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INTERNATIONAL NUCLEAR DATA COMMITTEE

IAEA Programme on Benchmark Neutron Fields Applications for Reactor Dosimetry

edited by
M. Vlasov

July 1976

IAEA NUCLEAR DATA SECTION, KÄRNTNER RING 11, A-1010 VIENNA

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FOREWORD

This document contains a description of the new IAEA programme on the applications of benchmark neutron fields for the improvement of nuclear data needed for reactor neutron dosimetry. It includes actions required for the start of the programme as well as the preliminary agenda of the IAEA Consultants' Meeting on Integral Cross Section Measurements in Standard Neutron Fields for Reactor Dosimetry, scheduled to be held at IAEA Headquarters in Vienna from 15 to 19 November 1976.

The cooperation of Messrs. A. Fabry*, J.A. Grundl** and W.N. McElroy+ in the development and formulation of this programme is most gratefully acknowledged.

Attached to the report is a questionnaire. All scientists and organizations who are interested and able to contribute to this programme are invited to reply to the questionnaire (part A, page 3). Please send replies to Mr. M. Vlasov, Nuclear Data Section, Division of Research and Laboratories, IAEA, P.O. Box 590, A-1011 Vienna, Austria. Comments on the preliminary agenda of the Consultants Meeting will also be welcome.

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IAEA PROGRAMME ON BENCHMARK NEUTRON FIELDS APPLICATIONS FOR REACTOR DOSIMETRY

The role and importance of benchmark neutron fields for reactor dosimetry have been recognized by the IAEA Consultants' Meeting on Nuclear Data for Reactor Neutron Dosimetry, September 1973, [INDC(NDS)-56/U] and were confirmed at the first ASTM-EURATOM Symposium on Reactor Dosimetry, Petten, the Netherlands, September 1975 [Petten Symposium, to be published]. The U.S. Interlaboratory LMFBR Reaction Rate (ILRR) program [Nuclear Technology Vol. 25, p. 177 (1975)] has so far best exemplified the need of benchmark neutron fields for dosimetry applications. In view of all this, the Agency has initiated a programme on "Benchmark Neutron Fields Applications for Reactor Dosimetry".

The two first steps in this programme will be

- to compile information on the basis of a survey and a questionnaire (included herewith), to be sent to all laboratories (part A of this document), and

- to convene a "Consultants' Meeting on Integral Cross-Section Measurements in Standard Neutron Fields for Reactor Dosimetry" in November 1976 (part B of this document).

Dosimetry benchmark neutron fields serve three general objectives:

1) validation and/or calibration of experimental techniques;

2) validation and/or improvement of cross section and other nuclear data (e.g. fission yields) needed for proper application of experimental techniques;

3) validation and/or improvement of analytical methods needed to extrapolate dosimetry data from a monitoring or surveillance position to the location of interest.

While these three objectives above are of equal importance, it is anticipated that the second one may be the focus of a working effort for this programme;data considerations as well as benchmarks pertinent to the two other purposes (i.e. the general objectives 1 and 3) are however not discarded especially when they are needed for operating or future reactor power plants.

There are two reasons why benchmark neutron fields are relevant to the establishment of an acceptable dosimetry cross section file:

a) A large amount of cross section data is needed and cannot all be obtained easily with the required accuracy from differential measurements alone.

b) Cross section data for dosimetry reactions must be integrally consistent.

Point b) is extremely important and is explained in Appendix I.

Three types of benchmark neutron fields have been identified at a workshop held during the Petten Symposium 1975:

Standard neutron field:

A permanent, stable and reproducible radiation field that is characterized to state-of-the-art accuracy in terms of neutron flux density and energy spectra, and spatial and angular flux distribution. Important field quantities must be verified by interlaboratory measurements.

Reference neutron field:

A permanent and reproducible radiation field reasonably well characterized in the above terms and accepted as a reference by a community of users.

Controlled neutron environment:

A radiation field employed for a restricted set of well-defined experiments.

Controlled environments are mostly useful for objective 3 above, but may provide also pertinent integral data for objective 2.

The important consideration for this programme is that a measurement performed in any of these "benchmark" fields must be well documented and its accuracy thoroughly assessed.

PART A

PROPOSED SURVEY OF BENCHMARK NEUTRON FIELDS

(FOR REACTOR FUELS AND MATERIALS DOSIMETRY DATA TESTING AND CALIBRATION)

Neutron dosimetry for fuels and materials exposures in nuclear reactors and other neutron environments generally involve flux, fluence and spectrum characterization with activation and other types of detectors. Proper interpretation of integral response from these detectors usually requires reference irradiations in well-characterized, benchmark neutron fields. Some requirements for such reference neutron fields are as follows:

1) Simple and well-defined geometry;
2) Adequate and stable flux densities;
3) Reproducible and accurately characterized neutron spectra based on careful spectrometry measurements and/or reliable calculations;
4) Sustained availability for measurements.

Integral detector responses obtained in these fields can be used to normalize or adjust differential neutron cross sections for dosimetry applications. Strong efforts are made to keep the adjustments within the errors of existing differential data. The best data for this purpose usually come from critically evaluated interlaboratory measurement efforts.

Other neutron fields which do not meet all of the requirements of a reference facility can also provide useful tests of differential data. Many activation reactions for neutron dosimetry have not been adequately measured by means of differential techniques and for these reactions, well-documented integral measurements in a variety of well-defined neutron fields are needed. In particular, neutron fields with spectra that complement the reference fields can be especially important. The relation of such supplementary neutron fields to the reference fields is a significant issue.

One aim of this inquiry is to gather facility information that will help establish these relationships.

Survey

The purpose of this survey is to ascertain the physical and radiation characteristics of both reference and supplementary neutron fields for referencing reactor dosimetry measurements which have been used in the past, are currently available, or which are in an advanced stage of development. The survey will at the same time collect existing documented results of dosimetry-related measurements performed with these facilities. A partial list of identified reactor dosimetry benchmarks for the investigation of fuel and materials property changes, updated and presented by Dr. W.N. Mc Elroy at the 1975 ASTM-Euratom Symposium for Reactor Dosimetry at Petten, is given in Table I. The questionnaire items listed below are structured so as to be as complete as possible in order to elicit maximum information for the most well-established neutron dosimetry benchmarks. It is recognized that many neutron fields in use or in preparation are not specified as completely as is requested in the questionnaire. A diversity of replies is therefore anticipated. The primary goal of the survey is to obtain the best information available for all neutron fields which provide reference measurements for neutron dosimetry.

QUESTIONNAIRE

1) Provide a physical description of the facility including material inventory and geometric arrangements. Include operational characteristics which are relevant for performing experiments.

2) Provide spatial description of the neutron flux and the available total flux and fluence for irradiation purposes. Include estimate of the neutron-to-gamma ratio.

3) Specify in numerical form a recommended neutron spectrum for the irradiation position in the facility. Include separately both calculational and experimental results upon which the recommendation is based. Statements of accuracy for the recommended spectrum are desireable and ultimately essential. If the spectrum can be more accurately determined, describe the required measurements and the estimated attainable accuracy.

4) Describe flux monitoring methods which are available for routine irradiations and the precision attainable for run-to-run normalizations. Indicate the general availability of the facility for interlaboratory experiments.

5) Provide documented experimental reaction rate and other integral experimental results obtained with the facility that are relevant for reactor fuels and materials dosimetry. Inclusion of unpublished or preliminary experimental data at your discretion is encouraged. No publication or general distribution of such information will be undertaken.

6) If it is appropriate, summarize unique features of the facility relevant for referencing dosimetry measurement techniques.

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Table 1. Reactor Dosimetry Benchmarks

for the Investigation of Fuels and Materials Property Changes

Benchmark	Approximate Data Testing Energy Range (MeV)	Primary Type of Reactor to which Benchmark Applies
1. Thermal spectrum	4×10^{-7}	LWR, HTGR
2. Epithermal-1/E	4×10^{-7} to 10^{-2}	All Types
3. ISNF[a]	10^{-3} to 5	Fast Breeder (FBR)
4. BIG-10[b] [d]	10^{-2} to 15	FBR
5. CFRMF[c]	10^{-4} to 15	FBR
6. ΣΣ[d] [e]	10^{-4} to 5	FBR
7. Tapiro, Yayoi, Godiva [b][d]	10^{-2} to 15	FBR
8. Fission spectrum (^{235}U, ^{252}Cf)	0.1 to 15	All Types
9. Accelerator spectra [^{9}Be (d,n)]	2,4,6,8,14	FBR, CTR

(a) Fission source driven system employing graphite and ^{10}B.

(b) Fast critical core center.

(c) Thermal-fast coupled reactor.

(d) Applicable to reactor physics, shielding and/or surveillance type dosimetry.

(e) Fission source driven system employing graphite, natural uranium and natural boron. Such systems are in operation at CEN/SCK Mol, Imperial College/London (NISUS), and Institute of Nuclear Technology, Bucarest.

References for Table 1 Benchmarks

Thermal and Epithermal - 1/E

Beckurts and Wirtz, "Neutron Physics", or equivalent

Intermediate-Energy Standard Neutron Field (ISNF)

J. Grundl and C. Eisenhauer, "Fission Rate Measurements For Materials Neutron Dosimetry in Reactor Environments", ASTM-EURATOM Symp. on Reactor Dosimetry (September 1975).

A. Fabry and J. Jenkins, "Wall Return Neutron Fluxes for High-and Intermediate-Energy Cavity Neutron Sources". ANS Trans. 15, no. 2, 975 (1973).

BIG-10 Fast Critical Assembly

E.J. Dowdy, E.J. Lozito, and E.A. Plassmann, "The Central Neutron Spectrum of the Fast Critical Assembly BIG-10," Nucl. Tech. 25, (2) p. 381 (Feb.1975).

L.J. Sapir, H.H. Helmick and J.D. Orndoff, "BIG TEN, A 10 % Enriched Uranium Critical Assembly: Kinetic Studies, "Trans. Am. Nucl. Soc., 15, 312 (1972); E.J. Lozito and E.J. Dowdy, "A Measurement of the Central Neutron Spectrum of 'BIG-10' Critical Assembly", Trans. Am. Nucl. Soc., 17, 529 (1973).

Coupled Fast Reactivity Measurement Facility (CFRMF)

J.W. Rogers, D.A. Millsap, and Y.D. Harker, "CFRMF Neutron Field Flux Spectral Characterization", Nucl. Tech. 25, (2) p. 330 (Feb. 1975).

Sigma Sigma Secondary Standard Fast Neutron Field ($\Sigma\Sigma$)

A. Fabry, G. DeLeeuw, S. DeLeeuw, "The Secondary Intermediate-Energy Standard Neutron Field at the MOL-$\Sigma\Sigma$ Facility, "Nuclear Technology, 25, p. 349 (Feb. 1975).

Tapiro

A. D'Angelo, M. Martini and M. Salvatores, "The TAPIRO Fast Source Reactor as a Benchmark for Nuclear Data Testing", Energia Nucleare (Milan), 20, n.11, 614-621 (1973);

A. D'Angelo, M. Martini and M. Salvatores, "^{235}U Compact Cu-Reflected TAPIRO Reactor Integral Experiments Results and a Check of Some High-Energy ENDF/B-III Data", Trans. Am. Nucl. Soc., 17, p. 498-499 (1973).

M. Martini, M. Belli, M.F. Sirito, "The TAPIRO Fast Source Reactor as a Benchmark to Test Activation Detector Cross Sections", ASTM-EURATOM Symposium on Reactor Dosimetry, Petten 1975, Paper 36.

Yayoi

I. Kimura and K. Kobayashi, "Intercomparison of Reactor Dosimetry Cross Sections Measured in a Thermal Reactor, in a Fast Reactor and with an Enriched Uranium Fission Plate", ASTM-EURATOM Symposium on Reactor Dosimetry, Petten 1975, Paper 46.

K. Kobayashi, I. Kimura, et al., to be published.

Godiva

R.E. Peterson and G.A. Newby, "An Unreflected U-235 Critical Assembly", Nucl. Sci. Eng. 1, 112 (1956).

T.F. Wimett, R.H. White, W.R. Stratton, and D.P. Wood, "Godiva-II An Unmoderated Pulse-Irradiation Reactor", Nucl. Sci. Eng. 8, 691 (1960).

J.A. Grundl and G.E. Hansen, "Measurement of Average Cross-Section Ratios in Fundamental Fast-Neutron Spectra", in Nuclear Data for Reactors, Conference Proceedings, Paris, 1966, pp. 321-336, International Atomic Energy Agency, Vienna, 1967 [STI-PUB/140 (Vol. 1)].

J. Grundl and A. Usner, "Spectral Comparisons with High-Energy Activation Detectors", Nucl. Sci. Eng. 8, p. 598 (1960).

Fission Spectrum

"Prompt Fission Neutron Spectra", Proc. of IAEA Consultants Meeting, Vienna (1971).

J. Grundl and C. Eisenhauer, "Fission Spectrum Neutrons for Cross Section Validation and Neutron Flux Transfer", Conf. on Nucl. Cross Sections and Technology, Washington, D.C. (March 1975).

Accelerator Spectra

L.R. Greenwood, R.R. Heinrich, and N.D. Dudey, "A Detailed Comparison of Differential and Integral Cross Section Measurements", Proceedings of the First ASTM-EURATOM Symposium on Reactor Dosimetry, Petten, 1975, paper 64.

PART B

THE IAEA CONSULTANTS' MEETING ON INTEGRAL CROSS-SECTION MEASUREMENTS IN STANDARD NEUTRON FIELDS FOR REACTOR DOSIMETRY

This meeting is being held on the recommendations of the September 1973 IAEA Consultants' Meeting on Neutron Data for Reactor Dosimetry [INDC(NDS)-56/U]. It is supported by the International Working Group on Reactor Radiation Measurements (IWGRRM) and the International Nuclear Data Committee (INDC).

The ultimate goals of the meeting are:

- to reach as far as possible a consensus regarding the status of reactor dosimetry neutron data;

- to work out specific recommendations for future efforts in this field.

The meeting is proposed to be divided into seven sessions. Each session should encompass the following typical sequences of presentations and discussions:

a) overviews: designated reviewers are proposed to present their appraisal of the status and future direction regarding the topic of the considered session (the names of suggested reviewers are given in the proposed Agenda, (see Appendix II).

b) work documents: these documents are aimed at providing, for each topic, current information at the time of the meeting; these reports would form the basis of a collective working effort to

c) confirm and/or update the results and conclusions of the reviewers' reports (see a)) and provide recommendations for future efforts.

The Agenda for the meeting is written with the intention to propose to the participants a framework for the preparation of their contributions.

Integral consistency of nuclear cross sections needed for reactor dosimetry

Reactor dosimetry is aimed at providing the capability to properly correlate, interpolate and/or extrapolate integral quantities, such as flux and fluence, fission rates, burn up, damage rates, doses, heating rates, etc.

The determination of flux-fluence neutron spectra is not a primary objective of reactor neutron dosimetry, but a necessary intermediate step in a more general correlation scheme between different, independent integral quantities, for ex. damage rate in a given material exposed at a given temperature in a test reactor to the damage rate of the same material under other exposure conditions. The reaction rates observed in neutron dosimeters are the correlation parameters in this scheme, and the flux-fluence neutron spectra are the corresponding transfer functions. These transfer functions must be defined to a reasonable accuracy and this requires in turn that differential-energy cross sections of dosimetry nuclear reactions be known accurately enough.

The above concept and the role of reference neutron fields are best illustrated (Fig. 1) by considering the crucial issue of applying to commercial power plants the materials property-change data obtained in high flux test reactors.

Fluence neutron spectra for test reactors are usually unfolded from a set of measured reaction rates R_i. Property changes observed in a series of such fluence neutron spectra are correlated by adjustment of the damage functions so as to provide consistent predictions for commercial plants. Such an adjustment procedure will be biased if the unfolded fluence neutron spectra are not consistent with the ones resulting from design computations.

The most practical and accurate way to avoid such bias is to adjust the differential-energy cross sections of dosimetry reactions within their assigned uncertainties so as to reproduce reaction rates observed in a set of benchmark and standard neutron fields. This is possible provided that:

1) the reference neutron fields span a relevant range of spectral hardness and shape,

2) the neutron spectrum characterization of these reference neutron fields is complete and accurate, and that

3) the available integral reaction rate data are accurate.

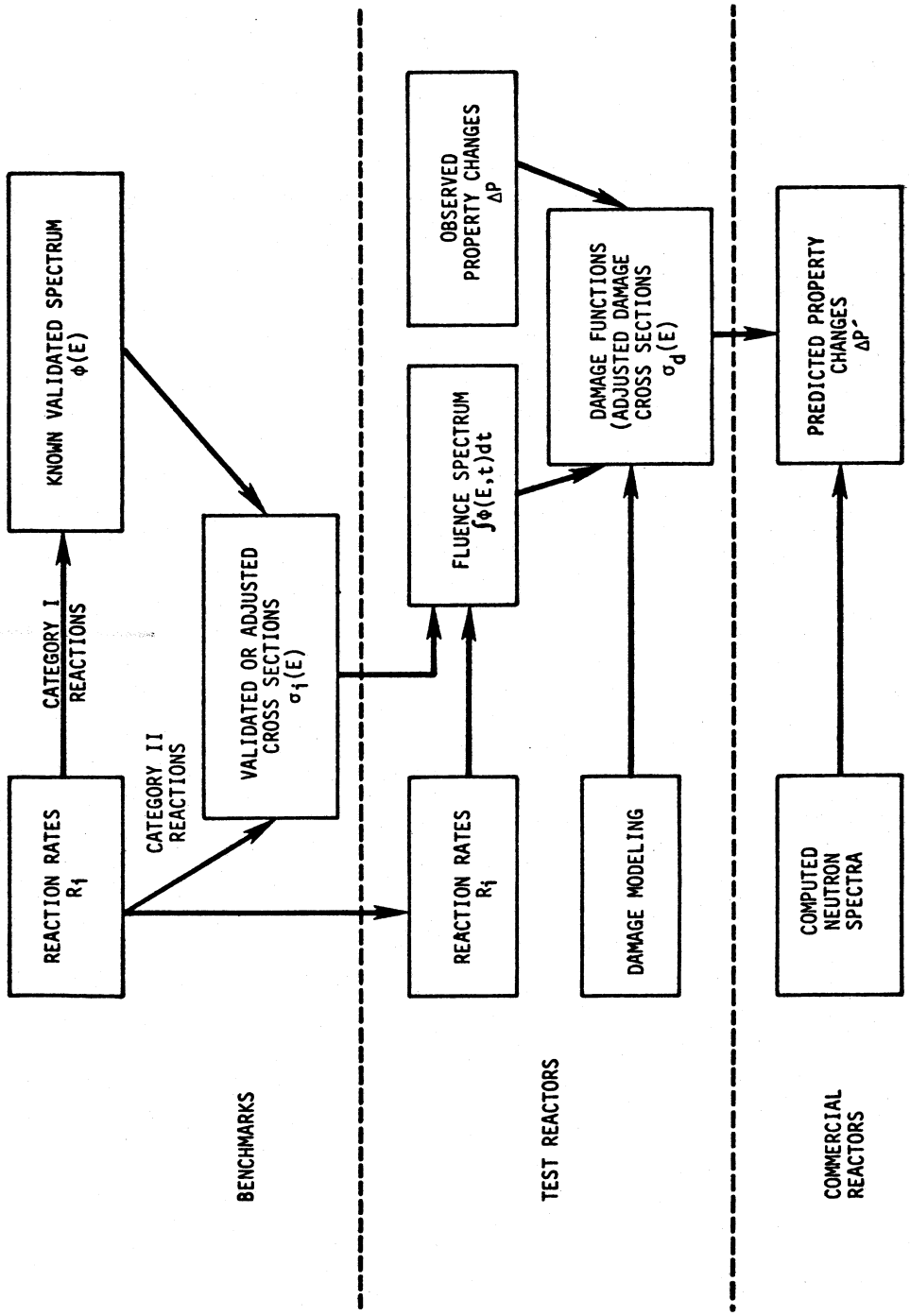
KNOWN VALIDATED SPECTRUM
$\phi(E)$
CATEGORY I
REACTIONS
REACTION RATES
R_i
CATEGORY II
REACTIONS
VALIDATED OR ADJUSTED
CROSS SECTIONS
$\sigma_i(E)$
BENCHMARKS
OBSERVED
PROPERTY CHANGES
ΔP
FLUENCE SPECTRUM
$\int\phi(E,t)dt$
REACTION RATES
R_i
DAMAGE FUNCTIONS
(ADJUSTED DAMAGE
CROSS SECTIONS
$\sigma_d(E)$
DAMAGE MODELING
TEST REACTORS
PREDICTED PROPERTY
CHANGES
$\Delta P'$
COMPUTED
NEUTRON
SPECTRA
COMMERCIAL
REACTORS

Fig. 1

Preliminary Agenda for the Consultants' Meeting on Integral Cross-Section Measurements in Standard Neutron Fields

Vienna, 15 - 19 November 1976

SESSION 1. Overview

1.1. Benchmark neutron fields for reactor dosimetry (J.A. GRUNDL, NBS).

1.2. Power reactor pressure vessel benchmarks (F. RAHN, EPRI).

1.3. Metrology requirements for standard neutron fields (R.D. VASIL'EV, MFTI).

SESSION 2. Spectral characterization of benchmark neutron fields

2.1. Review: Spectral characterization by combining neutron spectroscopy, analytical calculations and integral measurements (W.N. MC ELROY, HEDL).

2.2. In-pile neutron spectroscopy: Status (G. de LEEUW, CEN-SCK).

2.3. Contributed papers.

2.4. Recommendations

(Example: do we need more benchmarks?).

SESSION 3. Integral data in benchmark neutron fields

3.1. Review of the data and their accuracy (A. FABRY, CEN-SCK).

3.2. Contributed papers.

3.3. Data assessment (on the basis of presentations 3.1 and 3.2, attempt to derive a set of presently recommended integral data and their accuracies.

SESSION 4. Differential cross section data for reactor dosimetry

4.1. Review of the data and their accuracy (D. SMITH, ANL, M. VLASOV, IAEA).

4.2. Contributed papers (new differential data).

4.3. Recommendations.

SESSION 5. Comparison of differential and integral cross section data measured in benchmark neutron fields

5.1. Compilation and comparison of calculated and measured integral cross sections (W. ZIJP, RCN).

5.2. Contributed papers.

5.3. Discussion of the status of the compilation and its updating.

SESSION 6. Validation and adjustment of differential cross sections on the basis of integral data

6.1. Methodology: General proposals (U. FARINELLI, CSN Casaccia).

Codes for cross section adjustment (R. DIERCKX, EURATOM, ISPRA).

6.2. Contributed papers.

6.3. Discussion and recommendations.

SESSION 7. Conclusions and recommendations to the IAEA

(to be drafted by working groups after sessions 1–6 and discussed in plenary in session 7).