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DATA FORMATS AND PROCEDURES FOR THE ENDF NEUTRON CROSS SECTION LIBRARY*

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BNL 50066 (ENDF-102), "ENDF/B - Specifications for an Evaluated Nuclear Data File for Reactor Applications," May 1966; Rev. July 1967.

NATIONAL NEUTRON CROSS SECTION CENTER

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DATA FORMATS AND PROCEDURES
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1. INTRODUCTION

1.1. ENDF System

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

The ENDF system was developed for the purpose of providing a unified format that could be used to store and retrieve evaluated sets of neutron cross sections. This system was designed to allow easy exchange of cross section information between various laboratories. The initial system that was established 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 and photon production data (photons produced by neutron interactions).

The basic data formats that were developed for the library are versatile enough to allow for accurate description of the cross sections considered for a wide incident neutron energy range (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. The formats are restrictive in that for any given neutron reaction mechanism, only a limited number of different representations are allowed.

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. Volume I (this volume) of the report describes the formats and procedures for neutron interaction data. Volume II describes formats and procedures for photon production (due to neutron interactions) and photon interaction

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

data. 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 this report, but the codes themselves are described in separate documents.

This report is organized in the following manner. The first several sections describe the general features of the ENDF libraries, the relationship between ENDF and CSISRS, and the relationship between ENDF and its secondary libraries. Section 5 describes the standard formats used in all record types. Understanding the contents of Section 5 will facilitate understanding the data formats that are given in Sections 6-12. The formats described in Sections 6-12 are for the binary mode of representing the data. BCD card-image formats are given in Appendix N of this report.

1.3. A and B Libraries

Two different evaluated data libraries are maintained at the National Neutron Cross Section Center (NNCSC). The ENDF/A library contains either complete or incomplete data sets (incomplete in the sense that the data set may be, for example, an evaluation of the fission cross section for U-235 in the energy range 100 keV to 15 MeV). The ENDF/A 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 to 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 to 2 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 2 of this report.

2. GENERAL FEATURES OF THE EVALUATED NUCLEAR DATA FILE

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." The parameterization and reduction of the data to tabular form produces an evaluated data set. If the written description of how a unique data set is prepared from the data sources is available, it is referred to as a documented evaluation. The ENDF format was developed to store the results of this process in a form suitable for automated retrieval for further processing.

2.2. A and B Libraries

The demands on an evaluated data file vary according to the user's applications. Regardless of 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 that the data file will contain information such as the angular and energy distributions for secondary neutrons. Generally, evaluators do not supply the data in this form. Rather, they supply the "bits and pieces" which, when put together, form a fully evaluated set of data for each material. Thus there is a need for two storage systems---one to contain the "bits and pieces" as they are available, and a second to contain complete sets of data for each material. These two systems have been designated as ENDF/A and ENDF/B, respectively. 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 alternate 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 as is required for reactor physics or shielding calculations). The ENDF/A library is, in effect, a system for compiling evaluated data sets.

The formats used for the ENDF/A and ENDF/B libraries are essentially 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 of this report. The differences in the formats for the ENDF/A and ENDF/B libraries are given in Appendix H.

2.3. Choices of Data

The data sets contained on the ENDF/B library are those chosen by the Cross Section Evaluation Working Group (CSEWG). 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 based on (1) availability of new and significant experimental results, (2) integral data testing showing 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.

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 6-12. Basically there are only four different types of binary records (see Section 5.2). Each record type has a very 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 using 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 5.3 of this report. The formats used for BCD card images 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 example data sets.

2.5. Systematizing of Data

The ordering of nuclear data for a particular material is described in Section 4.1 of this report. Integral cross section data (for example, the total cross section) may be represented by giving tabulated values of σ_T 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. It is important to 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.

It should be noted 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.

3. RELATIONSHIP OF THE ENDF TO OTHER DATA SYSTEMS

3.1. Experimental Data Libraries

The National Neutron Cross Section Center (NNCSC) maintains a library for experimentally measured neutron cross section data. This library is known as SCISRS (Sigma Center Information Storage and Retrieval System). The SCISRS data library system is being replaced by a new library system, CSISRS (Cross Section Information Storage and Retrieval System) for experimental data. CSISRS offers considerable advantages over SCISRS, among which are the following:

(a) The CSISRS library is more flexible

(b) More types of data may be stored

(c) More bibliographic information may be stored

(d) More details about each experiment may be stored

(e) Data may be added to, checked, corrected, and retrieved by using computer codes that have been written in standard FORTRAN.

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 converted to the ENDF formats for subsequent incorporation into either the ENDF/A or ENDF/B libraries.

Recently, Dunford, et al.[1], from Atomics International, have developed a system for automating much of the time-consuming parts of the evaluation process. This system, known as SCORE (SCISRS Conversion Routine), utilizes the concept of interactive computer graphics. Thus, by permitting man/computer interaction through computer graphics, this system will shorten the length of time required for the evaluation process. Since SCORE will allow the evaluator to make more detailed analyses of the cross sections, the quality of the evaluation process should be improved.

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 quite different 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-averaged cross sections are obtained for use in the neutronics calculations. The available processing codes are summarized in Appendix I. Figure 3-A shows an example of 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.

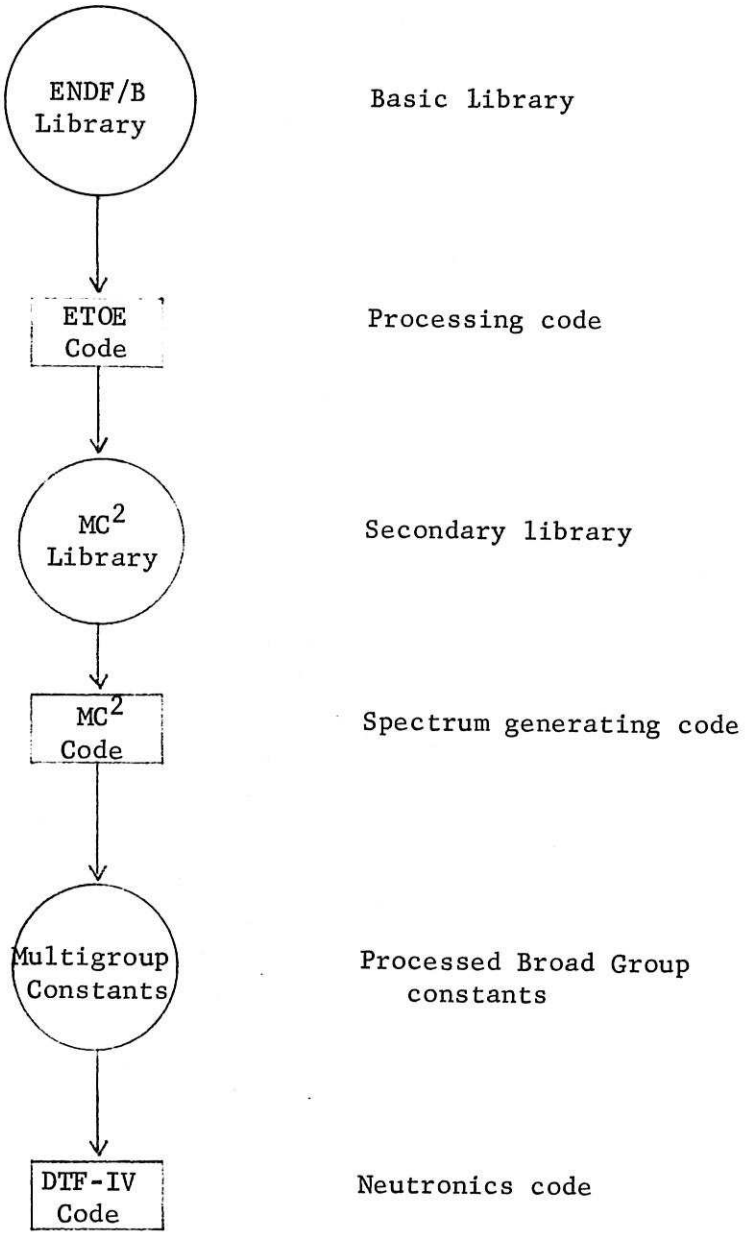

Fig. 3-A: Schematic of the flow of nuclear data from compilation to reactor calculation.

4. GENERAL DESCRIPTION OF THE ENDF LIBRARY

The ENDF library is a collection of documented data evaluations stored in a binary magnetic tape format that can be used as the main input into cross section processing programs. As such it is designed with the processing programs in mind and the reader of this report must be somewhat familiar with the FORTRAN programming language. The ordering of data on the tape allows the use of segmented as well as ordinary programs.

Punched cards are a nuisance, particularly when vast numbers of them are required, as in the case here. Unfortunately, it is not always possible to exchange data on magnetic tapes, particularly binary tapes. Therefore, 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.

4.1 Definitions and Conventions

A material is defined as either an isotope or a collection of isotopes. Therefore, 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 has a unique identification number assigned to it. These numbers are designated by the symbol MAT and they range from 1 to 9999. Two different evaluated sets of cross sections for U-235 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 in 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 National Neutron Cross Section Center, the center assignes a material number to this material. This number will be in the range 1000 to 9999. The assigned MAT number will be unique in that

it will never be assigned to another set of evaluated cross sections. The particular set of evaluated cross sections will retain this MAT number until significant modifications have been made to the data in the set. At this time, a new MAT number will be assigned. Material numbers from 1 - 999 are to be assigned by the user for data sets that he generates. As an example, consider the following sequence of events. User X evaluates a set of data for U-235 and assigns the material number 278 to this set. Within his installation the data set is always referred to as material 278. After a period of checking and testing, the user feels that the data set is satisfactory and transmits it to the NNCSC. The center adds the data set to its files and assigns it a MAT number of 4395. The center then issues a newsletter describing data received and available for distribution. User Y reads the newsletter and requests material 4395 from the centers files. Upon receipt of the data he adds it to his ENDF tape as material 4395 and refers to it in later processing programs by this number. Should user Y subsequently alter the data, he would assign a new material number between 1 and 999. The entire process might then start anew.

The evaluated data set for each material is divided into Files. These files are not physical files in the magnetic tape sense (i.e., there is no End-of-File mark at the end of each file). Each file contains data of a certain class. They are:

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
12	Multiplicities for photons (from neutron reactions)
13	Photon production cross sections (from neutron reactions)

(Con't on next page)

File Number (MF)	Class of Data
14	Angular distributions of photons (from neutron reactions)
15	Energy distributions of photons (from neutron reactions)
16	Energy-angular distributions of photons (from neutron reactions)
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)

The data formats and procedures for File 1 through 7 are described in this report. The formats and procedures for Files 12 through 27 are described in Volume II of this report.

Each file is divided into sections where each section contains 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 of this report.

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 U-238. 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, given the 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.

Parameter	Units
energies	electron volts
angles	dimensionless cosines of the angle
cross sections	barns
temperatures	^{o}Kelvin
mass	in units of the neutron mass
angular distributions	probability per unit cosine
energy distributions	probability per electron volt

4.2 Structure of an ENDF Data Tape

The structure of an ENDF binary tape is shown schematically in Figure 4.2.1. The structure of card deck or a BCD card image tape is exactly the same.

The tape contains a single record at the beginning which identifies the tape, and a single record at the end 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 is used or a BCD card image format is used. Every record on a tape contains three identification numbers: a material number (MAT), a file number (MF), and a reaction type number (MT). For a binary record, these numbers are given at the beginning of each record. For BCD card image records, these numbers are given in the last three fields of each record. These numbers are always in increasing numerical order, and the hierarchy is MAT, MF, and MT. The end of a section, file, and material are each signaled by special records.

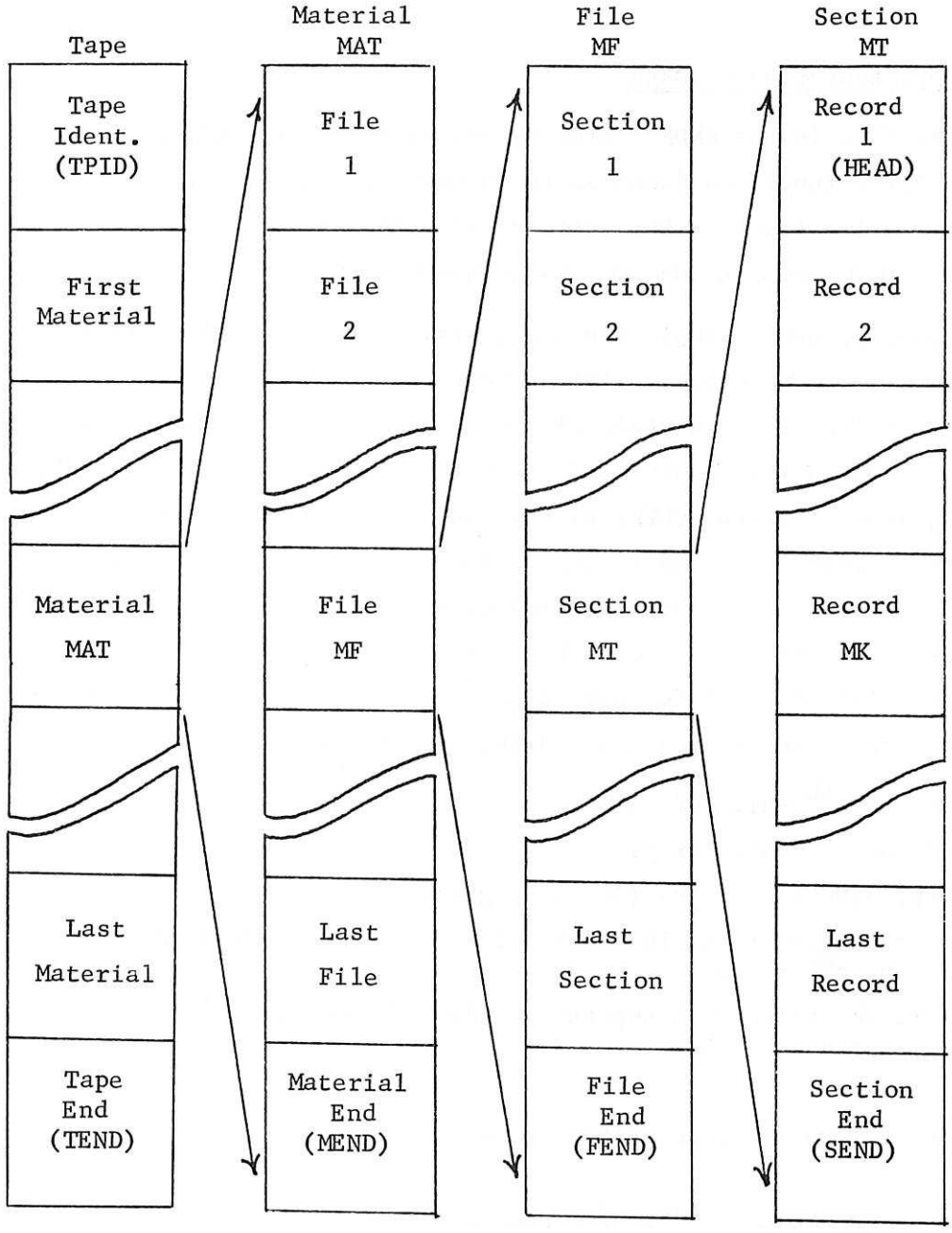

Figure 4.2.1. Arrangement of an ENDF Tape

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 the data 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 X (and Y(X)) given. The complete region over which x is defined is broken into NR interpolation ranges. An interpolation range is defined as a sequential series of 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. 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 ℓn x (linear-log)
4	ℓn y is linear in x (log-linear)
5	ℓn y is linear in ℓn x (log-log)

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

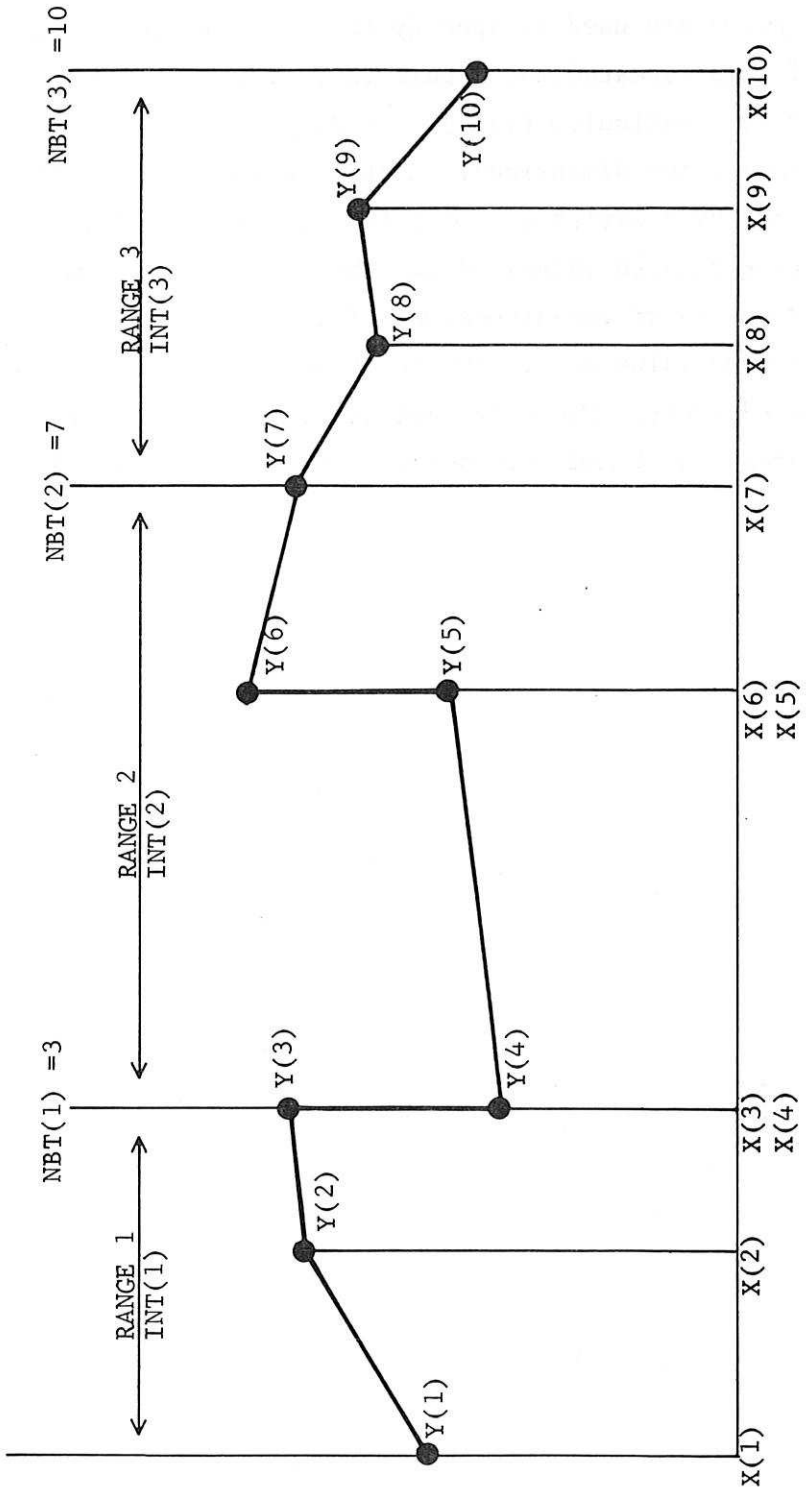

Figure 4.3.1 Tabulated one dimensional function illustrated for the case NP-10, NR-3

Note that in the case 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. 4.3.1).

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

5. GENERAL DESCRIPTION OF THE DATA FORMATS

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 symbols are frequently used and 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) which is contained in the same record

NR - The number of interpolation break points in a tabulation of y(x) which are contained in the same record

T - Temperature

E - Energy

μ - Cosine of an angle

LT - Temperature dependence (see Appendix F).

5.2 Types of Binary Records

All records on an ENDF binary tape are one of four possible types. These are denoted by CØNT, 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.

5.2.1 CØNT Records

The smallest possible record is a control (CØNT) record consisting of the nine numbers given above. For convenience, a CØNT record is denoted by:

[MAT, MF, MT/C1, C2; L1, L2; N1, N2]CØNT

The numbers contained within the brackets are symbolic of the numbers in a CØNT record. The semicolon punctuation is merely to remind the reader of the separation between floating point numbers,

test numbers, and counts. The slash punctuation is a reminder that the numbers MAT, MF, and MT appear in a different position in BCD card image records. The BCD card image format is described below in Section 5.3.

There are five special cases of a CØNT record, denoted by HEAD, SEND, FEND, MEND, and TEND. The HEAD record is the first record in a section and has the same form as a CØNT record. The numbers C1 and C2 are interpreted as ZA and AWR, respectively, on a HEAD record.

The SEND, FEND, MEND, and TEND records only use 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 IV statement to read any CØNT record from Tape LIB would be:

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

5.2.2 LIST Records

The second type of record is the LIST record, which is 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 IV 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.

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 IV 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 σ(E).

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 is used to specify how many values of Z are to be given and how to interpolate between successive value of Z. Tabulated values of Y(X) at each value of Z 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 IV 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:

$$[\text{MAT, MF, MT/C1, C2; L1, L2; NR, NZ/Z}_{\text{int}}]\text{TAB2.}$$

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.

5.3 Card Image (BCD) Formats

An alternate format is used when data are contained on punched cards or BCD card image tapes. Basically the data are stored in the same order for this format as was used in 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.

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	Material number (MAT)
8	71-72	File number (MF)
9	73-75	Reaction type (MT)
10	76-80	Sequence number, starting with 1 for the first card of a material

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(94)	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 IV 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 FØRMAT (2E11.4, 4I11, I4, I2, I3/(6I11))
      READ (INP, 20) (X(N), Y(N), N=1, NP)
20 FØRMAT (6E11.4)
```

It is obvious that a TAB2 record is the same as the TAB1 record, except that the list of x and y values is omitted. 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.

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

The FORTRAN IV 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 FØRMAT (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 IV "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 FØRMAT (2E11.4, 4I11, I4, I2, I3)
      NH = 17*NWD
      READ (INP, 50) (H(N), N=1, NH)
50 FORMAT (16A4, A2)
```

BCD card image formats are given in Appendix N. The following page illustrates how the four basic record types are punched. Fields 1-6 refer to the card Cols. 1-66 with 11 columns per field. Fields 7-10 (MAT, MF, MT, and sequence numbers) must also be punched but are omitted on these description sheets for convenience.

When arrays of numbers are punched, the first element of the array is in Field 1 (for example, X(1)). The last element may fall in any field, depending on how many values are in the array. Thus, the fact that X(NP) is shown in Field 6 should not be taken literally.

Illustration of standard record types								
Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Line	Type	Comments
C1	C2	L1	L2	N1	N2	1	CØNT	
ZA	AWR	L1	L2	N1	N2	2	HEAD	
C1	C2	L1	L2	NR	NP	3	TAB1	
NBT(1)	INT(1)	NBT(2)	INT(2)	NBT(3)	INT(3)	4		
NBT(4)	INT(4)	---	---	NBT(NR)	INT(NR)	5		
X(1)	Y(1)	X(2)	Y(2)	X(3)	Y(3)	6		
X(4)	Y(4)	---	---	X(NP)	Y(NP)	7		
C1	C2	L1	L2	NR	NP	8	TAB2	
NBT(1)	INT(1)	NBT(2)	INT(2)	NBT(3)	INT(3)	9		
NBT(4)	INT(4)	---	---	NBT(NR)	INT(NR)	10		
C1	C2	L1	L2	N1	N2	11	LIST	
B(1)	B(2)	B(3)	B(4)	B(5)	B(6)	12		
B(7)	B(8)	---	---	---	B(N1)	13		

Fig. 5.3 - A BCD Card Image Records

6. 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. File 1 consists of one or more sections that contain neutron cross section information and other related nuclear data. File 1 serves the purpose of providing a brief documentation of how the data were evaluated and a dictionary that summarizes the data files and cross-section types that are 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), the radioactive decay chains for the material, and the decay chains for the residual nuclei produced by neutron reactions with the material, fission product yield data (for fissile materials), and delayed neutron data (for fissile materials).

File 1 consists of at least one section and may contain as many as five sections for fissile materials. Each section in File 1 has been assigned an MT number (see below), and the sections are arranged in increasing MT numbers. Each section always starts with a HEAD record and ends with a SEND record. The end of File 1 (as well as all other files) is indicated by a FEND record. These record types are defined in detail in Section 5.2 of this report. 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, and

N2 is an integer to be used as a count of items in a second list to follow.

The symbolism used above to represent the HEAD record and to be used in the following format descriptions should be understood to mean that only data contents of each record are specified in the binary format. BCD card image formats for Files 1 to 7 are given in Appendix N.

6.1 Descriptive Data and Dictionary (MT = 451)

This section is always the first section of any material, and it is made up of 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 that were 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 66 characters (first card) should be a self-contained title for the material. The first card will be used to provide titles for listings and plots of the data for this material.

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.

NXC is an integer count of all of the sections to be found in the dictionary. Each section of this material is represented by a single card image in the dictionary. This card image contains the MF (file number), 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., equal to the sum of all the sections in the different files.

LDD is a flag to indicate whether radioactive decay data are given for this material:

LDD = 0, radioactive decay data not given for this material;

LDD = 1, radioactive decay data given.

LFP is a flag that indicates whether fission product yield data are given for this material:

LFP = 0, fission product yields not given;

LFP = 1, fission product yields are given.

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 $\leq$ 100). 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 $\leq$ 1700).

H(N) is the array that contains 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.

$\underline{MF}_n$, $\underline{MT}_n$, and $\underline{NC}_n$ are included in each of the NXC items contained in the dictionary.

$\underline{MF}_n$ is the File number (MF) of the n^{th} section.

$\underline{MT}_n$ is the reaction type number (MT) of the n^{th} section.

$\underline{NC}_n$ 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 + 2).

6.1.1 Formats

This section always begins with a HEAD record and ends with a SEND record. The structure of this section is

[MAT, 1, 451/ZA , AWR, LRP, LFI, 0, NXC]HEAD
[MAT, 1, 451/0.0, 0.0, LDD, LFP, NWD, 0/H(N)]LIST
[MAT, 1, 451/0.0, 0.0, MF_1, MT_1, NC_1, 0]CØNT
[MAT, 1, 451/0.0, 0.0, MF_2, MT_2, NC_2, 0]CØNT

[MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}, NC_{NXC}, 0]CØNT
[MAT, 1, 0 /0.0, 0.0, 0 , 0 , 0 , 0]SEND

6.1.2 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, but not every File 2 will contain resonance parameter data. File 2 for certain materials

will contain a scattering length (see 7.1 and 8.2.2 of this report). For those cases where File 2 only contains information on the scattering length, LRP will be set to zero.

The flag, LFI = 1, indicates that this material is fissionable. In this case, a section specifying the total number of neutrons per fission, $\bar{\nu}(E)$, must be given, i.e., MT = 452. Also sections may be given that specify fission product yields (MT = 454) and the number of delayed neutrons per fission (MT = 455).

The flag LLD indicates whether radioactive decay is given in MT = 453. Certain materials represent natural elements that contain more than one isotope or they represent molecules. For these cases, radioactive decay data my be ambiguous, and it is recommended that these data not be given.

The descriptive data given 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, and references. Also, any limitations of the use of the particular data set should be clearly pointed out along with other remarks which will assist the user in understanding the data. The 2200 m/sec cross sections that the data set contains 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.

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

If the material is fissionable (LFI = 1), then a section that specifies 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 given 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 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.

6.2.1 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 ≤ 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 on interpolation schemes.)

If LNU = 1 (polynomial representation used), 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 (tabulated values of $\bar{\nu}$), 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}/ν(E)]TAB1
[MAT, 1, 0 / 0.0, 0.0, 0, 0, 0, 0]SEND

6.2.2 Procedures

If a polynomial representation (LNU = 2) has been used to specify $\bar{\nu}(E)$, it is understood that 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 $\nu(E)$, the fit shall be limited to a third degree polynomial (NC = 4). If such a fit does not reproduce the recommended values of $\bar{\nu}(E)$, a tabulated form (LNU = 2) should be used.

If tabulated values of $\bar{\nu}(E)$ are specified (LNU = 2), then pairs of energy-ν 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.

It is understood that 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) may be given,

which specifies the delayed neutron from fission, $\bar{\nu}_d$, the number of delayed neutrons per fission must be included in the values of $\bar{\nu}(E)$ given in this section (MT = 452).

6.3 Radioactive Decay Data (MT = 453)*

If the material represents the nuclear data for a single nuclide, then a section (MT = 453) may be given which specifies the radioactive decay properties of this nuclide and the various reaction product nuclides produced by neutron interactions. This section is given if LDD = 1 in MT = 451 (see Section 6.1.1 of this report).

Data are given for the decay of the ground state(and/or any excited state) of the original nuclide. For each original state, any number of decay modes can be described that lead to different residual nuclides and to different excited states of the residual nuclides. Also data are given to specify the reaction products that result from various neutron reaction mechanisms. These data are given for neutron reactions on the ground state and/or any excited state of the original nuclides. One or more excited states of the reaction product nuclide may be given. The following quantities are defined:

ZA is the designation of the original nuclide (ZA = 1000.0*Z + A)

NS is the integer number of states of the original nuclide for which radioactive decay and/or reaction product data are given. (NS ≤ 5)

LIS designates the state of the original nuclide, ZA. (LIS = 0 means the ground state, LIS = 1 means the first excited state, etc.)

LFS designates the state of the product nuclide. (LFS = 0 means the ground state, LFS = 1 means the first excited state, etc.)

*Note that this section describes the revised formats for radioactive decay data. Recipients of ENDF/B-II data will be issued revised data sets that will conform to these formats.

NPR is the number of product nuclides and/or product nuclide states for which data is given for one state of the original nuclide. (the sum of all product nuclide/states formed by spontaneous decay and by neutron interactions)

RTYP is the designation of the reaction type leading to the described product nuclide/state. (0.0 indicates spontaneous decay. All other values of RTYP are floating point equivalents of MT numbers)

ZAP is the (Z,A) designation of the product nuclide (ZAP = 1000.0*Z + A)

DC is the decay constant (sec^{-1}) for the decay of a particular state of the original nuclide to a particular state of the product nuclide. (DC is only given when RTYP = 0.0, otherwise DC = 0.0)

EREL is the energy released (ev) by the decay of the original nuclide (EREL is the total energy of the gamma ray(s) and/or the particle(s) released by the decay. EREL = 0.0 when RTYP $\neq$ 0.0)

Q is the reaction Q value (ev) (the kinetic energy released by (positive) or required for (negative) a reaction)

ES(N) is the energy of the N^{th} point (ev) at which branching ratios are given.

BR(N) is the branching ratio at the N^{th} energy point giving the fraction of the original nuclide in a specified state that results in a specified product nuclide state for a specified reaction. At any particular energy point the sum of all branching ratios for a specified RTYP must be 1.0.

NE is the number of energy points at which branching ratios are given for a specified initial state.

6.3.1 Formats

The structure for this section always starts with a HEAD record and ends with a SEND record. The section is divided into subsections and each subsection contains the data for a particular state of the original nuclide. The subsections are ordered according to LIS, i.e., the data for the ground state (LIS = 0) of the original nuclide is given first.

Each subsection contains two or more LIST records, i.e., there will be (NPR + 1) LIST records. After the first LIST record (which specified NPR) the LIST records are first ordered by increasing values of RTYP. If there are more than two LIST records for the same RTYP, then the LIST records are first ordered by increasing values of ZAP (ZA designation of the product nuclide) and then by increasing values of LFS (product nuclide state designation).

The structure of a section is:

```
[MAT, 1, 453/ZA, AWR; 0, 0; NS, 0] HEAD
  <subsection for LIS = 0 (ground state)>
  <subsection for LIS = 1 (first excited state)>
     ---------------------------------
  <subsection for LIS = NS>
[MAT, 1, 0/0.0, 0.0; 0, 0; 0, 0 ] SEND
```

The structure of a subsection is:

```
[MAT, 1, 453/ZA, AWR; LIS, 0; NE, NPR/
            ES(1), ES(2),--------------
            ------------------, ES(NE)] LIST
[MAT, 1, 453/EREL, Q; LFS, 0, NE + 3, 0/
            RTYP, ZAP; DC, BR(1), BR(2), BR(3)/
            BR(4),------------------BR(NE)] LIST
          ------------------------------
          ------------------------------
          NPR such LIST records (of the second type)
```

Note that the first LIST record contains the set of energy points to describe the branching ratios of all final states from the particular initial state. Although this may lead to some superfluous zeros in the branching ratio lists, it will assure proper normalization. Linear-linear interpolation for branching ratios between the given energy points.

6.3.2 Procedures

Data should be given in MT = 453 for all materials that are single nuclides. Data should not be given for mistures of elements, molecules, or elements that have more than one naturally occurring isotope.

All spontaneous decay modes of the ground state should be described. Also, decay modes for the first few important excited states (isomeric states) should be given if they have significant half-lives ($T_{1/2}$ > several seconds). When a particular state decays by two or more modes, the decay constant for each mode must be given (not the decay constant for all modes together). Therefore it may be necessary to calculate the decay constants for each mode based on the observed branching ratio. Also, note that giving the branching ratio as a function of incident neutron energy for may not be meaningful.

When data are given to specify the residual nuclides formed by neutron reactions (RTYP > 0.0), they should not be given for reactions like the total cross sections (RTYP = 1.0) or the fission cross sections (RTYP = 18.0).Branching ratio data refer to a particular reaction type (RTYP). Therefore the sum of the branching ratios (at a particular energy point) is unity only for a specified reaction type (RTYP).

There will be a natural overlap of the same data (decay of a particular nuclide) being given in two or more different materials. It is important that consistency be preserved in the data given in various materials. Therefore, it is recommended that the current Chart of the Nuclides be used as the basis for obtaining the decay chain data for this section.

6.4 Fission Product Yield Data (MT = 454)*

This section (MT=454) is provided to specify the incident neutron energy-dependent fission product yield data and may be given if LFP = 1 in the first section (MT = 451). A complete set of fission

*This section specifies the revised formats for fission product yield data.

product yield data is given for a particular incident neutron energy. Data sets are then 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 and/or 3). The data are given by specifying fission product identifiers and fission product yields. Fractional yields are given, and the sum of all fractional yields for any particular incident neutron energy will be approximately 2.0.

The fission products are specified by giving an excited state designation (FPS) and a (charge, mass) indentifier (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, fission product identifiers- fission product yields). (NFP $\leq$ 150).

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.).

YLD is the fractional yield for a particular fission product.

$C_n(E_i)$ is the array of yield data for the i^{th} energy point. This array contains NFP sets of three parameters given in the order: ZAFP, FPS, YLD.

N1 is equal to 3*NFP, the count of 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 to be used between the E_{i-1} and E_i energy points (see details on interpolation scheme in Appendix E).

6.4.1 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 three parameters (FPS, ZAFP, YLD) for each fission product state. The data are first ordered by increasing values of ZAPF. If more than one yield is given for the same (Z,A), then the data are ordered by increasing value of the state designator (FPS).

The structure for a section is:

[MAT, 1, 454/ZA, AWR, LE + 1, 0, 0, 0]HEAD

[MAT, 1, 454/E_1, 0.0, LE, 0, N1, NFP/$C_n(E_1)$]LIST

[MAT, 1, 454/E_2, 0.0, I_2, 0, N1, NFP/$C_n(E_2)$]LIST

[MAT, 1, 454/E_3, 0.0, I3, 0, N1, NFP/$C_n(E_3)$]LIST

[MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0]SEND

There are (LE + 1) LIST records.

6.4.2 Procedures

The data sets for fission product yields should be given over the same energy range as that given in Files 2 and/or File 3 for the fission cross section. The yields are given as fractional values at each energy, and normally they will sum to approximately 2.0.

This format provides for the yields (YLD) to each excited state (FPS) of the nuclide designated by ZAFP, and hence accommodates the many metastable fission products which have 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 yield data are given for more than one nuclide for a particular mass chain, then it is understood that the yield for the lowest Z (charge) nuclide state for this mass chain should be a cumulative fractional yield, and all other yields for this same chain should be direct fractional yields.

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.

6.5 Delayed Neutron Data (MT = 455)

This section is provided to contain a description of the delayed neutrons that result from fission events. In this section the average total number of delayed neutron precursors emitted per fission, $\bar{\nu}_d$, are given along with the decay constants, λ_i for each of the precursor families. The fraction of $\bar{\nu}_d$ that is generated for each precursor's family is given in File 5 (Section 10 of this report). Also, the energy distributions of the secondary neutrons that are associated with each precursor family are given in File 5.

The total number of delayed neutron precursors are given as a function of incident neutron energy. Two representations are provided to specify the energy-dependence. These representations are the same as are used in this File, (MT = 452), to describe the average total number of neutrons produced per fission event (see Section 6.2). 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's families,

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

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 10), and is defined as

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

6.5.1 Formats

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

$$\bar{\nu}_d(E) = \sum_{m=1}^{NCD} CD_m E^{(m-1)} .$$

The following quantities are defined:

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

NCD is the number of terms in the polynomial expansion. (NCD ≤ 4).

CD_m are the coefficients for the polynomial.

NR is the number of interpolation ranges used.

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 for details about interpolation schemes).

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

NNF is the number of precursor's families considered.

λm is the decay constant(sec^{-1}) for the mth precursor.

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

⌈MAT, 1, 455/ ZA, AWR, 0, LND, 0, 0]HEAD ' LND = 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, NCD, 0/CD_1, CD_2,...CD_{NCD}]LIST
⌈MAT, 1, 0 /0.0, 0.0, 0, 0, 0, 0]SEND

The structure of a section when values of $\overline{\nu}_d$ are tabulated (LND = 2) is:

[MAT, 1, 455/ ZA, AWR, 0, LND, 0, 0]HEAD LND = 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}/$\overline{\nu}_d$(E)]TAB1
[MAT, 1, 0 / 0.0, 0.0, 0, 0, 0, 0]SEND

6.5.2 Procedures

When the polynomial representation is used (LND = 1), it is understood that the calculated values of $\overline{\nu}_d$(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 $\overline{\nu}_d$(E) are specified, they should be given for the same energy range as is used to specify the fission cross sections.

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 10 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 families should be ordered the same in both files. It is recommended that the families be ordered by decreasing half-lives ($\lambda_1 < \lambda_2 < \ldots < \lambda_{NNF}$).

7. FILE 2, RESONANCE PARAMETERS

7.1 General Description

File 2 contains data for both resolved and unresolved resonance parameters. There is only one section in this File, and this section has been assigned the reaction type number MT = 151. 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 contributions that are calculated from the resolved and/or unresolved parameters given in File 2 in order to obtain the correct reaction cross sections.

Every material will contain a File 2 even though no resolved and/or unresolved parameters are given. The purpose of a File 2, for those materials where no resonance parameter data are given, is to specify the effective scattering radius for the material. This scattering radius (to be used to obtain the potential scattering cross section) is required when resonance calculations are made for other materials and the presence of this material, i.e., the potential scattering cross sections, must be taken into consideration during analyses of the other materials.

The resonance parameter data for a material
fying 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 one another. Each energy range will contain a different representation of the resonance parameters. Normally two energy ranges will be specified for each isotope. The first energy range will contain resolved parameters and the second, unresolved resonance parameters.

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

The allowed representations for the resolved resonance parameters are:

LRF = 1, means single-level Breit-Wigner parameters given;

LRF = 2, means multilevel Breit-Wigner parameters are given (level-level interference effects are considered for the elastic scattering cross section and the total cross section);

LRF = 3, R-matrix (Reich-Moore) multilevel resonance parameters are given;

LRF = 4, Adler-Adler multilevel resonance parameters are given.

The data formats for each of the above representations are essentially the same, except for the format for the Adler-Adler multilevel parameters.

Each energy range contains a flag, LRU, that indicates whether the parameters in this energy range are resolved or unresolved resonance parameters. The flag LRU = 1 means that the data are for resolved resonance parameters. The flag LRU = 2 means that the data are for unresolved resonance parameters.

There is only one representation allowed for the unresolved resonance parameters, e.g., average single-level Breit-Wigner resonance parameters. However, several options exist for specifying the unresolved parameters. Using the first option LRF = 1, only the average fission width is allowed to be specified as a function of incident neutron energy. The second option LRF = 2 allows the following average parameters to be given as a function of incident neutron energy: 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 below in Sections 7.2.1 (resolved) and 7.3.1 (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. These are given below:

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.

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

LFW = 0, means average fission widths are not given;

LFW = 1, means average fission widths are given.

NER is the number of energy ranges given for this isotope. (NER ≤ 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 = 1, means resolved resonance parameters are given;

LRU = 2, means unresolved resonance parameters are given.

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

if LRU = 1 (resolved parameters), then

LRF = 1, Single-level B-W parameters
LRF = 2, Multilevel B-W parameters
LRF = 3, Reich-Moore 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. Resolved resonance levels may (of necessity) be outside of the limits, for example, bound levels).

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]CØNT (isotope)
[MAT, 2, 151/ EL, EH, LRU, LRF, 0, 0]CØNT (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]CØNT (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]CØNT (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 CØNT (range) record. The data for each isotope start with a CØNT (isotope) record. The specifications for the subsections are given in Sections 7.2.1 and 7.3.1, below.

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

7.2 Resolved Resonance Parameters (LRU = 1)

7.2.1 Formats

Four 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:

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

SPI is the nuclear spin of the target nucleus, I.

AP is the spin-dependent effective scattering radius A_+ (for spin-up) in units of 10^{-12} cm. AP is also given for the case of spin independence.

AM is the spin-dependent effective scattering radius, A_- (for spin-down). AM = 0.0 for spin independence.

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

L is the value of the ℓ-state (neutron angular momentum quantum number).

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

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.

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]CØNT
[MAT, 2, 151/ AWRI, AM, L, 0, 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 of the NLS ℓ-states 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.

The structure for a subsection, when R-Matrix (Reich-Moore) multilevel parameters are given (LRF = 3), is similar to that given above. The major difference is that the total resonance widths are not given and two fission widths are allowed for each resolved resonance. The quantities that are defined for use when LRF = 3 are given below:

Resolved Resonance Parameters
If LRF = 3 (Reich-Moore multilevel parameters)

SPI is the spin of the target nucleus I.

AP = A_+ is the spin-up effective scattering radius in units of 10^{-12}cm.

AM = A_- is the spin-down effective scattering radius in units of 10^{-12}cm. AM = 0.0 for spin independence.

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 the ℓ-state (neutron angular momentum quantum number).

AWRI is the ratio of the mass of a particular isotope to that of a 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 compound nucleus spin, 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 for Reich-Moore parameters.

GFB is the second partial fission width for Reich-Moore parameters. GFA and GFB are signed quantities; their signs being determined by the relative 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]CØNT
[MAT, 2, 151/ AWRI, AM, 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.

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 using the six constants in the following manner:

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

where $C = \pi \lambda\!\!\!^{-2} = \pi/k^2$ and $k = 2.19677 \times 10^{-3}\left(\frac{AWRI}{AWRI + 1.0}\right)\sqrt{E(eV)}$

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 the ℓ - state (neutron angular momentum quantum number).

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

NJS is the number of sets of resolved resonance parameters (each having the same J state) for a specified ℓ - state.

NLJ is the count of the number of levels for which parameters will be given (each level having a specified AJ and L).

*Reserved for use in ENDF/A only.

SPI is the spin of the target nucleus.

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

AP is the spin-dependent effective scattering radius, A_+ (for spin-up) in units of 10^{-12} cm.
AP is also given for the case of spin independence.

AM is the spin-dependent effective scattering radius, A_- (for spin-down). AM = 0.0 for spin independence.

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.

DET_n is the resonance energy (the n^{th} level) for the total cross section.

DEF_n is the resonance energy (the n^{th} level) for the fission cross section.

DEC_n is the resonance energy (the n^{th} level) for the radiative capture cross section.

DWT_n is the value of $\Gamma/2$, (ν), used for the total cross section (for the n^{th} level).

DWF_n is the value of $\Gamma/2$, (ν), used for the fission cross section (for the n^{th} level).

DWC_n is the value of $\Gamma/2$, (ν), used for the radiative capture cross section (for the n^{th} level).

GRT_n is the symmetrical total cross section parameter (for the n^{th} level).

GIT_n is the asymmetrical total cross section parameter (for the n^{th} level).

GRF_n is the symmetrical fission parameter (for the n^{th} level).

GIF_n is the asymmetrical fission parameter (for the n^{th} level).

GRC_n is the symmetrical capture parameter (for the n^{th} level).

GIC_n is the asymmetrical capture parameter (for the n^{th} level).

The structure of a subsection containing data for (LRU = 1 and LRF = 4, Adler-Adler multilevel parameters) 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, AM, 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 $\neq$ 7 then certain of the parameters for each level may be set to 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, it is to be obtained by subtracting the sum of capture and fission cross-sections from the total cross section.

7.2.2 Procedures

For certain resonances the value of ℓ is known but the resonance spin state J is not. In this case an average value of J and the corresponding average value of the statistical weight factor g_J should be used. $g_J = (2\ell + 1)/N$, where N is the number of possible combinations of ℓ and S (the channel spin).

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 of this energy range the cross sections must be obtained from the parameters given in another energy range and/or from data given in File 3. Therefore, it is sometimes necessary to give parameters whose energies lie outside of a specified energy range in order to accurately give 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 correct cross section.)

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. Also, for these materials it is recommended that the upper energy limit for the unresolved resonance range be the same for all isotopes. If resolved and/or unresolved resonance parameters are not given for all of the naturally occurring isotopes, some data should be given for the other isotopes. In particular, AP should be given for each of these isotopes.

If more than one energy range is used to describe the resonance parameters for any given isotope, it is recommended that the energy ranges be contiguous and that the energy ranges not overlap. It is further recommended that the data for each isotope be divided into two energy ranges: one for resolved resonance parameters and the other for unresolved resonance parameters.

7.3 Unresolved Resonance Parameters (LRU = 2)

7.3.1 Formats

There is only one representation of the unresolved resonance parameters allowed (see Appendix D for pertinent formulae). There are, however, several options available for specifying the average properties of the resonances.

The parameters given are for the single-level Breit-Wigner formula with interference, and they depend on both ℓ (neutron angular momentum) and J (compound nucleus spin) states. The widths are distributed according to a chi-squared distribution with a specified 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.

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

SPI is the nuclear spin I of the target nucleus.

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

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

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

ES(N) is the energy of the N^{th} point used to tabulate energy-dependent widths.

L is the value of ℓ (neutron angular momentum quantum number).

AWRI is the ratio of the mass of the 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-state.

D is the mean level spacing for a particular J-state. (This value is energy-dependent if LRF = 2.)

AMUX is the number of degrees of freedom used in the competitive width distribution. (If an actual value is not known or is extremely large, set AMUX = 0.0.)

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

AMUG is the number of degrees of freedom used in the radiation width distribution. (If this value is not known or is extremely large, set AMUG = 0.0.)

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

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

INT is the interpolation scheme to be used for interpolating the resonance parameters (normally, INT = 1).

GNO is the average reduced neutron width. This value is energy-dependent if LRU = 2.

GG is the average radiation width. This value is energy-dependent if LRU = 2.

GF is the average fission width. This value may be energy-dependent.

GX is the average competitive reaction width.

The structure of a subsection depends on whether LRF = 1 or LRF = 2. If LRF = 1, only the fission widths can be given as a function of neutron energy. If LRF = 1 and the average fission widths are not given (as indicated by LFW = 0), then a simple form of the unresolved resonance parameters is given. If LRF = 2, the energy-dependent average values may be given for the level density, a competitive reaction width, reduced neutron width, radiation width, and fission widths. Therefore, there are three different formats that are considered.

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, A, 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-states have been specified.

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,  A,   0,   0,   NE,   NLS/
              ES1,  ES2, ES3, .,   .,    .,
              .,    .,   .,   .,   ESNE  ]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,
              GF1,  GF2, GF3,  .,   .,    .,
              .,    .,   .,    GFNE       ]LIST
```

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

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,  A,   0,    0,     NLS,       0]CØNT
[MAT, 2, 151/ AWRI, 0.0, L     0,     NJS,       0]CØNT  (ℓ)
[MAT, 2, 151/ AJ,   0.0, INT,  0,     (6*NE)+6,  NE  /
              0.0,  0.0, AMUX, AMUN,  AMUG,      AMUF,
              E1,   D1,  GX1,  GNO1,  GG1,       GF1,
              E2,   D2,  GX2,  GNO2,  GG2,       GF2,
              ----------------------------------
              ENE,  DNE, GXNE, GNONE, GGNE,      GFNE]LIST
```

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

7.3.2 Procedures

The number of degrees of freedom for the distribution of the competitive reaction width (AMUX) and radiation widths (AMUG) may be meaningless or they may be extremely large. If AMUX and/or AMUG are zero, this is a flag that indicates the number of degrees of freedom is extremely large. The average competitive reaction width is given (LRF = 2) to account for all unspecified competitive reactions, i.e., other than scattering, capture, and fission.

Up to 250 energy points are allowed for giving energy-dependent average parameters. These data should allow for average cross sections to be computed that show any gross structure in the reaction cross sections. The unresolved resonance parameters should be provided for neutron energy regions where temperature or resonance self-shielding effects are important. Therefore, it is recommended that the unresolved resonance region extend up to at least 20 keV.

When preparing data for the unresolved resonance region, it is important that a consistent set of definitions be used in obtaining unresolved resonance parameters. These definitions are given in the Glossary (Appendix A) and the resonance region formulae (Appendix D). In particular, it should be noted that the neutron penetrability, $V_\ell(\rho)$, is defined as

$V_0(\rho) = 1$ for $\ell = 0$ neutrons (s-wave)

$V_1(\rho) = \rho^2/(1 + \rho^2)$ for $\ell = 1$ neutrons (p-wave)

$V_2(\rho) = \rho^4/(9 + 3\rho^2 + \rho^4)$ for $\ell = 2$ neutrons (d-wave)

and $\rho = ka$.

The wave number of the neutron in the center-of-mass system is:

$$k = 2.19685 \left(\frac{AWRI}{(AWRI + 1.0)}\right) \sqrt{E(eV)} \times 10^{-3}$$

and "a" is the radius that is used in calculating the penetration, shift, and hard-sphere phase factors,

$$a = [1.23(AWRI)^{1/3} + 0.8] \times 10^{-1}$$

in units of 10^{-12}cm.

8. FILE 3, NEUTRON CROSS SECTIONS

8.1 General Description

Neutron cross sections, such as the total cross section, elastic scattering cross section, radiative capture cross section, are given in File 3. Also, certain derived quantities are given in File 3. These data are given as a function of energy, E, where E is the incident neutron energy (in eV) in the Laboratory system. The data 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 to the next higher energy point.

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

The neutron cross sections (or derived quantities) 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 energy range to another energy range.

8.2 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 that are used in this File and other Files have been defined in Section 5.1 of this Report 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 has the following values:

LFS = 0: The final state is the ground state

= 1: The final state is the first excited state

(con't on next page)

LFS = 2: The final state is the second excited state
⋮ ⋮
LFS= 98: An unspecified range of final states
= 99: All final states

Q is the reaction Q-value (eV).

T is the temperature (^{o}K).

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

NR is the number of energy ranges that have been given. A different interpolation scheme may be given for each range. (NR $\leq$ 10).

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 4.3 of this Report.)

$\sigma(E)$ 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/T , Q , LT, 0 , NR, NP/E_{int}/σ(E)]TAB1
[MAT, 3, 0 /0.0, 0.0, 0, 0 , 0 , 0]SEND

8.2 Procedures

8.2.1 Reaction Types to Be Included

A complete list of possible reaction types and their definitions is given in Appendix B. Cross sections for all reaction types that are not zero or negligibly small should be given in File 3. 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

(con't on next page)

MT	Reaction
18	Fission cross section
51	Inelastic excitation cross section for the 1st level
52	" " " " " " 2nd level
⋮	
90	" " " " " " 40th level
91	" " " " " " 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

An initial state designator is defined in the HEAD record for each section. This flag has been added to facilitate the specification of certain cross sections for important metastable states (for example, the (n,γ) cross section for Pm^{148m}). Therefore, two or more sections may be given for the same reaction type (MT number). In this case the sections will be ordered by increasing state number (ground state data will be given first).

8.2.2 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 radiative capture cross section (MT = 102), fission cross section (MT = 18), and elastic scattering cross section (MT = 2), the cross sections calculated from the resolved or unresolved resonance parameters must be added to the appropriate data given in File 3. The contributions from File 2 and File 3 must be added together 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 multi-level interference effects that were apparent in the experimental data where it was not possible to construct a set of resonance parameters that adequately fit the measured data.

There will be some materials that do not have resonance parameters. However, these materials will have a scattering length given in File 2. This scattering length is intended to be used to calculate the potential elastic scattering cross section that is then used in the calculation of resonance self-shielding effects in other materials. For these materials the elastic scattering cross section given in File 3 must <u>not</u> 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. It is recommended that double-valued points always be given at the lower and upper energy limits of the resolved and unresolved resonance region.

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 9) 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 $\overline{\mu}_{Lab}$, ξ, and γ may be calculated entirely from File 4 angular distribution data for elastic scattering. These data are included in File 3 for convenience.

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. In this case 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-angle) scattering cross sections may be obtained by combining the data given in File 6 and the cross sections given in File 3.

8.2.3 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.* It is recommended that the evaluator not use more points than are necessary to represent the cross section accurately. Also, a limit of 10 must be adhered to for the number of interpolation regions (NR).

Cross section data for non-threshold reaction types should cover at least the energy range from 10^{-5} eV to 15 MeV. For shielding materials an upper energy limit of 20 MeV should be

*Note that at the present time some processing codes, such as the CHECKER, will only handle 3000 points.

used for data in File 3. For other reactions the cross section data should start at the reaction threshold energy (and have a value of 0.0 at this energy) and continue up to either 15 or 20 MeV. For non-threshold reactions a cross-section value should be given at an energy point of 0.0253 eV.

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

$$E_{th} = \left(\frac{AWR + 1}{AWR}\right)\left|Q\right| ,$$

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

For a material which is a mixture of several isotopes, the Q-value is not uniquely defined. The threshold energy would pertain to the particular isotope that contributes to the cross section at the lowest energy.

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 it 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 it has a non-zero value at the next higher energy point, a problem will be created by giving a log-linear or a log-log interpolation scheme.

9. FILE 4, ANGULAR DISTRIBUTIONS OF SECONDARY NEUTRONS

9.1 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 may 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 given in File 3. Angular distributions for several different reaction types (MT's) may be given in File 4 for each material. The data for each reaction type are given 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,

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

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;

$\sigma(E)$ = the scattering cross section, e.g., elastic scatterings 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;

f_ℓ = the ℓ^{th} Legendre polynomial coefficient and it is understood that $f_0 \equiv 1.0$.

The angular distributions may be given in one of two representations, and they may be given 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, $f_\ell(E)$, 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

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

where $\sigma_s(E)$ is given in File (for the same MT number) and the coefficients, $f_\ell(E)$ 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 described elastic scattering angular distributions, from one frame of reference to the other. The Legendre expansion coefficients, $f_\ell(E)$, in the two systems are related through an energy-independent transformation matrix, $U_{\ell m}$, and its inverse, $U^{-1}_{\ell m}$,

$$f_\ell^{Lab}(E) = \sum_{m=0}^{NM} U_{\ell m} \; f_m^{CM}(E)$$

and

$$f_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell m} \; f_m^{Lab}(E)$$

Expressions for the matrix elements of U and U^{-1} may be found in the papers by Zweifel and Hurwitz[1] and Amster[2]. Transformation matrices for nonelastic reactions are not incident energy-independent and <u>are not</u> given in File 4.

The transformation matrices should be square with the number of rows equal to NM + 1 where MN is the maximum order of the Legendre polynomial series that is used to describe any elastic angular distribution in this File. The transformation matrix is given as an array of numbers, V_K, where K = k, .,.,., 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 <u>column-wise</u> in the array V_K:

$$\begin{matrix}
U_{0,0} & U_{0,1} & \cdots & U_{0,NM} \\
U_{1,0} & U_{1,1} & \cdots & U_{1,NM} \\
\vdots & \vdots & \cdots & \vdots \\
U_{NM,0} & U_{NM,1} & \cdots & U_{NM,U_{NM}}
\end{matrix}$$

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1. P. F. Zweifel and H. Hurwitz, Jr., J. Appl. Phys. <u>25</u>, 1241 (1954).
2. H. Amster, J. Appl. Phys. <u>29</u>, 623 (1958).

9.2 Formats

File 4 is divided into sections, with each section containing data for a particular reaction type (MT number). The sections are 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.

The following quantities are defined:

LTT is a flag to specify the representation used and it may have the following values:

LTT = 1, the data are given as Legendre expansion coefficients, $f_\ell(E)$;

LTT = 2, the data are given as normalized probability distributions, $p(\mu,E)$.

LCT is a flag to specify the frame of reference used:

LCT = 1, the data are given in the laboratory (LAB) system;

LCT = 2, the data are given in the center-of-mass (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 ≤ 20) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system.

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 ≤ 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, $f_\ell(E)$ 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.

9.2.1 Legendre Polynomial Coefficients and Transformation Matrix Given: LTT = 1 and LVT = 1

For the case where 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, 0, LCT, NK, NM/V_K]LIST
[MAT, 4, MT/0.0, 0.0, 0, 0, NR, NE/E_{int}]TAB2
[MAT, 4, MT/T, , E_1 , LT, 0 , NL, 0/$f_\ell(E_1)$]LIST
[MAT, 4, MT/T , E_2 , LT, 0 , NL, 0/$f_\ell(E_2)$]LIST

[MAT, 4, MT/T , E_{NE}, LT, 0 , NL, 0/$f_\ell(E_{NE})$LIST
[MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0]SEND

Note that T and LT refer to temperature (in $^{\circ}K$) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases where temperature dependence is specified.

9.2.2 Legendre Polynomial Coefficients Given and the Transformation Matrix not Given: LTT = 1 and LVT = 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:

[0.0, AWR, 0, LCT, 0, 0]CØNT.

This form is always used for angular distributions of nonelastically scattered neutrons when Legendre polynomial expansion coefficients are used.

9.2.3 Tabulated Probability Distributions and Transformation Matrix Given: LTT = 2 and LVT = 1

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, 0, LCT, NK, NM/V_K]LIST

[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]SEND

T and LT are normally zero. See Appendix F for details about temperature dependence.

9.2.4 Tabulated Probability Distributions Given and Transformation Matrix not Given: LTT = 2 and LVT = 0

The structure of a section is the same as above, except that the second record (a LIST record) is replaced by:

[0.0, AWR, 0, LCT, 0, 0]CØNT.

This form is always used for angular distribution of non-elastically scattered neutrons when tabulated angular distributions are given.

9.3 Procedures

The angular distributions for elastic scattering should be given as Legendre polynomial coefficients, $f_\ell(E)$'s (LTT = 1), and they should be given in the center-of-mass system (LCT = 2). It is recommended that the angular distributions of neutrons from non-elastic reactions (such as inelastic discrete level data, continuum inelastic, fission, etc.) be given as tabulated distributions, $p(\mu,E)$'s, and that they be in the Laboratory 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 insure that the interpolated distribution is everywhere positive. The number of coefficients (NL), may vary from energy point to energy point and in general, NL will increase with increasing incident energies. A linear-linear interpolation scheme (INT = 2) must be used to obtain coefficients at intermediate energies. This is required to insure that the interpolated distribution is positive over the cosine interval from -1.0 to 1.0 and is also required because some coefficients may be negative. In no case should NL exceed a value of 20, and NL always should be an even value. If more than 20 coefficients should appear to be required to obtain a non-negative distribution, a constrained Legendre polynomial fit to the data should be used or a tabulated distribution should be given. NL = 1 is allowed at low energies to specify an isotopic 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 the interpolation scheme for $p(\mu,E)$ vs. E should be linear-linear (INT =2).

Representation of angular distribution 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. For example, free-atom data for carbon appear in MAT = 1010 and bound-atom data appear in MAT = 1065.

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 Laboratory to the center-of-mass system, or CM to LAB, a distribution should be generated and transformed pointwise to the desired frame of reference. The pointwise angular distributions can then be converted to Legendre polynomial coefficients in the new frame of reference.

10. FILE 5, ENERGY DISTRIBUTION OF SECONDARY NEUTRONS

10.1 General Description

File 5 contains data for the energy distributions of secondary neutrons. The energy distributions are expressed as normalized probability distributions. The File is divided into sections, each section giving the data for a particular reaction type (MT number). The sections are then ordered by increasing reaction type number (MT).

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 given in File 4. Also, no data will be given for neutrons that result from excitation of discrete inelastic levels when data are given for these reactions 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 such 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 <u>always</u> expressed in the <u>laboratory system</u>.

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 type (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 incident energy, E.

The partial energy distributions $f_k(E \rightarrow E')$ are represented by various analytical formulations. Each of these formulations 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.

$f(E \rightarrow E') = g(E \rightarrow E')$

A set of incident energy points is given, E_i, and $g(E_i \rightarrow E')$ is tabulated as a function of E'.

LF = 3, Excitation of Discrete Levels

$$f(E \rightarrow E') = \delta\left[E' - \frac{A^2+1}{(A+1)^2}\,E + \frac{A}{A+1}\,\theta\right]$$

A = AWR (the ratio of the mass of the target nucleus to that of the neutron);
θ = excitation energy of the energy level in the residual nucleus.

LF = 5, General evaporation spectrum.

$f(E \rightarrow E') = g[E'/\theta(E)]$

$\theta(E)$ 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,

$$I = \theta^{3/2}\left[\frac{\sqrt{\pi}}{2}\,\mathrm{erf}\left(\sqrt{(E-U)/\theta}\right) - \sqrt{(E-U)/\theta}\; e^{-(E-U)/\theta}\right];$$

θ is tabulated as a function of energy, E;
U is a constant and is introduced to define the proper upper limit for the final neutron energy such that $0 \le E' \le E - U$.

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 given as a constant and is introduced to define the proper upper limit for the final neutron energy such that $0 \le E' \le E - U$.

LF = 10, Watt spectrum.

$$f(E \rightarrow E') = \sqrt{4/\pi a^3 b}\; e^{-ab/4}\, e^{-E'/a} \sinh(\sqrt{bE'})$$

a and b are given as constants.

NOTE: Distribution laws are not presented for LF = 2, 4, 6, or 8. 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 partial energy distribution. The partial energy distributions may all use the same energy distribution law (LF number) or they may use different laws.

10.2 Formats

The File is made up of sections, and each section contains the data for a particular reaction type (MT number). Every section starts with a HEAD record and ends with a SEND record. The section is made up of subsections, and 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 such that $0 \leq E' \leq E - U$. (given in the Laboratory system)

θ is a parameter used to describe the secondary energy distribution. The definition of θ depends on the energy distribution law (LF) given; however, the units are always eV.

If LF = 3, θ is the excitation energy, $|Q|$, of a level in the residual nucleus.
LF = 5, 7, and 9, θ is an effective nuclear temperature.

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 above in Section 10.1).

$p_k(E_N)$ 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 (see Section 10.1, above).

NR is the number of interpolation ranges.

NP is the count of the number of incident energy points at which $p_k(E)$ is given.

a,b are constants used in the Watt spectrum. (LF = 10).

NE is the count of the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given.

NF is the count of the number of secondary energy points in a tabulation.

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 (energy distribution law). Subsections should be ordered by increasing values of LF number. For cases where more than one subsection contains data using the same LF, then 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→E'_1) , E'_2 , g(E_1→E'_2) , E'_3 , g(E_1→E'_3),

---

---, E'_{NF} , g(E_1→E'_{NF})]TAB1

[MAT, 5, MT/T , E_2 , LT , 0 , NR , NF/E'_{int} /

E'_1 , g(E_2→E'_1) , E'_2 , g(E_2→E'_2) , E'_3 , g(E_2→E'_3),

---

---, E'_{NF} , g(E_2→E'_{NF})]TAB1

[MAT, 5, MT/T , E_{NE} , LT , 0 , NR , NF/E'_{int} /

E'_1 , g(E_{NE}→E'_1), E'_2 , g(E_{NE}→E'_2), E'_3 , g(E_{NE}→E'_3),

---

---, E'_{NF} , g(E_{NE}→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 that is 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 = 3, Discrete level excitation

[MAT, 5, MT/ T , θ , LT ,LF =3, NR, NP/E_{int}/p(E)]TAB1

Only one record is given for each subsection.

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} /
$\theta(E_1)$, $\theta(E_2)$, ----------
----------, $\theta(E_{NE})$]TAB1

[MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NF/x_{int} /
x_1 , $g(x_1)$, x_2, $g(x_2)$, x_3 , $g(x_3)$

----------, x_{NF}, $g(x_{NF})$]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 = 10, Watt spectrum

[MAT, 5, MT/ 0.0 , 0.0, 0, LF=10 NR, NP/E_{int}/p(E)]TAB1
[MAT, 5, MT/ 0.0 , 0.0 , 0, 0 , 2 , 0 /
a , b , ----------]LIST

Note that no formats have been described for LF = 2, 4, 6, or 8. These laws are no longer defined.

10.3 Procedures

Up to 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. It is recommended that a linear-linear interpolation be used 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). Also, this energy mesh must include threshold energy values for all reactions which are 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 or 9). This constant is provided to define the proper upper limit for the secondary energy distribution such that $0 \leq E' \leq E - U$. The value of U depends on how the data are represented for a particular reaction type. Consider the constant U for inelastic scattering:

Case A: The total inelastic scattering cross section is described as a continuum. In this case, U would be the threshold energy for exciting the lowest level in the residual nucleus.

Case B: For the energy range being considered, the first three levels are described explicitly (either in File 3, MT = 51, 52, and 53, or in File 5, LF = 3), and the rest of the inelastic cross section is being treated as a continuum. In this case, 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). 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 point, in energy, 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 be 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.). Also, the angular distributions for the neutrons that result 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 given in Files 3 and 4. This procedure is the only way in which the angular distributions can be given for these neutrons. For inelastic scatter, the only data that are required in File 5 are for MT = 91 (continuum part).

11. FILE 6, ENERGY-ANGULAR DISTRIBUTIONS FOR SECONDARY NEUTRONS

11.1 General Description

This file is provided to represent energy-angular distributions of secondary neutrons. Data are given in this file 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 that result from certain neutron reactions with fairly light nuclei.

The file is divided into sections, each section contains the data for a particular reaction type (MT number). The sections are ordered by increasing reaction type numbers. If data are given in File 6 for a particular reaction, then no data will be given in Files 4 or File 5 for the same reaction. The secondary neutron energy-angle distribution are expressed as normalized probability distributions, $p(E \rightarrow E', \mu)$.

$$\int^{E'_{max}} dE' \int_{-1}^{1} p(E \rightarrow E', \mu)\, d\mu = 1$$

The differential cross section (in barns per steradian per eV) is obtained from

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \rightarrow E', \mu) = \frac{\sigma(E)}{2\pi}\, m\, p(E \rightarrow E', \mu)$$

where $\sigma(E)$ is the cross section for particular reaction as given in File 3 and/or File 2 for the same reaction type (MT number) and m is the implied neutron multiplicity.

The angular part of the distribution may be specified in one of two ways. First, secondary energy distributions may be tabulated at a set of secondary angles. Second, the probability distributions may be expressed as a Legendre polynomial expansion.

$$p(E \rightarrow E', \mu) = \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2}\, p_\ell(E \rightarrow E')\, P_\ell(\mu)$$

In this case, the zeroth coefficient, $p_0(E \rightarrow E')$, is not unity (as was used in File 4), but for a particular incident neutron energy, E,

$$\int p_0 \ (E \rightarrow E') \ dE' = 1$$

The secondary angles and energies may be given in either the laboratory or center-of-mass system. Incident neutron energies are always in the laboratory system.

If distributions, $p(E \rightarrow E', \mu)$ are tabulated at a series of angles, then a set of secondary angles (cosines of the scattered angles) are selected. This set of angles 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 same secondary energy distribution laws (LF numbers) are used in this file as were defined in Section 10.2 for File 5 data.

When the distributions are represented by Legendre polynomial expansion coefficients, then a subsection is given for each coefficient, $p_\ell(E \rightarrow E')$. The format for a subsection is similar to the format for a section in File 5 (section 10.2). The first subsection contains data for the zeroth coefficient, $p_0(E \rightarrow E')$. The subsections are then ordered by increasing ℓ-value of the coefficients.

The following quantities are defined:

LCT is a flag that indicates which reference frame is used for both secondary angles and energies.

LCT = 1: The data are given in the Laboratory system.

LCT = 2: (CM) Do not use.

LTT is a flag that indicates which representation is used.

LTT = 1: The data are given as Legendre expansion coefficients.

LTT = 2: The data are given as a tabulation.

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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 ≤ 101)

μ is the cosine of the scattered angle.

NK is the number of partial probability distributions used for this reaction type (used in the same manner as in File 5, Section 10.2 of this report).

LF is a flag that indicates which secondary energy distribution law is used for a particular partial probability distribution. (See Section 10.2).

11.2 Formats

The file is made up of sections where each section 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 $p_0(E \rightarrow E')$⟩

⟨Subsection for $p_{NL}(E \rightarrow E')$⟩
[MAT, 6, 0 /0.0, 0.0, 0, 0, 0, 0]SEND

The structure of a subsection is identical to the structure of a section for secondary energy distributions given in File 5 (Section 10.2 of this report) with the following exceptions. First, the SEND record is deleted (since the section in File 5 is used here as a subsection, and second, the HEAD record is changed to read:

[MAT, 6, MT/0.0, 0.0, L, 0, NK, 0] CØNT

The following is the structure for a typical section, e.g., LTT = 1 (Legendre expansion coefficients given), NK = 1 (one partial probability distribution, 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}/$p_0(E)$]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}/$p_0(E_1 \rightarrow E')$]TAB1

[MAT, 6, MT/T, E_2, LT, 0, NR, NF E'_{int}/$p_0(E_2 \rightarrow E')$]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p_0(E_{NE} \rightarrow E')$]TAB1

} Subsection for $p_0(E \rightarrow E')$

[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}/$p_1(E)$]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}/$p_1(E_1 \rightarrow E')$]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p_1(E_{NE} \rightarrow E')$]TAB2

} Subsection for $p_1(E \rightarrow E')$

⟨Subsection for $p_2(E \rightarrow E')$⟩

⟨Subsection for $p_{NL}(E \rightarrow E')$⟩

[MAT, 6, 0 /0.0, 0.0, 0, 0, 0, 0]SEND

T and LT refer to possible temperature dependence (see Appendix F for details on format for temperature dependence). 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 the structure of a section for secondary energy distributions as given in File 5(Section 10.2) with the following exceptions. First, the SEND record is deleted (since

the section in File 5 is used here as a subsection), and second, the HEAD record is changed to read:

$$[0.0, \mu, 0, 0, NK, 0]\text{CØNT}$$

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, μ_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($E_1 \rightarrow E'$, μ_1)]TAB1

[MAT, 6, MT/T, E_2, LT, 0, NR, NF/E'_{int}/p($E_2 \rightarrow E'$, μ_1)]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/p($E_{NE} \rightarrow E'$, μ_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, μ_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($E_1 \rightarrow E'$, μ_2)]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/p($E_{NE} \rightarrow E'$, μ_2)]TAB1

⟨Subsection for p($E \rightarrow E'$, μ_3)⟩

⟨Subsection for p($E \rightarrow E'$, μ_{NA})⟩

[MAT, 6, MT/0.0, 0.0, 0, 0, 0, 0]SEND

Again T and LT refer to possible temperature dependence.

11.3 Procedures

All interpolation schemes used in this section should be linear-linear to insure that the probability distributions will have the proper normalization everywhere. It is strongly recommended that an arbitary tabulated distribution law (LF = 1) be used for secondary energy distribution for both LTT = 1 and 2.

12. FILE 7, THERMAL NEUTRON SCATTERING LAW DATA

12.1 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 given in this file must be combined with data given in Files 2 and 4 (MT = 2) to obtain the total scattering cross sections for a certain materials.

Inelastic scattering is represented by the thermal neutron scattering law, $S(\alpha,\beta, T)$, and is defined (for a moderating molecule) by the equation

$$\frac{d^2\sigma}{d\Omega\, dE'}\ (E \rightarrow E', \mu, T) = \sum_{n=0}^{NS} \frac{M_n\, \sigma_{bn}}{4\pi T} \sqrt{\frac{E'}{E}}\ e^{-\beta/2}\ S_n(\alpha, \beta, T)$$

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 (^oK),

$\underline{E}$ 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 mass of the n^{th} type atom, A_o is the mass of the principle scattering atom in the molecule,

$\underline{\sigma_{bn}}$ is the bound atom scattering cross section of the n^{th} type atom,

$$\sigma_{bn} = \sigma_{fn}\,(A_n + 1)^2 / A_n^2$$

$\underline{\sigma_{fn}}$ is the free atom scattering cross section of the n^{th} type atom,

$\underline{k}$ is Boltzmann's constant,

$\underline{\mu}$ is the cosine of the scattering angle (in the laboratory system).

The data given in File 7 for any particular material only contains the scattering law for the principle scatterer, $S_o(\alpha,\beta,T)$, 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. It should be noted that the scattering properties of all atoms in the molecule may be represented by analytic functions. In this case there is no principle scattering atom.

The constants required for the scattering law data and the analytic representations for the non-principle 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 principle scattering atom, n = 0. The elements of the array, B(N) are defined as:

B(1) = $M_0\sigma_{f0}$, is the total free atom cross section for the principle scattering atom. If B(1) = 0.0, there is no principle scattering atom and the scattering properties for this material is completly 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_0, the ratio of the mass of the atom to that of the neutron that was used to compute α ($\alpha = (E' + E - 2\mu\sqrt{EE'})/A_0kT$)

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 are used to specify the analytic functions that are to be used to describe the scattering properties of the first non-principle scattering atom, (n = 1), i.e., for H_2O, this atom would be oxygen if the principle atom was hydrogen.

B(7) = a_1, is 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.

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B(8) = $M_1 \sigma_{fl}$, is the total free atom cross section for this atom type.

B(9) = A_1, effective mass for this atom type.

B(10)= 0.0, i.e., B(10) is not used.

B(11)= 0.0, i.e., B(11) is not used.

B(12)= 0.0, i.e., B(12) is not used.

The next six constants, B(13) through B(8) are used to describe the second non-principle 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 (^{o}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 and $S(\alpha, \beta)$ are given. (The data are given in this form only for the first temperature, see Appendix F for the formats for temperature dependent data). Three interpolation schemes are given to interpolate between values of β, α, and T.

In certain cases a more accurate temperature interpolation 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 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.

12.2 Formats

There is only one section in File 7, but the format is slightly different depending on whether temperature dependent data is 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.

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NS is the number of non-principle scattering atom types. For most moderating materials there will be (NS+1) types of atoms in the molecule. (NS ≤ 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 in Section 12.2.

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(1), B(2), ...B(NI)]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 is completely specified by analytic functions (no principle scattering atom type as indicated by B(1) = 0), tabulated values of $S_0(\alpha, \beta)$ are omitted. In this case the TAB2 and TAB1 records are not given.

12.3 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 given in File 3 shall at least cover the same energy range (constant, B(4)) as the scattering law data, $S_o(\alpha,\beta)$. 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.

It should be noted that the differential scattering cross section, as given in the equation in Section 12.1, 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 $N_n\sigma_{bn}$ by σ_{bn}.

13. PHOTON INTERACTION AND PHOTON PRODUCTION DATA

Formats have been established for storage of photon production (due to neutron interactions) and photon interaction data. The formats and procedures for entering these data into the ENDF/B library are described in Volume II of this report (Volume II has been written by D. Dudziak[1] and will be issued separately by Los Alamos Scientific Laboratory).

(1) D. Dudziak, LA-4549 (ENDF 102 Vol. II).

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A manual such as this could not have been prepared without the advice, cooperation, and assistance of numerous individuals. Special recognition is given to Henry C. Honeck who was the originator of the ENDF/B data library system and was the author of the original report upon which the present manual is based. Also special recognition is due Mulki Bhat and Sol Pearlstein who have made significant contributions to many parts of this manual. In addition Dr. Bhat is acknowledged for writing all parts of this manual pertaining to resonance region formats, procedures and formulae.

Almost all members of the Cross Section Evaluation Working Group (CSEWG) have made valuable contributions to this manual. In particular, the following individuals are recognized for their leadership in establishing the procedures in the indicated areas: Peter Aline (File 1), Tom Stephenson (File 2), Ed Pennington (File 3), Charles Dunford (File 4), Cecil Lubitz (File 5), and Don Finch (File 7). Many others, too numerous to mention,also contributed significantly to the preparation of this manual.

In addition, the clerical assistance provided by J. Wasson, J. Redzinak and G. Irving has been very much appreciated.

APPENDIX A

Glossary

The following is a list of symbols and variable names that are used to represent specific types of data in the ENDF format. These quantities are alphabetically ordered and the section numbers in which they are defined are given below.*

Name	Section	Name	Section
a	10.2	AJ	7.2
α_{int}	12.2	AM	7.2
A	7.3	AMUF	7.3
ABN	7.1	AMUG	7.3
AC_n	7.2	AMUN	7.3
AF_n	7.2	AMUX	7.3

*This appendix will be replaced at a later date.

Name	Section	Name	Section
AP	7.2	DWT	7.2
AT_n	7.3	E	5.1
AWR	4.1	$E_,$	9.2
AWRI	7.2	E'	10.2
b	10.2	E_{int}	6.2
β_{int}	12.2	E'_{int}	10.2
B(N)	12.2	EH	7.1
BC_n	7.2	EL	7.1
BF_n	7.2	ER	7.2
BT_n	7.2	ES(N)	7.3
C_n	6.2	$f_R(E \rightarrow E')$	10.1
$C_n(E_i)$	6.4	$f_\ell(E_n)$	9.2
CD_m	6.5	$f_\ell(E \rightarrow E')$	11.2
CØNT	5.2	FEND	5.2
D	7.3	$g(x_i)$	10.2
D_n	6.3	$g(E_i \rightarrow E'_n)$	10.2
DC1	6.3	GF	7.2, 7.3
DC2	6.3	GFA	7.2
DEC	7.2	GFB	7.2
DEF	7.2	GG	7.2, 7.3
DET	7.2	GIC	7.2
DWC	7.2	GIF	7.2
DWF	7.2	GIT	7.2

Name	Section
GN	7.2
GNO	7.3
GRC	7.2
GRF	7.2
GRT	7.2
GT	7.2
GX	7.3
H(N)	6.1
HEAD	5.2
I_i	6.4
INT	4.3
L	7.2
λ_m	6.5
LA	11.1
LAT	12.2
LCT	9.2
LDD	6.1
LE	6.4
LF	10.2
LFI	6.1
LFP	6.1
LFS	8.2
LFW	7.1
LI	7.2

Name	Section
LIST	5.2
LND	6.5
LNU	6.2
LRF	7.1
LRP	6.1
LRU	7.1
LT	5.1, App.-F
LTT	9.2
LVT	9.2
μ	5.1
μ_{int}	9.2
MAT	4.1
MEND	5.2
MF	4.1
MF_n	6.1
MT	App.-B
MT_n	6.1
MUF	7.3
$\bar{\nu}(E)$	6.2
$\nu_d(E)$	6.5
NB	12.2
NBT(M)	4.3
NC	6.2

Name	Section
NC_n	6.1
NCD	6.5
NE	7.3, 9.2
NER	7.1
NF	10.2
NFP	6.4
NIS	7.1
NJS	7.2, 7.3
NK	9.2, 10.2
NL	9.2, 12.2
NLJ	7.2
NLS	7.2
NM	9.2
NNF	6.5
NP	5.1
NR	5.1
NRS	7.2
NRT	6.3
NS	12.2
NWD	6.1
NX	7.2
NXC	6.1

Name	Section
$p(E)$	10.2
$p(\mu, E_i)$	9.2
$p(E, \mu_i)$	11.2
$p(E_i \rightarrow E', \mu_j)$	11.2
$P_k(E_n)$	10.2
$P_\ell(E)$	11.2
$P_\ell(E_i \rightarrow E')$	11.2
Q	8.2
RTYP	6.3
$S(\alpha, \beta_i)$	12.2
SEND	5.2
SPI	7.2, 7.3
T	5.1, App.-F
θ (E)	10.2
TAB1	5.2
TAB2	5.2
TEND	5.2
U_{nm}	9.2
U	10.2
V_k	9.2
X_i	10.2
YLD	6.4

Name	Section
ZA	4.1, App.-C
ZA1	6.3
ZA2	6.3
ZA3	6.3
ZAFP	6.4
ZAI	7.1
[ ]	5.1
< >	5.1

APPENDIX B

Definition of Reaction Types

Reaction types are identified by an integer, MT. A list of the allowed reaction types are given below. The reaction type number (MT) generally refers to a specific neutron-nucleus interaction mechanism, but occasionally it designates 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
16	(n,2n) cross section
17	(n,3n) cross section
18	Total fission cross section (sum of MT = 19, 20, 21, plus any undefined part)
19	(n,f) cross section
20	(n,n´f) cross section
21	(n,2nf) cross section
22	(n,n´)α cross section
23	(n,n´)3α cross section
24	(n,2n)α cross section
25	(n,3n)α cross section
26	(to be assigned)
27	Absorption cross section (sum of MT = 18 and 102) NOTE: MT = 27 no longer used
28	(n,n´)p cross section
29	Scattering (sum of MT = 2 and 4) NOTE: MT = 29 no longer used
30-50	(to be assigned)
51	(n,n´) to the 1st excited state
52	(n,n´) " " 2nd " "
⋮	
90	(n,n´) " " 40th " "
91	(n,n′) to the continuum

MT	Description
92-100	(to be assigned)
101	Parasitic absorption (redundant, sum of MT = 102, 103, 104, 105, 106, 107, 108, 109)
102	(n,γ) radiative capture cross section
103	(n,p) cross section
104	(n,d) cross section
105	(n,t) cross section
106	(n,He^3) cross section
107	(n,α) cross section
108	(n,2α) cross section
109	(n,3α) cross section
110-150	(to be assigned)
151	General designation for resonance information
152-200	(to be assigned for specific resonance information)
201-250	(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 (only given in File 1)
452	$\bar{\nu}$, average total (prompt plus delayed) number of neutrons released per fission event
453	~~Radioactive decay chain data~~ Induced reaction branching ratios
454	Fission product yield data
455	Delayed neutrons from fission
456-699	(to be assigned)
456	$\bar{\nu}_{prompt}$
457	Radioactive decay data

MT	Description
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 " "
703	(n,p_3) " " " 3rd " "
704	(n,p_4) " " " 4th " "
⋮	
718	(n,p_{18}) " " " 18th " "
719	(n,p_c) " " " continuum
720	(n,d_0) cross section for ground state
721	(n,d_1) cross section for 1st excited state
722	(n,d_2) " " " 2nd " "
⋮	
738	(n,d_{18}) " " " 18th " "
739	(n,d_c) " " " continuum
740	(n,t_0) cross section for ground state
741	(n,t_1) cross section for 1st excited state
742	(n,t_2) " " " 2nd " "
⋮	
758	(n,t_{18}) " " " 18th " "
759	(n,t_c) " " " continuum
760	(n,He_0^3) cross section for ground state
761	(n,He_1^3) cross section for 1st excited state
⋮	
779	(n,He_c^3) " " " continuum
780	(n,α_0) cross section for ground state
781	(n,α_1) cross section for 1st excited state
⋮	
799	(n,α_c) " " " continuum
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 U-238 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 materials, 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)

ZA	Material
100	Water, H_2O
101	Heavy water, D_2O
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	Beryllea, 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	(to be assigned)
304	304-type stainless steel
305-309	(to be assigned)
310	Uranium-dioxide, UO_2
311-314	(to be assigned)
315	Uranium-carbide, UC
316-399	(to be assigned)
400	U-233 Fission Products (rapidly saturating) for thermal reactors
401	U-235 " " " " " " "
402	Pu-239 " " " " " " "
403	Pu-241 " " " " " " "
404	Th-232 " " " " " " "
405	U-238 " " " " " " "
406	Pu-240 " " " " " " "

ZA	Material
407-409	(to be assigned)
410	U-233 Fission Products (slowly saturating) for thermal reactors
411	U-235 " " " " " " "
412	Pu-239 " " " " " " "
413	Pu-241 " " " " " " "
414	Th-232 " " " " " " "
415	U-238 " " " " " " "
416	Pu-240 " " " " " " "
417-419	(to be assigned)
420	U-233 Fission Products (non-saturating) for thermal reactors
421	U-235 " " " " " " "
422	Pu-239 " " " " " " "
423	Pu-241 " " " " " " "
424	Th-232 " " " " " " "
425	U-238 " " " " " " "
426	Pu-240 " " " " " " "
427-429	(to be assigned)
430	U-233 Fission Products (rapidly saturating) for fast reactors
431	U-235 " " " " " " "
432	Pu-239 " " " " " " "
433	Pu-241 " " " " " " "
434	Th-232 " " " " " " "
435	U-238 " " " " " " "
436	Pu-240 " " " " " " "
437-439	(to be assigned)
440	U-233 Fission Products (slowly saturating) for fast reactors
441	U-235 " " " " " " "
442	Pu-239 " " " " " " "
443	Pu-241 " " " " " " "
444	Th-232 " " " " " " "
445	U-238 " " " " " " "
446	Pu-240 " " " " " " "

ZA	Material
447-449	(to be assigned)
450	U-233 Fission Products (non-saturating) for fast reactors
451	U-235 " " " " " "
452	Pu-239 " " " " " "
453	Pu-241 " " " " " "
454	Th-232 " " " " " "
455	U-238 " " " " " "
456	Pu-240 " " " " " "
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] omitting 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

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,n}^{\ell}(E),$$

where

$$\sigma_{n,n}^{\ell}(E) = (2\ell+1)\frac{4\pi}{k^2}\sin^2\varphi_\ell$$

$$+ \frac{\pi}{k^2}\sum_J g_J \sum_{r=1}^{NR_J} \frac{\Gamma_{nr}^2\cos 2\varphi_\ell - 2\Gamma_{nr}(\Gamma_{\gamma r}+\Gamma_{fr})\sin^2\varphi_\ell + 2(E-E_r')\Gamma_{nr}\sin 2\varphi_\ell}{(E-E_r')^2 + \frac{1}{4}\Gamma_r^2}$$

2. Radiative Capture Cross Section

$$\sigma_{n,\gamma}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,\gamma}^{\ell}(E)$$

where

$$\sigma_{n,\gamma}^{\ell}(E) = \frac{\pi}{k^2}\sum_J g_J \sum_{r=1}^{NR_J} \frac{\Gamma_{nr}\Gamma_{\gamma r}}{(E-E_r')^2 + \frac{1}{4}\Gamma_r^2}$$

*Several processing codes have been developed to calculate cross sections using the formulae given in this appendix. These codes are given in Appendix I.

3. Fission Cross Section

$$\sigma_{n,f}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,f}^{\ell}(E) ,$$

where

$$\sigma_{n,f}^{\ell}(E) = \frac{\pi}{k^2} \sum_{J} g_J \sum_{r=1}^{NR_J} \frac{\Gamma_{nr}\Gamma_{fr}}{(E-E_r')^2 + \frac{1}{4}\Gamma_r^2} ,$$

where

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

I is the spin of the target nucleus and J is the spin of the compound nucleus for the resonance state.

I = SPI, as given in File 2 data for each isotope

The summation on ℓ extends over all ℓ-states described. There will be NLS terms in the summation.

NLS is given in File 2 for each isotope

The summation on J extends over all possible J-states for a particular ℓ-state. NR_J 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.

$$\Gamma_{nr} = \frac{P_\ell(E)\ \Gamma_{nr}(|E_r|)}{P_\ell(|E_r|)}$$

$\Gamma_r = \Gamma_{nr}(E) + \Gamma_{\gamma r} + \Gamma_{fr}$ is the total width.

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

$$E_r' = E_r + \frac{S_\ell(|E_r|) - S_\ell(E)}{2P_\ell(|E_r|)}\ \Gamma_{nr}(|E_r|)$$

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

where k is the neutron wave number and AWRI is the ratio of the mass of the particular isotope to that of the neutron.

<u>AWRI given in File 2 data for each isotope</u>

E is the incident neutron energy (Laboratory system);

S_ℓ is the shift factor,

$$S_0 = 0$$

$$S_1 = -\frac{1}{1 + \rho^2}$$

$$S_2 = -\frac{18 + 3\rho^2}{9 + 3\rho^2 + \rho^4}$$

P_ℓ is the penetration factor,

$$P_0 = \rho$$

$$P_1 = \frac{\rho^3}{1 + \rho^2}$$

$$P_2 = \frac{\rho^5}{9 + 3\rho^2 + \rho^4}$$

where $\rho = ka$ and "a" is the channel radius (in units of 10^{-12} cm) and is defined as

$$a = [1.23(AWRI)^{1/3} + 0.8] \times 10^{-1} \ ;$$

φ_ℓ is the phase shift,

$$\varphi_0 = \hat{\rho}$$

$$\varphi_1 = \hat{\rho} - \tan^{-1}\hat{\rho}$$

$$\varphi_2 = \hat{\rho} - \tan^{-1}\frac{3\hat{\rho}}{3-\hat{\rho}^2} \ ,$$

where $\hat{\rho} = k\hat{a}$ and $\hat{a}$ is the effective scattering radius.

$\hat{a}$ = AP, as given in File 2 data

D.1.2. Multilevel Breit-Wigner Formula: LRU=1, LRF=2

The equations are exactly the same as above, except that a level-level interference term is included in the equation for elastic scattering:

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

D.1.3. Reich-Moore Formulae

A detailed derivation of these formulae is to be found in a paper by Reich and Moore.[2] Neutron cross sections with an exit channel c are given by*

$$\sigma_{nc} = \pi \lambda\!\!\!^{-2}_{n} \sum_{J} g_J \left| \delta_{nc} - U^J_{nc} \right|^2 , \qquad (1)$$

where $\lambda\!\!\!^{-}_{n}$ is calculated in the center-of-mass system; and

$$\frac{1}{\lambda\!\!\!^{-}_{n}} = k_n = 2.196771 \left(\frac{AWRI}{AWRI+1.0} \right) \times 10^{-3} \sqrt{E(eV)} , \qquad (2)$$

where AWRI is the mass of the target nucleus in units of neutron mass. The statistical factor

$$g_J = \frac{(2J+1)}{2(2I+1)} , \qquad (3)$$

where J is the spin of the compound nucleus resonance and I is the target nucleus spin.

In terms of the Reich-Moore approximation one may write

$$U^J_{nc} = e^{-i(\varphi_n+\varphi_c)} \left\{ 2[(I-K)^{-1}]_{nc} - \delta_{nc} \right\} , \qquad (4)$$

where $(I-K)_{cc'} = \delta_{cc'} - \frac{i}{2} \sum_{\lambda} \frac{\Gamma^{1/2}_{\lambda c} \Gamma^{1/2}_{\lambda c'}}{E_\lambda - E - \frac{i}{2} \Gamma_{\lambda\gamma}}$, (5)

where the summation in Eq. (5) is over the resonance levels λ; E_λ is the resonance energy; $\Gamma_{\lambda\gamma}$, the corresponding radiation widths; and $\Gamma_{\lambda c}$ and $\Gamma_{\lambda c'}$ are the widths for the λ-th level and channels c and c′, respectively.

If we define

$$\rho_{nc} = \delta_{nc} - [(I-K)^{-1}]_{nc} = \delta_{nc} - \frac{m_{nc}}{\Delta} ,$$

*These formulae are to be used for the 0°K case (no Doppler broadening terms given).

where $\Delta = |I-K|$ the determinant of the matrix I-K and m_{nc} is the co-factor of the element $(I-K)_{nc}$ of the matrix I-K, we obtain

$$\sigma_{nT} = \sum_J \sigma_{nT}^J = 2\pi \lambda_n^2 \sum_J g_J \mathrm{Re}\left(1-U_{mn}^J\right)$$

$$= 2\pi \lambda_n^2 \sum_J \left\{ g_J(1-\cos 2\varphi_n) + 2g_J \mathrm{Re}\left(e^{-2i\varphi_n} \rho_{nn}\right)\right\} \quad (6)$$

$$\sigma_{nn} = \Pi \lambda_n^2 \sum_J g_J \left|1-U_{nn}^J\right|^2 \quad (7)$$

$$\sigma_{nAbs} = \sigma_{nT} - \sigma_{nn} = 4\Pi \lambda_n^2 \sum_J g_J \left[\mathrm{Re}\left(\rho_{nn}\right) - \left|\rho_{nn}\right|^2 \right] \quad (8)$$

$$\sigma_{nFiss} = 4\pi \lambda_n^2 \sum_J g_J \left(\sum_c \left|\rho_{nc}\right|^2 \right) \quad (9)$$

$$\sigma_{n\gamma} = \sigma_{nAbs} - \sigma_{nFiss} \quad (10)$$

For s-wave neutrons $\varphi_n = +k_n a$ where k_n has been defined by Eq. 2 and a is the channel radius. For p and d-wave resonances φ_n is defined in section D.1.1.

1. K. Gregson, M. F. James, and D. S. Norton, "MLBW - A Multilevel Breit-Wigner Computer Programme", UKAEA Report AEEW-M-517, March 1965.

2. C. W. Reich and M. S. Moore, Phys. Rev. 111, 929, 1958.

D.1.4. Adler-Adler Multilevel Resonance Parameters: LRU=1, LRF=2

The formulae for obtaining cross sections have been taken from Adler and Adler[3,4]. These are given for the total, radiative captive and fission cross sections (without Doppler broadening).

1. Total Cross Section

$$\sigma_T(E) = \frac{2C}{E}(1-\cos\omega) + \frac{C}{\sqrt{E}}\sum_{R=1}^{NRS}\frac{\nu_R^T\left[G_R^T\cos\omega + H_R^T\sin\omega\right] + \left(\mu_R^T - E\right)\left[H_R^T\cos\omega - G_R^T\sin\omega\right]}{\left(\mu_R^T - E\right)^2 + \left(\nu_R^T\right)^2}$$

$$+ \frac{C}{\sqrt{E}}\left(AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1 * E + BT_2 * E^2\right)$$

2. Captive Cross Section

$$\sigma_{n,\gamma}(E) = \frac{C}{\sqrt{E}}\sum_{R=1}^{NRS}\frac{\nu_R^\gamma\left[G_R^\gamma\cos\omega + H_R^\gamma\sin\omega\right] + \left(\mu_R^\gamma - E\right)\left[H_R^\gamma\cos\omega - G_R^\gamma\sin\omega\right]}{\left(\mu_R^\gamma - E\right)^2 + \left(\nu_R^\gamma\right)^2}$$

$$+ \frac{C}{\sqrt{E}}\left(AC_1 + AC_2/E + AC_3/E^2 + AC_1/E^3 + BC_1 * E + BC_2 * E^2\right)$$

3. Fission Cross Section

$$\sigma_{n,f}(E) = \frac{C}{\sqrt{E}}\sum_{R=1}^{NRS}\frac{\nu_R^f\left[G_R^f\cos\omega + H_R^f\sin\omega\right] + \left(\mu_R^f - E\right)\left[H_R^f\cos\omega - G_R^f\sin\omega\right]}{\left(\mu_R^f - E\right)^2 + \left(\nu_R^f\right)^2}$$

$$+ \frac{C}{\sqrt{E}}\left(AF_1 + AF_2/E + AF_3/E^2 + AF_4/E^3 + BF_1 * E + BF_2 * E^2\right)$$

$\omega = 2\ k\hat{a}$

where k is the neutron wave number

$$k = 2.196771\left(\frac{AWRI}{AWR + 1.0}\right) \times 10^{-3}\sqrt{E(eV)}$$

and

$\hat{a}$ = AP = effective scattering radius (in units of 10^{-12} cm) and

$$\frac{C}{E} = \pi \lambda\!\!\!^{-2} = \frac{\pi}{k^2}$$

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: $\bar{D}_{\ell,J}$, $\bar{\Gamma}^{\circ}_{n_{\ell,J}}$, $\bar{\Gamma}_{\gamma_{\ell,J}}$, $\bar{\Gamma}_{f_{\ell,J}}$. The parameters are for a single-level Breit-Wigner formula with interference. The widths are 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 Resonance Parameters for the Unresolved Region

The following gives a few definitions and formulae of use in the unresolved resonance region.

a. Level Spacing

The experimental value of mean spacing between resonances is determined as

$$D_{observed} = \frac{\Delta E_n}{\text{No. of resonances of given } \ell} \quad (1)$$

where ΔE_n is the neutron energy interval and ℓ is the angular momentum of the incident neutron. In using the above equation it is assumed that (a) corrections have been made for missed levels or (b) only that part of the energy range where a plot of the level position versus level number is linear. For most of the nuclei this quantity is determined by looking at the s-wave resonances.

If we assume that

$$\rho_J \sim (2J + 1)$$

where ρ_J is the density of compound nucleus levels of spin J, then

$$\frac{1}{D_{obs}} = \rho_{obs} = \Sigma \rho_J \quad (2)$$

If in addition I is the spin of the target nucleus, one can show that

$$D_J = \frac{D_o}{(2J + 1)} \quad \text{where } D_o = D_{obs} \times 2 \times (2I + 1) \times (2\ell + 1) \quad (3)$$

In the above we have neglected the exponential factors in the level density formula to get a simple expression. From these expressions we can calculate the level-spacing for the two sets of s-wave resonances.

Further, we assume that the level-spacing is independent of the neutron angular momentum ℓ and derive the spacings for the p and d-wave neutrons.

b. Neutron Widths

Since most of the resolved resonance data give the s - wave neutron widths for resonances of two spin states we can determine a mean neutron width. Suitable corrections have to be applied for missing levels. Or, the experimentor might give an s-wave strength-function. If we assume that the s-wave strength-function is independent of J we can write

$$\frac{\langle \Gamma_n^o \rangle_{J_1}^{\ell=0}}{\langle D \rangle_{J_1}} = \frac{\langle \Gamma_n^o \rangle_{J_2}^{\ell=0}}{\langle D \rangle_{J_2}} = S_0 \qquad (4)$$

for the two possible spin states J_1 and J_2 for s-wave resonances. Similarly if the p - wave strength-function is known we can write

$$\frac{\langle \Gamma_n^o \rangle_{J_1}^{\ell=1}}{\langle D \rangle_{J_1}} = \frac{\langle \Gamma_n^o \rangle_{J_2}^{\ell=1}}{\langle D \rangle_{J_2}} - - - - - = \frac{\langle \Gamma_n^o \rangle_{J_s}^{\ell=1}}{\langle D \rangle_{J_s}} = S_1 \qquad (5)$$

where J_1 J_2 ------ J_s are the possible spin values for the p-wave resonances. Here we would like to point out that some of the spin-states could be formed via two possible values of channel spin $I+\frac{1}{2}$ and $I-\frac{1}{2}$ and hence the corresponding neutron width could be thought of as following a Porter-Thomas distribution of μ = 2 degrees of freedom. If we calculate D_J in equation(5)from equation(3)we can calculate the corresponding reduced neutron width from the above equation. Here

we would like to point out another procedure which defines the ENDF/B convention. We define the neutron width Γ^{ℓ}_{nJ} for ℓ-wave neutrons and spin J - states as

$$\left\langle \Gamma^{\ell}_{nJ} \right\rangle = \left\langle \Gamma^{o\ell}_{nJ} \right\rangle \sqrt{E} \times V_{\ell} \times \mu \qquad (6)$$

where E is the neutron energy in eV and V_{ℓ} is defined below and μ is the number of degrees of freedom for the neutron width distribution

$V_{\ell}(\rho) = \frac{P_{\ell}(\rho)}{\rho}$ where $\rho = kr$ (k is the neutron wave-number and r the nuclear radius).

For $\ell = 0$ $\quad V_0(\rho) = 1$

$\ell = 1$ $\quad V_1(\rho) = \frac{\rho^2}{1 + \rho^2}$

$\ell = 2$ $\quad V_2(\rho) = \frac{\rho^4}{9 + 3\rho^2 + \rho^4}$

Sometimes it might happen that there are no experimental values of p-wave strength-functions available. In such a case one has to have recourse to the results of some model calculations and their predictions. A list of such calculations are

1. B.Buck and F. Perey. Phys. Rev. Letters 8, 444, 1962.

c. Gamma Widths

In the limited energy range of a few keV usually covered by the unresolved resonance region, the gamma widths may be assumed to be constant and equal to that obtained from an analyses of the resolved resonances. If however, this energy range is rather wide, an energy dependence as given by some of the well-known theoretical models[1] has to be built in. Since, the observed gamma width is a sum of a large number of primary gamma transitions each of which is assumed to have a chi-squared distribution of $\mu = 1$, it is found to have a $\mu \sim 20$ or larger. In effect this implies that the gamma width is a constant, as a chi-squared distribution with a large number of degrees of freedom approximates a δ-function.

1. J. E. Lynn, The Theory of Neutron Resonance Reactions, Chapter VII, Clarendon Press, Oxford (1968).

D.2.2 Cross Sections in the Unresolved Region

a. Elastic Scattering Cross Section

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,n}^{\ell}(E)$$

$$\sigma_{n,n}^{\ell}(E) = \frac{4\pi}{k^2} \sum_{J}^{NJS_\ell} g_J (2\ell+1) \sin^2\phi_\ell + \frac{2\pi^2}{k^2} \sum_{J}^{NJS_\ell} \frac{g_J}{\bar{D}_{\ell,J}} \left[\left\langle \frac{\Gamma_n \Gamma_n}{\Gamma} \right\rangle_{\ell,J} - 2\bar{\Gamma}_{n_{\ell,J}} \sin^2\phi_\ell \right]$$

b. Radiative Captive Cross Section

$$\sigma_{n,\gamma}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,\gamma}^{\ell}(E)$$

$$\sigma_{n,\gamma}^{\ell}(E) = \frac{2\pi^2}{k^2} \sum_{J}^{NJS_\ell} \frac{g_J}{\bar{D}_{\ell,J}} \left\langle \frac{\Gamma_n \Gamma_\gamma}{\Gamma} \right\rangle_{\ell,J}$$

c. Fission Cross Section

$$\sigma_{n,f}(E) = \sum_{\ell=0}^{NLS} \sigma_{n,f}^{\ell}(E)$$

$$\sigma_{n,f}^{\ell}(E) = \frac{2\pi^2}{k^2} \sum_{J}^{NJS_\ell} \frac{g_J}{\bar{D}_{\ell,J}} \left\langle \frac{\Gamma_n \Gamma_f}{\Gamma} \right\rangle_{\ell,J}$$

The summation over ℓ, in the above equations, extends up to $\ell = 2$ or to NLS (where NLS is the number of ℓ-states for which data are given). For each value of ℓ, the summation over J-states extends to NJS_ℓ (the number of J-states for a particular ℓ-state).

NLS and NJS are given in File 2.

$$\left\langle \frac{\Gamma_n \Gamma_n}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \; \bar{\Gamma}_{n_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{n\ell,J}$$

$$\left\langle \frac{\Gamma_n \Gamma_\gamma}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \; \bar{\Gamma}_{\gamma_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{\gamma\ell,J}$$

$$\left\langle \frac{\Gamma_n \Gamma_f}{\Gamma} \right\rangle_{\ell,J} = \left(\frac{\bar{\Gamma}_{n_{\ell,J}} \; \bar{\Gamma}_{f_{\ell,J}}}{\bar{\Gamma}_{\ell,J}} \right) R_{f\ell,J}$$

where $R_{\gamma\ell,J}$, $R_{f\ell,J}$, and $R_{n\ell,J}$ are fluctuation integrals for capture, fission, and elastic scattering, respectively. Associated with each of these integrals is the number of degrees of freedom for each of the average widths.

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

$\mu_{\gamma_{\ell,J}}$ = 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

$$\bar{\Gamma}_{n_{\ell,J}} = \bar{\Gamma}^{o}_{n\ell,J} \sqrt{E} \; V \mu_{n\ell,J}$$

where the penetrabilities, V_ℓ, are

$V_o = 1$ for s-wave neutrons, $\ell = 0$

$V_1 = \dfrac{\rho^2}{1+\rho^2}$ for p-wave neutrons, $\ell = 1$

$V_2 = \dfrac{\rho^4}{9+3\rho^2+\rho^4}$ for d=wave neutrons, $\ell = 2$

The statistical weight factor, g_J , is

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

The average total width, at energy E, is

$$\bar{\Gamma}_{\ell,J} = \bar{\Gamma}_{n\ell,J} + \bar{\Gamma}_{\gamma\ell,J} + \bar{\Gamma}_{f\ell,J} + \bar{\Gamma}_{x\ell,J}$$

where all widths are evaluated at energy, E.

J = AJ as given in File 2

I = SPI as given in File 2

ℓ = L as given in File 2

$$\rho = ka$$

Where k is the neutron wave number,

$$k = 2.196771\left(\frac{AWRI}{AWRI + 1.0}\right) X \ 10^{-3}\sqrt{E\,(eV)}$$

and

"a" is the channel radius (in units of 10^{-12}cm),

$$a = \left[\ 1.23 \ \ (AWRI)^{1/3} \ + \ 0.8\right] X \ 10^{-1}.$$

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

AWRI is given in File 2.

ϕ_ℓ is the phase shift and

$$\phi_0 = \hat{\rho} \qquad \ell = 0$$

$$\phi_1 = \hat{\rho} - \tan^{-1}\hat{\rho} \qquad \ell = 1$$

$$\phi 2 = \hat{\rho} - \tan^{-1}\left(\frac{3\hat{\rho}}{3-\hat{\rho}^2}\right) \qquad \ell = 2$$

where

$$\hat{\rho} = k\,\hat{a}$$

and $\hat{a}$ is the effective scattering radius (in units of 10^{-12}cm).

$\hat{a}$ = A as given in File 2.

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), Y(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 that are 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 that is 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 the point number 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-dimension 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 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 and each of these records 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.

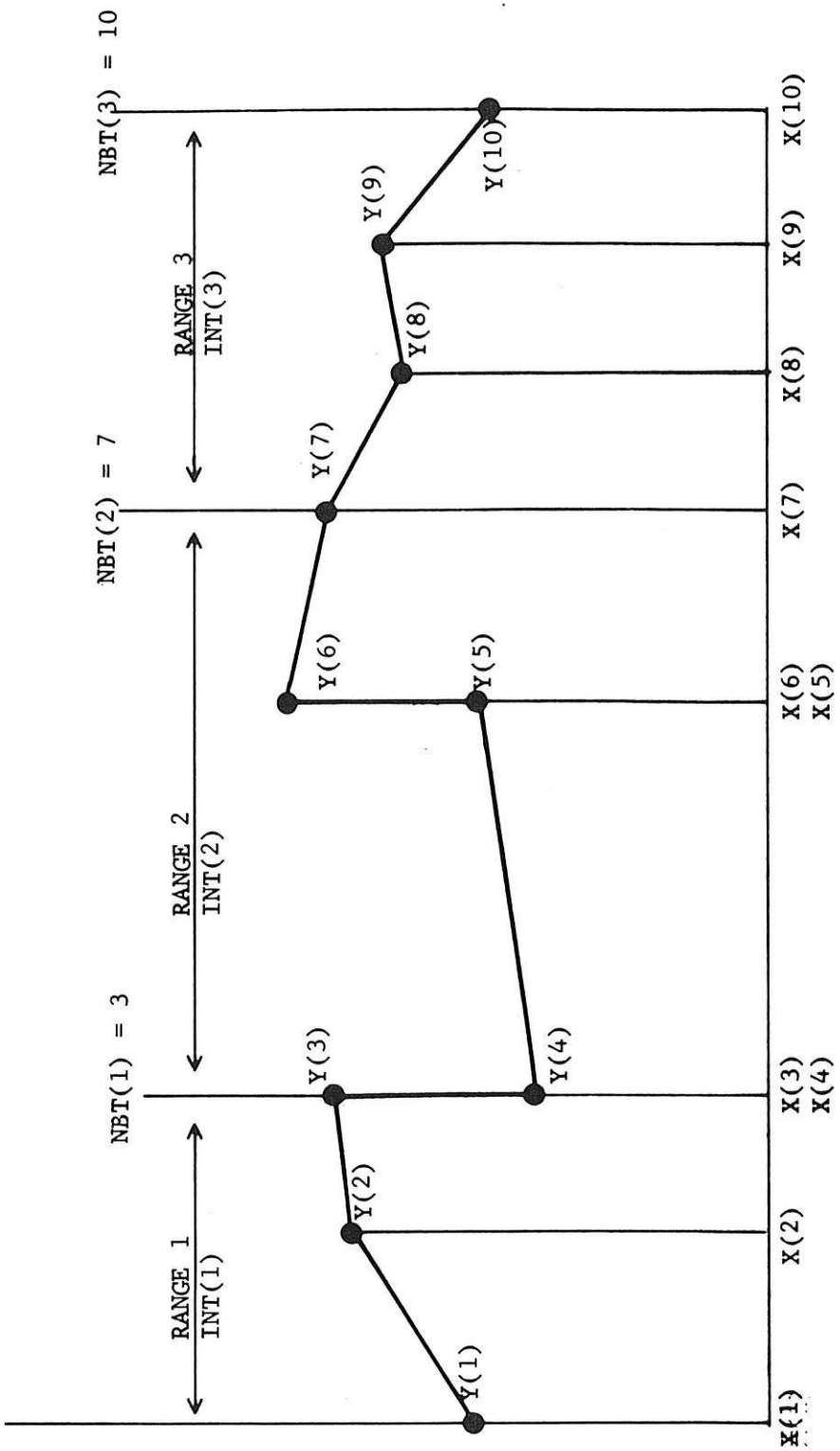

Figure E.1. Tabulated one dimensional function illustrated for the case NP-10, NR-3

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 is given 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, i.e., 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:

$$NP_1 \geq NP_2 \geq NP_3 \geq \ldots \geq NP_{LT+1} .$$

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.

[MAT, MF, MT/ T_1 , C2; LT , L2; NP_1 , 0/ $B_n(T_1)$]LIST
[MAT, MF, MT/ T_2 , C2; I_2 , L2; NP_2 , 0/ $B_n(T_2)$]LIST
[MAT, MF, MT/ T_3 , C2; 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 about NP, i.e.,

$$NP_1 \geq NP_2 \geq NP_3 \geq \ldots \geq NP_{LT+1} .$$

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

Alternate Structure for ENDF Data Tapes

The standard structure of an ENDF tape was described in Section 4.2 of this report. The standard structure is well suited for BCD (card image) and binary tapes.

An alternate structure of the ENDF data tapes has been developed for use in certain cross section processing codes. This alternate arrangement is illustrated in Fig. G.1 and 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).

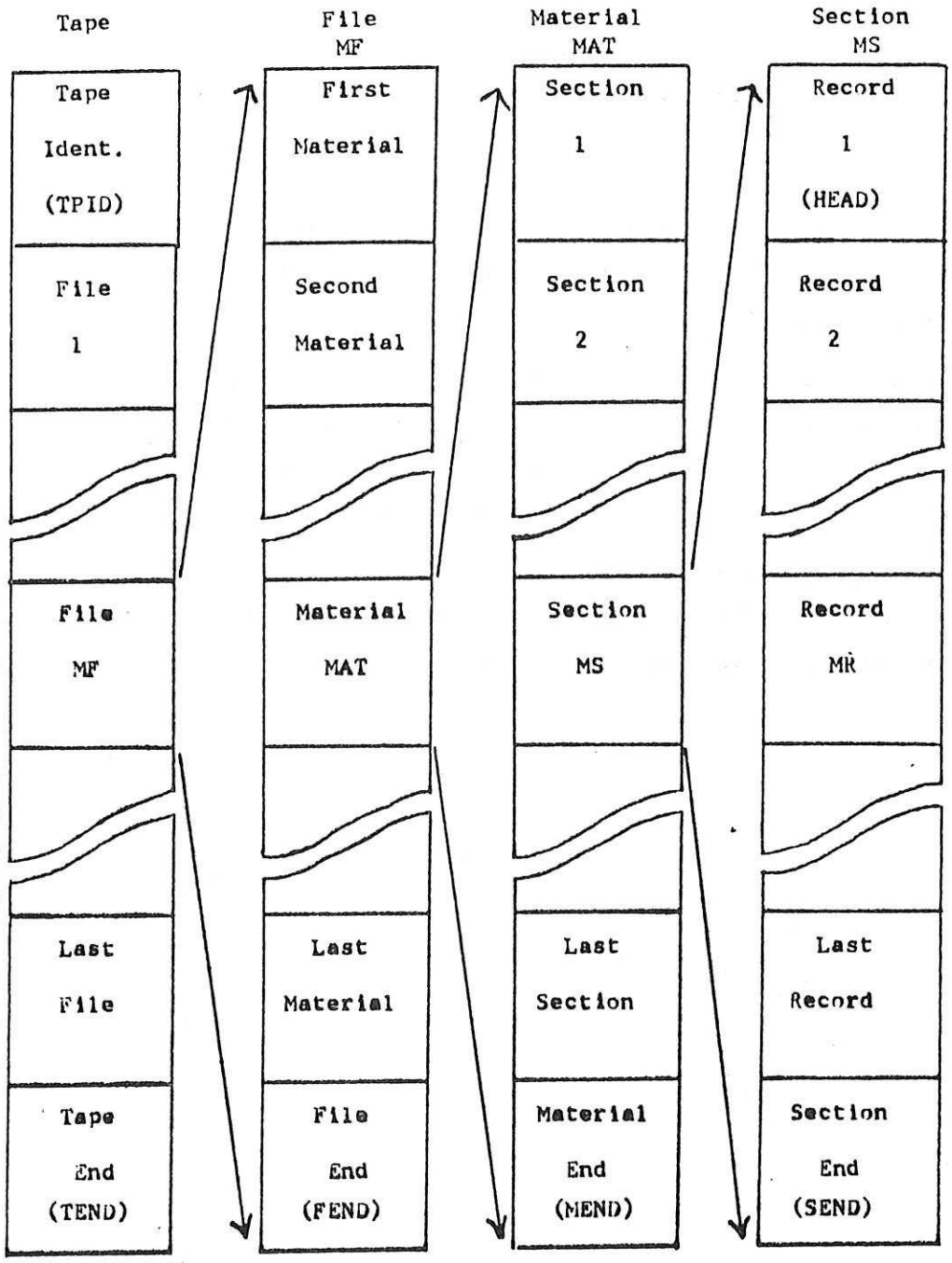

Figure G.1. Alternate Arrangement of an ENDF Tape

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, ETØE will be able to read the data tape regardless of 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, NTY, NXC] HEAD

[MAT, 1, 451/0.0, 0.0, LDD, LFP, NWD, 0/

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, 0]CØNT

[MAT, 1, 451/0.0, 0.0, MF_{NXC}, MT_{NXC}, NC_{NXC}, 0] CØNT

[MAT, 1, 0/0.0, 0.0, 0, 0, 0, 0]SEND

where

NTY is flag to indicate the type of data tape.

If NTY = 0 or bank - ENDF/B tape,

= 1 - ENDF/A tape,

= 2 - ENDF/A tape (translated from UK-AEA library),

= 3 - ENDF/A tape (translated from KEDAK library).

The first pat of the Hollerith information (first two BCD card records) has the structure:

Field	Cols.	Name	Description
		(First Card)	
1	2-11	AID	Material name (left adjusted)
2	12-22	ALAB	Originating laboratory (left adjusted)
3	23-33	DATE1	Date of evaluation (left adjusted)
4	34-66	AUTH	Author of evaluation (left adjusted)
		(Second Card)	
1	2-22	REF	Reference (left adjusted)
2	23-33	DATE2	Original distribution date (left adjusted)
3	34-44	DATE3	Date of last revision (left adjusted)
4	45-55	EMIN*	Lower limit of energy range (format is E11.4)
5	56-66	EMAX*	Upper limit of energy range (format is E11.4)

NWD has the same meaning as an ENDF/B tape, i.e., 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 of this material. NWD includes the count of the first two BCD card images. An example is given on the next page.

*Only given for materials that contain cross section data for one reaction type.

```
9.0232E+04 2.3004E+02                  1          1          1          3711117 1451    1
                                       0          0         51            1117 1451    2
TH-232    B AND W     EVAL-NOV66 WITTCOPF, ROY, AND LOVOLSI                1117 1451    3
BAW-317 (1970)        DIST-MAY67 REV-APR70                                 1117 1451    4
*         *           *          *          *                              1117 1451    5
 THORIUM-232       (B AND W)     WITTCOPF, ROY, AND LIVOLSI                1117 1451    6
          *           *          *          *                              1117 1451    7
           DATA COMPILED NOV., 1966 AND MODIFIED MAY, 1969                 1117 1451    8
           EVALUATION DESCRIBED IN BABCOCK AND WILCOX REPORT,              1117 1451    9
           BAW-317                                                         1117 1451   10
          *           *          *          *          *                   1117 1451   11
ENERGY RANGE (0.00001 TO 10 EV,                                           1117 1451   12
   TOTAL X/S ACCORDING TO MEMO FROM B.R. LEONARD TO CSEWG ON AUG,         1117 1451   13
   8, 1969,  A VALUE OF 10.15 B FOR THE POTENTIAL X/S WAS PROVID          1117 1451   14

   THE (N,2N) X/S WAS OBTAINED AS AN UNWEIGHTED AVERAGE THROUGH           1117 1451   50
   SEVERAL EXPERIMENTAL DATA SETS PUBLISHED BETWEEN 1964 AND 1956         1117 1451   51
   THE (N,3N) X/S WAS OBTINED FROM THE WORK OF M.H.TAGGART AND            1117 1451   52
   H. GOODFELLOW, JNE 17,437(1963),                                       1117 1451   53
                                  1        451         90                 1117 1451   54
                                  1        452          3                 1117 1451   55
                                  2        151        241                 1117 1451   56
                                  3          1        131                 1117 1451   57
                                  3          2        114                 1117 1451   58
                                  3          4         23                 1117 1451   59
                                  3         16          8                 1117 1451   60
                                  3         17          5                 1117 1451   61
                                  3         18         14                 1117 1451   62
                                  3         51         18                 1117 1451   63
                                  3         52         16                 1117 1451   64
                                  3         53         13                 1117 1451   65
                                  3         54         11                 1117 1451   66
                                  3         55         10                 1117 1451   67
                                  3         56          9                 1117 1451   68
                                  3         57          8                 1117 1451   69
                                  3         58          8                 1117 1451   70
                                  3         91         13                 1117 1451   71
                                  3        102        114                 1117 1451   72
                                  3        251         13                 1117 1451   73
                                  3        252         13                 1117 1451   74

                                  5         16          8                 1117 1451   86
                                  5         17          7                 1117 1451   87
                                  5         18          7                 1117 1451   88
                                  5         91         12                 1117 1451   89
                                  7          4          4                 1117 1451   90
                                                                          1117 1  0   91
```

APPENDIX I

Summary of Processing Codes Used with the ENDF Library

The following is a table listing computer codes which are directly related to the ENDF library and cognizant individual at originating laboratory. This list is not complete and includes only the more general codes.

Code Name	Description, Name (Laboratory)
ADLER	Calculates unshielded resonance cross sections using Adler-Adler multilevel formula, M.R. Bhat (BNL).
AVERAGE-II	Calculates infinitely-dilute cross sections using unresolved resonance parameters, M.R. Bhat (BNL).
BMCLIB	Generates BMC (Battelle Monte Carlo) library, D.H. Thompson (BNW).
CHECKER	Detects errors in ENDF formated data, D.E. Cullen (BNL). Available from ANL Code Center.
CRECT	Corrects ENDF data tapes, D.E. Cullen (BNL). Available from ANL Code Center.
DAMMET	Delete, alter mode, and merge data for ENDF library, D.E. Cullen (BNL). Available from ANL Code Center. This code will be replaced by RIGEL.
EDIT	Read, write, and plot data from ENDF library, R. Hubner (AI)
ENDF-GAND	Generates GAF-GAR library, D. Mathews (GGA) Available from ANL Code Center.
ENDRUN	Generate T DOWN library, B. Hutchins (GE-BRDO).
ENDT	Generate UNSAM-3 library, S. Kellman (WNES). Available from RSIC.
ETOE	Generate MC^2 library. E. Pennington (ANL). Available from ANL Code Center.

Code Name	Description, Name (Laboratory)
ETØG-1	Generate MUFT, GAM, ANISN libraries, S. Kellman (WNES). Available from ANL Code Center.
ETØJ	Prepare library for JØSHUA system. D. Finch (SRL).
ETØL	Generate photon production library for LAPH. D.J. Dudziak (LASL).
ETØM-1	Generate MUFT library. R.A. Dannels (WNES). Available from ANL Code Center.
ETØX	Generate IDX and FCC-IV libraries, R.B. Kidman (BNW).
FLANGE-II	Generates scattering kernel data from $S(\alpha,\beta)$, D. Finch (SRL), J. Neill (GGA). Available from ANL Code Center.
GENSIG	Prepare photon library data for ØGRE program, D. K. Trubey, S.K. Penny (ORNL). Available from RSIC.
NUSECT	Generate Ø6R library, E. Straker, S.K. Penny (ORNL).
PHØX	Physics checking code for photon production data, D. Dudziak (LASL). Available from RSIC.
PLØTFB	List and plot ENDF data, D.E. Cullen (BNL). Available from ANL Code Center.
RIGEL	Delete, alter mode, and merge ENDF data tapes, D.E. Cullen (BNL). Available from ANL Code Center.
SIGPLØT	Calculate Doppler-broadened cross sections from single level and multi-level Breit-Wigner formulas, M.R. Bhat (BNL).
SUPERTØG	Generate GAM, ANISN, DØT, and MØRSE libraries, D. Jenkins (ORNL). Available from RSIC.
UKE	Translates from UK-AWR data to ENDF formate data, D.K. Trubey (ORNL). Available from RSIC.
LISTFC	Generates an interpreted listing of an ENDF tape.

APPENDIX J

Materials in the ENDF/B-II Library

The following is a list of materials that are available in the ENDF/B-II library. This list of materials will be expanded as more evaluated data sets become available. The current contents of the ENDF/B-II library will be published from time to time in the NNCSC Newsletter.

Moderating Materials (Scattering Law Data)

Material	MAT Number	Temperatures (^{o}K)
H_2O	1002	296, 350, 400, 450, 500, 600, 800, 1000
D_2O	1004	296, 350, 400, 450, 500, 600, 800, 1000
Beryllium	1064	296, 400, 500, 600, 700, 800, 1000, 1200
BeO	1099	296, 400, 500, 600, 700, 800, 1000, 1200
Graphite	1065	296, 400, 500, 600, 700, 800, 1000, 1200, 1600, 2000
Polyethylene	1111	296, 350
Benzene	1095	296, 350, 400, 450, 500, 600, 800, 1000
H in ZrH	1097	296, 400, 500, 600, 700, 800, 1000, 1200
Zr in ZrH	1096	296, 400, 500, 600, 700, 800, 1000, 1200

Material	MAT Number	ENDF/B-II Tape Number
H	1001	202
D	1120	204
He	1088	204
Be	1007	204
B-10	1009	202
C	1140	202
O	1013	202
Na	1059	202
Mg	1014	202
Al	1015	202
Ti	1016	202
V	1017	202
Cr	1121	203
Mn	1019	202
Fe	1122	203
Ni	1123	203
Cu	1087	202
Cu-63	1085	202
Cu-65	1086	202
Nb	1112	203
Mo	1111	203
Xe-135	1026	204
Sm-149	1026	204
Eu-151	1028	204
Eu-153	1029	204
Gd	1030	204

Material	MAT Number	ENDF/B-II Tape Number
Dy-164	1031	204
Lu-175	1032	204
Lu-176	1033	204
Ta-181	1035	204
W-182	1060	204
W-183	1061	204
W-184	1062	204
W-186	1063	204
Re-185	1083	204
Re-187	1084	204
Au-197	1037	204
Th-232	1117	201
U-233	1041	201
U-234	1043	201
U-235	1102	201
U-236	1046	201
U-238	1103	201
Np-237	1048	204
Pu-238	1050	201
Pu-239	1104	201
Pu-240	1105	201
Pu-241	1106	203
Pu-242	1055	201
AM-241	1056	204
AM-243	1057	204

Material	MAT Number	ENDF/B-II Tape Number
Cm-244	1058	204
U-233 RSFP*	1042	204
U-235 RSFP	1045	204
Pu-239 RSFP	1052	204
U-233 SSFP**	1066	204
U-235 SSFP	1068	204
Pu-239 SSFP	1070	204
U-233 NSFP***	1067	204
U-235 NSFP	1069	204
Pu-239	1071	204

*Rapidly Saturating Fission Product aggregate (for thermal reactors).

**Slowly Saturating Fission Product aggregate (for thermal reactors).

***Non-Saturating Fission Product aggregate (for thermal reactors).

APPENDIX K

Sample Data Set

The following is a sample data set in the ENDF format. This sample was taken from an evaluation by E. Pennington (Argonne National Laboratory) and the data set contains neutron cross section data for the natural element, helium.

```
2,22222+ 3 3,96822+ 2          2          2          4        121288 1451    1
4,22222+ 3 3,22222+ 2          2          2         77         21288 1451    2
HELIUM     ANL          EVAL-JUN68 E.D.PENNINGTON (NATURAL ELEMENT) 1288 1451    3
ANL-7462 (OCT,1968)  DIST-JUN72 REV-JUN72                         1288 1451    4
 •          •          •          •          •          •          • 1288 1451    5
DATA MODIFIED JUNE,1972 TO CONFORM TO ENDF/B-II FORMATS           1288 1451    6
 •          •          •          •          •          •          • 1288 1451    7
HELIUM CROSS SECTIONS  MATERIAL 1288                              1288 1451    8
COMPILED BY ED PENNINGTON, ARGONNE NATIONAL LAB., IN JUNE 1968,   1288 1451    9
NATURAL HELIUM CONSISTS OF 1,00013 PER CENT HE-3 AND 99,99987 PER1288 1451   10
CENT HE-4,                                                        1288 1451   11
BECAUSE OF THE LOW ABUNDANCE OF HE-3, ONLY ITS (N,P) CROSS        1288 1451   12
SECTION, WHICH IS VERY LARGE AT LOW ENERGIES, NEED BE CONSIDERED.1288 1451   13
ELASTIC SCATTERING IS THE ONLY POSSIBLE REACTION FOR NEUTRONS     1288 1451   14
INCIDENT ON HE-4 AT ENERGIES BELOW 15 MEV, THUS THE ELASTIC       1288 1451   15
SCATTERING CROSS SECTION AND VALUES OF MU BAR(LAB), XI, AND GAMMA1288 1451   16
ARE GIVEN IN FILE 3, AND ELASTIC SCATTERING LEGENDRE COEFFICIENTS1288 1451   17
ARE GIVEN IN FILE 4, PARAMETERS FOR A FREE GAS THERMAL SCATTERING1288 1451   18
LAW ARE IN FILE 7,                                                1288 1451   19
THE ELASTIC SCATTERING CROSS SECTION AND THE LEGENDRE EXPANSION   1288 1451   20
COEFFICIENTS WERE CALCULATED FROM S-,P-, AND D-WAVE PHASE SHIFTS  1288 1451   21
USING A FORTRAN PROGRAM WRITTEN FOR THE PURPOSE, THE PHASE SHIFTS1288 1451   22
WERE READ FROM SMOOTH CURVES BASED ON TABLE I OF REF,1, AT        1288 1451   23
ENERGIES BELOW THE 700 KEV, LOWER LIMIT OF TABLE I, EACH OF THE   1288 1451   24
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                                1          451           89       1288 1451   80
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                                3          103           51       1288 1451   84
                                3          251           39       1288 1451   85
                                3          252           39       1288 1451   86
                                3          253           39       1288 1451   87
                                4            2          237       1288 1451   88
                                7            4            4       1288 1451   89
                                                                    1288 1  0   90
                                                                    1288 0  0   91
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0,0         2,2414             2            0            2      01288 2151   95
0,0         2,0                2            0            2      01288 2  0   96
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7,00000+ 5 2,57600+ 0 7,50000+ 5 3,07600+ 0 8,00000+ 5 3,75000+ 01088 3  2  174
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-2,33322- 1 1,92922- 1                                           1288 4  2  459
 0,22222+ 0 8,52222+ 5          0          0          2          21288 4  2  460
-1,88422- 1 1,96722- 1                                           1288 4  2  461
 0,22222+ 0 9,22222+ 5          0          0          2          21288 4  2  462
-1,41622- 1 1,99722- 1                                           1288 4  2  463
 0,22222+ 0 9,52222+ 5          0          0          2          21288 4  2  464
-9,71722- 2 2,22622- 1                                           1288 4  2  465
 2,22222+ 0 1,22222+ 6          0          0          2          21288 4  2  466
-5,43522- 2 2,22322- 1                                           1288 4  2  467
 2,22222+ 2 1,25222+ 6          0          0          2          21288 4  2  468
-1,57222- 2 1,98822- 1                                           1288 4  2  469
 2,22222+ 2 1,12222+ 6          0          0          2          21288 4  2  470
 1,66162- 2 1,97722- 1                                           1288 4  2  471
 2,22222+ 2 1,15222+ 6          0          0          2          21288 4  2  472
 3,63522- 2 1,95222- 1                                           1288 4  2  473
 2,22222+ 2 1,15222+ 6          0          0          2          21288 4  2  474
```

4,83120- 2	1,94120- 1									1288	4	2	475	
2,00000+ 0	1,17000+ 6			?		?		?		21088	4	2	476	
9,97000- 2	1,92900+ 1									1288	4	2	477	
2,00000+ 0	1,20000+ 6			0		0		2		21088	4	2	478	
7,67700- 2	1,91700- 1									1288	4	2	479	
2,00000+ 0	1,25000+ 6			0		0		2		21088	4	2	480	
1,22600- 1	1,87300- 1									1288	4	2	481	
2,00000+ 0	1,32000+ 6			0		0		2		21088	4	2	482	
1,24400- 1	1,83700- 1									1288	4	2	483	
2,00000+ 0	1,35000+ 6			0		0		2		21088	4	2	484	
1,44500- 1	1,80000- 1									1288	4	2	485	
2,00000+ 0	1,40000+ 6			0		0		2		21088	4	2	486	
1,62900- 1	1,76000- 1									1288	4	2	487	
0,00000+ 0	1,45000+ 6			0		0		2		21088	4	2	488	
1,78300- 1	1,72300- 1									1288	4	2	489	
2,00000+ 0	1,50000+ 6			0		0		2		21088	4	2	490	
1,91900- 1	1,68700- 1									1288	4	2	491	
2,00000+ 0	1,60000+ 6			0		0		2		21088	4	2	492	
2,17900- 1	1,60800- 1									1288	4	2	493	
0,00000+ 0	1,70000+ 6			0		0		2		21088	4	2	494	
2,37200- 1	1,53800- 1									1288	4	2	495	
0,00000+ 0	1,80000+ 6			0		0		2		21088	4	2	496	
2,52800- 1	1,47400- 1									1288	4	2	497	
0,00000+ 0	1,90000+ 6			0		0		2		21088	4	2	498	
2,64900- 1	1,41900- 1									1288	4	2	499	
0,00000+ 0	2,00000+ 6			0		0		4		21088	4	2	500	
2,74500- 1	1,35700- 1	-2,43000-	4	1,00000-	6					1288	4	2	501	
0,00000+ 0	2,10000+ 6			0		0		4		21088	4	2	502	
2,82100- 1	1,31400- 1	-2,44000-	4	1,00200-	6					1288	4	2	503	
0,00000+ 0	2,20000+ 6			0		0		4		21088	4	2	504	
2,88800- 1	1,27200- 1	-2,73000-	4	1,00000-	6					1288	4	2	505	
0,00000+ 0	2,30000+ 6			0		0		4		21088	4	2	506	
2,94400- 1	1,23900- 1	-3,21000-	4	2,00000-	6					1288	4	2	507	
0,00000+ 0	2,40000+ 6			0		0		4		21088	4	2	508	
2,98900- 1	1,21300- 1	-3,13000-	4	2,00000-	6					1288	4	2	509	
0,00000+ 0	2,50000+ 6			0		0		4		21088	4	2	510	
3,02800- 1	1,19500- 1	-3,51000-	4	2,00000-	6					1288	4	2	511	
0,00000+ 0	2,60000+ 6			0		0		4		21088	4	2	512	
3,05800- 1	1,18000- 1	-3,87000-	4	3,00000-	6					1288	4	2	513	
0,00000+ 0	2,70000+ 6			0		0		4		21088	4	2	514	
3,08400- 1	1,17100- 1	-3,94000-	4	3,00000-	6					1288	4	2	515	
0,00000+ 0	2,80000+ 6			0		0		4		21088	4	2	516	
3,11300- 1	1,16500- 1	-4,24000-	4	4,00000-	6					1288	4	2	517	
0,00000+ 0	2,90000+ 6			0		0		4		21088	4	2	518	
3,13700- 1	1,16600- 1	-4,44000-	4	5,00000-	6					1288	4	2	519	
0,00000+ 0	3,00000+ 6			0		0		4		21088	4	2	520	
3,15700- 1	1,17300- 1	-4,68000-	4	5,00000-	6					1288	4	2	521	
0,00000+ 0	3,10000+ 6			0		0		4		21088	4	2	522	
3,21600- 1	1,18600- 1	-4,81000-	4	7,00000-	6					1288	4	2	523	
0,00000+ 0	3,40000+ 6			0		0		4		21088	4	2	524	
3,27100- 1	1,21000- 1	-5,35000-	4	9,00000-	6					1288	4	2	525	
0,00000+ 0	3,60000+ 6			0		0		4		21088	4	2	526	
3,32100- 1	1,25200- 1	-5,48000-	4	1,20000-	5					1288	4	2	527	
0,00000+ 0	3,80000+ 6			0		0		4		21088	4	2	528	
3,36700- 1	1,31200- 1	-5,40000-	4	1,40000-	5					1288	4	2	529	
0,00000+ 0	4,00000+ 6			0		0		4		21088	4	2	530	
3,41800- 1	1,38100- 1	-5,78000-	4	1,70000-	5					1288	4	2	531	
0,00000+ 0	4,20000+ 6			0		0		4		21088	4	2	532	
3,47600- 1	1,43800- 1	-5,92000-	4	2,20000-	5					1288	4	2	533	
0,00000+ 0	4,40000+ 6			0		0		4		21088	4	2	534	

3,53700- 1	1,49500- 1	-6,16000-	4	2,60000-	5					1288	4	2	535	
0,00000+ 0	4,60000+ 6			0		0		4		21088	4	2	536	
3,59600- 1	1,55200- 1	-6,41000-	4	3,00000-	5					1288	4	2	537	
0,00000+ 0	4,80000+ 6			0		0		4		21088	4	2	538	
3,65400- 1	1,60400- 1	-6,74000-	4	3,60000-	5					1288	4	2	539	
0,00000+ 0	5,00000+ 6			0		0		4		21088	4	2	540	
3,71300- 1	1,65800- 1	-6,91000-	4	4,10000-	5					1288	4	2	541	
0,00000+ 0	5,20000+ 6			0		0		4		21088	4	2	542	
3,77000- 1	1,70700- 1	-7,03000-	4	4,70000-	5					1288	4	2	543	
0,00000+ 0	5,40000+ 6			0		0		4		21088	4	2	544	
3,82100- 1	1,75300- 1	-7,40000-	4	5,50000-	5					1288	4	2	545	
0,00000+ 0	5,60000+ 6			0		0		4		21088	4	2	546	
3,87000- 1	1,80000- 1	-7,32000-	4	6,30000-	5					1288	4	2	547	
0,00000+ 0	5,80000+ 6			0		0		4		21088	4	2	548	
3,91600- 1	1,84400- 1	-7,66000-	4	7,20000-	5					1288	4	2	549	
0,00000+ 0	6,00000+ 6			0		0		4		21088	4	2	550	
3,96600- 1	1,88400- 1	-8,19000-	4	8,40000-	5					1288	4	2	551	
0,00000+ 0	6,20000+ 6			0		0		4		21088	4	2	552	
4,00900- 1	1,92200- 1	-8,17000-	4	9,20000-	5					1288	4	2	553	
0,00000+ 0	6,40000+ 6			0		0		4		21088	4	2	554	
4,04900- 1	1,95300- 1	-8,20000-	4	1,05000-	4					1288	4	2	555	
0,00000+ 0	6,60000+ 6			0		0		4		21088	4	2	556	
4,09200- 1	1,98400- 1	-8,48000-	4	1,18000-	4					1288	4	2	557	
0,00000+ 0	6,80000+ 6			0		0		4		21088	4	2	558	
4,13000- 1	2,01100- 1	-8,70000-	4	1,33000-	4					1288	4	2	559	
0,00000+ 0	7,00000+ 6			0		0		4		21088	4	2	560	
4,16800- 1	2,03900- 1	-8,63000-	4	1,47000-	4					1288	4	2	561	
0,00000+ 0	7,20000+ 6			0		0		4		21088	4	2	562	
4,19800- 1	2,06200- 1	-8,14000-	4	1,61000-	4					1288	4	2	563	
0,00000+ 0	7,40000+ 6			0		0		4		21088	4	2	564	
4,23100- 1	2,08300- 1	-7,80000-	4	1,79000-	4					1288	4	2	565	
0,00000+ 0	7,60000+ 6			0		0		4		21088	4	2	566	
4,26000- 1	2,10600- 1	-7,50000-	4	1,95000-	4					1288	4	2	567	
0,00000+ 0	7,80000+ 6			0		0		4		21088	4	2	568	
4,29100- 1	2,12400- 1	-6,99000-	4	2,14000-	4					1288	4	2	569	
0,00000+ 0	8,00000+ 6			0		0		4		21088	4	2	570	
4,31600- 1	2,14200- 1	-6,41000-	4	2,39000-	4					1288	4	2	571	
0,00000+ 0	8,20000+ 6			0		0		4		21088	4	2	572	
4,34000- 1	2,15900- 1	-5,64000-	4	2,65000-	4					1288	4	2	573	
0,00000+ 0	8,40000+ 6			0		0		4		21088	4	2	574	
4,36200- 1	2,17500- 1	-4,67000-	4	2,90000-	4					1288	4	2	575	
0,00000+ 0	8,60000+ 6			0		0		4		21088	4	2	576	
4,38400- 1	2,18900- 1	-3,50000-	4	3,18000-	4					1288	4	2	577	
0,00000+ 0	8,80000+ 6			0		0		4		21088	4	2	578	
4,40300- 1	2,20200- 1	-2,49000-	4	3,48000-	4					1288	4	2	579	
0,00000+ 0	9,00000+ 6			0		0		4		21088	4	2	580	
4,42400- 1	2,21600- 1	-1,32000-	4	3,82000-	4					1288	4	2	581	
0,00000+ 0	9,20000+ 6			0		0		4		21088	4	2	582	
4,44500- 1	2,22300- 1	-1,22000-	5	4,11000-	4					1288	4	2	583	
0,00000+ 0	9,40000+ 6			0		0		4		21088	4	2	584	
4,46300- 1	2,23400- 1	-1,11000-	4	4,48000-	4					1288	4	2	585	
0,00000+ 0	9,60000+ 6			0		0		4		21088	4	2	586	
4,48000- 1	2,24500- 1	-2,70000-	4	4,84000-	4					1288	4	2	587	
0,00000+ 0	9,80000+ 6			0		0		4		21088	4	2	588	
4,49800- 1	2,25400- 1	-4,34000-	4	5,22000-	4					1288	4	2	589	
0,00000+ 0	1,00000+ 7			0		0		4		21088	4	2	590	
4,51700- 1	2,26000- 1	-5,14000-	4	5,62000-	4					1288	4	2	591	
0,00000+ 0	1,02000+ 7			0		0		4		21088	4	2	592	
4,53300- 1	2,26700- 1	-1,12000-	3	6,02000-	4					1288	4	2	593	
0,00000+ 0	1,04000+ 7			0		0		4		21088	4	2	594	

```
4,58900- 1 2,31222- 1 1,62922= 3 7,63222- 4                    1088 4  2  595  K-7
0,00000+ 0 1,15222+ 7          0          0          4         01088 4  2  596
4,62100- 1 2,32600- 1 2,31300= 3 9,25200- 4                    1088 4  2  597
0,00000+ 0 1,00000+ 7          0          0          4         01088 4  2  598
4,65100- 1 2,34200- 1 3,16900= 3 1,28300- 3                    1088 4  2  599
0,00000+ 0 1,25000+ 7          0          0          4         01088 4  2  600
4,67400- 1 2,35100- 1 4,00200= 3 1,23100- 3                    1088 4  2  601
0,00000+ 0 1,30000+ 7          0          0          4         01088 4  2  602
4,70100- 1 2,36200= 1 5,00100= 3 1,42900- 3                    1088 4  2  603
0,00000+ 0 1,35000+ 7          0          0          4         01088 4  2  604
4,72500- 1 2,37400= 1 6,13900= 3 1,60000- 3                    1088 4  2  605
0,00000+ 0 1,40000+ 7          0          0          4         01088 4  2  606
4,75000- 1 2,38800- 1 7,28400= 3 1,78100- 3                    1088 4  2  607
0,00000+ 0 1,45000+ 7          0          0          4         01088 4  2  608
4,77500- 1 2,40200= 1 8,51500= 3 1,97100- 3                    1088 4  2  609
0,00000+ 0 1,50000+ 7          0          0          4         01088 4  2  610
4,80100- 1 2,41800- 1 9,80900= 3 2,17100- 3                    1088 4  2  611
                                                               1088 4  0  612
                                                               1088 0  0  613
2,00000+ 3 3,96822+ 0          0          0          0         01088 7  4  614
0,00000+ 0 2,00000+ 0          0          0         12         11088 7  4  615
0,00000+ 0 5,93000+ 1 3,96822+ 0 1,50000+ 0 0,00000+ 0 0,00000+ 01088 7  4  616
1,00000+ 0 7,23800- 1 3,96822+ 0 0,00000+ 0 0,00000+ 0 0,00000+ 01088 7  4  617
                                                               1088 7  0  618
                                                               1088 0  0  619
                                                                  0 0  0  620
```

APPENDIX L

Sample of Interpreted Data Set

The following is an interpreted listing of an ENDF data set for the element helium. 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.

```
 HELIUM     ANL          EVAL-JUN68 E,H,PENNINGTON (NATURAL ELEMENT)
ANL-7462 (OCT,1968)  DIST-JUN70 REV-JUN72
.          .          .          .          .          .
DATA MODIFIED JUNE,1972 TO CONFORM TO ENDF/B-II FORMATS
.          .          .          .          .          .
HELIUM CROSS SECTIONS  MATERIAL 1288
COMPILED BY ED PENNINGTON, ARGONNE NATIONAL LAB,, IN JUNE 1968,
NATURAL HELIUM CONSISTS OF 0,00013 PER CENT HE-3 AND 99,99987 PER
CENT HE-4,
BECAUSE OF THE LOW ABUNDANCE OF HE-3, ONLY ITS (N,P) CROSS
SECTION, WHICH IS VERY LARGE AT LOW ENERGIES, NEED BE CONSIDERED,
ELASTIC SCATTERING IS THE ONLY POSSIBLE REACTION FOR NEUTRONS
INCIDENT ON HE-4 AT ENERGIES BELOW 15 MEV, THUS THE ELASTIC
SCATTERING CROSS SECTION AND VALUES OF MU BAR(LAB), XI, AND GAMMA
ARE GIVEN IN FILE 3, AND ELASTIC SCATTERING LEGENDRE COEFFICIENTS
ARE GIVEN IN FILE 4, PARAMETERS FOR A FREE GAS THERMAL SCATTERING
LAW ARE IN FILE 7,
THE ELASTIC SCATTERING CROSS SECTION AND THE LEGENDRE EXPANSION
COEFFICIENTS WERE CALCULATED FROM S-,P-, AND D-WAVE PHASE SHIFTS
USING A FORTRAN PROGRAM WRITTEN FOR THE PURPOSE, THE PHASE SHIFTS
WERE READ FROM SMOOTH CURVES BASED ON TABLE I OF REF,1, AT
ENERGIES BELOW THE 300 KEV, LOWER LIMIT OF TABLE I, EACH OF THE
TWO P-WAVE PHASE SHIFTS WAS OBTAINED BY ASSUMING A FUNCTIONAL
FORM BASED ON THE LOW ENERGY LIMIT FOR A SINGLE P-WAVE RESONANCE,
WITH PARAMETERS DETERMINED FROM FITTING THE LOW ENERGY PHASE
SHIFTS OF TABLE I, THE S-WAVE PHASE SHIFT BELOW 300 KEV, WAS
CALCULATED USING HARD SPHERE SCATTERING AND A NUCLEAR RADIUS, A=
2,4 FERMI, THIS YIELDS THE THERMAL SCATTERING CROSS SECTION =
4,*PI*A**2= 0,7238 BARNS IN AGREEMENT WITH THE EXPERIMENTAL VALUE
OF 0,73+- 0,05 BARNS (REF,2), THE LOW ENERGY S-WAVE PHASE SHIFTS
OF TABLE I ARE CONSISTENT WITH A NUCLEAR RADIUS OF ABOUT 2,48
FERMI, AND SO WOULD YIELD A SOMEWHAT HIGH THERMAL CROSS SECTION,
VALUES OF MU BAR(LAB), XI, AND GAMMA WERE CALCULATED FROM THE
LEGENDRE COEFFICIENTS USING A FORTRAN PROGRAM, MUXIGA, THIS
PROGRAM USES THE EQUATIONS OF REF,3-5,
AN ELASTIC SCATTERING TRANSFORMATION MATRIX FROM THE CENTER-OF-
MASS TO THE LABORATORY SYSTEM WAS COMPUTED USING CHAD (REF,6),
THE (N,P) CROSS SECTION FOR HE-3 IS THAT RECOMMENDED IN THE
EVALUATION OF HE-3 BY J, ALS-NIELSEN GIVEN IN REF,7, EXTENSION
FROM 10 TO 15 MEV, WAS MADE USING LINEAR EXTRAPOLATION ON A
LOG SIGMA- LOG E SCALE,
THE TOTAL CROSS SECTION IS THE SUM OF THE ELASTIC SCATTERING AND
(N,P) CROSS SECTIONS,
COMMENTS-
THE PHASE SHIFTS OF REF,1 ARE OPTICAL MODEL PHASE SHIFTS CHOSEN
TO FIT BOTH ANGULAR DISTRIBUTION AND POLARIZATION DATA AT MANY
ENERGIES, THE TOTAL SCATTERING CROSS SECTION IS ALSO FIT WITHIN
THE SCATTER OF THE EXPERIMENTAL POINTS, ANOTHER RECENT SET OF
PHASE SHIFTS (REF,8) IS NOT VERY DIFFERENT FROM THOSE USED HERE,
AND COULD ALSO HAVE BEEN USED IN THE PRESENT WORK, THERE SHOULD
 BE NO SERIOUS ERRORS IN THE HE-4 DATA CALCULATED FROM THE PHASE
SHIFTS,
AS DISCUSSED IN REF,7, THE HE-3 (N,P) CROSS SECTION IS RATHER
WELL KNOWN, PROBABLY MORE ERROR IS INTRODUCED INTO THE (N,P)
CROSS SECTION FOR NATURAL HELIUM BY THE UNCERTAINTY IN THE HE-3
ISOTOPIC ABUNDANCE THAN BY THE UNCERTAINTY IN THE HE-3 (N,P)
CROSS SECTION ITSELF,
PREVIOUS EVALUATIONS OF HELIUM FOR REACTOR CALCULATIONS INCLUDE
THOSE OF J,J,SCHMIDT (REF,9) AND B,R,S,BUCKINGHAM ET AL (REF,10),
SCHMIDT-S EVALUATION INCLUDES THE (N,P) CROSS SECTION FOR HE-3,
AND SIGMA ELASTIC, MU BAR(LAB), AND A SET OF PHASE SHIFTS FOR
HE-4, BUCKINGHAM ET AL GIVE SEPARATE EVALUATIONS FOR HE-3 AND
HE-4, FOR HE-3 ELASTIC, (N,P), (N,D), AND (N,2N) CROSS SECTIONS
ARE GIVEN, AS WELL AS ELASTIC ANGULAR DISTRIBUTIONS, THE HE-4
EVALUATION GIVES THE ELASTIC CROSS SECTION AND ANGULAR
DISTRIBUTIONS,
REFERENCES-
1, G,R,SATCHLER ET AL, NUCLEAR PHYSICS A112,1-31,(1968),
2, R,GENIN ET AL, JOURNAL DE PHYSIQUE ET LE RADIUM 24,21-26,1963,
3, H,AMSTER JOURNAL OF APPLIED PHYSICS 27,3,387,(1956),
4, H,AMSTER JOURNAL OF APPLIED PHYSICS 27,6,663,(1956),
5, H,AMSTER JOURNAL OF APPLIED PHYSICS 29,4,623-627,(1958),
6, R,F,BERLAND NAA-SR-11231,(1965),
7, E,N,E,A, NEUTRON DATA COMPILATION CENTRE NEWSLETTER NO,6,1967,
8, B,HOOP,JR, AND H,H,BARSCHALL NUCLEAR PHYSICS 83,65-79,(1966),
9, J,J,SCHMIDT KFK-120,PARTS II,III,(1962),PART I,(1966),
10,B,R,S,BUCKINGHAM ET AL AWRE O-28/60,(1961),
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NATURAL HELIUM | TABLE OF CONTENTS GENERAL INFORMATION | ENDF/B MATERIAL NO, 1088

NATURAL HELIUM | TOTAL SMOOTH NEUTRON CROSS SECTIONS | ENDF/B MATERIAL NO, 1088

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 142	Y LINEAR IN LN 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	1.0721E+00	2.0000E-05	9.7010E-01	3.0000E-05	9.2490E-01	5.0000E-05	8.7960E-01	7.0000E-05	8.5550E-01
6	1.0000E-04	8.3390E-01	2.0000E-04	8.0170E-01	3.0000E-04	7.8740E-01	5.0000E-04	7.7310E-01	7.0000E-04	7.6540E-01
11	1.0000E-03	7.5860E-01	2.0000E-03	7.4840E-01	3.0000E-03	7.4390E-01	5.0000E-03	7.3940E-01	7.0000E-03	7.3700E-01
16	1.0000E-02	7.3480E-01	2.0000E-02	7.3160E-01	2.5300E-02	7.3070E-01	3.0000E-02	7.3020E-01	5.0000E-02	7.2870E-01
21	7.0000E-02	7.2800E-01	1.0000E-01	7.2730E-01	2.0000E-01	7.2630E-01	3.0000E-01	7.2580E-01	5.0000E-01	7.2540E-01
26	7.0000E-01	7.2510E-01	1.0000E+00	7.2490E-01	2.0000E+00	7.2460E-01	3.0000E+00	7.2440E-01	5.0000E+00	7.2430E-01
31	7.0000E+00	7.2422E-01	1.0000E+01	7.2410E-01	2.0000E+01	7.2400E-01	3.0000E+01	7.2400E-01	5.0000E+01	7.2400E-01
36	7.0000E+01	7.2390E-01	1.0000E+02	7.2390E-01	2.0000E+02	7.2390E-01	3.0000E+02	7.2390E-01	5.0000E+02	7.2380E-01
41	7.0000E+02	7.2382E-01	1.0000E+03	7.2380E-01	2.0000E+03	7.2370E-01	3.0000E+03	7.2370E-01	5.0000E+03	7.2360E-01
46	7.0000E+03	7.2362E-01	1.0000E+04	7.2350E-01	2.0000E+04	7.2350E-01	3.0000E+04	7.2340E-01	5.0000E+04	7.2410E-01
51	7.0000E+04	7.2560E-01	1.0000E+05	7.2970E-01	1.5000E+05	7.4250E-01	2.0000E+05	7.6440E-01	2.5000E+05	7.7970E-01
56	3.0000E+05	8.9780E-01	3.5000E+05	9.6540E-01	4.0000E+05	1.0560E+00	4.5000E+05	1.1970E+00	5.0000E+05	1.3730E+00
61	5.5000E+05	1.5810E+00	6.0000E+05	1.8260E+00	6.5000E+05	2.1680E+00	7.0000E+05	2.5760E+00	7.5000E+05	3.0760E+00
66	8.0000E+05	3.7500E+00	8.5000E+05	4.3800E+00	9.0000E+05	5.1140E+00	9.5000E+05	5.8440E+00	1.0000E+06	6.5250E+00
71	1.0500E+06	7.0380E+00	1.1000E+06	7.3210E+00	1.1300E+06	7.4240E+00	1.1500E+06	7.4560E+00	1.1700E+06	7.4610E+00
76	1.2000E+06	7.4320E+00	1.2500E+06	7.2780E+00	1.3000E+06	7.0400E+00	1.3500E+06	6.7570E+00	1.4000E+06	6.4530E+00
81	1.4500E+06	6.1570E+00	1.5000E+06	5.8750E+00	1.6000E+06	5.3330E+00	1.7000E+06	4.8820E+00	1.8000E+06	4.5090E+00
86	1.9000E+06	4.2110E+00	2.0000E+06	3.9440E+00	2.1000E+06	3.7340E+00	2.2000E+06	3.5510E+00	2.3000E+06	3.3990E+00
91	2.4000E+06	3.2730E+00	2.5000E+06	3.1690E+00	2.6000E+06	3.0710E+00	2.7000E+06	2.9880E+00	2.8000E+06	2.9110E+00
96	2.9000E+06	2.8490E+00	3.0000E+06	2.7890E+00	3.2000E+06	2.6840E+00	3.4000E+06	2.5950E+00	3.6000E+06	2.5240E+00
101	3.8000E+06	2.4690E+00	4.0000E+06	2.4240E+00	4.2000E+06	2.3730E+00	4.4000E+06	2.3250E+00	4.6000E+06	2.2780E+00
106	4.8000E+06	2.2320E+00	5.0000E+06	2.1900E+00	5.2000E+06	2.1490E+00	5.4000E+06	2.1070E+00	5.6000E+06	2.0700E+00
111	5.8000E+06	2.0310E+00	6.0000E+06	1.9940E+00	6.2000E+06	1.9570E+00	6.4000E+06	1.9210E+00	6.6000E+06	1.8870E+00
116	6.8000E+06	1.8530E+00	7.0000E+06	1.8210E+00	7.2000E+06	1.7870E+00	7.4000E+06	1.7560E+00	7.6000E+06	1.7260E+00
121	7.8000E+06	1.6960E+00	8.0000E+06	1.6670E+00	8.2000E+06	1.6400E+00	8.4000E+06	1.6120E+00	8.6000E+06	1.5850E+00
126	8.8000E+06	1.5590E+00	9.0000E+06	1.5340E+00	9.2000E+06	1.5090E+00	9.4000E+06	1.4850E+00	9.6000E+06	1.4620E+00
131	9.8000E+06	1.4390E+00	1.0000E+07	1.4180E+00	1.0500E+07	1.3650E+00	1.1000E+07	1.3150E+00	1.1500E+07	1.2680E+00
136	1.2000E+07	1.2240E+00	1.2500E+07	1.1800E+00	1.3000E+07	1.1400E+00	1.3500E+07	1.1020E+00	1.4000E+07	1.0660E+00
141	1.4500E+07	1.0320E+00	1.5000E+07	9.9920E-01						

NATURAL HELIUM — ELASTIC — ENDF/B MATERIAL NO, 1088

SMOOTH NEUTRON CROSS SECTIONS

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 142	Y LINEAR IN LN 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	7,2380E+01	2,0000E-05	7,2380E+01	3,0000E-05	7,2380E+01	5,0000E-05	7,2380E+01	7,0000E-05	7,2380E+01
6	1,0000E-04	7,2380E+01	2,0000E-04	7,2380E+01	3,0000E-04	7,2380E+01	5,0000E-04	7,2380E+01	7,0000E-04	7,2380E+01
11	1,0000E-03	7,2380E+01	2,0000E-03	7,2380E+01	3,0000E-03	7,2380E+01	5,0000E-03	7,2380E+01	7,0000E-03	7,2380E+01
16	1,0000E-02	7,2380E+01	2,0000E-02	7,2380E+01	2,5300E-02	7,2380E+01	3,0000E-02	7,2380E+01	5,0000E-02	7,2380E+01
21	7,0000E-02	7,2360E+01	1,0000E+01	7,2380E+01	2,0000E+01	7,2380E+01	3,0000E+01	7,2380E+01	5,0000E+01	7,2380E+01
26	7,0000E+01	7,2380E+01	1,0000E+02	7,2380E+01	2,0000E+02	7,2380E+01	3,0000E+00	7,2380E+01	5,0000E+00	7,2380E+01
31	7,0000E+00	7,2380E+01	1,0000E+01	7,2380E+01	2,0000E+01	7,2380E+01	3,0000E+01	7,2380E+01	5,0000E+01	7,2380E+01
36	7,0000E+01	7,2380E+01	1,0000E+02	7,2360E+01	2,0000E+02	7,2380E+01	3,0000E+02	7,2380E+01	5,0000E+02	7,2380E+01
41	7,0000E+02	7,2380E+01	1,0000E+03	7,2380E+01	2,0000E+03	7,2370E+01	3,0000E+03	7,2370E+01	5,0000E+03	7,2360E+01
46	7,0000E+03	7,2360E+01	1,0000E+04	7,2350E+01	2,0000E+04	7,2350E+01	3,0000E+04	7,2340E+01	5,0000E+04	7,2410E+01
51	7,0000E+04	7,2560E+01	1,0000E+05	7,2970E+01	1,5000E+05	7,4250E+01	2,0000E+05	7,6440E+01	2,5000E+05	7,9770E+01
56	3,0000E+05	8,9780E+01	3,5000E+05	9,6540E+01	4,0000E+05	1,0560E+00	4,5000E+05	1,1970E+00	5,0000E+05	1,3730E+00
61	5,5000E+05	1,5810E+00	6,0000E+05	1,8260E+00	6,5000E+05	2,1660E+00	7,0000E+05	2,5760E+00	7,5000E+05	3,0760E+00
66	8,0000E+05	3,7520E+00	8,5000E+05	4,3800E+00	9,0000E+05	5,1140E+00	9,5000E+05	5,8440E+00	1,0000E+06	6,5250E+00
71	1,0500E+06	7,0380E+00	1,1000E+06	7,3210E+00	1,1300E+06	7,4240E+00	1,1500E+06	7,4560E+00	1,1700E+06	7,4610E+00
76	1,2000E+06	7,4320E+00	1,2500E+06	7,2780E+00	1,3000E+06	7,0400E+00	1,3500E+06	6,7570E+00	1,4000E+06	6,4530E+00
81	1,4500E+06	6,1570E+00	1,5000E+06	5,8750E+00	1,6000E+06	5,3330E+00	1,7000E+06	4,8020E+00	1,8000E+06	4,5090E+00
86	1,9000E+06	4,2110E+00	2,0000E+06	3,9450E+00	2,1000E+06	3,7340E+00	2,2000E+06	3,5510E+00	2,3000E+06	3,3990E+00
91	2,4000E+06	3,2730E+00	2,5000E+06	3,1690E+00	2,6000E+06	3,0710E+00	2,7000E+06	2,9880E+00	2,8000E+06	2,9110E+00
96	2,9000E+06	2,8490E+00	3,0000E+06	2,7890E+00	3,2000E+06	2,6840E+00	3,4000E+06	2,5950E+00	3,6000E+06	2,5240E+00
101	3,8000E+06	2,4690E+00	4,0000E+06	2,4240E+00	4,2000E+06	2,3730E+00	4,4000E+06	2,3250E+00	4,6000E+06	2,2780E+00
106	4,8000E+06	2,2320E+00	5,0000E+06	2,1900E+00	5,2000E+06	2,1490E+00	5,4000E+06	2,1070E+00	5,6000E+06	2,0700E+00
111	5,8000E+06	2,0310E+00	6,0000E+06	1,9940E+00	6,2000E+06	1,9570E+00	6,4000E+06	1,9210E+00	6,6000E+06	1,8870E+00
116	6,8000E+06	1,8530E+00	7,0000E+06	1,8210E+00	7,2000E+06	1,7870E+00	7,4000E+06	1,7560E+00	7,6000E+06	1,7260E+00
121	7,8000E+06	1,6960E+00	8,0000E+06	1,6670E+00	8,2000E+06	1,6400E+00	8,4000E+06	1,6120E+00	8,6000E+06	1,5850E+00
126	8,8000E+06	1,5590E+00	9,0000E+06	1,5340E+00	9,2000E+06	1,5090E+00	9,4000E+06	1,4850E+00	9,6000E+06	1,4620E+00
131	9,8000E+06	1,4390E+00	1,0000E+07	1,4160E+00	1,0500E+07	1,3650E+00	1,1000E+07	1,3150E+00	1,1500E+07	1,2680E+00
136	1,2000E+07	1,2240E+00	1,2500E+07	1,1800E+00	1,3000E+07	1,1400E+00	1,3500E+07	1,1022E+00	1,4000E+07	1,0660E+00
141	1,4500E+07	1,0320E+00	1,5000E+07	9,9920E-01						

NATURAL HELIUM — (N,P) — ENDF/B MATERIAL NO, 1088

SMOOTH NEUTRON CROSS SECTIONS

REACTION Q VALUE 7,6440E+05 EV

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 142	LN Y LINEAR IN LN 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	3,4830E+01	2,0000E-05	2,4630E+01	3,0000E-05	2,0110E+01	5,0000E-05	1,5580E+01	7,0000E-05	1,3170E+01
6	1,0000E-04	1,1010E+01	2,0000E-04	7,7890E+02	3,0000E-04	6,3590E+02	5,0000E-04	4,9260E+02	7,0000E-04	4,1630E+02
11	1,0000E-03	3,4830E+02	2,0000E-03	2,4630E+02	3,0000E-03	2,0110E+02	5,0000E-03	1,5580E+02	7,0000E-03	1,3160E+02
16	1,0000E-02	1,1010E+02	2,0000E-02	7,7870E+03	2,5300E-02	6,9240E+03	3,0000E-02	6,3580E+03	5,0000E-02	4,9240E+03
21	7,0000E-02	4,1610E+03	1,0000E-01	3,4810E+03	2,0000E-01	2,4610E+03	3,0000E-01	2,0100E+03	5,0000E-01	1,5560E+03
26	7,0000E-01	1,3160E+03	1,0000E+00	1,1000E+03	2,0000E+00	7,7730E+04	3,0000E+00	6,3440E+04	5,0000E+00	4,9100E+04
31	7,0000E+00	4,1470E+04	1,0000E+01	3,4670E+04	2,0000E+01	2,4480E+04	3,0000E+01	1,9960E+04	5,0000E+01	1,5420E+04
36	7,0000E+01	1,3010E+04	1,0000E+02	1,0860E+04	2,0000E+02	7,6310E+05	3,0000E+02	6,2010E+05	5,0000E+02	4,7710E+05
41	7,0000E+02	4,0040E+05	1,0000E+03	3,3280E+05	2,0000E+03	2,3140E+05	3,0000E+03	1,8590E+05	5,0000E+03	1,4170E+05
46	7,0000E+03	1,1830E+05	1,0000E+04	9,8820E+06	2,0000E+04	6,8250E+06	3,0000E+04	5,3300E+06	5,0000E+04	3,9000E+06
51	7,0000E+04	3,1850E+06	1,0000E+05	2,5740E+06	1,5000E+05	2,0670E+06	2,0000E+05	1,7550E+06	2,5000E+05	1,5600E+06
56	3,0000E+05	1,4300E+06	3,5000E+05	1,3130E+06	4,0000E+05	1,2480E+06	4,5000E+05	1,1960E+06	5,0000E+05	1,1830E+06
61	5,5000E+05	1,1780E+06	6,0000E+05	1,1570E+06	6,5000E+05	1,1570E+06	7,0000E+05	1,1570E+06	7,5000E+05	1,1570E+06
66	8,0000E+05	1,1570E+06	8,5000E+05	1,1570E+06	9,0000E+05	1,1570E+06	9,5000E+05	1,1570E+06	1,0000E+06	1,1570E+06
71	1,0500E+06	1,1570E+06	1,1000E+06	1,1570E+06	1,1300E+06	1,1570E+06	1,1500E+06	1,1570E+06	1,1700E+06	1,1570E+06
76	1,2000E+06	1,1570E+06	1,2500E+06	1,1570E+06	1,3000E+06	1,1570E+06	1,3500E+06	1,1570E+06	1,4000E+06	1,1570E+06
81	1,4500E+06	1,1570E+06	1,5000E+06	1,1570E+06	1,6000E+06	1,1510E+06	1,7000E+06	1,1440E+06	1,8000E+06	1,1380E+06
86	1,9000E+06	1,1180E+06	2,0000E+06	1,0920E+06	2,1000E+06	1,0660E+06	2,2000E+06	1,0400E+06	2,3000E+06	1,0140E+06
91	2,4000E+06	9,8800E+07	2,5000E+06	9,6200E+07	2,6000E+06	9,3600E+07	2,7000E+06	9,1000E+07	2,8000E+06	8,8400E+07
96	2,9000E+06	8,5800E+07	3,0000E+06	8,3200E+07	3,2000E+06	7,8000E+07	3,4000E+06	7,2800E+07	3,6000E+06	6,8250E+07
101	3,8000E+06	6,4350E+07	4,0000E+06	6,1100E+07	4,2000E+06	5,7850E+07	4,4000E+06	5,4600E+07	4,6000E+06	5,2650E+07
106	4,8000E+06	5,0050E+07	5,0000E+06	4,8100E+07	5,2000E+06	4,6150E+07	5,4000E+06	4,4200E+07	5,6000E+06	4,2250E+07
111	5,8000E+06	4,0950E+07	6,0000E+06	3,9650E+07	6,2000E+06	3,7700E+07	6,4000E+06	3,6400E+07	6,6000E+06	3,5100E+07
116	6,8000E+06	3,4060E+07	7,0000E+06	3,3150E+07	7,2000E+06	3,2110E+07	7,4000E+06	3,1330E+07	7,6000E+06	3,0550E+07
121	7,8000E+06	2,9770E+07	8,0000E+06	2,9120E+07	8,2000E+06	2,8470E+07	8,4000E+06	2,7690E+07	8,6000E+06	2,7170E+07
126	8,8000E+06	2,6650E+07	9,0000E+06	2,6000E+07	9,2000E+06	2,5480E+07	9,4000E+06	2,4830E+07	9,6000E+06	2,4310E+07
131	9,8000E+06	2,3790E+07	1,0000E+07	2,3420E+07	1,0500E+07	2,2360E+07	1,1000E+07	2,1450E+07	1,1500E+07	2,0540E+07
136	1,2000E+07	1,9760E+07	1,2500E+07	1,8980E+07	1,3000E+07	1,8330E+07	1,3500E+07	1,7680E+07	1,4000E+07	1,7030E+07
141	1,4500E+07	1,6510E+07	1,5000E+07	1,5990E+07						

NATURAL HELIUM

MU BAR
SMOOTH NEUTRON CROSS SECTIONS

ENDF/B MATERIAL NO. 1088

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 108	Y LINEAR IN LN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1,0000E-05	1,6800E-01	2,5300E-02	1,6800E-01	1,0000E+02	1,6792E-01	2,0000E+02	1,6780E-01	3,0000E+02	1,6770E-01
6	5,0000E+02	1,6750E-01	7,0000E+02	1,6720E-01	1,0000E+03	1,6690E-01	2,0000E+03	1,6580E-01	3,0000E+03	1,6470E-01
11	5,0000E+03	1,6250E-01	7,0000E+03	1,6030E-01	1,0000E+04	1,5720E-01	2,0000E+04	1,4580E-01	3,0000E+04	1,3440E-01
16	5,0000E+04	1,1160E-01	7,0000E+04	8,7980E-02	1,0000E+05	5,2280E-02	1,5000E+05	9,7400E-03	2,0000E+05	7,1610E-02
21	2,5000E+05	1,1040E-01	3,0000E+05	1,7390E-01	3,5000E+05	2,1850E-01	4,0000E+05	2,5180E-01	4,5000E+05	2,7020E-01
26	5,0000E+05	2,7460E-01	5,5000E+05	2,6980E-01	6,0000E+05	2,4570E-01	6,5000E+05	2,1430E-01	7,0000E+05	1,7760E-01
31	7,5000E+05	1,3570E-01	8,0000E+05	8,7240E-02	8,5000E+05	4,4410E-02	9,0000E+05	5,2000E-05	9,5000E+05	4,2670E-02
36	1,0000E+06	8,3900E-02	1,0500E+06	1,2130E-01	1,1000E+06	1,5290E-01	1,1500E+06	1,7220E-01	1,1500E+06	1,8360E-01
41	1,1700E+06	1,9480E-01	1,2000E+06	2,1150E-01	1,2500E+06	2,3700E-01	1,3000E+06	2,5850E-01	1,3500E+06	2,7840E-01
46	1,4000E+06	2,9670E-01	1,4500E+06	3,1220E-01	1,5000E+06	3,2580E-01	1,6000E+06	3,5200E-01	1,7000E+06	3,7180E-01
51	1,8000E+06	3,8780E-01	1,9000E+06	4,0220E-01	2,0000E+06	4,1040E-01	2,1000E+06	4,1850E-01	2,2000E+06	4,2550E-01
56	2,3000E+06	4,3150E-01	2,4000E+06	4,3630E-01	2,5000E+06	4,4020E-01	2,6000E+06	4,4340E-01	2,7000E+06	4,4600E-01
61	2,8000E+06	4,4890E-01	2,9000E+06	4,5120E-01	3,0000E+06	4,5320E-01	3,2000E+06	4,5850E-01	3,4000E+06	4,6340E-01
66	3,6000E+06	4,6750E-01	3,8000E+06	4,7100E-01	4,0000E+06	4,7490E-01	4,2000E+06	4,7950E-01	4,4000E+06	4,8450E-01
71	4,6000E+06	4,8920E-01	4,8000E+06	4,9400E-01	5,0000E+06	4,9870E-01	5,2000E+06	5,0350E-01	5,4000E+06	5,0770E-01
76	5,6000E+06	5,1170E-01	5,8000E+06	5,1530E-01	6,0000E+06	5,1950E-01	6,2000E+06	5,2310E-01	6,4000E+06	5,2640E-01
81	6,6000E+06	5,3210E-01	6,8000E+06	5,3330E-01	7,0000E+06	5,3650E-01	7,2000E+06	5,3900E-01	7,4000E+06	5,4180E-01
86	7,6000E+06	5,4430E-01	7,8000E+06	5,4700E-01	8,0000E+06	5,4920E-01	8,2000E+06	5,5130E-01	8,4000E+06	5,5300E-01
91	8,6000E+06	5,5490E-01	8,8000E+06	5,5650E-01	9,0000E+06	5,5840E-01	9,2000E+06	5,6030E-01	9,4000E+06	5,6180E-01
96	9,6000E+06	5,6350E-01	9,8000E+06	5,6490E-01	1,0000E+07	5,6660E-01	1,0500E+07	5,6960E-01	1,1000E+07	5,7280E-01
101	1,1500E+07	5,7560E-01	1,2000E+07	5,7830E-01	1,2500E+07	5,8040E-01	1,3000E+07	5,8280E-01	1,3500E+07	5,8500E-01
106	1,4000E+07	5,8720E-01	1,4500E+07	5,8940E-01	1,5000E+07	5,9170E-01				

NATURAL HELIUM

XI
SMOOTH NEUTRON CROSS SECTIONS

ENDF/B MATERIAL NO. 1088

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 108	Y LINEAR IN LN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1,0000E-05	4,2820E-01	2,5300E-02	4,2820E-01	1,0000E+02	4,2820E-01	2,0000E+02	4,2830E-01	3,0000E+02	4,2840E-01
6	5,0000E+02	4,2850E-01	7,0000E+02	4,2860E-01	1,0000E+03	4,2870E-01	2,0000E+03	4,2930E-01	3,0000E+03	4,2990E-01
11	5,0000E+03	4,3100E-01	7,0000E+03	4,3220E-01	1,0000E+04	4,3390E-01	2,0000E+04	4,3970E-01	3,0000E+04	4,4550E-01
16	5,0000E+04	4,5740E-01	7,0000E+04	4,6960E-01	1,0000E+05	4,8810E-01	1,5000E+05	5,2810E-01	2,0000E+05	5,5210E-01
21	2,5000E+05	5,8260E-01	3,0000E+05	6,0510E-01	3,5000E+05	6,2810E-01	4,0000E+05	6,4540E-01	4,5000E+05	6,5500E-01
26	5,0000E+05	6,5720E-01	5,5000E+05	6,5270E-01	6,0000E+05	6,4230E-01	6,5000E+05	6,2610E-01	7,0000E+05	6,0710E-01
31	7,5000E+05	5,8550E-01	8,0000E+05	5,6030E-01	8,5000E+05	5,3820E-01	9,0000E+05	5,1520E-01	9,5000E+05	4,9320E-01
36	1,0000E+06	4,7190E-01	1,0500E+06	4,5250E-01	1,1000E+06	4,3620E-01	1,1500E+06	4,2630E-01	1,1500E+06	4,2030E-01
41	1,1700E+06	4,1450E-01	1,2000E+06	4,0590E-01	1,2500E+06	3,9270E-01	1,3000E+06	3,8160E-01	1,3500E+06	3,7120E-01
46	1,4000E+06	3,6180E-01	1,4500E+06	3,5360E-01	1,5000E+06	3,4670E-01	1,6000E+06	3,3310E-01	1,7000E+06	3,2290E-01
51	1,8000E+06	3,1460E-01	1,9000E+06	3,0820E-01	2,0000E+06	3,0290E-01	2,1000E+06	2,9880E-01	2,2000E+06	2,9510E-01
56	2,3000E+06	2,9200E-01	2,4000E+06	2,8950E-01	2,5000E+06	2,8750E-01	2,6000E+06	2,8580E-01	2,7000E+06	2,8450E-01
61	2,8000E+06	2,8300E-01	2,9000E+06	2,8180E-01	3,0000E+06	2,8090E-01	3,2000E+06	2,7810E-01	3,4000E+06	2,7550E-01
66	3,6000E+06	2,7340E-01	3,8000E+06	2,7160E-01	4,0000E+06	2,6960E-01	4,2000E+06	2,6720E-01	4,4000E+06	2,6460E-01
71	4,6000E+06	2,6220E-01	4,8000E+06	2,5970E-01	5,0000E+06	2,5730E-01	5,2000E+06	2,5480E-01	5,4000E+06	2,5260E-01
76	5,6000E+06	2,5060E-01	5,8000E+06	2,4870E-01	6,0000E+06	2,4650E-01	6,2000E+06	2,4470E-01	6,4000E+06	2,4300E-01
81	6,6000E+06	2,4110E-01	6,8000E+06	2,3940E-01	7,0000E+06	2,3780E-01	7,2000E+06	2,3650E-01	7,4000E+06	2,3500E-01
86	7,6000E+06	2,3370E-01	7,8000E+06	2,3230E-01	8,0000E+06	2,3120E-01	8,2000E+06	2,3010E-01	8,4000E+06	2,2930E-01
91	8,6000E+06	2,2830E-01	8,8000E+06	2,2740E-01	9,0000E+06	2,2650E-01	9,2000E+06	2,2550E-01	9,4000E+06	2,2470E-01
96	9,6000E+06	2,2380E-01	9,8000E+06	2,2310E-01	1,0000E+07	2,2220E-01	1,0500E+07	2,2070E-01	1,1000E+07	2,1900E-01
101	1,1500E+07	2,1760E-01	1,2000E+07	2,1620E-01	1,2500E+07	2,1510E-01	1,3000E+07	2,1390E-01	1,3500E+07	2,1270E-01
106	1,4000E+07	2,1160E-01	1,4500E+07	2,1040E-01	1,5000E+07	2,0930E-01				

NATURAL HELIUM GAMMA ENDF/B MATERIAL NO, 1088

SMOOTH NEUTRON CROSS SECTIONS

INTERPOLATION LAW BETWEEN ENERGIES

RANGE		DESCRIPTION
1	TO 128	Y LINEAR IN LN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1,0000E-05	3,1210E-01	2,5300E-02	3,1210E-01	1,0000E+02	3,1210E-01	2,0000E+02	3,1220E-01	3,0000E+02	3,1220E-01
6	5,0000E-02	3,1220E-01	7,0000E-02	3,1222E-01	1,0000E+03	3,1230E-01	2,0000E+03	3,1250E-01	3,0000E+03	3,1270E-01
11	5,0000E+03	3,1310E-01	7,0000E+03	3,1350E-01	1,2700E+04	3,1412E-01	2,0000E+04	3,1610E-01	3,0000E+04	3,1800E-01
16	5,0000E+04	3,2190E-01	7,0000E+04	3,2570E-01	1,0000E+05	3,3122E-01	1,5000E+05	3,4020E-01	2,0000E+05	3,4850E-01
21	2,5000E+05	3,5610E-01	3,0000E+05	3,6170E-01	3,5000E+05	3,6752E-01	4,0000E+05	3,7240E-01	4,5000E+05	3,7620E-01
26	5,0000E+05	3,7900E-01	5,5000E+05	3,8080E-01	6,0000E+05	3,8180E-01	6,5000E+05	3,8170E-01	7,0000E+05	3,8090E-01
31	7,5000E+05	3,7910E-01	8,0000E+05	3,7650E-01	8,5000E+05	3,7360E-01	9,0000E+05	3,7020E-01	9,5000E+05	3,6620E-01
36	1,0000E+06	3,6180E-01	1,0500E+06	3,5722E-01	1,1000E+06	3,5320E-01	1,1500E+06	3,5010E-01	1,1500E+06	3,4820E-01
41	1,1700E+06	3,4640E-01	1,2000E+06	3,4350E-01	1,2500E+06	3,3880E-01	1,3000E+06	3,3440E-01	1,3500E+06	3,3000E-01
46	1,4000E+06	3,2560E-01	1,4500E+06	3,2160E-01	1,5000E+06	3,1790E-01	1,6000E+06	3,1000E-01	1,7000E+06	3,0340E-01
51	1,8000E+06	2,9750E-01	1,9000E+06	2,9250E-01	2,0000E+06	2,8790E-01	2,1000E+06	2,8420E-01	2,2000E+06	2,8080E-01
56	2,3000E+06	2,7790E-01	2,4000E+06	2,7560E-01	2,5000E+06	2,7380E-01	2,6000E+06	2,7220E-01	2,7000E+06	2,7110E-01
61	2,8000E+06	2,6990E-01	2,9000E+06	2,6910E-01	3,0000E+06	2,6870E-01	3,2000E+06	2,6710E-01	3,4000E+06	2,6610E-01
66	3,6000E+06	2,6580E-01	3,8000E+06	2,6640E-01	4,0000E+06	2,6710E-01	4,2000E+06	2,6720E-01	4,4000E+06	2,6710E-01
71	4,6000E+06	2,6710E-01	4,8000E+06	2,6700E-01	5,0000E+06	2,6690E-01	5,2000E+06	2,6660E-01	5,4000E+06	2,6650E-01
76	5,6000E+06	2,6640E-01	5,8000E+06	2,6640E-01	6,0000E+06	2,6610E-01	6,2000E+06	2,6590E-01	6,4000E+06	2,6560E-01
81	6,6000E+06	2,6520E-01	6,8000E+06	2,6480E-01	7,0000E+06	2,6440E-01	7,2000E+06	2,6410E-01	7,4000E+06	2,6360E-01
86	7,6000E+06	2,6340E-01	7,8000E+06	2,6280E-01	8,0000E+06	2,6240E-01	8,2000E+06	2,6220E-01	8,4000E+06	2,6200E-01
91	8,6000E+06	2,6160E-01	8,8000E+06	2,6140E-01	9,0000E+06	2,6120E-01	9,2000E+06	2,6040E-01	9,4000E+06	2,6010E-01
96	9,6000E+06	2,5970E-01	9,8000E+06	2,5940E-01	1,0000E+07	2,5920E-01	1,0500E+07	2,5830E-01	1,1000E+07	2,5760E-01
101	1,1500E+07	2,5680E-01	1,2000E+07	2,5590E-01	1,2500E+07	2,5520E-01	1,3000E+07	2,5430E-01	1,3500E+07	2,5360E-01
106	1,4000E+07	2,5290E-01	1,4500E+07	2,5220E-01	1,5000E+07	2,5150E-01				

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

TRANSFORMATION MATRIX FROM CENTER OF MASS TO LAB SYSTEM

F(E,L,LAB)=(SUM OVER M) U(L,M)*F(E,M,C,M,)

L/M	0	1	2	3	4	5	6	7	8	9
0	1,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00	0,0000E+00
1	1,6800E-01	9,6190E-01	-1,5886E-01	3,5863E-02	-8,9905E-03	2,1026E-03	-5,1992E-04	1,2929E-04	-3,2264E-05	8,0695E-06
2	1,2819E-02	2,9686E-01	9,0195E-01	-2,7304E-01	7,8689E-02	-2,2043E-02	6,0621E-03	-1,6456E-03	4,4243E-04	-1,1807E-04
3	0,0000E+00	4,3546E-02	4,1067E-01	8,1319E-01	-3,6481E-01	1,2928E-01	-4,1723E-02	1,2786E-02	-3,7901E-03	1,0976E-03
4	-6,6316E-05	3,1827E-03	8,8718E-02	5,0658E-01	7,0133E-01	-4,3373E-01	1,8451E-01	-6,7791E-02	2,2986E-02	-7,4043E-03
5	0,0000E+00	0,0000E+00	1,1085E-02	1,4568E-01	5,8110E-01	5,7179E-01	-4,7795E-01	2,4855E-01	-9,9735E-02	3,7198E-02
6	6,3701E-07	-1,4971E-05	7,1925E-04	2,5474E-02	2,1109E-01	6,3117E-01	4,3878E-01	-4,9642E-01	2,9341E-01	-1,3644E-01
7	0,0000E+00	0,0000E+00	0,0000E+00	2,8877E-03	4,7327E-02	2,8107E-01	6,5479E-01	2,8461E-01	-4,8921E-01	3,3928E-01
8	-7,4839E-09	1,3935E-07	-3,1239E-06	1,6752E-04	7,1261E-03	7,7131E-02	3,5139E-01	6,5122E-01	1,4018E-01	-4,5769E-01
9	0,0000E+00	0,0000E+00	0,0000E+00	7,8975E-04	1,4600E-02	1,1474E-01	4,1767E-01	6,2111E-01	3,8218E-03	

INTERPOLATION LAW BETWEEN ENERGIES

RANGE		DESCRIPTION
1	TO 108	Y LINEAR IN LN X

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,2200E-11 MEV IS ISOTROPIC
2 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 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								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 2,5300E-08 MEV IS ISOTROPIC
2 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 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								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,2022E+04 MEV
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1-1,1300E+04 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	4,9983E-01	9,7500E-01	4,9983E-01	9,5000E-01	4,9984E-01	9,2500E-01	4,9984E-01	9,0000E-01	4,9985E-01
6	8,7500E-01	4,9985E-01	8,5000E-01	4,9986E-01	8,2500E-01	4,9986E-01	8,0000E-01	4,9986E-01	7,7500E-01	4,9987E-01
11	7,5000E-01	4,9987E-01	7,2500E-01	4,9988E-01	7,0000E-01	4,9988E-01	6,7500E-01	4,9989E-01	6,5000E-01	4,9989E-01
16	6,2500E-01	4,9989E-01	6,0000E-01	4,9990E-01	5,7500E-01	4,9990E-01	5,5000E-01	4,9991E-01	5,2500E-01	4,9991E-01
21	5,0000E-01	4,9992E-01	4,7500E-01	4,9992E-01	4,5000E-01	4,9992E-01	4,2500E-01	4,9993E-01	4,0000E-01	4,9993E-01
26	3,7500E-01	4,9994E-01	3,5000E-01	4,9994E-01	3,2500E-01	4,9994E-01	3,0000E-01	4,9995E-01	2,7500E-01	4,9995E-01
31	2,5000E-01	4,9996E-01	2,2500E-01	4,9996E-01	2,0000E-01	4,9997E-01	1,7500E-01	4,9997E-01	1,5000E-01	4,9997E-01
36	1,2500E-01	4,9998E-01	1,0000E-01	4,9998E-01	7,5000E-02	4,9999E-01	5,0000E-02	4,9999E-01	2,5000E-02	5,0000E-01
41	4,4238E-08	5,0000E-01	-2,5000E-02	5,0000E-01	-5,0000E-02	5,0001E-01	-7,5000E-02	5,0001E-01	-1,0000E-01	5,0002E-01
46	-1,2500E-01	5,0002E-01	-1,5000E-01	5,0003E-01	-1,7500E-01	5,0003E-01	-2,0000E-01	5,0003E-01	-2,2500E-01	5,0004E-01
51	-2,5000E-01	5,0004E-01	-2,7500E-01	5,0005E-01	-3,0000E-01	5,0005E-01	-3,2500E-01	5,0006E-01	-3,5000E-01	5,0006E-01
56	-3,7500E-01	5,0006E-01	-4,0000E-01	5,0007E-01	-4,2500E-01	5,0007E-01	-4,5000E-01	5,0008E-01	-4,7500E-01	5,0008E-01
61	-5,0000E-01	5,0008E-01	-5,2500E-01	5,0009E-01	-5,5000E-01	5,0009E-01	-5,7500E-01	5,0010E-01	-6,0000E-01	5,0010E-01
66	-6,2500E-01	5,0011E-01	-6,5000E-01	5,0011E-01	-6,7500E-01	5,0011E-01	-7,0000E-01	5,0012E-01	-7,2500E-01	5,0012E-01
71	-7,5000E-01	5,0013E-01	-7,7500E-01	5,0013E-01	-8,0000E-01	5,0014E-01	-8,2500E-01	5,0014E-01	-8,5000E-01	5,0014E-01
76	-8,7500E-01	5,0015E-01	-9,0000E-01	5,0015E-01	-9,2500E-01	5,0016E-01	-9,5000E-01	5,0016E-01	-9,7500E-01	5,0017E-01
81	-1,0000E+00	5,0017E-01								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 2,0000E+04 MEV
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1-2,2700E+04 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	4,9966E-01	9,7500E-01	4,9967E-01	9,5000E-01	4,9968E-01	9,2500E-01	4,9969E-01	9,0000E-01	4,9969E-01
6	8,7500E-01	4,9970E-01	8,5000E-01	4,9971E-01	8,2500E-01	4,9972E-01	8,0000E-01	4,9973E-01	7,7500E-01	4,9974E-01
11	7,5000E-01	4,9974E-01	7,2500E-01	4,9975E-01	7,0000E-01	4,9976E-01	6,7500E-01	4,9977E-01	6,5000E-01	4,9978E-01
16	6,2500E-01	4,9979E-01	6,0000E-01	4,9980E-01	5,7500E-01	4,9980E-01	5,5000E-01	4,9981E-01	5,2500E-01	4,9982E-01
21	5,0000E-01	4,9983E-01	4,7500E-01	4,9984E-01	4,5000E-01	4,9985E-01	4,2500E-01	4,9986E-01	4,0000E-01	4,9986E-01
26	3,7500E-01	4,9987E-01	3,5000E-01	4,9988E-01	3,2500E-01	4,9989E-01	3,0000E-01	4,9990E-01	2,7500E-01	4,9991E-01
31	2,5000E-01	4,9991E-01	2,2500E-01	4,9992E-01	2,0000E-01	4,9993E-01	1,7500E-01	4,9994E-01	1,5000E-01	4,9995E-01
36	1,2500E-01	4,9996E-01	1,0000E-01	4,9997E-01	7,5000E-02	4,9997E-01	5,0000E-02	4,9998E-01	2,5000E-02	4,9999E-01
41	4,4238E-08	5,0000E-01	-2,5000E-02	5,0001E-01	-5,0000E-02	5,0002E-01	-7,5000E-02	5,0003E-01	-1,0000E-01	5,0003E-01
46	-1,2500E-01	5,0004E-01	-1,5000E-01	5,0005E-01	-1,7500E-01	5,0006E-01	-2,0000E-01	5,0007E-01	-2,2500E-01	5,0008E-01
51	-2,5000E-01	5,0009E-01	-2,7500E-01	5,0009E-01	-3,0000E-01	5,0010E-01	-3,2500E-01	5,0011E-01	-3,5000E-01	5,0012E-01
56	-3,7500E-01	5,0013E-01	-4,0000E-01	5,0014E-01	-4,2500E-01	5,0014E-01	-4,5000E-01	5,0015E-01	-4,7500E-01	5,0016E-01
61	-5,0000E-01	5,0017E-01	-5,2500E-01	5,0018E-01	-5,5000E-01	5,0019E-01	-5,7500E-01	5,0020E-01	-6,0000E-01	5,0020E-01
66	-6,2500E-01	5,0021E-01	-6,5000E-01	5,0022E-01	-6,7500E-01	5,0023E-01	-7,0000E-01	5,0024E-01	-7,2500E-01	5,0025E-01
71	-7,5000E-01	5,0026E-01	-7,7500E-01	5,0026E-01	-8,0000E-01	5,0027E-01	-8,2500E-01	5,0028E-01	-8,5000E-01	5,0029E-01
76	-8,7500E-01	5,0030E-01	-9,0000E-01	5,0031E-01	-9,2500E-01	5,0031E-01	-9,5000E-01	5,0032E-01	-9,7500E-01	5,0033E-01
81	-1,0000E+00	5,0034E-01								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 3,0000E-04 MEV
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1-3,4000E-04 2 2,0000E+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	4,9949E-01	9,7500E-01	4,9950E-01	9,5000E-01	4,9952E-01	9,2500E-01	4,9953E-01	9,0000E-01	4,9954E-01
6	8,7500E-01	4,9955E-01	8,5000E-01	4,9957E-01	8,2500E-01	4,9958E-01	8,0000E-01	4,9959E-01	7,7500E-01	4,9960E-01
11	7,5000E-01	4,9962E-01	7,2500E-01	4,9963E-01	7,0000E-01	4,9964E-01	6,7500E-01	4,9966E-01	6,5000E-01	4,9967E-01
16	6,2500E-01	4,9968E-01	6,0000E-01	4,9969E-01	5,7500E-01	4,9971E-01	5,5000E-01	4,9972E-01	5,2500E-01	4,9973E-01
21	5,0000E-01	4,9975E-01	4,7500E-01	4,9976E-01	4,5000E-01	4,9977E-01	4,2500E-01	4,9978E-01	4,0000E-01	4,9980E-01
26	3,7500E-01	4,9981E-01	3,5000E-01	4,9982E-01	3,2500E-01	4,9983E-01	3,0000E-01	4,9985E-01	2,7500E-01	4,9986E-01
31	2,5000E-01	4,9987E-01	2,2500E-01	4,9989E-01	2,0000E-01	4,9990E-01	1,7500E-01	4,9991E-01	1,5000E-01	4,9992E-01
36	1,2500E-01	4,9994E-01	1,0000E-01	4,9995E-01	7,5000E-02	4,9996E-01	5,0000E-02	4,9997E-01	2,5000E-02	4,9999E-01
41	4,4238E-08	5,0000E-01	-2,5000E-02	5,0001E-01	-5,0000E-02	5,0003E-01	-7,5000E-02	5,0004E-01	-1,0000E-01	5,0005E-01
46	-1,2500E-01	5,0006E-01	-1,5000E-01	5,0008E-01	-1,7500E-01	5,0009E-01	-2,0000E-01	5,0010E-01	-2,2500E-01	5,0011E-01
51	-2,5000E-01	5,0013E-01	-2,7500E-01	5,0014E-01	-3,0000E-01	5,0015E-01	-3,2500E-01	5,0017E-01	-3,5000E-01	5,0018E-01
56	-3,7500E-01	5,0019E-01	-4,0000E-01	5,0020E-01	-4,2500E-01	5,0022E-01	-4,5000E-01	5,0023E-01	-4,7500E-01	5,0024E-01
61	-5,0000E-01	5,0026E-01	-5,2500E-01	5,0027E-01	-5,5000E-01	5,0028E-01	-5,7500E-01	5,0029E-01	-6,0000E-01	5,0031E-01
66	-6,2500E-01	5,0032E-01	-6,5000E-01	5,0033E-01	-6,7500E-01	5,0034E-01	-7,0000E-01	5,0036E-01	-7,2500E-01	5,0037E-01
71	-7,5000E-01	5,0038E-01	-7,7500E-01	5,0040E-01	-8,0000E-01	5,0041E-01	-8,2500E-01	5,0042E-01	-8,5000E-01	5,0043E-01
76	-8,7500E-01	5,0045E-01	-9,0000E-01	5,0046E-01	-9,2500E-01	5,0047E-01	-9,5000E-01	5,0048E-01	-9,7500E-01	5,0050E-01
81	-1,0000E+00	5,0051E-01								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 5,0000E-04 MEV
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1-5,6700E-04 2 2,0000E+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	4,9915E-01	9,7500E-01	4,9917E-01	9,5000E-01	4,9919E-01	9,2500E-01	4,9921E-01	9,0000E-01	4,9923E-01
6	8,7500E-01	4,9926E-01	8,5000E-01	4,9928E-01	8,2500E-01	4,9930E-01	8,0000E-01	4,9932E-01	7,7500E-01	4,9934E-01
11	7,5000E-01	4,9936E-01	7,2500E-01	4,9938E-01	7,0000E-01	4,9940E-01	6,7500E-01	4,9943E-01	6,5000E-01	4,9945E-01
16	6,2500E-01	4,9947E-01	6,0000E-01	4,9949E-01	5,7500E-01	4,9951E-01	5,5000E-01	4,9953E-01	5,2500E-01	4,9955E-01
21	5,0000E-01	4,9957E-01	4,7500E-01	4,9960E-01	4,5000E-01	4,9962E-01	4,2500E-01	4,9964E-01	4,0000E-01	4,9966E-01
26	3,7500E-01	4,9968E-01	3,5000E-01	4,9970E-01	3,2500E-01	4,9972E-01	3,0000E-01	4,9974E-01	2,7500E-01	4,9977E-01
31	2,5000E-01	4,9979E-01	2,2500E-01	4,9981E-01	2,0000E-01	4,9983E-01	1,7500E-01	4,9985E-01	1,5000E-01	4,9987E-01
36	1,2500E-01	4,9989E-01	1,0000E-01	4,9991E-01	7,5000E-02	4,9994E-01	5,0000E-02	4,9996E-01	2,5000E-02	4,9998E-01
41	4,4238E-08	5,0000E-01	-2,5000E-02	5,0002E-01	-5,0000E-02	5,0004E-01	-7,5000E-02	5,0006E-01	-1,0000E-01	5,0009E-01
46	-1,2500E-01	5,0011E-01	-1,5000E-01	5,0013E-01	-1,7500E-01	5,0015E-01	-2,0000E-01	5,0017E-01	-2,2500E-01	5,0019E-01
51	-2,5000E-01	5,0021E-01	-2,7500E-01	5,0023E-01	-3,0000E-01	5,0026E-01	-3,2500E-01	5,0028E-01	-3,5000E-01	5,0030E-01
56	-3,7500E-01	5,0032E-01	-4,0000E-01	5,0034E-01	-4,2500E-01	5,0036E-01	-4,5000E-01	5,0038E-01	-4,7500E-01	5,0040E-01
61	-5,0000E-01	5,0043E-01	-5,2500E-01	5,0045E-01	-5,5000E-01	5,0047E-01	-5,7500E-01	5,0049E-01	-6,0000E-01	5,0051E-01
66	-6,2500E-01	5,0053E-01	-6,5000E-01	5,0055E-01	-6,7500E-01	5,0057E-01	-7,0000E-01	5,0060E-01	-7,2500E-01	5,0062E-01
71	-7,5000E-01	5,0064E-01	-7,7500E-01	5,0066E-01	-8,0000E-01	5,0068E-01	-8,2500E-01	5,0070E-01	-8,5000E-01	5,0072E-01
76	-8,7500E-01	5,0074E-01	-9,0000E-01	5,0077E-01	-9,2500E-01	5,0079E-01	-9,5000E-01	5,0081E-01	-9,7500E-01	5,0083E-01
81	-1,0000E+00	5,0085E-01								

NATURAL HELIUM — ELASTIC — ENDF/B MATERIAL NO, 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,3520E+01 MEV

5 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION

1 4,7252E-01 2 2,3748E-01 3 6,1397E-03 4 1,6222E-03

F(MU)=(SUM OVER L) (?,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	1,8329E+00	9,7500E-01	1,7644E+00	9,5000E-01	1,6994E+00	9,2500E-01	1,6359E+00	9,0000E-01	1,5739E+00
6	8,7500E-01	1,5133E+00	8,5000E-01	1,4541E+00	8,2500E-01	1,3964E+00	8,0000E-01	1,3401E+00	7,7500E-01	1,2851E+00
11	7,5000E-01	1,2315E+00	7,2500E-01	1,1793E+00	7,0000E-01	1,1285E+00	6,7500E-01	1,0790E+00	6,5000E-01	1,0308E+00
16	6,2500E-01	9,8390E-01	6,0000E-01	9,3832E-01	5,7500E-01	8,9402E-01	5,5000E-01	8,5099E-01	5,2500E-01	8,0922E-01
21	5,0000E-01	7,6871E-01	4,7500E-01	7,2943E-01	4,5000E-01	6,9138E-01	4,2500E-01	6,5455E-01	4,0000E-01	6,1892E-01
26	3,7500E-01	5,8450E-01	3,5000E-01	5,5126E-01	3,2500E-01	5,1922E-01	3,0000E-01	4,8838E-01	2,7500E-01	4,5857E-01
31	2,5000E-01	4,3000E-01	2,2500E-01	4,0296E-01	2,0000E-01	3,7626E-01	1,7500E-01	3,5118E-01	1,5000E-01	3,2705E-01
36	1,2500E-01	3,0412E-01	1,0000E-01	2,8229E-01	7,5000E-02	2,6157E-01	5,0000E-02	2,4194E-01	2,5000E-02	2,2340E-01
41	4,4238E-08	2,0595E-01	-2,5000E-02	1,8958E-01	-5,0000E-02	1,7428E-01	-7,5000E-02	1,6005E-01	-1,0000E-01	1,4688E-01
46	-1,2500E-01	1,3478E-01	-1,5000E-01	1,2373E-01	-1,7500E-01	1,1374E-01	-2,0000E-01	1,0480E-01	-2,2500E-01	9,6904E-02
51	-2,5000E-01	9,0057E-02	-2,7500E-01	8,4253E-02	-3,0000E-01	7,9491E-02	-3,2500E-01	7,5769E-02	-3,5000E-01	7,3086E-02
56	-3,7500E-01	7,1448E-02	-4,0000E-01	7,0838E-02	-4,2500E-01	7,1258E-02	-4,5000E-01	7,2721E-02	-4,7500E-01	7,5228E-02
61	-5,0000E-01	7,8757E-02	-5,2500E-01	8,3330E-02	-5,5000E-01	8,8942E-02	-5,7500E-01	9,5594E-02	-6,0000E-01	1,0329E-01
66	-6,2500E-01	1,1202E-01	-6,5000E-01	1,2181E-01	-6,7500E-01	1,3264E-01	-7,0000E-01	1,4452E-01	-7,2500E-01	1,5745E-01
71	-7,5000E-01	1,7144E-01	-7,7500E-01	1,8690E-01	-8,0000E-01	2,0261E-01	-8,2500E-01	2,1988E-01	-8,5000E-01	2,3806E-01
76	-8,7500E-01	2,5740E-01	-9,0000E-01	2,7782E-01	-9,2500E-01	2,9933E-01	-9,5000E-01	3,2194E-01	-9,7500E-01	3,4565E-01
81	-1,0000E+00	3,7046E-01								

NATURAL HELIUM — ELASTIC — ENDF/B MATERIAL NO, 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,4000E+01 MEV

5 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION

1 4,7500E-01 2 2,3880E-01 3 7,2840E-03 4 1,7818E-03

F(MU)=(SUM OVER L) (?,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	1,8438E+00	9,7500E-01	1,7754E+00	9,5000E-01	1,7093E+00	9,2500E-01	1,6448E+00	9,0000E-01	1,5818E+00
6	8,7500E-01	1,5203E+00	8,5000E-01	1,4604E+00	8,2500E-01	1,4018E+00	8,0000E-01	1,3448E+00	7,7500E-01	1,2892E+00
11	7,5000E-01	1,2350E+00	7,2500E-01	1,1822E+00	7,0000E-01	1,1308E+00	6,7500E-01	1,0808E+00	6,5000E-01	1,0322E+00
16	6,2500E-01	9,8489E-01	6,0000E-01	9,3893E-01	5,7500E-01	8,9429E-01	5,5000E-01	8,5096E-01	5,2500E-01	8,0892E-01
21	5,0000E-01	7,6819E-01	4,7500E-01	7,2866E-01	4,5000E-01	6,9042E-01	4,2500E-01	6,5342E-01	4,0000E-01	6,1766E-01
26	3,7500E-01	5,8311E-01	3,5000E-01	5,4977E-01	3,2500E-01	5,1763E-01	3,0000E-01	4,8668E-01	2,7500E-01	4,5692E-01
31	2,5000E-01	4,2829E-01	2,2500E-01	4,0084E-01	2,0000E-01	3,7454E-01	1,7500E-01	3,4938E-01	1,5000E-01	3,2535E-01
36	1,2500E-01	3,0244E-01	1,0000E-01	2,8065E-01	7,5000E-02	2,5997E-01	5,0000E-02	2,4039E-01	2,5000E-02	2,2190E-01
41	4,4238E-08	2,0451E-01	-2,5000E-02	1,8819E-01	-5,0000E-02	1,7295E-01	-7,5000E-02	1,5878E-01	-1,0000E-01	1,4567E-01
46	-1,2500E-01	1,3363E-01	-1,5000E-01	1,2264E-01	-1,7500E-01	1,1271E-01	-2,0000E-01	1,0382E-01	-2,2500E-01	9,5974E-02
51	-2,5000E-01	8,9172E-02	-2,7500E-01	8,3489E-02	-3,0000E-01	7,8681E-02	-3,2500E-01	7,4987E-02	-3,5000E-01	7,2325E-02
56	-3,7500E-01	7,0694E-02	-4,0000E-01	7,0092E-02	-4,2500E-01	7,0518E-02	-4,5000E-01	7,1972E-02	-4,7500E-01	7,4453E-02
61	-5,0000E-01	7,7962E-02	-5,2500E-01	8,2499E-02	-5,5000E-01	8,8065E-02	-5,7500E-01	9,4661E-02	-6,0000E-01	1,0229E-01
66	-6,2500E-01	1,1095E-01	-6,5000E-01	1,2064E-01	-6,7500E-01	1,3138E-01	-7,0000E-01	1,4315E-01	-7,2500E-01	1,5597E-01
71	-7,5000E-01	1,6983E-01	-7,7500E-01	1,8474E-01	-8,0000E-01	2,0071E-01	-8,2500E-01	2,1774E-01	-8,5000E-01	2,3583E-01
76	-8,7500E-01	2,5499E-01	-9,0000E-01	2,7523E-01	-9,2500E-01	2,9654E-01	-9,5000E-01	3,1894E-01	-9,7500E-01	3,4243E-01
81	-1,0000E+00	3,6782E-01								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,4522E+01 MEV
5 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 4,7752E-01 2 2,4720E-01 3 8,5150E-03 4 1,9712E-03

F(MU)=(SUM OVER L) (.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	1,8454E+00	9,7500E-01	1,7866E+00	9,5000E-01	1,7194E+00	9,2500E-01	1,6539E+00	9,0000E-01	1,5899E+00
6	8,7500E-01	1,5275E+00	8,5000E-01	1,4667E+00	8,2500E-01	1,4073E+00	8,0000E-01	1,3495E+00	7,7500E-01	1,2932E+00
11	7,5000E-01	1,2384E+00	7,2500E-01	1,1852E+00	7,0000E-01	1,1331E+00	6,7500E-01	1,0826E+00	6,5000E-01	1,0335E+00
16	6,2500E-01	9,8477E-01	6,0000E-01	9,3942E-01	5,7500E-01	8,9443E-01	5,5000E-01	8,5079E-01	5,2500E-01	8,0847E-01
21	5,0000E-01	7,6746E-01	4,7500E-01	7,2775E-01	4,5000E-01	6,8932E-01	4,2500E-01	6,5216E-01	4,0000E-01	6,1625E-01
26	3,7500E-01	5,8159E-01	3,5000E-01	5,4816E-01	3,2500E-01	5,1595E-01	3,0000E-01	4,8494E-01	2,7500E-01	4,5513E-01
31	2,5000E-01	4,2650E-01	2,2500E-01	3,9904E-01	2,0000E-01	3,7274E-01	1,7500E-01	3,4760E-01	1,5000E-01	3,2360E-01
36	1,2500E-01	3,0073E-01	1,0000E-01	2,7898E-01	7,5000E-02	2,5835E-01	5,0000E-02	2,3883E-01	2,5000E-02	2,2041E-01
41	4,4238E-08	2,0308E-01	-2,5000E-02	1,8603E-01	-5,0000E-02	1,7166E-01	-7,5000E-02	1,5756E-01	-1,0000E-01	1,4493E-01
46	-1,2500E-01	1,3255E-01	-1,5000E-01	1,2163E-01	-1,7500E-01	1,1176E-01	-2,0000E-01	1,0293E-01	-2,2500E-01	9,5146E-02
51	-2,5000E-01	8,8395E-02	-2,7500E-01	8,2677E-02	-3,0000E-01	7,7989E-02	-3,2500E-01	7,4327E-02	-3,5000E-01	7,1690E-02
56	-3,7500E-01	7,0075E-02	-4,0000E-01	6,9481E-02	-4,2500E-01	6,9906E-02	-4,5000E-01	7,1349E-02	-4,7500E-01	7,3809E-02
61	-5,0000E-01	7,7287E-02	-5,2500E-01	8,1783E-02	-5,5000E-01	8,7296E-02	-5,7500E-01	9,3828E-02	-6,0000E-01	1,0138E-01
66	-6,2500E-01	1,2995E-01	-6,5000E-01	1,1955E-01	-6,7500E-01	1,3017E-01	-7,0000E-01	1,4182E-01	-7,2500E-01	1,5451E-01
71	-7,5000E-01	1,6822E-01	-7,7500E-01	1,8298E-01	-8,0000E-01	1,9878E-01	-8,2500E-01	2,1562E-01	-8,5000E-01	2,3352E-01
76	-8,7500E-01	2,5248E-01	-9,0000E-01	2,7250E-01	-9,2500E-01	2,9358E-01	-9,5000E-01	3,1574E-01	-9,7500E-01	3,3899E-01
81	-1,0000E+00	3,6332E-01								

NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO, 1088
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1,5000E+01 MEV
5 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 4,8012E-01 2 2,4180E-01 3 9,8092E-03 4 2,1710E-03

F(MU)=(SUM OVER L) (.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	1,8688E+00	9,7500E-01	1,7987E+00	9,5000E-01	1,7303E+00	9,2500E-01	1,6636E+00	9,0000E-01	1,5986E+00
6	8,7500E-01	1,5352E+00	8,5000E-01	1,4734E+00	8,2500E-01	1,4133E+00	8,0000E-01	1,3547E+00	7,7500E-01	1,2976E+00
11	7,5000E-01	1,2421E+00	7,2500E-01	1,1881E+00	7,0000E-01	1,1355E+00	6,7500E-01	1,0845E+00	6,5000E-01	1,0349E+00
16	6,2500E-01	9,8669E-01	6,0000E-01	9,3992E-01	5,7500E-01	8,9456E-01	5,5000E-01	8,5057E-01	5,2500E-01	8,0795E-01
21	5,0000E-01	7,6667E-01	4,7500E-01	7,2672E-01	4,5000E-01	6,8808E-01	4,2500E-01	6,5074E-01	4,0000E-01	6,1468E-01
26	3,7500E-01	5,7989E-01	3,5000E-01	5,4635E-01	3,2500E-01	5,1405E-01	3,0000E-01	4,8298E-01	2,7500E-01	4,5312E-01
31	2,5000E-01	4,2447E-01	2,2500E-01	3,9700E-01	2,0000E-01	3,7270E-01	1,7500E-01	3,4558E-01	1,5000E-01	3,2160E-01
36	1,2500E-01	2,9877E-01	1,0000E-01	2,7707E-01	7,5000E-02	2,5649E-01	5,0000E-02	2,3703E-01	2,5000E-02	2,1868E-01
41	4,4238E-08	2,0141E-01	-2,5000E-02	1,8524E-01	-5,0000E-02	1,7215E-01	-7,5000E-02	1,5612E-01	-1,0000E-01	1,4317E-01
46	-1,2500E-01	1,3127E-01	-1,5000E-01	1,2043E-01	-1,7500E-01	1,1063E-01	-2,0000E-01	1,0187E-01	-2,2500E-01	9,4148E-02
51	-2,5000E-01	8,7458E-02	-2,7500E-01	8,1796E-02	-3,0000E-01	7,7157E-02	-3,2500E-01	7,3538E-02	-3,5000E-01	7,0936E-02
56	-3,7500E-01	6,9348E-02	-4,0000E-01	6,8772E-02	-4,2500E-01	6,9206E-02	-4,5000E-01	7,0650E-02	-4,7500E-01	7,3100E-02
61	-5,0000E-01	7,6559E-02	-5,2500E-01	8,1023E-02	-5,5000E-01	8,6495E-02	-5,7500E-01	9,2979E-02	-6,0000E-01	1,0046E-01
66	-6,2500E-01	1,0896E-01	-6,5000E-01	1,1847E-01	-6,7500E-01	1,2900E-01	-7,0000E-01	1,4054E-01	-7,2500E-01	1,5318E-01
71	-7,5000E-01	1,6668E-01	-7,7500E-01	1,8129E-01	-8,0000E-01	1,9693E-01	-8,2500E-01	2,1360E-01	-8,5000E-01	2,3132E-01
76	-8,7500E-01	2,5088E-01	-9,0000E-01	2,6989E-01	-9,2500E-01	2,9076E-01	-9,5000E-01	3,1270E-01	-9,7500E-01	3,3570E-01
81	-1,0000E+00	3,5979E-01								

APPENDIX M

Sample Graphical Display

The following is a sample graphical display of the cross sections for the element helium. These plots were obtained by using the PLOTFB code. Note that most of the angular distributions of secondary neutrons have been omitted.

HELIUM ANL EVAL-JUN68 E.M.PENNINGTON (NATURAL ELEMENT)
ANL-7462 (OCT,1968) DIST-JUN70 REV-JUN70
* * * * * * *
DATA MODIFIED JUNE,1970 TO CONFORM TO ENDF/B-II FORMATS
* * * * * * *
HELIUM CROSS SECTIONS MATERIAL 1088
COMPILED BY ED PENNINGTON, ARGONNE NATIONAL LAB., IN JUNE 1968.
NATURAL HELIUM CONSISTS OF 0.00013 PER CENT HE-3 AND 99.99987 PER CENT HE-4.
BECAUSE OF THE LOW ABUNDANCE OF HE-3, ONLY ITS (N,P) CROSS SECTION, WHICH IS VERY LARGE AT LOW ENERGIES, NEED BE CONSIDERED. ELASTIC SCATTERING IS THE ONLY POSSIBLE REACTION FOR NEUTRONS INCIDENT ON HE-4 AT ENERGIES BELOW 15 MEV. THUS THE ELASTIC SCATTERING CROSS SECTION AND VALUES OF MU BAR(LAB), XI, AND GAMMA ARE GIVEN IN FILE 3, AND ELASTIC SCATTERING LEGENDRE COEFFICIENTS ARE GIVEN IN FILE 4. PARAMETERS FOR A FREE GAS THERMAL SCATTERING LAW ARE IN FILE 7.
THE ELASTIC SCATTERING CROSS SECTION AND THE LEGENDRE EXPANSION COEFFICIENTS WERE CALCULATED FROM S-,P-, AND D-WAVE PHASE SHIFTS USING A FORTRAN PROGRAM WRITTEN FOR THE PURPOSE. THE PHASE SHIFTS WERE READ FROM SMOOTH CURVES BASED ON TABLE I OF REF.1. AT ENERGIES BELOW THE 300 KEV. LOWER LIMIT OF TABLE I, EACH OF THE TWO P-WAVE PHASE SHIFTS WAS OBTAINED BY ASSUMING A FUNCTIONAL FORM BASED ON THE LOW ENERGY LIMIT FOR A SINGLE P-WAVE RESONANCE, WITH PARAMETERS DETERMINED FROM FITTING THE LOW ENERGY PHASE SHIFTS OF TABLE I. THE S-WAVE PHASE SHIFT BELOW 300 KEV. WAS CALCULATED USING HARD SPHERE SCATTERING AND A NUCLEAR RADIUS, A= 2.4 FERMI. THIS YIELDS THE THERMAL SCATTERING CROSS SECTION = 4.*PI*A**2= 0.7238 BARNS IN AGREEMENT WITH THE EXPERIMENTAL VALUE OF 0.73+- 0.05 BARNS (REF.2). THE LOW ENERGY S-WAVE PHASE SHIFTS OF TABLE I ARE CONSISTENT WITH A NUCLEAR RADIUS OF ABOUT 2.48 FERMI, AND SO WOULD YIELD A SOMEWHAT HIGH THERMAL CROSS SECTION.
VALUES OF MU BAR(LAB), XI, AND GAMMA WERE CALCULATED FROM THE LEGENDRE COEFFICIENTS USING A FORTRAN PROGRAM, MUXIGA. THIS PROGRAM USES THE EQUATIONS OF REF.3-5.
AN ELASTIC SCATTERING TRANSFORMATION MATRIX FROM THE CENTER-OF-MASS TO THE LABORATORY SYSTEM WAS COMPUTED USING CHAD (REF.6).
THE (N,P) CROSS SECTION FOR HE-3 IS THAT RECOMMENDED IN THE EVALUATION OF HE-3 BY J. ALS-NIELSEN GIVEN IN REF.7. EXTENSION FROM 10 TO 15 MEV. WAS MADE USING LINEAR EXTRAPOLATION ON A LOG SIGMA- LOG E SCALE.
THE TOTAL CROSS SECTION IS THE SUM OF THE ELASTIC SCATTERING AND (N,P) CROSS SECTIONS.
COMMENTS-
THE PHASE SHIFTS OF REF.1 ARE OPTICAL MODEL PHASE SHIFTS CHOSEN TO FIT BOTH ANGULAR DISTRIBUTION AND POLARIZATION DATA AT MANY ENERGIES. THE TOTAL SCATTERING CROSS SECTION IS ALSO FIT WITHIN THE SCATTER OF THE EXPERIMENTAL POINTS. ANOTHER RECENT SET OF PHASE SHIFTS (REF.8) IS NOT VERY DIFFERENT FROM THOSE USED HERE, AND COULD ALSO HAVE BEEN USED IN THE PRESENT WORK. THERE SHOULD

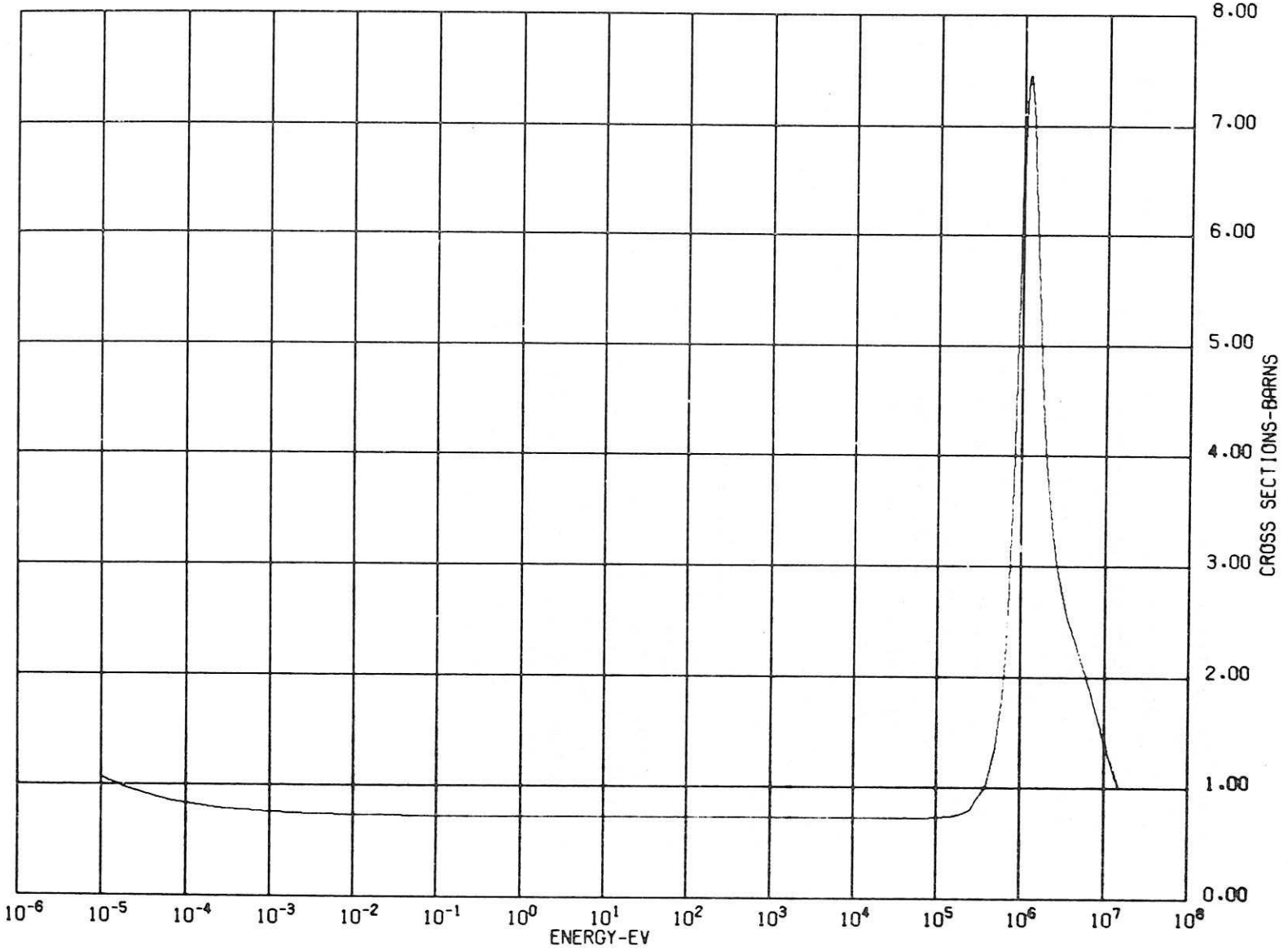
BE NO SERIOUS ERRORS IN THE HE-4 DATA CALCULATED FROM THE PHASE SHIFTS.
AS DISCUSSED IN REF.7, THE HE-3 (N,P) CROSS SECTION IS RATHER WELL KNOWN. PROBABLY MORE ERROR IS INTRODUCED INTO THE (N,P) CROSS SECTION FOR NATURAL HELIUM BY THE UNCERTAINTY IN THE HE-3 ISOTOPIC ABUNDANCE THAN BY THE UNCERTAINTY IN THE HE-3 (N,P) CROSS SECTION ITSELF.
PREVIOUS EVALUATIONS OF HELIUM FOR REACTOR CALCULATIONS INCLUDE THOSE OF J.J.SCHMIDT (REF.9) AND B.R.S.BUCKINGHAM ET AL (REF.10). SCHMIDT-S EVALUATION INCLUDES THE (N,P) CROSS SECTION FOR HE-3, AND SIGMA ELASTIC, MU BAR(LAB), AND A SET OF PHASE SHIFTS FOR HE-4. BUCKINGHAM ET AL GIVE SEPARATE EVALUATIONS FOR HE-3 AND HE-4. FOR HE-3 ELASTIC, (N,P), (N,D), AND (N,2N) CROSS SECTIONS ARE GIVEN, AS WELL AS ELASTIC ANGULAR DISTRIBUTIONS. THE HE-4 EVALUATION GIVES THE ELASTIC CROSS SECTION AND ANGULAR DISTRIBUTIONS.
REFERENCES-

1. G.R.SATCHLER ET AL. NUCLEAR PHYSICS A112,1-31,(1968).
2. R.GENIN ET AL. JOURNAL DE PHYSIQUE ET LE RADIUM 24,21-26,1963.
3. H.AMSTER JOURNAL OF APPLIED PHYSICS 27,3,307,(1956).
4. H.AMSTER JOURNAL OF APPLIED PHYSICS 27,6,663,(1956).
5. H.AMSTER JOURNAL OF APPLIED PHYSICS 29,4,623-627,(1958).
6. R.F.BERLAND NAA-SR-11231,(1965).
7. E.N.E.A. NEUTRON DATA COMPILATION CENTRE NEWSLETTER NO.6,1967.
8. B.HOOP,JR. AND H.H.BARSCHALL NUCLEAR PHYSICS 83,65-79,(1966).
9. J.J.SCHMIDT KFK-120,PARTS II,III,(1962),PART I,(1966).
10. B.R.S.BUCKINGHAM ET AL AWRE O-28/60,(1961).

NATURAL HELIUM

TOTAL
SMOOTH NEUTRON CROSS SECTIONS

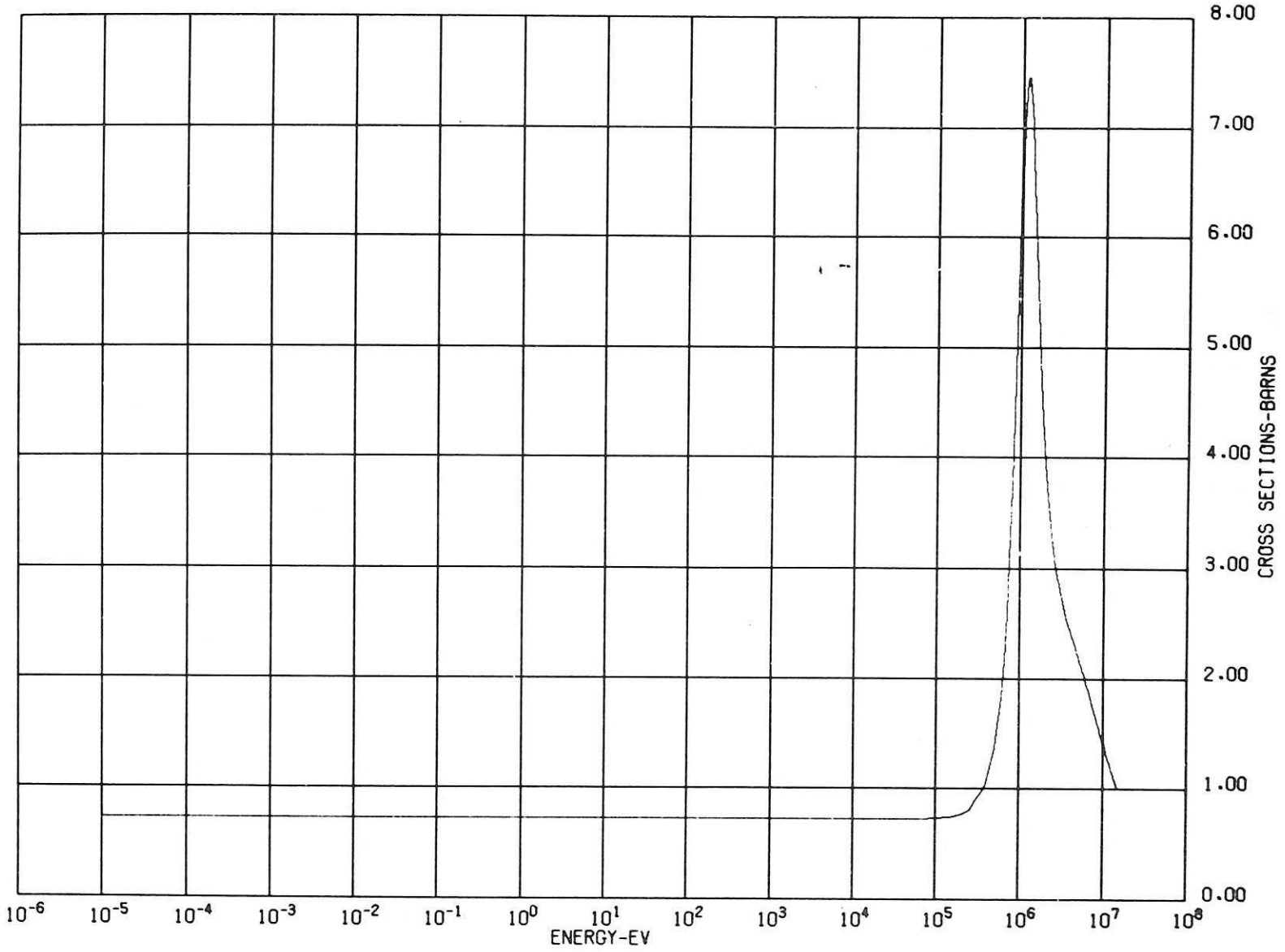
ENDF/B MATERIAL NO. 1288 1SS



NATURAL HELIUM

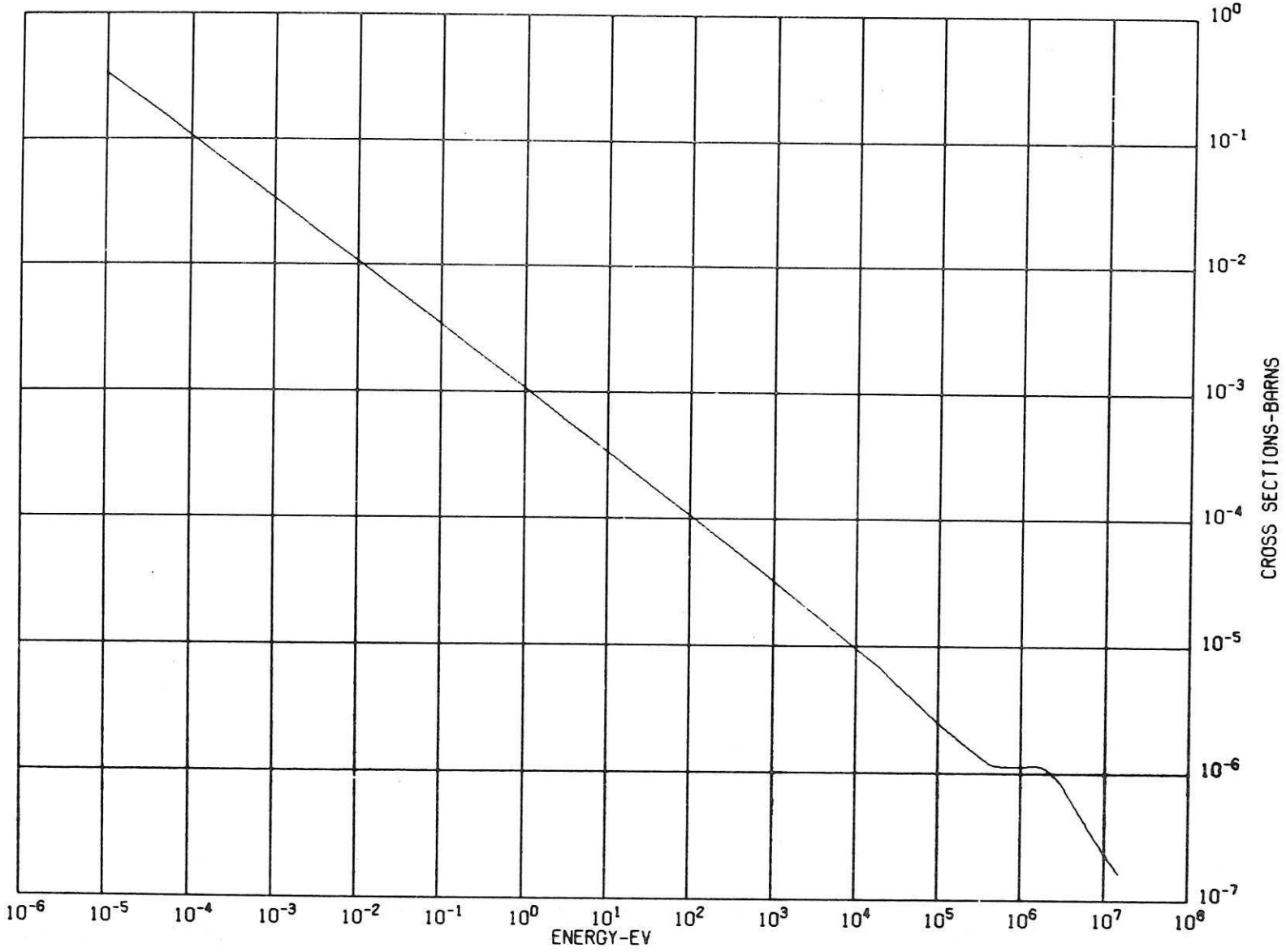
ELASTIC
SMOOTH NEUTRON CROSS SECTIONS

ENDF/B MATERIAL NO. 1088 189



NATURAL HELIUM (N,P) ENDF/B MATERIAL NO. 1388 19?

SMOOTH NEUTRON CROSS SECTIONS

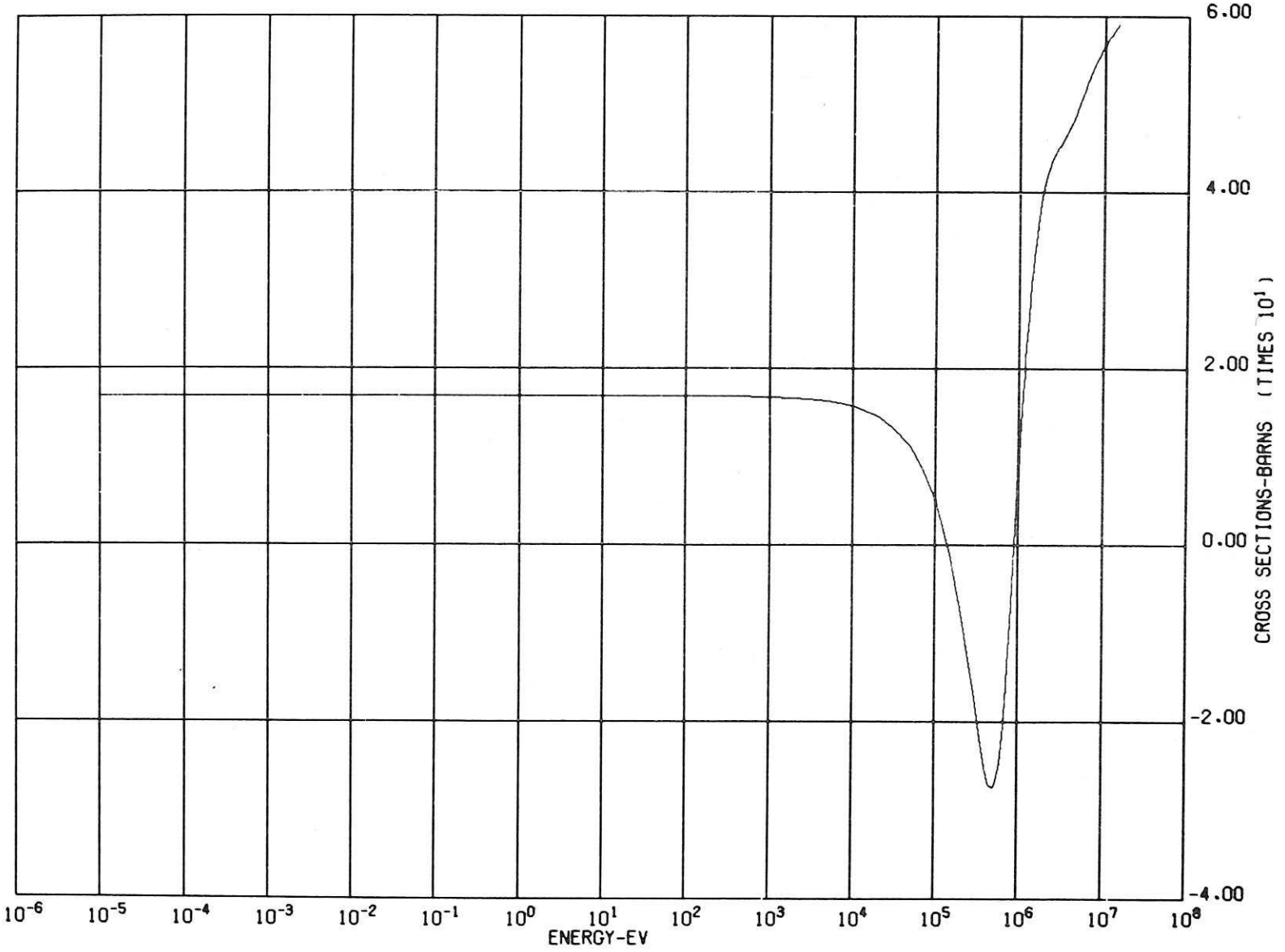


NATURAL HELIUM

MU BAR

ENDF/B MATERIAL NO. 1088

SMOOTH NEUTRON CROSS SECTIONS

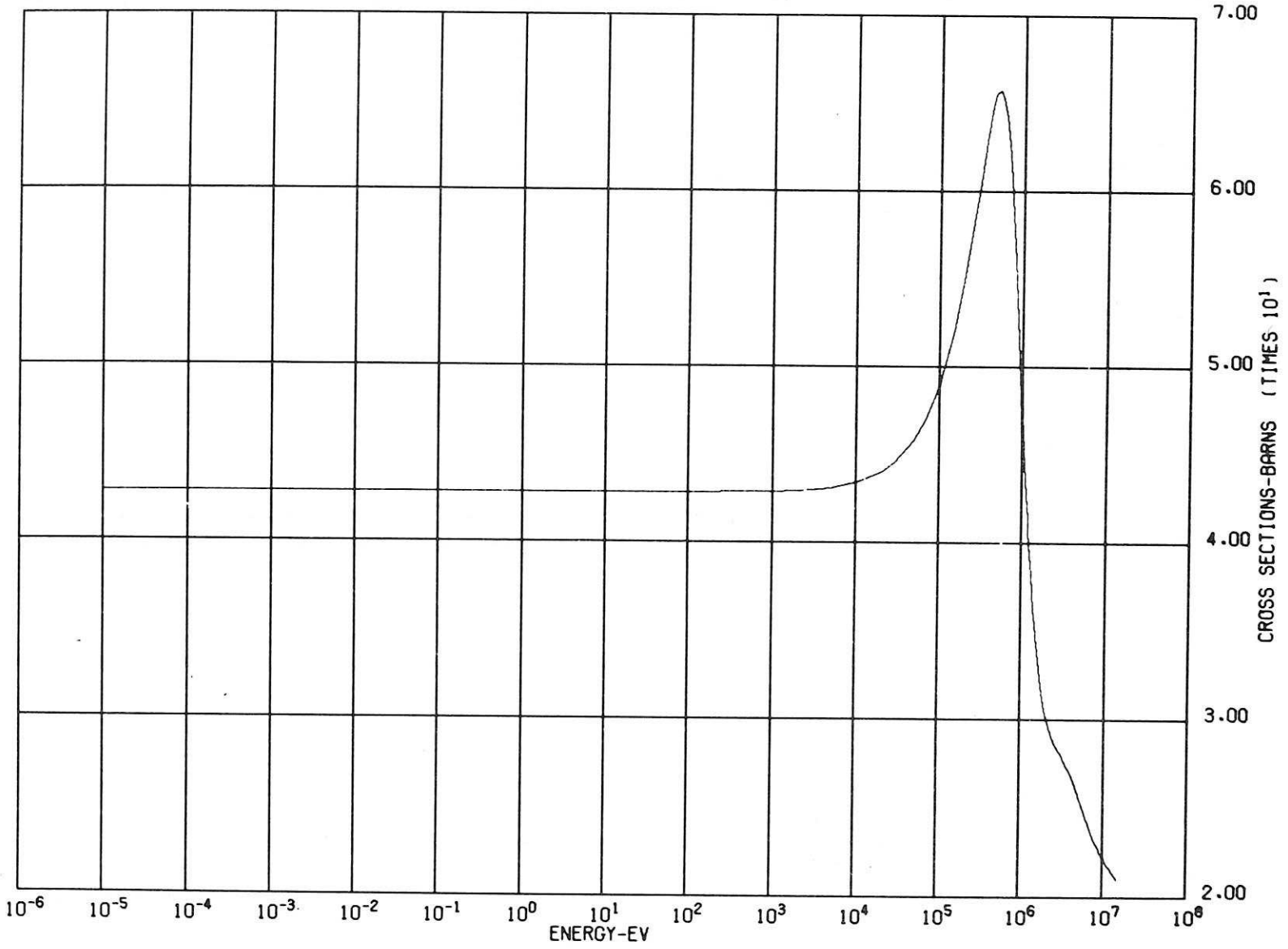


NATURAL HELIUM

XI

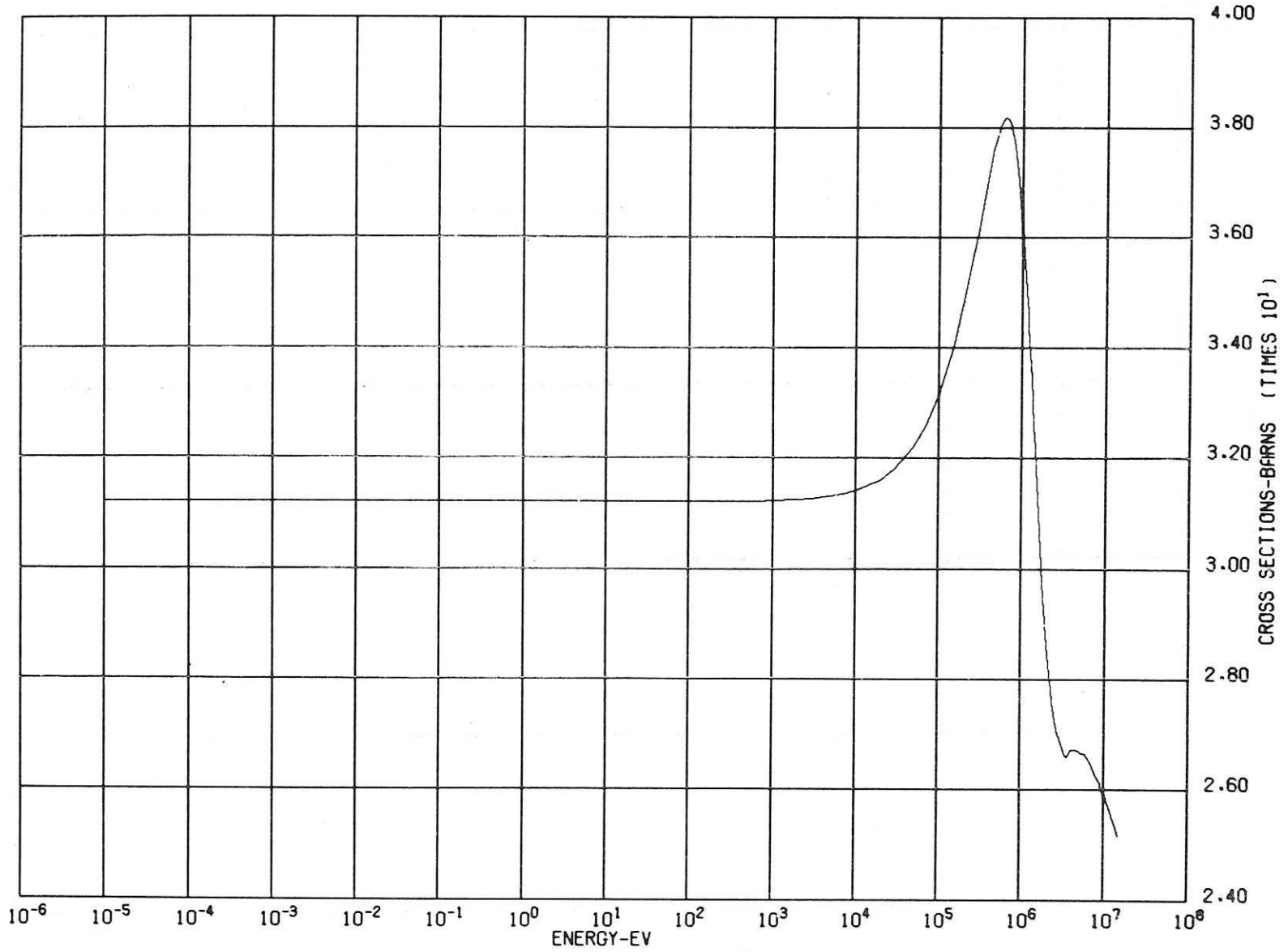
ENDF/B MATERIAL NO. 1088

SMOOTH NEUTRON CROSS SECTIONS



NATURAL HELIUM GAMMA ENDF/B MATERIAL NO. 1088 193

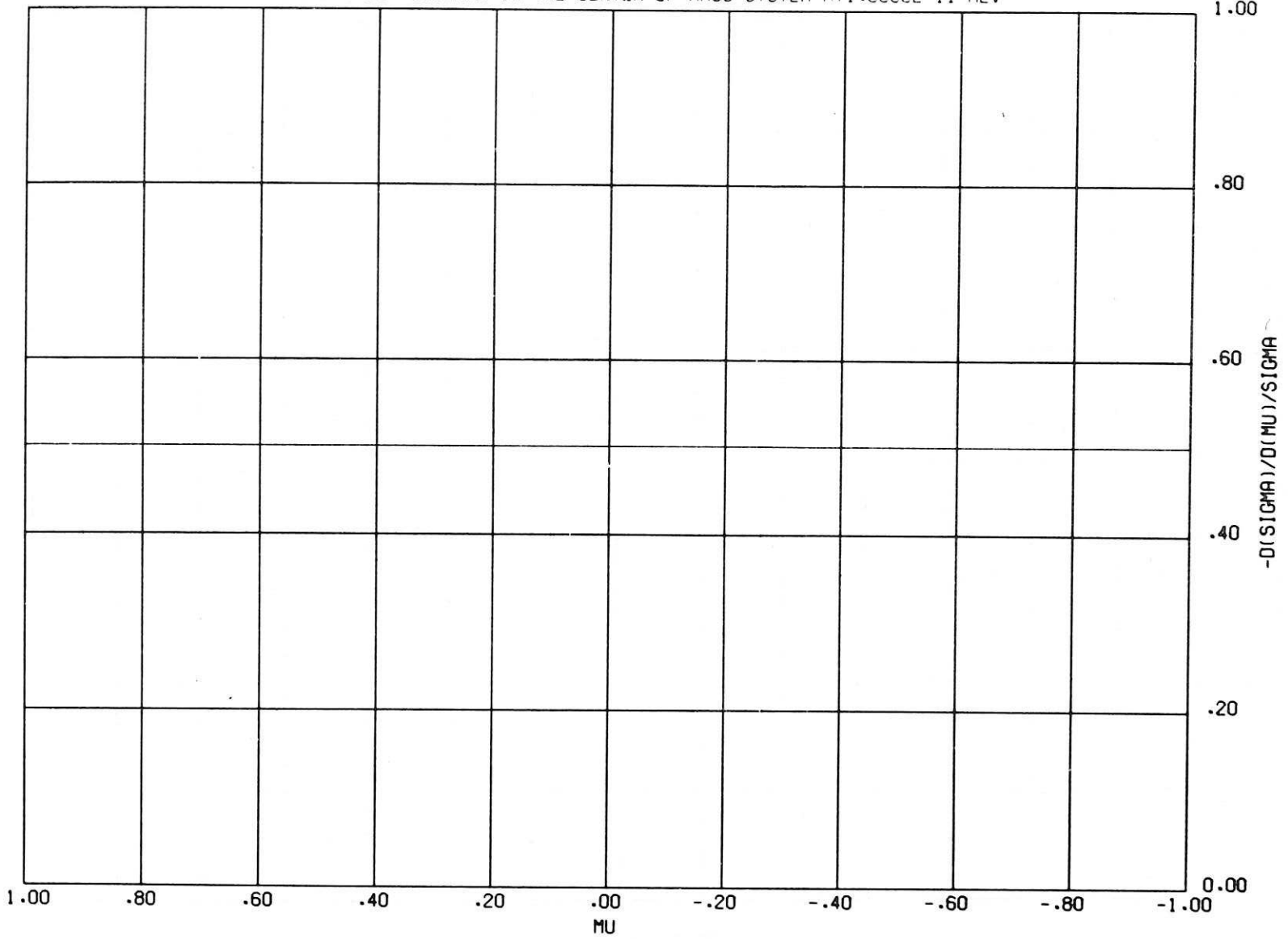
SMOOTH NEUTRON CROSS SECTIONS



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088 194

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

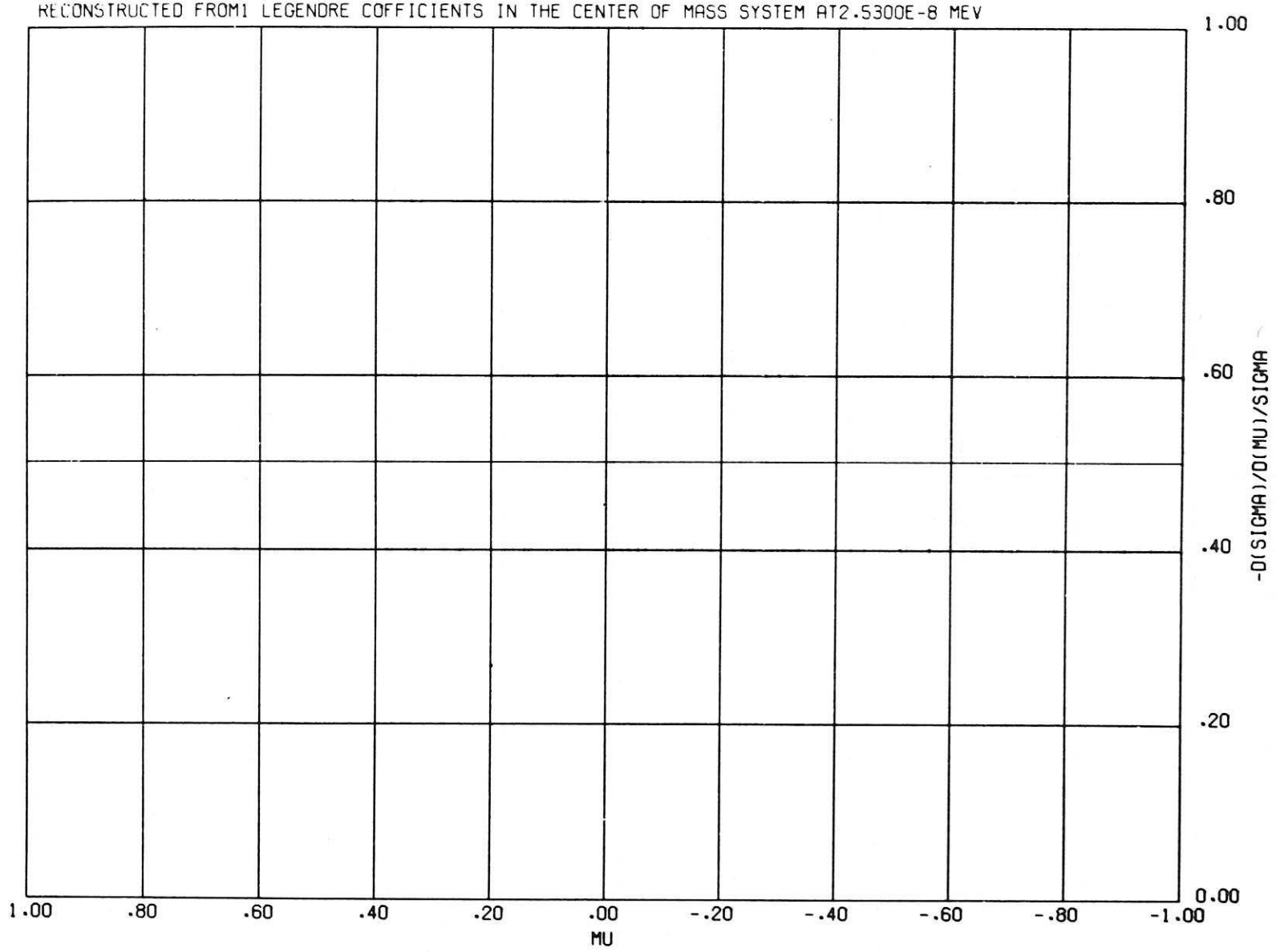
RECONSTRUCTED FROM1 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.0000E-11 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088 19

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

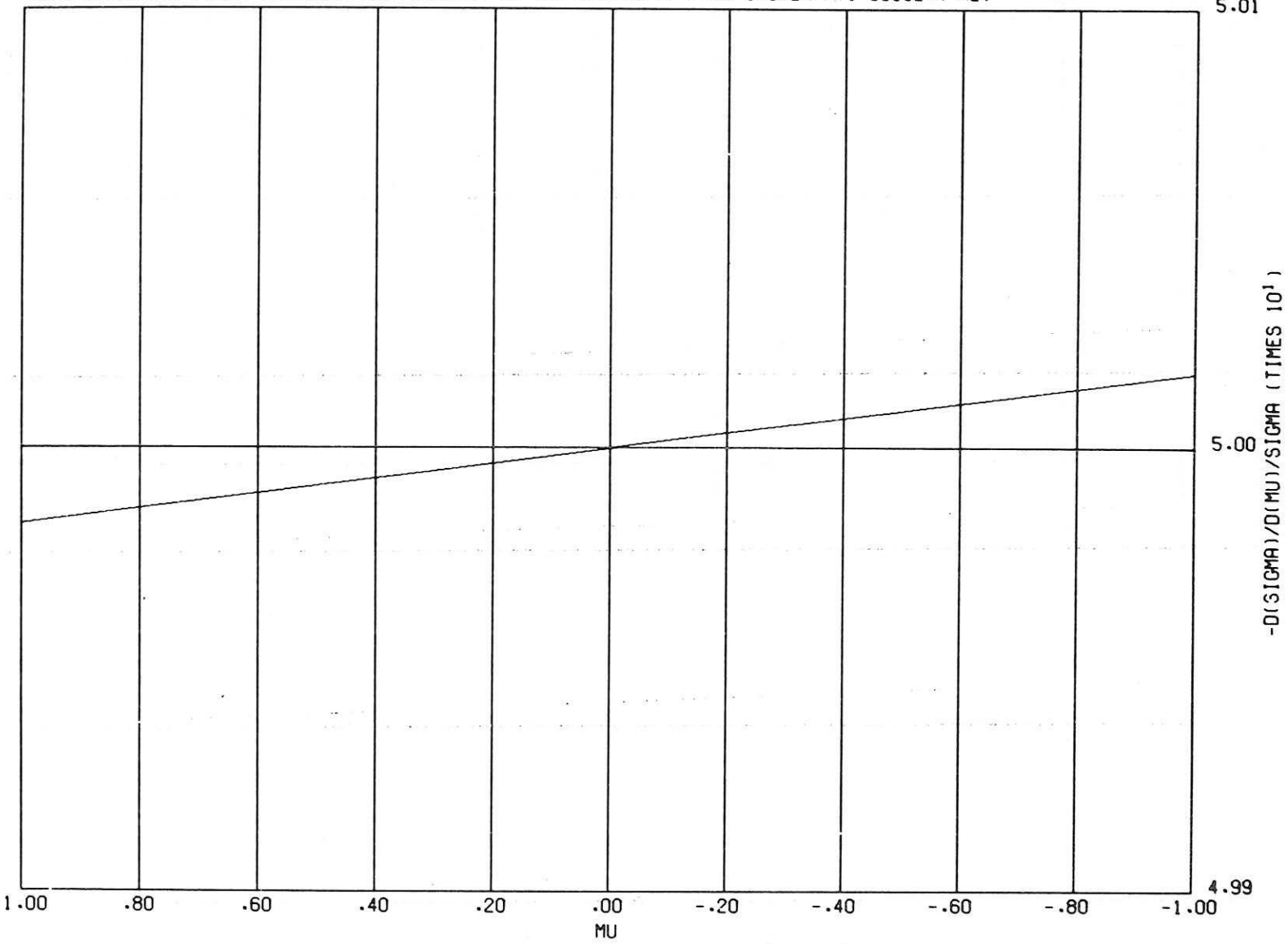
RECONSTRUCTED FROM1 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT2.5300E-8 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088 196

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

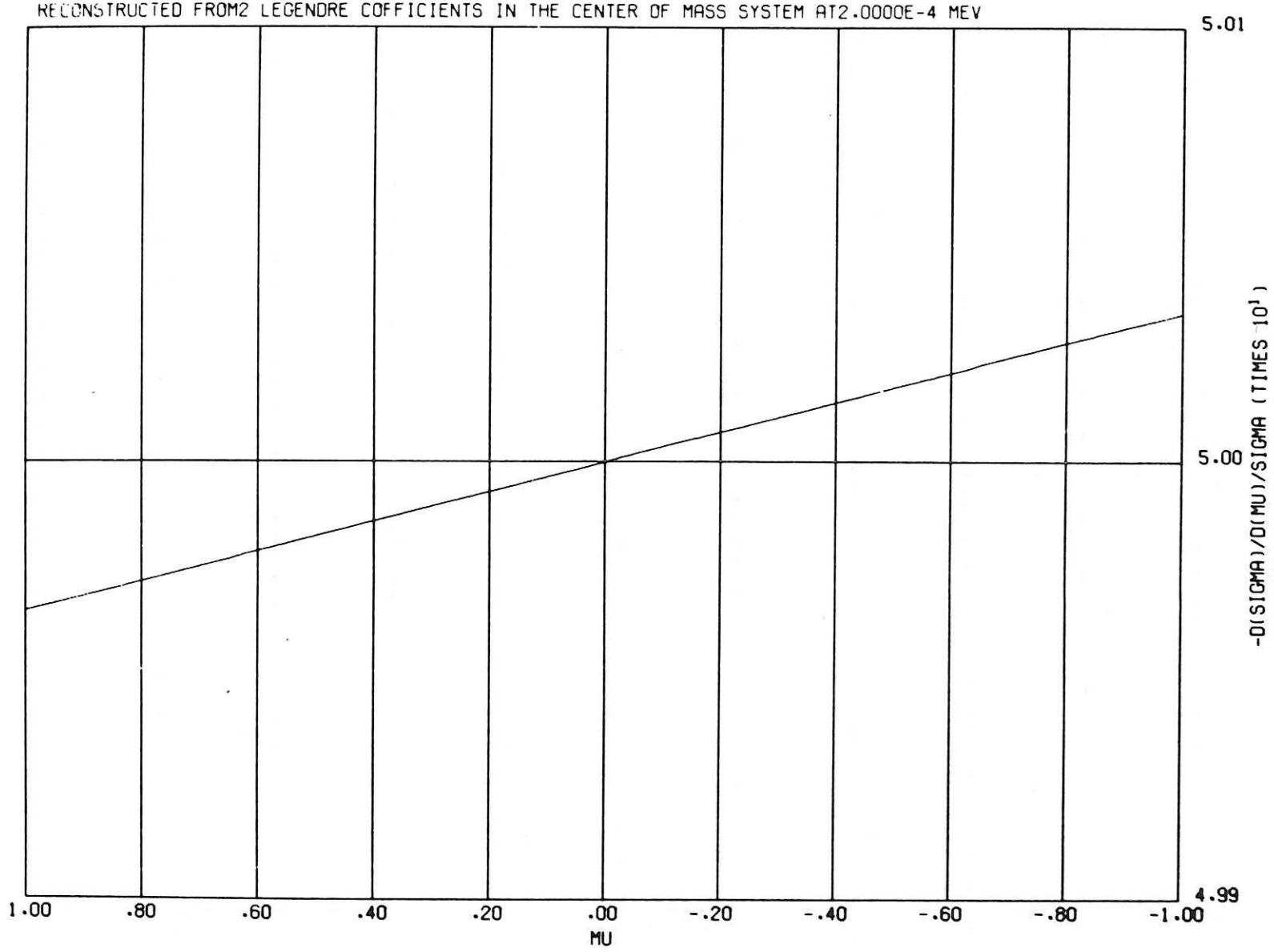
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.0000E-4 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

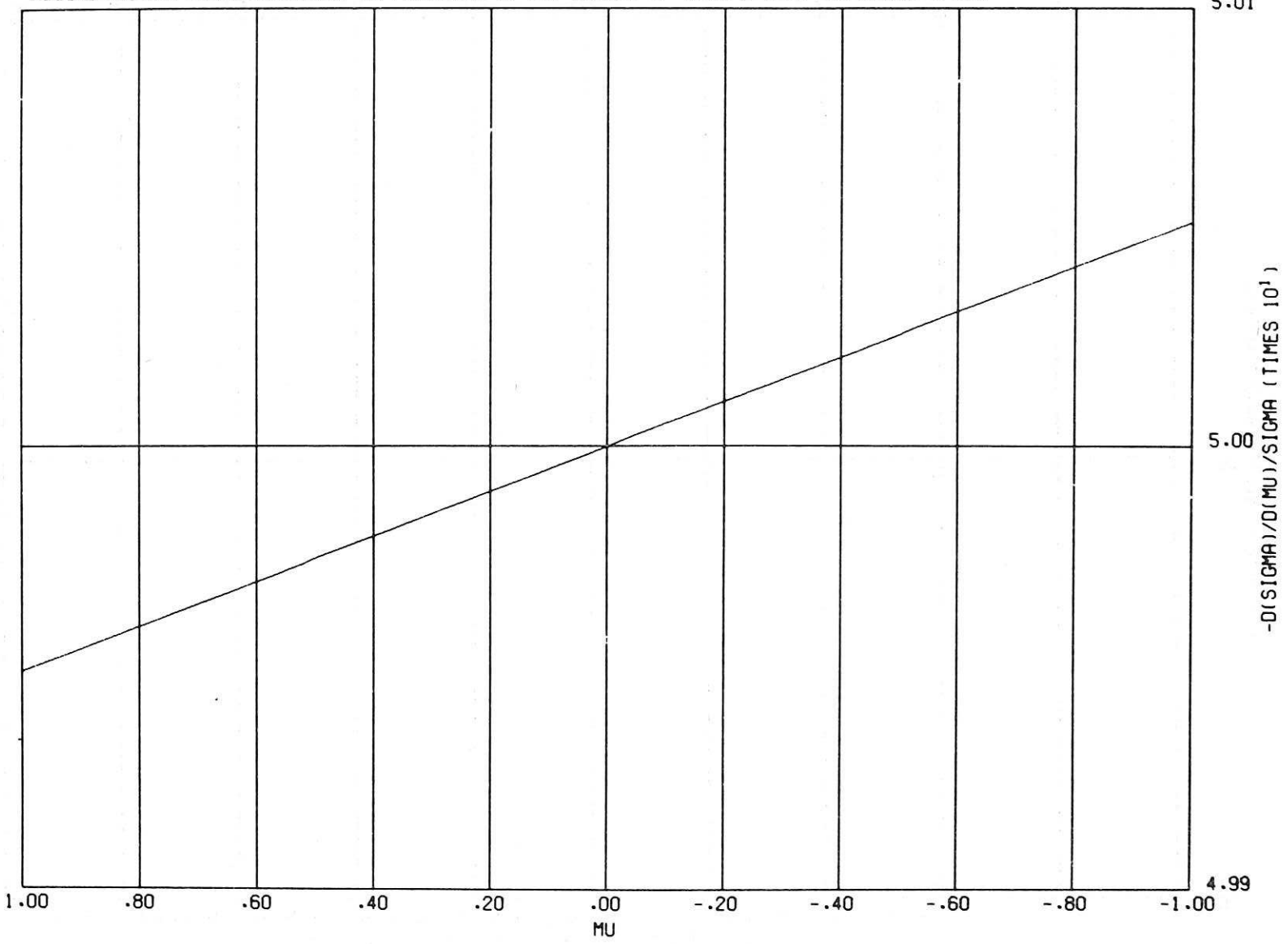
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT2.0000E-4 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288 1288

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

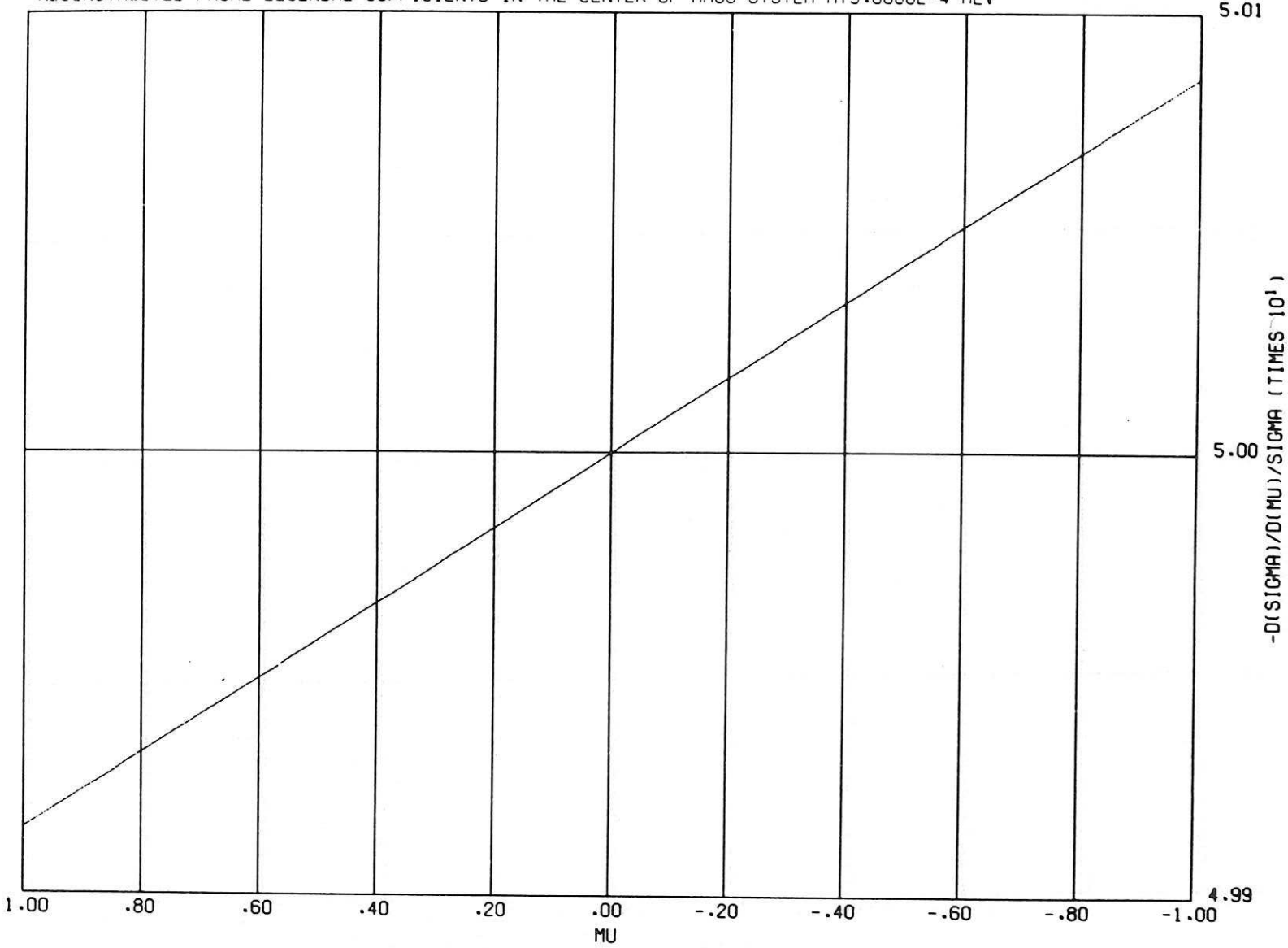
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT3.0000E-4 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

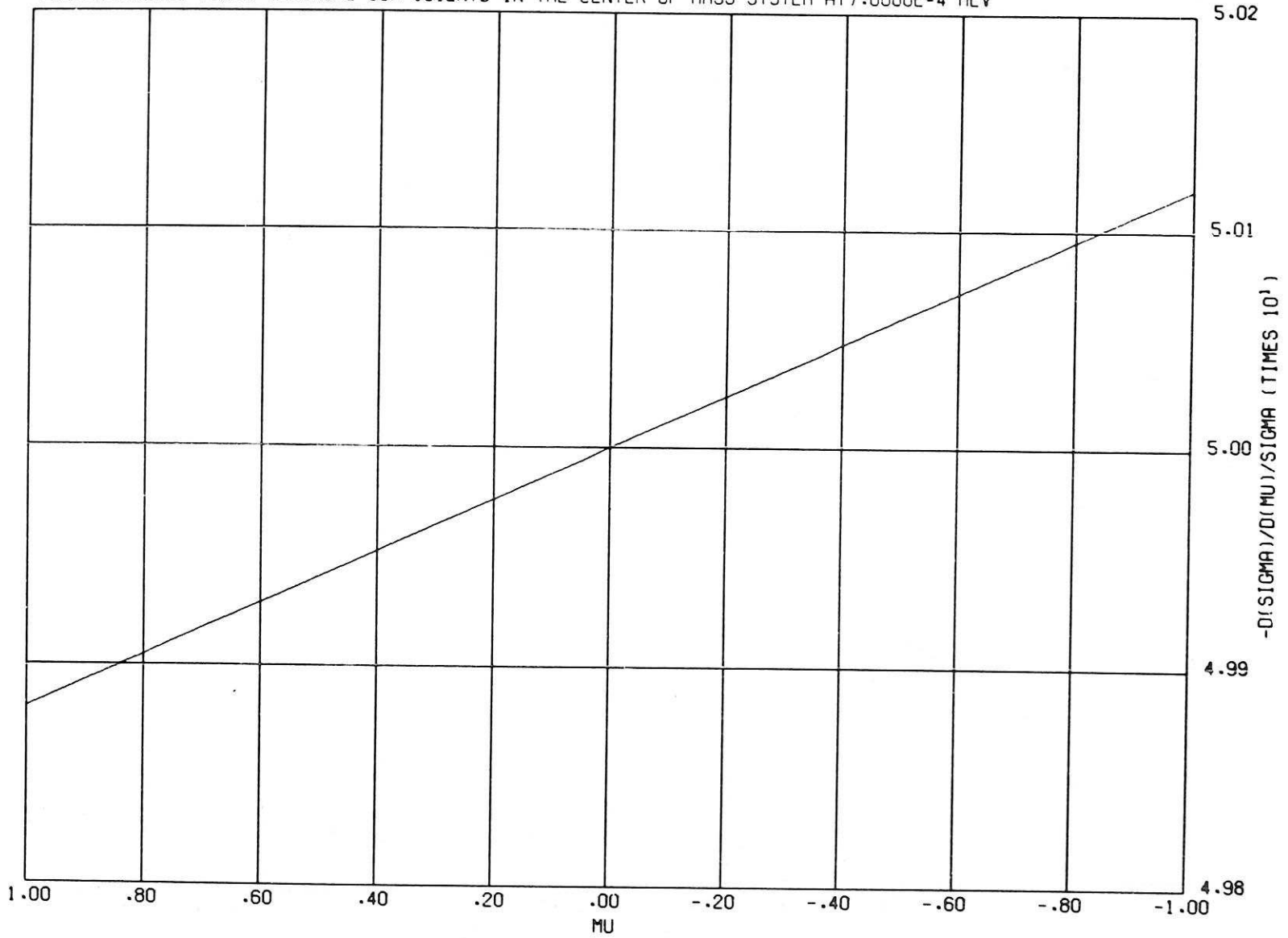
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT5.0000E-4 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1088 [illegible]

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

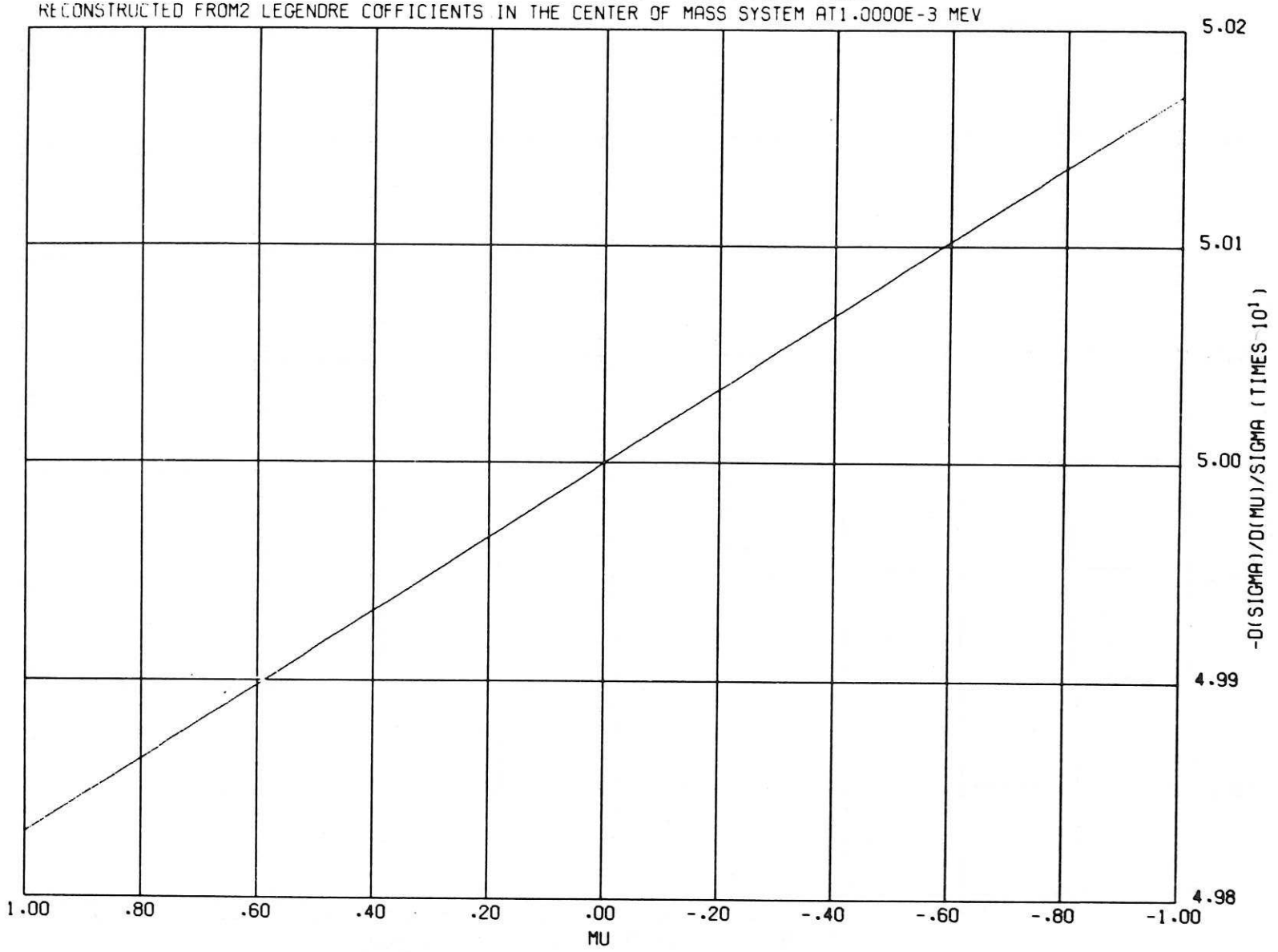
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT7.0000E-4 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

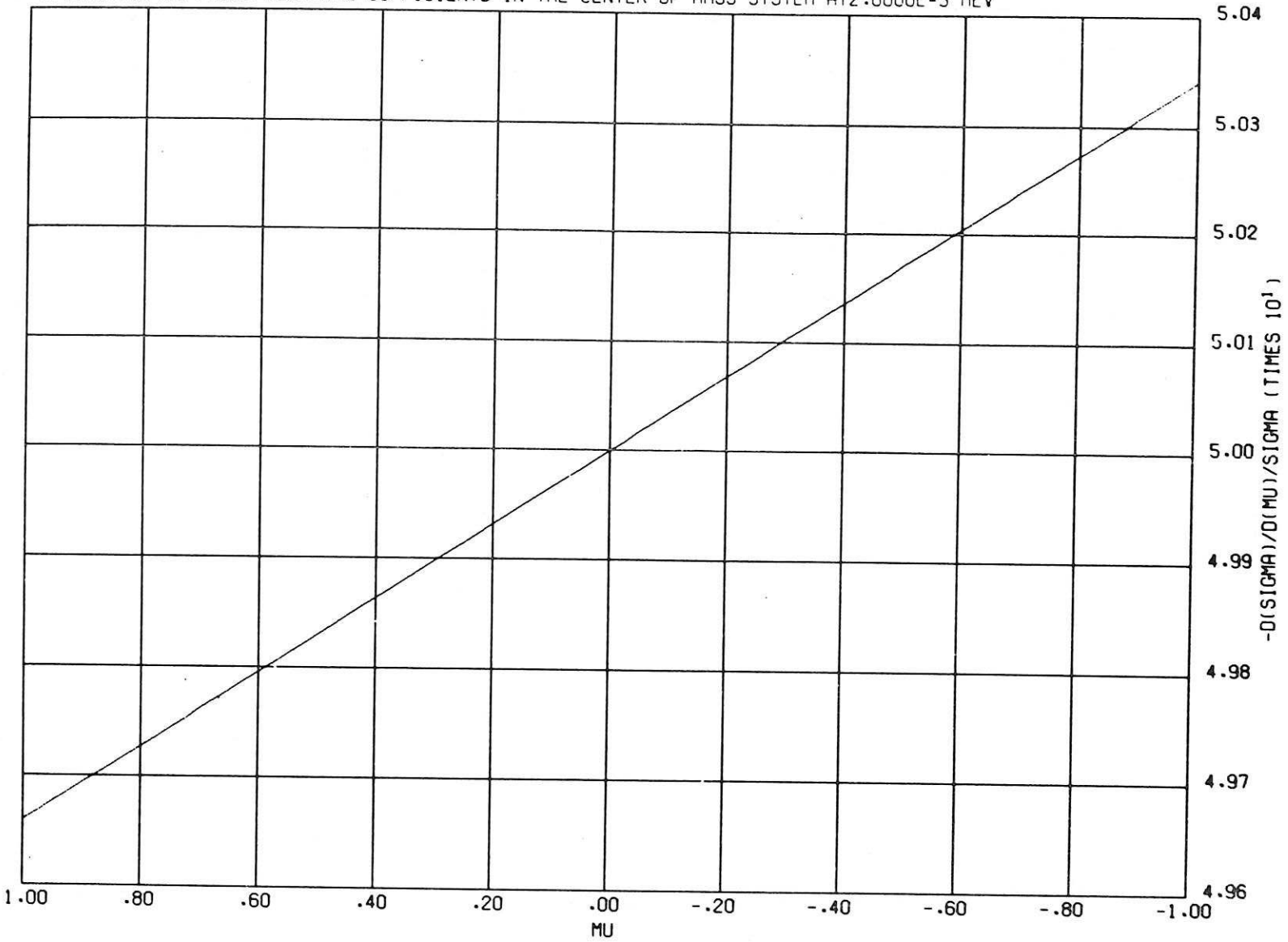
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.0000E-3 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

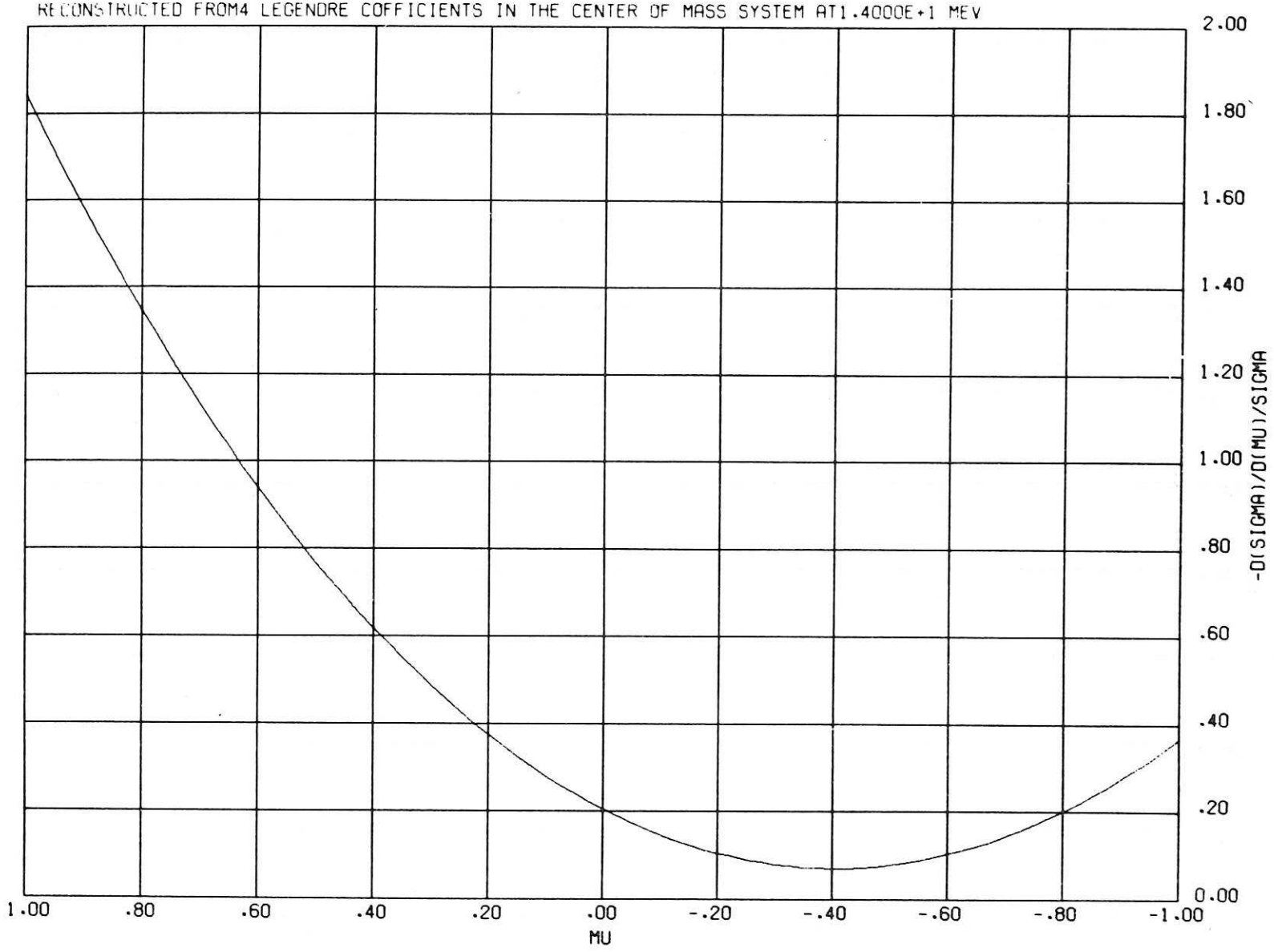
RECONSTRUCTED FROM2 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT2.0000E-3 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288 [illegible]

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

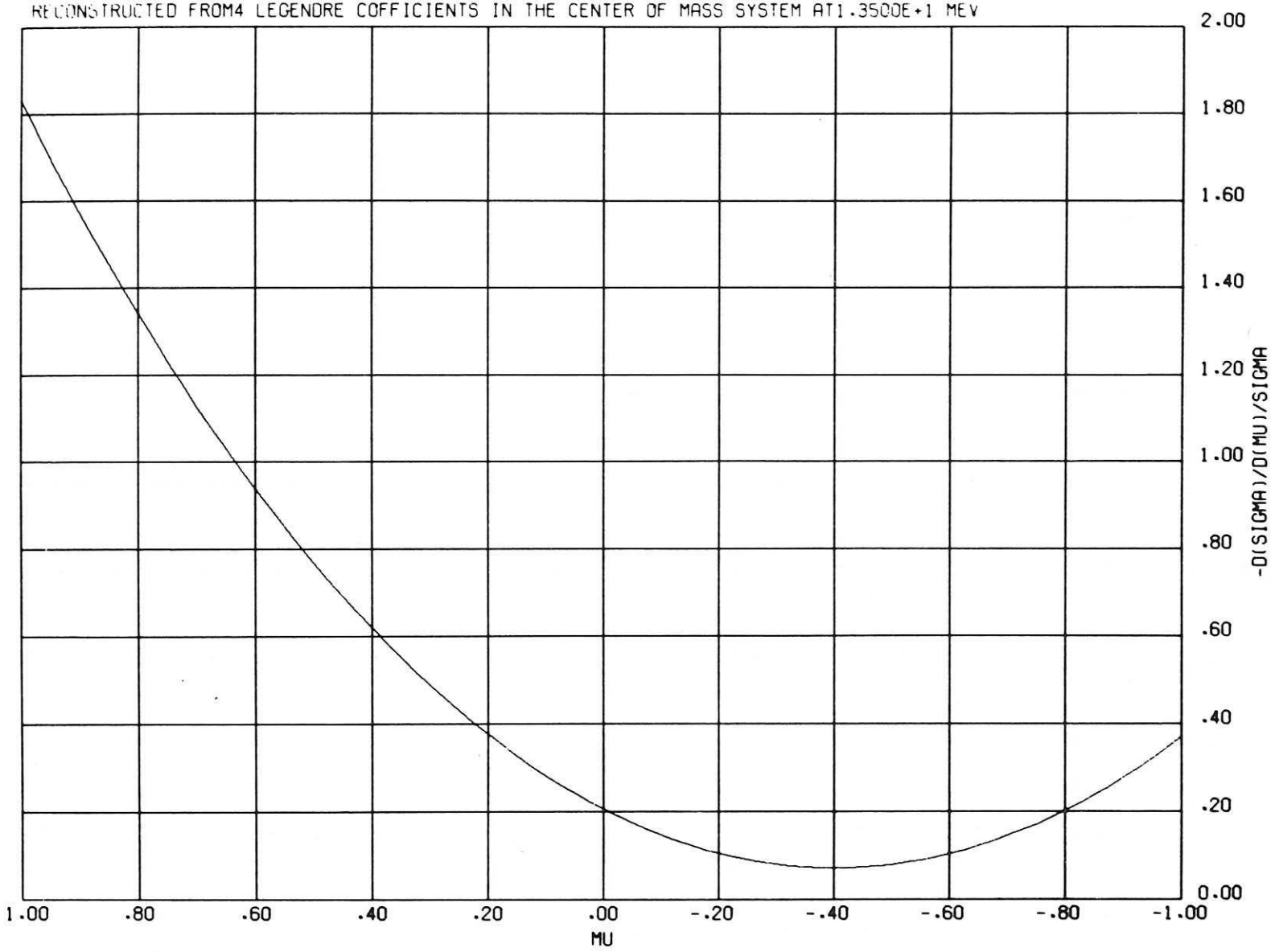
RECONSTRUCTED FROM4 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.4000E+1 MEV



NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288 1288

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

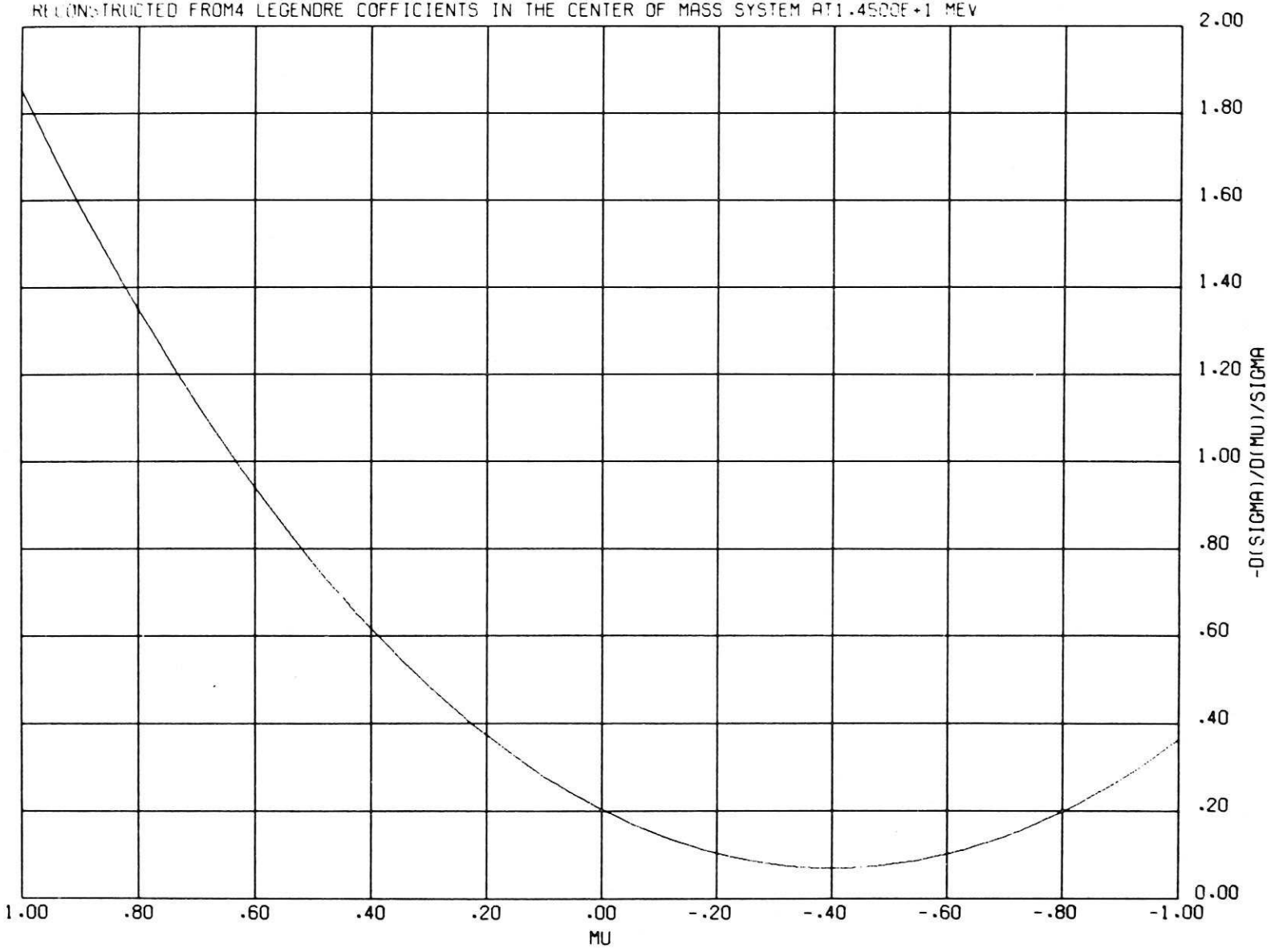
RECONSTRUCTED FROM4 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.3500E+1 MEV

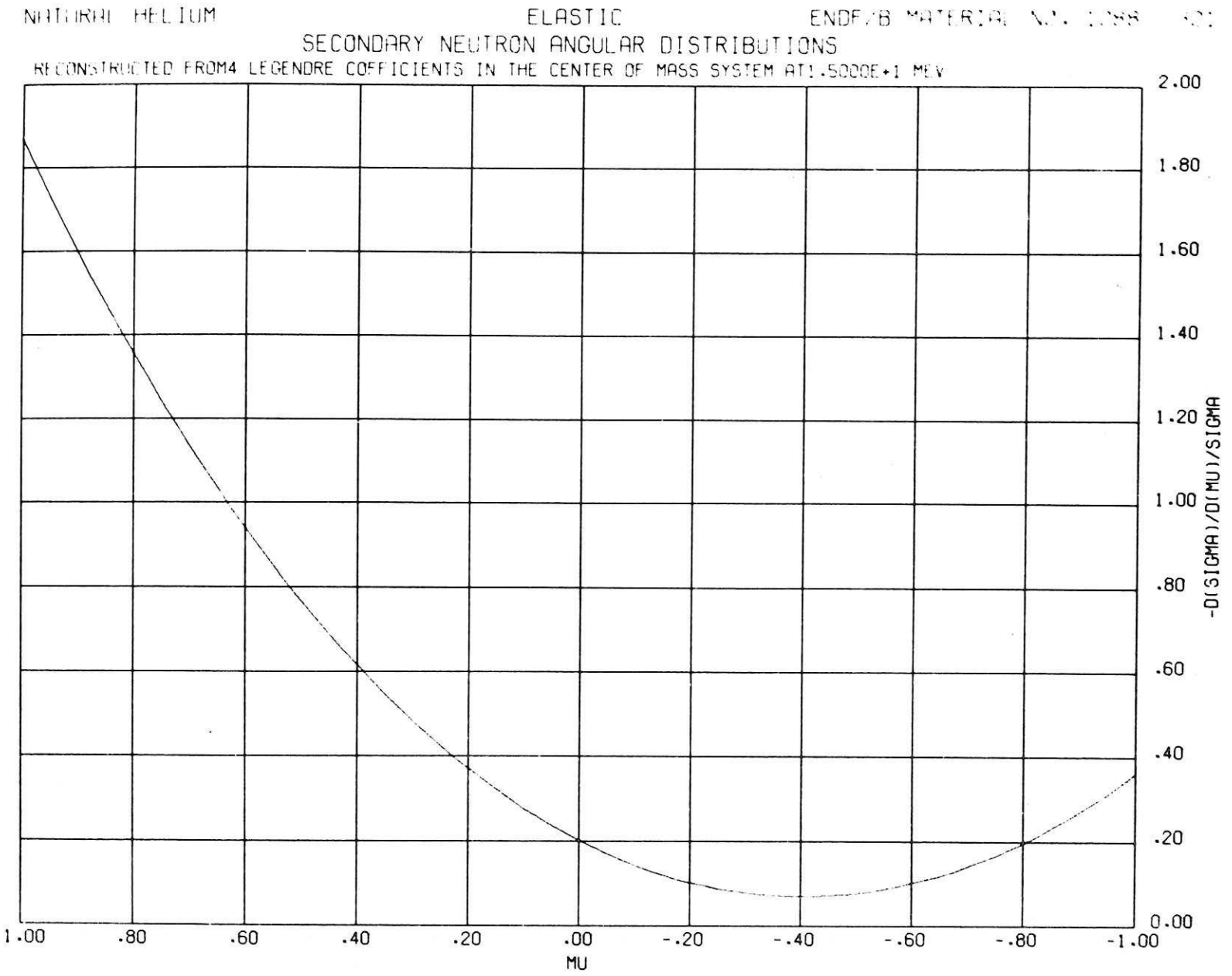


NATURAL HELIUM ELASTIC ENDF/B MATERIAL NO. 1288

SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED FROM4 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.4500E+1 MEV




NATURAL HELIUM
ELASTIC
ENDF/B MATERIAL
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS
RECONSTRUCTED FROM4 LEGENDRE COFFICIENTS IN THE CENTER OF MASS SYSTEM AT1.5000E+1 MEV
2.00
1.80
1.60
1.40
1.20
1.00
.80
.60
.40
.20
0.00
-D(SIGMA)/D(MU)/SIGMA
1.00
.80
.60
.40
.20
.00
-.20
-.40
-.60
-.80
-1.00
MU

Appendix N

BCD Card Image Formats

This appendix describes the data formats to be used when preparing magnetic tapes containing BCD card image records. The symbols and definitions used in describing BCD card image formats are the same as was used earlier in this report to describe binary record formats. The basic concepts of BCD card image formats was described in Section 5.3 of this report.

The following pages of this appendix contain the BCD card image formats for Files 1 through 7. The top of each page indicates the particular File and data format to be described. Four pages (A, B, C, and D) are used to describe each data type. When the appendix is opened to a particular data type, the user will see two pages. The left hand, Page A (the last page of the previous fold out sheet) will contain a review of the variable names and their definitions. The right hand, Page D(before unfolding) will contain the important formulae associated with the particular data type. When the right hand page is unfolded, this will expose pages B and C. Page B will give the BCD card image format. Page C will give an example of the particular BCD cards 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 in these cases, the cards of lesser importance have been omitted.

FILE 1

MT = 451 (Descriptive Information and Index)

LRP is a flag for resonance parameter data. If LRP=0 (no parameters), LRP=1 (parameters given in File 2).

LFI is a fissile material flag. If LFI=0 (no), LFI=1 (material is fissionable).

NXC is a count of cards in dictionary. Each section of this material is represented by a single card. This card contains the MF (File number), MT (reaction number), and NC (a count of the cards in the section).

LDD is a radioactive decay data flag. If LDD=0 (none), LDD=1 (decay data given in MT=453).

LFP is a fission product yield data flag. If LFP=0 (none), LFP=1 (data given in MT=454).

NWD is the number of cards used to describe the data set.

MF_n is the File number (MF) of the n^{th} section.

MT_n is the reaction type number (MT) of the n^{th} section.

NC_n is the number of cards in n^{th} section. This card count does not include SEND, FEND, or MEND cards.

BCD Card Image Format for File 1

MT = 451 (Descriptive Information and Index)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LRP	LFI	b*	NXC	HEAD
b	b	LDD	LFP	NWD	b	
Hollerith Information				1st Card		
				2nd Card		
--	--	--	--	--	--	
--	--	--	--	NWD^{th} Card		LIST
b	b	MF_1	MT_1	NC_1	b	CØNT
b	b	MF_2	MT_2	NC_2	b	CØNT
--	--	--	--	--	--	--
--	--	--	--	--	--	--
b	b	MF_{NXC}	MT_{NXC}	NC_{NXC}	b	CØNT
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 451 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
2,00000+ 3 3,96822+ 0          0           0           0          101088 1451    1
                               0           0          77            1088 1451    2
HELIUM    ANL         EVAL-JUN68 E,M,PENNINGTON (NATURAL ELEMENT) 1088 1451    3
ANL-7462 (OCT,1968)   DIST-JUN70 REV-JUN70                        1088 1451    4
*          *           *          *           *          *       * 1088 1451    5
DATA MODIFIED JUNE,1970 TO CONFORM TO ENDF/B-II FORMATS           1088 1451    6
*          *           *          *           *          *       * 1088 1451    7
HELIUM CROSS SECTIONS  MATERIAL 1088                              1088 1451    8
COMPILED BY ED PENNINGTON, ARGONNE NATIONAL LAB,, IN JUNE 1968,   1088 1451    9
NATURAL HELIUM CONSISTS OF 0,00013 PER CENT HE-3 AND 99,99987 PER1088 1451   10
CENT HE-4,                                                        1088 1451   11
BECAUSE OF THE LOW ABUNDANCE OF HE-3, ONLY ITS (N,P) CROSS        1088 1451   12
SECTION, WHICH IS VERY LARGE AT LOW ENERGIES, NEED BE CONSIDERED,1088 1451   13
ELASTIC SCATTERING IS THE ONLY POSSIBLE REACTION FOR NEUTRONS     1088 1451   14
INCIDENT ON HE-4 AT ENERGIES BELOW 15 MEV, THUS THE ELASTIC       1088 1451   15
SCATTERING CROSS SECTION AND VALUES OF MU BAR(LAB), XI, AND GAMMA1088 1451   16
ARE GIVEN IN FILE 3, AND ELASTIC SCATTERING LEGENDRE COEFFICIENTS1088 1451   17
ARE GIVEN IN FILE 4, PARAMETERS FOR A FREE GAS THERMAL SCATTERING1088 1451   18
LAW ARE IN FILE 7,                                                1088 1451   19
THE ELASTIC SCATTERING CROSS SECTION AND THE LEGENDRE EXPANSION   1088 1451   20
COEFFICIENTS WERE CALCULATED FROM S-,P-, AND D-WAVE PHASE SHIFTS  1088 1451   21
USING A FORTRAN PROGRAM WRITTEN FOR THE PURPOSE, THE PHASE SHIFTS1088 1451   22
WERE READ FROM SMOOTH CURVES BASED ON TABLE I OF REF,1, AT        1088 1451   23
ENERGIES BELOW THE 300 KEV, LOWER LIMIT OF TABLE I, EACH OF THE   1088 1451   24
TWO P-WAVE PHASE SHIFTS WAS OBTAINED BY ASSUMING A FUNCTIONAL     1088 1451   25
FORM BASED ON THE LOW ENERGY LIMIT FOR A SINGLE P-WAVE RESONANCE,1088 1451   26
WITH PARAMETERS DETERMINED FROM FITTING THE LOW ENERGY PHASE      1088 1451   27
SHIFTS OF TABLE I, THE S-WAVE PHASE SHIFT BELOW 300 KEV, WAS      1088 1451   28
--- --- --- --- --- --- --- --- --- --- --- --- --- --- --- ---

REFERENCES-                                                       1088 1451   69
1, G,R,SATCHLER ET AL, NUCLEAR PHYSICS A112,1-31,(1968),          1088 1451   70
2, R,GENIN ET AL, JOURNAL DE PHYSIQUE ET LE RADIUM 24,21-26,1963,1088 1451   71
3, H,AMSTER JOURNAL OF APPLIED PHYSICS 27,3,307,(1956),           1088 1451   72
4, H,AMSTER JOURNAL OF APPLIED PHYSICS 27,6,663,(1956),           1088 1451   73
5, H,AMSTER JOURNAL OF APPLIED PHYSICS 29,4,623-627,(1958),       1088 1451   74
6, R,F,BERLAND NAA-SR-11231,(1965),                               1088 1451   75
7, E,N,E,A, NEUTRON DATA COMPILATION CENTRE NEWSLETTER NO,6,1967,1088 1451   76
8, B,HOOP,JR, AND H,H,BARSCHALL NUCLEAR PHYSICS 83,65-79,(1966), 1088 1451   77
9, J,J,SCHMIDT KFK-120,PARTS II,III,(1962),PART I,(1966),         1088 1451   78
10,B,R,S,BUCKINGHAM ET AL AWRE O-28/60,(1961),                    1088 1451   79
                               1         451          89          1088 1451   80
                               2         151           4          1088 1451   81
                               3           1          51          1088 1451   82
                               3           2          51          1088 1451   83
                               3         103          51          1088 1451   84
                               3         251          39          1088 1451   85
                               3         252          39          1088 1451   86
                               3         253          39          1088 1451   87
                               4           2         237          1088 1451   88
                               7           4           4          1088 1451   89
                                                                  1088 1  0   90
```

FILE 1

MT = 451
Descriptive Information and Index

FILE 1

MT = 452 (Number of Neutrons per Fission, $\overline{\nu}$ (E)

LNU = 1 (Polynomial Representation Used)

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

LNU = 1, polynomial representation
LNU = 2, $\overline{\nu}$ (E) is tabulated

NC is the number of terms used in the polynomial expansion.

C_1, C_2, C_3, C_4 are the coefficients of the polynomial.

BCD Card Image Format for
File 1

MT = 452, Total number of neutrons per fission, $\bar{\nu}(E)$
LNU = 1, Polynomial representation used.

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	LNU=1	b	b	HEAD
b	b	b	b	NC	b	
C_1	C_2	--	--	C_{NC}		LIST
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 452 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9,22340+ 4 2,32029+ 2          1                      1043 1452   59
                                          2           1043 1452   60
2,37000+ 0 1,25000- 7                                 1043 1452   61
                                                      1043 1  0   62
```

FILE 1

MT = 452

Number of Neutrons per Fission

LNU = 1, polynomial representation of $\bar{\nu}$ (E)

$\bar{\nu}$ is the total (prompt plus delay) number of neutrons per fission.

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

FILE 1

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

LNU = 2 Tabulated Values of $\bar{\nu}$ (E)

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

NR is the number of interpolation ranges used.

NP is the total number of energy points used in the tabulation.

NBT(I), INT(I) is the interpolation scheme for $\bar{\nu}$(E).

$\bar{\nu}$ (E_i) is the average total (prompt plus delayed) number of neutrons per fission.

E_i is the energy (i^{th} point) of the neutrons causing fission.

BCD Card Image Format for File 1

MT = 452, Total number of neutrons per fission, $\overline{\nu}(E)$
LNU = 2, Tabulated values of $\overline{\nu}(E)$

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	LNU=2	b	b	HEAD
b	b	b	b	NR	NP	
NBT(1)	INT(1)	NBT(2)	INT(2)	--	--	
--	--	--	--	NBT(NR)	INT(NR)	
E1	$\overline{\nu}(E1)$	E2	$\overline{\nu}(E2)$	--	--	
--	--	--	--	E_{NP}	$\overline{\nu}(E_{NP})$	TAB1
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 452 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.4239 +04 236.999                             2                    1104 1452  163
                                                          1       121104 1452  164
        12          2                                               1104 1452  165
1.0    -05 2.880     2.53    -02 2.880    5.0    +05 2.945          1104 1452  166
1.0    +06 3.008     2.0     +06 3.143    3.0    +06 3.288          1104 1452  167
4.0    +06 3.442     6.0     +06 3.76     8.0    +06 4.084          1104 1452  168
1.0    +07 4.388     1.2     +07 4.660    1.5    +07 5.009          1104 1452  169
                                                                    1104 1  0  170
```

FILE 1

MT = 452

Number of Neutrons per Fission

LNU = 2, $\bar{\nu}$(E) is tabulated.

ν(E) is the average total (prompt plus delayed) number of neutrons per fission.

File 1

MT = 453 (Radioactive Decay Data)

NS is the number of excited states for which data will be given for the original nuclide (target nucleus).

LIS designate the state of the original nuclide (0 = ground state, etc.)

NE is the number of incident energy points at which branching ratios will be given.

NPR is the total number of product nuclide states that will be described.

ES(N) is the incident energy point (Nth point) at which branching ratios are given.

EREL is the total energy released by a specified decay mode (includes gamma rays and particles).

Q is the reaction Q value for a specified reaction.

LFS designates the state of the product nuclide (0 = ground state, etc.).

RTYP specifies the reaction types (floating point values of MT number. RTYP = 0.0 for spontaneous decay of the original nuclide.

DC is the decay constant (sec^{-1}) for the spontaneous decay of the original nuclide state to a particular product nuclide state.

BR(N) is the branching ratio at the Nth energy point for a specified reaction (fraction).

MT = 453 (Radioactive Decay Data)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	b	NS	b	HEAD
ZA	AWR	LIS	b	NE	NPR	(ground state)
ES(1)	ES(2)	ES(3)	---	---	ES(NE)	$LIST_1$
$EREL_1$	Q_1	LFS_1	b	NE + 3	b	
$RTYP_1$	ZAP_1	DC_1	BR(1)	BR(2)	BR(3)	
BR(4)	---	---	---	---	BR(NE)	$LIST_2$
$EREL_2$	Q_2	LFS_2	b	NE + 3	b	
$RTYP_2$	ZAP_2	DC_2	BR(1)	---	BR(NE)	$LIST_2$
---	---	---	---	---	---	
---	---	---	---	---	---	
$EREL_{NPR}$	Q_{NPR}	LFS_{NPR}	b	NE + 3	b	
$RTYP_{NPR}$	ZAP_{NPR}	DC_{NPR}	BR(1)	---	BR(NE)	$LIST_2$
	(structure is repeated for each original nuclide state until all NS states have been given. Start each state with the first LIST record)					

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 453 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
5.Ø1Ø   +Ø3 9.927Ø                                       1          1ØØ9 1453  122
5.Ø1Ø   +Ø3 9.927Ø                   Ø                   9         41ØØ9 1453  123
2.53    -Ø2 1.Ø         4.Ø        +Ø4 5.Ø      +Ø4 6.Ø        7.Ø    +Ø41ØØ9 1453  124
8.Ø     +Ø4 9.Ø    +Ø4 1.Ø         +Ø5                                  1ØØ9 1453  125
Ø.Ø         -4.3628+Ø6               Ø                  12          1ØØ9 1453  126
1Ø4.Ø       4.ØØ9  +Ø3 Ø.Ø           1.Ø        1.Ø         1.Ø     1ØØ9 1453  127
1.Ø         1.Ø        1.Ø           1.Ø        1.Ø         1.Ø     1ØØ9 1453  128
Ø.Ø         2.295Ø +Ø5               Ø                  12          12Ø9 1453  129
1Ø5.Ø       4.ØØ8  +Ø3 Ø.Ø           1.Ø        1.Ø         1.Ø     1ØØ9 1453  13Ø
1.Ø         1.Ø        1.Ø           1.Ø        1.Ø         1.Ø     1ØØ9 1453  131
Ø.Ø         2.7916 +Ø6               Ø                  12          1ØØ9 1453  132
1Ø7.Ø       3.ØØ7  +Ø3 Ø.Ø           Ø.Ø63Ø8    Ø.Ø63       Ø.Ø63   1ØØ9 1453  133
Ø.Ø64       Ø.Ø65      Ø.Ø66         Ø.Ø68      Ø.Ø7Ø       Ø.Ø72   1ØØ9 1453  134
Ø.Ø         2.3223 +Ø6               1                  12          1ØØ9 1453  135
1Ø7.Ø       3.ØØ7  +Ø3 Ø.Ø           Ø.93692    Ø.937       Ø.937   1ØØ9 1453  136
Ø.936       Ø.935      Ø.934         Ø.932      Ø.93Ø       Ø.928   1ØØ9 1453  137
                                                                    1ØØ9 1  Ø  138
```

FILE 1

MT = 453

Radioactive Decay Data

For a specified original nuclide state (LIS) and reaction type(RTYP) the branching ratios are

$$\sum \text{BR(N)} = 1.0$$

at each incident energy point, N.

FILE 1

MT = 454 (Fission Product Yield Data)

N1 is 3*NFP

NFP is the number of fission products to be specified at the i^{th} incident neutron energy point. (Sets of three parameters: ZAFP, FPS, YLD)

E_i is the incident neutron energy causing fission.

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 yields given).
LE > 0, means that (LE+1) sets of fission product yields are given.

I_i is the interpolation scheme to be used between E_{i-1} and E_i energy points.

ZAFP is the (Z,A) identifier for a particular fission product.

YLD is the fractional yield for a particular fission product.

FPS is the fission product nuclide state indicator (0.0 = ground state, 1.0 = 1st excited state, etc.).

BCD Card Image Format for
File 1

MT = 454 (Fission Product Yield Data)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LE + 1	b*	b	b	HEAD
E_1	b	LE	b	N1	NFP	(N1 = 3 * NFP)
$ZAFP_1$	FPS_1	YLD_1	$ZAFP_2$	FPS_2	YLD_2	
--	--	--	$ZAFP_{NFP}$	FPS_{NFP}	YLD_{NFP}	LIST
E_2	b	I_2	b	N1	NFP	
$ZAFP_1$	FPS_1	YLD_2	--	--	--	
--	--	--	$ZAFP_{NFP}$	FPS_{NFP}	YLD_{NFP}	LIST
E_N	b	I_N	b	N1	NFP	(N = LE + 1)
$ZAFP_1$	FPS_1	YLD_1	---	--	--	
--	--	--	$ZAFP_{NFP}$	FPS_{NFP}	YLD_{NFP}	LIST

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT= 454 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9,2235 +Ø4 233,Ø25                  1                                1102 1454  152
1,Ø    -Ø5                          Ø                  732        2441102 1454  153
3,ØØ72 +Ø4 Ø,Ø          1,596  -Ø7 3,ØØ73 +Ø4 Ø,Ø          1,Ø969 -Ø61102 1454  154
3,1Ø74 +Ø4 Ø,Ø          3,49Ø  -Ø6 3,1Ø75 +Ø4 Ø,Ø          1,1966 -Ø51102 1454  155
3,1Ø76 +Ø4 Ø,Ø          3,49Ø  -Ø6 3,1Ø77 +Ø4 Ø,Ø          4,7863 -Ø51102 1454  156
3,2Ø75 +Ø4 Ø,Ø          Ø,Ø        3,2Ø75 +Ø4 1,Ø          Ø,Ø        1102 1454  157
3,2Ø76 +Ø4 Ø,Ø          Ø,Ø        3,2Ø77 +Ø4 Ø,Ø          2,4Ø   -Ø51102 1454  158
3,2Ø77 +Ø4 1,Ø          1,Ø9   -Ø5 3,2Ø78 +Ø4 Ø,Ø          1,9945 -Ø41102 1454  159
-  -  -     -  -  -     -  -   -   -  -  -    -  -  -      -  -   -     -  -  -  -
6,516Ø +Ø4 Ø,Ø          2,991  -Ø8 6,5161 +Ø4 Ø,Ø          3,989  -Ø81102 1454  275
                                                                      1102 1  Ø  276
```

FILE 1

MT = 454

Fission Product Yield Data

At each incident energy point

$$\sum_{i=1}^{NFP} YLD_i \approx 2.000$$

FILE 1

MT = 455 (Delayed Neutrons from Fission)

LND = 1 (Polynomial representation used)

LND is a test that indicates which representation used:
LND = 1 (polynomial expansion)
LND = 2 (tabulated values of $\overline{\nu}_d$)

NNF is the number of precursor families given.

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

NCD is the number of terms in the polynomial expansion.

CD_1, CD_2, CD_3, CD_4 are the coefficients for the polynomial.

BCD Card Image Format for
File 1

MT = 455 (Delayed neutrons from fission, $\bar{\nu}_d(E)$)
LND = 1, Polynomial expansion used

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	LND=1	b	b	HEAD
b	b	b	b	NNF	b	LIST
λ_1	λ_2	--	--	--	λ_{NNF}	
b	b	b	b	NCD	b	LIST
CD_1	CD_2	--	--	--	CD_{NCD}	
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 455 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.2235 +Ø4 233.Ø25                        1                     11Ø2 1455  1Ø1
                                                     7          11Ø2 1455  1Ø2
1.2717 -Ø2 2.84Ø6 -Ø2 4.2521 -Ø2 1.5332 -Ø1 3.5363 -Ø1 1.5Ø67   11Ø2 1455  1Ø3
3.9833                                                          11Ø2 1455  1Ø4
                                                     2          11Ø2 1455  1Ø5
Ø.Ø158      8.1    -1Ø                                          11Ø2 1455  1Ø6
                                                                11Ø2 1  Ø  1Ø7
```

FILE 1

MT = 455

Delayed Neutron Data

LND = 1, polynomial prepresentation of $\bar{\nu}_d(E)$

$$\bar{\nu}_d(E) = \sum_{m=1}^{NCD} CD_m * E^{(m-1)}$$

The yield of the i^{th} precursor family is

$$\bar{\nu}_i(E) = \bar{\nu}_d(E) * p_i(E)$$

where $p_i(E)$ is the fractional yield given in File 5.
The time dependence of $\bar{\nu}_i(E)$ is

$$\bar{\nu}_i(E,t) = \bar{\nu}_i(E) e^{-\lambda_i t}$$

FILE 1

MT = 455 (Delayed Neutrons from Fission, $\bar{\nu}_d$)

LND = 2, tabulated values of $\bar{\nu}_d(E)$

LND is a test that indicates which representation used:
LND = 1 (polynomial expansion)
LND = 2 (tabulated values of $\bar{\nu}_d$)

NNF is the number of precursor families given.

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

NBT(I), INT(I) is the interpolation scheme for $\bar{\nu}_d$.

$\bar{\nu}_d(E_i)$ is the average total number of delayed neutrons that are emitted per fission event (due to neutrons of energy, E_i).

MT = 455, Delayed neutrons from fissions, $\bar{\nu}_d(E)$
LND = 2, Tabulated values of $\bar{\nu}_d(E)$

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	LND=2	b	b	HEAD
b	b	b	b	NNF	b	LIST
λ_1	λ_2	λ_3	--	--	λ_{NNF}	
b	b	b	b	NR	NP	TAB1
NBT(1)	INT(1)	NBT(2)	INT(2)	--	--	
--	--	--	--	NBT(NR)	INT(NR)	
E_1	$\nu_d(E_1)$	E_2	$\nu_d(E_2)$	--	--	
--	--	--	--	E_{NP}	$\nu_d(E_{NP})$	
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 1
Field 9 (Cols 73-75) = MT = 455 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.2235 +04 233.025                                 2                  1102 1455  101
                                                               7      1102 1455  102
1.2717 -02 2.8406 -02 4.2521 -02 1.5332 -01 3.5363 -01 1.5067         1102 1455  103
3.9833                                                                1102 1455  104
                                                               1     31102 1455  105
          3          2                                                1102 1455  106
1.0       -05 0.0158     1.5       +06 0.0165     1.5       +07 0.025 1102 1455  107
                                                                      1102 1  0  108
```

FILE 1

MT = 455

Delayed Neutron Data

LND = 2, $\bar{\nu}_d(E)$ is tabulated.

FILE 2: Resonance Parameter Data

LRP = 0 (only effective scattering radius given)

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

ABN is the abundance (weight fraction) of an isotope.

EL is the lower limit for the energy range.

EH is the upper limit for the energy range.

SPI is the nuclear spin of the target nucleus, I.

AP is the spin-independent effective scattering radius (in units of 10^{-12}cm).

BCD Card Image Format for
File 2

Resonance Parameter Data

LRP = 0 (No resolved and/ or unresolved parameters given, only effective scattering radius given)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	b	NIS=1	b	HEAD
ZAI	ABN	b	LFW=0	NER=1	b	CØNT
EL	EH	LRU=0	LRF=0	b	b	CØNT
SPI	AP	b	b	NLS=0	b	CØNT
b	b	b	b	b	b	SEND
b	b	b	b	b	b	FEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2 (except FEND card)
Field 9 (Cols 73-75) = MT= 151 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
2.Ø     +03 3.96822                             1          1Ø88 2151   92
2.ØØ4   +03 1.Ø                         Ø       1          1Ø88 2151   93
1.Ø     -Ø5 1.Ø     +Ø5         Ø       Ø                  1Ø88 2151   94
Ø.Ø         Ø.2414                              Ø          1Ø88 2151   95
                                                           1Ø88 2  Ø   96
                                                           1Ø88 Ø  Ø   97
```

FILE 2

Resonance Parameter Data

Special Case
LRP = 0 (In File 1, MT = 451)

Only data given is the effective scattering radius. The s-wave potential scattering cross section is

$$\sigma_p = \frac{4\pi}{k^2} \sin^2\varphi$$

where

$$k = 2.196771 \frac{AWRI}{AWRI + 1.0} \times 10^{-3}\sqrt{E}$$

E in electron volts.

and

$$\varphi = k*AP$$

FILE 2: Resonance Parameter Data
(General Structure)

NIS is the number of isotopes.

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

ABN is the abundance (weight fractions) of an isotope.

LFW is a flag that indicates whether average fission widths are given in the unresolved resonance region for this isotope:
LFW = 0, ave. fission widths not given.
LFW = 1, ave. fission widths given.

NER is the number of energy ranges for this isotope.

EL and EH are the lower and upper limits of the energy range.

LRU is a test for resolved or unresolved resonance parameters:
LRU = 1, resolved parameters given for this energy range.
LRU = 2, unresolved parameters given.

LRF is a test for the type of resonance formula used for this energy range:
if LRU = 1 (resolved parameters), then:
LRF = 1 SLBW parameters
LRF = 2 MLBW parameters
LRF = 3 Reich-Moore parameters
LRF = 4 Adler-Adler parameters
if LRF = 2 (unresolved parameters), then:
LRF = 1, only average fission widths are energy dependent.
LRF = 2, all parameters are energy dependent.

BCD Card Image Format for
File 2

Resonance Parameter Data
(General Structure)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	b	NIS	b	HEAD
ZAI	ABN	b	LFW	NER	b	CØNT (isotope)
EL	EH	LRU	LRF	b	b	CØNT (range)
<Subsection for the first energy range for the first isotope>						
EL	EH	LRU	LRF	b	b	CØNT (range)
<Subsection for the second energy range for the first isotope>						
--	--	--	---	--	--	
EL	EH	LRU	LRF	b	b	CØNT (range)
<Subsection for the last energy range for the first isotope>						
ZAI	ABN	b	LFW	NER	b	CØNT (isotope)
EL	EH	LRU	LRF	b	b	CØNT (range)
<Subsection for the first energy range for the second isotope>						
--	--	--	--	--	--	
--	--	--	--	--	--	
--	--	--	--	--	--	
EL	EH	LRU	LRF	b	b	CØNT (range)
<Subsection for the last energy range for the last isotope>						
b	b	b	b	b	b	SEND
b	b	b	b	b	b	FEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2 (except FEND card)
Field 9 (Cols 73-75) = MT= 151 (except SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
                                                                1043 0  0    69
9.22340+ 4 2.32029+ 2                                   1       1043 2151    70
9.22340+ 4 1.00000+ 0                         0         2       1043 2151    71
1.0    -05 3.78000+ 2           1             1                 1043 2151    72

3.78000+ 2 1.00000+ 3           2             1                 1043 2151    96

                                                                1043 2  0   103
                                                                1043 0  0   104
```

FILE 2

Resonance Parameter Data

General Structure

$$\sum_{i=1}^{NIS} ABN_i = 1.000$$

FILE 2: Resonance Parameters (Structure of a Subsection)

LRU= 1 (resolved parameters)
LRF=1 or 2 (single-level or multilevel Breit-Wigner resonance parameters)

SPI is the target spin, I.

AP is the spin-dependent (spin up) effective scattering radius (in units of 10^{-12} cm). AP is also given for spin-independence.

AM is the spin-dependent (spin down) effective scattering radius.

NLS is the number of ℓ-states given (a set of parameters are given for each ℓ-state).

L is the value of the ℓ-state.

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-state.

ER, AJ, GT, GN, GG, and GF are the resonance energy, spin of the resonance (J), total width, neutron width, radiation width, and fission width respectively. All widths evaluated at the resonance energy.

BCD Card Image Format for File 2

Resonance Parameters (Structure of a Subsection)

LRU = 1 (resolved parameters)
LRF = 1 or 2 (<u>Single-level</u> or <u>multilevel</u> Breit-Wigner resonance parameters)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	AP	b*	b	NLS	b	CØNT
AWRI	AM	L	b	6*NRS	NRS	
ER_1	AJ_1	GT_1	GN_1	GG_1	GF_1	(first
ER_2	AJ_2	GT_2	GN_2	GG_2	GF_2	ℓ-state)
--	--	--	--	--	--	
ER_{NRS}	AJ_{NRS}	GT_{NRS}	GN_{NRS}	GG_{NRS}	GF_{NRS}	LIST
AWRI	AM	L	b	6*NRS	NRS	
ER_1	AJ_1	GT_1	GN_1	GG_1	GF_1	(second
--	--	--	--	--	--	ℓ-state)
ER_{NRS}	AJ_{NRS}	GT_{NRS}	GN_{NRS}	GG_{NRS}	GF_{NRS}	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
--	--	--	--	--	--	
AWRI	AM	L	b	6*NRS	NRS	(last
ER_1	AJ_1	GT_1	GN_1	GG_1	GF_1	ℓ-state)
--	--	--	--	--	--	
ER_{NRS}	AJ_{NRS}	GT_{NRS}	GN_{NRS}	GG_{NRS}	GF_{NRS}	LIST

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT = 151
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
Ø,ØØØØØ+ Ø 8,93ØØØ- 1                                  1            1Ø43 2151    73
232,Ø29    Ø,ØØØØØ+ Ø           Ø                    126          211Ø43 2151    74
-1,6777    5,ØØØØØ- 1 Ø,Ø2822    Ø,ØØ322    2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    75
5,19ØØØ+ Ø 5,ØØØØØ- 1 2,9122Ø- 2 4,1ØØØØ- 3 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    76
3,14ØØØ+ 1 5,ØØØØØ- 1 3,27ØØØ- 2 7,7ØØØØ- 3 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    77
4,64ØØØ+ 1 5,ØØØØØ- 1 2,5Ø7ØØ- 2 7,ØØØØØ- 5 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    78
4,94ØØØ+ 1 5,ØØØØØ- 1 3,6ØØØØ- 2 1,1ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    79
7,83ØØØ+ 1 5,ØØØØØ- 1 3,14ØØØ- 2 6,4ØØØØ- 3 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    8Ø
8,87ØØØ+ 1 5,ØØØØØ- 1 2,59ØØØ- 2 9,ØØØØØ- 4 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    81
9,53ØØØ+ 1 5,ØØØØØ- 1 5,3ØØØØ- 2 2,8ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    82
1,069ØØ+ 2 5,ØØØØØ- 1 2,81ØØØ- 2 3,1ØØØØ- 3 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    83
1,121ØØ+ 2 5,ØØØØØ- 1 3,8ØØØØ- 2 1,3ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    84
1,329ØØ+ 2 5,ØØØØØ- 1 3,9ØØØØ- 2 1,4ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    85
1,459ØØ+ 2 5,ØØØØØ- 1 4,2ØØØØ- 2 1,7ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    86
1,54ØØØ+ 2 5,ØØØØØ- 1 4,4ØØØØ- 2 1,9ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    87
1,79ØØØ+ 2 5,ØØØØØ- 1 9,5ØØØØ- 2 7,ØØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    88
1,84ØØØ+ 2 5,ØØØØØ- 1 4,5ØØØØ- 2 2,ØØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    89
1,91ØØØ+ 2 5,ØØØØØ- 1 1,35ØØØ- 1 1,1ØØØØ- 1 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    9Ø
2,74ØØØ+ 2 5,ØØØØØ- 1 5,1ØØØØ- 2 2,6ØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    91
2,95ØØØ+ 2 5,ØØØØØ- 1 1,Ø5ØØØ- 1 8,ØØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    92
3,19ØØØ+ 2 5,ØØØØØ- 1 1,35ØØØ- 1 1,1ØØØØ- 1 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    93
3,57ØØØ+ 2 5,ØØØØØ- 1 5,5ØØØØ- 2 3,ØØØØØ- 2 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    94
3,69ØØØ+ 2 5,ØØØØØ- 1 2,45ØØØ- 1 2,2ØØØØ- 1 2,5ØØØØ- 2 Ø,ØØØØØ+ Ø1Ø43 2151    95
```

FILE 2

Resonance Parameter Data

LRU = 1, resolved parameters
LRF = 1 or 2, single or multilevel Breit-Wigner parameters

$$g_i = \frac{2*AJ_i + 1.0}{2(2*SPI + 1.0)}$$

$$GT_i = GN_i + GG_i + GF_i$$

FILE 2: Resonance Parameters (Structure of a Subsection)

LRU = 1 (resolved parameters)
LRF = 3 (Reich-Moore multilevel parameters)

SPI is the target spin, I.

AP and AM are the spin-dependent(spin up and spin down, respectively) effective scattering radii (10^{-12}cm).

NLS is the number of ℓ-states.

L is the value of the ℓ-state.

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

NRS is the number of resonances for a particular ℓ-state.

ER, AJ, GN, GG, GFA, and GFB are the resonance energy, resonance spin (J), neutron width, radiation width, the first and second fission widths, respectively.

BCD Card Image Format for
File 2

Resonance Parameters (Structure of a Subsection)

LRU = 1 (resolved parameters)
LRF = 3 (Reich-Moore multilevel parameters)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	AP	b*	b	NLS	b	CØNT
AWRI	AM	L	b	6*NRS	NRS	
ER_1	AJ_1	GN_1	GG_1	GFA_1	GFB_1	(first
ER_2	AJ_2	GN_2	GG_2	GFA_2	GFB_2	ℓ-state)
--	--	--	--	--	--	
ER_{NRS}	AJ_{NRS}	GN_{NRS}	GG_{NRS}	GFA_{NRS}	GFB_{NRS}	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
--	--	--	--	--	--	
AWRI	AM	L	b	6*NRS	NRS	
ER_1	AJ_1	GN_1	GG_1	GFA_1	GFB_1	(last
ER_2	AJ_2	GN_2	GG_2	GFA_2	GFB_2	ℓ-state)
--	--	--	--	--	--	
ER_{NRS}	AJ_{NRS}	GN_{NRS}	GG_{NRS}	GFA_{NRS}	GFB_{NRS}	LIST

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT= 151
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
1.ØØØØ +Ø2 1.786Ø6+Ø1                     1                       1002 7 4  21
                                                   12            11002 7 4  22
4.Ø72  +Ø1 7,9Ø514+Ø1 1,ØØ       2,ØØ                             1002 7 4  23
1.Ø        3,76       1,6Ø   +Ø1                                  1002 7 4  24
                                                   1             801002 7 4  25
       8Ø         4                                               1002 7 4  26
2.96   +Ø2 Ø,Ø                   Ø                 1             801002 7 4  27
       8Ø         4                                               1002 7 4  28
5.Ø4Ø6 -Ø2 5,28Ø3     1,ØØ812-Ø1 3,6841     1.51218-Ø1 2,9685Ø    1002 7 4  29
2.Ø1624-Ø1 2,5371     2,5203Ø-Ø1 2,2398     3.Ø2436-Ø1 2,Ø182Ø    1002 7 4  3Ø
 -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -
2.93141+Ø2 1.Ø    -33 3.21288+Ø2 1,Ø    -33                       1002 7 4  55
2.96   +Ø2 8,Ø6495-Ø2            Ø                 1             801002 7 4  56
       8Ø         4                                               1002 7 4  57
5.Ø4Ø6 -Ø2 2,9688Ø    1.ØØ812-Ø1 2,7628     1.51218-Ø1 2,45Ø5Ø    1002 7 4  58
 -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -
 -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -  -
```

FILE 2

Resonance Parameter Data

LRU = 1, resolved parameters
LRF = 3, Reich-Moore multilevel parameters

$$g_i = \frac{2*AJ_i + 1.0}{2(2*SPI + 1.0)}$$

$$\Gamma_i = GN_i + GG_i + |GFA_i| + |GFB_i|$$

FILE 2: Resonance Parameters (Structure of a Subsection)

LRU = 1 (resolved parameters)
LRF = 4 (Adler-Adler multilevel parameters).

SPI is the target spin, I.

AP and AM are the spin dependent (spin-up and spin-down, respectively) effective scattering radii (10^{-12}cm).

NLS is the number of ℓ-states given.

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 = 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 number of sets of background constants given:
NX = 2, constants for total and radiative capture cross sections
= 3, constants for total, capture and fission cross sections

L is the value of the ℓ-state.

AT_1, AT_2, AT_3, AT_4, BT_1, BT_2 are the background constants for the total cross section. There is a similar set, AF_i, BF_i for fission and AC_i, BC_i for radiative capture cross sections.

NJS is the number of sets of parameters (each resonance in a set has the same ℓ- and J-state).

AJ is the value of the resonance spin, (J).

NLJ is the number of resonances for each set.

DET_n, DEF_n, DEC_n are the resonance energies (for the n^{th} level) for the total, fission, and capture cross sections, respectively.

GRT_n and GIT_n are the symmetrical and asymmetrical cross section parameters, respectively (for the n^{th} level).

GRF_n and GIF_n are the symmetrical and asymmetrical fission parameters, respectively (for the n^{th} level).

GRC_n and GIC_n are the symmetrical and asymmetrical capture parameters, respectively (for the n^{th} level).

* Reserved for use in the ENDF/A Library only.

BCD Card Image Format for File 2

Resonance Parameters (Structure of a Subsection)

LRU = 1 (resolved parameters)
LRF = 4 (Adler-Adler multilevel parameters)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	AP	b*	b	NLS	b	CØNT
AWRI	b	LI	b	6*NX	NX	
AT_1	AT_2	AT_3	AT_4	BT_1	BT_2	
AF_1	AF_2	AF_3	AF_4	BF_1	BF_2	**
AC_1	AC_2	AC_3	AC_4	BC_1	BC_2	LIST
b	b	L	b	NJS	b	CØNT(ℓ)
AJ	AM	b	b	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	First ℓ-state
DET_2	DWT_2	GRT_2	GIT_2	DEF_2	DWF_2	
GRF_2	GIF_2	DEC_2	DWC_2	GRC_2	GIC_2	First J-state
--	--	--	--	--	--	
DET_{NLJ}	DWT_{NLJ}	GRT_{NLJ}	GIT_{NLJ}	DEF_{NLJ}	DWF_{NLJ}	
GRF_{NLJ}	GIF_{NLJ}	DEC_{NLJ}	DWC_{NLJ}	GRC_{NLJ}	GIC_{NLJ}	LIST
AJ	AM	b	b	12*NLJ	NLJ	
DET_2	DWT_1	GRT_2	GIT_1	DEF_1	DWF_1	First ℓ-state, Second J-state
--	--	--	--	--	--	
--	--	--	--	--	--	
GRF_{NLJ}	GIF_{NLJ}	DEC_{NLJ}	DWC_{NLJ}	GRC_{NLJ}	GIC_{NLJ}	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
AJ	AM	b	b	12*NLJ	NLJ	
DET_1	DWT_1	GRT_1	GIT_1	DEF_1	DWF_1	First ℓ-state, Second J-state
--	--	--	--	--	--	
--	--	--	--	--	--	
GRF_{NLJ}	GIF_{NLJ}	DEC_{NLJ}	DWC_{NLJ}	GRC_{NLJ}	GIC_{NLJ}	LIST

If NLS>1, repeat structure starting with CONT-ℓ card.

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT= 151
Field 10 (Cols 76-80) = Card sequence number

*b = blank
**This card is not present if NX=2.

3,5	Ø,8944				1	1102 2151	1Ø1
232,9812	Ø,Ø	Ø			6Ø	1Ø1102 2151	1Ø2
11,67	3,Ø	5,8Ø7ØE-Ø4	38,ØE-Ø3	4,ØE-Ø3	1Ø,ØE-031102	2151	1Ø3
12,39	4,Ø	1,3Ø24E-Ø3	45,ØE-Ø3	23,ØE-Ø3	Ø,ØE-031102	2151	1Ø4
12,9Ø	3,Ø	3,951ØE-Ø5	43,ØE-Ø3	4Ø,ØE-Ø3	-1Ø,ØE-031102	2151	1Ø5
13,34	4,Ø	5,844ØE-Ø5	4Ø,ØE-Ø3	1Ø,ØE-Ø3	1ØØ,ØE-031102	2151	1Ø6
13,7Ø	3,Ø	7,Ø32ØE-Ø5	4Ø,ØE-Ø3	7Ø,ØE-Ø3	-1ØØ,ØE-031102	2151	1Ø7
14,ØØ	4,Ø	2,993ØE-Ø4	4Ø,ØE-Ø3	7Ø,ØE-Ø3	1Ø,ØE-031102	2151	1Ø8
14,53	3,Ø	1,296ØE-Ø4	29,ØE-Ø3	23,ØE-Ø3	-3Ø,ØE-031102	2151	1Ø9
15,45	4,Ø	2,516ØE-Ø4	55,ØE-Ø3	45,ØE-Ø3	15Ø,ØE-031102	2151	11Ø
16,1Ø	3,Ø	3,691ØE-Ø4	4Ø,ØE-Ø3	16,ØE-Ø3	3ØØ,ØE-031102	2151	111
16,67	4,Ø	2,776ØE-Ø4	64,ØE-Ø3	76,ØE-Ø3	-2ØØ,ØE-031102	2151	112

FILE 2

Resonance Parameter Data

LRU = 1, resolved parameters
LRF = 4, Adler-Adler multilevel parameters

Background terms

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

$$C = \pi \lambda^2 = \pi/k^2 \text{ and } k = 2.196771 \times 10^{-3} \left(\frac{AWRI}{AWRI + 1.0}\right)\sqrt{E(eV)}$$

There are similar terms for fission and capture.

FILE 2: Resonance Parameter (Structure of a Subsection)

LRU = 2 (unresolved parameters)
LFW = 0 (fission widths not given)
LRF = 1 (all parameters are energy-independent)

SPI is the target spin, I.

A is the spin-independent effective scattering radius (10^{-12}cm).

NLS is the number of ℓ -states given.

AWRI is the ratio of the mass of an isotope to that of the neutron.

L is the value of the ℓ -state.

NJS is the number of J-states given for a particular ℓ -state.

AJ is the floating point value of the J-state.

AMUN is the number of degrees of freedom used in the neutron width distribution.

D_j, GNO_j, and GG_j are the mean level spacing, the average reduced neutron width, and radiation width for the j^{th} spin state.

BCD Card Image Format for File 2

Resonance Parameters (Structure of a Subsection)

LRU = 2 (unresolved parameters)
LFW = 0 (fission widths not given)
LRF = 1 (all parameters are energy-independent)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	A	b*	b	NLS	b	CØNT
AWRI	b	L	b	6*NJS	NJS	
D_1	AJ_1	$AMUN_1$	GNO_1	GG_1	b	(first ℓ-state)
D_2	AJ_2	$AMUN_2$	GNO_2	GG_2	b	
--	--	--	--	--	--	
D_{NJS}	AJ_{NJS}	$AMUN_{NJS}$	GNO_{NJS}	GG_{NJS}	b	LIST
AWRI	b	L	b	6*NJS	NJS	(second ℓ-state)
D_1	AJ_1	$AMUN_1$	GNO_2	GG_1	b	
--	--	--	--	--	--	
D_{NJS}	AJ_{NJS}	$AMUN_{NJS}$	GNO_{NJS}	GG_{NJS}	b	LIST
--	--	--	--	--	--	(third ℓ-state, if NLS=3)

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT = 151
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
0,00000+ 0 8,93000- 1                                    2          1043 2151    97
232,029    0,00000+ 0          0                         6         11043 2151    98
1,60000+ 1 5,00000- 1 1,00000+ 0 2,16000- 3 2,50000- 2 0,00000+ 01043 2151    99
232,029    0,00000+ 0          1                        12         21043 2151   100
1,60000+ 1 5,00000- 1 1,00000+ 0 1,60000- 3 2,50000- 2 0,00000+ 01043 2151   101
8,00000+ 0 1,50000+ 0 1,00000+ 0 1,20000- 3 2,50000- 2 0,00000+ 01043 2151   102
```

FILE 2

Resonance Parameter Data

LRU = 2, unresolved parameters
LFW = 0, fission widths not given.
LRF = 1, all parameters are energy-independent

FILE 2: Resonance Parameter (Structure of a Subsection)

LRU = 2 (unresolved parameters)
LFW = 1 (fission widths given)
LRF = 1 (only fission widths are energy-dependent)

SPI is the target spin, I.

A is the spin-independent effective scattering radius (10^{-12}cm).

NE is the number of energy points for the fission widths.

NLS is the number of ℓ -states given.

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

AWRI is the ratio of the mass of an isotope tc that of the neutron.

NJS is the number of J-states for a particular ℓ -state.

MUF is the number (integer) of degrees of freedom for fission width distribution.

AMUN is the number (floating point) of degrees of freedom for the neutron width distribution.

D, AJ, GNO, and GG are the mean level spacing, value of the spin state (J), average reduced neutron width, and radiation width for a specified J-state.

GF_i is the average fission width for the i^{th} energy point.

BCD Card Image Format for File 2

Resonance Parameters (Structure of a Subsection)

LRU = 2 (unresolved parameters)
LFW = 1 (fission widths given)
LRF = 1 (only fission widths energy-dependent)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	A	b*	b	NE	NLS	
ES_1	ES_2	ES_3	--	--	--	
--	--	--	--	--	ES_{NE}	LIST
AWRI	b	L	b	NJS	b	CØNT(ℓ)
b	b	L	MUF	NE+6	b	(first ℓ-state, first J-state)
D	AJ	AMUN	GNO	GG	b	
GF_1	GF_2	GF_3	--	--	--	
--	--	--	--	--	GF_{NE}	LIST
b	b	L	MUF	NE+6	b	(first ℓ-state, second J-state)
D	AJ	AMUN	GNO	GG	b	
GF_1	GF_2	GF_3	--	--	--	
--	--	--	--	--	GF_{NE}	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
b	b	L	MUF	NE+6	b	(first ℓ-state, last J-state)
D	AJ	AMUN	GNO	GG	b	
GF_1	GF_2	GF_3	--	--	--	
--	--	--	--	--	GF_{NE}	LIST

If NLS>1, repeat structure starting with CØNT-ℓ card.

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT= 151
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
0,00000+ 0 9,184  - 1                                          11          21105 2151  344
3,91   + 3 4,919  + 3 6,316  + 3 8,1103 + 3 1,0414 + 4 1,3372 + 41105 2151  345
1,7169 + 4 2,2046 + 4 2,8308 + 4 3,6348 + 4 4,0    +04          1105 2151  346
237,992                          0                  1          1105 2151  347
                                 0          1       17         1105 2151  348
1,35000+ 1 5,00000- 1 1,00000+ 0 1,41750- 3 2,30000- 2 0,00000+ 01105 2151  349
1,525  - 3 1,541  - 3 1,400  - 3 1,099  - 3 0,5462 - 3 0,4984 - 31105 2151  350
0,7209 - 3 1,080  - 3 1,671  - 3 1,992  - 3 1,929  -03          1105 2151  351
237,992                          1                  2          1105 2151  352
                                 1          1       17         1105 2151  353
1,35000+ 1 5,00000- 1 1,00000+ 0 2,36250- 3 2,30000- 2 0,00000+ 01105 2151  354
11,811 - 3 11,919 - 3 12,069 - 3 12,264 - 3 12,519 - 3 12,853 - 31105 2151  355
13,296 - 3 13,887 - 3 14,685 - 3 15,776 - 3 16,297 -03          1105 2151  356
                                 1          2       17         1105 2151  357
7,97000+ 0 1,50000+ 0 1,00000+ 0 1,39475- 3 2,30000- 2 0,00000+ 01105 2151  358
3,203  - 3 3,233  - 3 3,273  - 3 3,326  - 3 3,396  - 3 3,487  - 31105 2151  359
3,607  - 3 3,768  - 3 3,985  - 3 4,283  - 3 4,425  -03          1105 2151  360
```

FILE 2

Resonance Parameter Data

LRU = 2, unresolved parameters
LFW = 1, fission widths given
LRF = 1, only fission widths are energy-dependent

FILE 2: Resonance Parameters (Structure of a Subsection)

LRU = 2 (unresolved resonance parameters)
LRF = 2 (all parameters are energy dependent)
LFW = 1 or 2

SPI is the target spin, I.

A is the spin independent effective scattering radius (10^{-12}cm).

NLS is the number of ℓ-states given.

AWRI is the ratio fo the mass of an isotope to that of the neutron.

NJS is the number of J-states given for a particular ℓ-state.

AJ is the floating point value of the J-state.

INT is the interpolation scheme to be used between given values of the average parameters.

NE is the number of energy points at which parameters are given.

AMUX, AMUN, AMUG, and AMUF are the number (floating point) of degrees of freedom for the competitive width, neutron width, radiation width, and fission width distribution, respectively.

E_i, D_i, GX_i, GNO_i, GG_i, and GF ; are the incident energy, mean level spacing, average competitive width, reduced neutron width, radiation width, and fission width for the i^{th} point, respectively.

BCD Card Image Format for File 2

Resonance Parameters (Structure of a Subsection)

LRU = 2 (unresolved resonance parameters)
LRF = 2 (all parameters are energy dependent)
LFW = 1 or 2

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
SPI	A	b*	b	NLS	b	CØNT
AWRI	b	L	b	NJS	b	CØNT(ℓ)
AJ	b	INT	b	(6*NE)+6	NE	(first ℓ-state, first J-state)
b	b	AMUX	AMUN	AMUG	AMUF	
E_1	D_1	GX_1	GNO_1	GG_1	GF_1	
E_2	D_2	GX_2	GNO_2	GG_2	GF_2	
--	--	--	--	--	--	
E_{NE}	D_{NE}	GX_{NE}	GNO_{NE}	GG_{NE}	GF_{NE}	LIST
AJ	b	INT	b	(6*NE)+6	NE	(first ℓ-state, second J-state)
b	b	AMUX	AMUN	AMUG	AMUF	
E_1	D_1	GX_1	GNO_1	GG_1	GF_1	
--	--	--	--	--	--	
E_{NE}	D_{NE}	GX_{NE}	GNO_{NE}	GG_{NE}	GF_{NE}	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
AJ	b	INT	b	(6*NE)+6	NE	(first ℓ-state, last J-state)
b	b	AMUX	AMUN	AMUG	AMUF	
E_1	D_1	GX_1	GNO_1	GG_1	GF_1	
--	--	--	--	--	--	
E_{NE}	D_{NE}	GX_{NE}	GNO_{NE}	GG_{NE}	GF_{NE}	LIST

If NLS>1, repeat structure starting with CØNT(ℓ) card

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 2
Field 9 (Cols 73-75) = MT = 151
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
3,5         Ø,926                                      2             1102 2151  499
233,025                                Ø               2             1102 2151  500
3,ØØØØE+ØØ                             5             156           251102 2151  501
                             Ø,          1,ØØØØE+ØØ Ø,       2,ØØØØE+ØØ1102 2151  502
6,45Ø4E+Ø1 1,ØØØØE+ØØ Ø,Ø              7,18Ø7E-Ø5 3,5ØØØE-Ø2 2,9347E-Ø11102 2151  5Ø3
8,29Ø2E+Ø1 1,ØØØØE+ØØ Ø,Ø              8,1753E-Ø5 3,5ØØØE-Ø2 2,9133E-Ø11102 2151  5Ø4
1,1569E+Ø2 1,ØØØØE+ØØ Ø,Ø              9,Ø472E-Ø5 3,5ØØØE-Ø2 2,4665E-Ø11102 2151  5Ø5
1,4855E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,Ø486E-Ø4 3,5ØØØE-Ø2 2,1766E-Ø11102 2151  5Ø6
1,9Ø74E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,Ø389E-Ø4 3,5ØØØE-Ø2 2,6593E-Ø11102 2151  5Ø7
2,4491E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,3175E-Ø4 3,5ØØØE-Ø2 3,8Ø29E-Ø11102 2151  5Ø8
3,1447E+Ø2 1,ØØØØE+ØØ Ø,Ø              9,6Ø6ØE-Ø5 3,5ØØØE-Ø2 3,5141E-Ø11102 2151  5Ø9
4,Ø379E+Ø2 1,ØØØØE+ØØ Ø,Ø              8,8629E-Ø5 3,5ØØØE-Ø2 3,7344E-Ø11102 2151  51Ø
5,1848E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,1455E-Ø4 3,5ØØØE-Ø2 5,139ØE-Ø11102 2151  511
6,6574E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,1126E-Ø4 3,5ØØØE-Ø2 3,4713E-Ø11102 2151  512
8,5482E+Ø2 1,ØØØØE+ØØ Ø,Ø              1,Ø479E-Ø4 3,5ØØØE-Ø2 2,5Ø96E-Ø11102 2151  513
1,Ø976E+Ø3 1,ØØØØE+ØØ Ø,Ø              1,1267E-Ø4 3,5ØØØE-Ø2 2,5373E-Ø11102 2151  514
1,4Ø93E+Ø3 1,ØØØØE+ØØ Ø,Ø              9,989ØE-Ø5 3,5ØØØE-Ø2 3,4813E-Ø11102 2151  515
1.8Ø96E+Ø3 1,ØØØØE+ØØ Ø,Ø              1,Ø16ØE-Ø4 3,5ØØØE-Ø2 3,7083E-Ø11102 2151  516
2,3236E+Ø3 1,ØØØØE+ØØ Ø,Ø              9,8627E-Ø5 3,5ØØØE-Ø2 4,361ØE-Ø11102 2151  517
2,9836E+Ø3 1,ØØØØE+ØØ Ø,Ø              9,9Ø29E-Ø5 3,5ØØØE-Ø2 4,3348E-Ø11102 2151  518
____       ____       ____    ____     ____       ____       ____      ____  ____
2,2Ø46E+Ø4 1,ØØØØE+ØØ Ø,Ø              9,5268E-Ø5 3,5ØØØE-Ø2 4,7366E-Ø11102 2151  526
2,4788E+Ø4 1,ØØØØE+ØØ Ø,Ø              1,ØØ75E-Ø4 3,5ØØØE-Ø2 4,6219E-Ø11102 2151  527
4,ØØØØE+ØØ                             5             156           251102 2151  528
                             Ø,          1,ØØØØE+ØØ Ø,       1,ØØØØE+ØØ1102 2151  529
6,45Ø4E+Ø1 1,ØØØØE+ØØ Ø,Ø              7,18Ø7E-Ø5 3,5ØØØE-Ø2 1,4673E-Ø11102 2151  53Ø
8,29Ø2E+Ø1 1,ØØØØE+ØØ Ø,Ø              8,1753E-Ø5 3,5ØØØE-Ø2 1,4567E-Ø11102 2151  531
1,1569E+Ø2 1,ØØØØE+ØØ Ø,Ø              9,Ø472E-Ø5 3,5ØØØE-Ø2 1,2333E-Ø11102 2151  532

____       ____       ____    ____     ____       ____       ____      ____  ____

2,4788E+Ø4 1,ØØØØE+ØØ Ø,Ø              1,ØØ75E-Ø4 3,5ØØØE-Ø2 2,311ØE-Ø11102 2151  554
2,33Ø2E+Ø2                             1               4             1102 2151  555
2,ØØØØE+ØØ                             5             156           251102 2151  556
                             Ø,          1,ØØØØE+ØØ Ø,       2,ØØØØE+ØØ1102 2151  557
6,45Ø4E+Ø1 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  558
8,29Ø2E+Ø1 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  559
1,1569E+Ø2 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  56Ø
1,4855E+Ø2 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  561
1,9Ø74E+Ø2 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  562
2,4491E+Ø2 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  563
3,1447E+Ø2 1,16ØØE+ØØ Ø,Ø              2,32ØØE-Ø4 3,5ØØØE-Ø2 3,32ØØE-Ø11102 2151  564

____       ____       ____    ____     ____       ____       ____      ____  ____

2,3236E+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  653
2,9836E+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  654
3,831ØE+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  655
4,9191E+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  656
6,3163E+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  657
8,11Ø3E+Ø3 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  658
1,Ø414E+Ø4 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  659
1,3372E+Ø4 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  66Ø
1,7169E+Ø4 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  661
2,2Ø46E+Ø4 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  662
2,4788E+Ø4 1,12ØØE+ØØ Ø,Ø              2,24ØØE-Ø4 3,5ØØØE-Ø2 1,43ØØE-Ø11102 2151  663
```

FILE 2

Resonance Parameter Data

LRU = 2, unresolved parameters
LFW = 1 or 2
LRF = 2, all parameters are energy-dependent

FILE 3: Neutron Cross Sectons, $\sigma(E)$
(Structure for a Reaction Type)

LFS is an indicator that specifies the final excited state of the residual nucleus produced by a particular reaction, e.g., if LFS = 3, this means that the final state was the 3rd state.

Q is the reaction Q-value (eV).

NR is the number of interpolation ranges given.

NP is the total number of energy points used to specify the data (energy-cross section pairs).

NBT_i, INT_i is the interpolation scheme.

E_i, $\sigma(E_i)$ is the energy of the i^{th} point and the cross section for that energy point.

LIS designates the sate of the target nucleus (0 = ground state, 1 = 1st excited state, etc.)

LT = 0, no temperature dependence
>0, cross-sn. is given at LT+1 temperatures

BCD Card Image Format for File 3

Neutron Cross Section, $\sigma(E)$
(Structure for a reaction type)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LIS	LFS	b	b	HEAD
T	Q	LT	b	NR	NP	
NBT_1	INT_1	NBT_2	INT_2	NBT_3	INT_3	
--	--	--	--	NBT_{NR}	INT_{NR}	
E_1	$\sigma(E_1)$	E_2	$\sigma(E_2)$	E_3	$\sigma(E_3)$	
--	--	--	--	--	--	
--	--	--	--	E_{NP}	$\sigma(E_{NP})$	TAB1
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 3
Field 9 (Cols 73-75) = MT (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
2.00000+ 3 3.96822+ 0                      0                      1088 3103  202
0.00000+ 0 7.64400+ 5           0                     1        1421088 3103  203
       142          5                                             1088 3103  204
1.00000- 5 3.48300- 1 2.00000- 5 2.46300- 1 3.00000- 5 2.01100- 11088 3103  205
5.00000- 5 1.55800- 1 7.00000- 5 1.31700- 1 1.00000- 4 1.10100- 11088 3103  206
2.00000- 4 7.78900- 2 3.00000- 4 6.35900- 2 5.00000- 4 4.92600- 21088 3103  207
7.00000- 4 4.16300- 2 1.00000- 3 3.48300- 2 2.00000- 3 2.46300- 21088 3103  208
3.00000- 3 2.01100- 2 5.00000- 3 1.55800- 2 7.00000- 3 1.31600- 21088 3103  209
1.00000- 2 1.10100- 2 2.00000- 2 7.78700- 3 2.53000- 2 6.92400- 31088 3103  210
3.00000- 2 6.35800- 3 5.00000- 2 4.92400- 3 7.00000- 2 4.16100- 31088 3103  211
1.00000- 1 3.48100- 3 2.00000- 1 2.46100- 3 3.00000- 1 2.01000- 31088 3103  212
5.00000- 1 1.55600- 3 7.00000- 1 1.31600- 3 1.00000+ 0 1.10000- 31088 3103  213
2.00000+ 0 7.77300- 4 3.00000+ 0 6.34400- 4 5.00000+ 0 4.91000- 41088 3103  214
7.00000+ 0 4.14700- 4 1.00000+ 1 3.46700- 4 2.00000+ 1 2.44800- 41088 3103  215
3.00000+ 1 1.99600- 4 5.00000+ 1 1.54200- 4 7.00000+ 1 1.30100- 41088 3103  216
1.00000+ 2 1.08600- 4 2.00000+ 2 7.63100- 5 3.00000+ 2 6.20100- 51088 3103  217
5.00000+ 2 4.77100- 5 7.00000+ 2 4.00400- 5 1.00000+ 3 3.32800- 51088 3103  218
2.00000+ 3 2.31400- 5 3.00000+ 3 1.85900- 5 5.00000+ 3 1.41700- 51088 3103  219
7.00000+ 3 1.18300- 5 1.00000+ 4 9.88000- 6 2.00000+ 4 6.82500- 61088 3103  220
3.00000+ 4 5.33000- 6 5.00000+ 4 3.90000- 6 7.00000+ 4 3.18500- 61088 3103  221
1.00000+ 5 2.57400- 6 1.50000+ 5 2.06700- 6 2.00000+ 5 1.75500- 61088 3103  222
2.50000+ 5 1.56000- 6 3.00000+ 5 1.43000- 6 3.50000+ 5 1.31300- 61088 3103  223
4.00000+ 5 1.24800- 6 4.50000+ 5 1.19600- 6 5.00000+ 5 1.18300- 61088 3103  224
5.50000+ 5 1.17000- 6 6.00000+ 5 1.15700- 6 6.50000+ 5 1.15700- 61088 3103  225
7.00000+ 5 1.15700- 6 7.50000+ 5 1.15700- 6 8.00000+ 5 1.15700- 61088 3103  226
8.50000+ 5 1.15700- 6 9.00000+ 5 1.15700- 6 9.50000+ 5 1.15700- 61088 3103  227
1.00000+ 6 1.15700- 6 1.05000+ 6 1.15700- 6 1.10000+ 6 1.15700- 61088 3103  228
1.13000+ 6 1.15700- 6 1.15000+ 6 1.15700- 6 1.17000+ 6 1.15700- 61088 3103  229
1.20000+ 6 1.15700- 6 1.25000+ 6 1.15700- 6 1.30000+ 6 1.15700- 61088 3103  230
1.35000+ 6 1.15700- 6 1.40000+ 6 1.15700- 6 1.45000+ 6 1.15700- 61088 3103  231
1.50000+ 6 1.15700- 6 1.60000+ 6 1.15100- 6 1.70000+ 6 1.14400- 61088 3103  232
1.80000+ 6 1.13800- 6 1.90000+ 6 1.11800- 6 2.00000+ 6 1.09200- 61088 3103  233
2.10000+ 6 1.06600- 6 2.20000+ 6 1.04000- 6 2.30000+ 6 1.01400- 61088 3103  234
2.40000+ 6 9.88000- 7 2.50000+ 6 9.62000- 7 2.60000+ 6 9.36000- 71088 3103  235
2.70000+ 6 9.10000- 7 2.80000+ 6 8.84000- 7 2.90000+ 6 8.58000- 71088 3103  236
3.00000+ 6 8.32000- 7 3.20000+ 6 7.80000- 7 3.40000+ 6 7.28000- 71088 3103  237
3.60000+ 6 6.82500- 7 3.80000+ 6 6.43500- 7 4.00000+ 6 6.11000- 71088 3103  238
4.20000+ 6 5.78500- 7 4.40000+ 6 5.46000- 7 4.60000+ 6 5.26500- 71088 3103  239
4.80000+ 6 5.00500- 7 5.00000+ 6 4.81000- 7 5.20000+ 6 4.61500- 71088 3103  240
5.40000+ 6 4.42000- 7 5.60000+ 6 4.22500- 7 5.80000+ 6 4.09500- 71088 3103  241
6.00000+ 6 3.96500- 7 6.20000+ 6 3.77000- 7 6.40000+ 6 3.64000- 71088 3103  242
6.60000+ 6 3.51000- 7 6.80000+ 6 3.40600- 7 7.00000+ 6 3.31500- 71088 3103  243
7.20000+ 6 3.21100- 7 7.40000+ 6 3.13300- 7 7.60000+ 6 3.05500- 71088 3103  244
7.80000+ 6 2.97700- 7 8.00000+ 6 2.91200- 7 8.20000+ 6 2.84700- 71088 3103  245
8.40000+ 6 2.76900- 7 8.60000+ 6 2.71700- 7 8.80000+ 6 2.66500- 71088 3103  246
9.00000+ 6 2.60000- 7 9.20000+ 6 2.54800- 7 9.40000+ 6 2.48300- 71088 3103  247
9.60000+ 6 2.43100- 7 9.80000+ 6 2.37900- 7 1.00000+ 7 2.34000- 71088 3103  248
1.05000+ 7 2.23600- 7 1.10000+ 7 2.14500- 7 1.15000+ 7 2.05400- 71088 3103  249
1.20000+ 7 1.97600- 7 1.25000+ 7 1.89800- 7 1.30000+ 7 1.83300- 71088 3103  250
1.35000+ 7 1.76800- 7 1.40000+ 7 1.70300- 7 1.45000+ 7 1.65100- 71088 3103  251
1.50000+ 7 1.59900- 7                                             1088 3103  252
                                                                  1088 3  0  253
```

FILE 3

Neutron Cross Sections

Neutron cross sections, $\sigma_{n,x}(E)$, in barns, given as a function of incident neutron energy, E (in the laboratory system).

The threshold energy for a reaction is:

$$E_{th} = \left(\frac{AWR + 1.0}{AWR}\right) \left|Q\right|$$

FILE 4: Angular Distributions of Secondary Neutrons

LTT = 1 (Legendre expansion coefficients given)
LVT = 1 (Transformation matrix given)

LVT is a test for the transformation matrix:
LVT = 0 (not given)
LVT = 1 (transformation matrix given)

LTT is a flag to specify the representation used:
LTT = 1 (Legendre coefficients)
LTT = 2 (tabulated distributions)

LCT is a flag to specify the frame of reference:
LCT = 1 (laboratory system)
LCT = 2 (center-of-mass system)

NK is the number of elements in the transformation matrix.

NM is the maximum order Legendre polynomial that will be required for the angular distributions (in either the (CM) or (LAB) system.

$U_{\ell,m}$ are the elements of the transformation matrix.

NR is the number of interpolation ranges for the distribution.

NP is the number of energy points at which distribution will be given.

NBT_i and INT_i is the interpolation scheme for the i^{th} range (to interpolate distributions between given energy points).

E_i is the energy of the i^{th} point.

NL is the order of the Legendre expansion at a particular energy point.

$f_\ell(E_i)$ is the value of the ℓ^{th} coefficient for the i^{th} point.

BCD Card Image Format for File 4

Angular Distributions of Secondary Neutrons

LTT = 1 (Legendre expansion coefficients given)
LVT = 1 (transformation matrix given)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LVT=1	LTT=1	b*	b	HEAD
b	AWR	b	LCT	NK	NM	
$U_{0,0}$	$U_{1,0}$	$U_{2,0}$	--	--	$U_{NM,0}$	
$U_{0,1}$	$U_{1,1}$	$U_{2,1}$	--	--	$U_{NM,1}$	
$U_{0,2}$	$U_{1,2}$	$U_{2,2}$	--	--	$U_{NM,2}$	
--	--	--	--	--	--	
--	--	--	--	--	--	
$U_{0,NM}$	$U_{1,NM}$	$U_{2,NM}$	--	--	$U_{NM,NM}$	LIST
b	b	b	b	NR	NE	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NL	b	(f's for first energy point)
$f_1(E_1)$	$f_2(E_1)$	$f_3(E_1)$	--	--	--	
--	--	--	--	--	$f_{NL}(E_1)$	LIST
T	E_2	LT	b	NL	b	
$f_1(E_2)$	$f_2(E_2)$	$f_3(E_2)$	--	--	--	
--	--	--	--	--	$f_{NL}(E_2)$	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
T	E_{NE}	LT	b	NL	b	(f's for last energy point)
$f_1(E_{NE})$	$f_2(E_{NE})$	$f_3(E_{NE})$	--	--	--	
--	--	--	--	--	$f_{NL}(E_{NE})$	LIST
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 4
Field 9 (Cols 73-75) = MT (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
 2,ØØØØØ+ 3 3,96822+ Ø          1          1                     1Ø88 4  2  375
            3,96822+ Ø                     2          1ØØ       91Ø88 4  2  376
 1,ØØØØØ+ Ø 1,68ØØØ- 1 1,2819Ø- 2 Ø,ØØØØØ+ Ø-6,6316Ø- 5 Ø,ØØØØØ+ Ø1Ø88 4  2  377
 6,37Ø1Ø- 7 Ø,ØØØØØ+ Ø-7,4839Ø- 9 Ø,ØØØØØ+ Ø Ø,ØØØØØ+ Ø 9,619ØØ- 11Ø88 4  2  378
 2,9686Ø- 1 4,3546Ø- 2 3,1Ø27Ø- 3 Ø,ØØØØØ+ Ø-1,4971Ø- 5 Ø,ØØØØØ+ Ø1Ø88 4  2  379
 1,3935Ø- 7 Ø,ØØØØØ+ Ø Ø,ØØØØØ+ Ø-1,5886Ø- 1 9,Ø195Ø- 1 4,1Ø67Ø- 11Ø88 4  2  38Ø
 8,8718Ø- 2 1,1Ø85Ø- 2 7,1925Ø- 4 Ø,ØØØØØ+ Ø-3,1239Ø- 6 Ø,ØØØØØ+ Ø1Ø88 4  2  381
 Ø,ØØØØØ+ Ø 3,5863Ø- 2-2,73Ø4Ø- 1 8,1319Ø- 1 5,Ø658Ø- 1 1,4568Ø- 11Ø88 4  2  382
 2,5474Ø- 2 2,8Ø77Ø- 3 1,6752Ø- 4 Ø,ØØØØØ+ Ø Ø,ØØØØØ+ Ø-8,5905Ø- 31Ø88 4  2  383
 7,8689Ø- 2-3,6481Ø- 1 7,Ø133Ø- 1 5,811ØØ- 1 2,1109Ø- 1 4,7327Ø- 21Ø88 4  2  384
 7,1261Ø- 3 7,Ø975Ø- 4 Ø,ØØØØØ+ Ø 2,1Ø26Ø- 3-2,2Ø43Ø- 2 1,2928Ø- 11Ø88 4  2  385
-4,3373Ø- 1 5,7179Ø- 1 6,3117Ø- 1 2,81Ø7Ø- 1 7,7131Ø- 2 1,46ØØØ- 21Ø88 4  2  386
 Ø,ØØØØØ+ Ø-5,1992Ø- 4 6,Ø621Ø- 3-4,1723Ø- 2 1,8451Ø- 1-4,7795Ø- 11Ø88 4  2  387
 4,3Ø7ØØ- 1 6,5479Ø- 1 3,5139Ø- 1 1,1474Ø- 1 Ø,ØØØØØ+ Ø 1,2929Ø- 41Ø88 4  2  388
-1,6456Ø- 3 1,2786Ø- 2-6,7791Ø- 2 2,4Ø55Ø- 1-4,9642Ø- 1 2,8461Ø- 11Ø88 4  2  389
 6,5122Ø- 1 4,1767Ø- 1 Ø,ØØØØØ+ Ø-3,2264Ø- 5 4,4243Ø- 4-3,79Ø1Ø- 31Ø88 4  2  39Ø
 2,2986Ø- 2-9,9735Ø- 2 2,9341Ø- 1-4,8921Ø- 1 1,4Ø18Ø- 1 6,2111Ø- 11Ø88 4  2  391
 Ø,ØØØØØ+ Ø 8,Ø695Ø- 6-1,18Ø7Ø- 4 1,Ø976Ø- 3-7,4Ø43Ø- 3 3,7198Ø- 21Ø88 4  2  392
-1,3644Ø- 1 3,3928Ø- 1-4,5769Ø- 1 3,821ØØ- 3                     1Ø88 4  2  393
                                                      1        1Ø81Ø88 4  2  394
       1Ø8          3                                            1Ø88 4  2  395
 Ø,ØØØØØ+ Ø 1,ØØØØØ- 5          Ø                     1          1Ø88 4  2  396
 Ø,ØØØØØ+ Ø                                                      1Ø88 4  2  397
 Ø,ØØØØØ+ Ø 2,53ØØØ- 2          Ø                     1          1Ø88 4  2  398
 Ø,ØØØØØ+ Ø                                                      1Ø88 4  2  399
 Ø,ØØØØØ+ Ø 1,ØØØØØ+ 2          Ø                     2          1Ø88 4  2  4ØØ
-1,13ØØØ- 4 Ø,ØØØØØ+ Ø                                           1Ø88 4  2  4Ø1
 Ø,ØØØØØ+ Ø 2,ØØØØØ+ 2          Ø                     2          1Ø88 4  2  4Ø2
-2,27ØØØ- 4 Ø,ØØØØØ+ Ø                                           1Ø88 4  2  4Ø3
 Ø,ØØØØØ+ Ø 3,ØØØØØ+ 2          Ø                     2          1Ø88 4  2  4Ø4
-3,4ØØØØ- 4 Ø,ØØØØØ+ Ø                                           1Ø88 4  2  4Ø5
 Ø,ØØØØØ+ Ø 5,ØØØØØ+ 2          Ø                     2          1Ø88 4  2  4Ø6
-5,67ØØØ- 4 Ø,ØØØØØ+ Ø                                           1Ø88 4  2  4Ø7

 ____   ____   ____   ____   ____   ____   ____   ____   ____   ____

 Ø,ØØØØØ+ Ø 1,15ØØØ+ 7          Ø                     4          1Ø88 4  2  596
 4,621ØØ- 1 2,326ØØ- 1 2,313ØØ- 3 9,25ØØØ- 4                     1Ø88 4  2  597
 Ø,ØØØØØ+ Ø 1,2ØØØØ+ 7          Ø                     4          1Ø88 4  2  598
 4,651ØØ- 1 2,34ØØØ- 1 3,169ØØ- 3 1,Ø83ØØ- 3                     1Ø88 4  2  599
 Ø,ØØØØØ+ Ø 1,25ØØØ+ 7          Ø                     4          1Ø88 4  2  6ØØ
 4,674ØØ- 1 2,351ØØ- 1 4,Ø82ØØ- 3 1,231ØØ- 3                     1Ø88 4  2  6Ø1
 Ø,ØØØØØ+ Ø 1,3ØØØØ+ 7          Ø                     4          1Ø88 4  2  6Ø2
 4,7Ø1ØØ- 1 2,362ØØ- 1 5,ØØ1ØØ- 3 1,429ØØ- 3                     1Ø88 4  2  6Ø3
 Ø,ØØØØØ+ Ø 1,35ØØØ+ 7          Ø                     4          1Ø88 4  2  6Ø4
 4,725ØØ- 1 2,374ØØ- 1 6,139ØØ- 3 1,6ØØØØ- 3                     1Ø88 4  2  6Ø5
 Ø,ØØØØØ+ Ø 1,4ØØØØ+ 7          Ø                     4          1Ø88 4  2  6Ø6
 4,75ØØØ- 1 2,388ØØ- 1 7,284ØØ- 3 1,781ØØ- 3                     1Ø88 4  2  6Ø7
 Ø,ØØØØØ+ Ø 1,45ØØØ+ 7          Ø                     4          1Ø88 4  2  6Ø8
 4,775ØØ- 1 2,4Ø2ØØ- 1 8,515ØØ- 3 1,971ØØ- 3                     1Ø88 4  2  6Ø9
 Ø,ØØØØØ+ Ø 1,5ØØØØ+ 7          Ø                     4          1Ø88 4  2  61Ø
 4,8Ø1ØØ- 1 2,418ØØ- 1 9,8Ø9ØØ- 3 2,171ØØ- 3                     1Ø88 4  2  611
                                                                 1Ø88 4  Ø  612
                                                                 1Ø88 Ø  Ø  613
```

FILE 4

Angular Distributions of Secondary Neutrons

LTT = 1, Legendre coefficients given
LVT = 1, transformation matrix given.

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

$$f_o \cong 1.0$$

$$f_\ell(E) = \frac{2\pi}{\sigma_s(E)} \int_{-1}^{+1} \frac{d\sigma(\Omega, E)}{d\Omega} P_\ell(\mu) d\mu$$

$\sigma_s(E)$ given in File 3 (same MT number)

If LCT = 1 (laboratory system)

$$f_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell,m} f_m^L(E).$$

If LCT = 2 (center-of-mass system)

$$f_\ell^L(E) = \sum_{m=0}^{NM} U_{\ell,m} f_m^{CM}(E)$$

FILE 4: Angular Distributions of Secondary Neutrons

LTT = 1 (Legendre polynomial representation)
LVT = 0 (no transformation matrix)

LTT is a flag to specify representation use:
LTT = 1 (Legendre polynomial expansion)
LTT = 2 (tabulated distribution)

LVT is a flag to specify whether a transformation matrix has been given:
LVT = 0 (none)
LVT = 1 (transformation matrix given)

LCT is a flag to specify frame of reference given:
LCT = 1 (laboratory system)
LCT = 2 (center-of-mass system)

NR is the number of interpolation ranges.

NE is the number of energy points at which distributions are given.

NBT_i and INT_i is the scheme to interpolate the coefficient, f_ℓ, between given values.

E_i is the energy of point i at which a distribution is given.

NL is the order of Legendre polynomial expansion.

$f_\ell(E_i)$ is the ℓ^{th} coefficient given at point i.

BCD Card Image Format for File 4

Angular Distributions of Secondary Neutrons

LTT = 1 (Legendre polynomial coefficients given)
LVT = 0 (No transformation matrix)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LVT=0	LTT=1	b*	b	HEAD
b	AWR	b	LCT	NK=0	NM=0	CØNT
b	b	b	b	NR	NE	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NL	b	
$f_1(E_1)$	$f_2(E_1)$	--	--	--	$f_{NL}(E_1)$	LIST
T	E_2	LT	b	NL	b	
$f_1(E_2)$	$f_2(E_2)$	--	--	--	$f_{NL}(E_2)$	LIST
--	--	--	--	--	--	
--	--	--	--	--	--	
T	E_{NE}	LT	b	NL	b	
$f_1(E_{NE})$	$f_2(E_{NE})$	--	--	--	$f_{NL}(E_{NE})$	LIST
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 4
Field 9 (Cols 73-75) = MT (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.22330+04 2.31043+02          0          1                    1041 4 2 2371
           2.31043+02                     2          0         01041 4 2 2372
                                                        1      431041 4 2 2373
          43          2                                         1041 4 2 2374
0.0          1.0    -05          0                    1         1041 4 2 2375
0.0                                                             1041 4 2 2376
0.0          1.0                0                    1         1041 4 2 2377
0.0                                                             1041 4 2 2378
0.0          1.0    +02         0                    1         1041 4 2 2379
0.0                                                             1041 4 2 2380
0.0          5.0    +03         0                    1         1041 4 2 2381
0.0                                                             1041 4 2 2382
       0.0  .10000+05                                    7      1041 4 2 2383
 .33755-01  .58325-03 .12372-05 .48056-08 .38884-11 .10752-131041 4 2 2384
 .00000     .00000    .00000    .00000    .00000    .00000    1041 4 2 2385
       0.0  .20000+05                                    7      1041 4 2 2386
 .70415-01  .22608-02 .98636-05 .76459-07 .12585-09 .71100-121041 4 2 2387
 .00000     .00000    .00000    .00000    .00000    .00000    1041 4 2 2388
       0.0  .40000+05                                    9      1041 4 2 2389
 .14340-00  .83425-02 .76763-04 .11678-05 .53354-08 .48586-101041 4 2 2390
 .18037-12  .48244-17 .00000    .00000    .00000    .00000    1041 4 2 2391
       0.0  .50000+05                                    8      1041 4 2 2392
 .10545+00  .44994-01 .87053-04 .29361-05 .80295-08 .92793-081041 4 2 2393
 .19372-12  .42521-10 .00000    .00000    .00000    .00000    1041 4 2 2394
       0.0  .60000+05                                    8      1041 4 2 2395
 .13024-00  .44442-01 .15341-03 .52511-05 .20049-07 .82242-081041 4 2 2396
 .61352-12  .42395-11 .00000    .00000    .00000    .00000    1041 4 2 2397



       0.0  .12000+08                                   20      1041 4 2 2520
 .81210-00  .66847-00 .57574-00 .51803-00 .45993-00 .40346-001041 4 2 2521
 .33907-00  .28231-00 .23159-00 .20216-00 .16613-00 .11340+001041 4 2 2522
 .63329-01  .29519-01 .11035-01 .36845-02 .11638-02 .28698-031041 4 2 2523
 .74457-04  .17074-04 .00000    .00000    .00000    .00000    1041 4 2 2524
       0.0  .15000+08                                   20      1041 4 2 2525
 .84298-00  .69334-00 .57473-00 .49608-00 .43902-00 .39793-001041 4 2 2526
 .36039-00  .32774-00 .29587-00 .26710-00 .23635-00 .18941-001041 4 2 2527
 .12898-00  .74596-01 .37466-01 .17337-01 .72033-02 .23195-021041 4 2 2528
 .84942-03  .24189-03 .00000    .00000    .00000    .00000    1041 4 2 2529
                                                              1041 4 0 2530
                                                              1041 0 0 2531
```

FILE 4

Angular Distributions of Secondary Neutrons

LTT = 1, Legendre coefficients given.
LVT = 0, no transformation matrix

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

$\sigma_s(E)$ given in File 3 (same MT number)

$f_o = 1.00$

$$f_\ell(E) = \frac{2\pi}{\sigma_s(E)} \int_{-1}^{+1} \frac{d\sigma(\Omega\ ,E)}{d\Omega}\ P_\ell(\mu)\, d\mu$$

FILE 4: Angular Distributions of Secondary Neutrons

LTT = 2 (tabulated distributions given)
LVT = 1 (transformation matrix given)

LTT is a flag to specify representation used:
LTT = 1 (Legendre polynomial expansion)
LTT = 2 (tabulated distribution given)

LVT is a flag to specify whether transformation matrix given:
LVT = 0 (not given)
LVT = 1 (transformation matrix given)

LCT is a flag to specify frame of reference given:
LCT = 1 (laboratory system)
LCT = 2 (center-of-mass system)

NK is the number of elements in transformation matrix.

NM is the maximum order Legendre polynomial that will be required to represent angular distribution in either the (LAB) or (CM) system.

$U_{\ell,m}$ are elements of the transformation matrix.

NR is the number of interpolation ranges.

NE is the number of energy points at which distributions are given.

NBT_i and INT_i is the scheme for interpolating between given distributions.

E_i is the energy of point i at which a distribution is given.

NP is the number of cosine values for a particular distribution.

μ_j is the value of the cosine at point j.

$p(\mu_j, E_i)$ is the normalized angular probability at μ_j for energy point, E_i.

BCD Card Image Format for File 4

Angular Distributions of Secondary Neutrons

LTT = 2 (tabulated distributions given)
LVT = 1 (transformation matrix given)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LVT=1	LTT=2	b*	b	HEAD
b	AWR	b	LCT	NK	NM	
$U_{0,0}$	$U_{1,0}$	$U_{2,0}$	--	--	$U_{NM,0}$	
$U_{0,1}$	$U_{1,1}$	$U_{2,1}$	--	--	$U_{NM,1}$	
$U_{0.2}$	$U_{1,2}$	$U_{2,2}$	--	--	$U_{NM,2}$	
--	--	--	--	--	--	
--	--	--	--	--	--	
$U_{0,NM}$	$U_{1,NM}$	$U_{2,NM}$	--	--	U_{NM}, U_{NM}	LIST
b	b	b	b	NR	NE	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NR	NP	(first energy point)
μ_1	$p(\mu_1, E_1)$	μ_2	$p(\mu_2, E_1)$	μ_3	$p(\mu_3, E_1)$	
--	--	--	--	μ_{NP}	$p(\mu_{NP}, E_1)$	TAB1
T	E_2	LT	b	NR	NP	(second energy point)
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
μ_1	$p(\mu_1, E_2)$	μ_2	$p(\mu_2, E_2)$	--	--	
--	--	--	--	μ_{NP}	$p(\mu_{NP}, E_2)$	TAB1
--	--	--	--	--	--	
--	--	--	--	--	--	
T	E_{NE}	LT	b	NR	NP	(last energy point)
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
μ_1	$p(\mu_1, E_{NE})$	μ_2	$p(\mu_2, E_{NE})$	--	--	
--	--	--	--	μ_{NP}	$p(\mu_{NP}, E_{NE})$	TAB1
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 4
Field 9 (Cols 73-75) = MT(except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9,0232E+04 2,3004E+02          1          2                    1117 4  2  914
           2,3004E+02                     2          441        201117 4  2  915
1,0000E+00 2,8980E-03 3,7793E-06-8,8564E-17 0,0000E-01 0,0000E-011117 4  2  916
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  917
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  918
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 9,9999E-01 5,2164E-031117 4  2  919
1,2957E-05 1,5646E-08 1,0165E-11-1,1181E-14 0,0000E-01 0,0000E-011117 4  2  920
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  921
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  922
0,0000E-01-2,8979E-03 9,9997E-01 7,4519E-03 2,6995E-05 5,6895E-081117 4  2  923
7,4815E-11 3,4506E-14 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  924
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-011117 4  2  925
0,0000E-01 0,0000E-01 0,0000E-01 0,0000E-01 1,1338E-05-5,2162E-031117 4  2  926


0,0000E-01-2,5479E-30 1,9532E-27-1,0147E-24 4,4699E-22-1,8432E-191117 4  2  987
7,7630E-17-4,2615E-14 7,4293E-12-1,0896E-09 1,3148E-07-1,2570E-051117 4  2  988
8,9512E-04-4,2313E-02 9,9802E-01                         1117 4  2  989
                                                      1  121117 4  2  990
        12         3                                        1117 4  2  991
0,0000E-01 1,0000E-05          0                      1    21117 4  2  992
         2         2                                        1117 4  2  993
-1,0000E+00 5,0000E-01 1,0000E+00 5,0000E-01                1117 4  2  994
 0,0000E-01 1,0000E+04          0                      1    21117 4  2  995
         2         2                                        1117 4  2  996
-1,0000E+00 5,0000E-01 1,0000E+00 5,0000E-01                1117 4  2  997
 0,0000E-01 5,7000E+05          0                      1   191117 4  2  998
        19         2                                        1117 4  2  999
-1,0000E+00 2,8934E-01-9,8500E-01 2,7901E-01-9,4000E-01 2,6868E-011117 4  2 1000
-8,6600E-01 2,5834E-01-7,6600E-01 2,4801E-01-6,4300E-01 2,3767E-011117 4  2 1001
-5,0000E-01 2,2734E-01-3,4200E-01 2,3767E-01-1,7400E-01 2,5834E-011117 4  2 1002
 0,0000E-01 2,9968E-01 1,7400E-01 3,8235E-01 3,4200E-01 5,0635E-011117 4  2 1003
 5,0000E-01 6,7169E-01 6,4300E-01 8,4736E-01 7,6600E-01 1,0437E+001117 4  2 1004
 8,6600E-01 1,2400E+00 9,4000E-01 1,3640E+00 9,8500E-01 1,4570E+001117 4  2 1005


 0,0000E-01 1,5200E+07          0                      1   371117 4  2 1084
         37         2                                        1117 4  2 1085
-1,0000E+00 2,5170E-02-9,9600E-01 1,9780E-02-9,8500E-01 1,4390E-021117 4  2 1086
-9,6600E-01 1,7980E-02-9,4000E-01 3,0570E-02-9,0600E-01 1,9780E-021117 4  2 1087
-8,6600E-01 8,9900E-03-8,1900E-01 5,3900E-03-7,6600E-01 1,7980E-021117 4  2 1088
-7,0700E-01 2,5170E-02-6,4300E-01 2,3380E-02-5,7400E-01 1,9780E-021117 4  2 1089
-5,0000E-01 1,7620E-02-4,2300E-01 1,7980E-02-3,4200E-01 2,1580E-021117 4  2 1090
-2,5900E-01 3,2370E-02-1,7400E-01 5,0350E-02-8,7000E-02 5,5740E-021117 4  2 1091
 0,0000E-01 4,6750E-02 8,7000E-02 2,8770E-02 1,7400E-01 4,4950E-021117 4  2 1092
 2,5900E-01 8,9910E-02 3,4200E-01 1,5464E-01 4,2300E-01 1,9779E-011117 4  2 1093
 5,0000E-01 1,5284E-01 5,7400E-01 8,9910E-02 6,4300E-01 1,7981E-011117 4  2 1094
 7,0700E-01 4,4953E-01 7,6600E-01 8,9905E-01 8,1900E-01 1,0609E+001117 4  2 1095
 8,6600E-01 6,4732E-01 9,0600E-01 2,5174E-01 9,4000E-01 2,1577E+001117 4  2 1096
 9,6600E-01 7,1000E+00 9,8500E-01 1,7981E+01 9,9600E-01 2,4274E+011117 4  2 1097
 1,0000E+00 2,6972E+01                                    1117 4  2 1098
                                                          1117 4  0 1099
```

FILE 4

Angular Distributions of Secondary Neutrons

LTT = 2, tabulated distribution
LVT = 1, transformation matrix given

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

$$\int_{-1}^{+1} p(\mu,E)d\mu \equiv 1.0$$

$\sigma_s(E)$ is given in File 3 (same MT number)

If LCT = 1 (laboratory system)

$$f_\ell^{CM}(E) = \sum_{m=0}^{NM} U^{-1}_{\ell,m}\ f_m^L(E)$$

If LCT = 2 (center-of-mass system)

$$f_\ell^{L}(E) = \sum_{m=0}^{NM} U_{\ell,m} f_m^{CM}(E)$$

FILE 4: Angular Distributions of Secondary Neutrons

LTT = 2 (tabulated distributions given)
LVT = 0 (transformation matrix not given)

LTT is a flag to specify representation used:
LTT = 1 (Legendre polynomial expansion)
LTT = 2 (tabulated distributions given)

LVT is a flag to specify whether transformation matrix is given:
LVT = 0 (not given)
LVT = 1 (transformation matrix given)

LCT is a flag to specify frame of reference given:
LCT = 1 (laboratory system)
LCT = 2 (center-of-mass system)

NR is the number of interpolation ranges.

NE is the number of energy points at which distributions are given.

NBT_i and INT_i is the scheme for interpolating between given distributions.

E_i is the energy of point i at which a distribution is given.

NP is the number of cosine values for a particular distribution.

μ_i is the value of the cosine at point j.

$p(\mu_j,E_i)$ is the normalized angular probability at μ_j for energy point, E_i.

BCD Card Image Format for File 4

Angular Distributions of Secondary Neutrons

LTT = 2 (tabulated distributions given)
LVT = 0 (no transformation matrix)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	LVT=0	LTT=2	b*	b	HEAD
b	AWR	b	LCT	NK=0	NM=0	CØNT
b	b	b	b	NR	NE	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NR	NP	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
μ_1	$p(\mu_1,E_1)$	μ_2	$p(\mu_2,E_1)$	--	--	
--	--	--	--	μ_{NP}	$p(\mu_{NP},E_1)$	TAB1
T	E_2	LT	b	NR	NP	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
μ_1	$p(\mu_1,E_2)$	--	--	--	--	
--	--	--	--	μ_{NP}	$p(\mu_{NP},E_2)$	TAB1
--	--	--	--	--	--	
--	--	--	--	--	--	
T	E_{NE}	LT	b	NR	NP	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
μ_1	$p(\mu_1,E_{NE})$	--	--	--	--	
--	--	--	--	μ_{NP}	$p(\mu_{NP},E_{NE})$	TAB1
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 4
Field 9 (Cols 73-75) = MT (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9,2234 +04 2,32029+02          0          2                1043 4 51  715
           2,32029+02                     2          0     01043 4 51  716
                                          1     21043 4 51  717
           2          2                          1043 4 51  718
0,0        0,0443 +06             0          1     21043 4 51  719
           2          2                          1043 4 51  720
=1,0       0,5        1,0         0,5            1043 4 51  721
0,0        15,0   +06             0          1     21043 4 51  722
           2          2                          1043 4 51  723
=1,0       0,5        1,0         0,5            1043 4 51  724
                                                 1043 4  0  725
```

FILE 4

Angular Distribution of Secondary Neutrons

LTT = 2, tabulated distributions
LVT = 0, no transformation matrix

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

$$\int_{-1}^{+1} p(\mu,E)\ d\mu \equiv 1.00$$

$\sigma_s(E)$ is given in File 3 (same MT number).

FILE 5: Energy Distributions of Secondary Neutrons (General Structure)

NK is the number of partial energy distributions used for a particular reaction type (MT). There will be one subsection for each partial distribution.

The format for a subsection depends on the value of LF (a flag that specifies the type of distribution used).

LF	Distribution law used
1	Arbitrary tabulated function
3	Excitation of descrete level
5	General evaporation spectrum
7	Simple fission spectrum (Maxwellian)
9	Evaporation spectrum
10	Watt spectrum

BCD Card Image Format for File 5

Energy Distribution of Secondary Neutrons
(General Structure)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	b	NK	b	HEAD
<subsection for k = 1>						
<subsection for k = 2>						
<subsection for k = 3>						
--	--	--				
--	--	--				
<subsection for k = NK>						
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank

9,22340+ 4 2,32029+ 2 1 1043 5 17 801

1043 5 0 808

FILE 5

Energy Distributions of Secondary Neutrons
(General Structure)

$$\frac{d\sigma(E \rightarrow E')}{dE'} = m\sigma_x(E)\ p(E \rightarrow E')$$

m is the neutron multiplicity (e.g., m=2 for n,2n reactions) σ_x is given in File 3 (for the same MT number)

$$p(E \rightarrow E') = \sum_{k=1}^{NK} p_k(E)\ f_k(E \rightarrow E')$$

$p_k(E)$ and $f_k(E \rightarrow E')$ are given in File 5.

$$\int_0^{E'_{max}} p(E \rightarrow E')dE' \equiv 1.0$$

and at a particular incident neutron energy, E,

$$\sum_{k=1}^{NK} p_k(E) = 1.00$$

FILE 5: Energy Distributions of Secondary Neutron
(Structure of a Subsection)

LF = 1. (tabulated energy distributions)

NR is the number of interpolation ranges.

NP is the number of energy points at which probabilities, $p(E_i)$, are given.

NE is the number of inicident energy points at which distributions are given.

NF is the number of secondary energy points for a particular distribution.

$p(E_i)$ is the fractional probability for this distribution law (LF=1) at incident energy point, E_i.

$g(E_j \rightarrow E'_k)$ is the normalized probability that an incident neutron at energy, E_j, will end up at a secondary energy, E'_k.

NBT_i and INT_i is an interpolation scheme for:
(1) fractional probabilities, $p(E_i)$
(2) between distributions given at E .
(3) $g(E_j\text{-}E'_k)$, between given values of E'.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
(Structure of a subsection)

LF = 1 (tabulated energy distribution)

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	T	b*	LT	LF=1	NR	NP	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
	--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1
	b	b	b	b	NR	NE	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
	T	E_1	LT	b	NR	NF	(first energy point)
(3)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E'_1	$g(E_1 \to E'_1)$	E'_2	$g(E_1 \to E'_2)$	E'_3	$g(E_1 \to E'_3)$	
	--	--	--	--	E'_{NF}	$g(E_1 \to E'_{NF})$	TAB1
	T	E_2	LT	b	NR	NF	(second energy point)
(3)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E'_2	$g(E_2 \to E'_1)$	E'_2	$g(E_2 \to E'_2)$	E'_3	$g(E_2 \to E'_3)$	
	--	--	--	--	E'_{NF}	$g(E_2 \to E'_{NF})$	TAB1
	--	--	--	--	--	--	
	--	--	--	--	--	--	
	T	E_{NE}	LT	b	NR	NF	(last energy point)
(3)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E'_1	$g(E_{NE} \to E'_1)$	E'_2	$g(E_{NE} \to E'_2)$	--	--	
	--	--	--	--	E'_{NF}	$g(E_{NE} \to E'_{NF})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
Ø.Ø                                  Ø          1          1         25ØØ1 5 22 4696
           2           2                                              5ØØ1 5 22 4697
1.125    +Ø7 1.ØØØ        2.Ø      +Ø7 1.ØØØ                          5ØØ1 5 22 4698
                                                           1        2Ø5ØØ1 5 22 4699
          2Ø           2                                              5ØØ1 5 22 47ØØ
Ø.Ø          1.125   +Ø7             Ø                     1        235ØØ1 5 22 47Ø1
          23           2                                              5ØØ1 5 22 47Ø2
1.Ø      +Ø4 Ø.Ø          5.Ø      +Ø4 6.Ø534 -Ø8 1.Ø      +Ø5 1.1437 -Ø75ØØ1 5 22 47Ø3
2.Ø      +Ø5 2.Ø413 -Ø7 3.Ø        +Ø5 2.7326 -Ø7 4.Ø      +Ø5 3.2515 -Ø75ØØ1 5 22 47Ø4
5.Ø      +Ø5 3.6271 -Ø7 7.5        +Ø5 4.Ø934 -Ø7 1.Ø      +Ø6 4.1Ø62 -Ø75ØØ1 5 22 47Ø5
1.25     +Ø6 3.8617 -Ø7 1.5        +Ø6 3.4865 -Ø7 1.75     +Ø6 3.Ø6Ø3 -Ø75ØØ1 5 22 47Ø6
2.ØØ     +Ø6 2.6314 -Ø7 3.Ø        +Ø6 1.2647 -Ø7 4.ØØ     +Ø6 5.4Ø27 -Ø85ØØ1 5 22 47Ø7
5.ØØ     +Ø6 2.1638 -Ø8 6.Ø        +Ø6 8.3198 -Ø9 7.Ø      +Ø6 3.11ØØ -Ø95ØØ1 5 22 47Ø8
8.ØØ     +Ø6 1.1368 -Ø9 9.Ø        +Ø6 4.1Ø5  -1Ø 1.Ø      +Ø7 1.4614 -1Ø5ØØ1 5 22 47Ø9
1.1Ø     +Ø7 Ø.Ø          1.2      +Ø7 Ø.Ø                            5ØØ1 5 22 471Ø
Ø.Ø          1.15    +Ø7             Ø                     1        235ØØ1 5 22 4711
          23           2                                              5ØØ1 5 22 4712
1.ØØ     +Ø4 Ø.Ø          5.Ø      +Ø4 5.9266 -Ø8 1.Ø      +Ø5 1.12Ø4 -Ø75ØØ1 5 22 4713
 -  -    -   -  -  -    -  -  -    -  -  -    -  -  -    -  -  -   -  -  -  -  -
1.7Ø     +Ø7 6.1416 -12 1.8ØØ      +Ø7 2.7694 -12 1.9Ø     +Ø7 Ø.Ø            5ØØ1 5 22 4926
2.ØØ     +Ø7 Ø.Ø                                                      5ØØ1 5 22 4927
                                                                      5ØØ1 5  Ø 4928
```

FILE 5

Energy Distributions of Secondary Neutrons

LF = 1, tabulated distributions

$f(E \rightarrow E') = g(E \rightarrow E')$

FILE 5: Energy Distributions of Secondary Neutron (Structure of a Subsection)

LF = 3 (Discrete level excitation)

θ is the excitation energy of the level in the residual nucleus (positive value).

NR is the number of interpolation angles.

NP is the number of energy points at which, $p(E_i)$ is given.

NBT_i and INT_i is the interpolation scheme for $p(E_i)$.

E_i is the energy of point i.

$p(E_i)$ is the fractional probability for this law at energy, E_i.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a subsection for

LF = 3 (Discrete level excitation)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
T	θ	LT	LF=3	NR	NP	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
Ø.Ø          Ø.Ø4Ø4 +Ø6          Ø          3          1          29 233 5  4 1294
        29           2                                               233 5  4 1295
Ø.Ø4Ø7 +Ø6 1.ØØ       Ø.Ø95  +Ø6 1.Ø       Ø.1Ø   +Ø6 Ø.98592     233 5  4 1296
Ø.12   +Ø6 Ø.975      Ø.14   +Ø6 Ø.93636   Ø.16   +Ø6 Ø.89787     233 5  4 1297
Ø.18   +Ø6 Ø.84168    Ø.2Ø   +Ø6 Ø.8Ø583   Ø.22   +Ø6 Ø.7514Ø     233 5  4 1298
Ø.24   +Ø6 Ø.7Ø459    Ø.26   +Ø6 Ø.65714   Ø.28   +Ø6 Ø.6Ø522     233 5  4 1299
Ø.3Ø   +Ø6 Ø.56752    Ø.35   +Ø6 Ø.47541   Ø.4Ø   +Ø6 Ø.4ØØ       233 5  4 13ØØ
Ø.45   +Ø6 Ø.33799    Ø.5Ø   +Ø6 Ø.28636   Ø.55   +Ø6 Ø.24478     233 5  4 13Ø1
Ø.6Ø   +Ø6 Ø.2Ø735    Ø.65   +Ø6 Ø.16115   Ø.7Ø   +Ø6 Ø.14648     233 5  4 13Ø2
Ø.75   +Ø6 Ø.12639    Ø.8Ø   +Ø6 Ø.1Ø676   Ø.85   +Ø6 Ø.Ø9189     233 5  4 13Ø3
Ø.9Ø   +Ø6 Ø.Ø7792    Ø.95   +Ø6 Ø.Ø6456   1.ØØ   +Ø6 Ø.Ø5432     233 5  4 13Ø4
1.ØØ   +Ø6 Ø.Ø        15.Ø   +Ø6 Ø.Ø                              233 5  4 13Ø5
```

FILE 5

Energy Distributions of Secondary Neutrons

LF = 3, discrete level excitation

$$f(E \rightarrow E') = \delta\left[E' - \frac{A^2 + 1}{(A+1)^2} E + \frac{A}{A+1} \theta\right]$$

A = AWR

θ = -Q, θ given in File 5.

FILE 5: Energy Distributions of Secondary Neutrons
(Structure of a Subsection)

LF = 5 (General evaporation spectrum)

U — is a constant that defines the upper energy limit for the secondary neutrons such that:

$$0 \leq E' \leq E - U.$$

NR — is the number of interpolation ranges.

NP — is the number of incident energy points at which fractional probabilities are given.

NE — is the number of incident energy points at which effective nuclear temperatures are given.

NF — is the number of points at which X_i and $g(X_i)$ are given.

$p(E_i)$ — is the fractional probability for this law (LF = 5) at incident energy, E_i.

$\theta(E_j)$ — is the effective nuclear temperature at incident energy, E_j.

X_i — is defined as $X = E'/\theta(E)$

$g(X_i)$ is the normalize probability at X_i.

NBT_i and INT_i — is the interpolation scheme for:
(1) p(E) between incident energies, E.
(2) θ (E) between incident energies, E.
(3) g(X) between given values of X.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a Subsection for

LF = 5 (general evaporation spectrum)

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	U	b*	b	LF=5	NR	NP	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
	--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1
	b	b	b	b	NR	NE	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$\theta(E_1)$	E_2	$\theta(E_2)$	E_3	$\theta(E_3)$	
	--	--	--	--	E_{NE}	$\theta(E_{NE})$	TAB1
	b	b	b	b	NR	NF	
(3)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	X_1	$g(X_1)$	X_2	$g(X_2)$	X_3	$g(X_3)$	
	--	--	--	--	X_{NF}	$g(X_{NF})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
-2,Ø    +Ø7                                   5          1               233 5 18 1457
           2            2                                                233 5 18 1458
1.Ø     -Ø5 Ø,89766     1,5     +Ø7 Ø,89766                              233 5 18 1459
                                                         1             2 233 5 18 1460
           2            2                                                233 5 18 1461
1.Ø     -Ø5 1,ØØ    +Ø6 1.5     +Ø7 1,ØØ    +Ø6                          233 5 18 1462
                                                         1          1953 233 5 18 1463
        1953            2                                                233 5 18 1464
Ø.Ø         Ø.Ø         5,Ø     -Ø4 1,9Ø96  -Ø8 7.5     -Ø4 2,3388  -Ø8  233 5 18 1465
1.1717  -Ø3 2,9233  -Ø8 1.1776  -Ø3 2,93Ø6  -Ø8 1,1835  -Ø3 2,9379  -Ø8  233 5 18 1466
  -      -    -     -   -     -   -    -     -    -     -    -     -     -   -   -    -
1.98Ø1  +Ø1 2,Ø921  -16 1,99Ø   +Ø1 1,8748  -16 2,ØØ    +Ø1 1,679   -16  233 5 18 2115
                                                                         233 5  Ø 2116
```

File 5

Energy Distributions of Secondary Neutrons

LF = 5, general evaporation spectrum

$$f(E \rightarrow E') = g(x)$$
$$x = E'/\theta(E)$$
$$0 \leq E' \leq E - U$$

$\theta(E)$ is tabulated as a function of incident energy.
$g(x)$ is tabulated as a function of x
U is given in File 5.

FILE 5: Energy Distributions of Secondary Neutrons
(Structure of a Subsection)

LF = 7 (Simple fission spectrum, Maxwellian)

U is a constant that defines the upper energy limit for the secondary neutrons such that:

$$0 \leq E' \leq E - U$$

NR is the number of interpolation ranges.

NP is the number of incident energy points at which $p(E_i)$ are given.

NE is the number of incident energy points at which $\theta(E_j)$ are given.

$p(E_i)$ is the fractional probability for this law (LF=7) at incident energy, E_i.

$\theta(E_j)$ is the effective nuclear temperature at incident energy, E_j.

NBT_i and INT_i is the interpolation scheme for:
(1) $p(E_i)$ between values given at incident energies, E_i.
(2) $\theta(E_j)$ between values given at incident energies, E_j.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a subsection for

LF = 7 (simple fission spectrum, Maxwellian)

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	U	b*	b	LF=7	NR	NP	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
	--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1
	b	b	b	b	NR	NE	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$\theta(E_1)$	E_2	$\theta(E_2)$	E_3	$\theta(E_3)$	
	--	--	--	--	E_{NE}	$\theta(E_{NE})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF=5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
-1.0     +08                                      7          1            21043 5 18  810
            2            2                                                 1043 5 18  811
1.0      -05 1.0           1.5       +07 1.0                               1043 5 18  812
                                                             1            21043 5 18  813
            2            5                                                 1043 5 18  814
1.0      -05 1.2955 +06 1.5         +07 1.4923 +06                         1043 5 18  815
```

FILE 5

Energy Distributions of Secondary Neutrons

LF = 7, simple fission spectrum, Maxwellian

$$f(E - E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)}$$

$$I = \theta^{3/2}\left[\frac{\sqrt{\pi}}{2}\,\mathrm{erf}\left(\sqrt{(E-U)/\theta}\right) - \sqrt{(E-U)/\theta}\; e^{-(E-U)/\theta}\right]$$

$0 \le E' \le E-U$

U is large and negative for fission reactions.

FILE 5: Energy Distributions of Secondary Neutrons
(Strucutre of a Subsection)

LF = 9 (Evaporation spectrum)

U is a constant that defines the upper limit for secondary neutrons such that

$$0 \le E' \le E - U$$

NR is the number of interpolation ranges.

NP is the number of incident energy points at which $p(E_i)$ is given.

NE is the number of incident energy points at which $\theta(E_j)$ is given.

$p(E_i)$ is the fractional probability for this law (LF = 9) at incident energy, E_i.

$\theta(E_j)$ is the effective nuclear temperature at incident energy, E_j.

NBT_i and INT_i is the interpolation scheme for:
(1) $p(E_i)$ between values given at incident energies, E_i.
(2) $\theta(E_j)$ between values given at incident energies, E_j.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a subsection for

LF = 9 (Evaporation spectrum)

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	U	b*	b	LF=9	NR	NP	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
	--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1
	b	b	b	b	NR	NE	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$\theta(E_1)$	E_2	$\theta(E_2)$	E_3	$\theta(E_3)$	
	--	--	--	--	E_{NE}	$\theta(E_{NE})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.2234 +04 2.32029+02                                  1          1043 5 91  817
4.4    +04                                   9         1         21043 5 91  818
          2          2                                            1043 5 91  819
0.9    +06 1.0          15.0   +06 1.0                            1043 5 91  820
                                                       1         21043 5 91  821
          2          5                                            1043 5 91  822
0.9    +06 1.507  +05 1.5      +07 8.257  +05                     1043 5 91  823
                                                                  1043 5  0  824
```

FILE 5

Energy Distribution of Secondary Neutrons

LF = 9, Evaporation spectrum

$$f(E \rightarrow E') = \frac{E' e^{-E'/\theta(E)}}{I}$$

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta} \left(1 + \frac{E-U}{\theta} \right) \right]$$

$$0 \leq E' \leq E-U$$

FILE 5: Energy Distributions of Secondary Neutrons
(Structure of a Subsection)

LF = 10 (Watt spectrum)

NR is the number of interpolation ranges.

NP is the number of incident energy points at which $p(E_i)$ is given.

$p(E_i)$ is the fractional probability for this law at incident energy, E_i.

NBT_i and INT_i is the interpolation scheme for p(E) between values given at E_i.

"a" and "b" are constants for the Watt spectrum.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a subsection for

LF = 9 (Evaporation spectrum)

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	U	b*	b	LF=9	NR	NP	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	
	--	--	--	--	E_{NP}	$p(E_{NP})$	TAB1
	b	b	b	b	NR	NE	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	E_1	$\theta(E_1)$	E_2	$\theta(E_2)$	E_3	$\theta(E_3)$	
	--	--	--	--	E_{NE}	$\theta(E_{NE})$	TAB1

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
9.2234 +04 2.32029+02                                   1     1043 5 91  817
4.4    +04                                    9         1    21043 5 91  818
          2          2                                        1043 5 91  819
0.9    +06 1.0         15.0    +06 1.0                        1043 5 91  820
                                                        1    21043 5 91  821
          2          5                                        1043 5 91  822
0.9    +06 1.507   +05 1.5     +07 8.257   +05                1043 5 91  823
                                                              1043 5  0  824
```

FILE 5

Energy Distribution of Secondary Neutrons

LF = 9, Evaporation spectrum

$$f(E \rightarrow E') = \frac{E' e^{-E'/\theta(E)}}{I}$$

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta} \left(1 + \frac{E-U}{\theta} \right) \right]$$

$$0 \leq E' \leq E-U$$

FILE 5: Energy Distributions of Secondary Neutrons
(Structure of a Subsection)

LF = 10 (Watt spectrum)

NR is the number of interpolation ranges.

NP is the number of incident energy points at which $p(E_i)$ is given.

$p(E_i)$ is the fractional probability for this law at incident energy, E_i.

NBT_i and INT_i is the interpolation scheme for p(E) between values given at E_i.

"a" and "b" are constants for the Watt spectrum.

BCD Card Image Format for
File 5

Energy Distributions of Secondary Neutrons
Structure of a subsection for
(Watt spectrum)

LF=10

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
b*	b	b	LF=10	NR	NP	
NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
E_1	$p(E_1)$	E_2	$p(E_2)$	E_3	$p(E_3)$	TAB1
--	--	--	--	E_{NP}	$p(E_{NP})$	
b	b	b	b	2	0	
"a"	"b"	b	b	b	b	LIST

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 5
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank


```
                                               1Ø          1          2 235 5 18  1Ø1
                                                                        235 5 18  1Ø2
          2           2
1.Ø      -Ø5 1.ØØ         1.5     +Ø7 1.ØØ                              235 5 18  1Ø3
                                                           2            235 5 18  1Ø4
1.Ø      +Ø6 2.Ø     -Ø6                                                235 5 18  1Ø5
```

FILE 5

Energy Distribution of Secondary Neutrons

LF = 10, Watt spectrum

$$f(E \rightarrow E') = \sqrt{4/\pi a^3 b}\ e^{-ab/4}\ e^{-E'/a}\ \sinh(\sqrt{bE'})$$

FILE 6: Energy-Angular Distributions
LTT = 1 (Legendre Coefficients given)

LCT is a flag to specify reference system used (use LCT = 1, Laboratory system).

LTT is a flag to specify representations used.
LTT = 1 (Legendre polynomial expansion)
LTT = 2 (tabulated distributions given)

NL is the order of the Legendre expansion for a particular data set (all distributions).

LA is the value of ℓ, (for the ℓth coefficient).

NK is the number of partial probability distributions used for this reaction type (used in the same manner as in File 5). NK = 1 is recommended.

LF is a flag to specify which distribution law is used (same as in File 5). LF = 1 is recommended.

$p_{\ell,k}(E)$ is the fractional probability that the kth distribution law can be used to describe the ℓth Legendre coefficient. If NK = 1, then $p_\ell(E) = 1.0$ for all incident neutron energies.

$P_\ell(E \rightarrow E')$ is the ℓth Legendre coefficient used to describe the angular distribution of neutrons scattered from E to E'.

NR, NP, NE, NF, E_{int}, E'_{int} same as defined for File 5.

BCD Card Image Format for
File 6

Energy Angular Distributions of Secondary Neutrons

LTT = 1 (Legendre polynomia representation used)
NK = 1 (one distribution law used to represent Legendre coefficients)
LF = 1 (arbitrary tabulated function used to represent Legendre coefficients)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b*	LTT = 1	b	b	HEAD
b	b	b	LCT = 1	NL	b	CØNT
b	b	LA = 0	b	NK = 1	b	CØNT-ℓ
T	b	LT	LF = 1	NR	NP	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E_1	$P_o(E_1)$	---	---	E_{NP}	$P_o(E_{NP})$	TAB1
b	b	b	b	NR	NE	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NR	NF	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E'_1	$P_o(E_1 \rightarrow E'_2)$	E'_2	$P_o(E_1 \rightarrow E'_2)$	---	---	
---	---	---	---	E'_{NF}	$P_o(E \rightarrow E'_{NF})$	TAB1
---	---	---	---	---	---	
T	E_{NE}	LT	b	NR	NF	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E'_1	$P_o(E_{NE} \rightarrow E'_1)$	---	---	---	---	
---	---	---	---	E'_{NF}	$P_o(E_{NE} \rightarrow E'_{NF})$	TAB1
(Repeat structure starting with CØNT-ℓ record for each ℓ value, ℓ = 1 to NL)						
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 6
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank

No sample data set.

FILE 6

Energy-Angular Distributions
LTT = 1, Legendre coefficients given

$$\int_0^{E'_{max}} dE' \int_{-1}^{1} p(E \rightarrow E', \mu)\, d\mu = 1.0$$

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \rightarrow E', \mu) = \frac{\sigma(E)}{2\pi}\, m\, p(E \rightarrow E', \mu)$$

$\sigma(E)$ given in File 3; m is the implied neutron multiplicity for the reaction.

$$p(E \rightarrow E', \mu) = \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} P_\ell(E \rightarrow E')\, P_\ell(\mu)$$

if NK = 1 (NK is the number of partial distributions), $P_\ell(E \rightarrow E')$ is the ℓth Legendre coefficient. Otherwise,

$$P_\ell(E \rightarrow E') = \sum_{k=1}^{NK} P_{\ell,k}(E)\, P_{\ell,k}(E \rightarrow E')$$

File 6: Energy-Angular Distributions
LTT = 2 (Tabulated distribution given)

LCT is a flag to specify reference system used (use LCT = 1, Laboratory system).

LTT is a flag to specify representation used.
LTT = 1 (Legendre polynomial expansion)
LTT = 2 (tabulated distribution given)

NA is the number of angles (cosine values) at which secondary neutron distributions are given.

μ is the cosine of the scattered angle.

NK is the number of partial probability distributions given for a particular cosine value (used in the same manner as in File 5).

LF is a flag that specifies which secondary energy distribution law is used for a particular partial probability distribution (see section 10.2 for definitions)

$p_k(E,\mu)$ is the fraction probability that the k^{th} distribution low can be used at energy E, and cosine μ. If NK = 1, then $p_k(E,\mu) = p(E,\mu) = 1.0$ for all energies.

$p(E \rightarrow E', \mu)$ is the energy distribution for neutrons scattered into an angle whose cosine is μ.

NR, NP, NE, μ_{int}, E_{int} have same definitions as in File 4.

E'_{int} has the same definition as in File 5.

BCD Card Image Format for File 6

Energy-Angular Distributions of Secondary Neutrons

LTT = 2 (Tabulated distributions given)
NK = 1 (one distribution law used)
LF = 1 (arbitrary tabulated function to represent secondary neutron distribution)

Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
ZA	AWR	b	LTT = 2	b	b	HEAD
b	b	b	LCT = 1	NR	NA	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	TAB2
b	μ_1	b	b	NK = 1	b	CØNT-μ
T	b	LT	LF = 1	NR	NP	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E_1	$p(E_1,\mu_1)$	E_2	$p(E_2,\mu_1)$	---	---	
---	---	---	---	E_{NP}	$p(E_{NP},\mu_1)$	TAB1
b	b	b	b	NR	NE	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	TAB2
T	E_1	LT	b	NR	NF	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E'_1	$p(E_2 \rightarrow E'_2,\mu_1)$	E'_2	$p(E_1 \rightarrow E'_2,\mu_1)$	---	---	
---	---	---	---	E'_{NF}	$p(E_1 \rightarrow E'_{NF},\mu_1)$	TAB1
T	E_2	LT	b	NR	NF	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E'_2	$p(E_2 \rightarrow E'_1,\mu_1)$	---	---	---	---	
---	---	---	---	E'_{NF}	$p(E_2 \rightarrow E'_{NF},\mu_1)$	TAB1
---	---	---	---	---	---	
T	E_{NE}	LT	b	NR	NF	
NBT_1	INT_1	---	---	NBT_{NR}	INT_{NR}	
E'_1	$p(E_{NE} \rightarrow E'_1,\mu_1)$	---	---	---	---	
---	---	---	---	E'_{NF}	$p(E_{NE} \rightarrow E'_{NF},\mu_1)$	TAB1
	(repeat structure for each μ value, starting at CØNT-μ record, μ = 2 to NA)					
b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 6
Field 9 (Cols 73-75) = MT
Field 10 (Cols 76-80) = Card sequence number
*b = blank

No example data set.

FILE 6

Energy-Angular Distributions
LTT = 2 (Tabulated Distribution)

$$\int_0^{E'_{max}} dE' \int_{-1}^{1} p(E \rightarrow E', \mu)\, d\mu = 1.0$$

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \rightarrow E', \mu) = \frac{\sigma(E)}{2\pi}\, m\, p(E \rightarrow E', \mu)$$

$\sigma(E)$ given in File 3; m is the implied neutron multiplicity for the reaction.

$p(E \rightarrow E', \mu)$ is given in File 6.

If NK > 1, then

$$p(E \rightarrow E', \mu) = \sum_{k}^{NK} p_k(E, \mu)\, p_k(E \rightarrow E', \mu)$$

and $p_k(E,\mu)$ plus $p_k(E \rightarrow E', \mu)$ are given in File 6.

FILE 7: Scattering Law Data

LAT is a flag indicating which temperature has been used to calculate α and β.
LAT = 0, the actual temperature was used.
LAT = 1, the constant, T_o = 0.0253eV, has been used.

NI is the total number of items in the list, NI = 6*(NS+1).

NS is the number of non-principle scattering atom types (for H_2O,NS=1).

$M_o\sigma_{fo}$ is the product of M_o (the number of principle atoms per molecule) times (the free atom scattering cross section for the principle atom).

$\mathcal{E}$ is the value of E/kT above which the static model of elastic scattering is adequate.

A_o the ratio of the mass of the principle atom to that of the neutron. This value should be the same as that used to calculate α, i.e.,

$$\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$$

a_n is a test indicating the type of analytic function to be used for this atom type:
a = 1.0, use a free gas scattering law.
a = 2.0, use a diffusive motion scattering law.

$M_n\sigma_{fn}$ is for the n^{th} atom type and has the same definition as $M_o\sigma_{fo}$ for the principle atom type.

A_n is the effective mass for the n^{th} atom type.

E_{max} is the upper energy limit for the use of $S_o(\alpha,\beta_i)$

NR is the number of interpolation ranges.

NB is the number of β values at which sets of $S(\alpha,\beta_i)$ will be given.

NP is the number of α values at which $S(\alpha_j,\beta_i)$ will be given tabulated.

β_i is the value of β at which sets of α and $S(\alpha,\beta_i)$ are given.

α_j is the value of α at which $S(\alpha_j,\beta_i)$ is given.

$S(\alpha_j,\beta_i)$ is the value of the scattering law for β_i and α_j.

NBT_i and INT_i is the interpolation scheme for:
(1) $S(\alpha,\beta)$ between values given at β:
(2) $S(\alpha,\beta)$ between values given at α_j.

BCD Card Image Format for File 7

Scattering Law Data

	Field 1	Field 2	Field 3	Field 4	Field 5	Field 6	Record Type
	ZA	AWR	b*	LAT	b	b	HEAD
	b	b	b	b	NI	NS	
	$M_o\sigma_{fo}$	ϵ	A_0	E_{max}	b	b	
	a_1	$M_1\sigma_{f1}$	A_1	b	b	b	
	a_2	$M_2\sigma_{f2}$	A_2	b	b	b	
	a_3	$M_3\sigma_{f3}$	A_3	b	b	b	
	--	--	--	--	--	--	
	a_{NS}	$M_{NS}\sigma_{fNS}$	A_{NS}	b	b	b	LIST
	b	b	b	b	NR	NB	
(1)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	TAB2
	T	β_1	LT	b	NR	NP	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	α_1	$S(\alpha_1,\beta_1)$	α_2	$S(\alpha_2,\beta_1)$	α_3	$S(\alpha_3,\beta_1)$	
	--	--	--	--	α_{NP}	$S(\alpha_{NP},\beta_1)$	TAB1
	T	β_2	LT	b	NR	NP	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	α_1	$S(\alpha_2,\beta_2)$	α_2	$S(\alpha_2,\beta_2)$	α_3	$S(\alpha_3,\beta_2)$	
	--	--	--	--	α_{NP}	$S(\alpha_{NP},\beta_2)$	TAB1
	--	--	--	--	--	--	
	--	--	--	--	--	--	
	T	β_{NB}	LT	b	NR	NP	
(2)	NBT_1	INT_1	--	--	NBT_{NR}	INT_{NR}	
	α_1	$S(\alpha_1,\beta_{NB})$	α_2	$S(\alpha_2,\beta_{NB})$	--	--	
	--	--	--	--	α_{NP}	$S(\alpha_{NP},\beta_{NB})$	TAB1
	b	b	b	b	b	b	SEND

Field 7 (Cols 67-70) = MAT
Field 8 (Cols 71-72) = MF= 7
Field 9 (Cols 73-75) = MT = 4 (except for SEND card)
Field 10 (Cols 76-80) = Card sequence number
*b = blank

```
2.5Ø        1.ØØ92                                        1             111Ø 2151  1Ø7
2.31Ø4 +Ø2                        6                      18            3111Ø 2151  1Ø8
Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         111Ø 2151  1Ø9
Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         111Ø 2151  11Ø
Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         Ø.Ø         111Ø 2151  111
                                  Ø                       1             111Ø 2151  112
2.5         Ø.Ø                                         876           73111Ø 2151  113
Ø.          Ø.          Ø.          Ø.          -2.79       3.7ØØ   -Ø1111Ø 2151  114
2.5726 -Ø3 -8.2336-Ø2 -2.79Ø        3.7Ø    -Ø1 1.6119 -Ø4 -6.8Ø78-Ø6111Ø 2151  115
Ø.          Ø.          Ø.          Ø.          1.8Ø   -Ø1 7.ØØ    -Ø2111Ø 2151  116
3.1983 -Ø1 -1.2945-Ø6 1.8ØØ    -Ø1 7.ØØØ   -Ø2 4.2644 -Ø7 -6.Ø92Ø-Ø8111Ø 2151  117
Ø.          Ø.          Ø.          Ø.          1.42Ø       3.5ØØ   -Ø1111Ø 2151  118
9.84Ø1 -Ø5 -5.1965-Ø5 1.42Ø        3.5ØØ   -Ø1 3.5486 -Ø6 -1.11Ø3-Ø5111Ø 2151  119
Ø.          Ø.          Ø.          Ø.          1.78Ø       1.2Ø    -Ø1111Ø 2151  12Ø
1.6998 -Ø4 -4.4167-Ø7 1.78Ø        1.2ØØ   -Ø1 3.9887 -Ø5 8.8334  -Ø6111Ø 2151  121
 -  -  -    -  -  -    -  -  -    -  -  -    -  -  -    -  -  -    -  -  -
Ø.          Ø.          Ø.          Ø.          6.43Ø  +Ø1 4.ØØØ   -Ø1111Ø 2151  258
3.4727 -Ø4 -3.3582-Ø5 6.43Ø    +Ø1 4.ØØØ   -Ø1 5.4Ø82 -Ø5 -2.4642-Ø5111Ø 2151  259
```

FILE 7

Scattering Law Data

$$\frac{d^2\sigma}{d\Omega\, dE'}(E \rightarrow E',\mu,T) = \sum_{n=0}^{NS} \frac{M_n \sigma_{bn}}{4\pi T} \sqrt{\frac{E'}{E}}\, e^{-\beta/2}\, S_n(\alpha,\beta,T)$$

$$\beta = (E' - E)/kT$$

$$\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$$

$$\sigma_{bn} = \sigma_{fn}(A_n+1)^2/A_n^2$$

APPENDIX O

Format Difference Between Version I and II

The following is a summary of the format differences between version I and version II ENDF/B data tapes. These changes came about from a series of format modifications that have been made during the past three years.

ENDF/B version I was released July 1968 and version II was released 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 CØNT record. These records are given immediately after the Hollirith information.

2) The format for specifying radioactive decay data (MT = 453) has been extensively modified.

3) The format for specifying fission product yield data (MT = 454) has been modified. The new formats 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 6.5 of this report for details.

File 2 Changes

1) The test, LRF, indicating the type of resolved resonance formula used, have 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 where resolved and/or unresolved are not given, File 2 will contain the effective scattering radius. See Section 7.1 for details.

3) The test, LIS, has been removed. This means that the elastical scattering cross section always must be calculated using the resolved or unresolved resonance parameters.

4) The constant, "C" (a constant 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. "C" is to be calculated by using

$$C = 2.196771 \times 10^{-3}\left(\frac{AWRI}{AWRI + 1.0}\right) * AP$$

where AP is the effective scattering radius. This change effects both the resolved and unresolved resonance region.

5) A new option for specifying unresolved resonance parameters has been added. This new option is indicated by the test LRF = 2. This means that all average resonance parameters (level spacing, the width of an unspecified competitive reaction, the reduced neutrons width, the radiation width, and the fission width) 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 (positive Q values) or required for (negative Q values) a reaction. Also the threshold energy 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 have been increased from 2000 to 5000.

3) An initial state indicator has been added to the HEAD record. This will allow an inclusion of cross section data for metastable states; therefore 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:

$$f(E \rightarrow E') = \delta \left[E' - \frac{A^2 + 1}{(A + 1)^2} E + \frac{A}{A + 1} \theta \right]$$

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 the cases where LF = 5, 7, or 9. A value, U, replaces T. U was introduced to define the proper upper limit for the seondary neutron energy distributions such that

$$0 \leq E' \leq E - U$$

where E', E, and U are given in the laboratory system. Further, the

normalization constant for LF = 7 and LF = 9 have been re-defined to account for the use of U.

3) The use of LF = 2, 4, 6, and 8 have been deleted.

All Files

1) Certain reaction type numbers (MT numbers) have been changed. These are (see Appendix B for definitions):

Old MT Number	New MT Number	
5	51	
6	52	
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.