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SOME NOTES ABOUT THE NUCLEAR DATA LIBRARIES

COMITATO NAZIONALE PER LA RICERCA E PER LO SVILUPPO DELL'ENERGIA NUCLEARE E DELLE ENERGIE ALTERNATIVE

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FOREWORD

This paper is the body of a series of lectures presented at the IAEA Winter College on Nuclear Physics and Reactors held in Trieste from 25 January to 19 March 1982.

The author thinks it useful to re-present the lessons in report form in order to give the information greater diffusion.

The paper is mainly devoted to people who are at their first approach to the fascinating world of Nuclear Data such as young scientists or students. It does not intend to picture the status of the art on the matter, but rather to give a primer to the beginners in order to allow an approach in the proper sense.

Many thanks are due to Professor Mario Motta who, in my absence, carefully presented these lessons at the IAEA Winter College.

G.C.P.

DEFINITIONS

A library is a collection of data evaluations stored in a computer readable format that can be used as input into cross section processing programmes. Library data are subdivided into materials.

A material is defined either as 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. Each evaluated set of data for a material is assigned an identification number which is unique in that library. Each material is subdivided into sections which contain the same type of data.

A type of data defines the data for a certain category of information. Integral and differential data are of different categories. Resonance parameters (if any) constitute a category. Each type is assigned an identification number: not all materials have all types of data, but most have. Each type of data is subdivided into subsections which are called reactions.

A reaction generally refers to a specific neutron-nucleus interaction mechanism, but occasionally it indicates that a particular type of information is given. Total cross section is a reaction, but also the mean number of neutrons per fission is referred to as a reaction. The general information section (if any) is a reaction, too. Each reaction is assigned an identification number; a reaction may belong to different types of data: this is

the case of the inelastic process which appears as integral cross sections, secondary neutron angular distributions and secondary neutron energy distributions.

Within each reaction, data can be given in a variety of forms depending on the data type and the reaction itself. Integral neutron cross sections are often given in tabular form, i.e. with pairs of cross section vs. energy. Angular differential cross sections can be given both in tabular and parametric (Legendre polynomial) form. Resonance parameters can represent the cross section curves using different approaches.

It is generally agreed that all numerical quantities are given in increasing order of magnitude. Within some subsets a library is given:

a) in increasing order of the material identifier;
b) in increasing order of the data type identifier;
c) in increasing order of the reaction type identifier;
d) in increasing order of energy.

Nuclear Data Libraries are generally contained in magnetic tape support, the size of each library being so large that the use of other supports would be very expensive. Two basic formats exist for the libraries:

a) card image (decimal) data representation, which is suitable for immediate understanding of the contents and easy communication between different computers;

b) binary (machine dependent) representation, which speed up the run time of processing codes.

Most existing Libraries have the same Reaction Identification Number sequence, but not identical Data Type Identification numbers: the table listed in the following (which is particular of ENDF/B) is here presented as an example.

A common peculiarity of the Nuclear Data Libraries is that all use six field records in the card image format. This is due to some FORTRAN features: all Libraries have in fact been designed keeping FORTRAN processing and maintenance codes in mind.

DATA TYPE IDENTIFICATION NUMBERS

1. General Information
2. Resonance Parameters
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

REACTION CLASS DESCRIPTION

Range	
1-100	Reaction types in which secondary particles of the same type as the incident particles are emitted.
100-150	Reaction types in which no secondary particles of the same type as the incident particles are emitted.
201-450	Quantities derived from the basic data.
451-699	Miscellaneous quantities.
700-799	Excitation cross-sections for reactions that emit charged particles.

REACTION IDENTIFICATION NUMBERS

1. Total cross section (sum of all partial cross sections)
2. Elastic scattering cross section
3. Non-elastic cross section (1 - 2)
4. Total inelastic cross section (sum of 51 to 91)
16. (n,2n) cross section
17. (n,3n) cross section
18. Total fission cross section (sum of 19, 20, 21)
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'p) cross section
23. (n,n'3) cross section
28. (n,n'α) cross section
51. (n,n') to the first excited level
52. (n,n') to the second excited level

.....

90. (n,n') to the 40th excited level
91. (n,n') to the continuum
101. Neutron disappearance
102. (n,γ) cross section (radiative capture)
103. (n,p) cross section
104. (n,d) cross section
105. (n,t) cross section
106. (n,He-3) cross section
107. (n,α) cross section
151. Resonance parameters
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 energy decrement for elastic scattering

451. Alphanumerical information (Data type 1 only)

452. $\bar{\nu}$, the average number released per fission event

457. Radioactive decay data

459. Fission product yield data.

THE EVALUATED NUCLEAR DATA FILE (ENDF)

Country of origin: USA

Special features:

1) General

a) Data are specified over the entire range 10^{-5} eV to 20 MeV.

b) All energies are in eV, all cross sections in barns. Tabulated cross sections are given in the laboratory system.

c) The beginning of each material consists of a section of comments on the evaluation methods and references used as well as a few characteristic quantities. (Resonance integrals, 2200 m/sec cross sections, etc.).

d) An index of contents is given for each material (but not for an entire tape).

e) Five types of data can be given for each material. These are:

I General (Comments, radioactive decay schemes, fission product yields, number of neutrons produced in fission,etc.)

II Neutron data (cross sections, resonance parameters, secondary angle and energy distributions, thermal scattering law data, etc.)

III Photon production data

IV Photon interaction data

V Data Covariance Matrices

f) Either parametric representation of the data (resonance parameters, polynomial coefficients) or pointwise tabulations can be used; no redundancy in representation is, however,

allowed.

g) When a resonance parameter representation is used, up to two adjacent resonance regions can be defined (for resolved and unresolved resonance treatment). In the resolved region either single or multilevel formalisms are allowed (Breit-Wigner, Reich-More or Adler-Adler). Cross sections in the resonance range are obtained by calculating the resonance contribution and adding a pointwise "background" cross section.

h) Pointwise data tabulations are supplied with appropriate interpolation criteria. Each cross section can be given on a different energy grid. However, reactions which are the sum of partial cross sections must include all energy points given for their component reactions.

i) Angular distributions may be specified in tabular or polynomial form, the latter being the more usual.

j) For secondary energy distributions too, parametric representation is preferred to the tabular one.

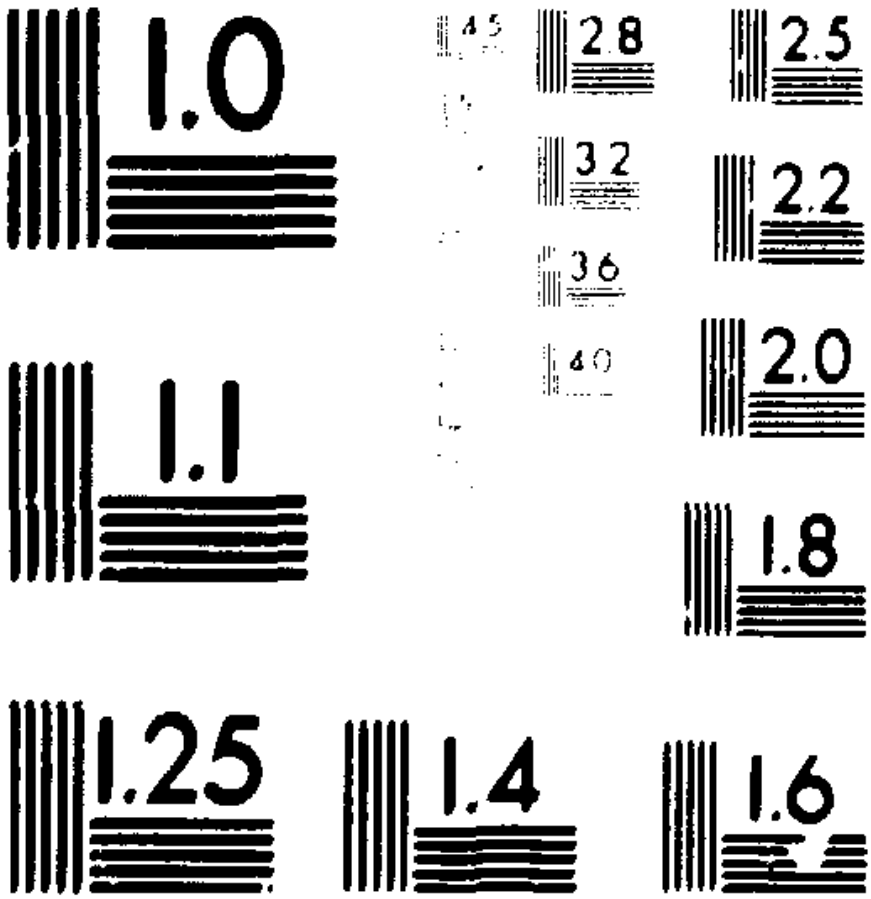
2) Card Image Format

a) The BCD card image format consists of six 11-column data fields per record followed by four identification and book-keeping fields. These are: Material identifier in columns 67-70, data type identifier in 71-72, reaction type number in 73-75 and material sequence number in 76-80.

c) The data for a material is divided into parts called "files" according to the type of information. The "files" are further divided into sections according to reaction type.
Materials, files and sections are separated by special cards.

3) Maintenance

a) Manipulative codes for checking, correcting, plotting, listing and retrieving the data are released together with the data file.



THE KARLSRUHE EVALUATED NUCLEAR DATA FILE

(KEDAK: KErneDAten Karlsruhe)

Country of origin: Federal Republic of Germany

Special features:

1) General:

a) Data are specified over the range 10^{-3} eV to 15 MeV.

b) All energies are in eV, all cross sections in barns.

c) In the resolved resonance region pointwise data as well as resonance parameters are given. The two representations are not necessarily consistent.

d) Breit-Wigner resonance parameter specifications only are allowed.

e) The linear-linear interpolation has been adopted throughout the file.

f) A limited number of format conventions for representation of secondary energy distributions is available.

g) Neutron data only are given.

2) Card Image Format

a) Alphanumerical information is included.

b) An index is presented for each tape and each material, but the total number of records is not given.

c) The order in which the reaction types appear within each material does not follow the increasing value of the numerical identifiers.

d) The card image format allows six 12-column fields per record plus a serial identifier of the material (2 digits), the reaction serial (2 digits) and four digits as record counter.

3) Maintenance

a) Manipulation codes (which do not form part of each release) are provided for listing and updating purposes.

THE LLL EVALUATED NUCLEAR DATA LIBRARY (ENDL)

Country of origin: USA

Special features

1) General

a) Data are specified over the entire range 10^{-4} eV to 20 MeV.

b) All energies are in MeV, all cross sections in barns.

c) Neutron, photon and charged particle induced reactions are included.

d) The method of interpolation to be used with each type of data is the same for all data of that type (e.g. neutron cross sections are given in sufficient detail to permit linear-linear interpolation).

e) With the only exception of Legendre coefficients for angular distributions, parametric representations are not used. Thus the library does not include resonance parameters.

2) Card Image Format

* Apart from the LLL, the ENDL is used in the version which has been translated into ENDF/B format by LTOE, an "ad hoc" code written for the convenience of those who wish to use ENDL data but who are more familiar with ENDF/B formats and procedures.

THE RECOMMENDED NUCLEAR DATA LIBRARY FOR REACTOR CALCULATIONS (SOKRATOR)

Country of origin: USSR

Special features

1) General

a) All energies are in MeV, all cross sections in barn.

b) An index of contents is given at the beginning of each material.

c) Format for neutron-induced reactions only are provided.

d) Besides parametric and tabular, also subgroup representation is included, which is particular to this Library.

e) All three data representations can be included in the same data type.

f) Square root scale interpolation is allowed.

g) Resonance parameters are given, but Breit-Wigner formula only is provided.

h) Alphanumerical information is not included in the library.

2) Card Image Format

a) The file is given in a card image format with six 11-column fields per record, separated by blank columns. Service flags for identification purposes are given in columns 73 to 80.

3) Maintenance

Checking code provided.

THE UNITED KINGDOM NUCLEAR DATA LIBRARY (UKNDL)

Country of origin: United Kingdom

Special features

1) General

a) Data are specified over the entire range 10^{-5} eV 15 MeV.

b) All energies are in MeV, all cross sections in barns.

c) In general, pointwise representation of the data is preferred to the parametric form.

d) Resonance parameters are collected in separate files and are not included in the normal release.

e) An index of the contents is presented at the beginning of each tape and, in addition, at the beginning of each material within a tape, so that the card count of each material and each reaction is known in advance.

f) Interpolation criteria are uniquely defined over the entire energy range both for the cross-sections (log-log) and for the angular distributions (linear-linear).

g) All the reactions for a given material are given at the same energy points.

h) Angular distributions of scattered neutrons (normalized to 1) are sometimes given rangewise as far as the energy dependence is concerned.

i) Secondary energy distributions are mostly given in tabular form.

j) Reaction cross sections are immediately followed by the pertinent angular and energy distributions,

if any.

1) Alphanumerical information is not included.

2) Card Image Format

a) The file is given in a card image format with six 11-column fields per record, separated by blank columns. The materials identification (Data File Number) appears (mod. 1000) in columns 73 to 75 of the label field; the section number, which is indexed in section 0 to the class of data, appears in columns 76 and 77, and the serial number (mod. 1000) of the card within that section is given in columns 78to 80.

3) Maintenance

a) Manipulation codes exists which perform all tasks necessary to maintain a library, though they do not form part of each release.

MAINTENANCE CODES

The maintenance of a Nuclear Data Library is one of the important chapters in its history. As well as each collection of information, a Library of Nuclear Data should be:

- checked
- updated
- retrieved
- displayed

In addition, due to the nature of the contents, the possibility of a point-by-point and integral quantities calculation should be available.

The checking includes a two-step process:

a) consistency check;

b) physical check.

In general, the first is referred to clerical errors which may arise from the physical nature of data (e.g. a cross section cannot be negative) or from the Fortran format (e.g. an exponent E+02 not right-adjusted leads to a 10^{20} quantity). The latter is more strictly concerned with the physics of data (e.g. the total cross section must be the sum of all partial reactions) or with its representation (e.g. a tabular angular distribution should extend from -1 to +1 cosine values). A checking code should, in addition, provide the reading of the data to be checked also when the format of some records is completely altered by manipulation errors.

Updating includes the possibility of:

- inserting
- deleting
- altering

information. It should be noted that inserting and deleting can modify the number of items included in some sets or subsets; in this case the indexes and tables of contents must be updated. Library updating occurs periodically when a sufficient number of modifications have been collected. A new release should be issued when completely new data from experiment or model calculations become available.

Retrieving is a necessary step when library data are to be processed or when portions of the library are to be transmitted outside. In addition, processing codes generally include tape scanning to search for some material, but when more than one material is requested and they reside on different tapes, they are to be merged together before the processing code reads them. Also retrieving a few items of a library may be useful.

Displaying a library can be done (mainly) in two ways:

- interpreted list;
- graphical output.

Interpreted lists can give a key to the library reading and are suitable for the retrieval of some usually required information such as e.g. the resonance parameter. Its disadvantage is that, in general, a large amount of paper with subsequent storage problems is required.

More compact is the graphical representation, which can be useful in studying the shape of some reactions, in an eye check of clerical errors and in a fast (but approximate) comparison between data from different sources.

Other codes allow the calculation of resonance integrals, 2200 m/sec cross sections, resonance peaks and other quantities which are often required.

Finally, when binary data representation is allowed, decimal to binary, and viceversa conversion programmes exist.

THE NEED CONVERSION BETWEEN DIFFERENT FILES

The need to translate all or part of one Nuclear Data Library from one into another may arise from different requirements.

a) When an Institute has adopted for its purpose a particular Nuclear Data Library, all maintenance and processing codes are, in general, devoted to that Library format. While users are gaining experience in using the various codes, their knowledge goes ever more deeply into the format. Thus, when a new Library becomes available, it generally seems the case to translate it into the home format, for comparison and/or improvement purposes, rather than use the new Library with its own codes.

b) Some formats are gaining more general adoption with respect to others, so as to be used as transmission format between different Institutes. Conversion into that format should then be available.

c) No data exist in a given Nuclear Data Library for some material, while a different one has.

d) Format conversion sometimes implies only the fact of translating data from parametric to pointwise representation.

Format conversion involves not only a change in the structure of a Library, but also in the logic configuration. A striking example of this assessment is the presence or absence of resonance

parameters, which implies the use of diferent approaches to reactor physics problems. In addition, when a material has been translated into a new format, comparison between the results of the processing codes of either formats (source and object) may lead to shocking results. When this happens, is the difference due to the translating code, to the procesing codes ot are both steps responsible ?

Another problem is the presence of some data for which no provisions are made in the object format or, the worst case, the absence in the source library of data needed in the object. A one-to-one correspondence is in this case highly desirable.

Codes exist which perform "good" conversions of data from one source format to another, but their results should be handled with care and, in general, amended from slight inconsistencies and completed with some kind of data.

The object of complete automation could probably be achieved provided data are tested for consistency errors and, mainly, for physical errors. Integral quantities, such as averages based on different functions should, in addition, be calculated from the original as well as the translated data for comparison. However, the best approach to conversion is to combine the use of a translation code with a certain amount of evaluation effort.

A limited number of codes exists with accepts several formats: its philosophy is structuring the data internally in a format which exists only for the time the code is in the computer. These programmes eliminate the ambiguity of using different

algorithms in processing the same data either in the original or in the translated version. They seem to be a useful tool during the data evaluation process, in checking and comparing data.

THE POINTWISE REPRESENTATION

The format adopted in the Nuclear Data Libraries should be sufficiently flexible to allow the inclusion of many different kinds of reaction data, both in point-by-point tabulations and in parametric forms. The first may include not only the simplest cross section vs. energy sample, but also three dimensional arrays as in the case of temperature dependence.

The parametric form includes a large variety of ways to represent data, such as resonance formulas, Legendre polynomials, analytical expressions such as Maxwell, Watt and Klein-Nishima formulas.

Although, sometimes, one can freely choose between the pointwise and parametric forms (as in the case of neutron angular distributions), in general the parametric form may have physical significance and may be used to correlate the results of a number of separate experiments. Examples are:

a) the resonance parameters, which may be derived from measurements;

b) the parameters of statistical distributions of resonance widths and spacings, which can be used to convey not only the average cross sections but also their probable fluctuations.

Use of parametric forms can bring about a great saving of storage space in the data library. Thus, representing the cross sections in the resolved resonance region accurately sometimes requires several thousand tabular points for each cross section at a well-defined temperature, whereas

a set of some hundreds of parameters would suffice to convey all the information needed to generate the cross sections at some required temperature. On the other hand, problems may arise which conflict with the need to reduce the computer time required for the generation of Doppler-broadened cross sections from resonance parameters.

A considerable saving of computer time can be achieved through the following procedure: by using a Nuclear Data Library with resonance parameters one may generate a certain number of libraries of pointwise Doppler-broadened cross sections, say at 300 °K, 900 °K, 1200 °K, for use in processing codes with the requested group scheme. Taking care to generate for each reaction a sufficient number of energy points so that the linear-linear interpolation law can be used, additional computer time saving in processing code running can be obtained.

Where resonance data are given, processing codes should normally join the resonance region to the continuum region by addition.

The pointwise section may have a "background" which should also be added to the resonance parameter computed cross section. This background generally has no physical meaning, but its aim is to obtain a correct cross section curve taking the following into account:

a) sometimes the resonance data are originated from different experiments and a "correction" should be made in order to homogenize the resulting cross section; or

b) a single level Breit-Wigner formula is used in the processing codes instead of a multi-level

one. This is, in fact, very time consuming so it is convenient to calculate (at the time when the evaluation is generated) both the MLBW and SLBW cross section curve once, to subtract the second from the first and assign the difference as background in the pointwise section, so that a SLBW calculation can be made by processing codes.

FROM THE NDL TO THE POINTWISE CROSS SECTION

Processing of Nuclear Data Libraries involves computation of the following basic quantities:

$$\int \sigma \phi \, dE \quad ; \iint \sigma \phi K \, dE' dE$$

where:

σ , ϕ are functions of the neutron incident energy and K is a transfer kernel function of E and of the outgoing energy E'. The second quantity is always very hard to compute in terms of numerical accuracy and computer time.

The first can take advantage of some simple expressions for ϕ (e.g. $\phi(E)$=constant or $\phi(E)=1/E$) in order to perform fast semianalytical computations. When σ is given in parametric form (resonance region) a full analytical computation of the resonance integral is allowed. Problems arise when σ is given in parametric form and ϕ is point-by-point or when σ is pointwise and ϕ assumes some complicated form. On the other hand the best situation occurs when any point in a reaction can be predicted from the adjacent ones by means of linear-linear interpolation; and possibly, ϕ is a simple expression. This latter opportunity does not often happen, while it is always possible to transform cross section data in such a way that they can be interpolated on a linear-linear basis, even though the source of the data are resonance parameters. It should here be remembered that the best representation of cross sections in the resolved resonance region is obtained through a log-log interpolation; nevertheless, by introducing several intermediate points, a linear-linear interpolation to a given accuracy degree can be

used.

Modern processing codes work on this basis.

Care should be taken in generating the resolved resonance region energy grid to avoid:

a) the curve not be described to a required accuracy degree where the shape is awkward to follow;

b) an excessive number of points being generated where the curve is smooth.

Algorithms have been designed in order to follow the shape of the cross section with a certain degree of accuracy, i.e. by increasing the number of energy points in the regions where the cross section curves rapidly vary, and decreasing it when a predetermined interpolation law can approximate them.

The algorithms generally consist of the following steps (see figure in the next page):

a) a set of initial energy "node points" is selected; node points are set at the peak of the resonances and at some optional energy;

b) the cross section is calculated at the node points E_1 and $E_2 (E_2 > E_1)$;

c) and additional cross section value S_3 is computed at the midpoint E_3 ,

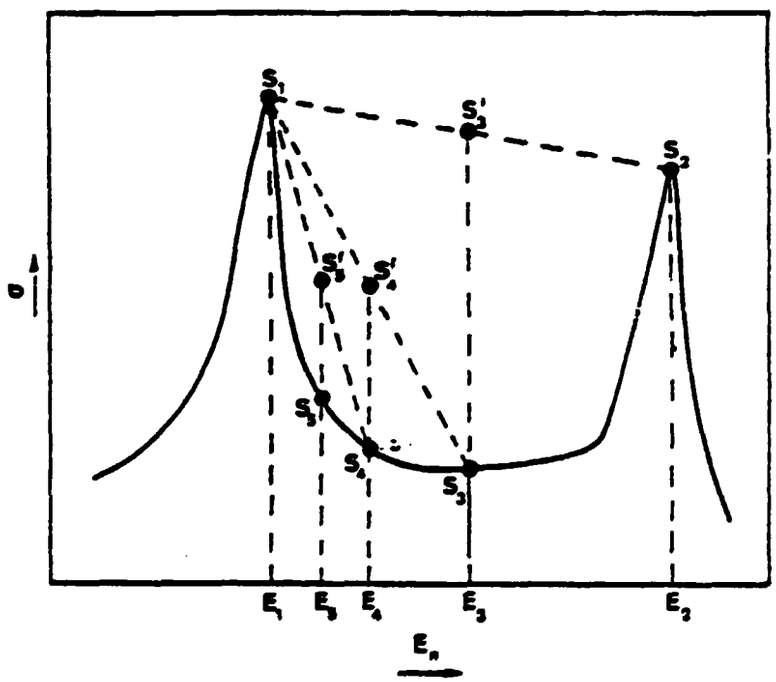
$$E_3 = \tfrac{1}{2}(E_1 + E_2) \quad ;$$

d) the value of the approximate cross section S'_3 at the mid-point is obtained by a given interpolation law and compared with the exact value S_3 at the same point;

e) if the fractional difference at the mid-point is no larger than the convergence criterion, i.e. if:

$$\frac{|S_3 - S'_3|}{S_3} < \varepsilon \qquad (1)$$

occurs, the interval is assumed to have converged and the process is repeated with the next pair of nodes;



If $\frac{|S_5 - S'_5|}{S_5} \leq \varepsilon$, S_5 is accepted and the range (E_4, E_3) is investigated.

f) if the fractional difference at the mid-point is larger than the convergence criterion, i.e. when (1) does not occur, the mid-point is defined as a new node and the process is repeated in the halved intervals.

Some very sophisticated codes have, in addition, an algorithm which further reduces the number of data after the generation process, which normally produces more points than are required for a given interpolation.

THE MOST COMMONLY USED GROUP SCHEMES AND WEIGHTING SPECTRA

Groupwise representation of nuclear data is the commonly used medium to perform reactor calculation. Group data span the entire energy range of interest and include, in general, 3 regions:

a) thermal (low energy) region

b) resonance region, split into b') resolved, b") unresolved

c) continuum (high energy) region.

Depending upon the type of calculation, each region may include more energy groups than others (e.g. a fast reactor calculation does not need an accurate description of the cross sections in the low energy range, while a thermal calculation does). All calculations are performed starting from a limited number of groups, say a few tens. On the contrary, schemes exist with a large number of groups (e.g. 600 or 2000) but those group cross sections are used as a starting Library in order to calculate reduced group scheme libraries with considerable time saving.

Groups are generally equally spaced in lethargy so as to have a balanced description of the phenomena in different energy regions. Care should be taken mainly in designing larger group schemes, in order to pick up some typical energies which are of interest from the point of view both of the materials involved and of the calculation to be performed. Some examples will explain this point.

a) The important p-wave resonance of Iron-56 at 1.15 keV should be included in its integrity

in a unique energy group;

b) the thermal energy range around 0.0253 eV should also be covered by a unique group;

c) thresholds of important reactions such as the inelastic processes should, on the contrary, coincide with some group boundary;

d) energies which are typical of some assembly should be considered too.

When a group scheme has to be used as a subsequent input to processing codes which calculate a new (reduced) group scheme or collapse it to a minor number of groups, then a large number of groups is used, say more than 50 and up to 2000; in this case equal lethargy spacing is adopted either in the whole range or, with constant Δu, in some subranges. On the contrary, working libraries use less than 30 energy groups, which are either equally spaced in lethargy in the entire range or with larger groups at the higher energies (continuum region).

Again, a variety of functions exists over which the cross sections are weighted. Typical measured fluxes of references assemblies are sometimes used and their shapes are not easily predictable (those weighting functions are normally given point by point to the processing codes). When a group scheme is to be used in subsequent cell calculation a "guess" is adopted to simulate the shape of the flux. One of the most widely used functions is so defined:

a) A Maxwellian shape with the maximum near the thermal energy (0.0253 eV) $F(E)=\exp(-E/\Theta)/\upsilon^2$,

b) $\phi(E) = 1/E$ in the intermediate energy range,

c) a fission spectrum at higher energies, say some hundred keV. $F(E) = \exp(-E/0.965) \sinh(\sqrt{2.29E})$

This describes with sufficient approximation the shape of the neutron flux. It should be noted that the 1/E function mainly applies in the resolved resonance region: this fact allows the speeding up of the integral quantity calculations by means of semi-analytcal formulas.

Basic multigroup libraries are generated at a given temperature and infinite dilution supposing, i.e. the material is uniformly distributed in the medium. Problem dependent libraries, also called working libraries, are then generated at the effective temperature and with the actual densities of the components by means of codes based upon different approaches, the main ones being the Bondarenko method and the Nordheim integral treatment.

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CONVERSION CODES

Programme	Translation from	Translation to	Author	Laboratory of origin
BRIGITTE	ENDF/B	KEDAK	J.C. Schepers	CEN, Mol, Belgium GFK, Karlsruhe
MISSIONARY	ENDF/B	UKNDL	J. Cameron	AWRE, Aldermaston,UK
TRAKEDAK	KEDAK	UKNDL	A. Beyeler	CEN, Saclay, France
UTOE	UKNDL	ENDF/B	G.C. Panini	ENEA, Bologna,Italy
KTOE	KEDAK	ENDF/B	G.C. Panini	ENEA, Bologna,Italy
UKTOA	UKNDL	ENDF/A		JAERI, Japan Report JAERI-Memo-3162
LTOE	ENDL	ENDF/B		LLL, Livermore, USA
LRL-UK	ENDL	UKNDL		LLL, Livermore, USA
ENDF-LRL	ENDF/B	ENDL		LLL, Livermore, USA
UK-LRL	UKNDL	ENDL		LLL, Livermore, USA
ETOS	ENDF/B	SPENG		AE, Sweden
UKE	UKNDL	ENDF/B	R.Q. Wright et al.	ORNL

OTHER CONVERSION CODES

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