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INDC INTERNATIONAL NUCLEAR DATA COMMITTEE

IAEA Advisory Group Meeting on Nuclear Data for Reactor Dosimetry

Vienna, 13-17 November 1978

SUMMARY REPORT

Edited by

A. Lorenz
Nuclear Data Section
International Atomic Energy Agency

January 1979

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Abstract

This Advisory Group Meeting on Nuclear Data for Reactor Dosimetry was convened by the IAEA Nuclear Data Section, at IAEA Headquarters in Vienna, Austria, from 13-17 November 1978. The meeting was attended by 20 representatives from 10 Member States and 2 international organizations.

The primary objective of this meeting was to finalize the details for the creation of a new international file of evaluated neutron cross section data for reactor dosimetry applications, and to establish a procedure for the testing and adjustment of these data.

I. SUMMARY OF THE MEETING

A. Introduction

The Advisory Group Meeting on Nuclear Data for Reactor Dosimetry was convened by the IAEA Nuclear Data Section at IAEA Headquarters in Vienna, Austria, from 13-17 November 1978. The meeting was attended by 20 representatives from 10 Member States and 2 international organizations. The list of participants is given in Appendix 1.

B. Objectives

The primary objective of this meeting was to finalize the details for the creation of a new international file of evaluated neutron cross section data for reactor dosimetry applications, and to establish a procedure for the testing and adjustment of these data.

The adopted agenda is given in Appendix 2 and the list of papers presented to the meeting by the participants is given in Appendix 3. Original papers prepared specifically for this meeting are planned to be published as a separate report.

C. Conclusions and Recommendations

The conclusions and recommendations of this meeting are contained in three separate reports produced at the meeting:

1. The General Guidelines for the Creation of the International Reactor Dosimetry File, consisting of a general set of recommendations on the creation, maintenance and up-keep of the basic reactor dosimetry data file, which contains the evaluated energy dependent cross section data and their uncertainties,

2. The Report of the Working Group on Benchmark Fields and Integral Data, recommending the creation of a complementary benchmark data file containing data on reference benchmark fields and recommended evaluated integral cross sections measured in these fields, and

3. The Report of the Working Group on Data Testing, Spectrum Unfolding and Data Adjustment, which recommends the methodology to be used in testing the data contained in the reactor dosimetry (differential) data file, in unfolding neutron flux density spectra, and in performing data adjustments on the basis of information from integral experiments.

The complete text of these recommendations is included in Part II of this report. The Actions which resulted from this meeting are listed in Appendix 4.

II. CONCLUSIONS AND RECOMMENDATIONS

A. General Guidelines for the Creation of the International Reactor Dosimetry File

1. Purpose of the International Reactor Dosimetry File (IRDF)

The purpose of this effort is to provide to the international reactor dosimetry community a comprehensive set of dosimetry reaction cross sections and their uncertainties which would be used for the spectral characterization of neutron fields, including for example the unfolding of neutron spectra from in-pile multiple foil reaction rate measurements, and e.g. for the estimation of radiation damage in fission and fusion reactors and for other applications.

2. Content of the IRDF

The IRDF is to be composed initially of the reactions included in the ENDF/B-V Dosimetry File (see Table I of paper AG-160/1 by B.A. Magurno presented at this meeting) and of an additional set of reaction cross sections selected at the previous meeting (see Table 7 of AG-160/17 by W.L. Zijp) which are evaluated by groups outside the USA partially coordinated by IAEA/NDS, with a possible extension of the file to include additional reactions of interest. The complete list of the reactions to be included in the initial file is given in Table I. After consideration of the two reactions $^{14}N(n,p)^{14}C$ and $^{199}Hg(n,n')^{199}Hg^{m}$, the meeting decided not to include them in the table because of the lack of sufficient data. The energy range encompassed by these reactions will initially extend to 20 MeV, with the understanding that the range will be extended to higher energies (up to 40 MeV for some reactions) as deemed necessary. The proposal to adopt the reactions $^{27}Al(n,\alpha)$ and $^{237}Np(n,f)$ as recommended standards for dosimetry cross section measurements was given special consideration; the summary of this discussion is given in Annex A.

IRDF will contain point-wise energy dependent cross section data (i.e. $\sigma(E)$). The file should contain (with the exception of Nb(n,n')data) only information derived from differential measurements and from theory. It is not clear at present that the data in the ENDF/B-V file will satisfy the above condition, but nevertheless, data from ENDF/B evaluations must be included at least in the first version of the IRDF and information be provided on the meaning and the extent of any adjustment made to the data.

Information from integral experiments will be included in a complementary file which may be used for data testing and adjustment. Angular distributions are needed at present only for standard reactions, and these are contained in the ENDF/B Standards File available from the international neutron data center network.

3. Structure of the IRDF

The reaction cross sections in the IRDF will be in the ENDF/B-V format. Although group structure representation of the IRDF is not the concern of this meeting, energy or lethargy averages of the cross-sections particularly in the resonance energy range could be calculated by the National Nuclear Data Center (NNDC) at the Brookhaven National Laboratory in the USA, and be made available upon request through IAEA/NDS.

4. The IRDF Uncertainty File

The dosimetry file IRDF will contain the cross section uncertainties in the form of covariance matrices in the format of the File 33 of the ENDF/B-V library.

All evaluators contributing to the IRDF are required to include in their evaluation the covariance matrices of the energy dependent cross-sections. No rules were fixed at this stage as regards the determination of these covariance matrices.

5. IRDF Review Procedures

The evaluation review procedures for energy dependent cross sections contained in IRDF will be similar to the procedure adopted by the Cross-Section Evaluation Working Group in the USA; it will consist of two steps:

a) Phase I Data Review,

b) Phase II Data Testing.

It will be the responsibility of the evaluator to have the evaluation subjected to the Phase I Data Review. This review is to be performed with the help of a "Phase I Data Review Kit" (available from the NNDC in the USA) by a scientist other than the evaluator.

The responsibility for Phase II Data Testing and the methods to be used are described in the report of the Working Group on Data Testing, Spectrum Unfolding, and Data Adjustment.

6. IRDF Documentation

A "Summary Documentation" of every data and covariance evaluation should be submitted with the evaluation. These summary documentations shall be collated and be available for distribution with the IRDF by the IAEA/NDS.

In addition, more extensive documentation of an evaluation will be required which should not be restricted to an internal report; it is encouraged that such documentations be submitted for publication in a journal having international distribution, such as the "Atomic Data and Nuclear Data Tables" journal.

Documentation of the ENDF/B-V format and of the associated covariance matrix file shall be made available through the IAEA/NDS.

7. IRDF Distribution

The initial IRDF, as well as future updated versions, will be maintained by IAEA/NDS, and will be freely available upon request from the IAEA/NDS.

8. IRDF Time Schedule

It is expected that the ENDF/B-V Dosimetry file will be released in March or April 1979, at which time it will be sent to IAEA/NDS.

Evaluations including covariance matrices contributed to IRDF by individual evaluators should be received by IAEA/NDS within six months from the time of this meeting, i.e. latest mid-May 1979.

IAEA/NDS will merge the ENDF/B-V file and the contributed evaluations as soon as it is feasible and strive to release the IRDF version I (IRDF-I) together with the benchmark file as soon as possible thereafter, (preferably before the move of the Agency to its new Headquarters in July 1979).

9. IRDF Updating

Updating of the file will be performed by the IAEA/NDS, and corrections released in a timely manner.

The US National Nuclear Data Center is prepared to supply IAEA/NDS with every update of the ENDF/B-V Dosimetry File as it occurs.

The individual contributors to the IRDF are expected to maintain their evaluations up-to-date and to submit revisions of their evaluations for new releases.

Evaluations of additional reactions can be added to the IRDF if they are deemed necessary by the reactor dosimetry community. A preliminary list of potential additions to the initial set of reactions contained in the IRDF is given in Table II.

10. IRDF Review

It appears advisable that a smaller group of people, representing both evaluators and users of the IRDF, meet in the future at timely intervals for the purpose of reviewing the status of the IRDF and recommending its future development. Such meetings could be held under the auspices of the IAEA.

Table I

List of Reactions to be included in the International Reactor Dosimetry File

Isotope/Reaction	Origin	Isotope/Reaction	Origin
^{6}Li (n, total He)	ENDF/B-V	^{60}Ni (n,2n)	ENDF/B-V
^{10}B (n, total He)	ENDF/B-V	^{63}Cu (n,γ)	ENDF/B-V
^{19}F (n,2n)	(Vonach)	^{63}Cu (n,α)	ENDF/B-V
^{23}Na (n,γ)	ENDF/B-V	^{63}Cu (n,2n)	(Vonach)
^{23}Na (n,2n)	(Marcinkowski)	^{65}Cu (n,2n)	ENDF/B-V
^{24}Mg (n,p)	(Vonach)	^{64}Zn (n,p)	(Vonach)
^{27}Al (n,p)	ENDF/B-V	^{90}Zr (n,2n)	(Vonach)
^{27}Al (n,α)	ENDF/B-V	^{93}Nb (n,n')	(Vonach)
^{31}P (n,p)	(Vonach)	^{93}Nb (n,2n)	(Philis)
^{32}S (n,p)	ENDF/B-V	^{103}Rh (n,n')	(Vonach)
^{45}Sc (n,γ)	ENDF/B-V	^{115}In (n,n')	ENDF/B-V
^{45}Sc (n,2n)	(Magurno)	^{115}In (n,γ)	ENDF/B-V
^{46}Ti (n,p)	ENDF/B-V	^{127}I (n,2n)	ENDF/B-V
^{47}Ti (n,p)(n,np)	ENDF/B-V	^{197}Au (n,γ)	ENDF/B-V
^{48}Ti (n,p)(n,np)	ENDF/B-V	^{197}Au (n,2n)	(Philis)
^{55}Mn (n,2n)	ENDF/B-V	^{197}Au (n,3n)	(Philis)
^{54}Fe (n,α)	(Vasiliu)	^{197}Au (n,4n)	(Philis)
^{54}Fe (n,p)	ENDF/B-V	^{232}Th (n,f)	ENDF/B-V
^{56}Fe (n,p)	ENDF/B-V	^{232}Th (n,γ)	ENDF/B-V
^{58}Fe (n,γ)	ENDF/B-V	^{235}U (n,f)	ENDF/B-V
^{59}Co (n,γ)	ENDF/B-V	^{238}U (n,f)	ENDF/B-V
^{59}Co (n,α)	ENDF/B-V	^{238}U (n,γ)	ENDF/B-V
^{59}Co (n,2n)	ENDF/B-V	^{237}Np (n,f)	ENDF/B-V
^{59}Co (n,p)	(Vasiliu)	^{239}Pu (n,f)	ENDF/B-V
^{58}Ni (n,p)	ENDF/B-V	^{241}Am (n,f)	(Patrick)
^{58}Ni (n,2n)	ENDF/B-V		

Table II

Preliminary List of Potential Additions to the International Reactor Dosimetry File

Reactions considered to be especially important are indicated by an asterisk.

Isotope/Reaction	E_{max}	Application
^{27}Al (n, total He)	30 MeV	fluence
^{35}Cl (n,γ)		
^{35}Cl (n,α)		
^{46}Ti (n,2n)	35 MeV	
* ^{51}V (n,γ)		
* ^{55}Mn (n,γ)		
^{107}Ag (n,2n)	30 MeV	
^{109}Ag (n,γ)		
^{127}I (n,γ)		
* ^{139}La (n,γ)		
^{164}Dy (n,γ)		
^{175}Lu (n,2n)		fluence
^{176}Lu (n,3n)		fluence
^{175}Lu (n,γ)		
^{176}Lu (n,γ)		
* ^{186}W (n,γ)		
^{234}U (n,f)		
^{236}U (n,f)		
^{238}U (n,2n)	30 MeV	

Note: In addition, for fusion applications and high energy neutron source spectrometry, the reactions listed in Table I will have to be complemented and evaluated to higher energies: (n,p) and (n,α) reactions to $\sim$ 30 MeV, (n,xn) reactions to 30-40 MeV, and other reactions to 25-30 MeV.

B. Report of the Working Group on Benchmark Fields and Integral Data

E.J. Axton (Chairman)
Z. Dezsoe
R. Dierckx
M.F. Hegedüs
R. Jahr
W. Mannhart
A. Marcinkowski
E. Menapace
F.G. Perey
J.J. Schmidt (IAEA)

Benchmark Data File

This file is intended to be a compilation of existing standard and reference fields, evaluated average cross sections measured in these fields, and their covariance matrices. At present the ^{252}Cf spontaneous fission field is recognized as the most important standard field. In the future the evaluated average cross sections measured in other fields should be analyzed to arrive at a final covariance matrix for all fields and for all measured average cross sections. The relevant fields are listed in Tables I and II, page 7 ff. volume 1 of the IAEA-208* report, and their detailed description on page 53 ff. of the same report. A new field should be added to cover fusion, and to correlate fusion reactor dosimetry with fission reactor dosimetry. This benchmark field should be a well defined DT source, with the deuteron energy, target geometry and construction specified.

A more detailed account of the considerations of this Working Group regarding the benchmark fields to be included in the proposed Benchmark File is given in Annex B.

Recommendations

1. In the Benchmark list of IAEA-208* (volume 1, page 7) a standard DT source should be added. Such a source has yet to be developed.

2. The International Reactor Dosimetry File should be complemented by a benchmark data file which will include
   1 - name of field
   2 - shape and magnitude of the neutron flux density spectrum, (determined by other methods than by unfolding multiple foil data), with all relevant covariance matrices
   3 - references and comments concerning the field

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* Neutron Cross Sections for Reactor Dosimetry, IAEA-208, Proceedings of a Consultants Meeting held at the International Atomic Energy Agency, Vienna, Austria, 15-19 November 1976.

4 - evaluated measured average cross sections including the covariance matrices of these cross sections

5 - references and comments concerning the evaluation and all relevant important covariances between evaluated measured cross sections and fields.

Formats for representing spectra and their covariance matrices will be the same as those used in ENDF/B-V. New formats must be defined for the evaluated experimental reaction rates. These will be formulated by F.G. Perey.

3. There is a need for additional evaluations and possibly measurements of spectra in standard and reference fields. There is a need for the evaluation and possibly the measurements of average cross sections in these fields which include covariances.

4. Future reporting of measurements should include full details of the uncertainties in the measurements so as to enable the generation of the covariance matrix of the results. Authors should provide lists of all the individual significant contributions to the uncertainty with particular emphasis on correlations to other data. For previously reported results, such details should be made available so as to generate the appropriate covariance matrices.

5. F.G. Perey will coordinate the putting together of the first version of this complementary benchmark data file and make it available to IAEA/NDS for dissemination together with the IRDF upon request.

C. Report of the Working Group on Data Testing, Spectrum Unfolding and Data Adjustment

J.G. Williams	(Chairman)	B. Magurno	
A. Gandini		W.L. Zijp	
A.K. McCracken		C. Ertek	(IAEA)
M. Najzer		K. Okamoto	(IAEA)

Data Testing

1. The information to be used in testing the energy dependent cross-section file is the supplementary information, supplied with the file, on standard and reference spectra and average cross-sections measured in those fields.

2. A check must be made that the evaluators have specified all relevant nuclear data used (such as half-lives, branching ratios, gamma ray energies and transition probabilities) and that these have been applied in a consistent way.

3. A test should be made of the agreement and consistency of each energy dependent cross-section set with the average value measured in the fields for which this information is supplied with the file. This test will involve folding the cross sections with the flux density spectra to obtain a calculated average cross-section together with an uncertainty obtained using the respective covariance matrices.

4. It is the prerogative of those doing the testing to make recommendations from the result of such tests as to which reactions belong to Category I, as defined below.

5. The definition of a Category I reaction is given in IAEA-208* (volume 1, page 32) and is here clarified as follows:

   (i) Reliable average cross section values, with uncertainty within 2.5% (1σ), must be available to at least one relevant standard field. The definition of a standard field is given in IAEA-208* (volume 1, page 6).

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* Neutron Cross Sections for Reactor Dosimetry, IAEA-208, Proceedings of a Consultants' Meeting, held at the International Atomic Energy Agency, Vienna, Austria, 15-19 November 1976.

(ii) The energy dependent cross section must have very well specified uncertainties, preferably within 5% (1σ) in the response range of major importance in the standard spectrum.

(iii) The value of χ^2_n (where n is the number of experiments) found in a comparison between measured and calculated average cross sections in the relevant standard fields must be within an interval corresponding to a 90% confidence level.

6. It should be the responsibility of the IAEA to have the testing carried out.

Unfolding

The working group only considered the problem of obtaining the most likely spectrum and its uncertainties. It may sometimes be desirable to derive from the dosimetry cross sections and the activation data some other type of spectrum; for instance, one which, while consistent with the activation data and the dosimetry cross sections, would also maximize some particular functional. The procedures for obtaining such spectra were not considered.

1. The practice of using "spectrum unfolding" algorithms which do not fully utilize all of the information available regarding the uncertainties in the input data (spectrum, activation and energy dependent cross-section) should be discontinued. Algorithms which make use of this information should be used instead since they produce the most likely spectrum. Examples of such algorithms can be found in ORNL/TM-6267 * and in Annex C to this report.

2. One cannot expect to derive any reasonably accurate representation of the spectrum in an energy range in which good coverage is not provided by detectors unless good information on the spectral shape (either from previous experiments or from reactor physics calculations) is used as input to the method proposed.

3. Codes which implement the mathematical approach recommended are in existence. One example is STAY'SL. It is noted that the important thing is the algorithm, not the specific implementation of it.

4. The covariance matrix of the output can easily be given by these methods and should be used in calculations of responses.

5. An understanding of the method is essential for its proper use.

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* F.G. Perey, Contributions to Few-Channel Spectrum Unfolding, Oak Ridge National Laboratory Report ORNL/TM-6267 (February 1978).

Data Adjustment

1. The method proposed for spectrum unfolding performs an adjustment of the input dosimetry data and is a suitable method for the adjustment of data to provide improved prediction of responses in specific applications.

2. The data chosen as input to such a procedure is the responsibility of users, but it is suggested that a simultaneous unfolding of the standard field data will be helpful provided there is an overlap between the standard and the target spectrum.

3. Because of problems associated with correlations, the adjusted data should not be output for application elsewhere, except to make them available to data testers.

Discussion Summary

At the request of the International Nuclear Data Committee the meeting discussed the suitability of the reactions ^{27}Al (n,α) and ^{237}Np (n,f) as standards for dosimetry cross section measurements and concluded the following:

1. The ^{27}Al (n,α) is not ideally suited as a standard for cross section measurements. However, for practical reasons, it has been used very frequently in the past for ratio measurements and as such is a de facto standard. If for this reason alone, the ^{27}Al (n,α) cross section needs to be better known in order to use more effectively a large class of existing data. We therefore recommend that the ^{27}Al (n,α) reaction continues to be considered as a standard until a better alternative is found.

2. The ^{237}Np (n,f) reaction is used for dosimetry. It is therefore necessary to provide cross sections for this reaction which are as accurate as possible. However, this reaction has not in the past been extensively used for ratio measurements and the group did not feel qualified at this stage to present convincing arguments as to its advantages and disadvantages. The group was aware that some arguments have been made for the use of this reaction as a standard, and that it had been tentatively included in the INDC standards file.

Annex B

Some Considerations on the Benchmark File

E. J. Axton

The first version of the file should contain a small number (perhaps four) of fields, but should not be restrictive to the extent that people with the necessary facilities and expertise would be discouraged from testing the file. Of the standard fields listed in table 1 of IAEA-208 only the 252-Cf is considered to exist in a permanent and reproducible form. Thermal Maxwellians may have different temperatures and different epithermal (1/E) components, and 1/E fields would terminate at different maxima. It would be premature to include Thermal Maxwellian and 1/E fields in the first version of the file.

However, in view of the importance of resonance activation integrals to spectrometry in the kilo-electron volt energy region these fields together with thermal neutron cross-sections and resonance activation integrals should be included in later versions.

Evaluations of thermal (2200 $m.s^{-1}$) cross sections and resonance activation integrals with comprehensive covariance matrices are badly needed, and consideration must be given to the manner in which the thermal and 1/E fields are described in the file. They are currently described in terms of the total neutron density. Maxwellian temperature, and non-Maxwellian fraction of the neutron density.

Annex C

Considerations of Spectrum Unfolding Algorithms

A. McCracken

The statement adopted in paragraph 1 of the section on unfolding identifies the joint probability distribution (p.d.f) of an input spectrum, measured detector activation-rates and energy-dependent detector cross-sections as the available information - all of which should be used - for the adjustment of the starting p.d.f. to its most probable value. In more casual terms one speaks of 'unfolding the spectrum with its uncertainty relations'; the more formal statement stresses that all input data is to be re-assessed and that one could just as well think of unfolding new values of detector cross sections with their uncertainties. We shall usually be concerned with discrete values of the variables appropriate to multigroup calculations of flux and so can write

$$X \phi = \underline{a} \qquad 1$$

for the calculation of reaction-rates $\underline{a}$ using X and ϕ, the original estimates of the multigroup detector cross-sections and fluxes respectively. Corresponding to the calculated activities $\underline{a}$ will be measured activities $\underline{m}$. The starting point of the unfolding, or adjustment, process is the observation that in general $\underline{a} \neq \underline{m}$. Perturbations $\delta X, \delta\phi$ and $\delta\underline{m}$ are therefore to be chosen such that $(X + \delta X)(\phi + \delta\phi) = \underline{a} + \delta\underline{a} = \underline{m} + \delta\underline{m}$

ie $\hat{X}\hat{\phi} = \hat{\underline{a}} = \hat{\underline{m}}$ where $\hat{X} = X + \delta X$ and the capped variables represent the best of the infinite choice available to satisfy the above equation of condition.

The best choice can readily be found by the application of the Maximum Likelihood Principle which requires the adjustments to be such that the probability of making the original measurements X, ϕ and $\underline{m}$ is a maximum. If now one writes

$$\underline{Y} = \begin{pmatrix} \underline{\phi} \\ \underline{x}_1 \\ \vdots \\ \underline{x}_n \\ \underline{m} \end{pmatrix} \qquad 2$$

where the $(\underline{x}_i)^T$ are the raw vectors which constitute the matrix X we wish to choose $\hat{\underline{Y}}$ to maximise $P_Y(\hat{\underline{Y}}, \underline{Y})$ where P_Y represents the joint p.d.f. of $\underline{Y}$. P_Y must be supplied by the unfolder and the following assumptions seem reasonable:-

(i) the vectors $\underline{\phi}$, $\underline{x}_i$ and $\underline{m}$, and the components of $\underline{m}$ are mutually independent,

(ii) correlations exist between the components of both $\underline{\phi}$ and $\underline{x}_i$.

With these simplifications we now wish to maximise

$$P_Y(\hat{\underline{Y}}, \underline{Y}) = P_{\underline{\phi}}(\hat{\underline{\phi}}, \underline{\phi})\, P_{\underline{m}}(\hat{\underline{m}}, \underline{m}) \prod_i P_{\underline{x}_i}(\hat{\underline{x}}_i, \underline{x}_i) \qquad 3$$

In practice each of the above distributions is probably represented fairly well by a multivariate normal distribution of the form

$$P_Y(\hat{\underline{Y}}, \underline{Y}) = \frac{1}{|2\pi V_Y|^{\frac{1}{2}}} \exp -\frac{1}{2}\left\{(\hat{\underline{Y}}-\underline{Y})^T V_Y^{-1} (\hat{\underline{Y}}-\underline{Y})\right\} \qquad 4$$

where V_Y is the dispersion matrix of $\underline{Y}$. This leads immediately to the easily recognisable least squares form:-

$$\left[P_Y(\hat{\underline{Y}}, \underline{Y})\right]_{MAX} = \left[(\hat{\underline{\phi}}-\underline{\phi})^T V_{\underline{\phi}}^{-1} (\hat{\underline{\phi}}-\underline{\phi}) + (\hat{\underline{m}}-\underline{m})^T V_{\underline{m}}^{-1} (\hat{\underline{m}}-\underline{m}) + \sum_i (\hat{\underline{x}}_i-\underline{x}_i)^T V_{\underline{x}_i}^{-1} (\hat{\underline{x}}_i-\underline{x}_i)\right]_{MIN} \qquad 5$$

A suitable unfolding algorithm, then, within the meaning of the statement adopted in Paragraph 1 is one in which the <u>complete</u> measurement vector $\underline{Y}$ of equation 2 is used in the maximisation of $P_Y(\hat{\underline{Y}}, \underline{Y})$, to obtain a <u>complete</u> probability distribution of $\hat{\underline{Y}}$. This requirement holds good for any distribution P_Y though in practice successive simplifications of the type leading to equations 3 and 5 may be adopted.

Examples of algorithms which satisfy the above criterion are:-

(i) STAYSL by Perey who considers the problem very fully in ORNL/TM-6267.

(ii) SENSAK I and SENSAK II by McCracken. The former has been widely used by the author for testing but has not been published - the improved version SENSAK II will be published shortly.

All methods of unfolding traditionally make use of $V_{\underline{m}}^{-1}$ and a few have used $V_{\underline{x}i}^{-1}$ for some time. It is the inclusion of $V_{\underline{x}i}^{-1}$ and $V_{\underline{\phi}}^{-1}$ in the process that the statement adopted seeks to stress. The fact that these may be imperfectly known does not justify their omission. In both of these distributions finite marginal variances and (largely) positive covariance terms are known to exist. Any sensible estimate of these distributions is therefore necessarily better than the use of $V_{\underline{x}i} = 0$ and/or $V_{\underline{\phi}}$ is diagonal with $v_{\phi ij} = 0$ and $v_{\phi ii} = \infty$ which is implied in some algorithms. The other fact which is stressed is the need to estimate the dispersion matrix $V_{\underline{\hat{y}}}$.

The above very brief sketch avoids some contention which was evident in the discussion, and which is partially unresolved, in an attempt to stress those basic principles on which there appeared to be a large measure of agreement.

Appendix 1

Advisory Group Meeting on Nuclear Data for Reactor Dosimetry

Vienna, 13-17 November 1978

List of Participants

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Appendix 2

Advisory Group Meeting on Nuclear Data for Reactor Dosimetry

Vienna, 13-17 November 1978

ADOPTED MEETING AGENDA

The topics I, II, IiI and IV referred to in the Agenda are identified in the List of Papers given in Appendix 3.

Monday AM	Introductory Items - Opening - Announcements - Selection of session and working group chairmen - Collection and distribution of papers
Monday AM+PM	Presentation of Part I Papers Chairman: W.L. Zijp
Tuesday AM+PM	Discussion of Part I Topics and formulation and drafting of conclusions and recommendations relating to Part I topics. Chairman: F.G. Perey
Wednesday AM+PM	Presentation of Part II, III and IV Papers Chairman: R. Dierckx
Thursday AM+PM	Full day meeting of working groups and formulation and drafting of conclusions and recommendations related to their respective topics. Working Group on Benchmark Fields and Integral Data Chairman: E.J. Axton Working Group on Data Testing, Spectrum Unfolding and Data Adjustment Chairman: J.G. Williams
Friday AM+PM	Plenary Session for the discussion and final approval of conclusions and recommendations related to Part I, II, III and IV topics. Chairman: J.J. Schmidt

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List of Presentations and Discussions

(Reference to the paper numbers is given in parenthesis after the author's name)

I. Differential Neutron Cross Section Data for Reactor Dosimetry

A. Status of the Dosimetry File for ENDF/B-V. B.A. Magurno (AG/160-1)

B. Status of data for important reactions not included in ENDF/B-V Dosimetry File: recent measurements and evaluations.

1) Evaluation of the Cross Sections for the Reactions $^{24}Mg(n.p)^{24}Na$, $^{64}Zn(n.p)^{64}Cu$, $^{63}Cu(n,2n)^{62}Cu$ and $^{90}Zr(n,2n)^{89}Zr$ H. Vonach, S. Tagesen, B. Strohmaier. (AG/160-2)

2) Measurement of the Activation Cross Section of the Reaction $^{93}Nb(n,n')^{93m}Nb$ for 0-25 MeV Neutrons. M.F. Hegedüs et al. (AG/160-3)

3) The Analysis of the $^{59}Co(n,p)$ and $^{54}Fe(n,\alpha)$ Cross Sections for their Use in Reactor Dosimetry. G. Vasiliu, S.Mateescu. (AG/160-4)

4) The Measurements and Evaluation of Fast Neutron Cross Section Data for Reactor Dosimetry. A. Marcinkowski. (AG/160-5)

5) Status of the $^{45}Sc(n,2n)^{44}Sc$ Reaction. K. Okamoto. (AG/160-6)

6) Informal communication of recent 14 MeV cross-section measurements and evaluations at the Institute of Experimental Physics at Debrecen. Z. Dezsoe (private communication)

7) Justification for the use of Al-27(n,α) and $^{237}Np(n.f)FP$ as standard. (Open discussion). (See Annex A of Meeting Report)

C. The internationally recommended neutron cross-section data file for dosimetry applications (purpose, contents, structure, error file, etc.)

1) Summary Remarks and Recommended Reactions for an International Data File for Dosimetry Applications for LWR, FBR, and MFR Reactor Research, Development and Testing Programmes. W.N. McElroy, et al. (Presented by R. Dierckx). (AG/160-7)

2) The Covariance Files of ENDF/B-V. F. Perey. (AG/160-8)

II. Spectral Characterization of Dosimetry Benchmark Neutron Fields

A. Spectral characterization of other dosimetry benchmark neutron fields: Progress since November 1976.

1) Neutron Fields Available at the National Physical Laboratory and the Need for Low Energy Neutron Standards. E.J. Axton, A.G. Bardell. (AG/160-9)

2) Fast Neutron Standard Fields at the Accelerators of the PTB. R. Jahr et al. (AG/160-10)

3) Accelerator-based Dosimetry Benchmarks. R. Dierckx. (AG/160-11)

B. Neutron Spectra unfolding

1a) Iteration Acceleration in the ITER-2 Unfolding Code. M. Panko and M. Najzer. (AG/160-12A)

1b) Multispectrometer Unfolding by the ITER Code. B. Glumac and M. Najzer. (AG/160-12B)

2a) Progress Report on the IAEA Programme on the Standardization of Reactor Dosimetry Measurements. C. Ertek, B. Cross, V. Chernyshev (AG/160-13A)

2b) Influence of Cross Section Structure on Unfolded Neutron Sepctra. C. Ertek, M.F. Vlasov, B. Cross, P.M. Smith. (AG/160-13B)

2c) Errors Involved in the Folding and the Unfolding of Neutron Spectrum by SAND-II. C. Ertek. (AG/160-13C)

2d) Seibersdorf-Helsinki Intercomparison of Neutron Flux Density Spectra by using the SAND-II and the LOUHI Unfolding Programmes. C. Ertek. (AG/160-13D)

III. Integral Data in Benchmark Neutron Fields

A. Progress in integral data and their accuracy.

1) Integral Cross Section Measurements in the Reference Field NISUS. J.G. Williams. (AG/160-14)

2) Average Cross Section in the Cf-252 Benchmark Field. W. Mannhart. (AG/160-15)

3) Integral Cross Section Measurements in the Cf-252 Neutron Spectrum. Z. Dezsoe, J. Csikai. (AG/160-16)

B. Measurements at higher energies for fast reactor dosimetry and CTR. (Open discussion).

IV. Differential-Integral Data Consistency and Progress in Neutron Cross-Section Data Adjustment on the Basis of Integral Experiments

A. On the Consistency between Integral and Differential Cross Section Data. W.L. Zijp. (AG/160-17)

B. New results in data adjustment.

1) Attempts at the Adjustment of Detector Cross Sections. A.K. McCracken, A. Packwood. (AG/160-18)

2) Review of Advanced Methods for Data Adjustments. A. Gandini. (AG/160-19)

Appendix 4

Advisory Group Meeting on Nuclear Data for Reactor Dosimetry

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List of Actions

(IRDF stands for International Reactor Dosimetry File)

1.	IAEA/NDS	Communicate all results of this meeting to Dr. Vlasov.
2.	IAEA/NDS	Communicate meeting results to Vasiliu and ask him to add covariance matrices to his evaluations.
3.	Individual Evaluators	Submit evaluated dosimetry cross sections together with their covariance matrices in ENDF/B format as well as summary documentation to IAEA/NDS by 1 April 1979.
4.	IAEA/NDS	Make the ENDF/B documentation for evaluated dosimetry cross-sections and covariance matrices available to individual evaluators.
5.	McCracken	Check with Patrick, whether he will include covariance matrices in his $^{241}Am(n,f)$ evaluation, and inform Perey and IAEA/NDS.
6.	Perey, IAEA/NDS	Inform Philis of the conclusions reached at this meeting about the IRDF and ask if his evaluations could be included in the IRDF.
7.	Perey, Schmidt	Communicate conclusions of meeting regarding the use of $^{27}Al(n,\alpha)$ and $^{237}Np(n,f)F.P.$ to A.B. Smith, Argonne.
8.	Magurno	Send "Phase I Review Kit" to IAEA/NDS for distribution to evaluators.
9.	Perey	Send 25 copies each of ORNL/TM-5938 and ORNL/TM-6267 to IAEA/NDS for distribution to the participants of this meeting.
10.	All participants	Send to IAEA/NDS a list of names and addresses of key scientists who should be on the distribution for the IRDF.
11.	Zijp	Send to Lorenz 10 copies of the Euratom "Nuclear Data Guide".
12.	IAEA/NDS	Supply to Magurno all information compiled on the $^{45}Sc(n,2n)\ ^{44}Sc^{gm}$ reaction.
13.	All participants	Inform IAEA/NDS before 15 December 1978 whether the "International Reactor Dosimetry File" is acceptable.

14.	Perey	Provide the format for the evaluated experimental reaction rates and their covariances.
15.	McCracken	Send to IAEA/NDS a revised write-up of the "Recommended Unfolding Method" for inclusion in the meeting report as an Appendix.
16.	Williams	Provide to Perey information on the $\Sigma\Sigma$ spectrum averaged cross sections.
17.	Mannhart	Provide to Perey information on the Californium spectrum averaged cross sections.
18.	Perey	Assemble the benchmark file, including the Cf and $\Sigma\Sigma$ spectra, and send it to IAEA/NDS before May 1979.