IAEA-NDS- 74

Rev. 0

INTERNATIONAL ATOMIC ENERGY AGENCY

NUCLEAR DATA SERVICES

DOCUMENTATION SERIES OF THE IAEA NUCLEAR DATA SECTION

XA9950019

Data Formats and Procedures

for the Evaluated Nuclear Data Format

ENDF-IV

Reprint of the report

BNL-NCS-50496 (ENDF-102), revised by
D. Garber, C. Dunford, S. Pearlstein, Oct. 1975

Reprint May 1984

30-21

IAEA NUCLEAR DATA SECTION, P.O. BOX 100, A-1400 VIENNA

BNL-NCS-50496
(ENDF 102)
(General Reactor Technology - TID-4500)

ENDF-102

DATA FORMATS AND PROCEDURES FOR THE EVALUATED NUCLEAR DATA FILE, ENDF

Revised by D. GARBER, C. DUNFORD, AND S. PEARLSTEIN

October 1975

This publication brings under one cover
what was formerly Vol. I (neutron formats) and Vol. II (photon formats)

INFORMATION ANALYSIS CENTER REPORT

NATIONAL NEUTRON CROSS SECTION CENTER

BROOKHAVEN NATIONAL LABORATORY

ASSOCIATED UNIVERSITIES, INC.

UPTON, NEW YORK 11973

UNDER CONTRACT NO. E(30-1)-16 WITH THE

UNITED STATES ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

This publication is based on previous reports:

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

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

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

It brings under one cover both the neutron and photon formats.

Printed in the United States of America
Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22161
Price: Domestic $12.75; Foreign $15.25;
Microfiche $2.25

March 1976 500 copies

DATA FORMATS AND PROCEDURES

FOR THE ENDF NEUTRON CROSS SECTION LIBRARY

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

DATA FORMATS AND PROCEDURES

0. ENDF/B - IV PREFACE

These revisions to Data Formats and Procedures for the ENDF Neutron Cross Section Library, ENDF-102, pertain to the latest version of ENDF/B-IV. The descriptions of the formats have been brought up to date and important procedural matters have been explained. Other explanations of formats will be made at a later time. Users of this manual who note deficiencies or have suggestions are encouraged to contact the National Neutron Cross Section Center at Brookhaven National Laboratory.

Three new appendices have been added. Appendix P contains a checklist of the most important rules for or misunderstandings about ENDF formats and procedures. Appendix Q summarizes the dimensions of important ENDF variables. Appendix R describes a new trial format approved for Files 62-76 that is now being tested for non-neutron reaction data.

Some of the data in the 1974 version of ENDF/B will contain files of data variances, sometimes called error files. Formats and procedures for the error files will be supplied at a later date.

The authors have been helped in these revisions by several people. Much material is based on an April 1973 draft memo entitled "Clarification of Existing Formats and Procedures" by M. K. Drake. The formats for radioactive decay data were largely constructed, clarified, and implemented by O. Ozer, C. W. Reich, and R. E. Schenter. Numerous people, in the U.S. and abroad, submitted suggestions and ideas, many of which were included in this edition.

1. INTRODUCTION

0.1.1. ENDF System

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

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

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

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

0.1.2. ENDF Documentation

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

This report is organized in the following manner. The first section describes the general features of the ENDF libraries, the relationship between ENDF and CSISRS, and the relationship between ENDF and its secondary libraries. Section 0.5 describes the standard formats used in all record types. An understanding of Section 0.5 will facilitate understanding the data formats given in Sections 1-15. BCD card-image formats are given in Appendix N.

The ENDF experience has prompted evaluation guidance to the evaluator in some instances to cope with familiar situations but cannot always be followed absolutely. Comments to the evaluators were primarily to improve data for shielding.

0.1.3. 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 partial data sets (partial in the sense that the data set may be, for example, an evaluation of the fission cross section for ^{235}U in the energy range 100 keV to 15 MeV only). This library may also contain several different evaluations of the cross sections for a particular nuclide. The ENDF/B library,

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

0.2. GENERAL FEATURES OF THE EVALUATED NUCLEAR DATA FILE

0.2.1. Evaluated Data

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

0.2.2. A and B Libraries

The demands on an evaluated data file vary according to the user's applications. Whether the user is interested in performing a reactor physics calcu-

lation 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. The ENDF/B library will contain only one representation (or interpretation) of the cross sections for a particular material at any given time. ENDF/A may contain several alternative sets of evaluated data for the materials on the ENDF/B library. The data sets that are contained on the ENDF/A library may or may not be complete (for the purposes of reactor physics or shielding calculations). The ENDF/A library is, in effect, a system for compiling evaluated data sets.

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

0.2.3. Choices of Data

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

0.2.4. Library Modes

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

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

0.2.5. Systematization of Data

The ordering of nuclear data for a particular material is described in Section 0.4.2. Integral cross section data (for example, the total cross section) may be represented by giving tabulated values of 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. Note that if resonance parameters are given, then contributions to a particular cross section from the resonance parameters must be added to the

integral cross sections to obtain the complete cross section. In other words, the integral cross sections and the resonance parameters are not redundant.

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

0.3 RELATIONSHIP OF THE ENDF TO OTHER DATA SYSTEMS

0.3.1. Experimental Data Libraries

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

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

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

(a) Listings of all or selected data sets.

(b) Magnetic tapes containing the requested data.

(c) Graphical displays containing selected data.

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

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

0.3.2. Processing Codes and Neutronics Calculations

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

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

0.4. GENERAL DESCRIPTION OF THE ENDF LIBRARY

The ENDF library is a collection of documented data evaluations stored in a defined computer readable format that can be used as the main input into

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

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

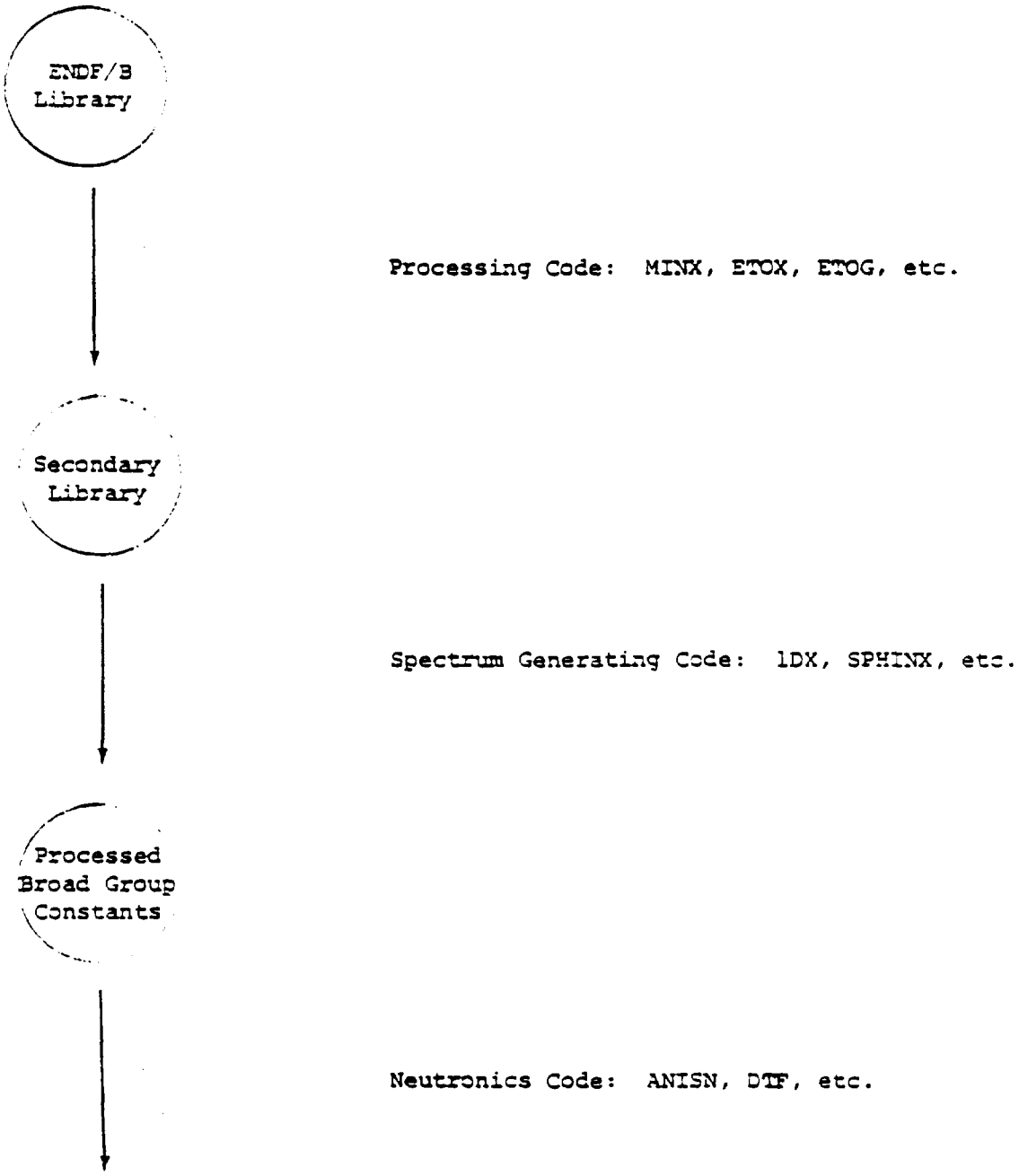


Fig. 0.3.2-1: Schematic of the flow of nuclear data from compilation to reactor calculation.

cross section processing programs. As such it has been designed with the processing programs in mind and requires some familiarity with the FORTRAN programming language. The ordering of data on the tape allows the use of segmented as well as ordinary programs.

Punched cards are a nuisance, particularly when required in vast numbers, as is 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.

0.4.1. Definitions and Conventions

A material is defined as either an isotope or a collection of isotopes. It may be a single nuclide, a natural element containing several isotopes, a molecule containing several elements, or a standard mixture of elements (such as 304 type stainless steel). Each evaluated set of cross sections for a material in the ENDF library is assigned a unique identification number. These numbers are designated by the symbol MAT and they range from 1 to 9999. Two different evaluated sets of cross sections for ^{235}U would have different MAT numbers even though they describe the cross sections for the same nuclide. A program that processes data from the ENDF library generally refers to the materials by their MAT number, but a (Z,A) designation is also given for each material and this value may be used.

When an evaluated set of cross sections for a material (in the ENDF format) is sent to the NNCSC, the Center assigns a MAT number to this material. This

number will never be assigned to another set of evaluated cross sections. If significant modifications are made to the data in this particular set, a new MAT number will be assigned. Material numbers from 1 to 999 are to be assigned by the user to data sets that he generates. As an example, consider the following sequence of events. User X evaluates a set of data for ^{235}U and assigns the material number 278 to this set. Within his installation the data set is always referred to as material 278. After checking and testing, the user feels that the data set is satisfactory and transmits it to the NNCSC. The Center adds the data set to its files and assigns it a MAT number of 1261. The Center then issues a newsletter describing data received and available for distribution. User Y reads the newsletter and requests material 1261 from the Center's files. Upon receipt of the data he adds it to his ENDF tape as material 1261 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.

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)
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)
33	Variance information (error files) Formats and procedures to come

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

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

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

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

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

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

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

0.4.2. Structure of an ENDF Data Tape

The structure of an ENDF tape is shown schematically in Figure 0.4.2-1. The structure of a card deck a BCD card image, tape or binary tape is exactly the same.

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

0.4.3. Representation of Data

The data in the ENDF library are given by providing parameters to known analytic functions (such as resonance formulae or secondary energy distribution laws), or are presented by tabulating the data in one [y(x)] or two [y(x,z)] dimensional arrays.

Consider how a simple function, y(x), which might be a cross section, $\sigma(E)$, is represented. y(x) is represented by a series of tabulated values, pairs of x and y(x), plus a method for interpolating between input values. The pairs are

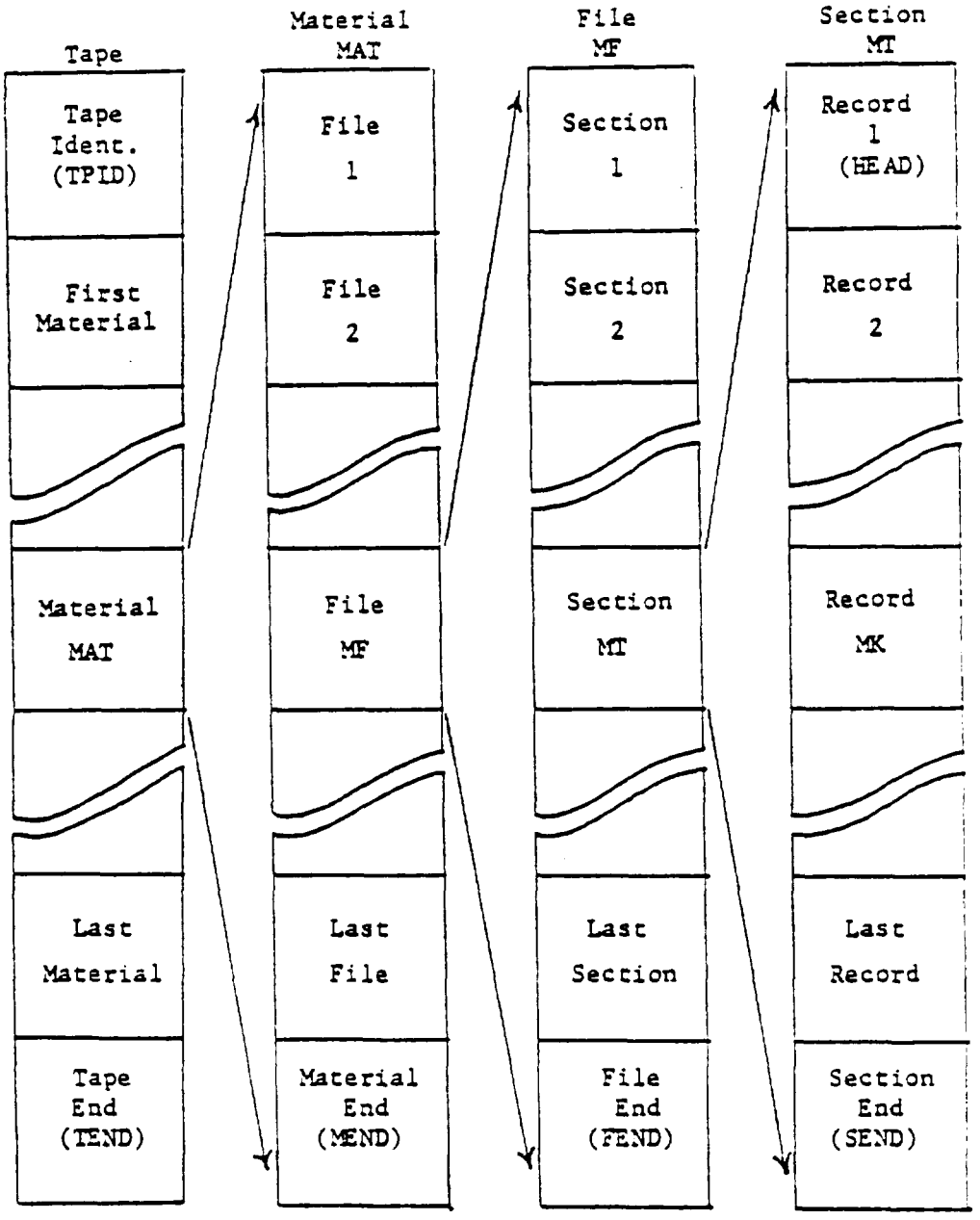


Figure 4.2.1. Arrangement of an ENDF Tape

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. 0.4.3-1 and the definitions below:

X(n) is the nth value of x.

Y(n) is the nth value of y.

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

INT(m) is the interpolation scheme identification number used in the mth range.

NBT(n) is the value of N separating the mth 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.

Note that where a function is discontinuous (for example, when resonance parameters are used to specify the cross section in one range), the value of X is

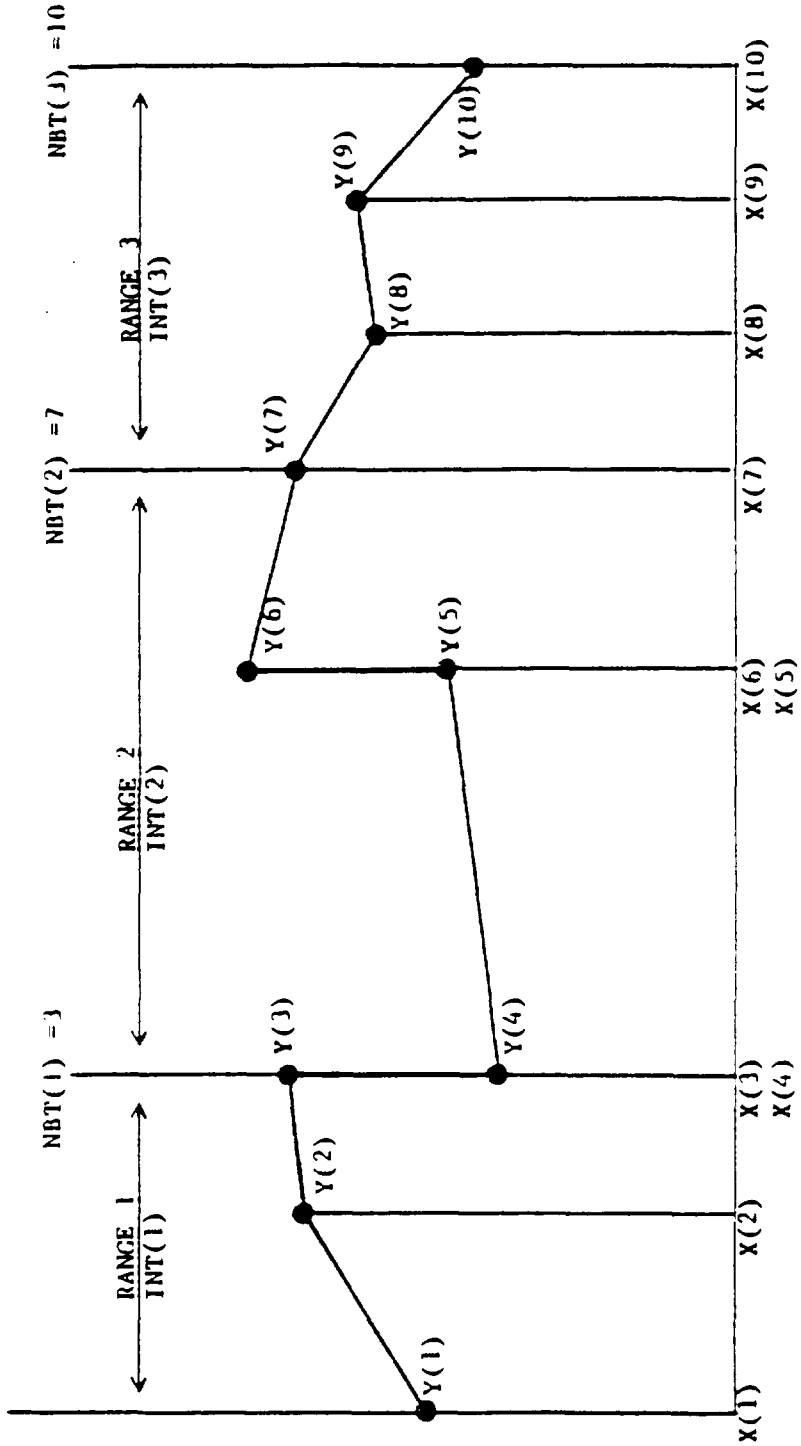


Figure 0.4.3-1. Tabulated one dimensional function illustrated for the case NP-10, NR-3

repeated and a pair (X,Y) given for each of the two values at the discontinuity (see Fig. 0.4.3-1).

Next consider a two-dimensional function of x and z. 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 breakpoint and interpolation table for interpolation between values of z.

0.5. GENERAL DESCRIPTION OF THE DATA FORMATS

0.5.1. Nomenclature

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

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

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

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

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

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

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

g) Brackets [] denote one record on a binary tape.

h) Brackets $\langle\ \rangle$ denote a group of records.

Several frequently used symbols are defined below.

MAT - Material number

MF - File number

MT - Reaction type number

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

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

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

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

T - Temperature

E - Energy

μ - Cosine of an angle

LT - Temperature dependence (see Appendix F).

0.5.2. Types of Binary Records

All records on an ENDF binary tape are one of four possible types, 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.

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

There are five special cases of a CØNT record, denoted by HEAD, SEND, FEND, MEND, and TEND. The HEAD record is the first 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 use only the first three numbers in the CØNT record, and they are used to signal the end of a section, file, material, and tape, respectively:

[MAT, MF, 0/0.0, 0.0; 0, 0; 0, 0]SEND

[MAT, 0 , 0/0.0, 0.0; 0, 0; 0, 0]FEND

[0 , 0 , 0/0.0, 0.0; 0, 0; 0, 0]MEND

[-1 , 0 , 0/0.0, 0.0; 0, 0; 0, 0]TEND

A FORTRAN IV statement to read any CØNT record from Tape LIB would be:

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

0.5.2.2. <u>LIST Records</u>

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

0.5.2.3. TAB1 Records

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

0.5.2.4. TAB2 Records

The last record type is the TAB 2 record, which is used to control the tabulation of a two-dimensional function, y(x,z). It specifies how many values of z are to be given and how to interpolate between successive value of z. Tabu-

lated 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

$$[MAT, MF, MT/C1, C2; L1, L2; NR, NZ/Z_{int}]TAB2$$

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

0.5.3. Card-Image (BCD) Formats

An alternative format is used when data are contained on punched cards or BCD card-image tapes. Basically the data are stored in the same order for this format as in the binary tape format. The major difference is the position of the three numbers MAT, MF, and MT. Also a card sequence number has been added to the card-image format. In general, more than one BCD card-image record will be required to contain the data in a binary record.

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

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

Consider a <u>TAB1 binary record</u> that was denoted by

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

This record would be punched on cards in the following way:

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

The FORTRAN 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)
```

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 <u>LIST record</u> 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
----	----	----	----	----	-----	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 FORMAT (2E11.4, 4I11, I4, I2, I3/6E11.4))
```

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

[MAT, MF, MT/ C1, C2; L1, L2; NWD, N2/ H_n]LIST

In this case the FORTRAN 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 FORMAT (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. Figure 0.5.3-1 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 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. 0.5.3-1 - A BCD Card-Image Record

1. FILE 1, GENERAL INFORMATION

File 1 is the first part of any set of evaluated cross section data for a material. Each material must have a File 1, which consists of one or more sections that contain neutron cross section information and other related nuclear data. File 1 provides a brief documentation of how the data were evaluated and a dictionary that summarizes the data files and cross section types given in Files 2, 3, 4, 5, etc. File 1 may also contain such basic nuclear data as the number of neutrons per fission (for fissile materials), 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 seven sections for fissile materials. Each section 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 (and all other files) is indicated by a FEND record. These record types are defined in detail in Section 0.5.2. 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,

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

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

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

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

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.

1.1 Descriptive Data and Dictionary (MT = 451)

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

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

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

(First BCD Card Image Record)

ZSYMA is a Hollerith representation of the material Z-chemical symbol - A with

Z right justified in col. 1 to 3

\- hyphen in col. 4

chemical symbol left justified in col. 5 and 6

- hyphen in col. 7

A right justified in 8 - 10 or blank

m, etc. indication of metastable state in col. 11

ALAB Mnemonic of originating laboratory(s) (left adjusted)

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

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

(Second BCD-Card Image Record)

REF reference 2-22

DDATE original distribution date (left adjusted DIST- followed by month-year as in EDATE

RDATE date and number of last revision REV1- followed by month-year as in EDATE

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 the sections to be found in the dictionary. Each section of this material is represented by a single card image that contains MF, MT, (reaction number), and NC (a count of

the number of cards in the section). NXC is the total number of sections for the complete material; i.e., it is equal to the sum of all the sections in the different files.

LDD is a flag to indicate whether induced reaction 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 $\leq$ 294). For binary tapes, NWD is the number of words containing the Hollerith information, and it is understood that 17 words are required for each card image (66 characters) and the format is (16A4, A2). (NWD $\leq$ 5000.)

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

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

MF_n is the MF of the nth section.

MT_n is the MT of the nth section.

NC_n is the number of BCD card images in a given section (the nth section). This card count does not include the SEND card. (Note that NC_1 = NXC + NWD + 2.)

1.1.1. Formats

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

[MAT, 1, 451/ZA , AWR, LPP, LFI, 0, NXC]HEAD

[MAT, 1, 451/0.0, 0.0, LDD, LFP, NWD, 0/ ZSYMA, ALAB, EDATE, AUTH (33 characters), REF (22 characters), DDATE, RDATE, b, b,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

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

1.1.2. Procedures

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

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

The flag LDD indicates whether induced reaction is given in MT = 453. Certain materials represent natural elements that contain more than one isotope or they represent molecules. For these cases radioactive nuclide production data may be ambiguous and are not allowed.

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

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

1.2. Number of Neutrons per Fission, (MT = 452)

If the material is fissionable (LFI = 1), then a section specifying the average total number of neutrons per fission, $\bar{\nu}$ (MT = 452) must be given. $\bar{\nu}$

is given as a function of incident neutron energy. The energy dependence of $\bar{\nu}$ may be found by tabulating $\bar{\nu}$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}(E)$,

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

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

1.2.1. Formats

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

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

LNU = 1, polynomial representation has been used;

LNU = 2, tabulated representation.

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

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

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

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

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

If LNU = 1, the structure of the section is

[MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0]HEAD LNU = 1

[MAT, 1, 452/ 0.0, 0.0, 0, 0 , NC, 0/C_1, C_2, ... C_{NC}]LIST

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

If LNU = 2, the structure of the section is

[MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0]HEAD LNU = 2

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

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

1.2.2. Procedures

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

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

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

1.3. Radioactive Nuclide Production (MT = 453)

When an evaluation represents the nuclear data for a single nuclide, then a section (MT = 453) may be given which specifies various radioactive product nuclides produced by neutron interactions. This section is given if LDD = 1 in MT = 451 (see section 1.1.1. of this report).

Data for the spontaneous decay of the ground state (and/or any excited state) of the original nuclide are given in MT = 457 (see section 1.7.1.).

Data are given in MT = 453 to specify the radioactive products resulting 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 reaction product data are given. (NS $\leq$ 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.)

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

*Although NS is limited to 5, the specific state number can be larger than 5 as long as the total number of states represented is no larger than 5.

RTYP is the designation of the reaction type leading to the described product nuclide state and is a floating-point equivalent of MT numbers (see Appendix B).

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 product nuclide (ZAP).

Q is the reaction Q-value (eV). Q = (rest mass of initial state - rest mass of final state.)

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

BR(N) is the branching ratio at the Nth 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.

1.3.1. Formats

The structure for this section always starts with a HEAD record and ends with a SEND record. The section is divided into subsections, each containing the data for a particular reaction (MT number). 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 - 1 >
[MAT, 1, 0/0.0, 0.0; 0, 0; 0, 0 ]SEND
```

There will be NS subsections.

The structure of a subsection is

```
[MAT, 1, 453/ZA, AWR; LIS, 0; NE, NPR/
            ES(1), ES(2),-------------
            -------------------, ES(NE)]LIST
[MAT, 1, 453/0.0, 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 ensure proper normalization. Linear-linear interpolation is implicit for branching ratios between the given energy points.

1.3.2. Procedures

1. Data should be given in MT = 453 for all isotopes for which radioactive products are produced in neutron interactions. Data should not be given for mixtures of elements, molecules, or elements that have more than one naturally occurring isotope.

2. All spontaneous decay modes of the ground state and important isomeric states should be described in section MT = 457. When branching ratios for the formation of particular final states are given as a function of incident neutron energy the information should be consistent with that in File 3. For example, the (n,α) reaction on ^{10}B that are given as energy-dependent branching ratios in File 1 should be consistent with the cross section information in File 3 for MT = 107, 780, and 781.

3. When data are given to specify the radioactive nuclides formed by neutron reactions (RTYP must be > 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 RTYP.

4. 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 the data given in various materials be consistent.

1.4. Fission Product Yield Data (MT = 454)

This section (MT = 454) specifies 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 product yield data is given for a par-

ticular incident neutron energy. Data sets should be given at sufficient incident energies to completely specify yield data for the energy range given for the fission cross section (as determined from Files 2 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 $\sim$ 2.0.

The fission products are specified by giving an excited state designation (FPS) and a (charge, mass) identifier (ZAFP). Thus, fission product nuclides are given, not mass chains. More than one (Z,A) may be used to represent the yields for a particular mass chain.

The following quantities are defined

NFP is the number of fission product nuclide states to be specified at each incident energy point (this is actually the number of sets of fission product identifiers - fission product yields). (NFP $\leq$ 1666.)

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 ith energy point. This array contains NFP sets of three parameters in the order ZAFP, FPS, YLD.

N1 is equal to 3*NFP, the number of items in the $C_n(E_i)$ array.

E_i is the incident neutron energy of the ith 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.

$\underline{I_i}$ is the interpolation scheme (see Appendix E) to be used between the E_{i-1} and E_i energy points.

1.4.1. 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), 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, I_3, 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.

1.4.2. Procedures

The data sets for fission product yields should be given over the same energy range as that in Files 2 and/or File 3 for the fission cross section. The yields are given as fractional values at each energy, and normally they will sum to $\sim$ 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 having direct fission yields. Data may be given for one or more fission product nuclide states to represent the yield for a particular mass chain. If yield data are given for more than one nuclide, the yield for the lowest Z (charge) nuclide state for a particular mass chain should be 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 faciliate interpolation of yield data between incident energy points. Also, a linear-linear interpolation scheme should be used.

1.5. Delayed Neutron Data (MT = 455)

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

The total number of delayed neutron precursors is given as a function of incident neutron energy. Two representations are provided to specify the energy

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

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

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

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

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

1.5.1. Formats

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

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

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 $\leq$ 4)

CD_m are the coefficients for the polynomial.

NR is the number of interpolation ranges used. (NR $\leq$ 200)

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

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

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

NNF is the number of precursor families considered.

λi is the decay constant (sec^{-1}) for the ith 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, $0/\lambda_1, \lambda_2, \ldots \lambda_{NNF}$]LIST

[MAT, 1, 455/ 0.0, 0.0, 0, 0, NCD, $0/CD_1, CD_2, \ldots CD_{NCD}$]LIST

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

The structure values of $\bar{\nu}_d$ 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, $0/\lambda_1, \lambda_2, \ldots \lambda_{NNF}$]LIST

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

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

1.5.2. Procedures

When the polynomial representation is used, the calculated values of $\bar{\nu}_d(E)$ may be used over any range in which the fission cross section has been given in

Files 2 and/or 3. When tabulated values of $\bar{\nu}_d(E)$ are specified, they should be given for the same energy range as that used to specify the fission cross section.

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

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

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

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

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

1.6.1. 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:

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

LNP = 1, polynomial representation has been used;

LNP = 2, tabulated representation.

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

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

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

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

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

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

[MAT, 1, 456/ZA, AWR, 0, LNP, 0, 0]HEAD LNP = 1

[MAT, 1, 456/0.0, 0.0, 0, 0, NCP, 0/CP_1, CP_2, ... CP_{NCP}]LIST

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

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

[MAT, 1, 456/ZA, AWR, 0, LNP, 0, 0]HEAD LNP = 2

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

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

1.6.2. Procedures

If a polynomial representation (LNP = 1) has been used to specify $\bar{\nu}_p(E)$, this representation is valid over any range in which the fission cross section is specified (as given in Files 2 and 3). When using a polynomial to fit $\bar{\nu}_p(E)$, the fit shall be limited to a third-degree polynomial (NCP = 4). If such a fit

does not reproduce the recommended values of $\bar{\nu}_p(E)$, a tabulated form (LNP = 2) should be used.

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

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

1.7. Radioactive Decay Data (MT = 457)

The spontaneous radioactive decay data are given in section 457.* This section is given for materials that are single nuclides in their ground state or an isomeric state (an isomeric state is defined as one having a half-life >0.1 sec.) The main purpose of MT = 457 is to describe absolutely the energy spectra resulting from radioactive decay and give average parameters useful for applications such as decay heat studies. The information in this section can be divided into three parts:

I. General information about the material

ZA = Designation of the original (radioactive) nuclide (=1000*Z + A)

LIS = Isomeric state flag for original nuclide (LIS = 0, ground state; LIS = 1, first isomeric state; etc.).

*The section MT = 453 is renamed Induced Reaction Branching Ratios.

$T_{1/2}$ = Half-life of the original nuclide (seconds).

$\Delta T_{1/2}$ = Uncertainty in the half-life (should be considered as one standard deviation).

NAV = Total number of decay modes for which average energies are given.

$\bar{E}_x, \Delta\bar{E}_x$ = Average decay energy (eV) of radiation of type x and its uncertainty (eV) for decay heat applications. The β, γ and α energies are given in that order, with space reserved for zero β or γ entries. All non-γ and non-α energies are presently included as β energy. The α energy includes the recoil nucleus energy.

II. Decay mode information for each mode of decay:

NDK = total number of decay modes given.

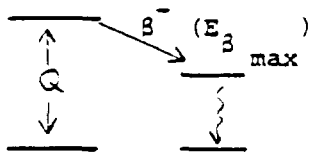
RTYP = Indicates the mode of decay.

Decay modes defined

Variable	Mode of decay	
0.0	γ	Gamma decay (not used for mode of decay)
1.0	β^-	Beta decay
2.0	β^+	Positron and/or electron capture decay
3.0	IT	Isomeric transition (in general, present only when the state being considered is an isomeric state)
4.0	α	Alpha decay
5.0	β^-,n	Neutron emission (generally given for delayed neutrons)
6.0	SF	Spontaneous fission

RFS = Isomeric state flag for daughter nuclide. (Fixed point number.)

Q = Total decay energy (eV) available in the corresponding decay process. (This is not necessarily the same as the maximum energy of the emitted radiation. In the case of an isomeric transition Q will be the difference in energy between the initial state and the isomeric state. For both β^+ and β^-, Q equals the energy corresponding to the mass difference between the initial and final atoms



ΔQ = Uncertainty in Q value (eV).

BR = Fraction of the decay which proceeds by the corresponding decay mode. (e.g., if only β^- occurs and no isomeric states in the daughter nucleus are excited, then BR = 1.0 for β^- decay.)

ΔBR = Uncertainty in BR (should be given as one standard deviation)

III. Resulting radiation spectra

STYP = Decay type (Use mode of decay variable list).

NSP = Total number of spectra. (NSP may be zero.)

E and ΔE = Energy (eV) or radiation produced ($E_{\beta-}$, $E_{\beta+}$, E_{γ}, etc.).

I and ΔI = Intensity of radiation produced (relative units).

ICC and ΔICC = Internal conversion coefficient.

F and ΔF = Normalization factor (absolute intensity/relative intensity).

NE = Total number of tabulated energies.

1.7.1. Formats

The structure of this section always starts with a HEAD record and ends with a SEND record. The section is divided into subsections as follows:

[MAT,1/457/	ZA	AWR	LIS	b	b	NSP	]	HEAD
[MAT,1/457/	$T_{1/2}$	$\Delta T_{1/2}$	b	b	2*NAV	NAV		
	$\bar{E}_\beta$	$\Delta\bar{E}_\beta$	$\bar{E}_\gamma$	$\Delta\bar{E}_\gamma$	$\bar{E}_\alpha$	$\Delta\bar{E}_\alpha$	]	LIST

[MAT,1/457/	ZA	AWR	b	b	6*NDK	NDK	/	
	$RTYP_1$	RFS_1	Q_1	ΔQ_1	BR_1	ΔBR_1		
	.							
	.							
	.							
	$RTYP_{NDK}$	RFS_{NDK}	Q_{NDK}	ΔQ_{NDK}	BR_{NDK}	ΔBR_{NDK}	]	LIST

[MAT,1,457/	STYP	b	b	b	6*(NE+1)	NE/	Repeat NSP times	
	F	ΔF	b	b	b	b	[omit if NSP=0]	
	E	ΔE	I	ΔI	ICC	ΔICC	]	LIST
[MAT,1,0/	b	b	b	b	b	b	]	SEND

1.7.2. Procedures

1. The initial state of the parent nucleus is designated by LIS, which equals 0 for the ground state and n for the nth isomeric state. Only isomeric states are included in the count of LIS. (In other files isomeric and non-isomeric states may be included in the count of levels.) Radioactive decay data need be given only for initial isomeric states with half-lives > 0.1 sec.

2. The average energy $\bar{E}$ and its uncertainty $\Delta\bar{E}$ is presently given for three types of radiation although the format does not limit the number. The average decay energy and its uncertainty for β, γ, and α radiation must be specified in that order, with space reserved for zero or unknown information. The average α energy also includes the recoil energy, but the α energy alone can be separated out by multiplying by the usual $M_R/(M_R + M_A)$ factor, where M_R and M_A are the masses of the recoil nucleus and α particle, respectively. The β radiation includes the contribution from β, positron, and conversion electron decay, and presently includes the average delayed neutron energy as well.

3. The symbol RTYP indicates the mode of decay as determined by the initial event. A nucleus undergoing β decay to an excited state of the daughter nucleus, which subsequently decays by γ emission, is in the β decay mode. In general, an RTYP = 0, indicating γ mode of decay, will not be used, since decay initiated by γ emission is classified as an isomeric transition requiring RTYP = 3. An isomeric state of the daughter nuclide resulting from the decay of parent nuclides is designated by RFS (floating point integer) following the procedures used for LIS. Q represents the total energy available in the decay process and is equal to the energy difference available between the initial and final states (both may be isomeric). The branching ratio BR for each decay mode is given as a fraction, and the sum over all decay modes must equal unity.

4. The energy spectra should be specified, if known. The decay type STYP should be specified using the RTYP variable list. γ spectra are described using STYP = 0. Relative intensities can be specified and normalized absolutely by multiplying by F. If absolute spectra are given, F must equal unity. The intensity I should be the total of the contributions from all decays lead-

ing to radiation within a particular decay type STYP having an energy $E \pm \Delta E$. The internal conversion coefficient should be the sum of all the partial conversion coefficients.

5. The specification of data uncertainties, an important quantity, is difficult to represent in a simple way. Although one standard deviation is desired, a number should be entered that at least indicates qualitatively how well the parameter is known.

6. One report detailing methods for specifying data in this section is ANCR-1157(1974), Radioactive-Nuclide Decay Data for ENDF/B, by C.W. Reich, R.G. Helmer, and M.H. Putnam.

2. FILE 2, RESONANCE PARAMETERS

2.1. General Description

File 2 contains data for both resolved and unresolved resonance parameters. It has only one section, which 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 corresponding contributions 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 in such cases 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 are obtained by specifying the parameters for each isotope in the material. The data for the various isotopes are ordered by increasing ZAI values (charge-isotopic mass number). The data for each isotope may be divided into several incident neutron energy ranges, and the data for the energy ranges are ordered by increasing energy. The energy ranges should not overlap; each will contain a different representation of the resonance parameters. Normally two energy ranges will be specified for each isotope. The first will contain resolved parameters, and the second, unresolved resonance parameters.

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

The allowed representations for the resolved resonance parameters are

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

LRF = 2, 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 basically the same, except for LRF = 4.

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

Only one representation is allowed for the unresolved resonance parameters, e.g., average single-level Breit-Wigner resonance parameters. However, several options exist for specifying the unresolved parameters. With 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 in Sections 2.2.1 (resolved) and 2.3.1 (unresolved). The formulae for

calculating cross sections for the various resonance region theories are given in Appendix D.

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

NIS is the number of isotopes in this material (NIS < 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 indicating whether average fission widths are given in the unresolved resonance region for this isotope:

LFW = 0, average fission widths are not given;

LFW = 1, average fission widths are given.

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

EL is the lower limit for an energy range.*

EH is the upper limit for an energy range.*

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

LRU = 0, means only effective scattering radius is given (LRF = 0, NLS = 0, LFW = 0 required)

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

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

If LRU = 2 (unresolved parameters), then

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

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

The general structure of a section is as follows:

[MAT, 2, 151/ ZA, AWR, 0, 0, NIS, 0]HEAD

[MAT, 2, 151/ ZAI, ABN, 0, LFW, NER, 0]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; those for each isotope, with a CØNT (isotope) record. The specifications for the subsections are given in Sections 2.2.1 and 2.3.1, below.

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

```
[MAT, 2, 151/ ZA,   AWR, 0,    0,    NIS, 0]HEAD    NIS = 1
[MAT, 2, 151/ ZAI,  ABN, 0,    LFW,  NER, 0]CØNT    LFW = 0, NER = 1
[MAT, 2, 151/ EL,   EH,  LRU,  LRF,  0,   0]CØNT    LRU = 0, LRF = 0
[MAT, 2, 151/ SPI,  AP,  0,    0,    NLS, 0]CØNT    NLS = 0
[MAT, 2,  0 / 0.0,  0.0, 0,    0,    0,   0]SEND
[MAT, 0,  0 / 0.0,  0.0, 0,    0,    0,   0]FEND
```

2.2. Resolved Resonance Parameters (LRU = 1)

2.2.1. 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 (see Appendix D for formulae):

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

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

AP is the 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. AP is defined in the relation $\sigma_{pot} = 4\Pi\ (AP)^2$.

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

NLS is the number of ℓ states 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 NLS ℓ-state 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 for use when LRF = 3 are defined 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. (AM = 0.0 required.)

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 $\Gamma\gamma$ 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]CONT
[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 by using the six constants in the following manner:

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

*Reserved for use in ENDF/A only.

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.

<u>AJ</u> is the floating-point value of J (the spin of the resonance).

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

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

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

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

<u>SPI</u> is the spin of the target nucleus.

<u>AWRI</u> is the ratio of the mass of a particular isotope to that of the neutron.

<u>AP</u> 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.

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

<u>AT_1, AT_2, AT_3, AT_4, BT_1, BT_2</u> 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 for the total cross section. Here and below, the subscript n denotes the nth level.

DEF_n is the resonance energy for the fission cross section.

DEC_n is the resonance energy for the radiative capture cross section.

DWT_n is the value of $\Gamma/2$, (ν), used for the total cross section.

DWF_n is the value of $\Gamma/2$, (ν), used for the fission cross section.

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

Note: $DET_n = DEF_n = DEC_n$ and $DWT_n = DWF_n = DWC_n$.

GRT_n is related to the symmetrical total cross section parameter.

GIT_n is related to the asymmetrical total cross section parameter.

GRF_n is the symmetrical fission parameter.

GIF_n is the asymmetrical fission parameter.

GRC_n is the symmetrical capture parameter.

GIC_n is the asymmetrical capture parameter.

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]CONT]
[MAT, 2, 151/AWRI,   0.0,   LI,   0,    6*NX,   NX/
          AT1,   AT2,   AT3,   AT4,   BT1,    BT2
          AF1,  ---------------------------,   BF2
          AC1,  ---------------------------,   BC2]LIST]
```

```
[MAT, 2, 151/0.0,    0.0,   L,      0,      NJS,      0]CØNT(ℓ)
[MAT, 2, 151/AJ,     AM,    0,      0,      12*NLJ,   NLJ/
            DET1,    DWT1,  GRT1,   GIT1,   DEF1,     DWF1,
            GRF1,    GIF1,  DEC1,   DWC1,   GRC1,     GIC1,
            DET2,    DWT2,  ------------------------------
            ----------------------------------,      GIC2,
            DET3,  --------------------------------------
            ----------------------------------------------
            ----------------------------------,      GICNLJ]LIST
```

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

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

2.2.2. Procedures

For certain resonances the value of ℓ is known but the resonance spin J is not. In such a case, the resonance spins J may be assigned to follow the level density law $\rho_J = \frac{1}{D_J} = \sim (2J+1)$ where ρ_J is the density of compound nucleus levels of spin J and D_J their spacing. The statistical weight factors g_J corresponding to resonances of spin J are such that $\sum_{J,S} g_J = (2\ell+1)$ where the summation is over the different spin states J and the channel spins S.

The resonance spins if not measured should not be set equal to the target nucleus 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 this energy range the cross sections must be obtained from the parameters given in another energy range and/or from data in File 3. Therefore, it is sometimes necessary to give parameters whose energies lie outside 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 at low energies, and resonances will usually be needed above EH to compensate the opposite, positive, bias at the high energy end.

For materials that contain more than one isotope, it is recommended that the lower energy limit of the resolved resonance region be the same for all isotopes. It is also recommended that the upper energy limit for the unresolved resonance range be the same for all isotopes. If resolved and/or unresolved resonance parameters are not given for all 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, the energy ranges must be contiguous and must not overlap. It is further required that the data for each isotope be divided into no more than two energy ranges, one for resolved and the other for unresolved resonance parameters.

With single-level Breit-Wigner parameters it is sometimes possible for negative cross sections to be calculated. Negative cross sections are not

physically possible and can be avoided by one of the following changes in representation:

1. Constraint of single-level parameters and scattering radius to produce non-negative cross sections.
2. Insertion of File 3 "background" to produce a File 2 + File 3 summation cross section everywhere positive.
3. Use of multilevel formalisms.

2.3. Unresolved Resonance Parameters (LRU = 2)

2.3.1. Formats

Only one representation of the unresolved resonance parameters is allowed (see Appendix D for pertinent formulae). However, several options are 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. This number 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 Nth 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 LFR = 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 between the cross-sections obtained from average resonance parameters (normally, INT = 1.)

GNO is the average reduced neutron width. It is energy dependent if LRU = 2.

GG is the average radiation width. It 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 (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, three different formats 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/
ES_1,	ES_2,	ES_3,	.,	.,	.,
.,	.,	.,	.,	ES_{NE}	]LIST
[MAT, 2, 151/AWRI,	0.0,	L,	0,	NJS,	0]CØNT(ℓ)
[MAT, 2, 151/0.0,	0.0,	L,	MUF,	NE+6,	0/
D,	AJ,	AMUN,	GNO,	GG,	0.0,
GF_1,	GF_2,	GF_3,	.,	.,	.,
.,	.,	.,	GF_{NE}		]LIST

In the above section, interpolation is assumed to be log-log.

If <u>LFW = 0 or 1</u> (does not depend on LFW).

<u>LRU = 2</u> (unresolved parameters).

<u>LRF = 2</u> (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
[MAT, 2, 151/AJ,	0.0,	INT,	0,	(6*NE)+6,	NE /
0.0,	0.0,	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

The LIST record is repeated until all 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.

2.3.2. Procedures

The number of degrees of freedom for the distribution of the competitive reaction width (AMUX) and radiation widths (AMUG) 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 other than scattering, capture, and fission.

Up to 250 energy points are allowed for giving energy-dependent average parameters. These data should allow average cross sections that show any gross structure in the reaction cross sections to be computed. 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 to use a consistent set of definitions in obtaining unresolved resonance parameters. These definitions are given in the Glossary (Appendix A) and the resonance region formulae (Appendix D). In particular, note 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)

$\rho = ka$.

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

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

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

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

in units of 10^{-12} cm. Note: A is usually approximated by AWRI (AWRI = $\frac{A}{m_n}$).

The Greebler-Hutchins* scheme for evaluating the width-fluctuation factor should be used in order to have uniformity between evaluators and users.

*P. Greebler and B. Hutchins. Physics of Fast and Intermediate Reactors. III, 121 (1962) International Atomic Energy Agency, Vienna, 1962.

3. FILE 3, NEUTRON CROSS SECTIONS

3.1. General Description

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

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

3.2. Formats

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

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

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

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

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

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

etc.

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

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

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

= 2, the final state is the excited state.

.

.

.

= 98, an unspecified range of final states.

= 99, all final states.

Q is the reaction Q-value (eV).

S is the temperature (°K). NOTE: If the LR flag is used, S becomes Q_1 for the reaction corresponding to LR.

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

LR is a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x). (See Section 3.24.4.)

NR is the number of energy ranges that have been given. A different interpolation scheme may be given for each range. (NR $\leq$ 200, but normally $\leq$ 20).

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

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

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

The structure of a section is

[MAT, 3, MT/ZA , AWR, LIS, LFS, 0 , 0]HEAD

[MAT, 3, MT/S , Q , LT, LR , NR, NP/E_{int}/σ(E)]TAB1

[MAT, 3, 0 /0.0, 0.0, 0, 0, 0 , 0]SEND

3.3 Procedures

3.2.1. Reaction Types to be Included

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

MT	Reaction
1	Total cross section
2	Elastic scattering cross section
4	Inelastic scattering cross section (total)
16	(n,2n) cross section
17	(n,3n) cross section
18	Fission cross section
51	Inelastic excitation cross section for the 1st level
52	" " " " " " 2nd level
.	
.	
.	

(continued on the next page)

MT	Reaction
90	Inelastic excitation cross section for the 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

3.2.2. General Procedures

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

Select the most appropriate MT number to represent the reactions. In many cases different MT numbers may be used to represent the same reaction mechanism, e.g., LI-6(n,t) and Li-6(n,α). This situation arises when the reaction produces multiple secondary particles or when the secondary particle and the residual nucleus are interchangeable. Many reactions of neutrons on light targets fall into this category. It is not possible to establish rigid ground

rules, but in general, the MT chosen includes the lightest nucleus in the exit channel.

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

3.2.3. Initial and Final States

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

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

For the (n,p), (n,d), (n,t), (n,^{3}He), and (n,α) reactions, the summation cross sections must be given in the MT = 103, 104, 105, 106, and 107 sections, respectively. Use of LFS = 99 in these cases is redundant. The cross sections to the ground and the first 17 discrete excited levels of the final nucleus must be given in the MT = 700 series. Use of LFS is redundant for a number of levels $\leq$17. For the (n,2n) reaction the summation cross section must be given

in the MT = 16 section. Use of LFS = 99 is redundant. The (n,2n) cross section to isomeric states should be given in the MT = 26 section, using the LFS flag to indicate the isomeric states (counting all states) designated.

3.2.4. Procedures for Specific Reactions

3.2.4.1. Index for Section 3.2.4.

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

3.2.4.1 Total Cross Sections

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

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

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

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

3.2.4.2. Elastic Scattering Cross Section

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

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

times very unrealistic) is placed in the elastic scattering cross section. Second, if the observed structure in the nonelastic cross section is improperly correlated with the structure in the total cross section, an unrealistic structure is generated in the elastic scattering cross section.

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

3.3.4.3. Nonelastic Cross Section (MT = 3)

The nonelastic cross section is not required unless any part of the photon production cross section data given in Files 12 and/or 13 uses MT = 3 to represent these data. In this case MT = 3 is required in File 3. If MT = 3 is given, then the set of points used to specify this cross section should be a union of the sets used for all its partials.

3.3.4.4. Inelastic Scattering Cross Sections

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

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

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

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

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

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

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

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

If a particular level, which has been left in an excited state, decays by emission of particles of more than one type, then several sections must be given in File 3. Consider the case in which an excited state decays by emission of a proton and an α particle. That part of the reaction that represents (n,n'α) would use LR = 22, and the other part would be given in the next section (next higher MT number) and would use LR = 28 (n,n'p). The angular distribution for the neutron would have to be given in two different MT numbers in File 4, even though they represent the same neutron. Competition for the de-excitation of a level should be considered only if it is at least 10% of the total de-excitation cross section for that level. The section must be ordered by increasing Q-values, i.e., increasing values of S in the TAB1 records.

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

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

2. If the (n,2n) cross section reaction produces an isomeric state, then in addition to the total (n,2n) cross section (given in MT = 16), the isomeric state production cross section can be given in MT = 26. Processing codes concerned only with the neutron cross sections can ignore the data given in MT = 26.

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

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

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

*This format is not restricted to ^9Be.

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

3.3.4.6. Fission Cross Sections

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

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

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

3.3.4.7. Absorption Cross Sections (MT = 27)

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

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

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

3.3.4.9. Neutron Disappearance Cross Section (MT - 101)

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

3.3.4.10. Target Destruction Cross Section (MT = 120)

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

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

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

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

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

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

For studies of radiation damage and/or long-lived activation in fission and fusion reactors, additional information about exit particles is natural. Exit protons will be used for illustration, although the same arguments hold for deuterons, tritons, and ^{3}He and ^{4}He particles. The MT = 700 series allows the cross sections and the energy and angular distributions for protons leaving the final nucleus in the ground through the 18th excited state to be described by using MT = 700 to MT = 718.

Data in sections MT = 700 to MT = 718 for (n,p_o) through (n,p_{18}) must add up to MT = 103. In some cases cross section information about the exit proton is needed that has already been included in the ENDF/B files. For example, the (n,n'p) cross section is usually found under MT = 51-91. It may be, however, that this proton energy distribution is more important for radiation damage studies than the energy of the neutron. In this case, a so-called redundant cross section that is already part of σ_{tot} is included in Section MT = 719 so that this section would refer to protons from both the (n,p) continua.

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

3.4. Relationship Between File 3 and Other Files

If File 2 (Resonance Data) contains resolved and/or unresolved resonance parameters, then in order to obtain the total cross section (MT=1) the radiative capture cross section (MT=102), fission cross section (MT=18), and elastic scattering cross section (MT=2), the cross sections calculated from these parameters must be added to the appropriate data given in File 3. Any contribution

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

Some materials will not have resonance parameters. However, they will have a scattering length, given in File 2, that can be used to calculate the potential elastic scattering cross section, which is then used to calculate resonance self-shielding effects in other materials. For these materials the elastic scattering cross section in File 3 must <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. They must always be given at the upper and lower energy limits of the resolved and unresolved resonance regions.

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

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

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

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

3.5. General Suggestions for Preparing Data for File 3

The limit on the number of energy points (NP) to be used to represent a particular cross section is 5000. The evaluator should not use more points than are necessary to represent the cross section accurately. Also, while the format limit of NR is 200, a limit of 20 is suggested for the number of interpolation regions (NR).

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

and continue up to 20 MeV. For nonthreshold reactions a cross section value must be given at 0.0253 eV.

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

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

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

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

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

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

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

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

4. FILE 4, ANGULAR DISTRIBUTIONS OF SECONDARY NEUTRONS

4.1. General Description

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

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

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

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

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

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

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;

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

ℓ = order of the Legendre polynomial;

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

f_ℓ = the ℓth Legendre polynomial coefficient and it is understood that $f_0 = 1.0$.

The angular distributions may be given in one of two representations, and in either the CM or LAB systems. In the first method the distributions are given by tabulating the normalized probability distribution, $p(\mu,E)$, as a function of incident neutron energy. Using the second method, the Legendre polynomial expansion coefficients, $f_\ell(E)$, 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 3 (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 describe 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 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 NM is the maximum order of the Legendre polynomial series used to describe any elastic angular distribution in this file. The transformation matrix is given as an array of numbers, V_K, where K = 1, ..., NK, and NK = $(NM + 1)^2$, and where $K = 1 + \ell + m\,(NM + 1)$. The values of K indicates how the $(\ell,m)^{th}$ element of the matrix may be found in array V_K. This means that the elements of the matrix $U_{\ell,m}$ or $U^{-1}_{\ell,m}$ are given <u>column-wise</u> in the array V_K:

---

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).

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

4.2. Formats

File 4 is divided into sections, each containing data for a particular reaction type (MT number) and ordered by increasing MT number. Each section always starts with a HEAD record and ends with a SEND record. If the section contains a description of the angular distributions for elastic scattering, the transformation matrix is given first (if present) and this is followed by the representation of the angular distributions.

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 LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NL is the highest order Legendre polynomial that is given at each energy ($NL \leq 20$).

NK is the number of elements in the transformation matrix ($NK \leq 441$). $NK = (NM + 1)^2$.

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system. NM should be an even number.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy ($NP \leq 101$).

Other commonly used variables are given in the Glossary (Appendix A).

The structure of a section depends on the values of LTT (representation used, $f_\ell(E)$ 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.

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

When LTT = 1 (angular distributions given in terms of Legendre polynomial coefficients) and LVT = 1, the structure of a section is

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0]HEAD LTT = 1, LVT = 1

[MAT, 4, MT/0.0, AWR, 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 °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified.

4.2.2. Legendre Polynomial Coefficients Given and the Transformation Matrix Not Given: LTT = 1 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.

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

If the angular distributions are given as tabulated probability distributions, LTT = 2, and a transformation matrix is given for elastic scattering, the structure of a section is

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0]HEAD LVT = 1, LTT = 2

[MAT, 4, MT/0.0, AWR, 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}/$p(\mu,E_1)$]TAB1

[MAT, 4, MT/T , E_2 , LT, 0, NR, NP/μ_{int}/$p(\mu,E_2)$]TAB1

[MAT, 4, MT/T , E_{NE}, LT, 0 , NR, NP/μ_{int}/$p(\mu,E_{NE})$]TAB1

[MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0]SEND

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

4.2.4. Tabulated Probability Distributions Given and Transformation Matrix Not Given: LTT = 2 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 nonelastically scattered neutrons when tabulated angular distributions are given.

4.3. 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 CM system (LCT = 2). It is recommended that the angular distributions of neutrons from nonelastic reactions (such as continuum inelastic, fission, etc.) be given as tabulated distributions, $p(\mu,E)$'s, and that they be in the Lab system. All angular distribution data should be given at the minimum number of incident energy points that will accurately describe the energy variation of the distributions.

When the angular distributions are represented as Legendre polynomial coefficients, certain procedures should be followed. Enough Legendre coefficients should be used to accurately represent the recommended angular distribution at a particular energy point and ensure that the interpolated distribution is everywhere positive. The number of coefficients (NL) may vary from energy point to energy point; in general, NL will increase with increasing incident energy. A linear-linear interpolation scheme (INT = 2) must be used to obtain coefficients at intermediate energies. This is required to ensure that the interpolated distribution is positive over the cosine interval from -1.0 to + 1.0; it is also required because some coefficients may be negative. In no case should NL exceed a value of 20. If more than 20 coefficients appear to be required to obtain a non-negative distribution, a constrained Legendre polynomial fit to the data should be given. NL = 1 is allowed at low energies to specify an isotropic angular distribution.

When angular distributions are represented as tabulated data, certain procedures should be followed. Sufficient angular points (cosine values) should be given to accurately represent the recommended distribution. The

number of angular points may vary from distribution to distribution. The cosine interval must be from -1.0 to +1.0. The interpolation scheme for $p(\mu,E)$ vs. μ should be log-linear (INT = 4), and that for $p(\mu,E)$ vs. E should be linear-linear (INT = 2).

Representation of angular distributions of neutrons for the thermal energy range presents a problem. Either free-atom or bound-atom scattering data may be given in File 4 for a material, but not both. For example, free-atom data for carbon appear in MAT = 1274 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 Lab to the CM system, or CM to LAB, a distribution should be generated and transformed point-wise to the desired frame of reference. The pointwise angular distributions can then be converted to Legendre polynomial coefficients in the new frame of reference.

The formats given above <u>do not allow both Legendre polynomial coefficients and tabulated data</u> to represent angular distributions for a given reaction type (MT number). If tabulated data are required to describe highly structured elastic scattering angular distributions at high energies, tabulated data must also be used to describe elastic scattering angular distributions at low energies.

4.4. Procedures for Specific Reactions

4.4.1. Elastic Scattering (MT = 2)

1. A transformation matrix should be given in File 4 for elastic scattering. If the angular distributions are given for the CM system, the matrix should

be for CM to LAB conversion. The parameter NM should be even, and it must be equal to or greater than ℓ_{max} used in any of the angular distributions (if Legendre coefficients are given). The parameter NK is equal to $(NM + 1)^2$.

2. Legendre polynomial representations should be used for elastic scattering angular distributions and discrete channel scattering and must be given in the CM system. When this representation is given, the maximum order of the polynomial for each incident energy should be even and ℓ_{max} must be ≤ 20.

3. Care must be exercised in selecting an incident energy mesh for certain light-to-medium mass materials. Here it is important to relate any known structure in the elastic scattering cross section to the energy dependent variations in the angular distributions. These two features of the cross sections cannot be analyzed independent of one another. Remember, processing codes operate on MT = 2 data given in Files 3 and 4. (Structure in the total cross section is not considered when generating energy transfer arrays.) It is better to maintain consistency in any structure effects between File 3 and File 4 data than to introduce structure in one file and ignore it in the other.

4. Consistency must be maintained between angular distribution data given for elastic and inelastic scattering. This applies not only to structural effects but also to how the distributions were obtained. Frequently, the evaluated elastic scattering angular distributions are based on experimental results that at times contain contributions from inelastic scattering to low-lying levels (which in turn may contain direct interaction effects). If inelastic contributions have been subtracted from the experimental angular distributions, this process must be done in a consistent manner. The same contributions must be subtracted from both the integrated elastic scattering and the angular distribution. Be sure that these contributions are included in

the inelastic scattering cross section (both integrated data and angular distributions). This is particularly important when the inelastic contributions are due to direct interaction, since the angular distributions are not isotropic or symmetric about 90° but they are generally forward peaked.

5. Do not use an excessive number of incident energy points for the angular distributions. The number used should be determined by the amount of variation in the angular distributions.

6. An incident energy point must be given at 10^{-5} eV. It is helpful, but not required, to include a point at 0.0253 eV. A point must be given at the highest energy point for which the angular distribution is isotropic. The highest incident energy point must be 20 MeV.

7. A relationship exists between the total cross section and the differential cross section at forward angles (Wick's limit or optical theorem).

$$\frac{d\sigma}{d\Omega}(0) \geq \sigma_w = (3.0560 \times 10^{-8} E_o) \frac{AWR^2}{(1 + AWR)^2} (\sigma_T)^2 \frac{\text{barns}}{\text{steradian}} ,$$

where E_o is in eV and σ_T in barns. Care should be taken to observe this inequality, especially at high energies.

4.4.2. Inelastic Scattering Cross Sections

1. Do not give angular distribution data for MT = 4.

2. Always give angular distribution data for any of the following if they are given in File 3: MT = 51, 52, 53, ..., 91.

3. Discrete level data should be given in the CM system, if possible. Some reactions, like MT = 91, must be given in the LAB system.

4. Discrete channel angular distributions (e.g., MT = 2, 51 - 90, 701 ...) should be given as Legendre coefficients in the CM system. Con-

tinuum angular distributions should be tabular in the LAB system.

5. Isotropic angular distributions should be given (at two energy points) unless the degree of anisotropy exceeds 5%. If any level excitation cross sections contain significant direct interaction contributions, angular distributions are very important.

6. Use the precautions outlined above when dealing with level excitation cross sections that contain a large amount of structure.

7. Do not overcomplicate the data files. Restrict the number of distributions to the minimum required to accurately represent the data.

8. Angular distributions for exit protons, etc., are given in the MT = 700 series, and for photons in File 14.

4.4.3. All Other Neutron Producing Reactions

Angular distribution data must be given for all neutron producing reactions. Make sure, giving these data, that they are realistic. Tabulated angular distributions are preferred.

5. FILE 5, ENERGY DISTRIBUTION OF SECONDARY NEUTRONS

5.1. General Description

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

Data will be given in File 5 for all reaction types that produce secondary neutrons, unless the secondary neutron energy distributions can be implicitly determined from data given in Files 3 and/or 4. No data will be given in File 5 for elastic scattering (MT = 2), since the secondary energy distributions can be obtained from the angular distributions in File 4. No data will be given for neutrons that result from excitation of discrete inelastic levels when data for these reactions are given in both File 3 and File 4 (MT = 51, 52, ..., 90). Data should be given in File 5 for MT = 91 (inelastic scattering to a continuum of levels), MT = 18 (fission), MT = 16 (n,2n), MT = 17 (n,3n), MT = 455 (delayed neutrons from fission), and certain other nonelastic reactions that produce secondary neutrons.

The energy distributions, $p(E \rightarrow E')$, are normalized so that

$$\int_{0}^{E'_{max}} p(E \rightarrow E')\, dE' = 1 ,$$

where E'_{max} is the maximum possible secondary neutron energy and its value depends on the incoming neutron energy E and the analytic representation of $p(E \rightarrow E')$. The secondary neutron energy E' is always expressed in the <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 <u>$\sigma(E)$ is the cross section as given in File 3</u> 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 E.

The partial energy distributions $f_k(E \rightarrow E')$ are represented by various analytical formulations. Each formulation is called an energy distribution law and has an identification number associated with it (LF number). The allowed energy distribution laws are given below.

<u>Secondary Energy Distribution Laws</u>

LF = 1, <u>Arbitrary tabulated function</u>:

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

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, <u>Excitation of discrete levels</u>:

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

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, <u>General evaporation spectrum</u>:

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

$\theta(E)$ is tabulated as a function of incident neutron energy, E;

$g(x)$ is tabulated as a function of x, $x = E'/\theta(E)$.

LF = 7, <u>Simple fission spectrum (Maxwellian)</u>:

$$f(E \rightarrow E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)} .$$

I is the normalization constant,

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

θ is tabulated as a function of energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy that $0 \leq E' \leq E - U$.

LF = 9, <u>Evaporation spectrum</u>:

$$f\ (E \rightarrow E') = \frac{E'}{I} e^{-E'/\theta} .$$

I is the normalization constant,

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta} \left(1 + \frac{E-U}{\theta}\right) \right] ;$$

θ is tabulated as a function of incident neutron energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy that $0 \leq E' \leq E - U$.

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 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 one. The partial energy distributions may all use the same energy distribution law (LF number) or they may use different laws.

5.2. Formats

Each section of File 5 contains the data for a particular reaction type (MT number), starts with a HEAD record, and ends with a SEND record. Each subsection contains the data for one partial energy distribution. The structure of a subsection depends on the value of LF (the energy distribution law).

The following quantities are defined

NK is the number of partial energy distributions. There will be one subsection for each partial distribution.

U is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ is a parameter used to describe the secondary energy distribution. 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.

If LF = 5, 7, or 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 in Section 5.1.).

$p_k(E_N)$ is the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ is the kth partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

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

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

NE is the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given. (NE $\leq$ 200.)

NF is the number of secondary energy points in a tabulation. (NF $\leq$ 1000.)

The structure of a section has the following form:

```
[MAT, 5, MT/ZA,  AWR, 0, 0, NK, 0]HEAD
      <subsection for k = 1>
      <subsection for k = 2>
       --------------------
      <subsection for k = NK>
[MAT, 5, 0 /0.0, 0.0, 0, 0, 0,  0]SEND
```

The structure of a subsection depends on the value of LF. Subsections should be ordered by increasing values of LF. For cases in which more than one subsection contains data using the same LF, these subsections should be ordered by increasing values of θ. The formats for the various values of LF are given below.

<u>LF = 1, Arbitrary tabulated function</u>

[MAT, 5, MT/T , 0.0 , LT , LF=1 , NR , NP/E_{int}/p(E)]TAB1

[MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int}]TAB2

[MAT, 5, MT/T , E_1 , LT , 0 , NR , NF/E'_{int}/

E'_1 , $g(E_1 \rightarrow E'_1)$, E'_2 , $g(E_1 \rightarrow E'_2)$, E'_3 , $g(E_1 \rightarrow E'_3)$,

---

----------, E'_{NF} , $g(E_1 \rightarrow E'_{NF})$]TAB1

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

E'_1 , $g(E_2 \rightarrow E'_1)$, E'_2 , $g(E_2 \rightarrow E'_2)$, E'_3 , $g(E_2 \rightarrow E'_3)$,

---

----------, E'_{NF} , $g(E_2 \rightarrow E'_{NF})$]TAB1

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

E'_1 , $g(E_{NE} \rightarrow E'_1)$, E'_2 , $g(E_{NE} \rightarrow E'_2)$, E'_3 , $g(E_{NE} \rightarrow E'_3)$,

---

----------, E'_{NF} , $g(E_{NE} \rightarrow E'_{NF})$]TAB1

Note that the incident energy mesh for $p_k(E)$ does not have to be the same as the E mesh used to specify the energy distributions. The interpolation scheme used between incident energy points, E, and between secondary energy points, E', should be linear-linear. T and LT refer to possible temperature (physical) dependence.

LF = 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 , $g(x_3)$ $\qquad x = \frac{E'}{\theta(E)}$

---------- , x_{NF}, $g(x_{NF})$]TAB1

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.

5.3. Procedures

As many as three different energy meshes may be required to describe the data in a subsection (one partial distribution). These are the incident energy mesh for $p_k(E)$, the incident energy mesh at which secondary distributions are given, $f_k(E \rightarrow E')$, and the secondary energy mesh for $f_k(E \rightarrow E')$. It is recommended that a linear-linear or a linear-log interpolation scheme be used for the first two energy meshes, and a linear-linear interpolation for the last energy mesh.

Double energy points must be given in the incident energy mesh whenever there is a discontinuity in any of the $p_k(E)$'s (this situation occurs fairly frequently). This energy mesh must also include threshold energy values for all reactions being described by the $p_k(E)$'s. Zero values for p_k must be given for energies below the threshold (if applicable).

Two nuclear temperatures may be given for the (n,2n) reaction. Each temperature, θ, may be given as a function of incident neutron energy. In this case $p_1(E) = p_2(E) = 0.5$. A similar procedure may be followed for the (n,3n) and other reactions.

A constant, U, is given for certain distribution laws (LF = 5, 7, or 9). Constant, U, is provided to define the proper upper limit for the secondary energy distribution so that $0 \leq E' \leq E - U$. The value of U depends on how the data are represented for a particular reaction type. Consider U for inelastic scattering:

<u>Case A</u> The total inelastic scattering cross section is described as a continuum. U is the threshold energy for exciting the lowest level in the residual nucleus.

<u>Case B</u> For the energy range considered, the first three levels are described explicitly (either in File 3, MT = 51, 52, and 53, or in File 5,

LF = 3), and the rest of the inelastic cross section is treated as a continuum. U is the threshold energy (known or estimated) for the fourth level in the residual nucleus.

If the reaction being described is fission, then U should be a large negative value ($U \sim -20.0 \times 10^6$ eV to $-30. \times 10^6$ eV). In this case neutrons can be born with energies much larger than the incident neutron energy. It is common practice to describe the inelastic cross section as the sum of excitation cross sections (for discrete levels) for neutron energies up to the point where level positions are no longer known. At this energy point, the total inelastic cross section is treated as a continuum. This practice can lead to erroneous secondary energy distributions for incident neutron energies just above the cutoff energy. It is recommended that the level excitation cross sections for the first several levels (e.g., 4 or 5 levels) be estimated for several MeV above the cutoff energy. The continuum portion of the inelastic cross section will be zero at the cutoff energy, and it will not become the total inelastic cross section until several MeV above the cutoff energy.

It is recommended that the cross sections for excitation of discrete inelastic levels be described in File 3 (MT = 51, 52, ..., etc.). The angular distributions for the neutrons resulting from these levels should be given in File 4 (same MT numbers). The secondary energy distributions for these neutrons can be obtained analytically from the data in Files 3 and 4. This procedure is the only way in which the energy distributions can be given for these neutrons. For inelastic scatter, the only data required in File 5 are for MT = 91 (continuum part).

5.4. Additional Procedures

5.4.1. General Comments

1. Do not give File 5 data for the discrete level excitation data given in File 3 as MT = 51, 52, ...,90. If MT = 91 is given in File 3, a section for MT = 91 must be given in File 5. A section must also be given in File 5 for all other neutron producing reactions. Energy distributions for exit protons, etc., are given in the MT = 700 series, and for photons, in File 15.

2. Care must be used in selecting the distribution law number (LF) to be used to represent the data. As a rule, use the simplest law that will accurately represent the data. Use only tabulated distributions (LF = 1) when the data cannot be represented by an evaporation spectrum (LF = 9) or a Maxwellian (LF = 7).

3. A section in File 5 must cover the same incident energy range as was used for the same MT number in File 3. The sum of the probabilities for all laws used must be equal to unity for all incident energy points.

4. If the incident neutron energy exceeds several MeV, pre-equilibrium particle emission can be important, as illustrated from high resolution neutron and proton spectra measurements and analysis of pulsed sphere experiments. In these cases either tabulated spectra or "mocked-up" levels can be constructed to supplement or replace simple evaporation spectra.

5.4.2. LF = 1 (Tabulated Distributions)

Use only tabulated distributions to represent complicated energy distributions. Use the minimum number of incident energy points and secondary neutron energy points to accurately represent the data. The integral over secondary neutron energies for each incident energy point must be unity to

within four significant figures. All interpolation schemes must be with linear-linear or log-linear (INT = 1, 2, or 3) to preserve probabilities upon interpolation. All secondary energy distributions must start and end with zero values for the distribution function $g(E \rightarrow E')$.

5.4.3. LF = 3 (Approximate Discrete Level Excitation)

The use of this law is discouraged. The use of the LR flag in File 3 eliminates most of the need to use this law.

5.4.4. LF = 5 (General Evaporation Spectrum)

This law is recommended for File 5 (MT = 455). Otherwise, LF = 1, 7, or 9 should be used.

5.4.5. LF = 7 (Maxwellian Spectrum)

This is the preferred law to use for the fission spectrum. A linear-linear interpolation scheme is preferred for specifying the nuclear temperature as a function of energy.

5.4.6. LF = 9 (Evaporation Spectrum)

An evaporation spectrum is preferred for most reactions. Care must be taken in describing the nuclear temperature near the threshold of a reaction. Nuclear temperatures that are too large can violate conservation of energy.

5.4.7. Selection of the Integration Constant, U

1. When LF = 5, 7, or 9 is used, an integration constant, U is required. This constant is used in defining the upper energy limit of secondary neutrons;

i.e., $E'_{max} = E_n - U$, where E_n is the incident neutron energy. U is a constant for the complete energy range covered by a subsection in File 5 and is given in the LAB system.

2. U is negative for fission reactions. The preferred value is -20 MeV.

3. In practice, U can be taken to be the threshold energy for the lowest level (known or estimated) that can be excited by the particular reaction within the incident energy range covered by the subsection.

4. The following three cases commonly occur in data files; procedures are given for obtaining U values.

Case A: The complete reaction is treated as a continuum.

$$U = -Q \frac{1 + AWR}{AWR},$$

where Q is the reaction Q-values, AWR is the ratio of the target mass to the neutron mass.

Case B: The reaction is described by excitation of three levels (in File 3 as MT = 51, 52, 53) and a continuum part where Q'_4 is the known or estimated Q-value for the fourth level.

Case C: The reaction is described by excitation of the first three levels (in File 3 as MT = 51, 52, 53) for neutron energies from the level thresholds up to 20 MeV, excitation of the next five levels (in File 3 as MT = 54, ..., 58) from their thresholds up to 8 MeV, and by a continuum part that starts at 5 MeV.

In this case two subsections should be used, one to describe the energy range from 5 to 8 MeV and another to describe the energy region from 8 to 20 MeV. In the first subsection (5 - 8 MeV),

$$U = -Q'_9 \frac{1 + AWR}{A},$$

and the second (8 - 20 MeV),

$$U = -Q'_4 \frac{1 + AWR}{A}.$$

5.4.8. Multiple Nuclear Temperatures

Certain reactions, such as (n,2n), may require specification of more than one nuclear temperature. $\theta(E)$ should be given for each neutron in the exit channels; this is done by using more than one subsection for a reaction. The U value is the same for all subsections. The upper energy limit is determined by the threshold energy and not by level densities in the residual nuclei.

5.4.9. Average Energy for a Distribution

The average energy of a secondary neutron distribution must be less than the available energy for the reaction:

$$E_{avail} = E + \frac{1 + AWR}{AWR} Q \ .$$

The mean energy should be calculated from the distribution at each value of E. This mean is analytic in the three cases given below.

LF	$\bar{E}$
3	$\frac{(AWR)^2 + 1}{(AWR + 1)^2} E - \frac{AWR}{AWR + 1} \theta$
7	$\frac{3}{2} \theta - \frac{\theta^{5/2}}{I} \left(\frac{E - U}{\theta}\right)^{3/2} e^{-(E - U)/\theta}$
9	$2\theta - \frac{\theta^3}{I} \left(\frac{E - U}{\theta}\right)^2 e^{-(E - U)/\theta}$

$$I = \int_0^{E'_{max}} f\,(E \rightarrow E')\, dE'$$

= the normalizing denominator (see 5.3). Thus $E_{avail}(E) >$ multiplicity * $\bar{E}$.

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

6.1. General Description

The use of File 6 is discouraged but the formats and procedures are presented here in case the File should be activated.

This file is provided to represent energy-angular distributions of secondary neutrons. Data are given here when it is not possible to provide accurate representation by using Files 4 and/or 5. This situation frequently arises when trying to provide a description of the secondary neutrons resulting from certain neutron reactions with fairly light nuclei.

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

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

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 a particular reaction as given in File 3 and/or File 2 for the same reaction type and m is the implied neutron multiplicity.

The angular part of the distribution may be specified in one of two ways: (1), secondary energy distributions may be tabulated at a set of secondary angles; (2), the probability distributions may be expressed as a Legendre polynomial expansion.

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

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

$$\int p_o \; (E \rightarrow E') \; dE' = 1 \; .$$

The incident neutron energies and secondary angles and energies must be given in the laboratory system.

If distributions $p(E \rightarrow E', \mu)$ are tabulated at a series of angles, a set of secondary angles (cosines of the scattered angles) is selected. This set is the same for all incident energy points and the data are ordered by increasing values of the cosine (-1.0 to +1.0). At each angular point, the probability distributions $p(E \rightarrow E', \mu)$ are given for a set of incident neutron energies; i.e., a subsection of data is given for each angle and the format of a subsection resembles the format of a section in File 5. The secondary energy distribution laws (LF numbers) defined in Section 5.2 are used in this file.

When the distributions are represented by Legendre polynomial expansion coefficients, then a subsection is given for each coefficient, $p_\ell(E \rightarrow E')$. Its format is similar to that for a section in File 5 (Section 5.2). The first subsection contains data for the zeroth coefficient, $p_o(E \rightarrow E')$. The subsections are then ordered by increasing ℓ-value of the coefficients.

The following quantities are defined.

<u>LCT</u> is a flag indicating which reference frame is used for both secondary angles and energies.

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

LCT = 2: (CM) <u>Do not use</u>.

LTT is a flag indicating which representation is used.

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

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

NL is the order of the expansion (when Legendre polynomial coefficients are given, LTT = 1).

LA is the value of ℓ (for the ℓth coefficient).

NA is the number of angles (cosines) at which the secondary distributions are given. (NA $\leq$ 101.)

μ is the cosine of the scattered angle.

NK is the number of partial probability distributions used for this reaction type (used as in File 5, Section 5.2.)

LF is a flag that indicates which secondary energy distribution law is used for a particular partial probability distribution. (See Section 5.2).

6.2. Formats

Each section of the file gives the data for a particular reaction type. The structure of a section depends on the value of LTT (representation used).

The structure of a section for LTT = 1 (Legendre polynomial expansion) is

[MAT, 6, MT/ZA , AWR, 0, LTT, 0, 0]HEAD LTT = 1

[MAT, 6, MT/0.0, 0.0, 0, LCT, NL, 0] CØNT

<Subsection for $p_0(E \rightarrow E')$>

<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 that of a section for secondary energy distributions in File 5 (Section 5.2) except that the SEND record is deleted (since the section in File 5 is used here as a subsection), and, the HEAD record is changed to read

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

The following is the structure for a typical section, where LTT = 1 (Legendre expansion coefficients given), NK = 1 (one partial probability distribution, and LF = 1 (an arbitrary tabulated distribution).

[MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0]HEAD LTT = 1

[MAT, 6, MT/0.0, 0.0, 0, LCT, NL, 0]CØNT

[MAT, 6, MT/0.0, 0.0, LA, 0, NK, 0]CØNT LA = 0, NK = 1

[MAT, 6, MT/T, 0.0, LT, LF, NR, NP/E_{int}/p_o(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_o(E_1 → E')]TAB1

[MAT, 6, MT/T, E_2, LT, 0, NR, NF/E'_{int}/p_o(E_2 → E')]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/p_o(E_{NE} → E')]TAB1

} Subsection for p_o(E → 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 → E')]TAB1

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

} Subsection for p_1(E → E')

<Subsection for p_2(E → E')>

<Subsection for p_{NL}(E → E')>

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

T and LT refer to possible temperature dependence (see Appendix F for format).

The structure of a section for LTT = 2 (tabulated distributions at a series of scattering angles) is

[MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0]HEAD LTT=2

[MAT, 6, MT/0.0, 0.0, 0, LCT, NR, NA/μ_{int}]TAB2

<Subsection for p(E → E', μ_1)>

<Subsection for p(E → E', μ_2)>

<Subsection for p(E → E', μ_{NA})>

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

Again the structure of a subsection is identical to that of a section for secondary energy distributions in File 5 except that the SEND record is deleted (since the section in File 5 is used here as a subsection), and the HEAD record is changed to read

[MAT, 6, MT/ 0.0, μ, 0, 0, NK, 0]CØNT

The structure of a typical section with LTT = 2 (tabulated distributions at a series of μ's), NK = 1 (one partial probability distribution) and LF = 1 (an arbitrary tabulated distributions) is

[MAT, 6, MT/ZA, AWR, 0, LTT, 0, 0]HEAD LTT = 2

[MAT, 6, MT/0.0, 0.0, 0, LCT, NR, NA/μ_{int}]TAB2

[MAT, 6, MT/0.0, μ_1 , 0, 0, NK, 0]CØNT NK = 1

[MAT, 6, MT/T, 0.0, LT, LF, NR, NP/E_{int}/p(E, 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 → E', μ_1)]TAB1

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

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p(E_{NE} \rightarrow E', \mu_1)$]TAB1

[MAT, 6, MT/0.0, μ_2, 0, 0, NK, 0]CØNT NK = 1

[MAT, 6, MT/T, 0.0, LT, LF, NR, NP/E_{int}/$p(E, \mu_2)$]TAB1 LF = 1

[MAT, 6, MT/0.0, 0.0, 0, 0, NR, NE/E_{int}]TAB2

[MAT, 6, MT/T, E_1, LT, 0, NR, NF/E'_{int}/$p(E_1 \rightarrow E', \mu_2)$]TAB1

[MAT, 6, MT/T, E_{NE}, LT, 0, NR, NF/E'_{int}/$p(E_{NE} \rightarrow E', \mu_2)$]TAB1

<Subsection for $p(E \rightarrow E', \mu_3)$>

<Subsection for $p(E \rightarrow E', \mu_{NA})$>

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

Again T and LT refer to possible temperature dependence.

6.3. Procedures

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

7. FILE 7, THERMAL NEUTRON SCATTERING LAW DATA

7.1. General Description

File 7 contains inelastic neutron scattering (MT = 4) data for the thermal neutron energy range for moderating materials (E < 5 eV). The data in this file must be combined with that in Files 2 and 4 (MT = 2) to obtain the total scattering cross sections for certain materials.

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

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

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 nth type in the molecule,

$\underline{T}$ is the moderator temperature (°K),

$\underline{E}$ is the incident neutron energy (eV),

$\underline{E'}$ is the secondary neutron energy (eV),

$\underline{\beta}$ is the energy transfer, $\beta = (E' - E)/kT$,

$\underline{\alpha}$ is the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$,

$\underline{A_n}$ is the mass of the nth type atom, A_o is the mass of the principal scattering atom in the molecule,

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

$$\sigma_{bn} = \sigma_{fn}\left(\frac{A_n + 1}{A_n}\right)^2$$

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

$\underline{k}$ is Boltzmann's contant, and

μ is the cosine of the scattering angle (in the lab system).

The data in File 7 for any particular material contain only the scattering law for the principal scatterer, $S_0(\alpha,\beta,T)$, i.e., the 0th atom in the molecule. These data are given as an arbitrary tabulated function. The scattering properties for the <u>other atom types</u> (n = 1, 2, ..., NS) are represented by <u>analytic functions</u>. Note that the scattering properties of all atoms in the molecule may be represented by analytic functions. In this case there is no principal scattering atom.

The constants required for the scattering law data and the analytic representations for the nonprincipal scattering atoms are given in an array, B(N), N = 1, 2..., NI, where NI = 6*(NS + 1). Six constants are required for each atom type (one BCD card-image record). The first six elements pertain to the principal scattering atom, n = 0. The elements of the array B(N) are defined as

B(1) = $M_o\sigma_{fo}$, the total free atom cross section for the principal scattering atom. If B(1) = 0.0, there is no principal scattering atom and the scattering properties for this material are 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_o, the ratio of the mass of the atom to that of the neutron that was used to compute α ($\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$).

B(4) = E_{max}, the upper energy limit for the constant σ_{f0} (upper energy limit in which $S_0(\alpha,\beta,T)$ may be used).

B(5), not used.

B(6), not used.

The next six constants specify the analytic functions to be used in describing the scattering properties of the first non-principal scattering atom, (n = 1); i.e., for H_2O, this atom would be oxygen if the principal atom was hydrogen.

B(7) = a_1, a test indicating the type of analytic function used for this atom type.

a_1 = 1.0, use a free gas scattering law.

a_1 = 2.0, use a diffusive motion scattering law.

B(8) = $M_1\sigma_{f1}$, the total free atom cross section for this atom type.

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

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

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

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

The next six constants, B(13) through B(18), are used to describe the second nonprincipal scattering atom (n = 2), if required. The constants are defined in the same way as for n = 1; e.g., B(13) is the same type of constant as B(7).

The scattering law is given by tabulating $S(\alpha,\beta)$ at a specific temperature (°K) or at a series of temperatures. Since scattering law data are generally given at more than one temperature, it is <u>extremely important</u> 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 $S(\alpha,\beta)$ 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 the equivalent depending on the units of Boltzmann's constant). A flag (LAT) is given for each material to indicate which temperature has been used in generating the $S(\alpha,\beta)$ data.

7.2. Formats

There is only one section in File 7, but the format varies slightly, depending on whether temperature-dependent data are given. The following quantities are defined:

LAT is a flag indicating which temperature has been used to compute α and β.

LAT = 0, the actual temperature was used.

LAT = 1, the constant T_o = 0.0253 eV has been used.

NS is the number of non-principal scattering atom types. For most moderating materials there will be (NS + 1) types of atoms in the molecule (NS $\leq$ 3).

NI is the total number of items in the B(N) list. NL = 6*(NS + 1).

B(N) is the list of constants. Definitions are given above (Section 7.1).

NR is the number of interpolation ranges for a particular parameter, either β or α.

NB is the total number of β values given.

NP is the number of α values given for each value of β for the first temperature described, NP is the number of pairs, α and $S(\alpha,\beta)$, given.

β_{int} and α_{int} are the interpolation schemes used (see Appendix E for interpolation formats).

The structure of a section is

[MAT, 7, MT/ZA, AWR, 0, LAT, 0, 0]HEAD

[MAT, 7, MT/0.0, 0.0, 0, 0, NI, NS/B(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 are completely specified by analytic functions [no principal scattering atom type, as indicated by B(1) = 0], tabulated values of $S_0(\alpha,\beta)$ are omitted and the TAB2 and TAB1 records are not given.

7.3. Procedures

Any material may contain a File 7 to describe inelastic scattering cross sections for the thermal neutron energy range. Except for moderating materials, a free gas scattering law is generally adequate.

File 7 is the most important part of the cross section data for moderator type materials. Moderator materials should also contain a File 3, and, as a minimum, the radiative capture cross section (MT = 102) should be given (as well as any other type of absorptive cross sections). If there are elastic scattering (i.e., coherent scattering) contributions to the total scattering cross section, then MT = 2 must be given in File 3. The data in File 3 shall at least cover the same energy range [constant, B(4)] as the scattering law

data, $S_0(\alpha,\beta)$. The scattering law data should cover the energy range in which thermal inelastic effects are important. The recommended energy range is 10^{-5} eV to 3.0 eV; however, it may not be possible to obtain scattering law data for every moderating material for this energy range. The β mesh for $S(\alpha,\beta)$ should be selected in such a manner as to accurately represent the scattering properties of the material with a minimum of β points. The α mesh at which $S(\alpha,\beta)$ is given should be the same for each value of β and for each temperature.

Note that the differential scattering cross section, as given in the equation in Section 7.1, represents the cross section for the complete molecule. The differential scattering cross section for a single atom of any component can be obtained by replacing $N_n\sigma_{bn}$ by σ_{bn}.

11. PHOTON PRODUCTION

Photon production data are divided into five distinct files.

File	Description
12	Multiplicities and transition probability arrays
13	Photon production cross sections
14	Photon angular distributions
15	Continuous photon energy spectra
16	Photon energy-angle distributions

With the exception of File 12, all the files are closely analogous to the corresponding neutron data files with the same number (modulo 10). The purpose of File 12 is to provide additional methods for representing the energy dependence of photon production cross sections. The allowed reaction type (MT) numbers are the same as those assigned for neutron reactions, Files 1 through 7. However, they may have somewhat different meanings for photon production that require additional explanation in some cases:

(1) MT = 3 should be used in Files 12 through 16 to represent composite cross sections, that is, photon production cross sections from more than one reaction type that have been lumped together.

(2) There is no apparent reason to have redundant or derived data for the photon production files, as is the case for the neutron files, i.e., MT = 3, 4, etc. Therefore, to avoid confusion, the join of all sections of Files 12 and 13 should represent the photon production, with each section being disjoint from all others.

(3) Let us consider how one might represent the inelastic γ-ray production data. The differential cross section for producing a γ-ray of energy E_γ resulting from the excitation of the m_o^{th} level of the

residual nucleus and the subsequent transition between two definite levels ($j \rightarrow i$), including the effects of cascading from the $m_o - j$ levels higher than j, is

$$\frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) = \delta(E_\gamma - \varepsilon_j + \varepsilon_i) A_{j,i}\, \sigma_{m_o}(E) \prod_{\ell=j}^{m_o-j} \sum_{m_\ell=j}^{m_{\ell-1}-1} TP_{m_{\ell-1},m_\ell} , \quad (1)$$

where

$\sigma_{m_o}(E)$ = neutron cross sections for exciting the m_o^{th} level with neutron energy E,

$\delta(E_\gamma - \varepsilon_j + \varepsilon_i)$ = delta function with ε_j, ε_i being energy levels of the residual nucleus,

$TP_{k,\ell}$ = probability of the residual nucleus having a transition to the ℓth level given that it was initially in the excited state corresponding to the kth level, and

$A_{k,\ell}$ = probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the kth to the ℓth level.

We are at once beset by the problem that no clear choice of ENDF representation in terms of section number is possible. The data may naturally be identified with both the m_o^{th} level and the jth level. To avoid this problem, we can sum Eq. (1) over m_o:

$$\frac{d\sigma_1}{dE_\gamma}(E_\gamma, E, i, j) = \sum_{m_o=j}^{N} \frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) , \quad (2)$$

where N is the highest level that can be excited by a neutron of incident energy E (i.e., $\varepsilon_N \leq \frac{AWR}{AWR + 1} E$). This gives a <u>de-excitation</u> cross section that can

single out a definite γ-ray transition and has the advantage when experimental data are to be represented. The de-excitation cross section is identified with the jth level. Alternatively, we can sum Eq. (1) over i and j:

$$\frac{d\sigma_2}{dE_\gamma}(E_\gamma,E,m_o) = \sum_{j=1}^{m_o} \sum_{i=0}^{j-1} \frac{d\sigma}{dE_\gamma}(E_\gamma,E,m_o,i,j) \ . \tag{3}$$

This gives an <u>excitation</u> cross section that can single out a definite excited state and has the advantage when calculated data are to be represented. The excitation cross section is identified with the m_o^{th} level. If Eq. (2) is summed over i and j, or if Eq. (3) is summed over m_o, then

$$\frac{d\sigma}{dE_\gamma}(E_\gamma,E) = \sum_{m_o=1}^{N} \frac{d\sigma_2}{dE_\gamma}(E_\gamma,E,m_o) \equiv \sum_{j=1}^{N} \sum_{i=0}^{j-1} \frac{d\sigma_1}{dE_\gamma}(E_\gamma,E,i,j) \ . \tag{4}$$

This gives a cross section for all possible excitations and transitions and thus corresponds to the total inelastic neutron cross section for discrete levels.

It is recommended that MT = 4 be used for the data represented by Eq. (4), as well as for the continuum. If, however, it is expedient or useful to use MT = 51 through 91, then one must use either the de-excitation cross sections of Eq. (2) or the excitation cross sections of Eq. (3), but not both. A restriction is imposed if the transition probability array option is used and if the entire neutron energy range is not covered by the known transition probabilities. Then, for MT = 51 through 90 in File 12 to be used for the remaining neutron energy range, a representation by <u>excitation</u> multiplicities must be used.

The integrated cross sections of File 13 are obtained by integrating Eqs. (1) through (4) over E_γ.

(4) The remarks in Item (3) apply for discrete rays from (n,pγ), (n,dγ), (n,tγ), (n, ^{3}Heγ), and (n,αγ) reactions, and the use of MT = 103, 104, 105, 106, and 107 is recommended for these cases.

12. FILE 12: MULTIPLICITIES AND TRANSITION PROBABILITY ARRAYS

File 12 can be used to represent the neutron energy dependence of photon production cross sections by means of either multiplicities or transition probability arrays. Both methods rely upon processing codes that use neutron cross sections from File 2 and/or File 3 to generate absolute photon production cross sections.

Multiplicities can be used to represent the cross sections of discrete photons and/or the integrated cross sections of continuous photon spectra. The MT numbers in File 12 designate the particular neutron cross sections (File 2 and/or File 3) to which the multiplicities are referred. The use of multiplicites is the recommended method of presenting (n,γ) capture ray cross sections, provided, of course, that the (n,γ) cross section is adequately represented in File 2 and/or File 3.

For well-established level decay schemes, the use of transition probability arrays offers a concise method for presenting (n,xγ) information. With this method, the actual decay scheme of the residual nucleus for a particular reaction (defined by MT number) is entered in File 12. This information can then be used by a processing code together with discrete level excitation cross sections from File 3 to calculate discrete ray production cross sections. This option cannot be used to represent the integrals of continuous photon spectra.

12.1. File 12 Format

Each section of File 12 gives information for a particular reaction type (MT number), either as multiplicities (LØ = 1) or as transition probability arrays (LØ = 2). Each section always starts with a HEAD record and ends with a SEND record.

12.1.1. Option 1 (LØ = 1): Multiplicities

The neutron energy dependence of photon production cross sections is represented by tabulating a set of neutron energy and multiplicity pairs $[E, y_k(E)]$ for each discrete photon and for the photon energy continuum.* The subscript k designates a particular discrete photon or a photon continuum, and the total number of such sets is represented by NK.

The multiplicity or yield $y_k(E)$ is defined by

$$y_k(E) = \frac{\sigma_k^\gamma(E)}{\sigma(E)} \quad \text{(photons)} ,$$

where E designates neutron energy and $\sigma(E)$ is the neutron cross section in File 2 and/or File 3 to which the multiplicity is referred (by the MT number). For discrete photons, $\sigma_k^\gamma(E)$ is the photon production cross section for the discrete photon designated by k. For photon continua, $\sigma_k^\gamma(E)$ is the cross section for the photon continuum integrated over photon energy. In the continuum case,

$$y_k(E) = \frac{\sigma_k^\gamma(E)}{\sigma(E)} = \frac{\int \frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)\,dE_\gamma}{\sigma(E)}$$

$$= \frac{\int \sigma(E)\; y_k(E_\gamma \leftarrow E)\; dE_\gamma}{\sigma(E)} = \int_0^{E_\gamma^{max}} y_k(E_\gamma \leftarrow E)\,dE_\gamma ,$$

*There should be no more than one energy continuum for each MT number used. If the decomposition of a continuum into several parts is desired, this can be accomplished in File 15.

where E_γ designates photon energy (eV), $\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)$ is the absolute photon energy distribution in barns/eV, and $y_k(E_\gamma \leftarrow E)$ is the relative energy distribution in photons/eV. The quantity $y_k(E_\gamma \leftarrow E)$ can be broken down further as

$$y_k(E_\gamma \leftarrow E) = y_k(E)\ f_k(E_\gamma \leftarrow E)\ ,$$

which results in the requirement that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)dE_\gamma = 1\ .$$

Any time a continuum representation is used for a given MT number in either File 12 or 13, then the normalized energy distribution $f_k(E_\gamma \leftarrow E)$ must be given in File 15 under the same MT number.

As a check quantity, the total yield

$$Y(E) = \sum_{k=1}^{NK} y_k(E) \quad \text{(photons)}$$

is also tabulated for each MT number if NK > 1.

The structure of a section for LØ = 1 is

[MAT, 12, MT/ZA, AWR; LØ=1, b; NK, b]HEAD

[MAT, 12, MT/ b, b; b, b; NR, NP/E_{int}/Y(E)]TAB1*

<subsection for k = 1>

<subsection for k = 2>

.

.

.

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

<subsection for k = NK>

[MAT, 12, 0/ b, b; b, b; b, b]SEND ,

and the structure of each subsection is

[MAT, 12, MT/EG_k, ES_k; LP, LF; NR, NP/E_{int}/y_k(E)]TAB1 ,

where

$\underline{ES_k}$ the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

$\underline{EG_k}$ the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k \equiv 0.0$ should be used.

$\underline{LP}$ indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons where the photon energy EG'_k is given by

$$EG'_k = EG_k + \frac{AWR}{AWR+1} E_n .$$

$\underline{LF}$ the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

= 2, a discrete photon energy.

12.1.2. Option 2 (LØ = 2): Transition Probability Arrays

With this option, the only data required are the level energies, de-excitation transition probabilities, and (where necessary) conditional photon emission probabilities. Given this information, the photon energies and their multiplicities can readily be calculated. Photon production cross sections can

then be computed for any given level from the excitation cross section in File 3, along with the transition probability array. Similarly, multiplicities and photon production cross sections can be constructed for the total cascade. For any given level, the transition and photon emission probability data given in the section are for photons originating at that level only; any further cascading is determined from the data for the lower levels.

Now define the following variables.

LG = 1, simple case (all transitions are γ emission).

= 2, complex case (internal conversion or other competing processes occur).

<u>NS</u> number of levels below the present one, including the ground state. (The present level is also uniquely defined by the MT number and by its energy level.)

<u>NT</u> number of transitions for which data are given in a list to follow (i.e., number of nonzero transition probabilities), NT $\leq$ NS.

<u>ES_i</u> energy of the ith level, i = 0,1,2... NS. ($ES_o \equiv 0.0$, the ground state.)

<u>TP_i</u> $TP_{NS,i}$, the probability of a direct transition from level NS to level i, i = 0,1,2... (NS-1).

<u>GP_i</u> $GP_{NS,i}$, the probability that, given a transition from level NS to level i, the transition is a photon transition (i.e., the conditional probability of photon emission).

<u>A_i</u> (TP_i) (GP_i).

Note that each level can be identified by its NS number. Then the energy of a photon from a transition to level i is given by $E_\gamma = ES_{NS} - ES_i$, and its multiplicity is given by $y(E_\gamma \leftarrow E) = (TP_i)$ (GP_i). It is implicitly assumed that the transition probability array is independent of incident neutron energy.

The structure of a section for LØ = 2 is

[MAT, 12, MT/ ZA, AWR; LØ=2, LG; NS, b]HEAD.

[MAT, 12, MT/ES_{NS}, b; LP, b; (LG+1)*NT, NT/B_i]LIST.

[MAT, 12 0/ b, b; b, b; b, b]SEND.

If LG = 1, the array B_i consists of NT doublets (ES_i,TP_i); if LG = 2, it consists of NT triplets (ES_i,TP_i,GP_i). Here the subscript i is a running index over the levels below the level for which the transition probability array is being given (i.e., below level NS). The doublets or triplets are given in decreasing magnitude of energy ES_i.

12.2. File 12 Procedures

1. Under Option 1, the subsections are given in decreasing magnitude of EG_k.

2. Under Option 1, the convention is that the subsection for the continuum photons, if present, is last. In this case, the last value of EG_k (EG_{NK}) is set equal to 0.0, and logical consistency with Procedure 1 is maintained.

3. Under Option 1, the values of EG_k should be consistent to within four significant figures with the corresponding EG_k values for the File 14 photon angular distributions. This allows processing and "physics" checking codes to match photon yields with the corresponding angular distributions.

4. Under Option 1, ES_k is the energy of the level from which the photon originates. If ES_k is unknown or not meaningful (as for the continuous photon spectrum), the value 0.0 should be entered.

5. If capture and fission resonance parameters are given in File 2, photon production for these reactions should be given by using Option 1 of

File 12, instead of using photon production cross sections in File 13. This is due to the voluminous data required to represent the resonance structure in File 13 and the difficulty of calculating multigroup photon production matrices from such data.

6. Under Option 1, the total yield table, Y(E), should exactly span the same energy range as the combined energy range of all the $y_k(E)$. Within that range,

$$Y(E) = \sum_{k=1}^{NK} y_k(E)$$

should hold within four significant figures.

7. The excitation cross sections for all the levels appearing in the transition probability arrays must, of course, be given in File 3.

8. The join of all sections, regardless of the option used, should represent the photon production data, with no redundancy. For example, MT = 4 cannot include any photons given elsewhere under MT = 51 through 91. Likewise, there can be no redundancy between Files 12 and 13.

9. If only one energy distribution is given under Option 1 (NK = 1), the TAB1 record for the Y(E) table is deleted to avoid repetitive entries.

10. Data should not be given in File 12 for reaction types that do not appear in Files 2 and/or 3.

11. Under Option 2, the level energies, ES_i, in the transition probability arrays are given in decreasing magnitude.

12. The MT numbers for which transition probability data are given should be for consecutive levels, beginning at the first level, with no embedded levels omitted.

13. The energies of photons arising from level transitions should be consistent within four significant figures with the corresponding EG_k values in File 14. Therefore, care must be taken to specify level energies to the appropriate number of significant figures.

14. Under Option 2, the sum of the transition probabilities (TP_i) over i should equal 1.0000 (that is, should be unity to within five significant figures).

15. The limit on the number of energy points in any tabulations of Y(E) or $y_k(E)$ is 1000. This is an upper limit that will rarely be approached in practice because yields are normally smoothly varying functions of incident neutron energy.

16. The limit on the number of interpolation regions is 10.

17. Tabulations of nonthreshold data should normally cover at least the energy range 10^{-5} eV $\leq E \leq 2 \times 10^7$ eV, where practical. Threshold data should be given from threshold energy up to 2×10^7 eV, where practical.

18. Transition Probability Arrays for (n,n'γ) photons.

a. The use of transition probability arrays (File 12, LØ = 2) is a convenient way to represent a portion of the rays produced by de-excitation of discrete levels populated by (n,n') and other reactions.

b. Several conditions must be met before this representation can be used. Level excitation cross sections (given in File 3 as MT = 51,...) must be given from threshold energies up to the same maximum energy (no exceptions). Decay properties of all n levels must be known. The information given in File 12 must be consistent with data given in File 3.

c. Usually, not all the conditions can be met. Part of the problem is the recommendation that level excitation cross sections for the first few levels be given for neutron energies up to 20 MeV. It is seldom that all level data can be given for neutron energies up to 15 MeV.

13. FILE 13: PHOTON PRODUCTION CROSS SECTIONS

The purpose of File 13 is the same as that of File 12; namely, it can be used to represent the neutron and photon energy dependence of photon production cross sections. In File 13, however, absolute cross sections in barns are tabulated, and there is no need to refer to the neutron files.

13.1. File 13 Format

As in File 12, each section in File 13 gives information for a particular reaction type (MT number). Each section always starts with a HEAD record and ends with a SEND record.

The representation of the energy dependence of the cross sections is accomplished by tabulating a set of neutron energy-cross section pairs $[E,\sigma_k^\gamma(E)]$ for each discrete photon and for the photon energy continuum. The subscript k designates a particular discrete photon or the photon continuum, and the total number of such sets is NK. For discrete photons, $\sigma_k^\gamma(E)$ is the photon production cross section (b) for the photon designated by k. For the photon continuum, $\sigma_k^\gamma(E)$ is the integrated (over photon energy) cross section for the photon continuum* designated by k. In the continuum case,

$$\sigma_k^\gamma(E) = \int_0^{E_\gamma^{max}} \frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)\, dE_\gamma \qquad \text{(b)} ,$$

where E_γ designates photon energy (eV), and $\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E)$ is the absolute photon energy distribution in b/eV. The energy distribution can be further broken down as

$$\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E) = \sigma_k^\gamma(E)\; f_k\;(E_\gamma \leftarrow E) ,$$

*There should be no more than one energy continuum for each MT number used. If the decomposition of a continuum into several parts is desired, this can be accomplished in File 15.

which obviously requires that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\, dE_\gamma = 1 \ .$$

Any time a continuum representation is used for a given MT number in File 13, the normalized energy distribution, $f_k(E_\gamma \leftarrow E)$, must be given in File 15 under the same MT number.

As a check quantity, the total photon production cross section,

$$\sigma_{TOT}^{\gamma}(E) = \sum_{k=1}^{NK} \sigma_k^{\gamma}(E) \qquad \text{(barns)} \ ,$$

is also tabulated for each MT number, unless only one subsection is present (i.e., NK = 1).

The structure of a section in File 13 is

[MAT, 13, MT/ZA, AWR; b, b; NK, b]HEAD

[MAT, 13, MT/ b, b; b, b; NR, NP/E_{int}/$\sigma_{TOT}^{\gamma}(E)$]TAB1*

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

[MAT, 13, 0/ b, b; b, b; b, b]SEND

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

and the structure of each subsection is

$$[MAT,\ 13,\ MT/EG_k,\ ES_k;\ LP,\ LF;\ NR,\ NP/E_{int}/\sigma_k^{\gamma}(E)]TAB1\ ,$$

where

<u>ES_k</u> the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced. $ES_k = 0.0$ should be used.

<u>EG_k</u> the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k = 0.0$ should be used.

<u>LP</u> Indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons, where the photon energy is given by $EG_k + \frac{AWR}{AWR + 1} E_n$.

<u>LF</u> the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

LF = 2, a discrete photon energy.

13.2. <u>File 13 Procedures</u>

1. The subsections are given in decreasing magnitude of EG_k.
2. The convention is that the subsection for the continuum photons, if present, is last. In this case, $EG_{NK} \equiv 0.0$.
3. The values of EG_k should be consistent to within four significant figures with the corresponding EG_k values in File 14.

4. ES_k is the energy of the level from which the photon originates, if known. Otherwise $ES_k \equiv 0.0$.

5. If capture and fission resonance parameters are given in File 2, the corresponding photon production should be given by using Option 1 of File 12, instead of using photon production cross sections.

6. The total photon production cross section table, $\sigma^{\gamma}_{TOT}(E)$, should exactly span the same energy range as the combined energy range of all the $\sigma^{\gamma}_{k}(E)$. Within that range,

$$\sigma^{\gamma}_{TOT}(E) = \sum_{k=1}^{NK} \sigma^{\gamma}_{k}(E)$$

should hold within four significant figures. If only one energy distribution is given, either discrete or continuous (NK = 1), the TAB1 record for the $\sigma^{\gamma}_{TOT}(E)$ is deleted.

7. The join of all sections in Files 12 and 13 combined should represent the photon production data with no redundancy. For example, MT = 4 cannot include any photons given elsewhere under MT = 51 through 91.

8. The limit on the number of energy points in a tabulation for any photon production subsection is 1000. This is an upper limit; in practice, the minimum number of points possible should be used. If there is extensive structure, the use of File 12 should be seriously considered, because yields are normally much smoother functions of incident neutron energy than cross sections.

9. The limit on the number of interpolation regions is 10.

10. Tabulations of nonthreshold data should normally cover at least the energy range 10^{-5} eV $\leq E \leq 2 \times 10^{7}$ eV, where practical. Threshold data should be given from threshold energy up to 2×10^{7} eV, where practical.

13.3. File 13 Preferred Representations

1. The recommended representation for (n,n'γ) reactions is photon production cross sections (File 13) using MT = 4. All discrete and continuum γ rays are given in a series of subsections.

2. Photon production cross sections resemble the frequently measured or reported results.

3. The use of MT = 4 eliminates confusion about whether the data represent an excitation or de-excitation cross section.

4. If for any reason MT = 51, 52 ... is used, it is understood that these data represent de-excitation and not excitation cross sections (see 3 above). MT = 51, 52, ... in File 3, of course, means excitation cross sections.

5. Combined use of MT = 4 and MT = 51, 52, ... is not allowed.

6. Above a certain energy point it probably will not be possible to separate the various components of the total γ production cross section. When this happens, it is preferred that the data be given as MT = 3.

7. All other reactions. Data for other reactions should be given as photon production cross sections (File 13) using the appropriate MT numbers. The same general rules outlined above should be used.

14. FILE 14: PHOTON ANGULAR DISTRIBUTIONS

The purpose of File 14 is to provide a means for representing the angular distributions of secondary photons produced in neutron interactions. Angular distributions should be given for each discrete photon and photon continuum appearing in Files 12 and 13, even if the distributions are isotropic.

The structure of File 14 is, with the exception of isotropic flag (LI), closely analogous to that of File 4. Angular distributions for a specific reaction type (MT number) are given for a series of incident neutron energies in order of increasing neutron energy. The energy range covered should be the same as that for the data given under the corresponding reaction type in File 12 or File 13. The data are given in ascending order of MT number.

The angular distributions are expressed as normalized probability distributions, that is,

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

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as a Legendre series expansion,

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} (\underline{\Omega},E)$$

$$= \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} a_{\ell}^{k}(E)\ P_{\ell}(\mu) \ ,$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^\gamma(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

ℓ = order of the Legendre polynomial.

$\frac{d\sigma_k^\gamma}{d\underline{\Omega}}$ = differential photon production cross section in barns/steradian.

$a_\ell^k(E)$ = the ℓth Legendre coefficient associated with the discrete photon or photon continuum specified by k. (It is understood that $a_o^k(E) \equiv 1.0$.)

$$a_\ell^k(E) = \int_{-1}^{1} p_k(\mu,E)\ P_\ell(\mu)d\mu\ .$$

Angular distributions may be given in File 14 by tabulating as a function of incident neutron energy either the normalized probability distribution function, $p_k(\mu,E)$, or the Legendre polynomial expansion coefficients, $a_\ell^k(E)$. Provision is made in the format for simple flags to denote isotropic angular distributions, either for a block of individual photons within a reaction type or for all photons within a reaction type taken as a group.

Note that File 14 assumes separability of the photon energy and angular distributions for the <u>continuum</u> spectrum. If this is not the case, File 16 (analogous to File 6) must be used instead of Files 14 and 15. (Since File 14 implicitly specifies an energy-angle distribution for discrete photons, File 16 is required only for the continuum spectrum.

14.1. File 14 Format

As usual, sections are ordered by increasing reaction type (MT) numbers. The following definitions are required.

LTT = 1, data are given as Legendre coefficients, where $a_o^k(E) \equiv 1.0$ is understood.

= 2, data are given as a tabulation.

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

= 1, distribution is isotropic for all photons from this reaction type.

NE number of neutron energy points given in a TAB2 record.

NI number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.

NL_i highest value of ℓ required at each neutron energy E_i.

a. LI = 1: Isotropic Distribution

If LI = 1, then all photons for the reaction type (MT) in question are assumed to be isotropic. This is a flag that the processing code can sense, and thus needless isotropic distribution data are not entered in the file. In this case, the section is composed of a HEAD card and a SEND card, as follows:

[MAT, 14, MT/ZA, AWR; LI=1, b; NK, b]HEAD

[MAT, 14, 0/ b, b; b, b; b, b]SEND .

b. LI = 0: Anisotropic Distribution

If LI = 0, there are two possible structures for a section, depending upon the value of LTT, but the section always starts with a HEAD record of the form

[MAT, 14, MT/ZA, AWR; LI=0, LTT; NK, NI]HEAD .

i. <u>LTT = 1: Legendre Coefficient Representation</u>

[MAT, 14, MT/ZA, AWR; LI=0, LTT=1; NK, NI]HEAD

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

[MAT, 14, 0/ b, b; b, b; b, b]SEND .

The structure of each record in the first block of NI subsections, which is for the NI isotropic photons, is

[MAT, 14, MT/EG_k, ES_k; b, b; b, b]CØNT .

There is just one CØNT record for each isotropic photon. (The set of CØNT records is empty if NI = 0.) The subsections are ordered in decreasing magnitude of EG_k (photon energy), and the continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons in decreasing magnitude of EG_k. The continuum, if present and anisotropic, appears last, with $EG_k \equiv 0.0$. The structure for the last NK-NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int}]TAB2

[MAT, 14, MT/ b, E_1; b, b; NL_1, b/$a_\ell^k(E_1)$]LIST

[MAT, 14, MT/ b, E_2; b, b; NL_2, b/$a_\ell^k(E_2)$]LIST

.

.

.

[MAT, 14, MT/ b, E_{NE}; b, b; NL_{NE},b/$a_\ell^k(E_{NE})$]LIST .

Note that lists of the $a_\ell^k(E)$ start at $\ell = 1$ because $a_o^k(E) \equiv 1.0$ is always understood.

ii. <u>LTT = 2: Tabulated Angular Distributions</u>

The structure of a section for LI = 0 and LTT = 2 is

[MAT, 14, MT/ZA, AWR; LI=0, LTT=2; NK, NI]HEAD

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

[MAT, 14, 0/ b, b; b, b; b, b]SEND .

The structure of the first block of NI subsections (where NI may be zero) is the same as for the case of a Legendre representation; i.e., it consists of one CØNT record for each of the NI isotropic photons in decreasing magnitude of EG_k. The continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

The structure of the first NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; b, b]CØNT .

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons, again in decreasing magnitude of EG_k, with the continuum, if present and anisotropic, appearing last, with $EG_k \equiv 0.0$. The structure of the last NK-NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int}]TAB2

[MAT, 14, MT/ b, E_1; b, b; NR, NP/μ_{int}/$P_k(\mu,E_1)$]TAB1

[MAT, 14, MT/ b, E_2; b, b; NR, NP/μ_{int}/$P_k(\mu,E_2)$]TAB1

.

.

.

[MAT, 14, MT/ b, E_{NE}; b, b; NR, NP/$\mu_{int}/p_k(\mu, E_{NE})$]TAB1 .

14.2. File 14 Procedures

1. The subsections are given in decreasing magnitude of EG_k within each of the isotropic and anisotropic blocks.

2. The convention is that the subsection for the continuous photon spectrum, if present, appears last in its block. In this case, $EG_{NK} \equiv 0.0$.

3. The values of EG_k should be consistent within four significant figures with the corresponding EG_k values in File 12 or 13. File 12, Option 2 (transition probability arrays), the values of EG_k are implicitly determined by the level energies.

4. ES_k is the energy of the level from which the photon originates, if known. Otherwise, $ES_k \equiv 0.0$ (as is always the case for the continuum).

5. Data should not appear in File 14 for photons that do not have production data given in File 12 or 13. Conversely, for every photon appearing in File 12 or 13 an angular distribution must be given in File 14. The neutron energy range for which the angular distributions are given should be the same as that for which the photon production data are given in File 12 or 13.

6. For LTT = 1 (Legendre coefficients), the value of NL should be the minimum number of coefficients that will reproduce the angular distribution with sufficient accuracy and be positive everywhere. In all cases, NL should be an even number, ≤ 20.

7. The TAB1 records for the $p_k(\mu,E_i)$ within a subsection are given in increasing order of neutron energy, E_i.

8. The tabulated probability functions, $p_k(\mu,E_i)$, should be normalized within four significant figures (to unity).

9. The interpolation scheme for $p_k(\mu,E)$ with respect to E must be linear-linear or log-linear (INT = 2 or 3) to preserve normality of the interpolated distributions. It is recommended that the interpolation in μ be linear-linear (INT = 2).

10. For LI = 1 (isotropic distribution), the parameter NK is the number of photons in that section and should be consistent with the NK values in Files 12 and 13. This parameter could be determined independently from Files 12 and 13, but it is useful in File 14 for the "physics" checking code.*

11. The minimum amount of data should be used that will accurately represent the angular distribution as a function of both μ and E.

12. If all photons for a reaction type (MT number) are isotropic, the LI = 1 flag should be used. The use of LI = 0 and NI = NK is strongly discouraged. Likewise, isotropic distributions should not be entered explicitly as a tabulation or as a Legendre expansion with $a_\ell^k(E) \equiv 0$, $\ell \geq 1$.

13. Angular distributions for photons must be given for all discrete and continuum photons. This can be done by specifying the data explicitly (by giving distributions) or implicitly by using a flag meaning that all photons for a particular reaction (MT number) are isotropic. Isotropic angular distributions should be specified unless the anisotropy is > 20%.

---

*Donald J. Dudziak and John F. Romero, "VIXEN, A Physical Consistency Checking Code for Photon Production Data in Revised ENDF Format," a Los Alamos Scientific Laboratory code LA-4739 (ENDF-155).

15. FILE 15: CONTINUOUS PHOTON ENERGY SPECTRA

File 15 provides a means for representing continuous energy distributions of secondary photons, expressed as normalized probability distributions. The energy distribution of each photon continuum occurring in Files 12 and 13 should be specified in File 15 over the same neutron energy range used in Files 12 and 13. Each section of File 15 gives the data for a particular reaction type (MT number) and the sections are ordered by increasing MT number.

The energy distributions, $f(E_\gamma \leftarrow E)$, are in units of eV^{-1} and are normalized so that

$$\int_0^{E_\gamma^{max}} f(E_\gamma \leftarrow E)\, dE_\gamma = 1 \quad ,$$

where E_γ^{max} is the maximum possible secondary photon energy and its value depends on the incoming neutron energy as well as the particular nuclei involved.* The energy distributions $f(E_\gamma \leftarrow E)$ can be broken down into the weighted sum of several different normalized distributions in the following manner:

$$f(E_\gamma \leftarrow E) = \sum_{j=1}^{NC} p_j(E)\; q_j(E_\gamma \leftarrow E) \qquad (eV)^{-1} \quad ,$$

where

$NC \equiv$ the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$,

$q_j(E_\gamma \leftarrow E) \equiv$ the jth normalized partial distribution in the units eV^{-1}, and

$p_j(E) \equiv$ the probability or weight given to the jth partial distribution, $q_j(E_\gamma \leftarrow E)$.

---

*Note that the subscript k used in describing Files 12 and 13 has been dropped from $f(E_\gamma \leftarrow E)$. This is done because only one energy continuum is allowed for each MT number, and the subscript k has no meaning in File 15. It is, in fact, the NK^{th} subsection in File 12 or 13 that contains the production data for the continuum.

The following norma ion condition is imposed.

$$\int_{o}^{E_{\gamma}^{max}} q_j(E_\gamma \; E_\gamma = 1 \quad .$$

Thus,

$$\sum_{j=1}^{NC} p_j(E)$$

The absolute ene tribution cross section, $\sigma^{\gamma}(E_\gamma \leftarrow E)$, can be constructed from the express

$$\sigma^{\gamma}(E_\gamma \leftarrow E)E) \; f(E_\gamma \leftarrow E) \qquad\qquad (b/eV),$$

where $\sigma^{\gamma}(E)$ is egrated cross section for the continuum given either directly in File through the combination of Files 2, 3, and 12.

The syst to represent continuous photon energy distributions in File 15 is similar used in File 5. At present, however, there is only one continuous dis on law activated for File 15, i.e.,

$$q_j(E_\gamma \leftarrow (E_\gamma \leftarrow E),$$

where $g(E_\gamma \leftarrow$ esents an arbitrary tabulated function. In the future, new laws (for exa he fission gamma-ray spectrum) may be added.

15.1. <u>File it</u> .

The st of a section is

[MMT/ZA, AWR; b, b; NC, b]HEAD

bsection for j = 1>

bsection for j = 2>

.

.

.

<subsection for j = NC>

[MAT, 15, 0/ b, b; b, b; b, b]SEND .

For LF = 1, the structure of a subsection is

[MAT, 15, MT/b, b; b, LF=1; NR, NP/E_{int}/p_j(E)]TAB1

[MAT, 15, MT/b, b; b, b; NR, NE/E_{int}]TAB2

[MAT, 15, MT/b, E_1; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ ← E_1)]TAB1

[MAT, 15, MT/b, E_2; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ ← E_2)]TAB1

.

.

.

[MAT, 15, MT/b, E_{NE}; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ ← E_{NE})]TAB1 .

Only one distribution law is presently available (tabulated secondary photon energy distribution). Therefore, formats for other laws remain to be defined, but their structures will probably closely parallel those in File 5 for LF = 5, 7, 9, and 10. When histogram representations are used (interpolation scheme, INT = 1), 0.25 to 0.5-MeV photon energy bands should be used. The incident energy ranges must agree with data given in Files 12 and/or 13. Other procedures are the same as those recommended for File 5 data (tabulated distribution).

15.2. File 15 Procedures

1. Photon energies, E_γ, within a subsection are given in order of increasing magnitude.

2. The TAB1 records for the $g(E_\gamma \leftarrow E_i)$ within a subsection are given in increasing order of neutron energy, E_i.

3. The tabulated functions, $g(E_\gamma \leftarrow E_i)$, should be normalized to unity within four significant figures.

4. The interpolation scheme for $p_j(E)$ must be either linear-linear or log-linear (INT = 1, 2, or 3) to preserve probabilities upon interpolation. Likewise, the interpolation scheme for $g(E_\gamma \leftarrow E)$ must be linear-linear or log-linear with respect to E.

5. The neutron energy mesh should be a subset of that used for the $y_{NK}(E)$ tabulation in File 12 or for the $\sigma^{\gamma}_{NK}(E)$ tabulation in File 13, and the energy ranges must be identical. However, the neutron energy mesh for $p_j(E)$ need not be the same as that for $g(E_\gamma \leftarrow E)$, as long as they span the same range.

6. For an MT number appearing in both File 12 and File 13, a continuous photon energy distribution (LF = 1) can appear in only one of those files. Otherwise the distribution as given in File 15 could not in general be uniquely associated with a corresponding multiplicity or production cross section.

7. Use the minimum amount of data that will accurately represent the energy distribution as a function of both E_γ and E. However, do not use too course a mesh for E, even if the distributions are slowly varying functions of E, since the interpolated distribution will always have a nonzero component up to the maximum energy at which either of the original distributions has a nonzero component.

8. The limit on the number of neutron energy points for either $p_j(E)$ or $g(E_\gamma \leftarrow E)$ is 200. The limit on the number of photon energy points for $g(E_\gamma \leftarrow E)$ is 1000.

16. FILE 16: PHOTON ENERGY-ANGLE DISTRIBUTIONS*

If the photon-energy and angular dependences of the cross section for production of the continuous spectrum are not separable, then a file analogous to File 6 is required instead of Files 14 and 15. The energy-angle distributions for discrete photons are completely determined in File 14 and should never appear in File 16. A knowledge of the formats of Files 14 and 15 is assumed in this discussion.

Consider a nonnormalized energy-angle distribution function $F(E_\gamma \leftarrow E,\mu)$, where the angular dependence is normalized so that

$$\int_{-1}^{1} F(E_\gamma \leftarrow E,\mu)d\mu = y(E_\gamma \leftarrow E) \quad .$$

Then the multiplicity (yield) can be separated out, leaving a function, $h(E_\gamma \leftarrow E, \mu)$, normalized in both E_γ and μ:

$$F(E_\gamma \leftarrow E,\mu) = y(E)\ h(E_\gamma \leftarrow E,\mu) \quad .$$

The differential photon production cross section is then obtained from

$$\frac{\partial^2 \sigma(E_\gamma \leftarrow E,\mu)}{\partial E_\gamma\ \partial\mu} = \sigma(E)\ y(E)\ h(E_\gamma \leftarrow E,\mu) \qquad \text{(b-photons/eV)},$$

where $\sigma(E)$ is the cross section for the reaction type being considered, as determined in Files 2 and 3.

As in File 14, the angular part of the distribution may be specified either in tabular form or as Legendre coefficients, $\eta_\ell(E_\gamma \leftarrow E)$. The Legendre expansion is

*The use of File 16 is discouraged but the formats and procedures are presented here in case the file should be activated.

$$h(E_\gamma \leftarrow E,\mu) = \sum_{\ell=0}^{NL} \frac{2\ell+1}{2} \eta_\ell(E_\gamma \leftarrow E)\, P_\ell(\mu) \quad .$$

16.1. File 16 Format

Two options are allowed, corresponding to the options in File 14: the angular distribution can be represented by either Legendre coefficients (LTT = 1) or by tabulated angular distributions (LTT = 2).

LTT = 1: Legendre Coefficient Representation. In this option, the Legendre coefficients are tabulated as functions of both incident neutron energy and photon energy. The structure of a section for LTT = 1 is

[MAT, 16, MT/ZA, AWR; b, LTT=1; b, b]HEAD

[MAT, 16, MT/ b, b; b, b; NL, b]CONT

<subsection for ℓ = 0>

<subsection for ℓ = 1>

.

.

.

<subsection for ℓ = NL>

[MAT, 16, 0/ b, b; b, b; b, b]SEND .

The subsections contain the energy distributions, and are identical in structure to a section for a continuous energy distribution (File 15), with the following exceptions:

a. The SEND record is deleted.

b. The HEAD record is changed to read

[MAT, 16, MT/b, b; b, b; NC, b]CØNT.

c. $g(E_\gamma \leftarrow E)$ is replaced by $\eta_\ell(E_\gamma \leftarrow E)$.

LTT = 2: Tabulated Angular Distribution. In the option, the subsections consist of tabulations for $h(E_\gamma \leftarrow E,\mu_m)$, m = 1,2...NA. The structure of a section for LTT = 2 is

[MAT, 16, MT/ZA, AWR; b, LTT=2; b, b]HEAD

[MAT, 16, MT/ b, b; b, b; NR, NA/μ_{int}]TAB2

<subsection for m = 1>

<subsection for m = 2>

.

.

.

<subsection for m = NA>

[MAT, 16, 0/ b, b; b, b; b, b]SEND .

As with the Legendre coefficient representation, this section for a tabulation contains subsections, identical in structure to a section for File 15, with the following exceptions:

a. The SEND record is deleted.

b. The HEAD record is changed to read

[MAT, 16, MT/b, μ_m; b, b; NC, b]CØNT .

c. $g(E_\gamma \leftarrow E)$ is replaced by $g(E_\gamma \leftarrow E,\mu_m)$, where each subsection is for a particular value of μ_m, m = 1,2...NA.

16.2. File 16 Procedures

The procedures for this file are the same as those for Files 14 and 15, where applicable.

22. GENERAL COMMENTS ON PHOTON INTERACTION

Photon interaction data are divided into five files, the first four analogous to Files 3 through 6.

File	Description
23	"Smooth" cross sections
24	Secondary angular distributions
25	Secondary energy distributions
26	Secondary energy-angle distributions
27	Coherent scattering form factors and incoherent scattering functions

As with the photon production data files, the photon interaction data formats parallel as closely as possible those for the neutron data files of the same number (modulo 20). This facilitates the use of existing retrieval routines in processing codes for photon interaction data (as in CHECKER). For Compton scattering at higher energies ($\gtrsim$1 MeV), the energy and angular distribution files would not normally be used because a simple analytical representation of these distributions is available. Also, provision is made for the entry of coherent scattering form factors as well as incoherent scattering functions. The secondary energy and angular distribution files can be used for both photon secondaries or particulate secondaries (e.g., photoneutrons).

Procedures are given for Files 23 and 27, but none will be given for Files 24, 25, and 26 until those files are activated. There are, at present, no data in these files.

23. FILE 23: "SMOOTH" PHOTON INTERACTION CROSS SECTIONS

This file is for the integrated photon interaction cross sections, including those usually called microscopic attenuation or energy-deposition coefficients, as well as photonuclear reaction cross sections. The reaction type (MT) numbers for photon interaction are in the 500 and 600 series. Several common photon interactions have been assigned MT numbers:

MT	Reaction Description
501	Total
502	Coherent scattering
504	Incoherent scattering
515	Pair production, electron field
516	Pair production, nuclear and electron field (i.e., pair plus triplet production)
517	Pair production, nuclear field
518	Photofission (γ,f)
532	Photoneutron (γ,n)
533	Total photonuclear
602	Photoelectric

Photon cross sections, such as the total cross section, coherent elastic scattering cross section, and incoherent (Compton) cross section, are given in File 23, which has essentially the same structure as File 3. These data are given as a function of energy, E_γ, where E_γ is the energy of the incident photon (in eV). The data are given as energy-cross-section pairs. An interpolation scheme is given that specifies the energy variation of the cross section for photon energies between a given energy point and the next higher energy point. The photon cross sections are given in one or more energy ranges. Within any one energy range,

the interpolation scheme is unchanged. The interpolation scheme may change from one to another energy range.

Each section in File 23 contains the data for a particular reaction type (MT number). The sections are ordered by increasing MT number.

23.1. File 23 Format

The format is almost identical to that of File 3, as follows.

[MAT, 23, MT/ZA, AWR; b, b; b, b]HEAD

[MAT, 23, MT/ b, b; b, b; NR, NP/$E_{\gamma\ int}$/$\sigma(E_\gamma)$]TAB1

[MAT, 23, 0/ b, b; b, b; b, b]]SEND .

23.2. File 23 Procedures

1. Values are usually for elements; hence, except for monoisotopic elements, ZA = Z x 1000; also, AWR should be for the naturally occurring element.

2. Photoelectric edges will not be multivalued. The edge will be defined by two energies differing in the fourth or fifth significant figure.

3. The total pair production values are given for reaction type MT = 516. Reaction type 517 is reserved for the portion of the pair production cross section due to the nuclear field, i.e., excluding triplet production.

4. Interpolation is normally log-log (INT = 5).

5. Kerma factor (energy deposition coefficients) libraries will normally be local because there is no universal definition. The application will determine whether annihilation or other radiation fractions are subtracted.

24. FILE 24: SECONDARY ANGULAR DISTRIBUTIONS

The structure of File 24 is identical to that for File 4, so the pertinent discussion from Section 4 of this report is reviewed here for convenience.

Secondary angular distributions are expressed as probability density functions, $p(\mu,E_\gamma)$. These functions can be represented either as a tabulation or as the Legendre coefficients, $f_\ell(E_\gamma)$, in

$$\frac{d\sigma(E_\gamma,\mu)}{d\mu} = \sigma(E_\gamma) \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} f_\ell(E_\gamma)\ P_\ell(\mu) ,$$

$$f_o(E_\gamma) \equiv 1.0 .$$

Here, $\mu = \cos\theta$, where θ is the polar angle of scattering in either the center-of-mass or the laboratory system. The secondary may be either a photon (coherently scattered) or a particle (e.g., photoneutrons). When the secondary distribution is for a photon, the laboratory system is always used.

24.1. File 24 Format

The format is identical to that for File 4 and will not be reproduced here (see section 4). However, for the case in which the secondary distribution is for a photon, the LCT flag is not relevant, and the following arbitrary convention is adopted:

a. LCT = 1, data are given in the laboratory system.

b. LVT = 0, transformation matrix is not given.

25. FILE 25: SECONDARY ENERGY DISTRIBUTIONS

The structure of the analogous File 5 appears to be entirely adequate (see Section 5). Thus, the format will not be reproduced here, but will be adopted by reference to File 5.

26. FILE 26: SECONDARY ENERGY-ANGLE DISTRIBUTIONS

The structure of the analogous File 6 appears to be entirely adequate (see Section 6). Thus, the format will not be reproduced here but adopted by reference to File 6. The inclusion of File 26 (as well as Files 6 and 16) is, at the present stage of development of cross section data, strictly *pro forma*.

27. FILE 27: ATOMIC FORM FACTORS OR SCATTERING FUNCTIONS

The ENDF system for neutron and photon production data allows two alternatives for storing angular distribution data. One is by probability per unit $\cos\theta$ vs $\cos\theta$, and the other is by Legendre coefficients. Actually, neither of these is a "natural" method for photons. The natural method would be atomic form factors or incoherent scattering functions. These are discussed briefly below.

a. Incoherent Scattering. The cross section for incoherent scattering is given by

$$\frac{d\sigma_i}{d\mu} = Z\, S(q;Z) \quad \frac{d\sigma_c}{d\mu} \quad ,$$

where $d\sigma_c/d\mu$ is the Klein-Nishina cross section, which can be written in closed form. The factor $S(q;Z)$ is the incoherent scattering function. At high ($\gtrsim 1$ MeV) energies, S approaches Z. In the other limit $S(0,Z) = 0$. The quantity q is the momentum of the recoil electron (in inverse angstroms*).

$$q = \alpha \left[1 + \left(\frac{\alpha'}{\alpha}\right)^2 - 2\mu \left(\frac{\alpha'}{\alpha}\right) \right]^{1/2} ,$$

where

$\alpha = E_\gamma/m_o c^2$,

E'_γ = scattered photon energy, and

$\mu = \cos\theta$.

*In ENDF, q is given in inverse angstroms as customarily reported in the literature. The above equations show q in "natural" m_oC units. Inverse angstroms, $\sin(\theta/2)/\lambda$, can be converted to m_oC units by the factor 2 x 12398.1/511006. = 0.0048524.

The angular distribution can then easily be calculated, given a table of $S(q;Z)$. Because S is a smoothly varying function of q, it can be represented by a reasonably small array of numbers. The quantities $S(q;Z)$ are tabulated as a function of q in File 27. The user presumably will have subroutines available for calculating q for energies and angles of interest and for calculating Klein-Nishina cross sections. He will then generate his cross sections for the appropriate cases by calculating q's, looking up the appropriate values of S, and substituting them in the above formula.

b. Coherent Scattering. The coherent scattering cross section is given by

$$\frac{d\sigma_{coh}}{d\mu} = \pi r_o^2 \, Z^2 (1 + \mu^2) \, F(q;Z) \ ,$$

where

$q = \alpha[2(1 - \mu)]^{1/2}$, the recoil momentum of the atom (in inverse angstroms, see note on previous page), and

$r_o = e^2/m_o c^2$, the classical radius of the electron.

The quantity $F(q;Z)$ is a form factor. This quantity is also easily tabulated. At high ($\gtrsim$1 MeV) energies, F approaches zero. In the other limit $F(0,Z) = Z$.

An alternative way of presenting the photon scattering data, then, would be to tabulate incoherent scattering functions and form factors. Users could then provide processing codes to generate the cross sections from this information. The calculation is quite straightforward and allows the user to generate all his scattering data from a relatively small table of numbers. The incoherent and coherent scattering data should always be presented as scattering

functions and form factors, respectively, whether or not data are included in Files 24, 25, or 26.

27.1. File 27 Format

The structure of a section is very similar to that of File 3 (and 23) and is

[MAT, 27, MT/ZA, AWR; b, b; b, b]HEAD

[MAT, 27, MT/ b, Z; b, b; NR, NP/q_{int}/H(q;Z)]TAB1

[MAT, 27, 0/ b, b; b, b; b, b]SEND.

The general symbol H(q;Z) is used for either F(q;Z) or S(q;Z) for coherent and incoherent scattering, respectively.

27.2. FILE 27 Procedures

1. Values of F(q;Z) should be entered in each case for the entire energy range for which integrated coherent and incoherent cross sections are given in File 23. This is true even though the respective values may be 0.0 or 1.0 over most of the (higher) energy range.
2. The value of Z is entered in floating-point format.

APPENDIX A

Glossary

		Section
A	the effective scattering radius.	2
A_i $(TP_i)(GP_i)$	the probability of a photon transition.	12
A_n	the mass of the nth type atom, A_o is the mass of the principal scattering atom in the molecule.	7
$A_{k,\ell}$	probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the kth to the ℓth level.	11
$a_\ell^k(E)$	ℓ^{th} Legendre coefficient associated with the discrete photon or photon continuum specified by k.	14
a,b	constants used in the Watt spectrum.	5
ABN	the abundance (weight fraction) of an isotope in this material.	2
AC_1, AC_2, AC_3, AC_4, BC_1, BC_2	the background constants for the Adler-Adler radiative capture cross section.	2
AF_1, AF_2, AF_3, AF_4, BF_1, BF_2	the background constants for the Adler-Adler fission cross section.	2
ALAB	Mnemonic of laboratory originating evaluation.	1
AJ	the compound nucleus spin, J (the spin of the resonance) (floating point).	2
AM	the spin-dependent effective scattering radius for spin-down, A_-.	2

		Section
AMUF	the number of degrees of freedom used in the fission width distribution.	2
AMUG	the number of degrees of freedom used in the radiation width distribution.	2
AMUN	the number of degrees of freedom used in the neutron width distribution.	2
AMUX	the number of degrees of freedom used in the competitive width distribution.	2
AT_1, AT_2, AT_3, AT_4, BT_1, BT_2	the background constants for the Adler-Adler total cross section.	2
AP	the spin-dependent effective scattering radius A_+ (for spin-up).	2
AUTH	author of evaluation.	1
AWR	the ratio of the mass of the atom (or molecule) to that of the neutron.	1
AWRI	the ratio of the mass of the particular isotope to that of the neutron.	2
B(N)	the list of constants.	7
BR	Fraction of the decay which proceeds by the corresponding decay mode.	1
ΔBR	Uncertainty in BR.	1

		Section
BR(N)	the branching ratio at the Nth 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.	1
C_n	the coefficients of a polynomial. There are NC coefficients given.	1
$C_n(E_i)$	the array of yield data for the ith energy point. This array contains NFP sets of three parameters in the order ZAFP, FPS, YLD.	1
CD_m	the coefficients for a polynomial.	1
CP_n	the coefficients of a polynomial. There are NC coefficients given.	1
D	the mean level spacing for a particular J-state.	2
DC	the decay constant (sec^{-1}) for the decay of a particular state of the product nuclide (ZAP).	1
DDATE	original distribution date of the evaluation.	1
DET_n	the Adler-Adler resonance energy for the total cross section. Here and below, the subscript n denotes the nth level.	2
DEF_n	the resonance energy for the fission cross section.	2
DEC_n	the resonance energy for the radiative capture cross section.	2

		Section
DWT_n	the value of $\Gamma/2$, (ν), used for the total cross section.	2
DWF_n	the value of $\Gamma/2$, (ν), used for the fission cross section.	2
DWC_n	the value of $\Gamma/2$, (ν), used for the radiative capture cross section.	2
$\frac{d\sigma}{d\Omega}(\Omega,E)$	differential scattering cross section in units of barns per steradian.	4
$\frac{d\sigma_k^\gamma}{d\Omega}$	differential photon production cross section in barns per steradian.	14
$\frac{d\sigma_{coh}}{du}$	differential photon coherent scattering.	27
E	energy of the incident neutron.	4,7,14
E and ΔE	Energy (eV) or radiation produced ($E_{\beta-}$, $E_{\beta+}$, E_γ, etc.).	1
E'	the secondary neutron energy (eV).	7
E_i	the incident neutron energy of the ith point (eV).	1
E_{avail}	Available Energy.	5
E_{int}	the interpolation scheme for each energy range. (Appendix E).	3

		Section
E_{th}	the threshold energy.	3
$\bar{E}_x, \Delta\bar{E}_x$	Average decay energy (eV) of radiation of type x and its uncertainty (eV). The β,γ and α energies are given in that order, with space reserved for zero β or γ entries. All non-γ and non-α energies are presently included as β energy. The α energy includes the recoil nucleus energy.	1
EDATE	date of evaluation.	1
EG_k	the photon energy or Binding Energy.	13
EL	the lower limit for a resonance region energy range.	2
EH	the upper limit for a resonance region energy range.	2
ER	the resonance energy (in the laboratory system).	2
ES_i	energy of the ith level.	12
ES_k	the energy of the level from which the photon originates.	13
ES(N)	the energy of the Nth incident energy (eV) at which branching ratios are given.	1
ES(N)	the energy of the Nth point used to tabulate energy-dependent widths.	2
F and ΔF	Normalization factor (absolute intensity/relative intensity).	1

		Section
$F(q;Z)$	The form factor for coherent photon scattering.	27
$F(E_\gamma \leftarrow E,\mu)$	An energy-angle distribution function for photon production (photons/eV).	12-15
$f_k(E \rightarrow E')$	the kth partial energy distribution. The definition depends on the value of LF.	5
$f_k(E_\gamma \leftarrow E)$	A normalized (to unity) photon energy distribution (or probability density) function at incident neutron energy E for the kth subsection within a reaction type (eV^{-1}).	12-15
f_ℓ	ℓ^{th} Legendre polynomial coefficient.	4
FPS	the state designator (floating-point number) for a fission product nuclide.	1
GRT_n	related to the symmetrical total cross section parameter.	2
GIT_n	related to the asymmetrical total cross section parameter.	2
GRF_n	the symmetrical fission parameter.	2
GIF_n	the asymmetrical fission parameter.	2
GRC_n	the symmetrical capture parameter.	2
GIC_n	the asymmetrical capture parameter.	2
GG	the average radiation width. It is energy dependent if LRU = 2.	2

		Section
GF	the average fission width. This value may be energy dependent.	2
GX	the average competitive reaction width.	2
GNO	the average <u>reduced neutron width</u>. It is energy dependent	2
$GP_{j,i} \equiv GP_i$	The conditional probability of photon emission in a direct transition from level j to level i, i < j.	12
GT	the resonance total width Γ evaluated at the resonance energy ER.	2
GN	the neutron width Γ_n evaluated at the resonance energy ER.	2
GG	the radiation width $\Gamma\gamma$ evaluated at the resonance energy ER.	2
GF	the fission width Γ_f evaluated at the resonance energy ER.	2
$g(E_\gamma \leftarrow E)$	A particular class of the functions $q_{j-1}(E_\gamma \leftarrow E)$ in File 15; those which are tabulated (eV^{-1}).	15
H(q;Z)	A general symbology for a form factor or incoherent scattering function; either F(q;Z) or S(q;Z), respectively.	27
H(N)	the array containing the Hollerith information that describes the particular evaluated data set.	1

		Section
$h(E_\gamma \leftarrow E,\mu)$	A normalized (to unity) energy-angle distribution function for photon production (eV^{-1}).	16
I and ΔI	Intensity of radiation produced.	1
I	the normalizing denominator (see 5.3).	5
I_i	the interpolation scheme (see Appendix E) to be used between the E_{i-1} and E_i energy points.	1
ICC and ΔICC	Internal conversion coefficient.	1
INT	the interpolation scheme to be used for interpolating between the cross-sections obtained from average resonance parameters.	2
INT(m)	the interpolation scheme identification number used in the mth range.	0
k	Boltzmann's contant.	7
ℓ	order of the Legendre polynomial.	1
L	the value of the ℓ-state (neutron angular momentum quantum number).	2
LA	the value of ℓ (for the ℓth coefficient).	6
LAT	a flag indicating which temperature has been used to compute α and β.	7
LCT	a flag indicating which reference frame is used for both secondary angles and energies.	4,6

		Section
LDD	a flag to indicate whether induced reaction decay data are given for this material.	1
LE	a test to determine whether energy-dependent fission product yields are given.	1
LF	a flag that specifies the energy distribution law that is used for a particular subsection (partial energy distribution).	5,16
LFI	a flag that indicates whether this material is fissionable.	1
LFP	a flag that indicates whether fission product yield data are given for this material.	1
LFS	an indicator that specifies the final excited state of the residual nucleus produced by a particular reaction.	1,3
LFW	a flag indicating whether average fission widths are given in the unresolved resonance region for this isotope.	2
LG	The transition probability array flag for distinguishing between doublet and triplet arrays in File 12.	12
LI	a flag to indicate the kind of Adler-Adler parameters given. The isotropy flag in File 14.	2,14
LIS	an indicator that specifies the initial state of the target nucleus (for materials that represent nuclides).	1,3

		Section
LND	a test that indicates whether polynomial or tabular representation is used.	1
LNP	a test that indicates what representation of $\bar{\nu}(E)$ has been used.	1
LNU	a test that indicates what representation of $\bar{\nu}(E)$ has been used.	1.2
LØ	The option flag to determine whether multiplicities or transition probability arrays are to be given in File 12.	12
LP	Indicator of whether or not the particular photon is a primary.	13
LR	a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x). (See Section 3.24.4.)	3
LRF	a flag indicating which resonance parameter representation has been used for this energy range. The definition of LRF depends on the value of LRU for this energy range.	2
LRP	a flag that indicates that resolved and/or unresolved resonance parameters are given in File 2.	1
LRU	a flag indicating whether an energy range contains data for resolved or unresolved resonance parameters.	2.1
LT	a flag to specify whether temperature-dependent data are given.	3

		Section
LT	Temperature dependence (see Appendix F).	0
LTT	a flag to specify whether Legendre or probability representation is used.	4,6,14
LVT	a flag to specify whether a transformation matrix is given for elastic scattering.	4
L1	an integer to be used as a flag or a test.	1
L2	an integer to be used as a flag or a test.	1
M_n	the number of atoms of the nth type in the molecule.	7
MAT	Material number.	0
MF	File number.	0
MT	Reaction type number.	0
MUF	the integer value of the number of degrees of freedom for fission widths.	2
MF_n	the MF of the nth section.	1
MT_n	the MT of the nth section.	1
NA	the number of angles (cosines) at which the secondary distributions are given.	6
NAV	Total number of decay modes for which average energies are given.	1

		Section
NB	the total number of β values given.	7
NBT(n)	the value of N separating the mth and $(m+1)^{th}$ interpolation ranges.	0
NC	the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$.	15
NC	a count of the number of terms used in the polynomial expansion.	1
NC_n	the number of BCD card images in a given section (the nth section).	1
NCD	the number of terms in the polynomial expansion.	1
NCP	a count of the number of terms used in the polynomial expansion.	1
NDK	total number of decay modes given.	1
NE	number of neutron energy points given in a TAB2 record.	0,14
NE	the number of energy points at which branching ratios are given for a specified initial state.	1
NE	the number of energy points at which energy-dependent widths are tabulated.	2
NE	the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given.	4,5

		Section
NER	the number of energy ranges given for this isotope	2
NF	the number of secondary energy points in a tabulation.	5
NFP	the number of fission product nuclide states to be specified at each incident energy point.	1
NI	the total number of items in the B(N) list. NL = 6* (NS + 1).	1
NI	number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.	14
NIS	the number of isotopes in this material.	2
NJS	the number of sets of resolved resonance parameters (each having the same J state) for a specified ℓ-state.	2
NK	the number of partial energy distributions. There will be one subsection for each partial distribution.	5,6
NK	the number of elements in the transformation matrix $NK = (NM + 1)^2$.	4
NL	the highest order Legendre polynomial that is given at each energy.	4,6
NL_i	highest value of ℓ required at each neutron energy E_i.	14

		Section
NLJ	the count of the number of levels for which parameters will be given.	2
NLS	the number of ℓ-states considered. A set of resolved resonance parameters is given for each ℓ-state.	2
NM	the maximum order Legendre polynomial that will be required to describe the angular distributions.	4
NNF	the number of precursor families considered.	1
NP	the number of points in a tabulation of $y(x)$ that are contained in the same record.	0.5
NP	the total number of energy points used to tabulate $\nu(E)$.	1
NP	the number of angular points (cosiness) used to give the tabulated probability distributions for each energy.	4
NP	the number of incident energy points at which $P_k(E)$ is given.	5
NP	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.	7
NPR	the number of product nuclides and/or product nuclide states for which data are given for <u>one</u> state of the original nuclide.	1

		Section
NR	the number of different interpolation intervals in a tabulation of y(x) that are contained in the same record.	0.5
NRS	the number of resolved resonances for a given ℓ-state.	2.2
NS	the integer number of states of the original nuclide for which reaction product data are given.	1
NS	the number of non-principal scattering atom types. For most moderating materials there will be (NS + 1) types of atoms in the molecule.	7.2
NS	number of levels below the present one, including the ground state.	12
NSP	Total number of spectra.	1
NT	number of transitions for which data are given in a list to follow.	12
NWD	the count of the number of elements in the Hollerith section.	1
NX	the count of the number of sets of background constants to be given.	2
NXC	an integer count of all the sections to be found in the dictionary.	1
N1	an integer to be used as a count of items in a list to follow except for MT 451.	1

		Section
N2	an integer to be used as a count of items in a second list to follow.	1
$P_j(E)$	the probability or weight given to the jth partial distribution, $q_j(E_\gamma \leftarrow E)$.	15
$P_k(E_N)$	the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.	5.2
$p(\mu,E)$	$\frac{2\pi}{\sigma_s(E)} \frac{d\sigma}{d\Omega}(\Omega,E)$	4
Q	the reaction Q-value (eV). Q = (rest mass of initial state - rest mass of final state.)	1
Q	Total decay energy (eV) available in the corresponding decay process. (This is not necessarily the same as the maximum energy of the emitted radiation.)	1.7
ΔQ	Uncertainty in Q value (eV).	1
$q = \alpha[2(1-\mu)]^{1/2}$	the recoil momentum of the atom (in inverse angstroms.	27
$q_j(E_\gamma \leftarrow E)$	the jth normalized partial distribution in the units eV^{-1}.	15
RDATE	date and number of last revision REV1- followed by month-year as in EDATE	1.1
REF	reference to evaluation	1.1

		Section
RFS	Isomeric state flag for daughter nuclide. (Fixed point number.)	1.7
RTYP	the designation of the reaction type leading to the described product nuclide state and is a floating-point equivalent of MT numbers.	1.3
$r_o = e^2/m_o c^2$	the classical radius of the electron.	27
S	the temperature (°K). NOTE: If the LR flag is used, S becomes Q_1 for the reaction corresponding to LR.	3
$S(\alpha,\beta,T)$	defined (for a moderating molecule) by the relation $$\frac{d^2\sigma}{d\Omega\, dE'}(E \rightarrow E',\mu,T) = \sum_{n=0}^{NS} \frac{M_n\, \sigma_{bn}}{4\pi\, T} \sqrt{\frac{E'}{E}}\; e^{-\beta/2}\, S_n(\alpha,\beta,T).$$	7
SPI	the nuclear spin of the target nucleus, I (positive number).	2
STYP	Decay type (Use mode of decay variable list).	1
T	the moderator temperature (°K).	7
$T_{1/2}$	Half-life of the original nuclide (seconds).	1
$\Delta T_{1/2}$	Uncertainty in the half-life (should be considered as one standard deviation).	1
TP_i $TP_{NS,i}$	the probability of a direct transition from level NS to level i.	12

		Section
$TP_{k,\ell}$	probability of the residual nucleus having a transition to the ℓth level given that it was initially in the excited state corresponding to the kth level.	11
U	a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).	5
V_K	the matrix elements of the transformation matrices.	4
X(n)	the nth value of x.	0.1
Y(n)	the nth value of y.	0.1
YLD	the fractional yield for a particular fission product.	1
ZA	the designation of the original nuclide (ZA = (1000.0*Z) + A)	1
ZA	the (Z,A) designation for a material (see Appendix C).	1
ZAI	the (Z,A) designation for an isotope.	2
ZAFP	the (Z,A) identifier for a particular fission product. (ZAFP = (1000.0*Z) + A).	1
ZAP	the (Z,A) designation of the product nuclide (ZAP = (1000.0*Z) + A).	1
ZSYMA	a Hollerith representation of the material Z-chemical symbol.	1

		Section
α	the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$.	7
β	the energy transfer, $\beta = (E' - E)/kT$.	7
β_{int} and α_{int}	the interpolation schemes used (see Appendix E for interpolation formats).	7
λi	the decay constant (sec^{-1}) for the ith precursor.	1
$\delta(E_\gamma - \varepsilon_j + \varepsilon_i$	delta function with ε_j, ε_i being energy levels of the residual nucleus.	11
$\bar{\nu}_d(E)$	the total average number of delayed neutron precursors formed per fission event.	1.4
θ	a parameter used to describe the secondary energy distribution. The definition of θ depends on the energy distribution law (LF).	5
$\sigma(E)$	the cross section (barns) for a particular reaction type at incident energy point, E, in (eV).	3
σ_{bn}	the bound atom scattering cross section of the nth type atom, $\sigma_{bn} = \sigma_{fn}\left(\frac{A_n + 1}{A_n}\right)^2$	1
σ_{fn}	the free atom scattering cross section of the nth type atom.	7
$\sigma_k^\gamma(E)$	photon production cross section for the discrete photon or photon continuum specified by k.	14
$\sigma_{m_o}(E)$	neutron cross sections for exciting the m_o^{th} level with neutron energy E.	11

		Section
$\sigma_s(E)$	the scattering cross section, e.g., elastic scattering at energy E as given in File 3 for the particular reaction type (MT).	4
σ_T (background)	$\frac{C}{\sqrt{E}}(AT_1 + AT_2/E + AT_3/E^2 + AT_4/E^3 + BT_1*E + BT_2*E^2)$.	2
σ_W	Wick's limit cross section in units of barns per steradian.	4
u	cosine of the scattered angle in either the laboratory or the center-of-mass system.	4

APPENDIX B

Definition of Reaction Types

Reaction types are identified by an integer, MT. The allowed reaction types are listed below. The reaction type number (MT) generally refers to a specific neutron-nucleus interaction mechanism, but occasionally it indicates that a particular type of information is given. The general rules for assignment of MT numbers are

MT (range)	Description of Class of Reactions
1-100	Reaction types in which secondary particles of the same type as the incident particles are emitted
101-150	Reaction types in which no secondary particles of the same type as the incident particles are emitted
151-200	Resonance region information
201-450	Quantities derived from the basic data
451-699	Miscellaneous quantities
700-799	Excitation cross sections for reactions that emit charged particles
800-999	(not assigned)

The specific MT assignments are given in the table below. For the most part, they are consistent with those used in the UKAEA Nuclear Data File.

MT	Description
1	Total cross section (redundant, equal to the sum of all partial cross sections)
2	Elastic scattering cross section
3	Nonelastic cross section (redundant, equal to the sum of all partial cross sections except elastic scattering)
4	Total inelastic cross section (redundant, equal to the sum of MT = 51, 52, 53, ..., 90, 91)

MT	Description
5	(to be assigned)
6	(n,2n) cross section for first excited state (describes first neutron)
7	(n,2n) cross section for second excited state (describes first neutron)
8	(n,2n) cross section for third excited state (describes first neutron)
9	(n,2n) cross section for fourth excited state (describes first neutron)
10-15	(to be assigned)
16	direct (n,2n) cross section [total (n,2n) cross section is sum of MT = 6, 7, 8, 9 and 16]
17	(n,3n) cross section
18	Total fission cross section (sum of MT = 19, 20, 21, 38)
19	(n,f) cross section (first chance fission)
20	(n,n'f) cross section (second chance fission)
21	(n,2nf) cross section (third chance fission)
22	(n,n´α) cross section
23	(n,n´3α) cross section
24	(n,2nα) cross section
25	(n,3nα) cross section
26	(n,2n) isomeric state cross section
27	Absorption cross section (sum of MT = 18 and 101) (includes particle reactions)
28	(n,n´p) cross section
29	(n,n´2α) cross section
30	(n,2n2α) cross section

MT	Description
31	to be used as LR flag only*
32	(n,n´d) cross section
33	(n,n´t) cross section
34	(n,n´^{3}He)
35	(n,n´d2α) cross section
36	(n,n´t2α) cross section
37	(n,4n) cross section
38	(n,3nf) cross section (fourth chance fission)
39	To be used as LR flag only*
40	To be used as LR flag only*
41-45	(to be assigned)
46	cross section for describing the second neutron from (n,2n) reaction for first excited state
47	cross section for describing the second neutron from (n,2n) reaction for second excited state
48	cross section for describing the second neutron from (n,2n) reaction for third excited state

*The following MT numbers are used only as LR flags in order to indicate the mode of decay of the residual nucleus:

LR	Description
31	Indicates that γ-emission is the mode of decay of the residual nucleus formed in the primary reaction.
39	Indicates that internal conversion is the mode of decay of the residual nucleus formed in the primary reaction.
40	Indicates that electron-positron pair formation is the mode of decay of the residual nucleus formed in the primary reaction.

(The "primary" reaction could be, for example, an (n,n´), (n,p), (n,α), (n,np), etc., reaction.)

MT	Description
49	cross section for describing the second neutron from (n,2n) reaction for fourth excited state (Note: MT = 46, 47, 48 and 49 should not be included in the sum for the total (n,2n) cross section)
50	(to be assigned)
51	(n,n') to the first excited state
52	(n,n') to the second excited state
. . .	
90	(n,n') to the 40th excited state
91	(n,n') to the continuum
92-100	(to be assigned)
101	neutron disappearance (sum of all cross sections in which a neutron is not in the exit channel). $MT = 101 \text{ is } \sum_{i=2}^{14} (MT-100+i)$
102	(n,γ) radiative capture cross section
103	(n,p) cross section
104	(n,d) cross section
105	(n,t) cross section
106	(n,^{3}He) cross section
107	(n,α) cross section
108	(n,2α) cross section
109	(n,3α) cross section
110	(to be assigned)
111	(n,2p) cross section
112	(n,pα) cross section
113	(n,t2α) cross section
114	(n,d2α) cross section

MT	Description
115-119	(to be assigned)
120	Target destruction = nonelastic less total $(n,n'\gamma)$
121-150	(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 (givenonly in File 1
452	$\bar{\nu}$, average total (prompt plus delayed) number of neutrons released per fission event
453	Radioactive nuclide production
454	Fission product yield data
455	Delayed neutrons from fission
456	Prompt neutrons from fission
457	Radioactive decay data
458-500	(to be assigned)
501	Total photon interaction cross section
502	Photon coherent scattering
503	(to be assigned)
504	Photon incoherent scattering

MT	Description
505-514	(to be assigned)
515	Pair production, electron field
516	Pair production, nuclear and electron field (i.e., pair plus triplet production)
517	Pair production, nuclear field
518	Photofission (γ,f)
519-531	(to be assigned)
532	Photoneutron (γ,n)
533	Total photonuclear
534-601	(to be assigned)
602	Photoelectric
603-699	(to be assigned)
700	(n,p_0) cross section (cross section for leaving the residual nucleus in the ground state)
701	(n,p_1) cross section for 1st excited state
702	(n,p_2) " " " 2nd " "
703	(n,p_3) " " " 3rd " "
704	(n,p_4) " " " 4th " "
.	
.	
.	
718	(n,p_c) " " " continuum excited state
719	(n,p_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing proton)
720	(n,d_0) cross section for ground state
721	(n,d_1) cross section for 1st excited state
722	(n,d_2) " " " 2nd " "
.	
.	
.	

MT	Description
738	(n,d_c) cross section for continuum excited state
739	(n,d_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing deuteron)
740	(n,t_0) cross section for ground state
741	(n,t_1) " " " 1st excited state
742	(n,t_2) " " " 2nd " "
.	
.	
.	
750	(n,t_c) " " " continuum excited state
759	(n,t_c') cross section for continuum specifically not included in σ total (redundant, used for describing outgoing triton)
760	$(n,{}^3He_0)$ cross section for ground state
761	$(n,{}^3He_1)$ cross section for 1st excited state
.	
.	
.	
778	$(n,{}^3He_c)$ cross section for continuum
779	$(n,{}^3He_c)$ cross section for continuum specifically not included in σ total (redundant, used for describing outgoing ${}^3He$)
780	(n,α_0) cross section for ground state
781	(n,α_1) cross section for 1st excited state
.	
.	
.	
798	(n,α_c) cross section for continuum
799	(n,α_c') cross section for continuum specifically not included in σ_T (redundant, used to describe outgoing α)
800-999	(to be assigned)

APPENDIX C

ZA Designation of Materials

A floating point number, ZA, is used to identify materials. If Z is the charge number and A the mass number, then ZA is computed from

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

For example, ZA for ^{238}U is 92238.0, and ZA for beryllium is 4009.0. For materials other than isotopes, the following rules apply.

(1) If the material is an element that has more than one naturally occurring isotope, then A is set to 0.0. For example, ZA for the element tungsten is 74000.0.

(2) For all other types of material, Z is set to zero, and the appropriate ZA is given in the following table. For example, ZA for H_2O is given as 100.0. The following classifications apply.

ZA (range)	Class of Materials
1-99	Hypothetical materials
100-199	Liquid moderators and coolants
200-299	Solid moderators
300-399	Metal alloys, cladding, and structural materials
400-499	Lumped fission products

Table of Appropriate ZA Designations

ZA	Material
1	Pure 1/v absorber. σ_{abs} (2200 m/sec) = 1.0
2	Pure scatterer. $\sigma_s(E) = 1.0$
3-99	(to be assigned)

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	Beryllium oxide, BeO
201	Beryllium carbide, Be_2C
202	Beryllium fluoride, BeF_2
203	Zirconium hydride, ZrH_x
204	Polystyrene, $(CH)_n$
205	Polyethylene $(CH_2)_n$
206-300	(to be assigned)
301	Zircalloy 1
302	Zircalloy 2
303	(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	^{233}U fission products (rapidly saturating) for thermal reactors
401	^{235}U " " " " " " "
402	^{239}U " " " " " " "

ZA	Material
403	^{241}Pu fission products (rapidly saturating) for thermal reactors
404	^{232}Th " " " " " " "
405	^{238}U " " " " " " "
406	^{240}Pu " " " " " " "
407-409	(to be assigned)
410	^{233}U fission products (slowly saturating) for thermal reactors
411	^{235}U " " " " " " "
412	^{239}Pu " " " " " " "
413	^{241}Pu " " " " " " "
414	^{232}Th " " " " " " "
415	^{238}U " " " " " " "
416	^{240}Pu " " " " " " "
417-419	(to be assigned)
420	^{233}U fission products (nonsaturating) for thermal reactors
421	^{235}U " " " " " "
422	^{239}Pu " " " " " "
423	^{241}Pu " " " " " "
424	^{232}Th " " " " " "
425	^{238}U " " " " " "
426	^{240}Pu " " " " " "
427-429	(to be assigned)
430	^{233}U fission products (rapidly saturating for fast reactors
431	^{235}U " " " " " " "
432	^{239}Pu " " " " " " "
433	^{241}Pu " " " " " " "
434	^{232}Th " " " " " " "

ZA	Materials
435	^{238}U fission products (rapidly saturating) for fast reactors
436	^{240}Pu " " " " " " "
437-439	(to be assigned)
440	^{233}U fission products (slowly saturating) for fast reactors
441	^{235}U " " " " " " "
442	^{239}Pu " " " " " " "
443	^{241}Pu " " " " " " "
444	^{232}Th " " " " " " "
445	^{238}U " " " " " " "
446	^{240}Pu " " " " " " "
447-449	(to be assigned)
450	^{233}U fission products (non-saturating) for fast reactors
451	^{235}U " " " " " "
452	^{239}Pu " " " " " "
453	^{241}Pu " " " " " "
454	^{232}Th " " " " " "
455	^{238}U " " " " " "
456	^{240}Pu " " " " " "
457-499	(to be assigned)

APPENDIX D

Resonance Region Formulae*

D.1. THE RESOLVED RESONANCE REGION

D.1.1. Single-Level Breit-Wigner Formula: LRU=1, LRF=1

The formulae appearing in Gregson et al.,[1] which omit the resonance-resonance interference terms, are adopted. These formulae, written in the laboratory system for all ℓ-values and without Doppler broadening, are (for a particular isotope)

1. Elastic Scattering Cross Section

$$\sigma_{n,n}(E) = \sum_{\ell=0}^{NLS-1} \sigma^{\ell}_{n,n}(E),$$

where

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

$$+ \frac{\pi}{k^2}\sum_J g_J \sum_{r=1}^{NR_J} \frac{\Gamma^2_{nr}\cos 2\phi_\ell - 2\Gamma_{nr}(\Gamma_{\gamma r}+\Gamma_{fr})\sin^2\phi_\ell + 2(E-E'_r)\Gamma_{nr}\sin 2\phi_\ell}{(E-E'_r)^2 + 1/4\,\Gamma^2_r}.$$

*Several processing codes have been developed to calculate cross sections with use of the formulae given here. These codes are given in Appendix I.

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

2. Radiative Capture Cross Section

$$\sigma_{n,\gamma}(E) = \sum_{\ell=0}^{NLS-1} \sigma_{n,\gamma}^{\ell}(E),$$

where

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

3. Fission Cross Section

$$\sigma_{n,f}(E) = \sum_{\ell=0}^{NLS-1} \sigma_{n,f}^{\ell}(E),$$

where

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

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.

<u>NLS is given in File 2 for each isotope</u>.

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.

$$NRS = \sum_J NR_J$$

<u>NRS is given in File 2 for each ℓ-value</u>.

$\Gamma_{nr}(|E_r|) \equiv GN_r$ is the neutron width, for the rth 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 is 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,

$$\phi_0 = \hat{\rho}$$

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

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

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

<u>$\hat{a}$ = AP, as given in File 2 data.</u>

D.1.2. <u>Multilevel Breit-Wigner Formula</u>: 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:

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

D.1.3. <u>Reich-Moore Formulae</u>

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

$$\sigma_{nc} = \pi \lambda\!\!\!^{-2}_n \sum_J g_J |\delta_{nc} - U^J_{nc}|^2 , \qquad (1)$$

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

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

(2) C.W. Reich and M.S. Moore, Phys. Rev. <u>111</u>, 929 (1958).

$$\frac{1}{\lambda\!\!\!^{-}_n} = k_n = 2.196771 \frac{AWRI}{AWRI+1.0} \times 10^{-3} \sqrt{E(eV)} , \tag{2}$$

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

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

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

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

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

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

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

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

$$\sigma_{nT} = \sum_J \sigma^J_{nT} = 2\pi \lambda^2_n \sum_J g_J \mathrm{Re}\left(1-U^J_{mn}\right)$$

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

$$\sigma_{nn} = \Pi \lambda^2_n \sum_J g_J \left|1-U^J_{nn}\right|^2 . \quad (7)$$

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

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

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

For s-wave neutrons $\phi_n = +k_n a$ 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.

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

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

(3) F.T. Adler and D.B. Adler, Conf. on Neutron Cross Section Technology, Vol. II, 873 (1967)

(4) D.B. Adler and F.T. Adler, ANL-6792, 695 (1963).

1. Total Cross Section

$$\sigma_T(E) = \frac{2C}{E}(1 - \cos\omega)$$

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

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

2. Capture Cross Section

$$\sigma_{n,\gamma}(E) =$$

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

$$+ \frac{C}{\sqrt{E}} (AC_1 + AC_2/E + AC_3/E^2 + AC_4/E^3 + BC_1 * E + BC_2 * E^2) \quad .$$

3. Fission Cross Section

$$\sigma_{n,f}(E) =$$

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

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

In all three formulae,

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

where k is the neutron wave number,

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

and

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

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

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: $\bar{D}_{\ell,J}$, $\bar{\Gamma}^{o}_{n_{\ell,J}}$, $\bar{\Gamma}_{\gamma_{\ell,J}}$, $\bar{\Gamma}_{f_{\ell,J}}$. 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

A few definitions and formulae useful in the unresolved resonance region are given below.

a. Level Spacing

The experimental value of mean spacing between resonances for a given ℓ state is determined as

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

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 corrections have been made for missed levels or that only that part of the energy range in which a plot of the level position vs. level number is linear has been used. For most of the nuclei this quantity is determined by looking at the s-wave resonances.

If we assume that

$\rho_{J,\ell} \sim (2J + 1)$,

where $\rho_{J,\ell}$ is the density of compound nucleus levels of spin J^{π} and given ℓ, then,

$$\frac{1}{D_{obs}} = \rho_{\ell,obs} = \sum_{J} \rho_{J,\ell} , \tag{2}$$

Note: all allowed ℓ values label the same set of resonances.

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

$$D_{J,\ell} = D_{\ell,obs} \times 2 \times (2I \times 1) \times (2\ell \times 1) \times \omega_{I,\ell} \Big/ \left(2J+1\right) \tag{3}$$

where $\omega_{I,\ell} = \frac{\ell + 1}{2\ell + 1}$ for $\ell \leq I$

and $\omega_{I,\ell} = \frac{I + 1}{2I + 1}$ for $\ell > I$

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 and also derive the spacings for the p and d-wave neutrons resonances.

b. Neutron Widths and Strength Functions

The definitions and usage of strength functions has been confused in the literature as has been pointed out by Gyulassy and Perkins[5]. These ambiguities, however, do not effect the results for s-wave neutrons ($\ell = 0$) or reactions on targets of spin zero ($I = 0$).

A microscopic strength function $S(\ell,J,s)$ can be defined, however, what is referred to in the literature as measurable is $S(\ell)$. The assumption is always made that $S(\ell,J,s)$ is independent of J and s. Two treatments which

(5) M. Gyolassy, S.T. Perkins, Nuc. Sci. Eng. 53, 482 (1974).

relate S_ℓ to $S_{\ell,J}$ are found in the literature and appear to differ in whether an explicit sum over s is included[5]. Except for the cases cited this can yield a factor of two difference.

The strength function is defined here as

$$S(\ell) = \frac{1}{(2\ell + 1)} \frac{\langle g\Gamma_n^\ell \rangle}{D(\ell)} \qquad (4)$$

where the statistical weight factor $g = \frac{2J + 1}{2\ (2I + 1)}$ for neutrons.

$$\langle g\Gamma_n^\ell \rangle = \frac{g\ \langle \Gamma_n^\ell\ (J) \rangle}{\mu_{\ell,J}\ \omega_{I,\ell}} = \frac{g\ \langle \Gamma_n^\ell\ (J,s) \rangle}{\omega_{I,\ell}} \qquad (5)$$

where $\omega_{I,\ell}$ is defined above

and $\mu_{\ell,J}$ is the number of ways to form a given J state of given ℓ (i.e. the multiplicity, either 1 or 2).

Note that the strength functions for a given ℓ-state but different spin states $J_1, J_2, \ldots J_s$ would all be equal. For more detail see ref. 5.

We define the neutron width $\Gamma_{n\ell J}$ for ℓ-wave neutrons and spin J - states as

$$\Gamma_{n\ell J} = \Gamma_{nJ}^\ell\ \sqrt{E} \times V_\ell \times \mu_{\ell,J}, \qquad (6)$$

where Γ_{nJ}^ℓ is the reduced neutron width, E is the neutron energy in eV, 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 $\ell = 0 \quad V_0(\rho) = 1$

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

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

In ENDF instead of summing over S a value of $\mu_{\ell,J}$ is introduced. If S has one or two values, $\mu_{\ell,J}$ is 1 or 2 respectively. (i.e. some of the spin states could be formed via two possible values of channel spin, I + 1/2 and I-1/2, and hence the corresponding neutron width could be thought of as following a Chi-squared distribution of $\mu = 2$ degrees of freedom.)

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 analysis of the resolved resonances. If, however, the energy range is rather wide, an energy dependence as given by some of the well-known theoretical models[6] has to be built in. Since the observed gamma width is a sum of a large number of primary gamma transitions, each assumed to have a chi-squared distribution of $\mu = 1$, it is found to have a $\mu \geq 20$. In effect this implies that the gamma width is a constant, since a chi-squared distribution with a large number of degrees of freedom approximates a δ-function.

D.2.2. Cross Sections in the Unresolved Region

a. Elastic Scattering Cross Section

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

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

$$+ \frac{2\pi^2}{k^2} \sum_{J}^{NJS_\ell} \frac{g_J}{\bar{D}_{\ell,J}} \left[\left\langle \frac{\Gamma_n \Gamma_n}{\Gamma} \right\rangle_{\ell,J} - 2\,\bar{\Gamma}_{n_{\ell,J}} \sin^2\phi_\ell \right] \quad .$$

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

b. Radiative Captive Cross Section

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

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

c. Fission Cross Section

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

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

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

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

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

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

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 integral 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

$\overline{\Gamma}_{x_{\ell,J}}$ = GX, the average competitive reaction width

$\overline{\Gamma}^{o}_{n_{\ell,J}}$ = GNO, the average reduced neutron width

$\overline{\Gamma}_{\gamma_{\ell,J}}$ = GG, the average radiation width

$\overline{\Gamma}_{f_{\ell,J}}$ = GF, the average fission width

$\overline{D}_{\ell,J}$ = D, the average level spacing

The average neutron widths are

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

where the penetrabilities, V_ℓ , are

$$V_0 = 1 \qquad \text{for s-wave neutrons, } \ell = 0$$

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

$$V_2 = \frac{\rho^4}{9 + 3\rho^2 + \rho^4} \qquad \text{for d-wave neutrons, } \ell = 2$$

The statistical weight factor, g_J , is

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

The average total width, at energy E, is

$$\overline{\Gamma}_{\ell,J} = \overline{\Gamma}_{n_{\ell,J}} + \overline{\Gamma}_{\gamma_{\ell,J}} + \overline{\Gamma}_{f_{\ell,J}} + \overline{\Gamma}_{x_{\ell,J}} \quad ,$$

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

ρ = ka,

where k is the neutron wave number,

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

and

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

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

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

<u>AWRI is given in File 2.</u>

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).

<u>$\hat{a}$ = A as given in File 2.</u>

APPENDIX E

Interpolation Schemes

Interpolation schemes are provided to obtain values of a function, y(x), from a tabulated series of X(N) and Y(N). The symbolism used to specify an interpolation scheme might be

[MAT, MF, MT/C1, C2; L1, L2; NR, NP/E_{int}/Y(E)]TAB1

where E_{int} implies an interpolation scheme and Y(E) implies pairs of values for E(N) and Y(N). The binary record would actually contain the following numbers:

[MAT, MF, MT, C1, C2, L1, L2, NR, NP, NBT(1), INT(1), NBT(2), INT(2), NBT(3), INT(3), ..., NBT(NR), INT(NR), E(1), Y(1), E(2), E(3), Y(3), ..., E(NP), Y(NP)]

NP is the number of pairs, E and Y, that are given. NR is the number of interpolation ranges given. NBT (1) is defined to mean that a particular interpolation scheme is to be used between point number one and the point number given by NBT(1). The interpolation scheme to be used in this range is specified by the value of INT(1). Likewise in the second interpolation region, between the point number given by NBT(1) and that given by NBT(2), the interpolation scheme is given by the value of INT(2). The procedure is followed until all interpolated regions have been specified. It should be obvious that the value of NBT(NR) is equal to the number NP. An illustration is shown in Figure E.1.

Interpolation schemes for a two-dimensional function y(E',E) are similar. The function is represented by a series of tabulated values and interpolation schemes. In this case two interpolation schemes must be given, one for E and

and another for E'. This is specified by a TAB2 record followed by several TAB1 or LIST records. An example might be

[MAT, MF, MT/C1, C2; L1, L2; NR, NE/E_{int}]TAB2

[MAT, MF, MT/C1, E(1); L1, L2; NR, NF/E'_{int}/g(E',E_1)]TAB1

[MAT, MF, MT/C1, E(2); L1, L2; NR, NF/E'_{int}/g(E', E_2)]TAB1

[MAT, MF, MT/C1, E(NE); L1, L2; NR, NF/E'_{int}/g(E', E_{NE})]TAB1

In this case NR, in the TAB2 record, indicates the number of interpolation ranges for (E). There will be NE TAB1 records, each will contain a value of E. E_{int} is the interpolation scheme used for the E mesh. NF in each TAB1 record indicates the number of pairs, E' and g(E',E) that will be given in the particular record. E'_{int} is the interpolation scheme to be used. The allowed interpolation schemes are given below.

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

*Note: INT = 1 (constant) implies that the function is constant and equal to the value given at the lower limit of the interval.

APPENDIX F

Temperature Dependence

Any of the data given in Files 3, 4, 5, 6, or 7 may have a temperature dependence (where it is physically realistic). The temperature dependence is specified by repeating the data for each temperature given and indicating how to interpolate the data between temperatures. LT is a flag that indicates whether or not temperature-dependent data are given.

The following quantities are defined.

T_m is the mth temperature (°K).

LT is a test for temperature dependence:

LT = 0 means no temperature dependence;

LT > 0 means that the function y(x,T) is given at (LT + 1) temperatures.

I_m is the interpolation scheme used between T_{m-1} and T_m. The values of I have the same definitions as INT given for other interpolation schemes (see Appendix E).

Since the data will always be given in a LIST or TAB1 record, consider a TAB1 record for a function y(x). In this case the functions must be y(x,T). The function at the first temperature $y(x,T_1)$ is given in a TAB1 record. The functions for the remaining temperatures are given in LIST records. The number of LIST records will be LT. An example might be

[MAT, MF, MT/ T_1, C2; LT, L2; NR, NP_1/ X_{int}/$Y(X,T_1)$]TAB1

[MAT, MF, MT/ T_2, C2; I_2, L2; NP_2, 0 / $Y_n(T_2)$]LIST

[MAT, MF, MT/ T_3, C2; I_3, L2; NP_3, 0 / $Y_n(T_3)$]LIST

[MAT, MF, MT/ T_{LT+1}, C2; I_{Lt+1}, L2, NP_{Lt+1}, 0/ $Y_n(T_{LT+1})$]LIST

The LIST records must be given in order of increasing value of the temperature T_m. Note that the interpolation scheme I_m is given in the same record position in the LIST record as LT in the TAB1 record. Also note that in the TAB1 record (for the first temperature) pairs of values are given, X(N) and $Y(X,T_1)$, while in the LIST record only values of $Y(X,T_2)$ are given. It is implied that $Y(X,T_2)$ given at the Nth point is for the same value of X(N) as is given for $Y(X,T_1)$. This means that the X mesh is given only once, for first temperature.

If a cross section exhibits a temperature dependence, it will generally occur only at low neutron energies, and the high energy data will be independent of temperature. Therefore, <u>the LIST records for the second and higher temperatures may contain NP's that are less than the NP given on the TAB1 record</u>. If the subscript n denotes the temperature, the following condition is defined:

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

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 as for NP, i.e.,

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

The above mechanism is used in File 1 to describe the <u>variation of fission product yields with incident neutron energy</u>. In this special case, the neutron energy replaces the temperature in the above illustration, and the interpolation code I_m refers to neutron energy.

APPENDIX G

Alternative Structure for ENDF Data Tapes

The standard structure of an ENDF tape was described in Section 0.4.2 of this report. The standard structure is well suited for BCD (card image) and binary tapes.

An alternative structure of the ENDF data tapes has been developed for use in certain cross section processing codes. This alternate arrangement, illustrated in Figure G.1, is simply an interchange of materials and files. The hierarchy is now MF, MAT, and MT.

Processing programs have been written that will convert an ENDF data tape (either BCD card image or binary) from the standard structure to the alternate structure (see Appendix I).

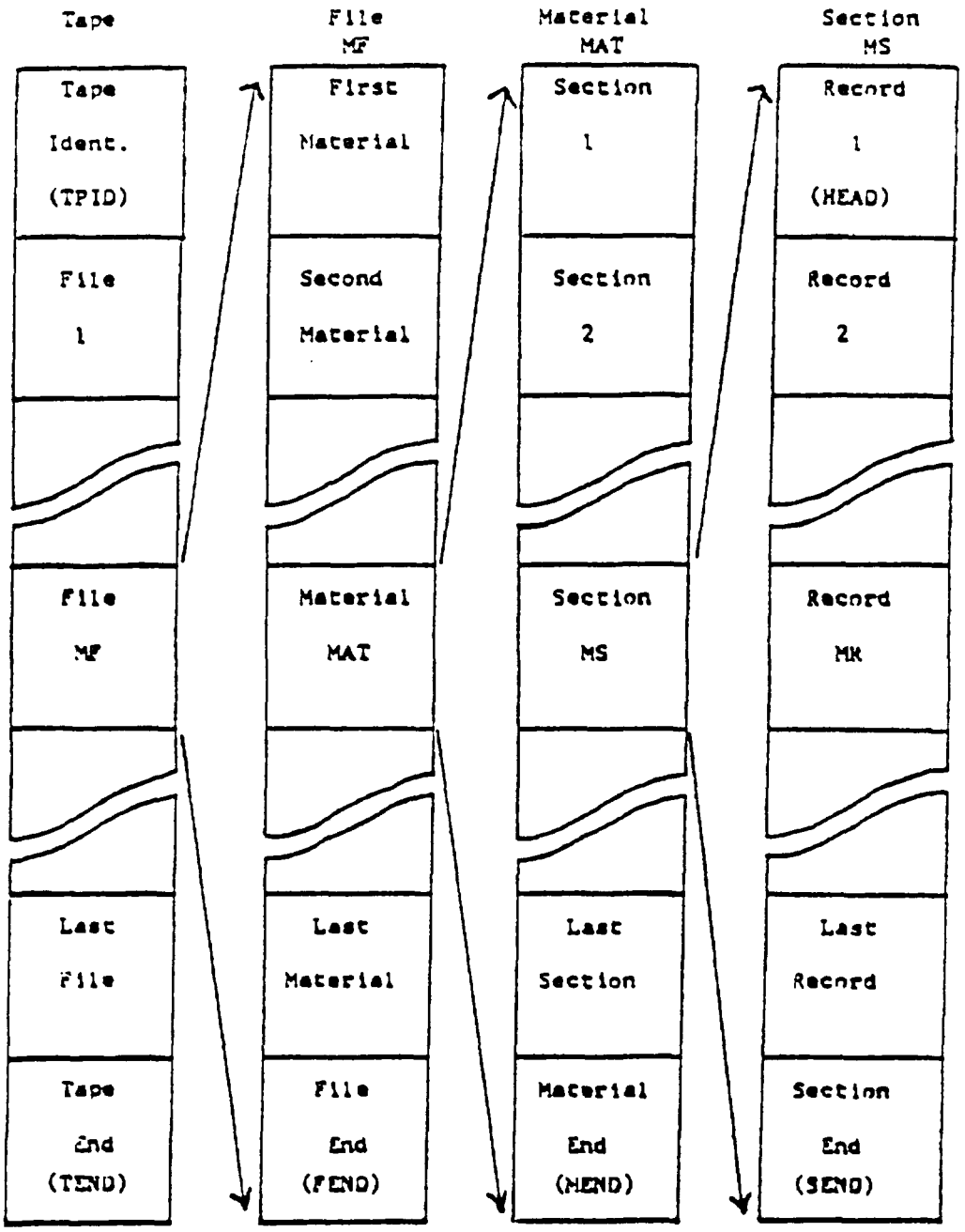


Figure G.1. Alternate Arrangement of an ENDF Tape

APPENDIX H

Data Formats for the ENDF/A Library

The data formats and procedures to be used for the ENDF/A library are essentially the same as those used for the ENDF/B. All processing codes, such as CHECKER, RIGEL, and ETØE, will be able to read the data tape, whether the tape is an ENDF/A or ENDF/B data type. The only difference between the two tapes is a flag in the HEAD record of the first section, MT = 451, in File 1. Also, the first part of the Hollerith information (first two BCD card-image records) will have an artificial structure. The modified structure for an ENDF/A tape is

[MAT, 1, 451/ A, 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 blank - ENDF/B tape,

= 1 - ENDF/A tape,

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

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

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

Field	Col.	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., it is 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 follows.

*Given only for materials that contain cross section data for one reaction type.

Appendix I

Cross Section Processing Codes Using ENDF/B-IV Data as Input

Code	Org.	Custodian	Comments
AMPX	ORNL	N. Greene	A large code system which will prepare neutron, photon production, and photon interaction cross sections for use in a variety of codes.
CAREN	BNL	NNCSC	Computes cross sections where resonance region representation changes. Used to check discontinuities.
CHEXER	BNL	NNCSC	Checks formats of an ENDF file for correctness.
ENDRUN	GE	C. Cowan	Prepares neutron cross sections and shielding factor tables for use in the TDOWN code.
ETOE-2/ MC^2-2	ANL	H. Henryson	Prepares broad group neutron cross sections for fast reactor calculations. Also used to prepare fine group neutron cross section libraries for use in the SDX code.
ETOG3	BNL	A. Aronson	Prepares neutron cross sections for use in the MUFT code.
ETOJ	SRL	D. Finch	Interfaces the JOSHUA system with ENDF.
ETOM	ANC	R. Grimesy	Prepares neutron cross sections for use in the MUFT code.
ETOM/ ETOG	WNES	M. Raymond	Prepares neutron cross sections for use in the MUFT and GAM codes. NO Adler-Adler capability.
ETOMX	BAPL	J. Hardy	Same as above.
ETOT	WNES	M. Raymond	Prepares pointwise thermal neutron cross sections from Files 2 and 3 of ENDF/B format data. No Adler-Adler capability.

Code	Org.	Custodian	Comments
ETOX	HEDL	R. Schenter	Prepares neutron cross sections and shielding factor tables for use in the 1DX code.
ETOX	LASL	R. MacFarland	Same as above.
FLANGE	SRL	D. Finch	Prepares thermal neutron cross sections from ENDF/B data including S(α,β) data in file 7. No Adler-Adler capability.
GAMLEG	LASL	R. Labauve	Prepares photon interaction cross sections for shielding analysis.
GFE4/ GAND3	GA	D. Mathews	Prepares neutron cross sections for use in the GGC-4, GGC-5 and MICROX codes.
INTEND/ INTER	BNL	NNCSC	Computes a variety of integral quantities from a pointwise ENDF file.
LAPHAND	LASL	R. Labauve	Prepares photon production cross section for use in shielding analysis.
LISTFC	BNL	NNCSC	Generates interpreted listings of ENDF files.
PUFF	ORNL	C. Weisbin	Processes covariance data for use in sensitivity analysis.
RESEND	BNL	NNCSC	Prepares infinitely dilute 0°K pointwise cross sections from File 2 + 3 information.
MINX	LASL	R. MacFarland	Prepares neutron cross section and shielding factor tables in the SPHINX code.
MINX	ORNL	C. Weisbin	Same as above.
NJOY	LASL	R. MacFarland	Extension of MINX to provide a coupled neutron/gamma ray capability for SPHINX.
PLOTFB	BNL	NNCSC	Automatic plotting code for ENDF.
RIGEL	BNL	NNCSC	ENDF file Editing code. Creates ENDF binary file formatted tapes.

Code	Org.	Custodian	Comments
SUPERTOG	ORNL	R. Q. Wright	Prepares neutron cross sections for codes of the GAM/MUFT type.
SUPERTON	GE	C. Stewert	Same as above.
SAMX	Magi	M. Beers	Prepares cross sections for Monte-Carlo codes.
SCOPEL	BNL	NNCSC	Interactive or instructed plotting code for ENDF.
SIGMA1	LLL	D. Cullen	Doppler broadens a linearized, pointwise ENDF file.
VIM	ANL	V. Prael	A Monte-Carlo slowing down code to prepare broad group cross sections for use in fast reactor calculations.
VIXEN	BNL	NNCSC	Checking code for photon files.

APPENDIX J

Materials in the ENDF/B-IV Library

The following is a list of materials that constitute the ENDF/B-IV Library. Those materials found on ENDF Tapes 401-411 are referred to as General Purpose Evaluations, Tape 412 as Dosimetry, and materials found on tapes 414-419 are referred to as Fission Product Evaluations.

Other materials exist and are available in the ENDF format. For a list of materials in the ENDF/A Library please contact NNCSC. Moderating Materials(i.e.Scattering Law Data) have been carried over from ENDF/B-III for H_2O, D_2O, Beryllium, BeO, Graphite, Polyethylene, Benzene, H in ZrH, and H in ZrH.

Additional materials such as a charged particle starter library are also available.

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
1-H- 1	1289	NEUT.+TH.SC.+GAM.PROD. DATA	411	LASL	LA-4574 (1971)	AUG78	L.STEWART,R.LABAUVE,P.YOUNG	484
1-H - 2	1128	NEUT.+TH.SC.+GAM.PROD. DATA	359	BNW	PRIV.COMM.(JUN.1967)	JUN67	B.R.LEONARD,JR.,K.S.STEWART	482
1-H - 3	1169	NEUT.+DECAY DATA	889	LASL		FEB65	L.STEWART	481
2-HE- 3	1144	NEUTRON CROSS SECTION DATA ONLY	481	LASL		1966	L.STEWART	482
2-HE- 4	1278	NEUTRON CROSS SECTION DATA ONLY	448	LASL		OCT73	R.A.NISLEY,G.M.HALE,P.G.YOUNG	481
3-LI- 6	1271	NEUT.+GAM.PROD. DATA	942	LASL		APR74	G.M.HALE, D.DODDER, P.YOUNG +	484
3-LI- 6	0271	NEUTRON CROSS SECTION DATA ONLY	138	LASL		NOV73	G.M.HALE,R.A.NISLEY,P.G.YOUNG	412
3-LI- 7	1272	NEUT.+GAM.PROD. DATA	679	LASL		OCT72	R.J.LABAUVE,L.STEWART,M.BATTAT	484
4-BE- 9	1289	NEUT.+GAM.PROD. DATA	2717	LLL		DEC71	R.J.HOMERTON,S.T.PERKINS	484
5-B - 10	1273	NEUT.+GAM.PROD. DATA	3282	LASL		NOV73	G.M.HALE,R.A.NISLEY,P.G.YOUNG	484
5-B - 10	0273	NEUTRON CROSS SECTION DATA ONLY	179	LASL		NOV73	G.M.HALE,R.A.NISLEY,P.G.YOUNG	412
5-B - 11	1148	NEUTRON CROSS SECTION DATA ONLY	977	GE-BNL		SEP71	G.COWAN	483
6- C- 12	1274	NEUT.+GAM.PROD.+ERR. DATA	1879	ORNL		DEC73	F.G.PEREY,C.Y.FU	488
7- N- 14	1275	NEUT.+TH.SC.+GAM.PROD.+ERR. DATA	6191	LASL		JUL73	P.YOUNG,D.FOSTER,JR.,S.HALE	485
8- O- 16	1276	NEUT.+TH.SC.+GAM.PROD.+ERR. DATA	6189	LASL		AUG73	P.YOUNG,D.FOSTER,JR.,S.HALE	488
9- F	1277	NEUT.+GAM.PROD. DATA	4189	ORNL		JUL74	C.Y.FU,D.C.LARSON,F.G.PEREY	411
11-NA- 23	0194	NEUTRON CROSS SECTION DATA ONLY	122	WARD-ORNL		1971	N PAIK,T.PITTERLE(WARD),PEREY	412
11-NA- 23	1194	NEUT.+TH.SC.+GAM.PROD. DATA	3196	WARD-ORNL		1971	N.PAIK,T.PITTERLE(WARD),PEREY	483
12-MG	1288	NEUT.+GAM.PROD. DATA	9888	ORT,BAI		FEB74	M.K.DRAKE,M.P.FRICKE	485
13-AL-27	0193	NEUTRON CROSS SECTION DATA ONLY	227	LASL		DEC73	P.G.YOUNG, D.G.FOSTER,JR.	412
13-AL-27	1193	NEUT.+GAM.PROD. DATA	5978	LASL		DEC73	P.G.YOUNG, D.G.FOSTER,JR.	485
14-SI	1194	NEUT.+GAM.PROD. DATA	18524	ORNL		FEB74	LARSON,PEREY,DRAKE,YOUNG	485
16- S- 32	0487	NEUTRON CROSS SECTION DATA ONLY	72	ANL		AUG72	N.O.DUDEY	412
17-CL	1149	NEUT.+GAM.PROD. DATA	4939	GGA	GA-7829 VOL-4(1967)	FEB67	M.S.ALLEN,M.K.DRAKE	483
19-K	1150	NEUT.+GAM.PROD. DATA	4172	GGA	GA-7829 VOL-5(1967)	FEB67	M.K.DRAKE	483
20-CA	1195	NEUT.+GAM.PROD. DATA	6161	ORNL		AUG71	C.Y.FU,F.G.PEREY	481
21-SC- 45	0413	NEUTRON CROSS SECTION DATA ONLY	113	BNL		JAN74	B.MAGURNO,S.F.MUGHABGHAB	412
22-TI	1286	NEUT.+GAM.PROD. DATA	1994	LLL		APR74	R.HOMERTON,R.MAIGNT,S.PERKINS	489
22-TI- 46	0481	NEUTRON CROSS SECTION DATA ONLY	68	BNL		APR72	B.MAGURNO	412
22-TI- 47	0482	NEUTRON CROSS SECTION DATA ONLY	72	BNL		AUG72	B.MAGURNO	412
22-TI- 48	0483	NEUTRON CROSS SECTION DATA ONLY	78	BNL		AUG72	B.MAGURNO	412
23-V	1196	NEUT.+GAM.PROD. DATA	4717	ORNL	ORNL-TM-4887(NOV72)	SEP72	S.K.PENNY, L.W.OWEN	482
24-CR	1191	NEUT.+GAM.PROD. DATA	8322	BNL		APR74	A.PRINCE	486
25-MN- 55	1197	NEUT.+GAM.PROD. DATA	4658	BNL		FEB74	M.TAKAHASHI	489
25-MN- 55	0197	NEUTRON CROSS SECTION DATA ONLY	42	BNL		OCT73	B.A.MAGURNO,M.TAKAHASHI	412
26-FE	1192	NEUT.+GAM.PROD. DATA	8286	ORNL	ORNL-4617(1970)	JAN74	PEREY,FU,PENNY,KINNEY,WRIGHT	486
26-FE- 54	0417	NEUTRON CROSS SECTION DATA ONLY	72	HEDL		AUG73	R.E.SCHENTER	412
26-FE- 56	0418	NEUTRON CROSS SECTION DATA ONLY	90	ANL		AUG72	N.O.DUDEY	412

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
26-FE- 58	0418	NEUTRON CROSS SECTION DATA ONLY	49	HEDL		OCT73	R.E.SCHENTER	412
27-CO- 59	0199	NEUTRON CROSS SECTION DATA ONLY	227	BNL,ANL		JAN74	T.KRIEGER(BNL)A.SMITH,D.SMITH	412
27-CO- 59	1199	NEUT.+GAM.PROD. DATA	2882	BNL,ANL		JAN74	T.KRIEGER(BNL),A.SMITH,D.SMITH	413
27-CO- 72	1	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
27-CO- 73	2	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
27-CO- 74	3	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
27-CO- 75	4	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI	1198	NEUT.+GAM.PROD. DATA	5688	BNL		DEC73	M.R.BHAT	405
28-NI- 58	0419	NEUTRON CROSS SECTION DATA ONLY	125	BNL,HEDL		AUG73	M.BHAT(BNL),R.SCHENTER(HEDL)	412
28-NI- 60	0420	NEUTRON CROSS SECTION DATA ONLY	49	BNL		NOV72	M.R.BHAT	412
28-NI- 72	5	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 73	6	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 74	7	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 75	8	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 76	9	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 77	10	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
28-NI- 78	11	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU	1299	NEUT.+GAM.PROD. DATA	3338	BAI		NOV73	M.K.DRAKE,H.P.FRICKE	412
29-CU- 63	0411	NEUTRON CROSS SECTION DATA ONLY	159	AI		APR72	H.ALTER,G.GIGAS,P.ROSE	412
29-CU- 65	0412	NEUTRON CROSS SECTION DATA ONLY	51	AI		APR72	H.ALTER,P.ROSE	412
29-CU- 72	12	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 73	13	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 74	14	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 75	15	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 76	16	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 77	17	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 78	18	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 79	19	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 80	20	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
29-CU- 81	21	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 72	22	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 73	23	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 74	24	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 75	25	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 76	26	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 77	27	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 78	28	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 79	29	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 80	30	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 81	31	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 82	32	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
30-ZN- 83	33	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 72	34	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 73	35	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 74	36	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 75	37	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
31-GA- 76	38	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
31-GA- 77	39	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
31-GA- 78	40	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 79	41	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	414
31-GA- 80	42	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	414
31-GA- 81	43	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 82	44	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 83	45	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 84	46	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
31-GA- 85	47	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 72	48	NEUTRON CROSS SECTION DATA ONLY	229	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
32-GE- 73	49	NEUTRON CROSS SECTION DATA ONLY	224	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
32-GE- 73M	50	RADIOACTIVE DECAY DATA ONLY	20	HEDL		APR74	R.E.SCHENTER	414
32-GE- 74	51	NEUTRON CROSS SECTION DATA ONLY	216	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
32-GE- 75	52	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 75M	53	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
32-GE- 76	54	NEUTRON CROSS SECTION DATA ONLY	188	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
32-GE- 77	55	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 77M	56	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414
32-GE- 78	57	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 79	58	RADIOACTIVE DECAY DATA ONLY	34	ANC		FEB74	C.W.REICH	414
32-GE- 80	59	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
32-GE- 81	60	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
32-GE- 82	61	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
32-GE- 83	62	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
32-GE- 84	63	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
32-GE- 85	64	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 86	65	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 87	66	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
32-GE- 88	67	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
33-AS- 75	68	NEUTRON CROSS SECTION DATA ONLY	364	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
33-AS- 76	69	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
33-AS- 77	70	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
33-AS- 78	71	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
33-AS- 79	72	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
33-AS- 80	73	RADIOACTIVE DECAY DATA ONLY	68	ANC		FEB74	C.W.REICH	414
33-AS- 81	74	RADIOACTIVE DECAY DATA ONLY	24	ANC		FEB74	C.W.REICH	414
33-AS- 82	75	RADIOACTIVE DECAY DATA ONLY	39	ANC		FEB74	C.W.REICH	414
33-AS- 82M	76	RADIOACTIVE DECAY DATA ONLY	48	ANC		FEB74	C.W.REICH	414
33-AS- 83	77	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414
33-AS- 84	78	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
33-AS- 85	79	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
33-AS- 86	80	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
33-AS- 87	81	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
33-AS- 88	82	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
33-AS- 89	83	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
33-AS- 90	84	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 76	85	NEUTRON CROSS SECTION DATA ONLY	228	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
34-SE- 77	86	NEUTRON CROSS SECTION DATA ONLY	319	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
34-SE- 77M	87	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
34-SE- 78	88	NEUTRON CROSS SECTION DATA ONLY	273	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
34-SE- 79	89	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 79M	90	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
34-SE- 80	91	NEUTRON CROSS SECTION DATA ONLY	257	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
34-SE- 81	92	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 81M	93	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
34-SE- 82	94	NEUTRON CROSS SECTION DATA ONLY	188	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
34-SE- 83	95	RADIOACTIVE DECAY DATA ONLY	112	ANC		FEB74	C.W.REICH	414
34-SE- 83M	96	RADIOACTIVE DECAY DATA ONLY	66	ANC		FEB74	C.W.REICH	414
34-SE- 84	97	RADIOACTIVE DECAY DATA ONLY	26	ANC		FEB74	C.W.REICH	414
34-SE- 85	98	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
34-SE- 85M	99	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
34-SE- 86	100	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414
34-SE- 87	101	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
34-SE- 88	102	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
34-SE- 89	103	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
34-SE- 90	104	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 91	105	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 92	106	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
34-SE- 93	107	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
35-BR- 79	108	NEUTRON CROSS SECTION DATA ONLY	318	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
35-BR- 79M	109	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
35-BR- 80	110	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
35-BR- 80M	111	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
35-BR- 81	112	NEUTRON CROSS SECTION DATA ONLY	216	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
35-BR- 82	113	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
35-BR- 82M	114	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414
35-BR- 83	115	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
35-BR- 84	116	RADIOACTIVE DECAY DATA ONLY	95	ANC		FEB74	C.W.REICH	414
35-BR- 84M	117	RADIOACTIVE DECAY DATA ONLY	33	ANC		FEB74	C.W.REICH	414
35-BR- 85	118	RADIOACTIVE DECAY DATA ONLY	33	ANC		JUL74	C.W.REICH	414
35-BR- 86	119	RADIOACTIVE DECAY DATA ONLY	55	ANC		FEB74	C.W.REICH	414
35-BR- 86M	120	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	414
35-BR- 87	121	RADIOACTIVE DECAY DATA ONLY	37	ANC		FEB74	C.W.REICH	414
35-BR- 88	122	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
35-BR- 89	123	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
35-BR- 90	124	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
35-BR- 91	125	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
35-BR- 92	126	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
35-BR- 93	127	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
35-BR- 94	128	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
35-BR- 95	129	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
35-BR- 96	130	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
36-KR- 78	1181	NEUTRON CROSS SECTION DATA ONLY	1848	BNL		JUL72	A.PRINCE	482

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
36-KR- 80	131	NEUTRON CROSS SECTION DATA ONLY	968	BNL		JUL72	A.PRINCE	414
36-KR- 80	1182	NEUTRON CROSS SECTION DATA ONLY	968	BNL		JUL72	A.PRINCE	482
36-KR- 81	132	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
36-KR- 81M	133	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
36-KR- 82	134	NEUTRON CROSS SECTION DATA ONLY	1126	BNL		JUL72	A.PRINCE	414
36-KR- 82	1183	NEUTRON CROSS SECTION DATA ONLY	1126	BNL		JUL72	A.PRINCE	482
36-KR- 83	135	NEUTRON CROSS SECTION DATA ONLY	1149	BNL		JUL72	A.PRINCE	414
36-KR- 83M	136	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	414
36-KR- 83	1184	NEUTRON CROSS SECTION DATA ONLY	1149	BNL		JUL72	A.PRINCE	482
36-KR- 84	137	NEUTRON CROSS SECTION DATA ONLY	896	BNL		JUL72	A.PRINCE	414
36-KR- 84	1185	NEUTRON CROSS SECTION DATA ONLY	896	BNL		JUL72	A.PRINCE	482
36-KR- 85	138	NEUT.+DECAY DATA	198	ANC,HEDL		FEB74	C.W.REICH	414
36-KR- 85M	139	RADIOACTIVE DECAY DATA ONLY	38	ANC		FEB74	C.W.REICH	414
36-KR- 86	140	NEUTRON CROSS SECTION DATA ONLY	735	BNL		JUL72	A.PRINCE	414
36-KR- 86	1186	NEUTRON CROSS SECTION DATA ONLY	735	BNL		JUL72	A.PRINCE	482
36-KR- 87	141	RADIOACTIVE DECAY DATA ONLY	57	ANC		FEB74	C.W.REICH	414
36-KR- 88	142	RADIOACTIVE DECAY DATA ONLY	148	ANC		FEB74	C.W.REICH	414
36-KR- 89	143	RADIOACTIVE DECAY DATA ONLY	375	ANC		FEB74	C.W.REICH	414
36-KR- 90	144	RADIOACTIVE DECAY DATA ONLY	88	ANC		FEB74	C.W.REICH	414
36-KR- 91	145	RADIOACTIVE DECAY DATA ONLY	43	ANC		FEB74	C.W.REICH	414
36-KR- 92	146	RADIOACTIVE DECAY DATA ONLY	148	ANC		FEB74	C.W.REICH	414
36-KR- 93	147	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
36-KR- 94	148	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
36-KR- 95	149	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
36-KR- 96	150	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
36-KR- 97	151	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
36-KR- 98	152	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
37-RB- 85	153	NEUTRON CROSS SECTION DATA ONLY	259	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	414
37-RB- 86	154	NEUT.+DECAY DATA	188	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
37-RB- 86M	155	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
37-RB- 87	156	NEUT.+DECAY DATA	282	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
37-RB- 88	157	RADIOACTIVE DECAY DATA ONLY	87	ANC		FEB74	C.W.REICH	414
37-RB- 89	158	RADIOACTIVE DECAY DATA ONLY	188	ANC		FEB74	C.W.REICH	414
37-RB- 90	159	RADIOACTIVE DECAY DATA ONLY	115	ANC		JUL74	C.W.REICH	414
37-RB- 90M	160	RADIOACTIVE DECAY DATA ONLY	69	ANC		JUL74	C.W.REICH	414
37-RB- 91	161	RADIOACTIVE DECAY DATA ONLY	79	ANC		FEB74	C.W.REICH	414
37-RB- 92	162	RADIOACTIVE DECAY DATA ONLY	186	ANC		FEB74	C.W.REICH	414
37-RB- 93	163	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
37-RB- 94	164	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
37-RB- 95	165	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	414
37-RB- 96	166	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
37-RB- 97	167	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	414
37-RB- 98	168	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414
37-RB- 99	169	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	414
37-RB-100	170	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
37-RB-101	171	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	414
38-SR- 86	172	NEUTRON CROSS SECTION DATA ONLY	298	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
38-SR- 87	173	NEUTRON CROSS SECTION DATA ONLY	261	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
38-SR- 87M	174	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	414

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
38-SR- 88	175	NEUTRON CROSS SECTION DATA ONLY	218	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
38-SR- 89	176	NEUT.+DECAY DATA	347	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
38-SR- 90	177	NEUT.+DECAY DATA	218	ANC,NEDL		FEB74	C.W.REICH	414
38-SR- 91	178	RADIOACTIVE DECAY DATA ONLY	75	ANC		FEB74	C.W.REICH	414
38-SR- 92	179	RADIOACTIVE DECAY DATA ONLY	39	ANC		FEB74	C.W.REICH	414
38-SR- 93	180	RADIOACTIVE DECAY DATA ONLY	68	ANC		FEB74	C.W.REICH	414
38-SR- 94	181	RADIOACTIVE DECAY DATA ONLY	37	ANC		FEB74	C.W.REICH	414
38-SR- 95	182	RADIOACTIVE DECAY DATA ONLY	23	NEDL		APR74	R.E.SCHENTER	414
38-SR- 96	183	RADIOACTIVE DECAY DATA ONLY	23	NEDL		APR74	R.E.SCHENTER	414
38-SR- 97	184	RADIOACTIVE DECAY DATA ONLY	29	NEDL		APR74	R.E.SCHENTER	414
38-SR- 98	185	RADIOACTIVE DECAY DATA ONLY	26	NEDL		APR74	R.E.SCHENTER	414
38-SR- 99	186	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
38-SR-100	187	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
38-SR-101	188	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
38-SR-102	189	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
38-SR-103	190	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
38-SR-104	191	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	414
39- Y- 89	192	NEUTRON CROSS SECTION DATA ONLY	237	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
39- Y- 89M	193	RADIOACTIVE DECAY DATA ONLY	28	NEDL		APR74	R.E.SCHENTER	414
39- Y- 90	194	NEUT.+DECAY DATA	287	ANC,NEDL		JUL74	C.W.REICH	414
39- Y- 90M	195	RADIOACTIVE DECAY DATA ONLY	33	ANC		JUL74	C.W.REICH	414
39- Y- 91	196	NEUT.+DECAY DATA	313	NEDL,ANC		OCT74	R.E.SCHENTER,F.SCHMITTROTH	414
39- Y- 91M	197	RADIOACTIVE DECAY DATA ONLY	23	ANC		FEB74	C.W.REICH	414
39- Y- 92	198	RADIOACTIVE DECAY DATA ONLY	58	ANC		FEB74	C.W.REICH	414
39- Y- 93	199	RADIOACTIVE DECAY DATA ONLY	62	ANC		FEB74	C.W.REICH	414
39- Y- 93M	200	RADIOACTIVE DECAY DATA ONLY	28	NEDL		APR74	R.E.SCHENTER	414
39- Y- 94	201	RADIOACTIVE DECAY DATA ONLY	59	ANC		JUL74	C.W.REICH	415
39- Y- 95	202	RADIOACTIVE DECAY DATA ONLY	62	ANC		FEB74	C.W.REICH	415
39- Y- 96	203	RADIOACTIVE DECAY DATA ONLY	23	NEDL		APR74	R.E.SCHENTER	415
39- Y- 97	204	RADIOACTIVE DECAY DATA ONLY	32	ANC		FEB74	C.W.REICH	415
39- Y- 98	205	RADIOACTIVE DECAY DATA ONLY	29	NEDL		APR74	R.E.SCHENTER	415
39- Y- 99	206	RADIOACTIVE DECAY DATA ONLY	26	NEDL		APR74	R.E.SCHENTER	415
39- Y-100	207	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	415
39- Y-101	208	RADIOACTIVE DECAY DATA ONLY	28	NEDL		APR74	R.E.SCHENTER	415
39- Y-102	209	RADIOACTIVE DECAY DATA ONLY	28	NEDL		APR74	R.E.SCHENTER	415
39- Y-103	210	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	415
39- Y-104	211	RADIOACTIVE DECAY DATA ONLY	28	NEDL		APR74	R.E.SCHENTER	415
39- Y-105	212	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	415
39- Y-106	213	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	415
39- Y-107	214	RADIOACTIVE DECAY DATA ONLY	22	NEDL		APR74	R.E.SCHENTER	415
40-ZR- 90	215	NEUTRON CROSS SECTION DATA ONLY	226	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
40-ZR- 90M	216	RADIOACTIVE DECAY DATA ONLY	24	ANC		JUL74	C.W.REICH	415
40-ZR- 91	217	NEUTRON CROSS SECTION DATA ONLY	219	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
40-ZR- 92	218	NEUTRON CROSS SECTION DATA ONLY	447	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
40-ZR- 93	219	NEUT.+DECAY DATA	338	NEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
40-ZR- 94	220	NEUTRON CROSS SECTION DATA ONLY	287	NEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
40-ZR- 95	221	NEUT.+DECAY DATA	266	NEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
40-ZR- 96	222	NEUTRON CROSS SECTION DATA ONLY	218	NEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
40-ZR- 97	223	RADIOACTIVE DECAY DATA ONLY	64	ANC		FEB74	C.W.REICH	415
40-ZR- 98	224	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
40-ZR- 99	225	RADIOACTIVE DECAY DATA ONLY	31	ANC		FEB74	C.W.REICH	415
40-ZR-128	226	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
40-ZR-181	227	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
40-ZR-182	228	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-183	229	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-184	230	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-185	231	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-186	232	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-187	233	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-188	234	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZR-189	235	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
40-ZIRC-2	1284	NEUTRON CROSS SECTION DATA ONLY	1863	BNW		OCT73	D.R.LEONARD,JR.	411
41-NB- 93	1189	NEUT.+GAM.PROD. DATA	2143	ANL,LLL		MAY74	R.HOWERTON(LLL),A.SMITH	411
41-NB- 93	236	NEUT.+GAM.PROD. DATA	2143	ANL,LLL		MAY74	R.HOWERTON(LLL),A.SMITH	415
41-NB- 93M	237	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	415
41-NB- 94	238	NEUT.+DECAY DATA	287	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
41-NB- 94M	239	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER,	415
41-NB- 95	240	NEUT.+DECAY DATA	288	HEDL,ANC		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
41-NB- 95M	241	RADIOACTIVE DECAY DATA ONLY	23	ANC		FEB74	C.W.REICH	415
41-NB- 96	242	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB- 97	243	RADIOACTIVE DECAY DATA ONLY	44	ANC		FEB74	C.W.REICH	415
41-NB- 97M	244	RADIOACTIVE DECAY DATA ONLY	23	ANC		FEB74	C.W.REICH	415
41-NB- 98	245	RADIOACTIVE DECAY DATA ONLY	64	ANC		FEB74	C.W.REICH	415
41-NB- 98M	246	RADIOACTIVE DECAY DATA ONLY	91	ANC		FEB74	C.W.REICH	415
41-NB- 99	247	RADIOACTIVE DECAY DATA ONLY	86	ANC		FEB74	C.W.REICH	415
41-NB- 99M	248	RADIOACTIVE DECAY DATA ONLY	94	ANC		FEB74	C.W.REICH	415
41-NB-120	249	RADIOACTIVE DECAY DATA ONLY	69	ANC		FEB74	C.W.REICH	415
41-NB-120M	250	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
41-NB-121	252	RADIOACTIVE DECAY DATA ONLY	38	ANC		FEB74	C.W.REICH	415
41-NB-122	253	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
41-NB-123	254	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-124	255	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
41-NB-125	256	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
41-NB-126	257	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-127	258	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-128	259	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-129	260	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-110	261	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-111	262	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
41-NB-112	263	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO	1287	NEUT.+GAM.PROD. DATA	1381	LLL		APR74	R.J.HOWERTON	489
42-MO- 94	264	NEUTRON CROSS SECTION DATA ONLY	244	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
42-MO- 95	265	NEUTRON CROSS SECTION DATA ONLY	517	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
42-MO- 96	266	NEUTRON CROSS SECTION DATA ONLY	241	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
42-MO- 97	267	NEUTRON CROSS SECTION DATA ONLY	538	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
42-MO- 98	268	NEUTRON CROSS SECTION DATA ONLY	348	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
42-MO- 99	269	NEUT.+DECAY DATA	275	ANC,HEDL		FEB74	C.W.REICH	415
42-MO-180	270	NEUTRON CROSS SECTION DATA ONLY	259	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
42-MO-181	271	RADIOACTIVE DECAY DATA ONLY	227	ANC		FEB74	C.W.REICH	415
42-MO-182	272	RADIOACTIVE DECAY DATA ONLY	25	ANC		FEB74	C.W.REICH	415
42-MO-183	273	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
42-MO-184	274	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
42-MO-185	275	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
42-MO-186	276	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
42-MO-187	277	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-188	278	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
42-MO-189	279	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-110	280	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-111	281	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-112	282	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-113	283	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-114	284	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
42-MO-115	285	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC- 99M	287	RADIOACTIVE DECAY DATA ONLY	34	ANC		FEB74	C.W.REICH	415
43-TC-180	288	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-181	289	RADIOACTIVE DECAY DATA ONLY	59	ANC		FEB74	C.W.REICH	415
43-TC-182	290	RADIOACTIVE DECAY DATA ONLY	33	ANC		FEB74	C.W.REICH	415
43-TC-182M	291	RADIOACTIVE DECAY DATA ONLY	62	ANC		FEB74	C.W.REICH	415
43-TC-183	292	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-184	293	RADIOACTIVE DECAY DATA ONLY	93	ANC		FEB74	C.W.REICH	415
43-TC-185	294	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-186	295	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-187	296	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-188	297	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-189	298	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-110	299	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
43-TC-111	300	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-112	301	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-113	302	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-114	303	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-115	304	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-116	305	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-117	306	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-118	307	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
43-TC-99	286	NEUTRON CROSS SECTION DATA ONLY	761	BAW	BAW-1367	OCT71	Z.LIVOLSI	415
43-TC-99	1137	NEUTRON CROSS SECTION DATA ONLY	761	BAW	BAW-1367	OCT71	Z.LIVOLSI	483
44-RU- 99	308	NEUTRON CROSS SECTION DATA ONLY	324	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
44-RU-100	309	NEUTRON CROSS SECTION DATA ONLY	246	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
44-RU-101	310	NEUTRON CROSS SECTION DATA ONLY	588	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
44-RU-102	311	NEUTRON CROSS SECTION DATA ONLY	449	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
44-RU-103	312	NEUT.+DECAY DATA	462	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
44-RU-104	313	NEUTRON CROSS SECTION DATA ONLY	215	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
44-RU-105	314	NEUT.+DECAY DATA	387	ANC,HEDL		FEB74	C.W.REICH	415
44-RU-106	315	NEUT.+DECAY DATA	255	HEDL,ANC		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
44-RU-127	316	RADIOACTIVE DECAY DATA ONLY	44	ANC		FEB74	C.W.REICH	415
44-RU-128	317	RADIOACTIVE DECAY DATA ONLY	29	ANC		FEB74	C.W.REICH	415
44-RU-129	318	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
44-RU-110	319	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
44-RU-111	320	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-112	321	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
44-RU-113	322	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-114	323	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-115	324	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-116	325	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-117	326	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-118	327	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-119	328	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
44-RU-120	329	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-103	1129	NEUTRON CROSS SECTION DATA ONLY	1129	BAN	BNW-1367 (ENDF-144)	OCT71	E.LIVOLSI	483
45-RH-103	330	NEUTRON CROSS SECTION DATA ONLY	1129	BAN	BNW-1367 (ENDF-144)	OCT71	E.LIVOLSI	415
45-RH-103M	331	RADIOACTIVE DECAY DATA ONLY	23	ANC		FEB74	C.W.REICH	415
45-RH-104	332	RADIOACTIVE DECAY DATA ONLY	59	ANC		FEB74	C.W.REICH	415
45-RH-104M	333	RADIOACTIVE DECAY DATA ONLY	68	ANC		FEB74	C.W.REICH	415
45-RH-105	334	NEUT.+DECAY DATA	239	ANC,HEDL		FEB74	C.W.REICH	415
45-RH-105M	335	RADIOACTIVE DECAY DATA ONLY	29	ANC		FEB74	C.W.REICH	415
45-RH-106	336	RADIOACTIVE DECAY DATA ONLY	67	ANC		FEB74	C.W.REICH	415
45-RH-106M	337	RADIOACTIVE DECAY DATA ONLY	69	ANC		FEB74	C.W.REICH	415
45-RH-107	338	RADIOACTIVE DECAY DATA ONLY	72	ANC		FEB74	C.W.REICH	415
45-RH-108	339	RADIOACTIVE DECAY DATA ONLY	34	ANC		FEB74	C.W.REICH	415
45-RH-108M	340	RADIOACTIVE DECAY DATA ONLY	43	ANC		FEB74	C.W.REICH	415
45-RH-109	341	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
45-RH-109M	342	RADIOACTIVE DECAY DATA ONLY	29	HEDL		APR74	R.E.SCHENTER	415
45-RH-110	343	RADIOACTIVE DECAY DATA ONLY	93	ANC		FEB74	C.W.REICH	415
45-RH-110M	344	RADIOACTIVE DECAY DATA ONLY	28	ANC		FEB74	C.W.REICH	415
45-RH-111	345	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
45-RH-112	346	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
45-RH-113	347	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
45-RH-114	348	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
45-RH-115	349	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-116	350	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-117	351	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-118	352	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-119	353	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-120	354	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-121	355	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-122	356	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
45-RH-123	357	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-104	358	NEUTRON CROSS SECTION DATA ONLY	259	HEDL		OCT74	R.E.SCHENTER AND F.SCHMITTROTH	415
46-PD-105	359	NEUTRON CROSS SECTION DATA ONLY	482	HEDL		OCT74	F.SCHMITTROTH AND R.E.SCHENTER	415
46-PD-106	360	NEUTRON CROSS SECTION DATA ONLY	326	HEDL		OCT74	F.SCHMITTROTH AND R.E.SCHENTER	415
46-PD-107	361	NEUT.+DECAY DATA	448	HEDL		OCT74	F.SCHMITTROTH AND R.E.SCHENTER	415
46-PD-107M	362	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
46-PD-188	363	NEUTRON CROSS SECTION DATA ONLY	328	HEDL		OCT74	F.SCHMITTROTH AND R.E.SCHENTER	415
46-PD-189	364	RADIOACTIVE DECAY DATA ONLY	49	ANC		FEB74	C.W.REICH	415
46-PD-189M	365	RADIOACTIVE DECAY DATA ONLY	24	ANC		FEB74	C.W.REICH	415
46-PD-110	366	NEUTRON CROSS SECTION DATA ONLY	348	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
46-PD-111	367	RADIOACTIVE DECAY DATA ONLY	131	ANC		FEB74	C.W.REICH	415
46-PD-111M	368	RADIOACTIVE DECAY DATA ONLY	126	ANC		FEB74	C.W.REICH	415
46-PD-112	369	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-113	370	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
46-PD-114	371	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-115	372	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
46-PD-116	373	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
46-PD-117	374	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
46-PD-118	375	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
46-PD-119	376	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-120	377	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-121	378	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-122	379	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-123	380	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-124	381	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-125	382	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
46-PD-126	383	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
47-AG-187	384	NEUTRON CROSS SECTION DATA ONLY	871	BNL	BNL-88383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	415
47-AG-187	1138	NEUTRON CROSS SECTION DATA ONLY	871	BNL	BNL-88383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	483
47-AG-188	385	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-188M	386	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
47-AG-189	387	NEUTRON CROSS SECTION DATA ONLY	889	BNL	BNL-88383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	415
47-AG-189M	388	RADIOACTIVE DECAY DATA ONLY	24	ANC		FEB74	C.W.REICH	415
47-AG-189	1139	NEUTRON CROSS SECTION DATA ONLY	889	BNL	BNL-88383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	483
47-AG-110	389	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-110M	390	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
47-AG-111	391	NEUT.+DECAY DATA	327	ANC,HEDL		FEB74	C.W.REICH	415
47-AG-111M	392	RADIOACTIVE DECAY DATA ONLY	24	ANC		FEB74	C.W.REICH	415
47-AG-112	393	RADIOACTIVE DECAY DATA ONLY	161	ANC		FEB74	C.W.REICH	415
47-AG-113	394	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-113M	395	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
47-AG-114	396	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-115	397	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-115M	398	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
47-AG-116	399	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-116M	400	RADIOACTIVE DECAY DATA ONLY	25	HEDL		APR74	R.E.SCHENTER	415
47-AG-117	401	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
47-AG-117M	402	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	416
47-AG-118	403	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
47-AG-118M	404	RADIOACTIVE DECAY DATA ONLY	25	HEDL		APR74	R.E.SCHENTER	416
47-AG-119	405	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	416
47-AG-120	406	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	416
47-AG-121	407	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
47-AG-122	408	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
47-AG-123	409	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
47-AG-124	418	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
47-AG-125	411	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
47-AG-126	412	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
47-AG-127	413	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
47-AG-128	414	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD	1281	NEUTRON CROSS SECTION DATA ONLY	2467	BNL		MAY74	S.PEARLSTEIN(TRANS FROM U.K.)	411
48-CD-108	415	NEUTRON CROSS SECTION DATA ONLY	386	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
48-CD-109	416	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-110	417	NEUTRON CROSS SECTION DATA ONLY	318	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
48-CD-111	418	NEUTRON CROSS SECTION DATA ONLY	443	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
48-CD-111M	419	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
48-CD-112	420	NEUTRON CROSS SECTION DATA ONLY	285	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
48-CD-113	1282	NEUTRON CROSS SECTION DATA ONLY	1333	BNL		MAY74	S.PEARLSTEIN(TRANS FROM U.K.)	411
48-CD-113	421	NEUTRON CROSS SECTION DATA ONLY	1333	BNL		MAY74	S.PEARLSTEIN(TRANS FROM U.K.)	415
48-CD-113M	422	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
48-CD-114	423	NEUTRON CROSS SECTION DATA ONLY	263	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
48-CD-115	424	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-115M	425	NEUT.+DECAY DATA	173	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
48-CD-116	426	NEUTRON CROSS SECTION DATA ONLY	215	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
48-CD-117	427	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
48-CD-117M	428	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
48-CD-118	429	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-119	430	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-119M	431	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
48-CD-120	432	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
48-CD-121	433	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
48-CD-122	434	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
48-CD-123	435	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
48-CD-124	436	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-125	437	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
48-CD-126	438	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-127	439	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
48-CD-128	440	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-129	441	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-130	442	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
48-CD-131	443	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
48-CD-132	444	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
49-IN-113	445	NEUTRON CROSS SECTION DATA ONLY	243	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
49-IN-113M	446	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
49-IN-114	447	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-114M	448	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-115	0486	NEUTRON CROSS SECTION DATA ONLY	67	STF		APR72	R.SHER	412
49-IN-115	0416	NEUTRON CROSS SECTION DATA ONLY	222	HEDL		DEC73	F.SCHMITTROTH	412
49-IN-115	449	NEUT.+DECAY DATA	483	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
49-IN-115M	450	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-116	451	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
49-IN-116M	452	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-116N	453	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	416

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
49-IN-117	494	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
49-IN-117M	495	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-118	496	RADIOACTIVE DECAY DATA ONLY	40	ANC		FEB74	C.W.REICH	415
49-IN-118M	497	RADIOACTIVE DECAY DATA ONLY	38	ANC		FEB74	C.W.REICH	415
49-IN-118N	498	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-119	499	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-119M	460	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-120	461	RADIOACTIVE DECAY DATA ONLY	65	ANC		FEB74	C.W.REICH	415
49-IN-120M	462	RADIOACTIVE DECAY DATA ONLY	28	ANC		FEB74	C.W.REICH	415
49-IN-121	463	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
49-IN-121M	464	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-122	465	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-122M	466	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-123	467	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-123M	468	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-124	469	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-125	470	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-125M	471	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
49-IN-126	472	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-127	473	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
49-IN-127M	474	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-128	475	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
49-IN-129	476	RADIOACTIVE DECAY DATA ONLY	27	HEDL		APR74	R.E.SCHENTER	415
49-IN-130	477	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
49-IN-131	478	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
49-IN-132	479	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
49-IN-133	480	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
49-IN-134	481	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
50-SN-115	482	NEUTRON CROSS SECTION DATA ONLY	214	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-116	483	NEUTRON CROSS SECTION DATA ONLY	389	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-117	484	NEUTRON CROSS SECTION DATA ONLY	249	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-117M	485	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
50-SN-118	486	NEUTRON CROSS SECTION DATA ONLY	343	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-119	487	NEUTRON CROSS SECTION DATA ONLY	274	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-119M	488	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
50-SN-120	489	NEUTRON CROSS SECTION DATA ONLY	455	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-121	490	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
50-SN-121M	491	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
50-SN-122	492	NEUTRON CROSS SECTION DATA ONLY	282	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-123	493	NEUT.+DECAY DATA	286	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-123M	494	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
50-SN-124	495	NEUTRON CROSS SECTION DATA ONLY	242	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-125	496	NEUT.+DECAY DATA	286	ANC,HEDL		FEB74	C.W.REICH	415
50-SN-125M	497	RADIOACTIVE DECAY DATA ONLY	64	ANC		FEB74	C.W.REICH	415
50-SN-126	498	NEUT.+DECAY DATA	286	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
50-SN-127	499	RADIOACTIVE DECAY DATA ONLY	241	ANC		FEB74	C.W.REICH	415
50-SN-127M	500	RADIOACTIVE DECAY DATA ONLY	27	ANC		FEB74	C.W.REICH	415
50-SN-128	501	RADIOACTIVE DECAY DATA ONLY	49	ANC		FEB74	C.W.REICH	415
50-SN-129	502	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
50-SN-129M	583	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
50-SN-130	584	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
50-SN-131	585	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
50-SN-132	586	RADIOACTIVE DECAY DATA ONLY	36	ANC		FEB74	C.W.REICH	415
50-SN-133	587	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
50-SN-134	588	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
50-SN-135	589	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
50-SN-136	590	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
51-SB-121	591	NEUTRON CROSS SECTION DATA ONLY	369	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
51-SB-122	592	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	416
51-SB-122M	593	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	416
51-SB-123	594	NEUTRON CROSS SECTION DATA ONLY	299	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
51-SB-124	595	NEUT.+DECAY DATA	298	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
51-SB-124M	596	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
51-SB-124N	597	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
51-SB-125	598	NEUT.+DECAY DATA	513	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
51-SB-126	599	NEUT.+DECAY DATA	242	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
51-SB-126M	600	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	416
51-SB-127	601	RADIOACTIVE DECAY DATA ONLY	75	ANC		FEB74	C.W.REICH	416
51-SB-128	602	RADIOACTIVE DECAY DATA ONLY	99	ANC		FEB74	C.W.REICH	415
51-SB-128M	603	RADIOACTIVE DECAY DATA ONLY	54	ANC		FEB74	C.W.REICH	416
51-SB-129	604	RADIOACTIVE DECAY DATA ONLY	144	ANC		FEB74	C.W.REICH	415
51-SB-130	605	RADIOACTIVE DECAY DATA ONLY	48	ANC		FEB74	C.W.REICH	416
51-SB-130M	606	RADIOACTIVE DECAY DATA ONLY	51	ANC		FEB74	C.W.REICH	416
51-SB-131	607	RADIOACTIVE DECAY DATA ONLY	189	ANC		FEB74	C.W.REICH	416
51-SB-132	608	RADIOACTIVE DECAY DATA ONLY	48	ANC		FEB74	C.W.REICH	415
51-SB-132M	609	RADIOACTIVE DECAY DATA ONLY	34	ANC		FEB74	C.W.REICH	415
51-SB-133	610	RADIOACTIVE DECAY DATA ONLY	118	ANC		FEB74	C.W.REICH	415
51-SB-134	611	RADIOACTIVE DECAY DATA ONLY	22	ANC		FEB74	C.W.REICH	416
51-SB-134M	612	RADIOACTIVE DECAY DATA ONLY	39	ANC		FEB74	C.W.REICH	415
51-SB-135	613	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
51-SB-136	614	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
51-SB-137	615	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
51-SB-138	616	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
51-SB-139	617	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
52-TE-122	618	NEUTRON CROSS SECTION DATA ONLY	224	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
52-TE-123	619	NEUT.+DECAY DATA	386	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
52-TE-123M	620	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	415
52-TE-124	621	NEUTRON CROSS SECTION DATA ONLY	329	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
52-TE-125	622	NEUTRON CROSS SECTION DATA ONLY	272	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
52-TE-125M	623	RADIOACTIVE DECAY DATA ONLY	23	ANC		FEB74	C.W.REICH	416
52-TE-126	624	NEUTRON CROSS SECTION DATA ONLY	344	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
52-TE-127	625	RADIOACTIVE DECAY DATA ONLY	48	ANC		FEB74	C.W.REICH	416
52-TE-127M	626	NEUT.+DECAY DATA	171	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
52-TE-128	627	NEUTRON CROSS SECTION DATA ONLY	236	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416
52-TE-129	628	RADIOACTIVE DECAY DATA ONLY	76	ANC		FEB74	C.W.REICH	416
52-TE-129M	629	NEUT.+DECAY DATA	192	ANC;HEDL		FEB74	C.W.REICH	416
52-TE-130	630	NEUTRON CROSS SECTION DATA ONLY	292	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	416

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
52-TE-131	551	RADIOACTIVE DECAY DATA ONLY	78	ANC		FEB74	C.W.REICH	415
52-TE-131M	552	RADIOACTIVE DECAY DATA ONLY	83	ANC		FEB74	C.W.REICH	415
52-TE-132	553	NEUT.+DECAY DATA	242	ANC,HEDL		JUL74	C.W.REICH	415
52-TE-133	554	RADIOACTIVE DECAY DATA ONLY	63	ANC		JUL74	C.W.REICH	415
52-TE-133M	555	RADIOACTIVE DECAY DATA ONLY	58	ANC		FEB74	C.W.REICH	415
52-TE-134	556	RADIOACTIVE DECAY DATA ONLY	46	ANC		JUL74	C.W.REICH	415
52-TE-135	557	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
52-TE-136	558	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
52-TE-137	559	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
52-TE-138	560	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
52-TE-139	561	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
52-TE-140	562	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	416
52-TE-141	563	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
52-TE-142	564	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	415
53- I-127	0414	NEUTRON CROSS SECTION DATA ONLY	61	STF		AUG72	R.SHER	412
53- I-127	565	NEUTRON CROSS SECTION DATA ONLY	564	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
53- I-128	566	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	415
53- I-129	567	NEUT.+DECAY DATA	341	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	415
53- I-130	568	NEUT.+DECAY DATA	174	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	415
53- I-130M	569	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	415
53- I-131	570	NEUT.+DECAY DATA	252	ANC,HEDL		FEB74	C.W.REICH	415
53- I-132	571	RADIOACTIVE DECAY DATA ONLY	76	ANC		FEB74	C.W.REICH	416
53- I-133	572	RADIOACTIVE DECAY DATA ONLY	49	ANC		FEB74	C.W.REICH	416
53- I-133M	573	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	416
53- I-134	574	RADIOACTIVE DECAY DATA ONLY	114	ANC		FEB74	C.W.REICH	415
53- I-134M	575	RADIOACTIVE DECAY DATA ONLY	24	ANC		FEB74	C.W.REICH	415
53- I-135	576	NEUT.+DECAY DATA	229	ANC,HEDL		FEB74	C.W.REICH	415
53- I-136	577	RADIOACTIVE DECAY DATA ONLY	44	ANC		FEB74	C.W.REICH	415
53- I-136M	578	RADIOACTIVE DECAY DATA ONLY	32	ANC		FEB74	C.W.REICH	415
53- I-137	579	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	415
53- I-139	581	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	417
53- I-140	582	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	417
53- I-141	583	RADIOACTIVE DECAY DATA ONLY	26	HEDL		APR74	R.E.SCHENTER	417
53- I-142	584	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
53- I-143	585	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
53- I-144	586	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
53- I-145	587	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
54-XE-124	1170	NEUTRON CROSS SECTION DATA ONLY	784	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-126	1171	NEUTRON CROSS SECTION DATA ONLY	788	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-128	1172	NEUTRON CROSS SECTION DATA ONLY	679	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-128	588	NEUTRON CROSS SECTION DATA ONLY	679	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-129	1173	NEUTRON CROSS SECTION DATA ONLY	842	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-129	589	NEUTRON CROSS SECTION DATA ONLY	842	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-129M	590	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	417
54-XE-130	1174	NEUTRON CROSS SECTION DATA ONLY	779	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-130	591	NEUTRON CROSS SECTION DATA ONLY	779	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-131	1175	NEUTRON CROSS SECTION DATA ONLY	814	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	482
54-XE-131	592	NEUTRON CROSS SECTION DATA ONLY	814	BNL	BNL-90374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
54-XE-131M	593	RADIOACTIVE DECAY DATA ONLY	29	ANC		FEB74	C.W.REICH	417
54-XE-132	1176	NEUTRON CROSS SECTION DATA ONLY	665	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT,S.F.MUGHABGHAB	482
54-XE-132	594	NEUTRON CROSS SECTION DATA ONLY	665	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-133	595	NEUT.+DECAY DATA	284	ANC,MEDL		FEB74	C.W.REICH DECAY DATA	417
54-XE-133M	596	RADIOACTIVE DECAY DATA ONLY	29	ANC		FEB74	C.W.REICH	417
54-XE-134	1177	NEUTRON CROSS SECTION DATA ONLY	648	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT,S.F.MUGHABGHAB	482
54-XE-134	597	NEUTRON CROSS SECTION DATA ONLY	648	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-134M	598	RADIOACTIVE DECAY DATA ONLY	28	MEDL		APR74	R.E.SCHENTER	417
54-XE-135	1294	NEUT.+DECAY DATA	889	BNW	PRIV.COM.JUNE,1967	JUN67	B.R.LEONARD,JR.,K.B.STEWART	484
54-XE-135	599	NEUT.+DECAY DATA	889	BNW	PRIV.COM.JUNE,1967	JUN67	B.R.LEONARD,JR.,K.B.STEWART	417
54-XE-135M	600	RADIOACTIVE DECAY DATA ONLY	25	ANC		FEB74	C.W.REICH	417
54-XE-136	1178	NEUTRON CROSS SECTION DATA ONLY	651	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT,S.F.MUGHABGHAB	482
54-XE-136	601	NEUTRON CROSS SECTION DATA ONLY	651	BNL	BNL-98374(FEB73)	JUN72	M.R.BHAT, S.F.MUGHABGHAB	417
54-XE-137	602	RADIOACTIVE DECAY DATA ONLY	55	ANC		FEB74	C.W.REICH	417
54-XE-138	603	RADIOACTIVE DECAY DATA ONLY	141	ANC		FEB74	C.W.REICH	417
54-XE-139	604	RADIOACTIVE DECAY DATA ONLY	295	ANC		FEB74	C.W.REICH	417
54-XE-140	605	RADIOACTIVE DECAY DATA ONLY	23	MEDL		APR74	R.E.SCHENTER	417
54-XE-141	606	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
54-XE-142	607	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
54-XE-143	608	RADIOACTIVE DECAY DATA ONLY	25	MEDL		APR74	R.E.SCHENTER	417
54-XE-144	609	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
54-XE-145	610	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
54-XE-146	611	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
54-XE-147	612	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
55-CS-133	613	NEUTRON CROSS SECTION DATA ONLY	982	BNL	BNL-38383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	417
55-CS-133	1141	NEUTRON CROSS SECTION DATA ONLY	982	BNL	BNL-38383(ENDF-163)	OCT71	M.R.BHAT, A.PRINCE	483
55-CS-134	614	NEUT.+DECAY DATA	268	ANC,MEDL		FEB74	C.W.REICH	417
55-CS-134M	615	RADIOACTIVE DECAY DATA ONLY	25	ANC		FEB74	C.W.REICH	417
55-CS-135	616	NEUT.+DECAY DATA	249	MEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
55-CS-135M	617	RADIOACTIVE DECAY DATA ONLY	28	MEDL		APR74	R.E.SCHENTER	417
55-CS-136	618	NEUT.+DECAY DATA	195	ANC,MEDL		FEB74	C.W.REICH	417
55-CS-137	619	NEUT.+DECAY DATA	366	MEDL,ANC		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
55-CS-138	620	RADIOACTIVE DECAY DATA ONLY	143	ANC		FEB74	C.W.REICH	417
55-CS-138M	621	RADIOACTIVE DECAY DATA ONLY	38	ANC		FEB74	C.W.REICH	417
55-CS-139	622	RADIOACTIVE DECAY DATA ONLY	269	ANC		FEB74	C.W.REICH	417
55-CS-140	623	RADIOACTIVE DECAY DATA ONLY	218	ANC		FEB74	C.W.REICH	417
55-CS-141	624	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-142	625	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-143	626	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-144	627	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-145	628	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-146	629	RADIOACTIVE DECAY DATA ONLY	26	MEDL		APR74	R.E.SCHENTER	417
55-CS-147	630	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
55-CS-148	631	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
55-CS-149	632	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
55-CS-150	633	RADIOACTIVE DECAY DATA ONLY	22	MEDL		APR74	R.E.SCHENTER	417
56-BA-134	634	NEUTRON CROSS SECTION DATA ONLY	258	MEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	417
56-BA-135	635	NEUTRON CROSS SECTION DATA ONLY	298	MEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	417
56-BA-135M	636	RADIOACTIVE DECAY DATA ONLY	28	MEDL		APR74	R.E.SCHENTER	417

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
56-BA-136	637	NEUTRON CROSS SECTION DATA ONLY	279	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	417
56-BA-136M	638	RADIOACTIVE DECAY DATA ONLY.	28	HEDL		APR74	R.E.SCHENTER	417
56-BA-137	639	NEUTRON CROSS SECTION DATA ONLY	268	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	417
56-BA-137M	640	RADIOACTIVE DECAY DATA ONLY	25	ANC		FEB74	C.W.REICH	417
56-BA-138	641	NEUTRON CROSS SECTION DATA ONLY	324	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	417
56-BA-139	642	RADIOACTIVE DECAY DATA ONLY	72	ANC		JUL74	C.W.REICH	417
56-BA-140	643	NEUT.+DECAY DATA	179	ANC,HEDL		JUL74	C.W.REICH	417
56-BA-141	644	RADIOACTIVE DECAY DATA ONLY	169	ANC		FEB74	C.W.REICH	417
56-BA-142	645	RADIOACTIVE DECAY DATA ONLY	97	ANC		FEB74	C.W.REICH	417
56-BA-143	646	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
56-BA-144	647	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
56-BA-145	648	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
56-BA-146	649	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
56-BA-147	650	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
56-BA-148	651	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
56-BA-149	652	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
56-BA-150	653	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
56-BA-151	654	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
56-BA-152	655	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-138	656	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-139	657	NEUTRON CROSS SECTION DATA ONLY	538	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
57-LA-140	658	NEUT.+DECAY DATA	294	ANC,HEDL		FEB74	C.W.REICH	417
57-LA-141	659	RADIOACTIVE DECAY DATA ONLY	72	ANC		FEB74	C.W.REICH	417
57-LA-142	660	RADIOACTIVE DECAY DATA ONLY	253	ANC		FEB74	C.W.REICH	417
57-LA-143	661	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-144	662	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-145	663	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-146	664	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-147	665	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-148	666	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	417
57-LA-149	667	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-150	668	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-151	669	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-152	670	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-153	671	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-154	672	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
57-LA-155	673	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	417
58-CE-140	674	NEUTRON CROSS SECTION DATA ONLY	272	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
58-CE-141	675	NEUT.+DECAY DATA	233	ANC,HEDL		FEB74	C.W.REICH	417
58-CE-142	676	NEUT.+DECAY DATA	238	HEDL		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
58-CE-143	677	NEUT.+DECAY DATA	211	ANC,HEDL		JUL74	C.W.REICH	417
58-CE-144	678	NEUT.+DECAY DATA	218	HEDL,ANC		OCT74	F.SCHMITTROTH,R.E.SCHENTER	417
58-CE-145	679	RADIOACTIVE DECAY DATA ONLY	37	ANC		FEB74	C.W.REICH	417
58-CE-146	680	RADIOACTIVE DECAY DATA ONLY	48	ANC		FEB74	C.W.REICH	417
58-CE-147	681	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	418
58-CE-148	682	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	418
58-CE-149	683	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	418
58-CE-150	684	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	418

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
58-CE-151	685	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	413
58-CE-152	686	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
58-CE-153	687	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
58-CE-154	688	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
58-CE-155	689	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
58-CE-156	690	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
58-CE-157	691	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-141	692	NEUTRON CROSS SECTION DATA ONLY	684	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
59-PR-142	693	NEUT.+DECAY DATA	174	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	413
59-PR-142M	694	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	413
59-PR-143	695	NEUT.+DECAY DATA	297	ANC,HEDL		FEB74	C.W.REICH	413
59-PR-144	696	RADIOACTIVE DECAY DATA ONLY	33	ANC		JUL74	C.W.REICH	413
59-PR-144M	697	RADIOACTIVE DECAY DATA ONLY	38	ANC		FEB74	C.W.REICH	413
59-PR-145	698	RADIOACTIVE DECAY DATA ONLY	37	ANC		FEB74	C.W.REICH	413
59-PR-146	699	RADIOACTIVE DECAY DATA ONLY	58	ANC		FEB74	C.W.REICH	413
59-PR-147	700	RADIOACTIVE DECAY DATA ONLY	49	ANC		FEB74	C.W.REICH	413
59-PR-148	701	RADIOACTIVE DECAY DATA ONLY	27	ANC		FEB74	C.W.REICH	413
59-PR-149	702	RADIOACTIVE DECAY DATA ONLY	34	ANC		FEB74	C.W.REICH	413
59-PR-150	703	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	413
59-PR-151	704	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	413
59-PR-152	705	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-153	706	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-154	707	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-155	708	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-156	709	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-157	710	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-158	711	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
59-PR-159	712	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-142	713	NEUTRON CROSS SECTION DATA ONLY	266	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	413
60-ND-143	714	NEUTRON CROSS SECTION DATA ONLY	535	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
60-ND-144	715	NEUT.+DECAY DATA	284	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	413
60-ND-145	716	NEUTRON CROSS SECTION DATA ONLY	868	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
60-ND-146	717	NEUTRON CROSS SECTION DATA ONLY	523	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
60-ND-147	718	NEUT.+DECAY DATA	239	ANC,HEDL		FEB74	C.W.REICH	413
60-ND-148	719	NEUTRON CROSS SECTION DATA ONLY	513	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
60-ND-149	720	RADIOACTIVE DECAY DATA ONLY	199	ANC		FEB74	C.W.REICH	413
60-ND-150	721	NEUTRON CROSS SECTION DATA ONLY	647	BNL,HEDL		NOV74	A.PRINCE,R.E.SCHENTER	413
60-ND-151	722	RADIOACTIVE DECAY DATA ONLY	382	ANC		FEB74	C.W.REICH	413
60-ND-152	723	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	413
60-ND-153	724	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-154	725	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-155	726	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-156	727	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-157	728	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-158	729	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-159	730	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-160	731	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413
60-ND-161	732	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	413

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
61-PM-147	733	NEUT.+DECAY DATA	626	HEDL,BNL		NOV74	R.E.SCHENTER AND A.PRINCE,	418
61-PM-148	734	NEUT.+DECAY DATA	166	ANC,HEDL		FEB74	C.W.REICH	418
61-PM-148M	735	NEUT.+DECAY DATA	281	ANC,HEDL		FEB74	C.W.REICH	418
61-PM-149	736	NEUT.+DECAY DATA	285	ANC,HEDL		FEB74	C.W.REICH	418
61-PM-150	737	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-151	738	NEUT.+DECAY DATA	316	ANC,HEDL		FEB74	C.W.REICH	418
61-PM-152	739	RADIOACTIVE DECAY DATA ONLY	71	ANC		FEB74	C.W.REICH	418
61-PM-152M	740	RADIOACTIVE DECAY DATA ONLY	79	ANC		FEB74	C.W.REICH	418
61-PM-152N	741	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	418
61-PM-153	742	RADIOACTIVE DECAY DATA ONLY	41	ANC		FEB74	C.W.REICH	418
61-PM-154	743	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	418
61-PM-154M	744	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	418
61-PM-155	745	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-156	746	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-157	747	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-158	748	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-159	749	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-160	750	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-161	751	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
61-PM-162	752	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	418
62-SM-147	753	NEUT.+DECAY DATA	886	HEDL,BNL		NOV74	R.E.SCHENTER,A.PRINCE	419
62-SM-148	754	NEUT.+DECAY DATA	386	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
62-SM-149	755	NEUT.+DECAY DATA	1210	BNW	PRIV.COM.JUNE,1967	JUN67	S.R.LEONARD,JR.,K.B.STEWART	418
62-SM-149	1027	NEUT.+DECAY DATA	1210	BNW	PRIV.COM.JUNE,1967	JUN67	S.R.LEONARD,JR.,K.B.STEWART	483
62-SM-150	756	NEUTRON CROSS SECTION DATA ONLY	398	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
62-SM-151	757	NEUT.+DECAY DATA	798	HEDL,BNL		NOV74	R.E.SCHENTER,A.PRINCE	419
62-SM-152	758	NEUTRON CROSS SECTION DATA ONLY	838	HEDL,BNL		NOV74	R.E.SCHENTER,A.PRINCE	419
62-SM-153	759	NEUT.+DECAY DATA	427	ANC,HEDL		FEB74	C.W.REICH	419
62-SM-154	760	NEUTRON CROSS SECTION DATA ONLY	439	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
62-SM-155	761	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-156	762	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-157	763	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
62-SM-158	764	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-159	765	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-160	766	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-161	767	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-162	768	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-163	769	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-164	770	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
62-SM-165	771	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-151	1290	NEUT.+GAM.PROD. DATA	3685	BNL		DEC73	M.TAKAHASHI	419
63-EU-151	772	NEUT.+GAM.PROD. DATA	3685	BNL		DEC73	M.TAKAHASHI	419
63-EU-152	1292	NEUT.+DECAY DATA	2822	BNL		DEC73	M.TAKAHASHI	419
63-EU-152	773	NEUT.+DECAY DATA	2822	BNL		DEC73	M.TAKAHASHI	419
63-EU-152M	774	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	419
63-EU-152N	775	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	419
63-EU-153	1291	NEUT.+GAM.PROD. DATA	3781	BNL		DEC73	M.TAKAHASHI	419
63-EU-153	776	NEUT.+GAM.PROD. DATA	3781	BNL		DEC73	M.TAKAHASHI	419

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
63-EU-154	1293	NEUT.+DECAY DATA	2867	BNL		DEC73	M.TAKAHASHI	419
63-EU-154	777	NEUT.+DECAY DATA	2849	BNL		DEC73	M.TAKAHASHI	419
63-EU-155	778	NEUT.+DECAY DATA	697	HEDL,BNL		NOV74	R.E.SCHENTER,A.PRINCE	419
63-EU-156	779	NEUT.+DECAY DATA	331	ANL,HEDL		FEB74	C.A.REICH	419
63-EU-157	780	NEUT.+DECAY DATA	169	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
63-EU-158	781	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-159	782	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-160	783	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
63-EU-161	784	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-162	785	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-163	786	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-164	787	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
63-EU-165	788	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD	1838	NEUTRON CROSS SECTION DATA ONLY	697	ANL	ANL-7387(MAR.68)	OCT68	E.M.PENNINGTON,J.C.GAJNIAK	483
64-GD-152	789	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	419
64-GD-153	790	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD-154	791	NEUTRON CROSS SECTION DATA ONLY	383	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-155	792	NEUTRON CROSS SECTION DATA ONLY	374	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-156	793	NEUTRON CROSS SECTION DATA ONLY	482	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-157	794	NEUTRON CROSS SECTION DATA ONLY	338	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-158	795	NEUTRON CROSS SECTION DATA ONLY	341	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-159	796	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD-160	797	NEUTRON CROSS SECTION DATA ONLY	249	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
64-GD-161	798	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD-162	799	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
64-GD-163	800	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD-164	801	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
64-GD-165	802	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
65-TB-159	803	NEUTRON CROSS SECTION DATA ONLY	415	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
65-TB-160	804	NEUT.+DECAY DATA	174	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
65-TB-161	805	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
65-TB-162	806	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
65-TB-162M	807	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
65-TB-163	808	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
65-TB-164	809	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
65-TB-165	810	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
66-DY-160	811	NEUTRON CROSS SECTION DATA ONLY	342	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
66-DY-161	812	NEUTRON CROSS SECTION DATA ONLY	288	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
66-DY-162	813	NEUTRON CROSS SECTION DATA ONLY	238	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
66-DY-163	814	NEUTRON CROSS SECTION DATA ONLY	381	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
66-DY-164	1931	NEUTRON CROSS SECTION DATA ONLY	1281	BNW	PRIV.COMM.JUNE.1967	JUN67	B.R.LEONARD.JR.,K.B.STEWART	482
66-DY-164	815	NEUTRON CROSS SECTION DATA ONLY	1281	BNW	PRIV.COMM.JUNE.1967	JUN67	B.R.LEONARD.JR.,K.B.STEWART	419
66-DY-165	816	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
66-DY-165M	817	RADIOACTIVE DECAY DATA ONLY	24	HEDL		APR74	R.E.SCHENTER	419
66-DY-165N	818	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
66-DY-166	819	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
67-HO-165	820	NEUTRON CROSS SECTION DATA ONLY	469	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419

TARGET	MAT	FILE CONTENT	# RECS	LAB	REFERENCE	DATE	AUTHOR	TAPE
67-HO-166	821	RADIOACTIVE DECAY DATA ONLY	22	HEDL		APR74	R.E.SCHENTER	419
67-HO-166M	822	RADIOACTIVE DECAY DATA ONLY	23	HEDL		APR74	R.E.SCHENTER	419
68-ER-166	823	NEUTRON CROSS SECTION DATA ONLY	472	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
68-ER-167	824	NEUTRON CROSS SECTION DATA ONLY	375	HEDL		OCT74	R.E.SCHENTER,F.SCHMITTROTH	419
68-ER-167M	825	RADIOACTIVE DECAY DATA ONLY	28	HEDL		APR74	R.E.SCHENTER	419
71-LU-175	1832	NEUTRON CROSS SECTION DATA ONLY	1863	BNW	PRIV.COMM.JUNE,1967	JUN67	B.R.LEONARD,JR.,K.B.STEWART	482
71-LU-176	1833	NEUTRON CROSS SECTION DATA ONLY	1149	BNW	PRIV.COMM.JUNE,1967	JUN67	B.R.LEONARD,JR.,K.B.STEWART	482
73-TA-181	1285	NEUT.+GAM.PROD. DATA	2799	LLL		JAN72	HOWERTON, PERKINS, MACGREGOR	411
73-TA-182	1127	NEUTRON CROSS SECTION DATA ONLY	761	AI	AI-AEC-12998 (1971)	APR71	J.OTTER,C.DUNFORD,E.OTTEWITTE	483
74-W-182	1128	NEUT.+GAM.PROD. DATA	3281	AI,LASL		JUN73	OTTER,OTTEWITTE,ROSE,YOUNG	481
74-W-183	1129	NEUT.+GAM.PROD. DATA	3535	AI,LASL		JUN73	OTTER,OTTEWITTE,ROSE,YOUNG	481
74-W-184	1130	NEUT.+GAM.PROD. DATA	3891	AI,LASL		JUN73	OTTER,OTTEWITTE,ROSE,YOUNG	481
74-W-186	1131	NEUT.+GAM.PROD. DATA	3873	AI,LASL		JUN73	OTTER,OTTEWITTE,ROSE,YOUNG	481
75-RE-185	1883	NEUTRON CROSS SECTION DATA ONLY	1375	GE(NMPO)	GEMP-567	JAN68	W.B.HENDERSON,J.W.ZWICK	483
75-RE-187	1884	NEUT.+DECAY DATA	1412	GE(NMPO)	GEMP-567	JAN68	W.B.HENDERSON,J.W.ZWICK	483
79-AU-197	1283	NEUTRON CROSS SECTION DATA ONLY	1447	BNL		APR74	MUGHABGHAB,PRINCE,GOLDBERG	411
79-AU-197	9283	NEUTRON CROSS SECTION DATA ONLY	232	BNL		OCT73	MUGHABGHAB ET.AL.	412
82-PB	1288	NEUT.+GAM.PROD. DATA	4288	ORNL		JUL71	C.Y.FU,F.G.PEREY	489
90-TH-232	1296	NEUT.+FPY+DECAY DATA	2823	B+W	BAW-317 (1978)	NOV66	W.WITTKOPF,D.ROY,E.LIVOLSI	484
90-TH-232	9296	NEUTRON CROSS SECTION DATA ONLY	426	B+W		NOV66	W.WITTKOPF,D.ROY,E.LIVOLSI	412
91-PA-233	1297	NEUT.+DECAY DATA	1886	BAPL		JAN78	P.C.YOUNG	487
92-U- 233	1268	NEUT.+FPY+DECAY DATA	5346	BAPL		JUL69	N.M.STEEN	489
92-U -234	1843	NEUT.+DECAY DATA	923	GGA	GA-8135 (SEP,1967)	JAN67	M.K.DRAKE,P.F.NICHOLS	434
92-U -235	9261	NEUTRON CROSS SECTION DATA ONLY	1273	LASL,AI		MAR74	L.STEWART,M.ALTER,R.HUNTER	412
92-U -235	1261	NEUT.+FPY+DECAY+GAM.PROD. DATA	6786	LASL,AI		MAR74	L.STEWART,M.ALTER,R.HUNTER	487
92-U -236	1163	NEUT.+DECAY DATA	925	SRL		OCT71	J. MCCROSSON	484
92-U -238	9262	NEUTRON CROSS SECTION DATA ONLY	788	WARD		SEP77	N.C.PAIK	412
92-U- 238	1262	NEUT.+FPY+DECAY+GAM.PROD. DATA	4687	WARD		SEP73	N.C.PAIK	489
93-NP-237	9263	NEUTRON CROSS SECTION DATA ONLY	1192	ANC,LASL		JUN73	J.R.SMITH(ANC),W.E.STEIN(LASL)	412
93-NP-237	1263	NEUT.+FPY+DECAY DATA	2547	ANC,LASL		JUN73	J.R.SMITH(ANC),W.E.STEIN(LASL)	489
94-PU-238	1898	NEUT.+DECAY DATA	1818	AI	NAA-SR-12271(MAY,67)	MAY67	M.ALTER,C.DUNFORD	484
94-PU-239	9264	NEUTRON CROSS SECTION DATA ONLY	887	GE-GRO,LA		MAR74	B.HUTCHINS,R.HUNTER,L.STEWART	412
94-PU-239,	1264	NEUT.+FPY+DECAY+GAM.PROD. DATA	9335	GE-GRO,LAB		MAR74	B.HUTCHINS,R.HUNTER,L.STEWART	487
94-PU-240	1265	NEUT.+DECAY+GAM.PROD. DATA	2278	ANL		APR74	E.PENNINGTON,M.HUMMEL	487
94-PU-241	1266	NEUT.+FPY+DECAY DATA	2648	ANL		DEC73	M.HUMMEL,E.PENNINGTON	487
94-PU-242	1161	NEUT.+DECAY DATA	999	AI,ANC	NAA-SR-12271(MAY-67)	MAY67	M.ALTER,C.DUNFORD(AI)MOD.-ANC	484
95-AM-241	1896	NEUT.+FPY+DECAY DATA	1868	ANC	PRIV.COMM.(NOV.1966)	NOV66	J.R.SMITH,R.A.GRIMESEY	484
95-AM-243	1897	NEUT.+FPY+DECAY DATA	599	ANC	PRIV.COMM.(NOV.1966)	NOV66	J.R.SMITH,R.A.GRIMESEY	484
96-CM-244	1162	NEUT.+DECAY DATA	1189	AI,ANC	NAA-SR-12271(MAY-67)	MAY67	M.ALTER,C.DUNFORD(AI)MOD.-ANC	484

APPENDIX K

Sample Data Set

The following is a sample data set in the ENDF format. This sample was taken from an evaluation by Nisley, et.al. and contains neutron cross section data for Helium-4. For other examples of data in the ENDF format see Appendix N.

```
2.00000+ 3 4.00130+ 0          0          0          0                  0 0  0    0
0.00000+ 0 0.00000+ 0          0          0          0          79      0127+ 1491    1
 0-0-0-  4 LASL        EVAL-OCT73 HIBLEY, HALE, YOUNG                     127+ 1491    2
                       DIST-FEB73                                         127+ 1491    3
                                                                          127+ 1491    4
                                                                          127+ 1491    5
*****************    MAT 4984    MOD 0    ********************           127+ 1491    6
                                                                          127+ 1491    7
MF=0     LASL-OCT,1973    R.A.HIBLEY, G.M.HALE, P.G.YOUNG                 127+ 1491    8
                                                                          127+ 1491    9
MF=1 ------------- SMOOTH CROSS SECTIONS ------------------------------  127+ 1491   10
                                                                          127+ 1491   11
        THE 2200 M/S CROSS SECTIONS ARE AS FOLLOWS,                       127+ 1491   12
            MT=1   SIGMA= 0.75916 BARNS                                   127+ 1491   13
            MT=2   SIGMA= 0.75916 BARNS                                   127+ 1491   14
                                                                          127+ 1491   15
  MT=1        TOTAL CROSS SECTION                                         127+ 1491   16
       SEE DISCUSSION UNDER MT=2 BELOW                                    127+ 1491   17
                                                                          127+ 1491   18
  MT=2        ELASTIC SCATTERING CROSS SECTION                            127+ 1491   19
       ALTHOUGH THE ONLY REACTION POSSIBLE FOR NEUTRONS INCIDENT127+ 1491   20
       ON HE=4 BELOW 20 MEV IS ELASTIC SCATTERING, THE MAJORITY   127+ 1491   21
       OF THE N-HE4 DATA IS RATHER IMPRECISE.  IN ORDER TO OVER-  127+ 1491   22
       COME THIS PROBLEM, AN R-MATRIX ANALYSIS WAS PERFORMED WITH 127+ 1491   23
       A DATA SET WHICH INCLUDED NOT ONLY THE N-HE4 DATA BUT ALSO 127+ 1491   24
       VERY PRECISE P-HE4 DATA.  ALL THE AVAILABLE N-HE4 AND P-HE4127+ 1491   25
       DATA BELOW 20 MEV WERE CONSIDERED IN THE ANALYSIS.  SINCE  127+ 1491   26
       THE PREVIOUS EVALUATION WAS COMPLETED IN 1966, SEVERAL     127+ 1491   27
       N-HE4 ELASTIC SCATTERING MEASUREMENTS HAVE BEEN DONE.  THE 127+ 1491   28
       MOST SIGNIFICANT OF THESE ARE THE LOW ENERGY NEUTRON CROSS 127+ 1491   29
       SECTION OF ROPER (R069),THE RPI TOTAL CROSS SECTION        127+ 1491   30
       MEASUREMENT (S073), WHICH COVER THE RANGE EN=0.7-30 MEV.   127+ 1491   31
       AND THE RELATIVE ANGULAR DISTRIBUTIONS OF MORGAN (M063).   127+ 1491   32
       A COMPLETE LIST OF REFERENCES FOR THE N-HE4 DATA USED IS   127+ 1491   33
       GIVEN BELOW.  THE P-HE4 DATA WAS SELECTED TO SATISFY VERY  127+ 1491   34
       STRINGENT STATISTICAL CRITERIA AND WE BELIEVE THE POSSIBLE 127+ 1491   35
       ERRORS OF THE PREDICTED VALUES FOR THE P-HE4 SCATTERING    127+ 1491   36
       TO BE LESS THAN 1.0 PER CENT.  A SIMPLE MODEL FOR THE      127+ 1491   37
       CHARGE DIFFERENCES BETWEEN THE N-HE4 AND P-HE4 SYSTEMS     127+ 1491   38
       WAS ASSUMED AND THE N-HE4 AND P-HE4 DATA SETS WERE         127+ 1491   39
       SIMULTANEOUSLY ANALYZED.  THE VALUES OF THE CROSS SECTIONS 127+ 1491   40
       AND ANGULAR DISTRIBUTIONS CONTAINED IN FILES 3 AND 4 ARE   127+ 1491   41
       PROBABLY ACCURATE TO WITHIN 2.0 PERCENT.                   127+ 1491   42
                                                                  127+ 1491   43
MF=4 ------------- NEUTRON ANGULAR DISTRIBUTIONS ---------------- 127+ 1491   44
                                                                  127+ 1491   45
  MT=2        ELASTIC SCATTERING ANGULAR DISTRIBUTIONS             127+ 1491   46
       OBTAINED FROM THE R-MATRIX ANALYSIS DESCRIBED ABOVE        127+ 1491   47
       UNDER MF=3,MT=2.  LEGENDRE POLYNOMIAL REPRESENTATION USED. 127+ 1491   48
                                                                  127+ 1491   49
------------------------- REFERENCES -------------------------- 127+ 1491   50
                                                                  127+ 1491   51
AU63  S.M.AUSTIN ET AL., PHYS.REV. 126(1962) 1932.                127+ 1491   52
BR72  H.B.BROSTE ET AL., PHYS.REV. C5(1972) 701.                  127+ 1491   53
BU66  F.W.BUSSER ET AL., NUCL.PHYS. 80(1966) 593.                 127+ 1491   54
CR72  O.S.CRAMER + L.CRANBERG, NUCL.PHYS. A180(1972) 273.         127+ 1491   55
FA63  U.FASOLI + G.ZAGO, NUOVO CIMENTO 30(1963) 1169.             127+ 1491   56
GO73  C.A.GOULDING ET AL., BULL.AM.PHYS.SOC. 18(1973) 538.        127+ 1491   57
HO66  R.HOOPER + H.H.BARSCHALL, NUCL.PHYS. 83(1966) 49.           127+ 1491   58
JE66  P.H.JEWELL ET AL., PHYS.REV. 142(1966) 867.                 127+ 1491   59
MA67  F.W.MAY ET AL., NUCL.PHYS. 49(1963) 177(REV. H066 + SA66)   127+ 1491   60
MO64  S.L.MORGAN + R.L.WALTER, PHYS.REV. 166(1968) 1114.          127+ 1491   61
NI71  A.NITLER ET AL., PHYS.REV. C4(1971) 30. (REV. W72)          127+ 1491   62
RO69  D.C.ROPER ET AL., NUCL.PHYS. 133(1969) 418.                 127+ 1491   63
SA66  J.R.SANDERS ET AL., PHYS.REV. 166(1968) 1182.               127+ 1491   64
SE73  J.J.SEAGRAVE, PHYS.REV. 92(1953) 1222.                      127+ 1491   65
SM73  D.P.SMAN, PROC.PHYS.SOC.(LONDON) 86(1973) 43.               127+ 1491   66
SN66  R.E.SNARD + J.S.JENKIN, PHYS.REV. 139(1966) 899.            127+ 1491   67
SM94  J.R.SMITH, PHYS.REV. 99(1954) 738.                          127+ 1491   68
ST70  T.STARMBACH ET AL., PHYS.REV. C2(1970) 434.                 127+ 1491   69
WH97  R.E.WHITE + F.J.M.FARLEY, NUCL.PHYS. 3(1957) 476.           127+ 1491   70
YO63  P.G.YOUNG ET AL., AUST.J.PHYS. 16(1963) 189.                127+ 1491   71
                                                                  127+ 1491   72
                                 1        491        80           127+ 1491   73
                                 2        191         4           127+ 1491   74
                                 3          1        89           127+ 1491   75
                                 3          2        89           127+ 1491   76
                                 3        291        19           127+ 1491   77
                                 3        292        19           127+ 1491   78
                                 3        293        19           127+ 1491   79
                                 4          2       113           127+ 1491   80
                                                                  127+ 1 0    81
                                                                  127+ 0 0    82
2.00000+ 3 4.00130+ 0          0          0          1           0127+ 2191   83
2.00000+ 3 1.00000+ 0          0          0          1           0127+ 2191   84
1.00000+ 3 1.00000+ 0          0          0          0           0127+ 2191   85
0.00000+ 0 2.49790+ 1          0          0          0           0127+ 2191   86
                                                                  127+ 2 0    87
                                                                  127+ 0 0    88
2.00000+ 3 4.00130+ 0          0         99          0           0127+ 3 1    89
0.00000+ 0 0.00000+ 0          0          0          1        2971270+ 3 1    90
     297                       2                                  127+ 3 1    91
1.00000+ 5 7.99160+ 1 2.33000+ 2 7.99160+ 1 1.30000+ 0 7.99160+ 1127+ 3 1    92
1.00000+ 1 7.99160+ 1 1.00000+ 2 7.99160+ 1 1.00000+ 2 7.99110+ 1127+ 3 1    93
0.00000+ 3 7.50010+ 1 1.00000+ 4 7.30720+ 1 2.00000+ 4 7.50400+ 1127+ 3 1    94
4.00000+ 4 7.00750+ 1 6.00000+ 4 7.00030+ 1 8.00000+ 4 7.62430+ 1127+ 3 1    95
1.00000+ 5 7.40090+ 1 1.50000+ 5 7.01500+ 1 2.00000+ 5 6.00170+ 1127+ 3 1    96
2.00000+ 5 8.40000+ 1 3.00000+ 5 9.00000+ 1 3.50000+ 5 9.00000+ 1127+ 3 1    97
4.00000+ 5 1.10200+ 0 4.50000+ 5 1.25300+ 0 5.00000+ 5 1.49200+ 0127+ 3 1    98
5.00000+ 5 1.71140+ 0 6.00000+ 5 2.24300+ 0 6.50000+ 5 2.24100+ 0127+ 3 1    99
6.00000+ 5 2.46300+ 0 6.75000+ 5 2.70870+ 0 7.00000+ 5 2.97000+ 0127+ 3 1   100
7.25000+ 5 3.27550+ 0 7.50000+ 5 3.59540+ 0 7.75000+ 5 3.93640+ 0127+ 3 1   101
8.00000+ 5 4.30060+ 0 8.25000+ 5 4.67790+ 0 8.50000+ 5 5.26160+ 0127+ 3 1   102
8.75000+ 5 5.44400+ 0 9.00000+ 5 6.02410+ 0 9.25000+ 5 6.18420+ 0127+ 3 1   103
9.50000+ 5 6.81000+ 0 9.75000+ 5 6.81730+ 0 1.00000+ 6 7.37910+ 0127+ 3 1   104
1.02500+ 6 7.28679+ 0 1.05000+ 6 7.44940+ 0 1.07500+ 6 7.56079+ 0127+ 3 1   105
1.10000+ 6 7.42030+ 0 1.12500+ 6 7.60010+ 0 1.15000+ 6 7.63210+ 0127+ 3 1   106
1.17500+ 6 7.57060+ 0 1.20000+ 6 7.46079+ 0 1.22500+ 6 7.39450+ 0127+ 3 1   107
1.25000+ 6 7.27210+ 0 1.27500+ 6 7.13020+ 0 1.30000+ 6 4.99400+ 0127+ 3 1   108
1.32500+ 6 4.63090+ 0 1.35000+ 6 4.06000+ 0 1.37500+ 6 4.52900+ 0127+ 3 1   109
1.40000+ 6 6.37400+ 0 1.45000+ 6 6.37330+ 0 1.50000+ 6 5.78820+ 0127+ 3 1   110
1.55000+ 6 5.32200+ 0 1.60000+ 6 5.27790+ 0 1.65000+ 6 5.05390+ 0127+ 3 1   111
1.70000+ 6 4.44000+ 0 1.75000+ 6 4.46360+ 0 1.80000+ 6 4.49430+ 0127+ 3 1   112
1.85000+ 6 4.34010+ 0 1.90000+ 6 4.19900+ 0 1.95000+ 6 4.07130+ 0127+ 3 1   113
2.00000+ 6 3.95410+ 0 2.05000+ 6 3.84070+ 0 2.10000+ 6 3.74020+ 0127+ 3 1   114
2.15000+ 6 3.67700+ 0 2.20000+ 6 3.57410+ 0 2.25000+ 6 3.49710+ 0127+ 3 1   115
2.30000+ 6 3.42590+ 0 2.35000+ 6 3.35980+ 0 2.40000+ 6 3.29060+ 0127+ 3 1   116
2.45000+ 6 3.34160+ 0 2.50000+ 6 3.18850+ 0 2.55000+ 6 3.13400+ 0127+ 3 1   117
2.60000+ 6 3.09250+ 0 2.65000+ 6 3.04010+ 0 2.70000+ 6 3.00600+ 0127+ 3 1   118
2.75000+ 6 2.97910+ 0 2.80000+ 6 2.93490+ 0 2.85000+ 6 2.90000+ 0127+ 3 1   119
```

2,90000-	6	2,94794-	8	2,99000-	6	2,83739-	8	3,00000-	6	2,88478-	8	127W	3	1	120
3,10000-	6	2,75944-	8	3,20000-	6	2,77878-	8	3,30000-	6	2,66273-	8	127E	3	1	121
3,40000-	6	2,62216-	8	3,50000-	6	2,56444-	8	3,60000-	6	2,54944-	8	127E	3	1	122
3,70000-	6	2,51450-	8	3,80000-	6	2,46550-	8	3,90000-	6	2,45611-	8	127F	3	1	123
4,00000-	6	2,42886-	8	4,10000-	6	2,40111-	8	4,20000-	6	2,37934-	8	127E	3	1	124
4,30000-	6	2,35830-	8	4,40000-	6	2,32600-	8	4,50000-	6	2,30240-	8	127E	3	1	125
4,60000-	6	2,27944-	8	4,70000-	6	2,25600-	8	4,80000-	6	2,23040-	8	127W	3	1	126
4,90000-	6	2,21324-	8	5,00000-	6	2,19199-	8	5,10000-	6	2,17190-	8	127F	3	1	127
5,20000-	6	2,15046-	8	5,30000-	6	2,13010-	8	5,40000-	6	2,11080-	8	127E	3	1	128
5,50000-	6	2,09030-	8	5,60000-	6	2,07070-	8	5,70000-	6	2,05150-	8	127F	3	1	129
5,80000-	6	2,03240-	8	5,90000-	6	2,01360-	8	6,00000-	6	1,99500-	8	127F	3	1	130
6,10000-	6	1,97670-	8	6,20000-	6	1,95960-	8	6,30000-	6	1,94279-	8	127E	3	1	131
6,40000-	6	1,92300-	8	6,50000-	6	1,90550-	8	6,60000-	6	1,88830-	8	127W	3	1	132
6,70000-	6	1,87120-	8	6,80000-	6	1,85440-	8	6,90000-	6	1,83780-	8	127E	3	1	133
7,00000-	6	1,82130-	8	7,10000-	6	1,80891-	8	7,20000-	6	1,78910-	8	127E	3	1	134
7,30000-	6	1,77330-	8	7,40000-	6	1,75777-	8	7,50000-	6	1,74233-	8	127E	3	1	135
7,60000-	6	1,72710-	8	7,70000-	6	1,71290-	8	7,80000-	6	1,69725-	8	127F	3	1	136
7,90000-	6	1,68250-	8	8,00000-	6	1,66810-	8	8,10000-	6	1,65380-	8	127E	3	1	137
8,20000-	6	1,63970-	8	8,30000-	6	1,62560-	8	8,40000-	6	1,61200-	8	127E	2	1	138
8,50000-	6	1,59850-	8	8,60000-	6	1,58510-	8	8,70000-	6	1,57180-	8	127E	3	1	139
8,80000-	6	1,55800-	8	8,90000-	6	1,54500-	8	9,00000-	6	1,53320-	8	127E	3	1	140
9,10000-	6	1,52260-	8	9,20000-	6	1,50020-	8	9,30000-	6	1,49900-	8	127F	3	1	141
9,40000-	6	1,46380-	8	9,50000-	6	1,47160-	8	9,60000-	6	1,46000-	8	127W	3	1	142
9,70000-	6	1,44840-	8	9,80000-	6	1,43690-	8	9,90000-	6	1,42990-	8	127E	3	1	143
1,00000-	7	1,41420-	8	1,01000-	7	1,40320-	8	1,02000-	7	1,39220-	8	127E	3	1	144
1,03000-	7	1,38140-	8	1,04000-	7	1,37070-	8	1,05000-	7	1,36010-	8	127F	3	1	145
1,06000-	7	1,34970-	8	1,07000-	7	1,33930-	8	1,08000-	7	1,32920-	8	127W	3	1	146
1,09000-	7	1,31910-	8	1,10000-	7	1,30910-	8	1,11000-	7	1,29930-	8	127W	3	1	147
1,12000-	7	1,28960-	8	1,13000-	7	1,28000-	8	1,14000-	7	1,27050-	8	127E	3	1	148
1,15000-	7	1,26120-	8	1,16000-	7	1,25190-	8	1,17000-	7	1,24280-	8	127F	3	1	149
1,18000-	7	1,23370-	8	1,19000-	7	1,22480-	8	1,20000-	7	1,21600-	8	127E	3	1	150
1,21000-	7	1,20720-	8	1,22000-	7	1,19860-	8	1,23000-	7	1,19010-	8	127F	3	1	151
1,24000-	7	1,18170-	8	1,25000-	7	1,17330-	8	1,26000-	7	1,16510-	8	127F	3	1	152
1,27000-	7	1,15700-	8	1,28000-	7	1,14900-	8	1,29000-	7	1,14100-	8	127W	3	1	153
1,30000-	7	1,13310-	8	1,31000-	7	1,12540-	8	1,32000-	7	1,11770-	8	127W	3	1	154
1,33000-	7	1,11010-	8	1,34000-	7	1,10260-	8	1,35000-	7	1,09520-	8	127E	3	1	155
1,36000-	7	1,08780-	8	1,37000-	7	1,08060-	8	1,38000-	7	1,07340-	8	127F	3	1	156
1,39000-	7	1,06640-	8	1,40000-	7	1,05940-	8	1,41000-	7	1,05250-	8	127W	3	1	157
1,42000-	7	1,04560-	8	1,43000-	7	1,03890-	8	1,44000-	7	1,03220-	8	127E	3	1	158
1,45000-	7	1,02560-	8	1,46000-	7	1,01900-	8	1,47000-	7	1,01260-	8	127E	3	1	159
1,48000-	7	1,00620-	8	1,49000-	7	9,99900-	1	1,50000-	7	9,93670-	1	127F	3	1	160
1,51000-	7	9,87910-	1	1,52000-	7	9,81420-	1	1,53000-	7	9,75480-	1	127E	3	1	161
1,54000-	7	9,69450-	1	1,55000-	7	9,63970-	1	1,56000-	7	9,57760-	1	127E	3	1	162
1,57000-	7	9,52010-	1	1,58000-	7	9,46340-	1	1,59000-	7	9,40725-	1	127E	3	1	163
1,60000-	7	9,39180-	1	1,61000-	7	9,29700-	1	1,62000-	7	9,24260-	1	127E	3	1	164
1,63000-	7	9,18930-	1	1,64000-	7	9,13640-	1	1,65000-	7	9,08410-	1	127E	3	1	165
1,66000-	7	9,03240-	1	1,67000-	7	8,98140-	1	1,68000-	7	8,93190-	1	127F	3	1	166
1,69000-	7	8,88110-	1	1,70000-	7	8,83190-	1	1,71000-	7	8,78320-	1	127E	3	1	167
1,72000-	7	8,73520-	1	1,73000-	7	8,68770-	1	1,74000-	7	8,64080-	1	127F	3	1	168
1,75000-	7	8,59440-	1	1,76000-	7	8,54660-	1	1,77000-	7	8,50340-	1	127E	3	1	169
1,78000-	7	8,45970-	1	1,79000-	7	8,41460-	1	1,80000-	7	8,37180-	1	127E	3	1	170
1,81000-	7	8,32800-	1	1,82000-	7	8,28950-	1	1,83000-	7	8,24990-	1	127F	3	1	171
1,84000-	7	8,20810-	1	1,85000-	7	8,16610-	1	1,86000-	7	8,12470-	1	127E	3	1	172
1,87000-	7	8,08460-	1	1,88000-	7	8,04410-	1	1,89000-	7	8,00390-	1	127E	3	1	173
1,90000-	7	7,96410-	1	1,91000-	7	7,92620-	1	1,92000-	7	7,88880-	1	127E	3	1	174
1,93000-	7	7,85180-	1	1,94000-	7	7,81540-	1	1,95000-	7	7,77940-	1	127F	3	1	175
1,96000-	7	7,74390-	1	1,97000-	7	7,70800-	1	1,98000-	7	7,67430-	1	127W	3	1	176
1,99000-	7	7,64010-	1	2,00000-	7	7,60690-	1					127W	3	1	177
												127E	3	2	178
2,04000-	3	4,00150-	0		0		99		8			0127W	3	2	179
8,00000-	0	8,00000-	0		0		0		1		297	127F	3	2	180
297			2									127E	3	2	181
1,00000-	5	7,99140-	1	2,50000-	2	7,90160-	1	1,00000-	8	7,90160-		1127F	3	2	182
1,00000-	1	7,90160-	1	1,00000-	2	7,90160-	1	1,00000-	3	7,90110-		1127E	3	2	183
5,00000-	3	7,58916-	1	1,00000-	4	7,58720-	1	2,00000-	4	7,56498-		1127E	3	2	184
4,00000-	4	7,38750-	1	6,00000-	4	7,60630-	1	8,00000-	4	7,62430-		1127E	3	2	185
1,00000-	5	7,66090-	1	2,00000-	5	7,81960-	1	3,00000-	5	8,00170-		1127E	3	2	186
2,50000-	5	8,40990-	1	3,00000-	5	9,00000-	1	3,50000-	5	9,08593-		1127E	3	2	187
4,00000-	5	1,10300-	0	4,00000-	5	1,25360-	0	5,00000-	5	1,45250-		0127E	3	2	188
5,50000-	5	1,71140-	0	6,00000-	5	2,74360-	0	6,25000-	5	2,20100-		0127E	3	2	189
6,50000-	5	2,46300-	0	6,75000-	5	2,70870-	0	7,00000-	0	2,97050-		0127E	3	2	190
7,25000-	5	3,27550-	0	7,50000-	5	3,50500-	0	7,75000-	5	3,93660-		0127E	3	2	191
8,00000-	5	4,30060-	0	8,25000-	5	4,67700-	0	8,50000-	5	5,06160-		0127E	3	2	192
8,75000-	5	5,44490-	0	9,00000-	5	5,82410-	0	9,25000-	5	6,18420-		0127E	3	2	193
9,50000-	5	6,51000-	0	9,75000-	5	6,81730-	0	1,00000-	6	7,07520-		0127E	3	2	194
1,07500-	6	7,28670-	0	1,15000-	6	7,44440-	0	1,22500-	6	7,58270-		0127E	3	2	195
1,30000-	6	7,42050-	0	1,37500-	6	7,40810-	0	1,45000-	6	7,43710-		0127E	3	2	196
1,37500-	6	7,57900-	0	1,30000-	6	7,40870-	0	1,22500-	6	7,38450-		0127E	3	2	197
1,20000-	6	7,27210-	0	1,27500-	6	7,13620-	0	1,30000-	6	6,99900-		0127E	3	2	198
1,32500-	6	6,63950-	0	1,35000-	6	6,00500-	0	1,37500-	6	6,52950-		0127W	3	2	199
1,40000-	6	6,37400-	0	1,45000-	6	6,07350-	0	1,50000-	6	6,76020-		0127E	3	2	200
1,55000-	6	5,92200-	0	1,60000-	6	5,27790-	0	1,65000-	6	5,05300-		0127E	3	2	201
1,70000-	6	4,84940-	0	1,75000-	6	4,66360-	0	1,80000-	6	4,49430-		0127E	3	2	202
1,85000-	6	4,34010-	0	1,90000-	6	4,19900-	0	1,95000-	6	4,37130-		0127E	3	2	203
2,00000-	6	3,95410-	0	2,05000-	6	3,86670-	0	2,10000-	6	3,74020-		0127E	3	2	204
2,15000-	6	3,65760-	0	2,20000-	6	3,57410-	0	2,25000-	6	3,49710-		0127E	3	2	205
2,30000-	6	3,42590-	0	2,35000-	6	3,35900-	0	2,40000-	6	3,29640-		0127E	3	2	206
2,45000-	6	3,24160-	0	2,50000-	6	3,18850-	0	2,55000-	6	3,13640-		0127E	3	2	207
2,60000-	6	3,08250-	0	2,65000-	6	3,04910-	0	2,70000-	6	3,00840-		0127E	3	2	208
2,75000-	6	2,97810-	0	2,80000-	6	2,93400-	0	2,85000-	6	2,90000-		0127E	3	2	209
2,90000-	6	2,86700-	0	2,95000-	6	2,83750-	0	3,00000-	6	2,80870-		0127E	3	2	210
3,10000-	6	2,75940-	0	3,20000-	6	2,70700-	0	3,30000-	6	2,66270-		0127E	3	2	211
3,40000-	6	2,62210-	0	3,50000-	6	2,56440-	0	3,60000-	6	2,54943-		0127E	3	2	212
3,70000-	6	2,51450-	0	3,80000-	6	2,46550-	0	3,90000-	6	2,45410-		0127W	3	2	213
4,00000-	6	2,42800-	0	4,10000-	6	2,40110-	0	4,20000-	6	2,37930-		0127E	3	2	214
4,30000-	6	2,35830-	0	4,40000-	6	2,32600-	0	4,50000-	6	2,30240-		0127E	3	2	215
4,60000-	6	2,27940-	0	4,70000-	6	2,25690-	0	4,80000-	6	2,23490-		0127E	3	2	216
4,90000-	6	2,21320-	0	5,00000-	6	2,19100-	0	5,10000-	6	2,17100-		0127E	3	2	217
5,20000-	6	2,15040-	0	5,30000-	6	2,13010-	0	5,40000-	6	2,11080-		0127E	3	2	218
5,50000-	6	2,09030-	0	5,60000-	6	2,07070-	0	5,70000-	6	2,05150-		0127E	3	2	219
5,80000-	6	2,03240-	0	5,90000-	6	2,01360-	0	6,00000-	6	1,99500-		0127E	3	2	220
6,10000-	6	1,97670-	0	6,20000-	6	1,91960-	0	6,30000-	6	1,94279-		0127W	3	2	221
6,40000-	6	1,92300-	0	6,50000-	6	1,90550-	0	6,60000-	6	1,88830-		0127E	3	2	222
6,70000-	6	1,87120-	0	6,80000-	6	1,85440-	0	6,90000-	6	1,83780-		0127W	3	2	223
7,00000-	6	1,82130-	0	7,10000-	6	1,80910-	0	7,20000-	6	1,78910-		0127F	3	2	224
7,30000-	6	1,77330-	0	7,40000-	6	1,75770-	0	7,50000-	6	1,74233-		0127E	3	2	225
7,60000-	6	1,72710-	0	7,70000-	6	1,71290-	0	7,80000-	6	1,69725-		0127E	3	2	226
7,90000-	6	1,68250-	0	8,00000-	6	1,66810-	0	8,10000-	6	1,65380-		0127E	3	2	227
8,20000-	6	1,63970-	0	8,30000-	6	1,62560-	0	8,40000-	6	1,61200-		0127E	3	2	228
8,50000-	6	1,59850-	0	8,60000-	6	1,58510-	0	8,70000-	6	1,57180-		0127E	3	2	229
8,80000-	6	1,55800-	0	8,90000-	6	1,54500-	0	9,00000-	6	1,53320-		0127E	3	2	230
9,10000-	6	1,52260-	0	9,20000-	6	1,50020-	0	9,30000-	6	1,49900-		0127E	3	2	231
9,40000-	6	1,46380-	0	9,50000-	6	1,47160-	0	9,60000-	6	1,46000-		0127E	3	2	232
9,70000-	6	1,44840-	0	9,80000-	6	1,43690-	0	9,90000-	6	1,42990-		0127E	3	2	233
1,00000-	7	1,41420-	0	1,01000-	7	1,40320-	0	1,02000-	7	1,39220-		0127W	3	2	234
1,03000-	7	1,38140-	0	1,04000-	7	1,37070-	0	1,05000-	7	1,36010-		0127W	3	2	235
1,06000-	7	1,34970-	0	1,07000-	7	1,33930-	0	1,08000-	7	1,32920-		0127W	3	2	236
1,09000-	7	1,31910-	0	1,10000-	7	1,30910-	0	1,11000-	7	1,29930-		0127F	3	2	237
1,12000-	7	1,28960-	0	1,13000-	7	1,28000-	0	1,14000-	7	1,27050-		0127E	3	2	238
1,15000-	7	1,26120-	0	1,16000-	7	1,25190-	0	1,17000-	7	1,24280-		0127E	3	2	239

```
1.140000e 7 1.23378e 8 1.19000e 7 1.22488e 8 1.28000e 7 1.21488e 01278 3  2  240
1.210000e 7 1.20778e 8 1.22000e 7 1.19860e 8 1.23000e 7 1.19712e 01278 3  2  241
1.240000e 7 1.18178e 8 1.25000e 7 1.17330e 8 1.26000e 7 1.16414e 01278 3  2  242
1.270000e 7 1.15760e 8 1.28000e 7 1.14499e 8 1.29000e 7 1.14188e 01278 3  2  243
1.300000e 7 1.13318e 8 1.31000e 7 1.12548e 8 1.32000e 7 1.11778e 01278 3  2  244
1.330000e 7 1.11518e 8 1.34000e 7 1.10268e 8 1.35000e 7 1.09928e 01278 3  2  245
1.360000e 7 1.08788e 8 1.37000e 7 1.08060e 8 1.38000e 7 1.07344e 01278 3  2  246
1.390000e 7 1.06640e 8 1.40000e 7 1.05994e 8 1.41000e 7 1.05298e 01278 3  2  247
1.420000e 7 1.04948e 8 1.43000e 7 1.03098e 8 1.44000e 7 1.03228e 01278 3  2  248
1.450000e 7 1.02598e 8 1.46000e 7 1.01968e 8 1.47000e 7 1.01268e 01278 3  2  249
1.480000e 7 1.00628e 8 1.49000e 7 9.99908e 1 1.50000e 7 9.93678e 11278 3  2  250
1.510000e 7 9.87918e 1 1.52000e 7 9.81428e 1 1.53000e 7 9.75488e 11278 3  2  251
1.540000e 7 9.69498e 1 1.55000e 7 9.63578e 1 1.56000e 7 9.57768e 11278 3  2  252
1.570000e 7 9.52018e 1 1.58000e 7 9.46348e 1 1.59000e 7 9.40928e 11278 3  2  253
1.600000e 7 9.35188e 1 1.61000e 7 9.29708e 1 1.62000e 7 9.24288e 11278 3  2  254
1.630000e 7 9.18938e 1 1.64000e 7 9.13648e 1 1.65000e 7 9.08418e 11278 3  2  255
1.660000e 7 9.03248e 1 1.67000e 7 8.98148e 1 1.68000e 7 8.93188e 11278 3  2  256
1.690000e 7 8.88118e 1 1.70000e 7 8.83198e 1 1.71000e 7 8.78328e 11278 3  2  257
1.720000e 7 8.73528e 1 1.73000e 7 8.68778e 1 1.74000e 7 8.64048e 11278 3  2  258
1.750000e 7 8.59448e 1 1.76000e 7 8.54868e 1 1.77000e 7 8.50348e 11278 3  2  259
1.780000e 7 8.45878e 1 1.79000e 7 8.41468e 1 1.80000e 7 8.37188e 11278 3  2  260
1.810000e 7 8.32888e 1 1.82000e 7 8.28598e 1 1.83000e 7 8.24398e 11278 3  2  261
1.840000e 7 8.20318e 1 1.85000e 7 8.16118e 1 1.86000e 7 8.12078e 11278 3  2  262
1.870000e 7 8.08008e 1 1.88000e 7 8.04148e 1 1.89000e 7 8.00258e 11278 3  2  263
1.900000e 7 7.96418e 1 1.91000e 7 7.92628e 1 1.92000e 7 7.88888e 11278 3  2  264
1.930000e 7 7.85188e 1 1.94000e 7 7.81548e 1 1.95000e 7 7.77948e 11278 3  2  265
1.960000e 7 7.74298e 1 1.97000e 7 7.70808e 1 1.98000e 7 7.67438e 11278 3  2  266
1.990000e 7 7.64018e 1 2.00000e 7 7.60698e 1                      1278 3  2  267
                                                                  1278 3  0  268
2.040000e 3 4.00158e 0          0          0          0          01278 3991 269
8.000000e 0 0.000000e 0          0          0          1         401278 3991 270
          46          2                                            1278 3991 271
1.000000e 5 1.49887e 1 1.00000e 0 1.49887e 1 6.00000e 4 9.04929e 11278 3991 272
1.000000e 5 4.81879e 2 1.50000e 5-8.72487e 2 2.00000e 5-8.51369e 21278 3991 273
2.000000e 5-1.49625e 1 3.00000e 5-1.70263e 1 3.50000e 5-2.39948e 11278 3991 274
4.000000e 5-4.66678e 1 4.50000e 5-8.76213e 1 5.00000e 5-4.74939e 11278 3991 275
5.500000e 5-2.96321e 1 6.00000e 5-8.27997e 1 6.50000e 5-1.89926e 11278 3991 276
7.000000e 5-1.46778e 1 7.75000e 5-7.85920e 2 8.50000e 5-1.18882e 21278 3991 277
9.250000e 5 9.17974e 2 1.00000e 6 1.70143e 1 1.27500e 6 1.57888e 11278 3991 278
1.175000e 6 2.13280e 1 1.27500e 6 2.36619e 1 1.40000e 6 3.23341e 11278 3991 279
1.550000e 6 3.43488e 1 1.70000e 6 3.72760e 1 1.85000e 6 3.94954e 11278 3991 280
2.050000e 6 4.14458e 1 2.25000e 6 4.28188e 1 2.50000e 6 4.39748e 11278 3991 281
2.750000e 6 4.47708e 1 3.00000e 6 4.53451e 1 3.40000e 6 4.62328e 11278 3991 282
4.100000e 6 4.77341e 1 5.30000e 6 5.24549e 1 6.20000e 6 5.20035e 11278 3991 283
6.500000e 6 5.34779e 1 7.70000e 6 5.46899e 1 8.50000e 6 5.50953e 11278 3991 284
9.000000e 6 5.64091e 1 1.07000e 7 5.72283e 1 1.20000e 7 5.78728e 11278 3991 285
1.350000e 7 5.84319e 1 1.54000e 7 5.80925e 1 1.81000e 7 5.97400e 11278 3991 286
2.000000e 7 6.03478e 1                                            1278 3991 287
                                                                  1278 3  0  288
2.040000e 3 4.00158e 0          0          0          0          01278 3992 289
8.000000e 0 0.000000e 0          0          0          1         401278 3992 290
          46          2                                            1278 3992 291
1.000000e 5 4.29188e 1 1.00000e 0 4.29188e 1 6.00000e 4 4.62662e 11278 3992 292
1.000000e 5 4.89932e 1 1.50000e 5 9.12044e 1 2.20000e 5 5.59770e 11278 3992 293
2.900000e 5 5.86731e 1 3.00000e 5 6.13659e 1 3.50000e 5 6.34761e 11278 3992 294
4.800000e 5 6.48600e 1 4.50000e 5 6.54441e 1 5.20000e 5 6.52360e 11278 3992 295
5.500000e 5 6.43133e 1 6.00000e 5 6.28892e 1 6.50000e 5 6.00021e 11278 3992 296
7.800000e 5 5.66664e 1 7.75000e 5 5.31404e 1 8.50000e 5 5.17302e 11278 3992 297
9.250000e 5 4.68097e 1 1.00000e 6 4.36289e 1 1.27500e 6 4.38467e 11278 3992 298
1.175000e 6 4.42356e 1 1.27500e 6 3.79139e 1 1.40000e 6 3.50748e 11278 3992 299
1.550000e 6 3.35790e 1 1.70000e 6 3.28738e 1 1.85000e 6 3.20764e 11278 3992 300
2.050000e 6 2.99416e 1 2.25000e 6 2.92402e 1 2.50000e 6 2.86498e 11278 3992 301
2.750000e 6 2.62423e 1 3.00000e 6 2.70274e 1 3.40000e 6 2.74419e 11278 3992 302
4.100000e 6 2.47174e 1 5.30000e 6 2.53139e 1 6.20000e 6 2.44692e 11278 3992 303
6.500000e 6 2.37540e 1 7.70000e 6 2.31732e 1 8.50000e 6 2.26900e 11278 3992 304
9.000000e 6 2.22811e 1 1.07000e 7 2.18260e 1 1.20000e 7 2.14944e 11278 3992 305
1.350000e 7 2.12877e 1 1.54000e 7 2.09109e 1 1.81000e 7 2.08389e 11278 3992 306
2.000000e 7 2.02246e 1                                            1278 3992 307
                                                                  1278 3  0  308
2.040000e 3 4.00158e 0          0          0          0          01278 3993 309
8.000000e 0 0.000000e 0          0          0          1         401278 3993 310
          46          2                                            1278 3993 311
1.000000e 5 3.89677e 1 1.00000e 0 3.89677e 1 6.00000e 4 3.82920e 11278 3993 312
1.000000e 5 3.29081e 1 1.50000e 5 3.39193e 1 2.20000e 5 3.47671e 11278 3993 313
2.900000e 5 3.55384e 1 3.00000e 5 3.61944e 1 3.50000e 5 3.67919e 11278 3993 314
4.800000e 5 3.71944e 1 4.50000e 5 3.75288e 1 5.20000e 5 3.77404e 11278 3993 315
5.500000e 5 3.78634e 1 6.00000e 5 3.78779e 1 6.50000e 5 3.78013e 11278 3993 316
7.800000e 5 3.76438e 1 7.75000e 5 3.73793e 1 8.50000e 5 3.67036e 11278 3993 317
9.250000e 5 3.62384e 1 1.00000e 6 3.55913e 1 1.27500e 6 3.49328e 11278 3993 318
1.175000e 6 3.40418e 1 1.27500e 6 3.31713e 1 1.40000e 6 3.21447e 11278 3993 319
1.550000e 6 3.10000e 1 1.70000e 6 3.01692e 1 1.85000e 6 2.92223e 11278 3993 320
2.050000e 6 2.84922e 1 2.25000e 6 2.76748e 1 2.50000e 6 2.73428e 11278 3993 321
2.750000e 6 2.70745e 1 3.00000e 6 2.68020e 1 3.40000e 6 2.66928e 11278 3993 322
4.100000e 6 2.66064e 1 5.30000e 6 2.65721e 1 6.20000e 6 2.64767e 11278 3993 323
6.500000e 6 2.63434e 1 7.70000e 6 2.61968e 1 8.50000e 6 2.60321e 11278 3993 324
9.000000e 6 2.58681e 1 1.07000e 7 2.57879e 1 1.20000e 7 2.55924e 11276 3993 325
1.350000e 7 2.54118e 1 1.54000e 7 2.52090e 1 1.81000e 7 2.50061e 11278 3993 326
2.000000e 7 2.48701e 1                                            1278 3993 327
                                                                  1278 3  0  328
                                                                  1278 0  0  329
2.040000e 3 4.00158e 0          1          1          0          01278 4  2  330
8.000000e 0 4.00158e 0          0          2        188          01278 4  2  331
1.000000e 0 1.44640e 1 1.26848e 2-4.97346e 9 8.00000e 0 8.00000e 01278 4  2  332
8.000000e 0 0.000000e 0 0.000000e 0 0.00000e 0 0.00000e 0 9.00028e 11278 4  2  333
2.944479e 1 4.28250e 2 3.82497e 3 1.11528e 4-4.66334e 6 8.00000e 01278 4  2  334
8.000000e 0 0.000000e 0 0.00000e 0-1.97486e 1 9.23949e 1 4.07601e 11278 4  2  335
8.720110e 2 1.07922e 2 8.20128e 4 8.00000e 0 8.00000e 0 8.00000e 01278 4  2  336
8.000000e 0 3.32958e 2-2.71243e 1 8.14168e 1 8.23231e 1 1.43408e 11278 4  2  337
2.441190e 2 7.80000e 0 8.00000e 0 8.00000e 0 8.00000e 0-8.38600e 31278 4  2  338
7.752921e 2-3.62041e 1 7.85900e 1 9.77948e 1 2.87957e 1 8.00000e 01278 4  2  339
8.000000e 0 9.00000e 0 8.00000e 0 2.23972e 3-8.19392e 2 1.27483e 11278 4  2  340
-4.32897e 1 9.70000e 1 8.00000e 0 8.00000e 0 8.00000e 0 8.00000e 01278 4  2  341
8.000000e 0-4.99220e 4 9.87300e 3-4.37999e 2 1.72141e 1-4.77077e 11278 4  2  342
6.325040e 1 3.14467e 2 3.82909e 3-4.90000e 4 8.00000e 0 1.23111e 41278 4  2  343
-1.59070e 3 1.32906e 2-4.63908e 2 2.37034e 1-4.76794e 1 3.76078e 11278 4  2  344
7.881960e 2 1.83375e 2 8.00000e 0-4.74408e 9 4.21349e 4-4.44983e 31278 4  2  345
2.238872e 2-9.77337e 2 2.90017e 1-9.87819e 1 1.84762e 1 1.17490e 11278 4  2  346
8.000000e 0 7.59631e 0-1.18200e 4 1.34444e 3-7.12109e 3 3.41398e 21278 4  2  347
-1.33892e 1 3.40418e 1-8.22781e 1 2.27390e 1                      1278 4  2  348
8.000000e 2 0.000000e 0          0          0          1         401278 4  2  349
          46          2                                            1278 4  2  350
8.000000e 0 1.00000e 0          0          0          2         01278 4  2  351
8.000000e 0 0.000000e 0                                           1278 4  2  352
8.000000e 0 1.00000e 0          0          0          2         01278 4  2  353
8.000000e 0 2.00000e 0                                            1278 4  2  354
8.000000e 0 4.00000e 4          0          0          4         01278 4  2  355
-7.97300e 2 1.93160e 3-3.39478e 8 4.31490e-11 4.93000e-12-4.18450e-121278 4  2  356
8.000000e 0 1.00000e 5          5          0          4         01278 4  2  357
-1.26640e 1 4.40440e 3-1.58878e 7 5.36499e-18 9.20018e-12-9.80178e-121278 4  2  358
8.000000e 0 1.00000e 5          5          0          4         01278 4  2  359
```

```
-1,99718- 1 1,07556- 1-9,48898- 7 2701240- 9 1740438-11-9,62988-12127F 4  2  360
 0,00000- 0 2,00000- 9             0            0            0        0127F 4  2  361
-2,01278- 1 2,01338- 2-1,28878- 6 6799418- 9 2716818-11-1,48418-11127F 4  2  362
 0,00000- 0 2,50000- 9             0            0            0        0127F 4  2  363
-3,21928- 1 3,27298- 2-2,46678- 6 2718568- 8 2789118-11-1,89998-11127F 4  2  364
 0,00000- 0 3,00000- 9             0            0            0        0127F 4  2  365
-3,73998- 1 4,03808- 2-4,13378- 6 4743298- 8 3761399-11-2,41888-11127F 4  2  366
 0,00000- 0 3,50000- 9             0            0            0        0127F 4  2  367
-4,17678- 1 4,63408- 2-6,24898- 6 7788268- 8 4727978-11-2,93988-11127F 4  2  368
 0,00000- 0 4,00000- 9             0            0            0        0127F 4  2  369
-4,38448- 1 5,95378- 2-8,70348- 6 1726918- 7 4782868-11-3,42918-11127F 4  2  370
 0,00000- 0 4,50000- 9             0            0            0        0127F 4  2  371
-4,47818- 1 1,06828- 1-1,13298- 5 1786788- 7 5721928-11-3,86668-11127F 4  2  372
 0,00000- 0 5,00000- 9             0            0            0        0127F 4  2  373
-4,30548- 1 1,25478- 1-1,39238- 5 2797158- 7 5741248-11-4,23318-11127F 4  2  374
 0,00000- 0 5,50000- 9             0            0            0        0127F 4  2  375
-4,17878- 1 1,43348- 1-1,62868- 5 3733618- 7 5741248-11-4,51148-11127F 4  2  376
 0,00000- 0 6,00000- 9             0            0            0        0127F 4  2  377
-3,84818- 1 1,59918- 1-1,82538- 5 4711278- 7 5724928-11-4,69798-11127F 4  2  378
 0,00000- 0 6,50000- 9             0            0            0        0127F 4  2  379
-3,43628- 1 1,71828- 1-1,97898- 5 4789198- 7 4793278-11-4,79738-11127F 4  2  380
 0,00000- 0 7,00000- 9             0            0            0        0127F 4  2  381
-2,97498- 1 1,81948- 1-2,09988- 5 5788718- 7 4769918-11-4,81998-11127F 4  2  382
 0,00000- 0 7,75000- 9             0            0            0        0127F 4  2  383
-2,24868- 1 1,92388- 1-2,28978- 5 6729998- 7 3764068-11-4,73988-11127F 4  2  384
 0,00000- 0 8,50000- 9             0            0            0        0127F 4  2  385
-1,53868- 1 1,97858- 1-1,99328- 5 6764768- 7 2756288-11-4,55778-11127F 4  2  386
 0,00000- 0 9,25000- 9             0            0            0        0127F 4  2  387
-8,83218- 2 1,99948- 1-1,08148- 5 6763918- 7 1713798-11-4,31958-11127F 4  2  388
 0,00000- 0 1,00000- 0             0            0            0        0127F 4  2  389
-2,99478- 2 1,98918- 1-1,53198- 5 6719898- 7-8725388-12-4,01828-11127F 4  2  390
 0,00000- 0 1,07500- 0             0            0            0        0127F 4  2  391
2,89188- 2 1,95438- 1-1,20148- 5 5733568- 7-3761308-11-3,68818-11127F 4  2  392
 0,00000- 0 1,17500- 0             0            0            0        0127F 4  2  393
7,77718- 2 1,90818- 1-6,07488- 4 3783738- 7-9726118-11-3,17988-11127F 4  2  394
 0,00000- 0 1,27500- 0             0            0            0        0127F 4  2  395
1,23678- 1 1,83388- 1-1,47388- 4 1783168- 7-1701198-10-2,55598-11127F 4  2  396
 0,00000- 0 1,40000- 0             0            0            0        0127F 4  2  397
1,68688- 1 1,74488- 1 8,71878- 4-3782638- 7-3736868-10-1,49988-11127F 4  2  398
 0,00000- 0 1,55000- 0             0            0            0        0127F 4  2  399
2,08488- 1 1,63918- 1 1,43798- 3-9711748- 7-7782218-10 4,64718-12127F 4  2  400
 0,00000- 0 1,70000- 0             0            0            0        0127F 4  2  401
2,37238- 1 1,54268- 1 2,27368- 3-1763938- 6-1724968- 9 3,69758-11127F 4  2  402
 0,00000- 0 1,85000- 0             0            0            0        0127F 4  2  403
2,58888- 1 1,45748- 1 3,09488- 3-2749148- 6-2736358- 9 8,94178-11127F 4  2  404
 0,00000- 0 2,05000- 0             0            0            0        0127F 4  2  405
2,77498- 1 1,36378- 1 3,98828- 3-3764438- 6-2768388- 9 2,08118-10127F 4  2  406
 0,00000- 0 2,25000- 0             0            0            0        0127F 4  2  407
2,95998- 1 1,29348- 1 4,75628- 3-4798889- 6-6786638- 9 4,12968-10127F 4  2  408
 0,00000- 0 2,50000- 0             0            0            0        0127F 4  2  409
3,01838- 1 1,23718- 1 5,30418- 3-4791998- 6-1783648- 8 8,92768-10127F 4  2  410
 0,00000- 0 2,75000- 0             0            0            0        0127F 4  2  411
3,09378- 1 1,21318- 1 6,81268- 3-5780878- 6-1763818- 8 1,48978- 9127F 4  2  412
 0,00000- 0 3,00000- 0             0            0            0        0127F 4  2  413
3,19848- 1 1,21678- 1 6,38288- 3-6799648- 6-2743148- 6 2,82868- 9127F 4  2  414
 0,00000- 0 3,40000- 0             0            0            0        0127F 4  2  415
3,25998- 1 1,26728- 1 6,37698- 3-1718818- 5-4711368- 8 6,22458- 9127F 4  2  416
 0,00000- 0 4,10000- 0             0            0            0        0127F 4  2  417
3,43918- 1 1,43158- 1 5,55138- 3-1798268- 5-8781898- 8 1,99238- 8127F 4  2  418
 0,00000- 0 5,30000- 0             0            0            0        0127F 4  2  419
3,77978- 1 1,74448- 1 1,44818- 3-2791768- 5-1713178- 7 9,88998- 8127F 4  2  420
 0,00000- 0 6,10000- 0             0            0            0        0127F 4  2  421
3,97478- 1 1,91448- 1-3,19398- 3-3727218- 5 2741698- 8 2,26818- 7127F 4  2  422
 0,00000- 0 6,90000- 0             0            0            0        0127F 4  2  423
4,14838- 1 2,04278- 1-9,02978- 3-3798338- 5 3731498- 7 4,81968- 7127F 4  2  424
 0,00000- 0 7,70000- 0             0            0            0        0127F 4  2  425
4,27438- 1 2,13448- 1-1,91188- 4-4796778- 5 1779888- 6 9,41698- 7127F 4  2  426
 0,00000- 0 8,60000- 0             0            0            0        0127F 4  2  427
4,39338- 1 2,21738- 1-2,85328- 4-3714528- 5 4786488- 6 1,84618- 6127F 4  2  428
 0,00000- 0 9,40000- 0             0            0            0        0127F 4  2  429
4,49998- 1 2,26088- 1-2,16748- 4 1741928- 6 1718888- 5 3,39738- 6127F 4  2  430
 0,00000- 0 1,47000- 7             0            0            0        0127F 4  2  431
4,57918- 1 2,32988- 1-1,19598- 4 7792128- 5 2778938- 5 6,91768- 6127F 4  2  432
 0,00000- 0 1,28000- 7             0            0            0        0127F 4  2  433
4,65388- 1 2,37848- 1 2,32988- 4 2792948- 4 6718888- 5 1,37258- 5127F 4  2  434
 0,00000- 0 1,35000- 7             0            0            0        0127F 4  2  435
4,71698- 1 2,40488- 1 1,18998- 3 6718448- 4 1739988- 4 2,74988- 5127F 4  2  436
 0,00000- 0 1,54000- 7             0            0            0        0127F 4  2  437
4,79148- 1 2,44798- 1 3,23998- 3 1749928- 3 3739868- 4 5,89868- 5127F 4  2  438
 0,00000- 0 1,41000- 7             0            0            0        0127F 4  2  439
4,86998- 1 2,51178- 1 8,89898- 3 3791798- 3 9731788- 4 1,45878- 4127F 4  2  440
 0,00000- 0 2,00000- 7             0            0            0        0127F 4  2  441
4,94878- 1 2,90878- 1 1,39498- 2 9792988- 3 1768778- 3 2,46398- 4127F 4  2  442
                                                                   127F 4  6  443
                                                                   127F 8  8  444
                                                                      0  0  0  445
                                                                     -1  0  0    0
```

APPENDIX L

Sample of Interpreted Data Set

The following is an interpreted listing of the ENDF data set for Helium-4. This listing was obtained using the LISTFC code (See Appendix I). Since this is an example, not all of the angular distributions have been listed.

```
2-HE-  4 LASL        EVAL-OCT73 WISLEY, HALE, YOUNG
                     DIST-FEB73

*****************   DNA 4504   MOD 6   ******************

HE-4    LASL-OCT.1973    R.A.WISLEY, G.M.HALE, P.G.YOUNG

MF=3 ------------- SMOOTH CROSS SECTIONS ----------------------

        THE 2200 M/S CROSS SECTIONS ARE AS FOLLOWS.
            MT=1   SIGMA= 2.73916 BARNS
            MT=2   SIGMA= 4.73916 BARNS

  MT=1        TOTAL CROSS SECTION
        SEE DISCUSSION UNDER MT=2 BELOW

  MT=2        ELASTIC SCATTERING CROSS SECTION
        ALTHOUGH THE ONLY REACTION POSSIBLE FOR NEUTRONS INCIDENT
      ON HE-4 BELOW 20 MEV IS ELASTIC SCATTERING, THE MAJORITY
      OF THE N-4HE DATA IS RATHER IMPRECISE.  IN ORDER TO OVER-
      COME THIS PROBLEM, AN R-MATRIX ANALYSIS WAS PERFORMED WITH
      A DATA SET WHICH INCLUDED NOT ONLY THE N-4HE DATA BUT ALSO
      VERY PRECISE P-4HE DATA.  ALL THE AVAILABLE N-4HE AND P-4HE
      DATA BELOW 20 MEV WERE CONSIDERED IN THE ANALYSIS.  SINCE
      THE PREVIOUS EVALUATION WAS COMPLETED IN 1968, SEVERAL
      N-4HE ELASTIC SCATTERING MEASUREMENTS HAVE BEEN DONE.  THE
      MOST SIGNIFICANT OF THESE ARE THE LOW ENERGY NEUTRON CROSS
      SECTION OF ROMER (R669),THE RPI TOTAL CROSS SECTION
      MEASUREMENT (G073), WHICH COVER THE RANGE ER=0.7-20 MEV,
      AND THE RELATIVE ANGULAR DISTRIBUTIONS OF MORGAN (M066).
      A COMPLETE LIST OF REFERENCES FOR THE N-4HE DATA USED IS
      GIVEN BELOW.  THE P-4HE DATA WAS SELECTED TO SATISFY VERY
      STRINGENT STATISTICAL CRITERIA AND WE BELIEVE THE POSSIBLE
      ERRORS OF THE PREDICTED VALUES FOR THE P-4HE SCATTERING
      TO BE LESS THAN 1.0 PER CENT.  A SIMPLE MODEL FOR THE
      CHARGE DIFFERENCES BETWEEN THE N-4HE AND P-4HE SYSTEMS
      WAS ASSUMED AND THE N-4HE AND P-4HE DATA SETS WERE
      SIMULTANEOUSLY ANALYZED.  THE VALUES OF THE CROSS SECTIONS
      AND ANGULAR DISTRIBUTIONS CONTAINED IN FILES 3 AND 4 ARE
      PROBABLY ACCURATE TO WITHIN 2.0 PERCENT.

MF=4 ------------- NEUTRON ANGULAR DISTRIBUTIONS ----------------

  MT=2        ELASTIC SCATTERING ANGULAR DISTRIBUTIONS
        OBTAINED FROM THE R-MATRIX ANALYSIS DESCRIBED ABOVE
      UNDER MF=3,MT=2.  LEGENDRE POLYNOMIAL REPRESENTATION USED.

------------------------ REFERENCES-----------------------------

AU62   S.M.AUSTIN ET AL., PHYS.REV. 126(1962) 1532.
BR72   W.B.BROSTE ET AL., PHYS.REV. C5(1972) 761.
BU64   F.W.BUESER ET AL., NUCL.PHYS. 58(1964) 593.
CR72   D.S.CRAMER + L.CRANBERG, NUCL.PHYS. A188(1972) 273.
FA63   U.FASOLI + G.ZAGO, NUOVO CIMENTO 30(1963) 1169.
GO73   C.A.GOULDING ET AL., BULL.AM.PHYS.SOC. 18(1973) 538.
HO66   B.HOOP,JR. + H.H.BARSCHALL, NUCL.PHYS. 83(1966) 65.
JE66   R.W.JEWELL ET AL., PHYS.REV. 142(1966) 647.
MA63   T.W.MAY ET AL., NUCL.PHYS. 49(1963) 17.(REV. MO68 + SA66)
MO66   G.L.MORGAN + R.L.WALTER, PHYS.REV. 166(1968) 1124.
MI71   A.MILLER ET AL., PHYS.REV. C4(1971) 30. (REV. 9/72)
RO69   D.C.ROMER ET AL., NUCL.PHYS. 133(1969) 418.
SA66   J.R.SANDERS ET AL., PHYS.REV. 148(1966) 1142.
SE53   J.D.SEAGRAVE, PHYS.REV. 92(1953) 1272.
SH59   G.F.SHAW, PROC.PHYS.SOC.(LONDON) 66(1959) 43.
SH64   R.E.SHAMU + J.D.ENGLISH, PHYS.REV. 133(1964) 899.
SM94   J.R.SMITH, PHYS.REV. 95(1954) 739.
ST78   T.STANNBACH ET AL., PHYS.REV. C2(1978) 434.
WH97   R.E.WHITE + F.J.W.FARLEY, NUCL.PHYS. 3(1957) 476.
YO63   P.G.YOUNG ET AL., AUST.J.PHYS. 16(1963) 139.
```

HELIUM-4 — TABLE OF CONTENTS — GENERAL INFORMATION — ENDF/B MATERIAL NO. 1278

DATA TYPE	REACTION	CARDS
GENERAL INFORMATION	COMMENTS	72
	TABLE OF CONTENTS	6
RESONANCE PARAMETERS	RESONANCE DATA	4
NEUTRON CROSS SECTION	TOTAL	89
	ELASTIC	89
	N, N'A	19
	N, I	19
	GAMMA	19
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS	ELASTIC	113

HELIUM-4 — RESONANCE DATA — RESONANCE PARAMETERS — ENDF/B MATERIAL NO. 1279

```
IDENTOPE-----------------------HELIUM-4
FRACTIONAL ABUNDANCE---------- 1.0000E+00
NUMBER OF ENERGY RANGES------------     1

ENERGY RANGE NUMBER----------------     1   RESOLVED SINGLE-LEVEL BREIT-WIGNER PARAMETERS
LOWER ENERGY LIMIT (EV)------------ 1.0000E-05
UPPER ENERGY LIMIT (EV)------------ 1.0000E+05
NUCLEAR SPIN----------------------- 0.0000E+00
SPIN SCATTERING LENGTH (AU)-------- 2.4579E-01
NUMBER OF L STATES----------------     0
```

NO RESONANCE PARAMETERS GIVEN FOR THIS MATERIAL

HELIUM-4 — TOTAL — NEUTRON CROSS SECTION — ENDF/B MATERIAL NO. 1278

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 297	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
1	1.0000E+05	7.9916E-01	2.5300E+02	7.9916E-01	1.0000E+00	7.9916E-01	1.0000E+01	7.9916E-01	1.0000E+02	7.9916E-01
6	1.0000E+03	7.9911E-01	5.0000E+03	7.9801E-01	1.0000E+04	7.9672E-01	2.0000E+04	7.5849E-01	4.2000E+04	7.5079E-01
11	6.0000E+04	7.6843E-01	8.0000E+04	7.6243E-01	1.0000E+05	7.4690E-01	1.5000E+05	7.0198E-01	2.0000E+05	6.2817E-01
16	2.5000E+05	5.4699E-01	3.0000E+05	4.8080E-01	3.5000E+05	9.9859E-01	4.0000E+05	1.1030E+00	4.5000E+05	1.2536E+00
21	5.0000E+05	1.4525E+00	5.5000E+05	1.7114E+00	6.0000E+05	2.0438E+00	6.2500E+05	2.2410E+00	6.5000E+05	2.4630E+00
26	6.7500E+05	2.7817E+00	7.0000E+05	2.9705E+00	7.2500E+05	3.2739E+00	7.5000E+05	3.5950E+00	7.7500E+05	3.9306E+00
31	8.0000E+05	4.3806E+00	8.2500E+05	4.6779E+00	8.5000E+05	5.0610E+00	8.7500E+05	5.4469E+00	9.0000E+05	5.8241E+00
36	9.2500E+05	6.1842E+00	9.5000E+05	6.5180E+00	9.7500E+05	6.8173E+00	1.0000E+06	7.2752E+00	1.0250E+06	7.2867E+00
41	1.0500E+06	7.4494E+00	1.0750E+06	7.9427E+00	1.1000E+06	7.6269E+00	1.1250E+06	7.6921E+00	1.1500E+06	7.6321E+00
46	1.1750E+06	7.5798E+00	1.2000E+06	7.4987E+00	1.2250E+06	7.3045E+00	1.2500E+06	7.2721E+00	1.2750E+06	7.1362E+00
51	1.3000E+06	6.9989E+00	1.3250E+06	6.8305E+00	1.3500E+06	6.6658E+00	1.3750E+06	6.5209E+00	1.4000E+06	6.3748E+00
56	1.4500E+06	6.0735E+00	1.5000E+06	5.7882E+00	1.5500E+06	5.5276E+00	1.6000E+06	5.2779E+00	1.6500E+06	5.0539E+00
61	1.7000E+06	4.8496E+00	1.7500E+06	4.6436E+00	1.8000E+06	4.4443E+00	1.8500E+06	4.3491E+00	1.9000E+06	4.1996E+00
66	1.9500E+06	4.0713E+00	2.0000E+06	3.9541E+00	2.0500E+06	3.8467E+00	2.1000E+06	3.7402E+00	2.1500E+06	3.6376E+00
71	2.2000E+06	3.5741E+00	2.2500E+06	3.4971E+00	2.3000E+06	3.4299E+00	2.3500E+06	3.3900E+00	2.4000E+06	3.2906E+00
76	2.4500E+06	3.2416E+00	2.5000E+06	3.1805E+00	2.5500E+06	3.1309E+00	2.6000E+06	3.0929E+00	2.6500E+06	3.0491E+00
81	2.7000E+06	3.0084E+00	2.7500E+06	2.9791E+00	2.8000E+06	2.9540E+00	2.8500E+06	2.9000E+00	2.9000E+06	2.8679E+00
86	2.9500E+06	2.8375E+00	3.0000E+06	2.8087E+00	3.1000E+06	2.7594E+00	3.2000E+06	2.7070E+00	3.3000E+06	2.6627E+00
91	3.4000E+06	2.6221E+00	3.5000E+06	2.5844E+00	3.6000E+06	2.5474E+00	3.7000E+06	2.5165E+00	3.8000E+06	2.4859E+00
96	3.9000E+06	2.4561E+00	4.0000E+06	2.4280E+00	4.1000E+06	2.4011E+00	4.2000E+06	2.3753E+00	4.3000E+06	2.3503E+00
101	4.4000E+06	2.3260E+00	4.5000E+06	2.3024E+00	4.6000E+06	2.2794E+00	4.7000E+06	2.2569E+00	4.8000E+06	2.2349E+00
106	4.9000E+06	2.2132E+00	5.0000E+06	2.1919E+00	5.1000E+06	2.1710E+00	5.2000E+06	2.1504E+00	5.3000E+06	2.1301E+00
111	5.4000E+06	2.1100E+00	5.5000E+06	2.0903E+00	5.6000E+06	2.0707E+00	5.7000E+06	2.0515E+00	5.8000E+06	2.0324E+00
116	5.9000E+06	2.0136E+00	6.0000E+06	1.9950E+00	6.1000E+06	1.9767E+00	6.2000E+06	1.9586E+00	6.3000E+06	1.9407E+00
121	6.4000E+06	1.9230E+00	6.5000E+06	1.9059E+00	6.6000E+06	1.8883E+00	6.7000E+06	1.8712E+00	6.8000E+06	1.8544E+00
126	6.9000E+06	1.8378E+00	7.0000E+06	1.8213E+00	7.1000E+06	1.8051E+00	7.2000E+06	1.7891E+00	7.3000E+06	1.7733E+00
131	7.4000E+06	1.7577E+00	7.5000E+06	1.7423E+00	7.6000E+06	1.7271E+00	7.7000E+06	1.7120E+00	7.8000E+06	1.6972E+00
136	7.9000E+06	1.6825E+00	8.0000E+06	1.6681E+00	8.1000E+06	1.6538E+00	8.2000E+06	1.6397E+00	8.3000E+06	1.6258E+00
141	8.4000E+06	1.6120E+00	8.5000E+06	1.5985E+00	8.6000E+06	1.5851E+00	8.7000E+06	1.5718E+00	8.8000E+06	1.5586E+00
146	8.9000E+06	1.5455E+00	9.0000E+06	1.5332E+00	9.1000E+06	1.5204E+00	9.2000E+06	1.5082E+00	9.3000E+06	1.4959E+00
151	9.4000E+06	1.4838E+00	9.5000E+06	1.4718E+00	9.6000E+06	1.4600E+00	9.7000E+06	1.4484E+00	9.8000E+06	1.4369E+00
156	9.9000E+06	1.4255E+00	1.0000E+07	1.4142E+00	1.0100E+07	1.4032E+00	1.0200E+07	1.3922E+00	1.0300E+07	1.3814E+00
161	1.0400E+07	1.3707E+00	1.0500E+07	1.3601E+00	1.0600E+07	1.3497E+00	1.0700E+07	1.3393E+00	1.0800E+07	1.3292E+00
166	1.0900E+07	1.3191E+00	1.1000E+07	1.3091E+00	1.1100E+07	1.2993E+00	1.1200E+07	1.2896E+00	1.1300E+07	1.2800E+00
171	1.1400E+07	1.2705E+00	1.1500E+07	1.2612E+00	1.1600E+07	1.2519E+00	1.1700E+07	1.2428E+00	1.1800E+07	1.2337E+00
176	1.1900E+07	1.2248E+00	1.2000E+07	1.2160E+00	1.2100E+07	1.2072E+00	1.2200E+07	1.1986E+00	1.2300E+07	1.1901E+00
181	1.2400E+07	1.1817E+00	1.2500E+07	1.1733E+00	1.2600E+07	1.1651E+00	1.2700E+07	1.1570E+00	1.2800E+07	1.1489E+00
186	1.2900E+07	1.1410E+00	1.3000E+07	1.1331E+00	1.3100E+07	1.1254E+00	1.3200E+07	1.1177E+00	1.3300E+07	1.1101E+00
191	1.3400E+07	1.1026E+00	1.3500E+07	1.0952E+00	1.3600E+07	1.0878E+00	1.3700E+07	1.0806E+00	1.3800E+07	1.0734E+00
196	1.3900E+07	1.0664E+00	1.4000E+07	1.0594E+00	1.4100E+07	1.0525E+00	1.4200E+07	1.0456E+00	1.4300E+07	1.0389E+00
201	1.4400E+07	1.0322E+00	1.4500E+07	1.0256E+00	1.4600E+07	1.0190E+00	1.4700E+07	1.0124E+00	1.4800E+07	1.0062E+00
206	1.4900E+07	9.9999E-01	1.5000E+07	9.9367E-01	1.5100E+07	9.8751E-01	1.5200E+07	9.8142E-01	1.5300E+07	9.7540E-01
211	1.5400E+07	9.6945E-01	1.5500E+07	9.6357E-01	1.5600E+07	9.5776E-01	1.5700E+07	9.5201E-01	1.5800E+07	9.4634E-01
216	1.5900E+07	9.4072E-01	1.6000E+07	9.3518E-01	1.6100E+07	9.2970E-01	1.6200E+07	9.2428E-01	1.6300E+07	9.1893E-01
221	1.6400E+07	9.1364E-01	1.6500E+07	9.0841E-01	1.6600E+07	9.0324E-01	1.6700E+07	8.9814E-01	1.6800E+07	8.9310E-01
226	1.6900E+07	8.8811E-01	1.7000E+07	8.8319E-01	1.7100E+07	8.7832E-01	1.7200E+07	8.7352E-01	1.7300E+07	8.6877E-01
231	1.7400E+07	8.6408E-01	1.7500E+07	8.5944E-01	1.7600E+07	8.5484E-01	1.7700E+07	8.5034E-01	1.7800E+07	8.4587E-01
236	1.7900E+07	8.4146E-01	1.8000E+07	8.3718E-01	1.8100E+07	8.3298E-01	1.8200E+07	8.2885E-01	1.8300E+07	8.2439E-01
241	1.8400E+07	8.2021E-01	1.8500E+07	8.1611E-01	1.8600E+07	8.1207E-01	1.8700E+07	8.0808E-01	1.8800E+07	8.0414E-01
246	1.8900E+07	8.0025E-01	1.9000E+07	7.9641E-01	1.9100E+07	7.9262E-01	1.9200E+07	7.8888E-01	1.9300E+07	7.8518E-01
251	1.9400E+07	7.8154E-01	1.9500E+07	7.7794E-01	1.9600E+07	7.7439E-01	1.9700E+07	7.7088E-01	1.9800E+07	7.6743E-01
256	1.9900E+07	7.6401E-01	2.0000E+07	7.6063E-01						

HELIUM-4 — ELASTIC NEUTRON CROSS SECTION — ENDF/B MATERIAL NO. 1278

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 297	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS	ENERGY EV	CROSS SECTION BARNS
1	1.2000E+03	7.9916E+01	2.5300E+02	7.5916E+01	1.2000E+00	7.5916E+01	1.2000E+01	7.5916E+01	1.2000E+02	7.5916E+01
6	1.2000E+03	7.5911E+01	5.2000E+03	7.5891E+01	1.2000E+04	7.5872E+01	2.0000E+04	7.5849E+01	4.2000E+04	7.5875E+01
11	6.0000E+04	7.6003E+01	8.0000E+04	7.6243E+01	1.0000E+05	7.6499E+01	1.5000E+05	7.6198E+01	2.0000E+05	8.7817E+01
16	2.5000E+05	8.4809E+01	3.0000E+05	9.0890E+01	3.5000E+05	9.9597E+01	4.0000E+05	1.1035E+00	4.5000E+05	1.2534E+00
21	5.0000E+05	1.4525E+00	5.5000E+05	1.7114E+00	6.0000E+05	2.0438E+00	6.5000E+05	2.2412E+00	6.5000E+05	2.4438E+00
26	6.7500E+05	2.7007E+00	7.0000E+05	2.9709E+00	7.2500E+05	3.2795E+00	7.5000E+05	3.5950E+00	7.7500E+05	3.9306E+00
31	8.0000E+05	4.2806E+00	8.2500E+05	4.6778E+00	8.5000E+05	5.0616E+00	8.7500E+05	5.4469E+00	9.0000E+05	5.8241E+00
36	9.2500E+05	6.1843E+00	9.5000E+05	6.5138E+00	9.7500E+05	6.8173E+00	1.0000E+06	7.0752E+00	1.0250E+06	7.2867E+00
41	1.0500E+06	7.4404E+00	1.0750E+06	7.5627E+00	1.1000E+06	7.6249E+00	1.1250E+06	7.6391E+00	1.1500E+06	7.6321E+00
46	1.1750E+06	7.5790E+00	1.2000E+06	7.4987E+00	1.2250E+06	7.3645E+00	1.2500E+06	7.2721E+00	1.2750E+06	7.1302E+00
51	1.3000E+06	6.9909E+00	1.3250E+06	6.8305E+00	1.3500E+06	6.6458E+00	1.3750E+06	6.5295E+00	1.4000E+06	6.3746E+00
56	1.4250E+06	6.2735E+00	1.4500E+06	5.7882E+00	1.4750E+06	5.3928E+00	1.5000E+06	5.1777E+00	1.5250E+06	5.0530E+00
61	1.5500E+06	6.4496E+00	1.5750E+06	4.8636E+00	1.6000E+06	4.4643E+00	1.6250E+06	4.3401E+00	1.6500E+06	4.1996E+00
66	1.6750E+06	4.0713E+00	1.7000E+06	3.9141E+00	1.7250E+06	3.8407E+00	1.7500E+06	3.7482E+00	1.7750E+06	3.6570E+00
71	1.8000E+06	3.9741E+00	1.8250E+06	3.4991E+00	1.8500E+06	3.4290E+00	1.8750E+06	3.3590E+00	1.9000E+06	3.2986E+00
76	1.9250E+06	3.2416E+00	1.9500E+06	3.1885E+00	1.9750E+06	3.1300E+00	2.0000E+06	3.0915E+00	2.0250E+06	3.0401E+00
81	2.0500E+06	3.0004E+00	2.0750E+06	2.9791E+00	2.1000E+06	2.9340E+00	2.1250E+06	2.9000E+00	2.1500E+06	2.8679E+00
86	2.1750E+06	2.8379E+00	2.2000E+06	2.8087E+00	2.2250E+06	2.7794E+00	2.2500E+06	2.7978E+00	2.2750E+06	2.4027E+00
91	2.3000E+06	2.6321E+00	2.3250E+06	2.5444E+00	2.3500E+06	2.5404E+00	2.3750E+06	2.5165E+00	2.4000E+06	2.4055E+00
96	2.4250E+06	2.4541E+00	2.4500E+06	2.4280E+00	2.4750E+06	2.4011E+00	2.5000E+06	2.3753E+00	2.5250E+06	2.3583E+00
101	2.5500E+06	2.3260E+00	2.5750E+06	2.3024E+00	2.6000E+06	2.2794E+00	2.6250E+06	2.2560E+00	2.6500E+06	2.2349E+00
106	2.6750E+06	2.2132E+00	2.7000E+06	2.1919E+00	2.7250E+06	2.1718E+00	2.7500E+06	2.1504E+00	2.7750E+06	2.1391E+00
111	2.8000E+06	2.1190E+00	2.8250E+06	2.0903E+00	2.8500E+06	2.0707E+00	2.8750E+06	2.0515E+00	2.9000E+06	2.0324E+00
116	2.9250E+06	2.0136E+00	2.9500E+06	1.9996E+00	2.9750E+06	1.9707E+00	3.0000E+06	1.9506E+00	3.0250E+06	1.9407E+00
121	3.0500E+06	1.9338E+00	3.0750E+06	1.9195E+00	3.1000E+06	1.8903E+00	3.1250E+06	1.8712E+00	3.1500E+06	1.8544E+00
126	3.1750E+06	1.8376E+00	3.2000E+06	1.8213E+00	3.2250E+06	1.8091E+00	3.2500E+06	1.7891E+00	3.2750E+06	1.7733E+00
131	3.3000E+06	1.7977E+00	3.3250E+06	1.7483E+00	3.3500E+06	1.7371E+00	3.3750E+06	1.7120E+00	3.4000E+06	1.6972E+00
136	3.4250E+06	1.6239E+00	3.4500E+06	1.6081E+00	3.4750E+06	1.6038E+00	3.5000E+06	1.6397E+00	3.5250E+06	1.6238E+00
141	3.5500E+06	1.6128E+00	3.5750E+06	1.5985E+00	3.6000E+06	1.5891E+00	3.6250E+06	1.5711E+00	3.6500E+06	1.5568E+00
146	3.6750E+06	1.5459E+00	3.7000E+06	1.5332E+00	3.7250E+06	1.5296E+00	3.7500E+06	1.5082E+00	3.7750E+06	1.4959E+00
151	3.8000E+06	1.4838E+00	3.8250E+06	1.4718E+00	3.8500E+06	1.4600E+00	3.8750E+06	1.4444E+00	3.9000E+06	1.4369E+00
156	3.9250E+06	1.4255E+00	3.9500E+06	1.4142E+00	3.9750E+06	1.3982E+00	4.0000E+06	1.3972E+00	4.0250E+06	1.3814E+00
161	4.0500E+06	1.3787E+00	4.0750E+06	1.3601E+00	4.1000E+06	1.3497E+00	4.1250E+06	1.3393E+00	4.1500E+06	1.3292E+00
166	4.1750E+06	1.3191E+00	4.2000E+06	1.3091E+00	4.2250E+06	1.2993E+00	4.2500E+06	1.2906E+00	4.2750E+06	1.2808E+00
171	4.3000E+06	1.2705E+00	4.3250E+06	1.2612E+00	4.3500E+06	1.2519E+00	4.3750E+06	1.2428E+00	4.4000E+06	1.2337E+00
176	4.4250E+06	1.2246E+00	4.4500E+06	1.2168E+00	4.4750E+06	1.2072E+00	4.5000E+06	1.1986E+00	4.5250E+06	1.1901E+00
181	4.5500E+06	1.1817E+00	4.5750E+06	1.1733E+00	4.6000E+06	1.1651E+00	4.6250E+06	1.1573E+00	4.6500E+06	1.1489E+00
186	4.6750E+06	1.1418E+00	4.7000E+06	1.1331E+00	4.7250E+06	1.1254E+00	4.7500E+06	1.1177E+00	4.7750E+06	1.1101E+00
191	4.8000E+06	1.1026E+00	4.8250E+06	1.0972E+00	4.8500E+06	1.0879E+00	4.8750E+06	1.0806E+00	4.9000E+06	1.0734E+00
196	4.9250E+06	1.0664E+00	4.9500E+06	1.0594E+00	4.9750E+06	1.0505E+00	5.0000E+06	1.0450E+00	5.0250E+06	1.0380E+00
201	5.0500E+06	1.0322E+00	5.0750E+06	1.0256E+00	5.1000E+06	1.0190E+00	5.1250E+06	1.0126E+00	5.1500E+06	1.0062E+00
206	5.1750E+06	9.9998E-01	5.2000E+06	9.9367E-01	5.2250E+06	9.8791E-01	5.2500E+06	9.8142E-01	5.2750E+06	9.7548E-01
211	5.3000E+06	9.6945E-01	5.3250E+06	9.6397E-01	5.3500E+06	9.5770E-01	5.3750E+06	9.5201E-01	5.4000E+06	9.4634E-01
216	5.4250E+06	9.4072E-01	5.4500E+06	9.3518E-01	5.4750E+06	9.2978E-01	5.5000E+06	9.2428E-01	5.5250E+06	9.1893E-01
221	5.5500E+06	9.1364E-01	5.5750E+06	9.0841E-01	5.6000E+06	9.0324E-01	5.6250E+06	8.9814E-01	5.6500E+06	8.9312E-01
226	5.6750E+06	8.8811E-01	5.7000E+06	8.8319E-01	5.7250E+06	8.7832E-01	5.7500E+06	8.7352E-01	5.7750E+06	8.6877E-01
231	5.8000E+06	8.6408E-01	5.8250E+06	8.5944E-01	5.8500E+06	8.5484E-01	5.8750E+06	8.5034E-01	5.9000E+06	8.4587E-01
236	5.9250E+06	8.4146E-01	5.9500E+06	8.3718E-01	5.9750E+06	8.3288E-01	6.0000E+06	8.2859E-01	6.0250E+06	8.2435E-01
241	6.0500E+06	8.2021E-01	6.0750E+06	8.1611E-01	6.1000E+06	8.1207E-01	6.1250E+06	8.0808E-01	6.1500E+06	8.0414E-01
246	6.1750E+06	8.0025E-01	6.2000E+06	7.9641E-01	6.2250E+06	7.9262E-01	6.2500E+06	7.8888E-01	6.2750E+06	7.8518E-01
251	6.3000E+06	7.8154E-01	6.3250E+06	7.7794E-01	6.3500E+06	7.7436E-01	6.3750E+06	7.7088E-01	6.4000E+06	7.6743E-01
256	6.4250E+06	7.6401E-01	6.4500E+06	7.6063E-01						

HELIUM-4 — FE BAR NEUTRON CROSS SECTION — ENDF/B MATERIAL NO. 1278

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1.0000E+03	1.6901E+01	1.0000E+00	1.6901E+01	6.0000E+04	9.6422E+02	1.0000E+05	4.5188E+02	1.5000E+05	2.6449E+02
6	2.0000E+03	1.9136E+02	2.5000E+03	1.4943E+01	3.0000E+03	1.9056E+01	3.5000E+03	2.3093E+01	4.2000E+03	2.6668E+01
11	6.3000E+03	2.7821E+01	9.0000E+03	8.7424E+01	5.0000E+03	2.3642E+01	6.0000E+03	2.2736E+01	6.5000E+03	2.4992E+01
16	7.0000E+03	2.4870E+01	7.7500E+03	2.6992E+01	8.5000E+03	2.1400E+02	9.2500E+03	9.1757E+02	1.0000E+04	1.7814E+01
21	1.0750E+04	1.5701E+01	1.1750E+04	1.1329E+01	1.2750E+04	2.3002E+01	1.4000E+04	3.3334E+01	1.5500E+04	3.4340E+01
26	1.7000E+04	3.7277E+01	1.8500E+04	3.9499E+01	2.0000E+04	4.1444E+01	2.2500E+04	4.2819E+01	2.5000E+04	4.3974E+01
31	2.7500E+04	4.4771E+01	3.0000E+04	4.5385E+01	3.4000E+04	4.6833E+01	4.1000E+04	4.7734E+01	5.3000E+04	5.0455E+01
36	6.1000E+04	5.2893E+01	6.9000E+04	5.3478E+01	7.7000E+04	5.4618E+01	8.6000E+04	5.5025E+01	9.6000E+04	5.6499E+01
41	1.0700E+05	5.7220E+01	1.2000E+05	5.7873E+01	1.3500E+05	5.8431E+01	1.5400E+05	5.8993E+01	1.8100E+05	5.9750E+01
46	2.2000E+05	6.9367E+01								

HELIUM-4 — NI NEUTRON CROSS SECTION — ENDF/B MATERIAL NO. 1278

INTERPOLATION LAW BETWEEN ENERGIES

RANGE	DESCRIPTION
1 TO 46	Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX,	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA	ENERGY EV	DATA
1	1.0000E+03	4.2518E+01	1.0000E+00	4.2518E+01	6.0000E+04	4.6266E+01	1.0000E+05	4.6903E+01	1.5000E+05	5.2269E+01
6	2.0000E+03	5.1577E+01	2.5000E+03	5.8693E+01	3.0000E+03	6.1366E+01	3.5000E+03	6.3476E+01	4.0000E+03	6.4601E+01
11	4.5000E+03	6.9446E+01	5.0000E+03	6.3237E+01	5.5000E+03	6.4113E+01	6.0000E+03	6.2889E+01	6.5000E+03	6.7812E+01
16	7.0000E+03	5.0689E+01	7.7500E+03	5.5189E+01	8.5000E+03	5.1730E+01	9.2500E+03	4.8918E+01	1.0000E+04	4.5028E+01
21	1.0750E+04	4.3807E+01	1.1750E+04	4.0235E+01	1.2750E+04	3.7014E+01	1.4000E+04	3.3025E+01	1.5500E+04	3.3571E+01
26	1.7000E+04	3.2873E+01	1.8500E+04	3.2074E+01	2.0500E+04	2.9941E+01	2.2500E+04	2.9248E+01	2.5000E+04	2.8658E+01
31	2.7500E+04	2.8842E+01	3.0000E+04	2.7927E+01	3.4000E+04	2.7491E+01	4.1000E+04	2.6717E+01	5.3000E+04	2.5314E+01
36	6.1000E+04	2.4449E+01	6.9000E+04	2.3750E+01	7.7000E+04	2.3173E+01	8.6000E+04	2.2691E+01	9.6000E+04	2.2281E+01
41	1.0700E+05	2.1826E+01	1.2000E+05	2.1499E+01	1.3500E+05	2.1200E+01	1.5400E+05	2.0919E+01	1.8100E+05	2.0531E+01
46	2.2000E+05	1.9225E+01								

HELIUM-4 GAMMA ENDF/B MATERIAL NO. 1270
NEUTRON CROSS SECTION

INTERPOLATION LAW BETWEEN ENERGIES
RANGE DESCRIPTION
1 TO 46 Y LINEAR IN X

NEUTRON CROSS SECTIONS

INDEX	ENERGY (EV)	DATA	ENERGY (EV)	DATA	ENERGY (EV)	DATA	ENERGY (EV)	DATA	ENERGY (EV)	DATA
1	1.0000E+05	3.0968E-01	1.0000E+05	3.0968E-01	6.0000E+04	3.2002E-01	1.0000E+05	3.2988E-01	1.5000E+05	3.3919E-01
6	2.0000E+05	3.4767E-01	2.5000E+05	3.5936E-01	3.0000E+05	3.6194E-01	3.5000E+05	3.6752E-01	4.0000E+05	3.7197E-01
11	4.5000E+05	3.7529E-01	5.0000E+05	3.7749E-01	5.5000E+05	3.7864E-01	6.0000E+05	3.7878E-01	6.5000E+05	3.7881E-01
16	7.0000E+05	3.7844E-01	7.5000E+05	3.7899E-01	8.0000E+05	3.6793E-01	8.5000E+05	3.6223E-01	1.0000E+06	3.5591E-01
21	1.0750E+06	3.4932E-01	1.1750E+06	3.4642E-01	1.2750E+06	3.3171E-01	1.4000E+06	3.2149E-01	1.5500E+06	3.1050E-01
26	1.7000E+06	3.0180E-01	1.8500E+06	2.8322E-01	2.0000E+06	2.8492E-01	2.2500E+06	2.7876E-01	2.5000E+06	2.7342E-01
31	2.7500E+06	2.7903E-01	3.0000E+06	2.6883E-01	3.4000E+06	2.6693E-01	4.1000E+06	2.6607E-01	5.3000E+06	2.6572E-01
36	6.1000E+06	2.6477E-01	6.9000E+06	2.6344E-01	7.7000E+06	2.6190E-01	8.6000E+06	2.6032E-01	9.6000E+06	2.5869E-01
41	1.0700E+07	2.5707E-01	1.2000E+07	2.5593E-01	1.3500E+07	2.5411E-01	1.5400E+07	2.5269E-01	1.0100E+07	2.5096E-01
46	2.0000E+07	2.4976E-01								

HELIUM-4 ELASTIC ENDF/B MATERIAL NO. 1270
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	1.6660E+01	0.6293E+01	-1.3740E+01	3.9389E-02	+0.3004E+03	2.0397E+03	-4.9921E+04	1.2311E+04	-2.0461E+05	7.9963E+06
2	1.2604E+02	2.9448E+01	0.8395E+01	-2.7124E+01	7.7520E+02	+8.1939E+02	5.8730E+03	+1.9410E+03	4.2137E+04	-1.1021E+04
3	-6.5735E+09	4.2829E+02	4.8760E+01	8.1616E+01	+3.4284E+01	1.2740E+01	+4.0796E+02	1.2397E+02	+2.6429E+03	1.8465E+03
4	6.0000E+00	3.9259E+03	6.7281E+02	9.8523E+01	7.8991E+01	+4.3200E+01	1.8216E+01	+6.8351E+02	2.2387E+02	+7.1211E+03
5	0.0000E+00	1.1163E+04	1.8792E+02	1.4148E+01	9.7709E+01	9.7004E+01	+4.7780E+01	2.3783E+01	+0.7734E+02	3.6139E+02
6	0.0000E+00	+4.6834E+04	8.2812E+04	2.4815E+02	2.8796E+01	0.0000E+00	4.3856E+01	+4.9679E+01	2.9062E+01	+1.3380E+01
7	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	3.1447E+02	3.7688E+01	+5.8782E+01	3.4042E+01
8	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	3.0307E+03	7.0816E+02	3.8476E+01	+5.8276E+01
9	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00	+4.9009E+04	1.8337E+02	1.1790E+01	2.2739E+01

INTERPOLATION LAW BETWEEN ENERGIES
RANGE DESCRIPTION
1 TO 46 Y LINEAR IN X

HELIUM-4 ELASTIC ENDF/B MATERIAL NO. 1270
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1.0000E+11 MEV IS ISOTROPIC
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 0.0000E+00 2 0.0000E+00

F(MU)=(SUM OVER L) (M,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-06	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								

HELIUM-4 ELASTIC ENDF/B MATERIAL NO. 1270
SECONDARY NEUTRON ANGULAR DISTRIBUTIONS

RECONSTRUCTED ANGULAR DISTRIBUTION IN THE CENTER OF MASS SYSTEM AT 1.0000E+06 MEV IS ISOTROPIC
3 LEGENDRE COEFFICIENTS WERE USED IN THE RECONSTRUCTION
1 0.0000E+00 2 0.0000E+00

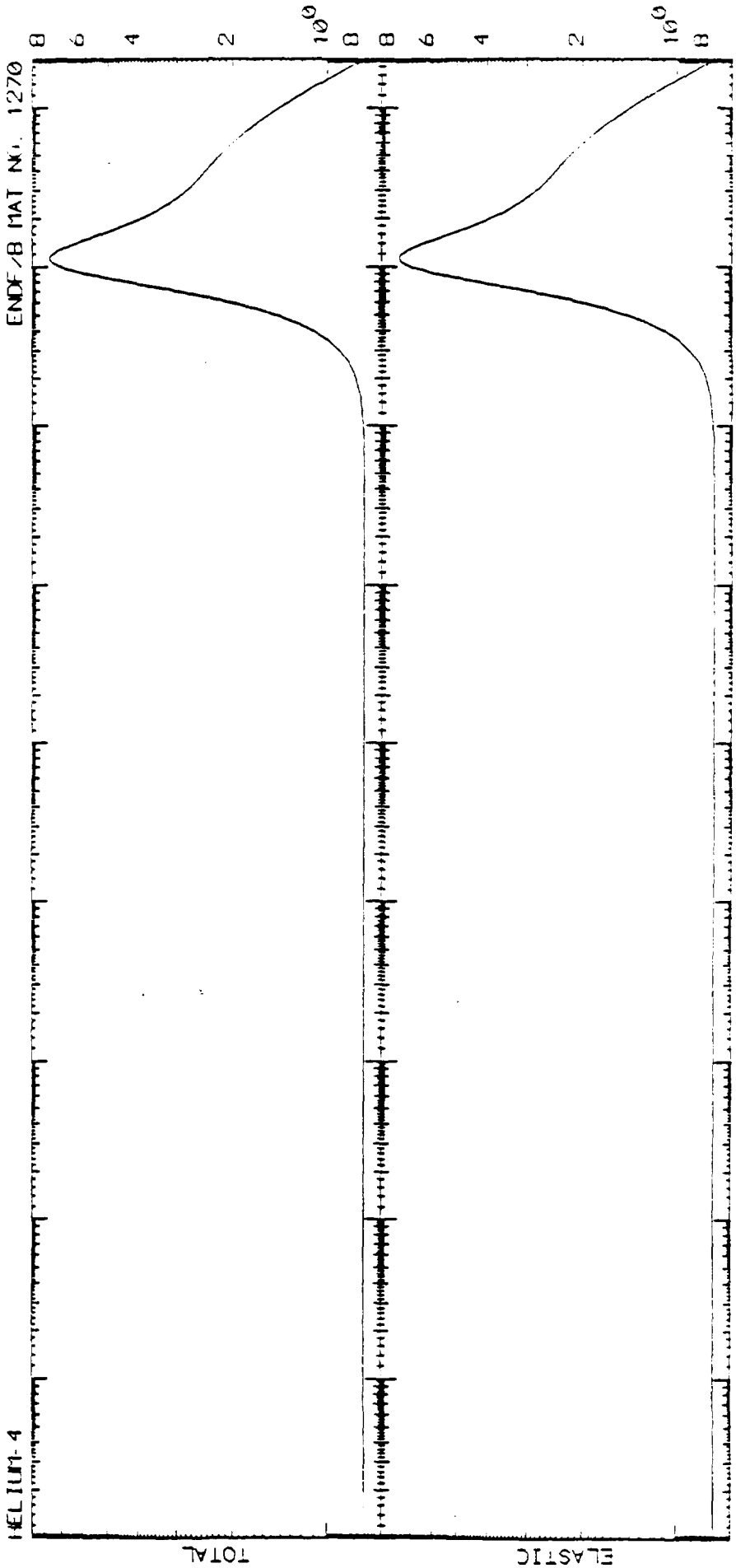
F(MU)=(SUM OVER L) (M,5*(2L+1)*F(L,E)*P(L,MU))

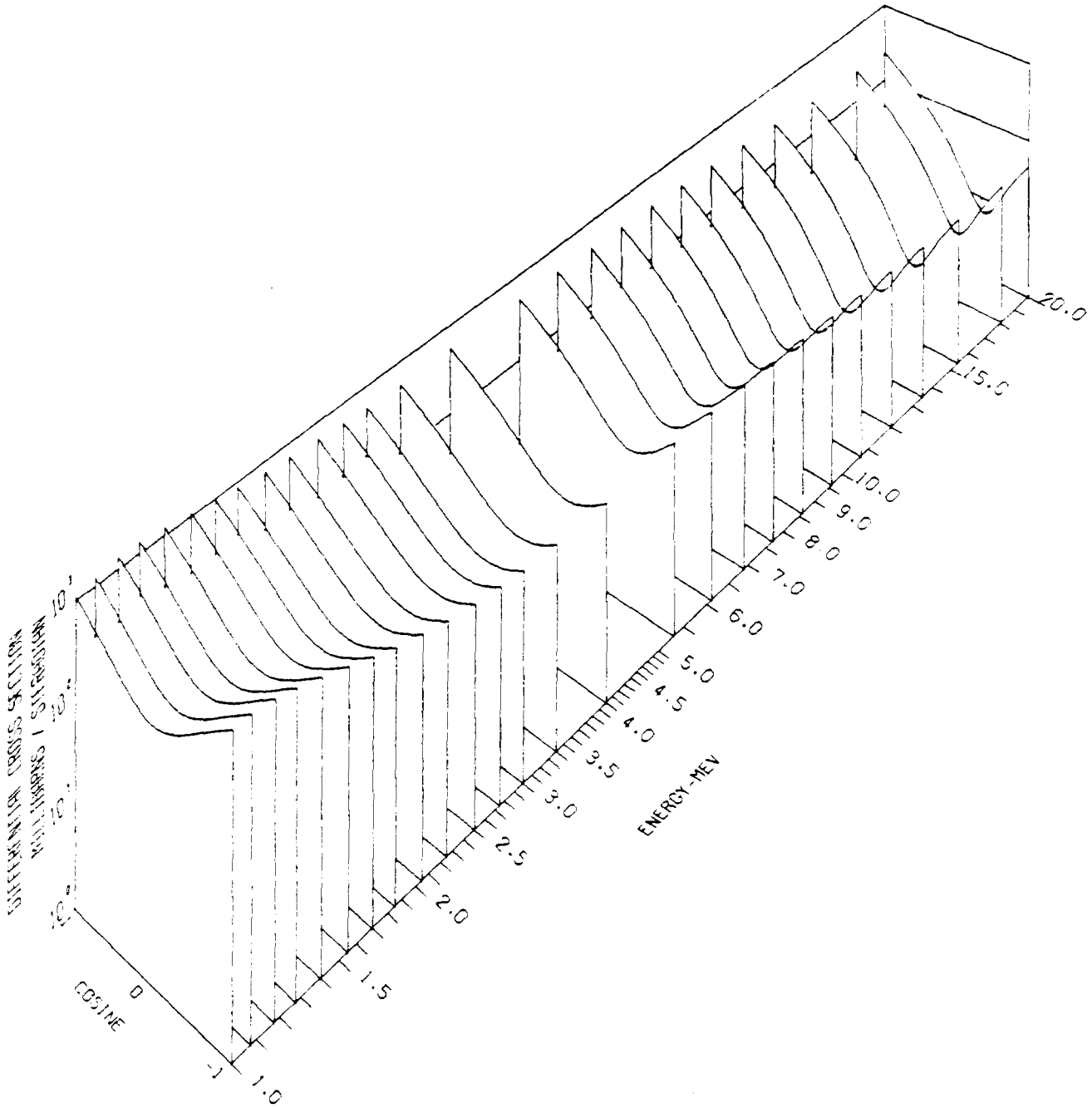
INDEX	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)	MU	F(MU)
1	1.0000E+00	5.0000E-01	9.7500E-01	5.0000E-01	9.5000E-01	5.0000E-01	9.2500E-01	5.0000E-01	9.0000E-01	5.0000E-01
6	8.7500E-01	5.0000E-01	8.5000E-01	5.0000E-01	8.2500E-01	5.0000E-01	8.0000E-01	5.0000E-01	7.7500E-01	5.0000E-01
11	7.5000E-01	5.0000E-01	7.2500E-01	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-06	5.0000E-01	-2.5000E-02	5.0000E-01	-5.0000E-02	5.0000E-01	-7.5000E-02	5.0000E-01	-1.0000E-01	5.0000E-01
46	-1.2500E-01	5.0000E-01	-1.5000E-01	5.0000E-01	-1.7500E-01	5.0000E-01	-2.0000E-01	5.0000E-01	-2.2500E-01	5.0000E-01
51	-2.5000E-01	5.0000E-01	-2.7500E-01	5.0000E-01	-3.0000E-01	5.0000E-01	-3.2500E-01	5.0000E-01	-3.5000E-01	5.0000E-01
56	-3.7500E-01	5.0000E-01	-4.0000E-01	5.0000E-01	-4.2500E-01	5.0000E-01	-4.5000E-01	5.0000E-01	-4.7500E-01	5.0000E-01
61	-5.0000E-01	5.0000E-01	-5.2500E-01	5.0000E-01	-5.5000E-01	5.0000E-01	-5.7500E-01	5.0000E-01	-6.0000E-01	5.0000E-01
66	-6.2500E-01	5.0000E-01	-6.5000E-01	5.0000E-01	-6.7500E-01	5.0000E-01	-7.0000E-01	5.0000E-01	-7.2500E-01	5.0000E-01
71	-7.5000E-01	5.0000E-01	-7.7500E-01	5.0000E-01	-8.0000E-01	5.0000E-01	-8.2500E-01	5.0000E-01	-8.5000E-01	5.0000E-01
76	-8.7500E-01	5.0000E-01	-9.0000E-01	5.0000E-01	-9.2500E-01	5.0000E-01	-9.5000E-01	5.0000E-01	-9.7500E-01	5.0000E-01
81	-1.0000E+00	5.0000E-01								

APPENDIX M

Sample Graphical Display

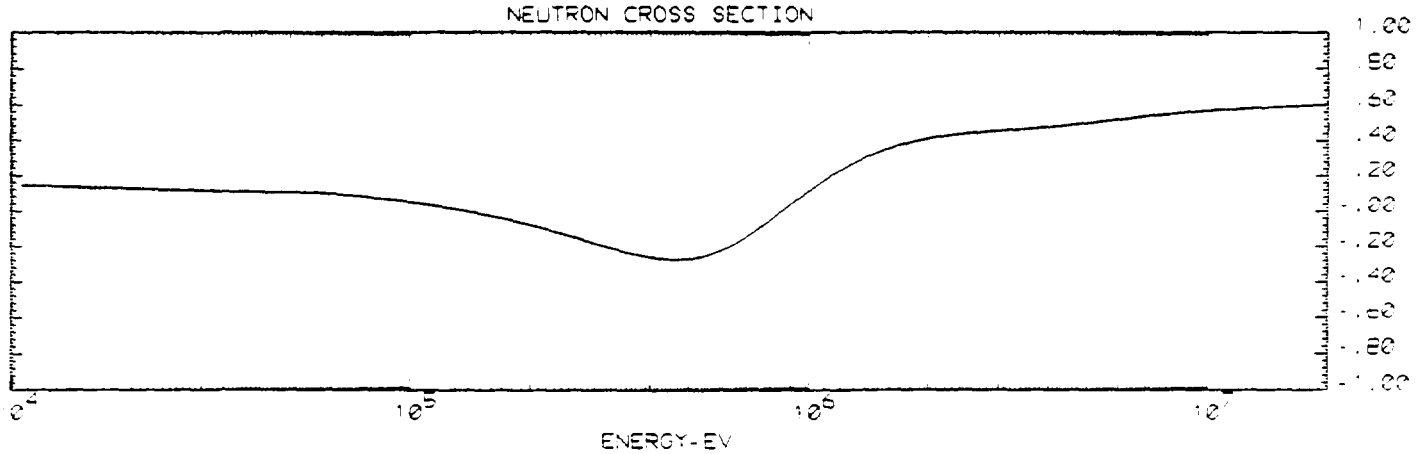
The following is a sample graphical display of the cross sections for Helium-4. A number of codes (see Appendix I) prepare graphical display of materials in the ENDF format. The examples shown here are taken from ENDF-200, ENDF CURVES.


HELIUM-4
ENDF/B MAT NO. 1270
TOTAL
ELASTIC
8
6
4
2
10^0
8
6
4
2
10^0
8



HELIUM-4 MU BAR (LAB) ENDF/B MAT NO. 1270

NEUTRON CROSS SECTION



Appendix N

Examples of Card-Image Formats

This appendix describes examples of formats containing BCD card-image records.

The following appendix begins with a review as to how ENDF record types are organized in card-image format. The remaining pages contain the card-image examples and formats.

The top of each page indicates the particular file and data format described. Four pages (A, B, C, D) are used to describe each data type. When the appendix is opened to a particular data type, the left-hand page, A (the last page of the previous foldout sheet), contains a review of the variable names and their definitions. The right-hand page, D (before unfolding) contains the important formulae associated with the particular data type. When the right-hand page is unfolded, pages B and C are exposed. Page B gives the format. For explanation of the notation describing the ENDF record types refer to either the beginning of this appendix or Section 0.5.3. Page C gives an example of ENDF card-images described by the formats. Note that in some cases, the example would have contained too many cards to have been displayed on one page; therefore, cards have been omitted.

Examples of Record Format and Notation used in Appendix

All records on an ENDF binary tape are one of four possible types, denoted by CONT, LIST, TAB1, and TAB2. A record always consists of nine numbers followed (depending on the record type) by one or two arrays of numbers. A general description of these nine numbers is given below, but the actual definition of each number will depend on its usage.

MAT is the material number (integer).

MF is the file number (integer).

MT is the reaction type number (integer).

C1 is a constant (floating point).

C2 is a constant (floating point).

L1 is an integer generally used as a test.

L2 is an integer generally used as a test.

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

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

0.5.3. Card-Image (BCD) Formats

An alternative 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 in the binary tape format. The major difference is the position of the three numbers MAT, MF, and MT. Also a card sequence number has been added to the card-image format. In general, more than one BCD card-image record will be required to contain the data in a binary record.

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

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

Examples of Record Format and Notation used in Appendix

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

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

The SEND, FEND, MEND, and TEND records use only the first three numbers in the 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

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.

Examples of Record Format and Notation used in Appendix

LIST Records

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

The LIST record denoted by

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

is punched in the following way:

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

Examples of Record Format and Notation used in Appendix

TAB1 Records

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

Consider a TAB1 binary record that was denoted by

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

This record would be punched on cards in the following way:

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

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

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

File 1 Descriptive Information and Index (MT=451)

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

LDD is a flag to indicate whether induced reaction 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 $\leq$ 294). For binary tapes, NWD is the number of words containing the Hollerith information, and it is understood that 17 words are required for each card image (66 characters) and the format is (16A4, A2). (NWD $\leq$ 5000.)

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

(First BCD Card Image Record)

ZSYMA is a Hollerith representation of the material Z-chemical symbol - A with

Z right justified in col. 1 to 3

- hyphen in col. 4

chemical symbol left justified in col. 5 and 6

- hyphen in col. 7

A right justified in 8 - 10 or blank

m, etc. indication of metastable state in col. 11

ALAB Mnemonic of originating laboratory(s) (left adjusted)

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

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

(Second BCD-Card Image Record)

REF reference 2-22

DDATE original distribution date (left adjusted DIST- followed by month-year as in EDATE

RDATE date and number of last revision REV1- followed by month-year as in EDATE

The following quantities are defined.

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

MF_n is the MF of the nth section.

MT_n is the MT of the nth section.

NC_n is the number of BCD card images in a given section (the nth section).

This card count does not include the SEND card. (Note that NC_1 = NXC + NWD + 1.)

File 1 Descriptive Information and Index (MT=451)

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

[MAT, 1, 451/ZA , AWR, LRP, LFI, 0, NXC]HEAD

[MAT, 1, 451/0.0, 0.0, LDD, LFP, NWD, 0/ ZSYMA, ALAB, EDATE, AUTH (33 characters), REF (22 characters), DDATE, RDATE, b, b,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

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

File 1 Descriptive Information and Index (MT=451)

```
 9.22350+ 4 2.33025+ 2          1          1          0         821261 1451    1
 0.00000+ 0 0.00000+ 0          1          1        263          01261 1451    2
 92-U -235 LASL,AI     EVAL-MAR74 L.STEWART, H.ALTER , R.HUNTER    1261 1451    3
                       DIST-JUL74 REV-JUN75                        1261 1451    4
PRINCIPAL EVALUATORS- L.STEWART LASL, H.ALTER AI, R.HUNTER LASL    1261 1451    5
                                                                   1261 1451    6
CONTRIBUTING EVALUATORS                                            1261 1451    7
                                                                   1261 1451    8
  NU-BAR—B.R. LEONARD BNW, L. STEWART AND RAY HUNTER LASL,         1261 1451    9
    HUMMEL ANL.                                                    1261 1451   10
  F.P.YIELDS—R.SCHENTER HEDL, FISSION PROD. SUBCOMMITTEE           1261 1451   11
  DELAYED NEUTRON DATA— S.A.COX(ANL)                               1261 1451   12
  RADIOACTIVE DECAY DATA—C.W.REICH ANC                             1261 1451   13
  RESOLVED RESONANCE DATA—J.R. SMITH ANC, R. GWIN, R. PEELE, AND   1261 1451   14
    G.DESAUSSURE ORNL                                              1261 1451   15
 UNRESOLVED RESONANCE DATA— R.PEELLE(ORNL) AND M.BHAT(BNL)         1261 1451   16
                                                                   1261 1451   17
  SMOOTH DATA                                                      1261 1451   18
                                                                   1261 1451   19
     THERMAL RANGE  C.LUBITZ KAPL, J.HARDY BAPL, B.R.LEONARD BNW   1261 1451   20
     82 EV -25 KEV—R.GWIN, G.DESAUSSURE ORNL, R.BLOCK RPI,         1261 1451   21
       J.R. SMITH ANC                                              1261 1451   22
     25 KEV-1 MEV A.CARLSON NBS, W.POENITZ ANL, L.STEWART          1261 1451   23
       LASL, H.ALTER                                               1261 1451   24
     1 MEV-20 MEV—R.HUNTER, L.STEWART LASL, H.ALTER                1261 1451   25
     INELASTIC SCAT—L.STEWART, R.HUNTER LASL                       1261 1451   26
     SECONDARY NEUTRON DIST.—L.STEWART, R.HUNTER LASL              1261 1451   27
     GAMMA PRODUCTION—R.HUNTER, L.STEWART LASL                     1261 1451   28
                                                                   1261 1451   29

     (MISSING LINES)

                                                                   1261 1451  172
                              MF = 3                               1261 1451  173
                                                                   1261 1451  174
  SMOOTH DATA                                                      1261 1451  175
       THERMAL DATA——THERMAL TASK FORCE                            1261 1451  176
       1 EV   TO 82 EV    J. R. SMITH                             1261 1451  177
       82 EV TO 25 KEV    PEELLE, BHAT                            1261 1451  178
       25 KEV TO 100 KEV  BIG THREE PLUS TWO TASK FORCE           1261 1451  179
       100KEV TO 1 MEV——FISSION CROSS SECTION TAKEN AS CURVE       1261 1451  180
     SUGGESTED BY U-235 TASK FORCE AND CSEWG STANDARDS AND         1261 1451  181
     NORMALIZATION SUBCOMMITTEE. IN THIS ENERGY REGION DATA TAKEN  1261 1451  182
     FROM REFERENCES 1 THROUGH 9. DATA OF REF.4 SZABO (71) RAISED  1261 1451  183
     BY 1.04. BETWEEN 1 AND 6 MEV CURVE DRAWN THROUGH DATA OF      1261 1451  184
     REFERENCES 3, 5, AND 7 THROUGH 11, WITH HEAVY WEIGHT GIVEN TO1261 1451  185
     REF. 11. ABOVE 6 MEV CURVE DRAWN THROUGH DATA OF REFERENCES   1261 1451  186
     7, 8, 12 AND 13. DATA OF REFS. 12 AND 13 NORMALIZED TO 2.132  1261 1451  187
     BARNS AT 14.0 MEV.——ALPHA CURVE BETWEEN 10 KEV AND 10 MEV     1261 1451  188
     BASED ON REFERENCES 1 AND 14 THROUGH 19 AS RECOMMENDED BY     1261 1451  189
     U-235 TASK FORCE. ABOVE 1 MEV ALPHA CURVE SMOOTHLY            1261 1451  190
     EXTRAPOLATED TO 20 MEV.——CAPTURE CROSS SECTION DERIVED AS     1261 1451  191
     THE PRODUCT OF THE FISSION CROSS SECTION WITH ALPHA——ABOVE    1261 1451  192
     0.5 MEV TOTAL CROSS SECTION TAKEN FROM SPLINE FIT TO DATA OF  1261 1451  193
     REFERENCES 20 AND 21. BETWEEN 23 KEV AND 0.5 MEV A SMOOTH     1261 1451  194
     CURVE WAS FIT TO THE TOTAL CROSS SECTION OF ENDF/B-3.         1261 1451  195
                                                                   1261 1451  196

     (MISSING LINES)

                          REFERENCES                               1261 1451  228
                                                                   1261 1451  229
  1. GWIN, R., ET.AL. PRIVATE COMMUNICATION (ORNL,1973)            1261 1451  230
  2. KAPPELER,F.SYMPOSIUM NEUT.STDS., (ANL) CONF.701002, 272(1970) 1261 1451  231
  3. SZABO,I., ET.AL., (AS REF.5) CONF-701002, 257(1970)           1261 1451  232
  4. SZABO,I., ET.AL., KNOXVILLE CONF. VOL.2 573(1971)             1261 1451  233
  5. KAPPELER,F., 2ND IAEA PANEL STANDARD X-SECTIONS,VIENNA(1972)  1261 1451  234
  6. GAYTHER, D.B., ET.AL., (AS REF.8) (1972)                      1261 1451  235
  7. HANSEN, G., ET.AL., PRIVATE COMMUNICATION,L.STEWART(LASL 1970)1261 1451  236
  8. WHITE, P.H.., J.NUCL.ENERGY 19,325(1965)                      1261 1451  237
  9. DIVEN, B.C., PHYS.REV.105,1350(1957)                          1261 1451  238
 10. POENITZ, W., PRIVATE COMMUNICATION (ANL,1973)                 1261 1451  239

     (MISSING LINES)

 32. R. W. PEELLE, ORNL, LETTER TO CSEWG, 9-24-73.                 1261 1451  264
                                                                   1261 1451  265
                              1        451        347              1261 1451  266
                              1        452          6              1261 1451  267
                              1        453          9              1261 1451  268
                              1        454       1699              1261 1451  269
                              1        455          8              1261 1451  270
                              1        456          6              1261 1451  271
                              1        457         60              1261 1451  272
                              2        151        871              1261 1451  273
                              3          1        388              1261 1451  274
                              3          2        339              1261 1451  275

     (MISSING LINES)

                             15          3        136              1261 1451  345
                             15         18         54              1261 1451  346
                             15        102         58              1261 1451  347
```

File 1 Descriptive Information and Index (MT=451)

File 1 Number of Neutrons per Fission (MT=452)

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

LNU = 1, polynomial representation has been used;

LNU = 2, tabulated representation.

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

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

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

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

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

File 1 Number of Neutrons per Fission (MT=452)

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

If <u>LNU = 1</u>, 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 <u>LNU = 2</u>, the structure of the section is

[MAT, 1, 452/ ZA, AWR, 0, LNU, 0, 0]HEAD LNU = 2

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

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

File 1 Number of Neutrons per Fission (MT=452)

```
9.22340+ 4 2.32029+ 2          0          1          0          01043 1452   81
0.00000+ 0 0.00000+ 0          0          0          2          01043 1452   82
2.37000+ 0 1.25000- 7                                             1043 1452   83
                                                                  1043 1  0   84
```

```
9.22350+ 4 2.33025+ 2          0          2          0          01261 1452  349
0.00000+ 0 0.00000+ 0          0          0          1          71261 1452  350
         7          2                                             1261 1452  351
1.00000- 5 2.41880+ 0 1.20000+ 6 2.57130+ 0 3.00000+ 6 2.78010+ 01261 1452  352
4.00000+ 6 2.94430+ 0 7.00000+ 6 3.42930+ 0 7.85000+ 6 3.56890+ 01261 1452  353
2.00000+ 7 5.19620+ 0                                             1261 1452  354
                                                                  1261 1  0  355
```

File 1 Number of Neutrons per Fission (MT=452)

The energy dependence of $\bar{\nu}$ may be found by tabulating $\bar{\nu}$ as a function of incident neutron energy or by providing the coefficients for a polynomial expansion of $\bar{\nu}(E)$,

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

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

File 1 Radioactive Nuclide Production (MT=453)

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 reaction product data are given. (NS $\leq$ 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.)

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

RTYP is the designation of the reaction type leading to the described product nuclide state and is a floating-point equivalent of MT numbers (see Appendix B).

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 product nuclide (ZAP).

Q is the reaction Q-value (eV). Q = (rest mass of initial state - rest mass of final state.)

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

BR(N) is the branching ratio at the Nth 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.

*Although NS is limited to 5, the specific state number can be larger than 5 as long as the total number of states represented is no larger than 5.

File 1 Radioactive Nuclide Production (MT=453)

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 - 1 >

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

There will be NS subsections.

The structure of a subsection is

[MAT, 1, 453/ZA, AWR; LIS, 0; NE, NPR/

ES(1), ES(2),----------

----------, ES(NE)]LIST

[MAT, 1, 453/0.0, 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).

File 1 Radioactive Nuclide Production (MT=453)

```
9.22350+ 4 2.33025+ 2          0          0          1          01261 1453  356
9.22350+ 4 2.33025+ 2          0          0          2          31261 1453  357
1.00000- 5 2.00000+ 7                                              1261 1453  358
0.00000+ 0-5.23000+ 6          0          0          5          01261 1453  359
1.60000+ 1 9.22340+ 4 8.85890-14 1.00000+ 0 1.00000+ 0           1261 1453  360
0.00000+ 0-1.18500+ 7          0          0          5          01261 1453  361
1.70000+ 1 9.22330+ 4 1.61540-13 1.00000+ 0 1.00000+ 0           1261 1453  362
0.00000+ 0 6.54510+ 6          0          0          5          01261 1453  363
1.02000+ 2 9.22360+ 4 9.19250-16 1.00000+ 0 1.00000+ 0           1261 1453  364
                                                                 1261 1  0  365
```

FILE 1

MT = 453

Radioactive Decay Data

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

$$\sum BR(N) = 1.0$$

at each incident energy point, N.

NFP is the number of fission product nuclide states to be specified at each incident energy point (this is actually the number of sets of fission product identifiers - fission product yields). (NFP $\leq$ 1666.)

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 ith energy point. This array contains NFP sets of three parameters in the order ZAFP, FPS, YLD.

N1 is equal to 3*NFP, the number of items in the $C_n(E_i)$ array.

E_i is the incident neutron energy of the ith point (eV).

LE is a test to determine whether energy-dependent fission product yields are given:

LE = 0 implies no energy-dependence (only one set of fission product yield data given);

LE > 0 means that (LE + 1) sets of fission product yield data are given at (LE + 1) incident neutron energies.

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

The structure for a section is

[MAT, 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, I_3, 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.

File 1 Fission Product Yield Data (MT=454)

```
9.22350+ 4 2.33025+ 2          3          0          0          01261 1454  366
2.53000- 2 0.00000+ 0          2          0       3390       11301261 1454  367
2.40660+ 4 0.00000+ 0 2.51136-13 2.40670+ 4 0.00000+ 0 7.69417-141261 1454  368
2.40680+ 4 0.00000+ 0 0.00000+ 0 2.40700+ 4 0.00000+ 0 0.00000+ 01261 1454  369
2.50660+ 4 0.00000+ 0 2.74149-11 2.50670+ 4 0.00000+ 0 2.65144-111261 1454  370
2.50680+ 4 0.00000+ 0 6.99379-12 2.50690+ 4 0.00000+ 0 1.16063-121261 1454  371
2.50700+ 4 0.00000+ 0 1.29070-13 2.50710+ 4 0.00000+ 0 1.00054-141261 1454  372
2.60660+ 4 0.00000+ 0 3.83208-10 2.60670+ 4 0.00000+ 0 1.06058- 91261 1454  373
2.60680+ 4 0.00000+ 0 8.37454-10 2.60690+ 4 0.00000+ 0 4.31234-101261 1454  374

    (MISSING LINES)

6.71690+ 4 0.00000+ 0 4.26231-11 6.71700+ 4 0.00000+ 0 1.04056-111261 1454  924
6.71700+ 4 1.00000+ 0 1.03056-11 6.71710+ 4 0.00000+ 0 1.59086-111261 1454  925
6.71720+ 4 0.00000+ 0 5.79314-12 6.81610+ 4 0.00000+ 0 0.00000+ 01261 1454  926
6.81620+ 4 0.00000+ 0 0.00000+ 0 6.81630+ 4 0.00000+ 0 0.00000+ 01261 1454  927
6.81640+ 4 0.00000+ 0 0.00000+ 0 6.81650+ 4 0.00000+ 0 0.00000+ 01261 1454  928
6.81660+ 4 0.00000+ 0 8.93485-14 6.81670+ 4 0.00000+ 0 3.95214-141261 1454  929
6.81670+ 4 1.00000+ 0 3.95214-14 6.81680+ 4 0.00000+ 0 2.79151-131261 1454  930
6.81690+ 4 0.00000+ 0 1.15062-12 6.81700+ 4 0.00000+ 0 1.64089-121261 1454  931
6.81710+ 4 0.00000+ 0 3.54192-12 6.81720+ 4 0.00000+ 0 3.48189-121261 1454  932
5.00000+ 5 0.00000+ 0          3          0       3390       11301261 1454  933
2.40660+ 4 0.00000+ 0 3.81062-12 2.40670+ 4 0.00000+ 0 7.96129-131261 1454  934
2.40680+ 4 0.00000+ 0 5.35087-14 2.40700+ 4 0.00000+ 0 0.00000+ 01261 1454  935
2.50660+ 4 0.00000+ 0 4.27069-10 2.50670+ 4 0.00000+ 0 2.82046-101261 1454  936
2.50680+ 4 0.00000+ 0 6.13099-11 2.50690+ 4 0.00000+ 0 1.00016-111261 1454  937
2.50700+ 4 0.00000+ 0 1.06017-12 2.50710+ 4 0.00000+ 0 8.49137-141261 1454  938
2.60660+ 4 0.00000+ 0 6.11099- 9 2.60670+ 4 0.00000+ 0 1.16019- 81261 1454  939
2.60680+ 4 0.00000+ 0 7.54122- 9 2.60690+ 4 0.00000+ 0 3.82062- 91261 1454  940
2.60700+ 4 0.00000+ 0 1.29021- 9 2.60710+ 4 0.00000+ 0 3.38055-101261 1454  941

    (MISSING LINES)

6.71690+ 4 0.00000+ 0 2.28037-10 6.71700+ 4 0.00000+ 0 7.83127-111261 1454 1490
6.71700+ 4 1.00000+ 0 7.87127-11 6.71710+ 4 0.00000+ 0 8.38136-111261 1454 1491
6.71720+ 4 0.00000+ 0 2.38039-11 6.81610+ 4 0.00000+ 0 0.00000+ 01261 1454 1492
6.81620+ 4 0.00000+ 0 0.00000+ 0 6.81630+ 4 0.00000+ 0 0.00000+ 01261 1454 1493
6.81640+ 4 0.00000+ 0 0.00000+ 0 6.81650+ 4 0.00000+ 0 0.00000+ 01261 1454 1494
6.81660+ 4 0.00000+ 0 5.62091-14 6.81670+ 4 0.00000+ 0 3.41055-131261 1454 1495
6.81670+ 4 1.00000+ 0 3.42055-13 6.81680+ 4 0.00000+ 0 2.71044-121261 1454 1496
6.81690+ 4 0.00000+ 0 6.32102-12 6.81700+ 4 0.00000+ 0 1.28021-111261 1454 1497
6.81710+ 4 0.00000+ 0 1.91031-11 6.81720+ 4 0.00000+ 0 1.45023-111261 1454 1498
1.40000+ 7 0.00000+ 0          3          0       3390       11301261 1454 1499
2.40660+ 4 0.00000+ 0 1.59889-10 2.40670+ 4 0.00000+ 0 1.59889-111261 1454 1500
2.40680+ 4 0.00000+ 0 5.88591-13 2.40700+ 4 0.00000+ 0 0.00000+ 01261 1454 1501
2.50660+ 4 0.00000+ 0 4.09715- 8 2.50670+ 4 0.00000+ 0 1.31908- 81261 1454 1502
2.50680+ 4 0.00000+ 0 1.58890- 9 2.50690+ 4 0.00000+ 0 1.34906-101261 1454 1503
2.50700+ 4 0.00000+ 0 8.36419-12 2.50710+ 4 0.00000+ 0 3.33768-131261 1454 1504
2.60660+ 4 0.00000+ 0 1.24913- 6 2.60670+ 4 0.00000+ 0 1.19917- 61261 1454 1505
2.60680+ 4 0.00000+ 0 4.43692- 7 2.60690+ 4 0.00000+ 0 1.19917- 71261 1454 1506
2.60700+ 4 0.00000+ 0 2.39833- 8 2.60710+ 4 0.00000+ 0 3.15781- 91261 1454 1507

    (MISSING LINES)

6.71670+ 4 0.00000+ 0 5.57613- 8 6.71680+ 4 0.00000+ 0 1.13921- 71261 1454 2055
6.71690+ 4 0.00000+ 0 2.01860- 7 6.71700+ 4 0.00000+ 0 7.76460- 81261 1454 2056
6.71700+ 4 1.00000+ 0 7.80458- 8 6.71710+ 4 0.00000+ 0 1.00930- 71261 1454 2057
6.71720+ 4 0.00000+ 0 7.12505- 8 6.81610+ 4 0.00000+ 0 0.00000+ 01261 1454 2058
6.81620+ 4 0.00000+ 0 0.00000+ 0 6.81630+ 4 0.00000+ 0 0.00000+ 01261 1454 2059
6.81640+ 4 0.00000+ 0 7.43483-14 6.81650+ 4 0.00000+ 0 1.91867-121261 1454 2060
6.81660+ 4 0.00000+ 0 2.95794-11 6.81670+ 4 0.00000+ 0 1.83872-101261 1454 2061
6.81670+ 4 1.00000+ 0 1.83872-10 6.81680+ 4 0.00000+ 0 2.31839- 91261 1454 2062
6.81690+ 4 0.00000+ 0 1.22915- 8 6.81700+ 4 0.00000+ 0 2.66815- 81261 1454 2063
                                                                 1261 1   0 2064
```

FILE 1

MT = 454

Fission Product Yield Data

At each incident energy point

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

<u>LND</u> 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.

<u>NCD</u> is the number of terms in the polynomial expansion. (NCD $\leq$ 4)

<u>CD_m</u> are the coefficients for the polynomial.

<u>NR</u> is the number of interpolation ranges used. (NR $\leq$ 200)

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

<u>E_{int}</u> is the interpolation scheme (see Appendix E).

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

<u>NNF</u> is the number of precursor families considered.

<u>λi</u> is the decay constant (sec^{-1}) for the ith precursor.

File 1 Delayed Neutron Data (MT=455)

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 when values of $\bar{\nu}_d$ 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}/$\bar{\nu}_d$(E)]TAB1

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

File 1 Delayed Neutron Data (MT=455)

```
9.2235> -04 233.725                         1                       1182 1455  181
                                                          7         1182 1455  182
1.2717 -02 2.8446 -02 4.2521 -02 1.5332 -01 3.5363 -01 1.5367       1182 1455  183
3.9833                                                              1182 1455  184
                                                          2         1182 1455  185
8.0158      0.1     -10                                             1182 1455  186
                                                                    1182 1  0  187
```

```
9.22350+ 4 2.33025+ 2          0          2          0          01261 1455 2066
0.00000+ 0 0.00000+ 0          0          0          6          01261 1455 2067
1.27200- 2 3.17400- 2 1.16000- 1 3.11000- 1 1.40000+ 0 3.87000+ 01261 1455 2068
0.00000+ 0 0.00000+ 0          0          0          1          71261 1455 2069
         7          2                                             1261 1455 2070
1.00000- 5 1.67000- 2 1.20000+ 6 1.67000- 2 2.00000+ 6 1.67000- 21261 1455 2071
4.00000+ 6 1.67000- 2 7.00000+ 6 9.00000- 3 7.85000+ 6 9.00000- 31261 1455 2072
2.00000+ 7 9.00000- 3                                             1261 1455 2073
                                                                  1261 1  0 2074
```

File 1 Delayed Neutron Data (MT=455)

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

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

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

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

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

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

File 1 Prompt Neutrons per Fission (MT=456)

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

LNP = 1, polynomial representation has been used;

LNP = 2, tabulated representation.

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

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

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

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

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

File 1 Prompt Neutrons per Fission (MT=456)

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

If <u>LNP = 1</u> (polynomial representation used), the structure of the section is

[MAT, 1, 456/ZA, AWR, 0, LNP, 0, 0]HEAD LNP = 1

[MAT, 1, 456/0.0, 0.0, 0, 0, NCP, 0/CP_1, CP_2, ... CP_{NCP}]LIST

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

If <u>LNP = 2</u> (tabulated values of $\bar{\nu}$), the structure of the section is

[MAT, 1, 456/ZA, AWR, 0, LNP, 0, 0]HEAD LNP = 2

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

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

File 1 Prompt Neutrons per Fission (MT=456)

NO EXAMPLE IN ENDF-IV

```
9.22350+ 4 2.33025+ 2          0          2          0          01261 1456 2075
0.00000+ 0 0.00000+ 0          0          0          1          71261 1456 2076
         7          2                                              1261 1456 2077
1.00000- 5 2.40210+ 0 1.20000+ 6 2.55460+ 0 3.00000+ 6 2.76340+ 01261 1456 2078
4.00000+ 6 2.92760+ 0 7.00000+ 6 3.42030+ 0 7.85000+ 6 3.35990+ 01261 1456 2079
2.00000+ 7 5.18720+ 0                                              1261 1456 2080
```

File 1 Prompt Neutrons per Fission (MT=456)

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

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

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

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

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

File 1 Radioactive Decay Data (MT=457)

I. General information about the material

ZA = Designation of the original (radioactive) nuclide (=1000*Z + A)

LIS = Isomeric state flag for original nuclide (LIS = 0, ground state; LIS = 1, first isomeric state; etc.).

$T_{1/2}$ = Half-life of the original nuclide (seconds).

$\Delta T_{1/2}$ = Uncertainty in the half-life (should be considered as one standard deviation).

NAV = Total number of decay modes for which average energies are given.

$\bar{E}_x, \Delta\bar{E}_x$ = Average decay energy (eV) of radiation of type x and its uncertainty (eV) for decay heat applications. The β, γ and α energies are given in that order, with space reserved for zero β or γ entries. All non-γ and non-α energies are presently included as β energy. The α energy includes the recoil nucleus energy.

II. Decay mode information for each mode of decay:

NDK = total number of decay modes given.

RTYP = Indicates the mode of decay.

RFS = Isomeric state flag for daughter nuclide. (Fixed point number.)

Q = Total decay energy (eV) available in the corresponding decay process.

ΔQ = Uncertainty in Q value (eV).

BR = Fraction of the decay which proceeds by the corresponding decay mode. (e.g., if only β^- occurs and no isomeric states in the daughter nucleus are excited, then BR = 1.0 for β^- decay.)

ΔBR = Uncertainty in BR (should be given as one standard deviation)

III. Resulting radiation spectra

STYP = Decay type (Use mode of decay variable list).

NSP = Total number of spectra. (NSP may be zero.)

E and ΔE = Energy (eV) or radiation produced (E_{β^-}, E_{β^+}, E_γ, etc.).

I and ΔI = Intensity of radiation produced (relative units).

ICC and ΔICC = Internal conversion coefficient.

F and ΔF = Normalization factor (absolute intensity/relative intensity).

NE = Total number of tabulated energies.

File 1 Radioactive Decay Data (MT=457)

Formats

The structure of this section always starts with a HEAD record and ends with a SEND record. The section is divided into subsections as follows:

[MAT,1/457/	ZA	AWR	LIS	b	b	NSP	]	HEAD
[MAT,1/457/	$T_{1/2}$	$\Delta T_{1/2}$	b	b	2*NAV	NAV		
	$\bar{E}_\beta$	$\Delta\bar{E}_\beta$	$\bar{E}_\gamma$	$\Delta\bar{E}_\gamma$	$\bar{E}_\alpha$	$\Delta\bar{E}_\alpha$	]	LIST
[MAT,1/457/	ZA	AWR	b	b	6*NDK	NDK	/	
	$RTYP_1$	RFS_1	Q_1	ΔQ_1	BR_1	ΔBR_1		
	.							
	.							
	.							
	$RTYP_{NDK}$	RFS_{NDK}	Q_{NDK}	ΔQ_{NDK}	BR_{NDK}	ΔBR_{NDK}	]	LIST
[MAT,1,457/	STYP	b	b	b	6*(NE+1)	NE/	Repeat NSP times	
	F	ΔF	b	b	b	b	[omit if NSP=0]	
	E	ΔE	I	ΔI	ICC	ΔICC	]	LIST
[MAT,1,0/	b	b	b	b	b	b	]	SEND

File 1 Radioactive Decay Data (MT=457)

```
9.22350+ 4 2.33025+ 2          0          0          0          21261 1457 2082
2.22013+16 3.15360+13          0          0          6          31261 1457 2083
0.00000+ 0 0.00000+ 0 1.61310+ 5 0.00000+ 0 4.46800+ 6 0.00000+ 01261 1457 2084
9.22350+ 4 2.33025+ 2          0          0         12          21261 1457 2085
4.00000+ 0 0.00000+ 0 4.67880+ 6 2.50000+ 3 1.00000+ 0 0.00000+ 01261 1457 2086
6.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 4.00000- 9 0.00000+ 01261 1457 2087
4.00000+ 0 0.00000+ 0          0          0        102          161261 1457 2088
1.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2089
4.14600+ 6 3.00000+ 3 5.00000- 1 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2090
4.19000+ 6 3.00000+ 3 1.30000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2091
4.21600+ 6 3.00000+ 3 6.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2092
4.27100+ 6 5.00000+ 3 4.00000- 1 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2093
4.32500+ 6 3.00000+ 3 3.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2094
4.34500+ 6 4.00000+ 3 1.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2095
4.36800+ 6 3.00000+ 3 1.20000+ 1 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2096
4.37400+ 6 5.00000+ 3 6.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2097
4.39800+ 6 3.00000+ 3 5.60000+ 1 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2098
4.41700+ 6 5.00000+ 3 2.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2099
4.43000+ 6 5.00000+ 3 1.50000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2100
4.44400+ 6 3.00000+ 3 1.50000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2101
4.48400+ 6 3.00000+ 3 1.60000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2102
4.50200+ 6 3.00000+ 3 1.40000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2103
4.55600+ 6 3.00000+ 3 3.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2104
4.59800+ 6 3.00000+ 3 4.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2105
0.00000+ 0 0.00000+ 0          0          0        210          341261 1457 2106
1.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2107
3.16000+ 4 1.00000+ 2 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2108
4.21000+ 4 2.00000+ 2 5.00000- 2 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2109
5.12000+ 4 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2110
7.49000+ 4 1.00000+ 2 1.30000- 1 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2111
1.09120+ 5 5.00000+ 1 1.50000+ 0 2.00000- 1 9.70000- 2 0.00000+ 01261 1457 2112
1.15200+ 5 5.00000+ 2 1.10000- 1 3.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2113
1.40750+ 5 1.00000+ 2 1.70000- 1 2.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2114
1.43780+ 5 2.00000+ 1 9.70000+ 0 5.00000- 1 2.07000- 1 0.00000+ 01261 1457 2115
1.50960+ 5 3.00000+ 1 1.10000- 1 3.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2116
1.63360+ 5 2.00000+ 1 4.60000+ 0 3.00000- 1 1.54000- 1 0.00000+ 01261 1457 2117
1.73400+ 5 1.00000+ 2 1.50000- 2 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2118
1.82720+ 5 2.00000+ 2 4.00000- 1 4.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2119
1.85720+ 5 2.00000+ 1 5.40000+ 1 0.00000+ 0 1.14000- 1 0.00000+ 01261 1457 2120
1.94940+ 5 2.00000+ 1 6.50000- 1 5.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2121
1.98910+ 5 6.00000+ 1 3.00000- 2 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2122
2.02130+ 5 3.00000+ 1 1.05000+ 0 5.00000- 2 2.58000+ 0 0.00000+ 01261 1457 2123
2.05310+ 5 2.00000+ 1 4.90000+ 0 3.00000- 1 9.30000- 2 0.00000+ 01261 1457 2124
2.15310+ 5 5.00000+ 1 3.00000- 2 3.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2125
2.21370+ 5 3.00000+ 1 1.20000- 1 1.00000- 2 0.00000+ 0 0.00000+ 01261 1457 2126
2.25700+ 5 2.00000+ 2 2.00000- 3 2.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2127
2.28800+ 5 1.00000+ 2 8.00000- 3 1.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2128
2.33530+ 5 4.00000+ 1 4.00000- 2 4.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2129
2.40930+ 5 4.00000+ 1 7.50000- 2 8.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2130
2.46830+ 5 4.00000+ 1 6.00000- 2 6.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2131
2.66440+ 5 8.00000+ 1 8.00000- 3 1.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2132
2.75400+ 5 1.00000+ 2 2.00000- 2 2.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2133
2.83100+ 5 2.00000+ 2 3.00000- 3 3.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2134
2.91600+ 5 1.00000+ 2 2.00000- 2 2.00000- 3 0.00000+ 0 0.00000+ 01261 1457 2135
3.11600+ 5 6.00000+ 2 5.00000- 3 5.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2136
3.43910+ 5 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2137
3.56000+ 5 2.00000+ 2 3.00000- 3 3.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2138
3.87850+ 5 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 1457 2139
4.10200+ 5 2.00000+ 2 1.60000- 3 2.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2140
4.47500+ 5 2.00000+ 2 1.60000- 3 2.00000- 4 0.00000+ 0 0.00000+ 01261 1457 2141
                                                                  1261 1  0 2142
```

File 1 Radioactive Decay Data (MT=457)

Decay modes defined

Variable	Mode of decay	
0.0	γ	Gamma decay (not used for mode of decay)
1.0	β^-	Beta decay
2.0	β^+	Positron and/or electron capture decay
3.0	IT	Isomeric transition (in general, present only when the state being considered is an isomeric state)
4.0	α	Alpha decay
5.0	β^-,n	Neutron emission (generally given for delayed neutrons)
6.0	SF	Spontaneous fission

File 2 Resonance Parameter Data (General Structure)

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

NIS is the number of isotopes in this material (NIS < 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 indicating whether average fission widths are given in the unresolved resonance region for this isotope:

LFW = 0, average fission widths are not given;

LFW = 1, average fission widths are given.

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

EL is the lower limit for an energy range.*

EH is the upper limit for an energy range.*

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

LRU = 0, means only effective scattering radius is given (LRF = 0, NLS = 0, LFW = 0 required)

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.

File 2 Resonance Parameter Data
(General Structure)

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; those for each isotope, with a CØNT (isotope) record.

File 2 Resonance Parameter Data
(General Structure)

```
9.22350+ 4 2.33025+ 2          0          0          1          01261 2151 2144
9.22350+ 4 1.00000+ 0          0          1          2          01261 2151 2145
1.00000+ 0 8.20000+ 1          1          1          0          01261 2151 2146
```

(SUBSECTION OF RESOLVED RESONANCE PARAMETERS)

```
8.20000+ 1 2.50000+ 4          2          2          0          01261 2151 2279
```

FILE 2

Resonance Parameter Data

General Structure

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

File 2 Resonance Parameter Data (Special Case, LRP=0)

The following quantities are defined

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

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. AP is defined in the relation $\sigma_{pot} = 4\Pi\ (AP)^2$.

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

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

File 2 Resonance Parameter Data (Special Case, LRP=0)

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 are given for this material):

```
[MAT, 2, 151/ ZA,   AWR, 0,    0,    NIS, 0]HEAD    NIS = 1
[MAT, 2, 151/ ZAI,  ABN, 0,    LFW,  NER, 0]CØNT    LFW = 0, NER = 1
[MAT, 2, 151/ EL,   EH,  LRU,  LRF,  0,   0]CØNT    LRU = 0, LRF = 0
[MAT, 2, 151/ SPI,  AP,  0,    0,    NLS, 0]CØNT    NLS = 0
[MAT, 2,  0 / 0.0,  0.0, 0,    0,    0,   0]SEND
[MAT, 0,  0 / 0.0,  0.0, 0,    0,    0,   0]FEND
```

File 2 Resonance Parameter Data
(Special Case, LRP=0)

```
2.00400+ 3 4.00150+ 0          0          0          1          01270 2151   83
2.00400+ 3 1.00000+ 0          0          0          1          01270 2151   84
1.00000- 5 1.00000+ 5          0          0          0          01270 2151   85
0.00000+ 0 2.45790- 1          0          0          0          01270 2151   86
```

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 Resolved Resonance Parameter Data (Single or Multilevel Breit-Wigner)

The following quantities are defined for use when LRF = 1 and 2 (see Appendix D for formulae):

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

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

AP is the 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. AP is defined in the relation $\sigma_{pot} = 4\Pi\ (AP)^2$.

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

NLS is the number of ℓ states 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 $\Gamma\gamma$ evaluated at the resonance energy ER.

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

File 2 Resolved Resonance Parameter Data
(Single or Multilevel Breit-Wigner)

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

[MAT, 2, 151/ SPI, AP, 0, 0, NLS, 0]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 NLS ℓ-state has been specified (in order of increasing value of ℓ). The values of ER for each ℓ-state shall be ordered by increasing neutron energy.

File 2 Resolved Resonance Parameter Data
(Single or Multilevel Breit-Wigner)

```
 3.50000+ 0 9.36630- 1          0          0          1          01261 2151 2147
 2.33020+ 2 0.00000+ 0          0          0        780        1301261 2151 2148
-1.49000+ 0 3.50000+ 0 2.37680- 1 3.68200- 3 2.70000- 2 2.07000- 11261 2151 2149
 2.90000- 1 3.50000+ 0 1.35000- 1 3.01570- 6 3.60000- 2 9.90000- 21261 2151 2150
 1.14000+ 0 3.50000+ 0 1.50820- 1 1.51610- 5 3.46000- 2 1.16200- 11261 2151 2151
 2.03500+ 0 3.50000+ 0 4.46960- 2 7.66050- 6 3.48740- 2 9.81400- 31261 2151 2152
 2.92000+ 0 3.50000+ 0 2.20000- 1 4.85300- 6 2.00000- 2 2.00000- 11261 2151 2153
 3.14700+ 0 3.50000+ 0 1.39610- 1 2.24050- 5 3.32100- 2 1.06370- 11261 2151 2154
 3.60900+ 0 3.50000+ 0 8.43790- 2 4.55940- 5 3.36960- 2 5.06370- 21261 2151 2155
 4.84800+ 0 3.50000+ 0 3.95920- 2 6.03520- 5 3.59450- 2 3.58700- 31261 2151 2156
 5.44800+ 0 3.50000+ 0 9.01200- 2 3.36110- 6 6.00000- 2 3.01170- 21261 2151 2157
 5.60000+ 0 3.50000+ 0 6.41920- 1 3.33190- 5 2.00000- 2 6.21890- 11261 2151 2158
 6.21000+ 0 3.50000+ 0 2.30900- 1 6.37950- 5 4.34690- 2 1.87360- 11261 2151 2159
 6.38200+ 0 3.50000+ 0 4.47880- 2 2.68340- 4 3.49720- 2 9.54800- 31261 2151 2160
 7.07700+ 0 3.50000+ 0 6.39340- 2 1.26600- 4 3.55740- 2 2.82330- 21261 2151 2161
 8.78100+ 0 3.50000+ 0 1.23290- 1 1.12340- 3 3.11700- 2 9.10000- 21261 2151 2162
 9.28600+ 0 3.50000+ 0 1.10760- 1 1.63640- 4 3.56000- 2 7.50000- 21261 2151 2163
 9.73000+ 0 3.50000+ 0 2.69050- 1 5.30280- 5 3.20000- 2 2.37000- 11261 2151 2164
 1.01800+ 1 3.50000+ 0 1.00560- 1 6.18980- 5 3.80000- 2 6.25000- 21261 2151 2165
 1.08000+ 1 3.50000+ 0 9.35090- 1 9.33320- 5 6.70000- 2 8.68000- 11261 2151 2166
 1.16660+ 1 3.50000+ 0 4.72770- 2 6.27440- 4 4.04000- 2 6.25000- 31261 2151 2167
 1.23960+ 1 3.50000+ 0 6.32620- 2 1.26220- 3 3.45000- 2 2.75000- 21261 2151 2168
 1.28610+ 1 3.50000+ 0 1.19550- 1 5.30760- 5 3.35000- 2 8.60000- 21261 2151 2169
 1.32750+ 1 3.50000+ 0 1.51440- 1 3.93500- 5 2.86000- 2 1.22800- 11261 2151 2170
 1.37000+ 1 3.50000+ 0 1.23940- 1 3.70130- 5 3.04000- 2 9.35000- 21261 2151 2171
 1.39960+ 1 3.50000+ 0 4.96540- 1 5.37230- 4 2.60000- 2 4.70000- 11261 2151 2172
   (MISSING RESONANCE PARAMETER ENTRIES)

 6.30200+ 1 3.50000+ 0 2.40090- 1 9.08960- 5 4.00000- 2 2.00000- 11261 2151 2253
 6.33200+ 1 3.50000+ 0 2.50100- 1 1.02090- 4 5.00000- 2 2.00000- 11261 2151 2254
 6.36900+ 1 3.50000+ 0 6.21070- 1 1.07440- 3 6.00000- 2 5.60000- 11261 2151 2255
 6.43000+ 1 3.50000+ 0 4.75450- 2 1.24470- 3 3.90000- 2 7.30000- 31261 2151 2256
 6.58000+ 1 3.50000+ 0 9.64230- 2 4.23270- 4 5.00000- 2 4.60000- 21261 2151 2257
 6.64020+ 1 3.50000+ 0 8.94490- 2 4.49480- 4 4.50000- 2 4.40000- 21261 2151 2258
 6.72470+ 1 3.50000+ 0 9.00810- 2 8.09380- 5 4.10000- 2 4.90000- 21261 2151 2259
 6.84400+ 1 3.50000+ 0 2.50040- 1 3.76410- 5 5.00000- 2 2.00000- 11261 2151 2260
 6.85300+ 1 3.50000+ 0 1.60110- 1 1.08360- 4 6.00000- 2 1.00000- 11261 2151 2261
 6.92930+ 1 3.50000+ 0 2.00720- 1 7.15300- 4 4.00000- 2 1.60000- 11261 2151 2262
 7.04040+ 1 3.50000+ 0 1.72720- 1 2.71560- 3 5.00000- 2 1.20000- 11261 2151 2263
 7.07500+ 1 3.50000+ 0 2.37410- 1 2.40910- 3 3.50000- 2 2.00000- 11261 2151 2264
 7.16100+ 1 3.50000+ 0 1.60290- 1 2.91360- 4 4.00000- 2 1.20000- 11261 2151 2265
 7.23900+ 1 3.50000+ 0 1.38610- 1 2.61150- 3 3.10000- 2 1.05000- 11261 2151 2266
 7.29100+ 1 3.50000+ 0 3.60370- 1 3.67170- 4 4.00000- 2 3.20000- 11261 2151 2267
 7.45440+ 1 3.50000+ 0 1.01670- 1 2.72870- 3 3.80000- 2 6.09370- 21261 2151 2268
 7.51700+ 1 3.50000+ 0 2.90890- 1 8.88330- 4 5.00000- 2 2.40000- 11261 2151 2269
 7.55410+ 1 3.50000+ 0 2.33360- 1 1.36210- 3 3.20000- 2 2.00000- 11261 2151 2270
 7.67500+ 1 3.50000+ 0 1.16110- 1 1.07320- 4 3.60000- 2 8.00000- 21261 2151 2271
 7.74920+ 1 3.50000+ 0 1.12990- 1 9.86810- 4 4.00000- 2 7.20000- 21261 2151 2272
 7.81170+ 1 3.50000+ 0 1.48220- 1 1.22450- 3 4.70000- 2 1.00000- 11261 2151 2273
 7.96720+ 1 3.50000+ 0 1.29790- 1 7.35570- 4 4.40000- 2 8.50000- 21261 2151 2274
 8.03570+ 1 3.50000+ 0 1.74840- 1 8.38510- 4 4.00000- 2 1.34000- 11261 2151 2275
 8.14340+ 1 3.50000+ 0 1.32040- 1 1.04330- 3 4.10000- 2 9.00000- 21261 2151 2276
 8.35900+ 1 3.50000+ 0 1.18270- 1 1.17030- 3 4.80000- 2 6.91000- 21261 2151 2277
 8.68800+ 1 3.50000+ 0 8.01200- 2 7.19580- 4 5.20000- 2 2.74000- 21261 2151 2278
```

File 2 Resolved Resonance Parameter Data
(Single or Multilevel Breit-Wigner)

FILE 2

Resonance Parameter Data

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

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

$$GT_i = GN_i + GG_i + GF_i$$

File 2 Resolved Resonance Parameter Data
(Reich-Moore Parameters)

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. (AM = 0.0 required.)

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 $\Gamma\gamma$ 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.

File 2 Resolved Resonance Parameter Data
(Reich-Moore Parameters)

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.

File 2 Resolved Resonance Parameter Data
(Reich-Moore Parameters)

```
2.50        1.0092                                  1          1113 2151  107
2.3174 -02                      6                  18         31118 2151  108
0.0         0.0         0.0         0.0         0.0         0.0         1113 2151  109
0.0         0.0         0.0         0.0         0.0         0.0         1110 2151  110
0.0         0.0         0.0         0.0         0.0         0.0         1110 2151  111
                                    0                       1           1113 2151  112
2.5         0.0                                           876        731113 2151  113
0.          0.          0.          0.          -2.79       3.700  -311113 2151  114
2.5726 -03  -0.2336-02  -2.790      3.78    -01 1.6110  -04 -6.8078-361113 2151  115
0.          0.          0.          0.          1.00    -01 7.86   -221113 2151  116
3.1983 -01  -1.2945-06  1.800   -01 7.900   -02 4.2644  -07 -0.5929-381113 2151  117
0.          0.          0.          0.          1.420       3.980  -411113 2151  118
9.8431 -05  -5.1965-05  1.426       3.500   -01 3.5486  -06 -1.1183-251113 2151  119
0.          0.          0.          0.          1.700       1.20   -011113 2151  120
1.6990 -04  -4.4167-07  1.700       1.280   -01 3.9887  -05 8.8334 -361113 2151  121
 -   -   -   -   -   -   -   -   -   -   -   -   -   -   -   -   -   -   -   -
0.          0.          0.          0.          6.438   -01 4.884  -211113 2151  258
3.4727 -04  -3.3582-05  6.438   -01 4.000   -01 5.4382  -05 -2.4642-351113 2151  259
```

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 Resolved Resonance Parameter Data (Adler-Adler Parameters)

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.

*Reserved for use in ENDF/A only.

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.

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

= 3, backyround 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 ≤ 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).

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.

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 for the total cross section. Here and below, the subscript n denotes the nth level.

DEF_n is the resonance energy for the fission cross section.

DEC_n is the resonance energy for the radiative capture cross section.

DWT_n is the value of Γ/2, (v), used for the total cross section.

DWF_n is the value of Γ/2, (v), used for the fission cross section.

DWC_n is the value of Γ/2, (v), used for the radiative capture cross section.

Note: $DET_n = DEF_n = DEC_n$ and $DWT_n = DWF_n = DWC_n$.

GRT_n is related to the symmetrical total cross section parameter.

GIT_n is related to the asymmetrical total cross section parameter.

GRF_n is the symmetrical fission parameter.

GIF_n is the asymmetrical fission parameter.

GRC_n is the symmetrical capture parameter.

GIC_n is the asymmetrical capture parameter.

File 2 Resolved Resonance Parameter Data
(Adler-Adler Parameters)

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/
            AT1,     AT2,   AT3,   AT4,   BT1,     BT2
            AF1,   ---------------------------,    BF2
            AC1,   ---------------------------,    BC2]LIST]
  [MAT, 2, 151/0.0,    0.0,   L,     0,     NJS,     0]CØNT(ℓ)
  [MAT, 2, 151/AJ,     AM,    0,     0,     12*NLJ,  NLJ/
              DET1,    DWT1,  GRT1,  GIT1,  DEF1,    DWF1,
              GRF1,    GIF1,  DEC1,  DWC1,  GRC1,    GIC1,
              DET2,    DWT2,  ----------------------------
              ---------------------------------,    GIC2,
              DET3,  -------------------------------------
              ---------------------------------------------
              ---------------------------------,    GICNLJ]LIST
```

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

File 2 Resolved Resonance Parameter Data
(Adler-Adler Parameters)

3.5	0.8944				1	1182	2151	101
232.9612	0.0	0			60	101182	2151	102
11.47	3.0	5.0270E-04	38.0E-03	4.0E-03		10.0E-0311182	2151	103
12.39	4.0	1.3024E-03	45.0E-03	23.0E-03		0.0E-0311182	2151	104
12.90	3.0	3.9910E-05	43.0E-03	40.0E-03		-10.0E-0311182	2151	105
13.34	4.0	9.8448E-05	48.0E-03	10.0E-03		100.0E-0311182	2151	106
13.70	3.0	7.2320E-05	48.0E-03	70.0E-03		-100.0E-0311182	2151	107
14.08	4.0	2.9936E-04	48.0E-03	70.0E-03		10.0E-0311182	2151	108
14.53	3.0	1.2960E-04	29.0E-03	23.0E-03		-30.0E-0311182	2151	109
15.45	4.0	2.5160E-04	55.0E-03	45.0E-03		150.0E-0311182	2151	110
16.15	3.0	3.6910E-04	48.0E-03	16.0E-03		300.0E-0311182	2151	111
16.67	4.0	2.7760E-04	64.0E-03	76.0E-03		-200.0E-0311182	2151	112

File 2 Resolved Resonance Parameter Data
(Adler-Adler Parameters)

The background correction for the total cross section is calculated by using the six constants in the following manner:

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

where $C = \pi\lambda\!\!\!^{-2} = \pi/k^2$ and $k = 2.19677 \times 10^{-3}\left(\frac{AWRI}{AWRI + 1.0}\right)\sqrt{E(eV)}$.
The background terms for the fission and radiative capture cross sections are calculated in a similar manner.

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

File 2 Unresolved Resonance Parameter Data
(Parameters are Energy Independent)
(Fission Widths not given)

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

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 Nth 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 LFR = 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 between the cross-sections obtained from average resonance parameters (normally, INT = 1.)

GNO is the average reduced neutron width. It is energy dependent if LRU = 2.

GG is the average radiation width. It 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.

File 2 Unresolved Resonance Parameter Data
(Parameters are Energy Independent)
(Fission Widths not given)

If <u>LFW = 0</u> (fission widths not given),

<u>LRU = 2</u> (unresolved parameters),

<u>LRF = 1</u> (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.

File 2 Unresolved Resonance Parameter Data
(Parameters are Energy Independent)
(Fission Widths not given)

```
2.00000+ 3 1.00000+ 4          2          1          0          01283 2151  299
1.50000+ 0 9.60000- 1          0          0          2          01283 2151  300
1.95274+ 2 0.00000+ 0          0          0         12          21283 2151  301
4.32000+ 1 1.00000+ 0 1.00000+ 0 9.07200- 3 1.25000- 1 0.00000+ 01283 2151  302
2.59000+ 1 2.00000+ 0 1.00000+ 0 5.43900- 3 1.25000- 1 0.00000+ 01283 2151  303
1.95274+ 2 0.00000+ 0          1          0         24          41283 2151  304
1.29600+ 2 0.00000+ 0 1.00000+ 0 5.18400- 3 1.25000- 1 0.00000+ 01283 2151  305
4.32000+ 1 1.00000+ 0 2.00000+ 0 1.72800- 3 1.25000- 1 0.00000+ 01283 2151  306
2.59000+ 1 2.00000+ 0 2.00000+ 0 1.03600- 3 1.25000- 1 0.00000+ 01283 2151  307
1.85000+ 1 3.00000+ 0 1.00000+ 0 7.40000- 4 1.25000- 1 0.00000+ 01283 2151  308
```

File 2 Unresolved Resonance Parameter Data
(Parameters are Energy Independent)
(Fission Widths not given)

File 2 Unresolved Resonance Parameter Data
(Fission Widths are given)
(Only Fission Widths are Energy Dependent)

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 Nth 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 LFR = 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 between the cross-sections obtained from average resonance parameters (normally, INT = 1.)

GNO is the average reduced neutron width. It is energy dependent if LRU = 2.

GG is the average radiation width. It 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.

File 2 Unresolved Resonance Parameter Data
(Fission Widths are given)
(Only Fission Widths are Energy Dependent)

If <u>LFW = 1</u> (fission widths given),

<u>LRU = 2</u> (unresolved parameters),

<u>LRF = 1</u> (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/
ES_1,	ES_2,	ES_3,	.,	.,	.,
.,	.,	.,	.,	ES_{NE}	]LIST
[MAT, 2, 151/AWRI,	0.0,	L,	0,	NJS,	0]CØNT(1)
[MAT, 2, 151/0.0,	0.0,	L,	MUF,	NE+6,	0/
D,	AJ,	AMUN,	GNO,	GG,	0.0,
GF_1,	GF_2,	GF_3,	.,	.,	.,
.,	.,	.,	GF_{NE}		]LIST

In the above section, interpolation is assumed to be log-log.

File 2 Unresolved Resonance Parameter Data
(Fission Widths are given)
(Only Fission Widths are Energy Dependent)

```
0.00000+ 0 9.22500- 1          0          0          2          21162 2151  261
5.00000+ 2 1.00000+ 4                                                1162 2151  262
2.42133+ 2 0.00000+ 0          0          0          1          01162 2151  263
0.00000+ 0 0.00000+ 0          0          3          8          01162 2151  264
1.82000+ 1 5.00000- 1 1.00000+ 0 3.70700- 3 3.90000- 2 0.00000+ 01162 2151  265
1.30300- 3 1.62700- 2                                                1162 2151  266
2.42133+ 2 0.00000+ 0          1          0          2          01162 2151  267
0.00000+ 0 0.00000+ 0          1          3          8          01162 2151  268
1.82000+ 1 5.00000- 1 1.00000+ 0 1.59300- 3 3.90000- 2 0.00000+ 01162 2151  269
1.30300- 3 1.62700- 2                                                1162 2151  270
0.00000+ 0 0.00000+ 0          1          3          8          01162 2151  271
9.10000+ 0 1.50000+ 0 1.00000+ 0 9.35000- 4 3.90000- 2 0.00000+ 01162 2151  272
6.51500- 4 8.13500- 3                                                1162 2151  273
```

File 2 Unresolved Resonance Parameter Data
(Fission Widths are given)
(Only Fission Widths are Energy Dependent)

File 2 Unresolved Resonance Parameter Data
(All Parameters are Energy Dependent)

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

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 Nth 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 LFR = 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 between the cross-sections obtained from average resonance parameters (normally, INT = 1.)

GNO is the average reduced neutron width. It is energy dependent if LRU = 2.

GG is the average radiation width. It 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.

File 2 Unresolved Resonance Parameter Data
(All Parameters are Energy Dependent)

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

LRU = 2 (unresolved parameters).

LRF = 2 (all energy-dependent parameters).

The structure of a subsection is:

[MAT, 2, 151/SPI,	A,	0,	0,	NLS,	0]CØNT
[MAT, 2, 151/AWRI,	0.0,	L	0,	NJS,	0]CØNT
[MAT, 2, 151/AJ,	0.0,	INT,	0,	(6*NE)+6,	NE /
0.0,	0.0,	AMUX,	AMUN,	AMUG,	AMUF,
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

The LIST record is repeated until all 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.

File 2 Unresolved Resonance Parameter Data
(All Parameters are Energy Dependent)

```
3.50000+ 0 9.56630- 1          0          0          2          01261 2151 2280
2.33025+ 2 0.00000+ 0          0          0          2          01261 2151 2281
3.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2282
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 2.00000+ 01261 2151 2283
8.20000+ 1 1.00000+ 0 0.00000+ 0 9.32780- 5 3.50000- 2 3.23080- 11261 2151 2284
8.65000+ 1 1.00000+ 0 0.00000+ 0 8.44860- 5 3.50000- 2 3.40560- 11261 2151 2285
9.10000+ 1 1.00000+ 0 0.00000+ 0 8.59390- 5 3.50000- 2 2.63940- 11261 2151 2286
9.55000+ 1 1.00000+ 0 0.00000+ 0 8.73180- 5 3.50000- 2 2.05980- 11261 2151 2287
1.00000+ 2 1.00000+ 0 0.00000+ 0 9.94340- 5 3.50000- 2 8.46200- 21261 2151 2288
1.10000+ 2 1.00000+ 0 0.00000+ 0 8.66200- 5 3.50000- 2 1.32250- 11261 2151 2289
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.42000+ 4 1.00000+ 0 0.00000+ 0 9.41230- 5 3.50000- 2 3.13230- 11261 2151 2400
2.44000+ 4 1.00000+ 0 0.00000+ 0 9.97630- 5 3.50000- 2 5.69080- 11261 2151 2401
2.46000+ 4 1.00000+ 0 0.00000+ 0 7.95710- 5 3.50000- 2 5.86170- 11261 2151 2402
2.50000+ 4 1.00000+ 0 0.00000+ 0 8.55930- 5 3.50000- 2 4.71280- 11261 2151 2403
4.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2404
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 1.00000+ 01261 2151 2405
8.20000+ 1 1.00000+ 0 0.00000+ 0 9.32780- 5 3.50000- 2 1.61550- 11261 2151 2406
8.65000+ 1 1.00000+ 0 0.00000+ 0 1.03340- 4 3.50000- 2 1.16900- 11261 2151 2407
9.10000+ 1 1.00000+ 0 0.00000+ 0 1.05120- 4 3.50000- 2 9.05980- 21261 2151 2408
9.55000+ 1 1.00000+ 0 0.00000+ 0 1.06810- 4 3.50000- 2 7.07030- 21261 2151 2409
1.00000+ 2 1.00000+ 0 0.00000+ 0 9.97820- 5 3.50000- 2 8.46440- 21261 2151 2410
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.34000+ 4 1.00000+ 0 0.00000+ 0 8.62980- 5 3.50000- 2 1.49700- 11261 2151 2521
2.42000+ 4 1.00000+ 0 0.00000+ 0 9.46010- 5 3.50000- 2 1.54750- 11261 2151 2522
2.44000+ 4 1.00000+ 0 0.00000+ 0 1.00270- 4 3.50000- 2 2.81160- 11261 2151 2523
2.46000+ 4 1.00000+ 0 0.00000+ 0 7.99750- 5 3.50000- 2 2.89600- 11261 2151 2524
2.50000+ 4 1.00000+ 0 0.00000+ 0 8.55930- 5 3.50000- 2 2.35080- 11261 2151 2525
2.33025+ 2 0.00000+ 0          1          0          4          01261 2151 2526
2.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2527
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 2.00000+ 01261 2151 2528
8.20000+ 1 1.16000+ 0 0.00000+ 0 2.32000- 4 3.50000- 2 3.32000- 11261 2151 2529
8.65000+ 1 1.16000+ 0 0.00000+ 0 2.32000- 4 3.50000- 2 3.32000- 11261 2151 2530
9.10000+ 1 1.16000+ 0 0.00000+ 0 2.32000- 4 3.50000- 2 3.32000- 11261 2151 2531
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.46000+ 4 1.16000+ 0 0.00000+ 0 2.32000- 4 3.50000- 2 3.32000- 11261 2151 2647
2.50000+ 4 1.16000+ 0 0.00000+ 0 2.32000- 4 3.50000- 2 3.32000- 11261 2151 2648
3.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2649
0.00000+ 0 0.00000+ 0 0.00000+ 0 2.00000+ 0 0.00000+ 0 1.00000+ 01261 2151 2650
8.20000+ 1 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 1.27000- 11261 2151 2651
8.65000+ 1 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 1.27000- 11261 2151 2652
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.46000+ 4 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 1.27000- 11261 2151 2769
2.50000+ 4 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 1.27000- 11261 2151 2770
4.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2771
0.00000+ 0 0.00000+ 0 0.00000+ 0 2.00000+ 0 0.00000+ 0 2.00000+ 01261 2151 2772
8.20000+ 1 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 2.86000- 11261 2151 2773
8.65000+ 1 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 2.86000- 11261 2151 2774
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.46000+ 4 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 2.86000- 11261 2151 2891
2.50000+ 4 1.00000+ 0 0.00000+ 0 2.00000- 4 3.50000- 2 2.86000- 11261 2151 2892
5.00000+ 0 0.00000+ 0          2          0        726        1201261 2151 2893
0.00000+ 0 0.00000+ 0 0.00000+ 0 1.00000+ 0 0.00000+ 0 1.00000+ 01261 2151 2894
8.20000+ 1 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 2895
```

(MISSING UNRESOLVED RESONANCE PARAMETERS)

```
2.32000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3009
2.34000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3010
2.42000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3011
2.44000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3012
2.46000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3013
2.50000+ 4 1.12000+ 0 0.00000+ 0 2.24000- 4 3.50000- 2 1.43000- 11261 2151 3014
```

File 2 Unresolved Resonance Parameter Data
(All Parameters are Energy Dependent)

File 3 Neutron Cross Sections

For File 3 the following quantities are defined:

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

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

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

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

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

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

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

= 2, the final state is the excited state.

. .

. .

. .

= 98, an unspecified range of final states.

= 99, all final states.

Q is the reaction Q-value (eV).

S is the temperature (°K). NOTE: If the LR flag is used, S becomes Q_1 for the reaction corresponding to LR.

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

LR is a flag to be used in the reactions MT = 51, 52, 53,...., 90, and 91, to define x in (n,n'x). (See Section 3.24.4.)

NR is the number of energy ranges that have been given. A different interpolation scheme may be given for each range. (NR $\leq$ 200, but normally $\leq$ 20).

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

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

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

File 3 Neutron Cross Sections

The structure of a section is

[MAT, 3, MT/ZA , AWR, LIS, LFS, 0 , 0]HEAD

[MAT, 3, MT/S , Q , LT, LR , NR, NP/E_{int}/σ(E)]TAB1

[MAT, 3, 0 /0.0, 0.0, 0, 0, 0 , 0]SEND

```
9.22350+ 4 2.33025+ 2          0         99          0          01261 3  1 3017
0.00000+ 0 0.00000+ 0          0          0          3       11341261 3  1 3018
       239          5        822          2       1154          31261 3  1 3019
1.00000- 5 3.67479+ 4 2.12500- 5 2.52034+ 4 4.09380- 5 1.81562+ 41261 3  1 3020
7.04690- 5 1.38382+ 4 1.00000- 4 1.16168+ 4 1.50000- 4 9.48589+ 31261 3  1 3021
1.93750- 4 8.34719+ 3 2.70310- 4 7.06804+ 3 3.85160- 4 5.92259+ 31261 3  1 3022
5.00000- 4 5.19931+ 3 6.25000- 4 4.65149+ 3 8.12500- 4 4.08096+ 31261 3  1 3023
1.00000- 3 3.67965+ 3 1.25000- 3 3.28818+ 3 1.43750- 3 3.06459+ 31261 3  1 3024
1.71880- 3 2.80078+ 3 2.00000- 3 2.59508+ 3 2.25000- 3 2.44573+ 31261 3  1 3025
2.62500- 3 2.26326+ 3 3.00000- 3 2.11631+ 3 3.00000- 3 2.11631+ 31261 3  1 3026
3.50000- 3 1.95857+ 3 4.00000- 3 1.83154+ 3 4.50000- 3 1.72640+ 31261 3  1 3027
5.00000- 3 1.63751+ 3 5.50000- 3 1.55928+ 3 6.00000- 3 1.49115+ 31261 3  1 3028
6.00000- 3 1.49115+ 3 7.00000- 3 1.37775+ 3 8.00000- 3 1.28659+ 31261 3  1 3029
9.00000- 3 1.21126+ 3 9.00000- 3 1.21126+ 3 1.00000- 2 1.14766+ 31261 3  1 3030

    (MISSING LINES) (THERMAL REGION)

9.29920- 1 7.93507+ 1 9.44920- 1 8.07723+ 1 9.49920- 1 8.14982+ 11261 3  1 3096
9.58670- 1 8.27784+ 1 9.65230- 1 8.37463+ 1 9.70150- 1 8.44769+ 11261 3  1 3097
9.77540- 1 8.55817+ 1 9.84920- 1 8.66940+ 1 9.88690- 1 8.76208+ 11261 3  1 3098
9.94340- 1 8.90252+ 1 1.00000+ 0 9.04507+ 1 1.00000+ 0 9.63800+ 01261 3  1 3099
1.03600+ 0 9.41382+ 0 1.05500+ 0 9.29550+ 0 1.07300+ 0 8.52700+ 01261 3  1 3100
1.09100+ 0 7.19610+ 0 1.18200+ 0-1.91285+ 1 1.20000+ 0-2.30500+ 11261 3  1 3101
1.21800+ 0-2.44439+ 1 1.23600+ 0-2.44662+ 1 1.32700+ 0-1.49745+ 11261 3  1 3102
1.36400+ 0-1.28195+ 1 1.45500+ 0-8.88020+ 0 1.54500+ 0-6.70980+ 01261 3  1 3103
1.63600+ 0-5.25670+ 0 1.72700+ 0-4.15820+ 0 1.81800+ 0-3.42100+ 01261 3  1 3104
1.90900+ 0-2.66550+ 0 2.00000+ 0-2.56000+ 0 2.18200+ 0-1.91980+ 01261 3  1 3105
2.36400+ 0-1.15200+ 0 2.45500+ 0-1.47840+ 0 2.54500+ 0-2.15980+ 01261 3  1 3106

    (MISSING LINES) (BACKGROUND IN RESONANCE REGION)

7.39910+ 1-8.04900- 1 7.44260+ 1-4.66500- 1 7.45710+ 1 3.48350+ 01261 3  1 3265
7.47890+ 1 1.60120+ 1 7.48250+ 1 1.58696+ 1 7.48620+ 1 1.33061+ 11261 3  1 3266
7.48980+ 1 8.24280+ 0 7.49340+ 1 6.40610+ 0 7.49710+ 1 4.57330+ 01261 3  1 3267
7.50430+ 1 2.06360+ 0 7.50790+ 1 1.03880+ 0 7.51880+ 1-1.24130+ 01261 3  1 3268
7.54420+ 1-3.07539+ 0 7.55870+ 1-3.26694+ 0 7.58050+ 1-2.07292+ 01261 3  1 3269
7.60230+ 1-9.47000- 1 7.61680+ 1-9.55400- 1 7.67480+ 1-1.02510+ 01261 3  1 3270
7.69660+ 1-8.99400- 1 7.71110+ 1-9.04900- 1 7.75620+ 1-1.10810+ 01261 3  1 3271
7.76190+ 1-1.06090+ 0 7.83000+ 1-1.21000+ 0 7.83630+ 1 4.62480+ 01261 3  1 3272
7.85440+ 1 5.93180+ 0 7.87260+ 1 4.83890+ 0 7.89070+ 1 4.24000+ 01261 3  1 3273
7.90880+ 1 2.50940+ 0 7.92700+ 1 1.31000+ 0 7.94510+ 1-1.12550+ 01261 3  1 3274
7.98140+ 1-1.11300+ 0 7.99950+ 1 1.18030+ 0 8.00680+ 1 1.11200+ 01261 3  1 3275
8.02490+ 1-1.52470+ 0 8.04670+ 1-1.43340+ 0 8.07570+ 1 1.78020+ 01261 3  1 3276
8.08660+ 1 1.86650+ 0 8.09750+ 1 1.24290+ 0 8.10840+ 1 3.96500- 11261 3  1 3277
8.13380+ 1-1.68580+ 0 8.15190+ 1-1.65140+ 0 8.16280+ 1-1.49200+ 01261 3  1 3278
8.18100+ 1 5.80600- 1 8.20000+ 1 1.04000+ 0 8.20000+ 1 0.00000+ 01261 3  1 3279
1.00000+ 3 0.00000+ 0 1.30000+ 4 0.00000+ 0 1.30550+ 4 0.00000+ 01261 3  1 3280
1.30553+ 4 3.24728- 7 1.32000+ 4 1.56990- 4 1.37000+ 4 6.98340- 41261 3  1 3281
1.42000+ 4 1.23970- 3 1.45000+ 4 1.56450- 3 1.47000+ 4 1.78100- 31261 3  1 3282
1.48000+ 4 1.88930- 3 1.50000+ 4 2.10580- 3 1.52000+ 4 2.32240- 31261 3  1 3283
1.55000+ 4 2.64720- 3 1.58000+ 4 2.97200- 3 1.61000+ 4 3.29680- 31261 3  1 3284
1.63000+ 4 3.51330- 3 1.72000+ 4 4.48780- 3 1.76000+ 4 4.92080- 31261 3  1 3285
    (MISSING LINES)

6.60000+ 5 7.56674+ 0 6.70000+ 5 7.53305+ 0 6.80000+ 5 7.50044+ 01261 3  1 3356
7.00000+ 5 7.43829+ 0 7.20000+ 5 7.38002+ 0 7.30000+ 5 7.35225+ 01261 3  1 3357
7.40000+ 5 7.32534+ 0 7.50000+ 5 7.29927+ 0 7.60000+ 5 7.27399+ 01261 3  1 3358
7.70000+ 5 7.24947+ 0 7.75000+ 5 7.23748+ 0 7.80000+ 5 7.22568+ 01261 3  1 3359
8.00000+ 5 7.18012+ 0 8.20000+ 5 7.13704+ 0 8.40000+ 5 7.09617+ 01261 3  1 3360
8.50000+ 5 7.07647+ 0 8.60000+ 5 7.03726+ 0 8.80000+ 5 7.02033+ 01261 3  1 3361
9.00000+ 5 6.98542+ 0 9.20000+ 5 6.95259+ 0 9.40000+ 5 6.92188+ 01261 3  1 3362
9.50000+ 5 6.90733+ 0 9.60000+ 5 6.89334+ 0 9.70000+ 5 6.87990+ 01261 3  1 3363
9.80000+ 5 6.86703+ 0 1.00000+ 6 6.84300+ 0 1.05000+ 6 6.79287+ 01261 3  1 3364
1.10000+ 6 6.75603+ 0 1.15000+ 6 6.73118+ 0 1.20000+ 6 6.71709+ 01261 3  1 3365

    (MISSING LINES)

1.00000+ 7 5.81839+ 0 1.05000+ 7 5.77124+ 0 1.10000+ 7 5.74233+ 01261 3  1 3397
1.15000+ 7 5.73068+ 0 1.19000+ 7 5.73287+ 0 1.20000+ 7 5.73481+ 01261 3  1 3398
1.21988+ 7 5.74215+ 0 1.25000+ 7 5.75138+ 0 1.30000+ 7 5.77654+ 01261 3  1 3399
1.35000+ 7 5.80645+ 0 1.40000+ 7 5.84000+ 0 1.45000+ 7 5.88900+ 01261 3  1 3400
1.50000+ 7 5.93400+ 0 1.55000+ 7 5.96800+ 0 1.60000+ 7 5.99800+ 01261 3  1 3401
1.65000+ 7 6.02200+ 0 1.70000+ 7 6.04700+ 0 1.75000+ 7 6.06700+ 01261 3  1 3402
1.80000+ 7 6.08700+ 0 1.85000+ 7 6.10500+ 0 1.90000+ 7 6.12100+ 01261 3  1 3403
1.95000+ 7 6.13600+ 0 2.00000+ 7 6.14953+ 0                     1261 3  1 3404
                                                                1261 3  0 3405
```

For a reaction having a threshold, the threshold energy E_{th} is given by

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

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

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

File 4 Angular Distributions of Secondary Neutrons (Legendre Coefficients and Transformation)

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 LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NL is the highest order Legendre polynomial that is given at each energy ($NL \leq 20$).

NK is the number of elements in the transformation matrix ($NK \leq 441$). $NK = (NM + 1)^2$.

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system. NM should be an even number.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons (Legendre Coefficients and Transformation)

Formats

File 4 is divided into sections, each containing data for a particular reaction type (MT number) and ordered by increasing MT number. Each section always starts with a HEAD record and ends with a SEND record. If the section contains a description of the angular distributions for elastic scattering, the transformation matrix is given first (if present) and this is followed by the representation of the angular distributions.

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

When LTT = 1 (angular distributions given in terms of Legendre polynomial coefficients) and LVT = 1, the structure of a section is

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0]HEAD LTT = 1, LVT = 1

[MAT, 4, MT/0.0, AWR, 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 °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified.

File 4 Angular Distributions of Secondary Neutrons (Legendre Coefficients and Transformation)

```
 9.22350+ 4 2.33025+ 2          1          1          0          01261 4  2 5440
 0.00000+ 0 2.33025+ 2          0          2        261         181261 4  2 5441
 1.00000+ 0 2.86092- 3 3.67731- 6-9.43608-11 0.00000+ 0 0.00000+ 01261 4  2 5442
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5443
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5444
 0.00000+ 0 0.00000+ 0 9.99989- 1 5.14963- 3 1.23972- 5 1.47746- 81261 4  2 5445
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5446
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5447
 0.00000+ 0 0.00000+ 0 0.00000+ 0-2.86088- 3 9.99971- 1 7.35655- 31261 4  2 5448
 2.62432- 5 5.43588- 8-1.63336- 8 1.46884-10 0.00000+ 0 0.00000+ 01261 4  2 5449
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5450
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 1.10494- 5-5.14951- 31261 4  2 5451
 9.99943- 1 9.53614- 3 4.45541- 5 1.28537- 7-5.71550- 8-2.66532-101261 4  2 5452
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5453

    (MISSING LINES)

 1.79937-18-1.25728-15 4.20810-13-1.03262-10 2.04013- 8-3.14407- 61261 4  2 5484
 3.57139- 4-2.67670- 2 9.99161- 1 3.10625- 2 4.81446- 4 4.93607- 61261 4  2 5485
 2.51865- 8-1.46454- 9 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5486
 0.00000+ 0 0.00000+ 0 3.70754-18-2.39212-15 6.89312-13-1.54348-101261 4  2 5487
 2.79849- 8-3.97363- 6 4.16968- 4-2.89131- 2 9.99032- 1 3.32076- 21261 4  2 5488
 5.50595- 4 6.04947- 6-2.18919- 8 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5489
 0.00000+ 0 0.00000+ 0 0.00000+ 0-1.29728-20 6.85286-18-4.21907-151261 4  2 5490
 1.08489-12-2.23626-10 3.74868- 8-4.93644- 6 4.81391- 4-3.10584- 21261 4  2 5491
 9.98894- 1 3.53519- 2 6.24137- 4 7.30598- 6 0.00000+ 0 0.00000+ 01261 4  2 5492
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0-2.40276-201261 4  2 5493
 1.18255-17-7.05278-13 1.65099-12-3.15518-10 4.92034- 8-6.04237- 61261 4  2 5494
 5.50408- 4-3.32028- 2 9.98747- 1 3.74955- 2 7.02335- 4 0.00000+ 01261 4  2 5495
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5496
 0.00000+ 0-4.14653-20 1.94174-17-1.13112-14 2.44124-12-4.35107-101261 4  2 5497
 6.34525- 8-7.30127- 6 6.24016- 4-3.53465- 2 9.98590- 1 3.96384- 21261 4  2 5498
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01261 4  2 5499
 0.00000+ 0 0.00000+ 0 0.00000+ 0-6.80077-20 3.06691-17-1.75392-141261 4  2 5500
 3.52090-12-5.88180-10 8.05727- 8-8.72297- 6 7.02215- 4-3.74894- 21261 4  2 5501
 9.98425- 1                                                         1261 4  2 5502
 0.00000+ 0 0.00000+ 0          0          0          1         211261 4  2 5503
        21          2                                                1261 4  2 5504
 0.00000+ 0 1.00000- 5          0          0          1          01261 4  2 5505
 0.00000+ 0                                                         1261 4  2 5506
 0.00000+ 0 1.00000+ 4          0          0          1          01261 4  2 5507
 0.00000+ 0                                                         1261 4  2 5508
 0.00000+ 0 2.00000+ 5          0          0          6          01261 4  2 5509
 2.34000- 1 6.07000- 2 5.50000- 3 3.37000- 3 3.32000- 3-7.70000- 41261 4  2 5510
 0.00000+ 0 5.00000+ 5          0          0          6          01261 4  2 5511
 3.34000- 1 1.24000- 1 4.11000- 2 1.61000- 2 4.20000- 3-3.85000- 31261 4  2 5512
 0.00000+ 0 8.00000+ 5          0          0         10          01261 4  2 5513
 3.97500- 1 1.91100- 1 1.25000- 1 4.47000- 2 6.10000- 3-1.13100- 21261 4  2 5514
-5.11000- 3 3.90000- 4-9.80000- 4-9.00000- 5                        1261 4  2 5515
 0.00000+ 0 1.00000+ 6          0          0         10          01261 4  2 5516
 4.35500- 1 2.35400- 1 1.78600- 1 7.51100- 2 1.12000- 2-1.36200- 21261 4  2 5517
-9.00000- 3 2.21000- 3-4.16000- 3-2.40000- 4                        1261 4  2 5518

    (MISSING LINES)

 0.00000+ 0 1.40000+ 7          0          0         18          01261 4  2 5561
 8.84700- 1 7.80000- 1 6.89300- 1 6.11300- 1 5.42400- 1 4.78600- 11261 4  2 5562
 4.19500- 1 3.63500- 1 3.08400- 1 2.53700- 1 1.97500- 1 1.42100- 11261 4  2 5563
 9.40000- 2 5.65000- 2 2.97400- 2 1.25200- 2 3.31000- 3 2.70000- 41261 4  2 5564
 0.00000+ 0 1.70000+ 7          0          0         20          01261 4  2 5565
 9.07600- 1 8.16900- 1 7.29500- 1 6.56300- 1 5.87000- 1 5.23900- 11261 4  2 5566
 4.65400- 1 4.10200- 1 3.59100- 1 3.14000- 1 2.63600- 1 2.07600- 11261 4  2 5567
 1.54400- 1 1.04100- 1 6.27200- 2 3.35100- 2 1.48700- 2 6.24000- 31261 4  2 5568
 1.87000- 3 6.10000- 4                                              1261 4  2 5569
 0.00000+ 0 2.00000+ 7          0          0         20          01261 4  2 5570
 9.45000- 1 8.75000- 1 7.96800- 1 7.18400- 1 6.41100- 1 5.71300- 11261 4  2 5571
 5.07600- 1 4.51000- 1 3.97500- 1 3.46700- 1 2.95800- 1 2.44200- 11261 4  2 5572
 1.90900- 1 1.38200- 1 9.26600- 2 5.78700- 2 3.30500- 2 1.63200- 21261 4  2 5573
 7.27000- 3 2.54000- 3                                              1261 4  2 5574
                                                                    1261 4  0 5575
```

File 4 Angular Distributions of Secondary Neutrons (Legendre Coefficients and Transformation)

If the angular distributions are represented as Legendre polynomial coefficients, the absolute differential cross sections are obtained by

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

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

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;

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

ℓ = order of the Legendre polynomial;

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

f_ℓ = the ℓth Legendre polynomial coefficient and it is understood that $\underline{f_o = 1.0}$.

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

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

File 4 Angular Distributions of Secondary Neutrons (Only Legendre Coefficients given)

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 LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NL is the highest order Legendre polynomial that is given at each energy ($NL \leq 20$).

NM is the maximum order Legendre polynomial that will be required ($NM \leq 20$) to describe the angular distributions of elastic scattering in either the center-of-mass or the laboratory system. NM should be an even number.

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons (Only Legendre Coefficients given)

Legendre Polynomial Coefficients Given and the Transformation Matrix Not Given: LTT = 1 and LVT = 0

File 4 is divided into sections, each containing data for a particular reaction type (MT number) and ordered by increasing MT number. Each section always starts with a HEAD record and ends with a SEND record. If the section contains a description of the angular distributions for elastic scattering, the transformation matrix is given first (if present) and this is followed by the representation of the angular distributions.

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0,0]HEAD

[MAT, 4, MT/0.0, AWR, 0, LCT, 0, 0] CØNT.

[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 °K) and a test for temperature dependence, respectively. These values are normally zero; however, see Appendix F for an explanation of cases in which temperature dependence is specified.

File 4 Angular Distributions of Secondary Neutrons
(Only Legendre Coefficients given)

```
 6.01200+ 3 1.18969+ 1          0          1          0          01274 4 51 1277
 0.00000+ 0 1.18970+ 1          0          2          0          01274 4 51 1278
 0.00000+ 0 0.00000+ 0          0          0          1         731274 4 51 1279
          73          2                                            1274 4 51 1280
 0.00000+ 0 4.80000+ 6          0          0          1          01274 4 51 1281
 0.00000+ 0                                                        1274 4 51 1282
 0.00000+ 0 5.25000+ 6          0          0          2          01274 4 51 1283
 1.50000- 2 2.00000- 3                                             1274 4 51 1284
 0.00000+ 0 5.37000+ 6          0          0          2          01274 4 51 1285
 3.35000- 2 1.20000- 2                                             1274 4 51 1286
 0.00000+ 0 5.47000+ 6          0          0          2          01274 4 51 1287
 8.25000- 2 4.10000- 2                                             1274 4 51 1288
 0.00000+ 0 5.59000+ 6          0          0          2          01274 4 51 1289
 1.41000- 1 2.28400- 2                                             1274 4 51 1290
 0.00000+ 0 5.67000+ 6          0          0          2          01274 4 51 1291
 1.50000- 1 7.68000- 3                                             1274 4 51 1292
 0.00000+ 0 5.73000+ 6          0          0          2          01274 4 51 1293
 1.41000- 1-2.00000- 3                                             1274 4 51 1294
 0.00000+ 0 5.83000+ 6          0          0          1          01274 4 51 1295
 1.09240- 1                                                        1274 4 51 1296
 0.00000+ 0 5.90000+ 6          0          0          2          01274 4 51 1297
 8.70000- 2 1.00000- 2                                             1274 4 51 1298
 0.00000+ 0 6.04000+ 6          0          0          2          01274 4 51 1299
 7.60000- 2 5.26000- 2                                             1274 4 51 1300
 0.00000+ 0 6.10000+ 6          0          0          2          01274 4 51 1301
 8.80000- 2 7.45000- 2                                             1274 4 51 1302
 0.00000+ 0 6.22000+ 6          0          0          3          01274 4 51 1303
 1.48000- 1 1.15670- 1-5.50000- 3                                  1274 4 51 1304
 0.00000+ 0 6.24000+ 6          0          0          3          01274 4 51 1305
 1.44000- 1 1.18000- 1-8.00000- 3                                  1274 4 51 1306
 0.00000+ 0 6.26000+ 6          0          0          3          01274 4 51 1307
 6.08000- 2 1.12000- 1-1.55600- 2                                  1274 4 51 1308
 0.00000+ 0 6.29000+ 6          0          0          3          01274 4 51 1309
-6.40000- 2 7.67500- 2-2.68900- 2                                  1274 4 51 1310
 0.00000+ 0 6.33000+ 6          0          0          4          01274 4 51 1311
-6.10000- 2 2.97500- 2-4.20000- 2-1.00000- 3                       1274 4 51 1312
 0.00000+ 0 6.34000+ 6          0          0          4          01274 4 51 1313
-5.40000- 2 1.80000- 2-4.30000- 2-1.33000- 3                       1274 4 51 1314
 0.00000+ 0 6.37000+ 6          0          0          4          01274 4 51 1315
-3.30000- 2 1.70000- 2-4.20000- 2-3.63000- 3                       1274 4 51 1316
 0.00000+ 0 6.38000+ 6          0          0          4          01274 4 51 1317
-2.60000- 2 1.80000- 2-3.10000- 2-5.26000- 3                       1274 4 51 1318

     (MISSING LINES)

 0.00000+ 0 8.48000+ 6          0          0          5          01274 4 51 1399
 9.08200- 2 1.56000- 1-1.40200- 2 2.71000- 3 2.61800- 2            1274 4 51 1400
 0.00000+ 0 8.55000+ 6          0          0          5          01274 4 51 1401
 1.04000- 1 1.70000- 1 4.82000- 3 1.21900- 2 2.46100- 2            1274 4 51 1402
 0.00000+ 0 8.67500+ 6          0          0          5          01274 4 51 1403
 1.03550- 1 1.85120- 1 3.40000- 2 2.91200- 2 2.18100- 2            1274 4 51 1404
 0.00000+ 0 8.72810+ 6          0          0          5          01274 4 51 1405
 1.03360- 1 1.82410- 1 2.97400- 2 2.83900- 2 2.06100- 2            1274 4 51 1406
 0.00000+ 0 8.80000+ 6          0          0          5          01274 4 51 1407
 5.49300- 2 1.78750- 1 2.39700- 2 2.31300- 2 1.90000- 2            1274 4 51 1408
 0.00000+ 0 8.85000+ 6          0          0          4          01274 4 51 1409
 2.12600- 2 1.76200- 1 1.99600- 2 1.94700- 2                       1274 4 51 1410
 0.00000+ 0 9.04660+ 6          0          0          4          01274 4 51 1411
-1.11160- 1 1.66180- 1 4.19000- 3 5.09000- 3                       1274 4 51 1412
 0.00000+ 0 9.09210+ 6          0          0          4          01274 4 51 1413
-9.22300- 2 1.63860- 1 5.40000- 4 3.99000- 3                       1274 4 51 1414
 0.00000+ 0 9.27150+ 6          0          0          4          01274 4 51 1415
-1.75900- 2 1.54710- 1 3.81000- 3 8.36000- 3                       1274 4 51 1416
 0.00000+ 0 9.91120+ 6          0          0          4          01274 4 51 1417
 2.48540- 1 1.54330- 1 1.54500- 2 2.39300- 2                       1274 4 51 1418
 0.00000+ 0 9.99460+ 6          0          0          4          01274 4 51 1419
 2.35330- 1 1.54550- 1 1.69700- 2 2.59700- 2                       1274 4 51 1420
 0.00000+ 0 1.09490+ 7          0          0          4          01274 4 51 1421
 9.18700- 2 2.02380- 1 3.43400- 2 2.70900- 2                       1274 4 51 1422
 0.00000+ 0 1.50000+ 7          0          0          4          01274 4 51 1423
 1.64180- 1 2.08310- 1 2.01900- 2 2.50000- 4                       1274 4 51 1424
 0.00000+ 0 2.00000+ 7          0          0          4          01274 4 51 1425
 1.64180- 1 2.08310- 1 2.01900- 2 2.50000- 4                       1274 4 51 1426
                                                                   1274 4  0 1427
```

File 4 Angular Distributions of Secondary Neutrons (Only Legendre Coefficients given)

If the angular distributions are represented as Legendre polynomial coefficients, the absolute differential cross sections are obtained by

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

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

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;

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

ℓ = order of the Legendre polynomial;

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

f_ℓ = the ℓth Legendre polynomial coefficient and it is understood that $\underline{f_o = 1.0}$.

File 4 Angular Distributions of Secondary Neutrons
(Tabulated Distribution and Transformation)

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 LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NK is the number of elements in the transformation matrix ($NK \leq 441$).

$NK = (NM + 1)^2$.

V_K are the matrix elements of the transformation matrices:

$V_K = U^{-1}_{\ell,m}$ if LCT = 1 (data given in LAB system); and

$V_K = U_{\ell,m}$ if LCT = 2 (data given in CM system).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy ($NP \leq 101$).

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons
(Tabulated Distribution and Transformation)

<u>Tabulated Probability Distributions and Transformation Matrix Given:</u>

<u>LTT = 2 and LVT = 1</u>

If the angular distributions are given as tabulated probability distributions, LTT = 2, and a transformation matrix is given for elastic scattering, the structure of a section is

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0]HEAD LVT = 1, LTT = 2

[MAT, 4, MT/0.0, AWR, 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 , 0]SEND

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

File 4 Angular Distributions of Secondary Neutrons
(Tabulated Distribution and Transformation)

```
 4.20000+ 4 9.51160+ 1          1          2          0          01287 4  2  377
 0.00000+ 0 9.51160+ 1          0          2        361         181287 4  2  378
 1.00000+ 0 7.00899- 3 2.21008- 5-1.38100-10 0.00000+ 0 0.00000+ 01287 4  2  379
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  380
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  381
 0.00000+ 0 0.00000+ 0 9.99934- 1 1.26158- 2 7.57680- 5 2.20939- 71287 4  2  382
-1.25663- 8-1.30840-10 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  383
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  384
 0.00000+ 0 0.00000+ 0 0.00000+ 0-7.00832- 3 9.99826- 1 1.80216- 21287 4  2  385
 1.57852- 4 8.04087- 7-2.70852- 8 2.26087-10 0.00000+ 0 0.00000+ 01287 4  2  386
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  387
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 6.63131- 5-1.26140- 21287 4  2  388
 9.99661- 1 2.33593- 2 2.67867- 4 1.89548- 6-6.55483- 8-1.27816- 91287 4  2  389
-6.44832-10 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  390
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0-6.63981- 71287 4  2  391
 1.51563- 4-1.80180- 2 9.99440- 1 2.86651- 2 4.05615- 4 3.63806- 61287 4  2  392

     (MISSING LINES)

 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 6.22845-19-1.34061-161287 4  2  430
 3.42066-14-3.81600-12 3.56704-10-2.78194- 8 1.77040- 6-8.87132- 51287 4  2  431
 3.29671- 3-8.10889- 2 9.92491- 1 9.15725- 2 4.20749- 3 0.00000+ 01287 4  2  432
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  433
 1.16935-18-3.67476-16 5.90580-14-6.07180-12 5.27386-10-3.83587- 81287 4  2  434
 2.28274- 6-1.07175- 4 3.73662- 3-8.62904- 2 9.91555- 1 9.67680- 21287 4  2  435
 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01287 4  2  436
 0.00000+ 0-1.05340-20 2.77279-18-7.61059-16 9.80913-14-9.36964-121287 4  2  437
 7.60542-10-5.18463- 8 2.89816- 6-1.28018- 4 4.20371- 3-9.14838- 21287 4  2  438
 9.90565- 1                                                      1287 4  2  439
 0.00000+ 0 0.00000+ 0          0          0          1         131287 4  2  440
        13          2                                             1287 4  2  441
 0.00000+ 0 1.00000- 5          0          0          1          21287 4  2  442
         2          2                                             1287 4  2  443
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                      1287 4  2  444
 0.00000+ 0 5.00000+ 4          0          0          1          21287 4  2  445
         2          2                                             1287 4  2  446
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                      1287 4  2  447
 0.00000+ 0 6.00000+ 4          0          0          1          21287 4  2  448
         2          2                                             1287 4  2  449
-1.00000+ 0 4.37000- 1 1.00000+ 0 5.63000- 1                      1287 4  2  450
 0.00000+ 0 1.50000+ 5          0          0          1          41287 4  2  451
         4          2                                             1287 4  2  452
-1.00000+ 0 4.71950- 1-6.00000- 1 4.26960- 1 0.00000+ 0 4.58950- 11287 4  2  453
 1.00000+ 0 6.49940- 1                                           1287 4  2  454
 0.00000+ 0 2.50000+ 5          0          0          1          61287 4  2  455
         6          2                                             1287 4  2  456
-1.00000+ 0 3.77080- 1-8.00000- 1 3.64070- 1-6.00000- 1 3.77080- 11287 4  2  457
-2.00000- 1 4.22080- 1 4.00000- 1 5.52110- 1 1.00000+ 0 7.80160- 11287 4  2  458

     (MISSING LINES)

 0.00000+ 0 2.00000+ 7          0          0          1         511287 4  2  532
        51          2                                             1287 4  2  533
-1.00000+ 0 5.64300- 3-9.90000- 1 1.93490- 2-9.80000- 1 2.71650- 21287 4  2  534
-9.70000- 1 3.07910- 2-9.60000- 1 3.15580- 2-9.40000- 1 2.83630- 21287 4  2  535
-9.00000- 1 1.83600- 2-8.80000- 1 1.54340- 2-8.60000- 1 1.44040- 21287 4  2  536
-8.30000- 1 1.56140- 2-7.30000- 1 2.48910- 2-6.60000- 1 2.66540- 21287 4  2  537
-4.10000- 1 2.40970- 2-3.50000- 1 2.55460- 2-2.90000- 1 3.10100- 21287 4  2  538
-2.40000- 1 3.92430- 2-1.20000- 1 6.43630- 2-5.00000- 2 7.10590- 21287 4  2  539
 1.00000- 2 6.67570- 2 1.30000- 1 4.22040- 2 1.80000- 1 3.82660- 21287 4  2  540
 2.20000- 1 4.31470- 2 2.60000- 1 5.65390- 2 4.00000- 1 1.40650- 11287 4  2  541
 4.60000- 1 1.57040- 1 5.20000- 1 1.39620- 1 6.20000- 1 6.16880- 21287 4  2  542
 6.50000- 1 4.30480- 2 6.80000- 1 3.24810- 2 7.10000- 1 2.97430- 21287 4  2  543
 7.70000- 1 3.44760- 2 7.90000- 1 3.91220- 2 8.00000- 1 4.46310- 21287 4  2  544
 8.10000- 1 5.45780- 2 8.20000- 1 7.17470- 2 8.30000- 1 1.00090- 11287 4  2  545
 8.40000- 1 1.45030- 1 8.50000- 1 2.13900- 1 8.60000- 1 3.16370- 11287 4  2  546
 8.70000- 1 4.65100- 1 8.80000- 1 6.76400- 1 8.90000- 1 9.71100- 11287 4  2  547
 9.00000- 1 1.37550+ 0 9.10000- 1 1.92270+ 0 9.20000- 1 2.65370+ 01287 4  2  548
 9.30000- 1 3.61960+ 0 9.40000- 1 4.88240+ 0 9.50000- 1 6.51960+ 01287 4  2  549
 9.70000- 1 1.13090+ 1 9.90000- 1 1.89960+ 1 1.00000+ 0 2.43600+ 11287 4  2  550
                                                                  1287 4  0  551
```

File 4 Angular Distributions of Secondary Neutrons (Tabulated Distribution and Transformation)

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

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

where $p(\mu,E)d\mu$ is the probability that a neutron of incident energy E will be scattered into the interval $d\mu$ about an angle whose cosine is μ.

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.

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

and

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

File 4 Angular Distributions of Secondary Neutrons (Only Tabulated Distributions given)

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 LAB system;

LCT = 2, the data are given in the CM system.

LVT is a flag to specify whether a transformation matrix is given for elastic scattering:

LVT = 0, a transformation matrix is not given (always use this value for all non-elastic scattering reactions);

LVT = 1, a transformation matrix is given.

NE is the number of incident energy points at which angular distributions are given ($NE \leq 500$).

NP is the number of angular points (cosines) used to give the tabulated probability distributions for each energy ($NP \leq 101$).

Other commonly used variables are given in the Glossary (Appendix A).

File 4 Angular Distributions of Secondary Neutrons
(Only Tabulated Distributions given)

Tabulated Probability Distributions Given and Transformation Matrix

Not Given: LTT = 2 and LVT = 0

If the angular distributions are given as tabulated probability distributions, LTT = 2, the structure of a section is

[MAT, 4, MT/ZA, AWR, LVT, LTT, 0, 0]HEAD

[MAT, 4, MT/0.0, AWR, 0, LCT, 0, 0]CØNT.

[MAT, 4, MT/0.0, 0.0, 0 , 0 , NR, NE/E_{int}]TAB2

[MAT, 4, MT/T , E_1 , LT, 0, NR, NP/μ_{int}/p(μ,E_1)]TAB1

[MAT, 4, MT/T , E_2 , LT, 0, NR, NP/μ_{int}/p(μ,E_2)]TAB1

[MAT, 4, MT/T , E_{NE}, LT, 0 , NR, NP/μ_{int}/p(μ,E_{NE})]TAB1

[MAT, 4, 0 /0.0, 0.0, 0 , 0 , 0 , 0]SEND

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

File 4 Angular Distributions of Secondary Neutrons
(Only Tabulated Distributions given)

```
 1.00100+ 3 9.99170- 1          0          2          0          01269 4  2  306
 0.00000+ 0 0.00000+ 0          0          2          0          01269 4  2  307
 0.00000+ 0 0.00000+ 0          0          0          1         141269 4  2  308
        14          2                                               1269 4  2  309
 0.00000+ 0 1.00000- 5          0          0          1         111269 4  2  310
        11          2                                               1269 4  2  311
-1.00000+ 0 5.00000- 1-8.00000- 1 5.00000- 1-6.00000- 1 5.00000- 11269 4  2  312
-4.00000- 1 5.00000- 1-2.00000- 1 5.00000- 1 0.00000+ 0 5.00000- 11269 4  2  313
 2.00000- 1 5.00000- 1 4.00000- 1 5.00000- 1 6.00000- 1 5.00000- 11269 4  2  314
 8.00000- 1 5.00000- 1 1.00000+ 0 5.00000- 1                      1269 4  2  315
 0.00000+ 0 1.00000+ 5          0          0          1         111269 4  2  316
        11          2                                               1269 4  2  317
-1.00000+ 0 5.00000- 1-8.00000- 1 5.00000- 1-6.00000- 1 5.00000- 11269 4  2  318
-4.00000- 1 5.00000- 1-2.00000- 1 5.00000- 1 0.00000+ 0 5.00000- 11269 4  2  319
 2.00000- 1 5.00000- 1 4.00000- 1 5.00000- 1 6.00000- 1 5.00000- 11269 4  2  320
 8.00000- 1 5.00000- 1 1.00000+ 0 5.00000- 1                      1269 4  2  321
 0.00000+ 0 5.00000+ 5          0          0          1         111269 4  2  322
        11          2                                               1269 4  2  323
-1.00000+ 0 5.01170- 1-8.00000- 1 5.00970- 1-6.00000- 1 5.00660- 11269 4  2  324
-4.00000- 1 5.00450- 1-2.00000- 1 5.00250- 1 0.00000+ 0 5.00050- 11269 4  2  325
 2.00000- 1 4.99740- 1 4.00000- 1 4.99530- 1 6.00000- 1 4.99330- 11269 4  2  326
 8.00000- 1 4.99020- 1 1.00000+ 0 4.98820- 1                      1269 4  2  327
 0.00000+ 0 1.00000+ 6          0          0          1         111269 4  2  328
        11          2                                               1269 4  2  329
-1.00000+ 0 5.02180- 1-8.00000- 1 5.01760- 1-6.00000- 1 5.01310- 11269 4  2  330
-4.00000- 1 5.00890- 1-2.00000- 1 5.00440- 1 0.00000+ 0 5.00000- 11269 4  2  331
 2.00000- 1 4.99570- 1 4.00000- 1 4.99130- 1 6.00000- 1 4.98690- 11269 4  2  332
 8.00000- 1 4.98230- 1 1.00000+ 0 4.97790- 1                      1269 4  2  333
 0.00000+ 0 2.00000+ 6          0          0          1         111269 4  2  334
        11          2                                               1269 4  2  335
-1.00000+ 0 5.03870- 1-8.00000- 1 5.03110- 1-6.00000- 1 5.02340- 11269 4  2  336
-4.00000- 1 5.01580- 1-2.00000- 1 5.00810- 1 0.00000+ 0 5.00030- 11269 4  2  337
 2.00000- 1 4.99250- 1 4.00000- 1 4.98460- 1 6.00000- 1 4.97640- 11269 4  2  338
 8.00000- 1 4.96820- 1 1.00000+ 0 4.95980- 1                      1269 4  2  339
 0.00000+ 0 4.00000+ 6          0          0          1         111269 4  2  340
        11          2                                               1269 4  2  341
-1.00000+ 0 5.06770- 1-8.00000- 1 5.05420- 1-6.00000- 1 5.04070- 11269 4  2  342
-4.00000- 1 5.02750- 1-2.00000- 1 5.01430- 1 0.00000+ 0 5.00110- 11269 4  2  343
 2.00000- 1 4.98730- 1 4.00000- 1 4.97340- 1 6.00000- 1 4.95890- 11269 4  2  344
 8.00000- 1 4.94410- 1 1.00000+ 0 4.92830- 1                      1269 4  2  345
 0.00000+ 0 6.00000+ 6          0          0          1         111269 4  2  346
        11          2                                               1269 4  2  347
-1.00000+ 0 5.09990- 1-8.00000- 1 5.07700- 1-6.00000- 1 5.05600- 11269 4  2  348
-4.00000- 1 5.03620- 1-2.00000- 1 5.01770- 1 0.00000+ 0 4.99930- 11269 4  2  349
 2.00000- 1 4.98080- 1 4.00000- 1 4.96280- 1 6.00000- 1 4.94390- 11269 4  2  350
 8.00000- 1 4.92460- 1 1.00000+ 0 4.90400- 1                      1269 4  2  351
 0.00000+ 0 8.00000+ 6          0          0          1         111269 4  2  352
        11          2                                               1269 4  2  353
-1.00000+ 0 5.12880- 1-8.00000- 1 5.09520- 1-6.00000- 1 5.06760- 11269 4  2  354
-4.00000- 1 5.04340- 1-2.00000- 1 5.02070- 1 0.00000+ 0 4.99870- 11269 4  2  355
 2.00000- 1 4.97660- 1 4.00000- 1 4.95460- 1 6.00000- 1 4.93140- 11269 4  2  356
 8.00000- 1 4.90770- 1 1.00000+ 0 4.88180- 1                      1269 4  2  357
 0.00000+ 0 1.00000+ 7          0          0          1         111269 4  2  358
        11          2                                               1269 4  2  359
-1.00000+ 0 5.17270- 1-8.00000- 1 5.12010- 1-6.00000- 1 5.07940- 11269 4  2  360
-4.00000- 1 5.04610- 1-2.00000- 1 5.01680- 1 0.00000+ 0 4.99080- 11269 4  2  361
 2.00000- 1 4.96690- 1 4.00000- 1 4.94350- 1 6.00000- 1 4.92020- 11269 4  2  362
 8.00000- 1 4.89690- 1 1.00000+ 0 4.87160- 1                      1269 4  2  363
 0.00000+ 0 1.20000+ 7          0          0          1         121269 4  2  364
        12          2                                               1269 4  2  365
-1.00000+ 0 5.22720- 1-9.00000- 1 5.18250- 1-8.00000- 1 5.14640- 11269 4  2  366
-6.00000- 1 5.08980- 1-4.00000- 1 5.04750- 1-2.00000- 1 5.01290- 11269 4  2  367
 0.00000+ 0 4.98230- 1 2.00000- 1 4.95560- 1 4.00000- 1 4.93130- 11269 4  2  368
 6.00000- 1 4.90850- 1 8.00000- 1 4.88660- 1 1.00000+ 0 4.86540- 11269 4  2  369
 0.00000+ 0 1.40000+ 7          0          0          1         121269 4  2  370
        12          2                                               1269 4  2  371
-1.00000+ 0 5.28230- 1-9.00000- 1 5.21880- 1-8.00000- 1 5.16980- 11269 4  2  372
-6.00000- 1 5.09820- 1-4.00000- 1 5.04560- 1-2.00000- 1 5.00480- 11269 4  2  373
 0.00000+ 0 4.97030- 1 2.00000- 1 4.94310- 1 4.00000- 1 4.91950- 11269 4  2  374
 6.00000- 1 4.90050- 1 8.00000- 1 4.88330- 1 1.00000+ 0 4.86880- 11269 4  2  375
 0.00000+ 0 1.60000+ 7          0          0          1         121269 4  2  376
        12          2                                               1269 4  2  377
-1.00000+ 0 5.34330- 1-9.00000- 1 5.25750- 1-8.00000- 1 5.19240- 11269 4  2  378
-6.00000- 1 5.10240- 1-4.00000- 1 5.04040- 1-2.00000- 1 4.99390- 11269 4  2  379
 0.00000+ 0 4.95670- 1 2.00000- 1 4.92880- 1 4.00000- 1 4.90810- 11269 4  2  380
 6.00000- 1 4.89360- 1 8.00000- 1 4.88530- 1 1.00000+ 0 4.88330- 11269 4  2  381
 0.00000+ 0 1.80000+ 7          0          0          1         121269 4  2  382
        12          2                                               1269 4  2  383
-1.00000+ 0 5.40920- 1-9.00000- 1 5.29620- 1-8.00000- 1 5.21340- 11269 4  2  384
-6.00000- 1 5.10380- 1-4.00000- 1 5.03270- 1-2.00000- 1 4.98020- 11269 4  2  385
 0.00000+ 0 4.94060- 1 2.00000- 1 4.91260- 1 4.00000- 1 4.89630- 11269 4  2  386
 6.00000- 1 4.89050- 1 8.00000- 1 4.89400- 1 1.00000+ 0 4.90910- 11269 4  2  387
 0.00000+ 0 2.00000+ 7          0          0          1         121269 4  2  388
        12          2                                               1269 4  2  389
-1.00000+ 0 5.48070- 1-9.00000- 1 5.33480- 1-8.00000- 1 5.23320- 11269 4  2  390
-6.00000- 1 5.10290- 1-4.00000- 1 5.02210- 1-2.00000- 1 4.96350- 11269 4  2  391
 0.00000+ 0 4.92180- 1 2.00000- 1 4.89580- 1 4.00000- 1 4.88400- 11269 4  2  392
 6.00000- 1 4.88660- 1 8.00000- 1 4.90750- 1 1.00000+ 0 4.94660- 11269 4  2  393
                                                                  1269 4  0  394
```

File 4 Angular Distributions of Secondary Neutrons (Only Tabulated Distributions given)

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

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

where $p(\mu,E)d\mu$ is the probability that a neutron of incident energy E will be scattered into the interval $d\mu$ about an angle whose cosine is μ.

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

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

where $\sigma_s(E)$ is given in File 3 (for the same MT number) and $p(\mu,E)$ is given in File 4.

File 5 Energy Distributions of Secondary Neutrons (General Structure)

The following quantities are defined

NK is the number of partial energy distributions. There will be one subsection for each partial distribution.

U is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ is a parameter used to describe the secondary energy distribution. 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.

If LF = 5, 7, or 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 in Section 5.1.).

$p_k(E_N)$ is the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ is the kth partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

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

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

NE is the number of incident energy points at which tabulated distributions are given. Also the number of points at which θ(E) is given. ($NE \leq 200$.)

NF is the number of secondary energy points in a tabulation. ($NF \leq 1000$.)

File 5 Energy Distributions of Secondary Neutrons (General Structure)

The structure of a section has the following form:

[MAT, 5, MT/ZA, AWR, 0, 0, NK, 0]HEAD

<subsection for k = 1>

<subsection for k = 2>

<subsection for k = NK>

[MAT, 5, 0 /0.0, 0.0, 0, 0, 0, 0]SEND

The structure of a subsection depends on the value of LF. Subsections should be ordered by increasing values of LF. For cases in which more than one subsection contains data using the same LF, these subsections should be ordered by increasing values of θ.

File 5 Energy Distributions of Secondary Neutrons
(General Structure)

```
4.80000+ 4 1.11460+ 2          0          0          2          01281 5 16 2426
7.50000+ 6 0.00000+ 0          0          9          1          41281 5 16 2427
         4          2                                             1281 5 16 2428
7.50000+ 6 0.00000+ 0 7.50000+ 6 5.00000- 1 2.00000+ 7 5.00000- 11281 5 16 2429
2.00000+ 7 0.00000+ 0                                             1281 5 16 2430
0.00000+ 0 0.00000+ 0          0          0          1          31281 5 16 2431
         3          5                                             1281 5 16 2432
7.50000+ 6 6.89960+ 5 1.50000+ 7 9.75750+ 5 2.00000+ 7 9.75750+ 51281 5 16 2433
0.00000+ 0 0.00000+ 0          0          1          1          41281 5 16 2434
         4          2                                             1281 5 16 2435
7.50000+ 6 0.00000+ 0 7.50000+ 6 5.00000- 1 2.00000+ 7 5.00000- 11281 5 16 2436
2.00000+ 7 0.00000+ 0                                             1281 5 16 2437
0.00000+ 0 0.00000+ 0          0          0          1          21281 5 16 2438
         2          2                                             1281 5 16 2439
0.00000+ 0 7.50000+ 6          0          0          1         171281 5 16 2440
        17          2                                             1281 5 16 2441
1.00000- 5 1.00000-16 1.00000+ 4 3.35370- 6 2.00000+ 4 4.14340- 61281 5 16 2442
4.00000+ 4 4.72520- 6 6.00000+ 4 4.78310- 6 8.00000+ 4 4.60920- 61281 5 16 2443
1.00000+ 5 4.31890- 6 1.20000+ 5 3.97090- 6 1.40000+ 5 3.59830- 61281 5 16 2444
1.60000+ 5 3.22120- 6 1.80000+ 5 2.85190- 6 2.00000+ 5 2.49810- 61281 5 16 2445
2.50000+ 5 1.70680- 6 3.00000+ 5 1.06400- 6 3.50000+ 5 5.65100- 71281 5 16 2446
4.00000+ 5 1.89300- 7 4.40000+ 5 1.00000-26                       1281 5 16 2447
0.00000+ 0 2.00000+ 7          0          0          1         171281 5 16 2448
        17          2                                             1281 5 16 2449
1.00000- 5 1.00000-16 1.00000+ 4 3.35370- 6 2.00000+ 4 4.14340- 61281 5 16 2450
4.00000+ 4 4.72520- 6 6.00000+ 4 4.78310- 6 8.00000+ 4 4.60920- 61281 5 16 2451
1.00000+ 5 4.31890- 6 1.20000+ 5 3.97090- 6 1.40000+ 5 3.59830- 61281 5 16 2452
1.60000+ 5 3.22120- 6 1.80000+ 5 2.85190- 6 2.00000+ 5 2.49810- 61281 5 16 2453
2.50000+ 5 1.70680- 6 3.00000+ 5 1.06400- 6 3.50000+ 5 5.65100- 71281 5 16 2454
4.00000+ 5 1.89300- 7 4.40000+ 5 1.00000-26                       1281 5 16 2455
                                                                  1281 5  0 2456
4.80000+ 4 1.11460+ 2          0          0          1          01281 5 91 2457
1.50000+ 6 0.00000+ 0          0          9          1          41281 5 91 2458
         4          2                                             1281 5 91 2459
1.50000+ 6 0.00000+ 0 1.50000+ 6 1.00000+ 0 2.00000+ 7 1.00000+ 01281 5 91 2460
2.00000+ 7 0.00000+ 0                                             1281 5 91 2461
0.00000+ 0 0.00000+ 0          0          0          1          31281 5 91 2462
         3          5                                             1281 5 91 2463
1.50000+ 6 3.08560+ 5 1.50000+ 7 9.75750+ 5 2.00000+ 7 9.75750+ 51281 5 91 2464
                                                                  1281 5  0 2465
                                                                  1281 0  0 2466
```

File 5 Energy Distributions of Secondary Neutrons (General Structure)

The energy distributions, $p(E \rightarrow E')$, are normalized so that

$$\int_{0}^{E'_{max}} p(E \rightarrow E')\, dE' = 1 ,$$

where E'_{max} is the maximum possible secondary neutron energy and its value depends on the incoming neutron energy E and the analytic representation of $p(E \rightarrow E')$. The secondary neutron energy E' is always expressed in the laboratory system.

The differential cross section is obtained from

$$\frac{d\sigma\ (E \rightarrow E')}{dE'} = m\ \sigma(E)\ p\ (E \rightarrow E'),$$

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

The partial energy distributions $f_k(E \rightarrow E')$ are represented by various analytical formulations. Each formulation is called an energy distribution law and has an identification number associated with it (LF number).

File 5 Energy Distributions of Secondary Neutrons (Tabulated Distributions)

The following quantities are defined

NK is the number of partial energy distributions. There will be one subsection for each partial distribution.

U is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ is a parameter used to describe the secondary energy distribution. 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.

If LF = 5, 7, or 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 in Section 5.1.).

$p_k(E_N)$ is the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ is the kth partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

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

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

NE is the number of incident energy points at which tabulated distributions are given. Also the number of points at which θ(E) is given. (NE ≤ 200.)

NF is the number of secondary energy points in a tabulation. (NF ≤ 1000.)

File 5 Energy Distributions of Secondary Neutrons
(Tabulated Distributions)

LF = 1, Arbitrary tabulated function

[MAT, 5, MT/T , 0.0 , LT , LF=1 , NR , NP/E_{int}/p(E)]TAB1

[MAT, 5, MT/0.0 , 0.0 , 0 , 0 , NR , NE/E_{int}]TAB2

[MAT, 5, MT/T , E_1 , LT , 0 , NR , NF/E'_{int}/

E'_1 , $g(E_1 \rightarrow E'_1)$, E'_2 , $g(E_1 \rightarrow E'_2)$, E'_3 , $g(E_1 \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_1 \rightarrow E'_{NF})$]TAB1

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

E'_1 , $g(E_2 \rightarrow E'_1)$, E'_2 , $g(E_2 \rightarrow E'_2)$, E'_3 , $g(E_2 \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_2 \rightarrow E'_{NF})$]TAB1

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

E'_1 , $g(E_{NE} \rightarrow E'_1)$, E'_2 , $g(E_{NE} \rightarrow E'_2)$, E'_3 , $g(E_{NE} \rightarrow E'_3)$,

---

---, E'_{NF} , $g(E_{NE} \rightarrow E'_{NF})$]TAB1

Note that the incident energy mesh for $p_k(E)$ does not have to be the same as the E mesh used to specify the energy distributions. The interpolation scheme used between incident energy points, E, and between secondary energy points, E', should be linear-linear. T and LT refer to possible temperature (physical) dependence.

File 5 Energy Distributions of Secondary Neutrons
(Tabulated Distributions)

```
0.00000+ 0 0.00000+ 0          0          1          1          31283 5 16 1281
         3          2                                                1283 5 16 1282
8.12138+ 6 0.00000+ 0 1.50000+ 7 5.00000- 1 2.00000+ 7 5.00000- 11283 5 16 1283
0.00000+ 0 0.00000+ 0          0          0          1          41283 5 16 1284
         4          2                                                1283 5 16 1285
0.00000+ 0 5.00000+ 6          0          0          1         261283 5 16 1286
        26          2                                                1283 5 16 1287
0.00000+ 0 0.00000+ 0 2.00000+ 5 6.39700- 8 4.00000+ 5 1.17540- 71283 5 16 1288
6.00000+ 5 1.62030- 7 8.00000+ 5 1.98570- 7 1.00000+ 6 2.28100- 71283 5 16 1289
1.20000+ 6 2.51430- 7 1.40000+ 6 2.69240- 7 1.60000+ 6 2.82100- 71283 5 16 1290
1.80000+ 6 2.90510- 7 2.00000+ 6 2.94880- 7 2.20000+ 6 2.95540- 71283 5 16 1291
2.40000+ 6 2.92780- 7 2.60000+ 6 2.86840- 7 2.80000+ 6 2.77890- 71283 5 16 1292
3.00000+ 6 2.66090- 7 3.20000+ 6 2.51550- 7 3.40000+ 6 2.34350- 71283 5 16 1293
3.60000+ 6 2.14510- 7 3.80000+ 6 1.92070- 7 4.00000+ 6 1.67010- 71283 5 16 1294
4.20000+ 6 1.39290- 7 4.40000+ 6 1.08830- 7 4.60000+ 6 7.55520- 81283 5 16 1295
4.80000+ 6 3.93240- 8 5.00000+ 6 3.65950-14                      1283 5 16 1296
0.00000+ 0 1.00000+ 7          0          0          1         411283 5 16 1297
        41          2                                                1283 5 16 1298
0.00000+ 0 0.00000+ 0 2.50000+ 5 3.16190- 8 5.00000+ 5 5.76040- 81283 5 16 1299
7.50000+ 5 7.88870- 8 1.00000+ 6 9.62310- 8 1.25000+ 6 1.10260- 71283 5 16 1300
1.50000+ 6 1.21500- 7 1.75000+ 6 1.30380- 7 2.00000+ 6 1.37240- 71283 5 16 1301
2.25000+ 6 1.42390- 7 2.50000+ 6 1.46090- 7 2.75000+ 6 1.48520- 71283 5 16 1302
3.00000+ 6 1.49880- 7 3.25000+ 6 1.50300- 7 3.50000+ 6 1.49920- 71283 5 16 1303
3.75000+ 6 1.48830- 7 4.00000+ 6 1.47110- 7 4.25000+ 6 1.44860- 71283 5 16 1304
4.50000+ 6 1.42110- 7 4.75000+ 6 1.38940- 7 5.00000+ 6 1.35370- 71283 5 16 1305
5.25000+ 6 1.31440- 7 5.50000+ 6 1.27190- 7 5.75000+ 6 1.22620- 71283 5 16 1306
6.00000+ 6 1.17760- 7 6.25000+ 6 1.12630- 7 6.50000+ 6 1.07220- 71283 5 16 1307
6.75000+ 6 1.01550- 7 7.00000+ 6 9.56090- 8 7.25000+ 6 8.94030- 81283 5 16 1308
7.50000+ 6 8.29240- 8 7.75000+ 6 7.61640- 8 8.00000+ 6 6.91130- 81283 5 16 1309
8.25000+ 6 6.17560- 8 8.50000+ 6 5.40790- 8 8.75000+ 6 4.60610- 81283 5 16 1310
9.00000+ 6 3.76810- 8 9.25000+ 6 2.89150- 8 9.50000+ 6 1.97340- 81283 5 16 1311
9.75000+ 6 1.01070- 8 1.00000+ 7 0.00000+ 0                      1283 5 16 1312
0.00000+ 0 1.50000+ 7          0          0          1         311283 5 16 1313
        31          2                                                1283 5 16 1314
0.00000+ 0 0.00000+ 0 5.00000+ 5 3.86080- 8 1.00000+ 6 6.48220- 81283 5 16 1315
1.50000+ 6 8.23490- 8 2.00000+ 6 9.37180- 8 2.50000+ 6 1.00680- 71283 5 16 1316
3.00000+ 6 1.04470- 7 3.50000+ 6 1.05960- 7 4.00000+ 6 1.05770- 71283 5 16 1317
4.50000+ 6 1.04360- 7 5.00000+ 6 1.02050- 7 5.50000+ 6 9.90810- 81283 5 16 1318
6.00000+ 6 9.56290- 8 6.50000+ 6 9.18190- 8 7.00000+ 6 8.77400- 81283 5 16 1319
7.50000+ 6 8.34570- 8 8.00000+ 6 7.90120- 8 8.50000+ 6 7.44340- 81283 5 16 1320
9.00000+ 6 6.97380- 8 9.50000+ 6 6.49310- 8 1.00000+ 7 6.00100- 81283 5 16 1321
1.05000+ 7 5.49670- 8 1.10000+ 7 4.97880- 8 1.15000+ 7 4.44520- 81283 5 16 1322
1.20000+ 7 3.89350- 8 1.25000+ 7 3.32080- 8 1.30000+ 7 2.72350- 81283 5 16 1323
1.35000+ 7 2.09780- 8 1.40000+ 7 1.43890- 8 1.45000+ 7 7.41590- 91283 5 16 1324
1.50000+ 7 0.00000+ 0                                            1283 5 16 1325
0.00000+ 0 2.00000+ 7          0          0          1         411283 5 16 1326
        41          2                                                1283 5 16 1327
0.00000+ 0 0.00000+ 0 5.00000+ 5 3.06700- 8 1.00000+ 6 5.15800- 81283 5 16 1328
1.50000+ 6 6.56480- 8 2.00000+ 6 7.48720- 8 2.50000+ 6 8.06380- 81283 5 16 1329
3.00000+ 6 8.39220- 8 3.50000+ 6 8.54150- 8 4.00000+ 6 8.56160- 81283 5 16 1330
4.50000+ 6 8.48830- 8 5.00000+ 6 8.34800- 8 5.50000+ 6 8.15990- 81283 5 16 1331
6.00000+ 6 7.93810- 8 6.50000+ 6 7.69300- 8 7.00000+ 6 7.43240- 81283 5 16 1332
7.50000+ 6 7.16170- 8 8.00000+ 6 6.88520- 8 8.50000+ 6 6.60570- 81283 5 16 1333
9.00000+ 6 6.32540- 8 9.50000+ 6 6.04560- 8 1.00000+ 7 5.76730- 81283 5 16 1334
1.05000+ 7 5.49100- 8 1.10000+ 7 5.21680- 8 1.15000+ 7 4.94480- 81283 5 16 1335
1.20000+ 7 4.67470- 8 1.25000+ 7 4.40620- 8 1.30000+ 7 4.13870- 81283 5 16 1336
1.35000+ 7 3.87170- 8 1.40000+ 7 3.60430- 8 1.45000+ 7 3.33600- 81283 5 16 1337
1.50000+ 7 3.06560- 8 1.55000+ 7 2.79240- 8 1.60000+ 7 2.51530- 81283 5 16 1338
1.65000+ 7 2.23310- 8 1.70000+ 7 1.94460- 8 1.75000+ 7 1.64850- 81283 5 16 1339
1.80000+ 7 1.34350- 8 1.85000+ 7 1.02790- 8 1.90000+ 7 7.00020- 91283 5 16 1340
1.95000+ 7 3.58080- 9 2.00000+ 7 0.00000+ 0                      1283 5 16 1341
```

File 5 Energy Distributions of Secondary Neutrons
(Tabulated Distributions)

LF = 1, <u>Arbitrary tabulated function</u>:

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

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

File 5 Energy Distributions of Secondary Neutrons
(Discrete Level & General Evaporation)

The following quantities are defined

NK is the number of partial energy distributions. There will be one subsection for each partial distribution.

U is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ is a parameter used to describe the secondary energy distribution. 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.

If LF = 5, 7, or 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 in Section 5.1.).

$p_k(E_N)$ is the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ is the kth partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

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

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

NE is the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given. ($NE \leq 200$.)

NF is the number of secondary energy points in a tabulation. ($NF \leq 1000$.)

File 5 Energy Distributions of Secondary Neutrons
(Discrete Level & General Evaporation)

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 , $g(x_3)$ $x = \frac{E'}{\theta(E)}$

---------- , x_{NF}, $g(x_{NF})$]TAB1

File 5 Energy Distributions of Secondary Neutrons
(Discrete Level & General Evaporation)

```
0.0          2.2474 +06          2          3          1         29 233 5  4 1294
          29          2                                                233 5  4 1295
0.+427 +06 1.72        0.395  +06 1.8        0.19   +06 0.90592     233 5  4 1296
0.12   +06 1.975       0.14   +06 0.93636    0.16   +06 0.89787     233 5  4 1297
0.18   +06 2.44169     0.26   +06 0.88583    0.22   +06 0.75148     233 5  4 1298
0.24   +06 2.72459     0.26   +06 0.65714    0.28   +06 0.60522     233 5  4 1299
0.30   +06 3.56752     0.35   +06 0.47541    0.40   +06 0.408       233 5  4 1300
0.45   +06 2.33799     0.50   +06 0.28036    0.55   +06 0.24478     233 5  4 1301
0.60   +06 2.20735     0.65   +06 0.16115    0.70   +06 0.14046     233 5  4 1302
0.75   +06 2.12439     0.80   +06 0.18676    0.85   +06 0.39189     233 5  4 1303
0.90   +06 2.27792     0.95   +06 0.26456    1.00   +06 0.85492     233 5  4 1304
1.00   +06 2.0         15.0   +06 0.5                               233 5  4 1305
```

```
-2.00000+ 7 0.00000+ 0          0          5          1          21260 5 18 4896
          2          2                                                1260 5 18 4897
 0.00000+ 0 8.97660- 1 2.00000+ 7 8.97660- 1                         1260 5 18 4898
```

File 5 Energy Distributions of Secondary Neutrons
(Discrete Level & General Evaporation)

LF = 3, <u>Excitation of discrete levels</u>:

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

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, <u>General evaporation spectrum</u>:

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

$\theta(E)$ is tabulated as a function of incident neutron energy, E;

$g(x)$ is tabulated as a function of x, $x = E'/\theta(E)$.

File 5 Energy Distributions of Secondary Neutrons (Maxwellian, Evaporation, & Watt Spectrum)

The following quantities are defined

NK is the number of partial energy distributions. There will be one subsection for each partial distribution.

U is a constant that defines the upper energy limit for the secondary neutron so that $0 \leq E' \leq E - U$ (given in the Lab system).

θ is a parameter used to describe the secondary energy distribution. 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.

If LF = 5, 7, or 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 in Section 5.1.).

$p_k(E_N)$ is the fractional part of the particular cross section that can be described by the kth partial energy distribution at the Nth incident energy point.

NOTE: $$\sum_{k=1}^{NK} p_k(E_N) = 1.0$$

$f_k(E \rightarrow E')$ is the kth partial energy distribution. The definition depends on the value of LF.

NR is the number of interpolation ranges.

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

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

NE is the number of incident energy points at which tabulated distributions are given. Also the number of points at which $\theta(E)$ is given. ($NE \leq 200$.)

NF is the number of secondary energy points in a tabulation. ($NF \leq 1000$.)

The structure of a section has the following form:

File 5 Energy Distributions of Secondary Neutrons
(Maxwellian, Evaporation, & Watt Spectrum)

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.

File 5 Energy Distributions of Secondary Neutrons
(Maxwellian, Evaporation, & Watt Spectrum)

```
-1.00000+10 0.00000+ 0          0          7          1          21260 5 18 4884
          2          2                                              1260 5 18 4885
 0.00000+ 0 2.60000- 3 2.00000+ 7 2.60000- 3                       1260 5 18 4886
 0.00000+ 0 0.00000+ 0          0          0          1          21260 5 18 4887
          2          2                                              1260 5 18 4888
 0.00000+ 0 3.00000+ 5 2.00000+ 7 3.00000+ 5                       1260 5 18 4889
```

```
-2.00000+ 7 0.00000+ 0          0          9          1          21260 5 18 4890
          2          2                                              1260 5 18 4891
 0.00000+ 0 9.97400- 2 2.00000+ 7 9.97400- 2                       1260 5 18 4892
 0.00000+ 0 0.00000+ 0          0          0          1          21260 5 18 4893
          2          2                                              1260 5 18 4894
 0.00000+ 0 1.52600+ 6 2.00000+ 7 1.52600+ 6                       1260 5 18 4895
```

```
                                             18          1        2 239 5 18  181
          2          2                                              239 5 18  182
 1.0    -85 1.70       1.5    -87 1.98                              239 5 18  183
                                                          2         239 5 18  184
 1.2    -86 2.2    -76                                              239 5 18  185
```

File 5 Energy Distributions of Secondary Neutrons
(Maxwellian, Evaporation, & Watt Spectrum)

LF = 7, Simple fission spectrum (Maxwellian):

$$f(E \rightarrow E') = \frac{\sqrt{E'}}{I} e^{-E'/\theta(E)} .$$

I is the normalization constant,

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

θ is tabulated as a function of energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy that $0 \leq E' \leq E - U$.

LF = 9, Evaporation spectrum:

$$f\ (E \rightarrow E') = \frac{E'}{I} e^{-E'/\theta} .$$

I is the normalization constant,

$$I = \theta^2 \left[1 - e^{-(E-U)/\theta} \left(1 + \frac{E-U}{\theta}\right)\right] ;$$

θ is tabulated as a function of incident neutron energy, E;

U is a constant introduced to define the proper upper limit for the final neutron energy that $0 \leq E' \leq E - U$.

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 constants.

NOTE: Distribution laws are not presented for LF = 2, 4, 6, or 8. These laws are no longer used.

There is only one section in File 7, but the format varies slightly, depending on whether temperature-dependent data are given. The following quantities are defined:

LAT is a flag indicating which temperature has been used to compute α and β.

LAT = 0, the actual temperature was used.

LAT = 1, the constant T_o = 0.0253 eV has been used.

NS is the number of non-principal scattering atom types. For most moderating materials there will be (NS + 1) types of atoms in the molecule (NS $\leq$ 3).

NI is the total number of items in the B(N) list. NL = 6*(NS + 1).

B(N) is the list of constants. Definitions are given above (Section 7.1).

NR is the number of interpolation ranges for a particular parameter, either β or α.

NB is the total number of β values given.

NP is the number of α values given for each value of β for the first temperature described, NP is the number of pairs, α and $S(\alpha,\beta)$, given.

β_{int} and α_{int} are the interpolation schemes used (see Appendix E for interpolation formats).

The structure of a section is

[MAT, 7, MT/ZA, AWR, 0, LAT, 0, 0]HEAD

[MAT, 7, MT/0.0, 0.0, 0, 0, NI, NS/B(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 are completely specified by analytic functions [no principal scattering atom type, as indicated by B(1) = 0], tabulated values of $S_0(\alpha,\beta)$ are omitted and the TAB2 and TAB1 records are not given.

File 7 Scattering Law Data

```
8.01600+ 3 1.58580+ 1          0          0          0          01276 7  4 4761
0.00000+ 0 0.00000+ 0          0          0         12          11276 7  4 4762
0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 0 0.00000+ 01276 7  4 4763
1.00000+ 0 3.74810+ 0 1.58580+ 1 0.00000+ 0 0.00000+ 0 0.00000+ 01276 7  4 4764
                                                                 1276 7  0 4765
                                                                 1276 0  0 4766
```

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

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

where there are (NS + 1) types of atoms in the molecule (i.e., for H_2O, NS = 1)

$\underline{M_n}$ is the number of atoms of the nth type in the molecule,

$\underline{T}$ is the moderator temperature (°K),

$\underline{E}$ is the *incident neutron energy* (eV),

$\underline{E'}$ is the secondary neutron energy (eV),

$\underline{\beta}$ is the energy transfer, $\beta = (E' - E)/kT$,

$\underline{\alpha}$ is the momentum transfer, $\alpha = (E' + E - 2\mu\sqrt{EE'})/A_o kT$,

$\underline{A_n}$ is the mass of the nth type atom, A_o is the mass of the principal scattering atom in the molecule,

$\underline{\sigma_{bn}}$ is the bound atom scattering cross section of the nth type atom, $\sigma_{bn} = \sigma_{fn}\left(\frac{A_n + 1}{A_n}\right)^2$

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

$\underline{k}$ is Boltzmann's contant, and

μ is the cosine of the scattering angle (in the lab system).

File 12 Multiplicities and Transition Probabilities
(Photon Multiplicities)

$\underline{ES_k}$ the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced, then $ES_k \equiv 0.0$ should be used.

$\underline{EG_k}$ the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k \equiv 0.0$ should be used.

<u>LP</u> indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons where the photon energy EG'_k is given by

$$EG'_k = EG_k + \frac{AWR}{AWR+1} E_n .$$

<u>LF</u> the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

= 2, a discrete photon energy.

File 12 Multiplicities and Transition Probabilities (Photon Multiplicities)

The structure of a section for LØ = 1 is

[MAT, 12, MT/ZA, AWR; LØ=1, b; NK, b]HEAD

[MAT, 12, MT/ b, b; b, b; NR, NP/E_{int}/Y(E)]TAB1*

<subsection for k = 1>

<subsection for k = 2>

.

.

.

<subsection for k = NK>

[MAT, 12, 0/ b, b; b, b; b, b]SEND ,

and the structure of each subsection is

[MAT, 12, MT/EG_k, ES_k; LP, LF; NR, NP/E_{int}/y_k(E)]TAB1 ,

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

File 12 Multiplicities and Transition Probabilities
(Photon Multiplicities)

```
9.22350+ 4 2.33025+ 2          1          0         30          0126112  4 6169
0.00000+ 0 0.00000+ 0          0          0          1         29126112  4 6170
        29         2                                                126112  4 6171
1.30550+ 4 0.00000+ 0 1.30560+ 4 1.00000+ 0 4.00000+ 4 1.00000+ 0126112  4 6172
5.22180+ 4 1.00000+ 0 8.23440+ 4 1.19210+ 0 1.03433+ 5 1.28190+ 0126112  4 6173
1.50630+ 5 1.42470+ 0 1.71718+ 5 1.43780+ 0 1.97827+ 5 1.46350+ 0126112  4 6174
2.25945+ 5 1.49120+ 0 2.96239+ 5 1.57010+ 0 3.34400+ 5 1.64930+ 0126112  4 6175
3.68541+ 5 1.73810+ 0 3.95655+ 5 1.79790+ 0 4.15739+ 5 1.79140+ 0126112  4 6176
4.28793+ 5 1.87680+ 0 4.75991+ 5 1.87530+ 0 5.00000+ 5 1.95490+ 0126112  4 6177
5.35239+ 5 1.93480+ 0 6.00000+ 5 2.13210+ 0 6.54738+ 5 2.23330+ 0126112  4 6178
7.00000+ 5 2.23460+ 0 7.74238+ 5 2.24560+ 0 8.00000+ 5 2.25100+ 0126112  4 6179
9.00000+ 5 2.16840+ 0 1.00000+ 6 2.14020+ 0 1.09000+ 6 2.12850+ 0126112  4 6180
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                     126112  4 6181
7.71000+ 5 7.71000+ 5          0          2          1         7126112  4 6182
         7         2                                                126112  4 6183
7.74238+ 5 0.00000+ 0 8.00000+ 5 1.00000- 2 9.00000+ 5 7.11000- 2126112  4 6184
1.00000+ 6 1.19000- 1 1.09000+ 6 1.23300- 1 1.09000+ 6 0.00000+ 0126112  4 6185
2.00000+ 7 0.00000+ 0                                           126112  4 6186
7.58000+ 5 7.71000+ 5          0          2          1         7126112  4 6187
         7         2                                                126112  4 6188
7.74238+ 5 0.00000+ 0 8.00000+ 5 3.00000- 3 9.00000+ 5 2.31000- 2126112  4 6189
1.00000+ 6 3.82000- 2 1.09000+ 6 3.97000- 2 1.09000+ 6 0.00000+ 0126112  4 6190
2.00000+ 7 0.00000+ 0                                           126112  4 6191

    (MISSING LINES)

        25         2                                                126112  4 6378
8.23440+ 4 0.00000+ 0 1.03433+ 5 7.81000- 2 1.50630+ 5 1.70900- 1126112  4 6379
1.71718+ 5 1.92700- 1 1.97827+ 5 2.13200- 1 2.25945+ 5 2.32500- 1126112  4 6380
2.96239+ 5 2.19300- 1 3.34400+ 5 2.16000- 1 3.68541+ 5 2.08200- 1126112  4 6381
3.95655+ 5 2.04700- 1 4.15739+ 5 1.94400- 1 4.28793+ 5 1.87400- 1126112  4 6382
4.75991+ 5 1.73200- 1 5.00000+ 5 1.68700- 1 5.35239+ 5 1.60600- 1126112  4 6383
6.00000+ 5 1.70400- 1 6.54738+ 5 1.80100- 1 7.00000+ 5 1.72100- 1126112  4 6384
7.74238+ 5 1.74100- 1 8.00000+ 5 1.66700- 1 9.00000+ 5 1.41200- 1126112  4 6385
1.00000+ 6 1.19800- 1 1.09000+ 6 1.34900- 1 1.09000+ 6 0.00000+ 0126112  4 6386
2.00000+ 7 0.00000+ 0                                           126112  4 6387
5.20000+ 4 5.20000+ 4          0          2          1        26126112  4 6388
        26         2                                                126112  4 6389
5.22180+ 4 0.00000+ 0 8.23440+ 4 5.22600- 1 1.03433+ 5 5.55300- 1126112  4 6390
1.50630+ 5 5.66800- 1 1.71718+ 5 5.64900- 1 1.97827+ 5 5.57800- 1126112  4 6391
2.25945+ 5 5.39400- 1 2.96239+ 5 5.59900- 1 3.34400+ 5 5.35100- 1126112  4 6392
3.68541+ 5 5.30900- 1 3.95655+ 5 5.26900- 1 4.15739+ 5 5.14400- 1126112  4 6393
4.28793+ 5 5.18500- 1 4.75991+ 5 4.93800- 1 5.00000+ 5 5.06100- 1126112  4 6394
5.35239+ 5 4.98600- 1 6.00000+ 5 4.99200- 1 6.54738+ 5 4.92100- 1126112  4 6395
7.00000+ 5 4.47800- 1 7.74238+ 5 4.09200- 1 8.00000+ 5 3.98600- 1126112  4 6396
9.00000+ 5 3.53500- 1 1.00000+ 6 3.12800- 1 1.09000+ 6 3.29500- 1126112  4 6397
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                     126112  4 6398
3.90000+ 4 5.20000+ 4          0          2          1        26126112  4 6399
        26         2                                                126112  4 6400
5.22180+ 4 0.00000+ 0 8.23440+ 4 1.91700- 1 1.03433+ 5 2.03900- 1126112  4 6401
1.50630+ 5 2.13200- 1 1.71718+ 5 1.97600- 1 1.97827+ 5 1.94000- 1126112  4 6402
2.25945+ 5 1.86000- 1 2.96239+ 5 1.94400- 1 3.34400+ 5 1.83600- 1126112  4 6403
3.68541+ 5 1.98400- 1 3.95655+ 5 1.75600- 1 4.15739+ 5 1.75100- 1126112  4 6404
4.28793+ 5 1.77800- 1 4.75991+ 5 1.69100- 1 5.00000+ 5 1.84800- 1126112  4 6405
5.35239+ 5 1.69600- 1 6.00000+ 5 1.82100- 1 6.54738+ 5 1.69800- 1126112  4 6406
7.00000+ 5 1.53500- 1 7.74238+ 5 1.43500- 1 8.00000+ 5 1.36500- 1126112  4 6407
9.00000+ 5 1.23700- 1 1.00000+ 6 1.08000- 1 1.09000+ 6 1.13400- 1126112  4 6408
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                     126112  4 6409
1.30000+ 4 1.30000+ 4          0          2          1        29126112  4 6410
        29         2                                                126112  4 6411
1.30550+ 4 0.00000+ 0 1.30560+ 4 1.00000+ 0 4.00000+ 4 1.00000+ 0126112  4 6412
5.22180+ 4 1.00000+ 0 8.23440+ 4 4.77800- 1 1.03433+ 5 4.44600- 1126112  4 6413
1.50630+ 5 4.67000- 1 1.71718+ 5 4.57900- 1 1.97827+ 5 4.63300- 1126112  4 6414
2.25945+ 5 4.66600- 1 2.96239+ 5 4.16700- 1 3.34400+ 5 4.34900- 1126112  4 6415
3.68541+ 5 4.37300- 1 3.95655+ 5 4.08000- 1 4.15739+ 5 3.99900- 1126112  4 6416
4.28793+ 5 4.09000- 1 4.75991+ 5 3.99100- 1 5.00000+ 5 3.89600- 1126112  4 6417
5.35239+ 5 3.93700- 1 6.00000+ 5 4.04400- 1 6.54738+ 5 3.98000- 1126112  4 6418
7.00000+ 5 4.13600- 1 7.74238+ 5 4.30600- 1 8.00000+ 5 4.36700- 1126112  4 6419
9.00000+ 5 4.20000- 1 1.00000+ 6 3.83400- 1 1.09000+ 6 4.10600- 1126112  4 6420
1.09000+ 6 0.00000+ 0 2.00000+ 7 0.00000+ 0                     126112  4 6421
                                                                126112  0 6422
```

File 12 Multiplicities and Transition Probabilities
(Photon Multiplicities)

The multiplicity or yield $y_k(E)$ is defined by

$$y_k(E) = \frac{\sigma_k^{\gamma}(E)}{\sigma(E)} \quad \text{(photons)} ,$$

where E designates neutron energy and $\sigma(E)$ is the neutron cross section in File 2 and/or File 3 to which the multiplicity is referred (by the MT number).

$$y_k(E_\gamma \leftarrow E) = y_k(E)\ f_k(E_\gamma \leftarrow E) ,$$

which results in the requirement that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\,dE_\gamma = 1 .$$

As a check quantity, the total yield

$$Y(E) = \sum_{k=1}^{NK} y_k(E) \quad \text{(photons)}$$

is also tabulated for each MT number if NK > 1.

File 12 Multiplicities and Transition Probabilities (Photon Transition Probability Arrays)

LG = 1, simple case (all transitions are γ emission).

= 2, complex case (internal conversion or other competing processes occur).

<u>NS</u> number of levels below the present one, including the ground state. (The present level is also uniquely defined by the MT number and by its energy level.)

<u>NT</u> number of transitions for which data are given in a list to follow (i.e., number of nonzero transition probabilities), NT $\leq$ NS.

<u>ES_i</u> energy of the ith level, i = 0,1,2... NS. ($ES_o \equiv 0.0$, the ground state.)

<u>TP_i</u> $TP_{NS,i}$, the probability of a direct transition from level NS to level i, i = 0,1,2... (NS-1).

<u>GP_i</u> $GP_{NS,i}$, the probability that, given a transition from level NS to level i, the transition is a photon transition (i.e., the conditional probability of photon emission).

<u>A_i</u> (TP_i) (GP_i).

Note that each level can be identified by its NS number. Then the energy of a photon from a transition to level i is given by $E_\gamma = ES_{NS} - ES_i$, and its multiplicity is given by $y(E_\gamma \leftarrow E) = (TP_i)$ (GP_i). It is implicitly assumed that the transition probability array is independent of incident neutron energy.

File 12 Multiplicities and Transition Probabilities
(Photon Transition Probability Arrays)

The structure of a section for LØ = 2 is

[MAT, 12, MT/ ZA, AWR; LØ=2, LG; NS, b]HEAD.

[MAT, 12, MT/ES_{NS}, b; LP, b; (LG+1)*NT, NT/B_i]LIST.

[MAT, 12 0/ b, b; b, b; b, b]SEND.

If LG = 1, the array B_i consists of NT doublets (ES_i,TP_i); if LG = 2, it consists of NT triplets (ES_i,TP_i,GP_i). Here the subscript i is a running index over the levels below the level for which the transition probability array is being given (i.e., below level NS). The doublets or triplets are given in decreasing magnitude of energy ES_i.

File 12 Multiplicities and Transition Probabilities
(Photon Transition Probability Arrays)

```
1.70000+ 4 3.51480+ 1          2          1          1          0114912 51 3401
1.22000+ 6 0.00000+ 0          0          0          2          1114912 51 3402
0.00000+ 0 1.00000+ 0                                            114912 51 3403
                                                                 114912  0 3404
1.70000+ 4 3.51480+ 1          2          1          2          0114912 52 3405
1.76200+ 6 0.00000+ 0          0          0          2          1114912 52 3406
0.00000+ 0 1.00000+ 0                                            114912 52 3407
                                                                 114912  0 3408
1.70000+ 4 3.51480+ 1          2          1          3          0114912 53 3409
2.64500+ 6 0.00000+ 0          0          0          2          1114912 53 3410
0.00000+ 0 1.00000+ 0                                            114912 53 3411
                                                                 114912  0 3412
1.70000+ 4 3.51480+ 1          2          1          4          0114912 54 3413
2.69500+ 6 0.00000+ 0          0          0          4          2114912 54 3414
1.22000+ 6 6.00000- 1 0.00000+ 0 4.00000- 1                      114912 54 3415
                                                                 114912  0 3416
1.70000+ 4 3.51480+ 1          2          1          5          0114912 55 3417
3.00600+ 6 0.00000+ 0          0          0          2          1114912 55 3418
0.00000+ 0 1.00000+ 0                                            114912 55 3419
                                                                 114912  0 3420
1.70000+ 4 3.51480+ 1          2          1          6          0114912 56 3421
3.16300+ 6 0.00000+ 0          0          0          4          2114912 56 3422
2.64500+ 6 1.60000- 1 0.00000+ 0 8.40000- 1                      114912 56 3423
                                                                 114912  0 3424
1.70000+ 4 3.51480+ 1          2          1          7          0114912 57 3425
4.05800+ 6 0.00000+ 0          0          0          6          3114912 57 3426
1.76200+ 6 1.80000- 1 1.22000+ 6 6.40000- 1 0.00000+ 0 1.80000- 1114912 57 3427
                                                                 114912  0 3428
1.70000+ 4 3.51480+ 1          2          1          8          0114912 58 3429
4.11300+ 6 0.00000+ 0          0          0          2          1114912 58 3430
1.22000+ 6 1.00000+ 0                                            114912 58 3431
                                                                 114912  0 3432
1.70000+ 4 3.51480+ 1          2          1          9          0114912 59 3433
4.17400+ 6 0.00000+ 0          0          0          4          2114912 59 3434
1.22000+ 6 6.20000- 1 0.00000+ 0 3.80000- 1                      114912 59 3435
                                                                 114912  0 3436
1.70000+ 4 3.51480+ 1          2          1         10          0114912 60 3437
5.13000+ 6 0.00000+ 0          0          0          2          1114912 60 3438
0.00000+ 0 1.00000+ 0                                            114912 60 3439
                                                                 114912  0 3440
1.70000+ 4 3.51480+ 1          2          1         11          0114912 61 3441
5.22000+ 6 0.00000+ 0          0          0          2          1114912 61 3442
0.00000+ 0 1.00000+ 0                                            114912 61 3443
                                                                 114912  0 3444
```

File 12 Multiplicities and Transition Probabilities
(Photon Transition Probability Arrays)

The differential cross section for producing a γ-ray of energy E_γ resulting from the excitation of the m_o^{th} level of the residual nucleus and the subsequent transition between two definite levels ($j \rightarrow i$), including the effects of cascading from the $m_o - j$ levels higher than j, is

$$\frac{d\sigma}{dE_\gamma}(E_\gamma, E, m_o, i, j) = \delta(E_\gamma - \varepsilon_j + \varepsilon_i) A_{j,i}\, \sigma_{m_o}(E) \prod_{\ell=j}^{m_o-j} \sum_{m_\ell=j}^{m_{\ell-1}-1} TP_{m_{\ell-1},m_\ell}\,, \quad (1)$$

where

$\sigma_{m_o}(E)$ = neutron cross sections for exciting the m_o^{th} level with neutron energy E,

$\delta(E_\gamma - \varepsilon_j + \varepsilon_i)$ = delta function with ε_j, ε_i being energy levels of the residual nucleus,

$TP_{k,\ell}$ = probability of the residual nucleus having a transition to the ℓth level given that it was initially in the excited state corresponding to the kth level, and

$A_{k,\ell}$ = probability of emission of a γ ray of energy $E_\gamma = \varepsilon_k - \varepsilon_\ell$ as a result of the residual nucleus having a transition from the kth to the ℓth level.

$\underline{ES_k}$ the energy of the level from which the photon originates. If the level is unknown or if a continuous photon spectrum is produced. $ES_k = 0.0$ should be used.

$\underline{EG_k}$ the photon energy for LP = 0 or 1 or Binding Energy for LP = 2. For a continuous photon energy distribution, $EG_k = 0.0$ should be used.

<u>LP</u> Indicator of whether or not the particular photon is a primary:

LP = 0, origin of photons is not designated or not known, and the photon energy is EG_k;

LP = 1, for nonprimary photons where the photon energy is again simply EG_k; and

LP = 2, for primary photons, where the photon energy is given by $EG_k + \frac{AWR}{AWR + 1} E_n$.

<u>LF</u> the photon energy distribution law number, which presently has only two values defined:

LF = 1, a normalized tabulated function (in File 15), and

LF = 2, a discrete photon energy.

The structure of a section in File 13 is

[MAT, 13, MT/ZA, AWR; b, b; NK, b]HEAD

[MAT, 13, MT/ b, b; b, b; NR, NP/$E_{int}/\sigma^{\gamma}_{TOT}(E)$]TAB1*

<subsection for k = 1>

<subsection for k = 2>

.

.

.

and the structure of each subsection is

[MAT, 13, MT/EG_k, ES_k; LP, LF; NR, NP/$E_{int}/\sigma^{\gamma}_k(E)$]TAB1 ,

<subsection for k = NK>

[MAT, 13, 0/ b, b; b, b; b, b]SEND

*If the total number of discrete photons and photon continua is one (NK = 1), this TAB1 record is omitted.

```
9.22350+ 4 2.33025+ 2          0          0          1          0126113  3 6436
0.00000+ 0 0.00000+ 0          0          1          1         12126113  3 6437
        12         2                                                126113  3 6438
1.00000- 5 0.00000+ 0 1.00000- 1 0.00000+ 0 1.09000+ 6 0.00000+ 0126113  3 6439
1.09000+ 6 1.38200+ 1 2.10000+ 6 1.48200+ 1 3.00000+ 6 1.48200+ 1126113  3 6440
4.00000+ 6 1.47700+ 1 5.00000+ 6 1.70000+ 1 6.00000+ 6 1.80000+ 1126113  3 6441
7.50000+ 6 2.16300+ 1 1.48000+ 7 2.50200+ 1 2.00000+ 7 2.82100+ 1126113  3 6442
                                                                 126113  0 6443
```

$$\frac{d\sigma_k^\gamma}{dE_\gamma}(E_\gamma \leftarrow E) = \sigma_k^\gamma(E)\ f_k\ (E_\gamma \leftarrow E)\ ,$$

which obviously requires that

$$\int_0^{E_\gamma^{max}} f_k(E_\gamma \leftarrow E)\ dE_\gamma = 1\ .$$

Any time a continuum representation is used for a given MT number in File 13, the normalized energy distribution, $f_k(E_\gamma \leftarrow E)$, must be given in File 15 under the same MT number.

As a check quantity, the total photon production cross section,

$$\sigma_{TOT}^\gamma(E) = \sum_{k=1}^{NK} \sigma_k^\gamma(E) \qquad \text{(barns)}\ ,$$

is also tabulated for each MT number, unless only one subsection is present (i.e., NK = 1).

File 14 Photon Angular Distributions
(Isotropic Distribution for all photons)

For LI = 1 (isotropic distribution), the parameter NK is the number of photons in that section and should be consistent with the NK values in Files 12 and 13.

File 14 Photon Angular Distributions (Isotropic Distribution for all photons)

LI = 1: Isotropic Distribution

If LI = 1, then all photons for the reaction type (MT) in question are assumed to be isotropic. This is a flag that the processing code can sense, and thus needless isotropic distribution data are not entered in the file. In this case, the section is composed of a HEAD card and a SEND card, as follows:

[MAT, 14, MT/ZA, AWR; LI=1, b; NK, b]HEAD

[MAT, 14, 0/ b, b; b, b; b, b]SEND

File 14 Photon Angular Distributions
(Isotropic Distribution for all photons)

9.22350+ 4	2.33025+ 2	1	0	30	0126114 4 6447
					126114 0 6448

File 14 Photon Angular Distributions
(Isotropic Distribution for all photons)

LI = 1 (Isotopic Distributions)

$P_K(\mu,E) \equiv 0.5$

$a_\ell^K(E) \equiv 0.0$ for all K

File 14 Photon Angular Distributions
(Legendre Coefficient Representation)

LTT = 1, data are given as Legendre coefficients, where $a_o^k(E) \equiv 1.0$ is understood.

= 2, data are given as a tabulation.

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

= 1, distribution is isotropic for <u>all</u> photons from this reaction type.

<u>NE</u> number of neutron energy points given in a TAB2 record.

<u>NI</u> number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.

<u>NL_i</u> highest value of ℓ required at each neutron energy E_i.

LTT = 1: Legendre Coefficient Representation

[MAT, 14, MT/ZA, AWR; LI=0, LTT=1; NK, NI]HEAD

<subsection for k = 1>

<subsection for k = 2>

.

.

.

The structure of each record in the first block of NI subsections, which is for the NI isotropic photons, is

[MAT, 14, MT/EG_k, ES_k; b, b; b, b]CØNT .

There is just one CØNT record for each isotropic photon. (The set of CØNT records is empty if NI = 0.) The subsections are ordered in decreasing magnitude of EG_k (photon energy), and the continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons in decreasing magnitude of EG_k. The continuum, if present and anisotropic, appears last, with $EG_k \equiv 0.0$. The structure for the last NK-NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int}]TAB2

[MAT, 14, MT/ b, E_1; b, b; NL_1, b/$a_\ell^k(E_1)$]LIST

[MAT, 14, MT/ b, E_2; b, b; NL_2, b/$a_\ell^k(E_2)$]LIST

.

.

.

[MAT, 14, MT/ b, E_{NE}; b, b; NL_{NE},b/$a_\ell^k(E_{NE})$]LIST .

Note that lists of the $a_\ell^k(E)$ start at $\ell = 1$ because $a_o^k(E) \equiv 1.0$ is always understood.

<subsection for k = NK>

[MAT, 14, 0/ b, b; b, b; b, b]SEND .

File 14 Photon Angular Distributions
(Legendre Coefficient Representation)

No example in ENDF-IV

File 14 Photon Angular Distributions (Legendre Coefficient Representation)

The angular distributions are expressed as normalized probability distributions, that is,

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

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as a Legendre series expansion,

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} (\underline{\Omega},E)$$

$$= \sum_{\ell=0}^{NL} \frac{2\ell + 1}{2} a_\ell^k(E)\ P_\ell(\mu) ,$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^{\gamma}(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

ℓ = order of the Legendre polynomial.

$\frac{d\sigma_k^{\gamma}}{d\underline{\Omega}}$ = differential photon production cross section in barns/steradian.

$a_\ell^k(E)$ = the ℓth Legendre coefficient associated with the discrete photon or photon continuum specified by k. (It is understood that $a_o^k(E) \equiv 1.0$.)

$$a_\ell^k(E) = \int_{-1}^{1} p_k(\mu,E)\ P_\ell(\mu)\,d\mu .$$

File 14 Photon Angular Distributions
(Tabulated Angular Distributions)

LTT = 1, data are given as Legendre coefficients, where $a_o^k(E) \equiv 1.0$ is understood.

= 2, data are given as a tabulation.

LI = 0, distribution is not isotropic for all photons from this reaction type, but may be for some photons.

= 1, distribution is isotropic for <u>all</u> photons from this reaction type.

<u>NE</u> number of neutron energy points given in a TAB2 record.

<u>NI</u> number of isotropic photon angular distributions given in a section (MT number) for which LI = 0, i.e., a section with at least one anisotropic distribution.

<u>NL_i</u> highest value of ℓ required at each neutron energy E_i.

LTT = 2: Tabulated Angular Distributions

The structure of a section for LI = 0 and LTT = 2 is

[MAT, 14, MT/ZA, AWR; LI=0, LTT=2; NK, NI]HEAD

<subsection for k = 1>

<subsection for k = 2>

.

.

.

The structure of the first block of NI subsections (where NI may be zero) is the same as for the case of a Legendre representation; i.e., it consists of one CØNT record for each of the NI isotropic photons in decreasing magnitude of EG_k. The continuum, if present and isotropic, appears last, with $EG_k \equiv 0.0$.

The structure of the first NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; b, b]CØNT .

This block of NI subsections is then followed by a block of NK-NI subsections for the anisotropic photons, again in decreasing magnitude of EG_k, with the continuum, if present and anisotropic, appearing last, with $EG_k \equiv 0.0$. The structure of the last NK-NI subsections is

[MAT, 14, MT/EG_k, ES_k; b, b; NR, NE/E_{int}]TAB2

[MAT, 14, MT/ b, E_1; b, b; NR, NP/μ_{int}/$P_k(\mu,E_1)$]TAB1

[MAT, 14, MT/ b, E_2; b, b; NR, NP/μ_{int}/$P_k(\mu,E_2)$]TAB1

.

.

[MAT, 14, MT/ b, E_{NE}; b, b; NR, NP/μ_{int}/$P_k(\mu,E_{NE})$]TAB1 .

<subsection for k = NK>

[MAT, 14, 0/ b, b; b, b; b, b]SEND .

File 14 Photon Angular Distributions
(Tabulated Angular Distributions)

```
 1.70000+ 4 3.51480+ 1          0          2          3          0114914 57 4406
 4.05800+ 6 4.05800+ 6          0          0          1          8114914 57 4407
          8          2                                             114914 57 4408
 0.00000+ 0 4.17500+ 6          0          0          1         21114914 57 4409
         21          2                                             114914 57 4410
-1.00000+ 0 5.21140- 1-9.00000- 1 5.14290- 1-8.00000- 1 5.09050- 1114914 57 4411
-7.00000- 1 5.04860- 1-6.00000- 1 5.01040- 1-5.00000- 1 4.97660- 1114914 57 4412
-4.00000- 1 4.94830- 1-3.00000- 1 4.92380- 1-2.00000- 1 4.90420- 1114914 57 4413
-1.00000- 1 4.89930- 1 0.00000+ 0 4.89980- 1 1.00000- 1 4.89930- 1114914 57 4414
 2.00000- 1 4.90420- 1 3.00000- 1 4.92380- 1 4.00000- 1 4.94830- 1114914 57 4415
 5.00000- 1 4.97660- 1 6.00000- 1 5.01040- 1 7.00000- 1 5.04860- 1114914 57 4416
 8.00000- 1 5.09050- 1 9.00000- 1 5.14290- 1 1.00000+ 0 5.21140- 1114914 57 4417
 0.00000+ 0 4.50000+ 6          0          0          1         21114914 57 4418
         21          2                                             114914 57 4419
-1.00000+ 0 5.23510- 1-9.00000- 1 5.16930- 1-8.00000- 1 5.10650- 1114914 57 4420
-7.00000- 1 5.05170- 1-6.00000- 1 5.00760- 1-5.00000- 1 4.97130- 1114914 57 4421
-4.00000- 1 4.93730- 1-3.00000- 1 4.91320- 1-2.00000- 1 4.89330- 1114914 57 4422
-1.00000- 1 4.88880- 1 0.00000+ 0 4.88630- 1 1.00000- 1 4.88880- 1114914 57 4423
 2.00000- 1 4.89330- 1 3.00000- 1 4.91320- 1 4.00000- 1 4.93730- 1114914 57 4424
 5.00000- 1 4.97130- 1 6.00000- 1 5.00760- 1 7.00000- 1 5.05170- 1114914 57 4425
 8.00000- 1 5.10650- 1 9.00000- 1 5.16930- 1 1.00000+ 0 5.23510- 1114914 57 4426

     (MISSING LINES)

 0.00000+ 0 7.00000+ 6          0          0          1         21114914 57 4463
         21          2                                             114914 57 4464
-1.00000+ 0 5.27090- 1-9.00000- 1 5.18810- 1-8.00000- 1 5.11990- 1114914 57 4465
-7.00000- 1 5.06160- 1-6.00000- 1 5.00800- 1-5.00000- 1 4.96910- 1114914 57 4466
-4.00000- 1 4.93010- 1-3.00000- 1 4.90090- 1-2.00000- 1 4.88140- 1114914 57 4467
-1.00000- 1 4.87170- 1 0.00000+ 0 4.86690- 1 1.00000- 1 4.87170- 1114914 57 4468
 2.00000- 1 4.88140- 1 3.00000- 1 4.90090- 1 4.00000- 1 4.93010- 1114914 57 4469
 5.00000- 1 4.96910- 1 6.00000- 1 5.00800- 1 7.00000- 1 5.06160- 1114914 57 4470
 8.00000- 1 5.11990- 1 9.00000- 1 5.18810- 1 1.00000+ 0 5.27090- 1114914 57 4471
 0.00000+ 0 2.00000+ 7          0          0          1          2114914 57 4472
          2          1                                             114914 57 4473
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                      114914 57 4474
 2.83800+ 6 4.05800+ 6          0          0          1          8114914 57 4475
          8          2                                             114914 57 4476
 0.00000+ 0 4.17500+ 6          0          0          1         21114914 57 4477
         21          2                                             114914 57 4478
-1.00000+ 0 4.75090- 1-9.00000- 1 4.82780- 1-8.00000- 1 4.88500- 1114914 57 4479
-7.00000- 1 4.93670- 1-6.00000- 1 4.98800- 1-5.00000- 1 5.03420- 1114914 57 4480
-4.00000- 1 5.06540- 1-3.00000- 1 5.09090- 1-2.00000- 1 5.11150- 1114914 57 4481
-1.00000- 1 5.12170- 1 0.00000+ 0 5.12690- 1 1.00000- 1 5.12170- 1114914 57 4482
 2.00000- 1 5.11150- 1 3.00000- 1 5.09090- 1 4.00000- 1 5.06540- 1114914 57 4483
 5.00000- 1 5.03420- 1 6.00000- 1 4.98800- 1 7.00000- 1 4.93670- 1114914 57 4484
 8.00000- 1 4.88500- 1 9.00000- 1 4.82780- 1 1.00000+ 0 4.75090- 1114914 57 4485

     (MISSING LINES)

 0.00000+ 0 2.00000+ 7          0          0          1          2114914 57 4540
          2          1                                             114914 57 4541
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                      114914 57 4542
 2.29600+ 6 4.05800+ 6          0          0          1          8114914 57 4543
          8          2                                             114914 57 4544
 0.00000+ 0 4.17500+ 6          0          0          1         21114914 57 4545
         21          2                                             114914 57 4546
-1.00000+ 0 4.94800- 1-9.00000- 1 4.96380- 1-8.00000- 1 4.97600- 1114914 57 4547
-7.00000- 1 4.98790- 1-6.00000- 1 4.99820- 1-5.00000- 1 5.00650- 1114914 57 4548
-4.00000- 1 5.01400- 1-3.00000- 1 5.01890- 1-2.00000- 1 5.02280- 1114914 57 4549
-1.00000- 1 5.02470- 1 0.00000+ 0 5.02570- 1 1.00000- 1 5.02470- 1114914 57 4550
 2.00000- 1 5.02280- 1 3.00000- 1 5.01890- 1 4.00000- 1 5.01400- 1114914 57 4551
 5.00000- 1 5.00650- 1 6.00000- 1 4.99820- 1 7.00000- 1 4.98790- 1114914 57 4552
 8.00000- 1 4.97600- 1 9.00000- 1 4.96380- 1 1.00000+ 0 4.94800- 1114914 57 4553

     (MISSING LINES)

 0.00000+ 0 7.00000+ 6          0          0          1         21114914 57 4599
         21          2                                             114914 57 4600
-1.00000+ 0 4.93350- 1-9.00000- 1 4.95070- 1-8.00000- 1 4.96930- 1114914 57 4601
-7.00000- 1 4.98340- 1-6.00000- 1 4.99750- 1-5.00000- 1 5.00910- 1114914 57 4602
-4.00000- 1 5.01760- 1-3.00000- 1 5.02520- 1-2.00000- 1 5.03020- 1114914 57 4603
-1.00000- 1 5.03330- 1 0.00000+ 0 5.03430- 1 1.00000- 1 5.03330- 1114914 57 4604
 2.00000- 1 5.03020- 1 3.00000- 1 5.02520- 1 4.00000- 1 5.01760- 1114914 57 4605
 5.00000- 1 5.00910- 1 6.00000- 1 4.99750- 1 7.00000- 1 4.98340- 1114914 57 4606
 8.00000- 1 4.96930- 1 9.00000- 1 4.95070- 1 1.00000+ 0 4.93350- 1114914 57 4607
 0.00000+ 0 2.00000+ 7          0          0          1          2114914 57 4608
          2          1                                             114914 57 4609
-1.00000+ 0 5.00000- 1 1.00000+ 0 5.00000- 1                      114914 57 4610
                                                                  114914  0 4611
```

File 14 Photon Angular Distributions
(Tabulated Angular Distributions)

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

where $p_k(\mu,E)$ is the probability that an incident neutron of energy E will result in a particular discrete photon or photon energy continuum (specified by k and MT number) being emitted into unit cosine about an angle whose cosine is μ. Because the photon angular distribution is assumed to have azimuthal symmetry, the distribution may be represented as

$$p_k(\mu,E) = \frac{2\pi}{\sigma_k^{\gamma}(E)} \frac{d\sigma_k^{\gamma}}{d\underline{\Omega}} (\underline{\Omega},E)$$

where

μ = cosine of the reaction angle in the lab system.

E = energy of the incident neutron in the laboratory system, and

$\sigma_k^{\gamma}(E)$ = photon production cross section for the discrete photon or photon continuum specified by k, as given in either File 13 or in Files 2, 3, and 12 combined.

NC ≡ the number of partial distributions used to represent $f(E_\gamma \leftarrow E)$,

$q_j(E_\gamma \leftarrow E)$ ≡ the jth normalized partial distribution in the units eV^{-1}, and

$p_j(E)$ ≡ the probability or weight given to the jth partial distribution, $q_j(E_\gamma \leftarrow E)$.

*Note that the subscript k used in describing Files 12 and 13 has been dropped from $f(E_\gamma \leftarrow E)$. This is done because only one energy continuum is allowed for each MT number, and the subscript k has no meaning in File 15. It is, in fact, the NK^{th} subsection in File 12 or 13 that contains the production data for the continuum.

The structure of a section is

[MAT, 15, MT/ZA, AWR; b, b; NC, b]HEAD

<subsection for j = 1>

<subsection for j = 2>

For LF = 1, the structure of a subsection is

[MAT, 15, MT/b, b; b, LF=1; NR, NP/E_{int}/p_j(E)]TAB1

[MAT, 15, MT/b, b; b, b; NR, NE/E_{int}]TAB2

[MAT, 15, MT/b, E_1; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ → E_1)]TAB1

[MAT, 15, MT/b, E_2; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ → E_2)]TAB1

.

.

.

[MAT, 15, MT/b, E_{NE}; b, b; NR, NP/$E_{\gamma\ int}$/g(E_γ → E_{NE})]TAB1 .

.

.

.

<subsection for j = NC>

[MAT, 15, 0/ b, b; b, b; b, b]SEND .

```
9.22350+ 4 2.33025+ 2          0          0          1          0126115102 6646
0.00000+ 0 0.00000+ 0          0          1          1          2126115102 6647
         2          2                                             126115102 6648
1.00000- 5 1.00000+ 0 2.00000+ 7 1.00000+ 0                       126115102 6649
0.00000+ 0 0.00000+ 0          0          0          1          4126115102 6650
         4          2                                             126115102 6651
0.00000+ 0 1.00000- 5          0          0          1         31126115102 6652
        31          2                                             126115102 6653
0.00000+ 0 1.06000- 7 1.00000+ 5 5.49000- 7 2.00000+ 5 5.10000- 7126115102 6654
3.00000+ 5 3.43000- 7 4.00000+ 5 2.71000- 7 5.00000+ 5 2.29000- 7126115102 6655
6.00000+ 5 2.09000- 7 7.00000+ 5 2.13000- 7 8.00000+ 5 2.31000- 7126115102 6656
9.00000+ 5 2.70000- 7 1.00000+ 6 3.45000- 7 1.25000+ 6 4.25000- 7126115102 6657
1.50000+ 6 3.74000- 7 1.75000+ 6 3.10000- 7 2.00000+ 6 2.54000- 7126115102 6658
2.25000+ 6 2.34000- 7 2.50000+ 6 1.95000- 7 2.75000+ 6 1.54000- 7126115102 6659
3.00000+ 6 1.25000- 7 3.25000+ 6 9.45000- 8 3.50000+ 6 7.96000- 8126115102 6660
3.75000+ 6 6.69000- 8 4.00000+ 6 4.88000- 8 4.25000+ 6 4.46000- 8126115102 5661
4.50000+ 6 3.08000- 8 4.75000+ 6 2.44000- 8 5.00000+ 6 2.12000- 8126115102 6662
5.50000+ 6 3.93000- 8 6.00000+ 6 1.17000- 8 6.50000+ 6 1.10000- 9126115102 6663
7.00000+ 6 0.00000+ 0                                             126115102 6664
0.00000+ 0 1.30000+ 4          0          0          1         31126115102 6665
        31          2                                             126115102 6666
0.00000+ 0 1.06000- 7 1.00000+ 5 5.49000- 7 2.00000+ 5 5.10000- 7126115102 6667
3.00000+ 5 3.43000- 7 4.00000+ 5 2.71000- 7 5.00000+ 5 2.29000- 7126115102 6668
6.00000+ 5 2.09000- 7 7.00000+ 5 2.13000- 7 8.00000+ 5 2.31000- 7126115102 6669
9.00000+ 5 2.70000- 7 1.00000+ 6 3.45000- 7 1.25000+ 6 4.25000- 7126115102 6670
1.50000+ 6 3.74000- 7 1.75000+ 6 3.10000- 7 2.00000+ 6 2.54000- 7126115102 6671
2.25000+ 6 2.34000- 7 2.50000+ 6 1.95000- 7 2.75000+ 6 1.54000- 7126115102 6672
3.00000+ 6 1.25000- 7 3.25000+ 6 9.45000- 8 3.50000+ 6 7.96000- 8126115102 6673
3.75000+ 6 6.69000- 8 4.00000+ 6 4.88000- 8 4.25000+ 6 4.46000- 8126115102 6674
4.50000+ 6 3.08000- 8 4.75000+ 6 2.44000- 8 5.00000+ 6 2.12000- 8126115102 6675
5.50000+ 6 3.93000- 8 6.00000+ 6 1.17000- 8 6.50000+ 6 1.10000- 9126115102 6676
7.00000+ 6 0.00000+ 0                                             126115102 6677
0.00000+ 0 1.09000+ 6          0          0          1         32126115102 6678
        32          2                                             126115102 6679
0.00000+ 0 9.44000- 8 1.00000+ 5 4.88000- 7 2.00000+ 5 4.53000- 7126115102 6680
3.00000+ 5 3.05000- 7 4.00000+ 5 2.41000- 7 5.00000+ 5 2.04000- 7126115102 6681
6.00000+ 5 1.86000- 7 7.00000+ 5 1.90000- 7 8.00000+ 5 2.06000- 7126115102 6682
9.00000+ 5 2.40000- 7 1.00000+ 6 3.07000- 7 1.25000+ 6 3.76000- 7126115102 6683
1.50000+ 6 3.31000- 7 1.75000+ 6 2.62000- 7 2.00000+ 6 2.23000- 7126115102 6684
2.25000+ 6 2.08000- 7 2.50000+ 6 1.90000- 7 2.75000+ 6 1.66000- 7126115102 6685
3.00000+ 6 1.44000- 7 3.25000+ 6 1.22000- 7 3.50000+ 6 1.02000- 7126115102 6686
3.75000+ 6 8.92000- 8 4.00000+ 6 7.37000- 8 4.25000+ 6 6.62000- 8126115102 6687
4.50000+ 6 5.40000- 8 4.75000+ 6 4.71000- 8 5.00000+ 6 4.20000- 8126115102 6688
5.50000+ 6 7.27000- 8 6.00000+ 6 2.23000- 8 6.50000+ 6 1.35000- 8126115102 6689
7.00000+ 6 4.50000- 9 7.50000+ 6 0.00000+ 0                       126115102 6690
0.00000+ 0 2.00000+ 7          0          0          1         32126115102 6691
        32          2                                             126115102 6692
0.00000+ 0 9.44000- 8 1.00000+ 5 4.88000- 7 2.00000+ 5 4.53000- 7126115102 6693
3.00000+ 5 3.05000- 7 4.00000+ 5 2.41000- 7 5.00000+ 5 2.04000- 7126115102 6694
6.00000+ 5 1.86000- 7 7.00000+ 5 1.90000- 7 8.00000+ 5 2.06000- 7126115102 6695
9.00000+ 5 2.40000- 7 1.00000+ 6 3.07000- 7 1.25000+ 6 3.76000- 7126115102 6696
1.50000+ 6 3.31000- 7 1.75000+ 6 2.62000- 7 2.00000+ 6 2.23000- 7126115102 6697
2.25000+ 6 2.08000- 7 2.50000+ 6 1.90000- 7 2.75000+ 6 1.66000- 7126115102 6698
3.00000+ 6 1.44000- 7 3.25000+ 6 1.22000- 7 3.50000+ 6 1.02000- 7126115102 6699
3.75000+ 6 8.92000- 8 4.00000+ 6 7.37000- 8 4.25000+ 6 6.62000- 8126115102 6700
4.50000+ 6 5.40000- 8 4.75000+ 6 4.71000- 8 5.00000+ 6 4.20000- 8126115102 6701
5.50000+ 6 7.27000- 8 6.00000+ 6 2.23000- 8 6.50000+ 6 1.35000- 8126115102 6702
7.00000+ 6 4.50000- 9 7.50000+ 6 0.00000+ 0                       126115102 6703
                                                                  126115   0 6704
```

$$f(E_\gamma \leftarrow E) = \sum_{j=1}^{NC} p_j(E)\ q_j(E_\gamma \leftarrow E) \qquad (eV)^{-1} ,$$

where

$$\int_0^{E_\gamma^{max}} f(E_\gamma \leftarrow E)\ dE_\gamma = 1 \quad ,$$

$$\int_0^{E_\gamma^{max}} q_j(E_\gamma \leftarrow E)\ dE_\gamma = 1 \quad .$$

Thus,

$$\sum_{j=1}^{NC} p_j(E) = 1 \quad .$$

APPENDIX O

Format Differences Between Successive Versions of ENDF/B

Versions I and II

The following is a summary of the format differences between Version I and Version II ENDF/B data tapes. ENDF/B Version I was released in July 1968, and Version II was released in August 1970.

File 1 Changes

1. An index has been added to MT = 451. Each record in this index contains a file number (MF), reaction type number (MT), and the number of BCD card images required to specify the data for each section to be given for the material The number of entries in the index is given by NXC, which is the last binary record (sixth field for BCD card-image format) of the HEAD record. Each index entry is given in a CØNT record. These records immediately follow the Hollerith information.

2. The format for specifying induced reaction branching (MT = 453) has been extensively modified.

3. The format for specifying fission product yield data (MT = 454) has been modified to allow the specification of yield data for metastable states.

4. A new section has been defined to contain data for delayed neutrons from fission (MT = 455). See Section 5.1 of this report for details.

File 2 Changes

1. LRF, the test indicating the type of resolved resonance formula used, has been expanded to include

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

= 2, multilevel Breit-Wigner parameters are given;

= 3, R-Matrix (Reich-Moore) multilevel parameters are given (added);

= 4, Adler-Adler multilevel resonance parameters are given (added).

2. All materials will contain a File 2. For those materials for which resolved and/or unresolved are not given, File 2 will contain the effective scattering radius, AP. See Section 2.1 for details.

3. The LIS test has been removed. This means that the elastic scattering cross section always must be calculated, using the resolved or unresolved resonance parameters.

4. The constant C (used in calculating the penetration factor) has been replaced by a quantity AWRI. AWRI is defined as the ratio of the mass of a particular isotope to that of a neutron.

5. A new option for specifying unresolved resonance parameters has been added. It is indicated by the test LRF = 2. This means that all average resonance parameters (level spacing, the width of an unspecified competitive reaction, and the reduced neutron, radiation, and fission widths) may be given as a function of incident neutron energy. Energy-dependent parameters may be given for each ℓ - J state. See Section 7.3 for details.

File 3 Changes

1. The reaction Q-value has been defined as the kinetic energy (in eV) released by a reaction (positive Q-values) or required for a reaction (negative Q-values). The threshold energy (negative Q only) is given by

$$E_{th} = \frac{AWR + 1.0}{AWR} |Q|,$$

where AWR is the atomic weight ratio given on the HEAD record.

2. The maximum number of allowed energy points per section has been increased from 2000 to 5000.

3. An initial-state indicator has been added to the HEAD record. This will allow the inclusion of cross section data for metastable states and thus more than one section may be given for the same reaction type (MT number).

File 4 (No Changes)

File 5 Changes

1. The definition of LF = 3 (discrete energy loss law) was changed to read

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

where A = AWR and θ is the level excitation energy (positive value).

2. T and LT have been removed from the TAB1 records that contain p(E) for cases in which LF = 5, 7, or 9. A value, U, replaces T. U was introduced to define the proper upper limit for the secondary neutron energy distributions so that

$$0 \leq E' \leq E - U ,$$

where E′, E, and U are given in the laboratory system. Further, the normalization constants for LF = 7 and LF = 9 have been redefined to account for the use of U.

3. LF = 2, 4, 6, and 8 have been deleted.

All Files

1. Certain reaction type (MT) numbers have been changed (see Appendix B for definitions):

Old MT Number	New MT Number
5	51
6	52
7	53
8	54
9	55
10	56
11	57
12	58
13	59
14	60
15	91
27	No longer used
29	No longer used
51	61
52	62
53	63
-	-
-	-
-	-
80	90
109 (Not assigned)	109 (n,3α) cross section
455 (Not assigned)	455 Delayed neutrons from fission
700-799 (Not assigned)	700-799 Assigned (see Appendix B)

2. The format for specifying temperature-dependent data has been modified so that the data for the second (and higher) temperatures may be given at a lesser number of points than was given for the first temperature. See Appendix F for details.

Versions II and IV

The following is a summary of the format differences between Versions II and IV ENDF/B data tapes. ENDF/B Version IV was released about February 1975.

General

1. The energy range for all general-purpose materials is 10^{-5} eV to 20 MeV.

File 1

1. The formats for specifying radioactive decay were changed. Section MT = 453 was changed to include only production of radioactive nuclides, and Section MT = 457 was added to include radioactive decay data.
2. Provision was made for supplying data for the number of prompt neutrons per fission ($\bar{\nu}p$) in added Section MT = 456.

File 3

1. The energy mesh for the total cross section must include the energy meshes for partial cross sections.
2. Time sequential (n,2n) reactions are described by using Sections MT = 6-9 and MT = 46-49.
3. An LR flag was added to designate x in the (n,n'x) reactions when x is not a photon. In this case, the temperature field S (formerly T) is used to designate the Q-value or energy difference of the combined reactions.

4. Sections MT = 718, 738, 758, 778, 798, and MT = 719, 739, 759, 779, 799 are redefined to describe continuum levels for (n,x') reactions. MT = 718 describes the (n,p_c') continuum cross sections as part of the (n,p) cross section and should be included in the total cross section. MT = 719 is used to describe a continuum cross section for exit protons whose cross section is already represented in the total cross section by other reaction types.

APPENDIX P

Summary of Important ENDF Rules

General

1. Cross sections for all significant reactions should be included.

2. The data in ENDF are specified over the entire energy range 10^{-5} eV to 20 MeV. It should be possible to determine values between tabulated points with use of the interpolation schemes provided.

3. All cross sections are in barns, all energies in eV, all temperatures in degrees Kelvin, and all times in seconds.

4. Summary documentation and unusual features of the evaluation should appear in the File 1 comments.

5. Threshold energies and Q-values must be consistent for all data presented in different files for a particular reaction.

File 2 - Resonance Parameters

1. Only one energy region containing resolved resonance parameters can be used, if needed.

2. The cross section from resonance parameters is calculated only within the energy range EL to EH, although some of the resonance parameters may lie outside the range.

3. Every ENDF Material has a File 2 even if no resonance parameters are given in order to specify the effective scattering radius.

4. In the unresolved resonance region interpolation should be done in cross section space and not in unresolved resonance parameter space. Any INT is allowed.

5. The Breit-Wigner single-level or multilevel formalisms should be used in the resolved resonance region unless experimental data prove that use of the other allowed formalisms is significantly better.

ile 3 - Tabulated Cross Sections

1. All File 3 data are given in the laboratory system.

2. The total cross section MT = 1 is the sum of all partial cross sections nd has an energy mesh that includes all energy meshes for partial cross sections.

(Exceptions: MT = 26, 46-49, 719, 739, 759, 779, and 799 are not included in the MT = 1.)

3. The following relationships among MT numbers are expected to be atisfied if data are presented:

1 = 2 + 3

3 (or 1 - 2) = 4 (or 51+...91) + (6+...9+16) + 17 + 18 (or 19+...21+38) + (22+...25) + (28+...37) + (102+...114)

4 = sum (51+...91)

18 = sum (19+...21) + 38

101 = sum (102+...114)

103 = sum (700+...718)

104 = sum (720+...738)

105 = sum (740+...758)

106 = sum (760+...778)

107 = sum (780+...798)

4. Threshold reactions begin at zero cross sections at the threshold energy.

Files 2 and 3

1. If there are resonance parameters in File 2, there are contributions to the total (MT = 1) and scattering (MT = 2) cross sections and to the fission (MT = 18) and capture (MT = 102) cross sections if fission and capture widths are also given. These must be added to the File 3 Sections MT = 1, 2, 18, and 102 over the resonance region in order to obtain summation values for these cross sections.

2. The cross sections in File 3 for MT = 1, 2, 18, and 102 in the resonance region are used to modify the cross section calculated from the resonance formalisms, if necessary. The File 3 "background" may be positive or negative or even zero if no modifications are required. The summation cross section (File 2 + File 3) should be everywhere positive.

3. Double-value points (discontinuities) are allowed anywhere but are required at resonance region boundaries. A typical situation for MT = 1, 2, 18, and 102 in File 3 is a tabulated cross section from 10^{-5} to 1 eV, tabulated "background" to the cross sections calculated in the resolved resonance region between EL1 and EH1, tabulated "background" to the cross sections calculated in the unresolved region between EL2 = EH1 and EH2, and tabulated cross sections from EH2 to 20 MeV. Double-value points occur at EL1, EL2, and EH2.

4. The tabulated "background" used in File 3 to modify the cross sections calculated from File 2 should not be highly structured or represent a large fraction of the cross sections calculated from File 2. It is assumed that the "background" cross section is assumed to be at 0° Kelvin. (The "background" cross section is usually obtained from room temperature comparisions, but this should be unimportant if the "background" cross section is either small or slowly varying).

5. The generalized procedure for Doppler-broadening cross sections from Files 2 + 3 is to generate a pointwise cross section from the resolved resonance region on an appropriate energy mesh at 0°K and add it to File 3. This summation cross section can be kernel-broadened to a higher temperature.

File 4 - Neutron Angular Distributions

1. Only relative angular distributions, normalized to an integrated probability of unity, are given in File 4. The differential scattering cross

section in barns per steradian is determined by multiplying File 4 values by the File 2 + File 3 summation scattering cross section σ_s divided by 2π.

2. Discrete channel angular distributions (e.g., MT = 2, 51-90,701...) should be given as Legendre coefficients in the center-of-mass system, with a maximum of 20 higher order terms, the last being even, in the expansion. If the angular distribution is highly structured and cannot be represented by a Legendre expansion, a tabular angular distribution in the CM system must be given.

3. When the elastic scattering is represented by Legendre coefficients, an energy-independent transformation matrix must be given to perform a CM to laboratory conversion.

4. Angular distributions for continuum and other reactions must be given as tabulated distributions in the Lab system.

5. The angular distribution, whether specified as a Legendre expansion or a tabulated distribution, must be everywhere positive.

6. Angular distribution data should be given at the minimum number of incident energy points that will accurately describe the energy variation of the distributions.

File 5 - Secondary Energy Distribution

1. Only relative energy spectra, normalized to an integrated probability of unity, are given in File 5. All spectra must be zero at the end points. The differential cross section in barns per eV is obtained by multiplying the File 5 values by the File 2 + File 3 cross section times its multiplicity [2 for the (n,2n) reaction].

2. While distribution laws 1, 3, 5, 7, 9, and 10 are allowed, distribution laws 3 and 5 are discouraged but can be used if others do not apply.

3. The sum of all probabilities for all laws used for a particular reaction must be unity at each incident energy.

4. The constant U must be specified, where applicable, to limit the energy range of emitted spectra to physical limits.

File 6

1. File 6 formats and procedures have been specified, but this File is not currently used in ENDF.

APPENDIX Q

Maximum Dimensions of Important ENDF Parameters

File	Section	Variable	Max	Definition
1	451	NCD	294	Card images containing Hollerith information
	452	NC	4	Polynomial terms in expansion of $\bar{\nu}$
	455	NCD	4	Polynomial terms in expansion of $\bar{\nu}_d$
	456	NCP	4	Polynomial terms in expansion of $\bar{\nu}_p$
2	151	NER	2	Energy ranges
	"	NIS	10	Isotopes
	"	NRS	500	Resonances per ℓ-state
	"	NLS	3	ℓ-states
	"	NE	250	Energy mesh in unresolved region
	"	AMUN	2	Degrees of freedom for neutron widths
	"	AMUF	4	Degrees of freedom for fission widths
3	All	NR	100	Interpolation ranges (< 20 usual)
	"	NP	5000	Mesh size
4	2	NL	21	Side dimension of transformation matrix
	All	NM	20	Higher order Legendre terms
	"	NE	500	Incident energies
	"	NP	101	Angular mesh size
5	All	NE	200	Incident energy mesh
	"	NF	1000	Final energy mesh
7	4	NS	3	Nonprincipal scattering atoms

APPENDIX R *

Trial Format for Non-Neutron Data

Purpose

It is desirable to expand the ENDF/B formats to include reactions relating to neutron physics and other applications such as fusion and space shielding studies. The major types needed are charged-particle-induced reactions arising in neutron source reactions, other inverse neutron reactions, and reactions arising from intense charged-particle fluxes produced by reactor, outer space, and common accelerator sources. It is also necessary to specify which secondary particle is designated in angular and energy distributions.

The solution proposed allows these data to be included with the following advantages:

1. No changes are required in existing neutron-induced data formats.

2. The same MAT number is used for both neutron and non-neutron-induced data. This is desirable, since ENDF/B should contain only one MAT per target material.

3. Where appropriate, the same MT numbers are used for both neutron and non-neutron data.

4. No changes are required in codes processing the neutron data files. If the non-neutron data files are merged with the neutron files, only minimal changes are required in some of these peripheral codes.

Proposed solution

Files 62-67 and 72-76 would be used in analogy with Files 2-7 and 12-16 (add 60 to the present MF numbers). Formats for data uncertainties will be contained in Files 82-87 and 92-96, in analogy with Files 32-27 and 42-46.

* Appendix R was previously approved as an ENDF format

1. MT numbers plus new ones as required would designate the exit channel(s). The exceptions would be MT = 1, 2, 3, and 4 where the exit channel would be taken to be the same as the entrance channel.*

2. Field 6 of the HEAD record is blank for all Files according to ENDF 102, Volumes 1 and II. This field will specify the ZA as (1000*Z+A of the incident particle (IZA) as a fixed point number. An IZA of 0 will designate a neutron-induced reaction. Other IZA's are

Incident Particle	IZA
g	1111 (defined)
B-	-1000
B+	1000
p	1001
d	1002
t	1003
^{3}He	2003
alpha	2004
^{12}C	6012
^{16}O	8016
^{32}S	16032

IZA's for molecules and strange particles can be invented as needed.

* MT = 4 would continue to equal the sum of MT = 51, 52..., 91. New MT's 800-819 would be defined to describe (x,n_0), (n,n_1)...(x,n_{18}), (x,n_c), where x represents the incident particle. MT = 15 would be used to define the total (x,n) cross section. Thus, for proton-induced reactions field 6 of the HEAD record would contain 1001 and (p,p') total would be described by MT = 4; (p,p') to discrete states and the continuum by MT = 51-91; (p,n) total by MT - 15; and (p,n') to discrete states and the continuum by MT = 800-819 cannot be used for neutron-induced reactions.

3. An MT number is repeated for as many sections as there are incident particles for which data are specified. The convention is followed that MT numbers are arranged in order of ascending IZA. ENDF/B tapes that do not contain merged neutron and non-neutron files can be supplied.

4. According to ENDF 102, field 5 of the HEAD record is blank for all files except Files5, for which field 4 is blank. It is proposed that this field contain JZA, the (1000*Z+A) of the particle designated in the angular or energy distribution following the same code as described above.

5. The structure of File 1 would be changed to include IZA and JZA in the dictionary. The CØNT record for a non-neutron data section would contain JZA and IZA as floating-point numbers in the first and second fields, respectively, to maintain the order in which they appear on the HEAD card of each section.

6. The structure of File 62 could be constructed in analogy with neutron-induced widths, with the incident particle designated as in item 4 and resonance energies, widths, and other data defined by new formats and procedures to be specified at a later time.

7. Photon-induced nuclear reactions are to be handled in the same way as other non-neutron-induced reactions. The word atomic should be added to the definitions for MT = 501, 502, and 504 and MF - 23-26. MT = 518, 532 and 533 should be cancelled.

Examples

The structure of File 1 of a MAT containing File 3 sections for (n,np), (n,γ), and File 63 (p,γ) data and a File 4 section for (n,np) outgoing portion angular distribution data only is

[EA,AWR,LRP,LFI,0,NXC]HEAD

[0.0,0.0,LDD,LFP,NWD,0/H(N)]LIST

[MAT,1,451,0.0,0.0,1,451,NC1,0]CØNT

[MAT,1,451,0.0,0.0,3,28,NC2,0]CØNT

[MAT,1,451,0.0,0.0,3,102,NC3,0]CØNT

[MAT,1,451,0.0,1001.0,63,102,NC4,0]CØNT

[MAT,1,451,1001 0,0,4,28,NC5,0]CØNT

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

The structure of File 63 containing (p,γ) data is

[MAT,63,MT-102/EA,AWR,LIS,0,IEA=100]HEAD

[MAT,63,MT-102/T,Q,LT,0,NR,NP/E_{int}/σ(E)]TAB1

[MAT,63,0/0.0,0.0,0,0,0,0]SEND

The structure of File 4 containing (n,np) and File 64 containing (p,pα) ngular distribution for the emerging proton is

[MAT,4,MT=28,/ZA,AWT,LVT,LTT,JZA=1001,0]HEAD

.....

.....

.....

[MAT,4,0/0.0,0.0,0,0]SEND

[MAT,64,MT-112/ZA,AWR,LVT,LTT,JZA=1001,IZA=1001]HEAD

.....

.....

.....

[MAT,64,0/0.0,0.0,0,0,0,0]SEND

The differential elastic distributions would be rationed to Rutherford scattering. Angular distributions for inelastic scattering and reaction data integrate to unity.

The structure of File 5 containing both (n,np) and File 65 containing energy distribution data for the emerging proton is

[MAT,5,MT=28/ZA,AWR,0,JZA=1001,NK,0] HEAD

.....

.....

.....

[MAT,5,0/0.0,0.0,0,0,0,0] SEND

[MAT,65,MT=112/ZA,AWR,0,JZA=1001,NK,IZA=1001] HEAD

.....

.....

.....

[MAT,65,0/0.0,0.0,0,0,0,0] SEND

The structure of File 6 would follow the rules for File 4.

Limitations

Simple additional tests would be required for DICTION and RIGEL and dictionary expansions made to the display codes. RIGEL can be modified by the NNCSC to output only neutron data files or only neutron and gamma files, etc. There would be no limitation on the user's receiving an ENDF tape containing only neutron-induced reaction and gamma-gamma interaction data. This will probably be the normal distribution recommended by CSEWG. However, for those users requiring them, other types of data may be included in the ENDF system with a minimum of modification to formats and processing codes.