321 4  2   3
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01321 4  2   4
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01321 4  2   5
                                (cards deleted)
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01321 4  2  60
 0.00000+ 0-1.05340-20 2.77279-18-7.61059-16 9.80913-14-9.36964-121321 4  2  61
 7.60542-10-5.18463- 8 2.89816- 6-1.28018- 4 4.20371- 3-9.14838- 21321 4  2  62
 9.90565- 1                                                       1321 4  2  63
 0.00000+ 0 0.00000+ 0          0          0          1         131321 4  2  64
        13          2                                            1321 4  2  65
 0.00000+ 0 1.00000- 5          0          0          1          21321 4  2  66
         2          2                                            1321 4  2  67
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                     1321 4  2  68
 0.00000+ 0 5.00000+ 4          0          0          1          21321 4  2  69
         2          2                                            1321 4  2  70
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                     1321 4  2  71
 0.00000+ 0 6.00000+ 4          0          0          1          21321 4  2  72
         2          2                                            1321 4  2  73
-1.00000+ 0 4.37000- 1 1.00000+ 0 5.63000- 1                     1321 4  2  74
 0.00000+ 0 1.50000+ 5          0          0          1          41321 4  2  75
         4          2                                            1321 4  2  76
-1.00000+ 0 4.71950- 1-6.00000- 1 4.26960- 1 0.00000+ 0 4.58950- 11321 4  2  77
 1.00000+ 0 6.49940- 1                                           1321 4  2  78
 0.00000+ 0 2.50000+ 5          0          0          1          61321 4  2  79
         6          2                                            1321 4  2  80
-1.00000+ 0 3.77080- 1-8.00000- 1 3.64070- 1-6.00000- 1 3.77080- 11321 4  2  81
-2.00000- 1 4.22080- 1 4.00000- 1 5.52110- 1 1.00000+ 0 7.80160- 11321 4  2  82
 0.00000+ 0 5.00000+ 5          0          0          1          81321 4  2  83
         8          2                                            1321 4  2  84
-1.00000+ 0 3.30070- 1-8.00000- 1 2.81060- 1-5.00000- 1 2.81060- 11321 4  2  85
-2.00000- 1 3.51070- 1 2.00000- 1 5.13100- 1 6.00000- 1 7.02140- 11321 4  2  86
 8.00000- 1 8.43170- 1 1.00000+ 0 1.05020+ 0                     1321 4  2  87
 0.00000+ 0 1.00000+ 6          0          0          1          91321 4  2  88
         9          2                                            1321 4  2  89
-1.00000+ 0 3.88200- 1-9.00000- 1 3.00170- 1-7.00000- 1 2.00110- 11321 4  2  90
-6.00000- 1 2.00110- 1-3.00000- 1 2.55140- 1 0.00000+ 0 3.33180- 11321 4  2  91
 4.00000- 1 5.55300- 1 8.00000- 1 1.02060+ 0 1.00000+ 0 1.44080+ 01321 4  2  92
                                (cards deleted)
 0.00000+ 0 2.00000+ 7          0          0          1         511321 4  2 156
        51          2                                            1321 4  2 157
-1.00000+ 0 5.64300- 3-9.90000- 1 1.93490- 2-9.80000- 1 2.71650- 21321 4  2 158
-9.70000- 1 3.07910- 2-9.60000- 1 3.15580- 2-9.40000- 1 2.83630- 21321 4  2 159
-9.00000- 1 1.83600- 2-8.80000- 1 1.54340- 2-8.60000- 1 1.44040- 21321 4  2 160
-8.30000- 1 1.56140- 2-7.30000- 1 2.48910- 2-6.60000- 1 2.66540- 21321 4  2 161
                                (cards deleted)
 8.70000- 1 4.65100- 1 8.80000- 1 6.76400- 1 8.90000- 1 9.71100- 11321 4  2 171
 9.00000- 1 1.37550+ 0 9.10000- 1 1.92270+ 0 9.20000- 1 2.65370+ 01321 4  2 172
 9.30000- 1 3.61960+ 0 9.40000- 1 4.88240+ 0 9.50000- 1 6.51960+ 01321 4  2 173
 9.70000- 1 1.13090+ 1 9.90000- 1 1.89960+ 1 1.00000+ 0 2.43600+ 11321 4  2 174
 0.00000+ 0 0.00000+ 0          0          0          0          01321 4  0 175
```

File 4 Angular Distributions of Secondary Neutrons

(Tabulated Distribution and Transformation)

The angular distributions are expressed as normalized probability distributions, i.e.,

$$\int_{-1}^{1} p(\mu,E)\,d\mu = 1,$$

where $p(\mu,E)\,d\mu$ is the probability that a neutron of incident energy E will be scattered into the interval $d\mu$ about an angle whose cosine is μ. The units of $p(\mu,E)$ are (unit cosine)$^{-1}$.

Absolute diffential cross sections are obtained by combining data from Files 3 and 4. The absolute differential cross section (in barns per steradian) is obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi}\,p(\mu,E)$$

where $\sigma_s(E)$ is given in File 3 (for the same MT number) and $p(\mu,E)$ is given in File 4.

The transformation matrix can be used to transform a set of Legendre expansion coefficients, which describe elastic scattering angular distributions, from one frame of reference to the other.

The Legendre expansion coefficients $a_\ell(E)$ in the two systems are related through an energy-independent transformation matrix, $U_{\ell m}$, and its inverse, $U^{-1}_{\ell m}$:

$$a_\ell^{Lab}(E) = \sum_{m=0}^{NM} U_{\ell m}\, a_m^{CM}(E)$$

and

$$a_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell m}\, a_m^{Lab}(E)\ .$$

File 4 Angular Distributions of Secondary Neutrons

(Only Tabulated Distributions given)

The following quantities are defined.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given.

LTT is a flag to specify the representation used.

LTT = 2, the data are given as normalized probability distributions, $p(\mu,E)$.

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 0, not all isotropic,

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy ($NP \leq 101$).

Other commonly used variables are given in the Glossary (Appendix A).

Tabulated Probability Distributions Given and Transformation Matrix

Not Given: LTT = 2, LVT = 0, and LI = 0

If the angular distributions are given as tabulated probability distributions, LTT = 2, and a transformation matrix is given for elastic scattering, the structure of a section is

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0)HEAD LVT = 1, LTT = 2

(MAT, 4, MT/0.0, AWR, LI, LCT, NK, NM/V_K)LIST LI = 0

(MAT, 4, MT/0.0, 0.0, 0 , 0 , NR, NE/E_{int})TAB2

(MAT, 4, MT/T , E_1 , LT, 0, NR, NP/μ_{int}/p(μ,E_1))TAB1

(MAT, 4, MT/T , E_2 , LT, 0, NR, NP/μ_{int}/p(μ,E_2))TAB1

(MAT, 4, MT/T , E_{NE}, LT, 0 , NR, NP/μ_{int}/p(μ,E_{NE}))TAB1

(MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0, 0)SEND

T and LT are normally zero. See Appendix F for details on temperature dependence.

This form is always used for angular distribution of nonelastically scattered neutrons when tabulated angular distributions are given.


```
 2.20000+ 4 4.74676+ 1          0          2          0          01322 4 53    1
 0.00000+ 0 4.74676+ 1          0          2          0          01322 4 53    2
 0.00000+ 0 0.00000+ 0          0          0          1          71322 4 53    3
         7          2                                            1322 4 53    4
 0.00000+ 0 1.00400+ 6          0          0          1         311322 4 53    5
        31          2                                            1322 4 53    6
-1.00000+ 0 5.00000- 1-9.94520- 1 5.00000- 1-9.78150- 1 5.00000- 11322 4 53    7
-9.51060- 1 5.00000- 1-9.13540- 1 5.00000- 1-8.66020- 1 5.00000- 11322 4 53    8
-8.09020- 1 5.00000- 1-7.43140- 1 5.00000- 1-6.69130- 1 5.00000- 11322 4 53    9
-5.87780- 1 5.00000- 1-5.00000- 1 5.00000- 1-4.06740- 1 5.00000- 11322 4 53   10
-3.09020- 1 5.00000- 1-2.07910- 1 5.00000- 1-1.04530- 1 5.00000- 11322 4 53   11
 1.32690- 6 5.00000- 1 1.04530- 1 5.00000- 1 2.07910- 1 5.00000- 11322 4 53   12
 3.09020- 1 5.00000- 1 4.06740- 1 5.00000- 1 5.00000- 1 5.00000- 11322 4 53   13
 5.87790- 1 5.00000- 1 6.69130- 1 5.00000- 1 7.43150- 1 5.00000- 11322 4 53   14
 8.09020- 1 5.00000- 1 8.66030- 1 5.00000- 1 9.13550- 1 5.00000- 11322 4 53   15
 9.51060- 1 5.00000- 1 9.78150- 1 5.00000- 1 9.94520- 1 5.00000- 11322 4 53   16
 1.00000+ 0 5.00000- 1                                          1322 4 53   17
 0.00000+ 0 2.00000+ 6          0          0          1         311322 4 53   18
        31          2                                            1322 4 53   19
-1.00000+ 0 4.67697- 1-9.94520- 1 4.69007- 1-9.78150- 1 4.72758- 11322 4 53   20
-9.51060- 1 4.78448- 1-9.13540- 1 4.85359- 1-8.66020- 1 4.92610- 11322 4 53   21
-8.09020- 1 4.99341- 1-7.43140- 1 5.04831- 1-6.69130- 1 5.08582- 11322 4 53   22
-5.87780- 1 5.10412- 1-5.00000- 1 5.10422- 1-4.06740- 1 5.09012- 11322 4 53   23
-3.09020- 1 5.06732- 1-2.07910- 1 5.04231- 1-1.04530- 1 5.02091- 11322 4 53   24
 1.32690- 6 5.00741- 1 1.04530- 1 5.00371- 1 2.07910- 1 5.00901- 11322 4 53   25
 3.09020- 1 5.02031- 1 4.06740- 1 5.03241- 1 5.00000- 1 5.03971- 11322 4 53   26
 5.87790- 1 5.03661- 1 6.69130- 1 5.01921- 1 7.43150- 1 4.98601- 11322 4 53   27
 8.09020- 1 4.93790- 1 8.66030- 1 4.87899- 1 9.13550- 1 4.81549- 11322 4 53   28
 9.51060- 1 4.75488- 1 9.78150- 1 4.70467- 1 9.94520- 1 4.67167- 11322 4 53   29
 1.00000+ 0 4.66017- 1                                          1322 4 53   30
                                (cards deleted)
 0.00000+ 0 2.00000+ 7          0          0          1         311322 4 53   83
        31          2                                            1322 4 53   84
-1.00000+ 0 3.37504- 1-9.94520- 1 3.13714- 1-9.78150- 1 2.72366- 11322 4 53   85
-9.51060- 1 2.70526- 1-9.13540- 1 3.27320- 1-8.66020- 1 3.95279- 11322 4 53   86
-8.09020- 1 4.00971- 1-7.43140- 1 3.18016- 1-6.69130- 1 2.00816- 11322 4 53   87
-5.87780- 1 1.42511- 1-5.00000- 1 1.93853- 1-4.06740- 1 3.17916- 11322 4 53   88
-3.09020- 1 4.22881- 1-2.07910- 1 4.41889- 1-1.04530- 1 3.86235- 11322 4 53   89
 1.32690- 6 3.25929- 1 1.04530- 1 3.21467- 1 2.07910- 1 3.74810- 11322 4 53   90
 3.09020- 1 4.44060- 1 4.06740- 1 5.00844- 1 5.00000- 1 5.67903- 11322 4 53   91
 5.87790- 1 6.95887- 1 6.69130- 1 9.01505- 1 7.43150- 1 1.12918+ 01322 4 53   92
 8.09020- 1 1.27705+ 0 8.66030- 1 1.26814+ 0 9.13550- 1 1.10417+ 01322 4 53   93
 9.51060- 1 8.60107- 1 9.78150- 1 6.30329- 1 9.94520- 1 4.78334- 11322 4 53   94
 1.00000+ 0 4.26722- 1                                          1322 4 53   95
 0.00000+ 0 0.00000+ 0          0          0          0          01322 4  0   96
```

File 4 Angular Distributions of Secondary Neutrons

(Only Tabulated Distributions given)

The angular distributions are expressed as normalized probability distributions, i.e.,

$$\int_{-1}^{1} p(\mu,E)\,d\mu = 1,$$

where $p(\mu,E)\,d\mu$ is the probability that a neutron of incident energy E will be scattered into the interval $d\mu$ about an angle whose cosine is μ. The units of $p(\mu,E)$ are (unit cosine)$^{-1}$.

Absolute diffential cross sections are obtained by combining data from Files 3 and 4. If tabulated distributions are given, the absolute differential cross section (in barns per steradian) is obtained by

$$\frac{d\sigma}{d\Omega}(\Omega,E) = \frac{\sigma_s(E)}{2\pi}\,p(\mu,E)$$

where $\sigma_s(E)$ is given in File 3 (for the same MT number) and $p(\mu,E)$ is given in File 4.

File 4 Angular Distributions of Secondary Neutrons

(All Distributions Isotropic)

The following quantities are defined.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given

LTT is a flag to specify the representation used and it may have the following values:

LTT = 0, all angular distributions are isotropic,

LI is a flag to specify whether all the angular distributions are isotropic:

LI = 1, all isotropic.

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the LAB system;

LCT = 2, the data are given in the CM system.

All Angular Distributions Isotropic:

LTT=0, LVT=0, LI=1

When all angular distributions for a given MT are assumed to be isotropic then the section structure is:

(MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0)HEAD LVT=0, LTT=0

(MAT, 4, MT/0.0, AWR, LI, LCT, 0,0)CØNT LI=1

(MAT, 4, 0 /0.0, 0.0, 0, 0, 0, 0)SEND


```
2.20000+ 4 4.74676+ 1          0          0          0          01322 4 51    1
0.00000+ 0 4.74676+ 1          1          2          0          01322 4 51    2
0.00000+ 0 0.00000+ 0          0          0          0          01322 4  0    3
```

File 4 Angular Distributions of Secondary Neutrons

(All Distributions Isotropic)

File 5 Energy Distributions of Secondary Neutrons

(General Structure)

The following quantities are defined

NK — is the number of partial energy distributions. There will be one subsection for each partial distribution.

LF — is a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution).

$p_k(E_N)$ — is the fractional part of the particular cross section that can be described by the k^{th} partial energy distribution at the N^{th} incident energy point.

NOTE: $\sum_{k=1}^{NK} p_k(E_N) = 1.0$

$f_k(E \rightarrow E')$ — is the k^{th} partial energy distribution. The definition depends on the value of LF.

File 5 Energy Distributions of Secondary Neutrons

(General Structure)

The structure of a section has the following form:

(MAT, 5, MT/ZA, AWR, 0, 0, NK, 0)HEAD

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

(MAT, 5, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

The structure of a subsection depends on the value of LF. Subsections should be ordered by increasing values of LF. For cases in which more than one subsection contains data using the same LF, these subsections should be ordered by increasing values of θ.

File 5 Energy Distributions of Secondary Neutrons
(General Structure)

```
4.80000+ 4 1.11460+ 2          0          0          2          01281 5 16    1

                                <subsection for k=1>

7.43300+ 6 0.00000+ 0          0          9          1          41281 5 16    2
         4          2                                                1281 5 16    3
7.50000+ 6 0.00000+ 0 7.50000+ 6 5.00000- 1 2.00000+ 7 5.00000- 11281 5 16    4
2.00000+ 7 0.00000+ 0                                                1281 5 16    5
0.00000+ 0 0.00000+ 0          0          0          1          31281 5 16    6
         3          5                                                1281 5 16    7
7.50000+ 6 6.89960+ 5 1.50000+ 7 9.75750+ 5 2.00000+ 7 9.75750+ 51281 5 16    8

                                <subsection for k=2>

0.00000+ 0 0.00000+ 0          0          1          1          41281 5 16    9
         4          2                                                1281 5 16   10
7.50000+ 6 0.00000+ 0 7.50000+ 6 5.00000- 1 2.00000+ 7 5.00000- 11281 5 16   11
2.00000+ 7 0.00000+ 0                                                1281 5 16   12
0.00000+ 0 0.00000+ 0          0          0          1          21281 5 16   13
         2          2                                                1281 5 16   14
0.00000+ 0 7.50000+ 6          0          0          1          71281 5 16   15
         7          2                                                1281 5 16   16
1.00000- 5 3.68607-16 1.00000+ 4 1.23620- 5 2.00000+ 4 1.52729- 51281 5 16   17
4.00000+ 4 1.74174- 5 6.00000+ 4 1.76308- 5 6.70000+ 4 1.74065- 51281 5 16   18
6.70000+ 4 0.00000+ 0                                                1281 5 16   19
0.00000+ 0 2.00000+ 7          0          0          1         171281 5 16   20
        17          2                                                1281 5 16   21
1.00000- 5 1.00000-16 1.00000+ 4 3.35370- 6 2.00000+ 4 4.14340- 61281 5 16   22
4.00000+ 4 4.72520- 6 6.00000+ 4 4.78310- 6 8.00000+ 4 4.60920- 61281 5 16   23
1.00000+ 5 4.31890- 6 1.20000+ 5 3.97090- 6 1.40000+ 5 3.59830- 61281 5 16   24
1.60000+ 5 3.22120- 6 1.80000+ 5 2.85190- 6 2.00000+ 5 2.49810- 61281 5 16   25
2.50000+ 5 1.70680- 6 3.00000+ 5 1.06400- 6 3.50000+ 5 5.65100- 71281 5 16   26
4.00000+ 5 1.89300- 7 4.40000+ 5 1.00000-26                          1281 5 16   27
0.00000+ 0 0.00000+ 0          0          0          0          01281 5  0   28
```

File 5 Energy Distributions of Secondary Neutrons

(General Structure)

The energy distributions, $p(E \rightarrow E')$, are normalized so that

$$\int_0^{E'_{max}} p(E \rightarrow E')\, dE' = 1 ,$$

where E'_{max} is the maximum possible secondary neutron energy and its value depends on the incoming neutron energy E and the analytic representation of $p(E \rightarrow E')$. The secondary neutron energy E' is always expressed in the laboratory system.

The differential cross section is obtained from

$$\frac{d\sigma\ (E \rightarrow E')}{dE'} = m\, \sigma(E)\ p\ (E \rightarrow E'),$$

where $\sigma(E)$ is the cross section as given in File 3 for the same reaction type number (MT) and m is the neutron multiplicity for this reaction (m is implicit; e.g., m = 2 for n,2n reactions).

The energy distributions $p(E \rightarrow E')$ can be broken down into partial energy distributions, $f_k(E \rightarrow E')$, where each of the partial distributions can be described by different analytic representations;

$$p(E \rightarrow E') = \sum_{k=1}^{NK} p_k(E)\ f_k(E \rightarrow E'),$$

and at a particular incident neutron energy E,

$$\sum_{k=1}^{NK} p_k(E) = 1,$$

where $p_k(E)$ is the fractional probability that the distribution $f_k(E \rightarrow E')$ can be used at E.

The partial energy distributions $f_k(E \rightarrow E')$ are represented by various analytical formulations. Each formulation is called an energy distribution law and has an identification number associated with it (LF number).

File 5 Energy Distributions of Secondary Neutrons

(Tabulated Distributions)

The following quantities are defined

LF	is a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution). LF = 1 for tabulated distributions.
NR	is the number of interpolation ranges.
NP	is the number of incident energy points at which p(E) is given.
E_{int}	is the interpolation scheme for the incident neutron energy, E.
E'_{int}	is the interpolation scheme for the secondary neutron energies, E'.
g(E→E')	is the tabulated secondary energy distribution as a function of E' for a given E.
p(E)	is the fractional part of the particular cross section that can be described by the tabulated partial energy distribution at incident neutron energy, E.
NE	is the number of incident energy points at which tabulated distributions are given. (NE ≤ 200.)
E_i	is the i^{th} incident energy of the NE energy points given.
NF	is the number of secondary energy points in a tabulation. (NF ≤ 1000.)

File 5 Energy Distributions of Secondary Neutrons

(Tabulated Distributions)

LF = 1, Arbitrary tabulated function

(MAT, 5, MT/T , 0.0 , LT , LF=1 , NR , NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int})TAB2

(MAT, 5, MT/T , E_1 , LT , 0 , NR , NF/E'_{int}/

E'_1, g($E_1 \rightarrow E'_1$) , E'_2 , g($E_1 \rightarrow E'_2$) , E'_3 , g($E_1 \rightarrow E'_3$),

---

---, E'_{NF} , g($E_1 \rightarrow E'_{NF}$))TAB1

(MAT, 5, MT/T , E_2, , LT , 0 , NR , NF/E'_{int}/

E'_1 , g($E_2 \rightarrow E'_1$) , E'_2 , g($E_2 \rightarrow E'_2$) , E'_3 , g($E_2 \rightarrow E'_3$),

---

---, E'_{NF} , g($E_2 \rightarrow E'_{NF}$))TAB1

(MAT, 5, MT/T , E_{NE} , LT , 0 , NR , NF/E'_{int}/

E'_1 , g($E_{NE} \rightarrow E'_1$) , E'_2 , g($E_{NE} \rightarrow E'_2$) , E'_3 , g($E_{NE} \rightarrow E'_3$),

---

---, E'_{NF} , g($E_{NE} \rightarrow E'_{NF}$))TAB1

Note that the incident energy mesh for p(E) does not have to be the same as the E mesh used to specify the energy distributions. The interpolation scheme used between incident energy points, E, and between secondary energy points, E', should be linear-linear. T and LT refer to possible temperature (physical) dependence.

File 5 Energy Distributions of Secondary Neutrons

(Tabulated Distributions)

```
0.00000+ 0 0.00000+ 0          0          1          1          21315 5 16   2
         2          2                                            1315 5 16   3
1.27110+ 7 1.00000+ 0 2.00000+ 7 1.00000+ 0                     1315 5 16   4
0.00000+ 0 0.00000+ 0          0          0          1          41315 5 16   5
         4          2                                            1315 5 16   6
0.00000+ 0 1.27110+ 7          0          0          1          31315 5 16   7
         3          2                                            1315 5 16   8
1.00000+ 3 0.00000+ 0 2.00000+ 3 1.00000- 3 3.00000+ 3 0.00000+ 01315 5 16   9
0.00000+ 0 1.30000+ 7          0          0          1         391315 5 16  10
        39          2                                            1315 5 16  11
1.00000+ 4 0.00000+ 0 2.00000+ 4 1.66690- 6 3.00000+ 4 2.26239- 61315 5 16  12
                                (cards deleted)
6.70000+ 5 8.39554- 8 6.90000+ 5 7.07867- 8 6.90000+ 5 0.00000+ 01315 5 16  24
0.00000+ 0 1.50000+ 7          0          0          1         361315 5 16  25
        36          2                                            1315 5 16  26
1.00000+ 4 0.00000+ 0 2.00000+ 4 7.92549- 8 3.00000+ 4 1.16539- 71315 5 16  27
5.00000+ 4 1.86594- 7 7.00000+ 4 2.50996- 7 9.00000+ 4 3.10056- 71315 5 16  28
1.20000+ 5 3.89336- 7 1.50000+ 5 4.58321- 7 1.80000+ 5 5.17961- 71315 5 16  29
2.10000+ 5 5.69087- 7 2.40000+ 5 6.12509- 7 2.70000+ 5 6.48927- 71315 5 16  30
3.00000+ 5 6.79042- 7 3.40000+ 5 7.10458- 7 3.80000+ 5 7.32969- 71315 5 16  31
4.20000+ 5 7.47827- 7 5.00000+ 5 7.58632- 7 6.10000+ 5 7.42774- 71315 5 16  32
7.30000+ 5 6.99252- 7 8.50000+ 5 6.40423- 7 1.16000+ 6 4.70177- 71315 5 16  33
1.32000+ 6 3.88516- 7 1.50000+ 6 3.08015- 7 1.58000+ 6 2.76469- 71315 5 16  34
1.67000+ 6 2.44093- 7 1.78000+ 6 2.08785- 7 1.89000+ 6 1.77920- 71315 5 16  35
2.01000+ 6 1.48835- 7 2.13000+ 6 1.24063- 7 2.26000+ 6 1.01501- 71315 5 16  36
2.39000+ 6 8.27617- 8 2.46000+ 6 7.40573- 8 2.53000+ 6 6.62134- 81315 5 16  37
2.61000+ 6 5.82093- 8 2.69000+ 6 5.11208- 8 2.69000+ 6 0.00000+ 01315 5 16  38
0.00000+ 0 2.00000+ 7          0          0          1         411315 5 16  39
        41          2                                            1315 5 16  40
1.00000+ 4 0.00000+ 0 2.00000+ 4 1.27970- 8 3.00000+ 4 1.90430- 81315 5 16  41
5.00000+ 4 3.12330- 8 9.00000+ 4 5.44500- 8 1.20000+ 5 7.08750- 81315 5 16  42
1.60000+ 5 9.15250- 8 2.00000+ 5 1.10810- 7 2.40000+ 5 1.28780- 71315 5 16  43
3.10000+ 5 1.57280- 7 3.90000+ 5 1.85600- 7 4.60000+ 5 2.06990- 71315 5 16  44
5.40000+ 5 2.27920- 7 6.20000+ 5 2.45470- 7 7.00000+ 5 2.59960- 71315 5 16  45
7.80000+ 5 2.71720- 7 8.70000+ 5 2.82010- 7 1.05000+ 6 2.94710- 71315 5 16  46
1.25000+ 6 2.98970- 7 1.50000+ 6 2.93740- 7 1.79000+ 6 2.77950- 71315 5 16  47
2.09000+ 6 2.55290- 7 2.95000+ 6 1.81090- 7 3.42000+ 6 1.44150- 71315 5 16  48
3.69000+ 6 1.25310- 7 3.95000+ 6 1.08960- 7 4.22000+ 6 9.37900- 81315 5 16  49
4.50000+ 6 7.99400- 8 4.77000+ 6 6.82750- 8 5.05000+ 6 5.77750- 81315 5 16  50
5.34000+ 6 4.84450- 8 5.65000+ 6 3.99990- 8 5.87000+ 6 3.48500- 81315 5 16  51
6.10000+ 6 3.01290- 8 6.33000+ 6 2.60110- 8 6.58000+ 6 2.21370- 81315 5 16  52
6.84000+ 6 1.86900- 8 7.11000+ 6 1.56540- 8 7.39000+ 6 1.30060- 81315 5 16  53
7.69000+ 6 1.06460- 8 7.69000+ 6 0.00000+ 0                     1315 5 16  54
```

File 5 Energy Distributions of Secondary Neutrons

(Tabulated Distributions)

LF = 1, Arbitrary tabulated function:

$f(E \rightarrow E') = g(E \rightarrow E')$.

A set of incident energy points is given, E_i, and $g(E_i \rightarrow E')$ is tabulated as a function of E'.

File 5 Energy Distributions of Secondary Neutrons

(General Evaporation, Simple Fission, Evaporation)

The following quantities are defined

U	is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).
LF	is a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution).
E_{int}	is the interpolation scheme for the incident neutron energy, E.
p(E)	is the fractional part of the particular cross section that can be described by the partial energy distribution at the incident neutron energy, E.
NR	is the number of interpolation ranges.
NP	is the number of incident energy points at which p(E) is given.
NE	is the number of incident energy points at which $\theta(E)$ is given. ($NE \leq 200$.)
$\theta(E)$	is the effective nuclear temperature in eV as a function of E.
NF	is the number of secondary energy points in a tabulation. ($NF \leq 1000$.)
x_{int}	is the interpolation scheme for $x \left(x = \frac{E'}{\theta(E)}\right)$ if LF = 5.
g(x)	is the tabulation secondary energy distribution as a function of x if LF = 5.

File 5 Energy Distributions of Secondary Neutrons

(General Evaporation, Simple Fission, Evaporation)

LF = 5, General evaporation spectrum

(MAT, 5, MT/ U , 0.0 , 0 , LF=5, NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR, NE/E_{int}/θ(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR, NF/x_{int}/g(x))TAB1 x = $\frac{E'}{\theta(E)}$

LF = 7, Simple fission spectrum (Maxwellian)

(MAT, 5, MT/ U , 0.0 , 0 , LF=7, NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int}/θ(E))TAB1

LF = 9, Evaporation spectrum

(MAT, 5, MT/ U , 0.0, 0, LF=9,NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/θ(E))TAB1

File 5 Energy Distributions of Secondary Neutrons

(General Evaporation, Simple Fission, Evaporation)

```
-3.00000+ 7 0.00000+ 0          0          5          1          21395 5455   2
          2          2                                            1395 5455   3
 1.00000- 5 3.80000- 2 2.00000+ 7 3.80000- 2                     1395 5455   4
 0.00000+ 0 0.00000+ 0          0          0          1          21395 5455   5
          2          2                                            1395 5455   6
 1.00000- 5 1.00000+ 0 2.00000+ 7 1.00000+ 0                     1395 5455   7
 0.00000+ 0 0.00000+ 0          0          0          2          281395 5455  8
          2          2         28          1                     1395 5455   9
 0.00000+ 0 0.00000+ 0 7.97300+ 4 2.70039- 6 8.85900+ 4 2.77519- 61395 5455  10
 9.84300+ 4 3.09089- 6 1.09370+ 5 3.25289- 6 1.21520+ 5 3.38999- 61395 5455  11
 1.35020+ 5 3.40659- 6 1.50020+ 5 3.37339- 6 1.66690+ 5 2.69209- 61395 5455  12
 1.85220+ 5 2.94959- 6 2.05800+ 5 3.07839- 6 2.28660+ 5 2.49269- 61395 5455  13
 2.54070+ 5 2.04399- 6 2.82300+ 5 1.28370- 6 3.13670+ 5 7.93498- 71395 5455  14
 3.48520+ 5 7.60258- 7 3.87240+ 5 1.04280- 6 4.30270+ 5 5.69159- 71395 5455  15
 4.78080+ 5 2.74189- 7 5.31200+ 5 2.16029- 7 5.90220+ 5 2.36799- 71395 5455  16
 6.55800+ 5 2.49269- 7 7.28660+ 5 3.28199- 7 8.09630+ 5 2.61729- 71395 5455  17
 8.99590+ 5 1.41250- 7 9.99540+ 5 7.89338- 8 1.11060+ 6 7.47788- 81395 5455  18
 1.23400+ 6 0.00000+ 0                                           1395 5455  19
```

```
-2.00000+ 7 0.00000+ 0          0          7          1          21396 5 18   2
          2          2                                            1396 5 18   3
 1.00000- 5 1.00000+ 0 2.00000+ 7 1.00000+ 0                     1396 5 18   4
 0.00000+ 0 0.00000+ 0          0          0          1          31396 5 18   5
          3          5                                            1396 5 18   6
 1.00000- 5 1.29550+ 6 1.50000+ 7 1.49230+ 6 2.00000+ 7 1.49447+ 61396 5 18   7
```

```
 5.30600+ 6 0.00000+ 0          0          9          1          21395 5 16   2
          2          2                                            1395 5 16   3
 5.32858+ 6 1.00000+ 0 2.00000+ 7 1.00000+ 0                     1395 5 16   4
 0.00000+ 0 0.00000+ 0          0          0          1          261395 5 16  5
         26          2                                            1395 5 16   6
 5.32858+ 6 1.00000+ 4 5.50000+ 6 5.20000+ 4 6.00000+ 6 1.82200+ 51395 5 16   7
 6.50000+ 6 2.35360+ 5 7.00000+ 6 2.78560+ 5 7.50000+ 6 3.15900+ 51395 5 16   8
 8.00000+ 6 3.49270+ 5 8.50000+ 6 3.79720+ 5 9.00000+ 6 4.07900+ 51395 5 16   9
 9.50000+ 6 4.34260+ 5 1.00000+ 7 4.59110+ 5 1.05000+ 7 4.82680+ 51395 5 16  10
 1.10000+ 7 5.05150+ 5 1.15000+ 7 5.26660+ 5 1.20000+ 7 5.47330+ 51395 5 16  11
 1.25000+ 7 5.67250+ 5 1.30000+ 7 5.86490+ 5 1.35000+ 7 6.05120+ 51395 5 16  12
 1.40000+ 7 6.23190+ 5 1.45000+ 7 6.40750+ 5 1.50000+ 7 6.57850+ 51395 5 16  13
 1.60000+ 7 6.86000+ 5 1.70000+ 7 7.14000+ 5 1.80000+ 7 7.36000+ 51395 5 16  14
 1.90000+ 7 7.51000+ 5 2.00000+ 7 7.57000+ 5                     1395 5 16  15
```

LF = 5, General evaporation spectrum:

$f(E \rightarrow E') = g\left(E'/\theta(E)\right)$.

$\theta(E)$ is tabulated as a function of incident neutron energy, E;

$g(x)$ is tabulated as a function of x, $x = E'/\theta(E)$.

LF = 7, Simple fission spectrum (Maxwellian):

$$f(E \rightarrow E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)} .$$

I is the normalization constant,

$$I = \theta^{3/2}\left[\frac{\sqrt{\pi}}{2} \operatorname{erf}\left(\sqrt{(E-U)/\theta}\right) - \sqrt{(E-U)/\theta}\ e^{-(E-U)/\theta}\right];$$

θ is tabulated as a function of energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

LF = 9, Evaporation spectrum:

$$f(E \rightarrow E') = \frac{E'}{I} e^{-E'/\theta}.$$

I is the normalization constant,

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta}\left(1 + \frac{E-U}{\theta}\right)\right];$$

θ is tabulated as a function of incident neutron energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

(Energy Dependent Watt Spectrum)

The following quantities are defined

U — is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

LF — is a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution).

NR — is the number of interpolation ranges.

NP — is the number of incident energy points at which $p_k(E)$ is given.

E_{int} — is the interpolation scheme for the incident neutron energy, E.

p(E) — is the fractional part of the particular cross section that can be described by the partial energy distribution at the incident energy, E

NE — is the number of incident energy points at which the parameters are given. ($NE \leq 200$.)

a,b — are parameters used in the energy dependent Watt spectrum. (LF = 11.)

File 5 Energy Distributions of Secondary Neutrons

(Energy Dependent Watt Spectrum)

LF = 11, Energy dependent Watt spectrum

(MAT, 5, MT/ U , 0.0, 0, LF=11,NR, NP/E_{int}/p(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/a(E))TAB1

(MAT, 5, MT/0.0 , 0.0 , 0, 0 , NR, NE/E_{int}/b(E))TAB1

File 5 Energy Distributions of Secondary Neutrons

(Energy Dependent Watt Spectrum)

```
-3.00000+ 7 0.00000+ 0          0          11         1          21395 5 18    2
          2                     2                                 1395 5 18    3
 1.00000- 5 1.00000+ 0 2.00000+ 7 1.00000+ 0                      1395 5 18    4
 0.00000+ 0 0.00000+ 0          0          0          1          241395 5 18   5
         24                    2                                 1395 5 18    6
 1.00000- 5 9.88000+ 5 1.00000+ 6 9.88000+ 5 1.50000+ 6 9.88000+ 51395 5 18    7
 2.00000+ 6 9.91000+ 5 3.00000+ 6 9.96000+ 5 4.00000+ 6 1.00100+ 61395 5 18    8
 5.00000+ 6 1.02500+ 6 5.50000+ 6 1.03600+ 6 6.00000+ 6 1.02600+ 61395 5 18    9
 7.00000+ 6 9.63000+ 5 8.00000+ 6 9.66000+ 5 9.00000+ 6 9.92000+ 51395 5 18   10
 1.00000+ 7 1.01600+ 6 1.10000+ 7 1.03800+ 6 1.20000+ 7 1.05600+ 61395 5 18   11
 1.21991+ 7 1.05800+ 6 1.30000+ 7 1.03400+ 6 1.40000+ 7 1.02800+ 61395 5 18   12
 1.50000+ 7 1.04700+ 6 1.60000+ 7 1.07500+ 6 1.70000+ 7 1.10700+ 61395 5 18   13
 1.80000+ 7 1.13200+ 6 1.90000+ 7 1.14700+ 6 2.00000+ 7 1.15800+ 61395 5 18   14
 0.00000+ 0 0.00000+ 0          0          0          1          241395 5 18   15
         24                    2                                 1395 5 18    16
 1.00000- 5 2.24900- 6 1.00000+ 6 2.24900- 6 1.50000+ 6 2.24900- 61395 5 18    17
 2.00000+ 6 2.23200- 6 3.00000+ 6 2.21000- 6 4.00000+ 6 2.18800- 61395 5 18    18
 5.00000+ 6 2.08800- 6 5.50000+ 6 2.04600- 6 6.00000+ 6 2.08600- 61395 5 18    19
 7.00000+ 6 2.36700- 6 8.00000+ 6 2.35500- 6 9.00000+ 6 2.23400- 61395 5 18    20
 1.00000+ 7 2.13200- 6 1.10000+ 7 2.04300- 6 1.20000+ 7 1.97600- 61395 5 18    21
 1.21991+ 7 1.97000- 6 1.30000+ 7 2.06200- 6 1.40000+ 7 2.08400- 61395 5 18    22
 1.50000+ 7 2.01000- 6 1.60000+ 7 1.90600- 6 1.70000+ 7 1.79900- 61395 5 18    23
 1.80000+ 7 1.72200- 6 1.90000+ 7 1.67600- 6 2.00000+ 7 1.64400- 61395 5 18    24
```

File 5 Energy Distributions of Secondary Neutrons

(Energy Dependent Watt Spectrum)

LF = 11, Energy dependent Watt spectrum:

$$f(E \rightarrow E') = \frac{e^{-E'/a}}{I} \sinh(\sqrt{bE'}).$$

I is the normalization constant,

$$I = \frac{1}{2}\sqrt{\frac{\pi a^3 b}{4}} \exp\left(\frac{ab}{4}\right)\left[\operatorname{erf}\left(\sqrt{\frac{E-U}{a}} - \sqrt{\frac{ab}{4}}\right) + \operatorname{erf}\left(\sqrt{\frac{E-U}{a}} + \sqrt{\frac{ab}{4}}\right) - a \exp\left(-\left(\frac{E-U}{a}\right)\right) \sinh\left(\sqrt{b(E-U)}\right)\right]$$

a and b are energy dependent;

U is a constant introduced to define the proper upper limit for the final neutron energy such that $0 \leq E' \leq E - U$.

File 7 Scattering Law Data

There is only one section in File 7, but the format varies slightly, depending on whether temperature-dependent data are given. The following quantities are defined:

LAT is a flag indicating which temperature has been used to compute α and β.

LAT = 0, the actual temperature was used.

LAT = 1, the constant T_o = 0.0253 eV has been used.

NS is the number of non-principal scattering atom types. For most moderating materials there will be (NS + 1) types of atoms in the molecule ($NS \leq 3$).

NI is the total number of items in the B(N) list. NL = 6*(NS + 1).

B(N) is the list of constants. Definitions are given above (Section 7.1).

NR is the number of interpolation ranges for a particular parameter, either β or α.

NB is the total number of β values given.

NP is the number of α values given for each value of β for the first temperature described, NP is the number of pairs, α and $S(\alpha,\beta)$, given.

β_{int} and α_{int} are the interpolation schemes used (see Appendix E for interpolation formats).

The structure of a section is

(MAT, 7, MT/ZA, AWR, 0, LAT, 0, 0)HEAD

(MAT, 7, MT/0.0, 0.0, 0, 0, NI, NS/B(N))LIST

(MAT, 7, MT/0.0, 0.0, 0, 0, NR, NB/β_{int})TAB2

(MAT, 7, MT/T, β_1, LT, 0, NR, NP/α_{int}/S(α,β_1))TAB1

(MAT, 7, MT/T, β_2, LT, 0, NR, NP/α_{int}/S(α,β_2))TAB1

(MAT, 7, MT/T, β_{NB}, LT, 0, NR, NP/α_{int}/S(α,β_{NB}))TAB1

(MAT, 7, 0 /0.0, 0.0, 0, 0, 0, 0)SEND

T and LT refer to possible temperature dependence. If the scattering law data are completely specified by analytic functions (no principal scattering atom type, as indicated by B(1) = 0), tabulated values of $S_0(\alpha,\beta)$ are omitted and the TAB2 and TAB1 records are not given.


```
8.01600+ 3 1.58580+ 1          0          0          0          01276 7  4   1
0.00000+ 0 0.00000+ 0          0          0         12          11276 7  4   2
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01276 7  4   3
1.00000+ 0 3.74810+ 0 1.58580+ 1 0.00000+ 0 0.00000+ 0 0.00000+ 01276 7  4   4
0.00000+ 0 0.00000+ 0          0          0          0          01276 7  0   5
```

Inelastic scattering is represented by the thermal neutron scattering law, $S(\alpha,\beta,T)$, and is defined (for a moderating molecule) by

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \to E',\mu,T) = \sum_{n=0}^{NS} \frac{M_n \sigma_{bn}}{4\pi T} \frac{\sqrt{E'}}{E} e^{-\beta/2} S_n(\alpha,\beta,T) ,$$

where there are (NS + 1) types of atoms in the molecule (i.e., for H_2O, NS = 1) and

M_n is the number of atoms of the n^{th} type in the molecule,

T is the moderator temperature (°K),

E is the incident neutron energy (eV),

E' is the secondary neutron energy (eV),

β is the energy transfer, $\beta = (E' - E)/kT$,

α is the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$,

A_n is the mass of the n^{th} type atom, A_o is the mass of the principal scattering atom in the molecule,

σ_{bn} is the bound atom scattering cross section of the n^{th} type atom,

$$\sigma_{bn} = \sigma_{fn} \frac{A_n + 1}{A_n}^2$$

σ_{fn} is the free atom scattering cross section of the n^{th} type atom,

k is Boltzmann's constant, and

μ is the cosine of the scattering angle (in the lab system).

The following quantities are defined:

ZAP is the designation of the nuclide produced in the reaction (ZAP = 1000.0 * Z + A).

MATP is the material number for the reaction product (ZAP).

NS is the total number of states of the reaction product given.

LMF is the file number in which the multiplicity or cross section for this MT number will be found.

LIS is the state number of the target.

LISØ is the isomeric state number of the target.

LFS is the level number of the state of ZAP.

ELFS is the excitation energy of the state of ZAP produced in the interaction (in eV above ground state).

NØ is the flag to allow the identification of a reaction product which is radioactive in its ground state and for which the cross section will not be found in File 9 or 10 (NØ=1).

ND is the number of branches into which the nuclide ZAP decays.

HL is the half-life of the nuclide ZAP in seconds.

ZAN is the Z and mass of the next nuclide produced along the chain.

BR is the branching ratio for the production of that particular ZAN and level.

ER is the end-point energy of the particle or quantum emitted.

CT is the chain terminator which gives minimal information about the formation and decay of ZAN.

RTYP is the mode of decay using the same definitions specified in MT=457.

The structure of each section always starts with a HEAD record and ends with a SEND record. If NØ=0, the section is divided into subsections, which contain data for a particular final state of the reaction product (LFS).

The number of subsections NS is given on the HEAD record for the section. Each subsection consists of a single LIST record. The subsections are ordered by increasing value of LFS.

The structure of a section is:

(MAT, 8, MT/ZA, AWR, LIS, LISØ, NS, NØ)HEAD
<subsection for LFS_1>
<subsection for LFS_2>
<subsection for LFS_{NS}>

The structure of a subsection is:

(MAT, 8, MT/ZAP, ELFS; LMF, LFS ; 6*ND, MATP/
HL(1), RTYP(1), ZAN(1), BR(1), ER(1), CT(1)
HL(2), RTYP(2), ZAN(2), BR(2), ER(2), CT(2)
.
.
.
HL(ND), RTYP(ND), ZAN(ND), BR(ND), ER(ND), CT(ND)) LIST
(MAT, 8, 0/0.0, 0.0; 0, 0, 0, 0)SEND

If NØ=1, then the reaction gives rise to a significant product which is radioactive in its ground state, and the evaluator wishes only to identify the radioactive product. The evaluator must supply MF=8, MT=457 data elsewhere to describe the decay of the product. It is understood that the cross section for producing the radioactive product is to be found in MF=3 for the given MT.

For NØ=1, the structure of a section is:

(MAT, 8, MT /ZA, AWR, LIS, LISO, NS, NØ=1)HEAD
(MAT, 8, MT /ZAP, ELFS=0., LMF=3, LFS=0, 6*ND=0, MATP)CONT
(MAT, 8, 0 /0.0, 0.0, 0, 0, 0, 0)SEND


```
1.10230+ 4 2.27920+ 1          0          0          1          01311 8103   1
1.00230+ 4 0.00000+ 0          9          0         18       71031311 8103   2
3.76000+ 1 1.00000+ 0 1.10230+ 4 6.70000- 1 4.37500+ 6 1.00000+ 01311 8103   3
3.76000+ 1 1.00000+ 0 1.10230+ 4 3.20000- 1 3.93500+ 6 1.01000+ 01311 8103   4
3.76000+ 1 1.00000+ 0 1.10230+ 4 1.00000- 2 2.29900+ 6 1.02000+ 01311 8103   5
0.00000+ 0 0.00000+ 0          0          0          0          01311 8  0   6
```

```
9.22350+ 4 2.33025+ 2          0          0          1          11395 8 16   1
9.12340+ 4 0.00000+ 0          3          0          0       13941395 8 16   2
0.00000+ 0 0.00000+ 0          0          0          0          01395 8  0   3
```

For example, consider the nuclide (ZAP) formed via a neutron reaction (MT number) in a final state (LFS number); ZAP then decays to a level in ZAN; the level number is part of the CT indicator and includes nonisomeric states in the count. The following examples may help explain the use of CT:

CT = 1.00 ZAN was formed in the ground state which is stable.

CT = 1.06 ZAN was formed in the sixth excited state; the sixth state decayed to the ground state which is stable.

CT = 2.00 ZAN was formed in the ground state which is unstable. (No delayed gammas are associated with the formation and decay of this particular ZAN.) The next decay in the chain is specified under the RTYP.

CT = 2.11 ZAN was formed in the 11th excited state but the chain does not terminate with that ZAN. The next decay in the chain is specified under the RTYP.

It is readily apparent from the above that CT = "1." indicates that the chain terminates with that particular ZAN and CT = "2." means that one or more decays are involved before stability is reached. Note, however, that stability can be reached instantaneously upon occasion with the emission of one or more light particles. The hundreths digits of the CT (06) indicate the state (sixth) in which ZAN is formed.

As an example, consider MT = 102. Then RTYP = 1.44 would be interpreted as follows:

> The first two columns of the RTYP (1.) indicates β^- decay of ZAP; the third and fourth columns (44) indicate that the nucleus ZAN (formed in the β^- decay) then immediately emits two α particles.

For this example: $n + {}^{7}Li \rightarrow \gamma + {}^{8}Li$ (ground state)

$ZA({}^{7}Li) = 3.007E + 03$ $LIS({}^{7}Li) = 0$ $LIS\emptyset({}^{7}Li) = 0$

$ZAP({}^{8}Li) = 3.008E + 03$ $LFS({}^{8}Li) = 0$

$ZAN({}^{8}Be) = 4.008E + 03$ $CT({}^{8}Be) = 2.01$

Since ^{8}Be has a half-life of the order of compound-nucleus formation times, decay data for MT = 457 are not required, and the complete chain can easily be represented and read from the information given here.

The following quantities are defined

NFP is the number of fission product nuclide states to be specified at each incident energy point (this is actually the number of sets of fission product identifiers - fission product yields). (NFP $\leq$ 1250.)

ZAFP is the (Z,A) identifier for a particular fission product. (ZAFP = (1000.0* Z) + A).

FPS is the state designator (floating-point number) for the fission product nuclide (FPS = 0.0 means the ground state, FPS = 1.0 means the first excited state, etc.)

YI (MT=454) is the fractional (independent) yield for a particular fission product prior to particle decay.

DYI (MT=454) is the 1σ uncertainty in YI.

YC (MT=459) is the cumulative yield.

DYC (MT=459) is the 1σ uncertainty in YC.

$C_n(E_i)$ is the array of yield data for the i^{th} energy point. This array contains NFP sets of four parameters in the order ZAFP, FPS, YI, and DYI in MT=454 and ZAFP, FPS, YC, and DYC in MT=459.

NN is equal to 4*NFP, the number of items in the $C_n(E_i)$ array.

E_i is the incident neutron energy of the i^{th} point (eV).

LE is a test to determine whether energy-dependent fission product yields are given:

LE = 0 implies no energy-dependence (only one set of fission product yield data given);

LE > 0 means that (LE + 1) sets of fission product yield data are given at (LE + 1) incident neutron energies.

I_i is the interpolation scheme (see Appendix E) to be used between the E_{i-1} and E_i energy points.

The structure for a section is

(MAT, 8, MT/ZA, AWR, LE + 1, 0, 0, 0)HEAD

(MAT, 8, MT/E_1, 0.0, LE, 0, NN, NFP/$C_n(E_1)$)LIST

(MAT, 8, MT/E_2, 0.0, I , 0, NN, NFP/$C_n(E_2)$)LIST

(MAT, 8, MT/E_3, 0.0, I , 0, NN, NFP/$C_n(E_3)$)LIST

(MAT, 8, 0/0.0, 0.0, 0, 0, 0, 0)SEND,

where MT = 454 for independent yield data and MT = 459 for cumulative yield data. There are (LE + 1) LIST records.


```
9.22350+ 4 2.33025+ 2          3          0          0          01395 8454    1
2.53000- 2 0.00000+ 0          2          0       4612       11531395 8454    2
2.30660+ 4 0.00000+ 0 1.93964-18 1.24137-18 2.30670+ 4 0.00000+ 01395 8454    3
1.07809-19 6.89974-20 2.40660+ 4 0.00000+ 0 2.10951-13 1.35010-131395 8454    4
2.40670+ 4 0.00000+ 0 6.14862-14 3.93512-14 2.40680+ 4 0.00000+ 01395 8454    5
3.69036-16 2.36183-16 2.40690+ 4 0.00000+ 0 7.88967-18 5.04939-181395 8454    6
2.40700+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 2.50660+ 4 0.00000+ 01395 8454    7
2.28949-11 1.46527-11 2.50670+ 4 0.00000+ 0 2.54943-11 1.63164-111395 8454    8
                                (cards deleted)
7.01710+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 7.01720+ 4 0.00000+ 01395 8454  765
0.00000+ 0 0.00000+ 0 7.11690+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454  766
7.11690+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 0 7.11710+ 4 0.00000+ 01395 8454  767
0.00000+ 0 0.00000+ 0 7.11710+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 01395 8454  768
7.11720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 7.11720+ 4 1.00000+ 01395 8454  769
0.00000+ 0 0.00000+ 0 7.21710+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454  770
7.21720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0                    1395 8454  771
5.00000+ 5 0.00000+ 0          2          0       4644       11611395 8454  772
2.30660+ 4 0.00000+ 0 9.72948-17 6.22687-17 2.30670+ 4 0.00000+ 01395 8454  773
3.58361-18 2.29351-18 2.40660+ 4 0.00000+ 0 4.88771-12 3.12820-121395 8454  774
2.40670+ 4 0.00000+ 0 1.03953-12 6.65302-13 2.40680+ 4 0.00000+ 01395 8454  775
8.07638-14 5.16888-14 2.40690+ 4 0.00000+ 0 2.32879-16 1.49043-161395 8454  776
2.40700+ 4 0.00000+ 0 4.44285-18 2.84342-18 2.50660+ 4 0.00000+ 01395 8454  777
4.54796-10 2.91070-10 2.50670+ 4 0.00000+ 0 3.41847-10 2.18782-101395 8454  778
                                (cards deleted)
7.01720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 7.11690+ 4 0.00000+ 01395 8454 1541
0.00000+ 0 0.00000+ 0 7.11690+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 1542
7.11710+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 7.11710+ 4 1.00000+ 01395 8454 1543
0.00000+ 0 0.00000+ 0 7.11720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 1544
7.11720+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 0 7.21710+ 4 0.00000+ 01395 8454 1545
0.00000+ 0 0.00000+ 0 7.21720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 1546
1.40000+ 7 0.00000+ 0          2          0       4548       11371395 8454 1547
2.30660+ 4 0.00000+ 0 1.26557-15 8.09965-16 2.40660+ 4 0.00000+ 01395 8454 1548
1.93009-10 1.23527-10 2.40670+ 4 0.00000+ 0 2.00011-11 1.28007-111395 8454 1549
2.40680+ 4 0.00000+ 0 7.89042-13 5.04987-13 2.40690+ 4 0.00000+ 01395 8454 1550
7.63964-16 4.88937-16 2.40700+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 1551
2.50660+ 4 0.00000+ 0 5.98032- 8 3.82740- 8 2.50670+ 4 0.00000+ 01395 8454 1552
2.02011- 8 1.29287- 8 2.50680+ 4 0.00000+ 0 2.57014- 9 1.64489- 91395 8454 1553
                                (cards deleted)
7.01690+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 0 7.01700+ 4 0.00000+ 01395 8454 2298
3.94021-14 2.52173-14 7.01710+ 4 0.00000+ 0 6.99037-13 4.47384-131395 8454 2299
7.01720+ 4 0.00000+ 0 1.21006-11 7.74441-12 7.11690+ 4 0.00000+ 01395 8454 2300
0.00000+ 0 0.00000+ 0 7.11690+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 2301
7.11710+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 7.11710+ 4 1.00000+ 01395 8454 2302
0.00000+ 0 0.00000+ 0 7.11720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 2303
7.11720+ 4 1.00000+ 0 0.00000+ 0 0.00000+ 0 7.21710+ 4 0.00000+ 01395 8454 2304
0.00000+ 0 0.00000+ 0 7.21720+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8454 2305
0.00000+ 0 0.00000+ 0          0          0          0          01395 8  0 2306
```

The yields are given as fractional values at each energy, and normally the independent yields will sum to ≈ 2.0.

NSP is the total number of radiation types (STYP) for which spectral information is given (NSP may be zero).

$T_{\frac{1}{2}}$ is the half-life of the original nuclide (seconds).

$E_{"x"}$ is the average decay energy (eV) of "x" radiation.

SPI is the spin of the nuclide in its LIS state.

PAR is the parity of the nuclide in its LIS state ($\pm$1.0).

NDK is the total number of decay modes given (cannot be zero).

RTYP is the mode of decay of the nuclide in its LIS state.

RFS is the isomeric state flag for daughter nuclide.

Q is the total decay energy (eV) available in the corresponding decay process.

BR is the fraction of the decay of the nuclide in its LIS state which proceeds by the corresponding decay mode.

STYP is the decay radiation type

ER is the energy (eV) of radiation produced (E_{γ}, $E_{\beta-}$, $E_{e.c.}$, etc.)

RI is the intensity of radiation produced (relative units)
RP is the spectrum of the continuum component of the radiation.
TYPE is the type of transition for beta and electron capture
RICC is the total internal conversion coefficient (STYP=0.0 only)
RICK is the K-shell internal conversion coefficient (STYP=0.0 only)
RICL is the L-shell internal conversion coefficient (STYP=0.0 only)
RIS is the internal pair formation coefficient or positron intensity.

LCON is the continuum spectrum flag

NT is the number of entries given for each discrete energy (ER)

FC is the continuum spectrum normalization factor.

FD is the discrete spectrum normalization factor.

NER is the total number of tabulated discrete energies.

$\overline{ER}$ is the average decay energy of radiation produced

The structure of this section always starts with a HEAD record and ends with a SEND record. This section is divided into subsections as follows:

(MAT, 8,457/	ZA	AWR	LIS	LISØ	b	NSP	) HEAD
(MAT, 8,457/	$T_{½}$	$\Delta T_{½}$	b	b	6	b /	
	$\overline{E}$"β"	$W\overline{E}$"β"	$\overline{E}$"γ"	$\Delta\overline{E}$"γ"	$\overline{E}$"α"	$\overline{\Delta E}$"α"	) LIST

(MAT,8,457/ SPI	PAR	b	b	6*NDK	NDK /	
	$RTYP_1$	RFS_1	Q_1	ΔQ_1	BR_1	ΔBR_1
	.					
	.					
	.					
	$RTYP_{NDK}$	RFS_{NDK}	Q_{NDK}	ΔQ_{NDK}	BR_{NDK}	ΔBR) LIST

Repeat NSP times

(MAT,8,457/	b	STYP	LCON	b	6	NER / (omit if NSP=0)	
	FD	ΔFD	$\overline{ER}$	$\overline{\Delta ER}$	FC	ΔFC)	LIST
(MAT,8,457/	ER_1	ΔER_1	b	b	NT	b / (omit if LCON=1)	
	$RTYP_1$	$TYPE_1$	RI_1	ΔRI	RIS_1	ΔRIS_1	
	$RICC_1$	$\Delta RICC_1$	$RICK_1$	ΔRICK	$RICL_1$	$\Delta RICL_1$) LIST	
	ER_{NER}	ΔER_{NER}	b	b	NT	b /	
	$RTYP_{NER}$	$TYPE_{NER}$	RI_{NER}	ΔRI_{NER}	----	) LIST	
(MAT,8,457/	RTYP	0.0	b	b	NR	NP / (omit if LCON=0)	
	ER_{int}	/	RP(ER)	)		TABl	

(MAT,8,0/	b	b	b	b	b	b	) SEND.


```
9.22350+ 4 2.33025+ 2          0          0          0          41395 8457    1
2.22097+16 1.57784+13          0          0          6          01395 8457    2
2.41000+ 4 6.00000+ 2 1.70000+ 5 4.00000+ 3 4.47100+ 6 1.68000+ 51395 8457    3
3.50000+ 0-1.00000+ 0          0          0          6          11395 8457    4
4.00000+ 0 0.00000+ 0 4.67900+ 6 2.50000+ 3 1.00000+ 0 0.00000+ 01395 8457    5
0.00000+ 0 0.00000+ 0          0          0          6         421395 8457    6
1.00000- 2 0.00000+ 0 1.41000+ 5 4.00000+ 3 0.00000+ 0 0.00000+ 01395 8457    7
                                (cards deleted)
5.17900+ 5 2.00000+ 2          0          0          6          01395 8457  111
4.00000+ 0 0.00000+ 0 4.00000- 4 0.00000+ 0 0.00000+ 0 0.00000+ 01395 8457  112
0.00000+ 0 4.00000+ 0          0          0          6         121395 8457  113
1.00000- 2 0.00000+ 0 4.47100+ 6 1.68000+ 5 0.00000+ 0 0.00000+ 01395 8457  114
4.15400+ 6 3.00000+ 3          0          0          6          01395 8457  115
4.00000+ 0 0.00000+ 0 9.00000- 1 2.00000- 1 0.00000+ 0 0.00000+ 01395 8457  116
                                (cards deleted)
4.59800+ 6 3.00000+ 3          0          0          6          01395 8457  137
4.00000+ 0 0.00000+ 0 5.40000+ 0 5.00000- 1 0.00000+ 0 0.00000+ 01395 8457  138
0.00000+ 0 8.00000+ 0          0          0          6         261395 8457  139
1.00000- 2 0.00000+ 0 2.41000+ 4 6.00000+ 2 0.00000+ 0 0.00000+ 01395 8457  140
9.47750+ 3 1.00000+ 0          0          0          6          01395 8457  141
4.00000+ 0 0.00000+ 0 1.98000+ 1 1.40000+ 0 0.00000+ 0 0.00000+ 01395 8457  142
1.11200+ 4 1.40000+ 2          0          0          6          01395 8457  143
4.00000+ 0 0.00000+ 0 2.80000+ 0 4.00000- 1 0.00000+ 0 0.00000+ 01395 8457  144
                                (cards deleted)
1.80850+ 5 2.00000+ 1          0          0          6          01395 8457  189
4.00000+ 0 0.00000+ 0 2.36000- 1 9.00000- 3 0.00000+ 0 0.00000+ 01395 8457  190
1.81650+ 5 2.00000+ 1          0          0          6          01395 8457  191
4.00000+ 0 0.00000+ 0 3.60000- 1 4.00000- 2 0.00000+ 0 0.00000+ 01395 8457  192
0.00000+ 0 9.00000+ 0          0          0          6         111395 8457  193
1.00000- 2 0.00000+ 0 2.95000+ 4 1.30000+ 3 0.00000+ 0 0.00000+ 01395 8457  194
1.11180+ 4 1.00000+ 1          0          0          6          01395 8457  195
4.00000+ 0 0.00000+ 0 1.70000+ 0 5.00000- 1 0.00000+ 0 0.00000+ 01395 8457  196
                                (cards deleted)
1.05212+ 5 0.00000+ 0          0          0          6          01395 8457  213
4.00000+ 0 0.00000+ 0 1.59000+ 0 6.00000- 2 0.00000+ 0 0.00000+ 01395 8457  214
1.09650+ 5 0.00000+ 0          0          0          6          01395 8457  215
4.00000+ 0 0.00000+ 0 5.40000- 1 2.00000- 2 0.00000+ 0 0.00000+ 01395 8457  216
0.00000+ 0 0.00000+ 0          0          0          0          01395 8  0  217
```

File 8 Radioactive Decay Data (MT=457)

Decay modes defined:

RTYP	Mode of decay	
0.	γ	γ-ray (not used in 457)
1.	β^-	Beta decay
2.	e.c., (β^+)	Electron capture and/or positron emission
3.	IT	Isomeric transition (will in general be present only when the state being considered is an isomeric state)
4.	α	Alpha decay
5.	n	Neutron emission (not "delayed neutron decay" considered below)
6.	SF	Spontaneous fission
7.	p	Proton emission.
10.	-	Unknown origin.

Decay radiations defined:

STYP	Radiation type	
0.	γ	Gamma rays
1.	β^-	Beta rays
2.	e.c., (β^+)	Electron capture and/or positron emission
4.	α	Alpha particles
5.	n	Neutrons
6.	SF	Spontaneous fission fragments
7.	p	Protons
8.	e^-	"Discrete electrons"
9.	x	X-rays and annihilation radiation (photons not arising as transitions between nuclear states)

File 9 Multiplicities for Production of Radioactive Nuclides

LIS is an indicator to specify the level number of the target.

LFS is an indicator to specify the level number of the nuclide (ZAP) (as defined in MF=8) produced in the neutron reaction (MT number).

LFS = 0 the final state is the ground state.

LFS = 1 the final state is the first excited state.

LFS = 2 the final state is the second excited state.

.
.
.

LFS = 98 an unspecified range of final states.

Q is the reaction Q-value (eV).

NS is the number of final states for each MT for which multiplicities are to be given.

NR is the number of energy ranges. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range. (For details, see Section 0.4.3.)

Y(E) is the multiplicity for a particular reaction type at incident energy E(eV). Data are given for energy-multiplicity pairs.

The structure of a section is:

(MAT, 9, MT/ZA, AWR, LIS, 0, NS, 0) HEAD

<NS subsections, one for each value of LFS>

(MAT, 9, 0/0.0, 0,0, 0, 0, 0, 0) SEND

The structure of a subsection is:

(MAT, 9, MT/0.0, Q, 0, LFS, NR, NP/E_{int}/Y(E))TABl


```
2.80600+ 4 5.94159+ 1          0          0          2          07280 9103   1
0.00000+ 0-2.04110+ 6          0          0          1         227280 9103   2
        22         2                                              7280 9103   3
2.07545+ 6 1.00000+ 0 2.50000+ 6 1.00000+ 0 3.00000+ 6 2.00000- 17280 9103   4
3.50000+ 6 2.25000- 1 4.00000+ 6 2.80000- 1 4.50000+ 6 3.20000- 17280 9103   5
5.00000+ 6 3.49000- 1 6.00000+ 6 3.50000- 1 7.00000+ 6 3.57000- 17280 9103   6
8.00000+ 6 3.82000- 1 9.00000+ 6 4.08000- 1 1.00000+ 7 4.40000- 17280 9103   7
1.10000+ 7 4.65000- 1 1.20000+ 7 4.90000- 1 1.30000+ 7 5.22000- 17280 9103   8
1.40000+ 7 5.40000- 1 1.50000+ 7 5.62000- 1 1.60000+ 7 5.86000- 17280 9103   9
1.70000+ 7 6.10000- 1 1.80000+ 7 6.22000- 1 1.90000+ 7 6.30000- 17280 9103  10
2.00000+ 7 6.25000- 1                                          7280 9103  11
0.00000+ 0-2.09971+ 6          0          1          1         227280 9103  12
        22         2                                              7280 9103  13
2.07545+ 6 0.00000+ 0 2.50000+ 6 0.00000+ 0 3.00000+ 6 8.00000- 17280 9103  14
3.50000+ 6 7.75000- 1 4.00000+ 6 7.20000- 1 4.50000+ 6 6.80000- 17280 9103  15
5.00000+ 6 6.51000- 1 6.00000+ 6 6.50000- 1 7.00000+ 6 6.43000- 17280 9103  16
8.00000+ 6 6.18000- 1 9.00000+ 6 5.92000- 1 1.00000+ 7 5.60000- 17280 9103  17
1.10000+ 7 5.35000- 1 1.20000+ 7 5.10000- 1 1.30000+ 7 4.78000- 17280 9103  18
1.40000+ 7 4.60000- 1 1.50000+ 7 4.38000- 1 1.60000+ 7 4.14000- 17280 9103  19
1.70000+ 7 3.90000- 1 1.80000+ 7 3.78000- 1 1.90000+ 7 3.70000- 17280 9103  20
2.00000+ 7 3.75000- 1                                          7280 9103  21
0.00000+ 0 0.00000+ 0          0          0          0          07280 9  0  22
```

LIS is an indicator to specify the level number of the target.

LFS is an indicator to specify the level number of the nuclide (ZAP) produced in the neutron reaction (MT) number.

LFS = 0 the final state is the ground state.

LFS = 1 the final state is the first excited state.

LFS = 2 the final state is the second excited state.

.
.
.

LFS = 98 an unspecified range of final states.

Q is the reaction Q-value (eV).

NS is the number of final states for each MT for which cross sections are to be given.

NR is the number of energy ranges. A different interpolation scheme may be given for each range. (NR $\leq$ 20).

NP is the total number of energy points used to specify the data (NP $\leq$ 5000).

E_{int} is the interpolation scheme for each energy range. For details, see Section 0.4.3.).

$\sigma(E)$ is the cross section in barns for a particular reaction type at incident energy E (eV). Data are given for energy, cross-section pairs.

The structure of a section is:

(MAT, 10, MT/ZA, AWR, LIS, 0, NS, 0) HEAD

<NS subsections one for each LFS>

(MAT, 10, 0/0.0, 0.0, 0, 0, 0, 0) SEND

The structure of a subsection is:

(MAT, 10, MT/0.0, Q, 0, LFS, NR, $NP/E_{int}/\sigma(E)$) TAB1


```
 7.91970+ 4 1.95274+ 2          0          0          1          0137910 16    1
 0.00000+ 0-8.67300+ 6          0          6          1         15137910 16    2
         15          2                                            137910 16    3
 8.71741+ 6 0.00000+ 0 8.80000+ 6 3.88000- 3 9.00000+ 6 5.00000- 3137910 16    4
 9.50000+ 6 1.00000- 2 1.00000+ 7 2.20000- 2 1.10000+ 7 5.50000- 2137910 16    5
 1.20000+ 7 9.20000- 2 1.30000+ 7 1.30000- 1 1.40000+ 7 1.52000- 1137910 16    6
 1.50000+ 7 1.65000- 1 1.60000+ 7 1.70000- 1 1.70000+ 7 1.67000- 1137910 16    7
 1.80000+ 7 1.60000- 1 1.90000+ 7 1.47000- 1 2.00000+ 7 1.30000- 1137910 16    8
 0.00000+ 0 0.00000+ 0          0          0          0          0137910  0    9
```

File 10 Cross Sections for Production of Radioactive Nuclides D

File 12 Multiplicities and Transition Probabilities A

(Photon Multiplicities)

NK the number of discrete photons including the continuum.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

EG_k the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k \equiv 0.0$ should be used.

LP indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons where the photon energy EG'_k is given by

$$EG'_k = EG_k + \frac{AWR}{AWR+1} E_n .$$

LF the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

= 2, a discrete photon energy.

(Photon Multiplicities)

The structure of a section for LØ = 1 is

(MAT, 12, MT/ZA, AWR; LØ=1, b; NK, b)HEAD

(MAT, 12, MT/ b, b; b, b; NR, NP/E_{int}/Y(E))TAB1*

<subsection for k = 1>

<subsection for k = 2>

.
.
.

<subsection for k = NK>

(MAT, 12, 0/ b, b; b, b; b, b)SEND ,

and the structure of each subsection is

(MAT, 12, MT/EG_k, ES_k; LP, LF; NR, NP/E_{int}/y_k(E))TAB1 ,

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

(Photon Multiplicities)

```
9.22350+ 4 2.33025+ 2          1          0         30          0139512  4    1
0.00000+ 0 0.00000+ 0          0          0          1         29139512  4    2
        29          2                                                139512  4    3
1.30550+ 4 0.00000+ 0 1.30560+ 4 1.00000+ 0 4.00000+ 4 1.00000+ 0139512  4    4
5.22180+ 4 1.00000+ 0 8.23440+ 4 1.19210+ 0 1.03433+ 5 1.28190+ 0139512  4    5
1.50630+ 5 1.42470+ 0 1.71718+ 5 1.43780+ 0 1.97827+ 5 1.46350+ 0139512  4    6
2.25945+ 5 1.49120+ 0 2.96239+ 5 1.57010+ 0 3.34400+ 5 1.64930+ 0139512  4    7
3.68541+ 5 1.73610+ 0 3.95655+ 5 1.79790+ 0 4.15739+ 5 1.79140+ 0139512  4    8
4.28793+ 5 1.87680+ 0 4.75991+ 5 1.87530+ 0 5.00000+ 5 1.95490+ 0139512  4    9
5.35239+ 5 1.93480+ 0 6.00000+ 5 2.13210+ 0 6.54738+ 5 2.23330+ 0139512  4   10
7.00000+ 5 2.23460+ 0 7.74238+ 5 2.24560+ 0 8.00000+ 5 2.25100+ 0139512  4   11
9.00000+ 5 2.16840+ 0 1.00000+ 6 2.14020+ 0 1.09000+ 6 2.12850+ 0139512  4   12
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                    139512  4   13
7.71000+ 5 7.71000+ 5          0          2          1         7139512  4   14
        7          2                                                139512  4   15
7.74238+ 5 0.00000+ 0 8.00000+ 5 1.00000- 2 9.00000+ 5 7.11000- 2139512  4   16
1.00000+ 6 1.19000- 1 1.09000+ 6 1.23300- 1 1.09000+ 6 0.00000+ 0139512  4   17
2.00000+ 7 0.00000+ 0                                          139512  4   18
7.58000+ 5 7.71000+ 5          0          2          1         7139512  4   19
        7          2                                                139512  4   20
7.74238+ 5 0.00000+ 0 8:00000+ 5 3.00000- 3 9.00000+ 5 2.31000- 2139512  4   21
1.00000+ 6 3.82000- 2 1.09000+ 6 3.97000- 2 1.09000+ 6 0.00000+ 0139512  4   22
2.00000+ 7 0.00000+ 0                                          139512  4   23
                                (cards deleted)
3.90000+ 4 5.20000+ 4          0          2          1        26139512  4  231
        26          2                                                139512  4  232
5.22180+ 4 0.00000+ 0 8.23440+ 4 1.91700- 1 1.03433+ 5 2.03900- 1139512  4  233
1.50630+ 5 2.13200- 1 1.71718+ 5 1.97600- 1 1.97827+ 5 1.94000- 1139512  4  234
2.25945+ 5 1.86000- 1 2.96239+ 5 1.94400- 1 3.34400+ 5 1.83600- 1139512  4  235
3.68541+ 5 1.98400- 1 3.95655+ 5 1.75600- 1 4.15739+ 5 1.75100- 1139512  4  236
4.28793+ 5 1.77800- 1 4.75991+ 5 1.69100- 1 5.00000+ 5 1.84800- 1139512  4  237
5.35239+ 5 1.69600- 1 6.00000+ 5 1.82100- 1 6.54738+ 5 1.69800- 1139512  4  238
7.00000+ 5 1.55500- 1 7.74238+ 5 1.43500- 1 8.00000+ 5 1.36500- 1139512  4  239
9.00000+ 5 1.23700- 1 1.00000+ 6 1.08000- 1 1.09000+ 6 1.13400- 1139512  4  240
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                    139512  4  241
1.30000+ 4 1.30000+ 4          0          2          1        29139512  4  242
        29          2                                                139512  4  243
1.30550+ 4 0.00000+ 0 1.30560+ 4 1.00000+ 0 4.00000+ 4 1.00000+ 0139512  4  244
5.22180+ 4 1.00000+ 0 8.23440+ 4 4.77800- 1 1.03433+ 5 4.44600- 1139512  4  245
1.50630+ 5 4.67000- 1 1.71718+ 5 4.57900- 1 1.97827+ 5 4.63300- 1139512  4  246
2.25945+ 5 4.66600- 1 2.96239+ 5 4.16700- 1 3.34400+ 5 4.34900- 1139512  4  247
3.68541+ 5 4.37300- 1 3.95655+ 5 4.08000- 1 4.15739+ 5 3.99900- 1139512  4  248
4.28793+ 5 4.09000- 1 4.75991+ 5 3.99100- 1 5.00000+ 5 3.89600- 1139512  4  249
5.35239+ 5 3.93700- 1 6.00000+ 5 4.04400- 1 6.54738+ 5 3.98000- 1139512  4  250
7.00000+ 5 4.13600- 1 7.74238+ 5 4.30600- 1 8.00000+ 5 4.36700- 1139512  4  251
9.00000+ 5 4.20000- 1 1.00000+ 6 3.83400- 1 1.09000+ 6 4.10600- 1139512  4  252
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                    139512  4  253
0.00000+ 0 0.00000+ 0          0          0          0         0139512  0  254
```

File 12 Multiplicities and Transition Probabilities

(Photon Multiplicities)

The multiplicity or yield $y_k(E)$ is defined by

$$y_k(E) = \frac{\sigma_k^{\gamma}(E)}{\sigma(E)} \quad \text{(photons)} \quad ,$$

where E designates neutron energy and $\sigma(E)$ is the neutron cross section in File 2 and/or File 3 to which the multiplicity is referred (by the MT number).

$$y_k(E_\gamma \leftarrow E) = y_k(E)\ f_k(E_\gamma \leftarrow E) \quad ,$$

which results in the requirement that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\, dE_\gamma = 1.$$

As a check quantity, the total yield

$$Y(E) = \sum_{k=1}^{NK} y_k(E) \quad \text{(photons)}$$

is also tabulated for each MT number if NK > 1.

File 12 Multiplicities and Transition Probabilities

(Photon Transition Probability Arrays)

LG = 1, simple case (all transitions are γ emission).

= 2, complex case (internal conversion or other competing processes occur).

NS number of levels below the present one, including the ground state. (The present level is also uniquely defined by the MT number and by its energy level.)

NT number of transitions for which data are given in a list to follow (i.e., number of nonzero transition probabilities), NT $\leq$ NS.

$\underline{ES}_i$ energy of the ith level, i = 0,1,2... NS. ($ES_o \equiv 0.0$, the ground state.)

$\underline{TP}_i$ $TP_{NS,i}$, the probability of a direct transition from level NS to level i, i = 0,1,2... (NS-1).

$\underline{GP}_i$ $GP_{NS,i}$, the probability that, given a transition from level NS to level i, the transition is a photon transition (i.e., the conditional probability of photon emission).

$\underline{B}_i$ an array of NT doublets or triplets depending on LG value.

Note that each level can be identified by its NS number. Then the energy of a photon from a transition to level i is given by $E_\gamma = ES_{NS} - ES_i$, and its multiplicity is given by $y(E_\gamma \leftarrow E) = (TP_i)\ (GP_i)$. It is implicitly assumed that the transition probability array is independent of incident neutron energy.

(Photon Transition Probability Arrays)

The structure of a section for LØ = 2 is

(MAT, 12, MT/ ZA, AWR; LØ=2, LG; NS, b)HEAD.

(MAT, 12, MT/ES_{NS}, b; LP, b; (LG+1)*NT, NT/B_i)LIST.

(MAT, 12 0/ b, b; b, b; b, b)SEND.

If LG = 1, the array B_i consists of NT doublets (ES_i,TP_i); if LG = 2, it consists of NT triplets (ES_i,TP_i,GP_i). Here the subscript i is a running index over the levels below the level for which the transition probability array is being given (i.e., below level NS). The doublets or triplets are given in decreasing magnitude of energy ES_i.

(Photon Transition Probability Arrays)

```
1.70000+ 4 3.51480+ 1          2          1          7          0114912 57   1
4.05800+ 6 0.00000+ 0          0          0          6          3114912 57   2
1.76200+ 6 1.80000- 1 1.22000+ 6 6.40000- 1 0.00000+ 0 1.80000- 1114912 57   3
0.00000+ 0 0.00000+ 0          0          0          0          0114912  0   4
```

File 12 Multiplicities and Transition Probabilities

(Photon Transition Probability Arrays)

The differential cross section for producing a γ-ray of energy E_γ resulting from the excitation of the m_o^{th} level of the residual nucleus and the subsequent transition between two definite levels ($j \rightarrow i$), including the effects of cascading from the $m_o - j$ levels higher than j, is

$$\frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) = \delta(E_\gamma - \varepsilon_j + \varepsilon_i) A_{j,i} \sigma_{m_o}(E) \sum_{\ell=j}^{m_o-j} \sum_{m_\ell=j}^{m_{\ell-1}-1} TP_{m_{\ell-1}, m_\ell} \quad (1)$$

where

$\sigma_{m_o}(E)$ = neutron cross sections for exciting the m_o^{th} level with neutron energy E,

$\delta(E_\gamma - \varepsilon_j + \varepsilon_i)$ = delta function with ε_j, ε_i being energy levels of the residual nucleus,

$TP_{k,\ell}$ = probability of the residual nucleus having a transition to the ℓ^{th} level given that it was initially in the excited state corresponding to the k^{th} level, and

$A_{k,\ell}$ = probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the k^{th} to the ℓ^{th} level.

File 13 Photon Production Cross Sections

NK the number of discrete photons including the continuum.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

EG_k the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k = 0.0$ should be used.

LP Indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons, where the photon energy is given by $EG_k + \frac{AWR}{AWR + 1} E_n$.

LF the photon energy distribution low number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

LF = 2, a discrete photon energy.

The structure of a section in File 13 is

(MAT, 13, MT/ZA, AWR; b, b; NK, b)HEAD

(MAT, 13, MT/ b, b; b, b; NR, NP/$E_{int}/\sigma^{\gamma}_{TOT}(E)$)TAB1*

<subsection for k = 1>

<subsection for k = 2>

.
.
.

<subsection for k = NK>

(MAT, 13, 0/ b, b; b, b; b, b)SEND

and the structure of each subsection is

(MAT, 13, MT/EG_k, ES_k; LP, LF; NR, NP/$E_{int}/\sigma^{\gamma}_k(E)$)TAB1 ,

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.


```
6.00000+ 3 1.18969+ 1          0          0          1          0130613 51   1
4.43900+ 6 4.43900+ 6          0          2          1        119130613 51   2
       119          2                                          130613 51   3
4.81200+ 6 0.00000+ 0 4.85000+ 6 8.00000- 3 4.90000+ 6 2.20000- 2130613 51   4
4.92000+ 6 2.80000- 2 4.93000+ 6 3.20000- 2 4.94000+ 6 3.50000- 2130613 51   5
4.95000+ 6 3.70000- 2 4.98000+ 6 4.70000- 2 5.00000+ 6 4.80000- 2130613 51   6
5.03000+ 6 4.70000- 2 5.10000+ 6 3.80000- 2 5.12000+ 6 3.60000- 2130613 51   7
5.15000+ 6 4.00000- 2 5.18000+ 6 4.50000- 2 5.20000+ 6 5.20000- 2130613 51   8
5.23000+ 6 6.60000- 2 5.28000+ 6 9.20000- 2 5.36000+ 6 1.48000- 1130613 51   9
5.37000+ 6 1.50000- 1 5.38000+ 6 1.49000- 1 5.43000+ 6 1.33000- 1130613 51  10
5.50000+ 6 1.24000- 1 5.55000+ 6 1.24000- 1 5.60000+ 6 1.28000- 1130613 51  11
5.65000+ 6 1.37000- 1 5.90000+ 6 1.97000- 1 6.05000+ 6 2.36000- 1130613 51  12
6.20000+ 6 2.52000- 1 6.25000+ 6 2.77000- 1 6.32000+ 6 3.40000- 1130613 51  13
6.34000+ 6 3.51000- 1 6.35000+ 6 3.49000- 1 6.36000+ 6 3.40000- 1130613 51  14
6.39000+ 6 2.88000- 1 6.41000+ 6 2.65000- 1 6.43000+ 6 2.55000- 1130613 51  15
6.45000+ 6 2.52000- 1 6.54000+ 6 2.72000- 1 6.56000+ 6 2.64000- 1130613 51  16
6.62000+ 6 2.00000- 1 6.64000+ 6 1.87000- 1 6.67000+ 6 1.75000- 1130613 51  17
6.75000+ 6 1.62000- 1 6.81000+ 6 1.56000- 1 6.92000+ 6 1.56000- 1130613 51  18
7.14000+ 6 1.67000- 1 7.18000+ 6 1.75000- 1 7.22000+ 6 1.92000- 1130613 51  19
7.25000+ 6 2.12000- 1 7.36000+ 6 3.10000- 1 7.42000+ 6 3.51000- 1130613 51  20
7.47000+ 6 3.48000- 1 7.54170+ 6 3.21560- 1 7.59370+ 6 3.09430- 1130613 51  21
7.66740+ 6 3.13760- 1 7.78870+ 6 3.85700- 1 7.81910+ 6 3.89170- 1130613 51  22
7.88840+ 6 3.58830- 1 7.93610+ 6 3.59700- 1 8.00000+ 6 3.77000- 1130613 51  23
8.01400+ 6 4.00000- 1 8.04400+ 6 4.50000- 1 8.10000+ 6 4.90000- 1130613 51  24
8.13800+ 6 4.60000- 1 8.16600+ 6 4.30000- 1 8.20000+ 6 4.00000- 1130613 51  25
8.24000+ 6 3.90000- 1 8.32000+ 6 3.45000- 1 8.42600+ 6 2.66000- 1130613 51  26
8.50000+ 6 2.45000- 1 8.61100+ 6 2.53000- 1 8.75000+ 6 2.60000- 1130613 51  27
8.83300+ 6 2.65000- 1 9.00000+ 6 2.80000- 1 9.04500+ 6 3.10000- 1130613 51  28
9.14900+ 6 3.14000- 1 9.25000+ 6 2.90000- 1 9.50000+ 6 2.80000- 1130613 51  29
9.52200+ 6 2.83000- 1 9.69200+ 6 3.22000- 1 9.75000+ 6 3.50000- 1130613 51  30
9.86800+ 6 3.40000- 1 1.00000+ 7 3.25000- 1 1.02500+ 7 3.15000- 1130613 51  31
1.05000+ 7 3.10000- 1 1.06900+ 7 3.30000- 1 1.07500+ 7 3.45000- 1130613 51  32
1.08300+ 7 3.65000- 1 1.10000+ 7 3.60000- 1 1.12500+ 7 3.35000- 1130613 51  33
1.15000+ 7 2.70000- 1 1.17500+ 7 2.60000- 1 1.19090+ 7 2.50000- 1130613 51  34
1.20000+ 7 2.39000- 1 1.22240+ 7 2.27000- 1 1.25990+ 7 2.14000- 1130613 51  35
1.30000+ 7 2.10000- 1 1.32500+ 7 2.05000- 1 1.35000+ 7 2.03000- 1130613 51  36
1.37480+ 7 1.95000- 1 1.40000+ 7 1.90000- 1 1.42500+ 7 1.80000- 1130613 51  37
1.45000+ 7 1.70000- 1 1.47500+ 7 1.65000- 1 1.48070+ 7 1.90000- 1130613 51  38
1.48630+ 7 2.15000- 1 1.49090+ 7 1.98300- 1 1.49540+ 7 1.81700- 1130613 51  39
1.50000+ 7 1.65000- 1 1.54770+ 7 1.45000- 1 1.59660+ 7 1.35000- 1130613 51  40
1.64430+ 7 1.25000- 1 1.69740+ 7 1.10000- 1 1.74670+ 7 1.00000- 1130613 51  41
1.80870+ 7 1.00000- 1 1.86320+ 7 9.90000- 2 1.90340+ 7 8.60000- 2130613 51  42
1.95110+ 7 7.00000- 2 2.00000+ 7 6.00000- 2                      130613 51  43
0.00000+ 0 0.00000+ 0          0          0          0          0130613  0  44
```

File 13 Photon Production Cross Sections

The representation of the energy dependence of the cross sections is accomplished by tabulating a set of neutron energy-cross section pairs $(E,\sigma_k^\gamma(E))$ for each discrete photon and for the photon energy continuum. For discrete photons, $\sigma_k^\gamma(E)$ is the photon production cross section (b) for the photon designated by k. For the photon continuum, $\sigma_k^\gamma(E)$ is the integrated (over photon energy) cross section for the photon continuum designated by k. In the continuum case,

$$\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E) = \sigma_k^\gamma(E)\ f_k\ (E_\gamma \leftarrow E)\ ,$$

which obviously requires that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\ dE_\gamma = 1\ .$$

Any time a continuum representation is used for a given MT number in File 13, the normalized energy distribution, $f_k(E_\gamma \leftarrow E)$, must be given in File 15 under the same MT number.

As a check quantity, the total photon production cross section,

$$\sigma_{TOT}^\gamma(E) = \sum_{k=1}^{NK} \sigma_k^\gamma(E) \qquad \text{(barns)}\ ,$$

is also tabulated for each MT number, unless only one subsection is present (i.e., NK = 1).

File 14 Photon Angular Distributions

(Isotropic Distribution for all Photons)

LI = 1: Isotropic Distribution

If LI = 1, then all photons for the reaction type (MT) in question are assumed to be isotropic. This is a flag that the processing code can sense, and thus needless isotropic distribution data are not entered in the file.

NK should be consistent with the NK value given in File 12 or 13 for the reaction type (MT).

(Isotropic Distribution for all Photons)

The section is composed of a HEAD card and a SEND card, as follows:

(MAT, 14, MT/ZA, AWR; LI=1, b; NK, b)HEAD

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

(Isotropic Distribution for all Photons)

```
9.22350+ 4 2.33025+ 2          1          0         30          0139514 4 1
0.00000+ 0 0.00000+ 0          0          0          0          0139514 0 2
```

File 14 Photon Angular Distributions

(Isotropic Distribution for all Photons)

File 14 Photon Angular Distributions

(Legendre Coefficient Representation)

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

LTT = 1, data are given as Legendre coefficients, where $a_o^k(E) \equiv 1.0$ is understood.

NK the number of discrete photons including the continuum (must equal the value given in File 12 or 13).

NI number of isotropic photon angular distributions given in a section.

EG_k the photon energy as given in File 12 or 13. For a continuous photon energy distribution, EG_k, = 0.0 should be used.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

NR the number of interpolation regions for the incident neutron energy.

NE number of neutron energy points given.

E_{int} the interpolation scheme of the incident neutron energy.

NL_i highest value of ℓ required at each neutron energy E_i.

$a_\ell(E)$ the ℓ^{th} Legendre coefficient at neutron energy E.

LTT = 1: Legendre Coefficient Representation

The structure of a section with LI = 0 and LTT = 1 is

(MAT, 14, MT/ZA, AWR; LI=0, LTT=1; NK, NI)HEAD

<subsection for k = 1>

<subsection for k = 2>

.
.
.

<subsection for k = NK>

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

The structure of each record in the first block of NI subsections, which is for the NI isotropic photons, is

(MAT, 14, MT/EG_k, ES_k; b, b; b, b)CONT .

There is just one CONT record for each isotropic photon. (The set of CONT records is empty if NI = 0.) The subsections are ordered in decreasing magnitude of EG_k (photon energy), and the continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons in decreasing magnitude of EG_k. The continuum, if present and anisotropic, appears last, with $EG_k \equiv 0.0$. The structure for the last NK-NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int})TAB2

(MAT, 14, MT/ b, E_1; b, b; NL_1, b/$a_\ell^k(E_1)$)LIST

(MAT, 14, MT/ b, E_2; b, b; NL_2, b/$a_\ell^k(E_2)$)LIST

.
.
.

(MAT, 14, MT/ b, E_{NE}; b, b; NL_{NE}, b/$a_\ell^k(E_{NE})$)LIST .

Note that lists of the $a_\ell^k(E)$ start at $\ell = 1$ because $a_o^k(E) \equiv 1.0$ is always understood.


```
8.01600+ 3 1.58580+ 1          0          1         13        11127614 4  1
8.87200+ 6 8.87200+ 6          0          0          0         0127614 4  2
7.11900+ 6 7.11900+ 6          0          0          0         0127614 4  3
4.94900+ 6 1.10800+ 7          0          0          0         0127614 4  4
```

(cards deleted for 5 isotropic photon distributions)

```
1.75300+ 6 8.87200+ 6          0          0          0         0127614 4 10
9.88000+ 5 7.11900+ 6          0          0          0         0127614 4 11
5.10000+ 5 6.05200+ 6          0          0          0         0127614 4 12
6.91700+ 6 6.91700+ 6          0          0          1        10127614 4 13
        10          2                                          127614 4 14
0.00000+ 0 6.43400+ 6          0          0          2         0127614 4 15
0.00000+ 0 0.00000+ 0                                         127614 4 16
0.00000+ 0 7.35300+ 6          0          0          2         0127614 4 17
0.00000+ 0 0.00000+ 0                                         127614 4 18
0.00000+ 0 7.93000+ 6          0          0          2         0127614 4 19
0.00000+ 0 2.00000- 1                                         127614 4 20
```

(cards deleted for 5 neutron energies)

```
0.00000+ 0 1.48000+ 7          0          0          2         0127614 4 31
0.00000+ 0 3.30000- 2                                         127614 4 32
0.00000+ 0 2.00000+ 7          0          0          2         0127614 4 33
0.00000+ 0 3.30000- 2                                         127614 4 34
6.13100+ 6 6.13100+ 6          0          0          1        12127614 4 35
        12          2                                          127614 4 36
0.00000+ 0 6.43400+ 6          0          0          6         0127614 4 37
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0127614 4 38
0.00000+ 0 6.51800+ 6          0          0          6         0127614 4 39
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0127614 4 40
0.00000+ 0 6.87000+ 6          0          0          6         0127614 4 41
0.00000+ 0 1.42000- 1 0.00000+ 0 2.50000- 2 0.00000+ 0-3.07000- 2127614 4 42
0.00000+ 0 7.50000+ 6          0          0          6         0127614 4 43
```

(cards deleted for 6 neutron energies)

```
0.00000+ 0 1.48000+ 7          0          0          2         0127614 4 57
0.00000+ 0 6.92000- 2                                         127614 4 58
0.00000+ 0 2.00000+ 7          0          0          2         0127614 4 59
0.00000+ 0 6.92000- 2                                         127614 4 60
0.00000+ 0 0.00000+ 0          0          0          0         0127614 0 61
```

The angular distributions are expressed as normalized probability distributions, that is,

$$\int_{-1}^{1} p_k(\mu,E)\,d\mu = 1 ,$$

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as a Legendre series expansion,

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} \quad (\underline{\Omega},E)$$

$$= \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} a_\ell^k(E)\ P_\ell(\mu) ,$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^{\gamma}(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

ℓ = order of the Legendre polynomial.

$\frac{d\sigma_k^{\gamma}}{d\underline{\Omega}}$ = differential photon production cross section in barns/steradian.

$a_\ell^k(E)$ = the ℓth Legendre coefficient associated with the discrete photon or photon continuum specified by k. (It is understood that $a_o^k(E) \equiv 1.0$.)

$$a_\ell^k(E) = \int_{-1}^{1} P_k(\mu,E)\ P_\ell(\mu)\,d\mu .$$

(Tabulated Angular Distributions)

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

LTT = 2, data are given as a tabulation.

NK the number of discrete photons including the continuum (must equal the value given in File 12 or 13).

NI number of isotropic photon angular distributions given in a section.

EG_k the photon energy as given in File 12 or 13. For a continuous photon energy distribution, EG_k = 0.0 should be used.

ES_k the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

NR the number of interpolation regions.

NE number of neutron energy points given in a TAB2 record.

E_{int} the interpolation scheme of the incident neutron energy.

NP the number of angular points (cosines) used to give the tabulated probability distributions for each energy (NP $\leq$ 101).

μ_{int} the interpolation scheme of μ.

p(μ,E) the tabulated probability as a function of μ for a given E.

LTT = 2: Tabulated Angular Distributions

The structure of a section for LI = 0 and LTT = 2 is

(MAT, 14, MT/ZA, AWR; LI=0, LTT=2; NK, NI)HEAD

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

(MAT, 14, 0/ b, b; b, b; b, b)SEND .

The structure of the first block of NI subsections (where NI may be zero) is the same as for the case of a Legendre representation; i.e., it consists of one CONT record for each of the NI isotropic photons in decreasing magnitude of EG_k. The continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

The structure of the first NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; b, b)CONT .

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons, again in decreasing magnitude of EG_k, with the continuum, if present and anisotropic, appearing last, with $EG_k \equiv 0.0$. The structure of the last NK-NI subsections is

(MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int})TAB2

(MAT, 14, MT/ b, E_1; b, b; NR, NP/μ_{int}/$p_k(\mu,E_1)$)TAB1

(MAT, 14, MT/ b, E_2; b, b; NR, NP/μ_{int}/$p_k(\mu,E_2)$)TAB1

.
.
.

(MAT, 14, MT/ b, E_{NE}; b, b; NR, NP/μ_{int}/$p_k(\mu,E_{NE})$)TAB1 .

NO EXAMPLE IN ENDF/B-V

The angular distributions are expressed as normalized probability distributions, that is,

$$\int_{-1}^{1} p_k(\mu,E)\,d\mu = 1 ,$$

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} \quad (\underline{\Omega},E)$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^{\gamma}(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

NC the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$

LF the flag for the energy distribution law;

LF = 1, tabulated energy distribution.

NR the number of interpolation regions

NP the number of energy points used to give p(E).

E_{int} the interpolation scheme of the incident neutron energy.

$p_j(E)$ the probability or weight given to the j^{th} partial distribution, $g_j(E_\gamma \leftarrow E)$.

NE the number of incident neutron energies at which $g(E_\gamma \leftarrow E)$ is given.

NP the number of photon energies at which $y(E_\gamma \leftarrow E)$ is given

$E_{\gamma int}$ the interpolation scheme for $g(E_\gamma \leftarrow E)$

$g_j(E_\gamma \leftarrow E)$ the j^{th} normalized partial distribution in the units eV^{-1}

*Note that the subscript k used in describing Files 12 and 13 has been dropped from $f(E_\gamma \leftarrow E)$. This is done because only one energy continuum is allowed for each MT number, and the subscript k has no meaning in File 15. It is, in fact, the NK^{th} subsection in File 12 or 13 that contains the production data for the continuum.

The structure of a section is

(MAT, 15, MT/ZA, AWR; b, b; NC, b)HEAD

<subsection for j = 1>

<subsection for j = 2>

.

.

.

<subsection for j = NC>

(MAT, 15, 0/ b, b; b, b; b, b)SEND .

For LF = 1, the structure of a subsection is

(MAT, 15, MT/b, b; b, LF=1; NR, NP/E_{int}/$p_j(E)$)TAB1

(MAT, 15, MT/b, b; b, b; NR, NE/E_{int})TAB2

(MAT, 15, MT/b, E_1; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_1)$)TAB1

(MAT, 15, MT/b, E_2; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_2)$)TAB1

.

.

.

(MAT, 15, MT/b, E_{NE}; b, b; NR, NP/$E_{\gamma\ int}$/$g(E_\gamma \leftarrow E_{NE})$)TAB1 .


```
9.22350+ 4 2.33025+ 2          0          0          1          0139515  3    1
0.00000+ 0 0.00000+ 0          0          1          1          2139515  3    2
         2          2                                                139515  3    3
1.00000- 5 1.00000+ 0 2.00000+ 7 1.00000+ 0                      139515  3    4
0.00000+ 0 0.00000+ 0          0          0          1          12139515  3    5
        12          2                                                139515  3    6
0.00000+ 0 1.00000- 5          0          0          1          32139515  3    7
        32          2                                                139515  3    8
0.00000+ 0 7.03591- 7 1.00000+ 5 1.09837- 6 2.00000+ 5 1.11995- 6139515  3    9
3.00000+ 5 1.02663- 6 4.00000+ 5 9.11723- 7 5.00000+ 5 7.96912- 7139515  3   10
6.00000+ 5 6.74838- 7 7.00000+ 5 5.59928- 7 8.00000+ 5 4.65213- 7139515  3   11
9.00000+ 5 3.84030- 7 1.00000+ 6 3.15880- 7 1.25000+ 6 2.06739- 7139515  3   12
1.50000+ 6 1.42171- 7 1.75000+ 6 1.01977- 7 2.00000+ 6 7.81988- 8139515  3   13
2.25000+ 6 5.95943- 8 2.50000+ 6 4.73571- 8 2.75000+ 6 3.66122- 8139515  3   14
3.00000+ 6 2.84540- 8 3.25000+ 6 2.20867- 8 3.50000+ 6 1.71122- 8139515  3   15
3.75000+ 6 1.32321- 8 4.00000+ 6 1.03469- 8 4.25000+ 6 8.18800- 9139515  3   16
4.50000+ 6 6.53647- 9 4.75000+ 6 5.17346- 9 5.00000+ 6 4.16862- 9139515  3   17
5.50000+ 6 2.64642- 9 6.00000+ 6 2.07933- 9 6.50000+ 6 1.07449- 9139515  3   18
7.00000+ 6 7.16325- 9 8.00000+ 6 0.00000+ 0                      139515  3   19
                                (cards deleted)
0.00000+ 0 1.42000+ 7          0          0          1          47139515  3   92
        47          1                                                139515  3   93
0.00000+ 0 8.12870- 7 3.00000+ 5 9.45530- 7 4.00000+ 5 1.02080- 6139515  3   94
5.00000+ 5 7.57360- 7 6.00000+ 5 6.11530- 7 7.00000+ 5 5.22160- 7139515  3   95
8.00000+ 5 4.37480- 7 9.00000+ 5 3.66920- 7 1.00000+ 6 2.96360- 7139515  3   96
                                (cards deleted)
4.50000+ 6 1.69350- 9 5.00000+ 6 1.12900- 9 5.50000+ 6 6.58580-10139515  3  107
6.00000+ 6 8.46740-10 6.50000+ 6 6.58580-10 7.00000+ 6 5.64490-10139515  3  108
7.50000+ 6 2.82250-10 8.00000+ 6 0.00000+ 0                      139515  3  109
0.00000+ 0 2.00000+ 7          0          0          1          47139515  3  110
        47          1                                                139515  3  111
0.00000+ 0 8.12870- 7 3.00000+ 5 9.45530- 7 4.00000+ 5 1.02080- 6139515  3  112
5.00000+ 5 7.57360- 7 6.00000+ 5 6.11530- 7 7.00000+ 5 5.22160- 7139515  3  113
8.00000+ 5 4.37480- 7 9.00000+ 5 3.66920- 7 1.00000+ 6 2.96360- 7139515  3  114
1.10000+ 6 2.82250- 7 1.20000+ 6 2.58730- 7 1.30000+ 6 2.25800- 7139515  3  115
1.40000+ 6 1.88160- 7 1.50000+ 6 1.74050- 7 1.60000+ 6 1.59940- 7139515  3  116
1.70000+ 6 1.50530- 7 1.80000+ 6 1.31720- 7 1.90000+ 6 1.17600- 7139515  3  117
2.00000+ 6 9.87860- 8 2.10000+ 6 8.93780- 8 2.20000+ 6 8.46740- 8139515  3  118
2.30000+ 6 7.05620- 8 2.40000+ 6 6.58580- 8 2.50000+ 6 6.11530- 8139515  3  119
2.60000+ 6 5.64490- 8 2.70000+ 6 4.65710- 8 2.80000+ 6 4.32780- 8139515  3  120
2.90000+ 6 4.04550- 8 3.00000+ 6 2.91650- 8 3.10000+ 6 2.72840- 8139515  3  121
3.20000+ 6 2.68130- 8 3.30000+ 6 2.49320- 8 3.40000+ 6 2.21090- 8139515  3  122
3.50000+ 6 1.74050- 8 3.60000+ 6 1.69350- 8 3.70000+ 6 1.88160- 8139515  3  123
3.80000+ 6 1.88160- 8 3.90000+ 6 1.36420- 8 4.00000+ 6 2.44610- 9139515  3  124
4.50000+ 6 1.69350- 9 5.00000+ 6 1.12900- 9 5.50000+ 6 6.58580-10139515  3  125
6.00000+ 6 8.46740-10 6.50000+ 6 6.58580-10 7.00000+ 6 5.64490-10139515  3  126
7.50000+ 6 2.82250-10 8.00000+ 6 0.00000+ 0                      139515  3  127
0.00000+ 0 0.00000+ 0          0          0          0          0139515  0  128
```

Only one distribution law is presently available (tabulated secondary photon energy distribution).

The energy distributions, $f(E_\gamma \leftarrow E)$, are in units of eV^{-1} and are normalized so that

$$\int_0^{E_\gamma^{max}} f(E_\gamma \leftarrow E)\, dE_\gamma = 1 \quad ,$$

where E_γ^{max} is the maximum possible secondary photon energy and its value depends on the incoming neutron energy as well as the particular nuclei involved. The energy distributions $f(E_\gamma \leftarrow E)$ can be broken down into the weighted sum of several different normalized distributions in the following manner:

$$f(E_\gamma \leftarrow E) = \sum_{j=1}^{NC} p_j(E)\, g_j(E_\gamma \leftarrow E) \qquad (eV)^{-1} \quad ,$$

The following normalization condition is imposed.

$$\int_0^{E_\gamma^{max}} g_j(E_\gamma \leftarrow E)\, dE_\gamma = 1 \ .$$

Thus,

$$\sum_{j=1}^{NC} p_j(E) = 1 \ .$$

The absolute energy distribution cross section, $\sigma^\gamma(E_\gamma \leftarrow E)$, can be constructed from the expression

$$\sigma^\gamma(E_\gamma \leftarrow E) = \sigma^\gamma(E)\, f(E_\gamma \leftarrow E) \qquad (b/eV),$$

where $\sigma^\gamma(E)$ is the integrated cross section for the continuum given either directly in File 13 or through the combination of Files 2, 3, and 12.

NIS is the number of isotopes in this material (NIS < 10).
ZAI is the (Z,A) designation for an isotope.
ABN is the abundance (weight fraction) of an isotope in this material.
EL is the lower energy limit of the energy range.
EH is the upper energy limit of the energy range.
LRF is a flag indicating which representation is used:
LRF=1, single-level B-W parameters.
LRF=2, multi-level B-W parameters.
SPI is the nuclear spin of the target nucleus, I (positive number).
AP is the scattering radius in units of 10^{-12} cm.
QX is an effective Q-value.
NLS is the number of sets of resonance parameters given.
A set of parameters is given for each ℓ-value (neutron orbital angular momentum (NLS $\leq$ 3).
L is the value of ℓ.
AWRI is the ratio of the mass of a particular isotope to that of a neutron.
NRS is the number of resonances for a given ℓ-value (NRS $\leq$ 500).

The symbols for the resonance parameters and the covariances of the resonance parameters are:

ER the resonance energy (in the laboratory system);
AJ the floating-point value of J (the spin, or total angular momentum of the resonance);
GT the resonance total width, Γ, evaluated at the resonance energy ER;
GN the neutron width, Γ_n, evaluated at the resonance energy ER;
GG the radiation width, Γ_γ, evaluated at the resonance energy ER;
GF the fission width, Γ_f, evaluated at the resonance energy ER;

DE^2 the variance of ER in units of eV squared;

DJ^2 the variance of AJ;
DJDN the covariance of AJ and of GN in units of eV;

DN^2 the variance of GN in units of eV squared;
DJDG the covariance of AJ and of GG in units of eV;
DNDG the covariance of GN and of GG in units of eV squared;

DG^2 the variance of GG in units of eV squared;
DJDF the covariance of AJ and of GF in uits of eV;
DNDF the covariance of GN and of GF in units of eV squared;
DGDF the covariance of GG and of GF in units of eV squared;

DF^2 the variance of GF in units of eV squared.

The general structure of File 32 is as follows:

(MAT, 32, 151/ ZA, AWR; 0, 0; NIS, 0) HEAD

(MAT, 32, 151/ ZAI, ABN; 0, 0; 1, 0) CONT (isotope)

(MAT, 32, 151/ EL, EH; 1, LRF; 0, 0) CONT (range)

(MAT, 32, 151/ SPI, AP; 0, 0; NLS, 0) CONT

(MAT, 32, 151/ AWRI, QX; L, 0; 18*NRS, NRS/

ER_1, AJ_1, GT_1, GN_1, GG_1, GF_1,
DE^2_1, DN^2_1, $DNDG_1$, DG^2_1, $DNDF_1$, $DGDF_1$,
DF^2_1, $DJDN_1$, $DJDG_1$, $DJDF_1$, DJ^2_1, 0.0,
ER_2, AJ_2, GT_2, GN_2, GG_2, GF_2,
DE^2_2, DN^2_2, $DNDG_2$, DG^2_2, $DNDF_2$, $DGDF_2$,
DF^2_2, $DJDN_2$, $DJDG_2$, $DJDF_2$, DJ^2_2, 0.0,

.
.
.

ER_{NRS}, AJ_{NRS}, GT_{NRS}, GN_{NRS}, GG_{NRS}, GF_{NRS},
DE^2_{NRS}, DN^2_{NRS}, $DNDG_{NRS}$, DG^2_{NRS}, $DNDF_{NRS}$, $DGDG_{NRS}$,
DF^2_{NRS}, $DJDN_{NRS}$, $DJDG_{NRS}$, $DJDF_{NRS}$, DJ^2_{NRS}, 0.0)LIST.

(MAT, 32, 151/ ZAI, ABN; 0, 0; 1, 0) CONT (isotope)

(MAT, 32, 151/ EL, EH; 1, LRF; 0, 0) CONT (range)

<subsection for the last isotope>

.
.
.

(MAT, 32, 0/ 0.0, 0.0; 0, 0; 0, 0) SEND


```
9.42400+ 4 2.37992+ 2          0          0          1          0138032151     1
9.42400+ 4 1.00000+ 0          0          0          1          0138032151     2
5.00000- 1 1.10000+ 2          1          1          0          0138032151     3
0.00000+ 0 9.18400- 1          0          0          1          0138032151     4
2.37990+ 2 0.00000+ 0          0          0        162          9138032151     5
1.05800+ 0 5.00000- 1 3.55860- 2 2.28000- 3 3.33000- 2 6.00000- 6138032151     6
4.00000- 6 2.25000- 8 0.00000+ 0 4.00000- 6 0.00000+ 0 0.00000+ 0138032151     7
3.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151     8
2.04600+ 1 5.00000- 1 3.50800- 2 2.64500- 3 3.22000- 2 2.30000- 4138032151     9
1.00000- 4 8.10000- 9 0.00000+ 0 1.23000- 5 0.00000+ 0 0.00000+ 0138032151    10
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    11
3.83200+ 1 5.00000- 1 4.39700- 2 1.73600- 2 2.65000- 2 1.10000- 4138032151    12
4.00000- 4 2.50000- 7 0.00000+ 0 4.00000- 6 0.00000+ 0 0.00000+ 0138032151    13
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    14
4.16200+ 1 5.00000- 1 4.69100- 2 1.66900- 2 3.02000- 2 2.00000- 5138032151    15
4.00000- 4 1.00000- 6 0.00000+ 0 1.00000- 6 0.00000+ 0 0.00000+ 0138032151    16
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    17
6.66200+ 1 5.00000- 1 8.33900- 2 5.41700- 2 2.92000- 2 2.00000- 5138032151    18
2.50000- 3 1.00000- 6 0.00000+ 0 4.00000- 6 0.00000+ 0 0.00000+ 0138032151    19
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    20
7.27800+ 1 5.00000- 1 4.89700- 2 2.14500- 2 2.75000- 2 2.00000- 5138032151    21
2.50000- 3 2.50000- 7 0.00000+ 0 2.25000- 6 0.00000+ 0 0.00000+ 0138032151    22
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    23
9.07700+ 1 5.00000- 1 5.11770- 2 1.28500- 2 3.83000- 2 2.00000- 5138032151    24
3.60000- 3 9.00000- 8 0.00000+ 0 3.60000- 5 0.00000+ 0 0.00000+ 0138032151    25
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    26
9.25100+ 1 5.00000- 1 3.26400- 2 3.12000- 3 2.95000- 2 2.00000- 5138032151    27
3.60000- 3 2.25000- 8 0.00000+ 0 4.00000- 6 0.00000+ 0 0.00000+ 0138032151    28
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    29
1.05000+ 2 5.00000- 1 8.16800- 2 4.48600- 2 3.68000- 2 2.00000- 5138032151    30
4.90000- 3 2.25000- 6 0.00000+ 0 1.70000- 6 0.00000+ 0 0.00000+ 0138032151    31
4.00000-11 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0138032151    32
0.00000+ 0 0.00000+ 0          0          0          0          0138032  0     33
```

NL - the number of subsections within a section.

NC - the number for "NC-type" sub-subsections which follow the CONT record.

NI - the number of "NI-type" sub-subsections which follow the "NC-type" sub-subsections.

E1 and E2 - energy range where the cross sections given in the section (MAT,3,MT) were "derived" in terms of other "evaluated" cross sections given in the sections (MAT,3,MTI)s for LTY=0 and of ratio measuremrnts to "evaluated" cross sections given in section (MATS,3,MTS) for LTYY1,2, and 3.

NCI - the number of pairs of values in the array {CI, XMTI}.

{CI, XMTI} - pairs of numbers. The coefficient CI is associated in the pair with a value of MTI, given as a floating point number and indicated as XMTI.

NEI - the number of pairs of values in the array {EI,WEI}.

{EI,WEI} - pairs of numbers. EI is the energy and WEI is the weight (must be {E1,1.;E2,0.3} for ENDF/B-V).

LB - a flag whose numerical value determines the meaning of the numbers given in the arrays $\{E_k , F_k\}\{E_\ell , F_\ell\}$.

NP - the total number of pairs of numbers in the arrays $\{E_k , F_k\}\{E_\ell , F_\ell\}$.

LT - the number of pairs of numbers in the second array, $\{E_\ell , F_\ell\}$.

NT - the total number of entries in the two arrays $\{E_k\}$ and $\{F_{k,k'}\}$.

NE - the number of entries in the array $\{E_k\}$ defining (NE-1) energy intervals.

LS - a flag indicating whether the $F_{k,k'}$ matrix is symmetric or not:

LS=0 Asymmetric matrix

Each section of File 33 starts with a HEAD record, ends with a SEND record, and has the following structure:

(MAT, 33, MT/ZA, AWR; 0, 0; 0, NL) HEAD

<subsection for L = 1>

The structure of a subsection is:

(MAT, 33, MT/ 0.0, 0.0; MAT1, MT1; NC, NI) CONT

<sub-subsection for n_c = 1>

The structure of an "NC-type" sub-subsection with LTY=1,2 and 3 is:

(MAT, 33, MT/ 0.0, 0.0; 0, LTY; 0, 0) CONT

(MAT, 33, MT/ E1, E2; MATS, MTS; 2*NEI, NEI/ {EI, WEI}) LIST

<sub-subsection for n_c = NC>

<sub-subsection for n_i = 1>

For values of the LB flag from 0 to 4, the "NI-type" sub-subsection has the following structure:

(MAT,33,MT/0.0,0.0; LT, LB; 2*NP, NP/{E_k , F_k}{E_ℓ , F_ℓ}) LIST

<sub-subsection for n_i = NI>

<subsection for L = NL>

(MAT, 33, 0/0.0, 0.0; 0, 0, 0, 0) SEND

The structure of an "NC-type" sub-subsection with LTY=0 is:

(MAT, 33, MT/ 0, 0; 0, LTY=0; 0, 0) CONT

(MAT, 33, MT/ E1, E2: 0, 0; 2*NCI, NCI/ {CI, XMTI}) LIST

The structure of an LB=5 sub-subsection is:

(MAT, 33, MT/ 0.0, 0.0; LS, LB=5; NT, NE/ {E_k}{$F_{k,k'}$}) LIST.


```
9.22380+ 4 2.36006+ 2          0          0          0          2139833102   1
0.00000+ 0 0.00000+ 0          0        102          1          2139833102   2
0.00000+ 0 0.00000+ 0          0          1          0          0139833102   3
4.00000+ 3 2.00000+ 4       1305        107          4          2139833102   4
4.00000+ 3 1.00000+ 0 2.00000+ 4 0.00000+ 0                      139833102   5
0.00000+ 0 0.00000+ 0          0          1         32         16139833102   6
1.00000- 5 5.49000- 5 1.00000+ 0 0.00000+ 0 4.00000+ 3 0.00000+ 0139833102   7
2.00000+ 4 2.03000- 3 3.00000+ 4 9.00000- 4 4.00000+ 4 1.37000- 3139833102   8
6.00000+ 4 2.03000- 3 8.00000+ 4 2.50000- 3 1.00000+ 5 1.44000- 3139833102   9
2.00000+ 5 7.29000- 4 4.00000+ 5 8.12000- 4 6.00000+ 5 1.16000- 3139833102  10
8.00000+ 5 2.20000- 3 1.00000+ 6 1.00000- 2 2.00000+ 6 9.00000- 2139833102  11
2.00000+ 7 0.00000+ 0                                            139833102  12
0.00000+ 0 0.00000+ 0          0          1         12          6139833102  13
1.00000- 5 0.00000+ 0 4.00000+ 3 2.92000- 3 7.00000+ 3 1.68000- 3139833102  14
1.00000+ 4 2.21000- 3 2.00000+ 4 0.00000+ 0 2.00000+ 7 0.00000+ 0139833102  15
0.00000+ 0 0.00000+ 0       1305        107          1          0139833102  16
0.00000+ 0 0.00000+ 0          0          2          0          0139833102  17
4.00000+ 3 2.00000+ 4       1305        107          4          2139833102  18
4.00000+ 3 1.00000+ 0 2.00000+ 4 0.00000+ 0                      139833102  19
0.00000+ 0 0.00000+ 0          0          0          0          0139833  0  20
```

```
0.00000+ 0 0.00000+ 0          0          0          0          0139833  2   3
4.00000+ 3 1.00000+ 6          0          0         20         10139833  2   4
1.00000+ 0 1.00000+ 0-1.00000+ 0 1.80000+ 1-1.00000+ 0 5.10000+ 1139833  2   5
-1.00000+ 0 5.20000+ 1-1.00000+ 0 5.30000+ 1-1.00000+ 0 5.40000+ 1139833  2   6
-1.00000+ 0 5.50000+ 1-1.00000+ 0 5.60000+ 1-1.00000+ 0 5.70000+ 1139833  2   7
-1.00000+ 0 1.02000+ 2                                           139833  2   8
```

```
0.00000+00 0.00000+00          1          5        231         21139533102   3
0.10000-04 0.62500+00 0.10000+01 0.18000+01 0.50000+01 0.10000+02139533102   4
0.20000+02 0.40000+02 0.80000+02 0.20000+03 0.40000+03 0.10000+04139533102   5
                              (cards deleted)
0.22500-01 0.19000-01 0.18000-01 0.18000-01 0.11999-01 0.62500-01139533102  40
0.50000-01 0.45000-01 0.40002-01 0.16000+00 0.12000+00 0.96022-01139533102  41
0.36000+00 0.23999+00 0.47997+00                                 139533102  42
```

File 33 Covariance of Neutron Cross Sections

In the energy range E1 to E2, the cross sections in file (MAT,3,MT), written as $^{MAT}\sigma_{MT}$ (E), are obtained in terms of the cross sections in files (MAT,3,MTI), written as $^{MAT}\sigma_{MT_i}$ (E), as follows:

$$^{MAT}\sigma_{MT}(E) = \sum_{i=1}^{NCI} C_i * {}^{MAT}\sigma_{MT_i}(E).$$

Let X_i refer to the cross section in (MAT,3,MT) at energy E_i and Y_j refer to the cross section in (MAT1,3,MT1) at energy E_j. The contribution of the sub-subsection to the covariance matrix $COV(X_i,Y_j)$, having the units of "barns squared," described in the subsection, is defined as follows for the different values of LB:

LB=0 Absolute components only correlated within each E_k interval

$$COV(X_i,Y_j) = \sum_k P^{i;k}_{j;k} F_{xy,k}$$

LB=1 Fractional components only correlated within each E_k interval

$$COV(X_i,Y_j) = \sum_k P^{i;k}_{j;k} F_{xy,k} X_i Y_j$$

LB=2 Fractional components correlated over all E_k intervals

$$COV(X_i,Y_j) = \sum_{k,k'} P^{i;k}_{j;k'} F_{xy,k} F_{xy,k'} X_i Y_j$$

LB=3 Fractional components correlated over E_k and E_ℓ intervals

$$COV(X_i,Y_j) = \sum_{k,\ell} P^{i;k}_{j;\ell} F_{x,k} F_{yk\ell} X_i Y_j$$

LB=4 Fractional components correlated over all E_ℓ intervals within each E_k interval

$$COV(X_i,Y_j) = \sum_{K,\ell,\ell'} P^{i;k,\ell}_{j;k,\ell'} F_k F_{xy,\ell} F_{xy,\ell'} X_i Y_j.$$

For LB=0, 1 and 2 we have LT=0, i.e., only one E_k table. For LB=3 and LB=4 we have LT≠0, i.e., two E tables, the E_k and the E_ℓ tables.

The dimensionless operators P in the above definitions are defined in terms of the operator S as follows:

$$P^{i;k,\ell,\ldots}_{j;m,n,\ldots} \equiv S^k_i S^\ell_i \cdots S^m_j S^n_j \cdots ,$$

where

$S^k_i \equiv 1$ when the energy E_i is in the interval E_k to E_{k+1} of an E_k table,

$S^k_i \equiv 0$ when the energy E_i is outside the range of E_k to E_{k+1} of an E_k table.

LB=5 Relative covariance matrix components

$$COV(X_i,Y_j) = \sum_{k,k'} P^{i;k}_{j;k'} F_{xy;k,k'} X_i Y_j$$

A single list of energies $\{E_k\}$ is required to specify the energy intervals labeled by the indices k and k'. The numbers $F_{xy;k,k'}$ represent fractional components correlated over the energy intervals E_k and $E_{k'}$.

Example of Lumped Reactions

MT52-MT59 are lumped into MT851 and MT60-MT64 are lumped into MT852. The important level MT51 is treated individually. MT51, MT851, and MT852 are "evaluated," while MT4 is "derived" as the sum of these 3 reactions.

```
 6.00000+ 3 1.18969+ 1          0          0          0          1130633  4  101
 0.00000+ 0 0.00000+ 0          0          4          1          0130633  4  102
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  4  103
 4.81200+ 6 2.00000+ 7          0          0          6          3130633  4  104
 1.00000+ 0 5.10000+ 1 1.00000+ 0 8.51000+ 2 1.00000+ 0 8.52000+ 2130633  4  105
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  106
 6.00000+ 3 1.18969+ 1          0          0          0          1130633 51  107
 0.00000+ 0 0.00000+ 0          0         51          0          2130633 51  108
 0.00000+ 0 0.00000+ 0          0          1         20         10130633 51  109
 1.00000- 5 0.00000+ 0 4.81200+ 6 1.00000- 2 8.45000+ 6 2.50000- 3130633 51  110
 8.95000+ 6 1.00000- 2 1.10000+ 7 4.00000- 2 1.20000+ 7 4.00000- 2130633 51  111
 1.30000+ 7 2.25000- 2 1.40000+ 7 1.00000- 2 1.50000+ 7 4.00000- 2130633 51  112
 2.00000+ 7 0.00000+ 0                                             130633 51  113
 0.00000+ 0 0.00000+ 0          0          1          6          3130633 51  114
 1.00000- 5 0.00000+ 0 4.81200+ 6 2.50000- 3 2.00000+ 7 0.00000+ 0130633 51  115
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  116
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 52  117
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  118
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 53  119
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  120
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 54  121
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  122
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 55  123
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  124
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 56  125
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  126
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 57  127
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  128
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 58  129
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  130
 6.00000+ 3 1.18969+ 1          0        851          0          0130633 59  131
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  132
 6.00000+ 3 1.18969+ 1          0        852          0          0130633 60  133
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  134
 6.00000+ 3 1.18969+ 1          0        852          0          0130633 61  135
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  136
 6.00000+ 3 1.18969+ 1          0        852          0          0130633 62  137
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  138
 6.00000+ 3 1.18969+ 1          0        852          0          0130633 63  139
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  140
 6.00000+ 3 1.18969+ 1          0        852          0          0130633 64  141
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  142
 6.00000+ 3 1.18969+ 1          0          0          0          1130633851  143
 0.00000+ 0 0.00000+ 0          0        851          0          1130633851  144
 0.00000+ 0 0.00000+ 0          0          1          6          3130633851  145
 1.00000- 5 0.00000+ 0 8.29600+ 6 4.00000- 2 2.00000+ 7 0.00000+ 0130633851  146
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  147
 6.00000+ 3 1.18969+ 1          0          0          0          1130633852  148
 0.00000+ 0 0.00000+ 0          0        852          0          1130633852  149
 0.00000+ 0 0.00000+ 0          0          1          6          3130633852  150
 1.00000- 5 0.00000+ 0 1.43640+ 7 4.00000- 2 2.00000+ 7 0.00000+ 0130633852  151
 0.00000+ 0 0.00000+ 0          0          0          0          0130633  0  152
```

APPENDIX O

Format Differences Between Successive Versions of ENDF/B

Versions I and II

The following is a summary of the format differences between Version I and Version II ENDF/B data tapes. ENDF/B Version I was released in July 1968, and Version II was released in August 1970.

File 1 Changes

1. An index has been added to MT = 451. Each record in this index contains a file number (MF), reaction type number (MT), and the number of BCD card images required to specify the data for each section to be given for the material. The number of entries in the index is given by NXC, which is the last binary record (sixth field for BCD card-image format) of the HEAD record. Each index entry is given in a CONT record. These records immediately follow the Hollerith information.

2. The format for specifying induced reaction branching (MT = 453) has been extensively modified.

3. The format for specifying fission product yield data (MT = 454) has been modified to allow the specification of yield data for metastable states.

4. A new section has been defined to contain data for delayed neutrons from fission (MT = 455). See Section 5.1 of this report for details.

File 2 Changes

1. LRF, the test indicating the type of resolved resonance formula used, has been expanded to include

LRF = 1, single-level Breit-Wigner parameters are given;

= 2, multilevel Breit-Wigner parameters are given;

= 3, R-Matrix (Reich-Moore) multilevel parameters are given (added);

= 4, Adler-Adler multilevel resonance parameters are given (added).

2. All materials will contain a File 2. For those materials for which resolved and/or unresolved are not given, File 2 will contain the effective scattering radius, AP. See Section 2.1 for details.

3. The LIS test has been removed. This means that the elastic scattering cross section always must be calculated, using the resolved or unresolved resonance parameters.

4. The constant C (used in calculating the penetration factor) has been replaced by a quantity AWRI. AWRI is defined as the ratio of the mass of a particular isotope to that of a neutron.

5. A new option for specifying unresolved resonance parameters has been added. It is indicated by the test LRF = 2. This means that all average resonance parameters (level spacing, the width of an unspecified competitive reaction, and the reduced neutron, radiation, and fission widths) may be given as a function of incident neutron energy. Energy-dependent parameters may be given for each ℓ - J state. See Section 7.3 for details.

File 3 Changes

1. The reaction Q-value has been defined as the kinetic energy (in eV) released by a reaction (positive Q-values) or required for a reaction (negative Q-values). The threshold energy (negative Q only) is given by

$$E_{th} = \frac{AWR + 1.0}{AWR} |Q|,$$

where AWR is the atomic weight ratio given on the HEAD record.

2. The maximum number of allowed energy points per section has been increased from 2000 to 5000.

3. An initial-state indicator has been added to the HEAD record. This will allow the inclusion of cross section data for metastable states and thus more than one section may be given for the same reaction type (MT number).

File 4 (No Changes)

File 5 Changes

1. The definition of LF = 3 (discrete energy loss law) was changed to read

$$f(E \rightarrow E') = \delta\left[E' - \frac{A^2 \times 1}{(A \times 1)^2}\, E + \frac{A}{A+1}\right]\theta$$

where A = AWR and θ is the level excitation energy (positive value).

2. T and LT have been removed from the TAB1 records that contain p(E) for cases in which LF = 5, 7, or 9. A value, U, replaces T. U was introduced to define the proper upper limit for the secondary neutron energy distributions so that

$$0 \leq E' \leq E - U ,$$

where E', E, and U are given in the laboratory system. Further, the normalization constants for LF = 7 and LF = 9 have been redefined to account for the use of U.

3. LF = 2, 4, 6, and 8 have been deleted.

All Files

1. Certain reaction type (MT) numbers have been changed (see Appendix B for definitions):

Old MT Number	New MT Number
5	51
6	52

Old MT Number	New MT Number
7	53
8	54
9	55
10	56
11	57
12	58
13	59
14	60
15	91
27	No longer used
29	No longer used
51	61
52	62
53	63
-	-
-	-
-	-
80	90
109 (Not assigned)	109 (n,3α) cross section
455 (Not assigned)	455 Delayed neutrons from fission
700-799 (Not assigned)	700-799 Assigned (see Appendix B)

2. The format for specifying temperature-dependent data has been modified so that the data for the second (and higher) temperatures may be given at a lesser number of points than was given for the first temperature. See Appendix F for details.

Versions II and IV

The following is a summary of the format differences between Versions II and IV ENDF/B data tapes. ENDF/B Version IV was released about February 1975.

General

1. The energy range for all general-purpose materials is 10^{-5} eV to 20 MeV.

File 1

1. The formats for specifying radioactive decay were changed. Section MT = 453 was changed to include only production of radioactive nuclides, and Section MT = 457 was added to include radioactive decay data.
2. Provision was made for supplying data for the number of prompt neutrons per fission ($\bar{\nu}p$) in added Section MT = 456.

File 3

1. The energy mesh for the total cross section must include the energy meshes for partial cross sections.
2. Time sequential (n,2n) reactions are described by using Sections MT = 6-9 and MT = 46-49.
3. An LR flag was added to designate x in the (n,n'x) reactions when x is not a photon. In this case, the temperature field S (formerly T) is used to designate the Q-value or energy difference of the combined reactions.
4. Sections MT = 718, 738, 758, 778, 798, and MT = 719, 739, 759, 779, 799 are redefined to describe continuum levels for (n,x') reactions. MT = 718 describes the (n,p'_c) continuum cross sections as part of the (n,p) cross section and should be included in the total cross section. MT = 719 is used to describe a continuum cross section for exit

protons whose cross section is already represented in the total cross section by other reaction types.

Versions IV and V

The following is a summary of the format differences between Versions IV and V ENDF/B data tapes. ENDF/B Version V was released about June 1979.

File 1

1. The HEAD card of MT=451 has been changed. NXC, the number of dictionary entries, has been moved to the sixth field of the Hollerith LIST record of MT=451. Field 5 now contains NLIB, the library identifier, and Field 6 now contains NMOD, the material modification number.
2. Following the HEAD card of MT=451 is a new CØNT card which contains information about the excitation energy, stability, state number, and isomeric state number of the target nucleus.
3. In the LIST record of MT=451, the LDD and LFP flags have been abolished. The number of dictionary entries, NXC, is now in the sixth field of the first card in this LIST record.
4. The fourth field on each dictionary card in MT=451 is now used to indicate the modification status (MOD) for the section described by the card.
5. Radioactive decay data (MT=453 and 457) has been removed from File 1. Entirely new formats have been devised and the radioactive decay data is given in MF=8, MT=457.
6. The fission product yields section (MT=454) has been removed from File 1. Fission product yield information is now given in File 8 using new formats.
7. A new section to describe energy release in fission (MF = 1, MT = 458) has been implemented.

File 2

1. The Reich-Moore resonance parameter representation is no longer permitted in ENDF/B, only in ENDF/A.

File 3

1. Total "gas production" MT's have been defined for H(203), D(204), T(205), He-3(206), and He-4(207).
2. The non-elastic cross section (MT = 3) is now optional and no longer required since total gamma ray production must be entered in File 13 and never as multiplicities in File 12.

File 4

1. A simplified format using a new flag, LI, has been introduced to indicate that all angular distributions for an MT are all isotropic.

File 5

1. Only the distribution laws given for LF=1, 5, 7, 9, and 11 are now allowed. LF=11 is a new format for an energy dependent Watt spectrum.

File 8

1. Information may be given for any MT specifying a reaction in which the end product is radioactive. The MT section contains information about the end product and how it decays. Files 9 and 10 may be used to give the cross section for the production of the end product.
2. Fission product yield information is given under MT=454 and 459. The format has been modified to include the 1σ uncertainty of the yields. MT=454 is for the independent yields and MT=459 is for the cumulative yields.
3. The spontaneous radioactive decay data is given in MT=457. This is an entirely new format.

Files 9 and 10

1. Isomer production is described in the new File 9 or File 10. In File 9 the cross sections are obtained by the use of multiplicities. In File 10, the absolute cross section is given.

Files 17 and 18

1. Formats for time dependent photon production data files have been defined. They may be used in ENDF/A only.

Files 19, 20, 21, and 22

1. The electron production data files have been implemented.

Files 31, 32, and 33

1. The formats for data covariance files first introduced in Version IV have been extensively modified and expanded. They are now included in this document for the first time.

APPENDIX P

Summary of Important ENDF Rules

General

1. Cross sections for all significant reactions should be included.

2. The data in ENDF are specified over the entire energy range 10^{-5} eV to 20 MeV. It should be possible to determine values between tabulated points with use of the interpolation schemes provided.

3. All cross sections are in barns, all energies in eV, all temperatures in degrees Kelvin, and all times in seconds.

4. Summary documentation and unusual features of the evaluation should appear in the File 1 comments.

5. Threshold energies and Q-values must be consistent for all data presented in different files for a particular reaction.

File 2 - Resonance Parameters

1. Only one energy region containing resolved resonance parameters can be used, if needed.

2. The cross section from resonance parameters is calculated only within the energy range EL to EH, although some of the resonance parameters may lie outside the range.

3. Every ENDF Material has a File 2 even if no resonance parameters are given in order to specify the effective scattering radius.

4. In the unresolved resonance region interpolation should be done in cross section space and not in unresolved resonance parameter space. Any INT is allowed.

5. The Breit-Wigner single-level or multilevel formalisms should be used in the resolved resonance region unless experimental data prove that use of the other allowed formalisms is significantly better.

File 3 - Tabulated Cross Sections

1. All File 3 data are given in the laboratory system.

2. The total cross section MT = 1 is the sum of all partial cross sections and has an energy mesh that includes all energy meshes for partial cross sections.

(Exceptions MT = 26, 46-49, 719, 739, 759, 779, and 799 are not included in the MT = 1.)

3. The following relationships among MT numbers are expected to be satisfied if data are presented:

$$1 = 2 + 3$$

$$3 \text{ (or } 1 - 2) = 4 \text{ (or } 51+...91) + (6+...9+16) + 17 + 18 \text{ (or } 19+...21+38) + (22+...25) + (28+...37) + (102+...114)$$

$$4 = \text{sum } (51+...91)$$

$$18 = \text{sum } (19+...21) + 38$$

$$101 = \text{sum } (102+...114)$$

$$103 = \text{sum } (700+...718)$$

$$104 = \text{sum } (720+...738)$$

$$105 = \text{sum } (740+...758)$$

$$106 = \text{sum } (760+...778)$$

$$107 = \text{sum } (780+...798)$$

4. Threshold reactions begin at zero cross sections at the threshold energy.

Files 2 and 3

1. If there are resonance parameters in File 2, there are contributions to the total (MT = 1) and scattering (MT = 2) cross sections and to the fission (MT = 18) and capture (MT = 102) cross sections if fission and capture widths

are also given. These must be added to the File 3 Sections MT = 1, 2, 18, and 102 over the resonance region in order to obtain summation values for these cross sections.

2. The cross sections in File 3 for MT = 1, 2, 18, and 102 in the resonance region are used to modify the cross section calculated from the resonance formalisms, if necessary. The File 3 "background" may be positive or negative or even zero if no modifications are required. The summation cross section (File 2 + File 3) should be everywhere positive.

3. Double-value points (discontinuities) are allowed anywhere but are required at resonance region boundaries. A typical situation for MT = 1, 2, 18, and 102 in File 3 is a tabulated cross section from 10^{-5} to 1 eV, tabulated "background" to the cross sections calculated in the resolved resonance region between EL1 and EH1, tabulated "background" to the cross sections calculated in the unresolved region between EL2 = EH1 and EH2, and tabulated cross sections from EH2 to 20 MeV. Double-value points occur at EL1, EL2, and EH2.

4. The tabulated "background" used in File 3 to modify the cross sections calculated from File 2 should not be highly structured or represent a large fraction of the cross sections calculated from File 2. It is assumed that the "background" cross section is assumed to be at 0 Kelvin. (The "background" cross section is usually obtained from room temperature comparisions, but this should be unimportant if the "background" cross section is either small or slowly varying).

5. The generalized procedure for Doppler-broadening cross sections from Files 2 + 3 is to generate a pointwise cross section from the resolved resonance region on an appropriate energy mesh at 0°K and add it to File 3. This summation cross section can be kernel-broadened to a higher temperature.

File 4 - Neutron Angular Distributions

1. Only relative angular distributions, normalized to an integrated probability of unity, are given in File 4. The differential scattering cross section in barns per steradian is determined by multiplying File 4 values by the File 2 + File 3 summation scattering cross section σ_s divided by 2π.

2. Discrete channel angular distributions (e.g., MT = 2, 51-90,701...) should be given as Legendre coefficients in the center-of-mass system, with a maximum of 20 higher order terms, the last being even, in the expansion. If the angular distribution is highly structured and cannot be represented by a Legendre expansion, a tabular angular distribution in the CM system must be given.

3. When the elastic scattering is represented by Legendre coefficients, an energy-independent transformation matrix must be given to perform a CM to laboratory conversion.

4. Angular distributions for continuum and other reactions must be given as tabulated distributions in the Lab system.

5. The angular distribution, whether specified as a Legendre expansion or a tabulated distribution, must be everywhere positive.

6. Angular distribution data should be given at the minimum number of incident energy points that will accurately describe the energy variation of the distributions.

File 5 - Secondary Energy Distribution

1. Only relative energy spectra, normalized to an integrated probability of unity, are given in File 5. All spectra must be zero at the end points. The differential cross section in barns per eV is obtained by multiplying the File 5 values by the File 2 + File 3 cross section times its multiplicity (2 for the (n,2n) reaction).

2. While distribution laws 1, 3, 5, 7, 9, and 10 are allowed, distribution laws 3 and 5 are discouraged but can be used if others do not apply.

3. The sum of all probabilities for all laws used for a particular reaction must be unity at each incident energy.

4. The constant U must be specified, where applicable, to limit the energy range of emitted spectra to physical limits.

File 6

1. File 6 formats and procedures have been specified, but this File is not currently used in ENDF.

APPENDIX Q

Maximum Dimensions of Important ENDF Parameters

File	Section	Variable	Max	Definition
1	451	NCD	294	Card images containing Hollerith information
	452	NC	4	Polynomial terms in expansion of $\bar{\nu}$
	455	NCD	4	Polynomial terms in expansion of $\bar{\nu}_d$
	456	NCP	4	Polynomial terms in expansion of $\bar{\nu}_p$
2	151	NER	2	Energy ranges
	"	NIS	10	Isotopes
	"	NRS	500	Resonances per ℓ-state
	"	NLS	3	ℓ-states
	"	NE	250	Energy mesh in unresolved region
	"	AMUN	2	Degrees of freedom for neutron widths
	"	AMUF	4	Degrees of freedom for fission widths
3	All	NR	20	Interpolation ranges
	"	NP	5000	Mesh size
4	2	NL	21	Side dimension of transformation matrix
	All	NM	20	Higher order Legendre terms
	"	NE	500	Incident energies
	"	NP	101	Angular mesh size

File	Section	Variable	Max	Definition
5	All	NE	200	Incident energy mesh
	"	NF	1000	Final energy mesh
7	4	NS	3	Nonprincipal scattering atoms