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I N D C INTERNATIONAL NUCLEAR DATA COMMITTEE

Summary Report of the Technical Meeting on "International Reactor Dosimetry File: IRDF-2002"

IAEA Headquarters
Vienna, Austria
27 – 29 August 2002

Prepared by
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Abstract

This report summarizes the presentations, recommendations and conclusions of the Technical Meeting on "International Reactor Dosimetry File: IRDF-2002." The purpose of this meeting was to discuss scientific and technical matters related to the subject and coordinate related tasks. Discussions were held and recommendations were given for the preparation of the files on topics related to: reactions to be included, need for new evaluations or revisions, decay data, radiation damage data, integral testing in benchmark fields, and computer codes to be included. Tasks were assigned and deadlines were set. The participants emphasized that accurate and complete knowledge of nuclear data for reactor dosimetry are essential for improving the accuracy of the reactor pressure vessel service life assessment of nuclear power plants as well as in other neutron metrology applications such as boron neutron capture therapy, therapeutic use of medical isotopes, nuclear physics measurements, and reactor safety applications.

TABLE OF CONTENTS

1. OBJECTIVE AND AGENDA

The Technical Meeting on "International Reactor Dosimetry File: IRDF-2002" was held at the IAEA Headquarters in Vienna, Austria, from 27 to 29 August 2002. The purpose of this meeting was to discuss scientific and technical matters related to the subject, coordinate related tasks and assign responsibilities.

Dr. W. Mannhart, PTB, Germany, was elected as the chairman of the meeting. Dr L. Greenwood, PNNL, USA, was selected as rapporteur of the meeting. The detailed approved Agenda is attached (see Appendix 1). Other experts taking part at the meeting, were Dr. O. Bersillon, CEA, France; Dr. K. Shibata, JAERI, Japan; Dr. K.I. Zolotarev, IPPE, Russia; Dr. E. M. Zsolnay, BUTE, Hungary; and Dr. A. Nouri, OECD-NEA, France (observer). For the complete list of participants including affiliations and addresses see Appendix 2.

Dr. Alan Nichols, Head of the Nuclear Data Section (NDS), welcomed the participants and Dr. R. Paviotti-Corcuera, Scientific Secretary of the Technical Meeting, summarized the mechanisms and objectives of the Data Development Project and the purpose of the meeting (see Appendix 3).

2. BACKGROUND

The last approved version of the reactor dosimetry file IRDF-90 V2 was released in 1993. Most of the evaluations for this file were prepared in the mid-eighties. Since then a large amount of new experimental data have been measured, and two new national reactor dosimetry libraries have been produced (RRDF-98 and JENDL/D-99). Some of the reaction cross sections and uncertainties included in these libraries may be of better quality than the data in the IRDF-90 file.

The reactor dosimetry community (at the workshop on "Cross Sections and Uncertainties" of the Tenth International Symposium on Reactor Dosimetry, and later via the chairman of the EWGRD and of the ASTM E10.05) expressed the need for an updated, consistent and tested reactor dosimetry library containing uncertainty information in the form of covariance matrices. Following a recommendation by the International Nuclear Data Committee (INDC), the Nuclear Data Section (NDS) of the IAEA initiated the data Development Project (DDP): International Reactor Dosimetry File IRDF-2002. The objective of the project is to prepare and distribute a standardized, updated and benchmarked neutron dosimetry reaction cross section library (IRDF-2002) for use in the Reactor Pressure Vessel service life assessment of nuclear power plants.

The IAEA-NDS had in the past, supported similar efforts to improve the knowledge of data for applications in Dosimetry. Some examples related to this support were documented in the following reports:

Intercomparison of predicted displacement rates based on neutron spectrum adjustments (REAL-80 exercise), W. L. Zijp, E. M. Zsolnay, H. J Nolthenius, E. J. Szondi, G.C. Verhaag, Nucl. Technol. (Nov 1984). v. 67(2) p. 282-301.

Information on the REAL-84 Exercise, W. L. Zijp, E. M. Zsolnay, D. E. Cullen, Proc. of the IAEA Cons. Meeting for Radiation Damage Estimates for Reactor Structural Materials, Santa Fe, NM, USA, May 20-22, 1985.

Nuclear data aspects encountered in the REAL80 and REAL84 intercomparisons. Nuclear data for radiation damage estimates for reactor structural materials, Ed.: V. Piksaikin, INDC(NDS)-179/G, Vienna, 1986, p. 95-105.

The assessment of the results of the REAL-84 exercise, E. M. Zsolnay, INDC(NDS) 190/G+F+R, Vienna, 1987. 36 p.

Nuclear data need for the covariance information used in the neutron spectrum adjustment. In: Covariance methods and practices in the field of nuclear data, Ed.: V. Piksaikin, INDC(NDS)-192/L, Vienna, 1988, p. 66-67.

Final report on the REAL-84 exercise, W. L. Zijp, E. M. Zsolnay, H. J Nolthenius, E. J. Szondi, ECN-212, BME-TR-RES-18/88, IAEA-NDS-212, Petten, 1988, 99 p.

Analysis of the REAL-84 intercomparison exercise, Summary of the specialists' meeting held in Jackson Hole, USA, 27-29 May 1987, IAEA, INDC(NDS)-198/LFR.

Nuclear Data for Radiation Damage Assessment and Related Safety Aspects, Proceedings of an advisory group meeting held in Vienna, 19-22 September 1989, IAEA-TECDOC-572-1990.

The International Reactor Dosimetry File (IRDF-90 Version 2), N. P. Kocherov, P. K. McLaughlin, Report IAEA-NDS-141 (rev.3), Mar 1996

Neutron metrology file NMF-90. An integrated database for performing neutron spectrum adjustment calculations, N. P Kocherov, Report INDC(NDS)-347, Jan 1996.

The Neutron Metrology File NMF–90, E. M. Zsolnay, E. J. Szondi, H. J. Nolthenius, Report IAEA-NDS-191. (Rev.1) January 1999.

3. SUMMARY OF PRESENTATIONS

Ms. R. Paviotti-Corcuera presented the agenda and principal goals of the meeting to discuss the preparation of the IRDF-2002. At the International Symposium on Reactor Dosimetry held in Brussels, Belgium, the week prior to this meeting, a supplementary workshop on benchmark fields discussed the best benchmark fields to be used for the testing of data in IRDF-2002. Ms. Paviotti-Corcuera presented the summary report of this workshop, which recommended that the only reference benchmark fields that should be used are a thermal neutron spectrum, a 1/E slowing down spectrum, the Cf-252 neutron spectrum in ENDF/B-VI, and a 14-MeV spectrum for high-energy neutrons (Appendix 4). Other neutron fields may also be used for consistency testing; however, only the above fields should be used for the primary testing of the dosimetry cross section data to avoid using neutron spectra that were based partly on reactor dosimetry cross sections.

The following presentations summarize extended abstracts that are given in Appendix 5.

Ms. Eva Zsolnay presented the results of cross section testing of the evaluated neutron cross section libraries IRDF-90, JENDL-D-99, and RRDF-98. Wherever possible, cross sections were compared to measurements performed in the ORR-PSF, CFRMF, RTNS-II, and HFR reactor. Detailed tables included in the attached report document the results of these integral consistency tests.

Mr. K. I. Zolotarev presented some new cross section evaluations and integral testing in Cf-252 and U-235 standard neutron fields. Of particular interest to retrospective reactor dosimetry, new evaluations will be completed for the 62Ni(n,γ)63Ni and 54Fe(n,γ)55Fe reactions. There was some discussion of which data to include in IRDF-2002 with a possible cut-off date on data evaluations. There was also a discussion of the best sources of nuclear data and consistency with the IRDF-2002 data files. Both questions are discussed in some detail later in this report.

V. Zerkin presented the cross section graphics software package ZVVIEW that was developed for the evaluators. ZVVIEW is a very powerful and complete package that simplifies the presentation of nuclear cross section data. A CD-ROM version of this computer package is available from the IAEA-NDS on request.

K. Shibata presented cross section comparisons contained in JENDL/D-99, IRDF-90V2, and RRDF-98 for Cf-252, U-235, ISNF, CFRMF, ΣΣ, YAYOI, JMTR, JOYO, and a d-Li accelerator neutron spectrum.

W. Mannhart presented the results of nuclear data testing in the Cf-252 and U-235 neutron fields. The spectrum of Cf-252 is the only field that is presently reliable. In the case of U-235, serious deficiencies exist in the spectral representation at higher neutron energies. The Cf-252 spectrum is determined by neutron time-of-flight measurements and is wholly independent of the reactor dosimetry cross sections. The consensus was that Cf-252 is the best reference neutron spectrum to use for the integral testing of the reactor dosimetry cross section files.

L. Greenwood presented nuclear data needs for retrospective reactor dosimetry, including requested evaluated cross sections for 54Fe(n,γ)55Fe, 62Ni(n,γ)63Ni, and 93Nb(n,γ)94Nb. The latest version of the SPECTER computer code, which calculates dpa, pka atomic recoil spectra, and gas production for 40 elements and selected compounds, has been made available to the IAEA-NDS for potential inclusion in IRDF-2002. A PC version of the STAY'SL computer code, which performs neutron spectral adjustments, has also been made available. The STAY'SL data libraries can be updated with the new IRDF-2002 cross sections and covariances, when these data become available.

O. Bersillon discussed the preparation of nuclear decay data files that have been requested for inclusion in IRDF-2002. He discussed a computer code, SDF2NDF, which he has developed that will read ENDSF data files and produce decay data files in the ENDF/B-VI format.

4. RECOMMENDATIONS AND DISTRIBUTION OF TASKS

Discussions were held on a number of topics concerning the future plans for the preparation of the IRDF-2002 data files. The following sections summarize the results of these discussions and include recommendations for specific tasks required to complete the work.

4.1. Structure and Dosimetry Reactions to be included in IRDF-2002

The IRDF-2002 file will include evaluated data taken from the special purpose dosimetry files of IRDF-90, JENDL/D-99, RRDF-98, or other appropriate sources. The selection of which reactions and evaluated data files will be based on the results of integral testing in primary benchmark fields, as discussed above. The selection will further include consistency testing in other selected neutron fields as well as a complete examination of the covariance data. Data files will not be included unless complete covariance files are available, except in special cases where such data may be incomplete or do not exist.

The group agreed that the data files would include 640-group reaction cross sections as well as the original data files in the general ENDF format and pointwise data, where available. Whereas the 640-group files may be adequate for most dosimetry applications, the general ENDF or pointwise data files may be needed in some cases, where self-shielding is important and/or for higher temperatures applications.

4.2. The Need for New Evaluations or Revisions to Existing Files

Eva Zsolnay has examined data files from the sources listed in 4.1 and performed integral testing, as described in her summary report in Appendix 5. The evaluated data files were found to have a number of inconsistencies or format errors, as described below. Furthermore, it was recommended that other data files such as JEFF-3.0, ENDF/B-VI, and CENDL/D should be studied to see if the files contain data that might be useful for IRDF-2002. A. Nouri and L. Greenwood agreed to look at JEFF-3.0 and ENDF/B-VI, respectively, to see if any new evaluations might be available that have not previously been considered.

K. Zolotarev has completed new evaluations for 56Fe(n,p)56Mn and 58Ni(n,p)58Co that will be considered for IRDF-2002, when they are released in December 2002. He is also working on new evaluations for 27Al(n,p)27Mg and 237Np(n,f) that will not be completed until March 2003. These latter evaluations may not be completed in time for the first version of IRDF-2002; however, they could be included in a later revision of the file. K. Zolotarev has also examined the available data for the 54Fe(n,g)55Fe and 62Ni(n,g)63Ni reactions that are useful for retrospective reactor dosimetry. However, these files need considerable work before they could be included in the new IRDF-2002 library.

4.3. New Data Evaluations Needed and Problems with Files

Eva Zsolnay tested the available dosimetry data files from IRDF-90, JENDL/D-99, and RRDF-98. A number of problems were found including file format errors as well as inconsistencies with the covariance files. The following table summarizes the problems that were found with the JENDL/D-99 dosimetry data file, as well as assigned actions to resolve the problems.

PROBLEMS WITH REACTION CROSS SECTIONS (IN THE REACTOR DOSIMETRY FILE JENDL/D-99) TO BE USED IN IRDF2002

4.3.1) Problems to be solved by Dr. SHIBATA (see detailed description in [1])

Reaction name	Problem to be solved
10B(n,a)	Format error (MT No.)
^{NAT}Ti (n,x)48Sc	Energy limits in Files 3 and 33 do not agree!
58 Fe(n,g)	-Zero or negative eigenvalues in the cov. info; -Eff. rank of the cov. matrix is smaller than the size; -Covariance information partly estimated, partly artificial.
54Fe(n,p)	Energy limits in Files 3 and 33 do not agree!
59Co(n,2n)	Energy limits in Files 3 and 33 do not agree!
63Cu(n,2n)	Energy limits in Files 3 and 33 do not agree!
65Cu(n,2n)	Energy limits in Files 3 and 33 do not agree!
199Hg(n,n')	Format error.

[1] E. M. Zsolnay, H. J. Nolthenius, E. J. Szondi: Nuclear Data for Dosimetry Libraries: Analysis, Intercomparison and Selection of Data. Progress Report, BME-NTI-251/2001. INT BUTE, Budapest, 2001 September

4.3.2) Further troublesome reaction cross sections in JENDL/D-99 to be substituted by other up-to-date evaluations

a) From ENDF/VI, if new revision is available (L. Greenwood assigned)

Reactions of interest: 45Sc(n,g), 50Cr(n,g), 47Ti(n,p), 60Ni(n,p).

b) From JEFF-3.0 (A. Nouri assigned)

Reactions of interest: 60Ni(n,p), 169Tm(n,2n), 235U(n,f) (also new uncertainty data from Oak Ridge)?

K. Zolotarev was notified of several problems with RRDF-98 and he has already corrected all of the errors. Some of the data files in JENDL/D-99 and RRDF-98 included new cross section evaluations; however, some of the covariance files in JENDL/D-99 were unfortunately taken from the older evaluations in IRDF-90 or other sources. Hence, in these cases, the covariance files do not match the cross section evaluations. It was agreed that all of the formatting errors and covariance file problems cannot be fully resolved for the first edition of IRDF-2002, although an effort will be made to fix the problems where possible. More difficult problems may require a new evaluation effort. In a few cases, it was agreed that L. Greenwood would look for new general data files in ENDF/B-VI and A. Nouri would look for files in JEFF-3.0 that might be better than the data in the three dosimetry libraries.

4.4 Evaluated Decay and Isotopic Abundance Data

The participants agreed that nuclear decay data should be included with IRDF-2002. O. Bersillon agreed to process the ENDSF data files to produce files in an ENDF 6 format for inclusion with IRDF-2002. There was also agreement that the main decay information should also be listed in plain text, if possible, to facilitate reading of the data by the users. This information consists of the half-life data and, for gamma emitters, only 3 or 4 of the main lines and intensities. For a few cases, positron emission, beta decay, or x-ray emission lines will be needed. Uncertainties should be provided for all of the data.

In principle, the nuclear decay data provided in IRDF-2002 should be the same data that were used to produce the evaluated cross section files. Unfortunately, these data are not present in the original data evaluations and it would take considerable effort to determine, if they can still be determined from the older evaluations. The participants agreed that this is a serious shortcoming of the present dosimetry data files that cannot be easily corrected in the new IRDF file. IRDF-2002 will thus only include the evaluated decay data that is currently available in ENDSF with the recognition that this may lead to some inconsistencies with the use of the cross section files.

Isotopic abundance data will also be included with IRDF-2002. O. Bersillon will provide the most recent list of recommended data.

4.5 Radiation Damage Data Files

It was agreed that IRDF-2002 should include the currently available standard dpa cross section files from ASTM, including C, Si, Fe, and GaAs. IRDF-90 also included dpa cross sections for Ni and Cr, and L. Greenwood agreed to provide updated files for these elements. More generally, the SPECTER computer code can be included with the new IRDF-2002 data package since SPECTER provides dpa and pka spectra for over 40 elements, although the data files were based on ENDF/B-V data.

4.6 Cross Section Data Needed for Retrospective Dosimetry

L. Greenwood discussed the need for additional reaction cross section evaluations that are useful for retrospective reactor dosimetry, namely, 54Fe(n,g)55Fe, 62Ni(n,g)63Ni, and 93Nb(n,g)94Nb. Mr. K. Zolotarev investigated the nuclear data that is available for the first two reactions. Unfortunately, the data quality is very poor and considerable work would be needed to provide evaluated cross sections for these reactions. Mr. Zolotarev agreed to look at this problem, but a new evaluation cannot be completed in time for IRDF-2002.

K. Zolotarev also presented the current status of the data for the 93Nb(n,g)94Nb reaction. This reaction was previously included in IRDF-90. However, there are some gaps in the available data and the present covariance data file is completely inadequate. More effort is needed to improve the quality of the covariance data; however, there was concern that this might not be possible in time for IRDF-2002.

4.7 Integral Testing in Benchmark Neutron Fields

As discussed above, R. Paviotti-Corcuera presented the preliminary summary of a supplementary workshop on benchmarks that was held at the International Symposium on Reactor Dosimetry in Brussels, Belgium, the week prior to our meeting (see Appendix 4). The workshop summary recommends that only four reference fields be used for the primary testing of the various dosimetry cross section libraries, namely, a thermal Maxwellian, a 1/E slowing down spectrum, the Cf-252 neutron source, and a 14 MeV source. W. Mannhart recently evaluated the currently available dosimetry files for the case of the Cf-252 neutron spectrum (see Appendix 5).

There was some discussion about the best source of experimental data for the other three cases cited in the workshop recommendation. The BNL website refers to the most recent version of BNL-325 as the best source of data on thermal cross sections and resonance integrals. This point needs to be investigated to see if any more recent data evaluations have been performed. In any case, the data to be considered for IRDF-2002 from the three dosimetry cross section files should be compared with the best available thermal and resonance integral data.

Similarly, there was discussion about the best source of information on 14 MeV experimental data. W. Mannhart mentioned several evaluations that might be considered and some effort will be needed to determine the best currently available source of 14 MeV data. These experimental data can then be easily compared with the various sources of dosimetry cross section data.

The above comparisons will form the primary basis for the selection of the best cross section data to be used for IRDF-2002. However, additional consistency checks on the data have already been performed in other available well-documented neutron fields including CFRMF, HFR, and RTN. P. Griffin presented a new neutron benchmark field ACRR (Annular Core Research Reactor) at the Brussels meeting and this field can also be used to test the consistency of the dosimetry cross sections.

4.8 Other Computer Codes for Distribution with IRDF-2002

Other computer codes will be considered for distribution with IRDF-2002; however, this decision can be delayed since such files can be provided independently from the main task of preparing the new IRDF-2002 file.

L. Greenwood has made available the most recent PC versions of the SPECTER computer code for radiation damage calculations and STAY'SL for spectral adjustment. O. Bersillon will provide the computer code SDF2NDF used to process the nuclear decay data from ENDSF to ENDF 6 format. It was agreed that these codes would be distributed only if they are adequately documented and provide a benefit to the potential users of IRDF-2002.

4.9 Assigned Tasks and Schedule

The summary report of this meeting should be issued by the end of October 2002. The plan is to then issue the test version of IRDF-2002 by May 2003. A TECDOC will accompany the data file to fully document the individual cross section files as well as the integral testing that was performed to validate the files. A detailed schedule is, as follows:

#	Action	Responsible Persons	Deadline
1	Correction of cross section data	K. Shibata	Oct. 29, 2002
2	Search for new data evaluations	L. Greenwood, A. Nouri	Oct. 29, 2002
3	Processing of data from new evaluations	A. Nouri, L. Greenwood	Nov. 18, 2002
4	Provide new evaluations for 56Fe(n,p)56Mn and 58Ni(n,p)58m+gCo	K. Zolotarev	Nov. 18, 2002
5	Review (analysis) of new data	E. Zsolnay	Nov. 30, 2002
6	Search for good quality measured data in reference fields	L. Greenwood, E. Zsolnay, W. Mannhart	Nov. 30, 2002
7	Provide list of reactions to O. Bersillon for decay data evaluation	E. Zsolnay	Nov. 30, 2002
8	Calculation of integral cross section data for reference neutron fields and send results to R. Paviotti-Corcuera	Cf-252 : W. Mannhart Thermal, 1/E: E. Zsolnay 14 MeV: L. Greenwood	Jan. 10, 2003
9	Send calculated and measured data for reference fields to W. Mannhart,	R. Paviotti-Corcuera	Jan. 27, 2003

#	Action	Responsible Persons	Deadline
	L. Greenwood, and E. Zsolnay		
10	Provide nuclear decay data for reactions in point 6	O. Bersillon	Jan. 30, 2003
11	Provide dpa cross sections for inclusion in IRDF-2002	L. Greenwood, P. Griffin	Jan. 30, 2003
12	Preliminary selection of cross sections for IRDF-2002	W. Mannhart, E. Zsolnay, and L. Greenwood	Feb. 20, 2003
13	Distribution of list of cross section selections for IRDF-2002 to K. Zolotarev and other participants	R. Paviotti-Corcuera	Mar. 15, 2003
14	Response from meeting participants on selection of cross sections to R. Paviotti-Corcuera	Meeting Participants	Mar. 31, 2003
15	Final selection of cross sections for IRDF-2002	W. Mannhart, E. Zsolnay, and L. Greenwood	April 15, 2003
16	Assemble, check, and edit IRDF-2002 Data File (ENDF 6 Format and 640 Group Format) and release test version of IRDF-2002 file	IAEA-NDS	May 31, 2003
17	Draft of sections of TECDOC submitted to R. Paviotti-Corcuera	All assigned actions in 4.10	June 15, 2003
18	Technical Meeting at IAEA Vienna	IAEA-NDS	July 15-17, 2003
19	Release of final version of IRDF-2002 and TECDOC	IAEA-NDS	Oct. 31, 2003

All relevant information related to the project (results, data, and important e-mail messages) will be interchanged and copied to the co-ordinator of the project (Raquel Paviotti-Corcuera).

4.10 Contents of the IAEA TECDOC

The IRDF-2002 data file will be documented in an IAEA TECDOC. The document will contain the following sections, with assigned actions:

Foreword – (R. Paviotti-Corcuera)

Introduction – (R. Paviotti-Corcuera)

Table of Contents – (R. Paviotti-Corcuera)

Content of the Data Library – this section should include the list of the reactions and the source of the evaluated data file (E. Zsolnay).

New Russian Data Evaluations – (K. Zolotarev)

Selection Process – Description of the process used to select data for IRDF-2002 with references to the detailed integral testing that were performed – (E. Zsolnay).

Validation of the Dosimetry Data with Integral Experiments – (W. Mannhart).

Benchmarking of the Dosimetry File in Reference Fields – (W. Mannhart and E. Zsolnay).

Radiation Damage Files and Computer Codes– (L. Greenwood and P. Griffin).

Decay Data and Isotopic Abundances for Dosimetry Applications – (O. Bersillon).

Plots of the Cross Sections (on CD-ROM) – (V. Zerkin).

Software Included on CD-ROM -(A. Nouri) – Section should contain a description of all the software such as programs to read ENDF 6 data files, plotting programs (ZVVIEW), radiation damage, spectral adjustment, etc.

5. CONCLUSIONS

The participants emphasized that accurate and complete knowledge of nuclear data for reactor dosimetry are essential for improving the accuracy of the reactor pressure vessel service life assessment of nuclear power plants and in other neutron metrology applications such as boron neutron capture therapy, the therapeutic use of medical isotopes, nuclear physics measurements, and reactor safety applications. It is important that the participants should maintain the schedule given in section 4.9 to ensure that the IRDF-2002 file will be issued in a timely manner and that benchmarking of the data can be completed in the available standard neutron fields before the end of October 2003.

6. APPENDICES

Appendix 1: Agenda

International Atomic Energy Agency

Data Development Project: Technical Meeting on

International Reactor Dosimetry File: IRDF-2002

IAEA Headquarters, Vienna, Austria

27 - 29 August 2002

Meeting Room A-0742

AGENDA

Tuesday, 27 August

08:30 - 09:30 **Registration** (at Gate 1, IAEA Headquarters)

09:30 - 10:30 **Opening Session:**

- Welcoming address - Alan Nichols, Head of Nuclear Data Section (NDS)
- Round table self-introduction by participants
- Election of Chairman and Rapporteur
- Discussion and adoption of Agenda (Chairman)
- General Considerations for IRDF-2002 (R. Paviotti, Scientific Secretary)

10:30 - 10:45 Coffee break

10:45 - 12:30 **Session 1: Presentations by Participants, and Discussions**
(15 minutes for each presentation, and 5 minutes for discussion):

1. Comparison of Cross Section Data and their Uncertainties, **Eva Zsolnay**, Budapest University of Technology and Economics, Budapest, Hungary.
2. Revision and Evaluation of Dosimetry Cross Sections, **K. I. Zolotarev**, Institute of Physics and Power Engineering, Obninsk, Russia.
3. Cross Sections Graphics and Specialized Software for Evaluators, **Viktor Zerkin**, Nuclear Data Section, International Atomic Energy Agency, Vienna, Austria.
4. Average Cross Sections Calculated in Various Neutron Fields, **Keechi Shibata**, Nuclear Data Centre, Japan Atomic Energy Research Institute, Japan.

12:30 - 14:00 **Lunch and Administrative/Financial Matters**

14:00 - 17:00 Session 1: Presentations by Participants, and Discussions (cont.)

5. Validation of Differential Cross Sections with Integral Data, **Wolfgang Mannhart**, Physikalisch-Technische Bundesanstalt, Braunschweig, Germany.
6. Neutron Spectral Adjustment and Radiation Damage Calculations For Reactor Dosimetry, **Larry R. Greenwood**, Pacific Northwest National Laboratory, USA.
7. Decay Data for Reactor Dosimetry Applications, **Olivier Bersillon**, CEA Bruyeres-le-Chatel, France.

17:15 Reception, NDS floor A-23 (adjacent to room A-2340)

Wednesday, 28 August

09:00 - 12:30 Session 2: Discussions

Reactions to be included in IRDF,

Need for new evaluations and revisions?

Evaluated decay data for what nuclides?

More dpa data, for what nuclides?

Extra data needed for retrospective dosimetry?

What benchmark fields should be used?

Other issues?

Assignment of tasks, including name and contents of the package,

TECDOC: structure and individual writing assignments.

[Coffee break when appropriate]

12:30 - 14:00 Lunch

14:00 - 18:00 Session 2: Discussions (cont.), drafting of the Meeting Report

Thursday, 29 August

09:00 - 12:30 Session 2: Drafting of the Meeting Report (cont.), and Conclusions

[Coffee break when appropriate]

12:30 - 14:00 Lunch

14:00 - 17:30 Concluding Session - Discussion and Approval of Meeting Report

[Coffee break when appropriate

Appendix 2: List of Participants

International Atomic Energy Agency

Technical Meeting on the Data Development Project

"International Reactor Dosimetry File: IRDF-2002 "

IAEA Headquarters, Vienna, Austria

27 to 29 August 2002

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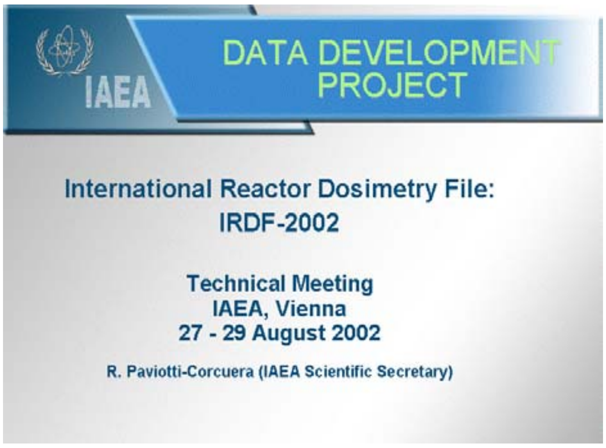
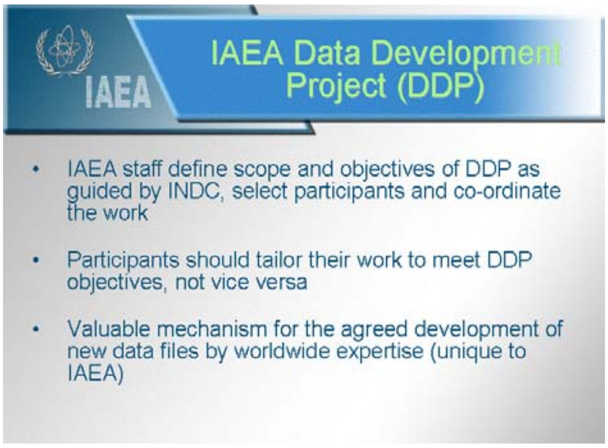
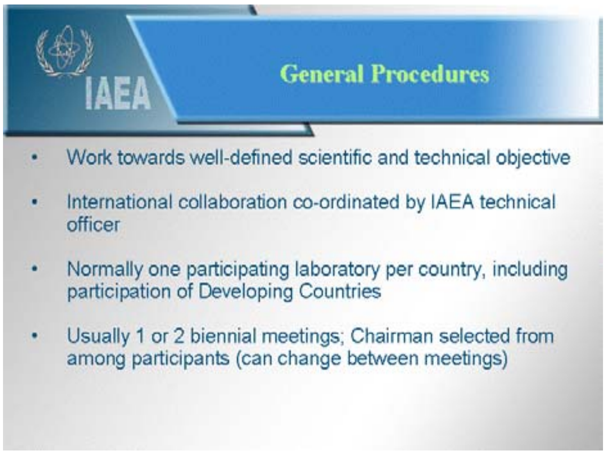
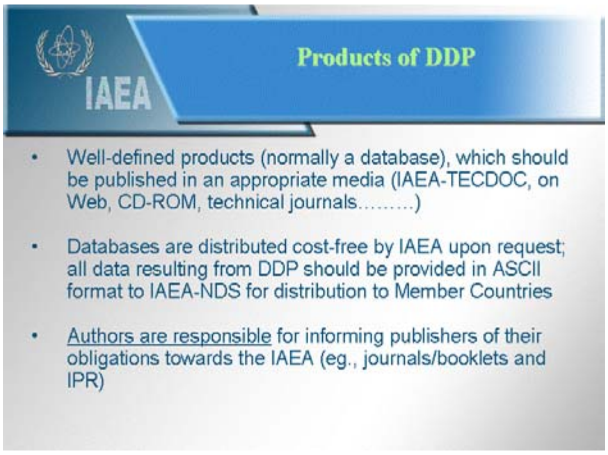
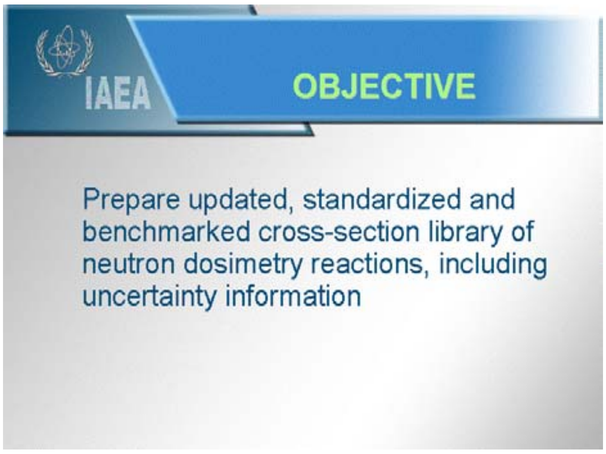
Appendix 3: Framework of the Data Development Project and Objective of the Meeting

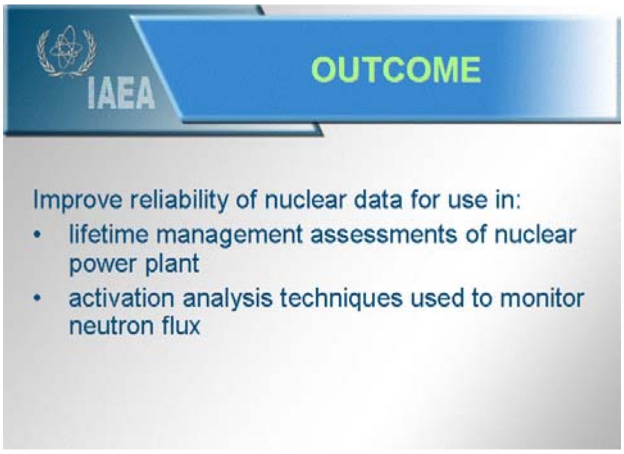












Participants

BUTE, Hungary

- Compare cross section and uncertainty data in IRDF-90.2, RRDF-98 and JENDL-DF-99
- Highlight specific needs for revision of physics consistency and format in both libraries

Participants

IPPE, Russian Federation

- New evaluations for
  $^{139}La(n,\gamma)^{140}La$, $^{186}W(n,\gamma)^{187}W$ and $^{204}Pb(n,n')^{204m}Pb$
- Revision of 14 reactions
  $^{19}F(n,2n)^{18}F$, $^{46}Ti(n,2n)^{45}Ti$, $^{46}Ti(n,p)^{46m+g}Sc$, $^{47}Ti(n,x)^{46m+g}Sc$, $^{48}Ti(n,p)^{48}Sc$, $^{48}Ti(n,x)^{47}Sc$, $^{49}Ti(n,x)^{48}Sc$, $^{51}V(n,\alpha)^{48}Sc$, $^{54}Fe(n,\alpha)^{51}Cr$, $^{54}Fe(n,2n)^{53m+g}Fe$, $^{59}Co(n,\alpha)^{56}Mn$, $^{63}Cu(n,\alpha)^{60m+g}Co$, $^{75}As(n,2n)^{74}As$ and $^{141}Pr(n,2n)^{140}Pr$

Participants

IAEA-NDS

- Specialized software for evaluators
- Compare evaluated cross sections from the different nuclear data libraries with available experimental data (EXFOR) – graphical plots

Participants

Physikalisch-Technische Bundesanstalt (PTB), Germany

- Experimental fission spectrum averaged cross sections for ^{252}Cf and ^{235}U + covariance information for 30 dosimetry reactions
- Compare experimental and calculated spectrum averaged cross sections for ^{252}Cf and ^{235}U (IRDF-90.2 and JENDL-DF-99

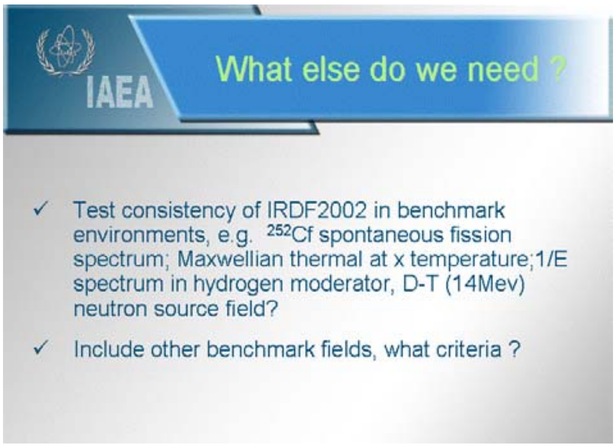
Participants

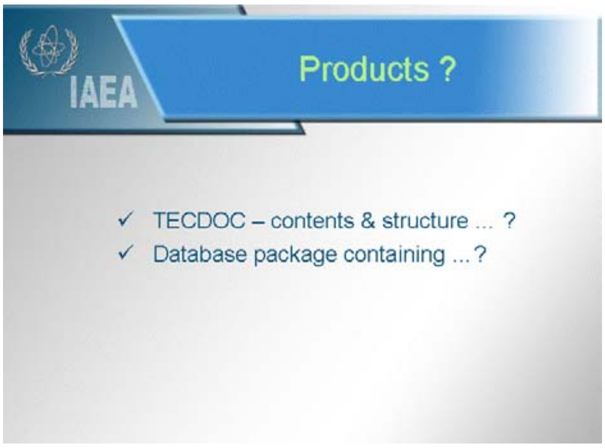
Pacific Northwest National Laboratory
and Sandia National Laboratories, USA

- Displacement per atom (dpa) data for Fe, Si, GaAs consistent with latest ASTM Standards, as derived from the ENDF/B-VI and NRT treatment
- SPECTER (dpa and gas production calculations) and STAY'SL (spectral adjustment)

What else do we need ?

- ✓ Corrections to JENDL/D-99 ?
- ✓ Include evaluated decay data in the files
- ✓ Evaluations or revisions of important reactions for retrospective reactor dosimetry, e.g. $^{54}Fe(n,\gamma)^{55}Fe$ and $^{62}Ni(n,\gamma)^{63}Ni$


IAEA
What else do we need ?
✓ Test consistency of IRDF2002 in benchmark environments, e.g. ^{252}Cf spontaneous fission spectrum; Maxwellian thermal at x temperature;1/E spectrum in hydrogen moderator, D-T (14Mev) neutron source field?
✓ Include other benchmark fields, what criteria ?


IAEA
Products ?
✓ TECDOC – contents & structure ... ?
✓ Database package containing ... ?

Appendix 4: Summary of the Supplementary Workshop on Testing of the IRDF-2002 File

DRAFT

Eleventh International Symposium on Reactor Dosimetry Summary of the Workshop on Benchmarks and Intercomparisons

Chairpersons: B. Osmera and J. Williams

Contributors:

Bohomil Osmera	Czech Republic	WWER Mock-up Benchmarks, EU Project REDOS
Sergei Zaritsky	Russia	New Tasks for WWER-1000 Benchmarking
Gregori Shimanski	Russia	The KORPUS Benchmark Facility
Patrick Griffin	USA	ASTM Standards on Neutron Benchmarks
Stan Anderson	USA	Benchmarks Listed in Reg. Guide 1.109
David Gilliam	USA	Measurement Assurance for Neutron Dosimetry
Klaas van der Meer	Belgium	VENUS Benchmark experiments
Parvin Lippincott	USA	BWR Benchmarks
Bohomil Jansky	Czech Republic	Iron Sphere Benchmark
Harm Wienke	IAEA	Cylindrical Slab Benchmark for Iron

Supplementary Workshop on Testing of the IRDF-2002 File

Contributors:
J. Williams, USA, and B. Osmera, Czech Republic, Co-Chairs
R. Paviotti-Corcuera, IAEA   Testing of the IRDF-2002 File
H. Nolthenius, Holland
P. Griffin ,USA
S. Zaritsky, Russia
D. Gilliam, USA
J. Wagschal, Israel
E. P. Lippincott, USA
B. Boehmer, Germany
K. van der Meer, Belgium
Other participants of the Workshop on Benchmarks and Intercomparisons

Dr. Paviotti-Corcuera requested recommendations concerning the benchmark-field testing to be undertaken prior to the release of the new International Reactor Dosimetry File IRDF-2002.

Dr Griffin, who co-chaired the Workshop on Cross Section Files and Uncertainties concurred that it was appropriate to discuss this question, since it had not been fully dealt with in that workshop. Accordingly a special workshop session of one hour was appended to the Workshop on Benchmarks and Intercomparisons, specifically to deal with this important question.

Regarding the proper selection of fields for testing of the new IRDF-2002 file prior to its release, with possible implications concerning the selection of which library files to include in it, the following recommendations were agreed.

1. Only standard neutron dosimetry fields are suitable for this purpose. Cross section consistency tests in the ^{235}U thermal-fission neutron reference field and in other benchmark fields should be done after the file contents have been established. Standard neutron fields, for this purpose, are understood to be those fields which are permanent and reproducible and which, in the energy range of their principal response, are characterized to state-of-the art accuracy by means of differential spectrometry and/or by fundamental physical laws. For the energy range of interest for the file (up to 20MeV) only four fields meet these criteria:
   - The Maxwellian thermal spectrum at specified neutron temperature;
   - The 1/E slowing down spectrum in hydrogenous moderator;
   - The spontaneous fission neutron field of ^{252}Cf;
   - The monoenergetic 14-MeV neutron field from a D-T source.
2. In each case evaluated data from worldwide implementations of the ideal embodied in the standards, accounting for necessary corrections applicable to their actualization, are suitable for the testing.
3. Only integral data with covariance information that have appeared in peer-reviewed evaluations may be used, and their values together with covariances should be included for distribution with the file.
4. Where the spectrum representations used in the testing depend on differential neutron spectrometry data, they must also be the result of peer-reviewed evaluations, including covariances, and they should be included for distribution with the file.
5. It is understood that some cross sections to be included in the file will not have suitable measurements of integral quantities meeting the above criteria. In those cases the selection of evaluated differential data will depend only on the qualities intrinsic to the differential data evaluations. In any case that should be the mean criterion for selection of evaluated differential data, even when suitable integral data in standard fields are also used.

These recommendations are based on those of the 1978 IAEA Consultants Meeting on Reactor Dosimetry. The definition of a Standarnd Neutron Field for reactor dosimetry appears on p.6 of the Proceedings and the definition of a Category I reaction appears on p.32 . With minor revisions these are applicable to the testing of IRDF-2002.

Reference: Proceedings of a Consultants Meeting on Reactor Dosimetry, Report IAEA-208, IAEA, Vienna, Vol. I, (1978).

Appendix 5: Presented Papers

ANALYSIS AND INTERCOMPARISON OF THE CROSS SECTION AND RELATED UNCERTAINTY DATA PRESENT IN THE REACTOR DOSIMETRY LIBRARIES IRDF-90, JENDL/D-99 AND RRDF-98+

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1. Introduction

The last and tested version (Ver.2) of the International Reactor Dosimetry File, IRDF-90 was released in 1993 [1-3], and another form of it was published in 1994, with improved format [4]. This library has already become old and, better quality cross section data have become available for a number of dosimetry reactions during the last years. The reliability of the reactor dosimetry results, used in the service life assessment of NPPs, requires that good quality input data should be applied in the related calculations. Therefore, the Nuclear Data Section of IAEA started a project in 2001 [5] for updating the old reactor dosimetry file
IRDF-90. The data for the new library (entitled IRDF-2002) should derive from tested, up-to-date reactor dosimetry files or evaluations.

Two new reactor dosimetry libraries have been published from the time of releasing IRDF-90: JENDL/D-99 (Japan, [6]) and RRDF-98 (Russia [7]). As the uncertainities of the cross sections in the new libraries are significantly smaller for a number of reactions, furthermore, several new reactions can be found in these files, as compared with IRDF-90, they can be considered as potential sources for the up-dating procedure. Therefore, the content of these files has been analysed, tested and, intercompared with the corresponding data of
IRDF-90.

This paper presents the results of the analysis and intercomparison, and gives a first proposal on the cross sections from the new files, to be included in IRDF-2002.

2. Methods of the Analysis and Intercomparison

The analysis and intercomparison of the cross section data involved the following actions:

a) ***For numerical characterization (and intercomparison) of the cross section data of the files JENDL/D-99 and RRDF-98,*** spectrum averaged cross section values were calculated for three theoretical neutron spectrum functions: Maxwellian thermal spectrum (at a neutron temperature of 293.58 K), 1/E spectrum (between 0.563 eV and 1.05 MeV) and Watt fission spectrum. The data obtained, were then compared with each other, and with the corresponding ones of the library IRDF-90.

+Modified version of the paper ″Results of Testing the Cross Section and Related Uncertainty Data to be Used in the New International Reactor Dosimetry File IRDF-2002″, presented at the Eleventh International Symposium on Reactor Dosimetry, Brussels, 18-24 October, 2002.

b) ***Detailed uncertainty analysis*** was performed involving the examination of the covariance matrices present in the libraries of question. For representation of the cross section uncertainty data the same energy group structure was used as described above (except, that the lower limit of the fast neutron energy group was 1.05 MeV). For derivation of the covariance information in the three energy groups of interest the cross section processing code X333 was used [8]. A typical MTR spectrum available in 640 SAND II groups format [9] was applied as weighting spectrum in the input of the cross section uncertainty processing. In order to show a more detailed picture as a function of energy on the inconsistencies found during the analysis, the relevant covariance information was calculated also in an extended (27 groups) ABBN group structure [10].

c) ***Consistency test was carried out in some reference neutron spectra*** of the Neutron Metrology File NMF-90 [11]. The spectra were derived for the following neutron fields:
- Pool Side Facility of the ORR reactor, simulated surveillance position (Oak Ridge, USA; spec. PS1, detectors covered by Gd)
- Coupled Fast Reactivity Measurement Facility (CFR, Idaho, USA; spec. CFR)
- Fusion Simulation Spectrum, Measured at the RTNS-II (LLL, USA, spec. RTN)
- High Flux Reactor (HFR, JRC Petten, NL; spec. HFR)

For the consistency test the neutron spectrum adjustment code STAYNL [12] was used.

d) ***Detecting of errors and inconsistencies in the new libraries*** involved the recognition both of the format errors and of the inconsistencies from physics and/or mathematics point of view.

3. Results

During the analysis and intercomparison outlined above, altogether 141 dosimetry reaction cross sections were investigated: 53 ones from IRDF-90, while 64 and 24 ones from the files JENDL/D-99 and RRDF-98, respectively (the reaction ^{12}C(n,2n) from RRDF-98, was not considered). The results obtained are as follows [10]:

1) In the three energy group representation, defined above, only a small number of cross sections present in the new libraries showed important (>5%) deviations from the corresponding IRDF-90 data (see Table 1).
2) At the same time, for a large number of reactions important differences appeared in the uncertainty values of the JENDL/D-99 and IRDF-98 cross sections, as compared with the old file IRDF-90. For about 50 % of the IRDF-90 cross sections improvement could be detected in the uncertainty information present in the new libraries. Table 2 shows the relative standard deviation values of the cross sections having a deviation larger than about a factor of 2 in the uncertainty data, as compared with the corresponding IRDF-90 values.

   It has to be noted that, in some cases of the JENDL/D-99 library diagonal covariance matrices were used, therefore, the improvement in the uncertainty values is not always unambiguous. For these cases revision of the corresponding covariance information was suggested to the evaluators [10].
3) In the consistency test (actually neutron spectrum adjustment) the C/E values for the reaction rates present in Table 3 were analysed (together with the corresponding uncertainty information) in the neutron fields defined above. It means that in case of RRDF-98 only four reaction cross sections (Ti46P, TI48P, FE56P, CU63A), in case of JENDL/D-99 twentyfour reaction cross sections (LI6A, B10A, NA23G, AL27P, AL27A, SC45G, TI46P, TI47P, TI48P, FE54P, FE56P, FE58G, CO592, CO59G, NI582, NI58P, NI60P, CU63G, CU63A, IN115N, IN115G, AU197G, TH232F, TH232G, U235F, U238F, U238G, NP237F, PU239F) could be tested.

For the four reactions of the RRDF-98 no significant difference was found in the C/E values, as compared with the corresponding IRDF-90 data. At the same time, in the uncertainty information the same trend could be seen as mentioned in the previous point.

Table 1. Ratio of average cross sections with a deviation >5 %,
to the corresponding IRDF-90 data.

JENDL-D-99/IRDF-90					RRDF-98/IRDF-90				
No	Reaction	Maxw. spec.	Res. int.	Watt spec.	No	Reaction	Maxw. spec.	Res. int.	Watt spec.
1	F192	-	-	1.092	1	F192	-	-	0.944
2	AL27P	-	-	1.104	2	TI46P	-	-	1.116
3	S32P	-	-	1.067	3	TI47NP	-	-	0.802
4	TI46P	-	-	1.103	4	TI48P	-	-	1.109
5	TI47NP	-	-	0.654	5	FE56P	-	-	1.079
6	FE58G	1.126	0.903	-	6	NB93N	-	0.899	1.025
7	CO59A	-	-	1.074					
8	NI60P	-	-	0.891					
9	CU63A	-	-	1.111					
10	ZN64P	-	-	0.902					
11	AG109G	0.889	1.035	-					
12	NP237F	1.214	1.443	0.984					

With the JENDL/D-99 cross section data *better consistency* (C/E values closer to unity) was obtained, than with the corresponding IRDF-90 values for the following reactions: in spectrum PS1 – SC45G, TI46P, FE58G, NP237F; in spectrum HFR – TI46P, FE58G; in spectrum CFR – TI46P, TI47P, FE58G, CU63G, IN115G, NP237F; and in spectrum RTN – TI46P, TI48P, FE54P. *Worse consistency* (C/E values deviating more from unity) was obtained with the JENDL/D-99 cross section data, than with the corresponding IRDF-90 values for the following reactions: in spectrum PS1 – NI58P; in spectrum CFR – AL27P, SC45G, NI58P, AU197G, TH232F, TH232G; and in spectrum RTN – NI60P. As the response region of the same detector is different in the various neutron spectra considered here, the results of the consistency test clearly show in what energy region the cross sections present in JENDL/D-99 have been changed as compared with the corresponding IRDF-90 data, and confirm the findings in Table 1.

4) Besides the investigations presented above, the contents of the libraries JENDL/D-99 and IRDF-98 were also compared with each other for the common reactions. The results obtained were taken into consideration in selecting the cross sections for the new reactor dosimetry file.
5) Several errors and discrepancies were detected during the analysis (lacking or erroneous cross section values in certain energy regions; lacking, insufficient and/or erroneous uncertainty information, etc.). The findings were communicated to the evaluators of the cross section files via IAEA NDS [10]. As a result, several cross section and uncertainty data in RRDF-98 have been revised, but no response has been received from the evaluators of JENDL/D-99 yet.

Table 2. Relative standard deviations for the cross sections deviating in their uncertainties by about a factor of 2 (or more) from the corresponding IRDF-90 data
(for a typical MTR spectrum)

No	Reaction name	Relative uncertainty for the spectrum part (%)								
		Maxwellian thermal			Intermediate			Fast		
		IRDF-90	JENDL/D-99	RRDF-98	IRDF-90	JENDL/D-99	RRDF-98	IRDF-90	JENDL/D-99	RRDF-98
1	LI6A	0.14	0.40	-	0.14	0.40	-	1.22	4.72	-
2	B10A	0.16	0.22	-	0.16	0.33	-	1.32	13.5	-
3	MG27P	-	-	-	-	-	-	2.26	1.24	1.14
4	AL27P	-	-	-	-	-	-	5.92	0.72	-
5	P31P	-	-	-	-	-	-	3.60	1.34	-
6	S32P	-	-	-	-	-	-	4.65	8.34	-
7	SC45G	2.65	0.83	-	3.92	1.13	-	12.8	8.85	-
8	TI46P	-	-	-	-	-	-	5.04	2.27	3.13
9	TI47NP	-	-	-	-	-	-	30.0	2.61	8.58
10	TI47P	-	-	-	-	-	-	3.84	1.43	-
11	TI48NP	-	-	-	-	-	-	30.0	2.65	8.59
12	TI48P	-	-	-	-	-	-	32.4	1.85	5.17
13	CR522	-	-	-	-	-	-	2.68	1.29	-
14	FE54P	-	-	-	-	-	-	2.18	0.99	-
15	FE58G	18.5	12.6	-	9.36	8.72	-	19.1	4.81	-
16	CO592	-	-	-	-	-	-	2.85	1.45	-
17	NI582	-	-	-	-	-	-	3.11	0.90	-
18	NI60P	-	-	-	-	-	-	40.3	19.4	-
19	CU63G	4.11	2.00	-	4.03	1.38	-	7.69	20.0	-
20	CU63A	-	-	-	-	-	-	10.8	1.46	2.84
21	ZN64P	-	-	-	-	-	-	4.80	1.63	-
22	Y892	-	-	-	-	-	-	4.28	1.45	-
23	ZR902	-	-	-	-	-	-	1.60	0.55	-
24	NB932	-	-	-	-	-	-	2.80	4.44	1.06
25	NB93N	-	-	-	7.37	20.1	4.68	3.01	2.78	2.80
26	AU1972	-	-	-	-	-	-	4.28	1.18	-
27	AU197G	0.14	0.74	-	0.17	3.04	-	-	-	-
28	U235F	0.19	0.32	-	0.27	2.13	-	0.35	2.22	-
29	U238F	-	-	-	-	-	-	0.54	2.09	-
30	U238G	0.35	0.74	-	0.37	3.45	-	2.03	10.0	-
31	NP237F	30.0	5.40	-	9.53	0.59	-	9.34	0.30	-
32	PU239F	0.25	0.71	-	0.26	4.06	-	0.43	2.05	-

Table 3. Reactions used in the consistency test

Spectrum	Reactions used
PS1	SC45G, TI46P, FE54P, FE58G, CO59G, NI58P, CU63A, U235F, NP237F
HFR	NA23G, AL27A, SC45G, TI46P, TI48P, FE54P, FE56P, FE58G, CO59G, NI58P, CU63G, IN115N, IN115G, AU197G, TH232G, U235F, U238G
CFR	LI6A, B10A, AL27P, AL27A, SC45G, TI46P, TI47P, TI48P, FE54P, FE58G, CO59G, NI58P, CU63G, IN115N, IN115G, AU197G, TH232F, TH232G, U235F, U238F, U238G, NP237F, PU239F
RTN	AL27A, SC45G, TI46P, TI47P, TI48P, FE54P, CO592, CO59G, NI582, NI58P, NI60P, AU197G

4. Conclusions – recommendations

Based on the results of the analysis and intercomparison outlined above, the following conclusions and recommendations have been made:

a) The majority of the cross sections present in the new libraries did not show important (>5%) deviations from the corresponding data of the file IRDF-90. The striking difference (in most cases improvement) could be detected in the uncertainty information of the new files, as compared with the relevant IRDF-90 values.
b) The first proposal on the cross sections from the analysed libraries, to be involved in the new International Reactor Dosimetry File IRDF-2002, can be seen in Table 4. This proposal contains cross sections together with uncertainty information in the form of covariance matrices, for 70 dosimetry reactions: 22 of them originating from the file IRDF-90, 26 ones from JENDL/D-99 and, 22 ones from the library RRDF-98.
c) However, the data of the dosimetry library JENDL/D-99 will have to be revised by the evaluators, and the discrepancies and errors [10] found in the cross section data of interest (see in Table 4) will have to be corrected. The evaluators of the JENDL/D-99 are kindly asked to do this task in the near future in order that, the new international reactor dosimetry library IRDF-2002 could be edited before the end of this year .
d) In the near future, further new, good quality cross section evaluations will have to be looked for inclusion in IRDF-2002.
e) After the evaluators of the file JENDL/D-99 (and RRDF-98) have corrected the discrepancies/format errors for the reactions of interest, the analysis and intercomparison outlined above, will have to be repeated both for the corrected and for the new data.
f) Based on the results obtained in poits d) to e), reconsideration of the list of the cross sections in Table 4 will be needed.
g) The radiation damage and gas production cross sections for the materials interesting for the fission (and fusion) reactor dosimetry will have to be collected from new evaluations (e.g. new ASTM standards, the library of the code SPECTER [13], and/or the Japanese "PKA/KERMA" file [14], if it is available), and after testing and editing, they will have to be involved in the file IRDF-2002.

Table 4. Cross sections suggested for IRDF-2002

From **IRDF-90**	From **JENDL/D-99**	From **RRDF-98**
LI6A, <u>B10A,</u> NA23G, AL27P, AL27A, S32P, MN55G, CO59G, NI58P, CU63G, NB93G, AG109G, IN1152, IN115G, AU197G,TH232F, TH232G, <u>U235F</u>, U238F, U238G, U238G, PU239F	NA232*, P31P, SC45G* TI0XSC46*, TI0XSC47*, TI0XSC48*, TI47P*, CR50G*,CR522, MN552, FE54P*, FE57NP*, FE58G*, CO592*, NI582, NI60P*, CU632*, CU652*, Y892, ZR902, I1272, TM1692*, AU1972, HG199N**, NP237F, AM241F	F192, MG246, TI462,TI46P, TI47NP**, TI48NP**, TI48P, TI49NP**, V51A, FE542, FE54A, FE56P, CO59A, CU63A, AS752, NB932, NB93N, RH103N, LA139G, PR1412, W189G, PB204N

NOTES
* = Poblems/errors in the cross section and/or in the uncertainty (covariance) information
**= Format error
The cross sections for the <u>underlined</u> reactions have errors in the format or in the uncertainty information in the library JENDL/D-99. After the errors have been corrected, the corresponding data of the file JENDL/D-99 will be used.

h) The final form of the new library IRDF-2002 will have to be edited by IAEA NDS in two forms, as concerned the cross sections:

- point values + resonance parameters in the format ENDF-6 (e.g. in order to make possible the calculation of the thermal cross sections and the Doppler broadening of the resonance cross sections at different neutron and/or fuel temperatures, respectively);
- in the format of IRDF-90 V2 (640 groups cross section values) at 0, 300 and 600 K.

i) The new cross section file IRDF-2002 will have then to be distributed by IAEA NDS to experienced laboratories for testing.

5. Acknowledgements

This work has been done in frame of the IAEA Research Contract No. 11455/R0 (2000, 2002).

6. References

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REVISIONS AND NEW EVALUATIONS OF CROSS SECTIONS FOR 19 DOSIMETRY REACTIONS

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New evaluations of cross sections for the reactions $^{139}La(n,\gamma)^{140}La$, $^{186}W(n,\gamma)^{187}W$, $^{204}Pb(n,n')^{204m}Pb$ and revisions of cross section data from RRDF-98 file [1] for the reactions $^{19}F(n,2n)^{18}F$, $^{46}Ti(n,2n)^{45}Ti$, $^{46}Ti(n,p)^{46m+g}Sc$, $^{47}Ti(n,x)^{46m+g}Sc$, $^{48}Ti(n,p)^{48}Sc$, $^{48}Ti(n,x)^{47}Sc$, $^{49}Ti(n,x)^{48}Sc$, $^{51}V(n,a)^{48}Sc$, $^{54}Fe(n,a)^{51}Cr$, $^{54}Fe(n,2n)^{53m+g}Fe$, $^{56}Fe(n,p)^{56}Mn$, $^{59}Co(n,a)^{56}Mn$, $^{58}Ni(n,p)^{58}Co$, $^{63}Cu(n,a)^{60m+g}Co$, $^{75}As(n,2n)^{74}As$, $^{141}Pr(n,2n)^{140}Pr$ were carried out at the Institute of Physics and Power Engineering (IPPE), Russia, Obninsk in 2001-2002 years. Re-evaluation of RRDF-98 data for 16 reactions were done with taking into account the results of the test performed by a team from the Budapest University of Technology and Economics, Hungary [2].

The activation detectors on the basis of $^{139}La(n,\gamma)^{140}La$ and $^{186}W(n,\gamma)^{187}W$ reactions are commonly used in the reactor dosimetry for determination of the neutron flux in the epithermal energy range. Reaction $^{204}Pb(n,n')^{204m}Pb$ looks very attractive for use in reactor dosimetry for neutron spectrum unfolding in the energy range higher 2.2 MeV. Reactions $^{46}Ti(n,2n)^{45}Ti$ and $^{54}Fe(n,2n)^{53m+g}Fe$ are very perspective for neutron dosimetry at $T(d,n)^{4}He$ sourses. The threshold reactions $^{47}Ti(n,x)^{46m+g}Sc$, $^{48}Ti(n,x)^{47}Sc$, $^{49}Ti(n,x)^{48}Sc$, $^{75}As(n,2n)^{74}As$ and $^{141}Pr(n,2n)^{140}Pr$ may be useful in the high energy neutron dosimetry. The two last reactions are using also as the monitor reactions for the cross sections measurements in the energy range 14 – 15 MeV.

At present the cross section data for $^{54}Fe(n,2n)^{53m+g}Fe$, $^{75}As(n,2n)^{74}As$, $^{139}La(n,\gamma)^{140}La$, $^{141}Pr(n,2n)^{140}Pr$ and $^{204}Pb(n,n')^{204m}Pb$ reactions are absent in the IRDF-90 ver.2 file and in national dosimetry libraries as well. Cross section data for $^{186}W(n,\gamma)^{187}W$ reaction are given in the Japanese Reactor Dosimetry File – JENDL/D-99 (MAT 7443) [3]. JENDL/D-99 data for $^{186}W(n,\gamma)^{187}W$ reaction have been evaluated in March 1987 and uncertainties in the cross section values estimated only via variations.

In the process of preparation of the input data for evaluation of cross sections and their uncertainties three information sources were used: available differential and integral experimental data, results of theoretical model calculations and predictions of the systematics.

Differential and integral experimental data were taken mainly from EXFOR Library (Version January 2001). In the cases then the data were absent in the EXFOR, information was taken from the original publications. In the first step of evaluation all experimental data were thoroughly analyzed. During this procedure the experimental data (if it was possible) were corrected to the new recommended cross section data for monitor reactions used in the measurements and to the new recommended decay data. Correction of experimental data to the new standards leads in generally to decreasing the discrepancies in the experimental data and thus to decreasing the uncertainty in the evaluated cross section values.

For theoretical description of excitation functions of $^{47}Ti(n,x)^{46m+g}Sc$, $^{48}Ti(n,x)^{47}Sc$, $^{49}Ti(n,x)^{48}Sc$, $^{139}La(n,\gamma)^{140}La$, $^{186}W(n,\gamma)^{187}W$ and $^{204}Pb(n,n')^{204m}Pb$ dosimetry reactions the optical-statistical method was used with taking into account consistently the contribution of the direct, preequilibrium and statistical equilibrium processes into different outgoing channels. The practical calculations of cross sections were made by means of modified version of the GNASH code [4] and STAPRE code [5]. Modified GNASH code differs mainly from original GNASH code [6] with having a subroutine for calculations of width fluctuation correction.

The calculation of penetrability coefficients for neutrons was made on the basis of generalized optical model, which permits to estimate the cross sections for the direct excitations of collective low-lying levels. The ECIS coupled channel deformed optical model code [7] was used for this calculations. The optical coefficients of proton and alpha particles penetrabilities were determined by means of the SCAT2 code [8].

The data on discrete levels parameters for all target and residual nuclei were obtained from the recent work [9]. Unknown branching ratios were estimated on the basis of statistical calculations of the possible E1, E2, and M1 gamma-ray transitions. Intensities of such transitions were calculated in accordance with the radiation strength functions recommended in Ref. [10].

Continuum level densities were represented with the Gilbert-Cameron [11] model using the Cook parameters [12]. The calculation of gamma-ray transition probabilities in continuum region of excited states of all nuclei under consideration was made in the frame of hypothesis of domination of giant dipole resonance with parameters of radiative strength function from Kopecky-Uhl systematics [13]. Recommended parameters of giant dipole resonances were taken from ref. [14].

By means of the modified GNASH code cross section values of $^{139}La(n,\gamma)^{140}La$ and $^{186}W(n,\gamma)^{187}W$ reactions were calculated from 1 keV to 20 MeV. The same data for the reactions $^{47}Ti(n,x)^{46m+g}Sc$, $^{48}Ti(n,x)^{47}Sc$, $^{49}Ti(n,x)^{48}Sc$ and $^{204}Pb(n,n')^{204m}Pb$ were obtained from threshold to 20 MeV.

The evaluation of excitation functions of dosimetry reactions had been carried out on the basis of prepared input data within the framework of generalized least squares method. Rational function was used as a model function [15]. Procedure of calculation recommended cross section data and related covariance matrixes of uncertainties was performed by means of PADE-2 code [16].

MLBW resonance parameters used for calculation $^{139}La(n,\gamma)^{140}La$ and $^{186}W(n,\gamma)^{187}W$ reactions excitation functions in the resolved resonance region were evaluated on the basis the data given in the compilations of S.F.Mughabghab [17], [18] and S.I.Sukhoruchkin [19]. Radiative capture cross sections for La-139 and W-186 nuclei in the unresolved resonance region were evaluated on the basis of calculations performed by means of EVPAR code [20].

Uncertainties in the evaluated excitation function for the $^{139}La(n,\gamma)^{140}La$ and $^{186}W(n,\gamma)^{187}W$ reactions are given by means of the three block matrixes. The first and the second block matrixes are used for description of the cross sections uncertainty in the resolved resonance region. The third block matrixes were used for description of reactions uncertainty from unresolved resonance region to 20 MeV. The first and the third block matrixes are the relative covariance matrixes, obtained by means of PADE code. Cross sections uncertainties in the second block matrixes are given via diagonal matrixes. This matrixes were prepared by means of DSIGNG code.[21]

Integral experimental data for U-235 neutron fission spectrum and Cf-252 spontaneous fission neutron spectrum were used for testing evaluated excitation functions of threshold reactions. Data for U-235 thermal fission neutron spectrum and Cf-252 spontaneous fission neutron spectrum were taken from ref.[22] and [23], respectively. The average cross section values for U-235 thermal fission neutron spectrum and Cf-252 spontaneous fission neutron spectrum calculated from the IPPE, JENDL/D-99 IRDF-90 Ver.2 evaluated excitation functions are given in Tables 1 and 2 in comparison with related experimental data. The comparison with the same data from IRDF-90v2 and JENDL/D-99 dosimetry libraries shows that the results of new re-evaluations agree better with integral experimental data.

The tested characteristics of the evaluated $^{139}La(n,\gamma)^{140}La$, $^{186}W(n,\gamma)^{187}W$ reaction excitation function - capture cross section at E_n=0.0253 eV and resonance integral (0.5 eV to 20 MeV) are agree well with data from compilations [17,18] and [38]. The averaged capture cross section of La-140 calculated from the new evaluation for neutron spectrum in the center of the Coupled Fast Reactivity Measurement Facility (CFRMF) agree within uncertainties with experimental data [39]. The data for neutron spectrum in the center of CFRM facility were taken from ref. [40].

The detailed description of the new evaluations $^{139}La(n,\gamma)^{140}La$, $^{186}W(n,\gamma)^{187}W$ and $^{204}Pb(n,n')^{204m}Pb$ reactions excitation function is given in the report [41]. Full describtion of the re-evaluded cross sections may be found on the IAEA Web site [42].

Data files prepared in the result of new evaluations and re-evaluations in the ENDF-6 format for 19 dosimetry reactions may be consider as candidates to the new International Reactor Dosimetry File: IRDF-2002.

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Table 1

Measured and calculated averaged cross sections
in ^{252}Cf spontaneous fission neutron spectrum

Reaction	**IPPE eval.** $\langle\sigma\rangle$**, mb**	**JENDL/D-99** $\langle\sigma\rangle$**, mb**	**IRDF-90** $\langle\sigma\rangle$**, mb**	**Experiment** $\langle\sigma\rangle$**, mb**
^{19}F(n,2n)^{18}F	0.01615	0.018417	0.017027	0.01628 ± 0.00054 [24]
^{46}Ti(n,2n)^{45}Ti	0.01198	0.012871		0.093 ± 0.031 [25]
^{46}Ti(n,p)$^{46m+g}$Sc	13.818	13.553	12.313	14.20 ± 0.24 [24]
^{47}Ti(n,x)$^{46m+g}$Sc	0.019201	0.016494		
^{48}Ti(n,p)^{48}Sc	0.42629	0.39483	0.3864	0.4275 ± 0.0078 [24]
^{48}Ti(n,x)^{47}Sc	0.0042891	0.0041188		
^{49}Ti(n,x)^{48}Sc	0.0026070	0.0027173		
^{51}V(n,α)^{48}Sc	0.038514		0.03872	0.03904 ± 0.00086 [24]
^{54}Fe(n,2n)$^{53m+g}$Fe	0.0036219			
^{54}Fe(n,α)^{51}Cr	1.1114			
^{56}Fe(n,p)^{56}Mn	1.4692	1.4088	1.368	1.471 ± 0.025 [24]
^{59}Co(n,α)^{56}Mn	0.22095	0.23034	0.2159	0.2221 ± 0.0039 [24]
				0.2208 ± 0.0014 [26]
^{58}Ni(n,p)^{58}Co	116.65	114.52	115.2	117.6 ± 1.5 [24]
^{63}Cu(n,α)$^{60m+g}$Co	0.6925	0.72831	0.6778	0.6897 ± 0.0130 [24]
^{75}As(n,2n)^{74}As	0.61804			
^{141}Pr(n,2n)^{140}Pr	1.9843			
^{139}La(n,g)^{140}La	6.650			
^{186}W(n,g)^{187}W	31.699	34.737		
^{204}Pb(n,n’) ^{204m}Pb	20.373			20.900 ± 1.202 [27]
				20.850 ± 0.920 [28]

* - evaluated by author

Table 2

Measured and calculated averaged cross sections in ^{235}U thermal fission neutron spectrum

Reaction	IPPE eval. <σ>, mb	JENDL/D-99 <σ>, mb	IRDF-90 <σ>, mb	Experiment <σ>, mb
$^{19}F(n,2n)^{18}F$	0.0072993	0.0084128	0.00772	0.007200±0.001000 [29] 0.006509±0.000300 [30] 0.008653±0.000464 [31]
$^{46}Ti(n,2n)^{45}Ti$	0.0044686	0.0048136		
$^{46}Ti(n,p)^{46m+g}Sc$	11.447	11.301	10.252	11.55 ± 0.20 [31] 11.51 ± 0.40 [32] 11.57 ± 0.37 [33]
$^{47}Ti(n,x)^{46m+g}Sc$	0.0081158	0.0067188		
$^{48}Ti(n,p)^{48}Sc$	0.3043	0.28257	0.2749	0.3007 ± 0.0054 [31] 0.302 ± 0.010 [32] 0.305 ± 0.020 [33]]
$^{48}Ti(n,x)^{47}Sc$	0.0016558	0.0016105		
$^{49}Ti(n,x)^{48}Sc$	0.0010041	0.0010174		
$^{51}V(n,\alpha)^{48}Sc$	0.024414		0.0246	0.02438 ±0. 00056 [31]
$^{54}Fe(n,2n)^{53m+g}Fe$	0.0012839			
$^{54}Fe(n,\alpha)^{51}Cr$	0.8459			0.850 ± 0.050 *
$^{56}Fe(n,p)^{56}Mn$	1.1070	1.0552	1.0297	1.083 ± 0.017 [31] 1.09 ± 0.04 [32]
$^{59}Co(n,\alpha)^{56}Mn$	0.15823	0.16608	0.1549	0.1568 ± 0.0035 [31]
$^{58}Ni(n,p)^{58}Co$	106.58	104.75	105.73	108.5 ± 1.4 [31]
$^{63}Cu(n,\alpha)^{609m+g}Co$	0.5329	0.57746	0.5214	0.5295 ± 0.0255 [34] 0.4935 ± 0.0242 [31]
$^{75}As(n,2n)^{74}As$	0.3092			0.309 ± 0.019 *
$^{141}Pr(n,2n)^{140}Pr$	1.0922			
$^{139}La(n,g)^{140}La$	6.737			5.30 [35]
$^{186}W(n,g)^{187}W$	32.267	35.298		
$^{204}Pb(n,n')\,^{204m}Pb$	17.770			18.900 ± 2.000 [36] 19.080 ± 1.524 [37]

* - evaluated by author

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Interactive Visual Analysis of Nuclear Data with ZVView

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1. Plotting program ZVView

ZVView [1] is a software designed for nuclear reactions data evaluators to perform efficient interactive visual analysis of experimental and theoretical nuclear data. The main function of ZVView is plotting and inter comparison of data, including variety of options for looking into numerous details of graphical, numerical and bibliographic information involved, along with a possibility to analyse results of own evaluation. ZVView allows to user change plotting attributes, logarithmic and linear scales, zooming, split plot to sub-windows, smoothing by least square method, choice and authors to be plotted and scan their points, changing language on the fly, saving picture in PS/EPS (PostScript), EMF (Windows Meta File) and PCX formats, to be imported into LaTeX, Word and other applications. And this all on many computer platforms.

ZVView is written on C and based on a universal graphic set of libraries (DINAMO). Originally, this package was developed to be common basic software for development of various applications for nuclear research, such as acquisition, data treatment, analysis and presentation. The main functions of package are related to plot on the screen, compare and analyse in interactive way functions of the type *f(x)* and *f(x,y)*. Variety of application fields, tasks, requests from number of users during several years of using, caused flexible structure of the package, universality related to data structures, rich and expandable functionality, fast speed of operations and small memory needed. One of the most attractive feature of the package is that it works on several computer platform, therefore, applications (such as ZVView) based on DINAMO are platform independent.

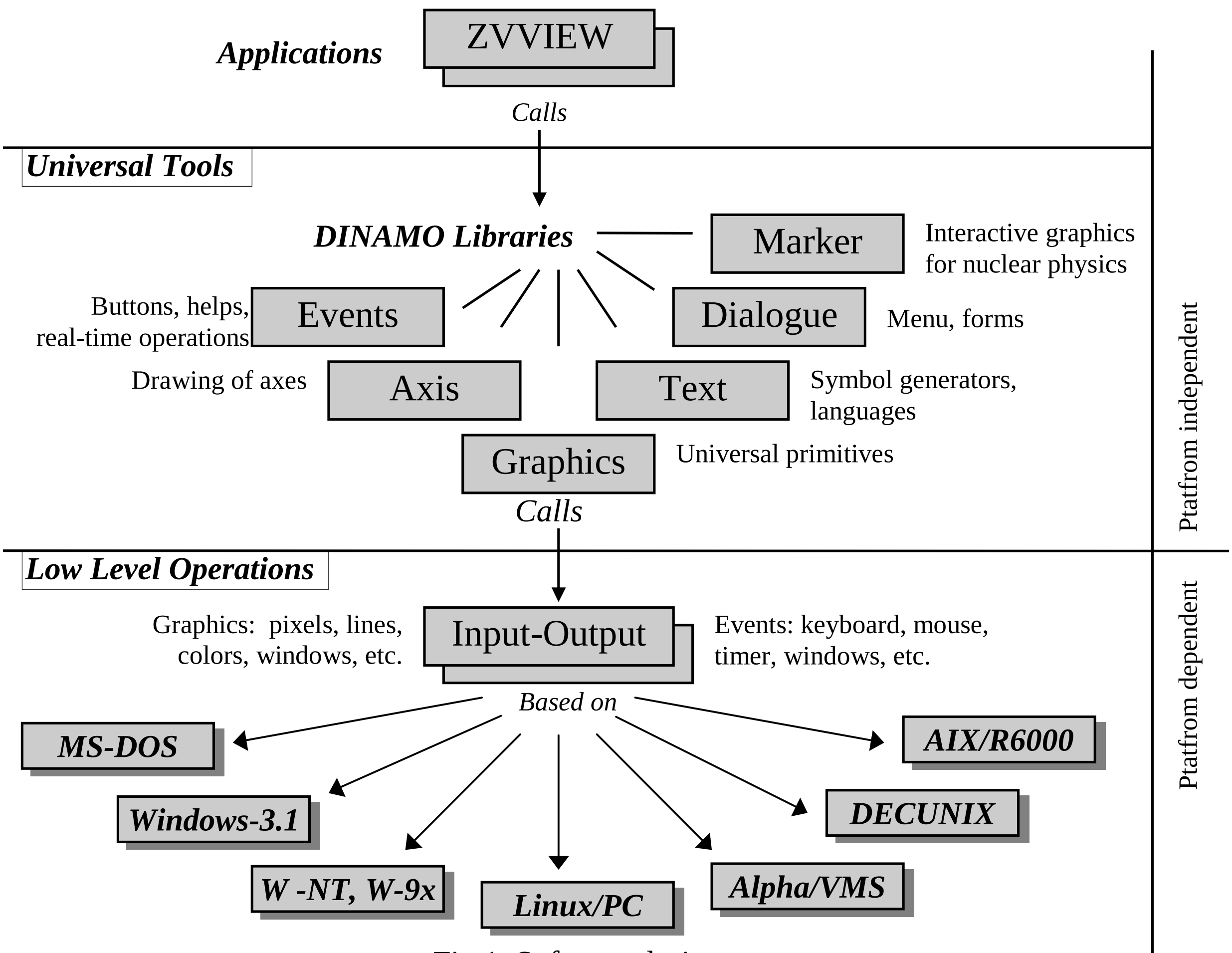


Fig.1. Software design

ZVView accept data in several formats, can be used in interactive and non-interactive mode. It can work as a standalone application and also as part of various packages (such as EXFOR/Access[2], EXFOR+ENDF/Web [3], Empire-II [4], etc.). Special arrangement of data in different formats allows to use ZVView via Internet as a helper-application running on local machine under Web-browser. It is actively used with Web-services in two ways: as part of complex retrieval system and Web-Atlases working via Internet and from CD-ROM. Several interactive Web-Atlases were designed in this way: “Fission Products in Pictures”, “Prompt Gamma Neutron Activation Analysis”, “NGAtlas-2”, “Fendl in Pictures”. This approach was also used in several works under project “IRDF-2002” ([5,6]).

2. IRDF-90, RRDF-98, JDOSM-98 vs. EXFOR-2000

In the preparation stage of IRDF-2002 project, current status of existing evaluations and experimental data has to be presented and analyzed. Plots with comparison of cross sections from different libraries and experimental data were organized as set of Web-pages (Atlas) available through Internet [7] and on CD-ROM. The Atlas contains page with full set of plots and pages for each isotope.

Special software generating html-pages and pictures was developed using existing packages EXFOR/Access and ZVView. Information from experimental data and all used evaluated libraries was organized as tables for search under MS-Access database management system. Retrieval code did a search for experimental data and corresponding evaluations, created html-file and input files for plotting program ZVView. This procedure was used in non-interactive mode to generate all sets of files automatically, but it has also interactive mode to correct plots individually (Fig.3-5).

Beside of plots, presented as static pictures in the Atlas, there is an optional possibility of interactive visual data analysis provided by ZVView. In order to use this option automatically, user should install on his computer ZVVIew as a helper for its Web browser.

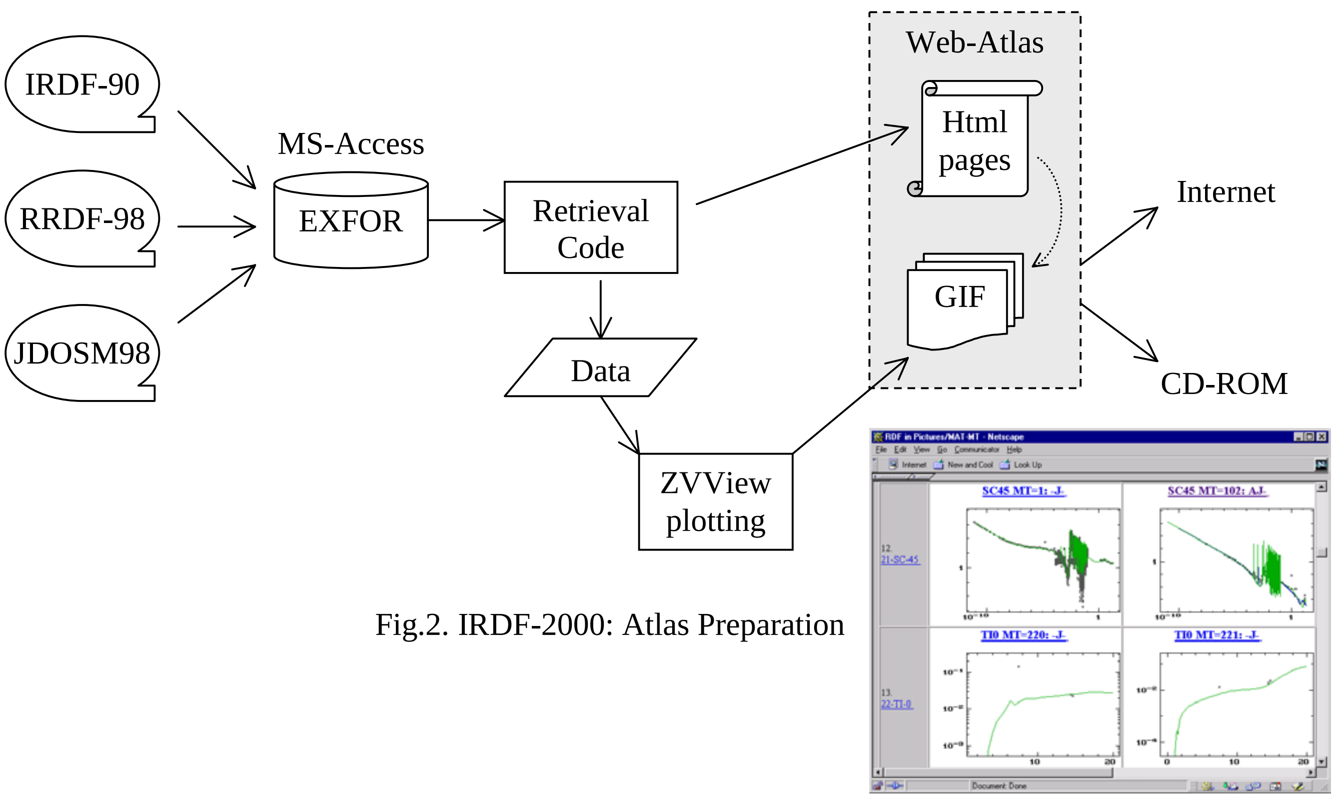


Fig.2. IRDF-2000: Atlas Preparation

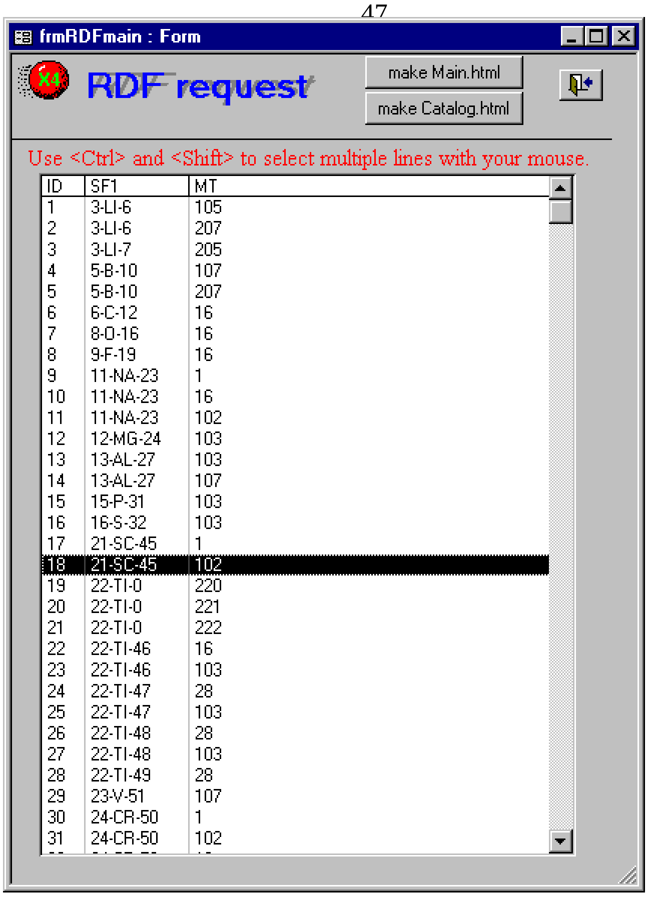


Fig.3. IRDF-2000/Access: Main form

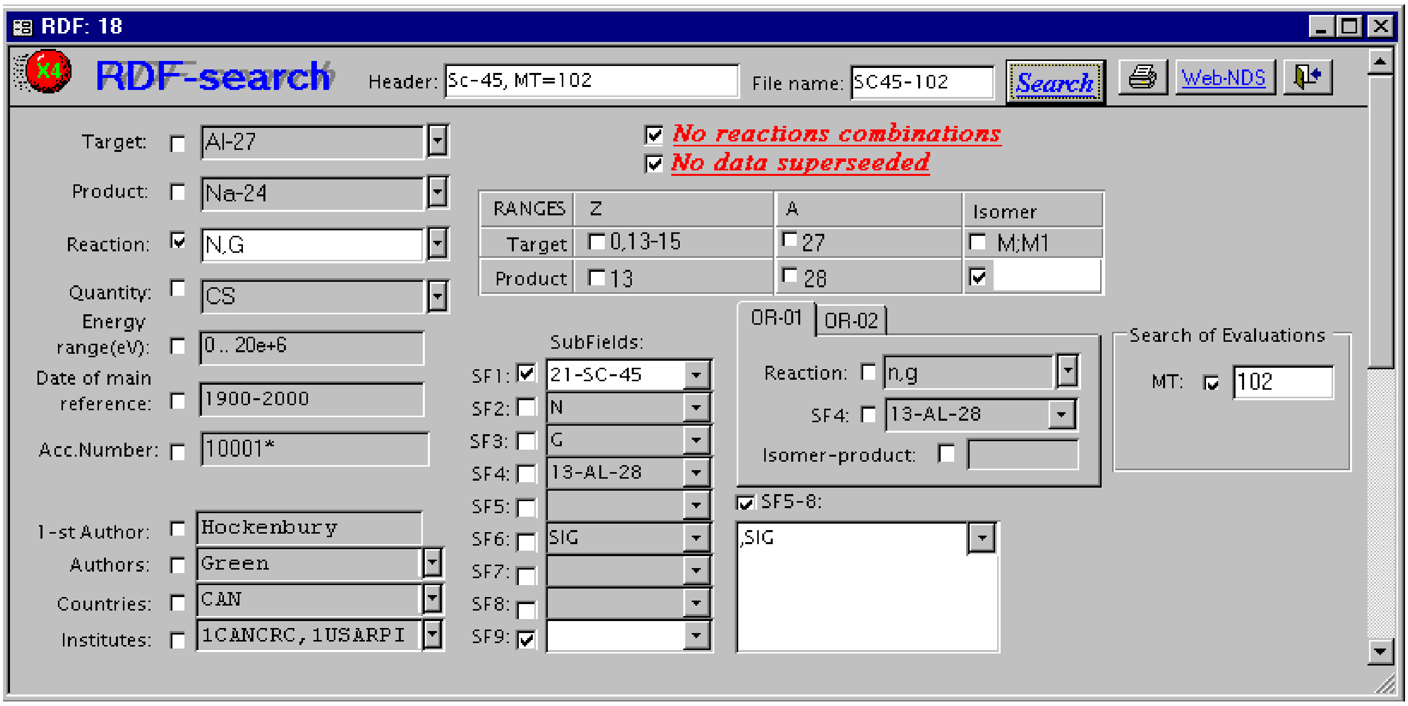
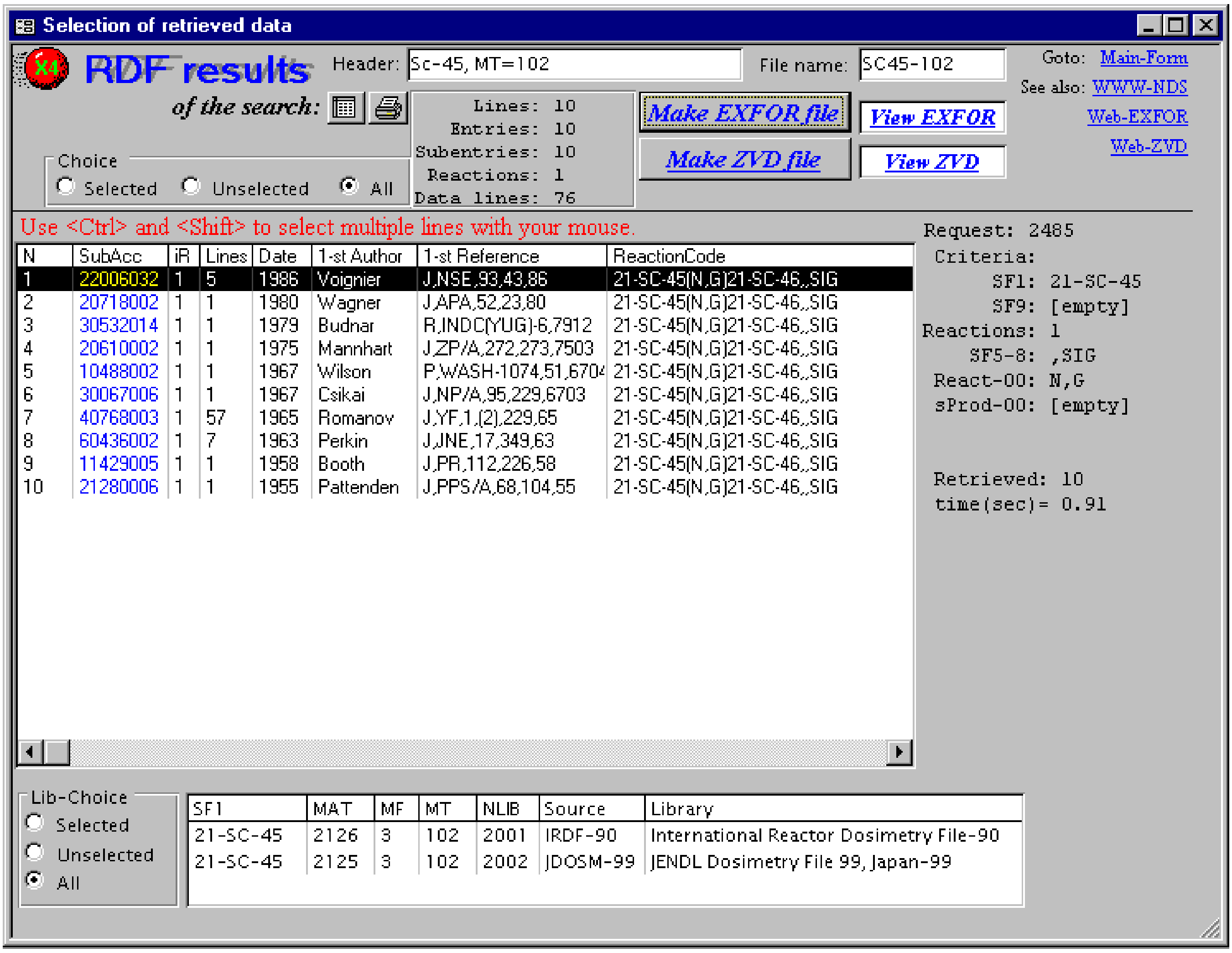


Fig.4. IRDF-2000/Access: Search form



Selection of retrieved data

RDF results of the search:

Header: Sc-45, MT=102 File name: SC45-102

Goto: Main-Form
See also: WWW-NDS
Web-EXFOR
Web-ZVD

Lines: 10
Entries: 10
Subentries: 10
Reactions: 1
Data lines: 76

Make EXFOR file View EXFOR
Make ZVD file View ZVD

Choice: Selected Unselected All

Use <Ctrl> and <Shift> to select multiple lines with your mouse.

N	SubAcc	iR	Lines	Date	1-st Author	1-st Reference	ReactionCode
1	22006032	1	5	1986	Voignier	J,NSE,93,43,86	21-SC-45(N,G)21-SC-46,,SIG
2	20718002	1	1	1980	Wagner	J,APA,52,23,80	21-SC-45(N,G)21-SC-46,,SIG
3	30532014	1	1	1979	Budnar	R,INDC(YUG)-6,7912	21-SC-45(N,G)21-SC-46,,SIG
4	20610002	1	1	1975	Mannhart	J,ZP/A,272,273,7503	21-SC-45(N,G)21-SC-46,,SIG
5	10488002	1	1	1967	Wilson	P,WASH-1074,51,6704	21-SC-45(N,G)21-SC-46,,SIG
6	30067006	1	1	1967	Csikai	J,NP/A,95,229,6703	21-SC-45(N,G)21-SC-46,,SIG
7	40768003	1	57	1965	Romanov	J,YF,1,(2),229,65	21-SC-45(N,G)21-SC-46,,SIG
8	60436002	1	7	1963	Perkin	J,JNE,17,349,63	21-SC-45(N,G)21-SC-46,,SIG
9	11429005	1	1	1958	Booth	J,PR,112,226,58	21-SC-45(N,G)21-SC-46,,SIG
10	21280006	1	1	1955	Pattenden	J,PPS/A,68,104,55	21-SC-45(N,G)21-SC-46,,SIG

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SF1	MAT	MF	MT	NLIB	Source	Library
21-SC-45	2126	3	102	2001	IRDF-90	International Reactor Dosimetry File-90
21-SC-45	2125	3	102	2002	JDOSM-99	JENDL Dosimetry File 99, Japan-99

Fig.5. IRDF-2000/Access: Select-Data form

References:

1. ZVView: http://www-nds.iaea.or.at/ndspub/zvview/
2. EXFOR/Access: IAEA-NDS-CD-09,
   User's Guide: http://www-nds.iaea.or.at/ndspub/exfor2/x4cd/guide/
3. EXFOR+ENDF/Web: http://www-nds.iaea.or.at/zvd/
4. EMPIRE-II: http://www-nds.iaea.or.at/empire/
5. R.Paviotti-Courcuera, V.Zerkin, E.M.Zsolnay, K.I.Zolotarev, W.Mannhart, L.R.Greenwood, P.J.Griffin *International Reactor Dosimetry File: IRDF-2002*, Proceedings of the 11th International Symposium on Reactor Dosimetry, p.161, Brussels, Belgium, 18-23 August 2002
6. O.Gritzay, M.Vlasov, L.Chervonna, G.Kolota, N.Klimova, V.Zerkin, *Neutron Excitation Function Guide for Reactor Dosimetry*, Proceedings of the 11th International Symposium on Reactor Dosimetry, p.127, Brussels, Belgium, 18-23 August 2002
7. IRDF-2000/Prep.: http://ndsalpha.iaea.or.at/~wwwdev/irdf/

Average Cross Sections Calculated in Various Neutron Fields

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Average cross sections have been calculated for the reactions contained in the dosimetry files, JENDL/D-99 [1], IRDF-90V2 [2], and RRDF-98 [3] in order to select the best data for the new library IRDF-2002. The neutron spectra used in the calculations are as follows:

1) ^{252}Cf spontaneous fission spectrum (NBS evaluation),
2) ^{235}U thermal fission spectrum (NBS evaluation),
3) Intermediate-energy Standard Neutron Field (ISNF),
4) Coupled Fast Reactivity Measurement Facility (CFRMF),
5) Coupled thermal/fast uranium and boron carbide spherical assembly (ΣΣ),
6) Fast neutron source reactor (YAYOI),
7) Experimental fast reactor (JOYO),
8) Japan Material Testing Reactor (JMTR),
9) d-Li neutron spectrum with a 2-MeV deuteron beam.

The items 3)-7) represent fast neutron spectra, while JMTR is a light water reactor. The Q-value for the d-Li reaction mentioned above is 15.02 MeV. Therefore, neutrons with energies up to 17 MeV can be produced in the d-Li reaction.

The calculated average cross sections were compared with the measurements. Figures 1-9 show the ratios of the calculations to the experimental data which are given in Ref. 1 It is found from these figures that the ^{58}Fe(n,γ) cross section in JENDL/D-99 reproduces the measurements in the thermal and fast reactor spectra better than that in IRDF-90V2.

References

[1] K. Kobayashi, T. Iguchi, S. Iwasaki, T. Aoyama, S. Shimakawa, Y. Ikeda, N. Odano, K. Sakurai, K. Shibata, T. Nakagawa, and M. Nakazawa: "JENDL Dosimetry File 99 (JENDL/D-99)," JAERI 1344 (2002).

[2] N.P. Kocherov and P.K. McLaughlin: "The International Reactor Dosiemtry File (IRDF-90 Version 2)," IAEA-NDS-141 Rev.3 (1996).

[3] K.I. Zolotarev, A.V. Ignatyuk, V.N. Manokhin, and A.B. Pashchenko: "RRDF-98 Russian Reactor Dosiemtry File," IAEA-NDS-193 Rev. 0 (1999).

* The present results are preliminary, and should not be referred elsewhere.

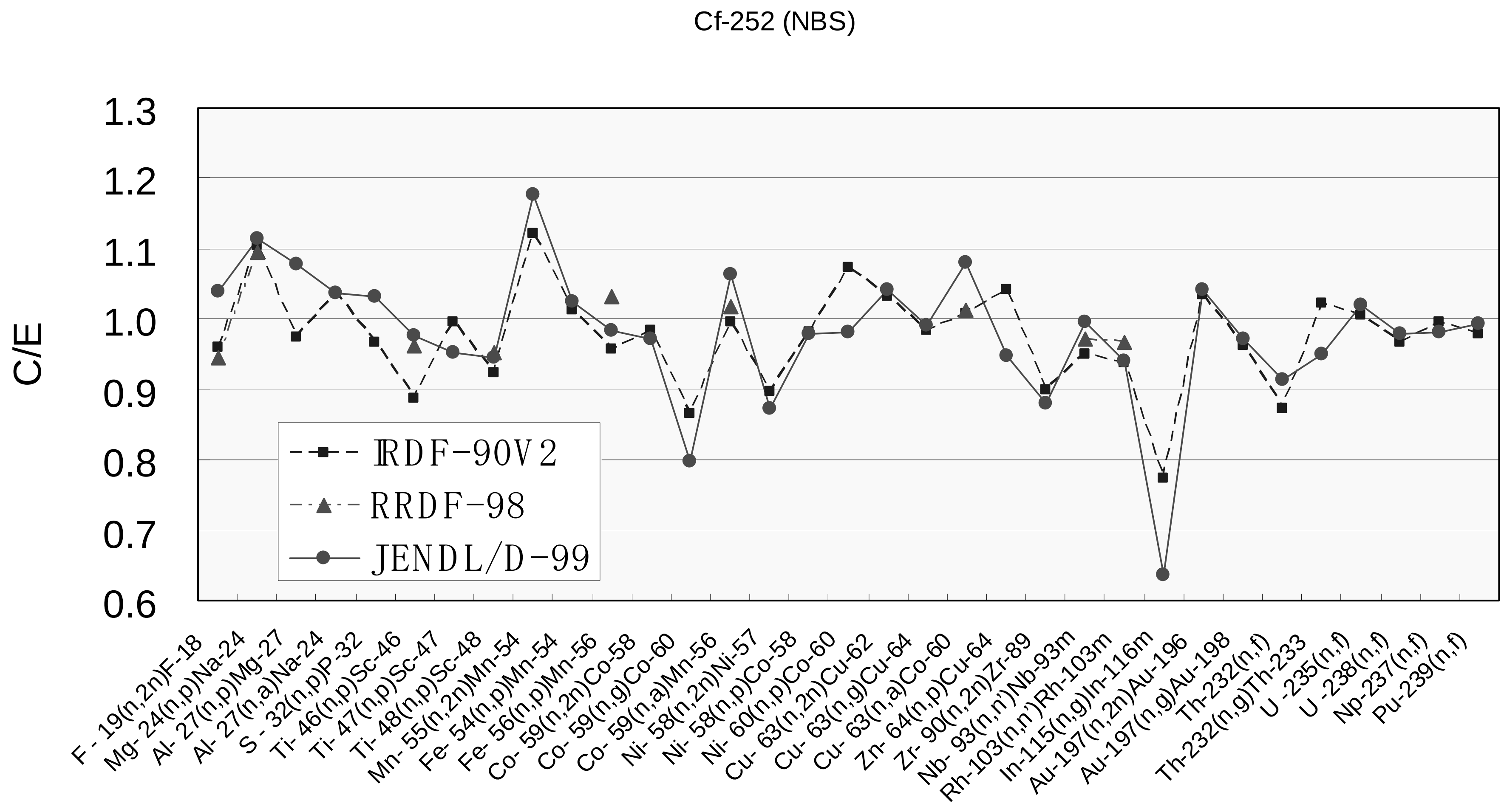

Fig. 1 C/E for ^{252}Cf spontaneous fission spectrum

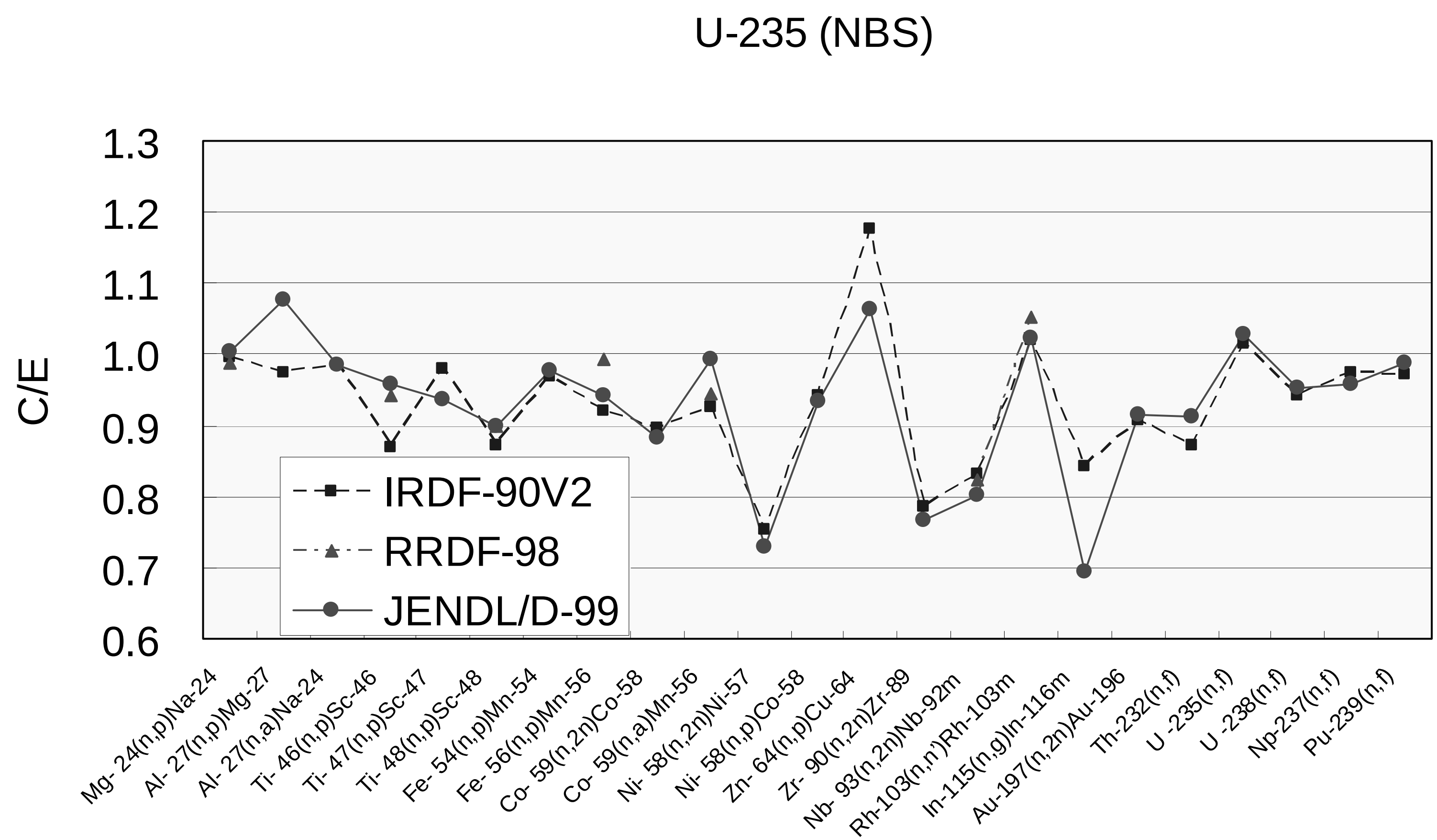

Fig. 2 C/E for ^{235}U thermal fission spectrum

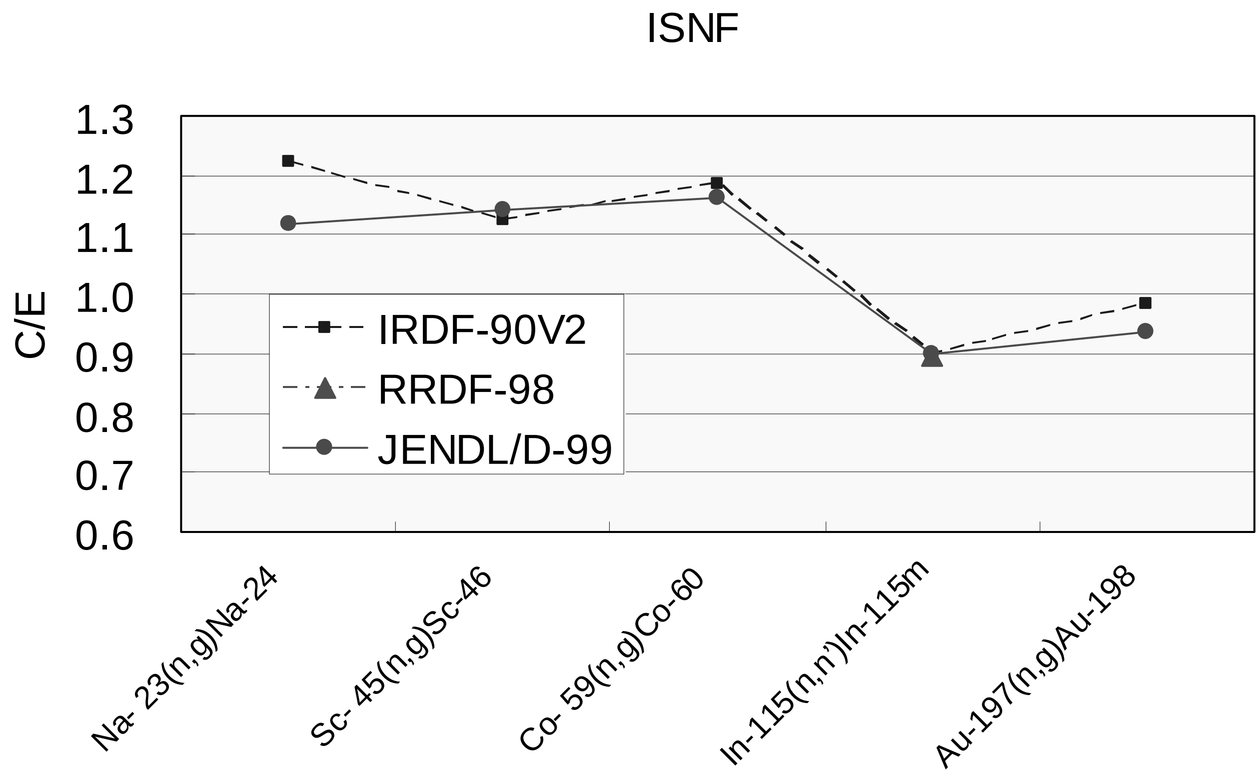

Fig. 3　C/E for ISNF spectrum

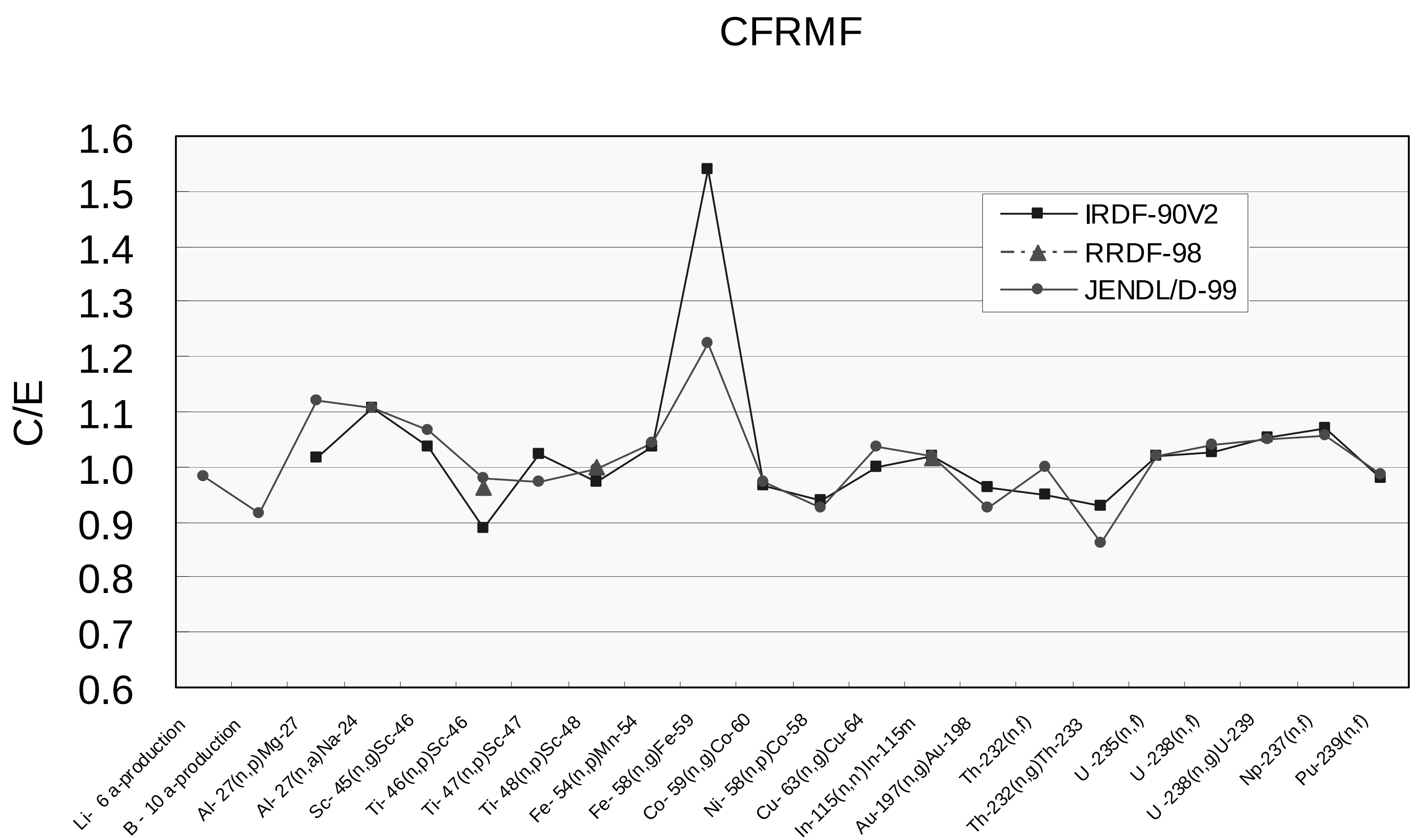

Fig. 4　C/E for CFRMF spectrum

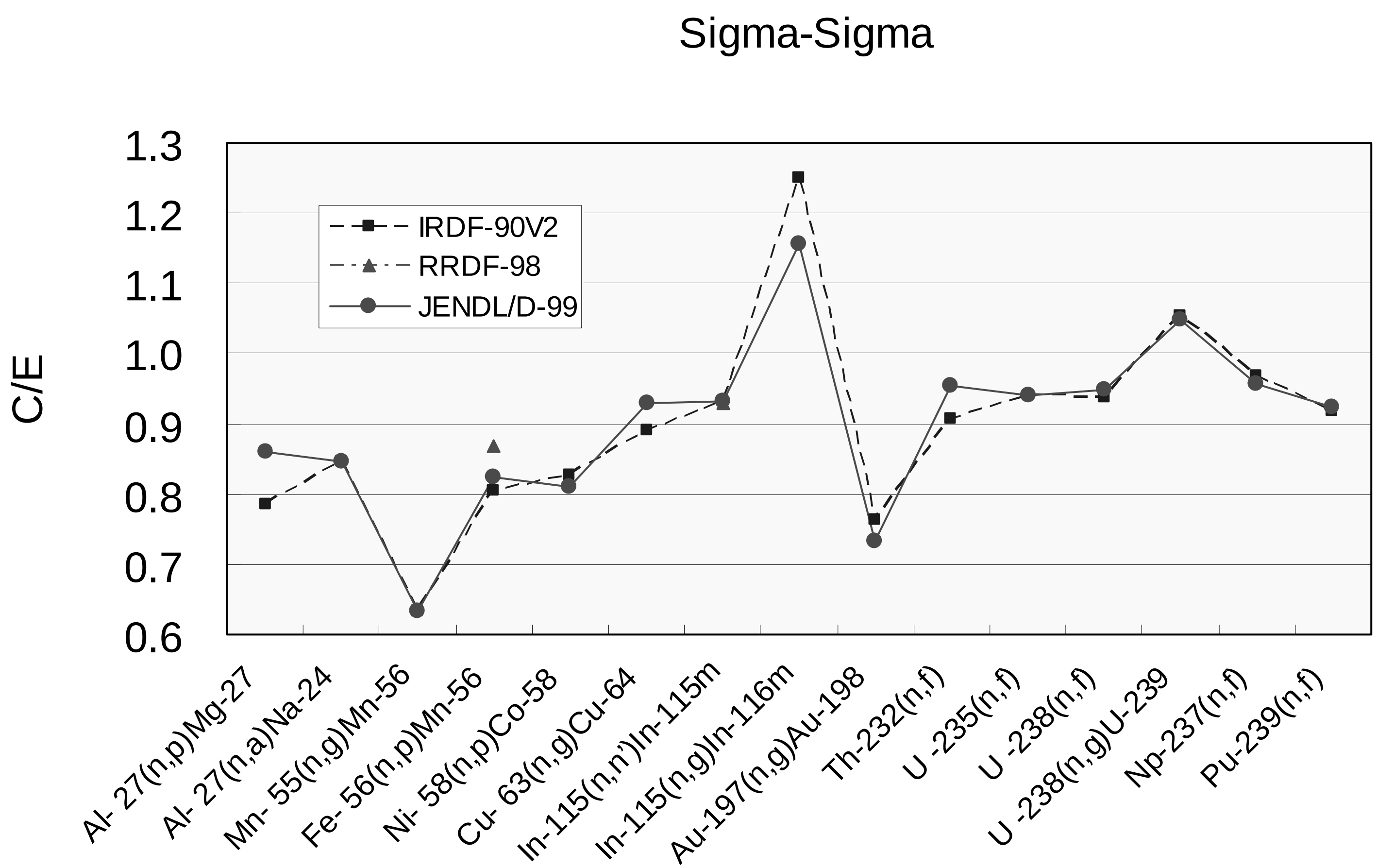

Fig. 5　C/E for ΣΣ spectrum

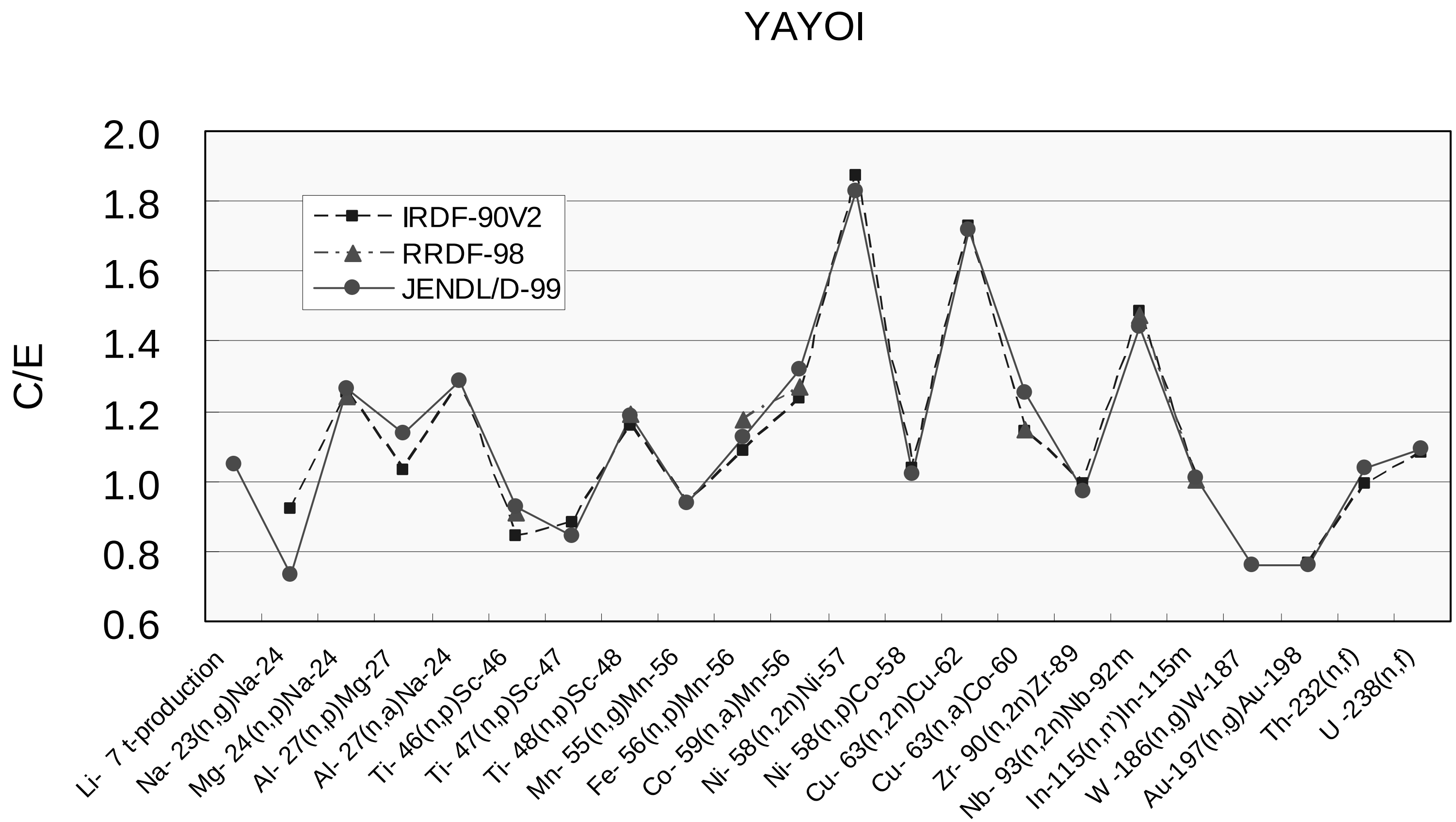

Fig. 6 C/E for YAYOI spectrum

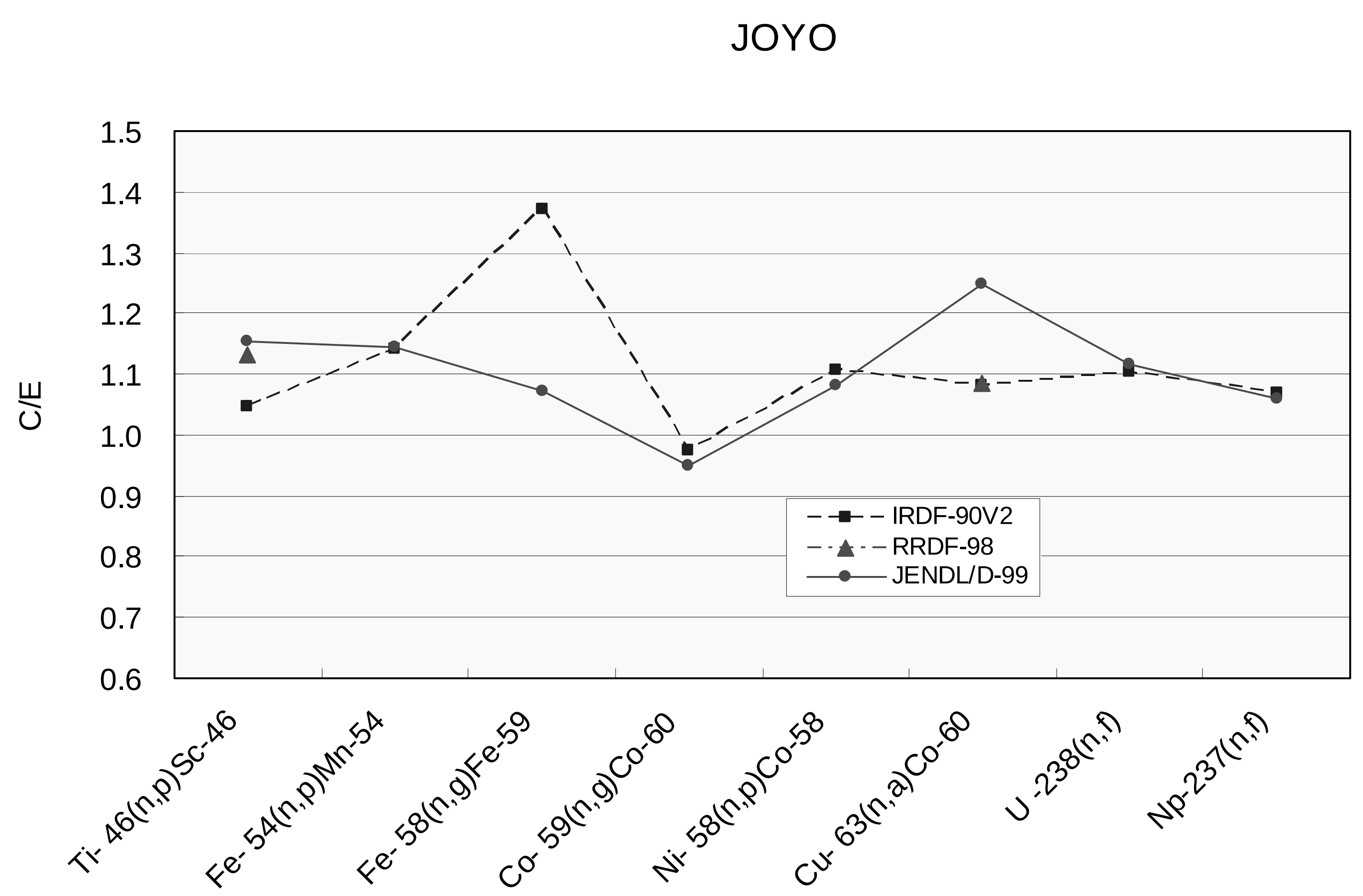

Fig. 7 C/E for JOYO spectrum

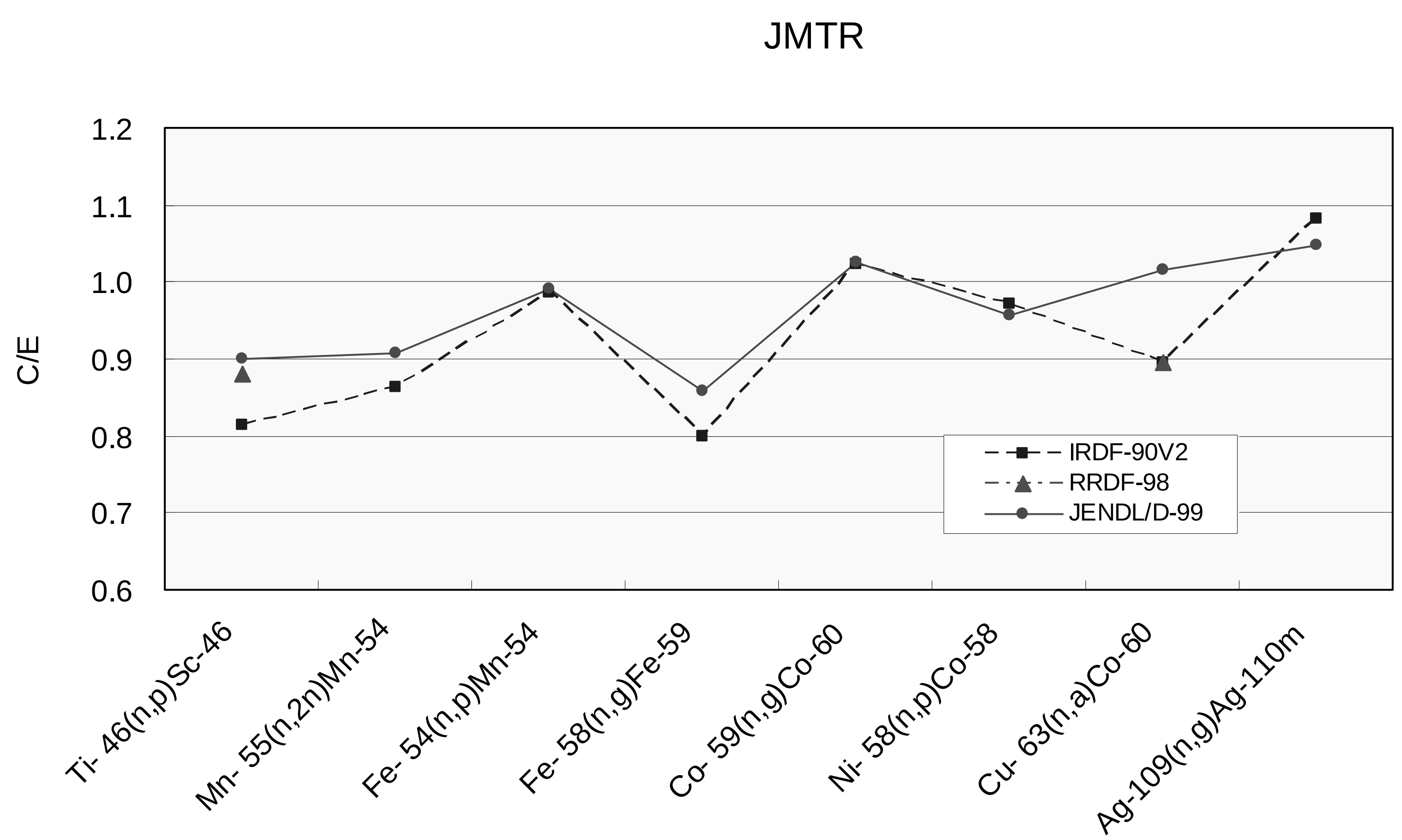

Fig. 8 C/E for JMTR spectrum

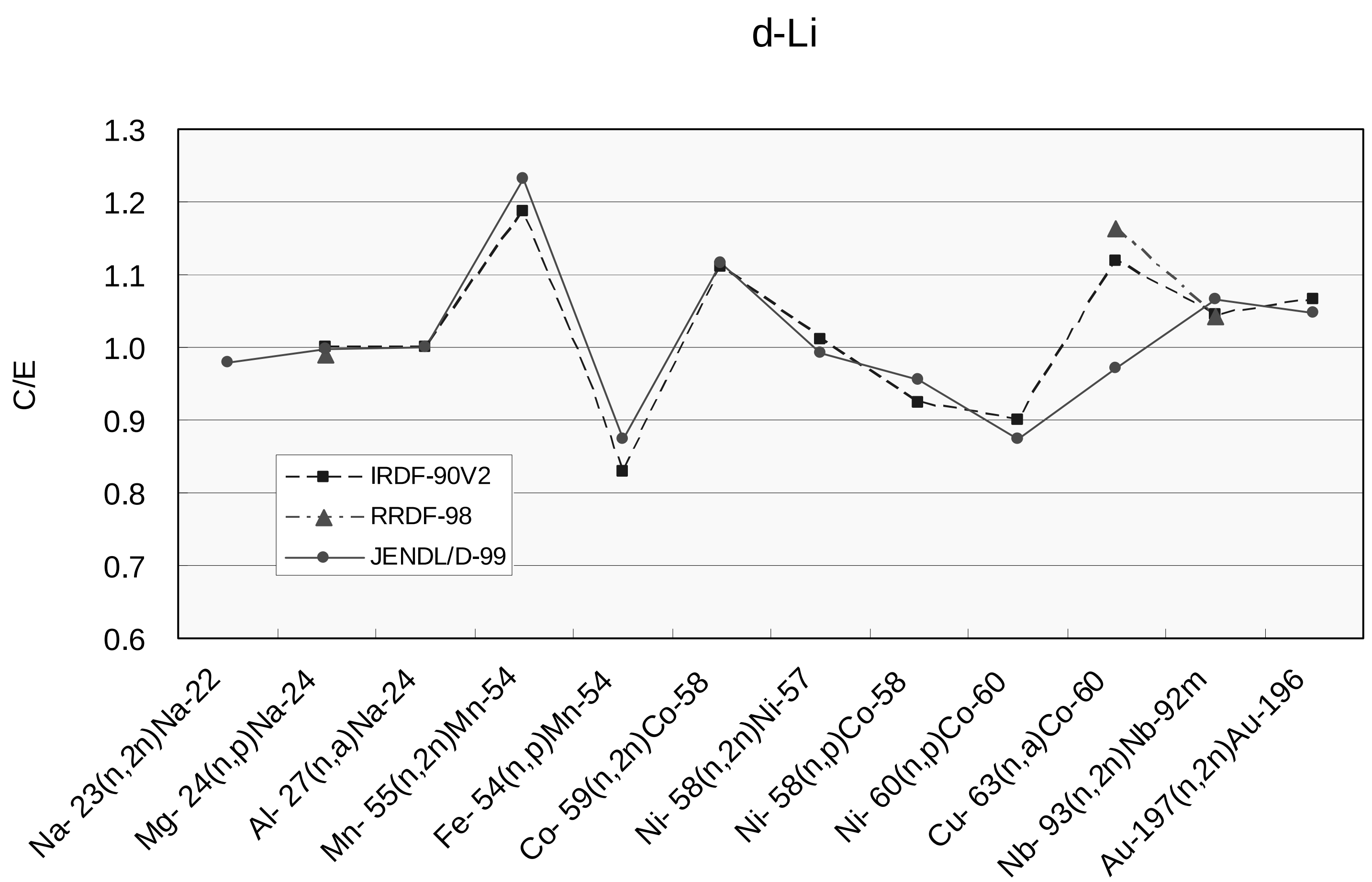

Fig. 9 C/E for d-Li spectrum

Validation of Differential Cross Sections with Integral Data

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The results of two systematic evaluations of spectrum-averaged cross section measurements performed in the fission neutron fields of ^{252}Cf and ^{235}U are shortly reviewed. These data were used to validate the σ(E) data of the IRDF-90.2 and JENDL/D-99 libraries. In the case of the ^{235}U neutron field, the lack of an adequate spectrum description valid over the whole energy range was identified. This fact presently limits the application of the spectrum-averaged data in the ^{235}U neutron field.

Integral data, i.e. spectrum-averaged cross sections <σ>, are a useful tool to check the validity of evaluated differential cross sections σ(E). An essential prerequisite to this procedure are well established <σ>-data and an adequate description of the spectral distribution N(E) of the corresponding neutron field. Quantified conclusions require in addition a consistent uncertainty handling, i.e. the uncertainties of <σ>, N(E) and σ(E) should be well defined.

In the neutron fields of spontaneous fission of ^{252}Cf and of thermal neutron-induced fission on ^{235}U a remarkable number of spectrum-averaged cross section measurements has been performed. These data were the basis of two evaluations of experimental integral data obtained in both neutron fields. The evaluation of integral data in the ^{252}Cf neutron field [1] comprises 35 different neutron reactions and covers the neutron energy response range between 0.05 and 18.1 MeV. Recently a similar evaluation has been done for the integral data measured in the ^{235}U neutron field [2]. The last systematic evaluation of integral data in the ^{235}U neutron field was performed in 1976 by Fabry et.al [3]. Due to the lack of complete documentation the evaluation was not suitable for a direct update. Therefore the whole database of integral measurements in the ^{235}U field has been reanalyzed. The new evaluation bases on 200 integral data of various neutron reactions determined in 38 different experiments (15 performed after 1976) and resulted in evaluated integral data of 30 different neutron reactions. In both evaluations [1,2] a complete covariance matrix was generated.

The results of both evaluations are compared in Fig. 1. Ratios of the evaluated spectrum-averaged cross sections in the ^{252}Cf and the ^{235}U neutron field were formed. The data of individual reactions were plotted as a function of the mean response energy ($E_{50\%}$). This is the neutron energy which corresponds to a value of 50% of the integrated energy response function in the neutron field. The given energy scale was calculated with the ^{252}Cf neutron spectrum and carefully selected σ(E) data. The energy response range of a specific neutron reaction is very similar in the ^{252}Cf and the ^{235}U neutron field which justifies the given representation. As expected, the calculated ratios show a smooth behaviour with increasing neutron energy and confirm the similarity of the spectral fission-neutron distributions of ^{252}Cf and ^{235}U. The figure is also helpful to identify outliers in the integral data, as the data of the Cu-63(n,2n) reaction, for example. Neglecting this data point, the remaining ratios were fitted to a polynomial (solid line) of the order of 2. The fit resulted in a value of the reduced chi-square of 0.98 and indicates consistency of the data with the simple fit model within the given uncertainties. The variation of the ratios as a function of energy up to a factor of two, reflects the hardness of the ^{252}Cf neutron spectrum compared to the ^{235}U spectrum.

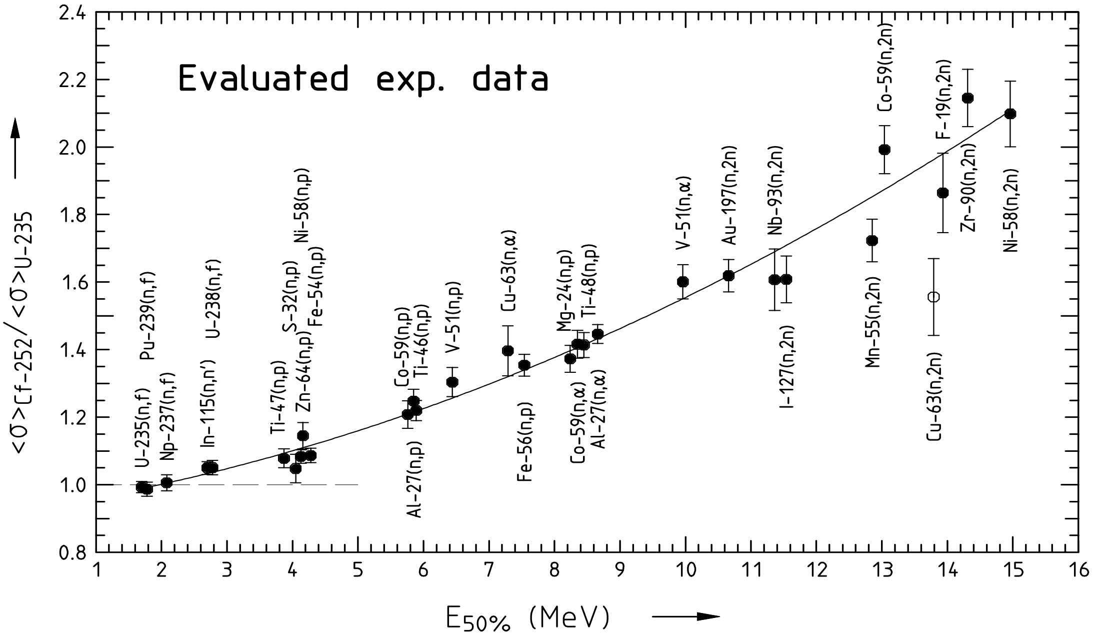

Fig. 1 Ratio of evaluated spectrum-averaged cross sections measured in the fission neutron fields of ^{252}Cf and ^{235}U (see text).

First, the differential cross sections of the IRDF-90.2 and the JENDL/D-99 libraries were tested in the ^{252}Cf neutron field. The fission neutron spectrum of ^{252}Cf is one of the best defined reference neutron spectra and has been evaluated based on a series of TOF measurements [4]. The numerical figures of the spectral distribution and the corresponding covariance matrix are given in the ENDF/B-VI library [5]. For all neutron reactions, the calculated spectrum-averaged cross sections $<\sigma> = \int N(E)\, \sigma(E)\, dE$ (for fission neutron spectra $\int N(E)\, dE = 1$ is valid) were compared with the (evaluated) experimental data and C/E ratios were formed. The uncertainties of the C/E-values were propagated from the covariances of $<\sigma>$, N(E) and σ(E). The results obtained for the various reactions were grouped into three different categories:

group A:	C/E-values being within the uncertainties consistent with unity
group B:	C/E-values not in group A, but within a band of ± 5% around unity
group C:	all remaining C/E-values

Of the 53 neutron reactions of the IRDF-90.2 library, 32 were tested in the ^{252}Cf neutron field. The results of 19 reactions were in group A: Cu-63(n,γ), U-235(n,f), Np-237 (n,f), Ti-47(n,p), S-32(n,p), Ni-58(n,p), Zn-64(n,p), Fe-54(n,p), Cu-63(n,α), Co-59(n,α), Al-27(n,α), V-51(n,α), Au-197(n,2n), Nb-93(n,2n), Cu-65(n,2n), Co-59(n,2n), F-19(n,2n), Zr-90(n,2n) and Ni-58(n,2n). In group B five reactions were identified: Au-197(n,γ), Pu-239 (n,f), In-115(n,n'), U-238(n,f) and Al-27(n,p). The uncertainties of the C/E of these reactions were too small to overlap with unity. The remaining 8 reactions belong to group C and indicate serious inconsistencies between integral and differential data: In-115(n,γ), Ti-46 (n,p), Fe-56 (n,p), Mg-24(n,p), Ti-48(n,p), I-127(n,2n), Mn-55(n,2n) and Cu-63(n,2n).

The same procedure was applied to the 61 neutron reactions of the JENDL/D-99 library. For 33 reactions experimental integral data in the ^{252}Cf neutron field were available. The C/E values

Table 1 C/E-values in the ^{252}Cf neutron field, calculated with σ(E) data of the IRDF-90.2 and JENDL/D-99 libraries (see also text).

Cf-252(sf)				
Reaction	Exp.		C/E	
	<σ> (mb)	%	IRDF-90.2	JENDL/D-99
Au-197(n,γ)	7.679E+1	1.59	0.966 ± 0.021	**0.977 ± 0.086**
Cu-63(n,γ)	1.044E+1	3.24	**0.996 ± 0.091**	**1.005 ± 0.196**
In-115(n,γ)	1.256E+1	2.23	1.227 ± 0.060	**1.003 ± 0.047**
U-235(n,f)	1.210E+3	1.20	**1.007 ± 0.012**	**1.021 ± 0.024**
Pu-239(n,f)	1.812E+3	1.37	0.980 ± 0.014	**0.996 ± 0.025**
Np-237(n,f)	1.361E+3	1.59	**0.999 ± 0.093**	**0.983 ± 0.016**
In-115(n,n')	1.974E+2	1.37	0.961 ± 0.025	0.961 ± 0.025
U-238(n,f)	3.257E+2	1.64	0.969 ±0.017	**0.980 ± 0.026**
Hg-199(n,n')	2.984E+2	1.81		0.833 ± 0.067
Ti-47(n,p)	1.927E+1	1.66	**1.006 ± 0.042**	0.962 ± 0.021
S-32(n,p)	7.254E+1	3.49	**0.969 ± 0.049**	**1.033 ± 0.090**
Ni-58(n,p)	1.175E+2	1.30	**0.982 ± 0.026**	0.975 ± 0.016
Zn-64(n,p)	4.059E+1	1.65	**1.037 ± 0.054**	0.942 ± 0.023
Fe-54(n,p)	8.684E+1	1.34	**1.015 ± 0.026**	1.027 ± 0.019
Co-59(n,p)	1.690E+0	2.48		
Al-27(n,p)	4.880E+0	2.14	0.958 ± 0.039	1.058 ± 0.027
Ti-46(n,p)	1.407E+1	1.77	0.876 ± 0.029	0.964 ± 0.030
V-51(n,p)	6.488E-1	1.97		
Cu-63(n,α)	6.887E-1	1.96	**0.986 ± 0.033**	1.059 ± 0.029
Fe-56(n,p)	1.465E+0	1.77	0.936 ± 0.030	**0.962 ± 0.048**
Mg-24(n,p)	1.996E+0	2.44	1.082 ± 0.040	1.092 ± 0.034
Co-59(n,α)	2.218E-1	1.88	**0.975 ± 0.036**	**1.040 ± 0.050**
Ti-48(n,p)	4.247E-1	1.89	0.912 ± 0.032	0.931 ± 0.028
Al-27(n,α)	1.016E+0	1.28	**1.022 ± 0.026**	**1.022 ± 0.026**
V-51(n,α)	3.900E-2	2.21	**0.995 ± 0.044**	
Tm-169(n,2n)	[6.690E+0]	6.28		**0.932 ± 0.065**
Au-197(n,2n)	5.506E+0	1.83	**1.044 ± 0.052**	1.049 ± 0.031
Nb-93(n,2n)	[7.490E-1]	5.07	**1.041 ± 0.064**	**1.011 ± 0.070**
I-127(n,2n)	2.069E+0	2.73	1.062 ± 0.045	1.096 ± 0.051
Cu-65(n,2n)	6.582E-1	2.22	**1.030 ± 0.042**	1.061 ± 0.039
Mn-55(n,2n)	4.075E-1	2.33	1.181 ± 0.115	1.237 ± 0.111
Co-59(n,2n)	4.051E-1	2.51	**1.044 ± 0.051**	**1.030 ± 0.045**
Cu-63(n,2n)	1.844E-1	3.98	1.134 ± 0.068	1.140 ± 0.066
F-19(n,2n)	1.612E-2	3.37	**1.065 ± 0.063**	1.151 ± 0.070
Zr-90(n,2n)	2.210E-1	2.89	**1.001 ± 0.061**	**0.979 ± 0.058**
Ni-58(n,2n)	8.952E-3	3.57	**1.033 ± 0.079**	**1.004 ± 0.072**

resulted in 16 reactions in group A: Au-197(n,γ), Cu-63(n,γ), In-115(n,γ), U-235(n,f), Pu-239(n,f), Np-237(n,f), U-238(n,f), S-32(n,p), Fe-56(n,p), Co-59(n,α), Al-27(n,α), Tm-169(n,2n), Nb-93 (n,2n), Co-59(n,2n), Zr-90(n,2n) and Ni-58(n,2n). Additional six reactions were in group B: In-115 (n,n'), Ti-47(n,p), Ni-58(n,p), Fe-54(n,p), Ti-46(n,p) and Au-197(n,2n). The remaining 11 reactions were in group C: Hg-199(n,n'), Zn-64(n,p), Al-27 (n,p), Cu-63(n,α), Mg-24(n,p), Ti-48(n,p), I-127 (n,2n), Cu-65(n,2n), Mn-55(n,2n), Cu-63 (n,2n) and F-19(n,2n).

The numerical results are summarized in Table 1. The reactions, given in column 1, are listed with rising mean response energies. In columns 2 and 3 of the table, the spectrum-averaged cross sections and the diagonal elements of the covariance matrix of Ref. [1] are given. The calculated C/E-values of the individual reactions of the IRDF-90.2 and JENDL/D-99 libraries are listed. Numerical values belonging to group A are given with bolded characters and those of group B with grey shadowing.

Some of the evaluated cross sections in the IRDF-90.2 and JENDL/D-99 libraries quote extremely low uncertainties of the σ(E) data. Such low values can easily originate from least-squares based evaluations with imperfect handling of the systematic components. Most of the experimental σ(E) data were measured as ratios relative to well-known reference cross sections. The uncertainty of the reference cross section is the irreducible uncertainty limit in such measurements. Absolute measurements are quite difficult and base on two techniques of the neutron fluence determination: the associated particle method and the proton recoil telescope. The proton recoil method is based on the n-p cross section and results in uncertainties of about 1%. The associated particle method, restricted to certain neutron energy ranges, gives uncertainties of ≥ 0.5%. (The Al-27(n,α) cross section at 14 MeV is an example of an associated particle measurement). Only very few cross sections can really reach the uncertainty level of the absolute measurements. In such cases all other systematic uncertainty components must be of negligible order of magnitude. With consideration of the status of the available experimental database, the magnitude of some of the recently quoted cross section uncertainties seems quite unreal.

Secondly, a test of the data of the IRDF-90.2 and JENDL/D-99 libraries in the ^{235}U neutron field was planned. In the course of work some complications did arise. As the most appropriate description of the spectral distribution of the fission neutrons of ^{235}U, the data of the Madland-Nix model [6] given in the ENDF/B-VI library [7] were identified. However, this model fails in adequately describing the neutron spectrum at energies > 10 MeV and uncertainty information is lacking. This situation is shown in Fig. 2. In the upper part of the figure the C/E-values of various neutron reactions in the ^{252}Cf neutron field are plotted as a function of the mean neutron energy of the response function in this field. The given uncertainties comprise the contributions of <σ>, σ(E) and N(E). The neutron spectrum is that of Ref. [5] and the <σ> data are the same as in column 2 of Table 1. However, the σ(E) data used in Fig. 2 were taken from various evaluations and are not identical with the data given in Table 1. With a few exceptions, most of the C/E-values in the ^{252}Cf neutron field are grouped within a band of ± 5% around unity, independent of the mean neutron energy of the individual reactions. In the lower part of Fig. 2, a similar representation is shown for the data in the ^{235}U neutron field. The neutron spectrum is that of Ref. [7]. The <σ> values are from Ref. [2], also given in Fig. 1, and the σ(E) data are the same as in the upper part of the figure. In the uncertainties of these C/E-values the spectral component is lacking. The C/E-values calculated in the ^{235}U neutron field show a pronounced trend at high neutron energies which originates from the spectral description with the Madland-Nix model. At low and medium neutron energies the Madland-Nix model is a fair description of the ^{235}U fission neutron spectrum. With increasing neutron energy, the C/E-value of the Al-27(n,α) reaction is the last one which is not influenced by the obvious underestimation of the high-energy portion of the ^{235}U fission neutron spectrum.

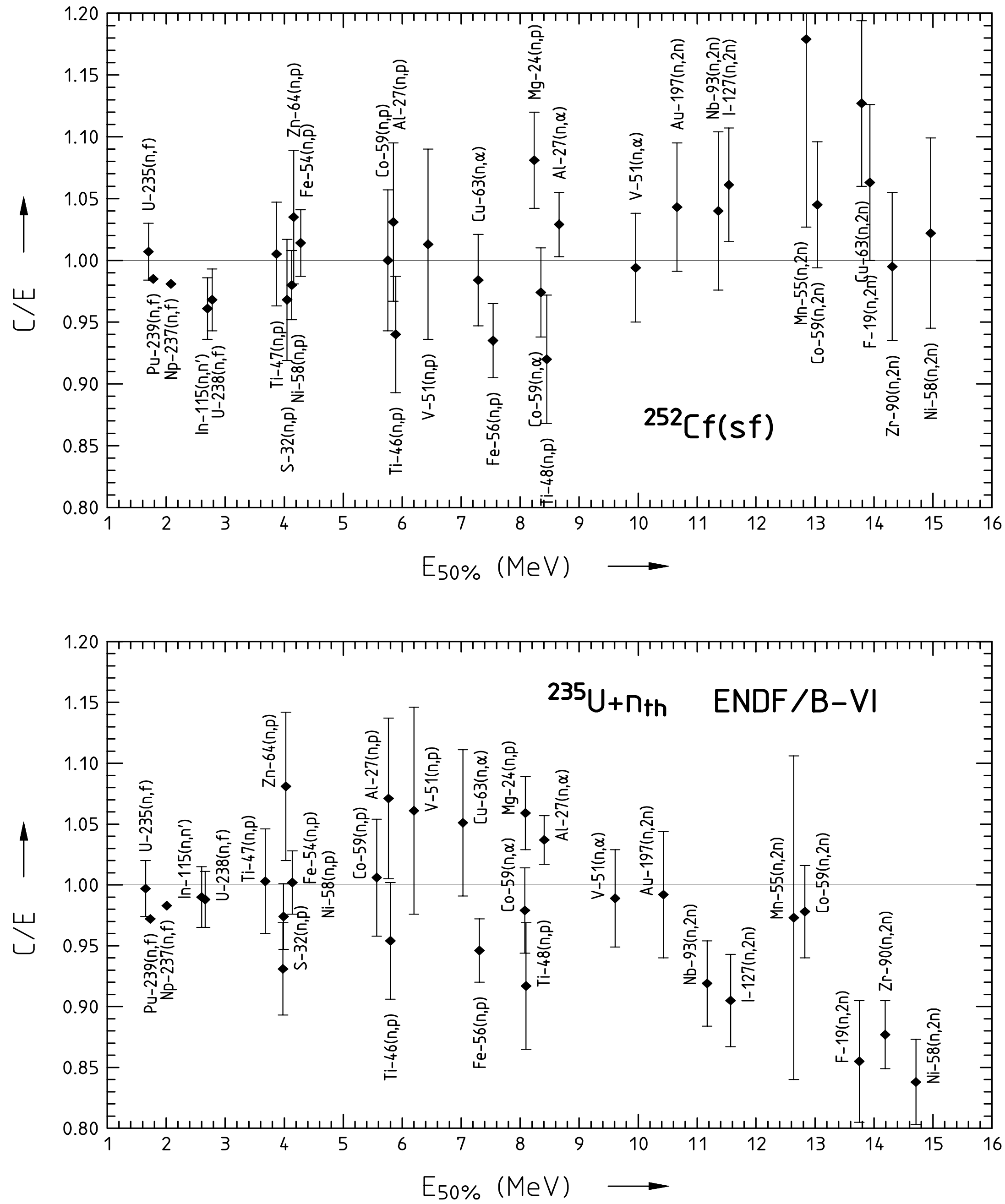

Fig. 2 C/E-values obtained in the fission neutron fields of ^{252}Cf and ^{235}U. The σ(E) data were the same in the upper and lower part of the figure. The ^{235}U spectrum is from Ref. [7].

In Table 2 some additional details are summarized by using the high-threshold reaction of ^{58}Ni(n,2n)^{57}Ni as an example. The 90% energy response range of this reaction is between 13.11 and 18.09 MeV in the ^{252}Cf field and between 13.03 and 17.73 MeV in the ^{235}U field.

The σ(E) data of ^{58}Ni(n,2n)^{57}Ni in the IRDF-90.2 library originate from the IRK-90 evaluation [9]. The C/E-value calculated with the original data of [9] in the ^{252}Cf field is 1.022 ± 0.077 and confirms the validity of σ(E). With the data of the IRDF-90.2 evaluation a slightly different result of

Table 2 Ratio of calculated-to-experimental spectrum-averaged cross sections <σ> of the ^{58}Ni(n,2n)^{57}Ni reaction obtained with various data of N(E) and σ(E).

Fission neutron field	C/E-values	Sources of	
		N(E)	σ(E)
Cf-252 (sf)	1.022 ± 0.077 a)	ENDF/B-VI [5]	IRK-90 [9]
	1.033 ± 0.079 a)	ENDF/B-VI [5]	IRDF-90.2
U-235 + n(thermal)	0.880 ± 0.037 b)	ENDF/B-VI c) [7]	IRK-90 [9]
	0.838 ± 0.035 b)	ENDF/B-VI d) [7]	IRK-90 [9]
	0.692	Watt spectrum [8]	IRDF-90.2

a) includes the uncertainty contributions of the spectrum, of σ(E) and of <σ>
b) without consideration of the spectral uncertainties
c) based on the given data, with lin-lin interpolation (INT=2) valid for all data
d) modified, by replacing the interpolation with INT=4 for the data above 10 MeV

1.033 is obtained. The difference between both evaluations is in the transformation of the original data to a 640 group structure. The C/E-value of 0.880 calculated in the ^{235}U field with the ENDF/B-VI spectrum [7] is an artefact. In ENDF/B-VI the number of data points given is insufficient to describe the shape of the original Madland-Nix model. After correction of this deficiency a value of 0.838 ± 0.035 is obtained in the ^{235}U neutron field. A comparison of this value with that obtained in the ^{252}Cf field shows the peculiarity of a lacking adequate model description of the high energy portion of the ^{235}U spectrum. The results of other high-threshold reactions confirm this trend and exclude the possibility that the underestimation is caused by erroneous integral experiments. At present, the C/E comparisons in the ^{235}U field are restricted to those σ(E) data with mean energy responses of < 10 MeV. When a Watt spectrum [8] is used, it is necessary to restrict the comparisons to σ(E) data with mean energy responses of < 7-8 MeV.

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Neutron Spectral Adjustment and Radiation Damage Calculations For Reactor Dosimetry

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The assessment of neutron exposure and radiation damage in materials is dependent on several key parameters including the measured neutron fluence spectrum and the derived radiation damage parameters displacements per atom (dpa) and gas production. Revised versions of the SPECTER [1] and STAY'SL [2] computer codes have been provided to the Nuclear Data Section of the IAEA for use with the production of IRDF-2002. STAY'SL performs a least-squares adjustment of neutron flux spectra, taking into account all known uncertainties and their associated covariances. SPECTER performs radiation damage calculations for any given neutron energy spectrum.

A new PC version of the SPECTER computer code provides dpa and gas production values, as well as the primary atomic recoil spectra, for over 40 elements and compounds and only requires the neutron flux spectrum and irradiation time as input data. A recent addition to SPECTER includes the new ASTM E693 standard for the dpa cross section of iron. The new ASTM standard iron dpa cross section has been compared with the previous ASTM standard, as shown in Figure 1, as well as the iron dpa cross section in SPECTER and, in general, spectral averaged dpa results differ by less than 10%. Nevertheless, it is important to consider such differences in the comparison of dpa values with previous work in the literature in order to prevent the creation of biases in the interpretation of radiation damage effects.

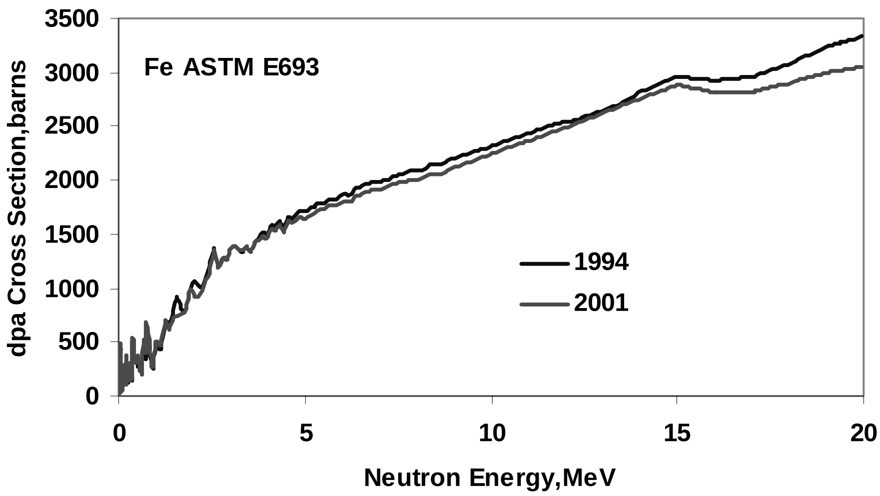

Figure 1 – Comparison of ASTM E693 dpa cross sections for iron from 1994 and 2001.

SPECTER has also been revised to include a calculation of the gas production and extra dpa due to the $^{58}Ni(n,\gamma)^{59}Ni(n,\alpha)$ and (n,p) reactions. These calculations are strongly non-linear due to the build up and burn out of ^{59}Ni and the results must be added to the fast neutron gas production. The production of helium or hydrogen from the two-step nickel reaction $^{58}Ni(n,\gamma)^{59}Ni$ (n,p or n,α) is given by:

$$N(x)/N_o(Ni) = 0.6808\, \sigma_X \{\sigma_\gamma(1-e^{-\sigma_T\phi t}) - \sigma_T(1-e^{-\sigma_\gamma\phi t})\} / \sigma_T (\sigma_\gamma - \sigma_T) \quad (1)$$

where N(x) = atoms of H or He at time t, $N_o(Ni)$ = initial atoms of Ni, 0.6808 is the abundance of ^{58}Ni, σ_x = spectral averaged reaction cross section for p or α production from ^{59}Ni, σ_γ = cross section for $^{58}Ni(n,\gamma)$, σ_T = total absorption cross section for ^{59}Ni, and ϕt = the total neutron fluence. Although the calculations in SPECTER average these reaction cross sections over the entire neutron spectrum, the largest contribution is due to the thermal neutrons.

The ^{59}Ni reaction calculations have recently been verified at very high neutron fluences in the High Flux Isotopes Reactor (HFIR) at Oak Ridge National Laboratory. [3] Figure 2 shows mass spectrometry data for various nickel samples irradiated to very high fluences in HFIR. The solid lines are calculations based on the evaluated ^{58}Ni and ^{59}Ni cross sections in ENDF/B-VI. [4] As can be seen, the data and measurements are in excellent agreement. However, helium measurements do not agree with the calculations at these high neutron fluences, as shown in Figure 3, suggesting a previously unknown source of helium due most likely to daughter or granddaughter isotopes. [3] Table 1 lists the Q-values and thermal neutron cross sections for several proton-rich isotopes in the mass range near nickel. Some of these isotopes may help explain the excess helium seen in Figure 3. However, further work is needed to determine the source of this excess helium.

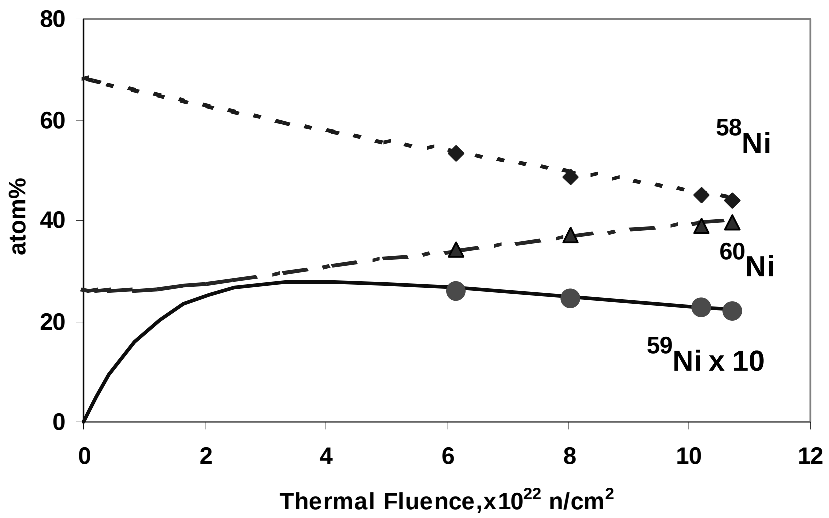

Figure 2 –Comparison of measured (points) and calculated (lines) nickel isotopic concentrations as a function of the thermal neutron fluence. Note that the ^{59}Ni values are multiplied times 10 for clarity.

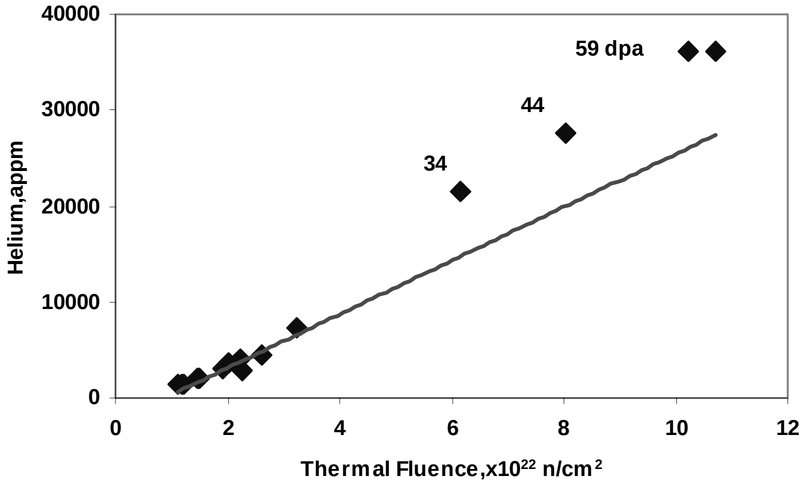

Figure 3 – Measured and calculated helium production from nickel irradiated in HFIR. The solid line is calculated using the evaluated ^{59}Ni cross sections. Dpa values are shown for 316 stainless steel (Ni dpa values are much higher).

Table 1 – Q-Values and Thermal Neutron Cross sections for H and He Production

Isotope	(n,α) Helium Reactions		(n,p) Hydrogen Reactions	
	Q,MeV	Thermal σ,b	Q,MeV	Thermal σ,b
Ni-59	5.096	12.0	1.855	1.96
Zn-65	6.481	4.7	2.134	?
Fe-55	3.584	0.011	1.014	?
Co-58	3.511	?	3.089	?
Co-57	1.858	?	1.618	?

Hydrogen measurements were also made on the high-dose HFIR nickel samples. The measurements were made using a new hydrogen measurement system developed at PNNL in 1999. [5] The system heats small radioactive samples up to ~ 1200^{o}C without melting or vaporizing them. Hydrogen leak standards are used to calibrate the system. In addition to rapid hydrogen release at a fixed temperature, hydrogen release can also be measured as a function of temperature. Hydrogen levels measured in the nickel samples are generally higher than calculated, as shown in Figure 4. The most surprising result, however, is that so much hydrogen is retained at the elevated temperatures of the HFIR samples. Hydrogen measurements for various steel samples irradiated to high neutron fluences also show surprisingly high retention of hydrogen at irradiation temperatures from 300 to 600^{o}C, contrary to expectations. With these data, there is a growing body of evidence that hydrogen can be trapped in voids associated with high levels of helium. [6,7] Although the measured hydrogen levels in Figure 4 are higher than the calculations, reactor samples are also exposed to high levels of hydrogen from environmental effects such as water radiolysis. Consequently, the excess hydrogen in these samples may be due to environmental rather than nuclear sources. Consequently, the main point is that the hydrogen is retained at these high temperatures.

A new PC version of the STAY'SL computer code is now available for distribution with IRDF-2002. This code performs neutron spectral adjustments based on the least squares technique,

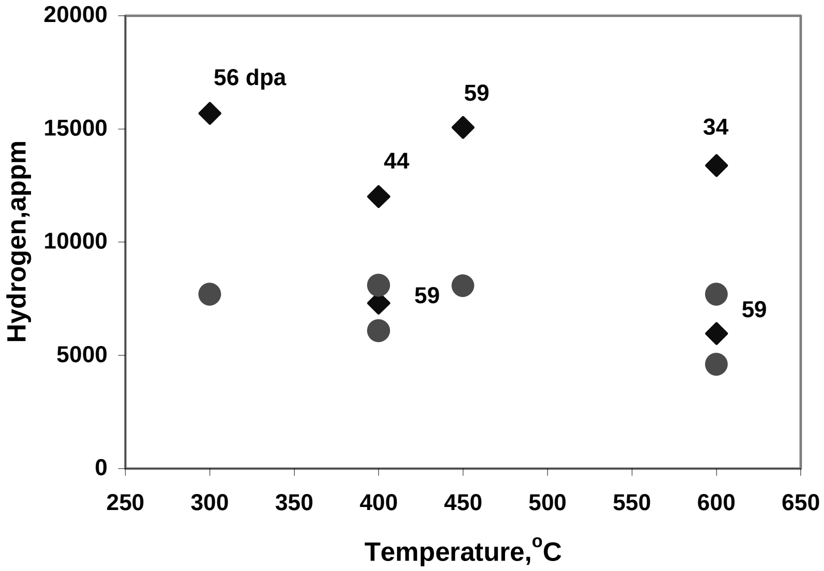

Figure 4 – Measured (diamonds) and calculated (dots) hydrogen in nickel irradiated in HFIR. Dpa values are shown for 316 stainless steel. Note that hydrogen is retained in the samples during irradiations from 300 to 600 °C.

given the input neutron flux spectrum and the measured reaction rates along with their uncertainties and covariances. The code package contains neutron activation cross sections and uncertainties compatible with IRDF-90. These libraries will be updated and tested with the new IRDF-2002 libraries when they become available.

Retrospective reactor dosimetry is rapidly becoming an accepted technique to assess materials exposure and radiation damage in operating power reactors. This technique analyzes small samples obtained from reactor components either during reactor outages or when components are removed from the reactor for replacement or decommissioning. Measurements have been made at various locations in BWR reactors and for baffle bolts removed from various PWR reactors. [8] The technique allows measurement of the neutron flux spectra and assessment of radiation damage in situations where standard reactor dosimetry capsules were not available. The most useful reactions include ^{54}Fe(n,p)^{54}Mn, ^{58}Ni(n,p)^{58}Co, and ^{59}Co(n,γ)^{60}Co, which are easily measured in most reactor steels. The concentration of the impurity cobalt, as well as exact values of the major elements, needs to be accurately determined by x-ray fluorescence or other techniques. It is also critically important to obtain an accurate power history for the reactor, taking into account any changes in the core edge fuel loading. There are a number of additional reactions which are very useful for this technique that are not generally used for reactor dosimetry, including ^{93}Nb(n,γ)^{94}Nb, ^{93}Nb(n,n')^{93m}Nb, ^{54}Fe(n,γ)^{55}Fe, and ^{62}Ni(n,γ)^{63}Ni. The latter three reactions require wet chemistry and x-ray or liquid scintillation counting. However, the longer half-lives of these reaction products make the reactions less dependent on details of the reactor power history. More work is needed to evaluate these cross sections and to provide the covariance data needed for neutron spectral adjustment so that the data can be added to IRDF-2002. Measurements of activation products in reactor components show reasonable agreement with calculations using activation cross sections from ENDF/B-V, as shown in Table 2. The spectral averaged cross sections and epithermal corrections for the thermal neutron reactions were calculated using a typical neutron spectrum for the last water node prior to the pressure vessel of a US BWR reactor near core midplane. Efforts are currently underway to compare these measurements with detailed neutron spectral calculations at the exact location of the samples, taking into account the fuel loading and power history of the reactor.

Although retrospective reactor dosimetry is relatively straightforward with stainless steel, Inconel, an alloy consisting of about 70% nickel and equal amounts of iron and manganese, produces significant interferences between several important activation reactions. The main interfering reactions are indicated by the footnotes in Table 2. More work is needed to develop standard procedures and evaluated cross sections for retrospective reactor dosimetry measurements. However, the technique has proven to be highly useful for determining neutron fluences and radiation damage for various locations in operating reactors, from core components to the reactor pressure vessel.

Table 2. Spectral Averaged Cross Sections and Neutron Fluences for Inconel Samples

	Neutron Cross Section, barns		Fluence, $x10^{17}$ n/cm^2
Reaction	Thermal	Epithermal factor	Thermal
$^{54}Fe(n,\gamma)^{55}Fe^{a}$	2.25	1.023	8.53 ±30%
$^{62}Ni(n,\gamma)^{63}Ni^{b}$	14.5	1.020	8.65 ±10%
$^{58}Fe(n,\gamma)^{59}Fe$	1.28	1.058	10.8 ± 5%
$^{50}Cr(n,\gamma)^{51}Cr$	15.9	1.021	9.86 ± 2%
$^{59}Co(n,\gamma)^{60}Co^{c}$	37.2	1.087	10.5 ± 12%
		Average =	**9.96 ± 10%**
Reaction	Fast > 1 MeV		Fast < 1 MeV
$^{93}Nb(n,n')^{93m}Nb$	0.211		8.28 ± 10%
$^{54}Fe(n,p)^{54}Mn^{d}$	0.185		7.48 ± 20%
$^{58}Ni(n,p)^{58}Co$	0.233		8.27 ± 15%
		Average =	**8.20 ± 10%**

[a] 50% correction applied for $^{58}Ni(n,\alpha)^{55}Fe$
[b] 3% correction applied for $^{62}Ni(n,\alpha)^{59}Fe$
[c] 12% correction applied for $^{60}Ni(n,p)^{60}Co$
[d] 40% correction applied for $^{55}Mn(n,2n)^{54}Mn$

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Decay Data for Reactor Dosimetry Applications

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One main objective of dosimetry applications is to determine by use of activation techniques the neutron fluence – sometimes also the neutron flux – at different places in a reactor. Other possible domains are activation and transmutation products, radiation damage, gaz production determination.

Up to now the nuclear data libraries devoted to these applications – like IRDF – consist of neutron induced cross sections only. As the main experimental method relies on the measurement of radiations emitted by the final radionuclides it was decided to complete the new IRDF library with an evaluated decay data section.

This contribution shortly describes:
– a possible list of radionuclides to be included in the library,
– the decay data of interest for dosimetry purposes,
– the main source of information for these data: the ENSDF[1] library,
– the processing of the data using the `SDF2NDF` code [BE02], and
– the control of the data before the transformation into the ENDF[2] format.
At present the conclusion is primarily a list of questions to be discussed during the meeting.

Which nuclei?

Considering the target elements and the nuclear reactions for which cross sections are given in the previous IRDF files and in other libraries like JENDL/D-99 or RRDF-98, a first list of radionuclides which can be considered in the decay data part of the IRDF-2002 library is given in Table 1 together with some primary decay data. The final list of radionuclides (including isomers, if any) will also have to be fully consistent with the final choice made for the cross section part of the library.

[1] Evaluated Nuclear Stucture Data File [ENSDF].
[2] Evaluated Nuclear Data File [ENDF].

Which type of data?

Besides the basic decay data – half-life, decay modes and intensities – which are necessary for any evolution calculation, the experimental data reduction also needs the characteristics – energy and intensity – of some specific emitted radiations (gamma, X-ray...).

The knowledge of the complete decay scheme is not necessary but it may gives more confidence in the partial data.

Origin of the data

Many of the data mentionned above were determined experimentally and published in the litterature. Within the Nuclear Structure and Decay Data (NSDD) international network these data are then collected, evaluated if necessary, and included in the ENSDF library using a specific format.

This format has the advantage that the presentation of the data closely follows the picture of a decay scheme and there is also room for many comments. Its major drawback is the lack of readability by a computer program because of its versatility and its softness. An example is given in Table 2 which describes the well-known ^{60}Co β^- decay.

Processing of the data

The decay data existing under the ENSDF format have to be converted into a more usual format for the reactor physicists: the ENDF format. The conversion is achieved by using the `SDF2NDF` code [BE02]. This code derives from the version 5.5 of the code RADLST [BU88] and was highly recoded, translated into double precision, and enhanced by several new features. It also calculates radiations emitted from the electronic cloud (X-rays, Auger electrons...). Several auxiliary output files were added in order to make data checking easier.

The ENDF file for the ^{60}Co β^- decay example is partly given in Table 3.

Control of the data

In addition to the format conversion aspect of the code a lot of physical checks are also performed by `SDF2NDF` to verify the data consistency. We can mention:

– the overall energy balance between the decay Q-value and the sum of the energies of all emitted particles (including recoils),
– the sum of the transition intensities depopulating an excited level must be equal to the feeding of this level,
– the transition intensity between two excited levels has to be the sum of the gamma intensity and the converted electron intensities,
– the total conversion coefficient has to be close to the sum of the partial coefficients for the different electron shells...

Conclusion

This note gives a short survey of the different components needed to produce a decay data section in the IRDF-2002 library but there are still a lot of open questions like:

- What are the key physical quantities for dosimetry purposes?
- What is the lower half-life limit for considering an excited state as isomeric?
- Which accuracy should be reached for which type of data?
- Which type of experimental result do we have to validate the data?

Answers to these questions are expected from the discussions during the meeting.

References

BE02 *The SDF2NDF Code*, O. BERSILLON, to be published.

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ENSDF *Evaluated Nuclear Structure Data File*, M.R. BHAT, Nuclear Data for Science and Technology, page 817, edited by S.M. QAIM, Springer-Verlag, Berlin, Germany, 1992.

Table 1: The decay data section of the IRDF-2002 library might include the following list of nuclides. The decay mode intensities are given in percentage. IT stands for internal transition and EC for electron capture.

Nucleus		Half-life		Mode	Int.
3- H -1	0	12.32	y	β^-	100.
11- C -6	0	20.39	m	β^+	100.
15- O -8	0	2.04	m	β^+	100.
18- F -9	0	1.83	h	β^+	100.
22- Na-11	0	2.6019	y	β^+	100.
24- Na-11	0	14.9590	h	β^-	100.
	1	20.02	ms	IT	99.95
				β^-	0.05
27-Mg-12	0	9.458	m	β^-	100.
31- Si-14	0	2.62	h	β^-	100.
32- P-15	0	14.262	d	β^-	100.
46- Sc-21	0	83.79	d	β^-	100.
	1	18.75	s	IT	100.
47- Sc-21	0	3.3492	d	β^-	100.
48- Sc-21	0	43.67	h	β^-	100.
45- Ti-22	0	3.08	h	EC	100.
51- Cr-24	0	27.7025	d	EC	100.
54-Mn-24	0	312.12	d	EC	$\sim$ 100.
				β^-	$<$2.9e-4
56-Mn-24	0	2.5789	h	β^-	100.
53- Fe-26	0	8.51	m	EC	100.
	1	2.526	m	IT	100.
59- Fe-26	0	44.503	d	β^-	100.
58- Co-27	0	70.86	d	EC	100.
	1	9.04	h	IT	100.
60- Co-27	0	5.2713	y	β^-	100.
	1	10.467	m	β^-	99.76
				IT	0.24
57- Ni-28	0	35.60	h	β^+	100.
62- Cu-29	0	9.726	m	β^+	100.
64- Cu-29	0	12.700	h	EC	61.0
				β^-	39.0
74- As-33	0	17.77	d	EC	66.
				β^-	34.
88- Y-39	0	106.65	d	β^+	100.
89- Zr-40	0	78.41	h	EC	100.
	1	4.161	m	IT	93.77
				EC	6.23
92- Nb-41	0	3.47e+7	y	EC	100.
	1	10.15	d	IT	100.
93- Nb-41	1	16.13	y	IT	100.
94- Nb-41	0	2.03e+4	y	EC	100.
	1	6.263	m	IT	99.50
				β^-	0.50

Nucleus		Half-life		Mode	Int.
103- Rh-45	1	56.114	m	IT	100.
110- Ag-47	0	24.6	s	β^-	99.70
				EC	0.30
	1	249.76	d	β^-	98.64
				IT	1.36
114- In-49	0	71.9	s	β^-	99.50
				EC	0.50
	1	49.51	d	IT	96.75
				EC	3.25
	2	43.1	ms	IT	100.
115- In-49	0	4.41e+14	y	β^-	100.
	1	4.486	h	IT	95.0
				EC	5.0
116- In-49	0	14.10	s	β^-	99.77
				EC	0.23
	1	54.29	m	β^-	100.
	2	2.18	s	IT	100.
126- In-49	0	1.60	s	β^-	100.
	1	1.64	s	β^-	100.
126- I-53	0	13.11	d	EC	56.3
				β^-	43.7
140- La-57	0	1.6781	d	β^-	100.
140- Pr-59	0	3.39	m	EC	100.
152- Eu-63	0	13.537	y	EC	72.1
				β^-	27.9
	1	9.3116	h	β^-	72.
				EC	28.
	2	96.	m	IT	100.
165- Dy-66	0	2.334	h	β^-	100.
	1	1.257	m	IT	97.76
				β^-	2.24
168-Tm-69	0	93.1	d	EC	99.99
				β^-	0.01
182- Ta-73	0	114.43	d	β^-	100.
	1	15.84	m	IT	100.
187- W-73	0	23.72	h	β^-	100.
196- Au-79	0	6.183	d	EC	92.8
				β^-	7.2
	1	8.1	s	IT	100.
	2	9.6	h	IT	100.
198- Au-79	0	2.69517	d	β^-	100.
	1	2.27	d	IT	100.
199- Hg-79	1	42.6	m	IT	100.
233- Th-90	0	22.3	m	β^-	100.
239- U-92	0	23.45	m	β^-	100.

Table 2: ENSDF format example (^{60}Co β^- decay) showing the close connection between the physical quantities and the data structure. Comments records are suppressed for clarity purpose.

```
60NI    60CO B- DECAY (1925.3 D)                                           200009
60NI  H TYP=UPD$AUT=R. Helmer$CIT=ENSDF$CUT=01-SEP-1996$DAT=12-SEP-2000$
60NI  N 1.0         1.0       1.0       1.0
60CO  P 0.0          5+             1925.3 D  3              2823.9   5
60NI  L 0            0+             STABLE
60NI  L 1332.508  4  2+             0.9 PS    3
60NI  B 1492      20 0.12   3           14.70 11                         2U
60NIS B EAV=625.87 21
60NI  G 1332.492  4 99.9826 6  E2                  1.28E-4 5
60NI2 G EKC=1.15E-4 5
60NI  L 2158.61   3  2+
60NI  B 670       20 0.000  2           14.0  GE                         2U
60NIS B EAV=274.93 21
60NI  G 826.10    3  0.0076  8 D+Q      +0.9     3    3.3E-4 4
60NI2 G KC=3.1E-4 4 $ LC=2.94E-5 17
60NI  G 2158.57   3 0.0012  2                         4.91E-5
60NI2 G KC=4.48E-5 14 $ LC=4.3E-6 2
60NI  L 2505.748  5  4+             0.30 PS    9
60NI  B 317.88    10 99.88  3          7.512  2
60NIS B EAV=95.77 15
60NI  G 347.14    7  0.0075 4                         5.54E-317
60NI2 G KC=5.03E-3 15 $ LC=5.08E-4 15
60NI  G 1173.228  3  99.85  3  E2(+M3)  -0.0025  22   1.68E-4 4
60NI2 G EKC=1.51E-4 7
60NI  G 2505.692  5  2.0E-6 4  E4                     8.6E-5 3
60NI2 G KC=7.8E-5 3 $ LC=7.6E-6 3
```

Table 3: ENDF format example (^{60}Co β^- decay) as converted from the ENSDF format. Only two sections are given for clarity purpose.

header section					
2.70600+04	5.94190+01	0	0	0	4
1.66346+08	2.59200+04	0	0	6	0
9.67355+04	2.42148+02	2.50384+06	3.52186+02	0.00000+00	0.00000+00
5.00000+00	1.00000+00	0	0	6	1
1.00000+00	0.00000+00	2.82390+06	5.00000+02	1.00000+00	0.00000+00
gamma section					
0.00000+00	0.00000+00	0	0	6	6
1.00000−02	0.00000+00	2.50384+06	3.52186+02	0.00000+00	0.00000+00
3.47140+05	7.00000+01	0	0	12	0
1.00000+00	0.00000+00	7.50000−03	4.00000−04	0.00000+00	0.00000+00
5.54000−03	1.70000−04	5.03000−03	2.12769−04	5.08000−04	2.13836−05
8.26100+05	3.00000+01	0	0	12	0
1.00000+00	0.00000+00	7.60000−03	8.00000−04	0.00000+00	0.00000+00
3.30000−04	4.00000−05	3.10000−04	4.10669−05	2.94000−05	1.91518−06
$\boxed{1.17323+06}$	3.00000+00	0	0	12	0
1.00000+00	0.00000+00	9.98500+01	3.00000−02	0.00000+00	0.00000+00
1.68000−04	4.00000−06	1.51000−04	7.00000−06	0.00000+00	0.00000+00
$\boxed{1.33249+06}$	4.00000+00	0	0	12	0
1.00000+00	0.00000+00	9.99826+01	6.00000−04	0.00000+00	0.00000+00
1.28000−04	5.00000−06	1.15000−04	5.00000−06	0.00000+00	0.00000+00
2.15857+06	3.00000+01	0	0	12	0
1.00000+00	0.00000+00	1.20000−03	2.00000−04	0.00000+00	0.00000+00
4.91000−05	0.00000+00	4.48000−05	1.94071−06	4.30000−06	2.37994−07
2.50569+06	5.00000+00	0	0	12	0
1.00000+00	0.00000+00	2.00000−06	4.00000−07	0.00000+00	0.00000+00
8.60000−05	3.00000−06	7.80000−05	3.80468−06	7.60000−06	3.76808−07

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