IAEA
International Atomic Energy Agency

INDC(NDS)-0682
Distr. AC/AD/G/IN/J/RD/SC/ST

INDC International Nuclear Data Committee

Summary Report of the Second Research Coordination Meeting on

Testing and Improving the International Reactor Dosimetry and Fusion File (IRDFF)

IAEA Headquarters, Vienna, Austria

16 – 20 March 2015

Prepared by

P.J. Griffin, Sandia National Laboratories, Albuquerque, NM, USA

L.R. Greenwood, Pacific Northwest National Laboratory, Richland, WA, USA

S.P. Simakov, Nuclear Data Section, IAEA, Vienna, Austria

June 2015

IAEA Nuclear Data Section
Vienna International Centre, P.O. Box 100, 1400 Vienna, Austria

Selected INDC documents may be downloaded in electronic form from
http://www-nds.iaea.org/publications
or sent as an e-mail attachment. Requests for hardcopy or e-mail transmittal should be directed to *NDS.Contact-Point@iaea.org*
or to:
Nuclear Data Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna
Austria

Printed by the IAEA in Austria
June 2015

IAEA
International Atomic Energy Agency

INDC(NDS)-0682
Distr. AC/AD/G/IN/J/RD/SC

INDC International Nuclear Data Committee

Summary Report of the First Research Coordination Meeting on

Testing and Improving the International Reactor Dosimetry and Fusion File (IRDFF)

IAEA Headquarters, Vienna, Austria

16 – 20 March 2015

Prepared by

P.J. Griffin, Sandia National Laboratories, Albuquerque, NM, USA

L.R. Greenwood, Pacific Northwest National Laboratory, Richland, WA, USA

S.P. Simakov, Nuclear Data Section, IAEA, Vienna, Austria

Abstract

The second Research Coordination Meeting (RCM-2) of the Coordinated Research Project (CRP F41031) "Testing and Improving the International Reactor Dosimetry and Fusion File (IRDFF)" was held 16 to 20 March 2015 in IAEA. At this meeting, the attendees presented their individual research contributions to CRP, summarised results and elaborated consolidated recommendations and actions for implementation for the period until RCM-3 which have to take place in the beginning of 2017. This Meeting also considered a perspective of the new Exercise on the verification of the neutron spectrum adjustment codes.

June 2015

Table of Contents

I. Introduction

1.1. The IAEA CRP on IRDFF

The Nuclear Data Section of IAEA, in accordance with the recommendation of the International Nuclear Data Committee Meeting 2012 [1], runs a Coordinated Research Project (CRP) with the main goal being to test, validate, and improve the international dosimetry library for fission and fusion, which during historical evolution was called the International Reactor Dosimetry and Fusion File (IRDFF) [2, 3]. The IRDFF extends the energy range up to 60 MeV, beyond the traditional 20 MeV limist of IRDF-2002 [4], and includes additional nuclear activation reactions. The CRP formally started at the 1st RCM held in July 2013 [5] and is planned to continue until 2017.

The participants of the CRP plan to measure new energy integrated (integral) and pointwise (energy-dependent) cross sections as well as to collect all other experimental information relevant to data validation, which has not historically been utilised. Additional and more detailed information is collected on the CRP web-page http://www-nds.iaea.org/IRDFFtest/.

The main CRP output will be the updated and validated database of dosimetry cross sections, decay and other data with corresponding documentation. This library will serve the needs of fission, fusion (International Thermonuclear Experimental Reactor, ITER; International Fusion Material Irradiation Facility, IFMIF) and accelerator applications (Accelerator Driven Systems, ADS; Spallation Neutron Sources, SNS).

1.2. The Second Research Coordination Meeting (RCM-2)

The second RCM of the IAEA CRP on "Testing and Improving the International Dosimetry library for Fission and Fusion (IRDFF)" was held at IAEA Headquarters, Vienna, Austria from 16 to 20 March 2015. The fourteen holders of CRP Agreements or Contracts attended this meeting: M. Angelone, V. Chechev, C. Destouches, I. Kodeli, C. Konno, L. Greenwood, P. Griffin, M. Majerle, P. Maleka, P. Mastinu, R. Nchodu, A. Plompen, H. Yashima, M. White. V. Pronyaev and K. Zolotarev participated as observers. The IAEA was represented by S. Simakov (Project Officer), R. Capote Noy (Alternative Project Officer), R. Forrest, N. Otuka, V. Semkova, A. Trkov, V. Zerkin (NDS Staff), and Benjaminas Marcinkevicius (NDS Intern).

The Meeting was opened by R. Forrest, Head of the Nuclear Data Section (NDS) of the Department of Nuclear Sciences and Applications of the IAEA, by welcoming the participants and explaining the importance of this RCM for the further coordination of the CRP work on testing and improvement of the IRDFF as a reference dosimetry library used in many practical applications.

Mr A. Öchs, who is responsible for administrative issues, made several announcements. This was followed by the self-introduction of new participants.

The participants elected P. Griffin as the Chairman and L. Greenwood as the Rapporteur of this Meeting and approved its Agenda (Appendix 1). The list of participants and their affiliations is summarized in Appendix 2.

The current objective and goals of the Meeting were outlined by S. Simakov in dedicated presentation.

During two and half days, participants gave the individual oral presentations (are available on the CRP web-page at http://www-nds.iaea.org/IRDFFtest/RCM2), summarised a progress achieved in IRDFF (Section II) and had discussions which resulted in a set of consolidated conclusions and recommendations (Section III). The summaries of participants' presentations which describes an individual contributions and future plans for this CRP are collected in Section IV.

The Nuclear Data Section acknowledged all participants for their cooperation and contributions to the Meeting and especially stressed the role of national experimental facilities in the production of new

experimental data for dosimetry and eventually for the successful implementation of this international project. We hope that the necessary resources for maintenance of these facilities and for carrying out precise measurements will be provided by the relevant authorities.

II. Evolution and Validation of IRDFF since RCM-1 (July 2013)

This section compactly summarizes the IRDFF database evolution and new validation results for the period since the previous RCM meeting. The full results and detailed descriptions are listed in the references given in this Section as well as in the Individual Summaries (Section IV).

Since validation of the dosimetry reaction cross sections is one of the main goals of this CRP we started systematically identifying and collecting the needs for new measurements in *https://www-nds.iaea.org/IRDFFtest/HPRL.pdf*. This list is submitted to the High Priority Request List (HPRL) maintained by the NEA Data Bank.

2.1. Evolution of the IRDFF database since RCM-1 and further perspectives

Several releases of the IRDFF cross section database have been issued since 2013 (see the versions and changes at *https://www-nds.iaea.org/IRDFFtest/Table4.pdf*):

- IRDFF-1.03 (Mar 2014) - included an evaluation of the new reaction ^{238}U(n,2n)^{237}U and updates of ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,2n)^{57}Ni, ^{93}Nb(n,γ)^{94}Nb,^{115}In(n,γ)^{116m}In made by K. Zolotarev [6];
- IRDFF-1.04 (Nov 2014) - had minor format corrections compared with IRDFF-1.03;
- IRDFF-1.05 (Oct 2014) - adopted an evaluation of new reactions ^{28}Si(n,p)^{28}Al, ^{29}Si(n,x)^{28}Al, ^{113}In(n,γ)^{114m}In and updates of ^{31}P(n,p)^{31}Si by K. Zolotarev [7].

The new or updated evaluations were extended, when needed, up to 60 MeV using the evaluated cross sections and covariance matrices from the TENDL-2013 library [8] as described in [9].

The new releases of IRDFF were processed using the PREPRO [10] and RR_UNC [11] codes to generate the point wise and spectrum average cross sections.

The ACE formatted files of the IRDFF library were generated by the NJOY-99 or NJOY-2012 processing codes [12] using the latest updates provided by A. Kahler. As additional testing of the whole processing procedure the NDS calculated SPA from the IRDFF ACE library using the MCNP-5 code (see example *SPA MCNP input*). The exact agreement of these results with results calculated analytically with the RR_UNC code confirms the correctness of the procedures that were applied in the preparation of the new IRDFF libraries.

The final IRDFF cross section data were made available on the NDS web-sites: *https://www-nds.iaea.org/IRDFF/* or *https://www-nds.iaea.org/IRDFFtest/*.

Following the recommendations of RCM-1, the requests from the fusion community experts were reviewed taking into account the availability of experimental data (https://www-nds.iaea.org/IRDFFtest/Reactions-to-Evaluate.pdf). It was decided that the next release of IRDFF will include:

- re-evaluation of the ^{23}Na(n,γ) and ^{23}Na(n,2n)^{22}Na reactions and
- a new evaluation of the ^{27}Al(n,2n)^{26}Al reaction cross sections.

Another recommendation is the evaluation and inclusion of the ^{37}Cl(n,γ)^{38}Cl and ^{127}I(n,γ)^{128}I reactions for the fission reactor dosimetry applications.

For extension of IRDFF **to the high energy** applications, the (n,xn) reactions on ^{197}Au, ^{169}Tm, ^{209}Bi, ^{59}Co, ^{63}Cu, ^{89}Y, ^{93}Nb, ^{103}Rh, ^{139}La are recommended additions to support proper energy coverage in this neutron energy region. Up to now, for many of these reactions, the existing experimental cross section data are scarce or discrepant.

At this meeting the results of a preliminary evaluation of the ^{209}Bi(n,xn) reactions (x = 2 - 10) up to 100 MeV were presented by V. Pronyaev and made available on the CRP web as ENDF formatted files: https://www-nds.iaea.org/IRDFFtest/RCM2/Pronyaev_209Bi(n,xn)-final-evaluation.txt. This evaluation is based on the mean square analysis of the existing experimental cross sections with inclusion of results from reaction modelling and constrains from the non-elastic and charged-particles production channels.

This investigation has indicated that there are large discrepancies between measured and ENDF/B-VII.1 [12], TENDL-2012 and other evaluated cross sections for reactions ^{209}Bi(n,xn) when $x \geq 4$. If these evaluated data are used for estimation of the contribution from the non-monoenergetic source neutrons, they result to the very different correction factors at the incident neutron energies above the position of the reaction cross section maximum.

2.2. Validation of IRDFF in ^{252}Cf(s.f.) and ^{235}U(n_{th},f) fields

The validation of the initial release IRDFF-1.00 in the ^{252}Cf(s.f,) and ^{235}U(n,f) fields was done previously in [2] and for an energy cut-off at 20 MeV. In the frame of this CRP the validation of the later versions of IRDFF is being continued [12].

The performance of the IRDFF cross section in the standard and reference fields was usually expressed in terms of the ratio of Calculated-to-Experimental (C/E) Spectrum-Averaged cross sections (SPA). For this the available experimental SPA data were systematically collected on the CRP web page:

https://www-nds.iaea.org/IRDFFtest/SPA_Exp_Cf252.pdf and
https://www-nds.iaea.org/IRDFFtest/SPA_Exp_U235.pdf.

These data comprise mainly data points recommended by experienced evaluators (W. Mannhart and K. Zolotorev) after their renormalisation to the current standards, uncertainty analysis and eventually weighting of known experimental results. Other data included in this collection are from individual measurements. NDS systematically checks the EXFOR content for SPA and various neutron source spectra and compiles relevant data still missing in this repository.

For the ^{252}Cf(s.f.) source, the C/E ratio fluctuates around unity as a rule within the associated total uncertainty and thus confirms the overall quality of the IRDFF cross sections. However there are several obvious outliers including ^{59}Co(n,γ), ^{92}Mo(n,p), ^{60}Ni(n,p) and ^{46}Ti(n,2n). The likely reason in these cases are experimental errors. Thus we recommend new measurements for these reactions.

For the non-threshold neutron capture reactions it is important to take into account the possible contribution of neutrons scattered by the room and ^{252}Cf source holder. The recent re-evaluation of two past experiments has shown a need for 30% corrections for the reported measured SPA [14].

For the ^{235}U(n_{th},f) field, the C/E ratios are close to unity for most of the reactions with $E_{50\%} < 10$ MeV. However there are several exceptions such as ^{10}B(n,α) and ^{6}Li(n,t)^{4}He reactions. Since known experimental SPA were measured by helium-gas counting technique, the other alpha producing reaction ^{10}B(n,t2α) or ^{6}Li(n,ndα) and (n,2npα) must be considered in the comparison. Calculation of the corresponding SPA taking into account addition reaction channels from ENDF/B-VII.1 results in C/E = 1.0.

The reactions ^{55}Mn(n,γ), ^{238}U(n,γ), ^{139}La(n,γ), ^{31}P(n,p) and ^{238}U(n,2n) were found to be outliers and thus need further investigations.

Above 10 MeV, the C/E shows a gradually increasing underestimation. The uncertainty associated with ENDF/B-VII.1 PFNS is rather large and begins to dominate in the C/E uncertainty above 4 MeV and even exceeds 50% above 8 MeV.

The impact of the spectrum extension above 20 MeV was studied: it amounts to 1 - 2% for reactions with $E_{50\%} > 12$ MeV, however it becomes stronger for the higher threshold reactions (whose SPA are not measured yet), e.g. it reaches a factor of 29X for the ^{59}Co(n,3n) reaction.

2.3. Validation in the reference reactor facilities

Another task of the CRP is to revisit the older reference data obtained at reactor facilities such as coupled thermal/fast uranium and boron carbide spherical assembly (Sigma-Sigma), Coupled Fast Reactivity Measurement Facility (CFRMF), Intermediate-energy Standard Neutron Fields (ISNF), and Glory hole of the Tokyo University research reactor (YAYOI). Their spectra were included in the IRDF-2002 database [4] however measurements in these neutron fields were not previously incorporated into the evidence package supporting validation of the IRDFF cross section library.

We selected these facilities and spectra from the IRDF-2002 collection for IRDFF validation [12]. It was found that the *a priori* calculated ISNF spectrum obviously has unphysical irregularities in the vicinity of 0.4 MeV. The re-simulation of this facility resulted in an updated neutron spectrum without such oscillations.

The deviation of C/E from unity, for most reactions and facilities (except YAYOI, whose spectrum is thought to be incorrectly presented in IRDF-2002), lies within two to three sigma uncertainty bars.

These research reactor benchmarks have much softer spectra than the fission source $^{235}U(n_{th},f)$ and thus they deliver validation for reactions not validated so far with the harder ^{252}Cf or $^{235}U(n_{th},f)$ spectra such as the (n,γ) reactions on ^{45}Sc, ^{55}Mn, ^{58}Fe, ^{109}Ag – the observed agreement is within 2-3 sigma uncertainty bars.

2.4. Validation against thermal capture and Fusion benchmarks

The comparison [16] of the thermal capture cross sections between the IRDFF, IRDF-2002, IRDF-90 and the experimental KAYZERO [17] and Mughabghab [18, 19] values was done for 15 reaction cross sections, referring either to the total radiative capture cross section, or the excitation of long-lived metastable states. It was shown that the latest versions of IRDFF exhibit similar performance as previous versions.

A series of shielding benchmarks available from the SINBAD database were used to check and validate the new IRDFF dosimetry file, version v. 1.04 (updated $^{6}Li(n,t)$ data were taken from v. 1.05). Several benchmark experiments performed at the Frascati Neutron Generator (FNG), ENEA Frascati, and in ASPIS, AEA Technology, Winfrith were analysed [16]. The main purpose of repeating the calculations with the new dosimetry cross sections was to check for any improvement between measured and calculated reaction rates (compared to IRDF-2002 as well as IRDF-90) and removal of some inconsistent trends in the results for different monitors. Since the dosimetry data represent a relatively small part of the overall uncertainty (the major part comes from the transport cross sections and model approximations) these results can be considered as an indirect validation of the new IRDFF dosimetry library. The compensation of errors between the transport cross sections and dosimetry data is likely. To obtain additional information potentially useful to conclude on the impact of transport and dosimetry cross-section uncertainties and their compensation, as well as on the computer code modelling uncertainties, the results using different transport cross-sections and computer codes (DOORS and MCNP) were presented for several benchmark analyses.

ENEA has performed the test of IRDFF via re-analysis of the previous benchmark experiments carried out at the 14 MeV Frascati Neutron Generator (FNG). A new experimental campaign was also performed at FNG to measure the Reaction Rates of several threshold reactions: $^{115}In(n,n')^{115}In$ (E_{thr} = 0.5 MeV), $^{58}Ni(n,p)^{58}Co$ (1 MeV), $^{27}Al(n,a)^{24}Na$ (4.2 MeV), $^{197}Au(n,2n)^{198}Au$ (8.2 MeV), $^{93}Nb(n,2n)^{92}Nb$ (8.9 MeV), $^{90}Zr(n,2n)^{89}Zr$ (12.0 MeV) and $^{58}Ni(n,2n)^{57}Ni$ (12.6 MeV). An overall conclusion is that IRDFF 1.05 shows similar performance as previous versions. Relatively large differences (10 - 15%) are still observed for several reactions such as $^{197}Au(n,\gamma)^{198}Au$ and $^{58}Ni(n,2n)$ (for more details see summary of M. Angelone et al. in the Section IV).

The Fusion Neutronics Source (FNS) facility of JAEA reported (i) the results of comparison between IRDFF and previously measured cross sections and new measurement of cross sections such as $^{nat}Ti(n,x)^{46}Sc$ near 14 MeV; (ii) reaction rate measurement inside the experimental assemblies such as

graphite and Li_2O, with well characterised neutron spectra. The IRDFF excitation functions in most cases agree well with activation cross-sections measured at FNS in the energy interval 13.5 to 15 MeV, though some disagreement was observed for $^{48}Ti(n,x)^{47}Sc$, $^{63}Cu(n,2n)^{62}Cu$, $^{64}Zn(n,p)^{64}Cu$ and $^{113}In(n,n')^{113m}In$. The validation of the 30 - 40 reactions from IRDFF in the graphite and Li_2O benchmarks indicates an agreement within 10% for most of them except $^{181}Ta(n,\gamma)^{182}Ta$, $^{204}Pb(n,n')^{204m}Pb$, $^{48}Ti(n,p)^{48}Sc$, $^{49}Ti(n,x)^{48}Sc$, and $^{51}V(n,\alpha)^{48}Sc$, correspondingly (for more details see summary of C. Konno et al. in the Section IV).

2.5. Measurements of the IRDFF cross sections at high energies

The extension of high energy cut of the dosimetry reaction cross sections from 20 MeV (IRDF-2002) to 60 MeV (IRDFF) required the experimental validations for both excitation function and spectrum average cross sections.

The Nuclear Physics Institute (NPI) at Řež near Prague has measured the (n,xn) reaction cross sections on ^{59}Co, ^{169}Tm, ^{209}Bi, ^{54}Fe and ^{197}Au employing the quasi-monoenergetic p+Li neutron sources at NPI (energy range 20 to 35 MeV) and at TSL in Uppsala (38, 50 and 62 MeV). The cross-sections were extracted from the measured reaction rates with the help of the SAND-II code. In such a way a contribution from the low energy neutrons produced in Li foil and carbon stopper were taken out.

The neutron activation measurements for $^{209}Bi(n,xn)^{203,204,205,206}Bi$ and $^{59}Co(n,xn)^{56,57,58}Co$ were performed using 140 MeV and 80 MeV p-Li quasi-monoenergetic neutron beams at the Research Center for Nuclear Physics (RCNP) of Osaka University. The experimental set-up allows to sweep protons passing through the Li target to the beam dump. Such a source produces less "parasitic" neutrons, i.e. low energy neutrons. Two different irradiation experiments were performed by using two different neutron beams to correct for low energy neutron production. Their contribution was "experimentally" subtracted by measuring the activities of foils located at 0° and 25° relative to the proton beam.

The iThemba LABS has presented the status of the measurements of (n,xn) reactions cross section on ^{59}Co, ^{169}Tm, ^{209}Bi and ^{197}Au. Ongoing work includes the analysis of the gamma-ray spectra, unfolding of the neutron spectra to determine the neutron fluence in the peak and continuum. Subsequently the cross-sections for various reactions will be calculated and reported.

2.6. Evolution of the IRDFF Decay sub-library

Besides the cross sections, the IRDFF library also includes the decay data for the produced unstable reaction residuals and for selected fission isotopes as well as isotopic abundances for target elements. The nuclear data of interest for dosimetry applications are half-lives, decay modes, energy and intensity of radiation used for the detection of the dosimetry reaction products.

The source of the decay information for IRDFF-1.00 (82 isotopes) and previous versions was the Evaluated Nuclear Structure Data Files (ENSDF) [20] (as Dec 2011), which has been converted to the ENDF-6 format for use together with cross sections in the same format by the SDF2NDF code [21].

The decay sub library requires further updating and expansion as a part of this CRP. The current version of IRDFF contains decay data for 90 isotopes and isomers https://www-nds.iaea.org/IRDFFtest/irdffnuclideslist.htm. Decay data for the new isotopes and some already existing in IRDFF (total of 33) were carefully re-analysed and evaluated by V. Chechev. The results of evaluation in the ENSDF and ASCII text formats are available on the CRP web. Many of the results were also submitted to and adopted by the Decay Data Evaluation Project (DDEP) working group [22].

The decay data sub-library of IRDFF is regularly converted from the ENSDF to ENDF-6 format by M. Verpelli (NDS) with a help of the updated code SDF2NDF (after proper modifications when needed). The resultant decay data file and output files from the format checking utilities are made available on the CRP web page.

2.7. Neutron induced fission reactions

The neutron induced fission cross sections for Th-232, U-235, U-238, Np-237, Pu-239, Am-241 are detected by measuring of the radiation emitted from the fission fragments - their decay characteristics were also considered in the IRDFF decay sub-library. For determination of the (n,f) reaction cross sections the fission fragments yield data are needed. Currently we recommend the use of the JEFF-3.1 evaluation for such purpose.

2.8. Competition between neutron- and photo-induced reactions

The (n,2n) and (γ,n) reactions on the initial target nucleus produce the same reaction residual isotope. This means that one should estimate the contribution of the each channel in the case of mixed neutron gamma fields. In this analysis, proper and reliable nuclear data should be used.

The preliminary estimations were done for the ^{238}U(n,2n)^{237}U and ^{238}U(γ,n)^{237}U reactions in the neutron-gamma field produced by thermal neutrons impinging on ^{235}U, see https://www-nds.iaea.org/IRDFFtest/photonuclear.pdf. The cross section for the (n,2n) reaction was taken from IRDFF, for (γ,n) - from the ENDF/B-VII.1 photonuclear library. The prompt neutron and gamma fission spectra for n_{th} + ^{235}U were taken from ENDF/B-VII.1. It was found that ^{238}U(γ,n) reaction produces ≈ 1% of ^{237}U produced by neutron reaction ^{238}U(n,2n).

References to Section II

1. D.H. Abriola and R.A. Forrest (Eds.), Report of the IAEA Nuclear Data Section to the International Nuclear Data Committee for the period January 2010 – December 2011, Report INDC(NDS)-0619, IAEA, April 2012
2. E.M. Zsolnay, R. Capote Noy, H.J. Nolthenius, A. Trkov, Summary description of the new International Reactor Dosimetry and Fusion File (IRDFF release 1.0), Report INDC(NDS)-0616, IAEA, May 2012
3. R. Capote, K.I. Zolotarev, V.G. Pronyaev, and A. Trkov, Updating and Extending the IRDF-2002 Dosimetry Library, Journal of ASTM International, v. 9, issue 4, April 2012, JAI104119
4. International Reactor Dosimetry File 2002 (IRDF-2002), *Tech. Report Ser*. **452**, IAEA, (2006); The numerical data are available on: https://www-nds.iaea.org/irdf2002/
5. A. Trkov, L.R. Greenwood, S.P. Simakov, RCM-1 Summary Report of the IAEA CRP, Report INDC(NDS)-0639, IAEA, 2013
6. K. Zolotarev, Report INDC(NDS)-0657, IAEA, Dec 2013
7. K. Zolotarev, Report INDC(NDS)-0668, IAEA, Oct 2014
8. A.J. Koning and D. Rochman, TENDL-2013: TALYS-based Evaluated Nuclear Data Library, data available on ftp://ftp.nrg.eu/pub/www/talys/tendl2013/tendl2013.html
9. A. Trkov, Report INDC(SEC)-0110, IAEA, 20141
10. PREPRO 2015 Home Page https://www-nds.iaea.org/public/endf/prepro/
11. Program RR_UNC to calculate uncertainties in reaction rates and cross sections: https://www-nds.iaea.org/IRDFF/rr_unc.for
12. NJOY 2012 Nuclear Data Processing System, *http://t2.lanl.gov/nis/codes/NJOY12/index.html*
13. M.B. Chadwick et al, ENDF/B-VII.1 Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data, Nucl. Data Sheets 112 (2012) 2887
14. S. Manojlovič, A. Trkov et al. Capture cross section measurement analysis in the Californium-252 spectrum with the Monte Carlo method", Appl. Rad. and Isotopes 101 (2015) 101
15. S. Simakov, R. Capote, L. Greenwood, P. Griffin, A. Kahler, V. Pronyaev, A. Trkov and K. Zolotarev, "Validation of IRDFF in ^{252}Cf standard and IRDF-2002 reference neutron fields", ISRD-15, May 2014, Aix-en-Provance, EPJ Web of Conferences, 2015, p. ???

16. I. Kodeli, Validation of IRDFF-1.04 (and 1.05) Dosimetry Library Using SINBAD Shielding Benchmark Experiments, Report INDC(SLO)-002, IAEA 2015
17. De Corte, F., Simonits, A., Recommended nuclear data for use in the k0 standardization of neutron activation analysis, At. Data Nuclear Data Tables 85, 47-67, 2003
18. Mughabghab, S.F., Atlas of Neutron Resonances, Elsevier, ISBN-13:978-0-444-52035-7, ISBN-10:0-444-52035-X, 2006
19. Mughabghab, S.F., Thermal Neutron Capture Cross Sections Resonance Integrals and g-factors, IAEA, Vienna, Austria, Technical report INDC(NDS)-440, 2003, http://www-nds.iaea.org/reports-new/indc-reports/indc-nds/indc-nds-0440.pdf
20. Evaluated Nuclear Structure Data File (ENSDF), see http://www.nndc.bnl.gov/ensdf/
21. O. Bersillon, *The SDF2NDF Code*, Commissariat à l'énergie atomique, unpublished.
22. Recommended Data by the Decay Data Evaluation Project working group: http://www.nucleide.org/DDEP_WG/DDEPdata.htm

III. Results of common discussions: research coordination, recommendations and actions.

3.1. Recommendation and Actions for IRDFF

High-priority request lists (HPRL) are already provided on the IRDFF/Test web site. There was agreement that the HPRL lists capture the reactions that need improvement, whether by measurements or evaluations. The list needs to be linked and prioritized to specific applications by connection to the reactor dosimetry and advanced reactor communities, accelerator neutron sources, and the fission and fusion material damage communities. Information on this list needs to be extended to include the value added if the additional measurements are made.

^{117}Sn(n,n')^{117m}Sn (14 day) would be useful to include in HPRL since it is a unique way to measure neutrons in the 300 keV energy range.

Action: NDS (S. Simakov), All

Participants agree that the set of reactions that need to be updated or require new evaluations that are included in the IRDFF database are ^{23}Na(n,γ), ^{23}Na(n,2n), and ^{27}Al(n,2n).

Action – K. Zolotarev

There are several cases where isotopic or natural element (n,x) reaction data would be highly useful. Examples include ^{6}Li(n,x)^{4}He, ^{7}Li(n,x)^{3}H, ^{7}Li(n,x)^{4}He, ^{10}B(n,x)^{4}He, ^{11}B(n,x)^{4}He, Ti(n,x)^{46}Sc, Ti(n,x)^{47}Sc, Ti(n,x)^{48}Sc; Zn(n,x)^{67}Cu, Fe(n,x)^{54}Mn and all cases where there are significant differences between the activation of elemental and isotopically-enriched targets or where there are additional reaction channels at higher neutron energies leading to the same residual isotope.

Partial cross sections for different isotopes need to be summed to create the elemental files.

Action: NDS, L. Greenwood

There is a significant discrepancy on the ^{55}Mn(n,γ) cross section from 10 keV to 1 MeV that needs to be addressed. Experiments at FNG will be useful in this regard. FNS will perform measurements to validate this reaction in the W bulk experiment.

Action: M. Angelone, C. Konno, I. Kodeli

The ^{58}Fe(n,γ) reaction, as well as many other capture reactions, tend to be discrepant in the 10 keV to 1 MeV energy region for fast reactor, shielding, and fusion applications. The FNS experiments will provide data on some of the reactions that were measured.

Action: C. Konno

More measurements and validation are needed in the higher neutron energy range above 20 MeV. Experiments are in progress for Bi and Co. We plan comparisons on more experiments at common neutron energies around 40 MeV for data evaluations.

Action: M. Majerle, H. Yashima, R. Nchodu

We encourage more measurements of some IRDFF reactions in well-characterized Cf-252 neutron fields. Such sources exist or are planned in EPFL, CV (Research Center) Rez, NPL (UK), and Institute of National Standards. In particular measurements are needed on high threshold reactions such as (n,2n) or (n,3n) to reduce uncertainties on the higher energy part of the 252Cf neutron spectrum.

Action: NDS, C. Destouches, A. Plompen

Re-evaluations of previous experiments are needed to resolve some discrepancies for the ^{235}U neutron field at neutron energies above 10 MeV. New measurements are encouraged to assist in improving the database. Potential facilities are BR1 (Mol), FRM2 (Munich), Budapest Research Reactor, NIST, Kyoto, and ILL.

Action: A. Plompen

Los Alamos is encouraged to continue to investigate and report the data and methods, particularly the expected uncertainties, associated with the historic critical assembly activation measurements. Planned new measurements in bare and natural uranium reflected highly-enriched uranium critical assemblies, particularly to obtain IRDFF activations not measured, would be very valuable.

Action: M. White

IRDFF should include some warnings on known problems such as the branching ratio to ^{58g}Co vs. ^{58m}Co for the ^{58}Ni(n,p) reaction and for ^{196g}Au vs. ^{196m}Au for the ^{197}Au(n,2n) reaction. Energy-dependent branching ratios for ^{58}Co will be added to IRDFF in order to support calculations of burnup during reactor irradiation.

Action: NDS

There are disagreements between JEFF and IRDFF evaluations for ^{237}Np(n,fission) and ^{241}Am(n,fission) measurements between LANL and n-TOF on the plateau region, the 2.5 keV to 100 keV energy range, and for ^{237}Np(n,γ) measurements in the 100 keV to 2 MeV energy range that need to be resolved.

Action: V. Pronayev, K. Zolotarev, G. Noguere

There is a 15% discrepancy between the IRDFF and measured spectrum averaged cross section for ^{238}U(n,2n) in the ^{235}U thermal neutron field. The reason could be due to the contribution from the competing reaction (γ,n) which also leads to ^{237}U. A similar problem may exist with (n,fission) and photo-fission reactions for ^{238}U. Additional effort is needed to simulate such experiments and to determine how to properly use current nuclear data. Validation of photonuclear data on SINBAD benchmarks with MCNP will be done.

Action: NDS, I. Kodeli

A verification of the MACS results for ^{238}U(n,γ) and ^{197}Au(n,γ) is recommended. There are new measurements by AMS (Wallner) that have to be considered.

A re-measurement of the ^{55}Mn(n,γ) reaction and comparison with neutron TOF data is needed. The ^{55}Mn(n,γ) measurements should be considered as a test case for this mass range.

Action: P. Mastinu, A. Plompen

In the frame of the CRP, the decay data library is being updated and all participants are strongly encouraged to use this library for consistency and document any differences with prior experiments. New experimental activation data should include full documentation of the experiment (half-life, gamma intensities, counting times, irradiation time, cooling times, reference reactions) and other data that are required for the later re-evaluation of the measurements.

Nuclear decay data problems were identified for the following nuclides:

Rh-103m - x-ray emission probability around 20 keV;

La-140 - gamma intensities for lines below 1596 keV;

W-187 - gamma intensities of 2 lines (473.53 keV and 685.81 keV);

Cu-64 - 511 keV annihilation gamma line intensity tends to be increased essentially.

Action: V. Chechev, All

There was a discussion on whether we should use integral experiments in data evaluations. There was agreement that integral data can't be used for both evaluation and validation at the same time. If evaluations consider integral data, they need to include comments regarding the impact on the evaluation.

The next 3rd meeting is recommended to convene in the 1st Quarter of 2017.

3.2. The new Neutron Spectral Adjustment Code Exercise

There was agreement that the NDS should organize a neutron spectral adjustment exercise similar to REAL-88, as requested in the Spectrum Adjustment Workshop summary at ISRD15.

The new Neutron Spectral Adjustment Exercise will comprise the following **elements:**

- There are several suggestions for datasets that might be used:
  - SINBAD has HB Robinson, VVER1000 and/or VVER400, Venus 3 cases
  - Critical assembly data from LANL for Flattop
  - 14 MeV from FNS graphite case
  - ^{9}Be(p,n) TOF data at 35 MeV
  - ^{9}Be(d,n) TOF data at various energies
  - D_2O(p,n) spectrum at 35 MeV
  - Commercial reactor RPV surveillance
  - Spallation source up to 60 MeV?
- Requirements for each case:
  - Saturated Activities with all corrections and uncertainties and covariances if available
  - *A priori* neutron spectrum (group flux or number fraction) and uncertainty and covariances (when available). When a covariance matrix is provided, the eigenvalues for the covariance matrix should also be included in order to verify that the matrix is positive semi-definite.
  - Neutron self-shielding correction factors in a fine group structure (non-threshold reactions)
  - Method of determination of the neutron spectrum and covariances
- Timing: Input data sets sent to NDS (S. Simakov) by 20 May 2015
- Cases: Downselect to 6 to 8 cases
- Reporting Requirements (with 1-sigma uncertainties):
  - Total fluence
  - Thermal fluence < 0.5 eV
  - Epithermal fluence between 0.5 eV and 0.1 MeV
  - Fluence > 0.1 MeV
  - Fluence > 1 MeV
  - For higher energies, request more fluence ranges > 20 MeV
  - Spectrum-averaged cross section (case dependent set of reactions)
  - dpa (need higher energy part)
  - Adjusted neutron spectrum and uncertainties (lethargy)
  - Metric for consistency of adjustment (χ^2 for a least-squares or relevant quality of fit metric for other spectrum adjustment approaches)
  - Adjusted reaction rates and uncertainties
- Requirements:
  - Strongly recommend use of IRDFF cross sections and covariances
  - Computer code (and type) that was used for the adjustment
  - Note on how data were processed and any problems or changes made to the data set
  - Cross section library and origin (if not IRDFF).

- Potentially applicable computer codes:
  STAYSL PNNL
  STAY'SL
  LSL
  MAXED
  CALMAR
  MS ITER
  GRUPINT
  FERRET
  NSVA
  GRAVEL/SANDII
  NEUPAC-JLOG
  MIEKE
  STAYNL
  NMF
- Steps for Conducting the Exercise:
  Collection of datasets (Summer 2015)
  Down selection of data for exercise
  Preparation of dataset to be issued (digital format)
  Small group analyses the data for consistency
  The NDS issues a letter soliciting participation in the exercise (Autumn 2015)
  Submission of results (June 2016) (anonymous spectra and participants)
  Analysis and final reporting of the exercise (April 2017)

IV. Participants' Summaries

Advanced UQ Approaches to the Validation of Dosimetry Cross Sections in Reactor Benchmark Fields: Input to 2nd RCM,

P. Griffin

Sandia National Laboratories[1], Radiation and Electrical Sciences, Albuquerque, NM, USA

Abstract. This report summaries the progress made by Sandia National Laboratories in its support of the of the International Atomic Energy Agency (IAEA) Nuclear Data Section (NDS) Cooperative Research Project (CRP) on the testing and improving of the International Reactor Dosimetry and Fusion File (IRDFF).

1 Planned Activities

This is the report from Sandia National Laboratories (SNL), presented at the 2nd Research Coordination Meeting (RCM) on Testing and Improving the IAEA International Reactor Dosimetry and Fusion File (IRDFF) Cooperative Research Project (CRP), on our first year's progress in this project. The 1st year plan submitted by SNL called for the following activities:

a) Survey set of UQ approaches applicable for an analysis of the consistency of integral cross section measurements in reactor benchmark fields
b) Apply UQ approaches to available data for ^{252}Cf spontaneous fission standard benchmark field and ^{235}U thermal fission reference benchmark fields
c) Determine what parameter dominates the uncertainty in the UQ metrics and address ways to reduce the uncertainty

The following sections of this report summarize actions taken in support of these areas of investigation.

2 Survey of Dosimetry Metrics

The IRDFF dosimetry cross section library [1] includes more reactions than the previous IRDF-2002 library and extends the applicable energy range for incident neutrons up to 60 MeV. As this library is released to the public it is important that the scope and extent of the validation evidence for these cross sections be documented and made available to the user community. When doing validation, it is important to note that any validation data must be accompanied by a quantitative statement on the uncertainty in the measured and in the calculated quantity. Since differential cross section data, e.g. monoenergetic neutron cross section measurements, were considered in the preparation of the evaluated cross sections, we look for validation, primarily, to comparisons with integral measurements obtained in benchmark neutron fields. The purpose of this paper is to explore various validation metrics and to compare the strength of the validation evidence that they can provide when applied to existing data in neutron benchmark fields.

Various metrics can be used to validate the underlying data. Each has its advantages and disadvantages. The relevant input data is a combination of energy-dependent reaction cross sections, activity measurements, benchmark field neutron source spectrum, and nuclear decay data (half-life, gamma emission rates). One must take into consideration correlations in the input data and properly propagate all uncertainty contributions. A statistically meaningful metric must be used to assess the level/fidelity of the validation evidence.

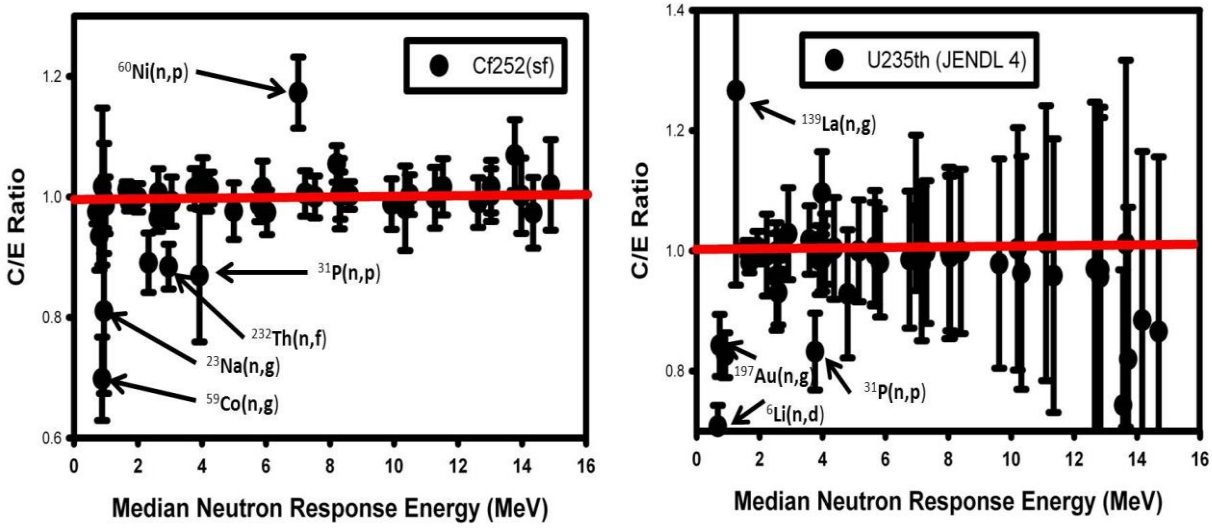
Commonly used validation metrics include the use of a calculated-to-experimental (C/E) ratio for spectrum-averaged cross section or a spectral index and a constrained least squares fit to activation

[1] *Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy under contract DE-AC04-94AL85000.*

data while reporting a chi-squared per degree of freedom (dof) for each measured activity. A discussion of the pros and cons of these various validation metrics has been captured in reference [2]:

Figures 1, 2 and 3 show the results of applying these metrics to the ^{252}Cf(sf) spontaneous fission standard neutron benchmark field and ^{235}U(th) thermal fission reference neutron benchmark fields. Figure 1a shows excellent agreement when using the C/E for the spectrum-averaged cross section metric in the ^{252}Cf(sf) field. This is because the ^{252}Cf reference neutron field is very well characterized neutron field based upon time-of-flight measurements and documented in Reference [3]. Figure 1b shows much larger uncertainty bounds when one applies this metric to the ^{235}U(th) reference neutron benchmark field. This is because there is a much larger uncertainty in the spectral characterization of this field – with use of either the ENDF/B-VII [4] or JENDL-4 [5] fission neutron spectrum representation. When the metric of the calculation-to-experiment is applied to the spectral indices in the ^{252}Cf(sf) and ^{235}U(th) fields, Figures 2 and 3 compare the original NBS analysis [6] to the results when version 1.03 of the IRDFF library is used [7]. The agreement of the integral metric with the IRDFF-calculated metric is seen to be excellent – for the limited available set of validation data using spectral indices. The cause of the deviation in the metric for the ^{47}Ti(n,p) reaction that is seen in the original NBS work was identified by Mannhart [8] and addressed in the latest cross section evaluations that are contained in the IRDFF library.

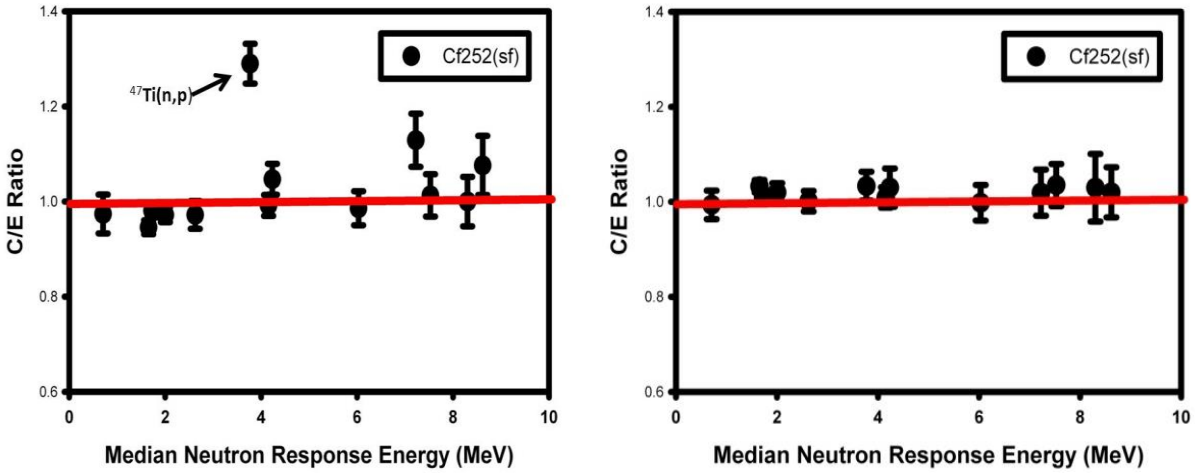
Figure 1: Metric: Spectrum-averaged Cross Section



a) ^{252}Cf(sf) Standard Benchmark Field

b) ^{235}U Reference Benchmark Field

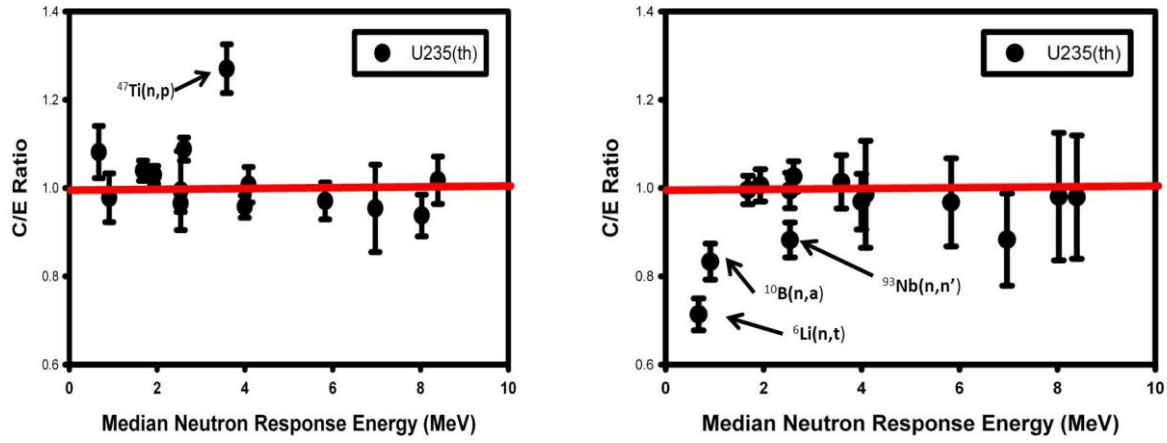
Figure 2: Metric: Spectral Index in ^{252}Cf(sf) Standard Neutron Benchmark Field



a) Original NBS Analysis

b) Updated Analysis with IRDFF Cross Sections

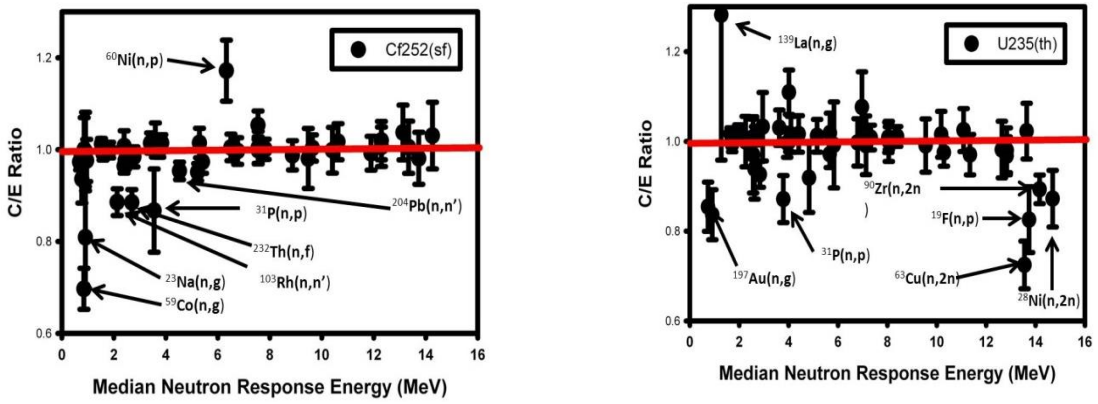
Figure 3: Metric: Spectral Index in ^{235}U(th) Reference Neutron Benchmark Field



a) Original NBS Analysis b) Updated Analysis with IRDFF Cross Sections

Figure 4 shows the results when a least squares analysis is applied to the available data in the ^{252}Cf(sf) and ^{235}U(th) neutron benchmark fields. The value in this metric is that the C/E ratios derived from the least squares analysis take into consideration the uncertainty due to knowledge of the neutron spectrum and that due to the activity measurement as well as examining the uncertainty due to knowledge of the cross section. A metric for the consistency of the a priori uncertainty input to the least squares analysis is given in the chi-squared per degree of freedom (χ^2/dof). The χ^2/dof for the complete set of ^{252}Cf(sf) data is 2.06. While this is an acceptable metric, a metric of 1.0 would indicate a self-consistent set of input data and the assigned uncertainties. With the elimination of three data points (Co592, Mo92p, Th252f), basically designating these measurements as being discrepant, the χ^2/dof is lowered to a more acceptable value of 1.28. The χ^2/dof for the complete set of ^{225}U(th) data is 2.32. Again, while this is an acceptable metric, the elimination of seven data points (B10a, Li6a, F192, P31p, La139gg, Zn64p, Cu632), lowers the χ^2/dof to a more acceptable value of 1.05. The issues with the B10a and Li6a measurements has been identified by Simakov in Reference 9 as being attributed to the failure of the current IRDFF cross section to include all of the reaction channels that produce alpha particles in the outgoing channel. If the other reaction channels are taken into account, this discrepancy is eliminated. The reason for the discrepancy in the other reactions remains to be identified – and motivates the community to seek additional measurements for the other five reactions in this neutron field.

Figure 4: Metric: Constrained Least Squares in Neutron Benchmark Field



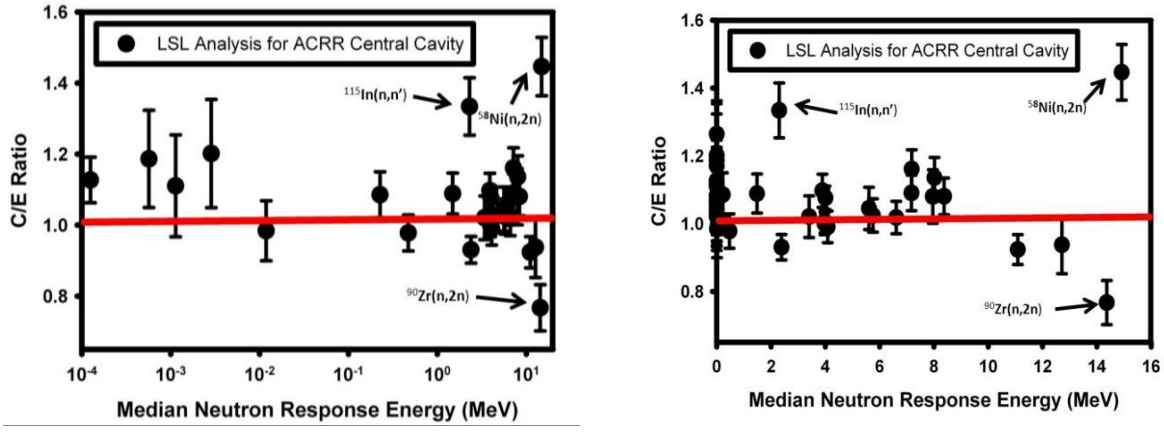
a) ^{252}Cf(sf) Standard Benchmark Field b) ^{235}U Reference Benchmark Field

3 Application of the Least Squares Dosimetry Metrics to Other Neutron Benchmark Fields

In addition to the ^{252}Cf(sf) and ^{235}U(th) neutron benchmark fields, there are other well-characterized neutron benchmark fields that can contribute to the validation of the dosimetry cross sections. The central cavities of the Annular Core Research Reactor (ACRR), a pool-type reactor, and of the Sandia Pulsed Reactor (SPR-III), a fast burst reactor, are two such reference benchmark fields used in radiation damage studies.

Figure 5a shows the C/E metric for a least squares analysis in the ACRR central cavity in a logarithmic and energy representation. This analysis included 40 measured activities with different reactions or different covers [10] on the dosimetry foil. It yielded a χ^2/dof of 2.06. In 62% of these reactions, the spectrum is the dominant contributor to the overall uncertainty in the C/E metric. This shows the importance of a balanced treatment of the uncertainty in the cross section and in the spectrum when one seeks validation evidence. In some cases a given reaction is consistent with one cover and inconsistent with a different cover e.g. ^{55}Mn(n,γ) and ^{56}Fe(n,p) reaction. The issue here may reside with the uncertainty in different portions of the cross section, or with the fidelity of the cover correction. The cover correction methodology is documented in Reference 9. The ^{58}Ni(n,2n) reaction is seen to be discrepant and is in conflict with three other reactions with response in the similar energy region. This reaction is a clear candidate for confirmatory measurements.

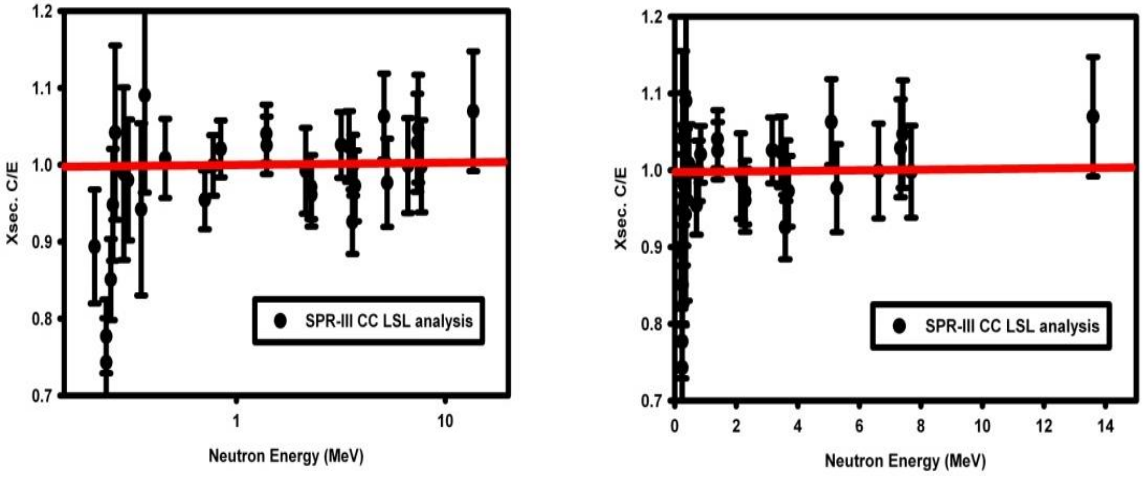
Figure 5: Metric: Constrained Least Squares in ACRR Reference Neutron Benchmark Field



a) Logarithmic Energy Representation

b) Linear Energy Representation

Figure 6: Metric: Constrained Least Squares in SPR-III Reference Neutron Benchmark Field



a) Logarithmic Energy Representation

b) Linear Energy Representation

Figure 6a shows the C/E metric for a least squares analysis in the SPR-III central cavity in a logarithmic and energy representation. This analysis included 31 measured activities with different reactions or different covers on the dosimetry foil. It yielded a χ^2/dof of 2.19. Since this fast burst reactor has been decommissioned, it is not feasible to gather additional or confirmatory measurements in this benchmark field.

4 Results of Least Squares Analysis as the Validation Metric

Table 1 shows the validation evidence available for all of the reactions in version1.05 of the IRDFF library when one applies the least squares methodology to the set of four benchmark neutron field discussed in the previous sections. A green shading indicates a good validation, i.e. a C/E within 1 standard deviation of unity for the spectrum-averaged cross section when one includes the uncertainty due to the neutron spectrum, dosimetry cross section, and activity measurement. A yellow shading indicated agreement within two standard deviations. A red shading indicates agreement outside of three standard deviations – a clear case of discrepant data that should be considered for rejection. The blue shading in the reaction number column indicates a case where a reaction has been suggested for addition to the library contents. The key to the color shading is indicated in the bottom of the table. The dosimetry reaction, as well as the cover and self-shielding correction used in the analysis, are indicated in the table. The metric presented is the C/E for the spectrum-averaged cross section and the uncertainty reflects consideration of the uncertainty contribution from the cross section, neutron spectrum, and activity measurement. The energy-dependent correlation in the spectrum and cross section are addressed in the analysis.

The analysis shows that the highest fidelity ^{252}Cf(sf) standard benchmark field uses 49 reactions and provided validation evidence for 41 different dosimetry reactions. The ^{235}U(th) reference neutron field includes data for 47 reactions and has validation evidence for 42 of the reactions. 9 of these 42 reactions are for reactions that had not been validated in the better characterized ^{252}Cf(sf) field. The ACRR central cavity has data for 41 different reactions and associated covers combinations and provides validation evidence for 31 of these reactions. 5 of these reactions are for reactions that had not been validated in either the ^{252}Cf(sf) or ^{235}U(th) fields. The SPR-III central cavity has data for 31 different reactions and covers combinations and provides validation evidence for 27 of these reactions. 3 of these reactions are for reactions that had not been validated in either the ^{252}Cf(sf), ^{235}U(th), or ACRR central cavity neutron fields.

Table 1: Status of Reaction Validation Evidence

#	Reaction ID	Cover	C/E in Neutron Benchmark Fields			
			^{252}Cf-SF (standard)	^{235}U-TF (reference)	SPR-III CC (reference)	ACRR CC (reference)
1	Li6t		---			
---	Li6(n,nd:2np)α		---			
---	Li6(n,X)He4			~1.03 requires xsec		
2	B10a		---			
---	B10(n,t)2α		---			
---	B10(n,X)He4			~1.0 requires xsec		
3	F192	void-bare	1.0183 +/- 6.48%	0.8257 +/- 8.87%		
4	Na232					
5	Na23g	void-bare pelt-bare pelt-cdna pelt-fiss	0.8216 +- 8.78%		1.0418 +/- 10.1% 0.9884 +/- 10.5% 1.0904 +/- 10.8%	1.1703 +/- 6.50% 1.0418 +/- 7.13% 1.2016 +/- 12.7%
6	Mg24p	void-bare mil5-cdnm	1.061 +/- 3.48%	1.007 +/- 2.54%	1.0467 +/- 6.20%	1.1358 +/- 5.23%
7	Al27p	void-bare ml3x-cdnm	0.9533 +/- 2.27%	0.9715 +/- 3.07%	1.0628 +/- 4.87%	
8	Al27a	void-bare ml3x-cdnm	1.009 +/- 2.87%	1.0110 +/- 2.18%	0.9981 +/- 5.57%	1.0805 +/- 5.00%

new	Si28p					
new	Si29d					
---	Sinat-dspk		---			
---	1-MeV(Si)					
9	P31p	void-bare	0.8670 +/- 11.2%	0.8716 +/- 6.01%		
10	S32p	void-bare sulf-bare	1.021 +/- 4.68%	0.9969 +/- 3.25%	1.0238 +/- 4.16%	1.0971 +/- 4.47%
11	Sc45g	void-bare mil5-cdnm mil5-fiss			0.8937 +/- 7.67% 0.9479 +/- 7.10%	1.0314 +/- 9.29% 1.0691 +/- 13.9%
12	Ti462	void-bare	2.0778 +/- 34.4%			
13	Ti46p		---			
14	Ti47np		---			
---	Tinat(n,X)Sc46	void-bare milx-cdnm	0.9753 +/- 4.08%	0.9921 +/- 9.66%	0.9766 +/- 5.44%	1.0241 +/- 4.80%
15	Ti47p		---			
16	Ti48np		---			
---	Tinat(n,X)Sc47	void-bare milx-cdnm	1.0139 +/- 3.69%	1.0304 +/- 3.83%	1.0258 +/- 3.87%	1.0206 +/- 6.02%
17	Ti48p		---			
18	Ti49np		---			
---	Tinat(n,X)Sc48	void-bare milx-cdnm	1.0114 +/- 6.09%	1.0088 +/- 3.33%	1.0286 +/- 5.74%	1.0801 +/- 7.32%
19	V51a	void-bare	0.9982 +/- 4.54%	0.9908 +/- 5.99%		
20	Cr522					
21	Mn55g	void-bare wcu2-cdnm wcu2-fiss			0.7430 +/- 7.17% 0.9803 +/- 7.43%	1.2641 +/- 7.78%
22	Mn552	void-bare	1.0280 +/- 4.70%	0.9802 +/- 5.09%		
23	Fe542					
24	Fe54p	void-bare mil5-cdnm	1.0144 +/- 3.13%	1.0156 +/- 3.94%	0.9726 +/- 4.38%	0.9905 +/- 4.72%
25	Fe54a			0.9913 +/- 6.60%		
26	Fe56p	void-bare mil5-cdnm mil5-fiss	1.0016 +/- 3.89%	1.0086 +/- 2.73%	0.9989 +/- 5.72%	1.0907 +/- 4.84% 1.1609 +/- 4.91%
27	Fe58g	void-bare void-cdnm				1.0576 +/- 10.6%
---	Fenat-dpa					
28	Co592	void-bare void-cdnm	1.0147 +/- 4.69%	0.9680 +/- 4.36%		0.9375 +/- 9.06%
29	Co593					
30	Co59g	void-bare void-cdnm	0.7461 +/- 5.18%			1.0429 +/- 5.98% 1.1272 +/- 5.70%
31	Co59p	void-bare void-cdnm	1.0169 +/- 4.74%	1.0186 +/- 4.34%		1.0450 +/- 5.99%
32	Co59a	void-bare	1.0064 +/- 4.29%	1.0000 +/- 4.66%		
33	Ni582	void-bare void-cdnm	1.0512 +/- 7.72%	0.8725 +/- 7.23%		1.4465 +/- 5.69%
34	*Ni58p	void-bare mil5-cdnm	1.0000 +/- 2.88%	1.0000 +/- 2.29%	0.9259 +/- 4.20%	1.0000 +/- 3.25% 1.0006 +/- 3.26%
35	Ni60p	void-bare void-cdnm	1.1762 +/- 6.14%	0.9978 +/- 5.28%		1.0180 +/- 4.70%
36	Cu632	void-bare	1.0503 +/- 6.13%	0.7249 +/- 7.32%		
37	Cu63g	void-bare mil5-cdnm	0.9995 +/- 6.52%		0.9420 +/- 11.0%	1.0249 +/- 7.36% 1.1865 +/- 11.6%
38	Cu63a	void-bare	1.0088 +/- 4.16%	1.0765 +/- 7.28%		
39	Cu652	void-bare	1.0003 +/- 4.47%			
40	Zn64p	void-bare milx-cdnm	1.008 +/- 3.39%	1.1095 +/- 4.45%	0.9994 +/- 3.66%	1.0761 +/- 3.18%
41	Zn67p			1.0166 +/- 4.00%		
42	As752			0.9818 +/- 6.43%		
---	GaAs-dspk		---			
---	1-MeV(GaAs)					

43	Y892			1.0228 +/- 6.07%		
44	Zr902	void-bare mil5-cdnm	0.9933 +/- 6.12%	0.8933 +/- 3.57%	1.0697 +/- 6.72%	0.7672 +/- 8.50%
45	Mo92p	void-bare	0.5168 +/- 4.73%	1.0125 +/- 3.59%		
46	Nb932	void-bare void-cdnm	1.0057 +/- 5.33%	1.0252 +/- 4.67%		0.9236 +/- 4.77%
47	Nb93n	void-bare	1.0057 +/- 4.31%	0.9700 +/- 8.75%		
48	Nb93g	void-bare void-cdnm				1.1105 +/- 12.9%
49	Rh103n	void-bare	1.1581 +/- 11.7%	0.9995 +/- 5.62%		
50	Ag109g					
	In113g	void-bare	0.9753 +/- 2.89%	0.9390 +/- 7.22%		
	In113gg					
new	In113gm					
51	In113n					
52	In1152m					
53	In115n	void-bare mil5-bare	0.9622 +/- 2.79%	1.0094 +/- 3.99%	0.9922 +/- 5.21%	1.3342 +/- 6.07%
54	In115g	void-bare mil5-cdnm	0.9779 +/- 5.81%		1.0083 +/- 4.73%	1.0207 +/- 8.68% 1.1166 +/- 9.73%
55	I1272	void-bare	1.0244 +/- 4.99%	0.9704 +/- 4.64%		
56	La139g			1.2822 +/- 25.3%		
57	Pr1412					
58	Tm1692	void-bare	0.9907 +/- 7.23%	1.0157 +/- 5.04%		
59	Tm1693					
60	Ta181g	void-bare	0.9489 +/- 4.22%			
61	W186g					
62	Au1972	void-bare	1.0137 +/- 3.72%	0.9753 +/- 3.15%		
63	Au197g	void-bare dil5-bare dil5-cdnm dil5-fiss	0.9844 +/- 3.15%	0.8545 +/- 6.41%	0.7769 +/- 5.72% 0.8507 +/- 5.75%	1.0588 +/- 7.26% 1.0835 +/- 7.57% 0.9841 +/- 8.59%
64	Hg199n	void-bare	0.9831 +/- 4.39%	1.0328 +/- 7.38%		
65	Pb204n	void-bare	0.9548 +/- 5.01%	0.9189 +/- 8.41%		
66	Bi2093					
67	Th232f	void-bare	0.8840 +/- 3.49%	0.9266 +/- 3.12%		
68	Th232g	void-bare	1.0025 +/- 5.40%			
69	U235f	void-bare void-cdtk void-fiss	1.0108 +/- 2.14%	1.0170 +/- 1.30%	0.9544 +/- 3.72% 0.9991 +/- 3.66%	1.0854 +/- 5.96%
70	U235g					
new	U2382					
71	U238f	void-bare void-cdnm void-fiss	0.9758 +/- 2.47%	1.0125 +/- 2.04%	0.9710 +/- 3.97% 0.9609 +/- 3.97%	0.9305 +/- 4.03%
72	U238g					
73	Np237f	void-bare void-cdtk void-fiss	0.9972 +/- 2.90%	1.0181 +/- 1.84%	1.0403 +/- 3.37% 1.0252 +/- 3.37%	1.0889 +/- 5.29%
74	Pu239f	void-bare void-cdtk void-fiss	0.9999 +/- 2.26%	0.9932 +/- 1.65%	1.0206 +/- 3.36%	0.9780 +/- 5.18%
75	Am241f					

*Ni58(n,p) is normalizing reaction in least squares analysis

75+6 = 81 reactions;

Plus 9 suggested new reactions – combinations/conversions; 2 from current responses.

C/E Validation Status:

Within ~1 std; within ~2 std.; outside ~3 std

Overall Reaction Status:

Well validated; Acceptable evidence.; Outstanding significant discrepancies validation required; N/A

Cover nomenclature defined in IEEE TNS Vol. 42, pp. 1878-1885, Dec. 1995, DOI:10.1109/23.489230

5 Library Status

The above analysis shows that application of the least squares approach to supplying validation evidence provides validation evidence for 58 of the 77 reactions in version 1.03 of the IRDFF library. The latest version of the IRDFF library, version 1.05, consists of 75 numbered reactions and 6 new reactions (Si28p, Si29d, In113gm, In113g, In113gg, U2352), for a total of 81 reactions. Table 2 shows the suggested modifications to the library content that were suggest by this work. Seven new reactions or reaction combinations have been suggested for inclusion to the library: ^{6}Li(n,nd:2np)α, ^{6}Li(n,X)^{4}He, ^{10}B(n,t)2α, ^{10}B(n,X)^{4}He, ^{nat}Ti(n,X)^{46}Sc, ^{nat}Ti(n,X)^{47}Sc, ^{nat}Ti(n,X)^{48}Sc. These additions address reaction channels that contribute/interfere with other library contents. We also suggest that two new responses be added as dosimeters based on their use in ASTM standards, 1-MeV(Si) and 1-MeV(GaAs). In addition, the response function corresponding to the ^{nat}Si displacement kerma should be updated to reflect the latest recommended kerma found in ASTM standard E722.

Table 2: Suggested Modifications to IRDFF Library

#	Reaction	Recommended Action
---	Li6(n,nd:2np)α	Add new reaction used in composite dosimeter response
---	Li6(n,X)He4	Add composite reaction to support use of direct measured dosimeter value
---	B10(n,t)2α	Add new reaction used in composite dosimeter response
---	B10(n,X)He4	Add composite reaction to support use of direct measured dosimeter value
---	Sinat-dspk	Update to latest ASTM E722 recommended values.
---	1-MeV(Si)	Add 1-MeV(Si)-Eqv. metric for silicon transistor gain degradation based on ASTM E722 and E1855. Add covariance matrix. This is being addressed.
---	Tinat(n,X)Sc46	Add composite reaction to support use of direct measured dosimeter value
---	Tinat(n,X)Sc47	Add composite reaction to support use of direct measured dosimeter value
---	Tinat(n,X)Sc48	Add composite reaction to support use of direct measured dosimeter value
---	1-MeV(GaAs)	Add 1-MeV(GaAs)-Eqv. metric for silicon transistor gain degradation based on ASTM E722 and E1855. Add covariance matrix. This is being addressed.

Table 3 summarizes the set of reactions for which discrepant data existed across the set of neutron benchmark environments used for this validation activity. A discrepancy in any one of the benchmark neutron fields was sufficient to place that reaction on this list of discrepant reactions where additional analysis or measurements are desired.

Table 3 shows that there are five reactions/measurements in the ACRR central cavity benchmark field where additional analysis is desired. Work on these reactions will be given priority in the next year's efforts conducted under this CRP. The table shows one reaction that needs additional analysis/measurements in the SPR-III benchmark neutron field. Since the SPR-III reactor has been decommissioned, it is not clear what steps can be taken to improve this data point. Analysis of previous spectral characterization efforts at the SPR-III reactor will be analysed to see if supplemental data can be found. Five reactions in the ^{252}Cf(sf) and three in the ^{235}U(th) field are seen to be desired.

Table 3: Discrepant Reactions Where Analysis or Re-measurement is Desired

#	Reaction ID	Foil/Cover Descriptor	C/E in Neutron Benchmark Fields			
			^{252}Cf-SF (standard)	^{235}U-TF (reference)	SPR-III CC (reference)	ACRR CC (reference)
12	Ti462	void-bare	2.0778 +/- 34.4%			
21	Mn55g	void-bare wcu2-cdnm wcu2-fiss			0.743+/-7.17% 0.980+/-7.43%	1.2641 +/- 7.78%

30	Co59g	void-bare void-cdnm	0.7461 +/- 5.18%			1.0429 +/- 5.98% 1.1272 +/- 5.70%
33	Ni582	void-bare void-cdnm	1.0512 +/- 7.72%	0.8725 +/- 7.23%		1.4465 +/- 5.69%
44	Zr902	void-bare mil5-cdnm	0.9933 +/- 6.12%	0.8933 +/- 3.57%	1.070+/- 6.72%	0.7672 +/- 8.50%
45	Mo92p	void-bare	0.5168 +/- 4.73%	1.0125 +/- 3.59%		
53	In115n	void-bare mil5-bare	0.9622 +/- 2.79%	1.0094 +/- 3.99%	0.992+/- 5.21%	1.3342 +/- 6.07%
63	Au197g	void-bare dil5-bare dil5-cdnm dil5-fiss	0.9844 +/- 3.15%	0.8545 +/- 6.41%	0.777+/- 5.72% 0.851+/- 5.75%	1.0588 +/- 7.26% 1.0835 +/- 7.57% 0.9841 +/- 8.59%
67	Th232f	void-bare	0.8840 +/- 3.49%	0.9266 +/- 3.12%		

The foil/cover identifier is described in Reference [10]. The foil/cover descriptor is composed of two portions: the first portion indicates the foil configuration/thickness which can affect self-shielding corrections; the second portion indicates use of a cover material that can be used to shift the response of the dosimeter to higher energies. The "void-bare" designator indicates use of a thin foil not requiring a self-shielding correct and that no foil cover was used.

Of particular interest is the ability to which additional activity measurements at Sandia reactor facilities can be used to support a more comprehensive validation of the IRDFF library. Table 4 captures some of the most critical reactions in the IRDFF library that are in need of validation evidence. In this table we capture the threshold energy for the reaction and the characteristics of the decay signature that would be likely used to perform the activity measurement. The column of data that gives the expected spectrum-averaged cross section in the SNL ACRR reactor environment is not yet completed. These spectrum-averaged cross sections will be evaluated in the next phase of our work. The Table 4 column labelled as "relevance" is color-coded to indicate the expected relevance of the ACRR Central Cavity environment for providing additional validation evidence for these critical reactions. An inspection of the table shows that most of the reactions of most interest either have too high a reaction threshold energy or too short of a half-life to expect that measurements in this reactor environment will be useful in supporting validation evidence.

Table 4: Relevance of ACRR Pool-type Reactor for Reactions Requiring Validation Evidence

#	Reaction ID	Relevance	Metric					
			Half-life	Threshold Energy (MeV)	ACRR CC Spect-Avg Cross Section (mb)	Decay Particle	Emission Energy (keV)	Emission Probability (%)
4	Na232	Energy		13.5				
new	Si28p	Half-life C/E data to be added to analysis	2.245 m	4.35		β^- γ	2863.27 EP 1778.987	99.990 100.
new	Si29d	Energy / Half-life	2.245 m	11.4		β^- γ	2863.27 EP 1778.987	99.990 100.
12	Ti462	Energy	184.8 m	13.55		β^+	438.93	84.80

20	Cr522	Energy	27.7010 d	12.4		γ	320.0824	9.910
29	Co593	Energy	271.74 d	19.6		γ	122.06065	85.60
50	Ag109g	Half-life	24.56 s	0.0		γ	657.50	4.50
	In113gg	Half-life	43.1 ms	0.0		γ	311.652	89.85
new	In113gm	Half-life	43.1 ms	0.0		γ	311.652	89.85
51	In113n	C/E data to be added to analysis	99.476 m	0.45		γ	391.698	64.94
52	In1152m	Half-life /Yield	43.1 ms 71.9 s 49.51 d	9.35		γ γ γ	190.34 558.43 725.24	15.56 0.03 4.4
57	Pr1412	Half-life	3.39 m	9.5		γ	1596.1 511.0	0.49 102.
59	Tm1693	Energy	93.1 d	??		γ	447.515 815.989	23.98 50.95
61	W186g		24 h	0.0		γ	685.81	33.2
66	Bi2093	Energy / Half-life	182 μs	14.6		γ	669.5	64.
70	U235g	Half-life	2.342E7 y	0.0		α	4494	74
72	U238g	S/N	23.45 m	0.0		γ	74.664	53.2
new	U2382		6.75 d	6.25		γ	208.005	21.2
75	Am241f		Various FP	0.0		---	---	---

While Table 4 examined the list of reactions where validation was lacking in any of the available neutron benchmark fields and examined the role that SNL facilities may play in addressing this deficiency, Table 5 shows a list of reactions where some high quality validation evidence is available in other neuron benchmark fields, but work can be done in the ACRR central cavity reference benchmark that could supplement the inventory of validation evidence.

Table 5: Other Redundant Reactions where Validation Data should be Gathered at the ACRR Reactor

#	Reaction ID	Status	C/E in Neutron Benchmark Fields			
			^{252}Cf-SF (standard)	^{235}U-TF (reference)	SPR-III CC (reference)	ACRR CC (reference)
---	Li6(n,X)He4			~1.03 requires xsec		
---	B10(n,X)He4			~1.0 requires xsec		
3	F192		1.0183 +/- 6.48%	0.8257 +/- 8.87%		
6	Mg24p		1.061 +/- 3.48%	1.007 +/- 2.54%	1.0467 +/- 6.20%	1.1358 +/- 5.23%
7	Al27p		0.9533 +/- 2.27%	0.9715 +/- 3.07%	1.0628 +/- 4.87%	
9	P31p		0.8670 +/- 11.2%	0.8716 +/- 6.01%		
10	S32p		1.021 +/- 4.68%	0.9969 +/- 3.25%	1.0238 +/- 4.16%	1.0971 +/- 4.47%
19	V51a		0.9982 +/- 4.54%	0.9908 +/- 5.99%		
22	Mn552		1.0280 +/- 4.70%	0.9802 +/- 5.09%		
23	Fe542					
25	Fe54a			0.9913 +/- 6.60%		

26	Fe56p		1.0016 +/- 3.89%	1.0086 +/- 2.73%	0.9989 +/- 5.72%	1.0907 +/- 4.84% 1.1609 +/- 4.91%
32	Co59a		1.0064 +/- 4.29%	1.0000 +/- 4.66%		
36	Cu632		1.0503 +/- 6.13%	0.7249 +/- 7.32%		
38	Cu63a		1.0088 +/- 4.16%	1.0765 +/- 7.28%		
39	Cu652		1.0003 +/- 4.47%			
41	Zn67p			1.0166 +/- 4.00%		
42	As752			0.9818 +/- 6.43%		
43	Y892			1.0228 +/- 6.07%		
45	Mo92p		0.5168 +/- 4.73%	1.0125 +/- 3.59%		
47	Nb93n		1.0057 +/- 4.31%	0.9700 +/- 8.75%		
49	Rh103n		1.1581 +/- 11.7%	0.9995 +/- 5.62%		
	In113g		0.9753 +/- 2.89%	0.9390 +/- 7.22%		
55	I1272		1.0244 +/- 4.99%	0.9704 +/- 4.64%		
56	La139g			1.2822 +/- 25.3%		
58	Tm1692		0.9907 +/- 7.23%	1.0157 +/- 5.04%		
60	Ta181g		0.9489 +/- 4.22%			
62	Au1972		1.0137 +/- 3.72%	0.9753 +/- 3.15%		
64	Hg199n		0.9831 +/- 4.39%	1.0328 +/- 7.38%		
65	Pb204n		0.9548 +/- 5.01%	0.9189 +/- 8.41%		
67	Th232f		0.8840 +/- 3.49%	0.9266 +/- 3.12%		
68	Th232g		1.0025 +/- 5.40%			
C/E Validation Status: Within ~1 std; within ~2 std.; outside ~3 std Overall Reaction Status: Well validated; Acceptable evidence.; Outstanding significant discrepancies validation required; N/A						

7 Conclusions and recommendations

The usefulness of several validation metrics has been addressed and the value of the metrics demonstrated by applying them to the ^{235}U thermal fission, ^{252}Cf spontaneous fission, and to various reactor reference fields for which there exists a good database of measured dosimetry cross sections. A coupled least squared analysis that includes the cross section along with the spectral representation and the measurements appears to be the most useful validation metric. The validation evidence using this least squares C/E metric has been summarized and reactions have been identified that are still in need of validation evidence. Follow-on steps that can be conducted at the Sandia National laboratories ACRR reactor have been identified for action in the next year of this CRP.

References

1. International Reactor Dosimetry and Fusion File, IRDFF v.1.05, October 9, 2014, see https://www-nds.iaea.org/IRDFF/
2. P.J. Griffin, "Use of Neutron Benchmark Fields for the Validation of Dosimetry Cross Sections," European Physical Journal, 2015, presented at ISRD15
3. Cf-252 Neutron Spectrum, Ed. H.D. Lemmel, IAEA report, IAEA-NDS-98, Dec. 1987 (includes W. Mannhart reports).
4. ENDF/B-VII
5. K. Shibata, et al., J. Katakura:"JENDL-4.0: A New Library for Nuclear Science and Engineering," J. Nucl. Sci. Technol., 48(1), 1-30 (2011).
6. J.A. Grundl, C.M. Eisenhauer, NBSIR 85-03151, *Compendium of Benchmark Neutron Fields for Reactor Dosimetry*, US Department of Commerce, National Bureau of Standards, January 1986.
7. P.J. Griffin, Covariance Propagation in Spectral Indices, presentation at the International Workshop on Nuclear Data Covariances, April 28 – May 1, 2014 in Santa Fe, New Mexico. Submitted as program proceedings for publication in Nuclear Data Sheets, January 2015.

8. W. Mannhart, D.L. Smith, J.W. Meadows, “The Discrepancy Between Differential and Integral Data on the Ti-47(n,p)”, in American Nuclear Society Proceedings on a Topical Meeting in Advances in Nuclear Engineering Computation and Radiation Shielding, Santa Fe, New Mexico, April 9-13, 1989, University of New Mexico Publications, Ed. M.L. Hall, pp. 577-580.
9. S. Simakov, R. Capote, L. Greenwood, P. Griffin, V. Pronyaev, A. Trkov, K. Zolotarev, “Validation of IRDFF in 252Cf standard and IRDF-2002 reference neutron fields”, European Physical Journal, 2015, presented at ISRD15.
10. P.J. Griffin, ”A rigorous treatment of self-shielding and covers in neutron spectra determinations”, IEEE TNS, pp. 1878-1885, Vol. 42, 1995.

Progress of the CEA contribution to IRDFF validation: experimental data and codes, *C. Destouches*

Centre d'Etudes Nucleaires de Cadarache, Saint-Paul-lez-Durance, France

The CEA presents the evolution status of its reactor facilities: CALIBAN (closed at end 2014), OSIRIS (to be closed at the end of 2015), EOLE and MINERVE (to be refurbished by mid2016), MASURCA (restarting 2019), JHR (starting end 2019) and ZEPHYR (replacement of EOLE-MINERVE, >2020). This important evolution explains the priority given to measurement campaigns. The progress of the CEA contribution is the following.

A. Nuclear Data

Concerning the CALIBAN’s experimental data, only ^{54}Fe(n,p), ^{60}Co(n,γ), ^{46}Ti(n,p), ^{63}Cu(n,α), ^{115}In(n,n’) measurements performed in the reactor cavity are selected. Fission chamber measurements ^{235}U(n,f), ^{238}U(n,f), ^{237}Np(n,f) could be added if they bring valuable information. The official ISCBEP benchmark (HEU-FAST-080) will be used for modelling. External measurements will not be used because there is no modelling benchmark published. These measurements could be used as test cases for a “REAL type” unfolding benchmark.

In addition, ^{58}Ni(n,p)^{58m}Co decay period and branching ratio experimental evaluations have been done and confirm previous published values in a representative reactor core spectrum. The branching ratio given by JEFF3.2 (0.53 versus 0.313) is different because measured at 14 MeV. These data will be published in 2015.

CEA is interested in the ^{117}Sn(n,n’)^{117m}Sn reaction because of its unique characteristics combination: $E_{threshold}$ = 0.314 MeV, $T_{1/2}$ = 14 d, E_{g1} = 158 keV (I_{g1} = 86%). Experimental irradiations of this dosimeter have been done with enriched Tin (93% at. ^{117}Sn) in different spectra: TRIGA MK-II – IJS – 2011, CALIBAN – CEA/Valduc 2011, OSIRIS/ISIS MTR CEA/Saclay in 2012 and 2014, EOLE (ZPR) 2014 – 2015 (to be continued). Measurement activity method has been upgraded at MADERE facility - CEA/Cadarache (coincidence summing correction factors estimated by modelling, calibration curves fitting,…) in order to reach a 5% uncertainty (1σ). However, nuclear data need upgrades (lack of uncertainties, discrepancies between the different library evaluations) to allow this reaction to be used. CEA supports a new evaluation of this reaction and its future integration to IRDFF. The experimental results and decay period evaluation will be published.

The previous analysis of the EOLE available experimental data (*G. Zerovnik, C. Destouches - Self-shielding factor calculations of heterogeneous samples in activation measurements - NENE2010*) need to be updated with recent nuclear data for activity measurements (^{103m}Rh, ^{116m}In) and with new IRDFF version for reaction rate evaluations. The following reactions: ^{27}Al(n,α)^{24}Na, ^{59}Co(n,γ)^{60}Co, ^{54}Fe(n,p)^{54}Mn, ^{115}In(n,n’)^{115m}In, ^{115}In(n,γ)^{116}In, ^{24}Mg(n,p)^{24}Na, ^{55}Mn(n,γ)^{56}Mn, ^{58}Ni(n,p)^{58}Co, ^{197}Au(n,γ)^{198}Au, ^{103}Rh(n,n’)^{103m}Rh, ^{51}V(n,α)^{48}Sc, ^{64}Zn(n,p)^{64}Cu, ^{64}Zn(n,γ)^{65}Zn will be available. Although, publication of the EOLE core description as a benchmark will not be possible, only local

calculated neutron spectrum will be provided. These measurements could be used as test cases for a "REAL type" unfolding benchmark.

The MUSE 4 experimental measurement data obtained in MASURCA are available (table below) (*M. Plaschy and C. Destouches - Investigation of ADS-Type Heterogeneities in the MUSE4 Critical Configuration - Journal of NUCLEAR SCIENCE and TECHNOLOGY, Vol. 42, No. 9, p. 779–787, September 2005*). These activity measurements must be re-evaluated with recent nuclear data.

Table 3 Experimental spatial indices for the M4CRIT configuration

Reaction	Threshold (MeV)	Location 2/1	Location 3/1	Location 4/1	Location 5/1	Location 6/1
$^{115}In(n,\gamma)$	/	1.16±0.02	1.12±0.02	1.13±0.02	1.02±0.02	0.88±0.01
$^{197}Au(n,\gamma)$	/	1.22±0.03	1.16±0.03	1.17±0.03	1.05±0.02	0.92±0.02
$^{59}Co(n,\gamma)$	/	1.54±0.04	1.33±0.04	1.29±0.04	1.06±0.03	1.23±0.04
$^{64}Zn(n,\gamma)$	/	1.19±0.02	1.14±0.02	1.11±0.02	1.02±0.04	0.83±0.01
$^{115}In(n,n')$	~0.4	0.74±0.01	0.88±0.01	0.86±0.01	0.90±0.01	0.75±0.01
$^{58}Ni(n,p)$	~1.0	0.52±0.01	0.74±0.01	0.74±0.01	0.81±0.01	0.71±0.01
$^{54}Fe(n,p)$	~1.6	0.49±0.01	0.73±0.02	0.74±0.02	0.81±0.02	0.70±0.02
$^{64}Zn(n,p)$	~1.7	0.48±0.02	0.71±0.03	0.69±0.03	0.79±0.03	0.67±0.03
$^{59}Co(n,p)$	~2.6	0.43±0.02	0.65±0.04	0.70±0.04	0.80±0.04	0.69±0.04
$^{56}Fe(n,p)$	~5.0	0.38±0.01	0.68±0.01	0.64±0.01	0.75±0.02	0.72±0.02
$^{24}Mg(n,p)$	~5.8	0.37±0.02	0.66±0.02	—	0.76±0.03	—
$^{27}Al(n,\alpha)$	~5.9	0.39±0.01	0.73±0.03	—	—	—
$^{51}V(n,\alpha)$	~7.1	0.46±0.03	0.69±0.05	0.72±0.05	0.77±0.06	0.70±0.05
$^{93}Nb(n,2n)$	~9.0	0.39±0.01	0.67±0.01	0.66±0.01	0.79±0.01	0.73±0.01

Calculation benchmark has been published and evaluated through the MUSE 4 project:

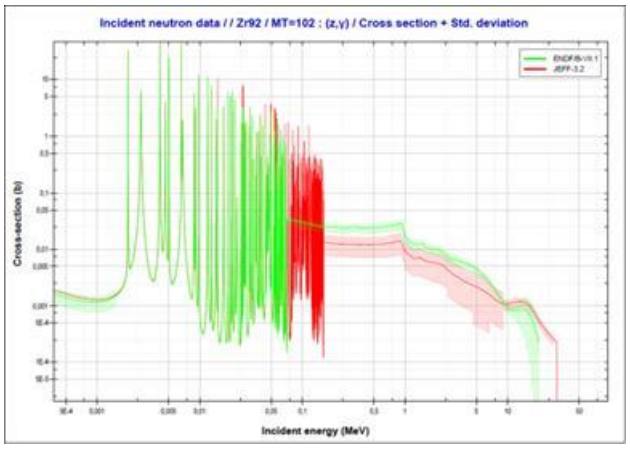
- *R. Soule "Neutronic Studies in Support of Accelerator-Driven Systems: The MUSE Experiments in the MASURCA Facility", Nucl. Sci. and Emg.: 148, 124–152 ~2004.*
- *D. Villamarin, "Benchmark on Computer Simulation of MASURCA Critical and Subcritical Experiments (MUSE-4 Benchmark)", WPPT(2001)5, NEA/OECD, Paris, France (2001).*
- *"Benchmark on Computer Simulation of MASURCA Critical and Subcritical Experiments MUSE-4 Benchmark", Nuclear Science ISBN 92-64-01086-6, NEA/NSC/DOC(2005)23.*

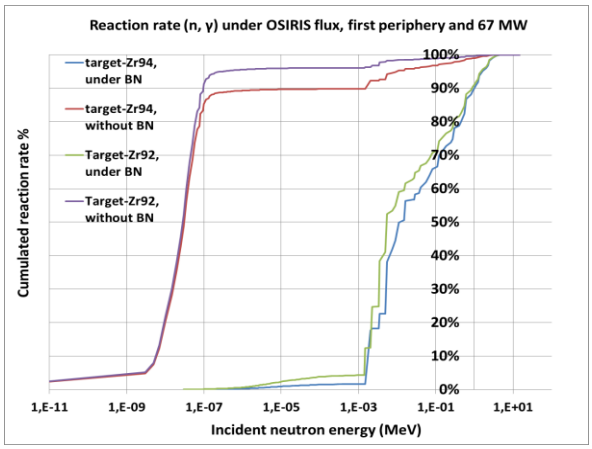
If the formalism and the quality of these data are good enough, a new version of the analysis could be done.

Concerning MINERVE facility, a publication on experimental measurements of $^{115}In(n,\gamma)^{116m}In$ relative gamma emission intensities and half-life estimation will be made at ANIMMA 2015 (April 2015) by A. Gruel et al.

In addition, CEA welcomes propositions for isotopes or materials to be tested by oscillation technique and activation measurements in the next 2016 experimental campaign.

CEA indicates that the study of using ^{92}Zr enriched Zirconium dosimeter is ongoing. The $^{92}Zr(n,\gamma)$ reaction study under BN filter allows to reach an interesting energy response ranging from 10 keV to 1 MeV (see figure below). A first irradiation test will be performed at OSIRIS by mid-2015. In addition, a publication of the preliminary results will be done at ANIMMA 2015. Nuclear data need to be upgraded due to large discrepancies between evaluations.



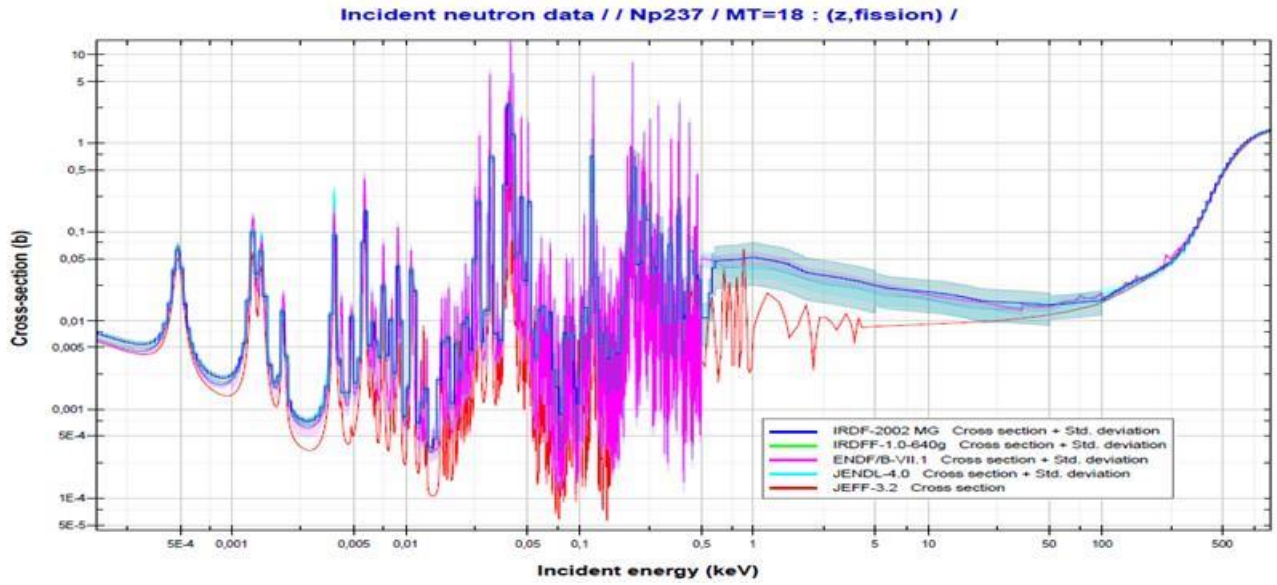


CEA is interested in participating to the validation of the nuclear data analysis. Feedback of the JEFF3.2 evaluation for some reactions, comparison of the covariance files with the COMAC process, systematic test of proposed evaluations on MASURCA and PROTEUS spectra (1966 grps) - X/U5 spectral indexes could be performed if needed.

Reactions	Contents (CEA/SPRC)	Availability
4. 23Na(n,2n)22Na	Evaluation	COMAC/JEFF32
5. 23Na(n,g)24Na	Evaluation	COMAC/JEFF32
7. 27Al(n,p)27Mg	Evaluation	COMAC
8 27Al(n,alpha)24Na	Evaluation	COMAC
21. 55Mn(n,g)56Mn	Measures : MAESTRO-1+ IRMM/ORNL+Evaluation ORNL	NT-LEPH-12-211
30. 59Co(n,g)60Co	Measures : MAESTRO-1 + CEA/SPRCLEPH Evaluation	COMAC
48. 93Nb(n,g)94Nb	Measures : MAESTRO-2	Under progress
55. 127I(n,2n)126I	Measures : IRMM	EXFOR ?
63. 197Au(n,g)198Au	Measures : IRMM + Evaluation IRMM	JEFF32
68. 232Th(n,g)233Th	Measures : OSMOSE	JEFDOC-1502
69. 235U(n,f)FP	Measures : ILL/ CEA	EXFOR ?
71. 238U(n,g)239U	Qualification CEA	COMAC
73. 239Pu(n,f)FP	Measures : ILL/ CEA	EXFOR ?

The status of the following reactions in the next IRDFF is still under investigation:

- The ^{93}Nb(n,n')^{93m}Nb reaction is one of the most important for MTR / reactor Vessel survey (Energy range, Period, ...). Up to now, the ^{93m}Nb(n,γ)^{94}Nb is still not known, leading to possible error in activity interpretation in the high level thermal flux.
- C/M discrepancies (10%) are observed for ^{103}Rh(n,n')^{103m}Rh reaction. This reaction is important for ZPR neutron calculation scheme validation.
- C/M discrepancies are still observed for ^{55}Mn(n,γ)^{56}Mn reaction.
- Choice of an unique benchmark value for ^{63}Cu(n,α)^{60}Co (10% between the 2 accepted values). Measured values at CALIBAN facility could be of interest.
- Remaining issues on the ^{237}Np(n,f) reaction (used for interpretation of the Vessel Surveillance capsules):
  - → Uncertainties are still relatively high in the 1 keV – 100 keV (> 5%);
  - → Discrepancy around 7% between the recent n-TOF (CERN) and (LANL) measurements has to be analysed.



The following inclusions in IRDFF are wished:

- ➔ Cross sections:
  - → ^{58}Co/^{58m}Co energy depending branching ratio inclusion
  - → ^{117}Sn(n,n')^{117m}Sn cross section
  - → ^{93m}Nb(n,γ)^{94}Nb cross section
- ➔ Fission yields associated to fission reactions
- ➔ Decay data:
  - → Re-evaluation of the ^{103m}Rh $Ix_{ka,b}$ emission probability.
  - → Evaluation of the existing nuclear decay data for ^{117m}Sn

B. Status of the unfolding code: ECORCES / CALMAR

Adjustment (Unfolding) process is considered by CEA as one of the best process to propagate uncertainties to the output data. It consists in the estimation of the most likely neutron spectrum and its covariance matrix from "measured" reaction rates (saturated activities) using nuclear data (cross sections), prior neutron spectrum and their respective covariance matrices. In this goal, **ECORCES** and **CALMAR** codes have been developed for Nuclear Reactor spectra application (E < 20 MeV).

ECORCES (Energy **CO**mputation of **R**eaction rate from **C**ross section & n**E**utron **S**pectrum) aims at providing an user friendly tool for data shaping operations (E_{max} = 20 MeV, 640 grp of SAND-II format):

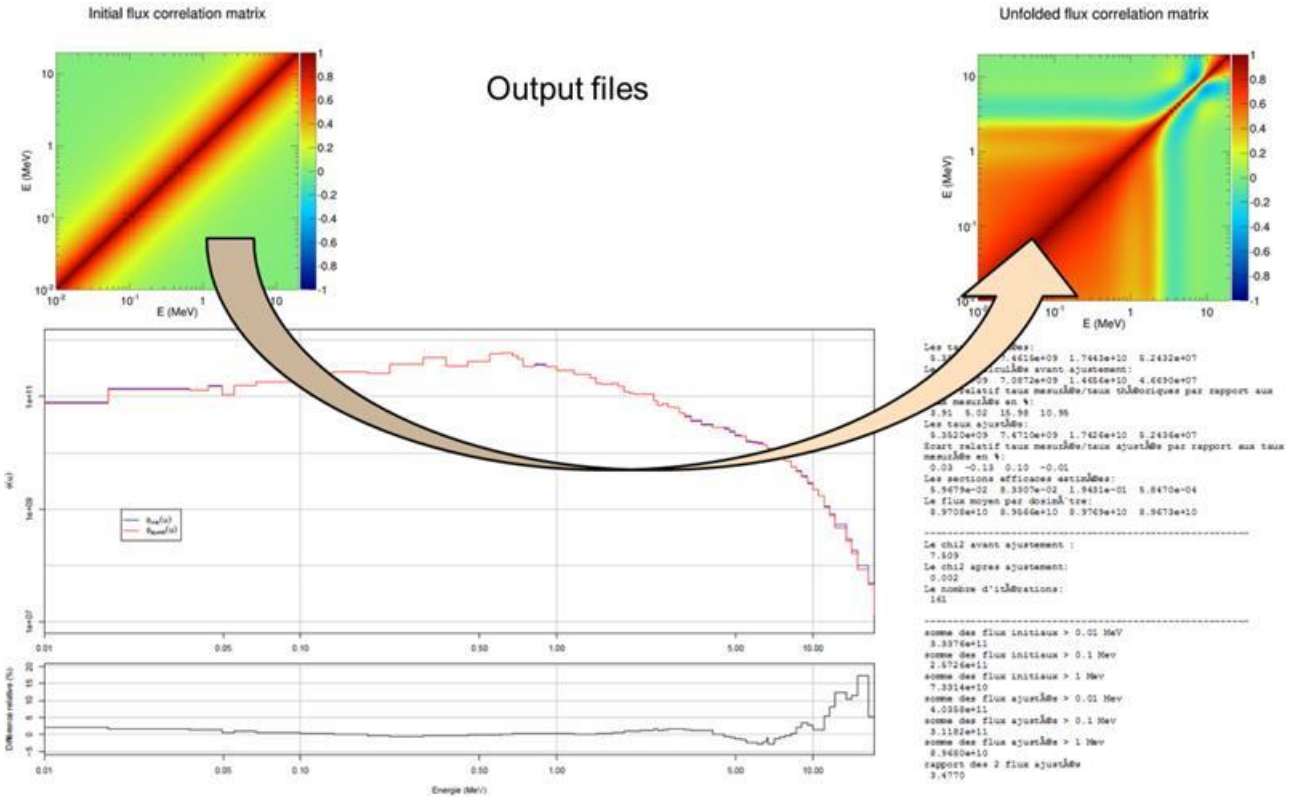
- Energy Mesh modification for neutron spectrum and Covariance Matrices;
- Condensation of cross section and Covariance Matrices on a selected Energy Mesh;
- Integrated Flux and Reaction Rate calculations with associated uncertainty quantification;
- Input data for CALMAR;
- GUI (WINDOWS LINUX);
- Encapsulated programming.

This program will be proposed for publication at NEA by the end of 2015. An upgrade to 60 MeV and 725 groups should be possible but has to be studied.

The **CALMAR** code is based on STAY'SL principle: Least squares, Gaussian conditioning methods. It is coded under a C++/ROOT platform and uses input data extracted from ECORCES results.

It should be noticed that the prior neutron spectrum covariance remains a key point of the data preparation. An optimization of the process has been performed in 2013 and 2014 with an implementation of parallel and iterative shape and normalization processes (based on MS-ITER option). A rigorous uncertainty propagation analysis has also been done (normalization, covariance inclusion). In addition, as a coding in C++ of STAY'SL, MSITER and GRAVEL/SANDII has been performed, a multi-code analysis is now possible. Year 2015 will be dedicated to code finalization and publication. An extension to 60 MeV and 725 groups could be studied if needed and possible. This code could be used for the new REAL adjustment benchmark.

Example of output CALMAR file



Updates on the LANL Efforts in Support of Testing and Improving the IRDFF Library, *M. White*

Los Alamos National Laboratory, Los Alamos, NM, USA

Los Alamos will continue to be an active participant within the Coordination Research Project (CRP number F41031) on Testing and Improving the International Reactor Dosimetry and Fusion File (IRDFF). Neutron dosimetry remains a vital part of our core missions and we view our participation within these activities as a key way to exchange best practices and develop new understandings that will benefit the wider community. As part of the discussions during the second CRM, we note the following recent work and ongoing plans.

A request was made to submit the historic Flattop-HEU activation measurements and associated data for consideration in a planned 'REAL84' like exercise to compare the current neutron spectra unfolding codes. We have committed to make such a submission.

Los Alamos has recently published many of the activation measurements made in the Godiva, Jezebel, Flattop and Bigten critical assemblies [Nuclear Data Sheets 118 (2014) 1–25] previously available only from informal reports and private communications. Systematic errors were given for a limited set of these values; only mean values for the remainder. Efforts continue at the laboratory to dig into the logbooks associated with the original measurements to understand the methods and data used to report these values and to assign modern means and uncertainty values. Activation will be an important part of the measurements made on new critical assemblies at the NCERC facility. Plans include new measurements in bare and natural uranium reflected (Godiva-like and Flattop-HEU) assemblies hopefully in 2016. We welcome input on prioritization on the specific activations to be made and would encourage potential collaborators to contact us with joint measurements that might be made.

The current status of several differential cross section measurements underway of interest to the activation community were reported. A new measurement of the U238(n,2n) reaction has been performed using 'mono'-energetic neutrons produced by the TUNL (Duke University) tandem accelerator. We appreciate feedback from the CRP members to the experimentalists helping get the analysis updated to a modern basis and expect that these data will be finalized and published in time to influence the IRDFF evaluation.

The TUNL-LANL-LLNL collaboration has been working to measure the energy dependence of the fission product yields. Data have been reported for 5 energies from 0.5 MeV to 14 MeV for the actinides U235, U238 and Pu239. Many of the cumulative fission product yields in the reactor fission range -- 200 keV to 1.5 MeV average energy causing fission -- show a strong energy dependence that must be taken into account in assessing the number of fissions inferred from measurements of such neutron fields. It was noted that many of the fission products recommended for measurements by IRDFF were not included in the data reported by TUNL. We will look into a request that was made to examine the gamma spectra that have been taken to see if yields for these can be extracted. It was noted that further measurements are planned to better understand the energy dependence near the shoulder at first chance fission and between fission and 14 MeV energies.

The issue of photonuclear reactions -- particularly (γ,n) and (γ,fission) -- contributing to activations used for neutron dosimetry was also discussed. There were concerns regarding the understanding of the gamma spectra produced by neutron induced fission as well as the photonuclear cross sections. A report of the recent measurements by DANCE on the gamma multiplicity and energy distributions from fission was provided. As the DANCE measurements have a threshold of many hundred keV, additional measurements of the keV energy distributions are also needed and we note these have been made by other institutes. Evaluations including all of these new data are needed. A report on new measurements of photo-fission cross sections using the photon source at the HIgS facility at TUNL was also made. Efforts should be made by the community to establish a list of high-priority photon induced measurements that are needed.

The californium-252 prompt fission neutron spectra (PFNS) is a standard and activation measurements in this spectrum are a key part of the validation of the IRDFF cross sections. Efforts are ongoing to define a uranium-235 thermal neutron induced PFNS spectra and use similar measurements made in such spectra for IRDFF validation. A report was made on the Chi-Nu experiment to measure these data in fast neutron induced fission PFNS with particular emphasis on the issues of multiple scattering that have plagued previous measurements. The difficulties making PFNS and activation measurements for neutron emission energies below 2 MeV was acknowledged and prompted good discussions of the issues that must be studied. This is an area of research we expect to be ongoing for some time.

It was noted by many participants that the lack of documentation on experiments continues to be a severe problem for the community. Journals continue to limit overall content such that key details needed to reanalyze an experiment are not included in the publication. Gaining access to logbooks or internal reports that might include these details is difficult.

We would encourage an IRDFF activity to produce a document that outlines what must be included in reports of activations -- in particular for benchmark measurements -- and that the IAEA NDS and other organizations work to encourage journals to require authors to submit such data to EXFOR before such articles are published.

IRMM projects related to IRDFF,

A. Plompen

on behalf of EC-JRC-IRMM and collaborators
Institute for Reference Materials, Geel, Belgium

In RCM-1 we expressed our interest in contributing to the knowledge of spectrum averaged cross sections for dosimetry reactions for which improved or new data are needed for IRDFF. This concerned spectrum averaged cross sections in the ^{252}Cf spectrum and maxwellian averaged cross sections in the quasi-maxwellian 30 keV spectrum due to Li(p,n) source reaction at threshold. Here I also report neutron time-of-flight measurements for gold and uranium-238 that are of interest to improved capture cross sections for IRDFF.

For spectrum averaged cross sections in the ^{252}Cf spectrum a proposal for METROFISSION was prepared and submitted to EMRP (European Metrology Research Programme). The project involved the use of the ^{252}Cf source in the low scattering room at NPL, UK and the use of the ^{27}Al(d,n) field at PTB. Despite the support of IAEA-NDS, EMRP decided not to fund METROFISSION so that insufficient resources were available to continue the foreseen activity.

^{252}Cf spectrum averaged cross sections were obtained at IRMM by N. Jovancevic, L. Daraban and S. Oberstedt and the results are shown in the presentation. The objective was two-fold: obtain experience with such measurements to obtain data complementary to time-of-flight measurements of the fission neutron spectrum and to develop experience with the unfolding technique using a well-known spectrum. C/E values of the cross sections were obtained with an uncertainty of about 10% in a field of just 200 n/cm2/s for Au197(n,γ), In115(n,n'), Ti47(n,p), Ni58(n,p), Fe54(n,p), Co59(n,p), Ti46(n,p), Mg24(n,p), Ti48(n,p), Al27(n,a), Zr96(n,2n), Au197(n,2n), Co59(n,2n) and Zr90(n,2n). Large underpredictions were found for Au197(n,γ), which can be understood from room return contributions, for Zr96(n,2n) which again is understood from room return leading to Zr95 by the Zr94(n,γ) reaction and Zr90(n,2n). The latter was quite a puzzle and the problem was eventually traced down to a close gamma-ray in the decay chain of Th. The low level activities were determined in the ultra-low-level underground HADES laboratory.

For the study of scattering modifications to the Li(p,n) neutron spectrum, measurements were done by collaborators from Goethe University, Hebrew University and SOREQ that will have results of interest

to activation studies in this recently very well characterized neutron spectrum. The data will be made available as soon as the analysis is completed.

We greatly welcome the summary of IRDFF related C/Es shown by S. Simakov for the 30 keV MACS based on the KADONIS database. It highlights the need for a better understanding. In our opinion this warrants the need to readdress some of the activation measurements in this spectrum with improved accuracy. However, it also requires in important cases a re-evaluation of the resonance range data and possibly accurate new measurements to establish the energy dependent cross sections.

For low level density nuclei the MACS may be dominated by resonance structure. An important example is the Au197(n,γ) reaction for which a recent accurate evaluation based on high accuracy measurements at IRMM shows that the MACS cross section evaluated from it should be higher than that of Ratynski and Kaeppeler. Also a recent activation measurement by Feinberg et al. made at IRMM shows that the Au197(n,γ) MACS should be higher. The IRMM evaluation is included in the recently released JEFF-3.2 library and includes both the resolved and unresolved resonance range. The URR was developed in close relation with V. Pronyaev and his proposed extension to the standards evaluation is very close to what is in JEFF-3.2. The extension below ~ 10 keV is however very important for the MACS. The JEFF-3.2 evaluation was validated with A. Trkov on the basis of lead slowing down spectrometer data and is convincingly better than earlier evaluations.

For cases like Mn55(n,γ) (discussed intensively at RCM-2) used in spectra where the reaction rates are dominated by the neutron energy range from 10 keV to 1 MeV a similar combination of experiments and evaluations is needed to come to agreement and a critical assessment of its use in hard neutron spectra. Data with the time-of-flight technique were taken with ORNL at IRMM up to 100 keV. Their impact on the evaluations should be verified. Given the present C/E of -15% and the rather large uncertainty this reaction may be an interesting test case for understanding the role of MACS at 30 keV and time-of-flight based data for the validation of libraries like IRDFF.

Finally a new evaluation was prepared at IRMM for the U238 resonance range, with emphasis on the unresolved energy range. As for the above-mentioned work for Au-197, this work was carried out by Peter Schillebeeckx and collaborators and the evaluation for U-238 was contributed at the end of 2014 to the JEFF and CIELO projects. First validation benchmarking shows excellent performance. The evaluation was prepared in close collaboration with V. Pronyaev who contributed to with GMA based evaluations of the total and capture cross sections in the URR. Above 10 keV the proposed evaluation therefore closely agrees with the capture cross section that is the by-product of the standards evaluation. As shown by a direct comparison made by A. Trkov during the RCM-2 meeting this new evaluation for 238U(n,γ) does not agree with the evaluation prepared by K. Zolotarev based on activation data. Differences amounting to 5% should be hard to reconcile with the benchmarks. We therefore strongly recommend to adopt the evaluation by Sirakov et al. contributed to JEFF and CIELO.

Summary report on the work done after the first CRP meeting,

P. Mastinu

Istututo Nazionale di Fisica Nucleare, Laboratori Nazionali di Legnaro, Italy

Since the previous meeting, the BELINA neutron beam line at Laboratori Nazionali di Legnaro of INFN has been completed and tested. It has been primary developed to perform the measurement of the neutron spectra for the production of 30 keV Maxwell-Boltzmann (MB) neutron energy distribution using the method developed for the LENOS project. The method consists in shaping the proton beam energy to a desired one in order to produce high quality MB neutron energy spectra, avoiding the use of moderators. The proton beam is shaped to a desired energy distribution using a thin Aluminium foil (75 μm thickness) on which a proton beam of 3661 keV impinge. The shaped proton energy beam impinge on a metal Lithium target and the produced neutron spectra resemble

very well a MB distribution at kT = 30 keV. Tuning the proton energy and foil thickness, different MB distribution can be obtained at different stellar temperature (kT ranging from 23 to 70 keV).

The pulsing system has been modified in order to have a variable repetition rate (not available before where the available working frequency of the accelerator was only 3 MHz), some bugs has been solved and we moved from the prototype to a final configuration. Since particular care have to be devoted to the calibration of the accelerator, we installed a system for measuring and monitoring the proton beam energy and energy distribution by Time of Flight (TOF) using a flight path of about 13 meters. The TOF is measured by two pickup detectors. The need of having a good energy calibration, as well as a continuous monitoring of it, is essential since the calibration with reactions (threshold or (p,γ)) can be performed only at too much different proton energy (the threshold Li(p,n) is at 1880 keV for instance). We have measured in our accelerator that, changing the voltage (in order to change the beam energy), the offset of the machine change, as well as the time and energy distribution of the pulses. This is quite usual in other accelerators of the same type and, as a consequence, can bring to a wrong beam energy determination.

The neutron beam line has been constructed and tested: particular care has been devoted to reduce as much as possible the background and/or using materials with a well known cross section.

In order to have very high statistic within a reasonable beam time, a new detector has been developed. It is basically a BaF2 scintillator (chosen because of the good timing and good energy resolution) coupled with a thick disc of ^{10}B. Neutrons are captured by the ^{10}B and the associated 480 keV gamma produced is detected in the BaF_2 scintillator. Thanks to the low absorption of gammas, we can use very thick Boron capsule, consequently increasing the capture cross section and thus the efficiency of the detector.

In the second half of 2013 we performed an experiment for measuring the *Li(p,n)* cross section near threshold using our beam line and the detector we developed.

Routines for data analysis has been written and tested: a paper on these results is writing. This experiment (basically very similar to the one for measuring the MB neutron spectra) proven that everything is ready for the measurement of the MB neutron spectra.

We already measured neutron spectra at 0 and 20 degrees at LNL, since there aren't other suitable facilities in Europe with pulsed beam.

We plan to finalize the spectra measurements at all relevant angles in 2015.

***The plan for next period*:**

As soon as the neutron field will be characterized, we are going to perform the activation measurement of gold, since it is used as a reference in the nuclear astrophysics cross section measurements as well as for all other measurements proposed and of interest for the CRP on IRDFF.

An activation of gold will be performed also in pulse mode, irradiating and acquiring at the same time the energy spectra at zero degree using the same setup used for the measurement of the neutron spectra.

After that, we will plan future experiments covering both needs of astrophysics (e.g $^{139}La(n,\gamma)$ reaction) and needs of fast reactor applications (e.g. $^{55}Mn(n,\gamma)$ reaction). Following RCM discussion we modified our setup: in order to reduce the multiple scattering in the target backing, a new semispherical target, with 100 μm thickness has been developed.

Integral Measurements for the Validation of the IRDFF Dosimetry Cross Sections,

I. Kodeli, G. Žerovnik, V. Radulović

Jožef Stefan Institute, Jamova 39, Ljubljana, Slovenia

The objective of the work done at JSI is to verify the consistency of the differential cross section data in the IRDFF library with integral measurements in thermal reactors and propose additional monitor materials for inclusion in the library. Activation measurements were performed in different irradiation channels in the TRIGA Mark-II Reactor at JSI with different foils to determine the constants for NAA using advanced methods.

1. TRIGA irradiations of ^{45}Sc, ^{55}Mn, ^{58}Fe, ^{59}Co, and ^{232}Th foils

The first set of measurements including ^{45}Sc, ^{55}Mn, ^{58}Fe, ^{59}Co, and ^{232}Th foils was performed in the channel F24 (with the pneumatic post system) in the outer ring of the TRIGA reactor core [1]. The initial spectrum in the channel F24 was calculated using a detailed (verified and validated) computational model of the JSI TRIGA reactor [2] developed using the Monte Carlo code MCNP5. The cadmium ratios of capture reactions on ^{45}Sc, ^{55}Mn, ^{58}Fe, ^{59}Co, and ^{232}Th were measured and compared with the calculated reaction rate ratios using the initial and unfolded MCNP spectra. The latter was obtained by unfolding using ^{27}Al(n,p)/^{197}Au(n,γ) and ^{27}Al(n,α)/^{197}Au(n,γ) activation rate ratios. Inelastic scattering on ^{117}Sn was also measured, however due to non-existent activation cross section in IRDFF it was ignored in this analysis. All samples were small enough to be able to neglect the self-shielding and self-absorption effects. The IRDFF v1.05 activation library was compared to the previous version IRDF-2002 [3, 4] and the general purpose library ENDF/B-VII.1 [5].

The results of the comparison are shown in Tables 1 and 2. In the absolute comparison (Table 1), the MCNP calculated neutron spectrum was normalized to the full power of 250 kW [6] taking into account a systematic normalization uncertainty of the order of ±10%. Due to large normalization uncertainty, the (relative) reaction rate ratios are considerably more accurate compared to the absolute values. The reaction rate ratio uncertainties were calculated taking into account both spectrum (for the unfolded spectrum only) and cross section (for IRDFF only) covariance matrices in the first-order approximation (the so-called "sandwich formula"). In all cases, the cadmium transmission function was calculated from the total cross section from the IRDFF library assuming Cd cover thickness of 1 mm.

Table 1: Comparison of the calculated specific reaction rates *C* (normalized **MCNP** spectrum and **IRDFF v1.05** library) to measured specific saturation activities *E* (per target atom) in the irradiation channel F24 in the JSI TRIGA reactor.

Reaction	*Reaction rate – Bare* [10^{-12} s^{-1}]			*Under Cd* [10^{-12} s^{-1}]		
	C	*E*	*C/E*	*C*	*E*	*C/E*
^{45}Sc(n,γ)	67.0 ± 8.1	69.5 ± 0.1	0.96 ± 12%	1.52 ± 0.19	1.65 ± 0.02	0.92 ± 12%
^{55}Mn(n,γ)	33.7 ± 3.4	35.7 ± 0.7	0.94 ± 10%	1.70 ± 0.18	1.79 ± 0.02	0.95 ± 10%
^{58}Fe(n,γ)	3.33 ± 0.37	3.45 ± 0.10	0.97 ± 12%	0.163 ± 0.018	0.153 ± 0.010	1.07 ± 11%
^{232}Th(n,γ)	28.6 ± 2.9	28.1 ± 0.3	1.02 ± 10%	10.7 ± 1.1	10.2 ± 0.2	1.05 ± 10%

The results are mostly in good agreement with the measurements. For ^{58}Fe and also ^{55}Mn, there is a significant improvement compared to the old IRDF-2002 library. The calculated cadmium ratios for some reactions are identical using different libraries; in these cases the evaluations are adopted from the IRDF-2002 or ENDF/B-VII.1 libraries. Slight inconsistencies for ^{55}Mn and ^{59}Co between IRDFF and measured cadmium ratios might imply slight underestimation uncertainties of the cross sections and/or the unfolded neutron spectrum. ^{45}Sc capture cross section from IRDFF has an impractically high uncertainty which is apparently highly overestimated.

Table 2: Comparison of calculated and measured reaction rate ratios in the irradiation channel F24 in the JSI TRIGA reactor.

React. rate ratio	Spectrum	IRDFF v1.05	IRDF-2002	ENDF/B-VII.1	Measured
^{45}Sc(n,γ)	MCNP	44.00 ± 17.25	44.00	43.36	
(Cd ratio)	unfolded	46.14 ± 19.01	46.14 ± 0.89	45.47 ± 0.85	49.02 ± 0.11
^{55}Mn(n,γ)	MCNP	19.79 ± 0.29	22.72	19.79	
(Cd ratio)	unfolded	20.78 ± 0.34	23.87 ± 0.14	20.78 ± 0.11	21.37 ± 0.15
^{58}Fe(n,γ)	MCNP	20.39 ± 0.45	19.04	15.48	
(Cd ratio)	unfolded	21.34 ± 0.52	19.94 ± 0.13	16.19 ± 0.10	21.88 ± 0.26
^{59}Co(n,γ)	MCNP	10.58 ± 0.00	10.59	10.58	
(Cd ratio)	unfolded	11.12 ± 0.03	11.12 ± 0.03	11.12 ± 0.03	11.44 ± 0.16
^{232}Th(n,γ)	MCNP	2.68 ± 0.00	2.67	2.68	
(Cd ratio)	unfolded	2.76 ± 0.00	2.76 ± 0.00	2.76 ± 0.00	2.75 ± 0.02

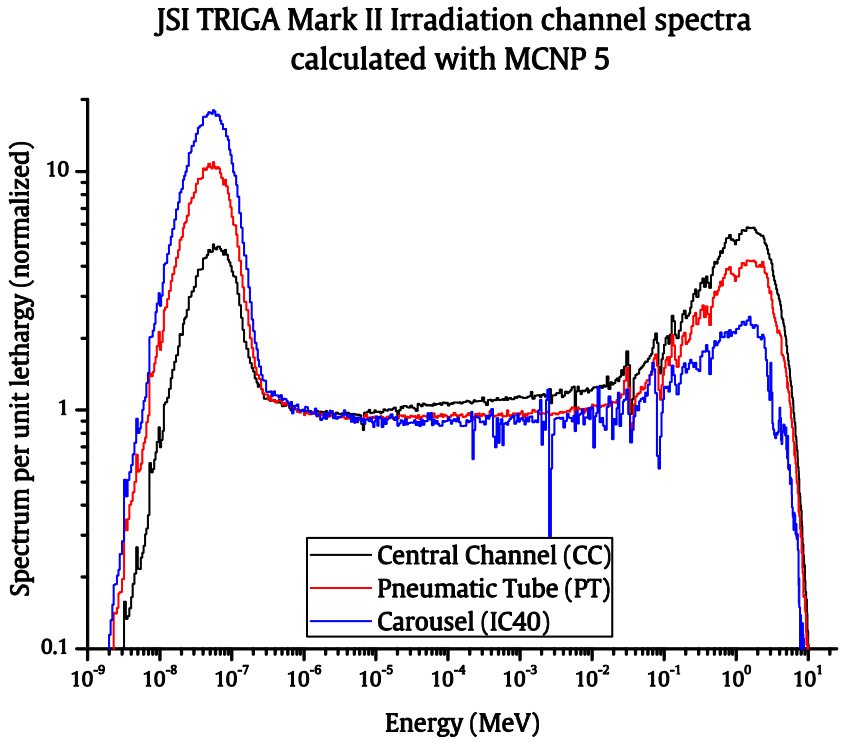


Figure 1: Neutron spectra in CC, PT and IC40 channels [3].

2. TRIGA irradiations of ^{55}Mn, ^{179}Au and TLDs

Due to its potential to be used as a tritium production monitor in fusion applications [7] a particular attention was put in the validation of the ^{55}Mn(n,γ) cross-sections. Foils of certified reference materials Al-1%Mn and Al-0.1%Au, as well as TLD(LiF) and LiPb samples were irradiated at different irradiation channels in the JSI TRIGA research reactor, i.e. in the central channel - CC, the pneumatic tube – PT in position F24 in the outer "F" ring of the reactor core and in the IC40 irradiation channel in the graphite reflector.

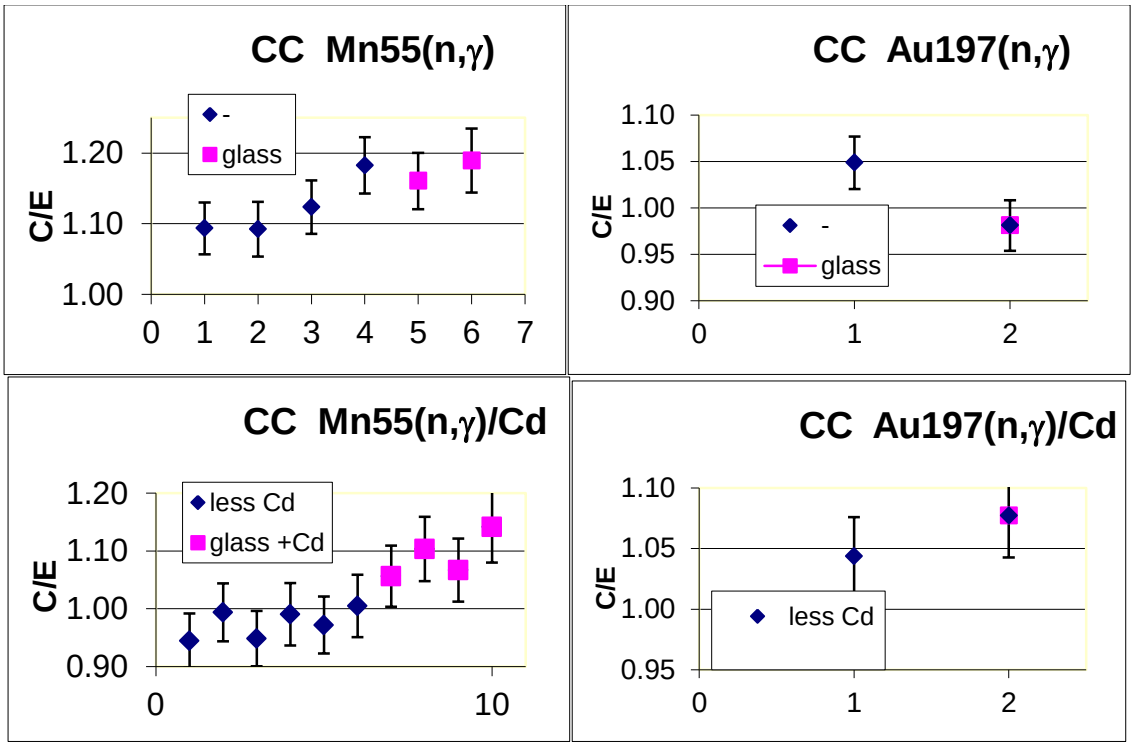
Altogether 11 irradiations were performed in spring 2014 [8] at the three TRIGA irradiation locations described above. The following dosimeters were used, all both bare and under Cd cover:

- Al-1%Mn and Al-0.1%Au foils,
- TLD(LiF) (3 x 30 mg), LiPb eutectic (~1g): in and without pyrex glass tube.

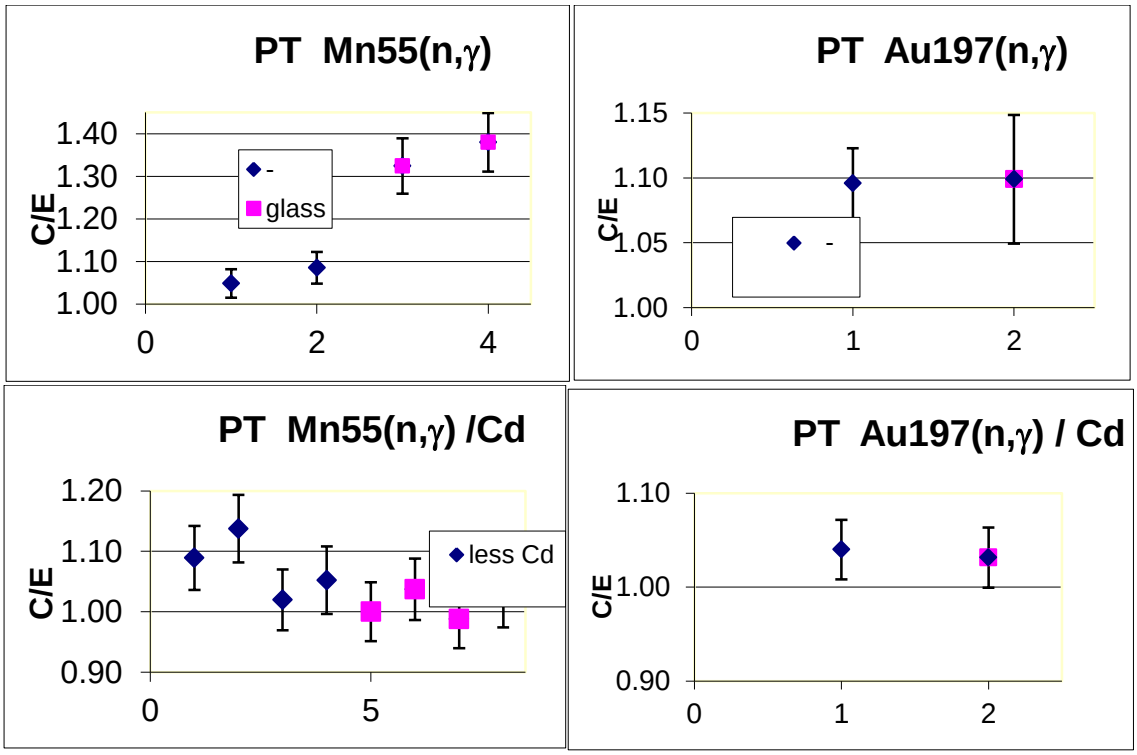
The analysis of the experiment is still underway. The first results of the comparison between the calculated and the measured saturated activities are presented in Figs. 4 - 6 below. Some bias can be observed in the results of the samples of Mn-Au and Al-Au irradiated in the presence of TLD, pyrex glass tube and larger quantities of Cd needed due to the size of the complete set-up. Contamination with B is likely to be present in the pyrex glass, but was not yet considered in this analysis. The perturbation of the flux due to the presence of pyrex glass tube and Cd were also not yet considered in the transport calculations. The measurements referred as "glass" and "glass+Cd" will therefore need supplementary analysis, contrary to bare samples without glass or with smaller quantities of Cd (referred as "-" and "less Cd", respectively).

In spite of the preliminary character of this comparison (note that the power variations during the irradiations and the perturbation of the flux and the reactor power due to glass and Cd were not taken into account here yet) is the agreement between the measurements and the calculations, as shown in Figs. 2 - 4, reasonably good, in particular for the CC and PT channels. Around 20 % overestimation of the calculated values in IC40 locations can be due to the more approximate spectra calculations at this location. On the other hand some systematic deviations in C/E observed for the TLD measurements (not yet shown here) are still under investigation.

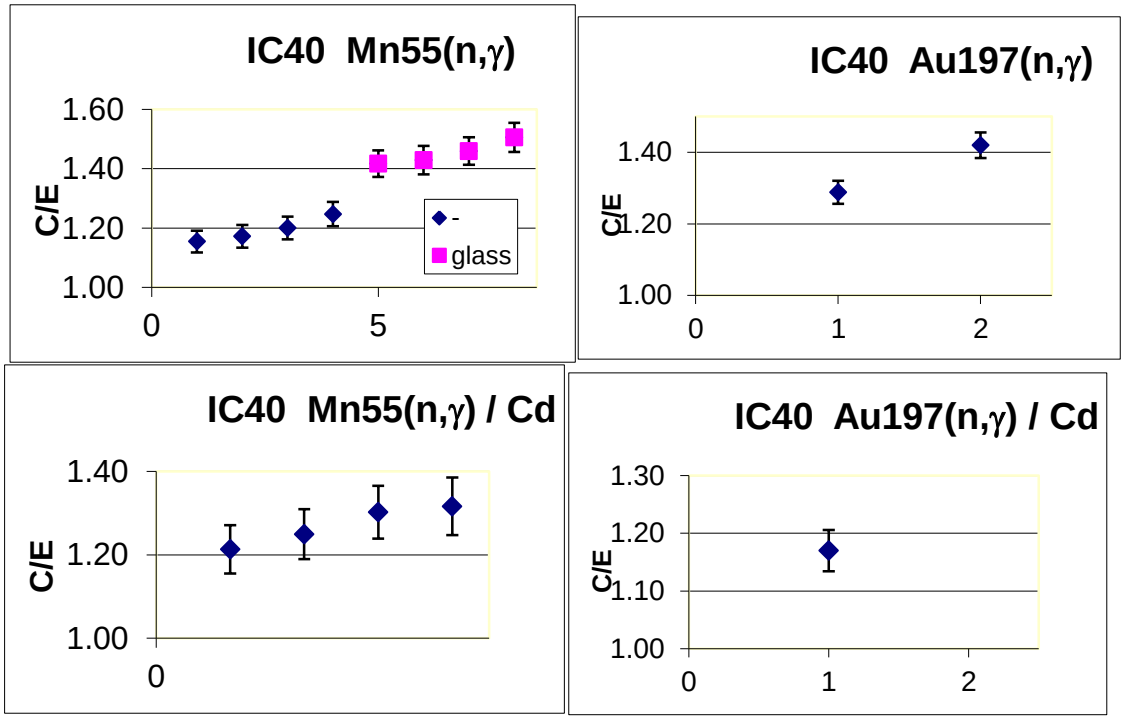
Mn foils were also prepared at JSI and ENEA Frascati to be irradiated in the ongoing FNG Copper benchmark experiment in spring 2015. The analysis of the experiment is still underway.



Figures 2: C/E comparison for the capture reaction measurements in CC. The two sets of results, with and without glass, correspond to different cooling times.



Figures 3: C/E comparison for the capture reaction measurements in PT.



Figures 4: C/E comparison for the capture reaction measurements in IC40.

References

[1] V. Radulović, "Validacija jedrskih presekov z uporabo aktivacijske metode moduliranih nevtronskih spektrov na reaktorju TRIGA," PhD thesis, Faculty for Mathematics and Physics, University of Ljubljana, September 2013 (in Slovene).

[2] G. Žerovnik et al., "Validation of the neutron and gamma fields in the JSI TRIGA reactor using in-core fission and ionization chambers," Applied Radiation and Isotopes 96 (2015) 27–35.

[3] R. Capote, K.I. Zolotarev, V.G. Pronyaev, A. Trkov, "Updating and Extending the IRDF-2002 Dosimetry Library," Journal of ASTM International 9 (2012) JAI104119.
[4] I. Kodeli, A. Trkov, "Validation of the IRDF-2002 dosimetry library," Nucl. Instruments and Methods A 577 (2007) 664–681.
[5] M.B. Chadwick et al., "ENDF/B-VII.1: Nuclear Data for Science and Technology: Cross Sections, Covariances, Fission Product Yields and Decay Data," Nucl. Data Sheets 112 (2011) 2887-2996.
[6] G. Žerovnik, M. Podvratnik, L. Snoj, "On normalization of fluxes and reaction rates in MCNP criticality calculations," Annals of Nuclear Energy 63 (2014) 126–128.
[7] I. Kodeli, F4E-2008- GRT-014 (ES-AC) – Action 2, Task 5: Sensitivity of the Tritium Production Rate to the Detector Position and Geometry Variation in the HCLL TBM Mock-Up. Examination of Covariance Matrices and Activation Foils for TPR Validation, IJS-DP-10457, 19 April 2010.
[8] I. Kodeli, V. Radulović, D. Kavšek, W. Pohorecki, T. Kuc, TRIGA Irradiations of Mn foils and TLD as Potential Tritium Production Monitors for Fusion Applications, 23rd Int. conf. Nuclear Energy for New Europe, sept. 8-11, 2014, Portorož, Slovenia.

Validation of IRDFF-v1.04 (&v1.05) Dosimetry Library using SINBAD Shielding Benchmark Experiments,

I. Kodeli

Jožef Stefan Institute, Jamova 39, Ljubljana, Slovenia

Similar validation exercises as performed in the past to validate the IRDF-2002 and IRDF-90 libraries was repeated for the new IAEA library of dosimetry cross sections "International Reactor Dosimetry File: IRDFF".

First, the radiative capture reactions in IRDFF were compared against the experimental KAYZERO and Mughabghab values as well as against the previous IRDFF-2002 and -90 data.

Furthermore, a series of shielding benchmarks available from the SINBAD database were used to check and validate the new IRDFF dosimetry file, version v. 1-04 (updated ^{6}Li(n,t) data were taken from v. 1.05). Several benchmark experiments performed at the Frascati Neutron Generator (FNG), ENEA Frascati, and in ASPIS, AEA Technology, Winfrith were analysed. The main purpose of repeating the calculations with the new dosimetry cross sections was to check for any improvement between measured and calculated reaction rates (compared to IRDF-2002 as well as IRDF-90) and removal of some inconsistent trends in the results for different monitors. Since the dosimetry data represent a relatively small part of the overall uncertainty, the major part coming from the transport cross section and model approximations, these results can be considered as an indirect validation of the new IRDFF dosimetry library. The compensation of errors between the transport cross sections and dosimetry data is likely.

Table 1: Dosimetry reactions measured at the benchmark experiments considered in this study.

Reactions	FNG-ITER Blanket	FNG SiC	FNG W	FNG HCPB TBM	FNG HCLL TBM	ASPIS Iron
^{93}Nb(n,2n) ^{92m}Nb	X	X	X	X	X	
^{58}Ni(n,2n) ^{57}Ni	X		X		X	
^{90}Zr(n,2n) ^{89}Zr			X		(X)	
^{27}Al(n,α) ^{24}Na	X	X	X	X	X	
^{32}S(n,p) ^{32}P						X
^{56}Fe(n,p) ^{56}Mn	X		X			
^{58}Ni(n,p) ^{58}Co	X	X	X	X	X	
^{115}In(n,n') ^{115m}In	X		X		X	X
^{103}Rh(n,n') ^{103m}Rh						X
^{55}Mn(n,γ) ^{56}Mn	X		X		X	
^{197}Au(n,γ) ^{198}Au	X	X	X	X	X	X
TPR(^{6}Li(n,t) ^{4}α)				X	X	

To obtain additional information potentially useful to conclude on the impact of transport and dosimetry cross-section uncertainties and their compensation, as well as on the computer code modelling uncertainties, the results using different transport cross-sections and computer codes (DOORS and MCNP) are presented for several benchmark analyses. The following dosimetry reactions were tested against the measurements: ^{6}Li(n,t), (^{7}Li(n,t)), ^{55}Mn(n,γ), ^{197}Au(n,γ), ^{58}Ni(n,p), ^{90}Zr(n,2n), ^{115}In(n,n'), ^{56}Fe(n,p), ^{93}Nb(n,2n), ^{103}Rh(n,n'), ^{32}S(n,p), ^{58}Ni(n,2n), ^{27}Al(n,α). Format errors were indicated in the ^{6}Li(n,t) IRDFF v.1.04 data and corrected at IAEA in the new v. 1.05 release.

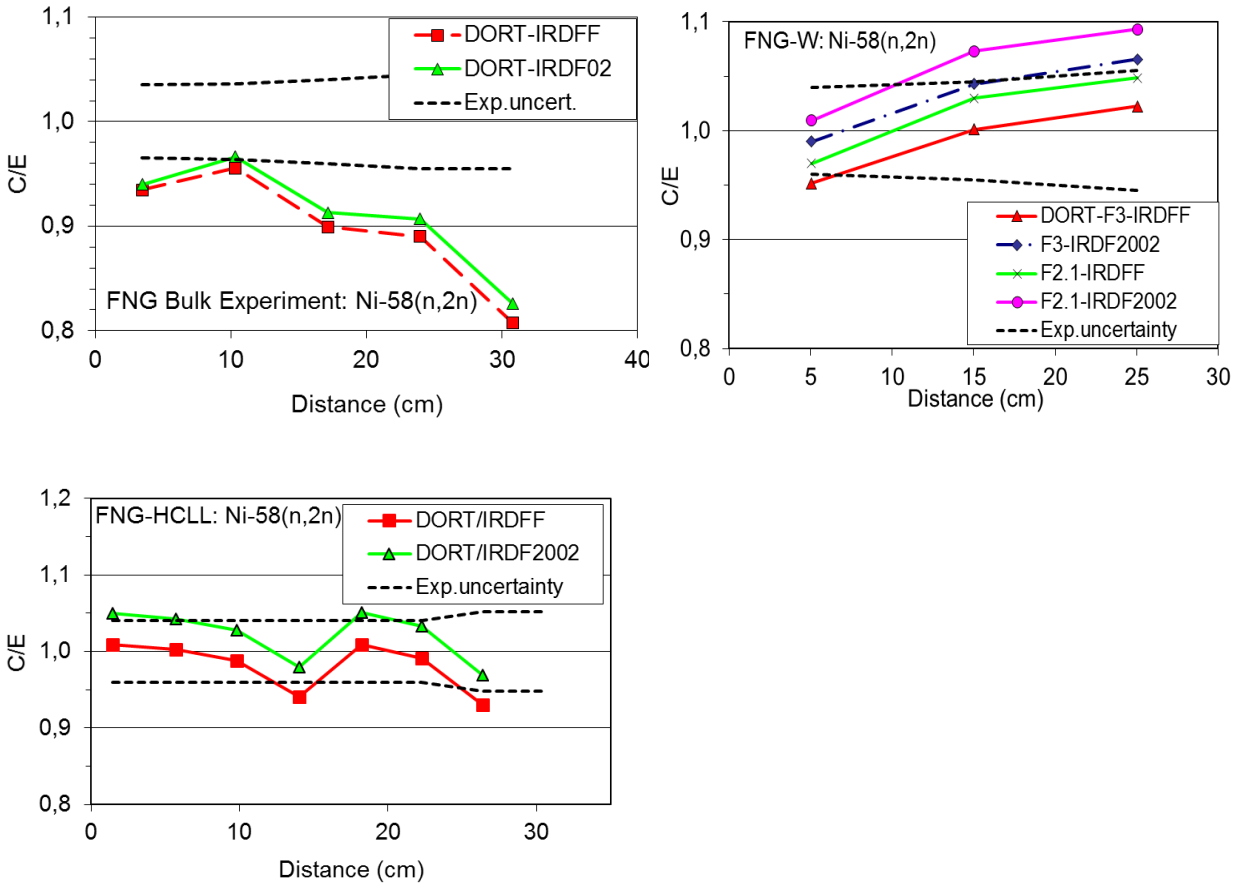


Fig. 1: ^{58}Ni(n,2n) detector cross section in IRDF evaluations: C/E detector responses for several FNG benchmarks based on calculations with different libraries and DORT deterministic computer code. FENDL-2.1 (F2.1), -3 (F3) and EFF3 cross sections were used in the analyses. Dashed lines delimit the ± 1 σ standard deviations of the measurements.

Except for the ^{55}Mn(n,γ) small differences were observed compared to the IRDF-2002 library (1% to 4%) with no clear conclusive indications concerning eventual improvements due to the uncertainties linked to the measurement and modelling uncertainties. Further experimental verifications are recommended for the ^{55}Mn(n,γ) reaction rates. We recommend to consider including the ^{7}Li(n,t) cross sections in a future IRDF evaluation (sum of MT > 52).

Benchmarking of IRDFF against 14 MeV Neutron Experiments,

M. Angelone, D. Flammini, R. Villari, S. Loreti, M. Pillon

ENEA UT- FUS, ENEA C.R. Frascati, Frascati (Rome) Italy

1. Introduction

During the first Research Coordination Meeting (RCM) on testing and improving the IAEA IRDFF file, it was agreed that ENEA Frascati participates in the testing of the IRDFF file. The Research Agreement No. 18207 was signed as part of CRP-F4.10.31

The ENEA proposal was mainly devoted to the testing of the IRDFF cross sections at and around 14 MeV, the neutron energy range of interest to fusion neutronics. The 14 MeV Frascati neutron generator (FNG) was proposed for the experimental test of IRDFF.

The ENEA proposal was based upon two main activities:

1) Test of IRDFF via re-analysis (using MCNP5 and FENDL-2.1 & 3 cross section libraries) of previous benchmark experiments carried out at the 14 MeV Frascati neutron generator (FNG) and comparison with measurements and results (C/E) already obtained for these benchmarks using IRDF-2002 evaluation;

2) Measurement, at FNG, of Reaction Rates for some high threshold reactions in the energy region around 14 MeV and comparison with the MCNP5 simulation (using JEFF 3.1.1 library) employing the FNG source routine developed at ENEA-Frascati.

2. Re-analysis of benchmark experiments performed using 14 MeV neutrons

For the purpose of the present activity the two mock-ups experiments concerning the European ITER-TBM modules performed at FNG with 14 MeV neutrons were re-analysed:

a) Helium cooled liquid lead (HCLL);
b) Helium cooled pebble bed (HCPB).

The HCLL mock-up is made of Lithium-lead bricks [1]. In the experiment performed at FNG the reaction rates (as well as the tritium production in Li_2CO_3 pellets) of selected activation materials were measured along the horizontal axis.

The HCPB mock-up is made of Berillium [2], the experiment was performed at FNG and, as above, the reaction rates (and tritium production) of selected activation materials were measured along the horizontal axis. For the purpose of present IRDFF validation only reaction rates were analysed.

In both cases, the experiment was analysed with MCNP5 and FENDL-2.1 & 3 libraries while the dosimetric cross sections used to get the reaction rates was the IRDF-2002 file (version 2005).

For this work, the MCNP inputs prepared for the two mock-ups were run without any modification but just adding the new IRDFF_v1.05 cross-sections for scoring the reaction rates of interest.

The new results were compared to those already published and obtained using IRDF-2002 (v. 2005).

Fig. 1 displays the comparison for the HCLL mock-up while Fig. 2 shows the results for the HCPB experiment.

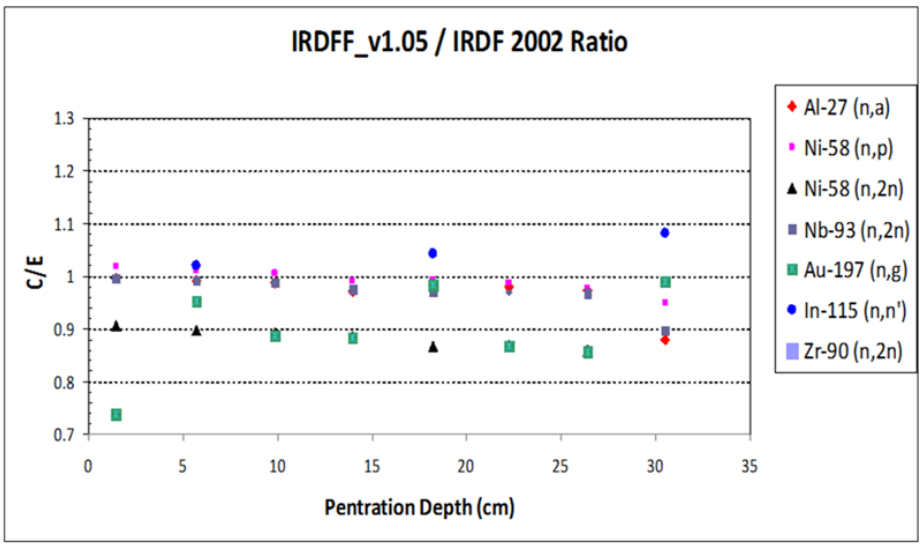


Fig. 1: HCLL - Comparison between results obtained with IRDF-2002 and IRDFF_v1.05 when using FENDL-2.1 library for neutron transport.

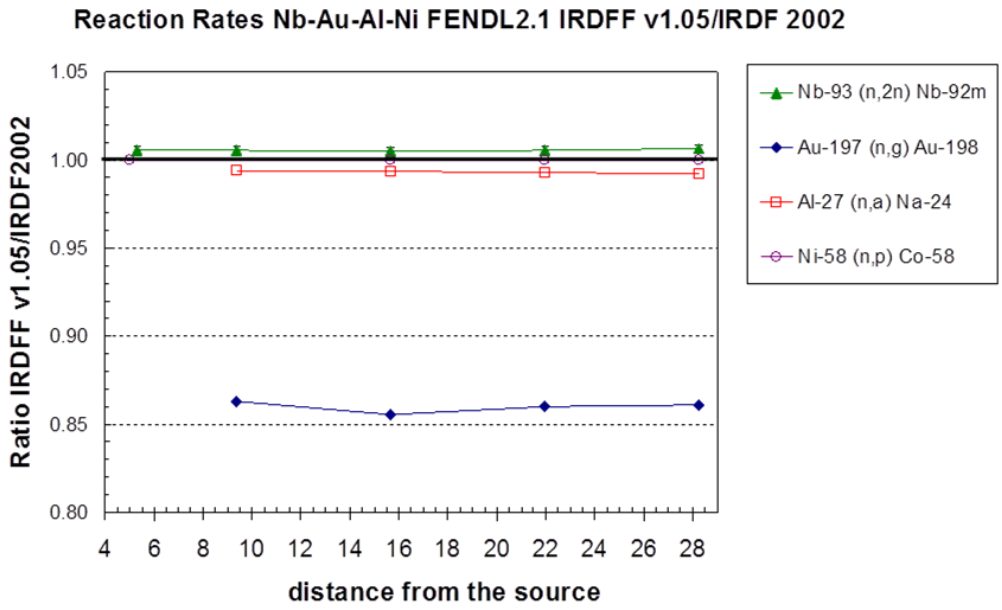


Fig. 2: HCPB - Comparison between results obtained with IRDF-2002 and IRDFF_v1.05 when using FENDL-2.1 library.

The results reported in Fig. 1 and 2 show that there is a very good agreement between IRDF-2002 and IRDFF_V1.05 dosimetric libraries for threshold reactions (differences within 2%). However a problem concerning $^{197}Au(n,\gamma)^{198}Au$ is clearly pointed out as a difference of 15-20% is found, with IRDFF_v1.05 underestimating the experimental results and thus the IRDF2002 results. This is independent from the used transport cross sections library. Furthermore, it was found that the difference between the two libraries depends upon the foil thickness. For void cells or Au foils thinner than 10 microns the difference is not noticeable, while it is increasing with the foil thickness (in Fig. 1 and 2 three location points show always C/E close to 1). These positions were hosting foils 19 microns thick, while the others where hosting Au foils 30 microns thick.

The reason for this difference is under investigation but it could rely upon the used version of the IRDF-2002 file which was a release of 2005.

3. Measurement of Reaction Rates

An experimental campaign was performed at the 14 MeV Frascati Neutron Generator (FNG) to measure the Reaction Rates of some high threshold reactions:

$^{115}In(n,n')^{115}In$ Thr. = 0.5 MeV
$^{58}Ni(n,p)^{58}Co$ Thr. = 1 MeV
$^{27}Al(n,\alpha)^{24}Na$ Thr. = 4.2 MeV
$^{197}Au(n,2n)^{198}Au$ Thr. = 8.2 MeV
$^{93}Nb(n,2n)^{92}Nb$ Thr. = 8.90 MeV
$^{90}Zr(n,2n)^{89}Zr$ Thr. = 12.0 eV
$^{58}Ni(n,2n)^{57}Ni$ Thr. = 12.6 MeV

These materials are routinely used for experiments in fusion neutronics so the test of the IRDF and IRDFF files (routinely used in the FNG experiments as dosimetric cross section files) is very relevant for validating of the benchmarks/mock-ups results.

The used foils were located around the FNG target at selected angles to take an advantage of the Energy vs. Angle variation of the DT beam-target reaction (Fig. 3).

The measurements were carried out in the energy range 13.5 MeV – 15.1 MeV at a distance (radius) of 15 cm from the target. The deuteron beams energy was 260 keV. The foils were located in the horizontal mid-plane facing the target. One HPGe 60% relative efficiency detector, absolutely calibrated (± 3.0%), was used for activation measurements. The nuclear data used for the γ-ray spectra analysis were taken from [3].

The measured Reaction Rates were normalized to the absolute neutron yield measured by the associated alpha particle monitor (Si diode) calibrated with uncertainty ±3.0% and routinely used as FNG absolute neutron yield monitor. The experimental total uncertainty is typically in between ± 4.0% and ± 4.5% (1 σ level).

The experiments (E) were simulated by MCNP5 using JEFF-3.1.1 library. The geometry of the irradiation set-up was accurately modelled and the reaction rates were calculated (C) using the IRDF-2002 (version 2005) and IRDFF (v-1.03 and v-1.05) dosimetry files. The results in term of the C/E ratio are reported in Figs. 4 and 5.

Tables 1a and b report the comparison between IRDF-2002 and IRDFF_v1.05. As far as the comparison between IRDFF_v1.05 and 1.03 is concerned, the differences for the measured high threshold reactions are negligible (< ± 2%).

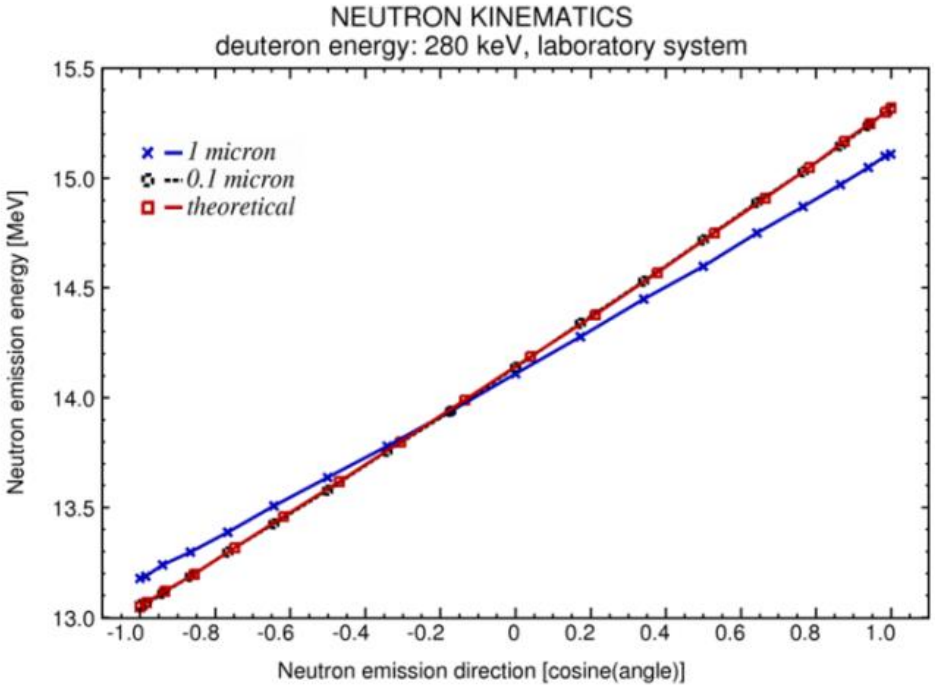


Fig. 3: Energy vs. angle for neutrons emitted by the beam-target reaction.

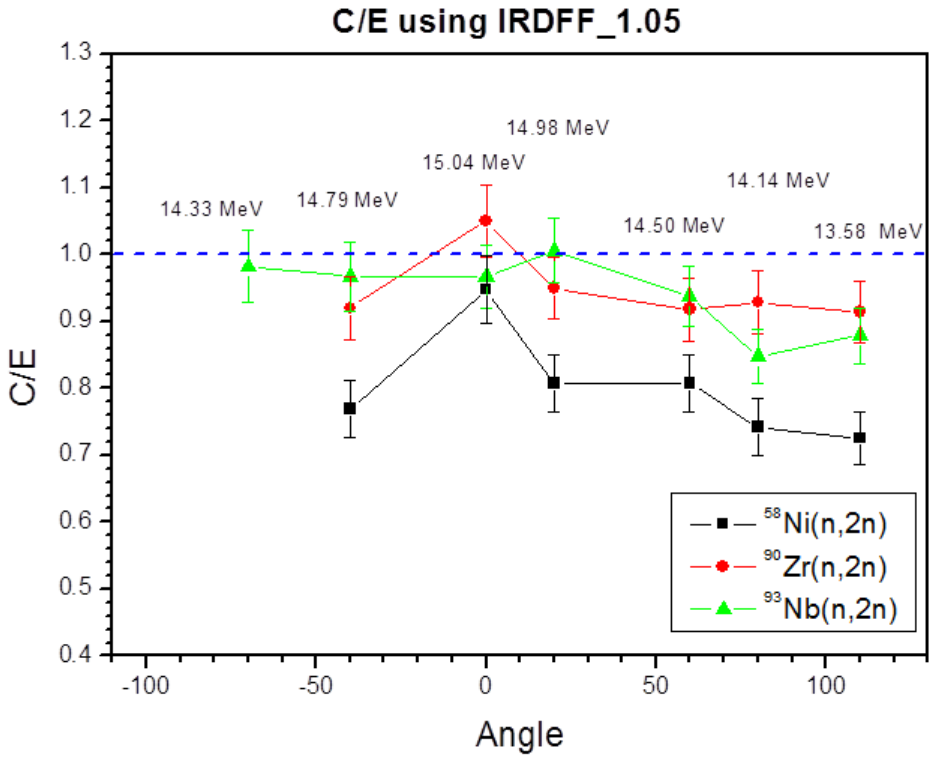


Fig. 4: C/E obtained with IRDFF_v1.05 for ^{58}Ni(n,2n), ^{90}Zr(n,2n) and ^{92}Nb(n,2n) reactions.

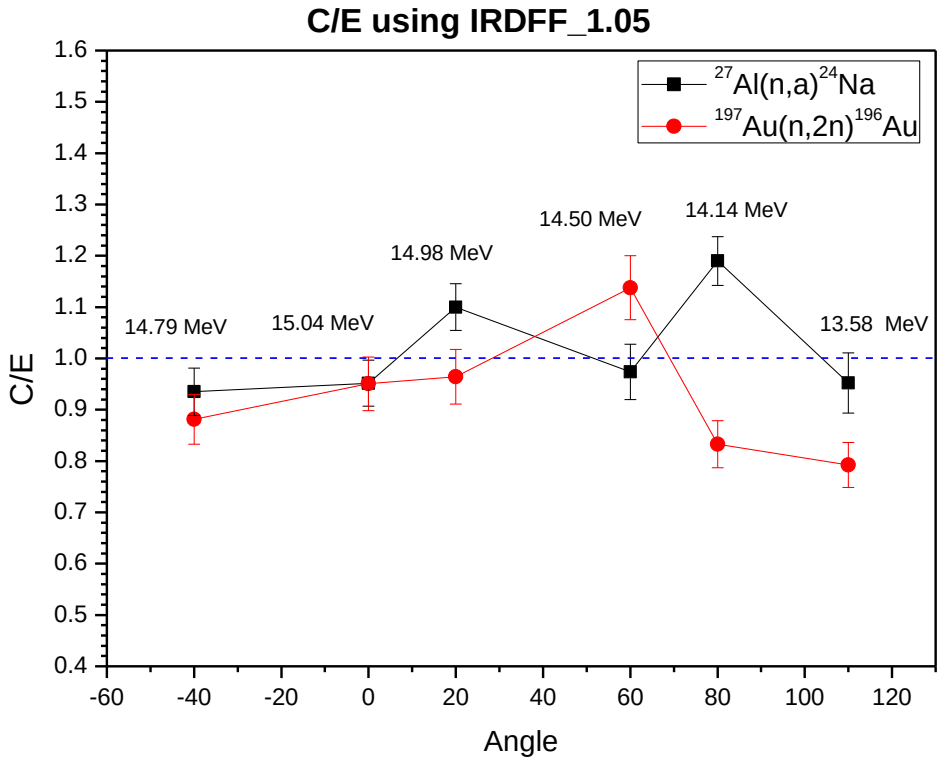


Fig. 5: C/E obtained with IRDFF_v1.05 for ^{27}Al(n,a) and ^{197}Au(n,2n) reactions.

Table-1a: IRDFF_v1.05/IRDF-2002 ratio for the measured reaction rates.

Experimental Position	Nb-93 (n,2n)	Err	Al-27 (n,a)	Err
-40°	**0.996**	0.010	**0.996**	0.011
0°	**1.006**	0.010	**1.004**	0.010
20°	**1.012**	0.010	**1.012**	0.010
60°	**1.014**	0.011	**1.010**	0.011
80°	**0.998**	0.012	**0.995**	0.012
110°	**1.008**	0.011	**1.009**	0.011

Table-1b: IRDFF_v1.05/IRDF-2002 ratio for the measured reaction rates.

Experimental Position	In-115(n,n')	Err	Ni-58 (n,2n)	Err	Ni-58 (n,p)	Err	Zr-90 (n,2n)	Err
-40°	**1.007**	0.015	**0.950**	0.015	**0.990**	0.015	**0.983**	0.011
0°	**1.010**	0.015	**0.993**	0.014	**1.014**	0.014	**1.004**	0.011
20°	**1.014**	0.015	**0.999**	0.014	**1.026**	0.014	**1.005**	0.011
60°	**1.009**	0.015	**0.952**	0.015	**0.999**	0.015	**0.982**	0.011
80°	**0.997**	0.017	**0.926**	0.017	**0.975**	0.017	**0.976**	0.012
110°	**0.995**	0.014	**0.947**	0.015	**1.006**	0.015	**1.016**	0.011

Conclusion

The re-analysis of the HCLL and HCPB ITER-TBM mock-ups using FENDL-2.1 & IRDF 2002 or IRDFF_v1.05 and FENDL-3 & IRDF-2002 or IRDFF_v1.05 does not show noticeable differences for THRESHOLD reactions (within ± 2%).

A large difference (≈ 15%) is observed for ^{197}Au(n,γ)^{198}Au reaction when using IRDFF_v1.05 respect to IRDF-2002 (version 2005) results. The reason for this is under investigation, among the possible explanations there is the possibility that a wrong version of IRDF-2002 was used. However, it worth to note that for "void or infinite dilution" as well as thin foils (≤ 10 μm) the agreement between the IRDF-2002 and IRDFF_v1.05 files is still good both for the HCLL and HCPB experiments.

As far as the reaction rates are concerned, the comparison among the calculated Reaction Rates using two different versions of IRDFF (v. 1.03 & v. 1.05) does not show notable differences (within ±2% max). The comparison between IRDF-2002 and IRDFF_v1.05 does not show the differences to be mentioned since these differences are always < 1-2% for the all investigated reaction rates.

More difficult is the comparison of the C/E values (using IRDFF_v1.05) since this comparison is largely affected by the FNG target structure which is not symmetric respect to the beam axis.

However, the results are satisfying for ^{93}Nb(n,2n) and ^{27}Al(n,α), still within ± 10% for ^{90}Zr(n,2n), ^{197}Au(n,2n) as well as for ^{58}Ni(n,p) and ^{115}In(n,n'). For ^{58}Ni(n,2n) the underestimation in the calculation is in the range 10 ÷ 15%.

References

1. P. Batistoni et. al., Fus. Eng. Des. 85 (2010) pp. 1675
2. P. Batistoni et. al., Fusion Eng. Des. 82 (2007) pp.2095-2104
3. J.H. Board, W.J. Zijp, H.J. Nolthenius, *Nuclear data guide for reactor neutron metrology*, Kluwen Academic Publisher, The Netherlands, 1989. See also ECN-Report ECN-89-027 (1989)

Preliminary results of IRDFF benchmark test at JAEA/FNS,

C. Konno

Japan Atomic Energy Agency, Tokai-mura, Naka-gun, Ibaraki-ken 319-1195, Japan

1. Introduction

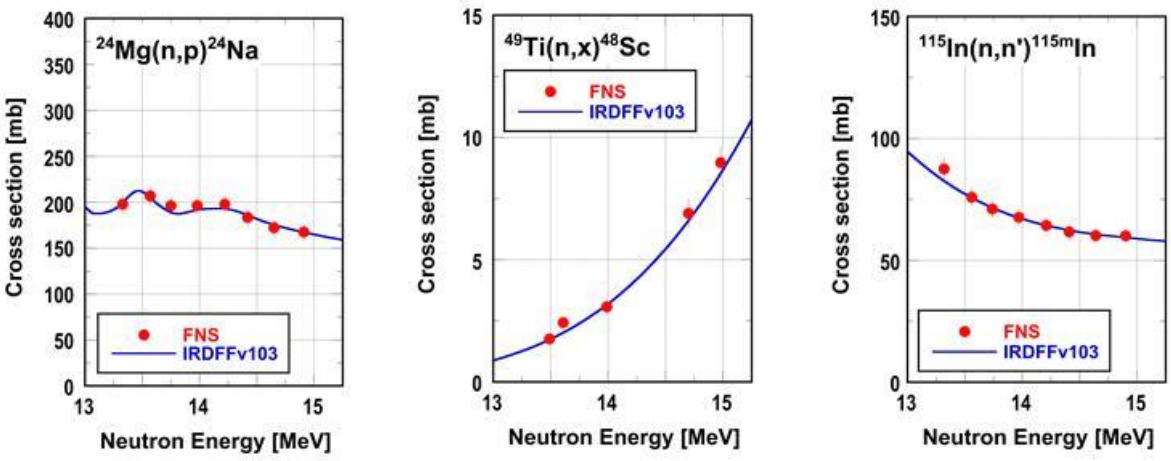
At the 1st RCM of CRP we proposed the following items.

1) Comparison between IRDFF and cross section data around 14 MeV previously measured at JAEA/FNS.
2) Measurement of cross section data such as $^{nat}Ti(n,x)^{46}Sc$ around 14 MeV with natural foils.
3) Reaction rate measurement inside some experimental assemblies, neutron spectra in which are well specified in our benchmark experiments, such as graphite and Li_2O.

However we skip the second work because the first work covers the second one and we have to complete our work before FNS shutdown. The detailed study is in progress. Preliminary results based on IRDFF v. 1.03 are presented here.

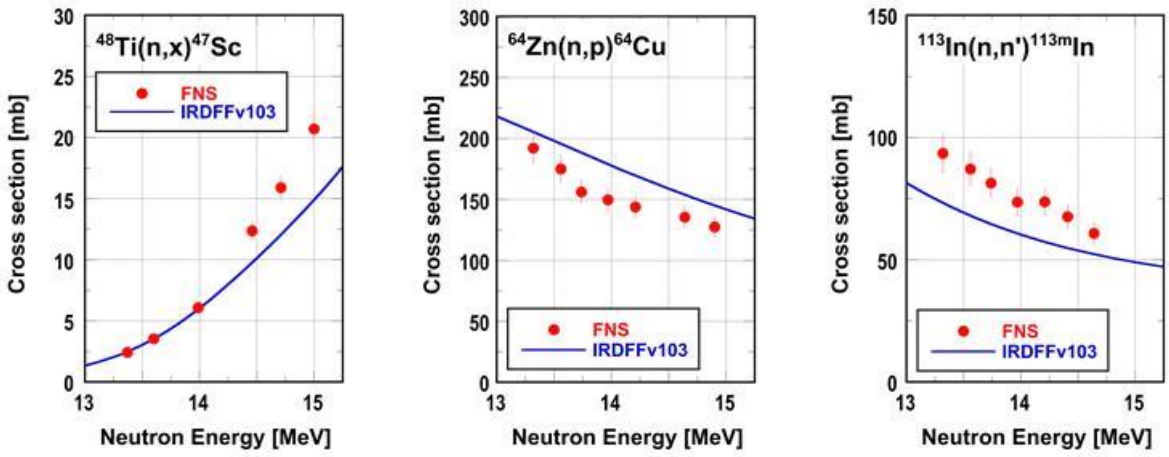
2. Comparison between IRDFF and activation cross-section data measured previously at FNS

More than 20 years ago we measured activation cross-section data of more than 200 reactions around 14 MeV at FNS. IRDFF and our measured activation cross-section data for 34 threshold reactions in IRDFF v. 1.03 were compared from 13 to 15.25 MeV. The agreement between IRDFF and our measured activation cross-section data was good for the most reactions as shown in Fig. 1, though it was not so good for $^{48}Ti(n,x)^{47}Sc$, $^{63}Cu(n,2n)^{62}Cu$, $^{64}Zn(n,p)^{64}Cu$ and $^{113}In(n,n')^{113m}In$ which are shown in Fig. 2.



(a) $^{24}Mg(n,p)^{24}Mg$ reaction (b) $^{49}Ti(n,x)^{48}Sc$ reaction (c) $^{115}In(n,n')^{115m}In$ reaction

Fig. 1: Comparison of IRDFF and cross-section data measured at JAEA/FNS



(a) ^{48}Ti(n,x)^{47}Sc reaction (b) ^{64}Zn(n,p)^{64}Cu reaction (c) ^{113}In(n,n')^{113m}In reaction

Fig. 2 : Comparison of IRDFF and cross-section data measured at JAEA/FNS

3. IRDFF benchmark experiments at FNS

A lot of nuclear data benchmark experiments have been carried out for 30 year at FNS. The graphite and Li_2O experiments showed much better agreement between measured and calculated data in the experiments. This means that the neutron characteristics are well specified with the calculation. The neutron spectra in the graphite and Li_2O assemblies are very different; that in the graphite assembly has not only DT neutrons but also low energy neutrons with thermal neutron peak, while that in the Li_2O assembly is very hard and has no thermal neutron peak, which is shown in Fig. 3. Thus we selected the graphite and Li_2O assemblies for benchmarking IRDFF.

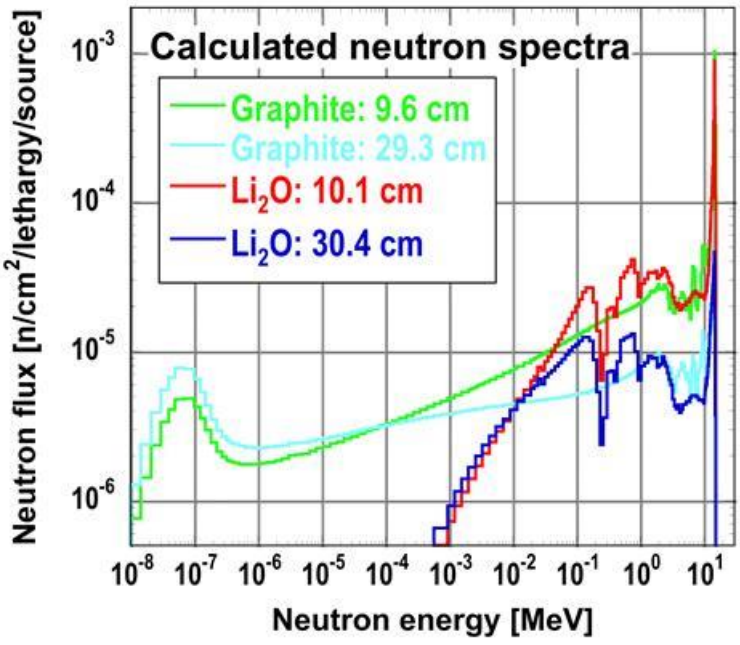


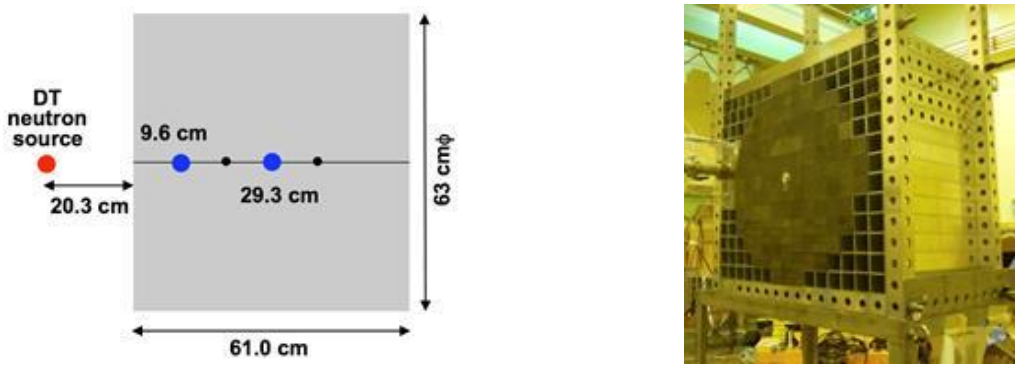
Fig. 3 : Calculated neutron spectra at the measuring positions in graphite and Li_2O assemblies.

3.1. Graphite experiment with DT neutron source

We used the same graphite assembly as our previous graphite assembly; a pseudo-cylindrical graphite assembly of 63 cm in equivalent diameter and 61 cm in thickness as shown in Fig. 3. 28 foils were set at the depths of 9.6 and 29.3 cm in order to measure reaction rates of the reactions in IRDFF. The

standard foils of Nb, In, and Au every ~ 10 cm were also set in order to check the neutron field in the graphite assembly. We irradiated the graphite assembly with DT neutron source of ~ 1.5 10^{11} n/sec for 5 hours twice in March 2014, because we had to set 28 foils at the same positions. Reaction rates of 40 reactions (half-lives of produced radio isotopes are longer than ~ 40 min.) in IRDFF v1.03 (77 reactions) were measured with Ge detectors. The experimental errors of the most measured reaction rates were less than 10%.

This experiment was analyzed by using a Monte Carlo neutron transport code MCNP5-1.40 with a nuclear data library ENDF/B-VII.1. The thermal scattering law data S(α,β) for graphite in ENDF/B-VII.1 was also used in the analysis. IRDFF-v1.03 was adopted as the cross section data for the reaction rate calculation. The reaction rates for the capture reactions with thicker foils were calculated by using the cell flux (F4) tally in order to correct neutron self-shielding. We did not carry out the error estimation for the calculated reaction rates with IRDFF covariance data yet. The measured reaction rates were compared with the calculated reaction rates, which suggested if IRDFF data was good or not. Most of the 40 measured reaction rates agreed with the calculated ones within 10%, while the measured reaction rates of the ^{181}Ta(n,γ)^{182}Ta and ^{204}Pb(n,n')^{204m}Pb did within 20%, which is shown in Fig. 4.



(a) Schematic view (b) Photo

Fig. 3: Graphite assembly.

3.2. Li_2O experiment with DT neutron source

We used a rectangular Li_2O assembly of 65.7 cm in width, 65.7 cm in height and 60.7 cm in thickness as shown in Fig. 5. We carried out the similar experiment with the graphite one last November. Note that the neutron spectra are harder than those in the graphite experiment. Reaction rates of 30 reactions in IRDFF (77 reactions) were measured. Reaction rates of several reactions measured in the graphite experiment were not finalized yet. Experimental error estimation and analysis method are similar with those in the graphite experiment.

Most of the 30 measured reaction rates agreed with the calculated ones within 10%, while the measured reaction rates of the ^{48}Ti(n,p)^{48}Sc and ^{49}Ti(n,x)^{48}Sc, and ^{51}V(n,α)^{48}Sc did within 20%, which is shown in Fig. 6.

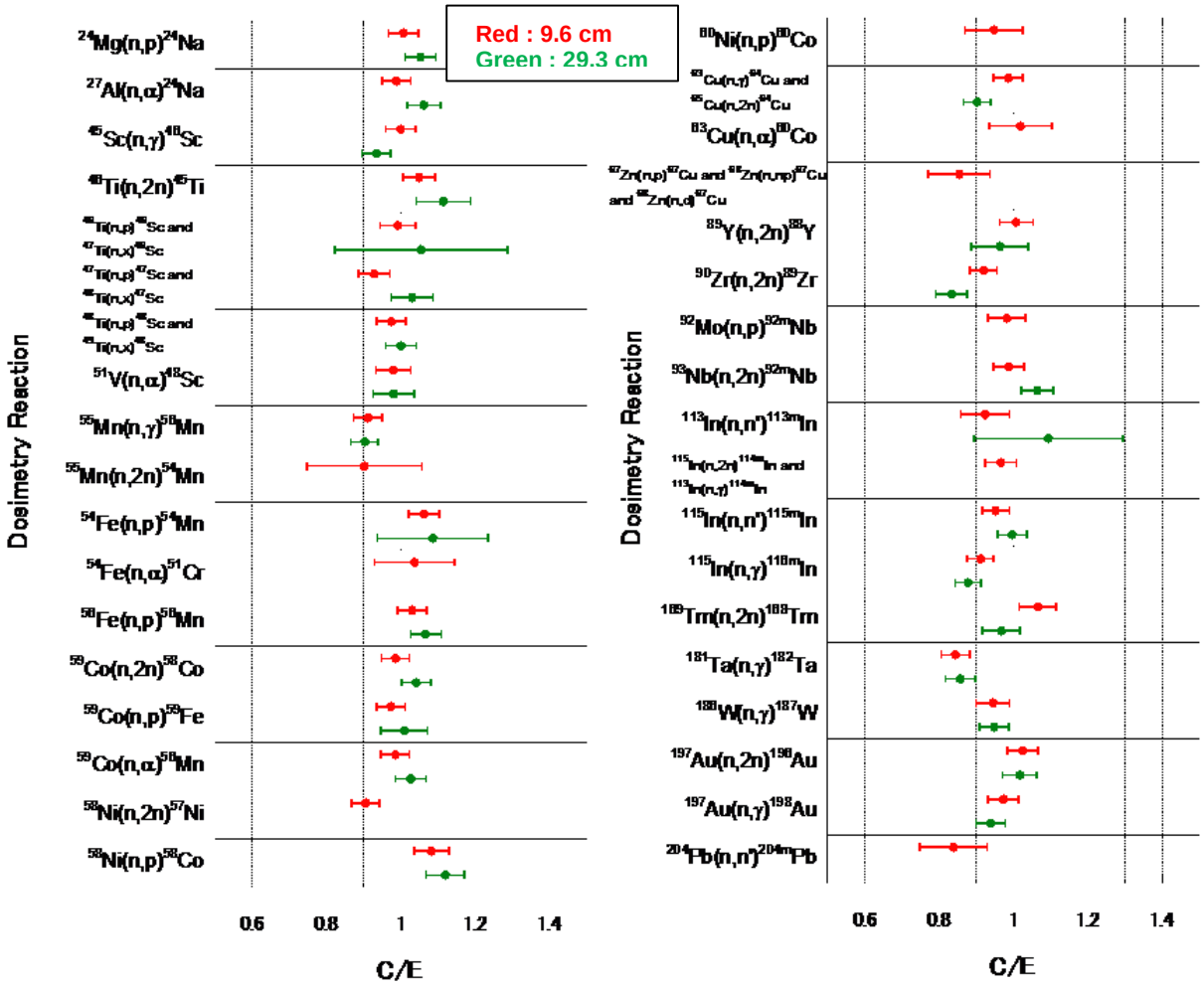


Fig. 4: Ratios of calculation to experiment (C/E).

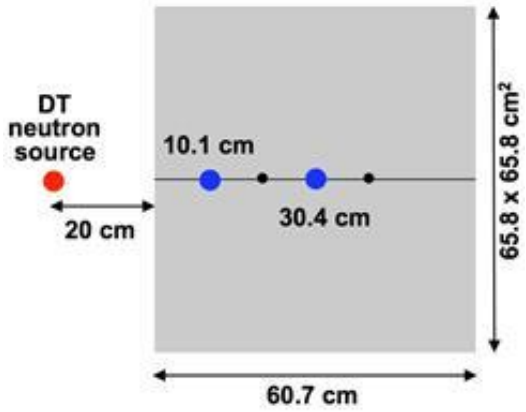


Fig. 5: Li_2O assembly (schematic view).

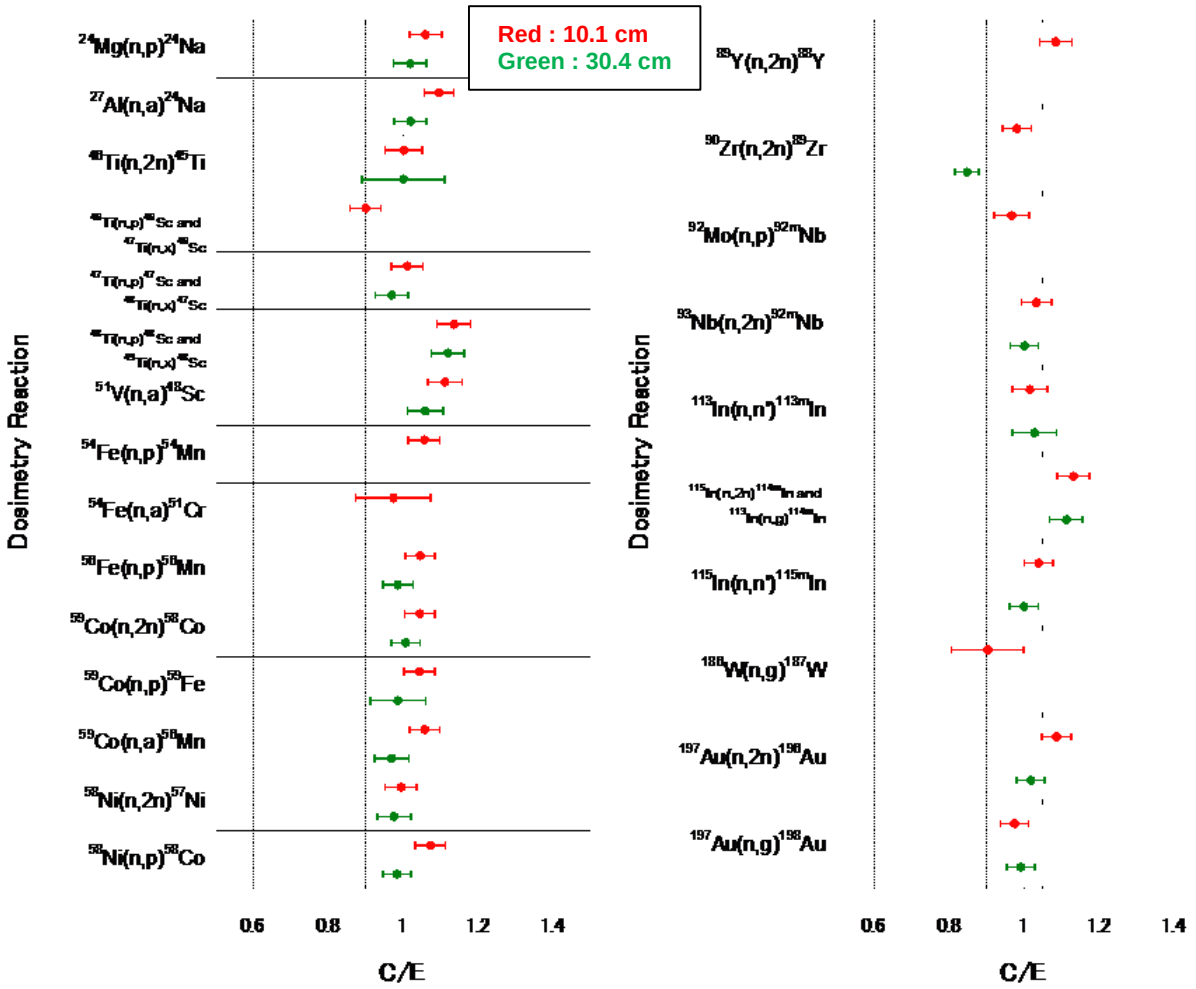


Fig. 6: Ratios of calculation to experiment (C/E).

4. Summary

Based on our work plan we carried out the following items.

Cross section comparison between IRDFF and our cross section data measured previously

Graphite experiment with DT neutrons

Li_2O experiment with DT neutrons.

It was confirmed that generally IRDFF had no big problems. We will check reasons of larger discrepancy between measured and calculated data and estimate calculation errors based on IRDFF error data. If necessary, additional experiments will be carried out before FNS shutdown; experiments for reactions, where half-lives of produced radio isotopes are shorter than ~ 40 min.

Evaluation of ^{238}U(n,γ) and ^{238}U(n,2n) Reactions Cross Sections, including Analysis of Microscopic and Integral Experimental Data,

K. I. Zolotarev

Institute of Physics and Power Engineering, Obninsk, Russian Federation

Uranium is the basic fuel element of fission reactors. For this reason, the data on the cross sections of neutron reactions on isotopes of uranium have prime importance. From the view of accuracy required to neutron data on the isotopes of uranium, the most important are the reactions (n,f), (n,γ), (n,el), (n,inl), (n,2n).

Currently, the most representative neutron data for the isotope ^{238}U are given in the US Library of ENDF/B-VII.1. Nevertheless, verification carried out for the ^{238}U(n,γ)^{239}U and ^{238}U(n,2n)^{237}U reaction cross sections from ENDF/B-VII.1 library showed that calculated averaged cross sections for the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra do not agree so well with the relevant experimental data.

The purpose of this contribution to the IAEA CRP on IRDFF is the analysis of all available microscopic and integral experimental data and new re-evaluation of the ^{238}U(n,γ)^{239}U and ^{238}U(n,2n)^{237}U reactions excitation functions in the neutron energy ranges 20 keV – 20 MeV and from threshold to 30 MeV, respectively.

At the first step of evaluation all experimental data were carefully analyzed. During this procedure all experimental data, when it was possible, were corrected to the new recommended cross section data for monitor reaction used in the measurements and to the new recommended decay data. The needed information about standards used for correction microscopic experimental data under investigation is given in the Table 1.

Table 1. Data used as standards for correction of microscopic experimental cross sections.

Monitor Reaction	Cross sections used as standards	Half-life for residual nucleus	Radiation Mode and Energy	Emission Probability per decay
^{1}H(n,n)^{1}H	Carlson+ [1]			
^{10}B(n,α)	Carlson+ [1]			
^{27}Al(n,α)^{24}Na	Zolotarev [3]	14.9590 (12) h	Gamma 1368.63 keV	0.999936(15) [10]
^{56}Fe(n,p)^{56}Mn	Zolotarev [4]	2.5789 (1) h	Gamma 846.754 keV Gamma 1810.72 keV	0.9887 (3) [11] 0.2719 (79) [11]
^{65}Cu(n,2n)^{64}Cu	Zolotarev [5]	12.700 (2) h	Beta+ 653.1 keV Beta– 578.7 keV Gamma 511 keV Gamma 1345.77 keV	0.1740 (22) [12] 0.390 (4) [12] 0.348 (4) [12] 0.00473 (10) [12]
^{93}Nb(n,2n)^{92m}Nb	Zolotarev [6]	10.15 (2) d	Gamma 934.44 keV	0.9907 (4) [11]
^{115}In(n,n')^{115m}In	Zolotarev+ [7]	4.486 (4) h	Gamma 336.24 keV	0.458 (22) [13]
^{115}In(n,γ)^{116m}In	Zolotarev+ [8]	54.29 (17) m	Gamma 1293.56 keV	0.848 (12) [14]
^{127}I(n,γ)^{128}I	Zolotarev+ [9]	24.99 (2) h	Gamma 442.90 keV	0.169 (17) [11]
^{197}Au(n,γ)^{198}Au	Carlson+ [1]	2.6947 (3) d	Gamma 411.80 keV	0.9562 (10) [15]
^{169}Tm(n,2n)^{168}Tm	Zolotarev [6]	93.1 (2) d	Gamma 198.25 keV	0.524 (16) [11]
^{235}U(n,f)	Carlson+ [1]			
^{235}U(n,γ)^{236}U	ENDF/B-VII.1 [2]			
^{235}U(n,abs)	ENDF/B-VII.1 [2]			
^{238}U(n,f)	Carlson+ [1]			

Comment: for beta transition the end-point value of energy is given**.**

The needed information about standards used for correction integral experimental data under investigation given in the Table 2.

Table 2. Data used as standards for correction of integral experimental cross sections measured in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra.

Monitor Reaction	Cross section used as standard, mb	Half-life for residual nucleus	Radiation Mode and Energy	Emission Probability per decay
^{27}Al(n,α)^{24}Na	0.7007±1.28% [16] U 1.016±1.28% [17] Cf	14.997 (12)H	Gamma 1368.63 keV	0.999936(15) [10]
^{46}Ti(n,p)^{46}Sc	11.51±1.70% [16] U	9.458 (12) M	Gamma 843.76 keV Gamma 1014.44 keV	0.718 (4) [11] 0.280 (4) [11]
^{54}Fe(n,p)^{54}Mn	78.09±1.50% [18] U	2.5789 (1) H	Gamma 846.754 keV Gamma 1810.72 keV	0.9887 (3) [11] 0.2719 (79) [11]
^{58}Ni(n,p)^{58}Co	108.2±1.30% [16] U	78.82 (3) D	Gamma 511 keV Gamma 810.78 keV	0.298 (4) [11] 0.99448 (8) [11]
^{115}In(n,n')^{115m}In	187.8±1.23% [16] U	4.486 (4) H	Gamma 336.24 keV	0.458 (22) [13]
^{238}U(n,f)	325.7±1.64% [17] Cf			

Comments: Symbol "U" – means ^{235}U thermal fission neutron spectrum,
Symbol "Cf" – means ^{252}Cf spontaneous fission neutron spectrum.

Decay data for residual nuclear ^{239}U and its daughter ^{239}Np were taken from [19]. Decay data for residual nuclear ^{237}U were taken from [20].

Evaluation of the ^{238}U(n,γ)^{239}U and ^{238}U(n,2n)^{237}U reactions excitation functions was carried out by means of the generalized least-squares method within the PADE-2 code [21].

Database for re-evaluation the excitation function of the ^{238}U(n,γ)^{239}U reaction in the neutron energy range 0.02 - 20 MeV was formed by analysis of all the available experimental microscopy data containing information about the reaction cross section in this energy range. In total, there was analyzed 57 works from EXFOR and original publications.

Uncertainties in the re-evaluated data of the ^{238}U(n,γ)^{239}U reaction cross section in the neutron energy range 0.02 - 20 MeV are in the range from 0.74 to 13.80%. The most accurately evaluated the cross section values are in the energy range 0.02 - 2.00 MeV, where the error in the data does not exceed 2%. The relatively large error from 3.28 to 13.8% corresponds to cross sections above 3 MeV. This is due to large errors in the experimental data.

The re-evaluated excitation function for the ^{238}U(n,γ)^{239}U reaction in the neutron energy range 0.02 – 0.3 MeV is shown in Fig. 1 in comparison with the equivalent data from the ENDF/B-VII.1 library and corrected experimental data. The same information in the energy range 0.3 – 20 MeV is shown in Fig. 2.

Experimental data on the integral cross sections measured in the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra are commonly used for testing of evaluated radiative capture excitation functions the of neutrons in the energy range 0.01 - 2.5 MeV. Unfortunately, the experimental data on the cross section for the ^{238}U(n,γ)^{239}U reaction in the ^{252}Cf spontaneous fission neutron spectrum are absent. The integral cross section of the ^{238}U(n,γ)^{239}U reaction in the ^{235}U thermal fission neutron spectrum was measured by Fabry and De Coster by activation method at the experimental facility of BR-1 reactor in Mol, Belgium (CEN, Mol , Belgium) [22].

The integral cross section value of the $^{238}U(n,\gamma)^{239}U$ reaction - $<\sigma>_{U-235}$ = (85 ± 8) mb measured by Fabry and De Coster [22] was renormalized to the new standards. Corrected to the new standards the experimental value of the cross section for the $^{238}U(n,\gamma)^{239}U$ reaction is equal to $<\sigma>_{U-235}$ = (71.69 ± 6.64) mb.

Microscopic experimental data for the $^{238}U(n,2n)^{237}U$ reaction excitation function are given in the 26 works and cover the neutron energy range from 5.28 MeV to 19.0 MeV.

The data base used for the evaluation of the $^{238}U(n,2n)^{237}U$ reaction excitation function in the energies range from threshold to 30 MeV was formed from the experimental cross section data obtained in 19 representative works. The relative shape of excitation function above 19 MeV was taken from the TENDL-2011 library [23]. The absolute cross sections used as input data in the interval 19 - 30 MeV were calculated from TENDL-2011 data by normalizing at 19 MeV to a value of 273.0 mb measured by Veeser and Arthur [24].

Uncertainties in the evaluated the $^{238}U(n,2n)^{237}U$ excitation function range from 1.75% to 31.57%. The highest uncertainty equal to 31.57% characterize data in the interval 6.18 – 6.4 MeV. The smallest uncertainties in the evaluated cross sections 1.75-1.91% are observed in the neutron energy range from 13.5 to 14.5 MeV. The cross sections uncertainties exceed 10% in the energy range 22 - 30 MeV.

The evaluated excitation function for the $^{238}U(n,2n)^{237}U$ reaction in the neutron energy range from threshold to 30 MeV is shown in Fig. 3 in comparison with the equivalent data from ENDF/B-VII.1 library and corrected experimental data. The same information but in the narrow neutron energy range from threshold to 12 MeV is shown in Fig. 4.

Integral experimental data for the $^{238}U(n,2n)^{237}U$ reaction are given in Refs. [25], [26], [27], [28]. Integral cross sections presented in the works [25], [26] were measured in neutron fields with spectra similar to the ^{235}U thermal fission neutron spectrum. Kobayashi et al. [25] measured the integral cross section in the core centre of fast reactor YAYOI at RRI of Kyoto University. Hashimoto et al. [26] measured the integral cross section of $^{238}U(n,2n)^{237}U$ reaction in the core of KUR reactor. Experiments which have been performed in a ^{252}Cf spontaneous fission neutron spectrum are described in the works [27] and [28]. The most representative experimental data in ^{252}Cf spontaneous fission neutron spectrum are obtained by Blinov et al. [28]. All of the integral experimental data were measured by activation method and were corrected to the new standards.

The re-evaluated $^{238}U(n,\gamma)^{239}U$ and $^{238}U(n,2n)^{237}U$ reactions excitation functions have been tested against the corrected experimental data for the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra. The results of testing are given in Tables 3 and 4. Calculated averaged cross sections from the re-evaluated excitation functions [A] are compared with the ENDF/B-VII.1 [B] and experimental data. Numerical data for the ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra were taken from the IRDFF-v.1.05 data file. The values of C/E in the tables - are the ratio of the calculated and measured values of the cross sections. The neutron energy range of 90% response function, where the reaction excitation function is basically tested, is indicated.

Table 3. Calculated and measured averaged cross sections of the $^{238}U(n,\gamma)^{239}U$ reaction in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra.

Type of neutron field	**Averaged cross section, mb**		**90% response function, MeV**	**C/E**
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	72.674 [A] 68.997 [B]	71.69 ± 6.64 [19]	0.084 – 2.5 0.088 – 2.5	1.01373 0.96244
^{252}Cf spontaneous fission neutron spectrum	67.541 [A] 70.535 [B]		0.076 – 2.5 0.076 – 2.6	

The C/E values of C/E obtained for the ^{235}U thermal fission neutron spectrum show that the averaged cross section calculated from the re-evaluated $^{238}U(n,\gamma)^{239}U$ reaction excitation function agree better with the measured cross section $<\sigma>_{U-235}$ than the equivalent data from the ENDF/B-VII.1 library. It should be noted that the relative number of acts of $^{238}U(n,\gamma)^{239}U$ reaction in the region 1.000E-05 eV –

20 keV is equal only to 0.656% at the ^{235}U thermal fission neutron spectrum and to 0.725% at the ^{252}Cf spontaneous fission neutron spectrum.

Table 4. Calculated and measured averaged cross sections of the ^{238}U(n,2n)^{237}U reaction in ^{235}U thermal fission and ^{252}Cf spontaneous fission neutron spectra.

Type of neutron field	Averaged cross section, mb		90% response function, MeV	C/E
	Calculated	Measured		
^{235}U thermal fission neutron spectrum	14.708 [A] 15.466 [B]	18.16 ± 0.93 [25] 18.98 ± 1.57 [26]	6.80 – 11.1 6.80 – 11.2	0.80991 0. 85165
^{252}Cf spontaneous fission neutron spectrum	20.607 [A] 21.363 [B]	20.61 ± 2.02 [28]	6.80 – 11.5 6.80 – 11.6	0.99985 1.03654

The C/E values obtained for the ^{252}Cf spontaneous fission neutron spectrum show that the ^{238}U(n,2n)^{239}U integral cross sections calculated from newly evaluated excitation function agree well with experimental data of Blinov et al. [28]. Integral cross section calculated from the ENDF/B-VII.1 excitation function is a 3.6% higher than experimental value.

The C/E values obtained for the ^{235}U thermal fission neutron spectrum shows a very big discrepancy between experimental and calculated data (Table 4 gives the ratio to the experimental data [25]) A very big discrepancy may be explained that measured cross sections $\langle\sigma\rangle_{U\text{-}235}$ in works [25] and [26] are the total values of two reactions ^{238}U(n,2n)^{237}U and ^{238}U(γ,n)^{237}U.

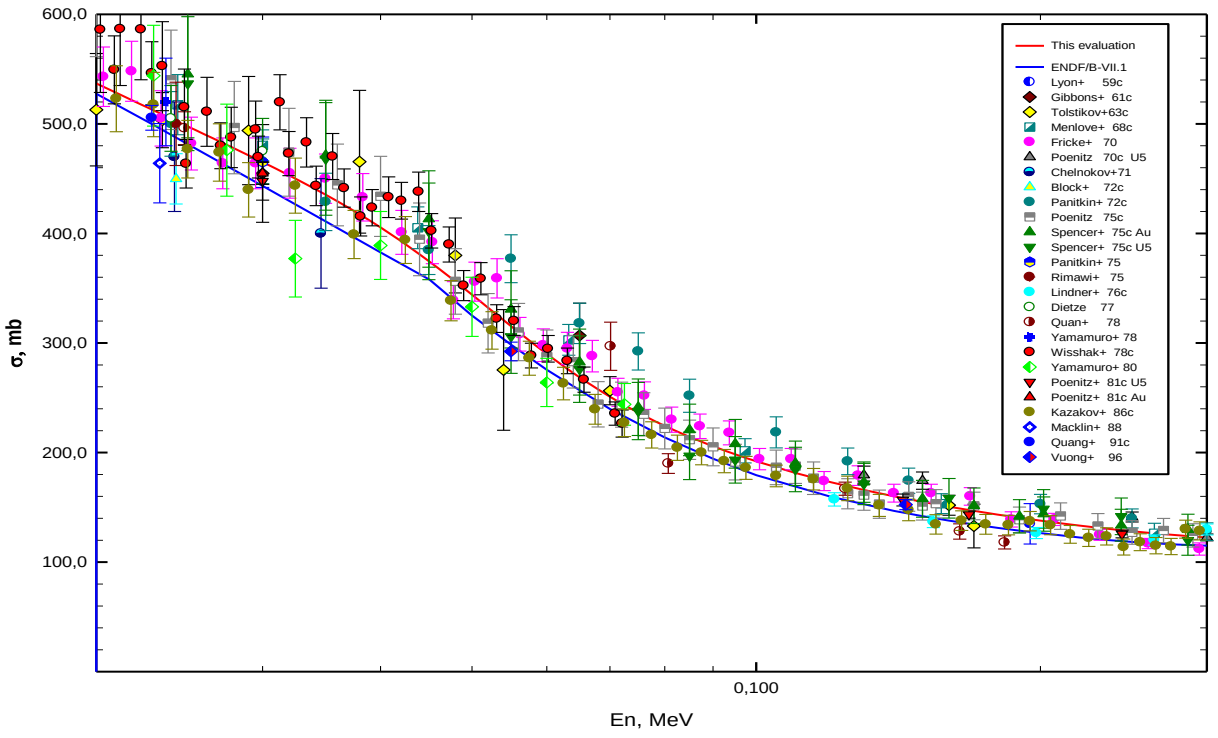


Fig. 1. Re-evaluated the U^{238}(n,γ)U^{239} reaction excitation function in the neutron energies range 0.02 – 0.3 MeV in comparison with ENDF/B-VII.1 and experimental data.

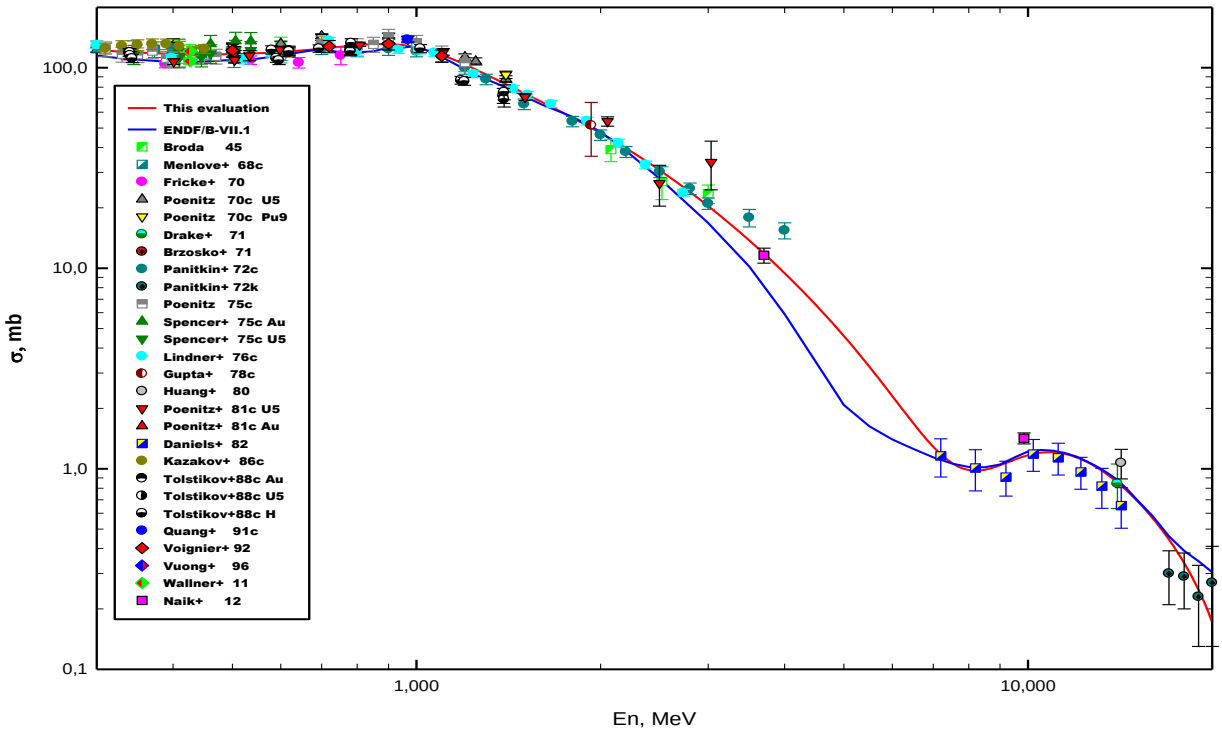


Fig. 2. Re-evaluated the $U^{238}(n,\gamma)U^{239}$ reaction excitation function in the neutron energies range 0.3 – 20 MeV in comparison with ENDF/B-VII.1 and experimental data.

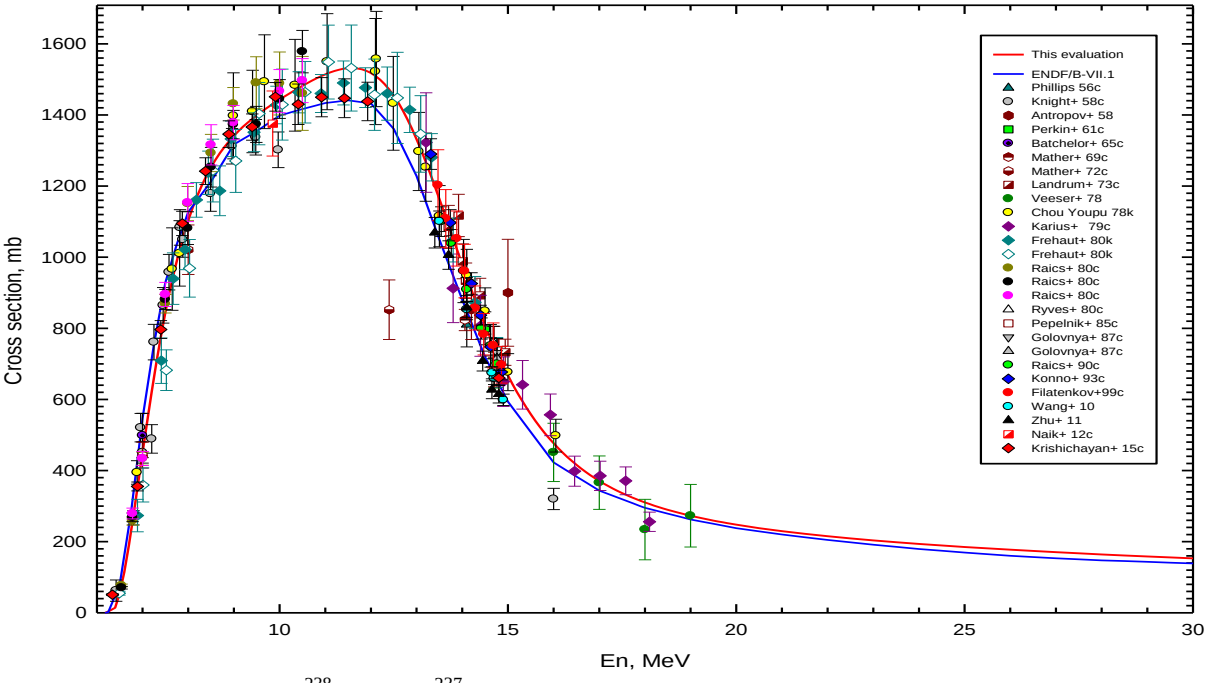


Fig. 3. Re-evaluated the $^{238}U(n,2n)\ ^{237}U$ reaction excitation function in the neutron energies range from threshold to 30 MeV in comparison with ENDF/B-VII.1 and experimental data.

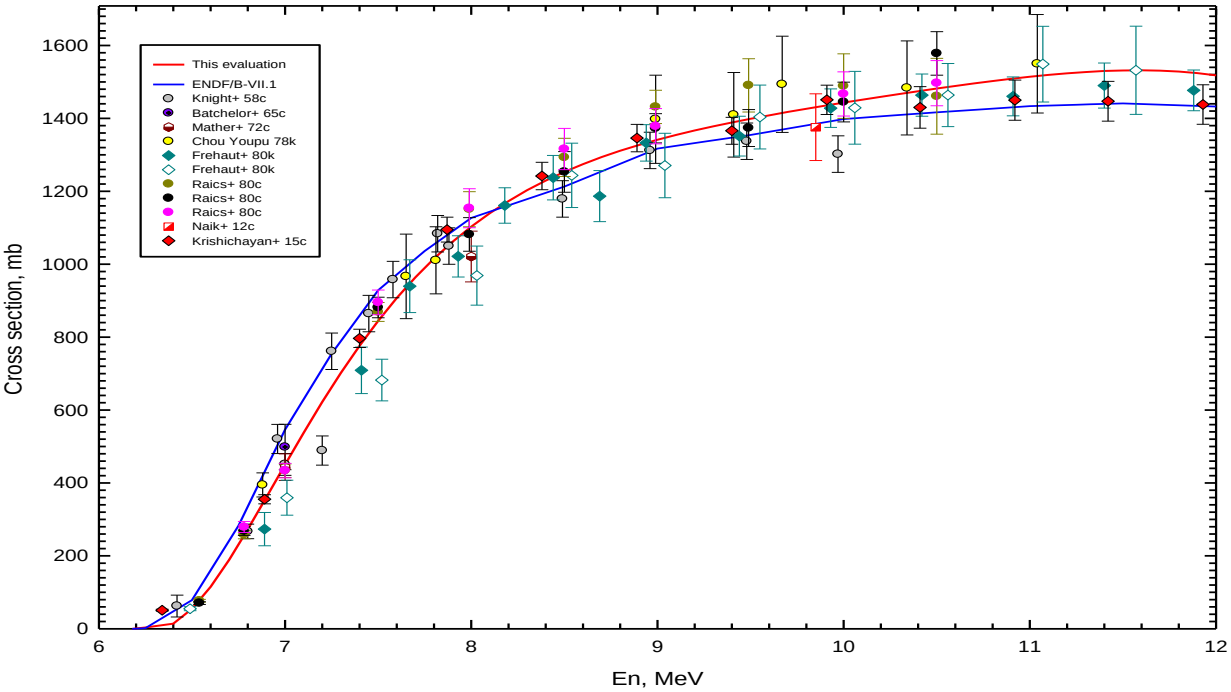


Fig. 4. Re-evaluated the ^{238}U(n,2n)^{237}U reaction excitation function in the neutron energies range from threshold to 12 MeV in comparison with ENDF/B-VII.1 and experimental data.

References

1. A.D. Carlson, V.G. Pronyaev, D.L. Smith, N.M. Larson, Chen Zhenpeng, G.M. Hale, F.-J. Hambsch, E.V. Gai, Soo-Youl Oh, S.A. Badikov, T. Kawano, H.M. Hofmann, H. Vonach, S. Tagesen, *International Evaluation of Neutron Cross Section Standards*, **Nuclear Data Sheets 110 (2009) 3215-3324**.
2. P.G. Young, M.B. Chadwick, R.E. MacFarlane, W.B. Wilson, D.G. Madland, P. Talou, T. Kawano, L. C. Leal, H. Derrien, N. M. Larson, R. Q. Wright, D.A. Brown, J. Pruet, Evaluated neutron data for U-235, ENDF/B-VII.1 library, MAT 9228, evaluated Sept 2006.
3. K.I. ZOLOTAREV, *Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Specific Neutron Reactions Important for Neutron Dosimetry Applications. Part 1. Evaluation of the excitation functions for the ^{27}Al(n,α)^{24}Na, ^{55}Mn(n,2n)^{54}Mn, ^{59}Co(n,p)^{59}Fe, ^{59}Co(n,2n)$^{58m+g}$Co and ^{90}Zr(n,2n)$^{89m+g}$Zr reactions.*, **INDC(NDS)-0546, IAEA, Vienna, Austria, April 2009**.
4. K.I. ZOLOTAREV, *Re-evaluation the ^{56}Fe(n,p)^{56}Mn reaction cross sections* (Rev. 3) **Private communication, IPPE, Obninsk, November 2003**
5. K.I. ZOLOTAREV, *Re-evaluation of Microscopic and Integral Cross-Section Data for Important Dosimetry Reactions. Re-evaluation of the excitation functions for the ^{24}Mg(n,p)^{24}Na, ^{32}S(n,p)^{32}P, ^{60}Ni(n,p)$^{60m+g}$Co, ^{63}Cu(n,2n)^{62}Cu, ^{65}Cu(n,2n)^{64}Cu, ^{64}Zn(n,p)^{64}Cu, ^{115}In(n,2n)^{114m}In, ^{127}I(n,2n)^{126}I, ^{197}Au(n,2n)^{196}Au and ^{199}Hg(n,n')^{199m}Hg reactions,* **INDC(NDS)-0526, IAEA, Vienna, Austria, August 2008**.
6. K.I. ZOLOTAREV, *Evaluation of Cross-Section Data from Threshold to 40-60 MeV for Specific Neutron Reactions Important for Neutron Dosimetry Applications. Part 2. Evaluation of the excitation functions for the ^{59}Co(n,3n)^{57}Co, ^{89}Y(n,2n)^{88}Y, ^{93}Nb(n,2n)^{92m}Nb, ^{169}Tm(n,2n)^{168}Tm and ^{209}Bi(n,3n)^{207}Bi reactions,* **INDC(NDS)-0584, IAEA, Vienna, Austria, August 2010**.
7. K.I. ZOLOTAREV, F.K. ZOLOTAREV, *Re-evaluation of the In115(n,n')In115m reaction excitation function from threshold to 20 MeV.* **Private communication, IPPE, Obninsk, March 2009**.
8. K.I. ZOLOTAREV, F.K. ZOLOTAREV, Evaluation of some (n,n'), (n,γ), (n,p), (n,2n) and (n,3n) reaction excitation functions for fission and fusion reactor dosimetry applications. **Report INDC(NDS)-0657, IAEA, Vienna, December 2013**.

9. K.I. ZOLOTAREV, *Re-evaluation of the I127(n,γ)I128 reaction excitation function from 4 keV to 20 MeV*. Private communication, IPPE, Obninsk, March 2013.
10. R.B. FIRESTONE, Nuclear Data Sheets **108 (2007) 2319**.
11. R.B. FIRESTONE, *Table of Isotopes CD-ROM*, Eighth Edition, Version 1.0, March 1996, S.Y. Frank Chu, CD-ROM Ed., V.S. Shirley, Ed., Wiley-Interscience, 1996.
12. BALRAJ SINGH, Nuclear Data Sheets **78 (1996) 407**.
13. J. BLACHOT, Nuclear Data Sheets **113 (2012) 2391**.
14. J. BLACHOT, Nuclear Data Sheets **111 (2010) 717**.
15. HUANG XIAOLONG, Nuclear Data Sheets **110 (2009) 2533**.
16. W. MANNHART, *Status of the Evaluation of the Neutron Spectrum of* $^{235}U + n_{th,,}$. p.29 Presentations-Web Links in Summary Report of the IAEA Consultants' Meeting, 13-15 October 2008, prepared by Pronyaev, V.G., Mengoni, A. and Carlson, A.D, IAEA report INDC(NDS)-0540, IAEA, Vienna, Austria, November 2008.
17. W. MANNHART, *Status of the Evaluation of the Neutron Spectrum of* $^{252}Cf(sf)$., p.29 Presentations-Web Links in Summary Report of the IAEA Consultants' Meeting, 13-15 October 2008, prepared by Pronyaev, V.G., Mengoni, A. and Carlson, A.D, IAEA report INDC(NDS)-0540, IAEA, Vienna, Austria, November 2008.
18. K.I. ZOLOTAREV, Private communication, IPPE, Obninsk, March 2013.
19. E. BROWNE, J.K. Tuli, Nuclear Data Sheets **122 (2014) 293**.
20. M.S. BASUNIA, Nuclear Data Sheets 107 (2006) 3323.
21. S.A. BADIKOV, V.N. VINOGRADOV, E.V. GAY, N.S. RABOTNOV, Report FEI-1686, Obninsk, 1985 (in Russian).
22. A. FABRY, M. De COSTER, *Integral test of capture cross-sections in the energy range 0.1 to 2 MeV*. Proc. of the 2nd Conf. on Nuclear Cross-Sections and Technology, Washington, D.C., 4-7 March 1968, NBS Special Publication 299, v. 2, p. 1263.
23. A.J. KONING, D. ROCHMAN, *Talys Evaluated Nuclear Data Library TENDL-2011*, NRG Petten, The Netherlands, December 2011.
24. L.R. VEESER, E.D. ARTHUR, *Measurement of n,3n Cross Sections for* ^{235}U *and* ^{238}U. Proc. of Int. Conf. on Neutron Physics and Nuclear Data for Reactors and Other Applied Purposes, 25-29 September 1978, AERE Harwell, UK, Publ. by OECD/NEA 1978, pp.1054-1058.
25. K. KOBAYASHI, I. KIMURA, M. NAKAZAWA, M. AKIYAMA, *Fission Spectrum Averaged Cross Sections of Some Threshold Reactions Measured with Fast Reactor YAYOI*. Nucl. Sci. Technology 13 (1976) 531-540.
26. T. HASHIMOTO, T. SOTOBAYASHI, K. KOBAYASHI, *Measurement of Average Cross Section for* $^{238}U(n,2n)^{237}U$ *Reaction*. Nucl. Sci. Technology 5 (1978) 626.
27. G. SHANI, *(n,2n) cross-section measurement of 93-Nb, 197-Au and 238-U with fission-neutron spectrum and its dependence on the asymmetry term*. Annals of Nuclear Energy 10 (1983) 473-476.
28. M.V. BLINOV, E.A. GROMOVA, S.S. KOVALENKO, B.D. Stsiborskiy, S.V. CHUVAEV, B.M. SHIRYAEV, $^{238}U(n,2n)^{237}U$ *Reaction Cross-Section Induced by The Cf-252 Spontaneous Fission Neutrons*. Atomnaja Energija (Sov.) 65 (1988) 206 (in Russian).

Experimental validation of IRDFF cross-sections in quasi-monoenergetic neutron fluxes in 20-35 MeV energy range,

M. Majerle, P. Bém, J. Novák, E. Šimečková, M. Štefanik

Nuclear Physics Institute of ASCR, Řež near Prague, Czech Republic

Abstract

The scope of the task is the validation of the IRDFF cross-sections for neutron dosimetric reactions (n,*x*n) on ^{59}Co, ^{169}Tm, ^{209}Bi, ^{54}Fe, and ^{197}Au in the energy range of 20 – 35 MeV. These materials in the form of thin foils are irradiated using the NPI quasi-monoenergetic neutron source based on the p+Li reaction at seven proton energies in the 20 - 35 MeV range. Irradiations above the energy of 35 MeV are realized with p+Li source at the TSL Uppsala. The activities of the produced nuclei are measured by the means of gamma-spectrometry. Cross-sections are extracted with SAND-II code from measured reaction rates.

Motivation

The experimentally measured cross-section data are scarce and uncertain above the neutron energy of 20 MeV and additional measurements are needed. In this work, the (n,3n) and (n,4n) reactions on ^{59}Co, ^{169}Tm, ^{209}Bi, ^{54}Fe and ^{197}Au are investigated at energies up to 60 MeV, at the NPI in the neutron energy range 20-35 MeV and at TSL under CHANDA program at energies above. The whole IRDFF energy interval is covered.

The uncertainties of the measured data at the energies above 20 MeV are usually above 10-15%, mostly because of not well known neutron spectra. At the NPI facility this can be partly overcome by the measurement of the ^{7}Be production in the lithium target after the irradiation and the neutron spectral shapes by the Time-Of-Flight method.

Experimental equipment and methods at the NPI

A basic experimental facility of the NPI is the isochronous Cyclotron U-120M. It provides the protons, deuterons, ^{3}He ions and alphas. In the negative ion mode of acceleration (extraction of the beam by the stripping foil), the protons and deuterons with energies of 6–37 MeV (15 μA) and 11 – 22 MeV (10 μA) with good beam-current stability are obtained and used for neutron production at the suitable targets.

For quasi-monoenergetic neutrons production, the p + ^{7}Li source reaction is used; the self-supporting 2 mm thin lithium foil together with the alcohol cooled carbon beam-stopper is bombarded by protons with energies up to 37 MeV and intensities of 8 μA. The generated quasi-monoenergetic neutron field presents the power-tool for cross-section data measurement in the neutron energy range of 18–36 MeV. The neutron flux is studied with MCNPX calculations and on-line detection techniques (the MCNPX neutron spectral flux was validated against both the time-of-flight measurement with the NE213 scintillation detector and the proton recoil telescope). The neutron flux density at the sample positions 85 mm from the Li target is around 10^{8} n/cm^{2}/s in the monoenergetic peak at the beam intensity 8 μA.

Similar facility is used at the TSL Uppsala where 3 irradiations at energies above 35 MeV were performed. The lithium target used at TSL was thicker (4 mm) and the magnetic deflection of the proton beam is used instead of the carbon beam stopper. This results in lower neutron fluxes, which were partly compensated with the longer irradiation time (24 hours).

After the irradiation (typically 8 hours at the NPI, 24 hours at the TSL), the studied materials are analyzed by several well calibrated HPGe detectors with good energy resolution.

Progress and preliminary results

By the beginning of 2015, seven irradiations were performed at the NPI facility. The used energies were 20.0, 22.5, 25, 27.5, 30, 32.5 and 35 MeV of proton energy. Some of the irradiations will be repeated to estimate the repeatability of the results and to overcome possible technical issues.

Three additional irradiations were performed at energies 38, 50 and 62 MeV of the proton energy at the TSL Uppsala. This work was supported by the CHANDA framework program.

At all irradiations the thin metal foils of the studied materials were irradiated and subsequently measured with several HPGe.

The studies performed with the measurements showed that the accuracy of the measured cross-section data should be improved by monitoring the ^{7}Be production in the lithium target. This enables to obtain the absolute number of the produced monoenergetic neutrons with the accuracy of the gamma spectroscopy methods (2 - 3%). The uncertainty of this number based only on the Li target thickness and charge measurements is higher and estimated to be around 7% as was confirmed with the repeatability of previous results.

The measured production of the ^{7}Be was compared with the data published by Schery et al. from 1977. Newly measured data are systematically lower for ca 8%. In 2015, we plan to repeat the Schery's experiment with the stacked foil technique to obtain an independent set of data and clarify the situation.

A good knowledge of the neutron spectra is obtained from the TOF measurements with the 2"x2" NE213 scintillator. The dynamic threshold technique is used to obtain the efficiency of the scintillator. The calculated efficiency depends only on the knowledge of the (n,el) reaction on ^{1}H, yielding very accurate shape of the efficiency curve with the absolute uncertainty within few percentage points (the number of the ^{1}H atoms in the NE213).

The TOF spectra are extracted from the maximal neutron energy down to the energies where the first frame overlap occurs, usually to some 10 - 15 MeV below the maximal neutron energy. The time resolution of 2 - 3 ns at distance 4 - 5 m means around 1 MeV resolution on the energy scale. The monoenergetic peak and shape of the spectra at lower energies is well characterized by this method.

The analysis of the data from the ^{196m2}Au isotope showed the disagreement of the gamma intensities between the LUND and ENSDF databases. Extra spectroscopy measurements are currently performed to determine accurately the absolute intensities of the gamma lines. The gamma line of ^{196m1}Au at 84.66 keV, which should be used for the absolute calibration of the intensities of other gammas, coincides with the X-ray lines produced on the lead shielding around the HPGe. The measurements will be performed with the removed lead shielding and on different HPGe to test for the integrity of the data.

Apart from the seven irradiations at the NPI, 3 irradiations at 38, 50 and 62 MeV proton energy were performed at the TSL Uppsala. Special effort was put to characterize the neutron flux at the position of the activation foils around 2 m from the lithium target (close user position). NE213 scintillator and thin film breakdown counters in the TOF mode were used, as well as extra activation foils to account for the proton contribution in the beam and the beam homogeneity as well as for comparison with more distant and well characterized position (standard user position). Higher order (n,xn) reaction products were observed in some foils.

The gamma spectra from all the measurements are currently being analyzed.

Facility development

During the experiments at the NPI it was realized that the facility could be used to measure isotopes with the decay constants in order of ms. The measurements should be performed with the online HPGe measurements, using the cyclotron duty cycle which can be set in the range 10-80% at frequencies 50,100 and 150 Hz. A pilot experiment with the HPGe placed behind the borated water door was performed and showed that such measurements should be feasible. Based on these results, the construction of the collimator and upgrade of the target station was started.

Activation cross section measurements for Bi and Co by 140 MeV p-Li quasi-monoenergetic neutrons,

H. Yashima

Kyoto University Research Reactor Institute, Kumatori-cho, JAPAN

Neutron activation experiment for Bi and Co by using 140 MeV p-Li quasi-monoenergetic neutron beam was performed. Experiment was performed at the Research Center for Nuclear Physics (RCNP), Osaka University.

Proton beams extracted from the ring cyclotron are transported to the neutron experimental hall and hit a 1.0 cm-thick ^{7}Li placed in the swinger which is in a vacuum chamber. Protons passing through a target are bent to the beam dump using a swinger magnet to measure the proton beam intensity with a Faraday cup. The neutrons emitted in the forward direction traveled into the time-of-flight (TOF) room through an iron collimator (12×10 cm^2 aperture) that was embedded inside a 150-cm-thick concrete wall. Charged particles such as secondary proton produced from Li target and accelerator component are deflected and stopped in the collimator with the aid of a bending magnet placed within the collimator. The movable collimator and the swinger magnet allow neutron emissions to be measured through angles between 0° and 25°.

The neutron spectra were measured using time-of-flight (TOF) method. A NE213 liquid scintillation detector was placed at the 0° position, then later at the 25° directions. The energy spectra of this neutron field are not purely monoenergetic, with both a high energy peak coming from ^{7}Li(p,n) reaction, and a low energy tail resulting from the consequent break-up reaction.

Thus, two different irradiation experiments were obtained by using two different neutron beams at 0˚ and 25˚. In irradiation experiments, activation samples were placed at an angle of 0˚ to the proton beam to measure activities induced by both high energy peak neutrons and low energy neutrons, while to measure activities induced by low energy neutrons only, samples were placed at an angle offset 25˚ from the direction of the proton beam. Average proton beam intensity was about 1 μA. During the irradiation time, the proton beam current at the beam dump was recorded with a digital current integrator, connected to a multi-channel scaler (MCS) to monitor the fluctuations of the proton beam currents.

After irradiation, the gamma rays emitted from the irradiated samples were measured with a high-purity germanium (HPGe) detector. The samples were measured several times in order to identify newly created radioactive nuclides by their half-lives. Reaction rates of radioactive nuclides produced in the samples were determined from the gamma-ray spectra and decay curves. Corrections to the data were made for beam current fluctuations using the digitized proton current and the peak efficiency of the HPGe detector.

To remove the component of nuclides produced by the low energy tail, the 25˚ spectra is normalized to the one measured at 0˚ by equalizing the neutron fluence in low energy region, then the corrected spectra is then obtained by subtracting the normalized 25˚ spectra from the 0˚ spectra.

But there are some region of which the neutron fluence are positive or negative in the low energy tail of the corrected spectra. To remove the contribution of low energy neutron completely, reaction rate were integrate for the low energy tail of the corrected spectra by using the cross section of the experimental data by Kim et al.[1] and the JENDL-HE data file[2,3]. Thus, the subtraction of the nuclide component produced by a beam angled at 25˚ from one angled at 0˚ gives a yield produced only by high energy peak neutrons after the correction of the low energy tail of the corrected spectra.

The contribution of low energy neutrons were also corrected by the ratio of peak reaction rate from minimum energy to maximum energy of peak energy region to total reaction rate from threshold energy to maximum energy of peak energy region by using the cross section of the experimental data by Kim et al. and the JENDL-HE data file. Contributing to the estimate of uncertainty in the cross section measurements are the counting statistics (< 25%), detector efficiency (Ge detector - 10%,

NE213 detector - 15% above 80 MeV, 10% below 80 MeV), beam current monitoring (5%) and correction for contribution of low energy tail (30%).

Preliminary results of neutron activation cross sections were obtained for ^{209}Bi(n,xn)^{203,204,205,206}Bi and ^{59}Co(n,xn)^{56,57,58}Co reactions. Cross section data obtained by subtraction method are larger than cross section data obtained by the correction using the ratio of peak reaction rate to total reaction rate, previous experimental data and evaluated data. Further analysis is needed to finalize cross section data.

Neutron activation cross sections for Bi and Co by using 80 MeV p-Li quasi-monoenergetic neutron beam was also performed. These results will be compared with other experimental data and cross section data will be finalized.

References

1. E. Kim et al., Nucl. Sci. and Eng., 129, 209-223 (1998).
2. Y. Watanabe et al., Proc. of Int. Conf. on Nuclear Data for Science and Technology, Santa Fe, USA, Sep.26-Oct.1,2004; AIP Conference Proc. 769, 326-331 (2005).
3. Y. Watanabe et al, T. J. Korean Phys. Soc., 59, 1040-1045 (2011).

Progress report on measurements of neutron cross sections with quasi-monoenergetic neutrons of 90 and 140 MeV,

P.P. Maleka, M.R. Nchodu

iThemba LABS, Cape Town, South Africa

Introduction

This report describes the status of the work performed at iThemba LABS up till now as presented at the RCM-II meeting held at the IAEA on the 16th – 20th March 2015. Emphasis is put on providing practical information for the recent data collected during irradiations and the preliminary gamma-ray spectra analysis to date.

Experimental status

Following the requirements of the institute, a project proposal requesting beam time for neutron irradiations on various targets in the high neutron energy field was submitted and presented to the iThemba LABS Project Advisory Committee (PAC) for evaluation. The proposal was approved and beam time was allocated for the two weekends of the 19-22 Sept. and 26-29 Sept. 2014. At the iThemba LABS neutron beam facility ^{nat}Li and ^{nat}Be targets of various thickness are normally used to produce quasi-monoenergetic neutron beams by (p,nx) reactions. The material which was preferred for this experiment for relative high peak fluence was using a metallic Li (8 mm thick) target. With the proton beams available from the separated sector cyclotron (SSC), the neutron energy range from 30 MeV to 200 MeV can be covered. Moreover, it was discussed and decided at the 1st CRM that measurements at iThemba LABS should include the 90 and 140 MeV neutron energies as to compare with similar experimental campaign that will be conducted at the Research Center for Nuclear Physics (RCNP) cyclotron facility of Osaka University

Facilities

As shown in fig. 1, the target area of the iThemba LABS neutron beam facility is separated from the experimental area by an iron shielding wall with collimator holes. The collimator channels have rectangular cross sections of about (48 × 48) mm^2 in size. The collimator exits are located 4 m behind the target. At a neutron emission angle of 0° the maximum distance from the target is about 10 m. The beam size at this position is about (130 × 130) mm^2. At 16°, the maximum distance is only about 8 m with a correspondingly smaller beam size. The availability of collimator at a neutron emission angle of

16° is a unique feature of this facility which allows an experimental subtraction of the detector readings effected by the neutrons from break-up continuum of the ^{7}Li(p,nx) reaction (Nolte *et al.*, 2002).

At higher beam currents (500 nA – 2.5 μA e.g. for this experiments), the beam position on the target can only be observed indirectly by monitoring the position of the deflected proton beam behind the neutron production target in a split Faraday cup. The time intervals between successive proton beam pulses can be enlarged from about 50 to 360 ns using an electrostatic deflector system, and at about 10 m flight path these time intervals correspond to frame-overlap neutron energies of about 3.5 MeV.

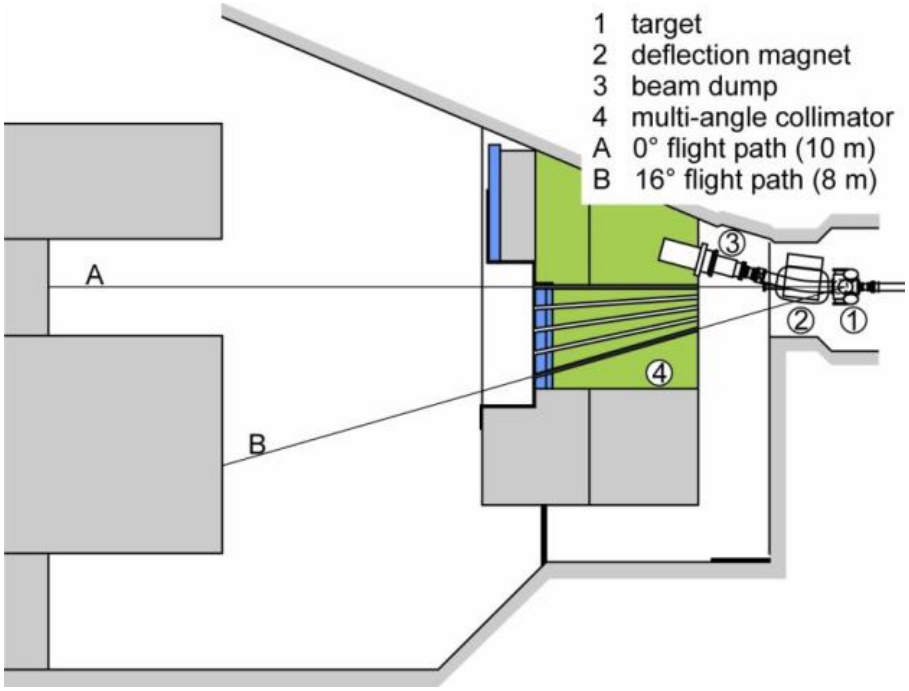


Fig. 1: *Layout of the iThemba LABS neutron vault facility with information explaining and identifying the various components*.

The meterological characterization of the beams is based on the time-of-flight (TOF) method. The neutron fluence per unit beam charge is measured using a calibrated ^{238}U fission ionization chamber with a total ^{238}U mass of about 250 mg. In addition, a 2 mm NE102 transmission detector is positioned in the 0° beam line to monitor the neutron yield.

The peak is made up of neutrons emitted at 0^0 from the ^{7}Li(p,n)^{7}Be reaction going to the ground and first excited states of ^{7}Be. The continuum is made up of neutrons from the breakup of ^{7}Li, which is mainly isotropic up to an angle of 16^0. The yield of any product radionuclide from an (n,x) reaction, produced by irradiation in the 0^0 beam, therefore includes components due to reactions initiated by both the high energy peak neutrons and the continuum, while the yield resulting from irradiations in the 16^0 beam is dominated by reactions initiated by the low-energy continuum alone. Thus subtracting the yield produced in the 16^0 beam (after appropriate normalization) from that simultaneously produced at 0^0 results in a yield determined for quasi-monoenergetic neutron energy, an example is shown in fig. 2 and see also Sisterson *et al.* (2005) for more details.

Two identical target (front to back arrangement, Al-Cu-Co-Tm-Bi-Au-Al) stacks were irradiated simultaneously in each measurement, one in the 0° beam and the other in the 16° beam. Both irradiation positions were located at a distance of about 5 m from the neutron production Li target. In addition Al and Cu targets are used as monitor. Laser beams are used to align the centre of the target

with the neutron beam line. As per manufacturer (GoodFellow Corp.) specifications, all target materials supplied were 99.9% pure and the discs were 25 mm in diameter and 0.5 mm thick.

The activated samples were measured by means of gamma-ray spectroscopy using the hyper-pure germanium (HPGe) detector (*p* type, 45 % relative efficiency, 2 keV FWHM resolution at 1.33 MeV) system. The detector is encased in a 10 cm thick lead castle fitted with a 2.0 mm thick copper inner lining in order to reduce the background in the sample spectra. Standard nuclear electronics are used to process the detector signals. The pulses from the amplifier are collected and sorted by the ATOMKI Palmtop software multi-channel analyzer. The system is energy and efficiency calibrated regularly with certified reference sources such that the centroid of the photo-peaks and the gamma-ray detection efficiency are continuously monitored. Depending on the type of the sample to be analysed, both point and volume reference sources are available for efficiency calibrations. Furthermore, these efficiency calibrations are also calculated using the MCNPX simulation code and compared with experimental data for estimates of various measuring geometries where sources are not available.

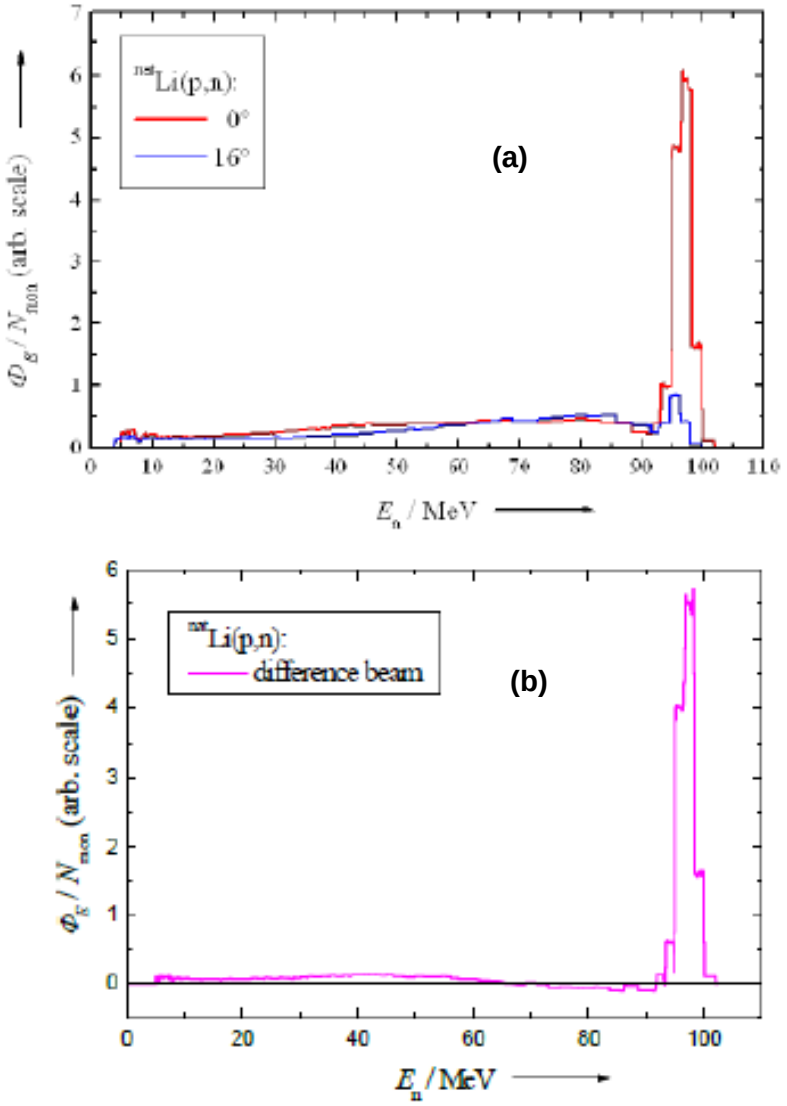


Fig. 2: (a) Neutron time-of-flight spectra obtained from the bombardment of a 5 mm thick Li target by 100 MeV protons. The two spectra were measured at neutron emission angles of 0^0 and at 16^0, respectively and are normalized so as to equalise the total number of counts in the continuum region. The neutron detector settings were identical for the two measurements. (b) Difference spectrum obtained by subtracting the 16^0 spectrum from the 0^0 spectrum.

Activity estimates

The activity $A(t)$ produced in a thin sample of mass m (with a size smaller than the beam area) after neutron irradiation for a time t_{irr} is:

$$A(t_{irr}) = (1 - e^{-\lambda \cdot t_{irr}}) \cdot x \cdot m \cdot \left(\frac{N_A}{M}\right) \cdot \sigma \cdot \left(\frac{d\Phi}{dt}\right)$$

And once the samples are counted in the HPGe detector, to include the decay during the irradiation, between irradiation and counting, and during the counting time, the activity can be calculated using the following expression;

$$A = N_T \cdot \sigma \cdot \phi \cdot \left(1 - e^{-\lambda \cdot t_{irr}}\right) \cdot e^{-\lambda \cdot t_d} \cdot \left(1 - e^{-\lambda \cdot t_c}\right)$$

A – Activity, λ - decay constant, x – isotopic percentage fraction in a sample, m – sample mass, N_A – Avogrado's constant, M – molecular weight, σ - cross-section, $d\Phi/dt$ – neutron fluence rate, N_T – number of target nuclei, ϕ - neutron flux, t_{irr} – irradiation time, t_d – delay time (time between end of irradiation and start of counting) and t_c – counting time (in the HPGe)

The total error estimate in the final cross section estimate will be affected mainly by the following components, ~ 7% due to the error estimate in the measurements of peak fluences, 4% peak to continuum ratio, 2% fluence monitor, ~4% for HPGe detection efficiency and 5 - 30% counting statistics (Nolte, 2013; Sisterson *et al.*, 2013).

Gamma-ray spectra analysis

Shown in figures 3 – 6 are the ratio of activity ($16^0/0^0$) estimates for the reactions identified and presented in the horizontal scales. These data indicates that the rates were consistent in the first weekend (E_n ~ 90MeV) and inconsistent with larger uncertainties for the second weekend (E_n ~ 140 MeV) of irradiations.

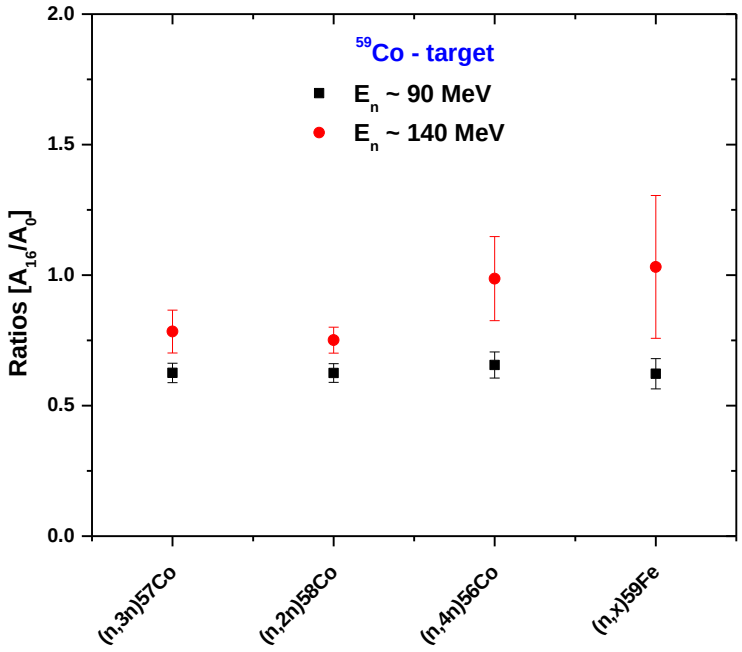


Fig. 3: Activity ratios (A_{16}/A_0) for various ^{59}Co(n,x) reactions.

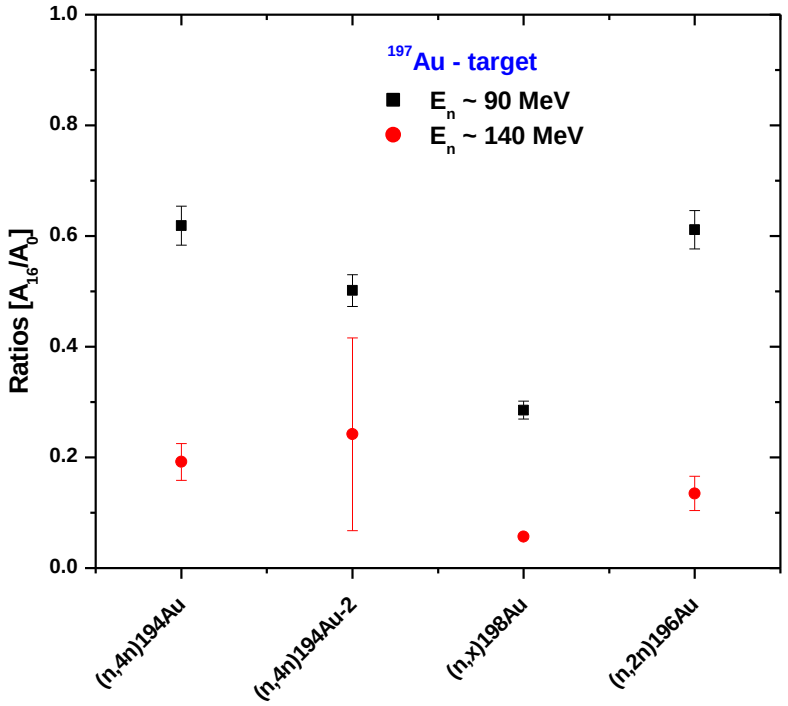


Fig. 4: Activity ratios (A_{16}/A_0) for various ^{197}Au(n,x) reactions.

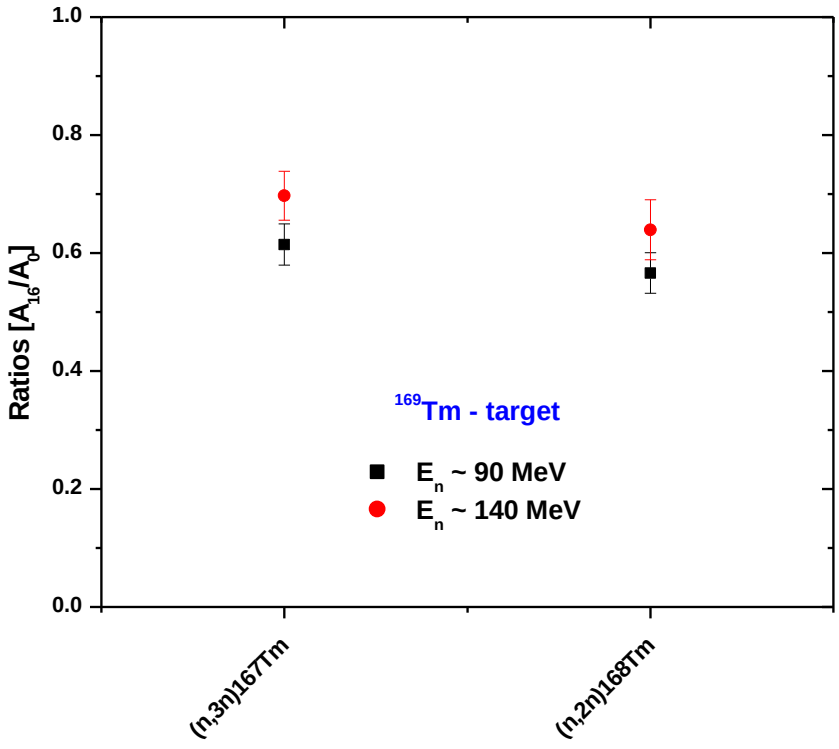


Fig. 5: Activity ratios (A_{16}/A_0) for various ^{169}Tm(n,x) reactions.

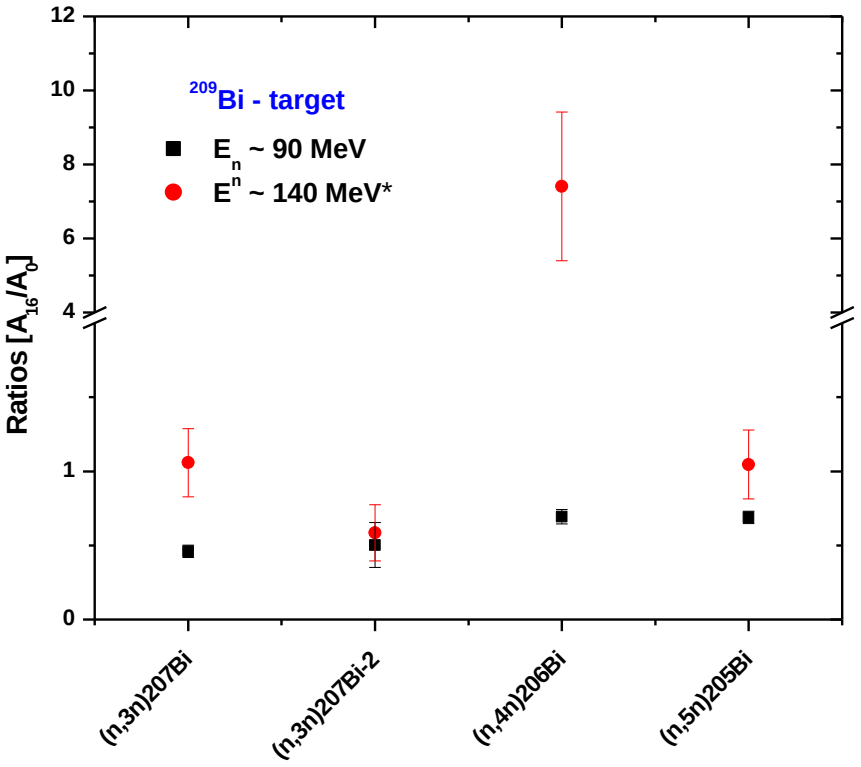


Fig. 6: Activity ratios (A_{16}/A_0) for various ^{209}Bi(n,x) reactions.

Summary and outlook

Ongoing gamma-ray spectra analysis is in progress, including the analysis of the monitor discs, Al and Cu. More importantly, in consultations with our collaborators the team is involved in unfolding the neutron spectra to determine the neutron fluences in the peak and the continuum. Subsequently cross-sections for various reactions will be calculated and reported.

References

R. Nolte *et al.*, (2002). High-Energy neutron reference fields for the calibration of detectors used in neutron spectrometry, *Nucl. Instrum. Meth.* A, 476, 369-373.

R. Nolte, (2013). Private communication.

J M. Sisterson *et al.*, (2005). Cross-section measurements for neutron-induced reactions in copper at neutron energies of 70.7 and 110.8 MeV, *Nucl. Instrum. Meth.* B, 240, 617-624

Evaluation of the ^{209}Bi(n,xn), x = 2, ... , 10 cross sections for high-energy neutron dosimetry,

V.G. Pronyaev, A.I. Poltavchenko

Institute of Physic and Power Engineering, Obninsk, Russia

Preliminary evaluation of the ^{209}Bi(n,xn) dosimetry reactions for x = 2 to 10 was carried out. The Bayesian code GLUCS was used for simultaneous fit of all partial cross sections in the energy range between 20 and 100 MeV. Evaluation of ^{209}Bi(n,2n) and ^{209}Bi(n,3n) reactions for energies below 20 MeV was done before and included in the files. ^{209}Bi(n,n') was added in the combined fit for completeness of constraint conditions, although it is not a dosimetry reaction. ^{209}Bi(n,nonel) reaction cross section reduced at the contribution from reaction channels with charge particle emission and fission was used as the constraint for the sum of (n,xn) reactions in the least-squares fit between 20 and 100 MeV. For this, the experimental data on non-elastic cross section for ^{nat}Pb and ^{209}Bi were analized in the energy range between 20 and 100 MeV and reduced by the contribution from channels with charge particle(s) emission.

A large difference in the model calculations of (n,xn) reactions with TALYS and GNASH codes was observed, which is probably caused by the different estimation of contribution of direct processes in the emission of the first neutron. The account of competition between neutron and gamma at higher sequential neutron emission steps as well as used level densities is also important.

A large discrepancy of the experimental data results is caused by large correction for the non-mono-energeticity of neutron source at the neutron energies above the maximum of the threshold of the reactions. Different results of calculations or evaluations of (n,xn) reactions introduce large uncertainty in the correction, which can reach 50 – 80%. It was found that existing long-lived metastable states in ^{204}Pb and ^{202}Pb with large population by β^+ decay of ^{204}Bi and ^{202}Bi are not mentioned in the description of the experimental data analysis. Moreover the proper taking into account of the activation and cooling times and activity measurements may influence at the determination of the cross sections.

Preliminary evaluation of ^{209}Bi(n,xn) reactions was done with a priory cross sections and covariance matrices of their uncertainties that will not be so much informative in the region where experimental data exist, but they provide some smoothness of the cross sections. A priori covariance matrices have rather strong correlations between neighbouring energy points but large uncertainties. Second iteration was used in the fit when a posteriori evaluation was used as a priori at the second step, without changes of the non-informative a priori covariance matrix.

The results of evaluation were compiled in the ENDF-6 formatted files. Plots of cross sections and correlation matrices were produced with the ERRORR module of the NJOY code.

The evaluation can be tested by unfolding of the spallation type spectra (for neutrons with energy below 60 MeV) using the results of activation rate measurements for ^{209}Bi sample, or by using the simulated spectra and rates.

Additional details of the cross sections evaluation for the ^{209}Bi(n,2-10n) reactions up to 100 MeV are available in the presentation at RCM-2 (https://www-nds.iaea.org/IRDFFtest/RCM2/Pronyaev_209Bi(n,xn)-final-evaluation.pdf), the ENDF formatted data – as the text file (https://www-nds.iaea.org/IRDFFtest/RCM2/Pronyaev_209Bi%28n,xn%29-final-evaluation.txt).

Action for improving IRDFF Decay Library: Updating decay data evaluations for the radionuclides produced by reactions included in IRDFF,

V.Chechev

Khlopin Radium Institute, St. Petersburg, Russia

This research as a part of the IAEA CRP F41031 [1] aims at establishing the reliable reference decay data of the dosimetry reaction residuals for consistent use in evaluation and applications.

A list of isotopes and isomers produced by reactions included and proposed for inclusion in IRDFF (versions 1.00 and 1.05) contains 102 (88+14) radioactive species. From this list, there are 55 ones with satisfactory ENSDF evaluations dated 2009-2014. For the remaining radionuclides, following discussion and recommendations of the 1st CRP Research Coordinated Meeting (Vienna, 1-5 July 2013) [2, 3] *updated decay data evaluations* would be beneficial. The half-lives and the absolute intensity of the gamma rays (per decay) may be regarded as the main decay data which are required for the IRDFF Decay Library.

Based on a comparison of the available ENSDF and DDEP evaluated data and analysis of the new published experimental information, we have determined **the first group of 20 radionuclides to update their half-lives and gamma ray intensities**: ^{3}H, ^{18}F, ^{22}Na, ^{24}Na, ^{46}Sc, ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{57}Co, ^{60}Co, ^{57}Ni, ^{64}Cu, ^{88}Y, ^{132}Te, ^{131}I, ^{140}Ba, ^{140}La, ^{141}Ce, ^{182}Ta, ^{198}Au. The new values of the decay characteristics recommended for the IRDFF library were obtained using the approaches and methodology adopted by the working group of the Decay Data Evaluation Project (DDEP) cooperation. The experimental data published up to 2014 were taken into account in updated evaluations. The evaluation results were presented at the 15th International Symposium on Reactor Dosimetry (ISRD) [4, 5] (see **Table**). New decay data obtained for the radionuclides of ^{18}F, ^{24}Na, ^{46}Sc, ^{51}Cr, ^{54}Mn, ^{59}Fe, ^{131}I, ^{141}Ce, ^{198}Au have been also placed on the DDEP web site [6].

Table. Recommended values of half-lives and prominent gamma ray intensities for 20 radionuclides – dosimetry reaction residuals[*)]

H-3	F-18	Na-22	Na-24	Sc-46	Cr-51
12.311 (25) y	1.82890 (23) h	2.6020 (4) y	14.9581 (20) h	83.787(16) d	27.704(4) d
	γ± 511 keV 193.72 (38) %	γ± 511 keV 180.71 (18) %	γ1368.63 keV 99.9934 (5) %	γ889.27 keV 99.98374 (25) %	γ320.08 keV 9.89 (2) %
		γ1274.54 keV 99.94 (13) %	γ2754.05 keV 99.863 (3) %	γ1120.54 keV 99.97 (2) %	

Mn-54	Fe-59	Co-57	Co-60	Ni-57	Cu-64
312.19 (3) d	44.494 (12) d	271.81 (4) d	5.2710 (8) y	35.9 (3) h	12.700 (2) h
γ834.85 keV 99.9752 (5) %	γ1099.24 keV 56.51 (31) %	γ122.06 keV 85.49 (15) %	γ1173.23 keV 99.85 (3) %	γ± 511 keV 86.8 (12) %	γ± 511 keV 35.04 (30) %
	γ1291.59 keV 43.23 (33) %	γ136.47 keV 10.71 (15) %	γ1332.49 keV 99.9826 (2) %	γ1377.6 keV 81.2 (6) %	γ1345.8 keV 0.4748 (34) %

Y-88	Te-132	I-131	Ba-140	La-140	Ce-141
106.63 (5) d	3.230 (13) d	8.0233 (19) d	12.753 (5) d	1.67858 (21) d	32.504 (13) d
γ898.04 keV 93.7 (3) %	γ49.72 keV 15.1 (3) %	γ364.5 keV 81.2 (5) %	γ29.964 keV 14.5 (4) %	γ487.02 keV 46.1 (5) %	γ145.44 keV 48.29 (30) %
γ1836.07 keV 99.346 (25) %	γ228.33 keV 88.12 (13) %		γ537.26 keV 24.6 (5) %	γ1596.2 keV 95.428 (25) %	

Ta-182	Au-198
114.61 (13) d	2.6943 (3) d
γ67.7497 keV 43.6 (15) %	γ411.8021 keV 95.62 (6) %
γ1121.29 keV 35.17 (33) %	

*) Compared to the paper in Proceedings of 15th ISRD, here minor corrections were done to some results of the evaluation for the radionuclides of ^{3}H, ^{22}Na, ^{51}Cr, ^{57}Co, ^{88}Y.

In relation to the problem discovered in the NIST calibration method [7], **the 2014 corrected NIST half-life values** [8] **were introduced** into the available experimental data sets, where required, and the re-evaluated half-life values were obtained for long-lived radionuclides.

For **absolute γ ray intensity values**, it is necessary to know not only the measured intensity values themselves but also the rest of the nuclide decay scheme. In some cases absolute γ ray intensity values may be deduced directly from adopted decay scheme parameters, and in all cases the overall consistency of the decay scheme is the best check of the quality of γ ray intensities [9]. The internal conversion coefficients (ICC) and their uncertainties used to obtain the absolute gamma ray intensities **were deduced with the BrIcc computer program** [10].

The updated evaluated decay data of the 20 radionuclides were presented in ENSDF format along with a table form and sent to the IAEA Nuclear Data Section to transfer them to the ENDF-6 format.

Based on the analysis of the new published experimental and evaluated data, the list was defined for **the second group of residual isotopes and isomers**, the key decay characteristics of which need to be updated: ^{28}Al, ^{47}Sc, ^{48}Sc, ^{67}Cu, ^{71}Zn, ^{74}As, ^{113m}In, ^{126}I, ^{137m}Ba, ^{167}Tm, ^{196}Au, ^{207}Bi. This group includes 6 radionuclides (underlined) from new reactions recommended for inclusion in IRDFF by the 1st RCM CRP. The updated evaluated decay data of the above 12 radionuclides were presented in ENSDF format along with a table form and also sent to the IAEA Nuclear Data Section to transfer them to the ENDF-6 format.

Proposed activities in the frame of CRP for 2015-2016:

- Analyzing of decay schemes and updating decay characteristics for the isotopes and isomers of ^{46m}Sc, ^{56}Mn, ^{58}Co, ^{58m}Co, ^{60m}Co, ^{93m}Nb, ^{106}Ru, ^{106}Rh, ^{106m}Rh, ^{131m}Xe, ^{144}Ce, ^{144}Pr, ^{144m}Pr, ^{182m1}Ta, ^{181m2}Ta, ^{196m1}Au, ^{196m2}Au, ^{198m}Au, ^{199m}Hg, ^{237}U.
- Testing and possible improving decay characteristics for the radionuclides from new reactions proposed for inclusion in IRDFF: ^{55}Co, ^{56}Co, ^{94}Nb, ^{114m}In, ^{117m}Sn, ^{195}Au.

References

1. The IAEA CRP F41031 Testing and Improving the IAEA International Dosimetry Library for Fission and Fusion (IRDFF)"". - *https://www-nds.iaea.org/IRDFFtest/*.
2. Summary Report of the 1st RCM CRP F41031. Prepared by A. Trkov, L.R. Greenwood, S.P. Simakov, Report INDC(NDS)-0639, IAEA, Vienna, September 2013.
3. V.P. Chechev. *Problem of Improving IRDFF Decay Library: List of Radionuclides, Status of the Available Evaluated Decay Data, Needs for New (Updated) Evaluations.* Summary for the 1st RCM CRP F41031. In: INDC(NDS)-0639, IAEA, Vienna, September 2013, p. 53-56.
4. Fifteenth International Symposium on Reactor Dosimetry (ISRD-15) - http://www.reactordosimetry.com/index.html.
5. V.P. Chechev and N.K. Kuzmenko. *Nuclear decay data for the International Reactor Dosimetry Library for Fission and Fusion (IRDFF): updated evaluations of the half-lives and gamma ray intensities.* Adopted for publication in Proceedings of the 15th ISRD.

6. *Recommended Data by the Decay Data Evaluation Project working group.* / URL: http://www.nucleide.org/DDEP_WG/DDEPdata.htm.
7. M.P. Unterweger, R. Fitzgerald, Appl. Radiat. Isot. 70, 1892 (2012).
8. M.P. Unterweger, R. Fitzgerald, Appl. Radiat. Isot. 87, 92 (2014).
9. M.-M. Bé, V.P. Chechev, Nucl. Instrum. Methods Phys. Res. A728, 157 (2013).
10. T. Kibédi, T.W. Burrows, M.B. Trzhaskovskaya, P.M. Davidson, C.W. Nestor Jr. Nucl. Instrum. and Methods in Phys. Res. A589, 202 (2008).

Neutron Spectral Adjustment with the STAYSL PNNL Software Suite with IRDFF to 60 MeV,

L.R. Greenwood, C.D. Johnson

Pacific Northwest National Laboratory, Richland, WA, US

The STAYSL PNNL [1, 2] suite of computer codes for neutron spectral adjustment provides a comprehensive suite of software tools that address all of the steps required to process raw neutron dosimetry data for neutron spectral adjustment. A generalized least-squares method is used by taking into account all known uncertainties and covariances in the input data, neutron activation cross sections, and neutron flux spectrum. Evaluated nuclear cross sections and covariances are first processed by NJOY. The NJpp code is then used to create nuclear data libraries that are used by STAYSL PNNL. BCF is used to process the irradiation history data from a variety of formats and the output file feeds into the SigPhi Calculator, which is an Excel-based spreadsheet that calculates the saturated reaction rates. The SigPhi Calculator corrects for decay during irradiation and gamma self-absorption and also uses an iterative technique to account for neutron burnup effects. The reaction rates calculated by the SigPhi Calculator can then be copied directly into a STAYSL PNNL input file. The SHIELD code is used to calculate neutron self-shielding corrections to the neutron activation cross sections in IRDFF for wire and foil geometries in both isotropic neutron fluxes and neutron beams.

When STAYSL PNNL was first issued in January 2013, the nuclear data libraries and covariance files were taken from IRDF-2002. These files have now been replaced using the evaluated data in IRDFF. Since IRDFF extends to 60 MeV, 40 additional energy groups, each 1 MeV wide, were added to the original 100-energy-group structure, resulting in a total of 140 energy groups. Since IRDFF added eight new nuclear reactions, the BCF, NJpp, SHIELD, and SigPhi Calculator programs were updated to incorporate the required nuclear data for processing of the new reactions.

In reviewing the data available in IRDFF, we have identified a need for the ^{56}Fe(n,nd+t)^{54}Mn cross section because this reaction contributes significant ^{54}Mn to the ^{54}Fe(n,p) reaction at higher neutron energies. Similarly, other reactions such as ^{60}Ni(n,nd+t)^{58}Co need to be reviewed to determine if other additional reactions are needed in IRDFF.

The STAYSL PNNL suite has been updated and testing with this “beta” version and the new IRDFF library has been conducted. Calculations for neutron spectra below 20 MeV show good agreement with the prior results using the IRDF-2002 libraries for reactions where the nuclear data has not changed significantly in IRDFF. Above 20 MeV, testing was conducted using available experiments at Be(d,n) neutron sources [3, 4]. In these high-energy cases, the neutron spectra are well-known from time-of-flight measurements. Hence, the spectral adjustment is small, as expected. An example of spectral adjustment at higher neutron energies is shown in Figure 1.

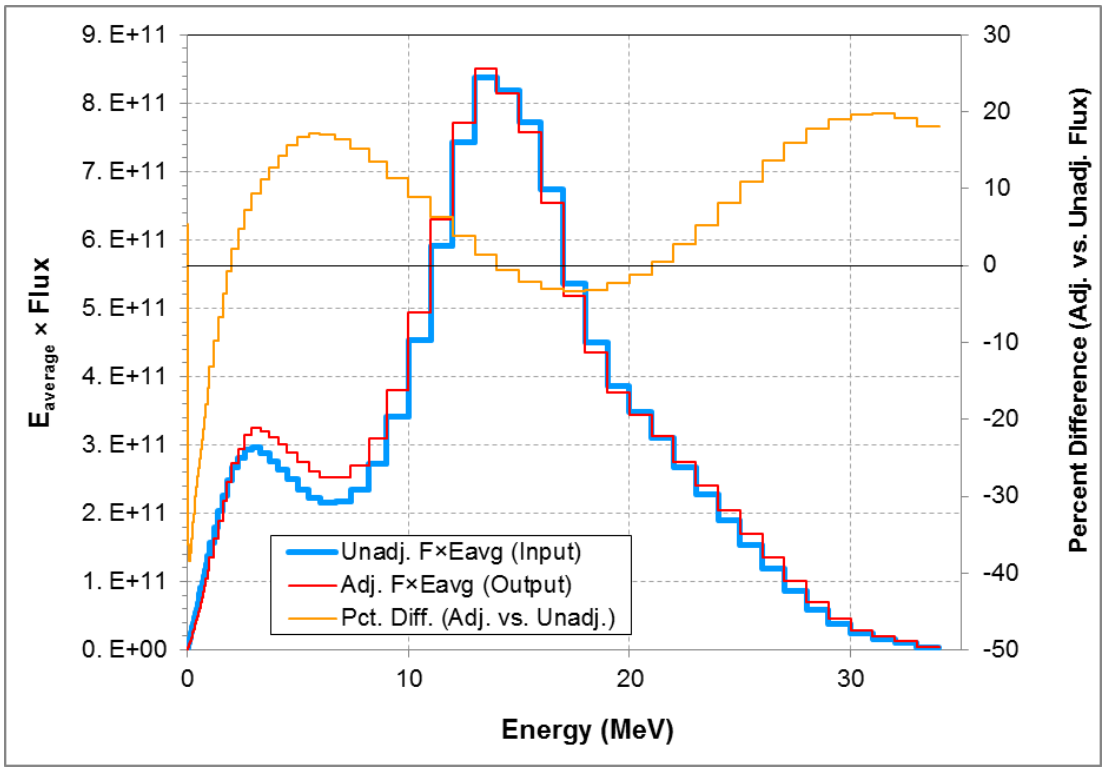


Figure 1 – Neutron spectral adjustment with STAYSL PNNL for a Be(d,n) neutron spectrum at E_d=30 MeV (Ref. 4).

The generalized least-squares technique used in STAYSL PNNL adjusts all of the input data according to the input uncertainties and covariances, including the neutron flux spectrum. A previous version of STAYSL [5] was modified to reverse the adjustment process so that neutron cross sections can be adjusted when saturated reaction rates are measured in a number of different neutron spectra that are well-characterized by neutron time-of-flight measurements or other standard neutron fields. We are currently revising STAYSL PNNL for this purpose and plan on releasing the updated STAYSL PNNL suite with these additional modifications in the near future.

References:

[1] LR Greenwood and CD Johnson, User Guide for the STAYSL PNNL Suite of Software Tools, Pacific Northwest National Laboratory Report PNNL-22253, January 2013.
[2] LR Greenwood and CD Johnson, Least-Squares Neutron Spectral Adjustment with STAYSL PNNL, proceedings of the 15th International Symposium on Reactor Dosimetry, Aix en Provence, France, May 2014.
[3] LR Greenwood, RR Heinrich, MJ Saltmarsh, and CB Fulmer, Integral Tests of Neutron Activation Cross Sections in a ^{9}Be(d,n) Field at E_d=40 MeV, Nucl. Sci. Eng. 72, p. 175-190 (1979).
[4] DW Kneff, H. Farrar IV, LR Greenwood, and MW Guinan, Characterization of the Be(d,n) Neutron Field by Passive Dosimetry Techniques, Symposium on Neutron Cross-Sections from 10 to 50 MeV, Brookhaven National Laboratory, BNL-NCS-51245, p. 113-132 (1980).
[5] LR Greenwood, Integral Testing of Spallation Cross Sections for Neutron Dosimetry at 113 and 256 MeV, Reactor Dosimetry: Radiation Metrology and Assessment, ASTM STP 1398, ppl. 409-416, 2001.

.

Appendix 1

Second Research Coordination Meeting of the IAEA CRP F41031

"Testing and Improving the International Dosimetry Library for Fission and Fusion (IRDFF)"

16 - 20 March 2015
IAEA Headquarters, Vienna, Austria
Meeting Room C 0739

AGENDA

(presentation's time is approximate and includes questions and breaks)

Monday, 16 March 2015

09:00 - 09:30 **Arriving**

09:30 - 10:30 **Opening session**
Welcome address - Robin Forrest, Section Head (NDS)
Administrative announcements - Alexander Öchs (NDS)
Election of Chairperson and Rapporteur - All
Approval of Agenda - All

Current organizational and technical issues of CRP - S. Simakov (NDS)

Session 1: Individual Presentations on Progress and Future Research Work

10:30 - 11:30 P. Griffin, "Advanced UQ Approaches to the Validation of Dosimetry Cross Sections in Reactor Benchmark Fields"

11:30 - 12:30 C. Destouches, "Progress of the CEA contribution to IRDFF validation: experimental data and codes"

13:00 - 14:00 Lunch break

14:00 - 15:00 P. Mastinu, "Measurement of 30keV Maxwell-Boltzmann neutron spectra: Status after the 1st CRP meeting"

15:00 - 16:00 I. Kodeli, "Validation of IRDFF-v1.04 (&v1.05) Dosimetry Library using SINBAD Shielding Benchmark Experiments"

16:00 - 17:00 M. Angelone, "Benchmarking of IRDFF against 14 MeV neutron Experiments"

17:00 - 18:00 C. Konno, "Preliminary results of IRDFF benchmark test at JAEA/FNS"

Coffee breaks as needed

Tuesday, 17 March 2015

Appendix 1

Session 1: Individual Presentations on Progress and Future Research Work

9:00 - 10:00	K. Zolotarev, "Evaluation of the excitation functions of the ^{238}U(n,γ) and ^{238}U(n,2n) reactions"
10:00 - 11:00	M. Majerle, "Experimental validation of IRDFF cross-sections in quasi-monoenergetic neutron fluxes in 20 - 35 MeV energy range"
11:00 - 12:00	R. Nchodu, "Progress report on measurements of neutron cross sections with quasi-monoenergetic neutrons of 90 and 140 MeV"
12:00 - 13:00	H. Yashima, "Activation cross section measurements for Bi and Co by 140 MeV p-Li quasi-monoenergetic neutrons"
13:00 - 14:00	*Lunch break*
14:00 - 15:00	V. Pronyaev, "On the task of evaluation of ^{209}Bi(n,xn), x = 2, …, 10 cross sections for the high-energy neutron dosimetry"
16:00 - 17:00	V. Chechev, "Action for improving IRDFF Decay Library: Updating decay data evaluations for the radionuclides produced by reactions included in IRDFF"
17:00 - 18:00	L. Greenwood, "Revision of the STAYSL PNNL Spectral Adjustment Code to use IRDFF and Extend the Energy Range to 60 MeV"

Coffee breaks as needed

Wednesday, 18 March 2015

9:00 - 10:00	A. Plompen, "**IRMM projects related to IRDFF**"
10:00 - 11:00	M. White, "Updates on the LANL Efforts in Support of Testing and Improving the IRDFF Library"
11:00 - 12:00	A. Trkov**,** "Ambiguity in ^{55}Mn(n,g) in the fast energy range", "Comparison of new ^{238}U(n,g) evaluations above resonance energy range", "On the use of spectrum-averaged cross sections and spectral indices in fast reactors"
12:00 - 13:00	R. Capote**,** "Prompt neutron spectra of thermal neutron induced fission of U-235 "
13:00 - 14:00	*Lunch break*
14:00 - 14:30	C. Destouches, "CALMAR – ECORCES unfolding code (under publishing process)"
14:00 - 17:00	**New Exercise on Spectrum Unfolding (and Radiation Damage ?) Codes ?:** specific issues, codes and potential users. (see previous "REAL-84": INDC(NDS)-212, INDC(NDS)-198, INDC(NDS)-190)

Coffee breaks as needed

19:00 - ***Hospitality event:*** *Visit Gösser Bierklinik http://www.goesser-bierklinik.at/*

Appendix 1

Thursday, 19 March 2015

Session 2: Joint Discussion of Progress and Future Research Work

9:00 - 12:30 **all**

12:30 - 14:00 *Lunch break*

Session 3: Drafting of the Summary Report of the Meeting

14:00 - 18:00 **all**

Coffee breaks as needed

Friday, 20 March 2015

Session 3: Finalisation of the Summary Report of the Meeting

09:00 - 13:00 **all**

13:00 ***Closing of the Meeting***

13:00 - 14:00 *Lunch break*

14:00 - ***Individual contacts between CRP Participants (who has needs and time) and NDS Experts (floor A-23, NDS offices)***

Coffee breaks as needed

Appendix 2

First Research Coordination Meeting on

Testing and Improving
the International Reactor Dosimetry and Fusion File (IRDFF)

16 – 20 March 2015, Vienna, Austria

LIST OF PARTICIPANTS

CZECH REPUBLIC Mr. Mitija MAJERLE Nuclear Research Institute Řež Husinec- Řež 130 Řež 25068 E-mail: majerle@ujf.cas.cz	Mr Hiroshi YASHIMA Kyoto University Research Reactor Institute Kumatori-cho, Sennan-gun Osaka 590-0494 E-mail: yashima@rri.kyoto-u.ac.jp
FRANCE Mr Cristophe DESTOUCHES CEA – Centre d'etudes nucleaires de Cadarache DER, B.P. 1 13108 St. Paul Lez Durance E-mail: chrisophe.destouches@cea.fr	**RUSSIAN FEDERATION** Mr Valery CHECHEV Khlopin Radium Institute 2 Murinsky prospekt 28 194021 St. Petersburg E-mail: chechev@khlopin.ru
ITALY Mr Maurizio ANGELONE Division of Fusion Centro Ricerche Energia-Frascati, ENEA C.P. 65, via Enrico Fermi 27 00044 Frascati E-mail: Maurizio.angelone@enea.it	**SLOVENIA** Mr Ivo Kodeli Jožef Stefan Institute Jamova cesta39 1000 Ljubljana E-mail: ivan.kodeli@ijs.si
ITALY-cont'd Mr Pierfrancesco MASTINU Italian Physical Society INFN, Laboratorio Nazionale di Legnaro Viale dell' Universita 2-35020 Legnaro E-mail: pierfrancesco.mastinu@lnl.infn.it	**SOUTH AFRICA** Mr Rudolph NCHODU iThemba Laboratory for Accelerator-based Sc. Old Faure Road, Faure 7131, P.O. Box 722 7129 Somerset West E-mail: nchodu@tlabs.ac.za
JAPAN Mr Chikara KONNO Japan Atomic energy Agency Tokai-mura, Naka-gun Ibaraki-ken 319-1195 E-mail: konno.chikara@jaea.go.jp	Mr Peane MALEKA iThemba Laboratory for Accelerator-based Sc. Old Faure Road, Faure 7131, P.O. Box 722 7129 Somerset West E-mail: pmaleka@tlabs.ac.za

Appendix 2

UNITED STATES OF AMERICA Mr Lawrence GREENWOOD Pacific Northwest National Laboratory 902 Battelle Boulevard, P.O. Box 999 Richland, WA 99352 E-mail: larry.greenwood@pnl.gov	**IAEA** Mr Stanislav SIMAKOV (CRP Officer) Head of the Nuclear Data Services Unit Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21717 E-mail: s.simakov@iaea.org
Mr Morgan White Los Alamos National Laboratory T-16, Nuclear Physics Group P.O. Box 1663, MS B243 Los Alamos, NM 87545 E-mail: morgan@lanl.gov	Mr Roberto CAPOTE NOY (CRP alternative Officer) Head of the Nuclear Data Development Unit Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21713 E-mail: r.capotenoy@iaea.org
Mr Patrick GRIFFIN Sandia National Laboratories P.O. Box 5820 87185 Albuquerque, NM UNITED STATES OF AMERICA E-mail: pjgriff@sandia.gov	Mr Robin FORREST Head of the Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21709 E-mail: r.forrest@iaea.org
CONSULTANTS Mr Konstantin ZOLOTAREV Institute for Physics and Power Engineering Bondarenko Sq. 1 249033 Obninsk RUSSIAN FEDERATION E-mail: zki1946@mail.ru	Mr Naohiko OTSUKA Nuclear Data Services Unit Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21715 E-mail: n.otsuka@iaea.org
Mr Vladimir PRONYAEV Institute for Physics and Power Engineering Bondarenko Sq. 1 249033 Obninsk RUSSIAN FEDERATION E-mail: pronyaev@ippe.ru	Ms Valentina SEMKOVA Nuclear Data Services Unit Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21727 E-mail: v.semkova@iaea.org
INTERNATIONAL ORGANSIATION Mr. Arjan PLOMPEN EC-JRC-IRMM Retiesweg 111 2440 Geel Belgium E-mail: arjan.plompen@ec.europa.eu	Mr Andrej TRKOV Nuclear Data Development Unit Nuclear Data Section Division of Physical and Chemical Sciences Tel.: +1 2600 21712 E-mail: a.trkov@iaea.org

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

E-mail: nds.contact-point@iaea.org
Fax: (43-1) 26007
Telephone: (43-1) 2600 21725
Web: http://www-nds.iaea.org