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Summary Report on the Consultants' Meeting
on

Neutron Sources Spectra for EXFOR

IAEA Headquarters, Vienna, Austria
13 - 15 April 2011

Prepared by

S.P. Simakov, IAEA, Vienna, Austria
and
F. Käppeler, KIT, Karlsruhe, Germany

October 2011

IAEA Nuclear Data Section, Vienna International Centre, A-1400 Vienna, Austria

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Nuclear Data Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
A-1400 Vienna
Austria

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Abstract

The participants highlighted the importance of complementing the averaged cross section data already stored in EXFOR by the incident neutron energy spectra. They shared their experience on measurement and simulation of neutron fields produced at reactors and accelerators over a wide energy range. The source characteristics, format and rules needed for storage in EXFOR were discussed. The participants submitted the numerical information on spectra that will essentially increase the number of "complete" data sets in EXFOR. The report additionally provides an overview of (i) neutron production cross sections and thick target yields missing from the EXFOR database; (ii) codes for neutron spectra calculations; (iii) informational resources for reactor, radioactive and spallation neutron sources; (iv) codes for spectrum unfolding and (v) EXFOR compilation rules for the Maxwellian averaged cross sections measured for the reactor and astrophysical applications.

October 2011

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Introduction

The purpose of the Meeting was to make the EXFOR database (http://www-nds.iaea.org/exfor/) more representative and complete by including information on neutron source energy spectra to allow correct interpretation of measured results, complete specification of physical process, comparison with other measurements and theoretical predictions. The necessity of knowing neutron spectra mainly appears when a neutron source has a relatively broad (non-monoenergetic) energy distribution and the cross section (XS) under investigation varies significantly over this interval.

The EXFOR database has already accumulated thousands of XS measured with such neutron sources produced by fission reactors (cold, thermal, fast and filtered fields) and by accelerators (thick targets, quasi monoenergetic fields), but practically all of them lack the source energy spectra except for a few cases (Entries 31868, 21816 and 22850).

Energy spectra from accelerator driven neutron sources are traditionally compiled in the EXFOR database even if they were not used for XS measurement. This is very valuable information for illustrating the typical energy distributions and intensities as well as for verifying charged particle induced reaction models.

Presentations and Proceedings

A Consultants' Meeting on "Neutron Source Spectra for EXFOR was held at IAEA Headquarters, Vienna, Austria from 13-15 April 2011. Seven consultants (T. Belgya, P. Bém, M. Drosg, O. Gritzay, F. Käppeler, M. Pillon and S.M. Qaim) attended the Meeting, the IAEA was represented by R. Forrest (Head, Nuclear Data Section), S.P. Simakov (served as Scientific Secretary), N. Otuka, V. Semkova and V. Zerkin. H. Harada, invited as a Consultant, was not able to come because of the Tsunami disaster in Japan and the related accident at the Fukushima Nuclear Power Station (his presentation was delivered by S.P. Simakov and was discussed by participants; the summary is included in this Report and the neutron spectra of the YOYOI reactor will be compiled in EXFOR).

S.P. Simakov welcomed the participants on behalf of R. Forrest, Head of the Nuclear Data Section of IAEA. He acknowledged the consultants expertise and emphasized the importance of improving the EXFOR database through the collection and inclusion of the source neutron spectra.

S.M. Qaim was elected as Chairman of the meeting and F. Käppeler as Rapporteur. The approved Agenda is attached (Appendix 1), as well as a list of participants and their affiliations (Appendix 2).

During the first day and half of the second day, participants gave summary presentations of relevant research activities within their institutes (Appendix 3). They shared their experience in the use of thermal and fast reactors, filtered beam techniques, low-energy accelerators, cyclotrons, linacs and radioactive isotopes as neutrons sources for measuring the data for nuclear physics, astrophysics and dosimetry, radionuclide production for medical diagnostics and therapy, energy-related technologies and materials research. They also presented the methods and techniques that were used to characterize the neutron source spectrum distributions.

Discussions then took place on recommendations and conclusions.

The Nuclear Data Section acknowledged all participants for cooperation and supplying of detailed information.

Conclusions and recommendations

Information on neutron source spectra and on the effective spectra at the sample position are a fundamental part of cross section measurements. Therefore, authors are advised to include such information in their publications and to provide numerical spectral information related to published work for inclusion in the EXFOR database.

The responsibility for the full description of the neutron source spectrum and related information to be included in EXFOR rests with the experimentalists. Experimental data should be provided for a complete characterization of the spectra according to the existing EXFOR rules and the recommendations listed below. Also codes for spectrum calculation (including the respective data bases used) are helpful in this respect and should be provided wherever appropriate.

The present practice of listing "representative" energies for broad spectra was discussed. In view of the fact that this concept may be sensitive to the cross sections and the spectra, representative energies should be supplemented with more specific spectral information.

The practice of including information on spectra that have not been used for cross section measurements so far, e.g. spectra obtained with triton-beams, should be continued – *the list of the energy differential neutron emission cross section and thick target yields, still missing in EXFOR, are collected in Appendix 4 and distributed among NRDC centers for compilation*.

Specific Recommendations for the submission of spectral information in EXFOR:

- provide spectra with appropriate resolution over the energy range covered in the experiment; provide spectrum uncertainties in E_n and $N(E_n)$, possibly with correlations;
- provide a good estimate of the absolute flux for supporting the plausibility of results;
- provide links to documented neutron spectra from averaged cross section Entries to avoid duplication.

Technical aspects:

- Codes for neutron spectrum calculations, e. g. NEUSDESC, PINO, TARGET, FNG Source Routine, DROSG2000, FILTER should be collected by NDS - *a short description of the codes, the range of applicability and their availability including hyperlinks are summarised in Appendix 5*.
- Additional links to the data base for reactors, which is kept by the Physics Section, and to the Spallation Benchmark site would be helpful as well – *an overview of the information resources is given in Appendix 6*.
- List of and links to codes and databases for neutron spectrum unfolding via multi-foil activation techniques should be provided by NDS – *summarised in Appendix 7.*
- NRDC Network must develop rules and formats for the storage of spectrum information in EXFOR (flux units, references to original publications) – *the NRDC proposals are presented by O. Gritzay et al. in Appendix 3 and in the NRDC working paper WP2011-22* http://www-nds.iaea.org/nrdc/nrdc_2011/ .
- Compilers of EXFOR entries are encouraged to contact authors to clarify inconsistencies or to add comments in case of doubtful data.
- It is noted that MXW average data must be given for the proper moderator temperature. These Entries should also include the normalization convention used ($2/\sqrt{\pi}$) – *the overview of EXFOR rules for compilation of the MXW averaged data is given in Appendix 8*.

The overall conclusion:

The participants expressed their views on the importance to store neutron spectra data as a supplement to energy averaged or smoothed cross section data measured in such neutron fields. They shared their experience of what neutron source information is necessary to be compiled in EXFOR for the correct interpretation of the energy averaged data.

They submitted the numerical information on neutron spectra for compilation together with the relevant cross sections. In this way the number of "complete" data sets in EXFOR will increase from 4 up to several thousand. The participants discussed rules and formats for convenient storage of incident source information in EXFOR.

The survey concerning information on (i) neutron production cross sections and thick target yields still missing in the EXFOR database; (ii) codes for neutron spectrum calculations; (iii) numerical data for reactor, radioactive and spallation neutron sources; (iv) spectra unfolding codes, and (v) EXFOR compilation rules for the Maxwellian averaged cross sections measured for the reactor and astrophysical applications has been collected by NDS/IAEA and is included in this Report.

APPENDIX 1

Consultants' Meeting on
„Neutron Sources Spectra for EXFOR"

IAEA Headquarters, Vienna, Austria
13 - 15 April 2011
Meeting Room G0E85

AGENDA

Wednesday, 13 April 2011

08:45-09:00	**Registration**
09:00-09:30	**Opening Session** Welcome Address - **Robin Forrest** Administrative Announcements - **Lidija Vrapcenjak** Self introduction of participants, Selection of Chairperson & Rapporteur, Approval of Agenda
09:30-09:45	**Mr Stanislav Simakov,** IAEA - Introduction & Objectives of the Meeting
Thermal, fast and filtered beams from Reactors	
09:45-10:30	**Mr Hideo Harada**, JAEA, Japan - "Measurement of neutron capture cross section of ^{237}Np for fast neutrons" - he could not come due to tsunami disaster in Japan – nevertheless he sent a talk and YAYOI spectra (was presented by S. Simakov)
10:30-11:00	*Coffee break*
11:00-12:30	**Mr Tamás Belgya**, KFKI, Budapest, Hungary - "Guided neutron beams" **Ms Olena Gritzay,** KINR, Kiev, Ukraine - "Neutron spectra after the interference neutron filters at Kyiv research reactor"
12:30-14:00	*Lunch break*
Accelerator driven neutron sources: kev, a few MeV and 14 MeV	
14:00-15:30	**Mr Franz Käppeler,** KIT, Karlsruhe, Germany - "Quasi-stellar spectra for activation studies in astrophysics" **Mr Manfred Drosg**, University of Vienna, Austria - "How did I (not) handle several neutron source related problems during 40+ years of fast neutron work"
15:30-16:00	*Coffee break*
16:00-17:30	**Mr Mario Pillon**, ENEA, Frascati, Italy - "Experimental techniques and computational methods used at the Frascati neutron generator in order to determine the neutron source spectrum " **Summary Discussion**

Thursday, 14 April 2011

Accelerator driven neutron sources: high energies (above 14 MeV)	
09:00-10:30	**Mr Syed Qaim**, FZJ, Juelich, Germany - "Neutron spectrum averaged activation cross section measurements" **Mr Pavel Bém**, NRI, Řez, Czech Republic, "Quasi-monoenergetic p-^{7}Li and white p-D_2O and ^{3}He-D_2O neutron sources"
10:30-11:00	*Coffee break*
Technical issues relevant to the EXFOR compilation	
11:00-12:30	**Ms Olena Gritzay**, KINR, Kiev, Ukraine - "EXFOR formats and rules: present status and proposals how to store neutron sources data" **Mr Naohiko Otsuka**, IAEA - "Possible improvement in EXFOR for spectrum averaged quantities"
12:30-14:00	*Lunch break*
14:00-15:30	**Discussion: Overview of neutron sources and their specific features**
15:30-16:00	*Coffee break*
16:00-17:30	**Discussion: Compilation practice and unresolved issues**
19:00 -	**Social event: visit to restaurant** www.zwoelf-apostelkeller.at

Friday, 15 April 2010

09:00-10:30	**Discussion: Drafting of conclusions & recommendations**
10:30-11:00	*Coffee break*
11:00-12:30	**Discussion: Finalizing conclusions & recommendations**
12:30-14:00	*Lunch break*
14:00-16:00	***Final remarks and end of the Meeting***

APPENDIX 2

Consultants' Meeting on
„Neutron Sources Spectra for EXFOR"

IAEA Headquarters, Vienna, Austria
13 - 15 April 2011
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LIST OF PARTICIPANTS

AUSTRIA

Manfred **Drosg**
University of Vienna
Faculty of Physics
Boltzmanngasse 5
1090 Vienna
Tel.: +43 1 320 6228
E-mail: manfred.drosg@univie.ac.at

CZECH REPUBLIC

Pavel **Bém**
Nuclear Physics Institute ASCR pri., Řež
plc. Husinec Rez 130
CZ-250 68 Rez
Tel.: +42 2 6617 2105
Fax: +42 2 2094 1130
E-mail: bem@ujf.cas.cz

GERMANY

Syed M. **Qaim** (Chair)
Institut fuer Nuclearchemie
Forschungszentrum Juelich GmbH
Postfach 1913
D-52425 Juelich
Tel.: +49 2461 61 3282
Fax: +49 2461 61 2535
E-mail: s.m.qaim@fz-juelich.de

HUNGARY

Támas **Belgya**
Hungarian Academy of Sciences
Head of Department
Department of Nuclear Research
Institute of Isotope
H-1525 Budapest
Tel.: +36 1 392 2539
Fax: +36 1 392 2584
E-mail: belgya@iki.kfki.hu

ITALY

Mario **Pillon**
ENEA-CR Frascati
Frascati Neutron Generator
Via Enrico Fermi 45
I-00044 Frascati
Tel.: +39 069 4005326
Fax : +39 069 4995147
E-mail: mario.pillon@enea.it

UKRAINE

Olena **Gritzay**
Institute for Nuclear Research
Prospekt Nauky 47
03680 Kyiv
Tel.: +380 44 525 3987
Fax: +380 44 525 4463
E-mail: ogritzay@kinr.kiev.ua

JAPAN

Hideo **Harada** *
Japan Atomic Energy Agency
Division of Nuclear Data and Reactor Engineering
Nuclear Science and Engineering Directorate
Shirakata Shirane 2-4
Tokaimura, Naka, IBARAKI 319-1195
Tel.: +81 29 282 6789
Fax: +81 29 282 5927
E-mail: harada.hideo@jaea.go.jp

**) – was not able to attend*

CONSULTANT

Franz **Käppeler** (Rapporteur)
Karlsruhe Institute of Technology
Campus Nord
Institut fűr Kernphysik
Postfach 3640
76021 Karlsruhe
Tel.: +49 7247 823991
E-mail: franz.kaeppeler@kit.edu

IAEA Staff

Robin A. **Forrest**
Head, Nuclear Data Section
Division of Physical and Chemical Sciences
Tel.: +43 1 2600 21709
Fax: +43 1 2600 7 21709
E-mail: r.forrest@iaea.org

Stanislav **Simakov** (Scientific Secretary)
Head of Nuclear Data Services Unit
Nuclear Data Section
Division of Physical and Chemical Sciences
Tel.: +43 1 2600 21717
Fax: +43 1 2600 7 21717
E-mail: s.simakov@iaea.org

Naohiko **Otsuka**
Nuclear Data Section
Division of Physical and Chemical Sciences
Tel.: +43 1 2600 21715
Fax: +43 1 2600 7 21715
E-mail: n.otsuka@iaea.org

Valentina **Semkova**
Nuclear Data Section
Division of Physical and Chemical Sciences
Tel.: +43 1 2600 21727
Fax: +43 1 2600 7 21727
E-mail: v.semkova@iaea.org

Viktor **Zerkin**
Nuclear Data Section
Division of Physical and Chemical Sciences
Tel.: +43 1 2600 21714
Fax: +43 1 2600 7 21714
E-mail: v.zerkin@iaea.org

APPENDIX 3

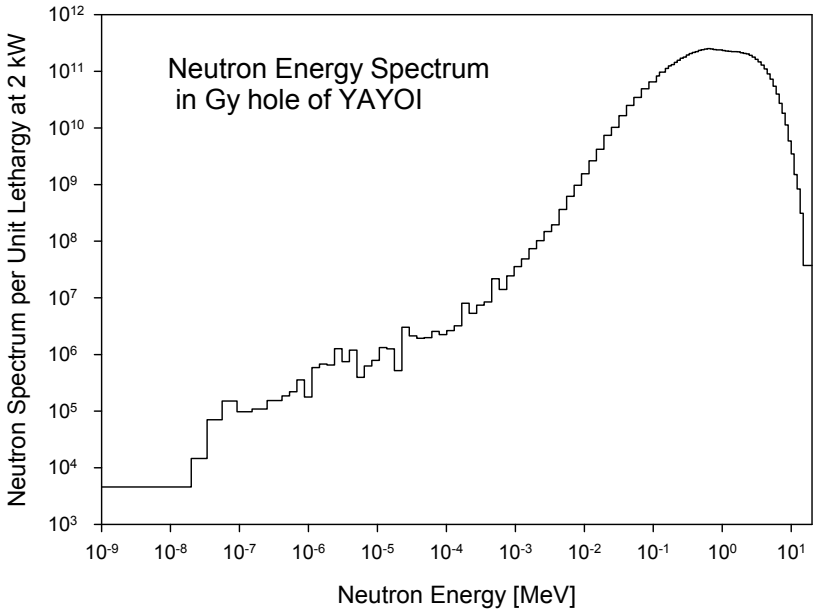
SUMMARIES OF PRESENTATIONS

The representative neutron energy and the corresponding capture cross section in activation measurement using fast neutron spectrum

Hideo Harada
Japan Atomic Energy Agency, 2-4 Shirane, Shirakata, Tokai-mura, Naka-gun,
Ibaraki 319-1195, Japan

The activation method using reactor neutrons has been utilized for the measurement of the capture cross sections. In order to deduce thermal neutron capture cross sections and resonance integrals, the Westcott convention has been successfully utilized, where a well moderated neutron spectrum is approximated by a Maxwell distribution plus a 1/E component. Since the energy dependence of neutron capture cross section is approximated by 1/v at thermal energy region for most nuclei, the thermal neutron capture cross section at thermal neutron energy (= 25.3 meV) is deduced.

On the other hand, a fast neutron spectrum cannot be approximated by a Maxwell distribution or simple formula. The energy dependence of the neutron capture cross section is also not approximated by 1/v or a simple formula. Therefore, it is difficult to give a mathematically rigid definition on the representative neutron energy used for measurements using a fast neutron spectrum. The simple average of neutron spectrum does not always give an appropriate representative neutron energy. In measurements using fast neutrons supplied by the Yayoi reactor [1], the representative neutron energy has been defined by a numerical calculation method. The representative neutron energy was defined as the energy where the integrated value of the reaction rate from 0 MeV to that energy is 50 % of the total reaction rate.



In order to make this calculation possible, the neutron spectrum used for the irradiation is required. The inclusion of the neutron spectrum in EXFOR makes this kind of evaluation possible. As an example, neutron spectrum used for the measurement at Yayoi reactor is shown in Fig. 1.

Fig. 1. Neutron spectrum in Gy hole of the Yayoi reactor.

Reference

1. H. Harada, S. Nakamura, Y. Hatsukawa et al., "Measurements of Neutron Capture Cross Section of ^{237}Np for Fast Neutrons", J. Nucl. Sci. Tech. 46, 460-468 (2009).

Cold neutron source spectra at the Budapest PGAA-NIPS facilities

T. Belgya, L. Szentmiklósi and Z. Kis
Institute of Isotopes Hungarian Academy of Sciences
H-1525 POB 77, Budapest, Hungary

The Budapest Research Reactor's Prompt Gamma Activation Analysis (PGAA) and Neutron-Induced Prompt gamma Spectroscopy (NIPS) facilities [1] were significantly upgraded during the last few years. The higher neutron flux, achieved by the partial replacement and realignment of the neutron guides, made feasible the automation and specialization of the two experimental stations. A new neutron flux monitor, computer-controlled beam shutters and a low-level counting chamber have been put into operation to assist with in-beam activation experiments. An automatic sample changer has been installed at the PGAA station, while the NIPS station was redesigned and upgraded with a Compton suppressor to use for the non-destructive analysis of bulky samples. In the near future the latter setup will be completed with a neutron tomograph and a moving table, to make possible the Neutron Radiography/Tomography-driven PGAA (see [2] and Figure 1).

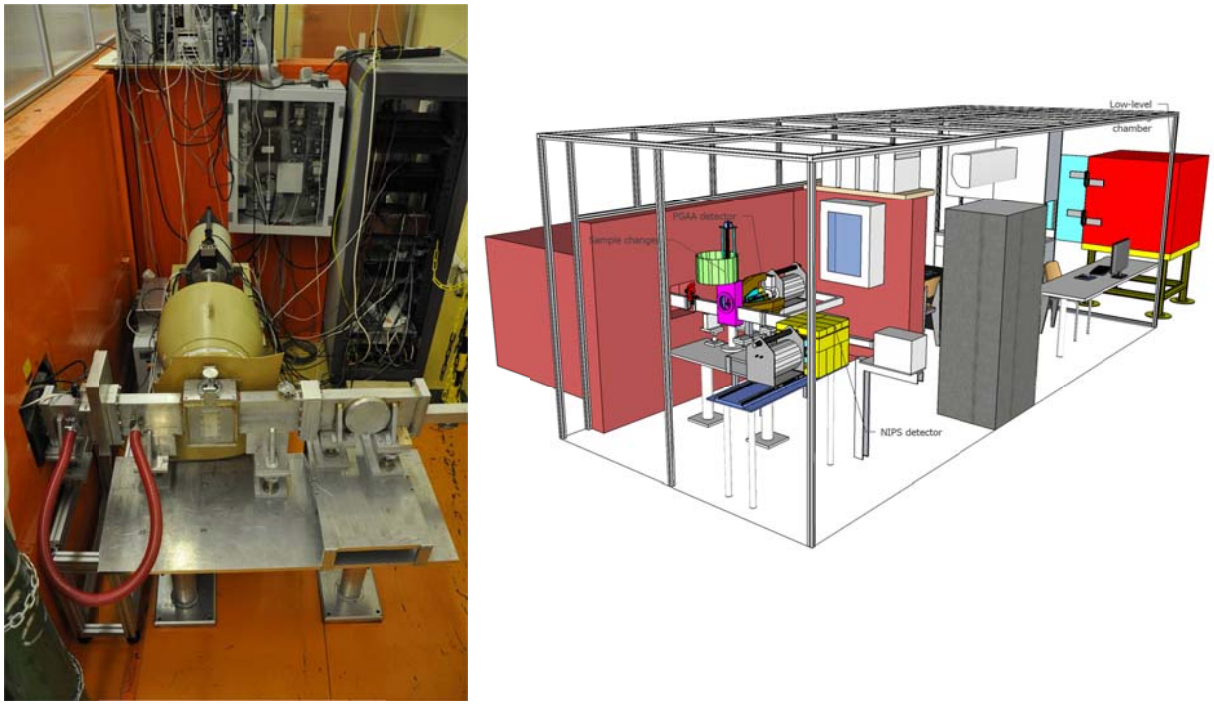


Fig. 1. The current status of PGAA-NIPS facilities.

Due to the changes in the neutron guide it is timely to repeat the neutron beam characterization using detailed time-of-flight (TOF) experiment. These experiments have been performed using a 2D Position Sensitive Neutron Detector (PSND). The PSND detector and the acquisition system [3] were provided by the Neutron Optics Department of Research Institute for Solid State Physics and Optics of the Hungarian Academy of Sciences. A flight path of 190 cm was used from the chopper blade rotated at 1250 rpm. On the chopper blade two 1.2 mm openings were set at 180 degree from each other. A standing slit of 1.2 mm was covered with a 5 mm^2 round-hole collimator at the middle of the upper PGAA beam. The dwell time of the acquisition system was set to 5 μs and the 2D positions were collected in 2048 channels. This made up a 256×256×2048 matrix with a position resolution of about 0.7 mm. To decrease the size the matrixes were re-binned into 512 channels, i.e. for 20 μs dwell time. The time of flight spectrum was calibrated with BeO and graphite Bragg filters for wavelength.

Fig. 2 shows neutron spectra obtained in this experiment. The pinhole geometry of the experimental system enables us to look into deep to the neutron guides and observe reflections arriving far back in the guide. Two dimensional position distributions are shown in Fig. 3 that were cut from time regions of the 3D matrix for the corresponding wavelengths regions shown in Angstrom for the figures.

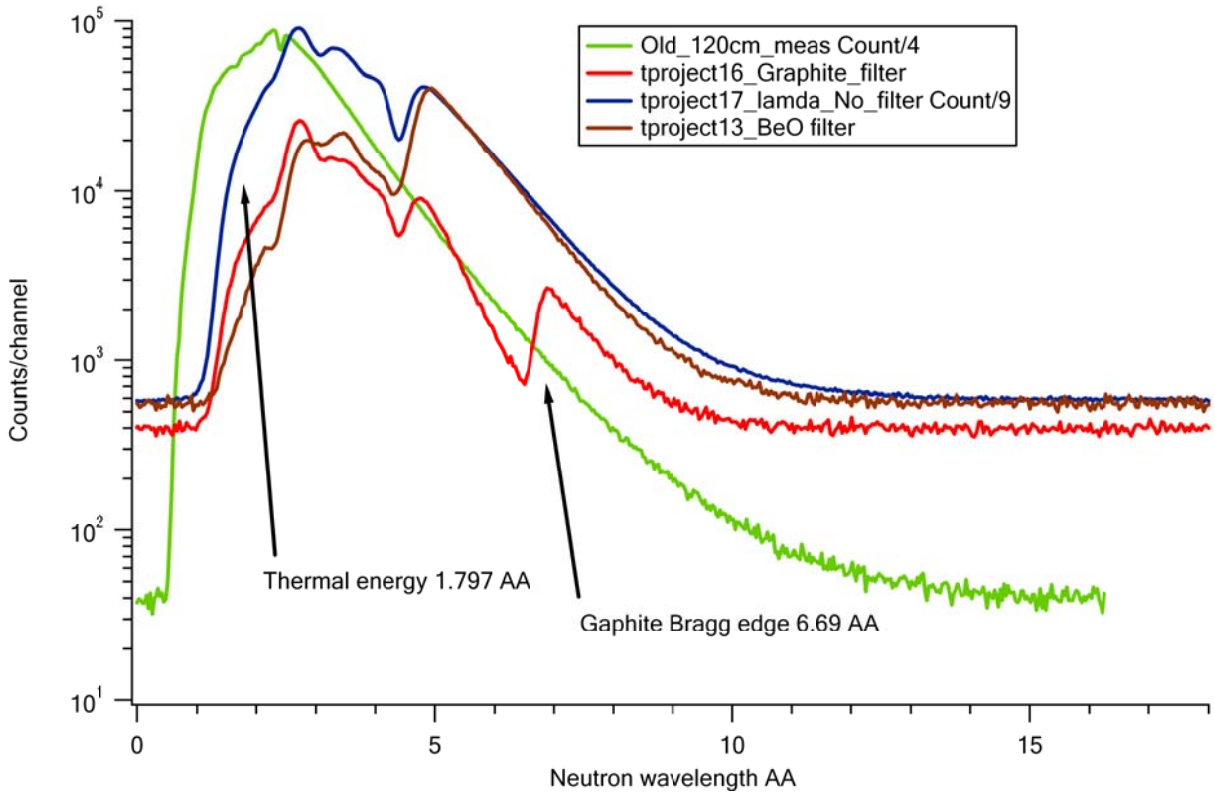


Fig. 2. Comparison of the new neutron spectra to the old spectrum taken in 2003 before the installation of new supermirror guides. It can be seen that the new guides cut the higher energy component of the beam and enhances the low energy component.

The energy (wavelength) distributions shown in Fig. 2 must be corrected for the energy dependent efficiency of the detector and the uncertainty of the wavelength measurement due to the finite time transfer function of the chopper and due to the different depth of absorption of the same energy neutrons.

All of this data can be supplied for the EXFOR database after publication in refereed journal. In case of thin, homogeneous and uniform targets with 1/v nuclei gives energy independent production rate ratios for different isotopes of the target. This can be used to determine thermal equivalent cross sections using the comparator method. This method is used in the simplified analysis of PGAA [4] and cross section measurements discussed by Belgya [5,6]. The PGAA-NIPS instrument is widely used to determine thermal equivalent cross sections that as discussed in [5, 6]. In the future cross section measurements the TOF method will be available to study deviations from the 1/v behaviour.

Finally a 4D representation of the TOF-matrix is shown in Figure 4.

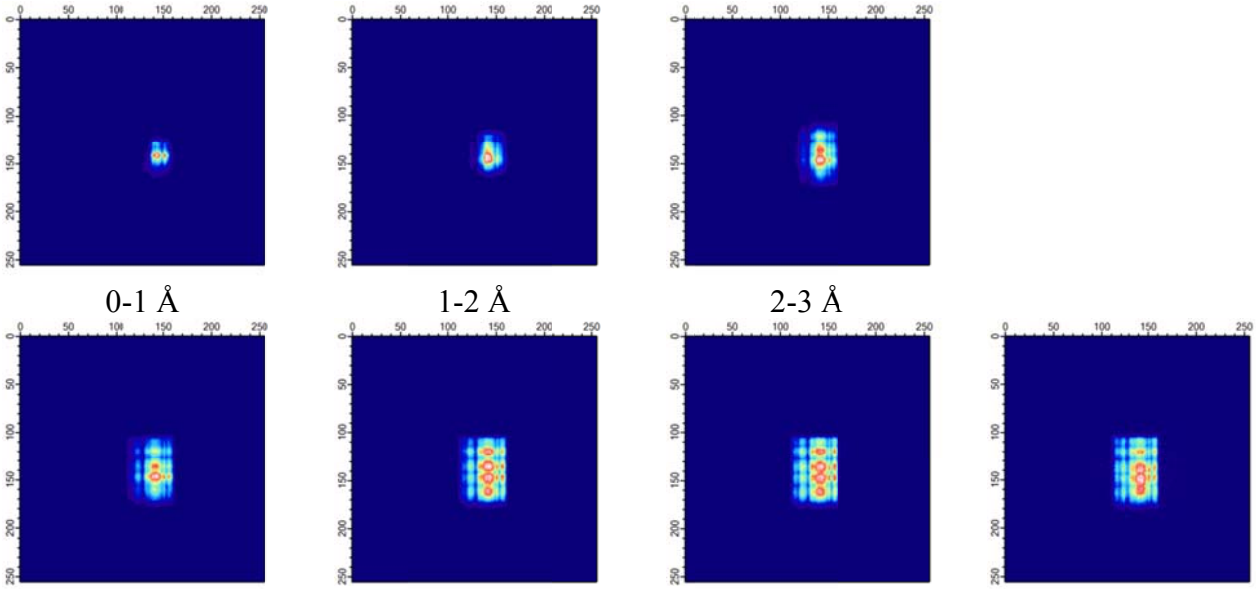


Fig. 3. Cuts from the 3D (X-Y-t) matrix for the shown Angstrom region corresponding time regions. The white square shows about a 2 cm x 2 cm area that is equal with the collimator sizes situated upstream of the beam, before the chopper blade. Repeated square regions in the figures belong to reflection of the neutron beam on the mirror walls. The guide is curved to the right looking in the directions of the neutrons. The picture shows the absorbed neutron positions from looking to it from the beam direction. The pinhole geometry reverses the up-down and left-right directions.

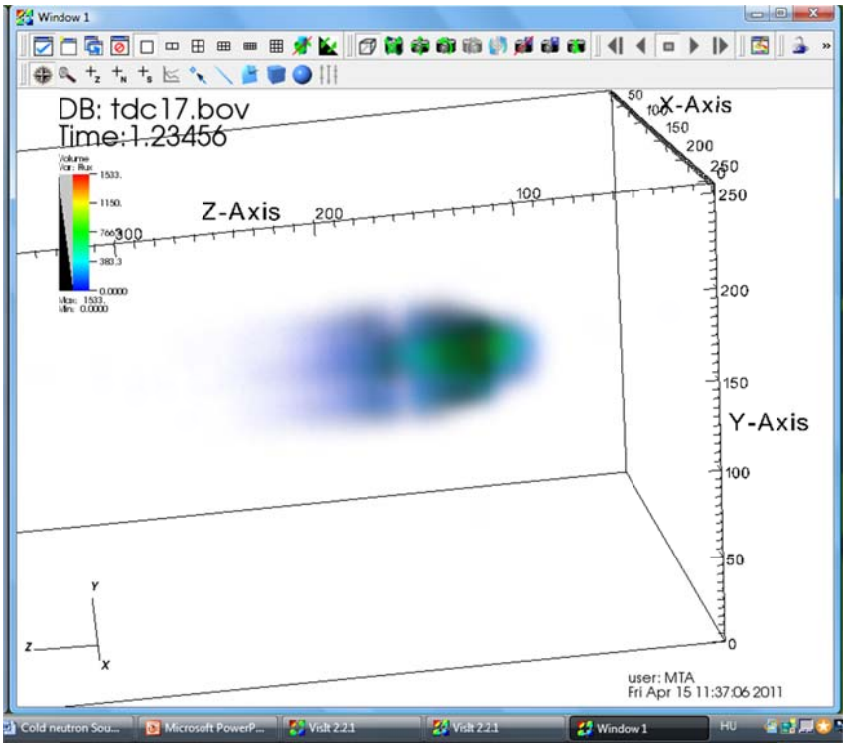


Fig. 4. 4D representation of the X-Y-T(Z axis on the figure) TOF matrix. The colour codes the number of neutron counts analyzed by the 2D position sensitive neutron detector at a given flight time T. The figure was made by the help of the VisIt 2.2.1 program.

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6. T. Belgya, *Determination of thermal radiative capture cross section*. (Eds. T. Belgya). Budapest, Hungary: (II-HAS), (2010), 115-120; http://www.iki.kfki.hu/efnudat/Proc.pdf

Neutron spectra after the interference neutron filters at Kyiv research reactor

Olena Gritzay
Institute for Nuclear Research of NAS of Ukraine, Prospekt Nauky, 47, Kyiv, Ukraine, 03680

At the Kyiv Research Reactor (KRR) a neutron filtered beam technique (NFBT) has been used for more than 30 years and its development is continued. Usage of the NFBT allows an extraction of neutron lines in the energy range from thermal energy to several hundred keV with intensity 10^6 –10^8 n/cm^{2}s from the white reactor spectrum. Such high intensity gives possibility to provide the measured values with good statistics and hence the averaged neutron cross sections with rather high accuracy: the total neutron cross sections with accuracy 1% and better, the neutron scattering cross sections with 3-6%, the neutron capture cross sections with 5-6%. These data may be very useful for practical applications and for improvement of the existing evaluated neutron files. It is necessary to remember that all data obtained by NFBT are averaged over neutron filtered spectrum. Thus information on the neutron spectrum after the filter is very important.

Let consider the neutron spectra after the interference neutron filters, which are used in fundamental investigations at KRR. The WWR-M KRR is a light water moderated and cooled tank-type reactor with a beryllium reflector. The reactor used 36% enriched uranium-235 WWR-M2 fuel assembly, today the replacement by 20% U^{235} fuel has been finished. The nominal thermal power is 10 MW, neutron flux in the core is about 10^{14} n/cm^{2}s. The reactor has 27 vertical and 10 horizontal experimental channels (HEC) for performing scientific and applied researches. Currently three of ten HEC are employed in the experimental investigations with NFBT.

The main idea of neutron filter development is the use of large quantities of materials, the nuclei of which have the deep interference minima in their total neutron cross sections. Transmission of the reactor neutrons, having a white energy distribution, through thick layer of such material yields the quasi-mono-energetic neutron lines.

Energy of a such quasi-mono-energetic neutron line may be in the range from thermal to several hundred kilo-electron-volts, and its intensity may reach 10^5- 10^8 n/cm^{2}s. Besides essential advantages NFBT has also some disadvantages: presence of the parasitic (non desirable energy) lines and γ-ray background in the filtered neutron spectrum.

To get only one quasi-mono-energetic neutron line (so-called, neutron filter beam with high purity) a composite filter is usually used. It consists of the “main filter material” and additional materials, for which the resonance maxima in their total neutron cross sections coincide with interference minima for filter material, except the most deep interference minimum energy.

The basic requirements for the neutron filter beam are the following:

- purity of the main energy line in neutron spectrum has to be close to 100%, as much as possible;
- neutron intensity has to be sufficient to obtain the necessary accuracy in experiment;
- construction and composition have to provide the minimal gamma-background;
- in necessary cases the construction and composition have to allow increasing or reducing of the base line width without essential worsening of filter quality;
- amount of enriched isotopes in filter components has to be as few as possible.

A wide set of natural elements and enriched isotopes are used as components for neutron filters in the Neutron Physics Department (NPD) at the KRR:

- natural elements: Si, Al, V, Sc, S, Mn, Fe, Ti, Mg, Co, Ce, Cr, Rh, Cu, B, Cd, LiF.
- enriched isotopes: ^{52}Cr (99.3%), ^{54}Fe (99.92%), ^{56}Fe (99.5%), ^{57}Fe (99.1%), ^{58}Ni (99.3%), ^{60}Ni (92.8 – 99.8%), ^{62}Ni (98.04%), ^{80}Se (99.2%), ^{10}B (85%), ^{7}Li (90%).

Availability of such wide set of materials, especially enriched isotopes, allows the creation of a unique set of neutron filters, providing more than 10 neutron lines in the energy range from thermal energy to several hundred keV with intensity up to $10^6 – 10^8$ n/cm^{2}s. This is much more than other methods (time of flight or others) can provide. The wide set of filter materials also allows us to modify filter parameters. So, the filtered neutron beams with the same main neutron line may have different parameters:

- purity (ratio of the main neutron line intensity to the intensity of all other lines);
- width of the main neutron line;
- shape of the main neutron line;
- intensity of main neutron line.

Chosen parameters depend on particular task. For example, if the measurement of the total neutron cross section is carried out using a hydrogen recoil counter, the purity is not very important, since we can separate effects from the different lines. If the measurement of the averaged radiative capture cross section is carried out using activation method, the purity is very important, since we are not able to separate effects from different lines. Hence in these two cases we have to use the different variants of filter though the energy of the main neutron line is the same.

The filter component optimization procedure to get the highest intensity of the main energy line and simultaneously the most possibly low intensity for parasitic lines in the filtered spectrum includes next three main steps:

- 1-st step: simulating calculation of the neutron filtered spectra;
- 2-nd step: creation of the filter with the calculated amount of the chosen components;
- 3-rd step: experimental testing of the created filter.

If necessary the sequence of these steps could be repeated to attain the desired quality of the filter.

1st step: *simulating calculation of neutron filtered spectra using information from ENDF.*

For simulating calculation of the neutron spectra formed by filters, special computer package FILTER was developed. It allows calculation for practically any material and isotope combinations to get the filtered neutron spectrum with necessary energy. The code allows obtaining two energy dependent quantities which image the filtered neutron spectrum:

(i) neutron transmission T(E) multiplied by the incident reactor neutron spectrum $\Phi(E)$:

$$F1(E) = T(E) * \Phi(E) \equiv \exp[-\sum n_i * \sigma_i(E)] * \Phi(E), \qquad (1)$$

where n_i – nuclear thickness of the i-th filter component; $\sigma_i(E)$ – total neutron cross section of the i-th nuclide.

(ii) neutron transmission T(E) multiplied by the incident reactor neutron spectrum $\Phi(E)$ and multiplied by the energy dependent cross section $\sigma_{react}(E)$ of the reaction used for neutron detection, i.e. it allows to take into account an efficiency of the neutron detector:

$$F2(E) = T(E) * \Phi(E) * \sigma_{react}(E). \qquad (2)$$

The incident reactor neutron spectrum $\Phi(E)$ is represented by 3 components: Maxwellian, 1/E – low and fission spectrum.
The total neutron cross sections for nuclides $\sigma_i(E)$ were obtained from ENDF libraries at the temperature 300 K in the energy range 10^{-5} eV to 20 MeV in point-wise form using the PREPRO or NJOY99 codes. JENDL-3.3 and ENDF/B-6 or -7 libraries were taken as the basis for forming this special library.

The use of these simple expressions for determination of the neutron spectrum shape after filter is justified by a strict collimation of the neutron filtration system, otherwise it is necessary to calculate a neutron transport between reactor core and sample taking into account all processes (scattering, absorption, etc.). Correctness of this statement was proved by calculation of the neutron spectrum shape after the 24 keV filter with the MCNP4C code. The MCNP4C results were similar to those obtained by our package FILTER.

2nd step: *creation of filter with components chosen by calculation.*

As a rule, when we create a new filter (or improve an existing one) with energy different from thermal, the first filter component is boron-10 to avoid excessive activation of the rest of the filter components. It has location on the beginning of the third beam shutter disk - the most close to core, see Fig. 1. Also in the first three shutter disks the main components of filter are inserted. The filter components, which are planned to alter, are placed, if it is possible, in the outside collimator. To facilitate and to accelerate the procedure of filter changing, special containers for filter were made. These containers are tubes from stainless steel with the length equal the length of beam shutter disks. Collimation materials (ordinarily paraffin with H_3BO_3 and lead) revolve and are replaced by the filter components in these tubes. One variant of the real 59 keV filter is presented in Fig. 2, as an example.

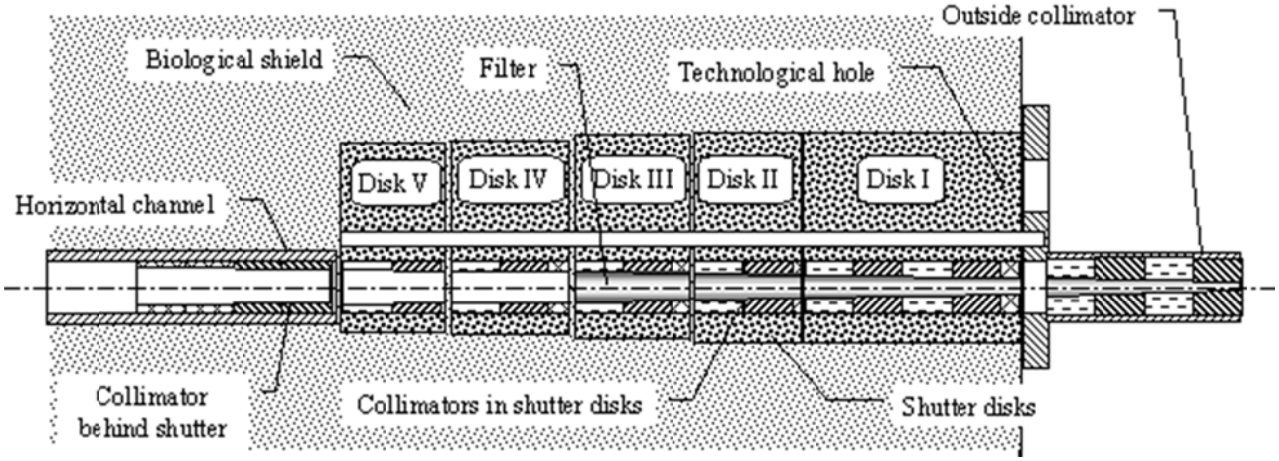

Fig. 1. Location of filter in horizontal reactor channel.

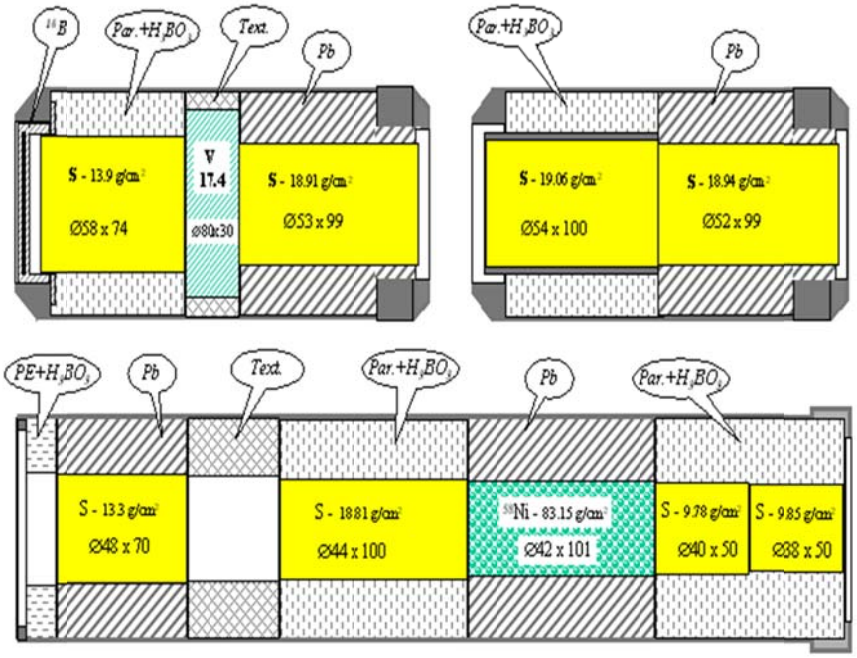

Fig. 2. One of variant of the 59 keV filter. Top – the III and II shutter discs (from left to right), bottom – the first shutter disc.

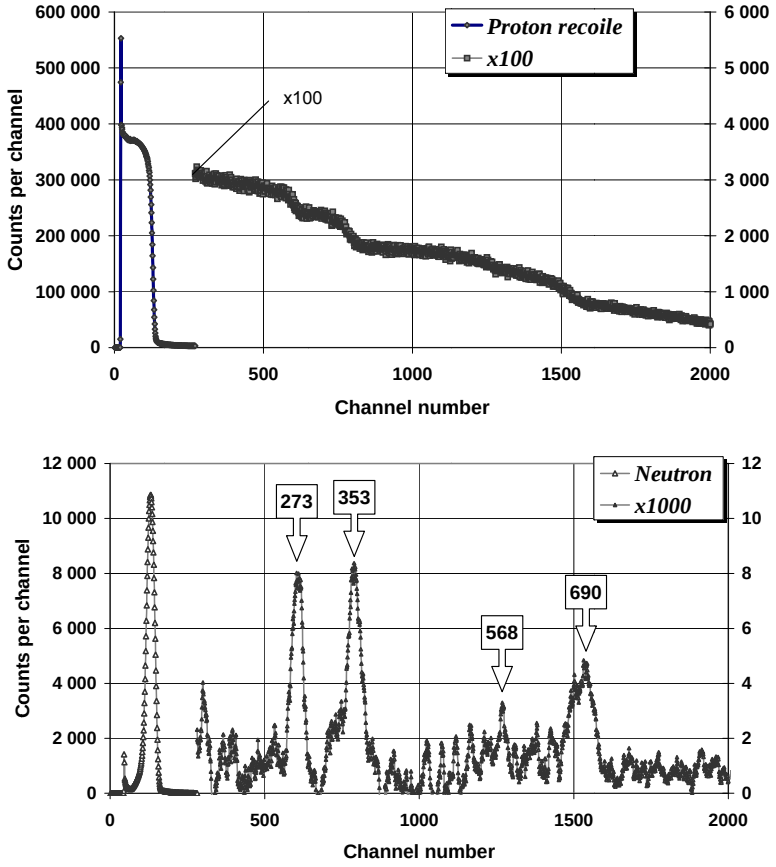

Fig. 3. Instrumental spectrum from proton recoil counter after neutron filter with energy 59 keV (top), and the shape of neutron filtered beam spectrum obtained by differentiation of previous curve (bottom).

Results of calculation with the code FILTER have shown that if we take ^{10}B - 0.281 g/cm^2, ^{58}Ni - 83.15 g/cm^2, V - 17.4 g/cm^2, S - 122.55 g/cm^2, Al - 5.4 g/cm^2 we can obtain a neuron filter with energy 59 keV and the purity of this filter will be better than 99%, other admixtures to the main neutron group will be negligible – less than 0.2% for each other line.

3-rd step: *experimental testing of the created filter.*

To demonstrate procedure of experimental testing of the created filter, consider the filter, main components of which are S, ^{58}Ni, V, ^{10}B, and Al. This filter was arranged as it is shown in Fig. 2 (Al was placed in outer collimator) and a neutron spectrum behind it was measured by proton recoil counter. The experimental results are shown in Fig. 3. The bottom part of this Figure displays the shape of neutron filtered beam spectrum obtained by differentiation of apparature spectrum. It seen that contribution of the parasitic lines with higher energy is negligible.

In summary, the type of information available about neutron spectrum after the interference neutron filters at Kyiv research reactor:

1. Neutron spectrum shape in relative units.

Calculated neutron spectrum shape is available for filters with all line energies, if the total neutron cross sections for all components of this filter are presented in the ENDF libraries.
Experimental neutron spectrum shape is available for filters with main neutron energy line $E_n \geq 12$ keV, where we can use hydrogen recoil counters.

2. Absolute intensity of the neutron filtered beam.

We can measure the absolute intensity of the neutron filtered beam in each experiment, but sometimes it is not important. The decision about such measurement depends upon research task.

In experiments with transmission method the information about the absolute intensity of the neutron filtered beam is not important, thus the measurements of the absolute value of the neutron flux are carried out only periodically (as a rule, after the creation a new filter). Usually we use the foil activation method (with foils Au,Mn, etc.).

In experiments with activation method the information about the absolute intensity of the neutron filtered beam is very important, therefore such measurements are carried out systematically, in each experiment. As a rule, the following measurements are carried out:

a) Flux measuring by ^{10}B(n,αγ)^{7}Li* reaction before and after irradiation. Flux monitoring during irradiation using He-3 counter; areas under He-3 peaks are used as the weight function for the flux averaging.
b) Flux measuring using the foil activation method (In, Au, …) during irradiation.

At the KRR a set of the neutron filtered beams was created by using the composition interference neutron filters. There were obtained about ten quasi-mono-energetic neutron lines with energies 2, 3.5, 7.5, 13, 24, 54, 59, 133, 149 keV with the intensity between 10^5 n/cm^{2}s and 10^8 n/cm^{2}s. There are several variants of the same filtered beam (purity, width of neutron line, intensity, etc.), that depend on the given research task. Information about parameters of the neutron filtered beams may be available in calculation and/or experimental forms. The new filters and the new variants of the existing filters are in development. The filtered neutron beams at the KRR are used and planned to be used in future for measuring the averaged neutron data with high accuracy.

Quasi-stellar spectra for activation studies in astrophysics

F. Käppeler
Karlsruhe Institute of Technology, Karlsruhe, Germany

Almost all of the elemental abundances between Fe and the actinides are produced by neutron reactions. The essential mechanisms are the slow (s) neutron capture process, which occurs in the He (and C) burning phases of stellar evolution and the rapid (r) neutron capture process that is characterized by explosive scenarios, presumably supernovae. Typical neutron capture times are years for the s- and milliseconds for the r-process, slow or rapid compared to average γ-decay times. Consequently, the reaction path of the s-process follows the valley of stability, and the so-produced abundances are mostly determined by the effective (n,γ) cross sections averaged over the stellar neutron spectrum. Because of the high density in the stellar interior, neutrons are instantly thermalized, resulting in a Maxwell-Boltzmann distribution. Thermal energies of current s-process scenarios range from kT=8 keV in low mass stars to 90 keV in massive stars. The required Maxwellian averaged (n,γ) cross sections (MACS) can either be determined by folding the energy-differential cross sections obtained in time-of-flight experiments with the stellar spectrum or by activation in a quasi-stellar neutron field. The latter method provides an attractive possibility for such measurements and has been extensively used so far.

The production of quasi-stellar spectra under laboratory conditions rests on the properties of (p,n) reactions. Near threshold these reactions provide kinematically collimated neutrons in forward direction with spectral distributions that are determined by the velocity of the compound system and the square of the mass ratio, $m_n/(A+m_n)$. Suited cases are the (p,n) reactions on ^{7}Li, ^{3}H, and ^{18}O, which yield quasi-stellar spectra for thermal energies kT = 25, 52, and 5 keV, right in the relevant temperature range of the s process. With proton beam currents of 100 μA, comparably large neutron source strengths of $3\cdot10^{9}$, $2\cdot10^{8}$, and $2\cdot10^{5}$ s^{-1} can be achieved with these reactions. So far, most activation measurements have been carried out with the ^{7}Li(p,n)^{7}Be reaction, which provides the highest neutron yield and has the additional advantage that it fits best to the temperature of the most important s-process scenarios.

The neutron fields produced by the above reactions were defined by means of the time-of-flight (TOF) technique, as described in Ref. [1] for example. The spectrum obtained with the ^{7}Li(p,n)^{7}Be reaction (Fig. 1) was recently confirmed with improved resolution by Feinberg, *et al*. [2] and Lederer, *et al.* [3]. As shown in [1] the spectrum is practically insensitive to variations of the proton energy within the 0.1% energy spread of the accelerator as well as to modifications of the ^{7}Li targets in diameter and thickness.

After an accurate activation measurement of the ^{197}Au(n,γ) cross section in the spectrum of Fig. 1, which was based on the comparison of the induced gold activity (E_γ = 412 keV) and that of the produced ^{7}Be (E_γ = 478 keV) [1], this technique was used for an extended series of activation measurements, mainly in the context of the astrophysical s-process. The measurements were carried out by irradiating a sample sandwich consisting of Au foils on both sides of the sample for monitoring the neutron flux. In principle, this would require a correction for the fact that neutrons emitted at larger angles see a “thicker” sample. The correction is very small and practically negligible for cases where the energy dependence of the investigated cross section and that of gold are similar. This was recently confirmed by comparison of results from TOF and activation measurements in Table II of Ref. [4].

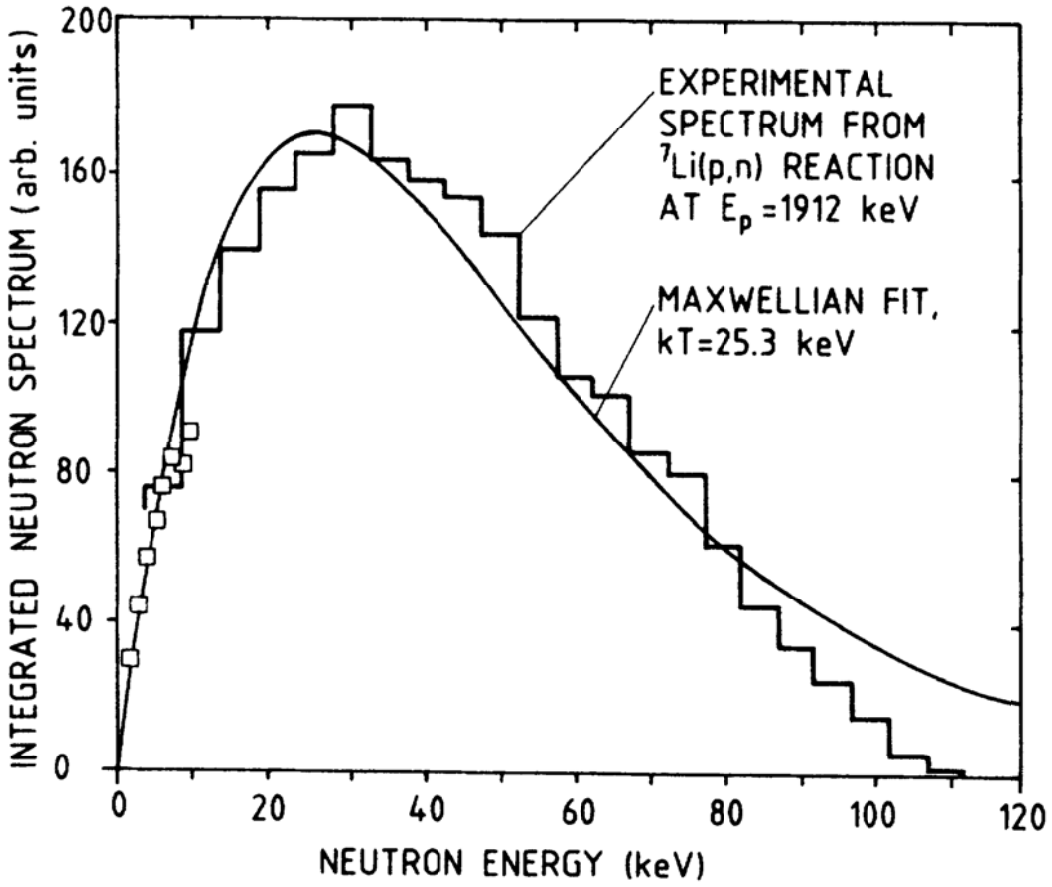

Fig. 1: The quasi-stellar neutron spectrum produced by the ^{7}Li(p,n)^{7}Be reaction at a proton energy E_p = 1912 keV after integration over all emission angles (histogram and symbols) compared to a true thermal spectrum for kT = 25 keV (from Ref. [1]).

As can be seen from Fig. 1, the quasi-stellar spectrum provides a good but not perfect representation of the true thermal spectrum. Therefore, the experimental results have to be corrected for the remaining differences by means of theoretical or evaluated energy-dependent cross sections. In practice, these data are normalized to the spectrum-averaged results obtained in the activation measurements, and are then used to calculate the Maxwellian averaged cross sections (MACS) required for applications in stellar s-process models. The measured cross sections are well documented in refereed journals and the deduced MACS results are collected in the *Karlsruhe Astrophysical Database of Nucleosynthesis in Stars* (see www.kadonis.org).

Accordingly, it is important to note that the relevant experimental data for the EXFOR entries are the original, experimental spectrum-averaged cross sections, whereas the deduced MACS values are containing additional information from secondary sources.

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Some white neutron (source) spectra encountered during 40+ years of fast neutron work

M. Drosg
Faculty of Physics, University of Vienna, A-1090 Wien, Austria

1. Introduction

Most of my neutron related experiments were time-of-flight measurements done at LANL, either at the Van-de-Graaffs, or at the spallation source of WNR. In particular, when investigating "monoenergetic" neutron sources I encountered quite a few neutron spectra with a wide neutron energy distribution stemming predominantly from deuteron or triton break-up. Based on this experience I now present my ideas on continuous neutron source spectra.

2. Definition of the Neutron Source Spectrum Relevant for EXFOR Compilation

One can discern (at least) three kinds of relevant definitions of neutron source spectra:

- spectrum delivered by the source,
- spectrum in the volume of the interaction sample, and
- effective spectrum in the volume of the interaction sample.

The last two types are derivatives of the first because the neutrons from the source reaction are their principal cause. However, from a practical point of view only the effective neutron spectrum in the volume of the sample is of interest.

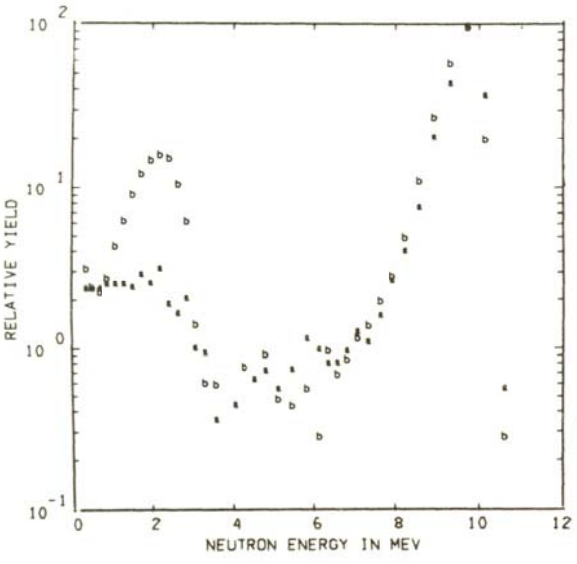

Fig. 1. Source intrinsic neutron spectrum for the production of 10 MeV neutrons by means of the $^{2}H(d,n)^{3}He$ (b) and the $^{3}H(p,n)^{3}He$ reactions (a) [1].

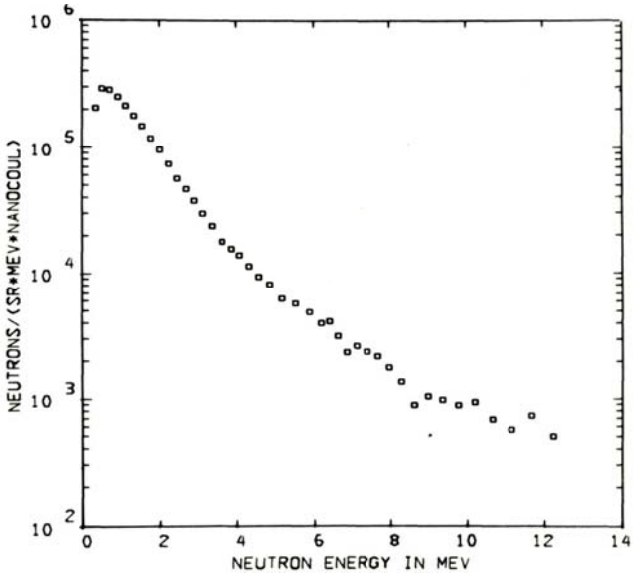

Fig. 2. Structural neutron background from the tritium gas cell when producing 12 MeV neutrons via the p-T reaction [1]

3. Effective Neutron Source Spectrum

Based on the experience with accelerator driven neutron sources 6 components of the effective neutron spectrum can be isolated:

a) Source Intrinsic Neutron Spectrum,
b) Structural Neutron Background,
c) Physical Spectrum Modification,
d) Experimental Spectrum Modification,
e) Modification of the Spectrum by the Measuring Process, and
f) Admixed Background (p, gammas).

It is practically impossible, to control all 6 components in a way that the effective neutron source spectrum is 100% transferable from one installation to another. Only in exceptional cases (self-supporting pure target materials) such a transfer will be possible to a sufficient degree.

Fig. 1 gives an example of two intrinsic neutron source spectra [1].

The structural neutron background stems from interactions of the beam with the target structure and from neutrons scattered from this structure. Fig. 2 shows the contribution of the tritium gas cell to the neutron source spectrum when 12 MeV neutrons are produced by the p-T reaction. By optimizing the target structure the structural background can be reduced as shown in Fig. 3. Such a target would allow measuring 12 MeV cross sections of reactions with thresholds above about 5 MeV as long as physical spectrum modifications can be disregarded. These are mostly the result of inscattering, from objects in the room and the room containment (ceiling, floor, walls), and to a small amount from ambient air.

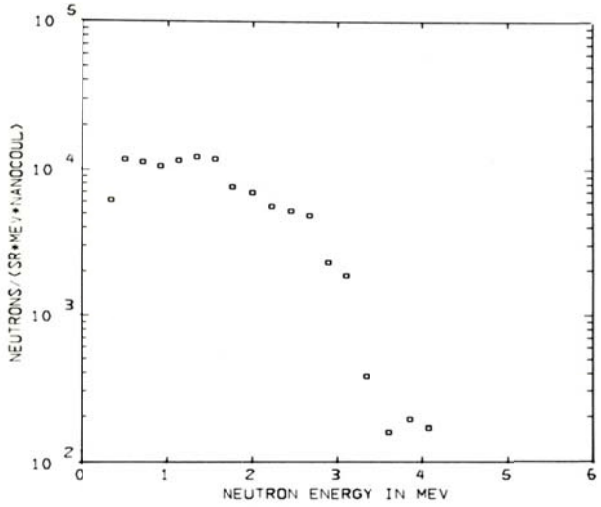

Fig. 3. Structural neutron background from a tritium gas cell with a ^{58}Ni entrance foil and a ^{58}Ni beam stop for producing 12 MeV neutrons via the p-T reaction [1].

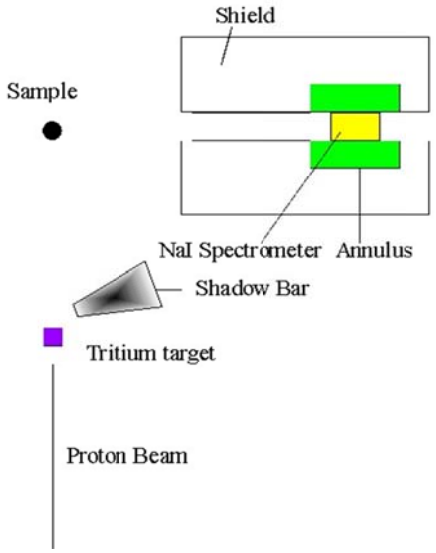

Fig. 4. Experimental spectrum modification. Time-of-flight method to modify the effective spectrum [2].

At the expense of intensity the 12 MeV neutron source mentioned above cannot only be used for the measurement of threshold reactions; Fig. 4 shows the principle of rejecting events from lower energy neutrons by applying the time-of-flight method in an (n-γ) experiment [2]. A distance of 96 cm provides enough time to reject the unwanted events.

In some cases the use of threshold detectors or a pulse-height bias in the neutron detector might help in "cleaning up" the low energy portion of the source spectrum, The use of thermal neutron detectors (e.g., ^{3}He counters) would make the higher energy portion of the effective source spectrum of little importance.

Radiation of other nature present in the source usually can be suppressed (n-γ-discrimination, absorbers against charged particles). However, if the intensity of such parasitic radiation is high as is the case with γ–rays when stopping higher energy protons in a ^{58}Ni beam stop (Fig. 3) the increase in detector and electronic dead time can cause a severe problem.

4. Reproducibility of the effective neutron source spectrum

An exact reproduction of a neutron source spectrum based on neutron source reactions is not possible, neither at different installations nor at different times. The first is mainly due to physical spectrum modifications, the second due to changes in the target composition. Using a pure self-supporting, fully stopping target will avoid background from the target structure and is a prerequisite for good reproducibility in time. However, in the course of time beam particles will be deposited in the beam stop disturbing the purity of the target material. As this deposition is at the very end of the particle range only the lowest energy part of the neutron spectrum is affected, as long as the beam energy is not raised. The d-Be source would deliver such a moderately well reproducible neutron spectrum. Other rather, exotic examples are neutron source spectra of fully stopped triton beams, e.g., in ice [3, 4]. Fig. 5 shows such a spectrum when 20.0 MeV tritons were fully stopped in heavy ice. At the upper edge one can see the effect of the limited energy resolution, the lowest energy neutrons were discriminated against by the detector pulse-height bias (modification of the effective spectrum by the measuring process). The measured spectrum can be unfolded into three components

- neutrons from the deuteron break-up,
- neutrons from the interaction with oxygen, and
- high energy neutrons from the two-body reaction ^{2}H(t,n)^{4}He.

The last component as shown in the figure is the thick-target yield prediction of the two-body neutron source reaction code WHIYIE [5].

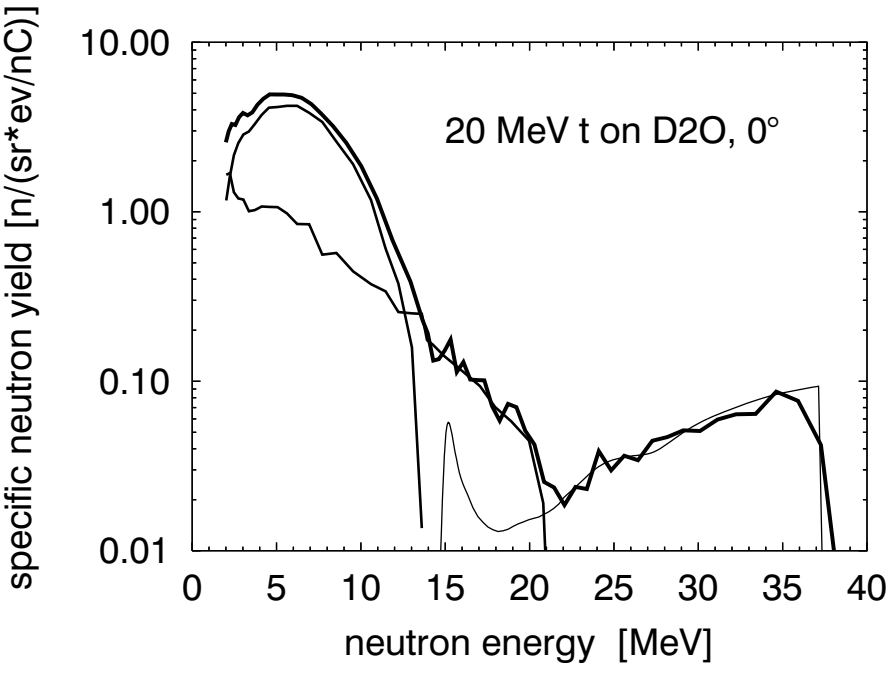

Fig. 5. Measured neutron (source) spectrum at 0° after fully stopping 20 MeV triton in heavy ice [4] (bold curve). The superimposed high-energy component is a prediction of WHIYIE [5].

Triton beam based neutron sources are presently of very little use. This is a pity because of the extraordinary properties of the kinematically collimated, very intense $^{1}H(t,n)^{3}He$ source [6] that, when fully stopped, would provide a flat white spectrum [3]. Neutron production cross section or double differential yield data (mainly at 0°) from triton interactions with targets of H_2, H_2O, D_2, D_2O, T_2, He, Li, LiF, Be, O, F, Si, Ni, Mo, Ta, W, Pt, and Au have been measured for triton energies up to 20.22 MeV [7].

Acknowledgement.

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Experimental techniques and computational methods used at the Frascati neutron generator in order to determine the neutron source spectrum

Mario Pillon
ENEA UT- Fusion CR Frascati, Italy,

FNG is a deuteron electrostatic accelerator of beam energy 260 KeV producing 10^{11} neutrons/s with energy 14 MeV from T(d,n)α or 10^{9} neutrons/s with energy 2.5 MeV from D(d,n)^{3}He reactions.

The absolute neutron intensity at FNG is measured by the associated particle method; accuracy is better than 5%.

The neutron source spectrum is calculated using an accurate source routine which is compiled inside MCNP5 transport code. This routine takes into account all the physical aspects of the neutron production from beam-target interaction, including ions scattering, fusion cross-section anisotropy, etc. The FNG source routine is now included in the SINBAD database, where several benchmark experiments performed at FNG are available. The source routine can be compiled inside MCNP5. In the MCNP geometry the target holder and the used detectors dimensions and materials are also described. The latest version of FNG source routine, not yet distributed, works for deuteron beam energy up to 10 MeV and for thick or thin, solid or gas target options.

The source routine has been recently validated for FNG with accurate neutron spectroscopy measurements using diamond detectors. Diamond detectors have been recently demonstrate to be a very high energy resolution fast neutron spectrometer for the energy range 8-20 MeV (FWHM 0.4% constant in the energy range). Validation of the FNG source routine is also in progress using the IRMM Van-der-Graaff of Geel (Belgium) for deuteron beam energy up to 5 MeV. Other mature source routine codes have been developed by other laboratories, i.e. TARGET code from PTB or NeuSDesc form JRC– Geel and they permit to obtain accurate calculated source spectra.

Spectra in the whole energy range are measured at FNG using activation foil unfolding techniques. The first guess input spectrum is calculated using MCNP5+source routine.

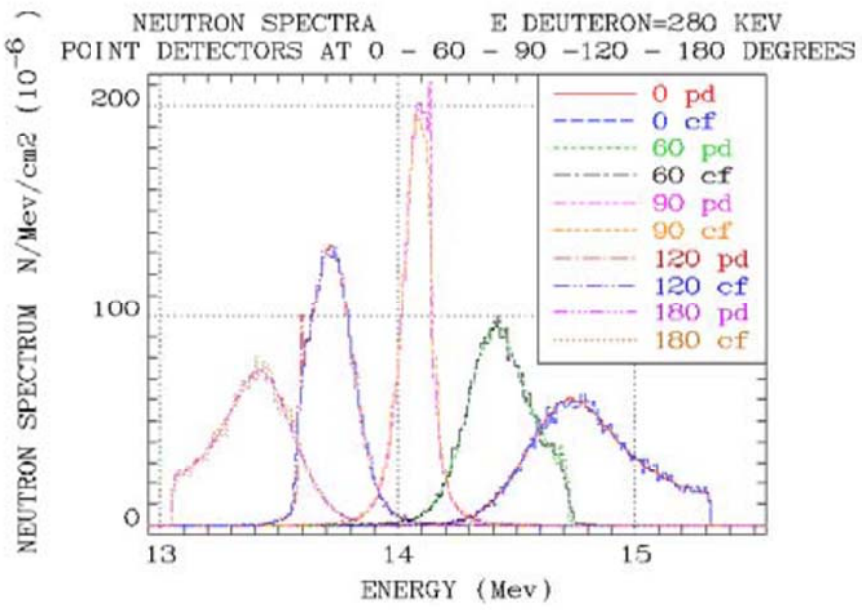

Fig. 1. Source routine was developed by ENEA Frascati and Institute Jozef Stefan in Ljubljana and is available in SINBAD database at NEA. Latest version up to 10 MeV beam energy (not yet in SINBAD): pd - point detector, cf - cell flux estimator.

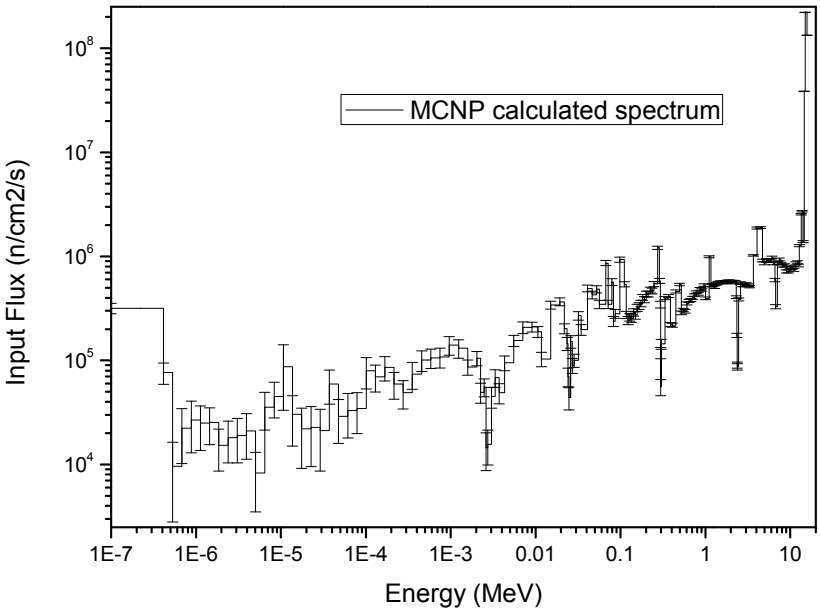

Fig. 2 Calculated whole neutron spectrum at the rhenium sample irradiation position. FNG experiment concerning irradiation of pure rhenium samples.

The unfolding codes used are SAND-II and STAY-NL (included in the neutron metrology file NMF-90 from IAEA) but using with IRDF-2002 dosimetry file instead of the IRDF-90 file which is included in the package.

The experience gain at FNG has shown that accurate cross-section measurements can be obtained only if the reaction under measure responds mainly in the energy range around the fusion peak (threshold reactions) since it is difficult to obtain an accurate measure of the spectrum tail. In the tail energy range significant differences of the reaction cross sections under measure (~ ±20 %) are obtain with the two unfolding codes. These differences appear only for no threshold reactions.

Table: Rhenium cross section measurements: comparison of the results using the unfolded spectra obtained by SAND-II and STAYNL codes

Reaction	Path	Effective cross section SAND	Effective cross sectionSTAY	RATIO
Re187(n,g)Re188m	100.0%	6.71E-03	7.83E-03	0.857
Re187(n,a)Ta184	100.0%	6.14E-04	6.15E-04	0.998
Re187(n,g)Re188	97.0%	2.46E-01	2.04E-01	1.206
Re187(n,g)Re188m(IT)Re188	3.0%			
Re187(n,p)W187	100.0%	3.81E-03	3.81E-03	1.001
Re185(n,g)Re186	14.5%	5.71E-01	4.82E-01	1.183
Re187(n,2n)Re186	85.4%	1.70E+00	1.69E+00	1.009
Re185(n,2n)Re184	100.0%	1.30E+00	1.29E+00	1.009
Re185(n,3n)Re183	100.0%	2.43E-02	2.41E-02	1.009
Re185(n,2n)Re184m	100.0%	3.13E-01	3.10E-01	1.009

One recommendation for the compilation of source spectra for EXFOR is to include, when available, the uncertainties on the measured spectra. The energy grid of the spectra included in EXFOR should be the original one of the experiments; otherwise some extra uncertainties should be introduced.

Neutron spectrum averaged activation cross section measurements

Syed M. Qaim
Institut für Neurowissenschaften und Medizin, INM-5: Nuklearchemie
Forschungszentrum Jülich GmbH, D-52425 Jülich, Germany

A brief description of neutron spectrum averaged activation cross sections, measured at Jülich over several decades, is given. The neutron source spectra and some other salient results are discussed.

1. Introduction

For several decades activation cross section measurements have been carried at Jülich using fast neutrons, both monoenergetic and spectral shaped. Here the discussion is limited to the use of spectral sources. In this connection, three types of facilities have been utilized: (1) d/Be neutrons, (2) fission neutrons, and (3) ^{241}Am/Be source neutrons. The d/Be neutron sources were available at two cyclotrons (CV 28 and JULIC) and were extensively used. The energy of the deuterons used at CV 28 was 14 MeV. At JULIC, however, deuteron energies could be varied between 17.5 and 30 MeV, so that various neutron spectra were available for the activation work. A yet another available facility was a 53 MeV d-Be breakup neutron source. As far as other spectral neutrons were concerned, use was made of fission neutrons in a nuclear reactor or those produced via the ^{241}Am/Be radionuclidic source. In this report the emphasis is on work done using the d/Be neutron sources. The publications from Jülich based on various spectral sources are listed at the end of this article [cf. 1–22]. Most of the studies were performed radiochemically, i.e. the product was separated chemically prior to the determination of the radioactivity. A brief outline of the various areas of investigations is given below.

2. Light Complex Particle Emission Reactions

Integral cross section measurement

Neutron induced light complex particle emission reactions, e.g. (n,t), (n,^{3}He) and (n,^{7}Be), have very high thresholds and are therefore extremely difficult to study with monoenergetic neutrons available up to 20 MeV. Extensive studies were therefore carried out using a 53 MeV d/Be neutron source. The (n,t) reaction was investigated on about 40 elements via separation and gas phase counting of accumulated tritium [1,2]. The neutron spectrum adopted was that described by Schweimer [23]. The (n,t) cross section was high for the light mass target nuclei but for elements with A > 40, it was practically constant at a value of about 3 mb. For comparison, the residual reaction product was also investigated. The cross section then entailed the sum of [(n,t) + (n,dn) + (n,p2n)] processes. It was found that the emission of three single particles is much stronger than the emission of a bound triton.

In the case of the (n,^{3}He) reaction, measurements were done on about 25 elements. On one hand, assay of the radioactive product was done and, on the other, quadrupole mass spectrometry on the accumulated ^{3}He and ^{4}He gases was applied [3,4]. The neutron spectrum adopted was slightly different from that in the case of the (n,t) reaction. The data by Meulders, *et al.* [24] covering the energy range 4 – 50 MeV were slightly corrected for the higher deuteron energy and then used in cross section determination. The cross section for the (n,^{3}He) reaction was found to be low in all cases and showed a decreasing trend with the increasing mass number of the target nucleus.

As regards the (n,^{7}Be) reaction, investigations were done on about 10 elements. The radiochemically separated ^{7}Be was assayed by low-level γ-ray spectrometry [8]. The neutron spectrum adopted was the same as for the (n,^{3}He) reaction. Compared to the other complex particle emission reactions described above, the (n,^{7}Be) reaction cross section was even lower but the trend over the whole mass range was somewhat similar.

Nuclear model calculations, based on the Hauser-Feshbach formalism, showed that the contribution of the statistical processes to the emission of complex particles ^{3}H, ^{3}He and ^{7}Be was small.

Determination of excitation function

Attempt was also made to deduce the excitation function of an (n,t) reaction up to 30 MeV via activation in diverse neutron fields and unfolding the spectra using certain codes. Metal discs together with sets of about 12 flux monitor foils having different reaction thresholds were irradiated in six different d/Be neutron fields (E_d = 17.5 to 30.0 MeV) [6,7]. The shapes of the neutron spectra were characterized by using the accumulated radioactivity in the foils combined with an unfolding code: SAND-II. The neutron energy range covered was from 2.0 MeV up to the maximum deuteron energy. In a second calculational step, from the measured tritium activities and neutron flux distributions, the excitation function for the (n,t) process was deduced. This methodology was used for two elements, namely Be [7] and Al [6]. The excitation function obtained in each case was in agreement with the cross section values obtained by using monoenergetic neutrons in the energy range of 13 to 20 MeV.

3. Application Oriented Work

Data related to radiation damage studies

In 1970s and 1980s considerable interest had arisen to measure cross section data related to materials damage in a 30 MeV d/Be neutron field. It was considered to be a possible alternative to an intense 14 MeV neutron source. Activation cross sections of (n,x) reactions on a large number of fusion reactor first wall and structural materials were therefore measured. Particular interest was on (n,p), (n,n'p), (n,α) and (n,n'α) reactions, i.e. hydrogen and helium producing reactions [9,10]. In addition, the (n,t) reaction was also investigated in detail [5] using this neutron spectrum.

Medical radionuclide production

Detailed cross section measurements were carried out using fission neutrons [11,12,14-17] available at FZ Jülich and PINSTECH (Pakistan).The major aim was to investigate the formation of some medically interesting radionuclides at reactors which are generally produced at accelerators. Some of the examples are ^{58}Ni(n,2n)^{57}Ni→^{57}Co, ^{123}Xe(n,2n)^{123}Xe→^{123}I, etc. [12,16,17].

A 14 MeV d/Be breakup neutron source was also used for measurement of data for medical radionuclide production. Since the spectrum is much harder than the fission spectrum, it was considered worthwhile to investigate the formation of ^{64}Cu, ^{67}Cu and ^{89}Sr via (n,p) reactions using those neutrons. In reality the cross section was found to be about three times higher than with fission neutrons [20]. Neutrons with a hard component (e.g. spallation neutrons) would thus be very suitable for production of some β^- emitting therapeutic radionuclides.

Data validation

Spectral neutrons, if well characterized in shape, can be used for data validation, i.e. testing the accuracy of excitation functions of neutron threshold reactions. Fission neutrons have often been used for this purpose but, as pointed out above, neutrons with a harder component than fission neutrons could be more advantageous. The technique of multiple-foil activation and spectrum unfolding via iterative methods has attained some more sophistication in recent years [cf. 21,22,25,26] and energy regions above about 1.5 MeV can be unfolded with sufficient accuracy. Thus a 14 MeV d/Be source was used to validate excitation functions of several reactions [19,21]. Very recently it has also been demonstrated [22] that an ^{241}Am/Be neutron source, such as the one recently installed at the Rajshahi University in Bangladesh, could be effectively used for validation of excitation functions of (n,p) and (n,α) reactions.

Acknowledgement

The summary report presented here is based on work done at Jülich and a few collaborating laboratories over many years, involving a large number of colleagues. It is a pleasure to thank them all for their respective contributions.

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Fast neutron sources with white and quasi-monoenergetic spectra of the NPI variable-energy cyclotron U-120M.

P. Bém, M. Honusek, M. Gotz, J. Novák, M. Majerle, E. Šimečková
Nuclear Physics Institute AS CR, p.r.i., 252 68 Řež, Czech Republic

in collaboration with S.P. Simakov, U. von Möllendorff, U. Fischer (KIT Karlsruhe)
and R.A. Forrest (Culham Science Centre)

The design and operation of the cyclotron-based fast neutron sources at Nuclear Physics Institute Řež were aimed to the integral- and differential benchmark tests of neutron cross-section data in the energy range relevant to IFMIF (International Fusion Material Irradiation Facility). The IFMIF neutron source reaction d(40 MeV)+Li produces a white spectrum with a high energy tail up to 35 MeV (50 MeV at lower intensity). To enable neutronic-design calculations for IFMIF, computational tool- and data tests are required for this energy range with neutrons having a realistic energy distribution (IFMIF-like). As deuterons up to 20 MeV energy only are available on the NPI cyclotron and two-fold energy could be reached for $H^{(-)}$ and $^{3,4}He^{(++)}$ beams, another reactions were used to simulate the d+Li neutron source spectrum of IFMIF.

IFMIF- like neutron spectrum simulated by the $^{3}He(40MeV)+D_2O$ source reaction [1-3]

High-yield emission of fast neutrons with a broad Gaussian-like energy spectrum from deuteron breakup reaction on ^{3}He target has been investigated before now. The same form of spectra and strongly forward directed neutron emission due to higher mass of the projectile could be presumed for the inverse reaction $D(^{3}He,np)^{3}He$. Thus, the $D_2O(^{3}He,xn)$ source reaction has been investigated experimentally for the first time. Measurements were carried out on external ^{3}He (40 MeV) beam of the cyclotron, using heavy water target of full-stop thickness. Open geometry arrangement of a scintillation detector and the conventional shadow-bare method were utilized for neutron spectrometry based on the pulse height technique with n-γ discrimination hardware. Neutron spectra were obtained using standard GRAVEL procedure (PTB Braunschweig), the detector response and efficiency were calculated by Monte Carlo code SCINFUL-R.

The angular distribution of spectral neutron yield was measured at $0\text{-}90^{O}$ angular range with an overall uncertainty from 8 to 25 %. In Fig. 1, resulting energy spectrum and angular distribution of energy integrated data are compared with prediction calculated for the IFMIF source reaction d(40 MeV)+Li and with available experimental data of d(40 MeV)+Li reaction, respectively. The observables of both reactions are found to be similar. Therefore, the $D_2O(^{3}He,xn)$ reaction was selected as a suitable source for a simulation of the IFMIF d+Li spectrum. Using the $^{3}He+D_2O$ neutron source, the set of neutron transmission experiments on iron, the main constituent of the IFMIF high-flux test module, has been performed in different configurations of sample arrangement.

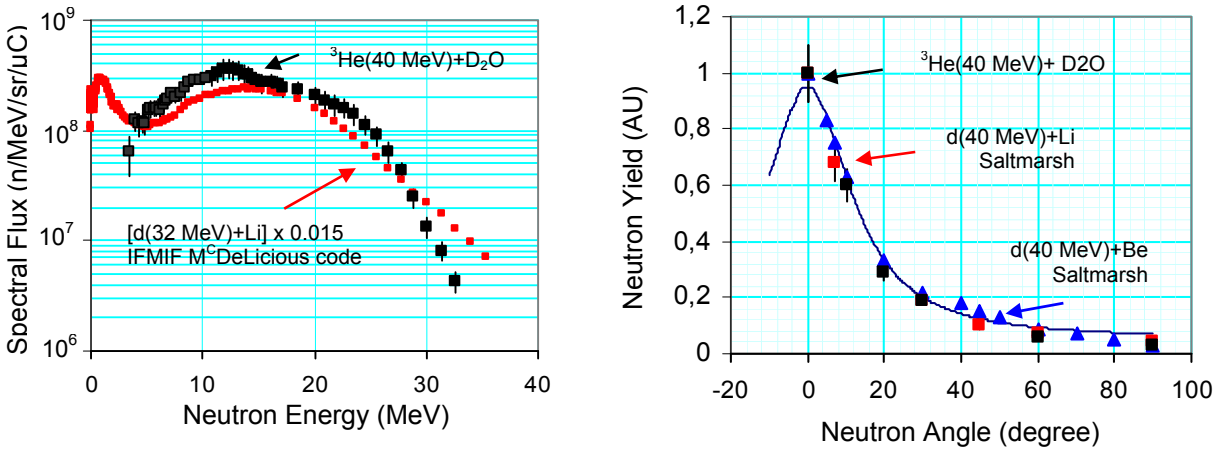

Fig. 1. Left: zero-degree spectral neutron yield from $D_2O(^3He,xn)$ reaction compared to the Li(d,xn) spectrum calculated for IFMIF. Right: relative angular dependence of neutron yield from $D_2O(^3He,xn)$ reaction versus experimental data for d+Li and d+Be reactions.

High-power white-spectrum neutrons from p(37 MeV)+D_2O source reaction [4-7]

The powerful fast neutron sources are usually built using proton or deuteron induced reactions on thick beryllium target. To avoid operation with toxic Be material, the D_2O(p,xn) reaction was investigated for the first-time using the experimental arrangement and data acquisition system described above. In Fig. 2 - left, resulting spectra measured at 24 MeV proton are compared with data for the p + Be (thick target) reaction. It is evident, that the p+D_2O reaction presents suitable alternative to p+Be source at proton energies under investigation.

Taking advantage in both the higher beam power (800 W) and larger energy range (up to 37 MeV) of protons from the cyclotron in the negative-ion mode of acceleration, the high-power neutron target station with special hardware for an operation of flowing D_2O target medium was build up.

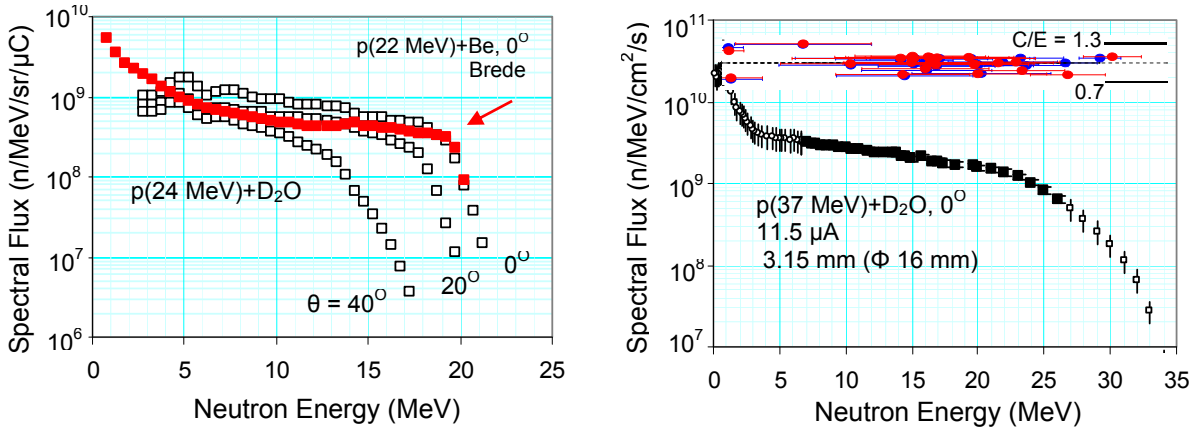

Fig. 2. Left: spectral neutron yield from p(24 MeV) + D_2O reaction measured by scintillation detector; zero-degree spectral yield from p(22 MeV) + Be reaction is given for a comparison. Right: p(37 MeV) + D_2O spectrum obtained by activation technique and SAND-II adjustment procedure; in upper part - resulting C/E ratios for different dosimetry reactions displayed at energies weighted with product of neutron flux and corresponding cross-section.

To determine the spectral flux from p(37 MeV) + D_2O neutron source at sample positions, the dosimetry activation foil technique was used instead of the MCNPX simulation so as a study has shown that the LA-150 proton library does not reproduce experimental double differential cross sections for the D(p,xn) reaction and hence the spectral yield from the thick heavy water target. Selected set of 11 foils was irradiated simultaneously at two distances of samples from the target. Up to 25 threshold activation reactions cover the full neutron energy range from 4 to 36 MeV. The SAND-II code was used for adjustment procedure after it had been modified to input dosimetry cross sections above 20 MeV from EAF-2007 library [9]. Standard adjustment procedure changes the shape and absolute value of the initial guess spectra (a combination of MCNPX simulation and experimental zero-degree spectral yield) bringing the C/E ratios close to unity.

Resulting adjusted neutron spectrum of p(37 MeV) + D_2O neutron source is given in the Fig. 2 (right). Standard relative deviation between calculated and experimental activities summed over all 25 detectors reach its minimum of 6-8% (see upper part of the Fig. 2, right). Estimated uncertainty is less than 10 % in the energy range below 25 MeV and increases above this energy due to errors of dosimetry cross sections. Below 4 MeV, the spectrum corresponds mainly to the MCNPX simulation. To determine the spectral flux at other distances from the target, the fit to measured reaction rate for selected dosimetry reactions at different distances was done.

Above 7 MeV neutron energy, the integral flux amounts 5.4 10^{10} n/cm^2/s for 15 μA proton beam current and for the nearest distance from the target. Utilizing this high-power neutron source with spectrum similar to one in the IFMIF test cell, the integral activation data for IFMIF relevant materials (low-activation steel Eurofer-97 and its components) were collected and used for the validation of EAF library. These investigations serve also for identification the set of nuclides for which the validation of cross-section with quasi-monoenergetic neutrons needs to be done.

Quasi-monoenergetic neutrons from p(20-37 MeV)+^{7}Li(C) source reaction [8-17]

To supply the integral validation experiments by differential cross-section measurements, the p+^{7}Li neutron source of quasi-monoenergetic neutrons was built on the NPI cyclotron. The standard ^{7}Li(p,n) reaction on thin lithium target induced by 20-38 MeV proton beam from the cyclotron is used for the production of quasi-monoenergetic neutron field at IFMIF relevant neutron energies. A self-supporting ^{7}Li foil (2 mm) and thick carbon beam-stopper (10 mm) are utilized in the target setup similar to one at CYRIC laboratory (Tohoku University, Japan). Up to 400 W/cm^3 of beam power could be dissipated in the Li target under separate setting of the foil and carbon disc in present modification. The neutron flux of 3 10^8 n/cm^2/s is calculated for 37 MeV and 5 μA proton beam for the target-to-sample distance 50 mm.

Neutron spectral flux at the sample positions is determined using the transport code MCNPX to simulate the proton interactions with the target assembly. The Los-Alamos evaluations for proton induced cross sections on ^{7}Li and ^{12}C from LA-150h were used. For other nuclides the relevant cross sections were represented by the MCNPX analytical models. The neutron scattering in the target assembly was simulated by making use of high energy neutron cross section data from ENDF/B-VII. This approach and nuclear data were validated against the neutron differential yields measured by time-of-flight technique at the CYRIC setup. As can be seen in the Fig. 3 (left), the MCNPX/LA-150h reasonably reproduces the whole Li/C spectra. Besides a mono-energetic peak ^{7}Li(p,n_{01}), the target simultaneously produces “parasitic” low energy neutrons from ^{7}Li and C discs. The analysis of impact of set-up arrangement on the extraction of activation cross sections from the measured induced radioactivities has been performed. The MCNPX simulation well represents the space-energy integration effect on neutron spectra due to comparable dimensions of the set-up, samples - and the sample-to-target distances. In Fig. 3 (right), the ratio of spectra calculated for two different “nominal” distances of sample from Li foil indicate the effect of different s-to-s distance for C disc and Li foil, respectively, the integration effect of ^{7}Li(p,n_{01}) cross section over solid angle of samples (the non uniform irradiation) and even the impact of ^{7}Li thickness on different emission of QME-neutrons as well.

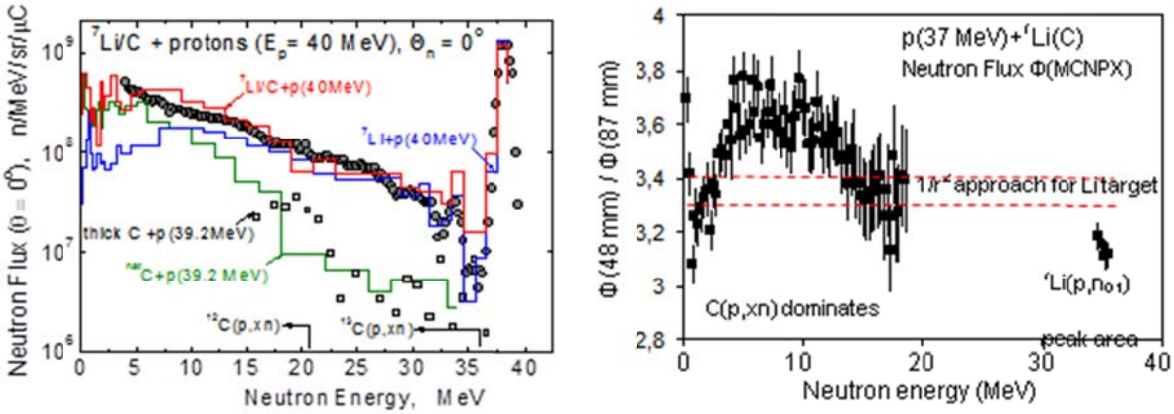

Fig. 3. Left: energy differential neutron yield measured for ^{7}Li/C and ^{nat}C targets (symbols) and calculated by MCNPX/LA150h (histograms); Right: neutron flux ratio Φ(49)/Φ(89) of spectra calculated at two distances ***r*** *(87 and 48 mm) from the ^{7}Li target. Red lines - the ratio calculated for neutrons from Li foil using* $\mathbf{1/r^2}$*-law.*

To improve the status of the dosimetry reaction data above 20 MeV for neutron spectral flux monitoring in IFMIF test cell and to complete data for gas production in the fusion-relevant materials, the activation cross sections on large set of nuclides is being carried out. Data are used for validation of the activation (EAF, IEAF, IRDF) and the general purposes (ENDF) cross sections files.

Acknowledgement

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EXFOR formats and rules: present status and proposals how to store neutron source data

Olena Gritzay
Institute for Nuclear Research of NAS of Ukraine, prospekt Nauky, 47, Kyiv, Ukraine, 03680
N. Otsuka, V. Semkova, S.P. Simakov, V. Zerkin
Nuclear Data Section, International Atomic Energy Agency
P.O. Box 100, A-1400 Vienna, Austria

In many cases the neutron cross sections, presented in EXFOR, are obtained with use of incident neutrons having broad energy distribution, thus such data are actually cross sections averaged over the neutron spectrum. In accordance with the existing EXFOR rules, this neutron spectrum should be entered in EXFOR as a free text. Analysis of data, available in EXFOR, has shown that information on neutron source spectra is practically absent. Lack of information makes it impossible to compare different experimental data correctly and to use the averaged cross sections for improvement of the evaluated neutron data libraries. Thus the situation has to be made better.

1. Existing EXFOR formats and rules.

EXFOR: Chapter 6. REACTION SPECIFICATION

The reaction and quantity for the data coded in the data table is specified using the information-identifier keyword REACTION. Therefore, this keyword must always be present in a data set.

A REACTION unit consists of three major fields (reaction, quantity, data-type):

Reaction field
- SF1. Target Nucleus
- SF2. Incident particle
- SF3. Process
- SF4. Reaction Product

Quantity field
- SF5. Branch
- SF6. Parameter
- SF7. Particle considered
- **SF8. Modifier**

Data-type field
- SF9. Data type

SF8 Modifier contains information on the representation of the data, for example, relative data, fitting coefficients. Code(s) are taken from Dictionary 34.

Dictionary 34: Modifiers (REACTION SF8)

EPI	epi-thermal neutron spectrum average
FIS	fission spectrum average
FST	fast reactor neutron spectrum average
MXW	Maxwellian average
SPA	**spectrum average**
AV	**average**

INC-SPECT provides information on the characteristics and resolution of the incident-projectile beam. It must be present when a spectrum average modifier (*e.g.*, MXW, SPA, or FIS) is present in REACTION SF8.

See also LEXFOR. Spectrum Average: cross sections averaged over a broad incident-projectile energy spectrum may be entered into EXFOR using the proper modifier to REACTION SF8. The type of spectrum and its characteristic should be entered in free text under the information-identifier keyword INC-SPECT.

To show the actual situation with information on neutron sources data in EXFOR, let consider several examples.

Example 1.

SUBENT	**22857001**
TITLE	Nuclear data for production of the therapeutic radio- nuclides P-32, …
AUTHOR	(M.AL-Abyad, …, S.M.Qaim, …)
FACILITY	(CYCLO,2GERJUL) Variable energy compact cyclotron CV28.
INC-SOURCE	**(D-BE)** Deuterons with primary energy 14 MeV impinging the thick beryllium target 19.7 mm diameter, 1.9 mm thick, 98% purity embedded into 2 mm thick copper holder cooled with water.
INC-SPECT	Continuous neutron energy spectrum evaluated using averaged cross section and the measured activity of the elements given below. Has two maxima - the first large at aprx 3 MeV and the second - small at 14 MeV neutron energy. Maximal flux density occurs at 2.5 - 3.5 MeV neutron energy.
SUBENT	**22857005**
REACTION	(16-S-32(N,P)15-P-32,,SIG,,**SPA**) Averaged over the neutron spectrum of (D-Be) source

Example 2.

SUBENT	**22995001**
TITLE	The 14-C(n,gamma) cross section between 10 keV and 1 MeV
AUTHOR	(R.Reifarth, … F.Kaeppeler, …)
FACILITY	(VDG,2GERKFK) 3.7 MV Van de Graaff accelerator
INC-SOURCE	**(P-LI7)** A metallic Li-7 target of different Li-7 layer thicknesses with protons of different energies above the (P,N) reaction threshold.
INC-SPECT	Spectrum at the place of the sample is determined by averaging of the down- and upstream gold monitor samples. Neutron fluxes between 1.E+7 - 1.E+9 1/cm**2/s .
SUBENT	**22995003**
REACTION	(6-C-14(N,G)6-C-15,,SIG,,**SPA**) Averaged over neutron spectrum width
INC-SOURCE	Li-7 target thickness is 5 micrometers, proton energies are 2001 - 2530 keV, above Li-7(p,n) reaction threshold
INC-SPECT	Asymmetric broad neutron spectrum with resolution taken by compiler from Fig.3 of main reference. Spectra are typical for mono-energetic experiment with wide energy uncertainty (for 150 and 500 keV).

Example 3.

SUBENT	**31611001**
TITLE	Thermal neutron capture cross sections of the palladium isotopes
AUTHOR	(M.Krticka, … T.Belgya, …)
FACILITY	(REAC,3HUNII)
INC-SOURCE	(REAC) The thermal-equivalent neutron flux was 2 x 10**6 n*cm2/s.
INC-SPECT	**ABSENT**
SUBENT	**31611015**
REACTION	(46-PD-106(N,G)46-PD-107,,SIG,,**SPA**)

Example 4.

SUBENT	**31633001**
TITLE	Neutron activation experiments on chromium and tantalum in the NPI p-7Li quasi-monoenergetic neutron field
AUTHOR	(P.Bem, …)
FACILITY	(CYCLO,3CZRUJF) The variable-energy proton beam of NPI cyclotron …
INC-SOURCE	**(P-LI7)** neutron flux density of about 10**9 n/cm2/s in peak at 30MeV neutron energy.
INC-SPECT	**ABSENT**
SUBENT	**31633002**
REACTION	(16-S-32(N,P)15-P-32,,SIG)

In subentry 31633002 the spectrum average modifier SF8 is absent.

Example 5.

SUBENT	**32217001**
TITLE	Measurements of neutron capture cross-section for tantalum at the neutron filtered beams
AUTHOR	(O.Gritzay, …)
FACILITY	(REAC) Reactor WWR-M
INC-SOURCE	(REAC) Neutron filters installed in horizontal channel of the reactor.
INC-SPECT	2) Filter 59 keV. Filter components: 10B, V, S and 58Ni. The main part of the filtered neutron spectrum is a line in the energy range 54.8-60.1 keV. The energy distribution centre of this line is at 58.96 keV; its relative intensity is 94.36% of the entire neutron flux. Besides, in the spectrum there are impurity lines: 3 groups of lines in the energy range 1-40 keV (0.01%), 70-230 keV (0.11%) and 235-1570 keV (5.52%).
SUBENT	**32217003**
REACTION	(73-TA-181(N,G)73-TA-182,,SIG,,**AV**)

In subentry 32217003 it was used AV instead of SPA.

It is noted that numerical information about neutron spectrum is absent in EXFOR, though the existing EXFOR rules recommend to include the type of spectrum and its characteristics in a free text under the information-identifier keyword INC-SPECT. As it was noted above the absence of this information leads to difficulties in comparison of the averaged data measured by different authors and use them in evaluation procedures. The numerical data on neutron spectrum thus have to be compiled in EXFOR.

2. Two possible options how to modify the existing EXFOR rules.

1) To change rule for INC-SPECT.
   Now: the type of spectrum and its characteristic should be entered **in free text** under the information-identifier keyword INC-SPECT.
   Alter: the type of spectrum and its characteristic should be entered **in numeric data type** under the information-identifier keyword INC-SPECT.

2) To introduce into EXFOR **new special ENTRY/SUBENTRY** for neutron spectrum.
   New: type of spectrum and its characteristic should be entered **in numeric data type** using **separate SUBENTRY** or **ENTRY** for neutron spectrum if the spectrum is commonly applied to measurements performed at the neutron source.
   2.1 Use special form of **REACTION** to define the neutron source (see table below) with the proper modifier **SPD** to REACTION **SF8**.
   2.2 Use **DATA** to enter **the numerical spectral data.**
   2.3 **Data**, **that are averaged** over broad incident-projectile energy spectrum and entered into the EXFOR system, should be labelled by the keyword **INC-SOURCE** with use of all relevant keywords from the Inc-Source Dictionary (#19) and **the cross-reference** to the EXFOR entry/subentry with **these numerical spectral data.**
   2.4 This cross-reference must be coded as an eight-digit integer.

The special form of REACTION to define the neutron spectrum

Name of neutron source	In INC-SOURCE (Dictionary #19)	SF1-SF8 in REACTION in Entry/SubEntry with spectrum
Alpha-Beryllium	A-BE	4-BE-9(A, X)0-NN-1,,DE,,SPD
Spont. fission of Californium-252	CF252	98-CF-252(0,F) ,, NU/DE,,SPD
Spont. fission of Curium-244	CM244	96-CM-244(0,F),, NU/DE,,SPD
Spont. fission of Curium-246	CM246	96-CM-246(0,F),, NU/DE,,SPD
Spont. fission of Curium-248	CM248	96-CM-248(0,F),,NU/DE,,SPD
Deuteron-Beryllium	D-BE	4-BE-9(D,X)0-NN-1,,DE,,SPD
Deuteron-Carbon 12	D-C12	6-C-12(D,X)0-NN-1,,DE,,SPD
Deuteron-Carbon 14	D-C14	6-C-14(D,X)0-NN-1,,DE,,SPD
Deuteron-Deuterium	D-D	1-H-2(D,X)0-NN-1,,DE,,SPD
Deuteron-Lithium	D-LI	3-LI-0(D,X)0-NN-1,,DE,,SPD
Deuteron-Lithium 7	D-LI7	3-LI-7(D,X)0-NN-1,,DE,,SPD
Deuteron-Nitrogen 14	D-N14	7-N-14(D,X)0-NN-1,,DE,,SPD
Deuteron-Nitrogen 15	D-N15	7-N-15(D,X)0-NN-1,,DE,,SPD
Deuteron-Tritium	D-T	1-H-3(D,X)0-NN-1,,DE,,SPD
Evaporation neutrons	EVAP	13-Al-27(P,X)0-NN-1,,DE,,SPD 74-W-0(P,X)0-NN-1,,DE,,SPD 82-Pb-0(P,X) 0-NN-1,,DE,,SPD 92-U-0(D,X) 0-NN-1,,DE,,SPD...
Nuclear explosive device	EXPLO	???
Proton-Beryllium	P-BE	4-BE-9(P, X)0-NN-1,,DE,,SPD
Proton-Deuterium	P-D	1-H-2(P,X)0-NN-1,,DE,N,SPD
Photo-neutron	PHOTO	1-H-2(G,X)0-NN-1,,DE,,SPD 13-Al-27(G,X)0-NN-1,,DE,,SPD 74-W-0(G,X)0-NN-1,,DE,,SPD 92-U-0(G,X)0-NN-1,,DE,,SPD ...
Proton-Lithium 7	P-LI7	3-LI-7(P,X)0-NN-1,,DE,,SPD
Polarized neutron source	POLNS	???
Proton-Tritium	P-T	1-H-3(P,X)0-NN-1,,DE,,SPD
Spont. fission of Plutonium-240	PU240	94-PU-240(0,F) ,, NU/DE,,SPD
Spont. fission of Plutonium-242	PU242	94-PU-242(0,F) ,, NU/DE,,SPD
Reactor	REAC	92-U-FUL(X,X) 0-NN-1,,DE,,SPD
Thermal column	THCOL	???

* The SF1-SF8 designation of source, marked by "???", is not defined today.

The fields **SF8, SF9** in **REACTION** may be used to indicate, if this spectrum is given in relative (SPD/**REL**) values, and if it was obtained by calculation (**CALC**).
In **COMMENT** or in the line with **REACTION** we can write the additional information about given spectrum: codes and libraries used in calculations, components on the used neutron filter, etc.

3. Proposal for the source spectral information storage in EXFOR.

We propose to use the second way, i.e., to introduce a **new separate special ENTRY or SUBENTRY** for neutron spectrum. The advantages would be:

1) to refrain from repeating the neutron spectrum information in Entries with data obtained with that neutron source spectrum.
2) to facilitate data search of neutron source spectrum - they could be found using the modifier **SPD** in REACTION **SF8** and the EXFOR retrieval system; this service is important for experimenters, evaluators and compilers.
3) to use keyword INC-SOURCE for cross-reference to the relevant spectrum (see above the point 2.3) - it will allow us to avoid introduction of new Dictionary and essential modification of the checking codes.

To demonstrate an example of such Entries, let consider subentry 32217003, where the filtered neutron spectrum was used to measure the averaged radiation cross section on Ta. There were two types of spectrum: calculated and experimental ones, the latter was obtained by differentiation of the instrumental proton recoil spectrum.

We can use **one new entry** (take for example **32**777) for the filtered neutron spectra. In subentry **32**777001, as usual, we describe general information using keywords TITLE, AUTHOR, INSTITUTE, …

ENTRY	32777	20110408				32777 0	1
SUBENT	32777001	20110408				32777 1	1
BIB	7	10				32777 1	2
TITLE	Measurements of neutron capture cross-section for					32777 1	3
	tantalum at the neutron filtered beams					32777 1	4
AUTHOR	(O.Gritzay,V.Libman,A.V.Chyzh,V.F.Razbudey)					32777 1	5
INSTITUTE	(4UKRIJD)					32777 1	6
REFERENCE	(C,2008KYIV,,548,2008) Result on 59 keV was					32777 1	7
	presented at the NPAE-Kyiv2008,ID# 86-95.					32777 1	8
FACILITY	(REAC,4UKRIJD) Reactor WWR-M					32777 1	9
INC-SOURCE	(REAC) Neutron filters installed in horizontal channel					32777 1	10
	of the reactor.					32777 1	11
HISTORY	(20110408) UKRNDC					32777 1	12
ENDBIB	10	0				32777 1	13
NOCOMMON	0	0				32777 1	14
ENDSUBENT	13	0				32777	199999

For calculated neutron spectrum we can use subentry **32**777**002** and for experimental one the subentry **32**777**003**. Note this neutron spectrum was created by filtering of the reactor spectrum. Since the reactor used uranium fuel, we propose to fill the fields **SF1-SF4** in **REACTION** as **92-U-FUL(X,X)0-NN-1**.

SUBENT	32777002	20110408	32777	2	1
BIB	2	8	32777	2	2
REACTION	(92-U-FUL(X,X)0-NN-1,,DE,,SPD/REL,CALC) Using JENDL-3.3		32777	2	3
	and CENDL-2		32777	2	4
COMMENT	Calculation was done by FILTER.5 using JENDL-3.3 for		32777	2	5
	Ni-58(83.15 g/cm2),V(24.44 g/cm2),Al(5.4 g/cm2),		32777	2	6
	B-10(0.5 g/cm2), and using CENDL-2 for S(147.78 g/cm2)		32777	2	7
	Calculated energy line is 58.9 keV, purity about 99%.		32777	2	8
	The limits of 95% response function for the 59 keV		32777	2	9
	filter spectrum were defined as 52.2 to 60.1 keV.		32777	2	10
ENDBIB	8	0	32777	2	11
NOCOMMON	0	0	32777	2	12
DATA	2	1543	32777	2	13
E	DATA		32777	2	14
EV	ARB-UNITS		32777	2	15
50000.15	7.05730E-11		32777	2	16
50019.84	7.85371E-11		32777	2	17
50039.52	8.42285E-11		32777	2	18
...					
63945.89	6.82944E-26		32777	2	1557
64018.24	7.01394E-26		32777	2	1558
ENDDATA	1545	0	32777	2	1559
ENDSUBENT	1558	0	32777		299999
SUBENT	32777003	20110408	32777	3	1
BIB	2	6	32777	3	2
REACTION	(92-U-FUL(X,X)0-NN-1,,DE,,SPD/REL)		32777	3	3
COMMENT	Ni-58(83.15 g/cm2),V(24.44 g/cm2),Al(5.4 g/cm2),		32777	3	4
	B-10(0.5 g/cm2), and S(147.78 g/cm2) were used as		32777	3	5
	filter components.		32777	3	6
	Experimental shape was obtained by differentiation of		32777	3	7
	the instrumental proton recoil spectrum LND-281.		32777	3	8
ENDBIB	6	0	32777	3	9
NOCOMMON	0	0	32777	3	10
DATA	3	431	32777	3	11
E	DATA	DATA-ERR	32777	3	12
EV	ARB-UNITS	ARB-UNITS	32777	3	13
48793.33	0.487	0.008	32777	3	14
48831.11	0.698	0.013	32777	3	15

In COMMENT or in the line with REACTION we can write the additional information about given spectrum: libraries used in calculations, components of the neutron filter, etc. To refer to the used neutron spectrum from the subentry 32217003, which contains the measured average cross section data, we can use there the keyword
INC-SOURCE (**REAC, 32**777**002**) and (**REAC, 32**777**003**):

SUBENT	32217003	20110318	20110323	20110323	3148	32217	3	1
BIB	4	24				32217	3	2
REACTION	(73-TA-181(N,G)73-TA-182,,SIG,,SPA)					32217	3	3
INC-SOURCE	(REAC,32777002) Calculated neutron spectrum					32217	3	4
	(REAC,32777003) Experimental neutron spectrum					32217	3	5
ANALYSIS	For determination of sample activities, nine gamma					32217	3	6
	lines of W-182 were selected: 152, 179, 222, 229, 264,					32217	3	7
	1121, 1189, 1221 and 1231 keV.					32217	3	8
ERR-ANALYS	(ERR-T)Absolute uncertainty of cross section-					32217	3	9
	it includes the uncertainties of-					32217	3	10
	(ERR-1)Error in extrapolated cross-section - it					32217	3	11
	includes the uncertainties of-					32217	3	12
	error in determination of sample activities 1.3-4.4%					32217	3	13
	statistical error in peak area 1.2-21.1%					32217	3	14
	error in gamma-line efficiency 4.2%					32217	3	15
	error in quantum yield gamma-lines 0.05-0.48					32217	3	16
	(ERR-2)Error in determination of Ta sample mass 0.057%					32217	3	17
	(ERR-3)Error in determination of neutron flux - it					32217	3	18

Such form of recording of the neutron source spectrum allows us to use the existing EXFOR rules, except small changes.

By analogy, for each of the above-mentioned examples 1-4 we may create the new special separate subentry (or create new entry) to enter spectrum numerical information. Thus, for the example 1 the numerical spectral information may be entered in SUBENTRY 22857008 and the keyword INC-SOURCE (D-BE, **22857008**) may be added in each subentries with averaged data (subentries 2-5).

Example 1.

…	
SUBENT	22857005
REACTION	(16-S-32(N,P)15-P-32,,SIG,,SPA) Averaged over the neutron spectrum of (D-Be) source
INC-SOURCE	**(D-BE, 22857008**)
…	
SUBENT	**22857008**
REACTION	**(4-BE-9(D,X)0-NN-1,,DE,,SPD)**
…	
DATA	

For the examples 2, 3, and 4 it may be written.

Example 2.

…	
SUBENT	22995003
REACTION	(6-C-14(N,G)6-C-15,,SIG,,SPA) Averaged over neutron spectrum width
INC-SOURCE	**(P-LI7, 22995004)**
…	
SUBENT	**22995004**
REACTION	**(3-LI-7(D,X)0-NN-1,,DE,,SPD)** Li-7 target thickness is 5 micrometers, proton energies are 2001 - 2530 keV, above Li-7(p,n) reaction threshold
…	
DATA	

Example 3.

…	
SUBENT	31611015
REACTION	(46-PD-106(N,G)46-PD-107,,SIG,,SPA)
INC-SOURCE	**(REAC, 31611021)**
…	
SUBENT	**31611021**
REACTION	**(92-U-FUL(X,X) 0-NN-1,,DE,,SPD)** The thermal-equivalent neutron flux was 2 x 10**6 n*cm2/s.
…	
DATA	

Example 4.

…	
SUBENT	31633002
REACTION	(16-S-32(N,P)15-P-32,,SIG,,**SPA**)
INC-SOURCE	**(REAC, 31633004)**
…	
SUBENT	**31633004**
REACTION	**(3-LI-7(P,X)0-NN-1,,DE,,SPD)**
COMMENT	Neutron flux density of about 10**9 n/cm2/s in peak at 30MeV neutron energy.
…	
DATA	

The given examples obviously show that new separate special subentries (not entries) can also be used for storage of the numerical spectral information.

APPENDIX 4

DIFFERENTIAL NEUTRON PRODUCTION CROSS SECTIONS AND THICK TARGET YIELD DATA MISSING IN THE EXFOR DATABASE

Neutrons generated by light-ion induced reactions on light or medium element targets (e.g., ^{7}Li(p,xn), ^{7}Li(d,xn), ^{9}Be(d,xn) etc.) have been recognized as useful neutron fields for cross section measurements, detector calibrations, material irradiation and validation of the neutron transport through shield materials. On the other hand, theoretical modelling of the differential cross sections for these reactions faces difficulties to verify them due to the lack of experimental data [1,2]. As a result, complete and validated evaluated data files are missing for many neutron production reactions.

Therefore, the collection of experimental differential neutron production cross sections and thick target yields in the EXFOR database continues to be a task of high importance. It turns out that still many experimental works are missing. Many measurements were done and published in the period 1960-1980 in Journals and Conference proceedings (where the results were usually presented in the graphs) or as laboratory Reports. There are also recent publications which were not included in EXFOR. Compilation of all relevant data by priority is proposed for NRDC (see also http://www-nds.iaea.org/nrdc/nrdc_2011/working/wp2011-13.pdf).

Table. *List of papers and reports containing neutron emission data from p, d, t, ^{3}He and α induced reactions, which were not included in EXFOR before this Meeting. The new Entries (or additional Sub-Entries) already assigned for compilation are shown in the last column. They will be available for users as the EXFOR database will be updated.*

	References	EXFOR
1.	H.I. Amols et al., Med. Phys. 4 (1977) 486, Report FermiLab-Pub-76/102-EXP	C1835
2.	T. Aoki et al., J. of Nucl. Sci. Tech. 41 (2004) 399-405	E1893
3.	D.L. Johnson, NIM85-86(1979)467, Report BNL-NCS-51245,99,1980	
4.	A.N. Goland, IEE Transactions on Nuclear Science 22 (1976) 1776	
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6.	H.J. Brede et al., Nucl. Instr. Meth A274 (1989) 332	D0523
7.	V.K. Daruga et al., Atomnaya Energiya, 30 (1971) 399, English Translation: Sov. Atom. Energy 30 (1971) 493	A0898
8.	M.A. Lone et al., Nucl. Instr. Meth. 143 (1977) 331, Report NBIR-77-1279, 1977, p. 5	C1837
9.	R. Madey et al., Med. Phys. 4 (1977) 322	C1833
10.	J.L. Ullmann et al., Med. Phys. 8 (1981) 396	C1834
11.	N. Colonna et al., Med. Phys. 26 (1999) 793	C1836
12.	J.P. Meulders et al., Phys. Med. Biol. 20 (1975) 235-243	
13.	Z. Radivojevic et al., Nucl. Instr. Meth. B 183 (2001) 212-220	O1050
14.	M.J. Saltmarsh et al., Nucl. Instr. Meth. 145 (1977) 81-90, Report NBIR-77-1279, 24, 1977	C1832
15.	G.W. Schweimer, Nucl. Phys. A100 (1967) 537-544	
16.	L.S. August et al., Report NBSIR 77-1279 (1977) 31	
17.	C.E. Nelson et al., Report NBIR-77-1279 (1977) 1	
18.	K.A. Weaver et al, NSE 52 (1973) 35; Report UCRL-51310, 1972	F0218
19.	K. Shin, Phys. Rev. C29 (1984) 1307	E2343
20.	H. Schuhmacher et al., NIM A421 (1999) 284	
21.	C. J. Batty et al., Nucl. Instr. and Meth. 68 (1969) 273	O0650
22.	A. S. Clough et al., Nucl. Instr. and Meth. A143 (1970) 385	F0789
23.	P. Bém et al., Nucl. Instr. and Meth. A 425 (1999) 522	D0273

24.	P. Bém et al., Nucl. Instr. and Meth. A466 (2001) 509	D0278
25.	P. Bém et al., Report NPI ASCR Řež, EXP(EFDA)-05/2004	
26.	M. Drosg et al., Nucl. Instr. Meth. B73 (1993) 387 and 392	F0710
27.	E. Kim et al., NSE 129 (1998) 209	22653
28.	G. Lhersonneau, et al., Nucl. Inst. Meth. A 603 (2009) 228	O1746
29.	M. Hagiwara et al., J. Nucl. Mat. 329-333 (2004) 218-222	E1985
30.	M. Hagiwara et al., Fus. Sci. and Technol. 48 (2005) 1320	E1986
31.	M. Hagiwara et al., J. Nucl. Mat. (2011), in press	E2322
32.	N. Nakao, NIM A420 (1999) 218	E2298
33.	Y. Uwamino et al., Nucl. Instrum. Meth. A 389 (1997) 463	E1826
34.	Y. Uwamino et al., NIM A 271 (1988) 546; NSE 111 (1992) 391	E2296
35.	M. Baba et al., NIM A428 (1999) 454	E1808
36.	M. Baba et al., JNM 307-311 (2002) 1715	E1893
37.	M. Baba et al., Int. Conf. ND-2004, Santa Fe, 2004, p. 884	E1856, E1986
38.	Y. Iwamoto et al. Nucl. Instrum. Meth. B 389 (1997) 463	E1826
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2. M. Majerle, S.P. Simakov, "Modelling d-Be and d-C Neutron Sources for SPIRAL-2", Proc. of Nuclear Measurements, Evaluations and Applications (NEMEA-6), 25-28 October 2010, Krakow, Poland, p. 131 - available on http://www.oecd-nea.org/science/wpec/nemea6/; 10th Int. Topical Meeting on Nuclear Applications of Accelerator (AccApp'11), 3-7 April 2011, Knoxville, TN, USA.

APPENDIX 5

CODES FOR NEUTRON SPECTRUM CALCULATIONS

This appendix gives an overview of codes developed elsewhere for the simulation of accelerator driven and reactor filtered neutron sources.

The most essential input for the modelling of accelerator driven neutron sources is double differential neutron production cross sections for the reactions under consideration. As this overview shows, the systematics by M. Drosg or by H. Liskien and A. Paulsen are most often used. Another essential aspect concerns the simulation of the slowing down process of charged particles in target materials. So far, this is predominantly based on the evaluation of J.M. Ziegler (SRIM) of electron and nuclear stopping powers.

One could also notice a tendency to implement the above models (or pass the neutron sampling results) in the sophisticated Monte-Carlo codes, like MCNP. It facilitates the simulation of the impact of neutron scattering on the target set-up, which depends on each facility and could be rather complex.

Particular attention was paid to the availability of these codes for practical use.

DROSG2000

The package was developed and regularly upgraded by M. Drosg [1] and contains data and three computer codes to calculate accelerator-based neutron production:

- neutron energies, differential cross-sections and differential yields for monoenergetic neutron production through the following reactions: $^{3}H(p,n)^{3}He$, $^{7}Li(p,n_0)^{7}Be$, $^{7}Li(p,n_1)^{7}Be^*$, $^{9}Be(p,n_0)^{9}B$, $^{11}B(p,n_0)^{11}C$, $^{13}C(p,n_0)^{13}N$, $^{15}N(p,n_0)^{15}O$, $^{2}H(d,n)^{3}He$, $^{3}H(d,n)^{4}He$, $^{3}H(\alpha,n_0)^{6}Li$, $^{36}Cl(p,n)^{36}Ar$, $^{39}Ar(p,n_0)^{39}K$, $^{59}Co(p,n)^{59}Ni$, $^{7}Li(d,n)^{8}Be$, $^{9}Be(d,n)^{10}B$, $^{11}B(d,n)^{12}C$, $^{13}C(d,n)^{14}N$, $^{15}N(d,n)^{16}O$, $^{18}O(d,n)^{19}F$, $^{7}Li(\alpha,n)^{10}B$, $^{11}B(\alpha,n)^{14}N$, $^{13}C(\alpha,n)^{16}O$, $^{22}Ne(\alpha,n)^{25}Mg$;
- energy-angular differential thick target yields and white neutron spectra from the two body reactions covered above;
- differential cross sections and energies of (n,p), (n,d), (n,t) and (n,α) reactions by detailed balance from the time reversed reactions listed above.

The reaction differential cross sections and their uncertainties are presented in [2]. The electronic stopping power needed to account for the slowing down of charged particles is based on data of Ziegler et al. [3] unless indicated otherwise.

Package is freely available from IAEA web page: http://www-nds.iaea.org/drosg2000.html

TARGET

The adjoint Monte-Carlo code TARGET [4] calculates the neutron energy spectrum produced by protons and deuterons impinging on solid (Li, LiF, LiOH, Sc, T-Ti, D-Ti) and gas (D_2) targets. The code computes the uncollided and target scattered neutrons contributions to the cylinder detector of given size and location, as well as for proton recoil telescope. It makes use of Drosg's neutron production cross section data [2]. The proton and deuteron transport in target materials is modelled on the base of Ziegler' data with inclusion of energy and angle straggling [4].

EnergySet and NeuSDesc

EnergySet [5] is a program developed by IRMM that calculates neutron energy spectra and fluencies for most commonly used quasi mono-energetic neutron sources employing the reactions: ^{7}Li(p,n)^{7}Be, T(p,n)^{3}He, D(d,n)^{3}He, ^{7}Li(α,n)^{10}B, T(d,n)^{4}He, ^{7}Li(p,n)^{7}Be*, ^{45}Sc(p,n)^{45}Ti and D(d,pn)D. The program uses kinematics for real-time neutron spectrum calculations as a function of ion energy, neutron emission angle, reaction target thickness, measurement distance and ion beam current. Also related accelerator parameters are calculated.

NeuSDesc (Neutron Source Description) [6] is an extended version of the EnergySet program including the option of creation an MCNP input file containing the description of the neutron field at an arbitrary surface (so called SDEF card). NeuSDesc does not, however, provide any accelerator parameters. Reaction cross section data are taken from H. Liskien and A. Paulsen [7], the ion stopping power - from SRIM-2008 [8].

Package is available from
http://www.irmm.jrc.be/about_IRMM/laboratories/Pages/the_van_de_graaff_laboratory.aspx.

PINO

The Monte-Carlo based program PINO (Protons In Neutrons Out) [9] simulates neutron spectra from the ^{7}Li(p,n) source (Li, LiF or LiO_2 thick targets) considering the geometry of the setup and the proton-energy distribution. The code was designed and validated for simulation of neutrons produced in moderately thick targets and for proton energies slightly exceeding the ^{7}Li(p,n) reaction threshold.
The double-differential cross-sections are taken from the compilation by Liskien and Paulsen [7], stopping power data are from SRIM [8]. The protons and the neutron transport are very simplified: protons do not scatter in the thin lithium layer, straggling is neglected, neutrons are not tracked - only position and angle at the production point are considered. The code estimates a spectrum seen by disk-like samples.

The calculations can be done on-line using the web interface
http://exp-astro.physik.uni-frankfurt.de/pino/.

FNG Source Subroutine

The FNG source subroutine was developed and validated at ENEA [10, 11] to simulate the neutron yield from thick titanium-tritium targets bombarded by hundred-keV deuterons (fusion D–T source). It was designed as a special subroutine for use (compilation) together with the MCNP code family: at a given incident deuteron energy it samples a neutron with particular energy, emission angle and weight for the further transport by MCNP.

The FNG source simulates the effect of the deuteron slowing down inside a titanium-tritium target by electrons using the models implemented in the TRIM/SRIM package. The T(d,n) double differential cross-sections were retrieved from the DROSG2000 code [1].

The latest version of this subroutine [12] includes several modifications: it takes D-T reaction anisotropies from ENDF/B–VII data, uses relativistic kinematics to convert data from centre of mass to target frame, drops the special treatment of the first deuteron collision that avoids artificial spikes in the spectrum.

The source subroutine is appropriate for analysis of 14 MeV neutron generator with solid TiT target, while the facility-dependent neutron scattering on the target holding materials could be simulated by exact representation of the target design in the MCNP input file.

The subroutines are available in SINBAD database
http://www.oecd-nea.org/science/shielding/sinbad/sinbadi.htm.

FILTER

The package was developed and then regularly upgraded by O. Gritzay [13]. It contains special data library and computer code to calculate neutron spectra formed by interference neutron filters installed in the beam line at Kyiv light water research reactor.

The special data library consists of a set of separate files with the total neutron cross section in the point-wise format. These total neutron cross sections were obtained from latest evaluated data libraries in the energy range from 10^{-5} eV to 20 MeV and a temperature of 300K by the PREPRO or NJOY codes.

The code allows simulation of practically any combination of materials and isotopes for the filter to get quasi-mono-energetic neutron spectrum with necessary energy, intensity and line purity.

The code uses simple expressions for calculations of the transmission through the filter materials. This approach is valid for the strong collimation of the reactor beam in the neutron filter system. It was tested by exact simulation of experimental set-up using the MCNP-4C code.

Detailed information is available from UKRNDC web page: http://ukrndc.kinr.kiev.ua/.

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APPENDIX 6

IAEA INFORMATION RESOURCES ON REACTOR, RADIOACTIVE AND SPALLATION NEUTRON SOURCES

The Nuclear Data Section and Physics Section of IAEA, undertaking the Agency activity in the areas of nuclear data development and relevant technology transfer, systematically collected information on the availability and characteristics of neutron sources at different facilities. This information is available from the IAEA reports and databases. A short overview of the IAEA resources is given below.

STANDARD AND REPRODUCIBLE NEUTRON FIELDS IN IRDF-2002

The International Reactor Dosimetry File IRDF-2002 [1] contains a set of neutron source spectra available in numerical form on http://www-nds.iaea.org/irdf2002/index.htmlx. These fields were used by the IRDF community to validate evaluated cross sections for 35 activation reactions selected for reactor dosimetry. The numeral data are presented in the ENDF format in the energy range from 0.0001–0.01 eV to 18-20 MeV. To characterise the energy distribution, the energy range comprising 90% of the spectrum (in parentheses) and the average energy are specified.

Regarding the quality, the neutron fields are sorted in two classes [2]:

Standard Fields - *a permanent and reproducible neutron field with neutron flux intensity, energy spectra and angular flux distributions characterized to state-of-the-art accuracy:*

- ^{252}Cf spontaneous fission spectrum (0.26 – 5.5 MeV), average energy 2.13 MeV, 122 energy groups and 71×71 energy-energy covariances matrix evaluated between 15 keV and 20 MeV [3];
- Thermal Maxwellian flux (0.01 - 0.4 eV) – neutron thermal equilibrium distribution $N(E) \approx E\exp(-E/kT)$ with atoms of moderator at temperature T, averaged energy 0.00253 eV at room temperature 293 K [2];
- Near-1/E (6 eV – 3 MeV) – epithermal slowing down spectrum in a hydrogenous moderator $N(E) \approx 1/E$, average energy 0.75 MeV [2].

Reference Fields - *a permanent and reproducible neutron field, less well characterized than a standard but accepted as a measurement reference by a community of users*

- ^{235}U thermal fission (0.24 – 5.1 MeV) – from National Bureau of Standards and ENDF/B-V evaluations, average energy 1.97 MeV, 620 groups [4];
- Sigma-Sigma or ΣΣ (0.02 – 2.9 MeV) - coupled Thermal/Fast Uranium + Boron Carbide spherical assembly located within a conventional graphite thermal column, evaluated at NBS, average energy 0.76 MeV, 431 groups [4];
- ISFN - Intermediate-Energy Standard Neutron Field (0.008 – 3.5 MeV) – from NBS, average energy 0.80 MeV, 620 groups [4];
- BIG-TEN (0.04 – 2.3 MeV) - 10 per cents enriched Uranium cylindrical assembly, average energy 0.58 MeV, 396 groups [4];
- CFRMF (0.02 – 2.9 MeV) - Coupled Fast Reactivity Measurement Facility is a zoned core critical assembly, average energy 0.76 MeV, 460 groups [5];
- ORR – Oak Ridge Research Reactor, 100 groups [6];
- YAYOI (0.1 – 4.2 MeV) - average energy 1.50 MeV, 460 groups [6];
- Central Zone Flux of the NEACRP benchmark, 208 groups [7].

EXFOR has Entries that have the spectral averaged cross sections measured with neutron fields listed above (for example, #30568 contains the measured dosimetry cross sections with Sigma-Sigma source). NDS will analyse the necessity to have cross references between the EXFOR Entries with energy averaged cross sections and the corresponding neuron spectra available in IRDF-2002.

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RADIOACTIVE NEUTRON AND GAMMA SOURCES

The IAEA has collected neutron energy spectra for 23 isotopic (natural radioactive) and more than 400 operational (driven by research and power reactors, medical and high energy accelerators, cosmic rays) neutron sources [1]. The purpose was to provide guidance to health physicists and dosimetry specialists on the proper selection and use of dosimeters and survey instruments on the basis of their energy response characteristics and the neutron spectra in the environment where the devices are to be used. This report includes dosimetric quantities, response functions and neutron spectra.

The compendium presents an interest for the physics and EXFOR community mainly by its collection of radioactive neutron source spectra, since operational ones have not been used for cross reaction measurements. Indeed the natural radioactive neutron sources are often used for detector calibration, cross section measurements, benchmarking of evaluated data files and as reference fields for metrological purposes. They consist of an α-particle emitting radioisotope (^{238,239}Pu, ^{226}Ra or ^{210}Po) and a neutron converter (^{9}Be, ^{10,11}B or ^{19}F). Technologically they are manufactured either as homogeneous mixture (isotropic sources) or as heterogeneous or “sandwich” (neutron yields depends on emission angle).

Compendium has clean (bare source) neutron fields for isotopic sources ^{252}Cf, ^{241}Am-B, ^{241}Am-Be, ^{238}Pu-Be and ^{238}Pu-F. These spectra were either evaluated by International Organization for Standardization [2] or measured at PTB [3], CERN [4] and GSF [5] labs. The numerical spectral data are presented in sixty energy bins in the energy range between 1 meV and 630 MeV.

Comprehensive and detailed survey of neutron and gamma isotopic sources was also given in monographs [6-8]. They cover many different sources like ^{252}Cf, ^{226}Ra-Be, ^{210}Po-Be, ^{242,244}Cm-Be, ^{238}Pu-Be, ^{210}Po-B, ^{238}Pu-B ^{210}Po-F, ^{238}Pu-Li, both homo- and hetero-geneous types. The numerical energy spectra are available, e.g., in [9].

Not only clean isotopic sources, but also those surrounded by moderators or bulk structural materials were used for the measurements of spectrum-averaged cross sections. In these cases, neutron scattering causes considerable contributions of low energy neutrons (which however could be neglected when threshold reactions are measured) [10, 11].

Regarding the essential number of cross section data measured with clean radioactive neutron sources the NDS has recommended to NRDC to start collection and compilation of the relevant isotopic neutron

fields in EXFOR [12]. It was also found that incident source code A-BE is used presently for both accelerator driven and all radioactive α-Be neutron sources. To improve the accessibility of spectrum averaged cross sections in the EXFOR database, the new codes sources (AM-BE, CM-BE, PU-B etc.) for radioactive were proposed, keeping the already existing code (A-BE) only for neutrons generated at accelerators.

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SPALLATION NEUTRON SOURCES

Spallation nuclear reactions as neutron sources play an important role in a wide domain of applications: nuclear and particle physics, astrophysics, condensed matter and material studies, transmutation of nuclear waste and rare isotope production, calibration of detectors and radiation protection issues near accelerators and in space.

IAEA and the Abdus Salam International Centre for Theoretical Physics (ICTP, http://www.ictp.it/) have organised a set of experts' meetings on international benchmarking of spallation model codes such as MCNP(X) and GEANT. This activity included collection of all available experimental data on spallation reaction products and demonstration of the capability of these codes to represent them.

As a result, all double differential cross sections for neutron production by protons with energies between 60 to 2500 MeV incident on target nuclei C, Al, Fe, Cu, Ni, Ta, Au, Pb, Bi, U were collected in EXFOR. The detailed graphical and numerical (in terms of Figure of Merit) intercomparisons give an overall status of existing experimental and theoretical results for spallation neutron spectra (for details see http://www-nds.iaea.org/spallations/).

APPENDIX 7

NEUTRON SPECTRA UNFOLDING CODES

The determination of energy spectra by the multiple foil activation technique is a rather universal and well establish method. It is of advantage for measuring the spectral fluence in harsh radiation environments, at high temperatures and in compact volumes, where other detectors and methods are not applicable.

Many codes, each employing different models and algorithms, have been developed during the last decades for unfolding an energy spectrum from measured reaction rates. Based on published reference documents and manuals, we would like to give a short overview of used methods, computing environment and availability of the most popular neutron spectra unfolding or adjustment codes.

SAND-II-SNL

Sandia National Laboratories, in the process of characterizing the neutron environments at its reactor facilities, has developed an enhanced version of W. McElroy's original SAND-II code [1, 2]. The enhanced input, output, and plotting interfaces make the code much easier to use. The basic physics and operation of the code remain unchanged. Important code enhancements include the interfaces to the latest dosimetry-quality cross sections and the ability to use silicon displacement-sensitive devices as dosimetry sensors.

The SAND II code provides a "best fit" neutron flux spectrum for a given input set of infinitely dilute foil activities. The calculational procedure (non-linear least squares) consists of the selection of a known flux spectrum form to serve as the initial approximation to the solution, and subsequent iteration to a form acceptable as an appropriate solution, no exact uncertainty analysis. A reaction cross-section library is provided with the code or could be retrieved from IRDF-2002, http://www-nds.iaea.org/irdf2002/index.htmlx .

Coding language Fortran 77, SUN Sparcstation; available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory http://www-rsicc.ornl.gov (package P00345/SUN04/00).

HEPROW Package

Computer codes based on Bayes theorem and the maximum entropy method have been developed in the PTB and have been successfully applied in many metrological fields to determine the spectral neutron or photon fluence [3, 4]. It contains a number of programs for data handling, plotting, folding and, in particular, for unfolding: GRAVELW - modified SAND-II code, UNFANAW and MIEKEW. The propagation of uncertainties is performed with a so-called ambiguity term. The HEPROW program system is an updated version of the PTB HEPRO system [5].

The codes run in the DOS-Box of a PC, tested under WINDOWS-98 and are available from http://www.matzke-bs.de/heprow/. The MIEKE code (predecessor of MIEKEW) is also available from http://www-nds.iaea.org/ndspub/libraries/nmf/.

STAY'SL

Computer program STAY'SL [6] solves the dosimetry unfolding problem by the method of linear least squares. The solution (the output spectrum and its covariance matrix) is calculated by minimizing chi-square based on the input data (the activation data, a priori spectrum, the dosimetry cross sections and their uncertainties given by covariance matrices). The solution reflects therefore the uncertainties in all of the input data and their correlations. The correlations among the various dosimetry cross sections are taken into account; however, the activation data, input spectrum and cross sections as classes are assumed to be uncorrelated with each other.

A FORTRAN code, sample problem and documentation are available from the Radiation Shielding Information Center (RSIC) at Oak Ridge National Laboratory (http://www-rsicc.ornl.gov, package P00113) or from NDS of IAEA (http://www-nds.iaea.org/irdf2002/codes/index.htmlx, IRDF-2002).

STAYNL

The code STAYNL [7] is a modified and extended follow-up of the code STAY'SL. It solves the neutron spectrum adjustment problem by the generalized least square method (GLSM). The spectrum normalization before adjustment is performed by GLSQM, and the cross-covariance terms derived from the calculation of the covariance matrix of the reaction rates also taken into account.

The program is written in standard FORTRAN-7 language, is a part of the Neutron Metrology File NMF-90 and could be downloaded from http://www-nds.iaea.org/ndspub/libraries/nmf/.

LSL

LSL-M2 - Least-Squares Logarithmic Adjustment of Neutron Spectra [8] - adjusts calculated neutron spectra to make the fluence values consistent with given neutron dosimetry measurements. In an adjustment procedure, consistency is achieved by adjusting the input data (a priori spectrum) in such a manner that a weighted sum of squares of adjustments is minimized. The weights are assigned according to the input uncertainties, i.e., the larger the uncertainty, the smaller the weight. This sum, which is further modified by correlations, represents the negative logarithm of the probability of the outcome of the experiment if the adjusted values are the ``true'' data.

Fortran 77, IBM PC or VAX family, available from RSICC http://www-rsicc.ornl.gov/ (package PSR-233) and from NDS of IAEA http://www-nds.iaea.org/ndspub/libraries/nmf/ (package NMF-90).

SULSA

Code SULSA [9] implements a new method for the neutron spectrum unfolding based on the improvement of the generalized least square method. It allows an iterative treatment of the problem and keep the number of the independent statistical variables of the solution lower than the number of the measured reaction rates. An advantage of the method is that the neutron spectrum can be determined when no prior information is available.

It uses the cross section data and their respective covariance matrices from the IRDF-90 (2002) Dosimetry File to get detailed representation of the energy grid with successive approximation.

NSVA-3

A new third version [10] of the spectrum least-squares adjustment code NSVA (Neutron Spectrum Validation and Adjustment) [11] has been developed and is being made available to the community. The designation of NSVA-3 is that it simultaneously adjusts spectra for multiple environments. Items of data may be easily swapped in or out of the calculation. The data input requires the fluence spectra, dosimetry measurements with the standard deviations and correlation matrices. In the case of multiple environments, the cross correlations between environments of the input fluence and dosimetry measurements can also be included. An 89-group cross section library including covariance matrices, based on SNLRML [12], is incorporated in the code package.

The code is written in MATLAB®, it allows variable-dimension matrix operations to be written as simple commands, and also provides facilities for the input and output of data to and from spreadsheets. The main advantage of the NSVA code is the usage of graphic user interfaces GUIs to assist with the data input and in interactive execution of adjustment cases.

NVA-3 is run under the MATLAB® (version 6.5 and higher) environment and is designed for a Windows operating system, but is can also be executed on UNIX machines.

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APPENDIX 8

Compilation of Data Averaged over Incident Neutron Spectra for EXFOR Library

N. Otuka, V. Semkova, S.P. Simakov
Nuclear Data Section, International Atomic Energy Agency, A-1400 Vienna, Austria

1. Introduction

Cross section and other experimental neutron-induced reaction data measured under broad neutron spectra have been compiled in the EXFOR Library maintained by the International Network of Nuclear Reaction Data Centres (NRDC) [1]. In the current EXFOR Formats rule, the neutron field used in the measurement is indicated by a "modifier" (Table 1) as a part of the code under the keyword **REACTION**, and its characteristic energy is coded with the measured cross section (Fig. 1). Following the introduction of Westcott's convention, two spectrum modifiers MXW and SPA and two definitions of Maxwellian averaged cross sections (MACS) are presently used in compilation practice. To remove ambiguity, the compilation rules for neutron spectrum averaged quantity and representative energy are discussed in this Appendix.

Table 1. *Selected modifiers in the EXFOR dictionary*

Code	Definition
MXW	Maxwellian average
EPI	Epithermal neutron spectrum average
FIS	Fission neutron spectrum average
FST	Fast reactor neutron spectrum average
SPA	Spectrum average (not specified above)
AV	Average

```
SUBENT         12438004   19830809
BIB                    1          1
REACTION   (92-U-235(N,F),,SIG,,MXW)
ENDBIB                 1
NOCOMMON               0          0
DATA                   3          1
EN-DUMMY   DATA         DATA-ERR
EV         B            B
0.0253     552.         55.
ENDDATA                3          0
ENDSUBENT              9          0
```

Fig. 1. EXFOR Entry: ^{235}U neutron-induced fission cross section measured in a thermal reactor and coded as a Maxwellian spectrum averaged (MXW) cross section [2].

2. Quantities defined with two codes MXW and SPA in EXFOR

For cross sections measured at thermal reactors, the spectrum averaged or effective cross section $\langle\sigma_{SPA}\rangle$ is introduced by the Westcott's convention [3] in terms of the reaction rate R:

$$R = \int_0^\infty n_0 n(v)\sigma(v)v\mathrm{d}v \equiv n_0 v_0 \langle\sigma_{\mathrm{SPA}}\rangle \quad (1)$$

with

$$\int_0^\infty n(v)\mathrm{d}v = 1 \quad (2)$$

where $n(v)$ is the neutron velocity distribution normalized to 1, n_0 is the total neutron density, and v_0 is the thermal neutron velocity (2200 m/s) at the room temperature kT_0 = 0.0253 eV. The effective cross section is related to the 2200 m/s cross section (thermal neutron cross section) $\sigma_0 = \sigma(v_0)$:

$$\left\langle \sigma_{\mathrm{SPA}} \right\rangle = \left\langle \sigma_{\mathrm{MXW}}(kT) \right\rangle (1 + rs_0 \sqrt{T/T_0}) = \sigma_0 g (1 + rs_0 \sqrt{T/T_0}), \quad (3)$$

Here $<\sigma_{\mathrm{MXW}}(kT)>$ is the effective cross section for incident neutrons with the Maxwell energy distribution (Maxwellian averaged cross section, MACS) at the temperature T:

$$\left\langle \sigma_{\mathrm{MXW}}(kT) \right\rangle = \frac{1}{v_0} \int_0^\infty n_{\mathrm{MXW}}(v) \sigma(v) v \mathrm{d}v \equiv \frac{\left\langle \sigma v \right\rangle_{kT}}{v_0}, \quad (4)$$

where g is the ratio of $<\sigma_{\mathrm{MXW}}(kT)>$ to the 2200 m/s cross section σ_0 (Westcott's g-factor), r is the epithermal index ($r = 0$ if there is no epithermal flux) and s_0 is a quantity depending on the temperature and resonance integral ($s_0 = 0$ if $\sigma \propto 1/v$). See ref. [4] for more details.
Note that the explicit form of Maxwellian velocity distribution is

$$n_{MXW}(v) = \frac{4}{v_T^3 \sqrt{\pi}} v^2 \exp\left(-\frac{v^2}{v_T^2} \right) \quad (5)$$

The velocity $v_T = \sqrt{2kT/m}$ is the most probable velocity corresponding to the peak of the velocity distribution, and it is equal to v_0 when $T = T_0$. If $\sigma \propto 1/v$, Eqs. (4) and (5) give (namely $g = 1$):

$$\left\langle \sigma_{\mathrm{MXW}}(kT) \right\rangle = \sigma_0 \quad (6)$$

Eq. (4) can be also expressed in terms of the neutron energy E:

$$\left\langle \sigma_{\mathrm{MXW}}(kT) \right\rangle = \frac{2}{\sqrt{\pi}} \frac{\int_0^\infty E \exp(-E/kT) \sigma(E) \mathrm{d}E}{\int_0^\infty E \exp(-E/kT) \mathrm{d}E} \quad (7)$$

Experimentally, $<\sigma_{\mathrm{MXW}}>$ is determined through determination of parameters r and s_0 by measurements with and without a Cd cover sheet around the sample.

3. Issues important for the EXFOR compilation

The moments essential for the correct compiling of the energy averaged data are considered in the next sub-sections.

3.1 Selection of modifiers - MXW or SPA

According to the EXFOR rules (Table 1), two modifiers `MXW` and `SPA` should be applied to the effective cross sections for Maxwellian and more generic spectra $n(v)$. However, it is not always obvious whether the cross section reported by the author as the "thermal cross section" is $<\sigma_{\mathrm{SPA}}>$, $<\sigma_{\mathrm{MXW}}>$ or σ_0. The situation is similar for other quantities (e.g., fission yields) measured in thermal reactors. Since `SPA` is too generic (any spectrum can be expressed by this code), some EXFOR compilers prefer to use `MXW` rather than `SPA` assuming that the effect of the epithermal flux is negligible. Moreover they sometimes omit both `MXW` and `SPA` codes when no description of the neutron spectrum is given in the article. EXFOR compilers do not have a clear prescription for these cases.

3.2 Two definitions of the Maxwellian Averaged Cross Sections (MACS)

A straightforward definition cross section averaged for a given spectrum is

$$\hat{\sigma}_{\mathrm{SPA}} = \frac{\int_0^\infty n(v) \sigma(v) v \mathrm{d}v}{\int_0^\infty n(v) v \mathrm{d}v} = \frac{1}{\bar{v}} \int_0^\infty n(v) \sigma(v) v \mathrm{d}v \quad (8)$$

For the Maxwellian distribution, this becomes

$$\hat{\sigma}_{\mathrm{MXW}}(kT) = \frac{\int_0^\infty n_{\mathrm{MXW}}(v) \sigma(v) v \mathrm{d}v}{\int_0^\infty n_{\mathrm{MXW}}(v) v \mathrm{d}v} = \frac{\left\langle \sigma v \right\rangle_{kT}}{\bar{v}} \quad (9)$$

Since the mean velocity in the Maxwellian distribution $\bar{v} = 2\sqrt{2kT/m\pi}$ is a factor $2/\sqrt{\pi}$ (~1.128) larger than v_T, it results to the following difference of MACS at the room temperature $T=T_0$ in accordance with two definitions Eqs. (4) and (9):

$$\langle \sigma_{\mathrm{MXW}}(kT_0) \rangle = \frac{2}{\sqrt{\pi}} \hat{\sigma}_{\mathrm{MXW}}(kT_0) \tag{10}$$

Usually the definition based on Westcott's convention $<\sigma_{MXW}(kT_0)>$ is adopted as the definition of MACS in the thermal neutron region. The situation is the same for the MACS at the stellar temperature $kT \sim 30$ keV (see, e.g. Eq. (2) in [5]). However, Eq. (9) is also used as another definition of MACS at the room temperature (see, e.g. Table II in [6]). Currently both MACS are coded with the same modifier MXW in the EXFOR Library.

3.3 Renormalization of existing data by new spectrum averaged experimental data

Various reactions relevant to the s-process nucleosynthesis have been studied by using $kT = 25$ keV quasi-Maxwellian spectrum neutrons produced by the ^{7}Li(p,n)^{7}Be reaction at the Karlsruhe 3.7 MV Van de Graaff accelerator. Though the neutron source simulates the Maxwellian spectrum almost perfectly, it has a maximum neutron energy (high energy limit), e.g., $E_{n,max} = 106$ keV at the incident proton beam energy $E_p = 1912$ keV [5]. Note that genuine Maxwellian distribution extends to the infinite energies. The Karlsruhe group has utilized their experimental quasi-Maxwellian spectrum averaged cross sections $\hat{\sigma}_{\mathrm{exp,\,qMXW}}$ for renormalization of the Maxwellian averaged cross sections constructed from an existing point-wise cross section data sets folded by the Maxwellian spectrum $\hat{\sigma}_{\mathrm{pw,MXW}}$:

$$\hat{\sigma}_{\mathrm{pw,\,MXW,renorm}} = \frac{\hat{\sigma}_{\mathrm{exp,\,qMXW}}}{\hat{\sigma}_{\mathrm{pw,qMXW}}} \hat{\sigma}_{\mathrm{pw,\,MXW}} \tag{11}$$

where $\hat{\sigma}_{\mathrm{pw,qMXW}}$ is an existing point-wise cross section data set folded by the quasi-Maxwellian spectrum.

For example, the measured spectrum averaged ^{197}Au(n,γ)^{198}Au cross section under the quasi-Maxwellian spectrum $\hat{\sigma}_{\mathrm{exp,qMXW}} = 586$ mb [5] was used to renormalize Macklin's experimental point-wise cross section [7], which give $\hat{\sigma}_{\mathrm{pw,MXW}} = 557$ mb and $\hat{\sigma}_{\mathrm{pw,qMXW}} = 568$ mb. Applying a renormalization factor $\hat{\sigma}_{\mathrm{exp,qMXW}} / \hat{\sigma}_{\mathrm{pw,qMXW}} = 586/568 = 1.04$ to $\hat{\sigma}_{\mathrm{pw,MXW}}$, and then correcting by the factor $2/\sqrt{\pi}$, the MACS derived from Macklin's experimental data was finally renormalized to

$$\langle \sigma_{\mathrm{MXW}}(kT = 25\,\mathrm{keV}) \rangle = \frac{2}{\sqrt{\pi}} \sigma_{\mathrm{pw,renorm}} = 648\,\mathrm{mb} \tag{12}$$

MACS obtained from original Macklin's point-wise cross sections at various temperatures were also renormalized by the same normalization factor. A similar approach is seen in renormalization of MACS derived from point-wise evaluated nuclear data libraries (e.g., the normalization factor *NF* introduced in [8]).

EXFOR compilers have compiled these renormalized MACS as an original experimental data set. F. Käppeler has reported at this Consultant' Meeting that the MACS renormalized by the Karlsruhe group have been compiled in the KADONiS (Karlsruhe Astrophysical Database of Nucleosynthesis in Stars) database [9], and he proposed to exclude these renormalized values from EXFOR compilation. Note that $\hat{\sigma}_{\mathrm{pw,qMXW}} = 586$ mb mentioned above is currently coded with MXW in EXFOR 22099, and it must be changed to SPA.

4. Characteristic energy of spectrum averaged quantities in EXFOR

The EXFOR Format rules always require the incident energy as an independent variable of each experimental data set except for spontaneous fission data. For the spectrum averaged quantities, a characteristic energy of the spectrum is coded and used as a key to search data at specific energy ranges. If authors do not define any characteristic neutron energy, the EXFOR compiler gives an assumed characteristic energy under the heading `EN-DUMMY`. For typical spectra, assumed values are defined in the EXFOR coding rule (Table 2). Currently we do not have a common assumed value for fast reactor neutron fields.

Table 2. *Characteristic energies used in EXFOR under heading* `EN-DUMMY` [10]

Energy	Neutron field
0.0005 eV	Cold neutron field
0.0253 eV	Thermal reactor neutron field
1.5 MeV	Prompt fission neutron field
4.5 MeV	Decay α-Be neutron field

Below two proposals on characteristic energies are considered.

4.1. Cold and thermal reactor neutron fields

Usually the cold and thermal reactor neutron fields are approximated by Maxwellian distribution at a given temperature. The dummy energy 0.0253 eV in Table 2 is equal to the room temperature *kT*. Therefore the heading `KT` would be more suitable than `EN-DUMMY` for the characteristic energies when the contribution of the epithermal neutrons is negligible. Note that numerous EXFOR entries coded with 0.0253 eV under the heading `EN-DUMMY` are already exist, and we should not do retroactive corrections only for this issue.

4.2. Prompt fission neutron fields

The prompt neutron fission spectrum averaged cross section is defined according to Eq. (8):

$$\hat{\sigma}_{\mathrm{PFNS}} = \frac{\int_0^\infty n_{\mathrm{PFNS}}(v)\sigma(v)v\mathrm{d}v}{\int_0^\infty n_{\mathrm{PFNS}}(v)v\mathrm{d}v} = \frac{\int_0^\infty n_{\mathrm{PFNS}}(E)\sigma(E)\mathrm{d}E}{\int_0^\infty n_{\mathrm{PFNS}}(E)\mathrm{d}E} \qquad (13)$$

It is known that the prompt fission neutron spectrum (PFNS) is well approximated by Maxwellian spectrum at a given temperature. Therefore the heading `KT` can be used to express a characteristic energy of the prompt fission neutron field, $\langle E \rangle = 3/2kT$. For the spontaneous fission of ^{252}Cf and neutron-induced fission of ^{235}U (by both thermal and fast neutrons), the temperatures *kT* are about 1.42 MeV and 1.32 MeV, respectively. These temperatures coded under the heading `KT` would be more appropriate as characteristic energies of PFNS averaged cross sections in EXFOR.

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Nuclear Data Section
International Atomic Energy Agency
P.O. Box 100
A-1400 Vienna
Austria

e-mail: services@iaeand.iaea.org
fax: (43-1) 26007
telephone: (43-1) 2600-21710
Web: http://www-nds.iaea.org/