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

IAEA Headquarters
Vienna, Austria
1 – 3 October 2003

Prepared by
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Abstract

Presentations, recommendations and conclusions of the Final Technical Meeting on "International Reactor Dosimetry File: IRDF-2002" are summarized in this report. The main aims of this meeting were to discuss scientific and technical matters related to reactor dosimetry and to assign responsibilities for the preparation of the final version of the IRDF-2002 library and the associated TECDOC. Tasks were assigned and deadlines were agreed. Participants emphasized that accurate and complete nuclear data for reactor dosimetry are essential to improve the assessment accuracies for reactor pressure vessel service lifetimes in nuclear power plants, as well as for other neutron metrology applications such as boron neutron capture therapy, therapeutic use of medical isotopes, nuclear physics measurements, and reactor safety applications.

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1. OBJECTIVES AND AGENDA

The Final Technical Meeting on "International Reactor Dosimetry File: IRDF-2002" was held at the IAEA Headquarters in Vienna, Austria, from 1 to 3 October 2003. The primary aims of this meeting were to debate scientific and technical matters related to reactor dosimetry, and to assign responsibilities for the preparation of the final IRDF-2002 library and the associated TECDOC.

Dr. Lawrence R. Greenwood, PNL, USA was elected Chairman of the meeting, and Dr. Patrick J. Griffin, SNL USA agreed to act as rapporteur. The approved Agenda is attached (see Appendix 1). Other experts attending the meeting were Dr. Olivier Bersillon, CEA, France; Dr. K.I. Zolotarev, IPPE, Russia; Dr. E. M. Zsolnay, BUTE, Hungary; Mr. Kevin McLaughlin IAEA; and Mr. Henk J. Nolthenius, The Netherlands (observer). Dr. Wolf Mannhart (PTB, Germany) and Dr Keiichi Shibata (JAERI, Japan) were also invited, but were unable to attend; however, they provided their contributions as assigned at the previous meeting (INDC(NDS)-443). Dr. A. Nouri (OECD-NEA, France) who was invited as an observer was also unable to participate. The complete list of participants including affiliations and addresses are given in Appendix 2.

Dr. Andrej Trkov, Deputy Head of the Nuclear Data Section, welcomed the participants, and Dr. R. Paviotti-Corcuera, Scientific Secretary for the Technical Meeting, summarized the mechanisms and objectives of the Data Development Project (see page 23 of INDC(NDS)-443) and presented details of the WebPages of the project that describe the status of the current contributions. Requirements for this Data Development Project (DDP) are a new IRDF library and a TECDOC to be published prior to 1 January 2004. All written contributions from participants must be received prior to 31 October 2003 in order to meet this deadline. In addition, the final form of the proposed library must be prepared and endorsed by all participants at this same time.

All written contributions to the meeting by DDP participants are given in Appendix 3.

2. SUMMARY OF PRESENTATIONS AND DISCUSSIONS

K. Zolotorev

New updates to the RRDF-98 library were presented. Problems with the covariance matrix in previously submitted evaluations have been eliminated by using extended precision, which resolved the issue of negative Eigenvalues in some of the covariance matrices. A new ^{237}Np(n, f) evaluation was also provided; this file does not yet have a File 1 comments section, but includes the data required to finalize the contents of the IRDF-2002 library. Several example cases were presented where excessive scatter in the experimental data had been resolved by carefully tracing the standards used by the experimenter and re-normalizing the data using the current best estimates of the reference standard cross sections. This renormalization required that Zolotarev track down the experimental details, often contacting the actual experimenters since details were not provided in their written documentation. Cases were shown where the resulting re-normalization procedure dramatically collapsed the spread in the experimental data.

Some of Zolotorev's tables have the latest RRDF-98 contributions labelled as IRDF-2002 evaluations. Since the final IRDF-2002 library contents will not be decided until the end of this meeting, these contributions will be relabelled Updated RRDF-98.

Zsolnay expressed the views of the whole meeting when she thanked Zolotarev for his extensive contributions to this DDP, and for his quick response to requests to resolve the issues that arose from the reviews.

The issue of adding cumulative fission yields to the IRDF-2002 library was discussed since these data are part of the database required in using the new dosimetry cross sections. Trkov noted that the IAEA has an on-going project that addresses this need, but the results will not be available within the timeframe of the IRDF-2002 release. The issue of the addition of fission yields to the IRDF-2002 library was set aside for consideration as part of any future revision.

L. Greenwood

Greenwood presented plots of the 14-MeV experimental data and the IRDF-2002 candidate cross sections for the high-energy dosimetry reactions. A complete version of this comparison is available in Appendix 3. A survey of these data shows that the spread in the 14-MeV reference neutron field does not provide significant help in the selection of suitable data sets for IRDF-2002. Discussions highlighted the fact that the 14-MeV neutron spectrum is not monoenergetic - there is a variation in the out-going neutron energy based on the energy of the incident deuteron in the DT reaction, and a spread in the neutron energy based on the solid angle subtended by the target and scattering within the target. Good experimental data in this area provide a quantitative estimate of the uncertainty/spread in the neutron energy. In general, the mean neutron energy varies from 13.5 to 15 MeV, with an energy spread of 50-500 keV (FWHM). Data spreads of the order of 1 MeV were generally discarded in his analysis.

Greenwood provided information on the neutron displacement damage. This information is available on the DDP website and was not discussed in detail. The website and the IRDF-2002 release will provide a version of the SPECTER code that can be used to produce damage response functions for a 40-element library.

In response to a question, Greenwood indicated that an extended version of the SPECTER code that produces the detailed recoil spectra for the major residual nuclei can also be provided in the IRDF-2002 release. Detailed recoil spectra are used to estimate the neutron-induced damage for some responses that exhibit an LET dependence (e.g., damage described by track structure modeling such as the neutron response of alanine dosimeters using an electron spin resonance readout procedure and response of CaF_2:Mn TLDs).

W. Mannhart

Mannhart could not attend the meeting, but had submitted a written contribution on the testing of candidate cross sections in the ^{252}Cf standard neutron field. A detailed discussion of this contribution was held in conjunction with the next presentation.

E. Zsolnay

Zsolnay presented the results of a preliminary cross-section selection procedure for the IRDF-2002 candidates, both for the thermal-intermediate and for the fast neutron reactions. In the low neutron energy region, cross-section selection is based (after a detailed analysis of the data of interest) on comparisons of the calculated integral cross sections with the experimental data in the thermal and resonance-region standard neutron benchmark fields (Maxwellian and 1/E fields), and on the quality of the uncertainty information accompanying the cross-section data of interest (see details in Appendix 3). For the fast neutron reactions, cross-section selection is based on the results of Mannhart, obtained by comparing the calculated integral cross sections with experimental data in the ^{252}Cf spontaneous fission standard benchmark neutron field, and on the quality of the uncertainty information accompanying these cross-section data (see details in Appendix 3).

Practically no new cross-section evaluations have been made in the low neutron energy region over the previous one-two decades, except the ^{139}La(n, γ) and ^{186}W(n, γ), reactions evaluated for RRDF. At the same time, cross sections and information on their uncertainties need to be improved for several reactions (e.g., only diagonal matrices are present for the ^{93}Nb(n, γ), ^{115}In(n, γ), ^{181}Ta(n, γ) and ^{232}Th(n, γ) reactions below 15 eV).

No C/E values could be derived for a number of reactions because no (or no reliable) experimental data are available in the Maxwellian, 1/E or ^{252}Cf spontaneous fission neutron field.

O. Bersillon

Bersillon presented his recommended nuclear data, based on analyses of ENSDF data (see Appendix 3). ENSDF data are updated regularly, and Bersillon will provide the date of issue of the ENSDF library used in his studies.

A problem was raised concerning the ^{64}Cu and ^{62}Cu for which decay data were only given for the 511-keV emission. Since this line can be difficult to use because of interference from annihilation radiation from other sources, Bersillon agreed to add data for other low-yield lines that may be employed in counting by IRDF-2002 users.

Paviotti requested that all contributions to the final TECDOC be provided in Word format. Bersillon had used Latex to present his results and decay-data recommendations. Despite the merits of Latex, the Scientific Secretary would be faced with the problem of format conversion. No obvious solution had been found in previous exercises of this type, although methods to resolve this issue will be addressed by all parties.

Bersillon presented nuclear data for all potential residual nuclei, based on a list provided by Zsolnay. During the course of this meeting, several additional reactions were added to the proposed library (e.g., ^{22}Na, ^{168}Tm, ^{182}Ta, ^{199m}Hg, ^{204m}Pb, ^{233}Th and ^{239}U). Bersillon noted these additions, and promised to add these data to his contribution prior to the agreed deadline.

ENSDF data are not fully consistent with the recommendations emerging from the on-going IAEA-CRP on "Update of X-ray and Gamma-ray Decay Data Standards for Detector Calibration and Other Applications". Bersillon has examined these data, and endorsed the detailed considerations that went into the CRP recommendations for the subset of isotopes considered. However, he noted that differences in decay data between ENSDF and the CRP-

recommended values are not statistically significant for any dosimetry application involving the IRDF-2002 cross sections. There was a desire to be consistent with the detector calibration recommendations, but these CRP data are currently in draft from only. In view of this situation, the inclusion of the detector calibration data should be addressed in any future revision of IRDF-2002, and the current release will use the data from ENSDF as compiled by Bersillon; a note will be placed in the IRDF-2002 documentation stating that users desiring higher fidelity nuclear data for isotopes used as detector calibration standards should consult "Update of X-ray and Gamma-ray Decay Data Standards for Detector Calibration and Other Applications". Citation to the most recently published version of this IAEA database will be given, along with a statement that an IAEA-sponsored CRP is in the process of formulating an update to these data.

The consistency of the abundance data recommended by Bersillon with the abundances used by the evaluators for IRDF-2002 was addressed. Actions were assigned to determine whether any statistically meaningful inconsistencies would arise due to the differences.

The need for full decay schemes in the nuclear data summary released with IRDF-2002 was discussed - full decay schemes would be given in electronic form on the companion CD and web page.

P. Griffin

Griffin addressed the absence of a candidate file for the ^{23}Na(n, γ) reaction for IRDF-2002. No suitable file has been identified because all available covariance matrices only consisted of the main diagonal. While a diagonal covariance matrix indicates significant deficiencies in the treatment of data correlations, the presence of such a covariance matrix provides sufficient quantification of the uncertainty to permit the intelligent use of the reaction for dosimetry applications. Therefore, the participants were justifiably able to soften their position from that of the pervious DDP meeting, and not reject candidate evaluations that had only a diagonal component to the covariance matrix (but downgrade the fidelity of such evaluated files). This relaxation permits IRDF-2002 to include the best available evaluations for some important reactions, such as ^{23}Na(n, γ), in which only diagonal covariance matrices are available. This acceptability criterion was carried over into the candidate-selection discussions that were conducted at later sessions of this IRDF-2002 meeting. Other reactions affected included ^{93}Nb(n, γ), ^{115}In(n, γ), ^{181}Ta(n, γ) and ^{232}Th(n, γ).

Griffin provided details of an evaluation of the potential ^{23}Na(n, γ) candidates and concluded by recommending that the ENDF/B-VI evaluation be used (older ENDF version/evaluation incorporated into the ENDF/B-VI library).

Participants endorsed the detailed and quantified methodology used to select the ^{23}Na(n, γ) reaction. We may implement this procedure in future for the selection of all candidates, and the example of ^{23}Na(n, γ) may provide a basis for the development of a refined methodology.

Griffin addressed the results of testing the IRDF-2002 candidate files with the NJOY99 code. While NJOY99 data reduction of IRDF-2002 was not required, problems discovered during these tests may highlight inconsistencies in the adherence of the evaluation formats with current ENDF-6 format rules

While Griffin discussed the types of problem, the fixes for specific evaluations were not clearly identified. Many of these compatibility issues were identified and fixed during format

checks performed by McLaughlin, and reported in Appendix 3. The covariance matrices for several evaluations could be processed with NJOY97, but not with NJOY99, suggesting that some patches may have been applied to NJOY99 that resolves some of the issues but were not included into the latest version of the code. Potential NJOY99 defects will be submitted to the web-based defect tracking system: significant issues were associated with the covariance file for the IRDF90 ^{32}S(n, p) evaluation, cross-section processing of the IRDF90 ^{46}Ti(n, p) evaluation, and cross-section processing for the JEFF-3.0 ^{58}Ni evaluation. Griffin agreed to use NJOY99 to test the final IRDF-2002 candidate files, and to report any remaining incompatibilities along with suggested steps to resolve them. This action requires completion by 20 October 2003.

Griffin presented the results of validation tests on the IRDF-2002 candidate files using an ACRR water-moderated reference neutron field – this form of testing in a reference field should NOT be used to select IRDF-2002 evaluations, but can be used to ensure the consistency of selection. These studies indicate some areas for future analysis, but (with one exception) did not indicate that any of the IRDF-2002 candidate reactions could not be validated in this reference neutron field. The one exception related to the failure to apply the 0.79 branching ratio to the ^{115}In(n, n′) reaction. This deficiency had been previously identified in tests of IRDF-90v1, and was fixed for the IRDF-90v2 release – the problem arose again when some IRDF-90 point cross-section files were reconstituted in support of the IRDF-2002 DDP, and can be easily corrected. Zsolnay will ensure that the branching ratio is applied to the latest IRDF-90 candidate file if selected for use in IRDF-2002 [Secretary's note: this reaction was subsequently selected].

ACRR testing indicated that the lack of a high-quality ^{nat}Cd absorption cross section file above the cut-off energy may be responsible for some observed discrepancies in Cd-covered foil activities. Trkov requested a high-priority action be placed with the WPEC requesting refined cross-section data in this area. Griffin was asked to compose the request and address the required accuracy and energy range, while Trkov will ensure that the request is placed with the WPEC.

Griffin presented data on candidate file validations in the SPR-III reference neutron field (see details in Appendix 3) – there were no file rejections, and the results strongly favour JENDL for the ^{58}Fe(n, γ) reaction. He also presented data on the standard neutron damage exposure functions that will be included within the IRDF-2002 release (see details in Appendix 3).

K. Shibata

Although Shibata was unable to attend the meeting, his contributions were presented. An updated ^{23}Na(n, 2n) cross-section evaluation has resolved discrepancies in the covariance matrix, and this file is now available for consideration by the DDP. Shibata's written contribution was distributed, along with a report on this new ^{23}Na cross-section evaluation published recently in the Journal of Nuclear Science and Technology.

E-mails from Shibata were distributed indicating that a revision of the JENDL evaluation of the ^{23}Na(n, γ) reaction was not warranted because no new experimental data have become appeared since the preparation of IRDF-90v2.

P. K. McLaughlin

McLaughlin presented discrepancies uncovered in the format testing of the IRDF-2002 candidate evaluations. A significant issue in this work related to the use of MT numbers for reactions resulting in metastable residual nuclei. After much debate, the group decided that the IRDF-2002 point cross-section library will use File 10 (MF) to present these data, while the group-wise IRDF-2002 companion file will use File 3 but with MT numbers taken from the unassigned region. The following convention will be adopted for MT designations:

(n, np) + (n, pn) + (n, d)	231
inelastic to metastable	291
(n, 2n)to metastable	292
(n, γ) to metastable	293

Groupwise assignments for reactions on elemental targets resulting in a specific residual nucleus will be assigned unique MT numbers. The assignments (selected to be consistent with previous applications by other dosimetry libraries) include:

^{nat}Ti(n, X)^{46}Sc	220
^{nat}Ti(n, X)^{47}Sc	221
^{nat}Ti(n, X)^{48}Sc	222

In several cases, there were minor problems identified with Q-values and energy bounds. A suitable procedure to resolve these issues was identified, and McLaughlin will apply this process to the selected IRDF-2002 candidate reactions.

The participants emphasized that accurate and complete nuclear data for reactor dosimetry are essential for improving the assessment accuracy for reactor pressure vessel service lifetimes in 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.

3. LIBRARY SELECTION

The initial list of candidate reactions was based upon the results of the selection procedure performed by Zsolnay and Nolthenius. These proposed reactions are discussed in the contributions detailed in Appendix 3. The group discussed the candidate reactions individually, and arrived at a consensus for the recommended evaluations based on the nuclear data and C/E values in the standard benchmark neutron fields. Because of the quality of the benchmark data and the variations within candidate evaluations, the results in the ^{252}Cf spontaneous fission field proved to be the prime contributor to the selection process for fast cross sections. Maxwellian (2200 m/s, 0.02253 eV) and 1/E standard neutron fields were used for the thermal and epithermal regions.

The selected evaluations, along with caveats/comments are listed in Table I.

Table I. Selected Evaluations and Comments.

Reaction	Selected Evaluation	Comments
^{6}Li(n, t)	IRDF-90	Contact CSWEG for covariance update
^{10}B(n, α)	IRDF-90	Contact CSWEG for covariance update
^{19}F(n, 2n)	RRDF	
^{24}Mg(n, p)	IRDF-90	
^{27}Al(n, p)	IRDF-90	Complete check of RRDF required and reconsideration may be necessary
^{27}Al(n, α)	IRDF-90	
^{31}P(n, p)	IRDF-90	Request measurement in ^{252}Cf field
^{32}S(n, p)	IRDF-90	
^{45}Sc(n, γ)	IRDF-90	
^{46}Ti(n, 2n)	RRDF	Request measurement in ^{252}Cf field
^{46}Ti(n, p)	RRDF	
^{47}Ti(n, np)	RRDF	
^{nat}Ti(n, X)^{46}Sc	Not currently selected	Will be generated in a consistent fashion by additional component selections
^{47}Ti(n, p)	IRDF-90	
^{48}Ti(n, np)	RRDF	
^{nat}Ti(n, X)^{47}Sc	Not currently selected	Will be generated in a consistent fashion by additional component selections
^{48}Ti(n, p)	RRDF	
^{49}Ti(n, np)	RRDF	
^{nat}Ti(n, X)^{48}Sc	Not currently selected	Will be generated in a consistent fashion by additional component selections
^{51}V(n, α)	RRDF	
^{52}Cr(n, 2n)	IRDF-90	
^{55}Mn(n, γ)	Special JENDL-D99/ JENDL-3.3 combination	Combination used to restore better RI parameters and maintain consistency with activation community
^{54}Fe(n, 2n)	RRDF	
^{54}Fe(n, α)	RRDF	
^{54}Fe(n, p)	IRDF-90	
^{56}Fe(n, p)	RRDF	
^{58}Fe(n, γ)	JENDL-D99	Need improved evaluation in future; work currently in progress expected to be complete by mid-October
^{59}Co(n, 2n)	IRDF-90	
^{59}Co(n, α)	RRDF	
^{59}Co(n, γ)	IRDF-90	
^{58}Ni(n, 2n)	JEFF 3.0	
^{58}Ni(n, p)	RRDF	
^{60}Ni(n, p)	ENDF/B-VI	
^{63}Cu(n, 2n)	ENDF/B-VI	
^{63}Cu(n, γ)	IRDF-90	
^{63}Cu(n, α)	RRDF	

Reaction	Selected Evaluation	Comments
^{65}Cu(n, 2n)	IRDF-90	
^{64}Zn(n, p)	IRDF-90	Need improved evaluation in future
^{75}As(n, 2n)	RRDF	
^{89}Y(n, 2n)	JENDL-D99	
^{90}Zr(n, 2n)	IRDF-90	
^{93}Nb(n, 2n)	RRDF	
^{93}Nb(n, n′)	RRDF	
^{93}Nb(n, γ)	IRDF-90	Diagonal covariance needs to be improved
^{103}Rh(n, n′)	RRDF	Future requirement to resolve discrepancy between microscopic data and integral field data
^{109}Ag(n, γ)	IRDF-90	Need improved evaluation in future; poor structure in resonance region
^{115}In(n, 2n)	IRDF-90	
^{115}In(n, n′)	RRDF	
^{115}In(n, γ)	IRDF-90	Fix diagonal covariance in future update.
^{127}I(n, 2n)	IRDF-90	
^{139}La(n, γ)	RRDF	Require update of experimental RI data
^{141}Pr(n, 2n)	RRDF	Require ^{252}Cf measurement
^{169}Tm(n, 2n)	JENDL/D99	
^{186}W(n, γ)	RRDF	
^{181}Ta(n, γ)	JENDL-D99	Diagonal covariance needs to be improved
^{197}Au(n, 2n)	IRDF-90	
^{197}Au(n, γ)	IRDF-90	Contact CSWEG for covariance update
^{199}Hg(n, n′)	JENDL-D99	Very poor agreement in ^{252}Cf field - requires attention
^{204}Pb(n, n′)	RRDF	Require ^{252}Cf measurement
^{232}Th(n, γ)	IRDF-90	Requirement to improve diagonal covariance in future update
^{232}Th(n, f)	IRDF-90	
^{235}U(n, f)	IRDF-90	Contact CSWEG for covariance update
^{238}U(n, f)	JENDL/D99	
^{238}U(n, γ)	IRDF-90	Need experimental RI update
^{237}Np(n, f)	JENDL-D99	RRDF considered after metrics studied
^{239}Pu(n, f)	JENDL-D99	
^{23}Na(n, γ)	Open issue	Possible requirement to improve diagonal covariance in future update
^{23}Na(n, 2n)	JENDL-D99	Evaluation to be analysed before a decision can be taken
^{nat}Cd(n, X)	ENDF/B-VI	Cover - no covariance available
^{nat}Gd(n, X)	IRDF-90	Cover - no covariance available
^{241}Am(n, f)	JENDL-D99	Need experimental data for validation

Miscellaneous Issue

The final name of the IRDF-2002 library was discussed. Consideration was given to adopting "IRDF-2003" for the final release since this library represents consideration of available evaluations up to 2003. However, the final consensus position was to retain "IRDF-2002" as the title despite the library being released in 2003/04, because this name has already been used a number of times in several publications and in the IAEA programme.

4. ASSIGNED TASKS AND SCHEDULE

Contents of IAEA TECDOC:

The proposed contents list drafted by Zsolnay was used in these discussions (see Appendix 4).

Paviotti will write the **Introduction** – include a discussion of the name of the release, and the pertinent dates covering the preparation of the data for the library.

Bersillon will provide a short discussion on **abundances and decay constants** for Chapter 2.

Zsolnay will produce a list of the **main characteristics and sources of the cross sections**to be found in IRDF-2002.

McLaughlin will write a description of the **format**.

Both the group and pointwise libraries need to be clearly designated by "G" and "P" as part of the IRDF-2002 library name.

The decay constants will be in ENDF format (MF=8, MT=457). Details of the final form of the decay data remain to be reviewed by Trkov and McLaughlin. Bersillon will send the final decay data to McLaughlin by 9 October 2003; these data will only be included in the point cross-section library (not in the group library).

All new evaluations that are not part of the previously documented library will be described in the TECDOC.

New Russian Evaluations: Zolotarev's most recent contribution constitutes the main body of Chapter 3, which describes the methodology used for these new evaluations - full descriptions will be included as appendices in the library.

Selection Procedures: Derivation depended heavily on the C/E values in standard benchmark fields; ^{252}Cf field was the most useful. Include caveat on importance of standard renormalization of data in 14-MeV region, based on changing reference standard cross section values. Greenwood will update his 14-MeV discussion using information provided by Zolotarev (to provide this information by early November 2003). Discussion will only addresses the selected evaluations for the final IRDF-2002 library.

Consistency Test of Cross Section Data: Griffin will provide the text for Chapter 5.

Radiation Damage: Greenwood and Griffin will confer on combining their sections on this topic as Chapter 6.

Who will update STAYSL libraries with the new IRDF-2002 selections? - there is no time available for this exercise, which will be delayed until a later update.

Chapter 7 will address **abundances and decay data** - most of the tables will be included in an appendix.

Chapter 8 will contain the **cross section plots**. McLaughlin will provide these figures for selected reactions (using Zerkin's code). Uncertainties for the evaluations will also be included whenever possible.

Appendix 1 – Zolotarev: detailed descriptions of the new evaluations, including Al, Fe, La, W, Pb and Np.

Appendix 2 – 14-MeV plots – Greenwood.

Appendix 3 – Plots of cross sections. – McLaughlin.

Appendix 4 – Covariance data – Nolthenius.

Appendix 5 - Nuclear decay data – Bersillon.

The agreed contents page of the TECDOC is shown in Appendix 5, as discussed above.

Deadlines:

End of October 2003 for all written material (definitive cut-off of 10 November 2003).

Need final files now so McLaughlin will have them all by 17 October 2003.

Actions:

Greenwood - new Holden thermal and resonance integral data file has been published in a CRC handbook, and should be used in IRDF activities. Analysis presented at this meeting was based on the 1996 version of these Holden data. . Greenwood* will contact Holden and clarify the status of this updated compendium on resonance and thermal nuclear data. The latest version of the document will be passed on to Paviotti* for distribution to DDP participants via the Web for use in support of the final documentation of IRDF-2002. The Holden document will not be distributed as part of the library – but the publication will be referenced, and the website will be cited at which the document can be viewed.

> [Secretary's note: this task has been completed; correct reference is:
>
> N.E. Holden, "Neutron Scattering and Absorption Properties (Revised 2003)", CRC Handbook of Chemistry and Physics, 84th Edition, Chapter 11, pp 198-213, D.R. Lide, Editor-in-Chief, CRC Press, 2000 NW Corporate Blvd., Boca Raton, Florida 33431, (2003)].

Griffin* - perform NJOY99 tests on the final IRDF-2002 reaction files resulting from this meeting, and report on any remaining incompatibility issues along with suggested steps to resolve these problems. Action to be completed by 20 October 2003.

McLaughlin* – ensure that the 0.79 branching ratio is applied to the ^{115}In(n, γ) file selected for inclusion in IRDF-2002.

Griffin* – e-mail Trkov by 15 October 2003 with a statement concerning the required accuracy and energy range for refined ^{nat}Cd absorption cross-section data.

Greenwood* – same task as above for Gd (co-ordinate response with Griffin before transmission to Trkov).

Trkov* – communicate high-priority requests to WPEC for refined absorption cross-section measurements for ^{nat}Cd and Gd.

McLaughlin* – apply format adjustments to the selected IRDF-2002 evaluations, and post them on the DDP web site by 15 October 2003.

Greenwood* – inspect the selected cross sections for the the ^{58}Fe(n, γ), Ni, Cu, Ti and V reactions, and ensure that the abundances used by the evaluator are not in conflict with the recommended of Bersillon (as recommended for the release of IRDF-2002).

[Secretary's note: this task has been completed – comments received from Greenwood:

"Ideally, applications of the reactor dosimetry cross sections in IRDF-2002 should use the same isotopic abundance values as were used in the cross-section evaluations. Unfortunately, this information is not readily available in any of the evaluated cross-section files for multi-element isotopes, such as Ti, Fe, Ni or Cu. In cases where the isotopic abundance is a high value, this does not introduce any significant error since the abundance value is known to a high accuracy. However, a significant error is possible when the isotopic abundance is relatively low or may have changed in recent years, such as Fe-58. We were not able to verify that the recommended cross section file for the ^{58}Fe(n, γ)^{59}Fe reaction from JENDL/D-99 used the same isotopic abundance for ^{58}Fe as recommended in IRDF-2002. However, we note that the abundance and thermal cross section values recommended by N. E. Holden of 0.28% and 1.3 barns agree with the isotopic abundance value recommended in IRDF-2002 and the thermal neutron cross section in JENDL/D-99".]

Trkov* – update the resonance description for the ^{55}Mn(n, γ) reaction using the JENDL 3.3 evaluation while preserving the JENDL-D99 covariance matrix; this action to be completed by 17 October 2003, and e-mailed to Zsolnay.

Zsolnay* – analyse the ^{55}Mn(n, γ) evaluation received from Trkov - if this file conforms with the various checks, these data will become the selected evaluation; required one week after receipt of the file from Trkov.

Griffin* – combine the covariance files for components of the ^{nat}Ti(n, X)^{46,47,48}Sc reactions so that these reactions on a natural element can be reported in the next update to the IRDF library.

Zolotarev* – will give priority to completing the ^{47}Ti(n, p) evaluation in support of a consistent definition of the ^{nat}Ti(n, X)^{47}Sc cross section.

Griffin* – will contact Carlson (NIST) and determine the status of the CSWEG-recommended covariance files for the standard cross sections. Recommended covariance files for ENDF/B-VI were withdrawn due to problems in the treatment of the correlations in the R-matrix resonance data. Updated recommendations were to have been released, but did not appear. Griffin will determine from Carlson what data are available, and inform Paviotti who will e-mail DDP participants and post the data on the website. In some cases (indicated in the comments associated with the IRDF-2002 selection and restricted to ENDF/B-VI candidate evaluations), these data may be used to update the older IRDF-90v2 cross-section covariances to be used in IRDF-2002.

Nolthenius – e-mail Griffin the processed IRDF90 S32P covariance results to help resolve the covariance problem with this reaction (as detected during NJOY99 processing).

Zsolnay* – process the latest RRDF Np237F file and report the results; then reconsider the selection of the JENDL-D99 or RRDF evaluation for this reaction and e-mail conclusions to DDP participants (Zsolnay will coordinate this activity with Mannhart); required by 17 October 2003.

Zsolnay* - do same as above for ^{27}Al(n, p) and ^{23}Na(n, 2n).

Mannhart* - requested to test some additional reactions: ^{27}Al(n, p) (new evaluation of Zolotarev), ^{23}Na(n, 2n) (new evaluation of Shibata), and ^{237}Np(n, f)) (new evaluation of Zolotarev) in the ^{252}Cf field. If compiled data doe not currently exist, data gathering is requested; if compiled data exist for these reactions, requested deadline is 30 October 2003.

**Starred items are assigned tasks.*

Further issues to be addressed:

Trkov noted the requirement to provide total cross sections for the dosimetry reactions so that users could apply consistent self-shielding corrections. This need may apply to 19 capture reactions in the library, and has been raised in an e-mail from Dean (Serco Assurance, UK).

The proposal was to add both total and elastic cross sections to IRDF-2002. Someone must write an appropriate chapter for the final TECDOC to provide guidance on how these total and elastic cross sections should be used. The library will also include recommended cover materials for Cd and Gd - ^{nat}Cd cross sections from the ENDF/B-VI library will probably be used, and the IRDF-90 library contains Gd cross section from NDS.

Greenwood will prepare the chapter on the correction methodology, while McLaughlin will provide the updated files for the IRDF-2002 release. A disclaimer will also be required to indicate that the data arising from the self-shielding correction have not been tested.

Group and point cross-section files will be provided for the self-shielding corrections.

Appendix 1: Agenda

International Atomic Energy Agency

Data Development Project: Technical Meeting

International Reactor Dosimetry File: IRDF-2002

IAEA Headquarters, Vienna, Austria

1-3 October 2003

Meeting Room A-2774

AGENDA

Wednesday 1 October

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

09:30 - 10:30 Opening Session:

- Welcoming address –Andrej Trkov, Acting Section Head, (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 - 11:00 Administrative/Financial Matters and Coffee break

11:00 - 12:45 Session 1: Presentations by Participants, and Discussions

(15 minutes for each presentation, and 5 minutes for discussion)

1. Evaluated Cross Section Data for the Russian Reactor Dosimetry, ***K. I. Zolotarev***, Institute of Physics and Power Engineering, Obninsk, Russia.
2. *Neutron Spectral Adjustment and Radiation Damage Calculations for Reactor Dosimetry,* ***Larry R. Greenwood***, Pacific Northwest National Laboratory, USA.
3. *Selection of Cross Sections for Thermal Neutron Capture and Fission Reactions for the File RDF-2002, and Characterization of the Selected Data,* ***Eva Zsolnay***, and ***H. J. Nolthenius***, Budapest University of Technology and Economics, Budapest, Hungary.
4. *Response of Activation Reactions in the Neutron fields of Spontaneous Fission of* ^{252}*Cf,* ***Wolfgang Mannhart***, Physikalisch-Technische Bundesanstalt, Braunschweig, Germany.

5. *Selection of Fast Neutron Reaction Cross Sections for the File IRDF-2002 and Characterization of the Selected Data,* ***Eva Zsolnay and H. J. Nolthenius****,* Budapest University of Technology and Economics, Budapest, Hungary

12:45- 14:00 Lunch and Administrative/Financial Matters

14:00 - 18:00 Session 1: Presentations by Participants, and Discussions (cont.)

6. *DecayData and Isotopic Abundances for Dosimetry Applications,* ***Olivier Bersillon*****,** CEA Bruyeres-le-Chatel, France.
7. *Selection of NA23G Dosimetry Cross Sections; Review of the IRDF-2002 Candidate Cross; Validation in Reference Neutron Fields; Damage Response Functions in the IRDF-2002 Library,* ***Patrick Griffin***, Sandia National Laboratories, USA.
8. *Checking and Corrections made to the Files of Candidate Cross Sections, Graphic Representation,* ***Patrick McLaughlin*****,** IAEA-NDS, Vienna, Austria.

Thursday 2 October

09:00 - 12:30 Session 2: Discussions

Data for IRDF and other related CRPs (consistency of data in IAEA/NDS publications)

Final Selection of Reactions to be included in IRDF

TECDOC: structure and individual writing assignments with deadlines.

Coffee break as appropriate.

12:30 - 14:00 Lunch

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

Coffee break as appropriate.

Friday 3 October

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

Coffee break as appropriate.

12:30 - 14:00 Lunch

14:00 - 16:30 Concluding Session - Discussion and Approval of Meeting Report

Coffee break as appropriate.

Appendix 2: List of Participants

IAEA Final Technical Meeting on

Data Development Project "International Reactor Dosimetry File: IRDF-2002"

1–3 October 2003, IAEA Headquarters, Vienna, Austria

Meeting Room A-2774

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Appendix 3: Papers Presented

Evaluated Cross Section Data from Russian Reactor Dosimetry File, ***Konstantin I. Zolotarev****, Institute of Physics and Power Engineering, Obninsk, Russia.*

Evaluation of Cross Sections for IRDF-2002 at 14 MeV, ***Larry R. Greenwood****, Pacific Northwest National Laboratory, USA.*

Response of Activation Reactions in The Neutron Field of Spontaneous Fission of ^{252}*Cf,* ***Wolfgang Mannhart****, Physikalisch-Technische Bundesanstalt, Braunschweig, Germany.*

Selection of Cross Sections in the Thermal and Epithermal Neutron Energy Region for the File IRDF-2002, and Characterization of the Selected Data, ***Eva Zsolnay and H. J. Nolthenius****, Budapest University of Technology and Economics, Budapest, Hungary.*

Selection of Fast Neutron Reaction Cross Sections for the File IRDF-2002 and Characterization of the Selected Data, ***Eva Zsolnay and H. J. Nolthenius****, Budapest University of Technology and Economics, Budapest, Hungary.*

Decay Data and Isotopic Abundances for Dosimetry Applications, ***Olivier Bersillon****, CEA Bruyeres-le-Chatel, France.*

Selection of ^{23}*Na(n,*γ*) Dosimetry of NA23G Dosimetry Cross Sections,* ***Patrick Griffin****, Sandia National Laboratories, USA.*

Inspection of the IRDF-2002 Candidate Cross Sections, ***Patrick Griffin****, Sandia National Laboratories, USA.*

Results of Validation of the IRDF-2002 Using the ACRR Reference Neutron Field, ***Patrick Griffin****, Sandia National Laboratories, USA.*

Results of Validation of the IRDF-2002Using the SPR-III Reference Neutron Field, ***Patrick Griffin****, Sandia National Laboratories, USA.*

Damage Response Functions in the IRDF-2002 Library, ***Patrick Griffin****, Sandia National Laboratories, USA.*

Checking and Corrections made to the Files of Candidate Cross Sections, Graphic Representation, ***Kevin McLaughlin****, IAEA-NDS, Vienna, Austria.*

EVALUATED CROSS SECTION DATA FROM RUSSIAN REACTOR DOSIMETRY FILE

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Cross section data for dosimetry reactions: ^{19}F(n,2n)^{18}F, ^{24}Mg(n,p)^{24}Na, ^{27}Al(n,p)^{27}Mg, ^{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, ^{93}Nb(n,n')^{93m}Nb, ^{103}Rh(n,n')^{103m}Rh, ^{115}In(n,n')^{115m}In, ^{139}La(n,γ)^{140}La, ^{141}Pr(n,2n)^{140}Pr, ^{186}W(n,γ)^{187}W, ^{204}Pb(n,n')^{204m}Pb were taken to the International Reactor Dosimetry File IRDF-2002 from the new version of Russian Rector Dosimetry File. This version is the improved and extended version of the previes version of Russian Reactor Dosimetry File - RRDF-98 [1].

New evaluations of cross sections for the reactions ^{27}Al(n,p)^{27}Mg, ^{56}Fe(n,p)^{56}Mn, ^{58}Ni(n,p)^{58}Co, ^{103}Rh(n,n')^{103m}Rh, ^{115}In(n,n')^{115m}In, ^{139}La(n,γ)^{140}La, ^{139}La(n,γ)^{140}La, ^{204}Pb(n,n')^{204m}Pb and revisions of cross section data from RRDF-98 file 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, ^{59}Co(n,a)^{56}Mn, ^{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 2001-2003 years. Re-evaluation of RRDF-98 data for 14 reactions were done with taking into account the results of the test performed by É.M. Zsolnay, H.J. Nolthenius and E.J. Szondi [2].

Cross section data for ^{46}Ti(n,2n)^{45}Ti, ^{49}Ti(n,x)^{48}Sc ^{54}Fe(n,2n)$^{53m+g}$Fe, ^{54}Fe(n,a)^{51}Cr, ^{75}As(n,2n)^{74}As, ^{139}La(n,γ)^{140}La, ^{186}W(n,γ)^{187}W, ^{141}Pr(n,2n)^{140}Pr and ^{204}Pb(n,n')^{204m}Pb reactions were absent in the IRDF-90 ver.2 file [3]. The activation detectors on the basis of ^{139}La(n,γ)^{140}La and ^{186}W(n,γ)^{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 ^{49}Ti(n,x)^{48}Sc, ^{75}As(n,2n)^{74}As and ^{141}Pr(n,2n)^{140}Pr as the reactions ^{47}Ti(n,x)$^{46m+g}$Sc and ^{48}Ti(n,x)^{47}Sc may be useful in the high energy neutron dosimetry. In addition to the dosimetry application ^{75}As(n,2n)^{74}As and ^{141}Pr(n,2n)^{140}Pr reactions are using also in the experimental nuclear physics as the monitor reactions for the measurement of unknown cross sections in the neutron energy range 14 – 15 MeV.

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 from EXFOR Library (Version May 2003) and 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.

Additional information about 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,γ)^{140}La, ^{186}W(n,γ)^{187}W and ^{204}Pb(n,n')^{204m}Pb dosimetry reactions was obtained from theoretical model calculations.

For theoretical description of excitation functions of above mentioned 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 generalised 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].

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 [9]. Procedure of calculation recommended cross section data and related covariance matrixes of uncertainties was performed by means of PADE-2 code [10].

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 [11] and S.I.Sukhoruchkin [12]. 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 [13].

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-2 code. Cross sections uncertainties in the second block matrixes are given via diagonal matrixes. This matrixes were prepared by means of DSIGNG code.[14]

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.[15] and [16], respectively. The average cross section values for U-235 thermal fission neutron spectrum and Cf-252 spontaneous fission neutron spectrum calculated from the IRDF-2002 (IPPE) and IRDF-90 Ver.2 evaluated excitation functions are given in Tables 1 and 2 in comparison with relevant experimental data. Integral experimental data [17-29] were corrected to the new recommended cross sections for monitor reactions from ref. [30] and [31].

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 [11] and [32]. 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 [33]. The data for neutron spectrum in the center of CFRM facility were taken from ref. [34].

The detail description of the cross section evaluation and revision for the reactions adopted for IRDF-2002 from new version of Russian Rector Dosimetry File is given in refs. [35-37].

Table 1. Measured and calculated averaged cross sections in ^{252}Cf spontaneous fission neutron spectrum

Reaction	Updated RRDF-98 <σ>, mb	IRDF-90 <σ>, mb	Experiment <σ>, mb
^{19}F(n,2n)^{18}F	0.01615	0.01703	0.01612 ± 0.00054 [31]
^{24}Mg(n,p)^{24}Na	2.1398	2.1564	1.996 ± 0.049 [31]
^{27}Al(n,p)^{27}Mg	4.9070		4.880 ± 0.105 [31]
^{46}Ti(n,2n)^{45}Ti	0.01198		0.093 ± 0.031 [23]
^{46}Ti(n,p)$^{46m+g}$Sc	13.818	12.313	14.07 ± 0.25 [31]
^{47}Ti(n,x)$^{46m+g}$Sc	0.019201		
^{48}Ti(n,p)^{48}Sc	0.42629	0.3864	0.4247 ± 0.0080 [31]
^{48}Ti(n,x)^{47}Sc	0.0042891		
^{49}Ti(n,x)^{48}Sc	0.0026070		
^{51}V(n,α)^{48}Sc	0.038514	0.03872	0.03900 ± 0.00086 [31]
^{54}Fe(n,2n)$^{53m+g}$Fe	0.0036219		
^{54}Fe(n,α)^{51}Cr	1.1114		
^{56}Fe(n,p)^{56}Mn	1.4730	1.368	1.465 ± 0.026 [31]
^{59}Co(n,α)^{56}Mn	0.22095	0.2159	0.2218 ± 0.0042 [31]
			0.2208 ± 0.0014 [24]
^{58}Ni(n,p)$^{58m+g}$Co	117.36	115.2	117.5 ± 1.5 [31]
^{63}Cu(n,α)$^{60m+g}$Co	0.6925	0.6778	0.6887 ± 0.0135 [31]
^{75}As(n,2n)^{74}As	0.61804		
^{93}Nb(n,2n)^{92m}Nb	0.7701	0.7773	0.749 ± 0.038 [31]
^{93}Nb(n,n')^{93m}Nb	146.02	142.55	147.5 ± 2.5 *
^{103}Rh(n,n')^{103m}Rh	724.83	714.1	620.8 ± 67.2 [20]
			813.2 ± 24.2 [27]
^{115}In(n,n')^{115m}In	191.66	189.7	197.4 ± 2.7 [31]
^{141}Pr(n,2n)^{140}Pr	1.9843		
^{139}La(n,g)^{140}La	6.650		
^{186}W(n,g)^{187}W	31.699		
^{204}Pb(n,n') ^{204m}Pb	20.373		20.900 ± 1.202 [21]
			20.850 ± 0.920 [25]

- - evaluated by author

Table 2. Measured and calculated averaged cross sections in ^{235}U thermal fission neutron spectrum

Reaction	Updated RRDF-98 <σ>, mb	IRDF-90 <σ>, mb	Experiment <σ>, mb
^{19}F(n,2n)^{18}F	0.007299	0.00772	0.007200 ± 0.00100 [18]
			0.008624 ± 0.00046 [31]
^{24}Mg(n,p)^{24}Na	1.5396	1.5517	1.455 ± 0.023 [30]
			1.451 ± 0.023 [31]
^{27}Al(n,p)^{27}Mg	4.0768		4.133 ± 0.074 [28]
			3.914 ± 0.070 [30]
			3.902 ± 0.069 [31]
^{46}Ti(n,2n)^{45}Ti	0.004469		
^{46}Ti(n,p)$^{46m+g}$Sc	11.447	10.252	11.51 ± 0.20 [31]
^{47}Ti(n,x)$^{46m+g}$Sc	0.008116		
^{48}Ti(n,p)^{48}Sc	0.3043	0.2749	0.305 ± 0.020 [28]
			0.2996 ± 0.0054 [31]
^{48}Ti(n,x)^{47}Sc	0.001656		
^{49}Ti(n,x)^{48}Sc	0.001004		
^{51}V(n,α)^{48}Sc	0.02441	0.0246	0.02429 ±0. 00056 [31]
^{54}Fe(n,2n)$^{53m+g}$Fe	0.001284		
^{54}Fe(n,α)^{51}Cr	0.8459		0.850 ± 0.050 *
^{56}Fe(n,p)^{56}Mn	1.1085	1.0297	1.130 ± 0.070 [28]
			1.083 ± 0.017 [30]
			1.079 ± 0.017 [31]
^{59}Co(n,α)^{56}Mn	0.1582	0.1549	0.1563 ± 0.0035 [31]
^{58}Ni(n,p)^{58}Co	107.44	105.73	108.2 ± 1.4 [31]
^{63}Cu(n,α)$^{60m+g}$Co	0.5329	0.5214	0.5295 ± 0.0255 [29]
			0.4918 ± 0.0242 [31]
^{75}As(n,2n)^{74}As	0.3092		0.309 ± 0.019 *
^{93}Nb(n,2n)^{92m}Nb	0.4416	0.4459	0.4576 ± 0.0226 *
			0.4645 ±0.0117 [31]
^{93}Nb(n,n')^{93m}Nb	143.46	139.97	147.6 ± 7.0 *

Table 2. Measured and calculated averaged cross sections in ^{235}U thermal fission neutron spectrum (continued)

Reaction	Updated RRDF-98 <σ>, mb	IRDF-90 <σ>, mb	Experiment <σ>, mb
$^{103}Rh(n,n')^{103m}Rh$	715.85	706.03	702.2 ± 28.1 [26]
			721.2 ± 38.7 [28]
$^{115}In(n,n')^{115m}In$	188.40	186.35	188.2 ± 2.3 [30]
			187.8 ± 2.3 [31]
$^{141}Pr(n,2n)^{140}Pr$	1.0922		
$^{139}La(n,g)^{140}La$	6.737		5.30 [17]
$^{186}W(n,g)^{187}W$	32.267		
$^{204}Pb(n,n')^{204m}Pb$	17.770		18.900 ± 2.000 [19]
			19.080 ± 1.524 [22]

* - evaluated by author

References :

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Evaluation of Cross Sections for IRDF-2002 at 14 MeV

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Summary

The selection of evaluated neutron activation cross sections for IRDF-2002 depends, in part, on how well the various evaluations fit experimental data for thermal cross sections, resonance integrals, and near 14 MeV. The candidate evaluated neutron cross section libraries included IRDF-90, JENDL/D-99, RRDF-98, ENDF/B-VI, and JEFF-3.0. In order to assess differences in these evaluations, the evaluated cross sections were plotted along with the available experimental data in the 14 MeV region. These plots are shown for all of the reactions considered for IRDF-2002 in the appendix. These comparisons were then used to determine whether there were any significant differences between the different evaluations such that a selection could be made for IRDF-2002. In most cases, the differences between the different evaluations were negligible such that no clear preference could be made based solely on the fit to the experimental data near 14 MeV. Detailed comments are provided for each of the reactions that were considered. It is important to note that this cursory evaluation of experimental data and cross section evaluations had the very limited objective of aiding in the selection of cross sections for use in IRDF-2002. More detailed discussions of the data and cross sections by the evaluators are readily available in the report section for each reaction in the cross section libraries.

Plots of Experimental Data and Evaluated Cross Sections

Most neutron activation reactions have been extensively studied near 14 MeV due to the widespread availability of d+t 14 MeV neutron sources as well as other accelerator based neutron sources. However, it must first be recognized that such data has been measured at a significant range of energies around 14 MeV due to the characteristics of the various accelerators that have been used. Although the interaction of deuterium and tritium produces a neutron near 14 MeV at low energies, many “14 MeV” neutron sources accelerate the deuteron to some energy and thicken the target containing the tritium in order to substantially increase the neutron yield. Furthermore, the neutron energy distribution depends on the angle between the incoming deuteron beam and the location of the measurement. Consequently, these effects lead to both a range of neutron energies as well as a distinct spread in the neutron energies. Looking at the available experimental neutron data, it was thus decided to plot experimental data in the range of 13.5 to 15.0 MeV. The experimental data were taken from EXFOR, which is available on the IAEA website as the Nuclear Reaction Database Retrieval System by V. Zerkin (2003). The cross section evaluations were taken from 640 group representations processed by the IAEA Nuclear Data Section.

Initially it was planned to plot all available experimental data in the 13.5 to 15 MeV region. However, due to the scatter in the data, it was deemed more appropriate to add

some selection criteria to limit the data that was plotted. The following criteria were used to select the data for the plots:

-In most cases, experimental data were taken directly from EXFOR. However, it is important to note that the cross section evaluators examined all the data in more detail and renormalized data based on changes in monitor reactions. There was insufficient time to perform this task for all of the reactions in IRDF-2002. However, K. Zolotarev kindly provided evaluated and renormalized data for a number of reactions as indicated in the discussion below for each of the reactions. Comparison of the raw data and the renormalized data shows a significant reduction in the scatter of the data in most cases, as would be expected.

-Data that are clearly discrepant from the preponderance of data were omitted for clarity in the plots. Only data that differed from most of the other data by significantly more than the stated data uncertainties were omitted. In cases where only a few data measurements were available, no data were omitted.

-Data with very large energy uncertainties or energy resolution were generally omitted. Such data can be difficult to interpret, especially for reactions where the cross section is rapidly changing in the 14 MeV region since such data are more truly integral rather than differential in nature.

-Data prior to 1970 tended to be omitted since more recent data tended to supersede these older measurements or were found to be of much higher quality than newer data. Again, no data were omitted in cases where only a few measurements were available.

Although such data omission is generally not the best practice, it is clear from the agreement between the data and the cross section evaluations that the evaluators appeared to have made similar data selections. In any case, it should be pointed out that all available data can readily be plotted using the EXFOR software, anytime a researcher wishes to see it. Most of the plots are presented on expanded, linear scales with suppressed zeros in order to show the relatively small differences between the various cross section evaluations.

The list of reactions and cross section evaluations is given in Table I. The X symbol means that plots were prepared and/or that cross sections were evaluated from the various data libraries. The N symbol means that no experimental data were available at 14 MeV and consequently plots were not prepared. The D symbol was used to indicate that some of the cross section files were duplicates of the cross sections found in IRDF-90.

Detailed Comments on the Candidate Cross Section Libraries near 14 MeV

Detailed comments are provided for each of the plots shown in the appendix. In the following comments, IRDF means IRDF-90, JENDL means JENDL/D-99, RRDF refers to either RRDF-98 or new evaluations, ENDF refers to ENDF/B-VI, and JEFF refers to JEFF-3.0.

^{6}Li(n,α)^{3}H and ^{10}B(n,α)^{7}Li – No experimental data were available in EXFOR so no plots were prepared.

^{19}F(n,2n)^{18}F - The experimental data was evaluated and renormalized by K. Zolotarev. Evaluated cross section files were available in JENDL and RRDF. The RRDF-98 file clearly gives the best fit to the data with the lowest uncertainties.

^{24}Mg(n,p)^{24}Na - The IRDF and JENDL files are virtually the same and both agree well with the data.

^{27}Al(n,p)^{27}Mg – The experimental data was evaluated and renormalized by K. Zolotarev. The IRDF, JENDL, and new RRDF evaluations are very similar and appear to be slightly lower than most of the experimental data.

^{27}Al(n,α)^{24}Na – The JENDL file is a duplicate of the file in IRDF, which does an excellent job of fitting the available data.

^{31}P(n,p)^{31}Si – The IRDF and JENDL files are nearly identical and fit the data equally well, although there is one data point that appears to be discrepant and should probably be rejected.

^{32}S(n,p)^{32}P – IRDF was the only file available and the cross section appears to fit the data quite well, neglecting one apparently discrepant data point.

^{45}Sc(n,g)^{46}Sc – IRDF was the only file available. Neglecting the data point with very high uncertainties, the evaluation appears to fit the data quite well.

^{46}Ti(n,2n)^{45}Ti- The experimental data were evaluated and renormalized by K. Zolotarev. The JENDL and RRDF evaluations are very similar. Both fit the data reasonably well, although the RRDF file appears to give the best fit.

^{46}Ti(n,p)^{46}Sc – Evaluations were available in IRDF, JENDL, and RRDF. All of the evaluations appear to be lower than the average of the data. JENDL does the best job of fitting all the data, although IRDF and RRDF fit some of the data with the lowest uncertainties. The comments of the evaluators should be studied to help with this selection.

Ti(n,x)^{46}Sc – JENDL is the only file available. The evaluated cross section appears to be slightly higher than the available data from natural titanium.

^{47}Ti(n,np+pn+d)^{46}Sc – The experimental data were evaluated and renormalized by K. Zolotarev. The limited data include measurements of (n,np+d). The IRDF and RRDF files are distinctly different. RRDF clearly gives the best fit to the data.

^{47}Ti(n,p)^{47}Sc – The experimental data were evaluated and renormalized by K. Zolotarev. IRDF is the only available cross section file. The evaluation appears to be somewhat lower than the available data, although the data have considerable scatter.

^{48}Ti(n,np+pn+d)^{47}Sc - The experimental data were evaluated and renormalized by K. Zolotarev. The limited data include measurements of (n,np+d). Evaluations are available from IRDF, RRDF, and JENDL. All of the data sets do an equally good job of fitting the data.

Ti(n,x)^{48}Sc – Only one data point was available from natural titanium and the only evaluated file is from JENDL. The evaluation appears to be higher than the sole data point.

^{48}Ti(n,p)^{48}Sc – The experimental data were evaluated and renormalized by K. Zolotarev. The available files from IRDF and RRDF are quite similar and both fit the average of the available data, which have a considerable scatter.

^{49}Ti(n,np+pn+d)^{48}Ti – The experimental data were evaluated and renormalized by K. Zolotarev. The JENDL and RRDF files are similar; however, the RRDF file appears to do a better job of fitting the available data.

^{51}V(n,α)^{48}Sc – The experimental data were evaluated and renormalized by K. Zolotarev. RRDF is the only available file and it closely matches the available data.

^{55}Cr(n,2n)^{51}Cr – The IRDF, JENDL, and ENDF files are very nearly identical. All of the files appear to be slightly higher than the average of the available data, although the files appear to be a good fit to the data with the lowest uncertainties.

^{55}Mn(n,γ)^{56}Mn – The JENDL file is essentially identical to IRDF and both fit the data with the lowest uncertainties quite well.

^{54}Fe(n,2n)^{53}Fe - The experimental data were evaluated and renormalized by K. Zolotarev. The only available file in RRDF is a good fit to the available data, although there is some scatter in the data.

^{54}Fe(n,α)^{51}Cr – The experimental data were evaluated and renormalized by K. Zolotarev. The only available cross section file from RRDF is a good fit to the available data.

^{54}Fe(n,p)^{54}Mn - The experimental data were evaluated and renormalized by K. Zolotarev. Evaluated files are available from IRDF and JENDL. The IRDF file gives a better fit to the entire energy range, although JENDL may be closer to the average of the data near 14.7 MeV.

^{56}Fe(n,p)^{56}Mn – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF and JEFF files are nearly identical and both appear to fit the data quite well.

^{58}Fe(n,γ)^{59}Fe – No experimental data is available near 14 MeV.

^{59}Co(n,γ)^{60}Co – The IRDF file is the only one available and it appears to fit the data, ignoring the one high data point.

^{59}Co(n,2n)^{58}Co – IRDF is the only file and it appears to fit the average of the data, which have a significant scatter.

^{59}Co(n,α)^{56}Mn – The experimental data were evaluated and renormalized by K. Zolotarev. RRDF is the only available cross section file and it fits the data reasonably well.

^{58}Ni(n,2n)^{57}Ni - The IRDF, JENDL, and JEFF files are in good agreement and all fit the data reasonably well.

^{58}Ni(n,p)^{58}Co – The experimental data were evaluated and renormalized by K. Zolotarev. Evaluated cross sections are available from IRDF, JENDL, JEFF, and a new evaluation in RRDF. The file in ENDF is the same as IRDF. The evaluations differ by about 10%; however, it is difficult to select one or the other since the differences are generally less than the scatter in the experimental data. The JEFF evaluation appears to be too high, especially at the lower energies.

^{60}Ni(n,p)^{60}Co – The experimental data were evaluated and renormalized by K. Zolotarev. The evaluated files in ENDF and JEFF are nearly identical and both appear to fit the data with the lowest uncertainties, although there is considerable scatter in the data.

^{63}Cu(n,γ)^{64}Cu – The single experimental data point is well fit by the IRDF evaluation, which is the only one available.

^{63}Cu(n,2n)^{62}Cu – ENDF and JENDL files are available. The ENDF file appears to be a better fit to the data with the lowest uncertainties.

^{63}Cu(n,α)^{60}Co – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF file is about 5% higher than the IRDF file and clearly gives a better fit to the data with the lowest uncertainties.

^{65}Cu(n,2n)^{64}Cu – The IRDF and JENDL files are very nearly the same and both appear to fit the data equally well.

^{64}Zn(n,p)^{64}Cu – IRDF is the only file available. There is considerable scatter in the available data, although the evaluation is reasonably close to the average of the data with the lowest uncertainties.

^{75}As(n,2n)^{74}As – The experimental data were evaluated and renormalized by K. Zolotarev. The new RRDF evaluation appears to fit the data reasonably well, although the data has considerable scatter.

^{89}Y(n,2n)^{88}Y – JENDL is the only available file and it appears to be a very good fit to the available data.

^{90}Zr(n,2n)^{89}Zr – The IRDF file is slightly higher than the JENDL file. JENDL appears to give a slightly improved fit to the data.

^{93}Nb(n,2n)^{92m}Nb – The experimental data were evaluated and renormalized by K. Zolotarev. The IRDF and RRDF files are essentially identical and both fit the data quite well.

^{93}Nb(n,n′)^{93m}Nb – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF file is slightly higher than the IRDF file. Unfortunately, there is only one credible data point near 14 MeV and the RRDF file gives the closest fit.

^{103}Rh(n,n′)^{103m}Rh – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF file is nearly the same as the IRDF file, although it is a bit higher above 14.5 MeV. Both files are slightly lower than the experimental data.

^{109}Ag(n,γ)^{110m}Ag – No experimental data is available near 14 MeV.

^{115}In(n,2n)^{114m}In – IRDF is the only available file and it seems to fit the data with the lowest uncertainties.

^{115}In(n,n′)^{115m}In - The experimental data were evaluated and renormalized by K. Zolotarev. The JENDL file is the same as IRDF. The RRDF file appears to give the best fit to the experimental data.

^{127}I(n,2n)^{126}I - The JENDL and IRDF files are essentially identical and both give a reasonable fit to the data, which have relatively large uncertainties.

^{139}La(n,γ)^{140}La – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF evaluation gives a reasonable fit to the data with the lowest uncertainties, although the data have considerable scatter near 14 MeV.

^{141}Pr(n,2n)^{140}Pr – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF evaluation gives a reasonable fit to the data, which have relatively large scatter and uncertainties.

^{169}Tm(n,2n)^{168}Tm – The JENDL file gives a good fit to the available data.

^{186}W(n,γ)^{187}W – The experimental data were evaluated and renormalized by K. Zolotarev. The RRDF file fits the data, ignoring the one high data point.

^{197}Au(n,γ)^{198}Au – The sparse experimental data have considerable scatter near 14 MeV and the IRDF evaluation gives a reasonable fit.

^{197}Au(n,2n)^{196}Au – The IRDF file is slightly higher than the JENDL file, although both give reasonably good fits to the available data.

^{199}Hg(n,n′)^{119m}Hg – JENDL gives a good fit to the sole data point for this reaction.

^{204}Pb(n,n′)^{204m}Pb - The RRDF file appears to be somewhat lower than the available data would suggest, although there is some scatter in the data.

^{232}Th(n,f) – IRDF is the only available file and it gives a reasonable fit to the available data.

^{237}Np(n,f) – The experimental data were evaluated and renormalized by K. Zolotarev. JENDL is the only available file and it seems to give a good fit to the available data with the lowest uncertainties.

^{235}U(n,f) – The IRDF file appears to give a good fit to the data with the lowest uncertainties.

^{238}U(n,γ)^{239}U(β$^-$)^{239}Np – IRDF provides the only evaluated data file and it fits the data quite well, ignoring one high data point near 14.5 MeV.

^{238}U(n,f) – The JENDL and IRDF files are nearly identical and both appear to be slightly lower than would be suggested by the available data.

^{239}Pu(n,f) – The JENDL file appears to give a better fit to the data than the IRDF file.

^{241}Am(n,f) – JENDL is the only available file and it seems to give a good fit to the average of the available data.

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K. I. Zolotarev of the Institute of Physics and Power Engineering, Obninsk, Russia, kindly made available his evaluations and renormalizations of the experimental data for many of the reactions, as noted in the text. Such evaluation provides the best basis for the evaluation of the various cross section files and is normally taken into account during the cross section evaluation process.

References

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[3] Russian Reactor Dosimetry File 98 (RRDF-98)

[4] Joint European Fusion File 3.0 (JEFF-3.0)

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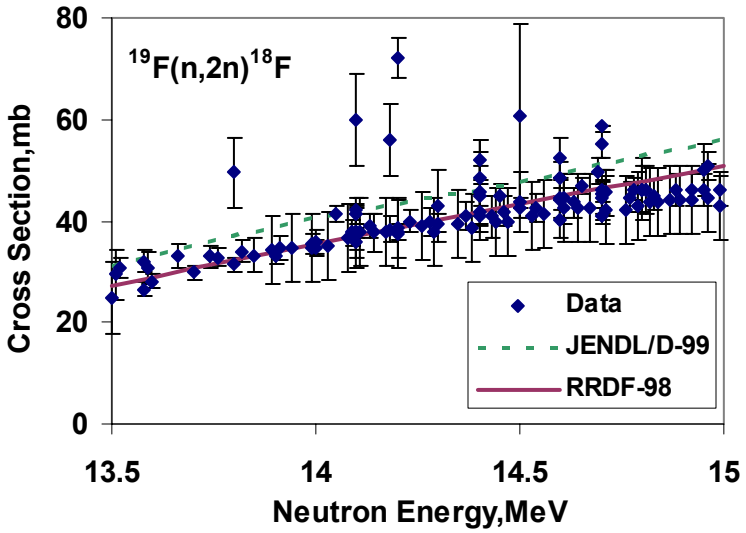
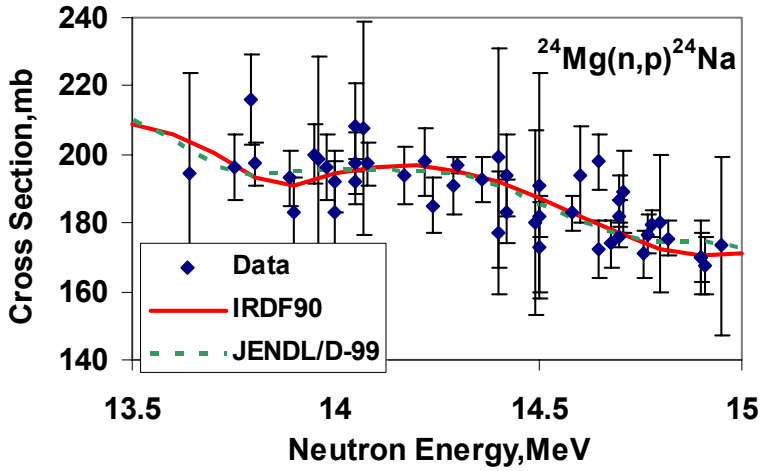
Cross Section Evaluations at 14 MeV

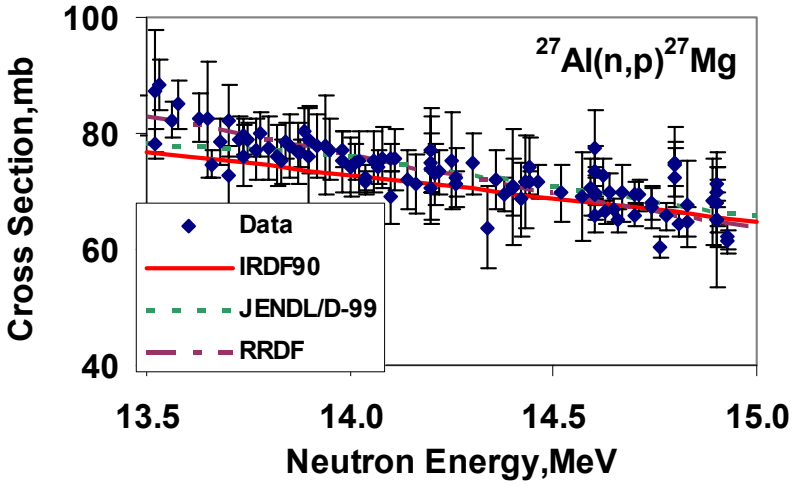
Reaction	Plot	IRDF-90	JENDL/D-99	RRDF-98 or New	ENDF/B-VI	JEFF-3.0
$^{6}Li(n,\alpha)^{3}H$	N	X				
$^{10}B(n,\alpha)^{7}Li$	N	X				
$^{19}F(n,2n)^{18}F$	X		X	X		
$^{23}Na(n,\gamma)^{24}Na^{*}$					X	
$^{23}Na(n,2n)^{22}Na^{*}$			X			
$^{24}Mg(n,p)^{24}Na$	X	X	X	Q		
$^{27}Al(n,p)^{27}Mg$	X	X	X	X		
$^{27}Al(n,\alpha)^{24}Na$	X	X	D			
$^{31}P(n,p)^{31}Si$	X	X	X			
$^{32}S(n,p)^{32}P$	X	X				
$^{45}Sc(n,g)^{46}Sc$	X	X				
$^{46}Ti(n,2n)^{45}Ti$	X		X	X		
$^{46}Ti(n,p)^{46}Sc$	X	X	X	X		
$Ti(n,x)^{46}Sc$	X		X			
$^{47}Ti(n,np+pn+d)^{46}Sc$	X	X		X		
$^{47}Ti(n,p)^{47}Sc$	X	X				
$^{48}Ti(n,np+pn+d)^{47}Sc$	X	X	X	X		
$Ti(n,x)^{48}Sc$	X		X			
$^{48}Ti(n,p)^{48}Sc$	X	X		X		
$^{49}Ti(n,np+pn+d)^{48}Ti$	X		X	X		
$^{51}V(n,\alpha)^{48}Sc$	X			X		
$^{55}Cr(n,2n)^{51}Cr$	X	X	X		X	
$^{55}Mn(n,\gamma)^{56}Mn$	X	X	D			
$^{54}Fe(n,2n)^{53}Fe$	X			X		
$^{54}Fe(n,\alpha)^{51}Cr$	X			X		
$^{54}Fe(n,p)^{54}Mn$	X	X	X			
$^{56}Fe(n,p)^{56}Mn$	X			X		X
$^{58}Fe(n,\gamma)^{59}Fe$	N	X	X			
$^{59}Co(n,2n)^{58}Co$	X	X				
$^{59}Co(n,\alpha)^{56}Mn$	X			X		
$^{59}Co(n,\gamma)^{60}Co$	X	X				
$^{58}Ni(n,2n)^{57}Ni$	X	X	X			X
$^{58}Ni(n,p)^{58}Co$	X	X	X	X	D	X
$^{60}Ni(n,p)^{60}Co$	X				X	X
$^{63}Cu(n,2n)^{62}Cu$	X		X		X	
$^{63}Cu(n,\gamma)^{64}Cu$	X	X			D	
$^{63}Cu(n,\alpha)^{60}Co$	X	X	X	X		
$^{65}Cu(n,2n)^{64}Cu$	X	X			D	
$^{64}Zn(n,p)^{64}Cu$	X	X				

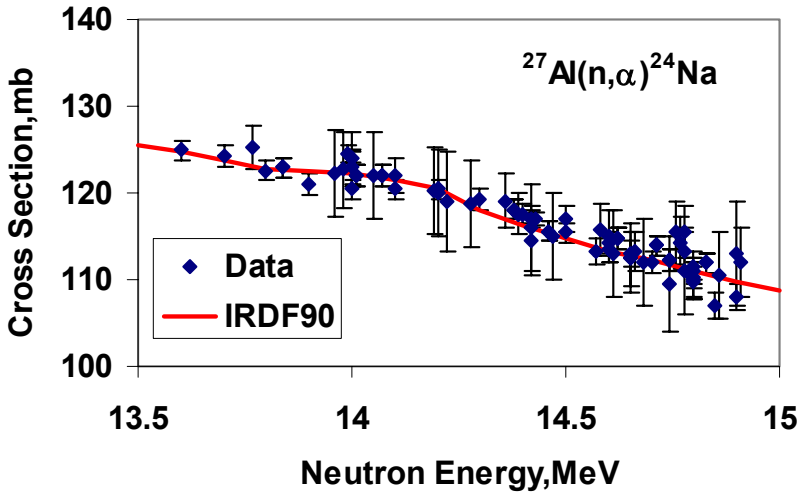
Reaction	Plot	IRDF-90	JENDL/D-99	RRDF-98 or New	ENDF/B-VI	JEFF-3.0
^{75}As(n,2n)^{74}As	X			X		
^{89}Y(n,2n)^{88}Y	X		X			
^{90}Zr(n,2n)^{89}Zr	X	X	X			
^{93}Nb(n,γ)^{94}Nb*		X				
^{93}Nb(n,2n)^{92m}Nb	X	X		X		
^{93}Nb(n,n′)^{93m}Nb	X	X		X		
^{103}Rh(n,n′)^{103m}Rh	X	X		X		
^{109}Ag(n,γ)^{110m}Ag	N	X				
^{115}In(n,2n)^{114m}In	X	X				
^{115}In(n,γ)^{116m}In*		X				
^{115}In(n,n′)^{115m}In	X	X	D	X		
^{127}I(n,2n)^{126}I	X	X	X			
^{139}La(n,γ)^{140}La	X			X		
^{141}Pr(n,2n)^{140}Pr	X			X		
^{169}Tm(n,2n)^{168}Tm	X		X			
^{181}Ta(n,γ)^{182}Ta*			X			
^{186}W(n,γ)^{187}W	X			X		
^{197}Au(n,2n)^{196}Au	X	X	X			
^{197}Au(n,γ)^{198}Au	X	X				
^{199}Hg(n,n′)^{119m}Hg	X		X			
^{204}Pb(n,n′)^{204m}Pb	X			X		
^{232}Th(n,γ)^{233}Th*		X				
^{232}Th(n,f)	X	X				
^{235}U(n,f)	X	X				
^{238}U(n,f)	X	X	X			
^{238}U(n,γ)^{239}U	X	X				
^{237}Np(n,f)	X		X	Q		
^{239}Pu(n,f)	X	X	X			
^{241}Am(n,f)	X		X			

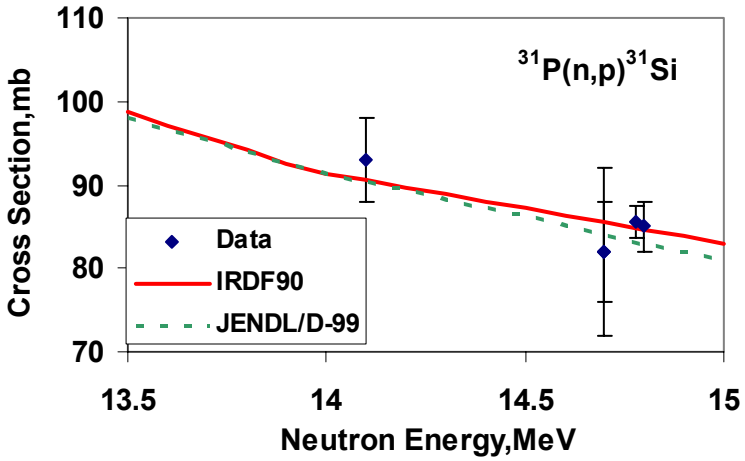
D – These files are duplicates of IRDF-90 files.
N – No cross section data were available; plots were not prepared.
Q – A new evaluation is nearly complete, but not yet available for consideration.
* - These files did not meet the requirements of the covariance matrices, but were included due to their importance for reactor dosimetry.

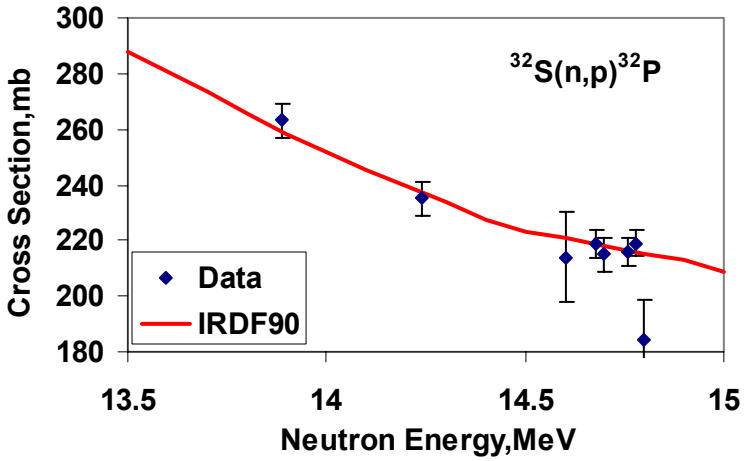

$^{19}F(n,2n)^{18}F$
Cross Section,mb
Neutron Energy,MeV
Data
JENDL/D-99
RRDF-98
0
20
40
60
80
13.5
14
14.5
15

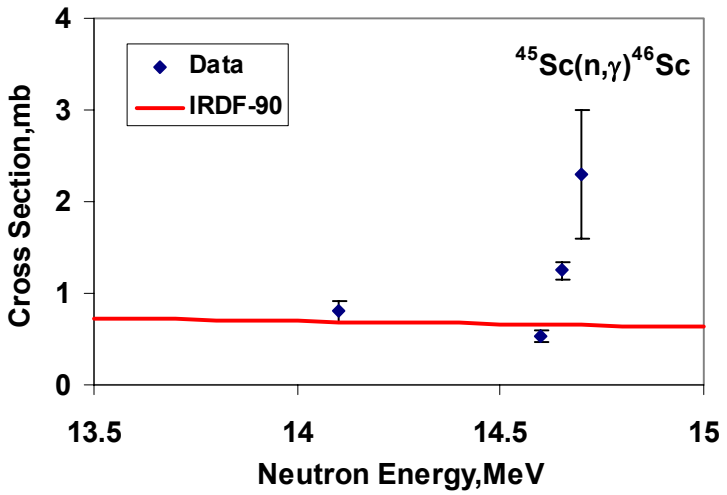

$^{24}Mg(n,p)^{24}Na$
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
JENDL/D-99
140
160
180
200
220
240
13.5
14
14.5
15

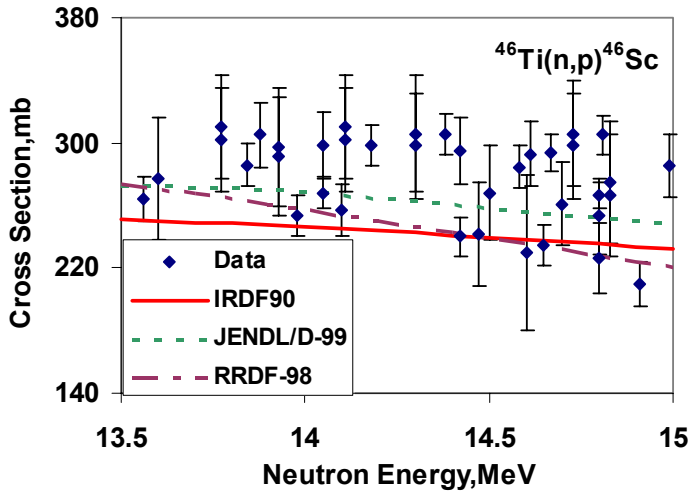

$^{27}Al(n,p)^{27}Mg$
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
JENDL/D-99
RRDF
40
60
80
100
13.5
14.0
14.5
15.0

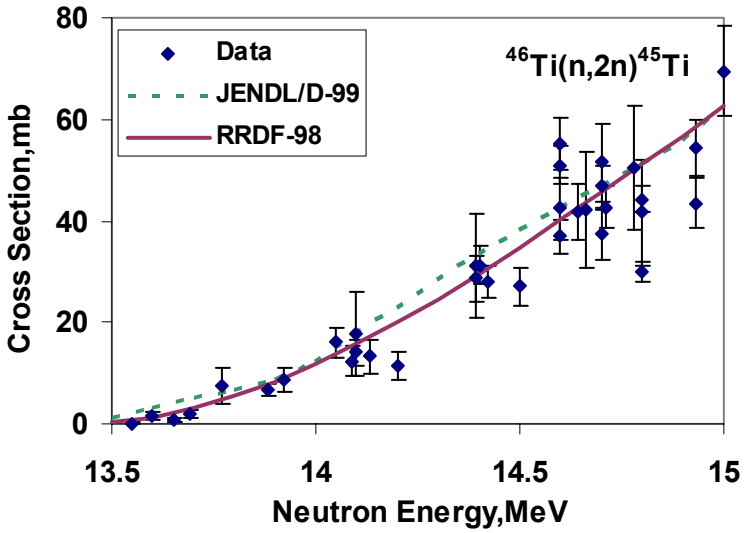

^{27}Al(n,α)^{24}Na
Data
IRDF90
Cross Section,mb
Neutron Energy,MeV
140
130
120
110
100
13.5
14
14.5
15

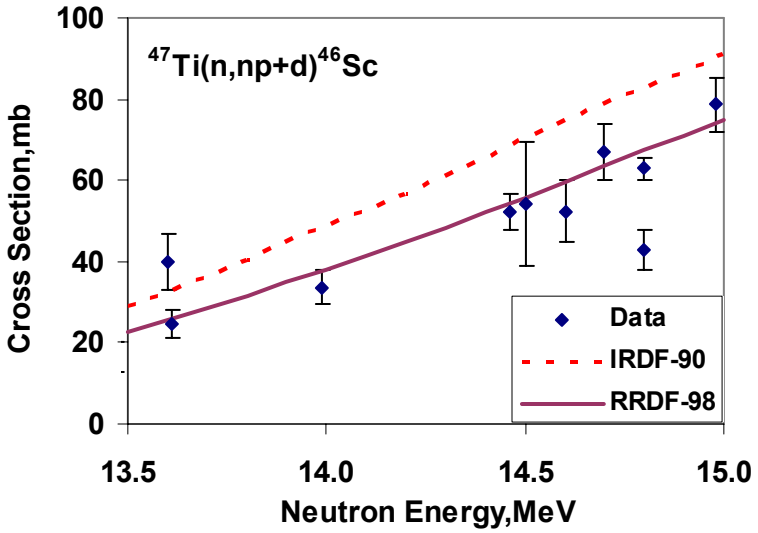

^{31}P(n,p)^{31}Si
Data
IRDF90
JENDL/D-99
Cross Section,mb
Neutron Energy,MeV
110
100
90
80
70
13.5
14
14.5
15

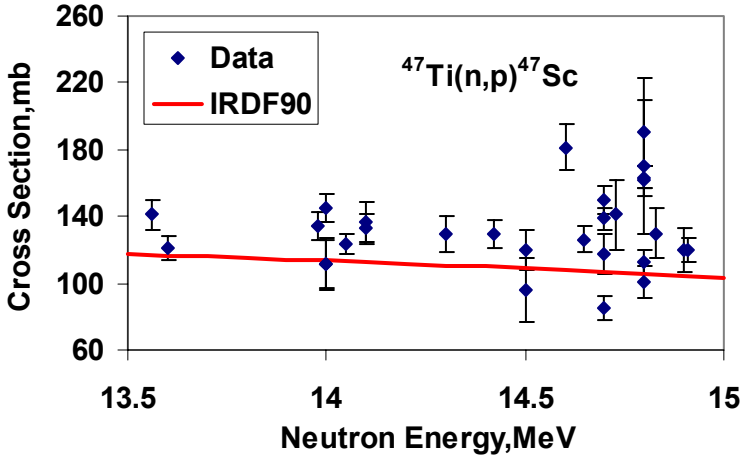

^{32}S(n,p)^{32}P
Data
IRDF90
Cross Section,mb
Neutron Energy,MeV
300
280
260
240
220
200
180
13.5
14
14.5
15

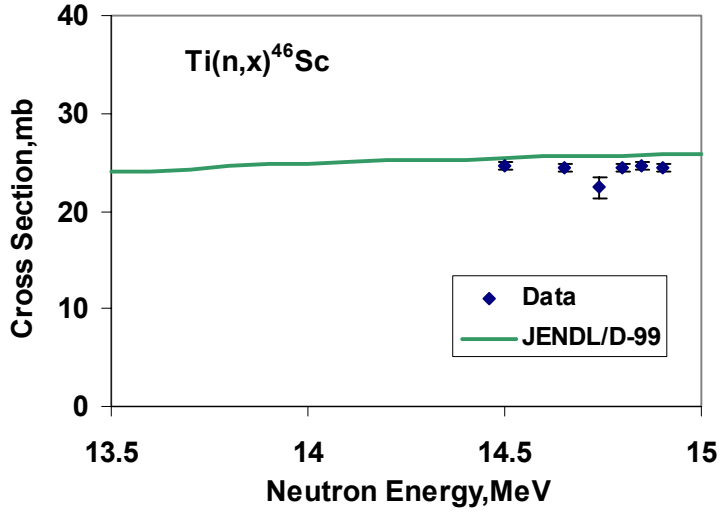

Data
IRDF-90
$^{45}Sc(n,\gamma)^{46}Sc$
Cross Section,mb
Neutron Energy,MeV
0
1
2
3
4
13.5
14
14.5
15

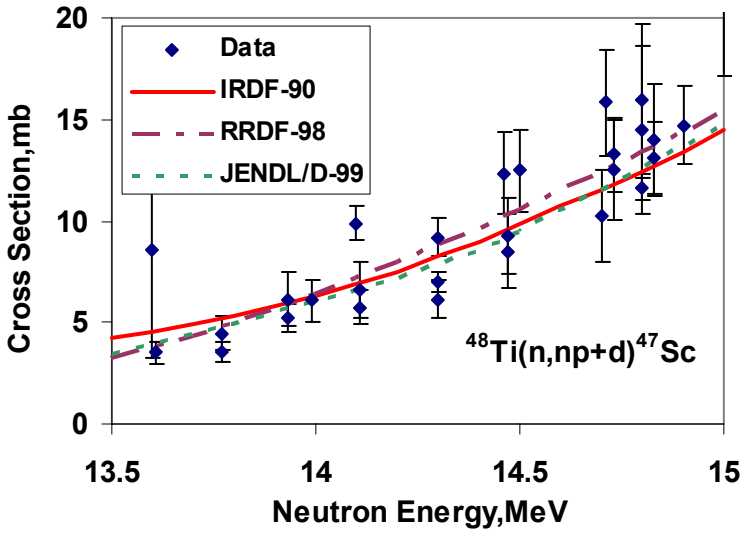

$^{46}Ti(n,p)^{46}Sc$
Data
IRDF90
JENDL/D-99
RRDF-98
Cross Section,mb
Neutron Energy,MeV
140
220
300
380
13.5
14
14.5
15

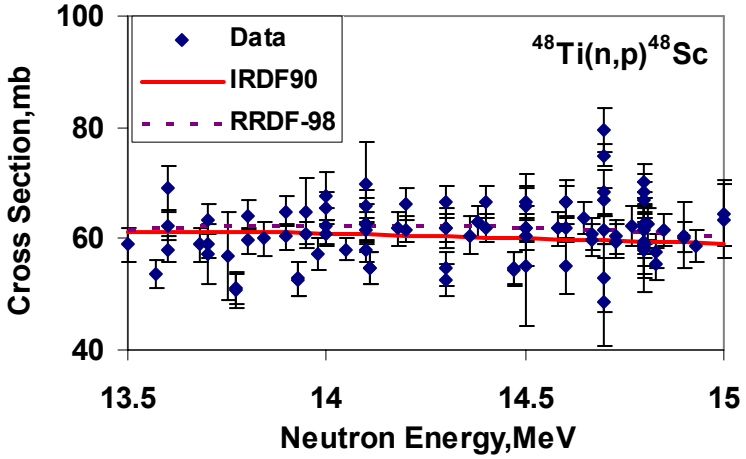

Data
JENDL/D-99
RRDF-98
$^{46}Ti(n,2n)^{45}Ti$
Cross Section,mb
Neutron Energy,MeV
0
20
40
60
80
13.5
14
14.5
15

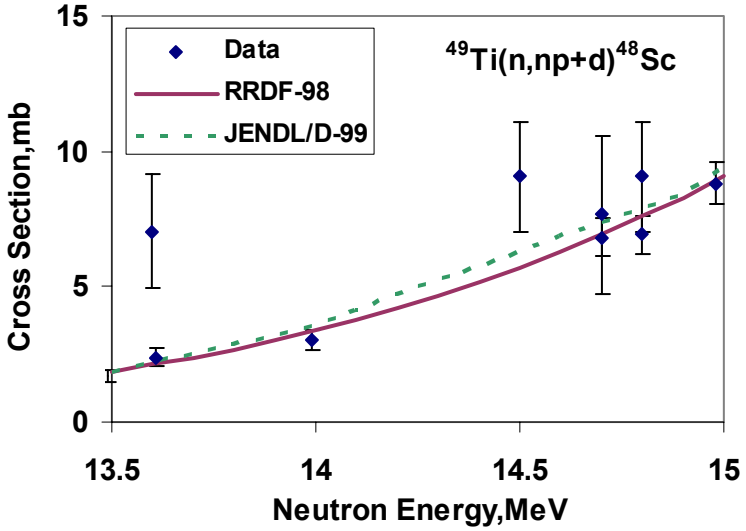

^{47}Ti(n,np+d)^{46}Sc
Cross Section,mb
Neutron Energy,MeV
Data
IRDF-90
RRDF-98
100
80
60
40
20
0
13.5
14.0
14.5
15.0

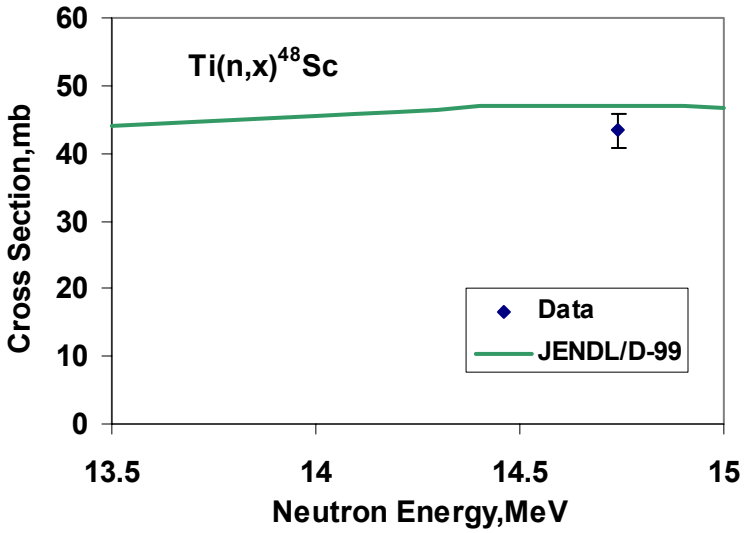

^{47}Ti(n,p)^{47}Sc
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
260
220
180
140
100
60
13.5
14
14.5
15

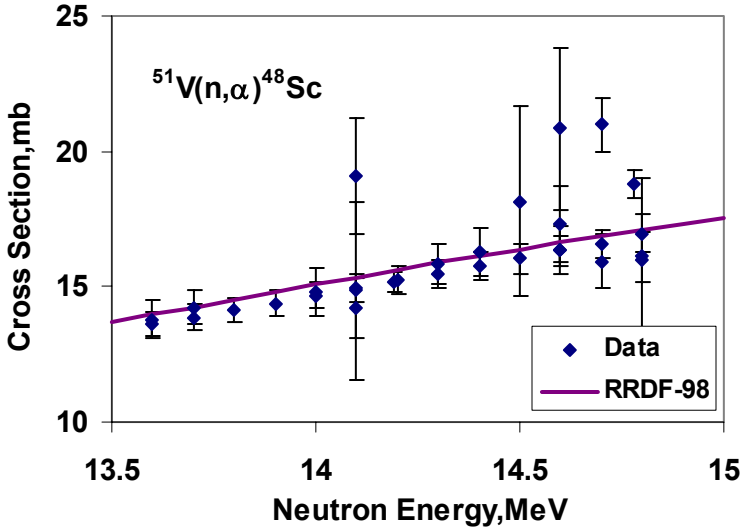

Ti(n,x)^{46}Sc
Cross Section,mb
Neutron Energy,MeV
Data
JENDL/D-99
40
30
20
10
0
13.5
14
14.5
15

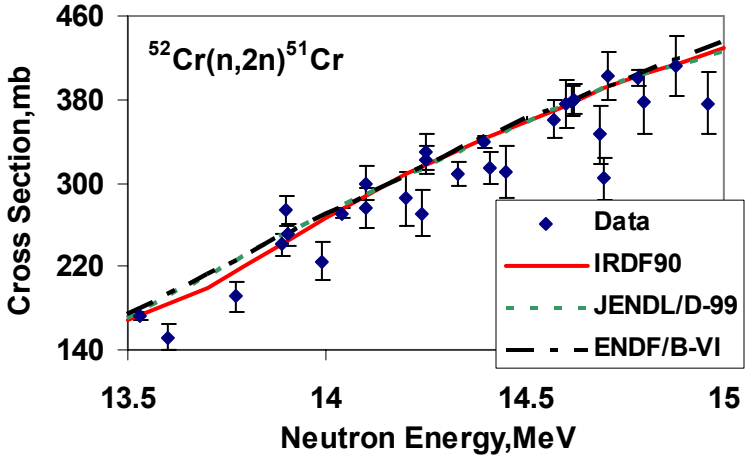

Data
IRDF-90
RRDF-98
JENDL/D-99
^{48}Ti(n,np+d)^{47}Sc
Cross Section,mb
Neutron Energy,MeV
0
5
10
15
20
13.5
14
14.5
15

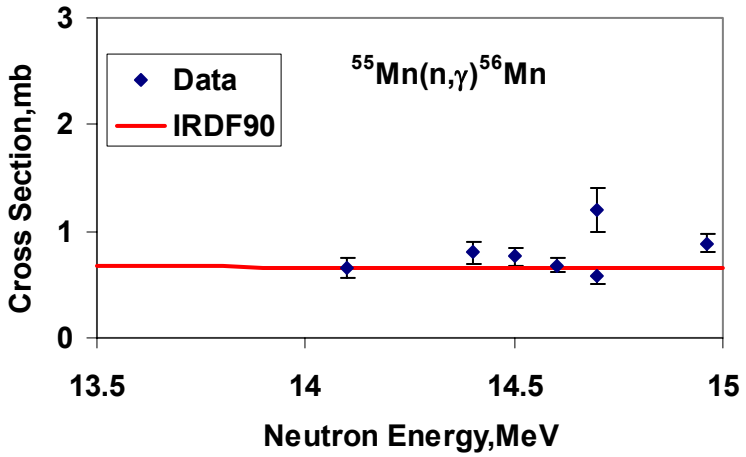

Data
IRDF90
RRDF-98
^{48}Ti(n,p)^{48}Sc
Cross Section,mb
Neutron Energy,MeV
40
60
80
100
13.5
14
14.5
15

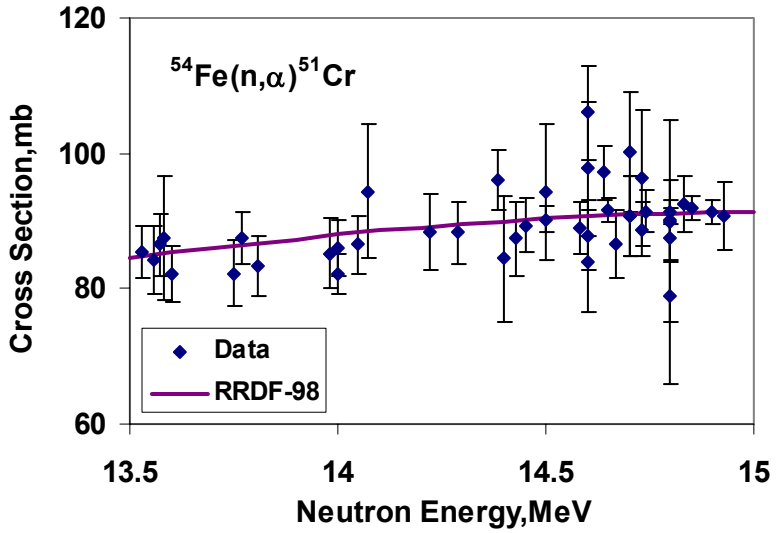

Data
RRDF-98
JENDL/D-99
^{49}Ti(n,np+d)^{48}Sc
Cross Section,mb
Neutron Energy,MeV
0
5
10
15
13.5
14
14.5
15

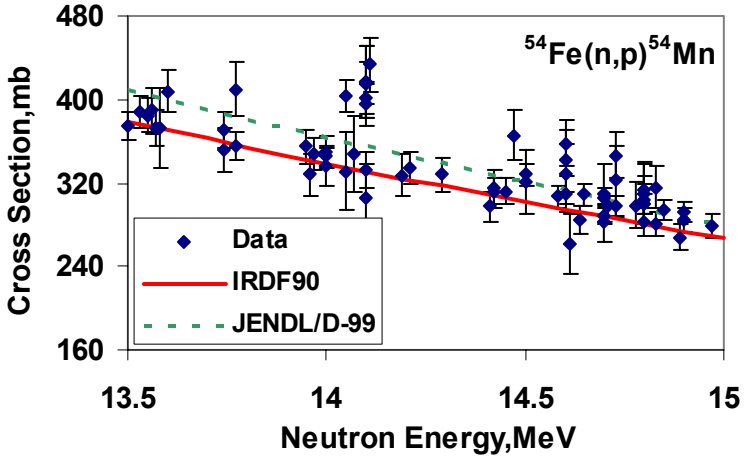

Ti(n,x)48Sc
Cross Section,mb
Neutron Energy,MeV
Data
JENDL/D-99
60
50
40
30
20
10
0
13.5
14
14.5
15

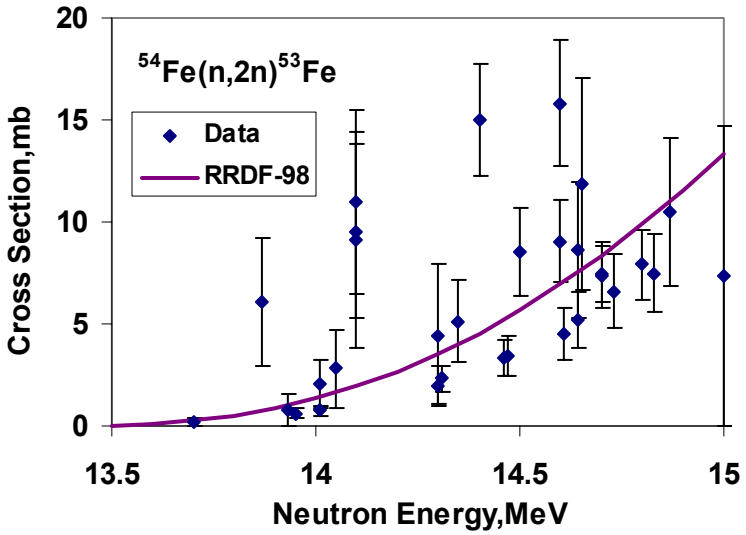

51V(n,α)48Sc
Cross Section,mb
Neutron Energy,MeV
Data
RRDF-98
25
20
15
10
13.5
14
14.5
15

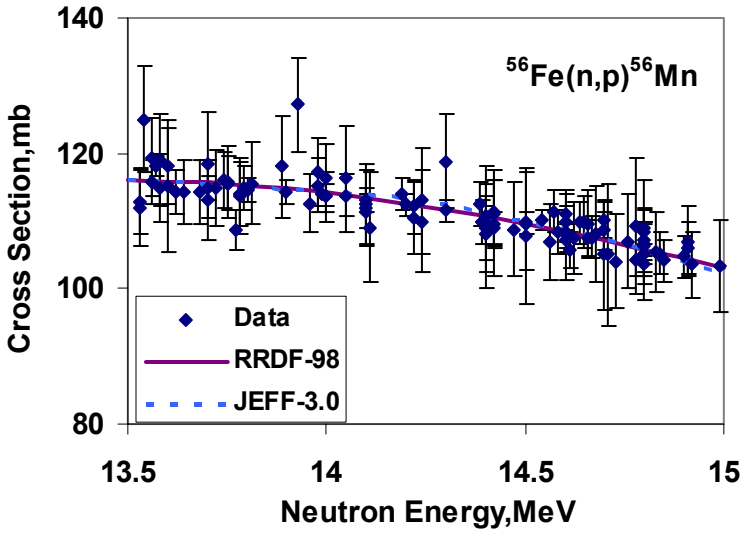

52Cr(n,2n)51Cr
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
JENDL/D-99
ENDF/B-VI
460
380
300
220
140
13.5
14
14.5
15

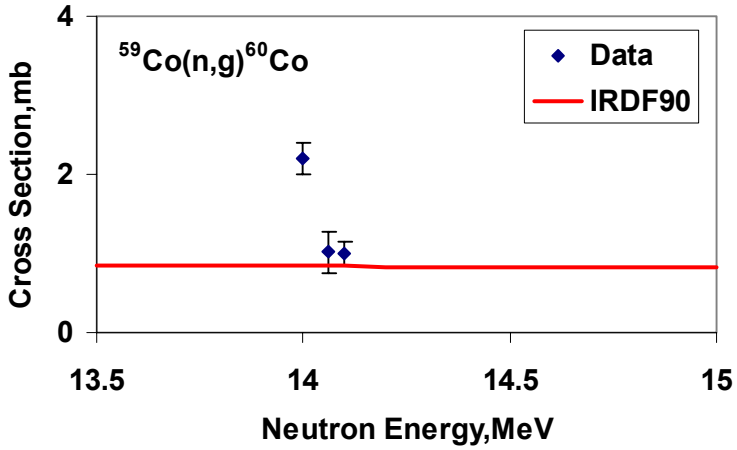

Data
IRDF90
^{55}Mn(n,γ)^{56}Mn
Cross Section,mb
Neutron Energy,MeV
0
1
2
3
13.5
14
14.5
15

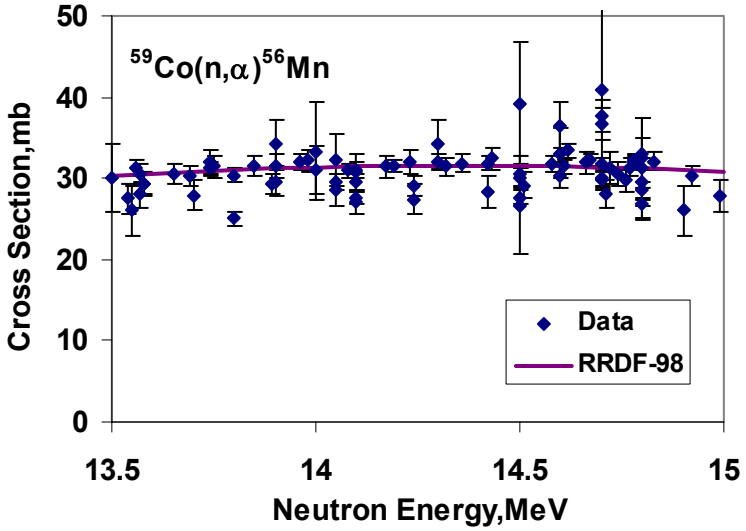

^{54}Fe(n,α)^{51}Cr
Data
RRDF-98
Cross Section,mb
Neutron Energy,MeV
60
80
100
120
13.5
14
14.5
15

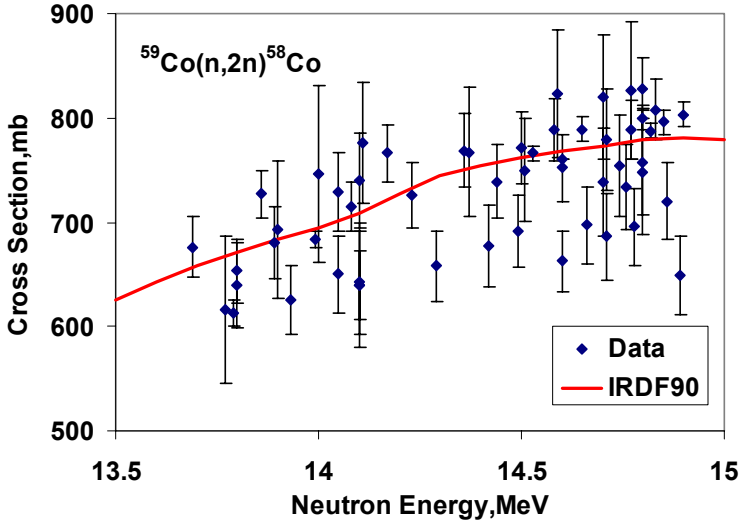

^{54}Fe(n,p)^{54}Mn
Data
IRDF90
JENDL/D-99
Cross Section,mb
Neutron Energy,MeV
160
240
320
400
480
13.5
14
14.5
15

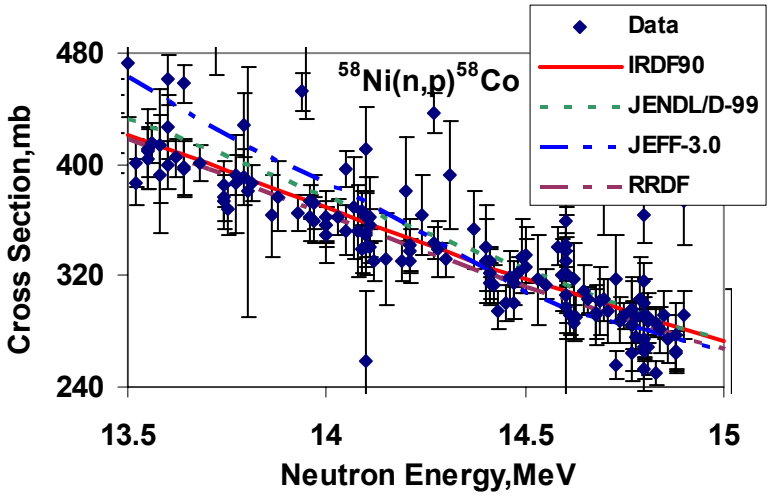

^{54}Fe(n,2n)^{53}Fe
Data
RRDF-98
Cross Section,mb
Neutron Energy,MeV
20
15
10
5
0
13.5
14
14.5
15

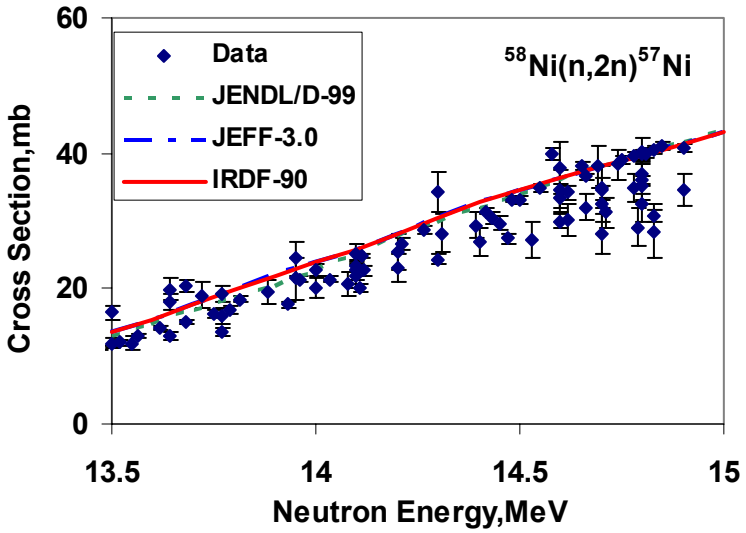

^{56}Fe(n,p)^{56}Mn
Data
RRDF-98
JEFF-3.0
Cross Section,mb
Neutron Energy,MeV
140
120
100
80
13.5
14
14.5
15

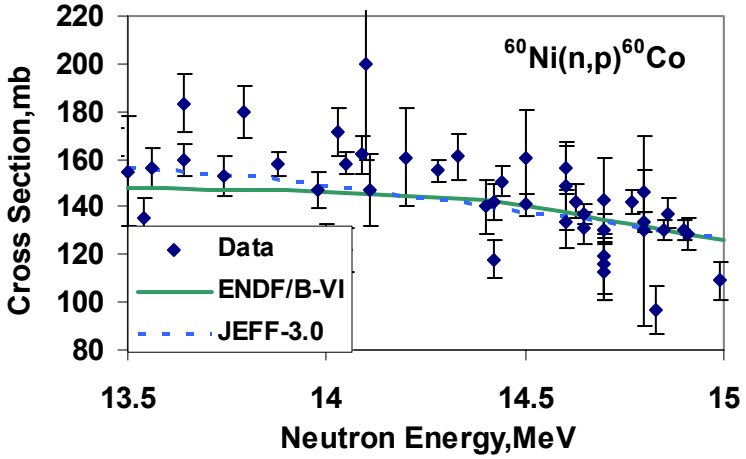

^{59}Co(n,g)^{60}Co
Data
IRDF90
Cross Section,mb
Neutron Energy,MeV
4
2
0
13.5
14
14.5
15

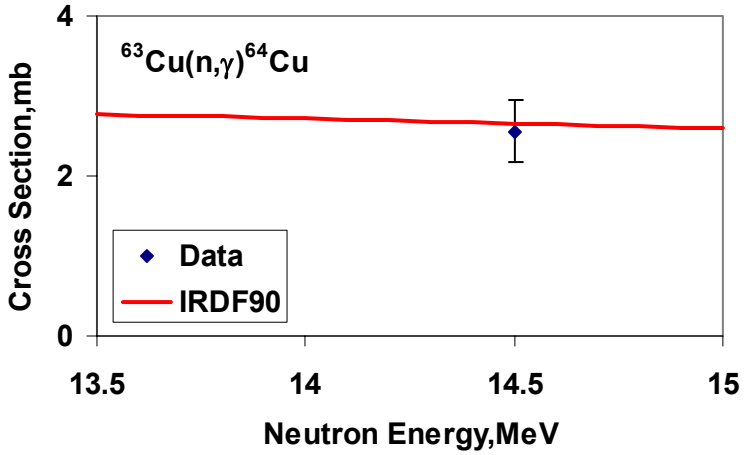

^{59}Co(n,α)^{56}Mn
Cross Section,mb
Neutron Energy,MeV
Data
RRDF-98
50
40
30
20
10
0
13.5
14
14.5
15

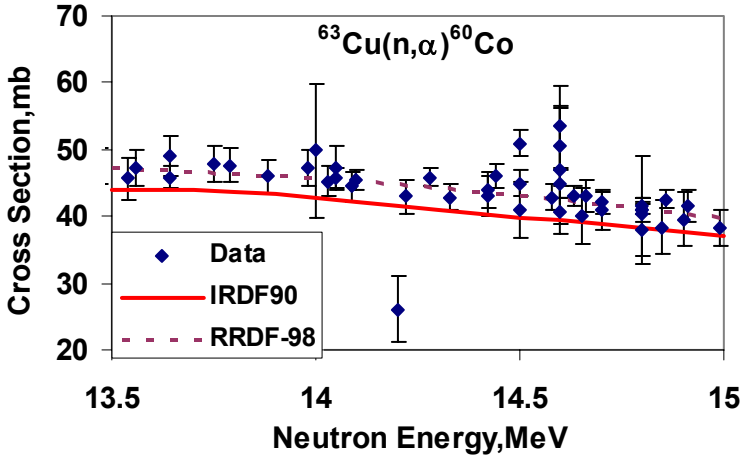

^{59}Co(n,2n)^{58}Co
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
900
800
700
600
500
13.5
14
14.5
15

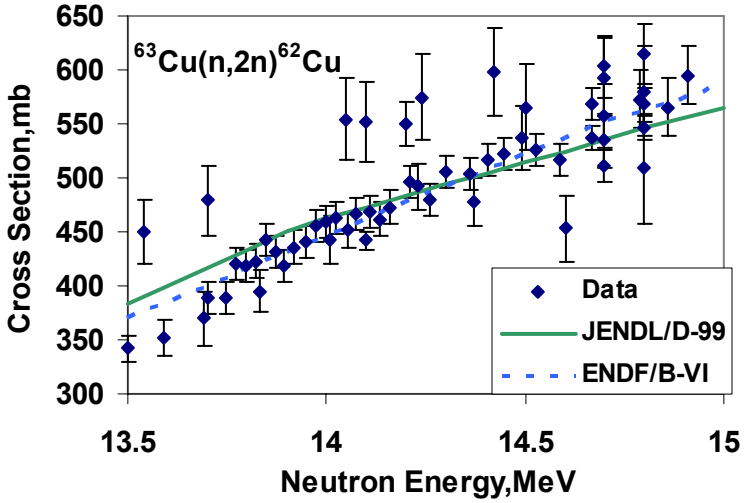

^{58}Ni(n,p)^{58}Co
Cross Section,mb
Neutron Energy,MeV
Data
IRDF90
JENDL/D-99
JEFF-3.0
RRDF
480
400
320
240
13.5
14
14.5
15

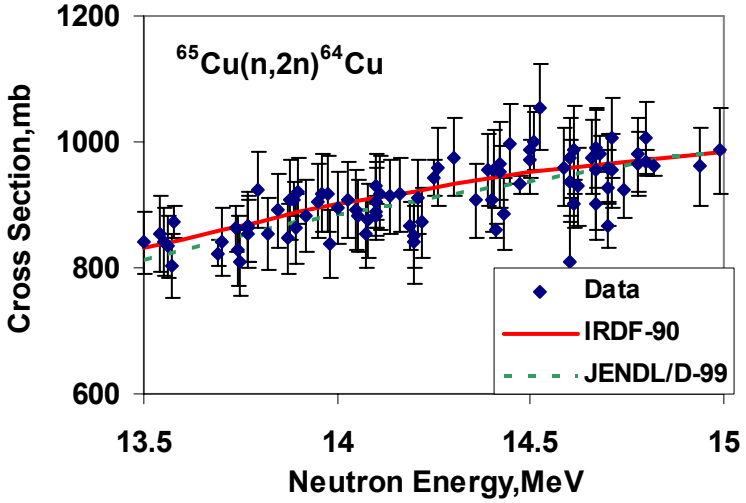

Data
JENDL/D-99
JEFF-3.0
IRDF-90
58Ni(n,2n)57Ni
Cross Section,mb
Neutron Energy,MeV
0
20
40
60
13.5
14
14.5
15

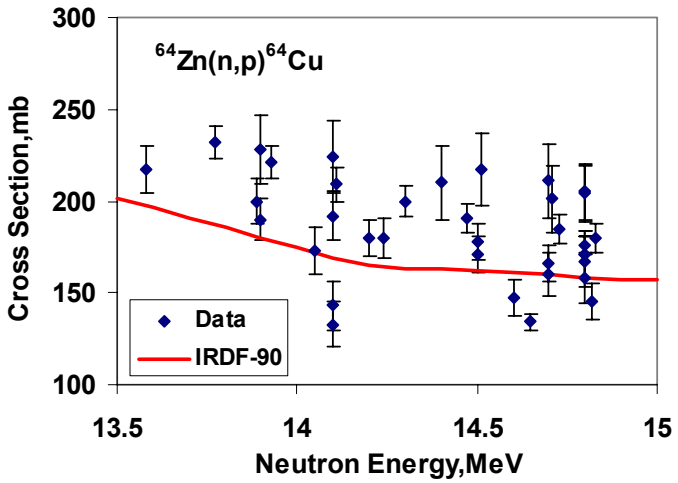

60Ni(n,p)60Co
Data
ENDF/B-VI
JEFF-3.0
Cross Section,mb
Neutron Energy,MeV
80
100
120
140
160
180
200
220
13.5
14
14.5
15

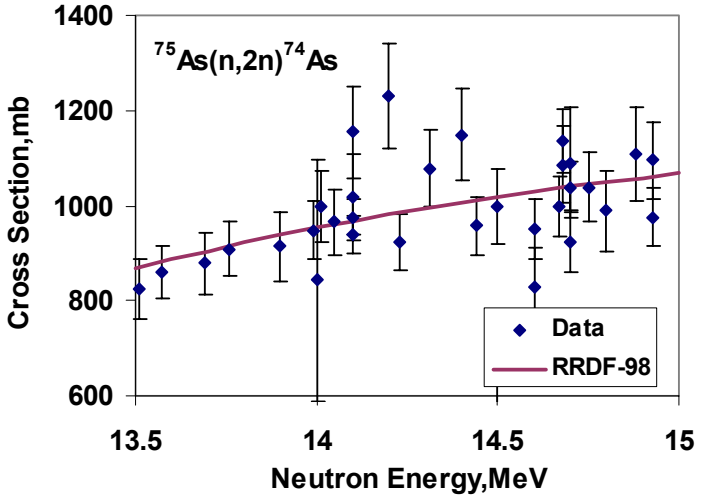

63Cu(n,γ)64Cu
Data
IRDF90
Cross Section,mb
Neutron Energy,MeV
0
2
4
13.5
14
14.5
15

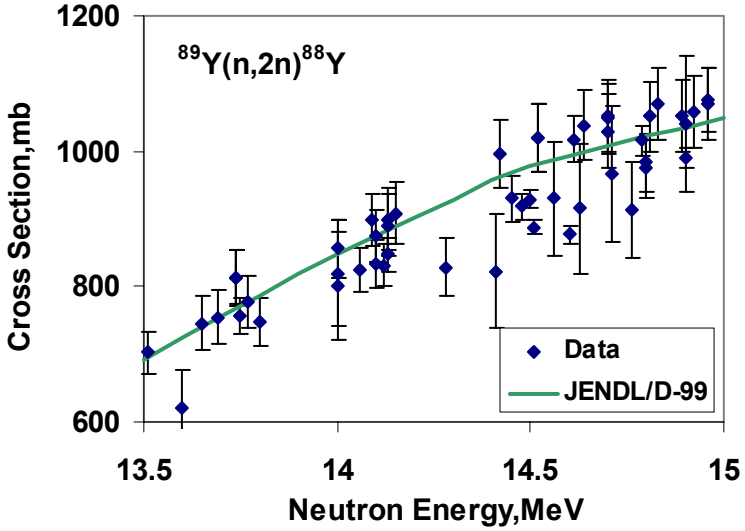

^{63}Cu(n,α)^{60}Co
Cross Section,mb
70
60
50
40
30
20
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF90
RRDF-98

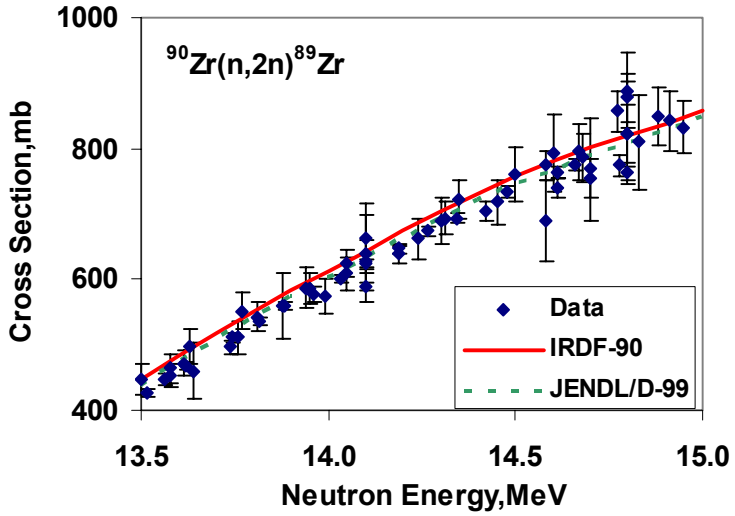

^{63}Cu(n,2n)^{62}Cu
Cross Section,mb
650
600
550
500
450
400
350
300
13.5
14
14.5
15
Neutron Energy,MeV
Data
JENDL/D-99
ENDF/B-VI

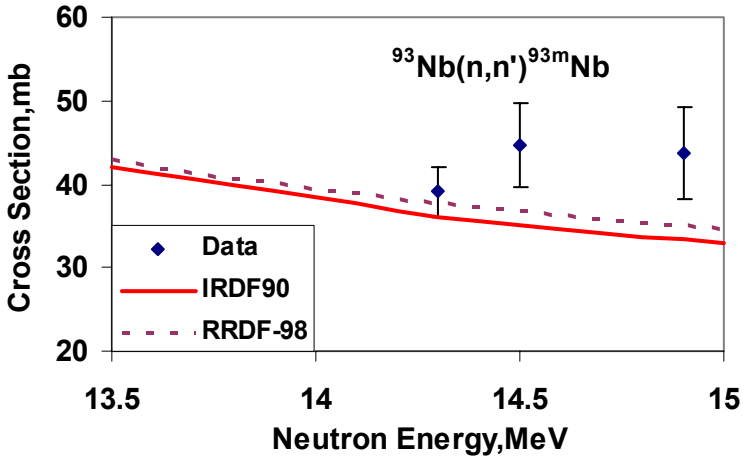

^{65}Cu(n,2n)^{64}Cu
Cross Section,mb
1200
1000
800
600
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF-90
JENDL/D-99

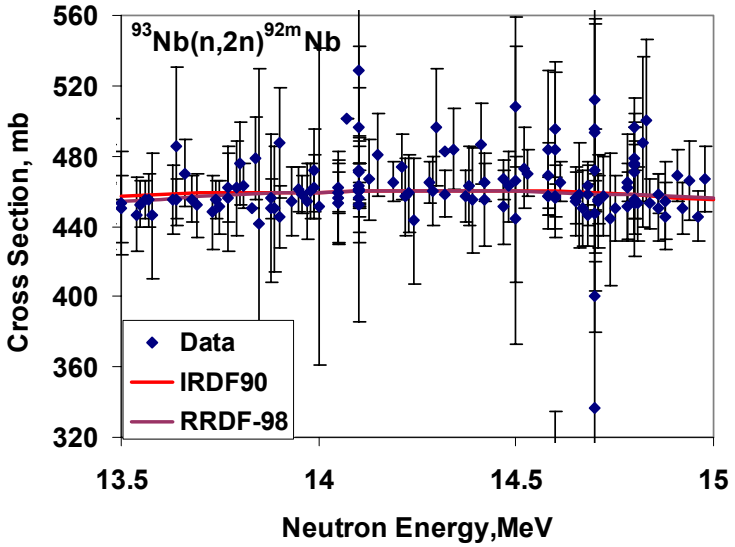

^{64}Zn(n,p)^{64}Cu
Cross Section,mb
300
250
200
150
100
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF-90

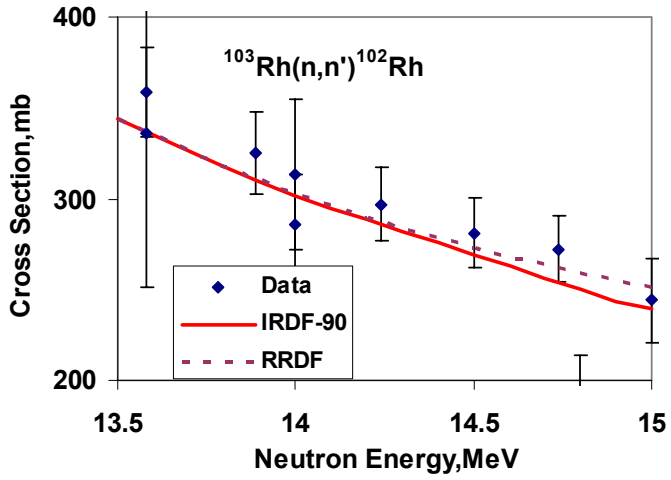

^{75}As(n,2n)^{74}As
Cross Section,mb
1400
1200
1000
800
600
13.5
14
14.5
15
Neutron Energy,MeV
Data
RRDF-98

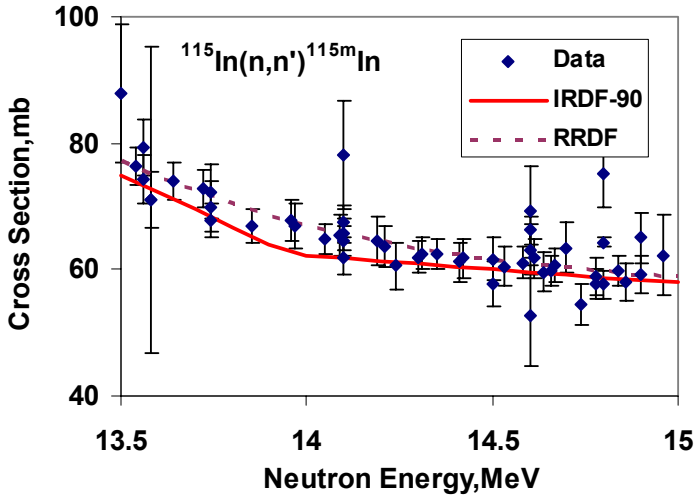

^{89}Y(n,2n)^{88}Y
Cross Section,mb
1200
1000
800
600
13.5
14
14.5
15
Neutron Energy,MeV
Data
JENDL/D-99

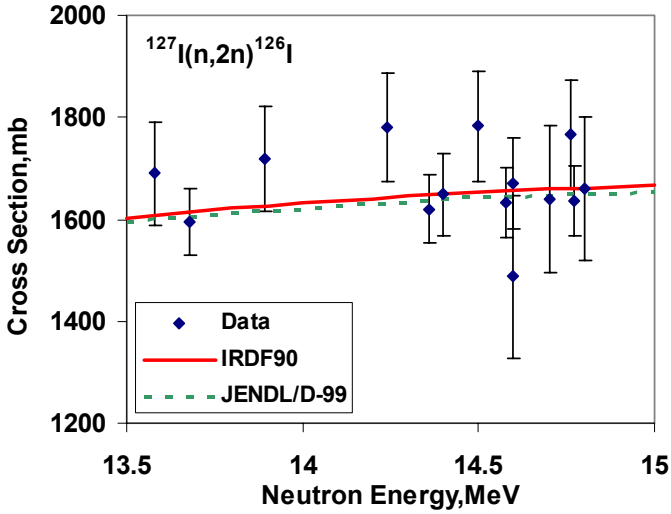

^{90}Zr(n,2n)^{89}Zr
Cross Section,mb
1000
800
600
400
13.5
14.0
14.5
15.0
Neutron Energy,MeV
Data
IRDF-90
JENDL/D-99

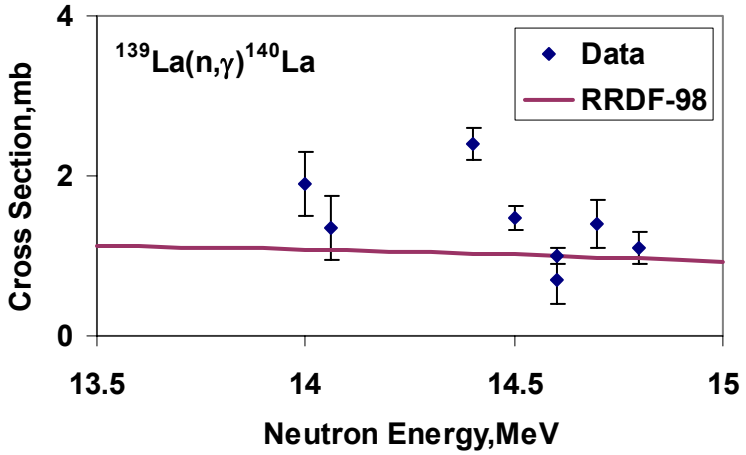

^{93}Nb(n,n')^{93m}Nb
Cross Section,mb
60
50
40
30
20
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF90
RRDF-98

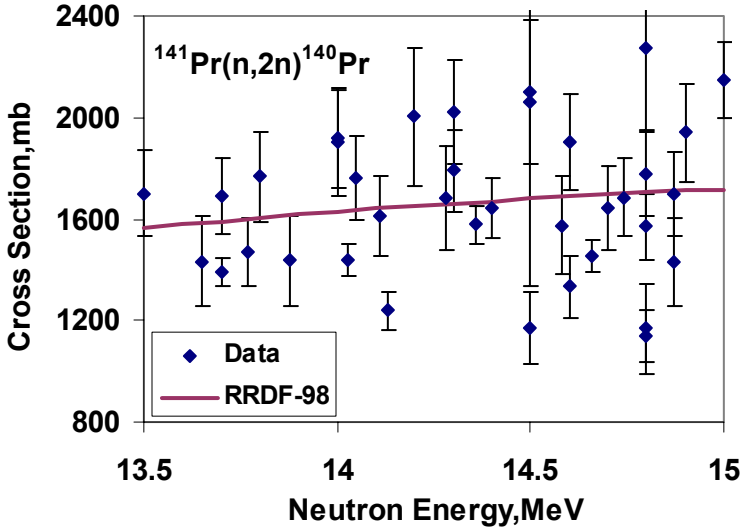

^{93}Nb(n,2n)^{92m}Nb
Cross Section, mb
560
520
480
440
400
360
320
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF90
RRDF-98

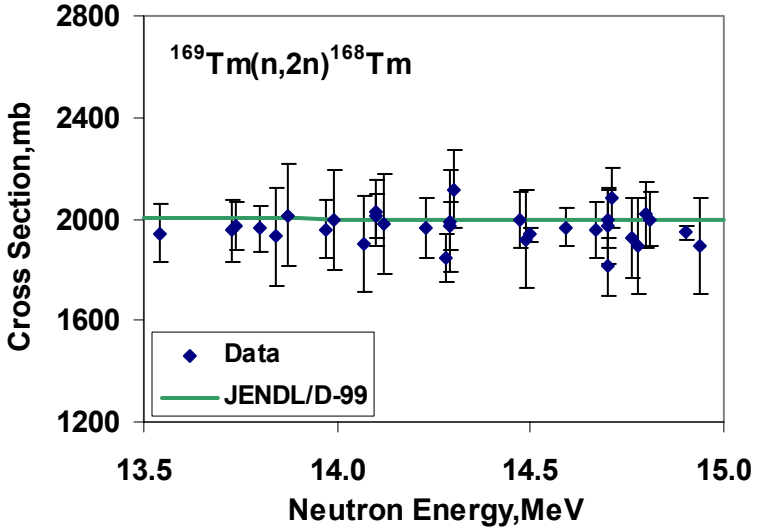

400
300
200
Cross Section,mb
^{103}Rh(n,n')^{102}Rh
Data
IRDF-90
RRDF
13.5
14
14.5
15
Neutron Energy,MeV

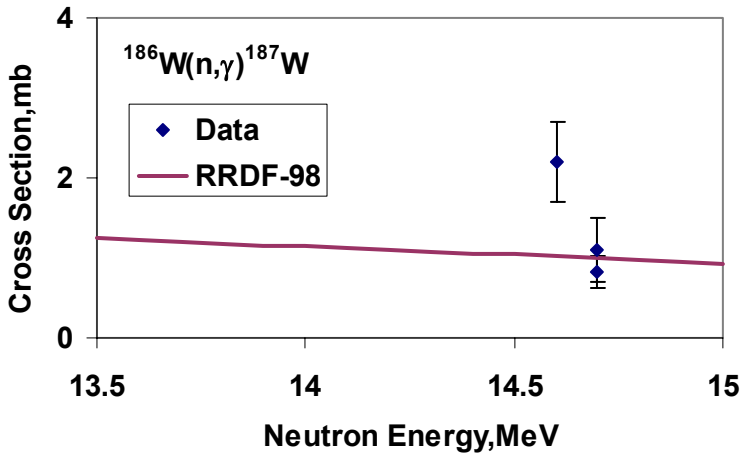

100
80
60
40
Cross Section,mb
^{115}In(n,n')^{115m}In
Data
IRDF-90
RRDF
13.5
14
14.5
15
Neutron Energy,MeV

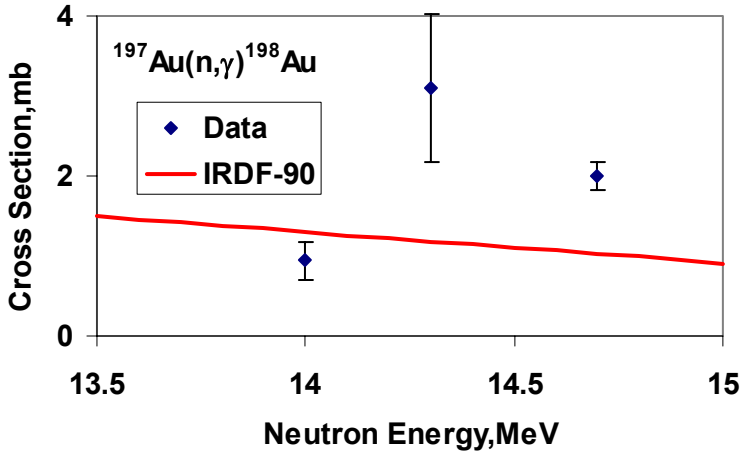

2000
1800
1600
1400
1200
Cross Section,mb
^{127}I(n,2n)^{126}I
Data
IRDF90
JENDL/D-99
13.5
14
14.5
15
Neutron Energy,MeV

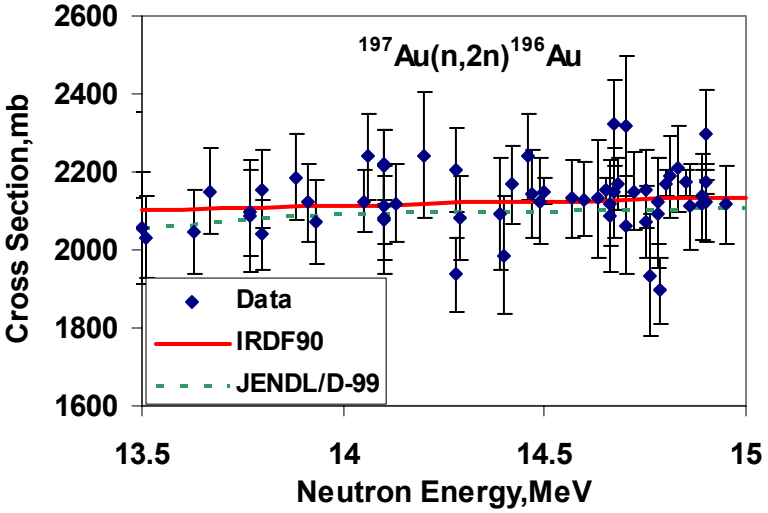

^{139}La(n,γ)^{140}La
Data
RRDF-98
Cross Section,mb
Neutron Energy,MeV
0
2
4
13.5
14
14.5
15

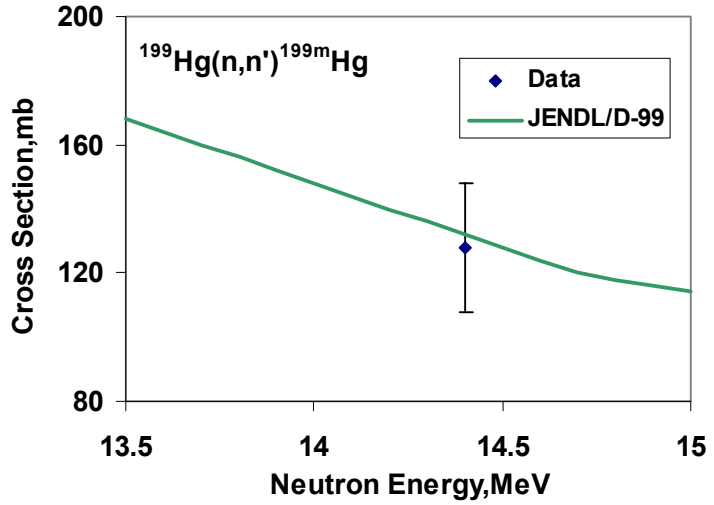

^{141}Pr(n,2n)^{140}Pr
Data
RRDF-98
Cross Section,mb
Neutron Energy,MeV
800
1200
1600
2000
2400
13.5
14
14.5
15

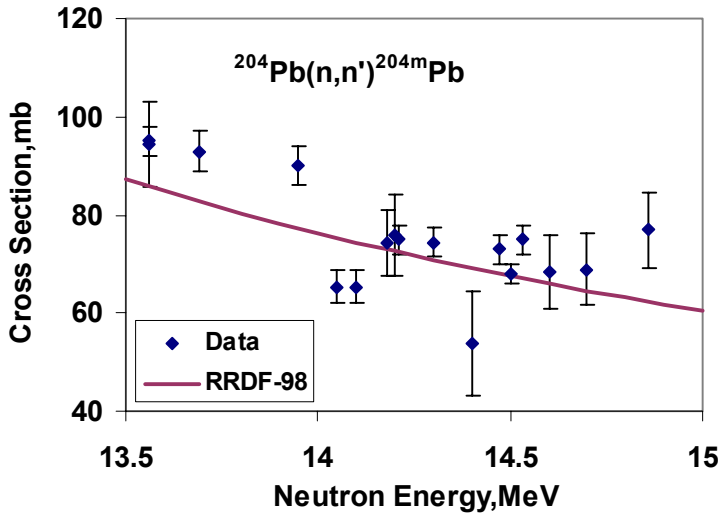

^{169}Tm(n,2n)^{168}Tm
Data
JENDL/D-99
Cross Section,mb
Neutron Energy,MeV
1200
1600
2000
2400
2800
13.5
14.0
14.5
15.0

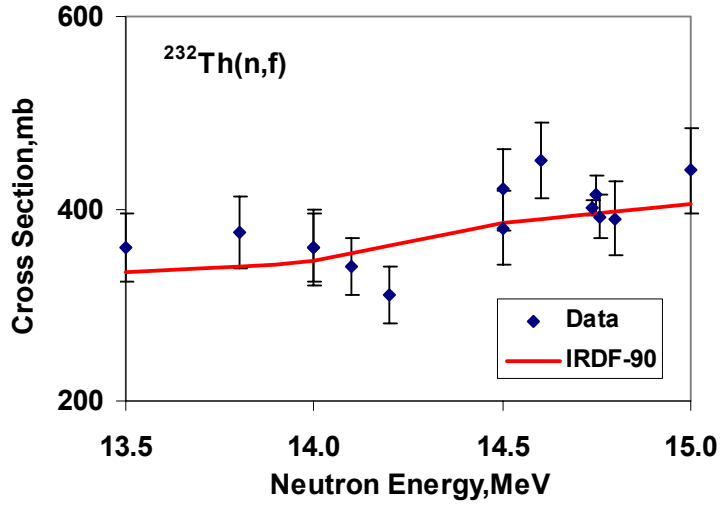

^{186}W(n,γ)^{187}W
Data
RRDF-98
Cross Section,mb
Neutron Energy,MeV
0
2
4
13.5
14
14.5
15

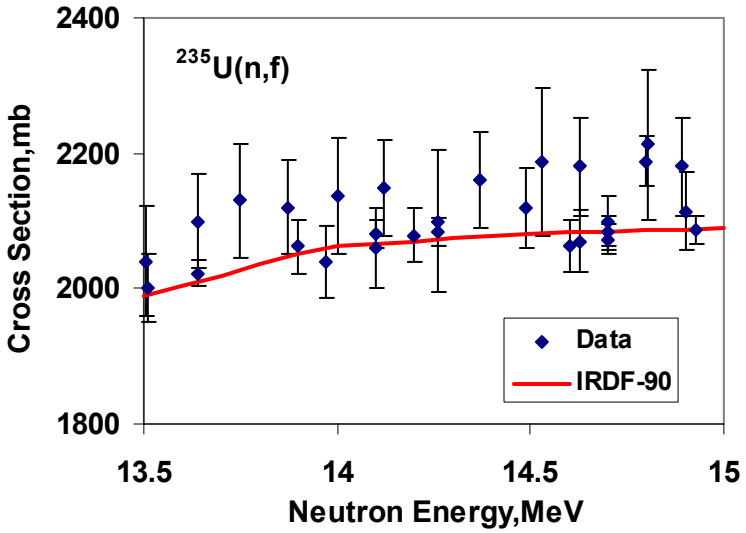

^{197}Au(n,γ)^{198}Au
Data
IRDF-90
Cross Section,mb
Neutron Energy,MeV
0
2
4
13.5
14
14.5
15

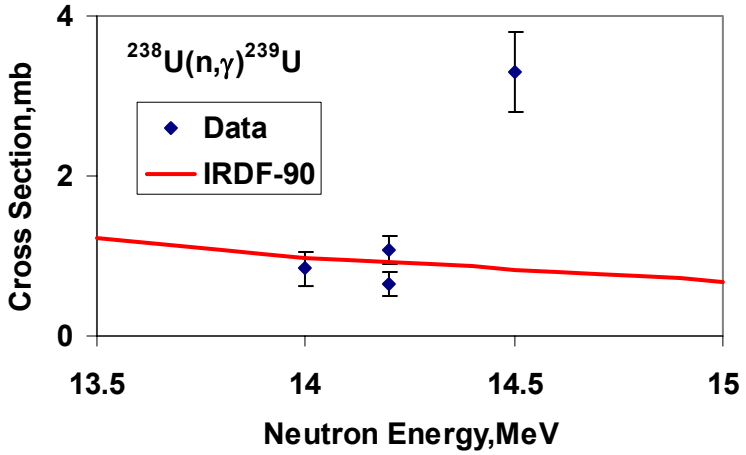

^{197}Au(n,2n)^{196}Au
Data
IRDF90
JENDL/D-99
Cross Section,mb
Neutron Energy,MeV
1600
1800
2000
2200
2400
2600
13.5
14
14.5
15

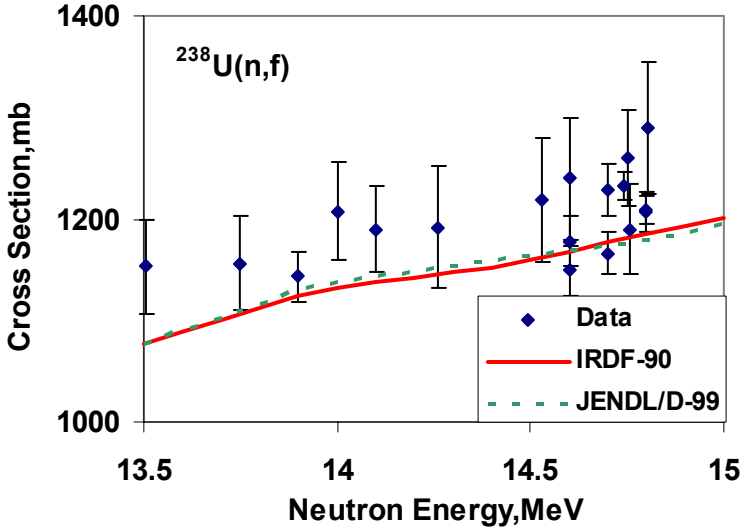

^{199}Hg(n,n')^{199m}Hg
Data
JENDL/D-99
Cross Section,mb
200
160
120
80
13.5
14
14.5
15
Neutron Energy,MeV

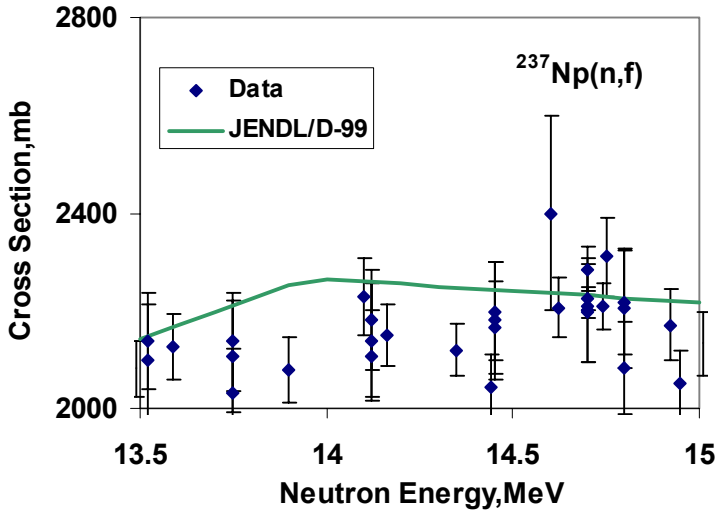

^{204}Pb(n,n')^{204m}Pb
Data
RRDF-98
Cross Section,mb
120
100
80
60
40
13.5
14
14.5
15
Neutron Energy,MeV

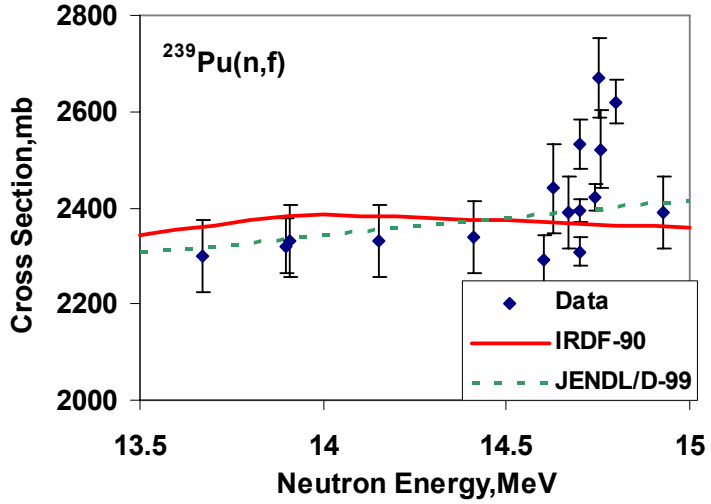

^{232}Th(n,f)
Data
IRDF-90
Cross Section,mb
600
400
200
13.5
14.0
14.5
15.0
Neutron Energy,MeV

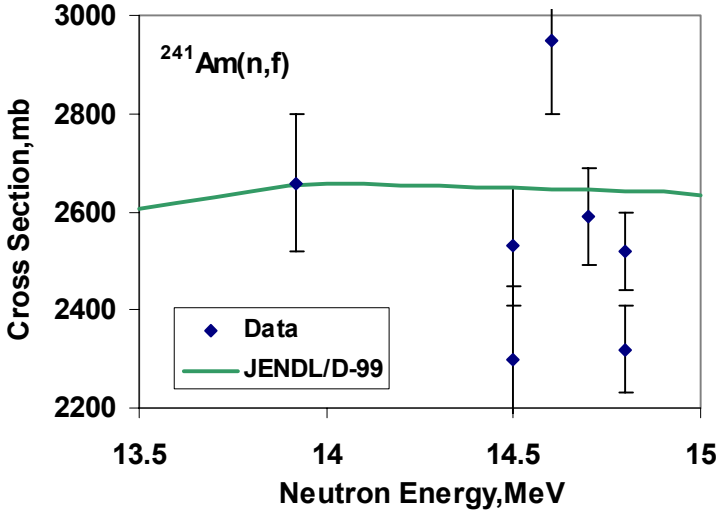

^{235}U(n,f)
Cross Section,mb
2400
2200
2000
1800
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF-90


^{238}U(n,γ)^{239}U
Cross Section,mb
4
2
0
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF-90


^{238}U(n,f)
Cross Section,mb
1400
1200
1000
13.5
14
14.5
15
Neutron Energy,MeV
Data
IRDF-90
JENDL/D-99


2800
2400
2000
Cross Section,mb
Data
JENDL/D-99
237Np(n,f)
13.5
14
14.5
15
Neutron Energy,MeV


2800
2600
2400
2200
2000
Cross Section,mb
239Pu(n,f)
Data
IRDF-90
JENDL/D-99
13.5
14
14.5
15
Neutron Energy,MeV


3000
2800
2600
2400
2200
Cross Section,mb
241Am(n,f)
Data
JENDL/D-99
13.5
14
14.5
15
Neutron Energy,MeV

Response of activation reactions in the neutron field of spontaneous fission of ^{252}Cf

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Physikalisch-Technische Bundesanstalt, Braunschweig, Germany

The response of evaluated cross section data for neutron activation reactions in the reference neutron field of spontaneous fission of ^{252}Cf has been calculated. The bulk of cross section data investigated stems from the previous version of the International Reactor Dosimetry File (IRDF-90.2) [1], the JENDL Dosimetry File (JENDL/D-99) [2], and an updated version of the Russian Reactor Dosimetry File (RRDF-98) [3]. Also, a few selected data sets of the ENDF/B-VI and JEFF-3.0 libraries were used.

Presently, the neutron field of spontaneous fission of ^{252}Cf is the only neutron field which meets all criteria of a reference field with a well-established and accurate spectral distribution valid up to 20 MeV and a complete uncertainty description. The spectral distribution N(E) of the fission neutrons of ^{252}Cf is the result of an evaluation based on modern TOF measurements of this neutron spectrum [4]. The numerical figures and the associated covariance matrix are given in Ref. [5].

Calculated spectrum-averaged cross sections of $\int \sigma(E)\, N(E)\, dE / \int N(E)\, dE$ were determined for the various $\sigma(E)$ data. The associated uncertainties were obtained from the propagated uncertainties of $\sigma(E)$ and N(E). The calculated data are compared with experimental data, and C/E values were derived. The experimental data are from a detailed evaluation [6] of the integral experiments available.

The results are summarized in Table 1. The investigated neutron reactions are given in column 1. The reactions are listed with increasing energy response ranges. In column 2 the mean neutron energy E(50%) of the integrated response of each neutron reaction in the fission neutron field is given. The experimental data of spectrum-averaged cross sections and the uncertainties are given in columns 3 and 4. Experimental data given in brackets are from single experiments which were not included in the evaluation. Such data are listed in Ref. [7], and in a few cases more recent data of the EXFOR database were used. In columns 5 to 7 the C/E values obtained with the IRDF-90.2, the JENDL/D-99 and the RRDF-98 library are given. With the exception of ^{24}Mg(n,p)^{24}Na and ^{93}Nb(n,2n)^{92}Nb^{m}, the original $\sigma(E)$ data of the RRDF-98 library were replaced by data from recent updates [3].

The rigorous inclusion of all uncertainty components contributing to the given C/E values allows quantitative statements to be made on the quality of the evaluated $\sigma(E)$ data valid for the energy response range of the reaction. The calculated C/E values which show agreement with unity within the given uncertainties are printed in bold and indicate optimum agreement between integral and differential data. Besides such values, C/E values were also accepted which were within a band of ± 5% around unity even if the calculated uncertainties were too small to overlap with unity. These values are shaded in the table in gray. For most of the investigated reactions a suitable data set of $\sigma(E)$ is identified in one of the investigated libraries. Only for the reactions ^{199}Hg(n,n'), ^{24}Mg(n,p), ^{127}I(n,2n), ^{55}Mn(n,2n) and ^{63}Cu(n,2n) does this statement fail.

Table 2 summarizes the results obtained with selected data sets of the libraries ENDF/B-VI and JEFF-3.0. The structure of the table is identical with that of table 1.

Table 1 C/E-values in the ^{252}Cf neutron field, calculated with σ(E) data of the cross section libraries IRDF-90.2 , JENDL/D-99 and RRDF-98.

Cf-252(sf)						
Reaction	**E(50%)**	**Exp.**		**C/E**		
	MeV	**<σ> (mb)**	**%**	**IRDF-90.2**	**JENDL/D-99**	**RRDF-98 (Update)**
Au-197(n,γ)Au-198	0.75	**7.679E+1**	1.59	0.966 ± 0.021	**0.977 ± 0.086**	
Cu-63(n,γ)Cu-64	0.93	**1.044E+1**	3.24	**0.996 ± 0.091**	**1.005 ± 0.196**	
In-115(n,γ)In-116m1+m2	1.06	**1.256E+1**	2.23	**0.969 ± 0.047**	**1.003 ± 0.047**	
U-235(n,f)	1.70	**1.210E+3**	1.20	**1.007 ± 0.012**	**1.021 ± 0.024**	
Pu-239(n,f)	1.78	**1.812E+3**	1.37	0.980 ± 0.014	**0.996 ± 0.025**	
Np-237(n,f)	2.07	**1.361E+3**	1.59	**0.999 ± 0.093**	**0.983 ± 0.016**	**0.999 ± 0.024**
In-115(n,n')In-115m	2.68	**1.974E+2**	1.37	0.961 ± 0.025	0.961 ± 0.025	0.972 ± 0.021
U-238(n,f)	2.78	**3.257E+2**	1.64	0.969 ±0.017	**0.980 ± 0.026**	
Hg-199(n,n')Hg-199m	3.10	**2.984E+2**	1.81		0.833 ± 0.067	
Ti-47(n,p)Sc-47	3.84	**1.927E+1**	1.66	**1.006 ± 0.042**	0.962 ± 0.021	
S-32(n,p)P-32	4.06	**7.254E+1**	3.49	**0.969 ± 0.049**	**1.033 ± 0.090**	
Ni-58(n,p)Co-58	4.17	**1.175E+2**	1.30	**0.982 ± 0.026**	0.975 ± 0.016	**1.000 ± 0.023**
Zn-64(n,p)Cu-64	4.26	**4.059E+1**	1.65	**1.037 ± 0.054**	0.942 ± 0.023	
Fe-54(n,p)Mn-54	4.32	**8.684E+1**	1.34	**1.015 ± 0.026**	1.027 ± 0.019	
Co-59(n,p)Fe-59	5.76	**1.690E+0**	2.48			
Al-27(n,p)Mg-27	5.87	**4.880E+0**	2.14	0.958 ± 0.039	1.058 ± 0.027	**1.007 ± 0.032**
Ti-46(n,p)Sc-46	6.08	**1.407E+1**	1.77	0.876 ± 0.029	0.964 ± 0.030	**0.983 ± 0.037**
V-51(n,p)Ti-51	6.44	**6.488E-1**	1.97			
Cu-63(n,α)Co-60	7.28	**6.887E-1**	1.96	**0.986 ± 0.033**	1.059 ± 0.029	**1.007 ± 0.037**
Fe-56(n,p)Mn-56	7.56	**1.465E+0**	1.77	0.936 ± 0.030	**0.962 ± 0.048**	**1.007 ± 0.035**
Mg-24(n,p)Na-24	8.25	**1.996E+0**	2.44	1.082 ± 0.040	1.092 ± 0.034	1.073 ± 0.034
Co-59(n,α)Mn-56	8.36	**2.218E-1**	1.88	**0.975 ± 0.036**	**1.040 ± 0.050**	**0.997 ± 0.043**
Ti-48(n,p)Sc-48	8.38	**4.247E-1**	1.89	0.912 ± 0.032	0.931 ± 0.028	**1.005 ± 0.057**
Al-27(n,α)Na-24	8.66	**1.016E+0**	1.28	**1.022 ± 0.026**	**1.022 ± 0.026**	
V-51(n,α)Sc-48	9.97	**3.900E-2**	2.21	**0.995 ± 0.044**		**0.989 ± 0.041**
Tm-169(n,2n)Tm-168	10.34	**[6.690E+0]**	6.28		**0.932 ± 0.065**	
Au-197(n,2n)Au-196	10.61	**5.506E+0**	1.83	**1.044 ± 0.052**	1.049 ± 0.031	
Nb-93(n,2n)Nb-92m	11.47	**[7.490E-1]**	5.07	**1.041 ± 0.064**	**1.011 ± 0.070**	**1.030 ± 0.058**
I-127(n,2n)I-126	11.75	**2.069E+0**	2.73	1.062 ± 0.045	1.096 ± 0.051	
Cu-65(n,2n)Cu-64	12.64	**6.582E-1**	2.22	**1.030 ± 0.042**	1.061 ± 0.039	
Mn-55(n,2n)Mn-54	12.84	**4.075E-1**	2.33	1.181 ± 0.115	1.237 ± 0.111	
Co-59(n,2n)Co-58	13.06	**4.051E-1**	2.51	**1.044 ± 0.051**	**1.030 ± 0.045**	
Cu-63(n,2n)Cu-62	13.75	**1.844E-1**	3.98	1.134 ± 0.068	1.140 ± 0.066	
F-19(n,2n)F-18	14.02	**1.612E-2**	3.37	**1.065 ± 0.063**	1.151 ± 0.070	**1.009 ± 0.064**
Zr-90(n,2n)Zr-89	14.41	**2.210E-1**	2.89	**1.001 ± 0.061**	**0.979 ± 0.058**	
Ni-58(n,2n)Ni-57	14.98	**8.952E-3**	3.57	**1.033 ± 0.079**	**1.004 ± 0.072**	

Table 2 C/E-values in the ^{252}Cf neutron field, calculated with selected σ(E) data of the cross section libraries ENDF/B-VI and JEFF-3.0.

Cf-252(sf)						
Reaction	**E(50%)**	**Exp.**		**C/E**		
	MeV	**<σ> (mb)**	**%**	**ENDF/B-VI**	**JEFF-3.0**	
Ni-58(n,p)Co-58	4.17	**1.175E+2**	1.30	**0.981 ± 0.028**	**0.997 ± 0.037**	
Ni-60(n,p)Co-60	7.05	**[2.390E+0]**	5.44	**1.044 ± 0.121**	1.170 ± 0.117	
Fe-56(n,p)Mn-56	7.56	**1.465E+0**	1.77		**0.981 ± 0.025**	
Cu-65(n,2n)Cu-64	12.64	**6.582E-1**	2.22	**1.030 ± 0.044**		
Cu-63(n,2n)Cu-62	13.75	**1.844E-1**	3.98	1.115 ± 0.078		
Cr-52(n,2n)Cr-51	14.69					
Ni-58(n,2n)Ni-57	14.98	**8.952E-3**	3.57	**1.034 ± 0.077**	**1.034 ± 0.078**	

The energy response of the various reactions strongly depends on the threshold and the shape of the σ(E) data. The energy response range, covering 90% of the total response of a reaction, in the ^{252}Cf neutron field is between 0.21 MeV and 5.70 MeV for ^{235}U(n,f) and between 13.12 MeV and 18.25 MeV for ^{58}Ni(n,2n)^{57}Ni , with mean values E(50%) of 1.70 MeV and 14.98 MeV, respectively. This means the C/E values of Table 1 and 2 cover quite different energy regions and highlight only selected portions of the cross section curve. Complete proof of the validity of a cross section requires additional investigations in the remaining energy regions.

More details of the summary given in Table 1 and Table 2 are quoted in Tables 3 - 6. For each of the investigated cross section libraries, a complete list of considered reactions and all calculated spectrum-averaged data are given, independent of the availability of appropriate experimental data. In column 4 the numerical values for the calculated spectrum-averaged cross sections are given and in column 5 the corresponding uncertainties. In addition, the individual uncertainty contributions of the σ(E) data and of the spectral distribution N(E) to the calculated values are separately listed in columns 6 and 7. Due to the averaging process in the calculation of spectrum-averaged data, the original uncertainties of the σ(E) data are often further reduced.

In a number of cases very low uncertainties are found in column 6 of the tables, indicating that the quoted uncertainties of the evaluated σ(E) data likely are extremely small. An analysis of the covariance files for these evaluations shows uncertainty values which often approach the accuracy level of the best-known reference cross sections. Considering the experimental database of the individual reactions and the spread of the available experimental data, it has to be concluded that only a minority of the evaluated data sets with uncertainty values of < 2% will really meet the accuracy level quoted. Unfortunately, such low uncertainties can also originate from cross section evaluations based on least-squares principles if the cross-correlations between the different experimental data sets or the correlations between data belonging to the same experiment are neglected or improperly handled.

One should not underestimate the impact of unreliable uncertainty values on practical applications. In reactor dosimetry, for example, the response of a number of activation reactions in a typical neutron field is used to derive the spectral fluence distribution with unfolding methods. The response of each of the activation reactions represents a broad resolution experiment with a strong

Table 3 IRDF-90.2 Library (<σ> data in the ^{252}Cf neutron field)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
Sc-45(n,γ)Sc-46			**4.850**	9.06	9.00	1.07		0.56
Li-6(n,t)He-4			**311.8**	0.85	0.63	0.57		0.62
Nb-93(n,γ)Nb-94m			**26.33**	7.52	7.48	0.84		0.68
Au-197(n,γ)Au-198	**76.79**	1.59	**74.19**	1.43	1.23	0.73	0.966 ± 0.021	0.73
Ag-109(n,γ)Ag-110m			**9.344**	7.58	7.56	0.53		0.74
Mn-55(n,γ)Mn-56			**2.843**	11.71	11.66	1.08		0.80
B-10(n,α)Li-7			**444.1**	6.23	6.20	0.56		0.92
U-238(n,γ)U-239			**67.97**	1.06	0.94	0.48		0.92
Na-23(n,γ)Na-24	[0.335]	4.48	**0.2696**	12.92	12.90	0.85	0.805 ± 0.110	0.93
Th-232(n,γ)Th-233	**[87.8]**	4.56	**89.68**	11.83	11.82	0.42	**1.021 ± 0.129**	0.93
Cu-63(n,γ)Cu-64	**10.44**	3.24	**10.40**	8.51	8.49	0.62	**0.996 ± 0.091**	0.97
Fe-58(n,γ)Fe-59			**2.469**	29.17	29.17	0.60		1.07
Co-59(n,γ)Co-60	[6.97]	4.88	**6.062**	4.76	4.70	0.71	0.870 ± 0.059	1.08
In-115(n,γ)In-116m1+m2	**125.6**	2.23	**121.7**	4.35	4.33	0.33	**0.969 ± 0.047**	1.13
U-235(n,f)	**1210**	1.20	**1218**	0.32	0.32	0.06	**1.007 ± 0.012**	1.70
Pu-239(n,f)	**1812**	1.37	**1775**	0.41	0.41	0.04	0.980 ± 0.014	1.78
Np-237(n,f)	**1361**	1.59	**1360**	9.21	9.21	0.21	**0.999 ± 0.093**	2.07
Rh-103(n,n')Rh-103m	**[809]**	2.97	**714.4**	3.08	3.07	0.25	0.883 ± 0.038	2.39
In-115(n,n')In-115m	**197.4**	1.37	**189.8**	2.19	2.16	0.37	0.961 ± 0.025	2.68
Nb-93(n,n')Nb-93m	**[146]**	3.45	**142.6**	3.01	2.99	0.35	**0.977 ± 0.045**	2.72
U-238(n,f)	**325.7**	1.64	**315.5**	0.67	0.54	0.39	0.969 ± 0.017	2.78
Th-232(n,f)	**[89.4]**	3.02	**78.55**	5.11	5.09	0.42	0.879 ± 0.052	2.99
Ti-47(n,p)Sc-47	**19.27**	1.66	**19.38**	3.83	3.78	0.62	**1.006 ± 0.042**	3.85
P-31(n,p)Si-31			**30.68**	3.65	3.58	0.69		3.97
S-32(n,p)P-32	**72.54**	3.49	**70.30**	3.67	3.60	0.74	**0.969 ± 0.049**	4.08
Ni-58(n,p)Co-58	**117.5**	1.30	**115.4**	2.32	2.21	0.72	**0.982 ± 0.026**	4.13
Zn-64(n,p)Cu-64	**40.59**	1.65	**42.10**	4.93	4.87	0.78	**1.037 ± 0.054**	4.16
Fe-54(n,p)Mn-54	**86.84**	1.34	**88.16**	2.23	2.09	0.78	**1.015 ± 0.026**	4.28

Table 3 (continued)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
Al-27(n,p)Mg-27	**4.880**	2.14	**4.674**	3.44	3.24	1.14	**0.958 ± 0.039**	5.85
Ti-46(n,p)Sc-46	**14.07**	1.77	**12.33**	2.74	2.47	1.17	0.876 ± 0.029	5.93
Ni-60(n,p)Co-60	**[2.39]**	5.44	**2.495**	10.14	10.05	1.37	**1.044 ± 0.120**	7.09
Cu-63(n,α)Co-60	**0.6887**	1.96	**0.6789**	2.75	2.38	1.39	**0.986 ± 0.033**	7.29
Fe-56(n,p)Mn-56	**1.465**	1.77	**1.371**	2.62	2.18	1.46	0.936 ± 0.030	7.54
Mg-24(n,p)Na-24	**1.996**	2.44	**2.160**	2.75	2.24	1.59	1.082 ± 0.040	8.25
Co-59(n,α)Mn-56	**0.2218**	1.88	**0.2163**	3.14	2.73	1.56	**0.975 ± 0.036**	8.35
Ti-48(n,p)Sc-48	**0.4247**	1.89	**0.3872**	3.02	2.58	1.57	0.912 ± 0.032	8.40
Al-27(n,α)Na-24	**1.016**	1.47	**1.038**	2.12	1.36	1.62	**1.022 ± 0.026**	8.66
V-51(n,α)Sc-48	**3.900E-2**	2.21	**3.882E-2**	3.81	3.32	1.88	**0.995 ± 0.044**	9.95
Au-197(n,2n)Au-196	**5.506**	1.83	**5.747**	4.65	4.19	2.02	**1.044 ± 0.052**	10.62
Nb-93(n,2n)Nb-92m	**[0.749]**	5.07	**0.7794**	3.48	2.67	2.23	**1.041 ± 0.064**	11.35
In-115(n,2n)In-114			**1.586**	4.02	3.23	2.39		11.74
I-127(n,2n)I-126	**2.069**	2.73	**2.197**	3.30	2.28	2.38	1.062 ± 0.045	11.75
Cu-65(n,2n)Cu-64	**0.6582**	2.22	**0.6779**	3.44	1.83	2.92	**1.030 ± 0.042**	12.64
Mn-55(n,2n)Mn-54	**0.4075**	2.33	**0.4811**	9.42	8.90	3.07	1.181 ± 0.115	12.85
Co-59(n,2n)Co-58	**0.4051**	2.51	**0.4228**	4.20	2.67	3.24	**1.044 ± 0.051**	13.03
Cu-63(n,2n)Cu-62	**0.1844**	3.98	**0.2091**	4.44	1.66	4.11	1.134 ± 0.068	13.79
Y-89(n,2n)Y-88			**0.3446**	5.91	4.05	4.30		13.94
F-19(n,2n)F-18	**1.612E-2**	3.37	**1.716E-2**	4.86	2.09	4.39	**1.065 ± 0.063**	14.00
Zr-90(n,2n)Zr-89	**0.2210**	2.89	**0.2212**	5.31	1.57	5.07	**1.001 ± 0.061**	14.41
Ti-47(n,np)Sc-46			**2.316E-2**	30.45	30.00	5.24		14.61
Cr-52(n,2n)Cr-51			**9.703E-2**	6.23	2.72	5.60		14.70
Ni-58(n,2n)Ni-57	**8.952E-3**	3.57	**9.243E-3**	6.80	3.05	6.07	**1.033 ± 0.079**	14.95
Ti-48(n,np)Sc-47			**4.059E-3**	30.61	30.00	6.09		15.13

Table 4 JENDL/D-99 Library (<σ> data in the ^{252}Cf neutron field)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
Eu-151(n,γ)Eu-152			**360.0**	2.76	2.63	0.86		0.62
Li-6(n,t)He-4			**323.3**	2.83	2.77	0.56		0.64
Sc-45(n,γ)Sc-46			**6.023**	3.51	3.40	0.86		0.68
Au-197(n,γ)Au-198	**76.79**	1.59	**75.01**	8.71	8.69	0.66	**0.977 ± 0.086**	0.77
Mn-55(n,γ)Mn-56			**2.843**	11.71	11.66	1.08		0.80
Fe-58(n,γ)Fe-59			**1.795**	3.05	2.96	0.75		0.80
Ta-181(n,γ)Ta-182	**[106]**	5.78	**83.34**	5.46	5.41	0.77	0.786 ± 0.062	0.82
B-10(n,α)Li-7			**427.2**	6.19	6.17	0.58		0.84
U-238(n,γ)U-239			**64.68**	5.74	5.72	0.51		0.87
Cu-63(n,γ)Cu-64	**10.44**	3.24	**10.49**	19.19	19.18	0.65	**1.005 ± 0.196**	0.88
Cr-50(n,γ)Cr-51			**6.230**	10.65	10.64	0.62		0.92
Th-232(n,γ)Th-233	**[87.8]**	4.56	**83.48**	11.92	11.91	0.43	**0.951 ± 0.121**	0.95
In-115(n,γ)In-116m1+m2	**125.6**	2.23	**126.0**	4.11	4.09	0.41	**1.003 ± 0.047**	0.99
W-186(n,γ)W-187			**34.74**	4.61	4.59	0.38		1.00
Co-59(n,γ)Co-60	[6.97]	4.88	**5.591**	4.58	4.52	0.74	0.802 ± 0.054	1.04
Na-23(n,γ)Na-24	[0.335]	4.48	**0.2236**	12.89	12.86	0.95	0.667 ± 0.091	1.08
Ag-109(n,γ)Ag-110m			**6.657**	9.85	9.84	0.39		1.09
U-235(n,f)	**1210**	1.20	**1236**	1.97	1.97	0.05	**1.021 ± 0.024**	1.70
Pu-239(n,f)	**1812**	1.37	**1804**	2.04	2.04	0.05	**0.996 ± 0.025**	1.79
Np-237(n,f)	**1361**	1.59	**1338**	0.34	0.27	0.21	**0.983 ± 0.016**	2.07
Am-241(n,f)			**1396**	2.81	2.90	0.29		2.23
Rh-103(n,n')Rh-103m	**[809]**	2.97	**716.3**	3.08	3.07	0.25	0.885 ± 0.038	2.39
In-115(n,n')In-115m	**197.4**	1.37	**189.8**	2.19	2.16	0.37	0.961 ± 0.025	2.68
Nb-93(n,n')Nb-93m	**[146]**	3.45	**149.7**	3.18	3.16	0.35	**1.025 ± 0.048**	2.69
U-238(n,f)	**325.7**	1.64	**319.2**	2.04	2.00	0.40	0.980 ± 0.026	2.77
Th-232(n,f)	**[89.4]**	3.02	**82.14**	5.10	5.08	0.42	0.919 ± 0.054	2.98
Hg-199(n,n')Hg-199m	**298.4**	1.81	**248.6**	7.83	7.82	0.43	0.833 ± 0.067	3.10
Ti-47(n,p)Sc-47	**19.27**	1.66	**18.54**	1.42	1.28	0.61	0.962 ± 0.021	3.82

Table 4 (continued)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
P-31(n,p)Si-31			32.24	1.55	1.40	0.67		3.83
S-32(n,p)P-32	72.54	3.49	74.97	7.96	7.93	0.73	1.033 ± 0.090	4.04
Ni-58(n,p)Co-58	117.5	1.30	114.6	0.95	0.60	0.74	0.975 ± 0.016	4.22
Zn-64(n,p)Cu-64	40.59	1.65	38.23	1.78	1.60	0.79	0.942 ± 0.023	4.35
Fe-54(n,p)Mn-54	86.84	1.34	89.22	1.25	0.97	0.79	1.027 ± 0.019	4.35
Al-27(n,p)Mg-27	4.880	2.14	5.163	1.37	0.73	1.16	1.058 ± 0.027	5.89
Ti-46(n,p)Sc-46	14.07	1.77	13.57	2.59	2.31	1.18	0.964 ± 0.030	5.99
Cu-63(n,α)Co-60	0.6887	1.96	0.7291	1.95	1.49	1.26	1.059 ± 0.029	7.00
Ni-60(n,p)Co-60	[2.39]	5.44	2.281	18.75	18.70	1.45	0.954 ± 0.186	7.42
Fe-56(n,p)Mn-56	1.465	1.77	1.410	4.66	4.42	1.47	0.962 ± 0.048	7.68
Co-59(n,α)Mn-56	0.2218	1.88	0.2306	4.46	4.18	1.55	1.040 ± 0.050	8.23
Mg-24(n,p)Na-24	1.996	2.44	2.179	2.00	1.23	1.59	1.092 ± 0.034	8.25
Ti-48(n,p)Sc-48	0.4247	1.89	0.3954	2.39	1.82	1.56	0.931 ± 0.028	8.33
Al-27(n,α)Na-24	1.016	1.47	1.038	2.12	1.36	1.63	1.022 ± 0.026	8.66
Tm-169(n,2n)Tm-168	[6.69]	6.28	6.233	3.01	2.26	1.98	0.932 ± 0.065	10.34
Au-197(n,2n)Au-196	5.506	1.83	5.776	2.33	1.16	2.02	1.049 ± 0.031	10.60
Nb-93(n,2n)Nb-92m	[0.749]	5.07	0.7576	4.65	4.07	2.25	1.011 ± 0.070	11.47
I-127(n,2n)I-126	2.069	2.73	2.268	3.78	2.97	2.35	1.096 ± 0.051	11.69
Cu-65(n,2n)Cu-64	0.6582	2.22	0.6985	2.97	0.87	2.84	1.061 ± 0.039	12.55
Mn-55(n,2n)Mn-54	0.4075	2.33	0.5041	8.70	8.15	3.06	1.237 ± 0.111	12.82
Co-59(n,2n)Co-58	0.4051	2.51	0.4171	3.56	1.36	3.29	1.030 ± 0.045	13.09
Cu-63(n,2n)Cu-62	0.1844	3.98	0.2102	4.19	1.32	3.97	1.140 ± 0.066	13.70
Y-89(n,2n)Y-88			0.3440	4.47	1.40	4.25		13.90
F-19(n,2n)F-18	1.612E-2	3.37	1.855E-2	5.04	2.61	4.31	1.151 ± 0.070	13.94
Zr-90(n,2n)Zr-89	0.2210	2.89	0.2164	5.15	0.56	5.12	0.979 ± 0.058	14.44
Cr-52(n,2n)Cr-51			9.555E-2	5.75	1.29	5.60		14.69
Ni-58(n,2n)Ni-57	8.952E-3	3.57	8.985E-3	6.24	0.85	6.19	1.004 ± 0.072	15.00

Table 4 (continued)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
Ti-47(n,np)Sc-46			**1.670E-2**	6.90	2.58	6.40		15.19
Na-23(n,2n)Na-22			**8.611E-3**	8.16	3.90	7.17		15.40
Ti-48(n,np)Sc-47			**4.175E-3**	8.32	2.90	7.80		15.76
Ti-46(n,2n)Ti-45			**1.308E-2**	8.58	1.86	8.38		16.01
Ti-49(n,np)Sc-48			**2.759E-3**	13.49	10.45	8.52		16.15
Fe-57(n,np)Mn-56			**2.112E-3**	18.74	16.47	8.93		16.20

Table 5 Updated RRDF-98 Library (<σ> data in the ^{252}Cf neutron field)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
Np-237(n,f)	**1361**	1.59	**1359**	1.74	1.72	0.21	**0.999 ± 0.024**	2.06
Rh-103(n,n')Rh-103m	**[809]**	2.97	**725.1**	3.95	3.94	0.25	0.896 ± 0.044	2.38
In-115(n,n')In-115m	**197.4**	1.37	**191.8**	1.70	1.66	0.37	0.972 ± 0.021	2.67
Nb-93(n,n')Nb-93m	**[146]**	3.45	**146.1**	2.61	2.59	0.35	**1.001 ± 0.043**	2.69
Ni-58(n,p)Co-58	**117.5**	1.30	**117.5**	1.89	1.74	0.74	**1.000 ± 0.023**	4.20
Pb-204(n,n')Pb-204m	**[20.85]**	4.41	**20.39**	4.67	4.57	0.98	**0.978 ± 0.063**	5.04
Al-27(n,p)Mg-27	**4.880**	2.14	**4.912**	2.37	2.06	1.17	**1.007 ± 0.032**	6.02
Ti-46(n,p)Sc-46	**14.07**	1.77	**13.83**	3.28	3.05	1.19	**0.983 ± 0.037**	6.08
Cu-63(n,α)Co-60	**0.6887**	1.96	**0.6933**	3.15	2.83	1.39	**1.007 ± 0.037**	7.27
Fe-54(n,α)Cr-51			**1.113**	3.48	3.18	1.42		7.43
Fe-56(n,p)Mn-56	**1.465**	1.77	**1.475**	2.99	2.61	1.46	**1.007 ± 0.035**	7.56
Mg-24(n,p)Na-24	**1.996**	2.44	**2.142**	1.96	1.14	1.59	1.073 ± 0.034	8.25
Ti-48(n,p)Sc-48	**0.4247**	1.89	**0.4268**	5.32	5.08	1.57	**1.005 ± 0.057**	8.35
Co-59(n,α)Mn-56	**0.2218**	1.88	**0.2212**	3.87	3.54	1.56	**0.997 ± 0.043**	8.37
V-51(n,α)Sc-48	**3.900E-2**	2.21	**3.859E-2**	3.56	3.02	1.89	**0.989 ± 0.041**	9.98
Nb-93(n,2n)Nb-92m	**[0.749]**	5.07	**0.7717**	2.46	1.03	2.24	**1.030 ± 0.058**	11.36
Pr-141(n,2n)Pr-140			**1.990**	11.37	11.03	2.45		11.85
As-75(n,2n)As-74			**0.6209**	6.55	5.76	3.14		12.91
F-19(n,2n)F-18	**1.612E-2**	3.37	**1.627E-2**	5.33	2.92	4.46	**1.009 ± 0.064**	14.04
Ti-47(n,np)Sc-46			**1.941E-2**	9.58	7.57	5.88		14.93
Ti-48(n,np)Sc-47			**4.349E-3**	11.62	8.20	8.24		15.88
Ti-49(n,np)Sc-48			**2.644E-3**	10.84	7.18	8.13		15.96
Ti-46(n,2n)Ti-45			**1.218E-2**	9.55	4.41	8.47		16.03
Fe-54(n,2n)Fe-53			**3.498E-3**	10.71	4.87	9.54		16.48

Remarks:

a) The data of the reactions ^{93}Nb(n,n')^{93}Nb^{m}, ^{24}Mg(n,p)^{24}Na and ^{93}Nb(n,2n)^{92}Nb^{m} base on the original version of RRDF-98.

b) The calculated <σ>-value of ^{54}Fe(n,2n)^{53}Fe is incomplete with an upper limit of the neutron spectrum of 20 MeV.

Table 6 ENDF/B-VI and JEFF-3.0 Library (<σ> data in the ^{252}Cf neutron field)

Reaction	Experiment		Calculation		σ(E)	N(E)	C/E	E(50%)
	<σ> in mb	error (%)	<σ> in mb	error (%)	error (%)	error (%)		MeV
ENDF/B-VI								
Ni-58(n,p)Co-58	**117.5**	1.30	**115.3**	2.52	2.41	0.72	**0.981 ± 0.028**	4.13
Ni-60(n,p)Co-60	**[2.39]**	5.44	**2.494**	10.20	10.11	1.37	**1.044 ± 0.121**	7.10
Cu-65(n,2n)Cu-64	**0.6582**	2.22	**0.6777**	3.69	2.25	2.92	**1.030 ± 0.044**	12.65
Cu-63(n,2n)Cu-62	**0.1844**	3.98	**0.2056**	5.81	4.10	4.12	1.115 ± 0.078	13.79
Cr-52(n,2n)Cr-51			**9.841E-2**	9.83	8.11	5.56		14.69
Ni-58(n,2n)Ni-57	**8.952E-3**	3.57	**9.254E-3**	6.58	2.48	6.10	**1.034 ± 0.077**	14.96
JEFF-3.0								
Ni-58(n,p)Co-58	**117.5**	1.30	**117.1**	3.46	3.38	0.72	**0.997 ± 0.037**	4.16
Ni-60(n,p)Co-60	**[2.39]**	5.44	**2.796**	8.37	8.26	1.36	1.170 ± 0.117	6.99
Fe-56(n,p)Mn-56	**1.465**	1.77	**1.437**	1.82	1.08	1.46	**0.981 ± 0.025**	7.57
Ni-58(n,2n)Ni-57	**8.952E-3**	3.57	**9.256E-3**	6.67	2.72	6.09	**1.034 ± 0.078**	14.96

overlap in the energy response range between the various reactions. Unfolding implicitly requires the σ(E) data of the various reactions be consistent within the uncertainties quoted. If this consistency is not valid, the derived spectral fluences will show strong discontinuities which seriously hamper the result of the unfolding process.

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SELECTION OF CROSS SECTIONS IN THE THERMAL AND EPITHERMAL NEUTRON ENERGY REGION FOR THE FILE IRDF-2002, AND CHARACTERIZATION OF THE SELECTED DATA

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1. INTRODUCTION

As a part of the up-dating procedure of the International Reactor Dosimetry File, IRDF-90, the data of several reactor dosimetry files (JENDL/D-99 [1], RRDF-98[2], ENDF/B-VI (V.8), JEFF-3.0 and CENDL-2 [3]) have been analysed in order to select the best quality cross section and related uncertainty information for the new International Reactor Dosimetry File, IRDF-2002 [4-6]. As a result of the analysis [5], the evaluators reviewed numerous data in the cross section files JENDL/D-99 and RRDF-98, furthermore several new cross section evaluations have been made [7]. After the repeated investigation of the revised data and analysis of the new data, a collection was prepared for each of the libraries mentioned above, containing the reactions with cross sections suitable for inclusion in IRDF-2002 [8]. Table 1 [8] shows the list of these candidate reactions. The cross sections together with the uncertainty information in this Table, are the best quality data available in the open literature at the moment. At the same time, there are some reactions which no cross section information – among the reviewed data of the investigated libraries – applicable in the file IRDF-2002 have been found for. These reactions are also listed in Table 1.

The cross section selection procedure for the new reactor dosimetry file consisted of comparison of the appropriate integral values of these cross sections with experimental data in standard neutron fields. An other important criterion of the selection was the quality of the uncertainty information accompanying the cross sections of interest.

This paper presents the results of the cross section selection in the thermal and epithermal neutron energy regions.

2. THE SELECTION PROCEDURE AND RESULTS

Surveying the content of Table 1, it is striking that the majority of the reactions with correct cross section and uncertainty information in the thermal and epithermal neutron energy region can be found in the library IRDF-90 only. In the other considered libraries either no data of this type are available or the cross sections and/or the related uncertainty information (covariance matrices) are erroneous or incomplete [5,6].

New evaluations have been prepared in the low neutron energy region for two reactions, LA139G and W186G (missing reactions with complete uncertainty information from the dosimetry libraries so far) by the evaluators of the Russian Reactor Dosimetry File, and a serious revision was made for the covariance information of the reaction FE58G (for substitution of earlier erroneous data in JENDL/D-99) by the Japanese evaluators [7,9]

The base of the selection procedure was the comparison of the integral values of the cross sections and the related uncertainty information from the libraries of interest with each other, and with experimental data of standard neutron fields. It means that in our case experimental data in a Maxwellian thermal neutron spectrum and in an 1/E neutron field had to be considered [10]. Unfortunately, up-to date experimental data in these neutron fields can not easily be found in the literature. After an extensive literature overview, two sources were found for the purpose: the evaluated experimental data of S. F. Mughabgbab [11] and of N. E. Holden [12]. The thermal neutron cross sections in both evaluations of experiments refer to a neutron energy of 0.0253 eV (v_o=2200 m/s), while the resonance integrals were calculated by Mughabgbab with a lower energy limit of 0.5 eV and, with an upper energy limit corresponding to the upper resonance with known scattering width [13]. Holden calculated the resonance integrals from 0.5 eV to 0.1 MeV. In our calculations the thermal neutron cross sections refer to 0.0253 eV neutron energy (v_o=2200 m/s), and the resonance integrals were calculated from 0.5 eV to 1.05 MeV (see also [5] and [6]), with using a multigroup representation (SAND type 640 energy groups) of the the cross sections of interest. A similar comparison of the thermal neutron cross sections and resonance integrals with the corresponding Mughabgbab data [12] was made for a series of capture cross sections of different dosimetry libraries by A. Trkov [14]. His results agree with the ones presented in this paper.

The uncertainty information for the cross sections of interest have been represented by the corresponding relative standard deviation values (with the same energy boundaries as used in the cross section calculations), weighted with a typical MTR spectrum (see Figure 1 [15]). Here we notice, that we have discovered a shortcoming in our uncertainty processig code X333 [16], leading in some cases (when LB = 0 or LB=8 sub-sub-sections were present in the uncertainty data) to biased results. The present calculations have been performed with the corrected version of the program.

The results of the comparison can be seen in Table 2. From the data of this Table one can see that, for a few reactions the same cross section information was present both in IRDF-90 and in the other libraries in question, while the related uncertainty information was sometimes different. These type of reactions are: MN55G (IRDF-90 and JENDL/D-99) and CU63G (IRDF-90 and ENDF/B-VI). In these cases the file IRDF-90 was taken as the source of the data for IRDF-2002.

In case of the reaction FE58G the resonance integral for both cross section files of interest (IRDF-90 and JENDL/D-99) meaningfully deviates from the corresponding data of Mughabgbab, while the JENDL/D-99 value shows a good agreement (as compared with the relevant uncertainties) with the corresponding data of Holden. The situation needs further clarification, eg. by comparison with experimental data in benchmark neutron fields. Nevertheless, for the JENDL/D-99 data generally a better agreement was

found with the experimental values than for the corresponding IRDF-90 ones. Taking into consideration the corresponding uncertainty values as well, the JENDL/D-99 data seem to be more realistic, therefore, the cross section to be included in IRDF-2002 for this reaction has been selected from the file JENDL/D-99.

The other reaction in Table 2, being the subject of selection, is PU239F. As it can be seen, the cross section values for this reaction are practically the same in the considered libraries, but the uncertainties in the file JENDL/D-99 seem to be more reliable, than the corresponding IRDF-90 values. Therefore, again the JENDL/D-99 data have been selected for IRDF-2002.

3. CHARACTERIZATION OF THE CROSS SECTIONS IN THE THERMAL AND EPITHERMAL ENERGY REGION, SELECTED FOR IRDF-2002

The thermal and epithermal neutron cross sections, selected for the file IRDF-2002, are presented by Table 3. For numerical characterization of the data the thermal cross sections (σ_L) at 2200 m/s (0.0253 eV) and the resonance integral (IR_L) values from 0.5 eV to 1.05 MeV have been calculated. All the cross section and resonance integral values are compared with the evaluated experimental data of S. F. Mughabgbab [11] and of N. E. Holden [12], as described in the previous chapter.

For representation of the uncertainty information of the selected cross sections, the relative standard deviation values (weighted with an MTR spectrum) were calculated for the thermal and intermediate neutron energy regions separately, using the same energy boundaries as in case of the cross section characterization. The results can be seen in Table 4.

Evaluating the data in Tables 3 and 4, the following statements can be made:

1. The thermal neutron cross sections for the selected reactions in general agree with the evaluated experimental data within one standard deviation of the corresponding library and experimental values.
2. However, the resonance integrals calculated from the library data deviate from the evaluated experimental ones more than one standard deviation of the corresponding library and experimental values, for several reactions (details see below).
3. List of the problems by reactions, related to the data in Tables 3 and 4:

   MN55G: The C/E value for the resonance integral is deviating by 16 % from the unity. It is too large deviation also as compared with the related uncertainty values. New cross section evaluation is needed for this reaction in the intermediate neutron energy region!

   FE58G: The C/E value for the resonance integral with the Mughabghab data is deviating by 19 % from unity. At the same time, a large difference is present between the experimental data of the sources considered. Clarification of the situation is needed, as this reaction is one of the most frequently used detectors in the reactor dosimetry! Maybe also new cross section evaluation is needed in the intermediate neutron energy region!

NB93G: The C/E value for the resonance integral is deviating by 17% from unity, furthermore, the uncertainty information contains only a diagonal matrix. New evaluation is needed!

AG109G: The C/E value shows a large deviation from unity both in the thermal and in the intermediate neutron energy regions, furthermore, the evaluated cross section in the library IRDF-90 is given in a rough energy group structure. Re-evaluation of the data is needed! (For this reaction Mughabgbab gave the sum of the cross sections of reactions leading to Ag110(m+g), while the dosimetry libraries contain the cross sections for the reaction leading to Ag110m. Therefore, no comparison with the data of Mughabgbab was possible in this case.)

IN115G: The uncertainty information contains only diagonal matrix. New evaluation is needed!

TA181G: The uncertainty information contains only a diagonal matrix. New evaluation is needed!

AU197G: The available uncertainty information for this reaction is not reliable, it has been withdrawn from ENDF/B-VI. The uncertainty data in IRDF-90 are deriving from the same source. New evaluation is needed!

TH232G: In the uncertainty information below 15 eV diagonal matrix is present. New evaluation is needed!

U235F: The uncertainty information has been declared to be not reliable and has been withdrawn from ENDF/B-VI. The data in the library IRDF-90 have the same origin. New evaluation is needed!

AM241F: No up-to-date experimental data are available for this reaction, therefore, the corresponding C/E values could not be derived.

4. CONCLUSIONS

Based on the results of the cross section selection procedure outlined above, the following conclusions can be drawn, related to the data of Tables 3 and 4:

1. Practically no new cross section evaluations have been made in the low neutron energy region during the last one-two decades, except the reactions ^{139}La(n,γ) and ^{186}W(n, γ), evaluated for the Russian Reactor Dosimetry File [7].
2. In the thermal neutron energy region the selected cross sections show in most cases a very good agreement with the corresponding evaluated experimental values.
3. At the same time, the resonance integrals of the reactions MN55G, FE58G and NB93G, meaningfully (>10%) deviate from the corresponding experimental data. This deviation is too large even in comparison with the corresponding uncertainty information. Further investigations (eg. testing the data also in benchmark neutron fields) and new cross section evaluations will be needed in these cases.

4. For the reactions NB93G, IN115G, TA181G and for TH232G below 15 eV, the uncertainty information consists of diagonal covariance matrices only. New evaluations with complete covariance information are needed in these cases.
5. Unreliable uncertainty information (withdrawn from ENDF/B-VI) is present in all the investigated cross section libtraries for the reactions AU197G and U235F. Therefore, new cross section evaluations with complete covariance information are needed for these reactions!
6. The selected cross sections in Tables 3 and 4 will have to go through a consistency test as well, by comparing the relevent integral data with experimental values in benchmark neutron fields.

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Table 1. Reactions from the different libraries with cross sections suitable for IRDF-2002 [8]

REACTIONS FROM IRDF-90						
LI6T	B10A	MG24P	AL27P	AL27A	P31P	S32P
SC45G	TI46P	TI47NP	TI47P	TI48NP	TI48P	CR522
MN55G	FE54P	FE58G	CO592	CO59G	NI582	NI58P
CU632	CU63G	CU63A	CU652	ZN64P	ZR902	NB932
NB93N	RH103N	AG109G	IN1152	IN115N	I1272	AU1972
AU197G	TH232F	U235F	U238F	U238G	PU239F	
REACTIONS FROM JENDL/D-99						
F192	MG24P	AL27P	AL27A	P31P	TI0XSC46	TI0XSC48
TI462	TI46P	TI48NP	TI48P	TI49NP	CR522	MN55G
FE54P	FE58G	NI582	NI58P	CU632	CU652	Y892
ZR902	IN115N	I1272	TM1692	AU1972	HG199N	U238F
NP237F	PU239F	AM241F				
REACTIONS FROM RRDF-98						
F192	TI462	TI46P	TI47NP*	TI48NP*	TI48P	TI49NP*
V51A	FE542	FE54A	FE56P	CO59A	CU63A	AS752
NB932	NB93N	LA139G	PR1412	W186G	PB204N	
AL27P•	NI58P•	RH103N•	IN115N•			
REACTIONS FROM ENDF/B-VI (V.8)						
CR522	NI58P	NI60P	CU632	CU63G	CU652	
REACTIONS FROM JEFF-3.0						
FE56P	NI582	NI58P	NI60P			

REMARKS

* Format error

• New evaluations or updates, 2003.

No suitable cross section data were found for the reactions♣						
NA232	NA23G	TI0XSC47	CR50G	MN552	FE57NP	NB93G
IN115G	EU151G	TA181G	TH232G			

REMARK

♣ due to discrepancies in the cross section and/or in the corresponding uncertainty information

Table 2. Comparison of the cross section characteristics for some thermal and epithermal neutron reactions, with evaluated experimental data

Reaction code	Library	Calc.cross. sec. σ_L (2200 m/s) (m^2)	Rel. std. of σ_L* (%)	Calc. res. int. IR_L♣ (m^2)	Rel. std. in intermed. E region** (%)	Thermal cross sec. ratio		Res. int. ratio	
						σ_L/σ_M	σ_L/σ_H	IR_L/IR_M	IR_L/IR_H
MN55G	IRDF-90	1.34E-27	4.18	1.18E-27	3.84	1.00	1.01	0.84	0.84
MN55G	JENDL/D-99	1.34E-27	6.31	1.18E-27	8.04	1.00	1.01	0.84	0.84
FE58G	IRDF-90	1.15E-28	5.07	1.51E-28	5.12	0.88	0.88	0.89	1.16
FE58G	JENDL/D-99	1.30E-28	12.60	1.37E-28	8.75	1.00	1.00	0.81	1.05
CU63G	IRDF-90	4.48E-28	4.11	4.96E-28	3.86	0.99	1.00	1.00	0.99
CU63G	ENDF/B-VI	4.48E-28	4.11	4.95E-28	3.86	0.99	1.00	1.00	0.99
PU239F	IRDF-90	7.48E-26	0.25	2.93E-26	0.26	---	0.99	---	0.98
PU239F	JENDL/D-99	7.47E-26	0.71	2.97E-26	3.82	---	0.99	---	0.99

REMARKS

σ_L and IR_L are calculated values from the corresponding library data.

* Calculated for a typical MTR spectrum from 1E-4 eV to 0.5 eV.

** Calculated for a typical MTR spectrum from 0.5 eV to 1.05 MeV.

.♣ Calculated from 0.5 eV to 1.05 MeV.

σ_M and IR_M are evaluated experimental data of S.F. Mughabghab [11].

σ_H and IR_H are evaluated experimental data of N.E. Holden [12].

Table 3. Thermal neutron cross sections and resonance integrals for the reactions, selected for IRDF-2002

Reaction code	Mat No.	MT No.	Library, source of selection	Libr.cross sec.at 2200 m/s σ_L (m^2)	Evaluated exp. data (2200 m/s) σ_M (m^2)	Evaluated exp. data (2200 m/s) σ_H (m^2)	Cross sec. ratio σ_L/σ_M	Cross sec. ratio σ_L/σ_H	Res. Int. from libr.data IR_L (m^2)	Evaluated res.integral IR_M (m^2)	Evaluated res.integral IR_H (m^2)	Res. int. ratio IR_L/IR_M	Res. int. ratio IR_L/IR_H
LI6T	325	105	IRDF-90	9.41E-26	Not available	9.4(.1)E-26	---	1.00	4.25E-26	Not available	4.22(.04)E-26	---	1.01
B10A	525	107	IRDF-90	3.84E-25	Not available	3.84(.01)E-25	---	1.00	1.72E-25	Not available	1.73(.01)E-25	---	0.99
SC45G	2126	102	IRDF-90	2.72E-27	2.72(.02)E-27	2.7E-27	1.00	1.01	1.19E-27	1.20(.05)E-27	1.20E-27	0.99	0.99
MN55G	2525	102	IRDF-90	1.34E-27	1.336(.005)E-27	1.33(.01)E-27	1.00	1.01	1.18E-27	1.40(.03)E-27	1.40(3)E-27	0.84	0.84
FE58G	2637	102	JENDL/D-99u	1.30E-28	1.30(.03)E-28	1.3(.1)E-28	1.00	1.00	1.37E-28	1.7(.1)E-28	1.3(.2)E-28	0.81	1.05
CO59G	2725	102	IRDF-90	3.73E-27	3.718(.006)E-27	3.72E-27	1.00	1.00	7.45E-27	7.59(.02)E-27	7.4E-27	0.98	1.01
CU63G	2925	102	IRDF-90	4.48E-28	4.52(.02)E-28	4.5(.2)E-28	0.99	1.00	4.96E-28	4.97(.08)E-28	5.0(1)E-28	1.00	0.99
NB93G+	4125	102	IRDF-90	1.16E-28	1.15(.05)E-28	1.1E-28	1.01	1.05	9.92E-28	8.5(.5)E-28	8.5E-28	1.17	1.17
AG109G	4731	102	IRDF-90	4.69E-28	---	4.2E-28	----	1.12	6.56E-27	---	7.0E-27	---	0.94
IN115G*+	4931	102	IRDF-90	2.11E-26	2.02(.02)E-26	2.05E-26	1.04	1.03	3.28E-25	3.3(.1)E-25	3.4E-25	0.99	0.96
LA139G	5712	102	RRDF-98 u	8.89E-28	9.04(.04)E-28	9.2(.2)E-28	0.98	0.97	1.19E-27	1.21(.06)E-27	1.2(.1)E-27	0.98	0.99
TA181G+	7328	102	JENDL/D-99	2.07E-27	2.05(.05)E-27	2.01E-27	1.01	1.04	6.59E-26	6.6(.23)E-26	6.504E-26	1.00	1.01
W186G	7452	102	RRDF-98 u	3.79E-27	3.85(.05)E-27	3.7(.2)E-27	0.98	1.02	4.79E-26	4.85(.15)E-26	5.10(.50)E-26	0.99	0.94
AU197G♣	7925	102	IRDF-90	9.89E-27	9.865(.09)E-27	9.87(.1)E-27	1.00	1.00	1.57E-25	1.55(.028)E-25	1.55(.03)E-25	1.01	1.01
TH232G♦	9040	102	IRDF-90	7.40E-28	7.35(.03)-28	7.37(.04)-28	1.01	1.00	8.57E-27	8.5(.3)E-27	8.5(.3)E-27	1.01	1.01
U235F♥	9228	18	IRDF-90	5.86E-26	Not available	5.86(.02)E-26	---	1.00	2.74E-26	Not available	2.75(.05)E-26	---	1.00
U238G	9237	102	IRDF-90	2.71E-28	2.68(.019)E-28	2.7(.1)E-28	1.01	1.00	2.77E-26	2.77(.03)E-26	2.77(.03)E-26	1.00	1.00
PU239F	9437	18	JENDL/D-99	7.47E-26	Not available	7.52(.03)E-26	---	0.99	2.97E-26	Not available	3.0(.1)E-26	---	0.99
AM241F	9543	18	JENDL/D-99	3.03E-28	Not available	3.15(.1)E-28	---	0.99	7.84E-28	Not available	Not available	---	---

REMARKS

At the evaluated experimental cross section data the values in brackets mean the absolute uncertainties.

\+ Diagonal matrix

♣ For the reaction AU197G the uncertainty information has been withdrawn from ENDF/B-VI (similar old evaluation is present in IRDF-90) .

♦ For the reaction TH232G below 15 eV diagonal matrix is present.

♥ The uncertainty information for the reaction U235F is not reliable, it has been withdrawn from ENDF/B-VI.

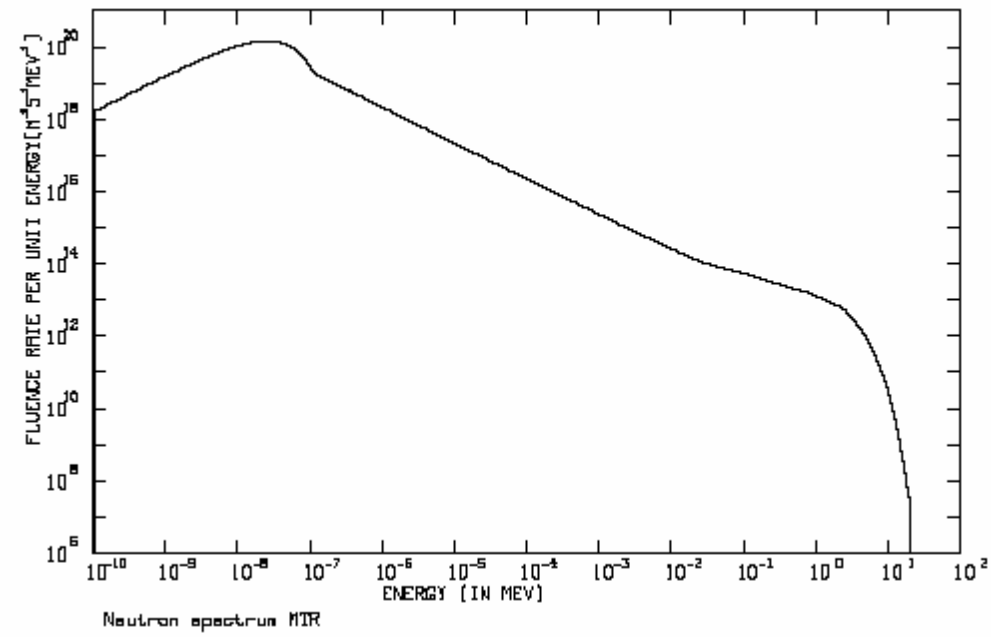
u Means up-dated data. The subscripts L, M and H mean library data, and the evaluated experimental data of S.F. Mughabghab [11] and N.E. Holden [12], respectively.

Table 4. Relative standard deviation values averaged over a typical MTR spectrum for the reactions in Table 3, selected for IRDF-2002.

Reaction code	Library, source of selection	Mat.MT No.	Relative std. (%) For the spectrum part	
			Thermal°	Intermediate*
LI6T	IRDF-90	0325.105	0.14	0.14
B10A	IRDF-90	0525.107	0.16	0.16
SC45G	IRDF-90	2126.102	0.73	0.76
MN55G	IRDF-90	2525.102	4.18	3.84
FE58G	JENDL/D-99u	2637.102	12.60	8.75
CO59G	IRDF-90	2725.102	0.66	0.77
CU63G	IRDF-90	2925.102	4.11	3.86
NB93G+	IRDF-90	4125.102	10.00	9.49
AG109G	IRDF-90	4731.102	5.10	6.90
IN115G+	IRDF-90	4931.102	6.00	5.98
LA139G	RRDF-98 u	5712.102	3.87	5.50
TA181G+	JENDL/D-99	7328.102	3.00	3.77
W186G	RRDF-98 u	7452.102	2.31	3.32
AU197G♣	IRDF-90	7925.102	0.14	0.17
TH232G♦	IRDF-90	9040.102	4.33	10.92
U235F♥	IRDF-90	9228.018	0.19	0.27
U238G	IRDF-90	9237.102	0.35	0.37
PU239F	JENDL/D-99	9437.018	0.71	3.82
AM241F	JENDL/D-99	9543.018	2.00	1.56

REMARKS

° From 1E-4 eV to 0.5 eV.
* From 0.5 eV to 1.05 MeV.
u Means up-dated data.
+ Diagonal matrix
♣ For the reaction AU197G the uncertainty information has been withdrawn from ENDF/B-VI (similar old evaluation is present in IRDF-90) .
♦ For the reaction TH232G below 15 eV diagonal matrix is present.
♥ The uncertainty information for the reaction U235F is not reliable, it has been withdrawn from ENDF/B-VI.



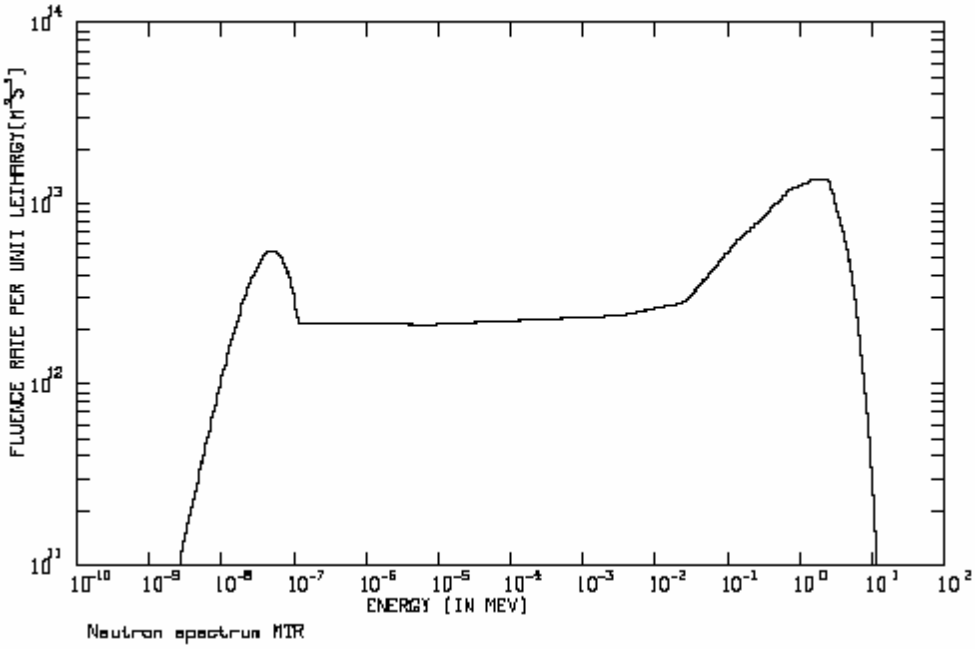


Figure 1. Neutron spectrum MTR – in two different representations – used in the calculations ([15]).

SELECTION OF FAST NEUTRON REACTION CROSS SECTIONS FOR THE FILE IRDF-2002 AND CHARACTERIZATION OF THE SELECTED DATA

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1. INTRODUCTION

Updating of the International Reactor Dosimetry File IRDF-90 [1] has been in progress with the coordination of IAEA NDS [2]. The new library will be named IRDF-2002. Analysis of the data of two new national dosimetry files JENDL/D-99 [3] and RRDF-98 [4], furthermore, new evaluations from the files ENDF/B-VI (V.8), JEFF-3.0 and CENDL-2 [3]) has been performed in order to select the best quality cross section and uncertainty information for IRDF-2002. The results, together with the detected errors and discrepancies were presented in the form of Progress Reports [5-7], and communicated to the evaluators of the libraries via IAEA. The evaluators have then revised and modified several data, furthermore, also a number of new evaluations have been prepared [8,9].

Table 1 (from [7]) shows the list of the reactions – for each of the libraries – having cross sections suitable for inclusion in IRDF-2002. The cross sections together with the uncertainty information in this Table, are the best quality data available in the open literature at the moment. For characterizing and comparing the cross section data of the fast neutron (threshold) reactions present in Table 1, spectrum averaged cross sections were calculated for the theoretical function of the Watt fission spectrum [5,6], with using a multigroup representation (SAND type 640 energy groups) of the the cross sections of interest. The uncertainty information for the cross sections was represented by the corresponding standard deviation values above 1.05 MeV, weighted with a typical MTR spectrum [10]. The results obtained can be seen in Table 2. Here we notice, that we have discovered a shortcoming in our uncertainty processig code X333 [11], leading in some cases (when LB = 0 or LB=8 sub-sub-sections were present in the uncertainty data) to biased results. The present calculations have been performed with the corrected version of the program.

W. Mannhart calculated the responses of activation reactions in the standard neutron field of spontaneous fission of ^{252}Cf and, compared them with experimental data obtained in that neutron field [12]. Spectrum averaged cross sections were calculated together with the related standard deviations, and C/E values were derived. Also qualification of the considered cross section information was given. He investigated the cross section data of the files IRDF-90.v2, JENDL/D-99, and an updated version of RRDF-98, furthermore, the ones of the reactions selected in Table 1 from ENDF/B-VI and JEFF-3.0. His results – with consideration of the differences between the two spectrum functions – show a good agreement with the data in Table 2.

This paper presents a recommendation on the fast neutron cross sections to be included in the file IRDF-2002, and characterizes the selected data.

2. RECOMMENDED FAST NEUTRON CROSS SECTIONS FOR IRDF-2002

The cross section selection procedure for the new reactor dosimetry file consisted in one side, of comparison of the appropriate integral values of the cross sections in Table 1, with experimental data in standard neutron fields [7,13]. Another important criterion of the selection was the quality of the uncertainty information accompanying the cross sections of interest.

In the fast neutron energy region the applicable standard neutron field is the one of the spontaneous fission of ^{252}Cf. Therefore, our earlier findings [5,6] were combined with the ones of W. Mannhart [12], and the procedure resulted in the data of Table 3. This Table contains the list of the 46 fast neutron cross sections recommended to be included in IRDF-2002, together with their characteristics, taken from [12]. The column with the uncertainty values of the calculated average cross sections ($<\sigma_c>$) shows the standard deviation values in $<\sigma_c>$ due to the cross sections, while the values in brackets give the total standard deviation of $<\sigma_c>$, including the contribution of the uncertainty of the ^{252}Cf spectrum function as well. The uncertainty of the C/E values involves the standard deviations present in the experimental data, in the cross sections of interest and, in the ^{252}Cf spectrum function. So they can be calculated based on the data of column 4 in brackets and, on the data of column 6.

The following conclusions can be drawn regarding the cross section values of this Table:

1. No up-to-date experimental cross section data were available for the reactions P31P, TI0XSC46, TI0XSC48, TI462, TI47NP, TI48NP, TI49NP, CR522, FE542, FE54A, AS752, Y892, IN1152 and PR1412. Therefore, the corresponding C/E values could not be derived. At the same time, a large deviation was found between the cross section values of the reaction TI47NP in the libraries RRDF-98 and IRDF-90. Clarification of the situation is needed!
2. C/E values larger than 5 per cent are present for the following reactions: CU632, RH103N, I1272, TM1692, HG199N, and TH232F. The relatively large deviation between the experimental and calculated cross section values originates from the side of the library cross sections, except the reaction TM1692, where the measured cross section has a large (~ 6%) uncertainty. Improvement of the situation would be useful!
3. Inconsistency is present between the C/E value and the corresponding uncertainties for the reaction MG24P. Clarification of the situation is needed!
4. The uncertainty of the cross sections in Table 3 is in most cases below 4 %. Larger uncertainty values are present in case of the ractions TI47NP , TI48NP, TI48P, TI49NP, NI60P, AS752, PR1412, HG199N, TH232F. During the neutron spectrum adjustment procedure in the energy regions with responses of more detectors, these reactions will have a much smaller weight than the other ones with meaningfully smaller uncertainties.
5. Format problems are still present in case of the reactions TI47NP, TI48NP and TI49NP. They have to be corrected!
6. Zolotarev et al. prepared a new evaluation for the cross section of the reaction AL27P [9]. The data were communicated to IAEA in September, too late to be included in this selection procedure. It will be analysed and taken into consideration in the final content of the library IRDF-2002.

7. The selected cross sections in Table 3 will have to go through a consistency test as well, by comparing the relevent integral data with experimental values in benchmark neutron fields.

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Table 1. Reactions from the different libraries with cross sections suitable for IRDF-2002 [7]

REACTIONS FROM IRDF-90						
LI6T	B10A	MG24P	AL27P	AL27A	P31P	S32P
SC45G	TI46P	TI47NP	TI47P	TI48NP	TI48P	CR522
MN55G	FE54P	FE58G	CO592	CO59G	NI582	NI58P
CU632	CU63G	CU63A	CU652	ZN64P	ZR902	NB932
NB93N	RH103N	AG109G	IN1152	IN115N	I1272	AU1972
AU197G	TH232F	U235F	U238F	U238G	PU239F	
REACTIONS FROM JENDL/D-99						
F192	MG24P	AL27P	AL27A	P31P	TI0XSC46	TI0XSC48
TI462	TI46P	TI48NP	TI48P	TI49NP	CR522	MN55G
FE54P	FE58G	NI582	NI58P	CU632	CU652	Y892
ZR902	IN115N	I1272	TM1692	AU1972	HG199N	U238F
NP237F	PU239F	AM241F				
REACTIONS FROM RRDF-98						
F192	TI462	TI46P	TI47NP*	TI48NP*	TI48P	TI49NP*
V51A	FE542	FE54A	FE56P	CO59A	CU63A	AS752
NB932	NB93N	LA139G	PR1412	W186G	PB204N	
AL27P•	NI58P•	RH103N•	IN115N•			
REACTIONS FROM ENDF/B-VI (V.8)						
CR522	NI58P	NI60P	CU632	CU63G	CU652	
REACTIONS FROM JEFF-3.0						
FE56P	NI582	NI58P	NI60P			

REMARKS

* Format error

• New evaluations or updates, 2003.

No suitable cross section data were found for the reactions♣						
NA232	NA23G	TI0XSC47	CR50G	MN552	FE57NP	NB93G
IN115G	EU151G	TA181G	TH232G			

REMARK

♣ due to discrepancies in the cross section and/or in the corresponding uncertainty information

Table 2. Comparison of the cross section characteristics for the fast neutron reactions, candidates for IRDF-2002

Reaction code	Library	Cross. sec. $<\sigma_f>$ (m^2)	Rel. std. of $<\sigma_f>$* (%)
F192	JENDL/D-99	6.773E-34	2.92
F192	RRDF-98(u)	5.855E-34	3.02
MG24P	IRDF-90	1.473E-31	2.26
MG24P	JENDL/D-99	1.488E-31	1.24
AL27P	IRDF-90	3.825E-31	3.31
AL27P	JENDL/D-99	4.224E-31	0.72
AL27P	RRDF-98(new)	3.980E-31	2.06
AL27A	IRDF-90	6.860E-32	1.37
AL27A	JENDL/D-99	6.860E-32	1.37
P31P	IRDF-90	2.783E-30	3.60
P31P	JENDL/D-99	2.938E-30	1.34
S32P	IRDF-90	6.345E-30	3.54
TI0XSC46	JENDL/D-99	9.117E-32	2.28
TI0XSC48	JENDL/D-99 (u)	1.971E-32	2.10
TI462	JENDL/D-99	3.621E-34	1.84
TI462	RRDF-98(u)	3.359E-34	4.40
TI46P	IRDF-90	1.002E-30	2.43
TI46P	JENDL/D-99	1.105E-30	2.27
TI46P	RRDF-98(u)	1.118E-30	3.13
TI47NP	IRDF-90	7.958E-34	30.00
TI47NP	RRDF-98(u)	6.380E-34	8.53
TI47P	IRDF-90	1.760E-30	3.69
TI48NP	IRDF-90	1.302E-34	30.00
TI48NP	JENDL/D-99	1.235E-34	2.65
TI48NP	RRDF-98 (u)	1.264E-34	8.59
TI48P	IRDF-90	2.596E-32	2.54
TI48P	JENDL/D-99	2.673E-32	1.85
TI48P	RRDF-98(u)	2.878E-32	5.17
TI49NP	JENDL/D-99	7.668E-35	10.01
TI49NP	RRDF-98(u)	7.657E-35	7.31
V51A	RRDF-98(u)	2.231E-33	3.13
CR522	IRDF-90	3.194E-33	2.68
CR522	JENDL/D-99	3.149E-33	1.29
CR522	ENDF/B-VI	3.248E-33	8.09
FE542	RRDF-98(u)	9.138E-35	4.96
FE54A	RRDF-98(u)	8.122E-32	3.28
FE54P	IRDF-90	7.880E-30	2.13
FE54P	JENDL/D-99 (u)	7.955E-30	0.99
FE56P	RRDF-98(u)	1.022E-31	2.62
CO592	IRDF-90	1.719E-32	2.85
CO59A	RRDF-98(u)	1.498E-32	3.76
NI582	IRDF-90	2.947E-34	3.11
NI582	JENDL/D-99	2.850E-34	0.90
NI582	JEFF-3.0	2.946E-34	2.75
NI58P	IRDF-90	1.038E-29	2.20
NI58P	JENDL/D-99	1.029E-29	0.61
NI58P	RRDF-98(new)	1.055E-29	1.73
NI58P	ENDF/B-VI	1.038E-29	2.45
NI58P	JEFF-3.0	1.054E-29	3.56

NI60P	ENDF/B-VI	1.867E-31	10.15
NI60P	JEFF-3.0	2.111E-31	8.83
CU632	IRDF-90	7.738E-33	1.75
CU632	JENDL/D-99(u)	7.877E-33	1.36
CU632	ENDF/B-VI	7.608E-33	4.43
CU63A	IRDF-90	5.017E-32	2.34
CU63A	RRDF-98(u)	5.128E-32	2.84
CU652	IRDF-90	2.894E-32	1.84
CU652	JENDL/D-99 (u)	3.024E-32	0.92
CU652	ENDF/B-VI	2.894E-32	2.31
ZN64P	IRDF-90	3.774E-30	4.80
AS752	RRDF-98(u)	2.562E-32	6.12
Y892	JENDL/D-99	1.255E-32	1.45
ZR902	IRDF-90	7.536E-33	1.60
ZR902	JENDL/D-99	7.355E-33	0.55
NB932	IRDF-90	3.878E-32	2.80
NB932	RRDF-98	3.839E-32	1.06
NB93N	IRDF-90	1.376E-29	3.01
NB93N	RRDF-98	1.410E-29	2.80
RH103N	IRDF-90	6.968E-29	3.01
RH103N	RRDF-98(nu)	7.061E-29	3.95
IN1152	IRDF-90	7.535E-32	3.57
IN115N	IRDF-90	1.828E-29	2.18
IN115N	JENDL/D-99	1.828E-29	2.18
IN115N	RRDF-98(nu)	1.848E-29	1.71
I1272	IRDF-90	1.045E-31	2.53
I1272	JENDL/D-99	1.090E-31	3.09
PR1412	RRDF-98(u)	9.328E-32	11.68
TM1692	JENDL/D-99	3.458E-31	2.33
AU1972	IRDF-90	3.112E-31	4.28
AU1972	JENDL/D-99	3.140E-31	1.18
HG199N	JENDL/D-99 (u)	2.354E-29	8.08
PB204N	RRDF-98(new)	1.744E-30	4.64
TH232F	IRDF-90	7.372E-30	5.18
U238F	IRDF-90	2.997E-29	0.54
U238F	JENDL/D-99	3.034E-29	2.09

REMARKS

$<\sigma_f>$ Cross section, averaged over the Watt fission spectrum.

* Weighted with a typical MTR spectrum from 1.05 MeV to 20.MeV.

u – update

nu – new update

new – new evaluation

Table 3. Cross section characteristics and C/E values in the ^{252}Cf fission neutron spectrum, for the fast neutron reactions selected for IRDF-2002*

Reaction code	Library	Calc.cross. sec.<σ_c> (mb)	Uncertainty in <σ_c> (%)	Exp. value <σ_e> (mb)	Uncertainty in <σ_e> (%)	C/E
F192	RRDF-98(u)	1.627E-2	2.92 (5.33)	1.612E-2	3.37	1.009±0.064
MG24P	IRDF-90	2.160	2.24 (2.75)	1.996	2.44	1.082±0.040
AL27P	IRDF-90	4.674	3.24 (3.44)	4.880	2.14	0.958±0.039
AL27A	IRDF-90	1.038	1.36 (2.12)	1.016	1.47	1.022±0.026
P31P	JENDL/D-99	32.24	1.40 (1.55)	not available	---	---
S32P	IRDF-90	70.30	3.60 (3.67)	72.54	3.49	0.969±0.049
TI0XSC46	JENDL/D-99	No infor-	mation	is	avail-	able!
TI0XSC48	JENDL/D-99 (u)	No infor-	mation	is	avail-	able!
TI462	JENDL/D-99	1.308E-2	1.86 (8.58)	not available	---	---
TI46P	RRDF-98(u)	13.83	3.05 (3.28)	14.07	1.77	0.983±0.037
TI47NP	RRDF-98(u)	1.941E-2	7.57 (9.58)	not available	---	---
TI47P	IRDF-90	19.38	3.78 (3.83)	19.27	1.66	1.006±0.042
TI48NP	RRDF-98 (u)	4.359E-3	8.20 (11.62)	not available	---	---
TI48P	RRDF-98(u)	0.4268	5.08 (5.32)	0.4247	1.89	1.005±0.057
TI49NP	RRDF-98(u)	2.644E-3	7.18 (10.84)	not available	---	---
V51A	RRDF-98(u)	3.859E-2	3.02 (3.56)	3.900E-2	2.21	0.989±0.041
CR522	JENDL/D-99	9.555E-2	1.29 (5.75)	not available	---	---
FE542	RRDF-98(u)	3.498E-3	4.87 (10.71)	not available	---	---
FE54A	RRDF-98(u)	1.113	3.18 (3.48)	not available	---	---
FE54P	IRDF-90	88.16	2.09 (2.23)	86.84	1.34	1.015±0.026
FE56P	RRDF-98(u)	1.475	2.61 (2.99)	1.465	1.77	1.007±0.035
CO592	IRDF-90	0.4228	2.67 (4.20)	0.405	2.51	1.044±0.051
CO59A	RRDF-98(u)	0.2212	3.54 (3.87)	0.2218	1.88	0.997±0.043
NI582	JENDL/D-99	8.985E-3	0.85 (6.24)	8.952E-3	3.57	1.004±0.072
NI58P	RRDF-98(new)	117.5	1.74 (1.89)	117.5	1.30	1.000±0.023
NI60P	ENDF/B-VI	2.494	10.11 (10.20)	(2.39)	5.44	1.044±0.121
CU632	ENDF/B-VI	0.2056	4.10 (4.81)	0.1844	3.98	1.115±0.078
CU63A	RRDF-98(u)	0.6933	2.83 (3.15)	0.6887	1.96	1.007±0.037
CU652	ENDF/B-VI	0.6777	2.25 (3.69)	0.6582	2.22	1.030±0.044
ZN64P	IRDF-90	42.10	4.87 (4.93)	40.59	1.65	1.037±0.054
AS752	RRDF-98(u)	0.6209	5.76 (6.55)	not available	---	---
Y892	JENDL/D-99	0.344	1.40 (4.47)	not available	---	---
ZR902	IRDF-90	0.2212	1.57 (5.31)	0.2210	2.89	1.001±0.061
NB932	RRDF-98	0.7717	1.03 (2.46)	(0.749)	5.07	1.030±0.058
NB93N	RRDF-98	146.1	2.59 (2.61)	(146)	3.45	1.001±0.043
RH103N	RRDF-98(nu)	725.1	3.94 (3.95)	(809)	2.97	0.896±0.044
IN1152	IRDF-90	1.586	3.23 (4.02)	not available	---	---
IN115N	RRDF-98(nu)	191.8	1.66 (1.70)	197.4	1.37	0.972±0.021
I1272	IRDF-90	2.197	2.28 (3.30)	2.069	2.73	1.062±0.045
PR1412	RRDF-98(u)	1.990	11.03 (11.37)	not available	---	---
TM1692	JENDL/D-99	6.233	2.26 (3.01)	(6.69)	6.28	0.932±0.065
AU1972	IRDF-90	5.747	4.19 (4.65)	5.506	1.83	1.044±0.052
HG199N	JENDL/D-99 (u)	248.6	7.82 (7.83)	298.4	1.81	0.833±0.067
PB204N	RRDF-98(u)	20.39	4.57 (4.67)	(20.58)	4.41	0.978±0.063
TH232F	IRDF-90	78.55	5.09 (5.11)	(89.4)	3.02	0.879±0.052
U238F	JENDL/D-99	319.2	2.00 (2.04)	325.7	1.64	0.980±0.026

Remarks see on the next page

REMARKS

* Data taken from [12]

$<\sigma_c>$ – calculated cross section, averaged over the ^{252}Cf fission neutron spectrum

$<\sigma_e>$ – experimental value, ^{252}Cf fission spectrum-averaged cross section

u – update

nu – new update

new – new evaluation

"Experimental data given in brackets are from single experiments which were not part of the evaluation process" – information from [12]

Decay Data and Isotopic Abundances for 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. The main experimental method relies on the measurement of some of the radiations emitted by the radionuclides produced by the neutron irradiation. So, it was decided to complete the new IRDF-2002 library with an evaluated decay data section containing all the necessary data to reduce and process the experimental results.

This Chapter explains the successive steps followed between the basic data as they are given in the ENSDF [1] library and the final results (in ENDF [2] format), and whose the most significative part is presented in Appendix 5. Also, some recommandations are made for the use of a recent evaluation of the isotopic abundances.

8.1 Decay Data

8.1.1 Which radionuclides?

Chapter 2 of this report selects the target elements and the nuclear reactions for which cross sections are given in the present IRDF-2002 library. From this a first list of radionuclides which have to be considered in the new decay data part of the library was established. In addition, the fission channel is characterized by the selected fission products [3] given in the following Table.

^{95}Zr+^{95}Nb	^{97}Zr+^{97}Nb	^{103}Ru	^{106}Ru+^{106}Rh	^{131}I	^{132}Te+^{132}I
^{137}Cs+^{137}Ba^{m}	^{140}Ba+^{140}La	^{141}Ce	^{143}Ce+^{143}Pr	^{144}Ce+^{144}Pr	

Finally, the list of radionuclides is completed by the intermediate radionuclides necessary to reach the stability valley. So, the decay data part of the library contains a total of 85 radionuclides: 58 of them are ground-states (of which 7 have two decay modes), 25 are first isomeric states (of which 8 have two decay modes) and 2 are second isomeric states (^{116}In^{n} and ^{196}Au^{n}).

8.1.2 Which type of data are needed?

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

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

8.1.3 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 data organization 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.

8.1.4 Processing of the data

The decay data existing under the ENSDF format have to be converted into the more usual ENDF format already used in the first part of the library for the cross sections. The conversion from one format to the other is achieved by using the SDF2NDF code [4]. This code derives from the version 5.5 of the code RADLST [5] 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.

8.1.5 Control of the data

Besides the format conversion itself, the code SDF2NDF also performs a lot of physical checks 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...

8.1.6 Results

The most intense emitted radiations are presented in Appendix 5 and some explanations are given in the header of the table.

Nine radio nuclides (^{95}Zr, ^{97}Zr, ^{103}Ru, ^{106}Ru, ^{116}In^{m}, ^{131}I, ^{132}Te, ^{137}Cs, ^{144}Ce), have a decay branch leading to a daughter nucleus with an isomeric state. In these cases the total decay intensity in this mode is given together with the fraction of the decay feeding the ground- and the isomeric-states.

In addition to the partials results of Appendix 5, the complete decay section (in ENDF format) is contained in the CD version of the IRDF-2002 library.

For about ¼ of the radio nuclides considered here, the main gamma-rays have received special attention in the frame of a CRP [6] and the users desiring higher fidelity nuclear data for isotopes used as detector calibration standards should consult the final document of the *X-ray and gamma-ray Decay Data Standards for the Detector Efficiency calibration and Other Applications* CRP [6].

8.2 Isotopic abundances

The isotopic composition of the elements is a very important quantity because it is directly related to the number of nuclei irradiated by the neutron flux. Three major evaluations of the isotopic composition were published in the last ten years [7], [8], [9].

These three tables give very similar values of the isotopic abundance for the 287 stable isotopes, except the four isotopes given in the following Table for which the deviation exceeds 1 %.

^{2}H	3.04 %	^{124}Xe	1.11 %	^{187}Os	-1.84 %	^{190}Pt	-2.86 %

Needless to repeat that Technetium (Z=43) and Promethium (Z=61) do not have any stable isotope.

In [9] the ^{231}Pa is given with a 100 % abundance which is certainly an error due to the fact that this nucleus has a rather short half-life ($T_{1/2}$ = 32760 years).

Despite this remark, we encourage the use of this recent compilation, more precisely the values given in column 9 of the main Table in [9]. The numerical data can be found at the Web address given in [9]

Conclusions

For many radionuclides involved in reactor dosimetry applications we propose recent values for the decay data. These data originate from the ENSDF library and are then transformed, checked, and given in the ENDF format. They complete the cross section part of the IRDF-2002 library.
We also recommend the use of the isotopic abundances which are given in [9].

Table 2. ENSDF format example (^{60}Co β^- decay) showing the close connection between the physical quantities and the data structure (L is used for level description, B for branching, G for gamma-ray...). Comments records are suppressed for clarity purpose. Arrows on the right, not part of the format, indicate the two well-known γ-rays.

```
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. For clarity purpose only two sections are given. The two underlined numbers are the energies (in eV) of the two well-known γ-rays.

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
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
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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`http://physics.nist.gov/PhysRefData/Compositions`.

Selection of ^{23}Na(n,γ) Dosimetry Cross Sections for the IRDF-2002 Library

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I. PURPOSE

The proposed candidates for the IRDF-2002 library do not include any recommendations for the ^{23}Na(n,γ) reaction. Since this is a very important reaction for many users of the library, it is suggested that the IAEA Data Development Project members examine the potential candidate cross section evaluations in more detail and make a recommendation or explicitly state that dosimetry users should not use this reaction in any dosimetry application.

In support of this task, this paper presents a solid quantitative methodology for selecting among the candidate cross sections. The dosimetry cross section selection procedure has been a controversial topic for many years. It was addressed as early as 1976 in an IAEA-sponsored International Consultant's meeting [3] and was recently the focus of a special workshop at the 11th International Symposium on Reactor Dosimetry (ISRD11) [4]. This paper builds on the international consensus that resulted from this recent workshop and proposes a concrete structure for the selection process that is compliant with the ISRD11 workshop recommendations and applies the process to the selection procedure to the recommendation for the ^{23}Na(n,γ)^{24}Na reaction.

Table 1: Available Dosimetry Cross Section Libraries

Library	Comment	Year
ENDF/B-VI, Release 8	329 evaluations	Orig. 1991 Oct. 2001, Rel. 8
IRDF-90	44 partial evaluations for reactor dosimetry.	1990 Oct. 1993, Rev. 1
GLUCS	ORNL dosimetry set, 14 reactions with cross isotope correlations, source for new ENDF/B-VI, update covariance in 1993.	1990 1993
JENDL-3.2	318 nuclide Japanese Evaluated Library.	1994
JENDL-3.3	337/211 nuclide Japanese Evaluated Library	2002
JENDL-D99	47 nuclide 67 reaction JENDL Dosimetry Library.	1999
JEF-2.2	313 nuclide Joint Evaluated File by NEA.	1992
JEFF 3.0	350 nuclide Joint Evaluated Fission and Fusion File by OECD/NEA, units EFF and JEF efforts	Org. 1997 2002
BROND 2.2	121 nuclide Russian evaluated cross sections.	Orig. 1987 1997, Rev. 2.2 BROND-3 pending
CENDL-2	50 nuclide Chinese Evaluated Nuclear Data Library.	1992 CENDL-3 pending
RRDF	Russian Dosimetry File	1998
RNAL	Reference Neutron Activation File	255 reactions for fusion applications
FENDL2- Dos	Fusion Evaluated Nuclear Data Library, dosimetry sublibrary. Adopted IRDF-90	1995
FENDL2-Act	636 nuclide Fusion Evaluated Nuclear Data Library, activation sublibrary	1995

II. SELECTION PROCESS

Five steps constitute the proposed selection procedure. These steps include: culling, evaluating, comparing, recommending, and evaluating. The following sections detail these five steps and apply them to an example application for the $^{23}Na(n,\gamma)^{24}Na$ reaction.

A. *Culling Candidate Evaluations*

The first step in recommending a cross section evaluation is to survey the population of candidate evaluations. Fourteen cross section libraries, listed in Table 1, are considered in this paper. From these libraries, twelve candidate evaluations were available for the $^{23}Na(n,\gamma)$ reaction. There are two minimal requirements for a high quality dosimetry cross section. If any of these elements are missing, then the evaluation is removed/culled from further consideration. These requirements include:

- The file represents a point cross section evaluation and is available in the ENDF-6 format specification.
- The evaluation includes a covariance matrix (ENDF-6 File 33) for the reaction of interest. This covariance matrix must be associated with the actual evaluation and cannot merely be "inherited" from a prior evaluation.

Of the original twelve $^{23}Na(n,\gamma)$ candidate evaluations, when the go/no-go requirements are applied, 7 evaluations are eliminated for lack of a covariance matrix. This leaves 5 candidate evaluations that are shaded in Table 2.

Table 2: Candidate Evaluations for $^{23}Na(n,\gamma)$ Reaction

$^{23}Na(n,\gamma)^{24}Na$			Xsec. and Contrib. to Spct- Avg. Unc.				Comment
Library	Cov .	Orig. Year/ Rev. Year	2200 m/s (b)	1/E (b)	^{252}Cf (mb)	14- MeV (mb)	
ENDF/B- VI	Yes	1977/ 2000	0.5281	0.3166	0.2721	0.2247	Rel. 8
JENDL- D99	Yes/ Invalid	1996/ 1998	0.5330	0.3121	0.2238	0.1342	from JENDL 3.2, cov. taken from IRDF-85
JENDL- 3.2	No	1987/ 1989	0.5315	0.3117	0.2243	0.1343	reac. calc. with CASTHY code
JENDL- 3.3	Yes	2000	0.5316	0.3119	0.2796	0.1900	Cov based on expt. data, same as JENDL 3.2
FENDL- Dos.	Yes	1990	0.5282	0.3167	0.2685	0.2354	CNDC eval., used in IRDF-90
IRDF90	Yes	1990	0.5282	0.3165	0.2681	0.2353	CNDC, same as FENDL-Dos
JEF 2.2	No	1987/ 1992	0.5316	0.3117	0.2243	0.1343	taken from JENDL-3
JEFF 3.0	No	1999	0.5316	0.3117	0.2243	0.1342	From JEF 2.2
CENDL- 2	No	1983/ 1994	0.5315	0.3117	0.2387	0.2351	JENDL-3 resonance params.
BROND 2.2	No	1978/ 1990	0.5300	0.3242	0.2692	0.2179	GRUKON code
GLUCS	NA						
RRDF	NA						
RNAL	No	2000	0.5318	0.3117	0.2243	0.1342	Reac. from EAF- 4.1' grnd and meta. states
FENDL- Act.	No	1996	0.5318	0.3117	0.2243	0.1342	Reac. from JEF 2.2; grnd and meta. states

The "NA" entries in Table 1 indicate that the GLUCS and RRDF libraries do not have a cross section for the ^{23}Na(n,γ) reaction. The JENDL-D99 evaluation appears to have adopted the IRDF-85 covariance matrix while using a different cross section. This "inherited" covariance matrix does not appear to be valid, but is passed onto the next step for a more detailed evaluation. The FENDL-Dos cross section is a duplicate of the IRDF-90 entry for both the cross section and covariance file.

B. Evaluate Culled Cross Sections

The previous set of metrics were go/no-go metrics to cull out the subset of dosimetry-quality cross sections from the available data. This next set of metrics is used to rank the remaining candidate evaluations. These metrics, ranked by relative importance, include:

- Small cross section uncertainty over the complete energy range of the reaction.
- A recent evaluation date.
- A graphical inspection of the energy-dependent cross section should not reveal any unphysical features.
- Evaluate the fidelity (one indicator is the recent vintage) of the nuclear codes used by the evaluator to supplement the available experimental data (e.g. TNG, GNASH, SAMMY, etc.).

These metrics should be assigned a weight and an overall ranking and a quantitative metric for each of the cross section evaluations should result from this second step. Fig. 1 compares the cross section candidates for the ^{23}Na(n,γ) reactions. Fig. 2 show a representative covariance matrix. Table 3 gives the metric scores and weight factors.

Figure 1: ^{23}Na(n,γ) Candidate Cross Sections

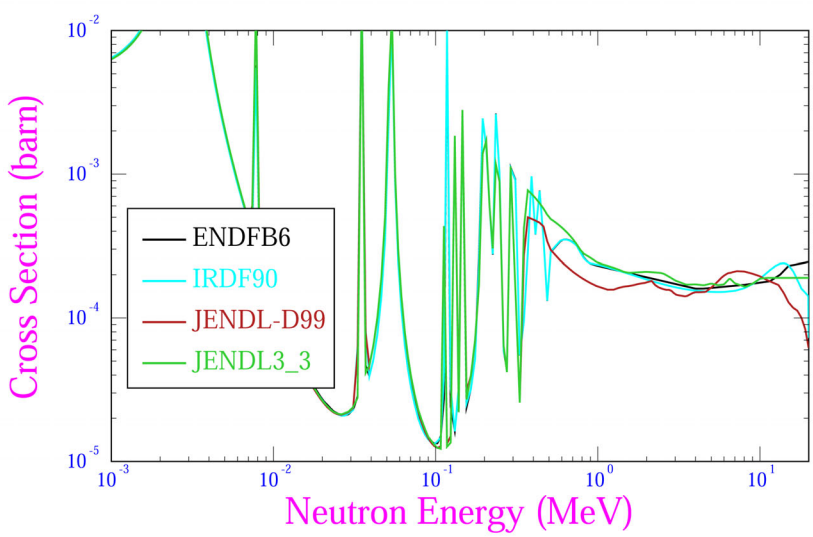

Figure 2: ^{23}Na(n,γ) ENDF/V-VI Relative Covariance Matrix

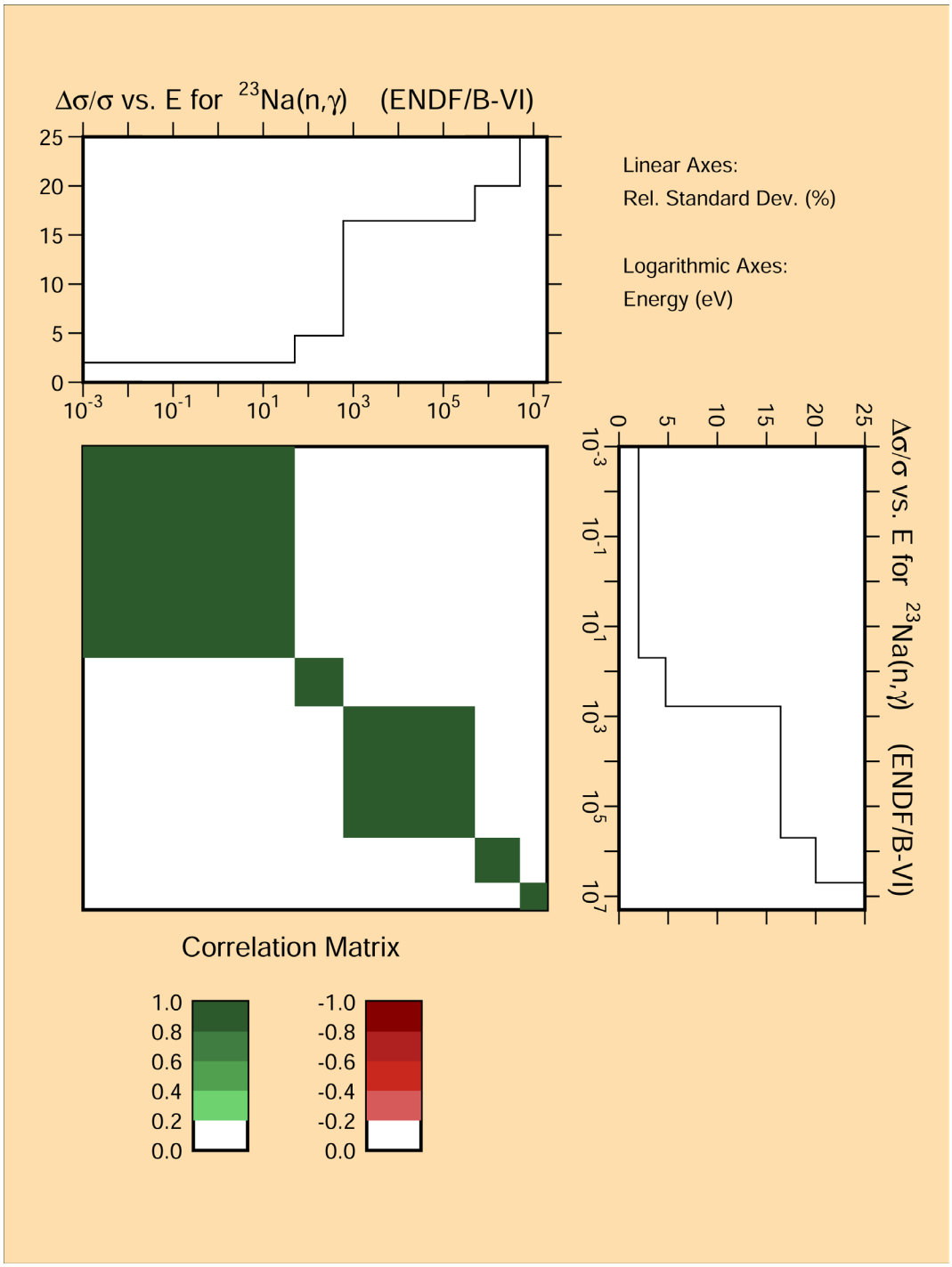

Table 3: Metric Scores and Weighting Factors

Metric	Normalized Weight Factors	Library			
		ENDF/B-VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
Small xsec unc.	1	1	1	0.7	1
Recent evaluation	1	0.5	0.8	1	0.7
Visual inspection	1	1	1	1	1
Fidelity of codes used	0.3	0.8	0.8	0.8	0.8
Fidelity of covariance	1.0	1	0.1	1	0.3
Defn.of residual product	0.1	0.2	0.2	0.2	0.2
Evaluation documentation	0.2	0.8	0.8	0.8	0.8
Score		3.92	3.32	4.12	3.42

The weighting factor for each evaluation criteria is seen in column 2 of Table 3. The JENDL-D99 entry is penalized for using a covariance matrix that is not derived from the cross section. The JENDL-3.3 covariance has a zero uncertainty for energies less than 1.2E-2 eV while showing a non-zero cross section. This is obviously a problem with the File 33 data. The CNDC evaluation (used in IRDF-90) and the JENDL-3.2 evaluation have a covariance matrix that is identical to the much older ENDF/B-VI evaluation. These covariance matrices are penalized for appearing to be "inherited" without apparent regard to the actual evaluation process. The ^{24}Na isotope has a ground state with a half-life of 14.969 hr. and a metastable state with a half-life of 20.20 ms. None of the five culled evaluations has distinguished the (n,γ) branching ratios to these two states. Some of the other libraries (RNAL and FENDL-Act) do distinguish between these two states but did not pass the covariance culling criteria from Step 1.

The evaluation step metrics, shown in Table 3, show that the JENDL-3.3 evaluation rates the highest, followed by the ENDF/B-VI evaluation.

C. *Comparisons With Measurements from Standard Benchmark Neutron Fields*

This third step, the comparison with measurements in standard benchmark neutron fields, has been very controversial. The 1976 IAEA Consultants Meeting [2] and the recent ISRD11 Workshop reflect a consensus that **ONLY** standard benchmark fields can be used for this step in differentiating between candidate evaluations. 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. Only four benchmark fields are recognized by the community. These benchmark fields are [4]:

- The spontaneous fission neutron field of ^{252}Cf
- The 1/E slowing down spectrum in hydrogenous moderator
- The Maxwellian thermal spectrum at specified neutron temperature
- The monoenergetic 14-MeV neutron field from a D-T source

Note that the ^{235}U thermal fission benchmark field is not included in this list. This field has been designated as a "reference" rather than a "standard" benchmark field because only one "standard" is permitted in a given energy region and the ^{252}Cf field is a much better characterized neutron field. The data of interest in a standard field are typically the spectrum-averaged cross sections. For the thermal Maxwellian spectrum at a temperature of 293.6 ^{o}C, the spectrum-averaged cross section is uniquely related to the 2200 m/s cross section. In the case of the 1/E field, the measured data are corrected for the

thermal contribution of the spectrum and are used to deduce the resonance integrals, typically corrected to represent the integral between the energy bounds from 0.5 eV to 100 keV. This paper also discusses the sensitivity of the resonance integral (for dosimetry cross sections) to these bounds and to deviations from a 1/E spectrum.

When comparing a measurement to a calculated quantity it is critical that one addresses the uncertainty in both measurement and calculation and that all sources of uncertainty are taken into account. The evaluation covariance file is folded with the neutron spectrum to obtain the cross section contribution to the uncertainty. All acceptable measurement data must include a measurement uncertainty. Great care must be taken in combining the experimental data in a statistically valid manner while addressing the issue of discrepant data [5]. A criteria for flagging and rejecting discrepant data must be established prior to forming the recommended experimental metrics. In the case of a resonance integral, the uncertainty analysis should include the added uncertainty associated with reducing the measured quantity to an inferred resonance integral. Finally, the uncertainty in the representation of the neutron spectrum in the standard reference field must be taken into account. The comparison quantity of interest is the C/E (calculated-to-experimental) ratio. Each of the sources of uncertainty term should be combined to provide an uncertainty in the C/E ratio. The extracted metric used to compare evaluations with measured data should be the C/E ratio and the number of standard deviations between this ratio and unity.

The importance of the C/E ratios in the various standard benchmark fields can be weighted according to the energy response of a reaction and the proposed application for a given cross section compendium. For a generic dosimetry cross section library, there is no reason to weight any one standard field as being more important than another field. If one considers a reactor application, such as embrittlement of a pressure vessel, then one would tend to heavily weight the ^{252}Cf field. For a fusion application, one would most heavily weight the 14-MeV field. Note that no special credit should be given for a C/E ratio with a deviation from unity of less than one standard deviation. These C/E ratios for standard benchmark neutron fields should be assigned a weight and the result of this third step should be an overall ranking and quantitative metric for each of the cross section evaluations.

Table 4 shows the C/E ratios for in the ^{252}Cf spontaneous fission standard neutron field. Data for thermal capture reactions in this fast fission field is very sparse and only one datapoint is reported for the ^{23}Na(n,γ) reaction [6]. None of the cross section evaluations does a good job in matching the C/E in this field. Table 4 shows the various contributors to the uncertainty in the C/E ratio. The ENDF/B-VI and CNDC evaluations have the best C/E even though they differ from unity by about two standard deviations. The JENDL-3.3 C/E is closer to unity in absolute value but not in terms of the number of standard deviations.

There are at least 14 datapoints for the resonance integral for the ^{23}Na(n,γ) reaction, but only nine of these datapoints have an adequate experimental uncertainty. The best datapoint (most recent, lowest uncertainty, best characterized standard field) was made by Kimura in a graphite moderated high quality and well characterized 1/E spectrum [7]. The data in the literature shows considerable variation. A least squares fit to this data has a chi-squared per degree of freedom of 4.25 where a critical chi-squared of 2.8 corresponds to an acceptable probability with a 1% confidence interval. When the LWEIGHT [8] algorithms were used to handle the discrepant data, the least squares recommended datapoint agrees very well with the "best" datapoint of Kimura. Table 5 shows that all of the cross sections have a good C/E ratio (within one standard deviation), but that the JENDL-3.3 evaluation has a very large uncertainty (consistent with the cross section uncertainty from Fig. 2).

Table 4: C/E for ^{252}CF Standard Neutron Field

Parameter	Library			
	ENDF/B- VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
Spectrum-averaged cross section (mb)	0.2721	0.2239	0.2796	0.2681
Cross section uncertainty component	13.7%	12.84%	8.77%	12.97%
Spectrum uncertainty component	1.23%	1.39%	1.12%	1.25%
Measurement uncertainty component	4%	4%	4%	4%
C/E ratio and uncer tainty	0.8122 +/- 14.3%	0.6683 +/- 13.5%	0.8346 +/- 9.70%	0.8003 +/- 13.6%

Six datapoints with an adequate uncertainty characterization are found for the 2200 m/s thermal cross section. The "best" datapoint (based on uncertainty and date) is from Steinnes [9]. Using the LWEIGHT code for the identification and removal of discrepant data the least squares analysis yields a chi-squared per degree of freedom is 0.09 - a very small value suggesting correlations in the data or poor uncertainty quantification. Table 6 shows that all the evaluations are in good agreement with the measurements in this standard neutron field.

Table 5: C/E for Resonance Integral in 1/E Field

Parameter	Library			
	ENDF/B-VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
Spectrum-averaged resonance integral	0.31667	0.31217	0.31194	0.31646
Cross section uncertainty component	4.42%	2.79%	21.8%	2.91%
Spectrum uncertainty component	--	--	--	--
Measurement uncertainty component	4.1%	4.1%	4.1%	4.1%
C/E ratio and uncer tainty	0.99052 +/- 6.0%	0.97645 +/- 5.0%	0.975727 +/- 22.2%	0.989865 +/- 5.0%

Table 6: C/E in Thermal Maxwellian Field

Parameter	Library			
	ENDF/B-VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
2200 m/s cross section	0.52813	0.53306	0.53156	0.52819
Cross section uncertainty component	2.0%	2.0%	7.3%	2.0%
Spectrum uncertainty component	---	---	---	---
Measurement uncertainty component	0.68%	0.68%	0.68%	0.68%
C/E ratio and uncer tainty	0.99273 +/- 2.1%	1.0020 +/- 2.1%	0.99917 +/- 7.3%	0.9928 +/- 2.1%

Table 7 shows the C/E comparison in a 14-MeV neutron field. Seven datapoints were considered for this reaction in a 14-MeV DT neutron field. The exact neutron energy depends upon energy of the incident deuteron and the angle of the tritiated target. Thus the analysis had to form C/E ratios for the individual energy points and then average the C/E ratios to obtain an evaluation-dependent metric. The uncertainty in the neutron energy, the experimental cross section measurement, and the cross section evaluation were all convoluted in forming the uncertainty metrics. The uncertainty in the cross sections at this high energy were large. The uncertainty in the neutron energy was also fairly large, but the cross sections were fairly flat in this region., so this uncertainty in neutron energy did not dominate the C/E uncertainty. Table 7 shows that the good agreement for the ENDF/B-VI and CNDC evaluations but poor agreement for the JENDL-D99 and JENDL-3.3 evaluations.

Table 7: C/E in a 14-MeV Standard Neutron Field

Parameter	Library			
	ENDF/B-VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
14-MeV cross section (mb)	0.22467	0.1342	0.190	0.2353
Cross section uncertainty	25%	25%	12.1%	25%
Rep. expt. avg. energy uncertainty	15%			
Cross section uncertainty due to this energy spread	3.2%	6.3%	0%%	1.8%
Expt. measurement uncertainty	13%			
C/E ratio and uncer tainty	0.9136 +/- 14.7%	0.52685 +/- 12%	0.7125 +/- 7.6%	0.90621 +/- 11%

Table 8 shows a composite of the scores for the culled evaluations in the standard neutron benchmark fields. The ENDF/B-VI and CNDC evaluations are preferred based on this comparison.

Table 8: Composite Metrics in Standard Benchmark Fields

Metric	Normalized Weight Factors	Library			
		ENDF/ B-VI	JENDL-D99	JENDL-3.3	IRDF- 90/ CNDC
^{252}Cf spontaneous fission	0.6	0.8	0.2	0.6	0.8
1/E slowing-down	1	1	1	0.6	1
296.3 ^{o}C Thermal Maxwellian	1	1	1	0.7	1
14-MeV	0.3	1	0.6	1	1
Score		2.78	2.3	1.96	2.78

D. Make Recommendation for Nuclear Data

The next step is to use the Step 2 and 3 metrics to recommend a specific evaluation. First, the ranking of the culled evaluations should be examined. If the top ranked evaluation from steps 2 and 3 are identical, then the recommendation is clear. If the top ranking differs, then the step 2 and 3 metrics should be weighted according to the significance of the variation of a given metric within the range of candidate evaluations. The weighting needs to be clearly established prior to the actual selection process. If most of the candidate evaluations are similar with respect to a metric, then this metric will not be important in the selection procedure.

The task is not completely finished with this down-selection of the cross section evaluation. Since this step involves defining all relevant nuclear data, the associated nuclear data should also be defined in a manner that reflects the latest nuclear data information while being consistent with the selected evaluation. The relevant nuclear data includes such quantities as the atomic weight, relevant natural abundance, isotopic half-live, and the gamma/beta decay energies and probabilities. The spectrum-averaged cross sections and C/E ratios in benchmark fields should also be reported. The IRDF-2002 library is the first cross section compendium the author is aware of that will clearly define this associated nuclear data required for the consistent application of a dosimetry cross section.

Table 9 compares the metrics for the ^{23}Na(n,γ) reaction. The shaded entries show the leading evaluation in each category. Using the pre-assigned weighting factors for this cross section, the ENDF/B-VI evaluation is seen to be the best evaluation. Thus this is the recommended evaluation. Tables 10 and 11 identify the “best” values for the other nuclear data that are “not inconsistent” with any of the documented assumptions that went into the recommended evaluation.

Table 9: Metrics and Weights for Final Selection

Metric	Normalized Weight Factors	Library			
		ENDF/B-VI	JENDL-D99	JENDL-3.3	IRDF-90/ CNDC
Step 2: fidelity	1	3.92	3.32	4.12	3.42
Step 3: Std. Benchmarks	1	2.78	2.3	1.96	2.78
Score		3.35	2.81	3.04	3.10

Table 10: Target and Residual Atom Nuclear Data

Parameter	Value	Reference
Tgt. Elemental atomic weight	22.989770 (2)	Nuclear Wallet Cards, Jan. 2000 [10]
Abundance	100%	IUPAC, 1997 [11]
Mass Excess (MeV)	-9.530	Nuclear Wallet Cards, Jan. 2000 [10]
Rsd. Half-life	14.9574 (20) h	BNM-CEA/LNHB [12]

Table 11: Residual Isotope Decay Nuclear Data

Gamma energy (keV)	Emission Probability (%)	Reference
1368.626 (5)	99.9935 (5)	BNM-CEA/LNHB [12]
2754.007 (11)	99.872 (8)	BNM-CEA/LNHB [12]

E. Evaluate Consistency of Cross Section Library

After the selection of the recommended cross section evaluation, the process is still not finished. The final part step is to validate the fidelity of the selection and to clearly document the intermediate metrics and weighting used to select the cross section evaluation so that it is available to users.

There are other well characterized neutron fields, called "reference" neutron benchmark fields, that should be used to validate the cross section selection. One such neutron field is the ^{235}U thermal fission field. Other reference neutron fields include the ACRR central cavity [13, 14] and the YAYOI neutron field [15]. The C/E ratios in these reference neutron fields should be determined and reported as a final confirmation. This step has not been carried out at the present time, but it should be pursued if the recommended cross section is adopted by the IRDF-2002 compilation. Note that this is a "beta test" step

and in no case should this "confirmation" step result in a change in the selected cross section. It should only serve to indicate areas where either the cross section or the reference field characterization should be further examined. Even if it were not for the need to have only one 'standard" field in a given energy region, the "reference" neutron fields typically have neutron spectrum characterizations that were derived from activation foils in conjunction with spectrum unfold or adjustment methods. Since this spectrum characterization process introduces correlations between the spectrum and cross section evaluations that are not properly accounted for in the application of the step 3 metrics, this data can not be used in the cross section selection process, only in the validation process.

In addition to the comparison with "reference" neutron fields, one should also validate the selection by making a comparison to other community recommended quantities. Again, this step can not change the recommendation, it can only flag a metric for additional review. BNL-325 [16] is the best and most recently updated compilation of recommended thermal cross sections and resonance integrals. This data source was not considered in step 3 because its recommended values reflect calculations as well as measurements [17]. The correlation of these "BNL-325 considered calculations" with the calculations that supported the cross section evaluation could not be properly taken into account.

Another compendium of resonance integral data and thermal cross sections comes from Gryntakis and Kim [18]. This is a much earlier compendium that BNL-325 but clearly distinguishes experimental data from calculated "recommendations".

Data for the ^{23}Na(n,γ) reaction is not reported in the ^{235}U thermal fission reference benchmark field. Table 12 shows that the C/E comparison for the community-standard recommendations for resonance integrals and thermal cross sections. These community standards and the other reference neutron fields do not provide any comparisons that are in conflict with the recommendation to use the ENDF/B-VI evaluation for the ^{23}Na(n,γ) reaction.

Table 12: C/E For Community-Recommended Metrics

Recommended Metrics	Value	C/E for Selected Cross Section
G&K resonance integral (mb) [18]	---	---
Mughabghab thermal cross section (b) [16]	0.530 +/- 0.005	0.9647 +/- 2.2%
Mughabghab resonance integral (mb) [16]	0.311 +/- 0.01	1.018 +/- 5.5%

III. CONCLUSION

This paper details the rationale for the selection of ^{23}Na(n,γ) dosimetry cross sections to be used in the IRDF-2002 library. This analysis addresses 12 candidate cross section evaluations, culls the dosimetry- quality evaluations down to 5, and makes a final recommendation of the ENDF/B-VI cross section evaluation. This recommendation is then validated by examining the deviation of measured quantities from current community recommendations. The quality of this cross section is consistent with the requirements for a dosimetry cross section to be included in the IRDF-2002 library.

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Inspection of the IRDF-2002 Candidate Cross Sections

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I. PURPOSE

The IRDF-2002 library candidate cross sections have been selected and distributed to "beta" testers for review. The purpose of this paper is to report on issues affecting the use of the candidate cross section that arose as part of this "beta" test evaluation of the library. These issues are divided into major and minor issues as well as being subdivided into the level of difficulty in resolving the issues.

II. MINOR ISSUES

1. The files do not have enough information (MF=2 resonance data) to permit resonance reconstruction and broadening. This is critical to (n,γ) and (n,f) cross sections. In this absence, it is critical that the temperature of the cross section be specified. This has not been done in the current library files. My recommendation is to present cross sections at 298 ^{o}K (the temperature where they are typically used) rather than 0 ^{o}K. The critical requirement is that the relevant temperature be presented in library documentation.

2. Several candidate cross sections could not have their covariance matrix processed using the NJOY99 code but were processed correctly while using the NOY97 code. This behavior suggests a format issue that may have been addressed in a patch to the NJOY97 code and not integrated into the ongoing NJOY code series. This omission is important only if it also follows that the file does not adhere to the ENDF-6 format specification.

   The recommendation here is to verify that the processing is correct using the LISTEF code and, if so, to warn users about the potential NJOY99 processing issues.

3. The ENDF/B-VI evaluation of the ^{93}Nb(n,2n) reaction was said to be rejected because it differed by a factor of 3X from the other evaluations. While this observed difference is true - the source of the problem should be clearly stated. The ENDF/B-VI cross section is for the total (n.2n) reaction - not just the probability of branching to the metastable state. The ENDF/B-VI cross section correctly adheres to the ENDF-6 format specifications - it is the evaluations that present only the metastable but still label it as the MT=16 component that violate the ENDF-6 conventions.

   Problems similar to this were previously encountered for the IN115N reaction when the IRDF-90 library was first published. The material processing failed to apply the MF=51 branching ratio of 0.79 to the processed cross section. I believe that this problem was remedied in the IRDF-90v2 release.

   The recommendation here is to have all of the files either strictly adhere to the ENDF- 6 format specifications or to clearly state that the cross section presented in the library only applies to a metastable branching component of the reaction. I believe that this is consistent with the plans for the IRDF-2002 library.

4. The IRDF-90 S32P reaction has comments that suggest that it is derived from the ENDF/B-VI cross sections. But the current ENDF/B-VI S32 cross section does not even have a covariance matrix. The project needs to modify the header cards in this material to clearly state the origin of the evaluation and covariance.

5. The nuclear data used in the IRDF-2002 library should be consistent with the recommended data used for detector calibration. In particular, the IAEA has a group updating the "X-ray and Gamma-ray standards for Detector Calibration". The old document was IAEA-TECDOC-619. I strongly suggest that this group buy into the nuclear data we are using or that some compromise be reached. It is crucial that the dosimetry-users' nuclear data be consistent with what is used for detector calibration.

6. On the nuclear data, the 511 keV line is not useful for most dosimetry applications. Accordingly, I suggest that:

   For Cu-62, give additional lines - even if they have low yield, e.g. 1173 and 875 keV lines with <1% yield.

   For Cu-64, give additional lines, even if they have low yield, e.g. 1345 keV line with 0.47% yield.

III. MAJOR ISSUES, EASILY CORRECTED

7. Several isotope evaluations failed to process due to format problems in the MF=2 MT=151 file.

8. One file had to have a dummy MF=2 MT=151 resonance file added in order to process it with NJOY99.

9. None of the IRDF-90 files could be processed from the files as distributed. In some cases the errors in one reaction prevented the NJOY99 code from continuing to even search the file for other isotopes. Several files had format issues that needed to be fixed before any processing could be performed.

10. Given all the previous causes, problems were found and changes had to be made in many files in order to use NJOY99 in the processing. The changes files include:

    ENDF/B-VI - ni60p, cu652
    JEFF30 - fe56p, ni58p, ni582, ni60p
    RRDF - pb204
    JENDL/D-99 -
    IRDF-90 - li6t, b10a, mg24p, al27p, al27a, p31p, s32p, sc45g, ti46p, ti47np, ti47p, ti48np, ti48p, cr522, mn55g, fe54p, ni582, ni58p, pu239f, cu652, zn64p, zr902, nb932, nb93n, rh103n, ag109g, in1152, in115n, i1272, au1972, au197g, fe58g, cu63a, co59g, co582, cu63g, th232f, u235f, u238f, u238g

IV. MAJOR ISSUES TO BE RESOLVED

11. There is no IRDF-2002 candidate for the Na23G reaction. This is a very important reaction and should clearly be included in the library. The preliminary analysis states that there are no acceptable candidates because all potential candidates had a diagonal covariance matrix. This rejection criteria should be re-examined if there are no candidates that meet the criteria

and if the the diagonal covariance that is available has realistic uncertainties.

12. There are also no candidates identified for the NB93G and IN115G reactions. Since we had previous dosimetry-quality entries for these important reactions I can not understand not having them in the new library. The community does have covariance matrices for these reactions. Unless significant problems have recently been identified, the previous reactions appeared to be adequate and should be used if newer updated cross sections are not available.

13. There are no library entries for the ^{98}Mo(n,γ) reaction. I do not know of a suitable candidate, but this reaction should be placed on the list of desired dosimetry reactions.

14. The ^{nat}Cd(n,X) absorption cross section should also be added to the list of desired cross sections supporting dosimetry applications. Although this reaction is not used directly as a dosimetry reaction, cadmium is used as a cover for many dosimeters. The Cd cover is intended to reduce the thermal response of a dosimeter and to move the sensitive region above the 0.2 eV Cd-cutoff energy.

15. The covariance file for S32P can not be processed with either NJOY99 or NJOY97.

16. The cross section processing of the JEFF 3.0 material ni58p has errors. Our testing went back to the full file as distributed in the JEFF 3.0 library and processed the MAT=2825 file.

17. The material IRDF90 ti46p cross section had nsub problems during cross section processing.

V. CONCLUSION

The beta testing of the IRDF-2002 library has been very productive. With the exception of the IRDF- 90 S32P covariance, IRDF-90 TI46p cross section, JEF30 ni582, and the JEFF30 ni58p, all of the problems found were easily corrected. These four evaluations need further attention before the library is distributed.

It is suggested that the final IRDF-2002 library be tested again for consistency with NJOY99 processing before it is distributed. I note that the errors may have been in the NJOY99 processing code and not in the library format since many of these files appeared to pass inspection by ENDF6 format checking routines. However, since NJOY99 is used by a large customer base and changes to the code are readily submitted on an internet-based bug-tracking system, changes to either the IRDF-2002 library or the NJOY99 code should be made before the final version of the IRDF-2002 library is distributed. This is expected to be an easily accomplished task. If this cannot be done, then the code documentation should warn users that some NJOY99 incompatibilities exist for specific evaluations and that the problem will be addressed in future revisions of the library or the NJOY99 code.

In summary, this was a very useful and productive check of the library and will ensure that the released IRDF-2002 version is ready for the user community. Such beta testing checks should be a standard part of the release process for future updates of the IRDF libraries.

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Results of Validation of the IRDF-2002 Dosimetry Library Using the ACRR Reference Neutron Field

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I. PURPOSE

After the selection of the recommended cross section evaluations for inclusion in the IRDF-2002 library, the process is not quite finished. The final step before library release is to validate the fidelity of the selection There are well characterized neutron fields, called "reference" neutron benchmark fields, that should be used to validate the dosimetry cross section selection. This paper address the validation of the IRDF-2002 dosimetry selections in one such "reference" neutron field, the ACRR central cavity [1].

II. BACKGROUND

These "reference" fields should not be confused with the "standard" benchmark fields. The 1976 IAEA Consultants Meeting [2] and the recent ISRD11 Workshop [3] reflect a consensus that **ONLY** standard benchmark fields can be used for this step in differentiating between candidate evaluations. 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. Only four benchmark fields are recognized by the community. These benchmark fields are [3]:

- The spontaneous fission neutron field of ^{252}Cf
- The 1/E slowing down spectrum in an hydrogenous moderator
- The Maxwellian thermal spectrum at a specified neutron temperature
- The monoenergetic 14-MeV neutron field from a D-T source

Note that the ^{235}U thermal fission benchmark field is not included in this list of "standard" neutron fields. This field has been also designated as a "reference" rather than a "standard" benchmark field because only one "standard" is permitted in a given energy region and the ^{252}Cf field is a much better characterized neutron field. The data of interest in a standard field are typically the spectrum-averaged cross sections. For the thermal Maxwellian spectrum at a temperature of 293.6 ^{o}C, the spectrum-averaged cross section is uniquely related to the 2200 m/s cross section. In the case of the 1/E field, the measured data are corrected for the thermal contribution of the spectrum and are used to deduce the resonance integrals, typically corrected to represent the integral between the energy bounds from 0.5 eV to 100 keV.

When comparing a measurement to a calculated quantity it is critical that one addresses the uncertainty on both measurement and calculation and that all sources of uncertainty are taken into

account. The evaluation covariance file is folded with the neutron spectrum to obtain the cross section contribution to the uncertainty. All acceptable measurement data must include a measurement uncertainty. Great care must be taken in combining the experimental data in a statistically valid manner while addressing the issue of discrepant data [4]. A criteria for flagging and rejecting discrepant data must be established prior to forming the recommended experimental metrics.

Finally, the uncertainty in the representation of the neutron spectrum in the standard and reference fields must be taken into account. The comparison quantity of interest is the C/E (calculated-to-experimental) ratio. Each of the sources of uncertainty should be combined to provide an uncertainty in the C/E ratio. The extracted metric used to compare evaluations with measured data should be the C/E ratio and the number of standard deviations between this ratio and unity.

Note that this paper reports on a "beta test" step for the library selection using a "reference" neutron field and **in no case** should this "confirmation" step result in a change in the selected cross section. It should only serve to indicate areas where either the cross section or the reference field characterization should be further examined. Even if it were not for the need to have only one "standard" field in a given energy region, the "reference" neutron fields typically have neutron spectrum characterizations that were derived from activation foils in conjunction with spectrum unfold or adjustment methods. Since this spectrum characterization process introduces correlations between the spectrum and cross section evaluations that are not properly accounted for in the spectrum adjustment, this data cannot be used in the cross section selection process, only in the validation process.

III. ACRR REFERENCE NEUTRON FIELD

This paper presents the Annular Core Research Reactor (ACRR) as a well characterized reference neutron field appropriate for validating the selection of the IRDF-2002 cross sections. This water-moderated neutron spectrum is a very good complement to the fast fission "standard" benchmark fields that were used to support the down-selection of the candidate IRDF-2002 dosimetry cross sections. A more detailed description of this reference field appears in Reference [1]. The details of the radiation transport models and 640-group representation of the neutron spectrum in the ACRR central cavity appears in published laboratory reports [5].

III.i Experimental Characterization

The ACRR is a water-moderated test reactor with a dry central cavity and a fueled external cavity. The 236 cylindrical fuel elements contain a unique BeO-UO_2 fuel with 35% enriched ^{235}U that allows operation at fuel temperatures up to 1400 ^{o}C in pulse and steady- state modes. The reactor is capable of steady-state operation at 2-MW, intermittent operation at 4-MW, and pulsed operation with a maximum pulse of 300 MJ, a peak power of 30,000 MW, and a minimum pulse width of 6.5 ms. The reactor core is located in a 3.1 meter diameter and 8.5 meter deep pool. The reactor has a 9-inch diameter dry central cavity that supports large test fixtures and provides good uniformity.

The first step in a neutron field characterization is to establish the uniformity of the field. A mapping was done with the fast neutron (> 3-MeV) $^{32}S(n,p)$ reaction to identify a 6-inch diameter region with excellent uniformity (<5% variation). A second mapping was performed with CaF_2:Mn thermoluminesent dosimeters (TLDs) to characterize the uniformity of the gamma field. Vertical as well as horizontal profiles were measured.

Once this uniform exposure region was clearly established, dosimetry foils were fielded. Even though the uniform region was fairly large, the activity measurements were performed in a series of runs.

Separate reactor operations were required since only eight high purity Germanium detectors were available for reading the dosimetry foils and many of the activated foils have short half-lives or require long counting times. Cd and ^{10}B covers were used to alter the region of energy response for some of the activation foils. The ^{10}B cover was a 5- cm diameter 91% ^{10}B- enriched B_4C ball. This boron ball was large enough to alter the neutron field in the surrounding region, so each boron-covered activation foil was exposed on a separate operation. Fission foils were not stacked in the boron ball since previous testing had shown that the neutron scattering between adjacent 1-gram fission foils thermalized enough neutrons to alter the dosimeter response of ^{235}U and ^{239}Pu foils. The $^{58}Ni(n,p)$ reaction was used to normalize the separate reactor operations to a uniform neutron fluence. When a boron ball was used, internal as well as external Ni foils were used for normalization. Table 1 details the 44 dosimetry reactions and cover combinations that were used in the spectrum characterization.

Table 1. Spectrum-averaged Dosimetry Cross Sections for ACRR Central Cavity

Reaction/ Cover 1	Median Eng. Rsp. (eV)	Cross Section (mb)			Uncertainty (%)		
		Expt.	Calc.	C/E	Expt.	Xsec.	Spct.
Nb93g[Cd]	852.3	2.054E-12	2.517E-12	1.225	7.83	9.5	21.12
Ni58p	3.765E6	1.572E-13	1.557E-13	0.990	5.43	2.48	12.06
Ni58p[Cd]	3.765E6	1.552E-13	1.557E-13	1.003	5.44	2.48	12.06
S32p	3.856E6	9.040E-14	9.279E-14	1.026	5.94	3.52	12.91
Na23g	7.138E-2	2.599E-13	3.052E-13	1.174	5.45	2.17	163.2
Na23g[Cd]	8.180	5.527E-14	6.813E-14	1.233	5.45	5.40	37.62
Na23g[Fi]	2717.	5.091E-15	7.197E-15	1.414	5.90	15.3	19.63
Mg24p[Cd]	8.026E6	1.754E-15	1.952E-15	1.113	6.33	2.36	20.67
Al27a[Cd]	8.346E6	8.717E-16	8.891E-16	1.13	6.28	2.18	20.60
Sc45g[Cd]	1.673	2.290E-12	2.628E-12	1.148	6.09	1.13	59.84
Sc45g	6.729E-2	1.437E-11	1.460E-11	1.016	6.09	0.98	175.4
Ti46p[Cd]	5.623E6	1.498E-14	1.347E-14	0.899	5.43	2.46	16.05
Ti48p[Cd]	8.01E6	3.699E-16	3.415E-16	1.083	5.95	2.54	19.34
Ti47p[Cd]	3.290E6	2.691E-14	2.777E-14	1.032	7.06	3.64	11.24
Mn55g[Cd]	236.9	2.093E-12	2.779E-12	1.329	5.38	4.48	39.97
Fe54p[Cd]	4.011E6	1.157E-13	1.144E-13	0.989	6.30	2.14	12.78
Fe56p[Cd]	7.155E6	1.326E-15	1.302E-15	0.982	5.39	2.29	18.80
Fe56p[Fi]	7.1598E6	1.195E-15	1.255E-15	1.050	5.42	2.29	18.80
Fe58g[Cd]	229.5	2.593E-13	2.796E-13	1.078	5.76	5.88	38.84
Co59p[Cd]	5.454E6	1.828E-15	1.863E-15	1.019	6.33	4.10	15.43
Co59g[Cd]	113.	8.225E-12	1.807E-11	2.197	5.53	0.77	53.78
Co59g	0.6426	2.529E-11	3.473E-11	1.373	6.09	0.73	105.1
Co592[Cd]	1.326E7	2.799E-16	2.909E-16	1.039	9.78	2.54	33.02
Ni582[Cd]	1.631E7	5.010E-18	8.740E-18	1.745	6.50	2.74	35.69
Ni60p[Cd]	6.739E6	2.847E-15	2.484E-15	0.872	5.68	10.49	17.6
Cu63g[Cd]	528.2	8.655E-13	1.249E-12	1.44	6.42	4.17	35.52
Cu63g	8.589E-2	2.998E-12	3.220E-12	1.074	5.57	4.00	131.3
Zn64p[Cd]	3.919E6	5.146E-14	5.432E-14	1.056	5.37	4.79	12.95
Zr902[Cd]	1.536E7	2.039E-16	1.798E-16	0.882	9.41	1.56	34.97
Nb932[Cd]	1.137E7	5.744E-16	5.072E-16	0.883	6.24	2.60	30.33
In115g	1.497	3.969E-10	5.853E-10	1.475	5.91	5.98	71.38

In115g[Cd]	1.586	2.820E-10	4.966E-10	1.761	6.30	5.98	73.91
In115n[Cd]	2.269E6	1.924E-13	2.731E-13	1.420	7.04	2.18	10.55
Au197g	3.099	3.243E-10	3.421E-10	1.055	6.70	0.16	63.90
Au197g[Cd]	3.292	2.702E-10	2.971E-10	1.099	6.31	0.17	66.03
Au197g[Fi]	6318.	1.491E-12	1.432E-12	0.960	6.71	0.49	7.66
Np237f[Fi]	1.497E6	2.480E-12	2.910E-12	1.173	5.66	9.58	8.59
U235f[Fi]	2.463E5	5.733E-12	5.874E-12	1.025	5.64	0.29	4.90
U238f[Fi]	2.336E6	5.300E-13	5.212E-13	0.983	5.66	0.53	11.26
Pu239f[Fi]	5.961E5	6.986E-12	6.635E-12	0.950	5.42	0.39	5.35
Mo98g[Cd] 2#	---	2.453E-16	---	---	7.12	---	---
Ag109g 2#	---	8.828E-18	---	---	2.46	---	---
Ag109g[Cd] 2#	---	5.816E-18	---	---	3.33	---	---
W186g 2#	---	1.464E-14	---	---	6.54	---	---

1 Cross section identifier is the target isotope with a reaction symbol followed by a cover in square brackets. Reaction symbols include g=(n,γ), p=(n,p), 2=(n,2n), a=(n,α). f=(n,f), n=(n,n'). Covers include [Cd]=Cadmium and [Fi]=^{10}B-enriched boron ball.

2 These data were not used in the spectrum adjustment due to the lack of cross section covariance matrices. The experimental data given for these reactions is the activity in Bq/atom.

Reported measurement is an activity, not a cross section.

III.ii Spectrum Modeling

High fidelity calculations were performed to provide the best possible calculated neutron spectrum. This spectrum was then used as input to the least-squares spectrum adjustment codes and as a trial function to the iterative unfolding codes. Intuition and experience was used to estimate an *a priori* calculated spectrum uncertainty and covariance matrix. Calculations were also performed to ensure that the highest fidelity dosimeter responses were used.

The radiation transport was done with the 3D general geometry Monte Carlo point cross section MCNP code (version 4C) [6]. The reactor, support structures and dosimetry test fixture were modeled in detail. Figures 1 and 2 show a combinatorial geometry model for a horizontal and vertical cut through the reactor. The neutron spectrum was scored in an 89-energy and in a 640-energy group structure.

The Monte Carlo calculations in this fine energy group structure used $>10^9$ source particles. Figure 3 shows the calculated spectrum in a typical logarithmic fluence plot (notated as dn/dE or $\Phi(E)$). Figure 4 shows the calculated spectrum in a linear lethargy plot (often notated as dE/dE, $\Phi(\mu)$, or $E\Phi(E)$). In the lethargy plot (with linear y-axis and logarithmic energy x- axis or lethargy), equal areas under the curve correspond to equal neutron content.

The Monte Carlo scoring statistics are so small (typically < 1%) that they cannot be easily resolved even in enlargements of the figures.

Figure 4 shows some prominent resonance absorption structure in the 0.5 - 2 MeV region. The excellent sampling statistics (<1%) indicated that this structure related to some aspect of the nuclear data used in the transport model. Given the smooth ^{235}U fission neutron spectrum, such resonance structures in this high energy region of the neutron spectrum are clearly not in the source term. This structure is not as evident in the more common dn/dE plot in Figure 3 due to the logarithmic y-axis. Some of the high energy structure was traced to the presence of high energy resonances in the ^{16}O elastic cross section. The oxygen is present in the oxide fuel form and in the ACRR water moderator. This source of the

structures comes from a high energy elastic rather than an absorption event. This source identification was further identified and confirmed by altering the ^{16}O cross sections to smooth out these resonances, reprocessing the cross section, repeating the spectrum calculation, and observing the elimination of the structures in the spectrum. The presence of this structures as a meaningful feature poses problems for iterative spectrum unfolding codes that depend upon a local smoothness criteria in the spectrum [9] for their convergence methodology.

In keeping with the desire to provide the highest quality neutron field characterization, a high fidelity treatment is applied to address the influence by the dosimetry packaging on the measured activity. Responses from a 640 group calculation are used to account for the detailed response of the dosimetry covers and to account for self-shielding in resonance regions of the activation foil [10].

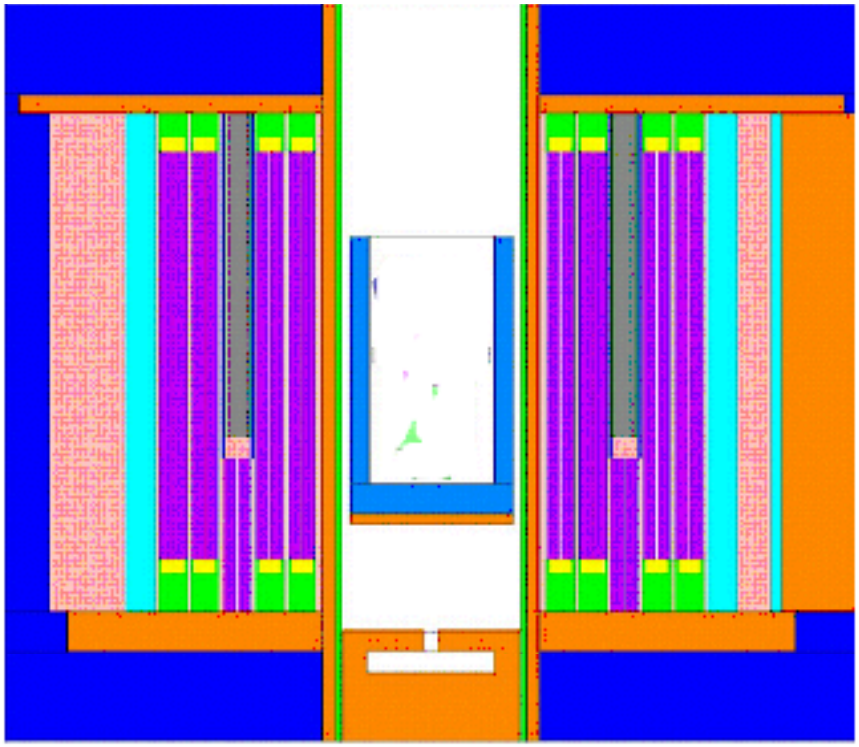

Figure 1: Horizontal Cut Through ACRR Cavity

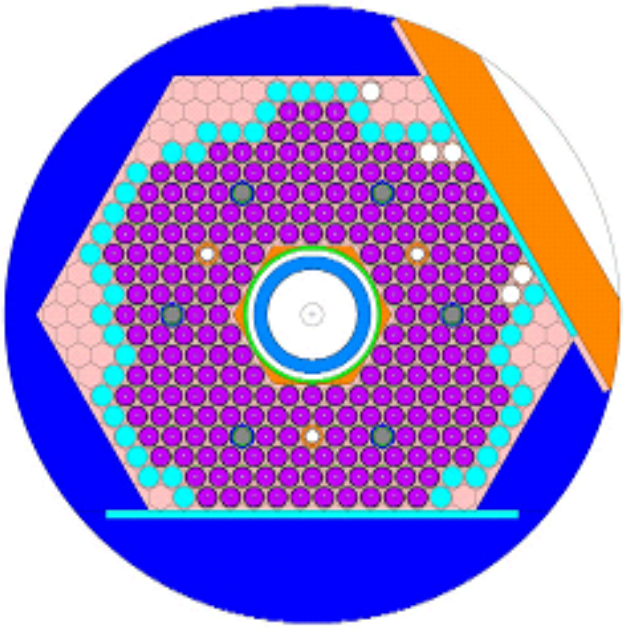

Figure 2: Horizontal Cut Through ACRR Cavity

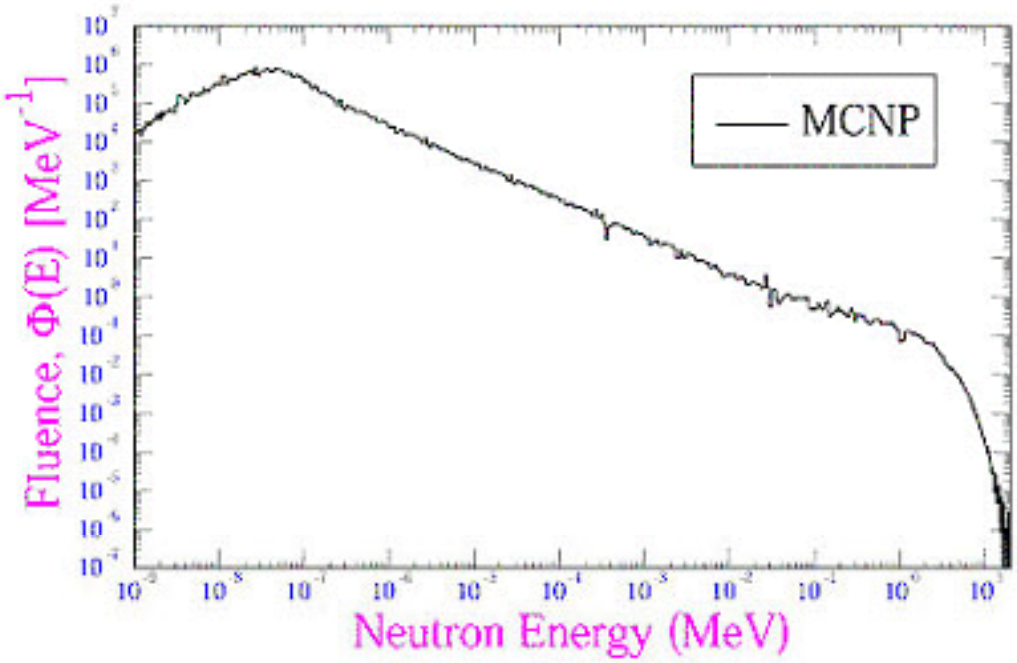

Figure 3: dE/dE Representation of Calculated Spectrum

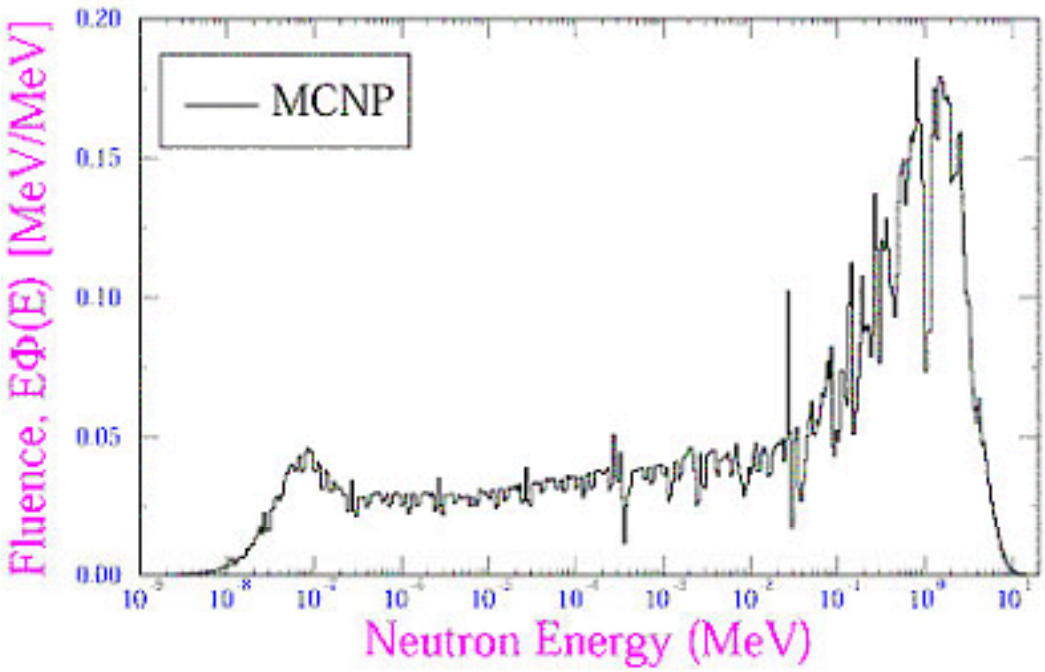

Figure 4: dE/dE Representation of Calculated Spectrum

III.iii Spectrum Adjustment/Unfold

Both least-squares spectrum adjustment and iterative spectrum unfolding codes were used to analyze the neutron spectrum and its associated uncertainty. The nuclear data was taken from the SNLRML dosimetry library [7]. This library is based, in large part, on the IRDF-90 dosimetry library.

A Sandia modified version of the LSL code [8] was used in the analysis. This code has been modified to interface with the SNLRML dosimetry compendium and to use foil covers. This code was run using both 89 and 366 energy group representations. The 366-group representation was selected to include the energy break points from all of the representations of the reaction cross section covariance matrices. The much faster running 89-group calculations were used for parametric studies. A 640-group LSL calculation was not done since it resulted in singularities in the matrix inversion operations.

Figure 5 shows the ratios of the calculated-to-experimental (C/E) activity values along with the associated uncertainty. All 40 dosimetry-quality reactions were used in the spectrum adjustment. Four reactions (^{98}Mo(n,γ)[Cd], ^{186}W(n,γ), ^{109}Ag(n,γ), and ^{109}Ag(n,γ)[Cd]) were not used since dosimetry-quality cross sections with associated covariance matrices were not available. The resulting adjustment in the calculated neutron spectrum is shown in Figure 6. The χ^2 per degree of freedom (dof) for the adjustment was 1.68, a very acceptable value. Figure 7 shows the effects of successively removing the reactions that contributed most to the χ^2. An examination of the χ^2 contributions revealed a serious discrepancy between the ^{58}Ni(n,2n) and ^{90}Zr(n,2n) reaction, both of which have thresholds near 14-MeV. There are some issues about the dilution of the indium activation foil that are still being investigated. A systematic difference was also found between bare and Cd-covered dosimeters with their primary response in the 100 - 500 eV region. Since the ^{nat}Cd(n,γ) reaction used for the cover does not have an associated covariance matrix and since there are a number of resonances in this region, additional experimental confirmation of the ^{nat}Cd(n,γ) cross section is desired in this region.

A SNL modified version [11] of the SAND-II code [12] was also used. This iterative unfolding code used the SNLRML dosimetry cross sections along with covers, and was modified to determine a neutron spectrum covariance matrix from a Monte Carlo-based variation of the correlated input trial spectrum and the measured activities. These results were in good agreement with the LSL spectrum adjustment.

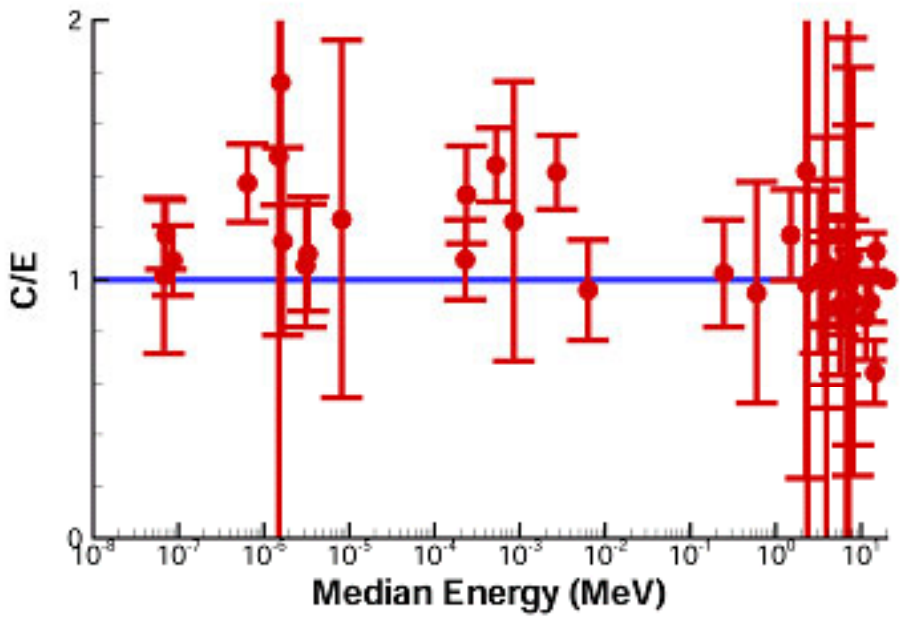

Figure 5: C/E for Dosimetry Reactions

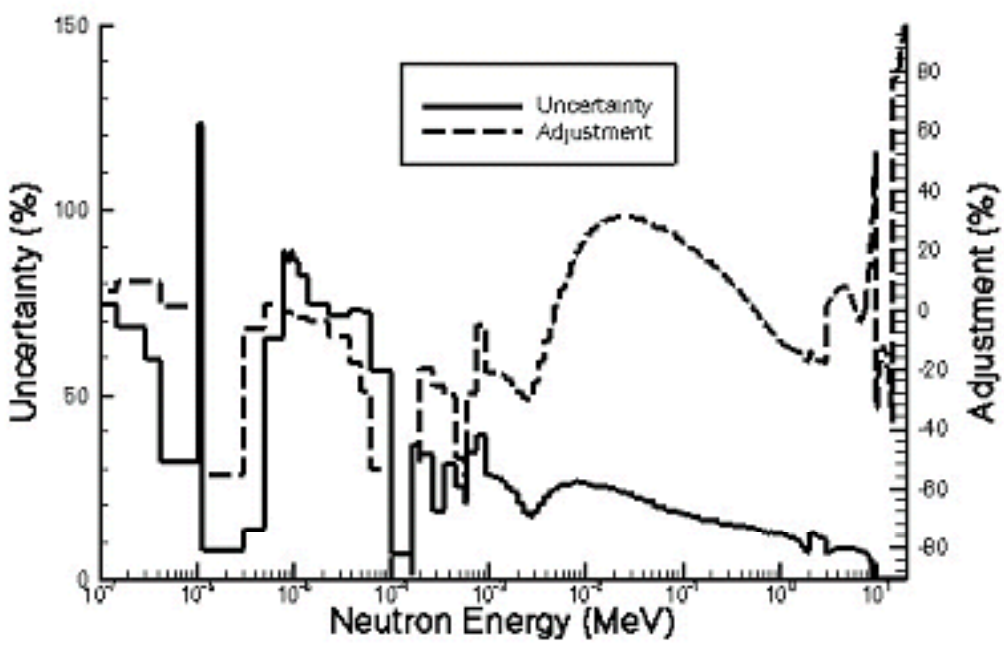

Figure 6: Spectrum Adjustment and Uncertainty

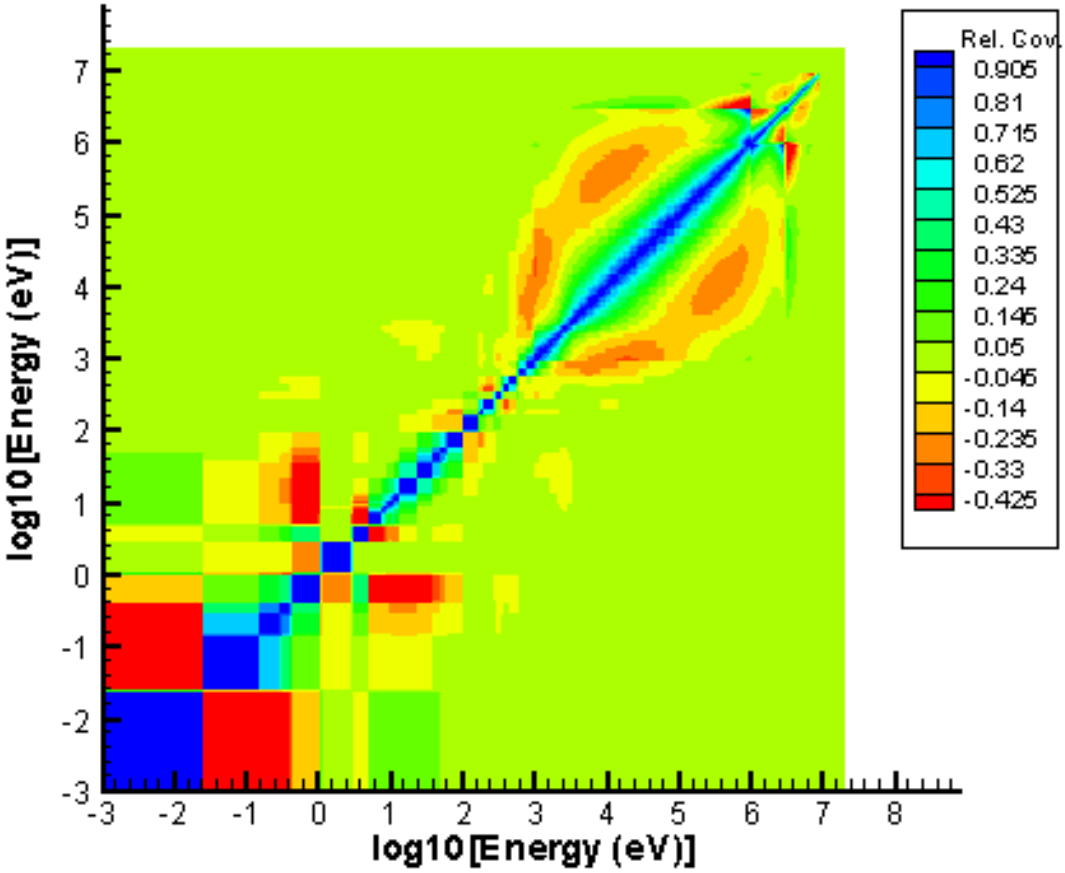

Figure 7: ACRR Spectrum Relative Covariance matrix

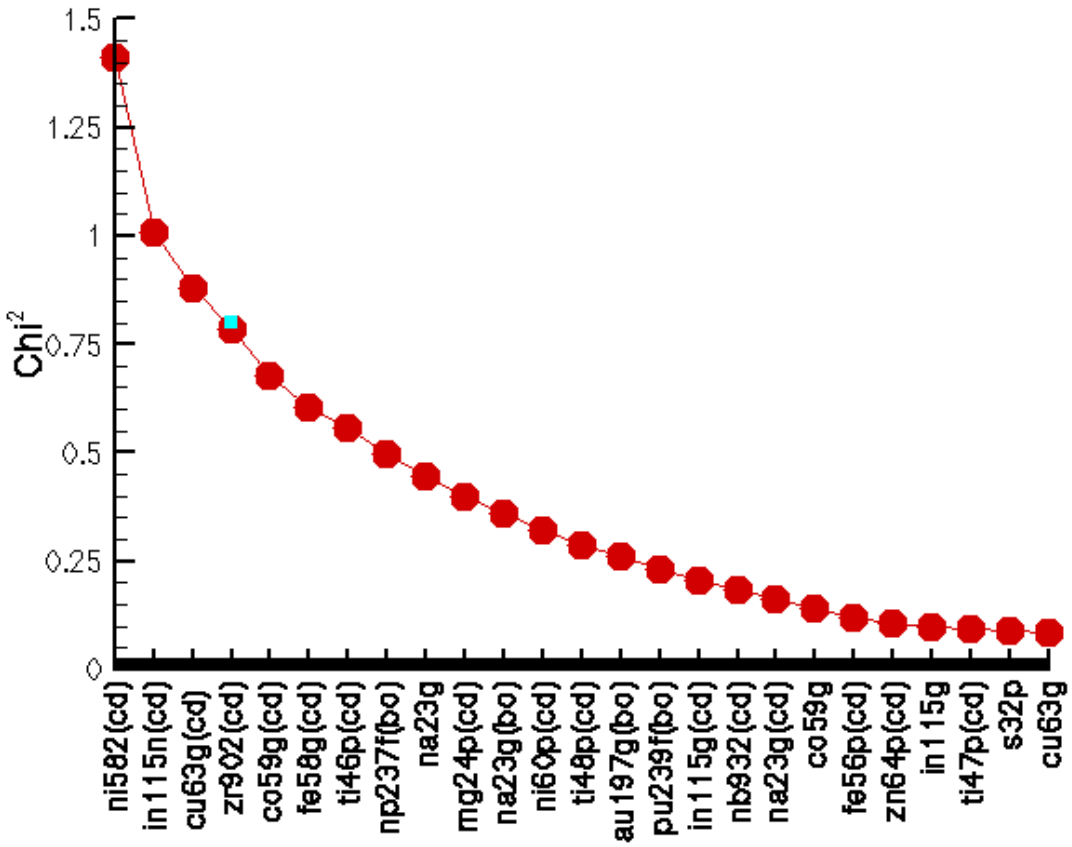

Figure 8: Chi^2 Per Degree of Freedom for Successively Removed Reactions

IV. COMPARISON OF IRDF-2002 CANDIDATE REACTIONS

At this time there has been an initial "culling" of the potential cross section candidates but no final selection. Despite this lack of a "final" selection, the "reference" ACRR neutron spectrum can still be examined to see the role that it may play in "validating" the final selection.

IV.i Variation Between Candidate Cross Sections

The first step in this process is to examine the overlap of the IRDF-2002 cross section candidates with the measured activities in the ACRR neutron field. The variation between the spectrum-averaged cross sections for the candidate cross sections is compared to the uncertainty in the spectrum-averaged cross section in this field. Table 2 shows the reactions for which there are measured activities along with the cross sections. For each intersection of a measurement and a candidate cross section, the spectrum-averaged cross section along with the uncertainty is presented. The total uncertainty is composed of the combination (in quadrature since they are uncorrelated) of the uncertainty due to knowledge of the neutron spectrum and due to knowledge of the dosimetry cross section. To facilitate the examination of the usefulness of this neutron field in the "validation", both of these uncertainty components are identified.

Table 2: Spectrum-averaged Cross Sections for Candidate Evaluations in the ACRR Central Cavity

Reaction Number	Reaction/ Cover1	Candidate Xsec	Cross Section (mb)	5% - 95% Response Region (MeV)	Uncertainty (%)			Variation Between Candidates (%) (max - min)/min
					Xsec.	Spct.	Total	
1	Na23g	---	---	---	---	---	---	0
2	Na23g[Cd]	---	---	---	---	---	---	0
3	Na23g[Fi]	---	---	---	---	---	---	0
4	Mg24p[Cd]	IRDF90	2.77E-04	6.46 - 11.6	2.36	3.51	4.23	1.1
		JENDL/D-99	2.80E-04	6.42 - 11.5	1.28	3.48	3.71	
5	Al27a[Cd]	IRDF90	1.288E-04	6.4 - 12.0	1.41	3.64	3.9	0
		JENDL/D-99	1.288E-04	6.4 - 12.0	1.41	3.64	3.9	
6	S32p	IRDF90	1.32E-02	2.2 - 7.2	error	2.97	error	0
7	Sc45g[Cd]	IRDF90	0.4006	5E-7 - 4E-3	0.88	5.79	5.86	0
8	Sc45g	IRDF90	2.066	1.E-8 - 4.E-6	0.82	9.77	9.80	0
9	Ti46p[Cd]	JENDL/D-99	2.18E-03	3.77 - 9.2	2.23	3.67	4.29	1.4
		RRDF-98	2.21E-03	3.8 - 9.4	3.20	3.56	4.79	
10	Ti48p[Cd]	IRDF90	4.98E-05	5.85 - 12.2	2.53	2.69	3.69	10.
		JENDL/D-99	5.133E-5	5.7 - 12.2	2.07	2.61	3.33	
		RRDF-98	5.50E-05	5.84 - 12.1	5.26	2.74	5.93	
11	Ti47p[Cd]	IRDF90	3.77E-03	1.54 - 7.4	3.67	2.22	4.29	0
12	Mn55g[Cd]	IRDF90	0.2959	6E-7 - 2.E-3	4.73	5.32	7.12	0
		JENDL/D-99	0.2959	6E-7 - 2E-3	4.73	5.32	7.12	
13	Fe54p[Cd]	IRDF90	1.60E-02	2.17 - 7.3	2.12	2.62	3.37	0.6
		JENDL/D-99	1.61E-02	2.22 - 7.4	1.00	2.63	2.81	
14	Fe56p[Cd]	RRDF-98	2.05E-04	5.44 - 11.2	2.72	2.99	4.04	2.5
		JEFF-3.0	2.00E-04	5.7 - 11.3	1.11	2.91	3.11	
15	Fe56p[Fi]	RRDF-98	1.86E-04	5.46 - 11.3	2.71	2.94	4.00	2.8
		JEFF-3.0	1.81E-04	5.4 - 11.3	1.11	2.86	3.07	
16	Fe58g[Cd]	IRDF90	3.41E-02	7E-7 - 9E-2	6.40	8.23	10.4	5.2
		JENDL/D-99	3.24E-02	6E-7 - 6E-2	9.88	7.45	12.4	
17	Co59p[Cd]	---	---	---	---	---	---	0
18	Co59g[Cd]	IRDF90	1.3491	7E-7 - 1E-4	0.76	4.18	4.25	0
19	Co59g	IRDF90	4.1706	1E-8 - 1E-4	0.72	7.55	7.58	0
20	Co592[Cd]	IRDF90	3.93E-05	11.2 - 16.5	8.27	4.64	9.48	0
21	Ni58p	IRDF90	0.022	1.8 – 7.3	2.19	2.34	3.20	1.6
		JENDL/D-99	0.02172	1.8 - 7.3	0.6	2.38	2.45	
		ENDF/B-VI	0.02207	1.8 - 7.3	2.43	2.34	3.37	
		JEFF-3.0	0.02238	1.8 - 7.3	3.49	2.32	4.19	
22	Ni58p[Cd]	IRDF90	2.15E-02	1.8 - 7.3	2.18	2.34	3.20	3.3

		JENDL/D-99	2.12E-02	1.8 - 7.3	0.60	2.38	2.45	
		ENDF/B-VI	2.15E-02	1.8 - 7.3	2.44	2.34	3.38	
		JEFF-3.0	2.19E-02	1.8 - 7.3	3.49	2.32	4.19	
23	Ni582[Cd]	IRDF90	7.26E-07	13.0 – 18.3	3.06	5.14	5.98	
		JENDL/D-99	7.03E-07	13.0 - 18.3	1.01	5.19	5.29	3.3
		JEFF-3.0	7.25E-07	13.0 - 18.4	2.70	5.14	5.81	
24	Ni60p[Cd]	ENDF/B-VI	3.75E-04	4.7 - 10.8	10.5	3.16	11.0	1.3
		JEFF-3.0	4.25E-04	4.58 - 10.7	9.67	3.21	10.2	
25	Cu63g[Cd]	IRDF90	1.65E-01	8E-7 - 6E-2	3.83	7.88	8.76	0
		ENDF/B-VI	1.65E-01	8E-7 - 6E-2	3.83	7.88	8.76	
26	Cu63g	IRDF90	0.4476	1E-8 - 5E-3	3.77	8.12	8.95	0.02
		ENDF/B-VI	0.4475	1E-8 - 5E-3	4.96	8.12	9.52	
27	Zn64p[Cd]	IRDF90	7.58E-03	2.28 - 7.3	4.72	2.85	5.51	0
28	Zr902[Cd]	IRDF90	1.72E-05	12.6 - 17.9	1.67	4.75	5.04	1.8
		JENDL/D-99	1.69E-05	12.6 - 17.9	0.57	4.75	4.78	
29	Nb93g[Cd]	---	---	---	---	---	---	---
30	Nb932[Cd]	IRDF90	8.60E-05	9.7 - 14.3	2.93	5.72	6.43	0.82
		RRDF-98	8.53E-05	9.7 - 14.3	5.71???	5.71		
31	In115g	---	---	---	---	---	---	---
32	In115g[Cd]	---	---	---	---	---	---	---
33	In115n[Cd]	IRDF90	4.45E-02	0.98 - 5.6	0	3.18		0
		JENDL/D-99	4.45E-02	0.98 - 5.6	2.8	3.18	4.24	
34	Au197g	IRDF90	45.3324	4E-8 - 8E-6	0.16	6.37	6.37	0
35	Au197g[Cd]	IRDF90	16.60949	1E-6 - 3E-4	0.17	4.16	4.16	0
36	Au197g[Fi]	IRDF90	0.1867	6E-4 - 0.9	0.53	4.07	4.10	0
37	Np237f[Fi]	JENDL/D-99	0.3714	0.51 - 4.9	0.26	3.96	3.97	0
38	U235f[Fi]	IRDF90	0.8315	2E-3 - 3.1	0.29	2.96	2.97	0
39	U238f[Fi]	IRDF90	6.457E-2	1.39 - 6.37	0.53	3.36	3.40	1.2
		JENDL/D-99	6.535E-2	1.39 - 6.36	2.08	3.37	3.96	
40	Pu239f[Fi]	IRDF90	0.9338	4E-3 - 3.6	0.39	2.97	3.0	1.1
		JENDL/D-99	0.9438	4E-3 - 3.6	2.36	2.95	3.78	
41	Mo98g[Cd]2	---	---	---	---	---	---	
42	Ag109g2	IRDF90	1.4589	3E-8 - 6E-5	6.22	44.5	45.	0
43	Ag109g[Cd]2	IRDF90	1.14	1E-6 - 2E-4	6.75	52.5	52.9	0
44	W186g2	RRDF-98	9.446	2E-8 - 2E-5	2.89	23.3	23.5	0
Footnotes are identical to those in Table 1								

Table 2 shows that only for the ^{48}Ti(n,p) reaction does the variation between candidate cross sections exceed the uncertainty of the spectrum-averaged cross section in the reference field where both the reference field spectrum uncertainty and the dosimetry cross section uncertainty components are considered.

IV.ii Normalized C/E Ratios in Reference Neutron Field

Since the ACRR spectrum does not have time-of-flight spectrum measurement (as all standard benchmarks must have), an absolute calculated-to-experimental ratio (C/E) can not be formed. The ACRR reactor exposures typically use the ^{58}Ni(n,p) reaction as an irradiation monitor, thus the LSL

spectrum adjustment was normalized to this measured activity. In the absence of a spectrum unfold for each combination of cross section candidates, we can form ratios of the individual activities to the ^{58}Ni(n,p) reference/monitor and then examine the C/E ratio of this metric. Table 3 shows the results of this approach.

Table 3: Ratio of Spectrum-averaged Cross Sections to Monitor Ni58p Reaction for Candidate Evaluations in the ACRR Central Cavity

Reaction Number	Reaction/ Cover1	Candidate Xsec	Measured Ratio[2] (mb/mb)	Calc. Ratio (mb/mb)	C/E
1	Na23g	---	1.6533 +/- 5.45%	---	---
2	Na23g[Cd]	---	0.35159 +/- 5.45%	---	---
3	Na23g[Fi]	---	0.032385 +/- 5.9%	---	---
4	Mg24p[Cd]	IRDF90	11.158E-3 +/- 6.3%	12.55E-3 +/-4.23%	1.12 +/- 7.6%
		JENDL/D-99		12.69E-3 +/- 3.7%	1.137 +/- 7.3%
5	Al27a[Cd]	IRDF90	5.545E-3 +/- 6.28%	5.836E-3 +/- 3.9%	1.052 +/- 7.4%
		JENDL/D-99		5.836E-3 +/- 3.9%	1.052 +/- 7.4%
6	S32p	IRDF-90	0.57506 +/- 5.9%	0.5981 +/- err%	1.04 +/- err%
7	Sc45g[Cd]	IRDF90	14.5674 +/- 6.1%	18.1515 +/- 5.86%	1.25 +/- 8.5%
8	Sc45g	IRDF90	91.4122 +/- 6.1%	93.617 +/- 9.8%	1.02 +/- 11.5%
9	Ti46p[Cd]	JENDL/D-99	9.529E-2 +/- 5.43%	9.88E-2 +/- 4.3%	1.04 +/- 6.9%
		RRDF-98		10.01E-2 +/- 4.8%	1.05 +/- 7.25%
10	Ti48p[Cd]	IRDF90		2.256E-3 +/- 3.7%	0.959 +/- 7.0%
		JENDL/D-99	2.353E-3 +/- 5.95%	2.326E-5 +/- 3.33	0.989 +/- 6.8%
		RRDF-98		2.492E-3 +/- 5.9%	1.059 +/- 8.4%
11	Ti47p[Cd]	IRDF90	0.171183 +/- 7.1%	0.1708 +/- 4.3%	0.9978 +/- 8.3%
12	Mn55g[Cd]	IRDF90	13.31425 +/- 5.4%	13.408 +/- 7.1%	1.007 +/- 8.9%
		JENDL/D-99		13.408 +/- 7.1%	1.007 +/- 8.9%
13	Fe54p[Cd]	IRDF90	0.736005 +/- 6.3%	0.725 +/- 3.4%	0.985 +/- 9.7%
		JENDL/D-99		0.7295 +/- 2.8%	0.991 +/- 6.9%
14	Fe56p[Cd]	RRDF-98	8.4351E-3 +/- 5.4%	9.29E-3 +/- 4.0%	1.10 +/- 6.7%
		JEFF-3.0		9.062E-3 +/- 3.1%	1.07 +/- 6.2%
15	Fe56p[Fi]	RRDF-98	7.6018E-3 +/- 5.4%	8.428E-3 +/- 4.0%	1.11 +/- 6.7%
		JEFF-3.0		8.201E-3 +/- 3.1%	1.08 +/- 6.2%
16	Fe58g[Cd]	IRDF90	1.64949 +/- 5.76%	1.5451 +/- 10.4%	0.937 +/- 11.9%
		JENDL/D-99		1.468 +/- 12.4%	0.89 +/- 13.7%
17	Co59p[Cd]	---	1.1629E-2 +/- 6.3%	---	---
18	Co59g[Cd]	IRDF90	52.32188 +/- 5.5%	61.131 +/- 4.25%	1.17 +/- 7.0%
19	Co59g	IRDF90	160.87786 +/-6.1%	188.97 +/- 7.58%	1.17 +/- 9.7%
20	Co592[Cd]	IRDF90	1.7805E-3 +/- 9.8%	1.781E-3 +/-9.48%	1.00 +/- 13.6%
21	Ni58p	IRDF90		0.9968 +/- 3.2%	0.9968 +/- 6.3%
		JENDL/D-99	1.00 +/- 5.4%	0.9841 +/- 2.45%	0.9841 +/- 5.9%
		ENDF/B-VI		1.0 +/- 3.37%	1.0 +/- 6.4%
		JEFF-3.0		1.014 +/- 4.19%	1.014 +/- 6.8%
22	Ni58p[Cd]	IRDF90		0.974 +/- 3.2%	0.987 +/- 6.3%
		JENDL/D-99	0.9873 +/- 5.4%	0.9605 +/- 2.45%	0.973 +/- 5.9%
		ENDF/B-VI		0.9742 +/- 3.38%	0.987 +/- 6.4%

		JEFF-3.0		0.9923 +/- 4.2%	1.005 +/- 6.8%
23	Ni582[Cd]	IRDF90	3.1870E-5 +/- 6.5%	3.290E-5 +/-5.98%	1.03 +/- 8.8%
		JENDL/D-99		3.185E-5 +/- 5.29%	0.999 +/- 8.4%
		JEFF-3.0		3.285E-5 +/- 5.81%	1.031 +/- 8.7%
24	Ni60p[Cd]	ENDF/B-VI	1.8111E-2 +/- 5.7%	1.699E-2 +/- 11%	0.938 +/- 12.4%
		JEFF-3.0		1.926E-2 +/- 10%	1.06 +/- 11.5%
25	Cu63g[Cd]	IRDF90	5.505725 +/- 6.4%	7.476 +/- 8.76%	1.36 +/- 10.8%
		ENDF/B-VI		7.476 +/- 8.76%	1.36 +/- 10.8%
26	Cu63g	IRDF90	19.071247 +/- 5.6%	20.28 +/- 8.95%	1.06 +/- 10.6%
		ENDF/B-VI		20.276 +/- 9.52%	1.06 +/- 11.0%
27	Zn64p[Cd]	IRDF90	0.327354 +/- 5.4%	0.3435 +/- 5.51%	1.05 +/- 7.7%
28	Zr902[Cd]	IRDF90	1.2971E-3 +/- 9.4%	0.779E-3 +/- 5.04%	0.600 +/- 10.7%
		JENDL/D-99		0.766E-3 +/- 4.78%	0.591 +/- 10.5%
29	Nb93g[Cd]	---	13.066 +/- 7.8%	---	---
30	Nb932[Cd]	IRDF90	3.6539E-3 +/- 6.2%	3.90E-3 +/- 6.43%	1.07 +/- 8.9%
		RRDF-98		3.86E-3 +/- x%	1.06 +/- x%
31	In115g	---	2524.809 +/- 5.9%	---	---
32	In115g[Cd]	---	1793.893 +/- 6.3%	---	---
33	In115n[Cd]	IRDF90	1.223918 +/- 7.0%	2.016 +/- x%	1.65 +/- x%
		JENDL/D-99		2.016 +/- 4.24%	1.65 +/- 8.2%
34	Au197g	IRDF90	2062.977 +/- 6.7%	2054. +/- 6.37%	0.9956 +/- 9.2%
35	Au197g[Cd]	IRDF90	1718.8295 +/- 6.3%	752.58 +/- 4.36%	0.4378 +/- 7.7%
36	Au197g[Fi]	IRDF90	9.48473 +/- 6.7%	8.461 +/- 4.1%	0.892 +/- 7.9%
37	Np237f[Fi]	JENDL/D-99	15.77608 +/- 5.7%	16.827 +/- 3.97%	1.07 +/- 6.9%
38	U235f[Fi]	IRDF90	36.46947 +/- 5.6%	37.68 +/- 2.97%	1.03 +/- 6.3%
39	U238f[Fi]	IRDF90	3.37150 +/- 5.7%	2.926 +/- 3.4%	0.868 +/- 6.6%
		JENDL/D-99		2.961 +/- 3.96%	0.893 +/- 6.9%
40	Pu239f[Fi]	IRDF90	44.4402 +/- 5.4%	42.309 +/- 3%	0.952 +/- 6.2%
		JENDL/D-99		42.767 +/- 3.78%	0.952 +/- 6.6%
41	Mo98g[Cd][2]	---	1.5604E-3 +/- 7.1%	---	---
42	Ag109g[2]	IRDF90	31.03 +/- 2.5%	66.103 +/- 45%	2.13 +/- 45%
43	Ag109g[Cd][2]	IRDF90	20.446 +/- 3.3%	51.65 +/- 53	2.53 +/- 53%
44	W186g[2]	RRDF-98	203.7 +/- 6.5%	428.0 +/- 23.5%	2.10 +/- 24%

1 Cover nomenclature is identical to that used in Table 1

2 Uncertainty only includes that of main reaction cross section. The Ni58p/ENDF/B-VI cross section is treat ed as a reference with zero uncertainty

An examination of these C/E ratios in Table 3 clearly indicate that there are serious problems for several reactions where the C/E deviation from unity exceeds a three-sigma variation. These reactions are shaded. The problem may reside in 1) the spectrum characterization, 2) the candidate dosimetry cross sections, or 3) the cross section processing used in this analysis - but this analysis clearly fails to validate the following reactions:

- ^{90}Zr(n,2n) - problem appears to be in high energy portion of spectrum
- ^{115}In(n,n') - problem appears to be with the cross section. An identical problem was seen in the original ACRR spectrum adjustment. In that case the discrepancy was traced to the failure

to apply the 0.79 branching ratio to the metastable residual isotope. We suspect that the problem has re-occurred here. When this branching ratio adjustment is backed out the results are a C/E of 1.3 +/- 8%. This is not a good C/E but it is consistent with that from the original ACRR spectrum adjustment.

- ^{197}Au(n,γ) - problem appears to be with the Cadmium cover cross section. Difference is not inconsistent with the observations during the original ACRR spectrum adjustment.
- ^{109}Ag(n,γ) - problem is not identified, may be cross section or activity measurement. This activity was not used or validated in the original ACRR spectrum characterization. Foil self-shielding may be an issue - no correction was made or investigated since there was no available covariance file at the time.
- ^{186}W(n,γ) - problem is not identified, may be cross section or activity measurement. This activity was not used or validated in the original ACRR spectrum characterization. Foil self-shielding may be an issue - no correction was made or investigated since there was no available covariance file at the time.

We conclude that the ^{115}In(n,n'), ^{109}Ag(n,γ), and ^{186}W(n,γ) cross sections need to be examined in more detail. We note that the ^{115}In(n,n') cross section appears to have failed to apply the 0.79 metastable branching ratio - a duplication of a problem addressed in the original IRDF-90 library.

Minor issues are also raised for some reactions. These reactions are also shaded and correspond to cases where the C/E ratio deviated from unity by about two standard deviations. These reactions include:

- ^{45}Sc(n,γ) - problem appears to be with the Cadmium cover cross section. Difference is not inconsistent with the observations during the original ACRR spectrum adjustment.
- ^{63}Cu(n,γ) - problem appears to be with the Cadmium cover cross section. Difference is not inconsistent with the observations during the original ACRR spectrum adjustment.
- ^{197}Au(n,γ) - problem appears to be with the Cadmium cover cross section. Difference is not inconsistent with the observations during the original ACRR spectrum adjustment.
- ^{238}U(n,f) - problem is not currently understood but may, despite the boron cover, be related to contaminant levels of ^{235}U in the dosimeter (0.00205 atom fraction).

In each of these cases, the bare foil reaction activity appears to be acceptable and the problem only occurs when a Cd cover is used on the sensor. Thus the problem with most of these reactions appears to reside in the uncertainty in the ^{nat}Cd(n,abs) cross section for the dosimetry cover. The issue of the uncertainty in the ^{nat}Cd(n,abs) cross section just above (higher in energy) the large cadmium cut-off absorption energy was raised during the original spectrum adjustment [1].

IV.iii Spectrum Adjustment Using the Final IRDF-2002 Library

There are too many combinations of reactions to attempt to "validate" the cross sections through a spectrum adjustment for the ACRR using all of the variations in candidate cross sections. Table 4 shows the uncertainty components from the original spectrum adjustment. When a final selection of IRDF-2002 dosimetry cross sections has been made, a new spectrum adjustment can be performed. An examination of the chi-squared for each reaction can then be used to highlight cross sections that have improved the adjustment process as well as cross sections that may degrade the previous adjustment Note however, that the C/E and uncertainty values for this new spectrum adjustment should ONLY be used for the purpose validating the cross sectionselection and not to alter any of the selections.

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Table 4: Impact of IRDF-2002 Cross Sections on ACRR Spectrum Adjustment

Reaction[1]	Xsec. Lib.	expt . unc.	Original Spectrum Adj.[2]			
			xsec. unc.	c/e (%)	std. (%)	χ^2/dof
nb93g#-void-cdnm	ENDF-VI	7.83	9.50	20.35	30.23	-0.49
ni58p#-void-bare	JDL-D99	5.43	2.48	-0.99	18.25	0.09
ni58p#-void-cdnm	JDL-D99	5.44	2.48	0.27	18.25	-0.01
s32p#-void-bare	IRDF90	5.94	3.52	2.60	19.41	0.10
na23g#-void-bare	ENDF-VI	5.45	2.17	16.05	142.0	4.86
na23g#-void-cdna	ENDF-VI	5.45	5.40	20.92	40.41	-0.35
na23g#-void-fiss	ENDF-VI	5.90	15.30	34.62	37.28	2.03
mg24p#-void-cdnm	IRDF90	6.33	2.37	8.55	27.43	2.49
al27a#-void-cdnm	IRDF90	6.28	2.19	1.74	27.59	0.29
sc45g#-void-cdnm	IRDF90	6.09	1.13	13.78	60.74	-0.95
sc45g#-void-bare	IRDF90	6.09	0.98	1.55	152.6	-0.09
ti46p#-void-cdnm	RRDF-98	5.43	2.46	-10.64	23.69	1.93
ti48p#-void-cdnm	IRDF90	5.95	2.55	-8.48	26.90	0.99
ti47p#-void-cdnm	IRDF90	7.06	3.64	3.15	17.86	-0.05
mn55g#-void-cdnm	IRDF90	5.38	4.48	28.36	41.36	3.53
fe54p#-void-cdnm	JDL-D99	6.30	2.14	-1.10	19.43	0.05
fe56p-void-cdnm	JEFF-30	5.39	2.29	-2.05	26.05	-0.04
fe56p-void-fiss	JDL-D99	5.42	2.29	4.69	26.06	1.03
fe58g-void-cdnm	JDL-D99	5.76	5.88	7.51	40.05	-0.98
co59p-void-cdnm	ENDF-VI	6.33	4.10	1.79	23.61	0.18
co59g-void-cdnm	IRDF90	5.53	0.77	78.73	54.98	11.75
co59g-void-bare	IRDF90	6.09	0.73	31.70	92.70	-6.73
co592-void-cdnm	IRDF90	9.78	2.78	-9.08	34.77	1.20
ni582-void-cdnm	JDL-D99	6.50	2.75	10.32	36.03	7.47
ni60p-void-cdnm	JEFF-30	5.68	10.49	-13.76	27.27	0.99
cu63g-void-cdnm	ENDF-VI	6.42	4.17	36.64	37.93	4.19
cu63g-void-bare	ENDF-VI	5.57	4.00	7.13	114.5	-0.79
zn64p-void-cdnm	IRDF90	5.37	4.79	5.41	19.76	0.53
zr902-void-cdnm	JDL-D99	9.41	1.57	-44.23	36.13	15.02
nb932-void-cdnm	IRDF90	6.24	2.67	-15.52	32.04	8.91
in115g-void-bare	ENDF-VI	5.91	5.98	38.85	70.11	-2.85
in115g-void-cdnm	ENDF-VI	6.30	5.98	56.58	74.25	4.20
in115n-void-cdnm	JDL-D99	7.04	2.18	35.02	16.39	16.36
au197g-void-bare	IRDF90	6.70	0.16	5.34	62.73	-0.22
au197g-void-cdnm	IRDF90	6.31	0.17	9.50	66.30	0.33
au197g-void-fiss	IRDF90	6.71	0.49	-4.00	22.12	0.59
np237f-void-fiss	JDL-D99	5.66	9.58	15.99	18.67	0.91
u235f-void-fiss	IRDF90	5.64	0.29	2.43	15.45	0.30

u238f-void-fiss	JDL-D99	5.66	0.53	-1.69	16.04	0.59
pu239f-void-fiss	JDL-D99	5.42	0.39	-5.17	14.70	0.96
Total Adjustment			---			1.975

1 Reaction identifies include the foil descriptor and cover designation. See Reference 8 for a description of the nomenclature.

2 The chi-squared reported here (1.975) is for the original 366-group spectrum adjustment. A smaller chi-squared (1.68) resulted from the 89-group adjustment used for the recursive removal analysis reported in Section III.iii

V. CONCLUSION

The well characterized and documented ACRR Central Cavity reference neutron field with measured activity data for 44 dosimetry-quality reactions has been used to "validate" the final selection of IRDF-2002 cross sections. In support of this "validation" uncertainty and covariance data were available for both the spectrum and the experimental activity measurements. Since the final cross section down-selection was not available at this time, a final "validation" spectrum adjustment has not yet been performed.

While surveying all potential "trial" selections, the IRDF-2002 library candidates were found to be in good agreement with the previous ACRR spectrum characterization and the measured activities except in the cases of the following reactions:

- ^{115}In(n,n') - need to apply the 0.79 branching factor correction
- ^{109}Ag(n,γ) - need to review experiment activity, self-shielding, and any other potential validation cases
- ^{186}W(n,γ) - need to review experiment activity, self-shielding, and any other potential validation cases

These three reactions need to be considered in more detail before the IRDF-2002 library is released. The ^{115}In(n,n') correction is easily accomplished. The ^{109}Ag(n,γ) and ^{186}W(n,γ) reaction conflicts may reflect a problem with the ACRR measurements since this data was not validated in the original spectrum adjustment, but, if these foil activities have not been addressed in the "standard" benchmark fields, then some type of validation or a warning needs to be given to potential users that these reactions need further validation before their application is endorsed.

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Results of Validation of the IRDF-2002 Dosimetry Library Using the SPR-III Reference Neutron Field

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I. PURPOSE

After the selection of the recommended cross section evaluations for inclusion in the IRDF-2002 library, the process is not quite finished. The final step before library release is to validate the fidelity of the selection There are well characterized neutron fields, called "reference" neutron benchmark fields, that should be used to validate the dosimetry cross section selection. This paper address the validation of the IRDF-2002 dosimetry selections in one such "reference" neutron field, the SPR-III central cavity [1].

II. BACKGROUND

These "reference" fields should not be confused with the "standard" benchmark fields. The 1976 IAEA Consultants Meeting [2] and the recent ISRD11 Workshop [3] reflect a consensus that **ONLY** standard benchmark fields can be used for this step in differentiating between candidate evaluations. 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. Only four benchmark fields are recognized by the dosimetry community. These benchmark fields are [3]:

- The spontaneous fission neutron field of ^{252}Cf
- The 1/E slowing down spectrum in hydrogenous moderator
- The Maxwellian thermal spectrum at specified neutron temperature
- The monoenergetic 14-MeV neutron field from a D-T source

Note that the ^{235}U thermal fission benchmark field is not included in this list of "standard" neutron fields. This field has been also designated as a "reference" rather than a "standard" benchmark field because only one "standard" is permitted in a given energy region and the ^{252}Cf field is a much better characterized neutron field. The data of interest in a standard field are typically the spectrum-averaged cross sections. For the thermal Maxwellian spectrum at a temperature of 293.6 ^{o}C, the spectrum-averaged cross section is uniquely related to the 2200 m/s cross section. In the case of the 1/E field, the measured data are corrected for the thermal contribution of the spectrum and are used to deduce the resonance integrals, typically corrected to represent the integral between the energy bounds from 0.5 eV to 100 keV.

When comparing a measurement to a calculated quantity it is critical that one addresses the uncertainty on both measurement and calculation and that all sources of uncertainty are taken into account. The evaluation covariance file is folded with the neutron spectrum to obtain the cross section contribution to the uncertainty. All acceptable measurement data must include a measurement uncertainty. Great care must be taken in combining the experimental data in a statistically valid manner while addressing the issue of discrepant data [4]. A criteria for flagging and rejecting discrepant data must be established prior to forming the recommended experimental metrics.

Finally, the uncertainty in the representation of the neutron spectrum in the standard and reference fields must be taken into account. The comparison quantity of interest is the C/E (calculated-to-experimental) ratio. Each of the sources of uncertainty term should be combined to provide an uncertainty in the C/E ratio. The extracted metric used to compare evaluations with measured data should be the C/E ratio and the number of standard deviations between this ratio and unity.

Note that this paper reports on a "beta test" step for the library selection using a "reference" neutron field and **in no case** should this "confirmation"' step result in a change in the selected cross section. It should only serve to indicate areas where either the cross section or the reference field characterization should be further examined. Even if it were not for the need to have only one "standard" field in a given energy region, the "reference" neutron fields typically have neutron spectrum characterizations that were derived from activation foils in conjunction with spectrum unfold or adjustment methods. Since this spectrum characterization process introduces correlations between the spectrum and cross section evaluations that are not properly accounted for in the spectrum adjustment, this data can not be used in the cross section selection process, only in the validation process.

III. SPR-III CENTRAL CAVITY REFERENCE NEUTRON FIELD

This paper presents the Sandia Pulsed Reactor (SPR-III) as a well characterized reference neutron field appropriate for validating the selection of the IRDF-2002 cross sections. This fast burst ^{235}U metal assembly has a very similar neutron spectrum to the ^{235}U thermal fission reference benchmark field. A more detailed description of this reference field appears in Reference [1]. The details of the radiation transport models and 640-group representation of the neutron spectrum in the SPR-III central cavity appears in published laboratory reports [5].

III.i Experimental Characterization

SPR-III, shown in Figure 1, is an advanced fast-burst Godiva-type reactor with a large 16.5-cm central cavity. It was commissioned in 1975. Although it was developed primarily for the radiation testing of electronic parts and systems, it has been used in a wide variety of research activities. It is positioned in the center of an air-filed shield building called a Kiva. Experiments are conducted not only in the cavity but also outside the core at distances between 0.3 and 3.0 meters from the core axis. There are also ports in the shield wall for fielding of experiments that require collimated beam geometries. The reactor can be operated in steady-state (up to 10 kW power) or pulsed mode (10 MJ in an 80 μs FWHM pulse that yields approximately 5×10^{14} n/cm^2 in the cavity).

Once the uniform exposure region was clearly established, dosimetry foils were fielded. Even though the uniform region was fairly large, the activity measurements were performed in a series of runs. Separate reactor operations were required since only eight high purity Germanium detectors were available for reading the dosimetry foils and since many of the activated products have short half-lives or require long counting times. Each detector is calibrated against a mixed radionuclide gamma reference source, QCD.1, from Amersham International. Typically the detectors will yield values for the activity of the same foil that are reproducible to within 1-3%. If detector counts total more than 10^4 and background counts are low so that statistical uncertainties are small, then the uncertainty of an activity measurement is less than ~5%. Cd and ^{10}B covers were used to alter the region of energy response for some of the activation foils. The ^{10}B cover was a 5-cm diameter 91% ^{10}B- enriched B_4C ball. This boron ball was large enough to alter the neutron field in the surrounding region, so each boron-covered activation foil was exposed on a separate operation. Fission foils were not stacked in the boron ball since previous testing had shown that the neutron scattering between adjacent 1-gram fission foils thermalized enough neutrons to alter the dosimeter response of ^{235}U and ^{239}Pu foils. The ^{58}Ni(n,p) reaction was used to normalize the separate reactor operations to a uniform neutron fluence. When a boron ball was used, internal as well as external Ni foils were used for normalization. Table 1 details the 34 dosimetry reactions and cover combinations that were measured in support of the spectrum characterization.

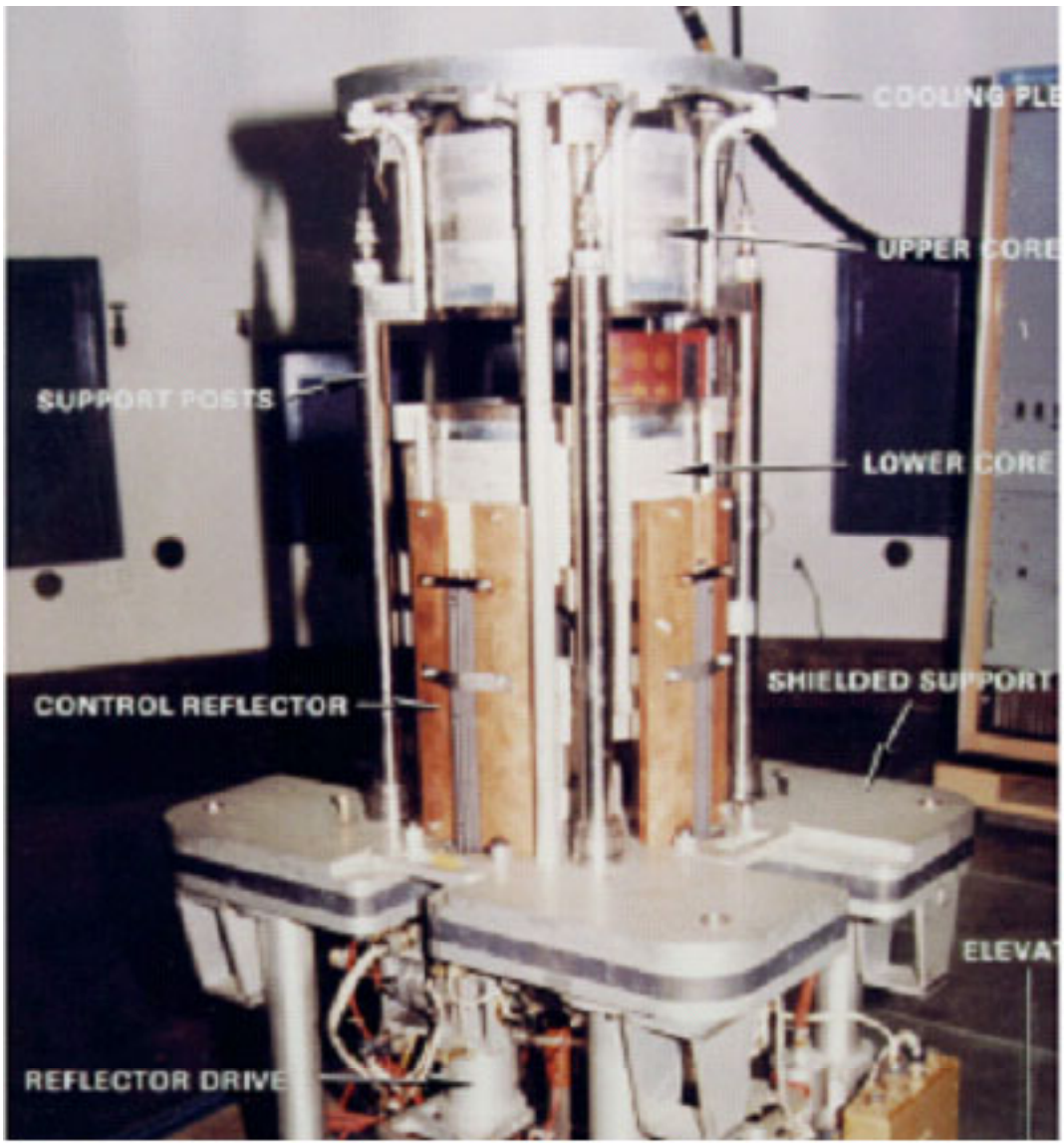


Figure 1: SPR-III Reactor

Table 1: Spectrum-averaged Dosimetry Cross Sections for SPR-III Central Cavity

Reac. #	Sensor Reaction	Foil Cover	Measured		MCNP Calculated		SAND-II Unfold	
			Activity (Bq/ nucleus)	$\Delta\sigma_{cnt}$ (%)	C/E Ratio	$\Delta\sigma_{score}$ (%)	C/E Ratio	Meas.-to-Calc. Dev. (%)
1	$^{197}Au(n,\gamma)^{198}Au$	Cd	6.574E-18	4.5	0.7428	1.16	1.0269	-2.621
2	$^{197}Au(n,\gamma)^{198}Au$		7.414E-18	4.5	0.6931	1.12	0.9766	2.410
3	$^{59}Co(n,\gamma)^{60}Co$	Cd	6.923E-22	2.9	0.5928	1.69	0.9747	2.568
4	$^{63}Cu(n,\gamma)^{64}Cu$	Cd	3.302E-18	2.2	0.8776	0.13	1.0266	-2.585
5	$^{115}In(n,n')^{115m}In$		6.204E-17	4.7	1.1280	0.14	0.9766	2.350
6	$^{Nat}Ti(n,X)^{46}Sc$	Cd	7.325E-21	3.4	0.9416	0.62	0.9524	5.009
7	$^{Nat}Ti(n,p)^{47}Sc$	Cd	2.882E-19	2.7	1.1590	0.27	1.0673	-6.314
8	$^{48}Ti(n,p)^{48}Sc$	Cd	8.684E-21	1.2	0.9456	1.43	0.9443	5.869
9	$^{32}S(n,p)^{32}P$		2.508E-19	3.0	1.0746	0.31	1.0050	-0.486
10	$^{58}Ni(n,p)^{58}Co$	Cd	8.752E-20	3.1	1.0311	0.26	0.9662	3.462
11	$^{54}Fe(n,p)^{54}Mn$	Cd	1.400E-20	3.2	1.0893	0.33	1.0331	-3.197
12	$^{56}Fe(n,p)^{56}Mn$	Cd	5.408E-19	2.4	0.9791	1.10	1.0097	-0.956
13	$^{64}Zn(n,p)^{64}Cu$	Cd	3.882E-18	2.2	1.1090	0.27	1.0452	-4.323
14	$^{24}Mg(n,p)^{24}Na$	Cd	1.242E-19	3.0	1.1055	1.48	1.0648	-6.086
15	$^{27}Al(n,\alpha)^{24}Na$	Cd	6.135E-20	1.9	1.0386	1.63	0.9940	0.561
16	$^{90}Zr(n,2n)^{89}Zr$	Cd	1.616E-21	3.4	0.8843	10.38	1.0016	-0.161
17	$^{235}U(n,f)^{140}La$	Cd	1.755E-11	3.0	0.9595	0.08	0.9872	1.280
18	EU: $^{235}U(n,f)^{140}La$	B_4C, Cd	1.500E-11		--tbd--	--tbd--	0.9872	1.266
19	$^{238}U(n,f)^{140}La$	Cd	2.317E-12	3.2	1.1104	0.15	0.9747	2.624
20	DU: $^{238}U(n,f)^{140}La$	B_4C, Cd	2.223E-12		--tbd--	--tbd--	0.9833	1.718
21	$^{239}Pu(n,f)^{140}La$	Cd	2.233E-11	2.7	1.0518	0.08	1.0377	-3.633
22	PU: $^{239}Pu(n,f)^{140}La$	B_4C, Cd	1.912E-11		--tbd--	0.08	1.0524	-4.979
23	$^{237}Np(n,f)^{140}La$	Cd	1.234E-11	2.8	1.1336	0.01	0.9709	3.039
24	$^{237}Np(n,f)^{140}La$	B_4C, Cd	1.182E-11		1.1336	0.01	0.9690	3.183
25	$^{45}Sc(n,\gamma)^{46}Sc$	B_4C, Cd	1.192E-20	3.5	0.8163	0.14	----	----
26	$^{45}Sc(n,\gamma)^{46}Sc$	Cd	1.372E-20	3.3	0.8149	0.16	----	----
27	$^{23}Na(n,\gamma)^{24}Na$	B_4C, Cd	5.963E-20	2.1	0.9551	0.18	----	----
28	$^{23}Na(n,\gamma)^{24}Na$	Cd	7.170E-20	2.1	0.8944	2.65	----	----
29	$^{55}Mn(n,\gamma)^{56}Mn$	B_4C, Cd	4.557E-18	2.7	0.7950	0.13	----	----
30	$^{55}Mn(n,\gamma)^{56}Mn$	Cd	6.618E-18	2.6	0.6383	0.4	----	----
31	$^{58}Fe(n,\gamma)^{59}Fe$	Cd	7.974E-21	2.8	--tbd--	--tbd--	----	----
32	$^{27}Al(n,p)^{27}Mg$	Cd	2.872E-17	3.0	--tbd--	--tbd--	----	----
33	$^{115}In(n,\gamma)^{115m}In$	Cd	4.916E-16	1.3	--tbd--	--tbd--	----	----
34	$^{23}Na(n,\gamma)^{24}Na$		7.087E-20	2.1	--tbd--	--tbd--	----	----

@ Cover composition: 91.67% ^{10}B-enriched B_4C=0.1481 atoms/bar; Cd=2.587E-3 atoms/barn

* Assumes the use of the 1993 GLUCS $^{32}S(n,p)^{32}P$ cross section, MAT=1625. This is consistent with use of the SNLRML cross section library.

$ This reaction was not used in the SAND-II spectrum unfold due to conflicts between reactions.

III.ii Spectrum Modeling

High fidelity calculations were performed to provide the best possible calculated neutron spectrum. This spectrum was then used as input to the least-squares spectrum adjustment codes and as a trial function to the iterative unfolding codes. Intuition and experience was used to estimate an *a priori* calculated spectrum uncertainty and covariance matrix. Calculations were also performed to ensure that the highest fidelity dosimeter responses were used.

The radiation transport was done with the 3D general geometry Monte Carlo point cross section MCNP code (version 4C) [6]. The reactor, support structures, Kiva, and dosimetry test fixture were modeled in detail. Figures 2 and 3 show a combinatorial geometry model for a cut-away model of the reactor. The figures show a test object in the central cavity. The neutron spectrum was scored in an 89-energy and in a 640-energy group structure. The 640-group spectrum was the input used in the LSL analysis.

Figure 2: SPR-III Reactor, Shroud, and Support Structure

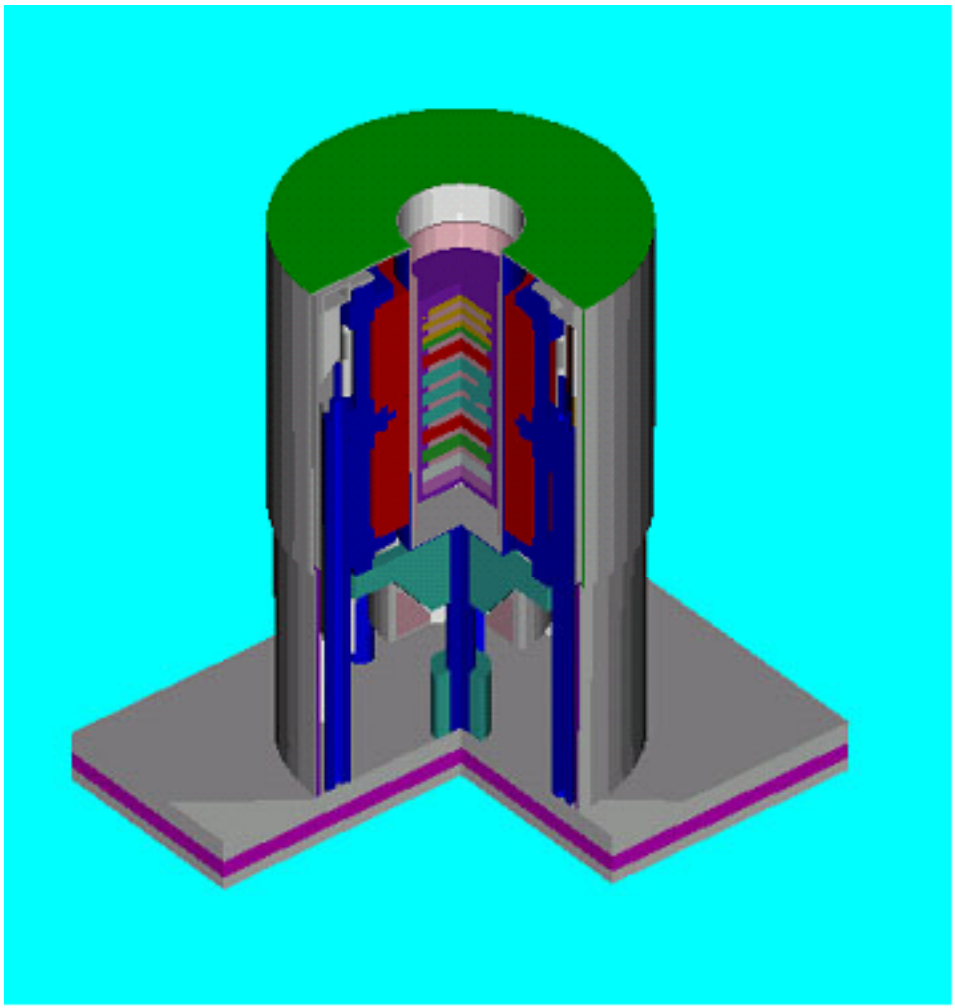
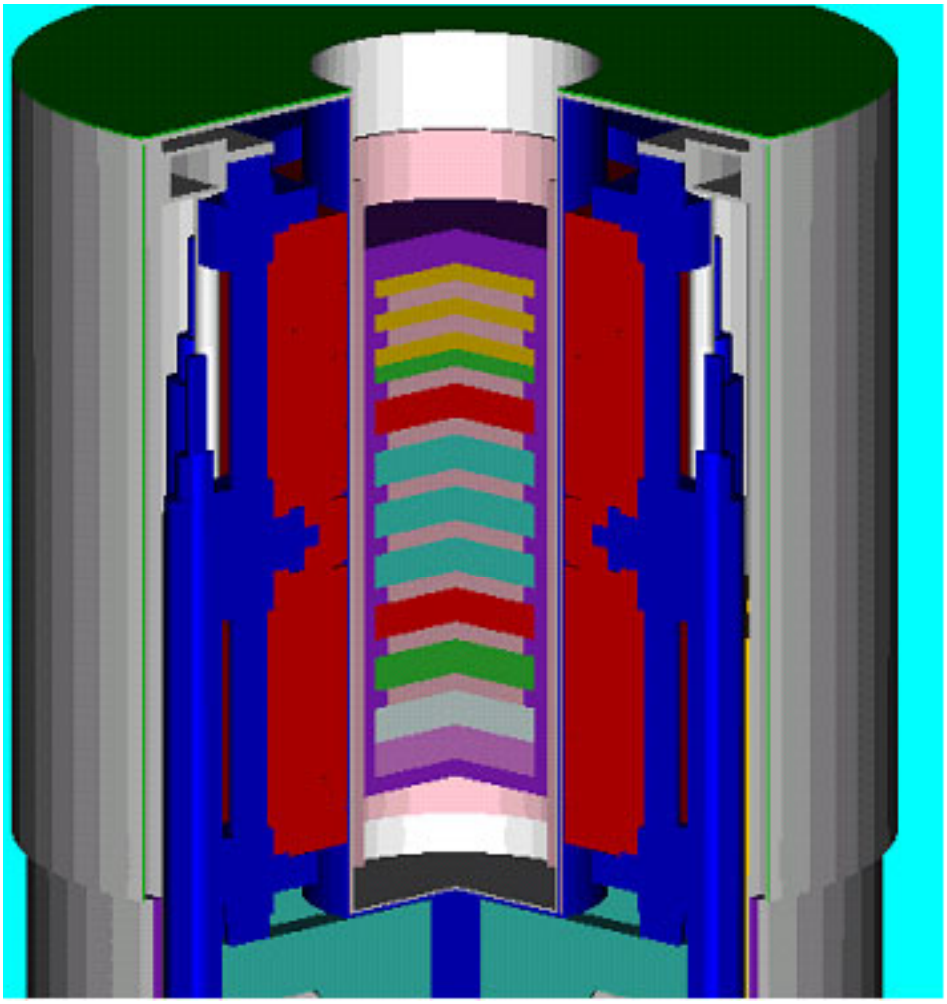
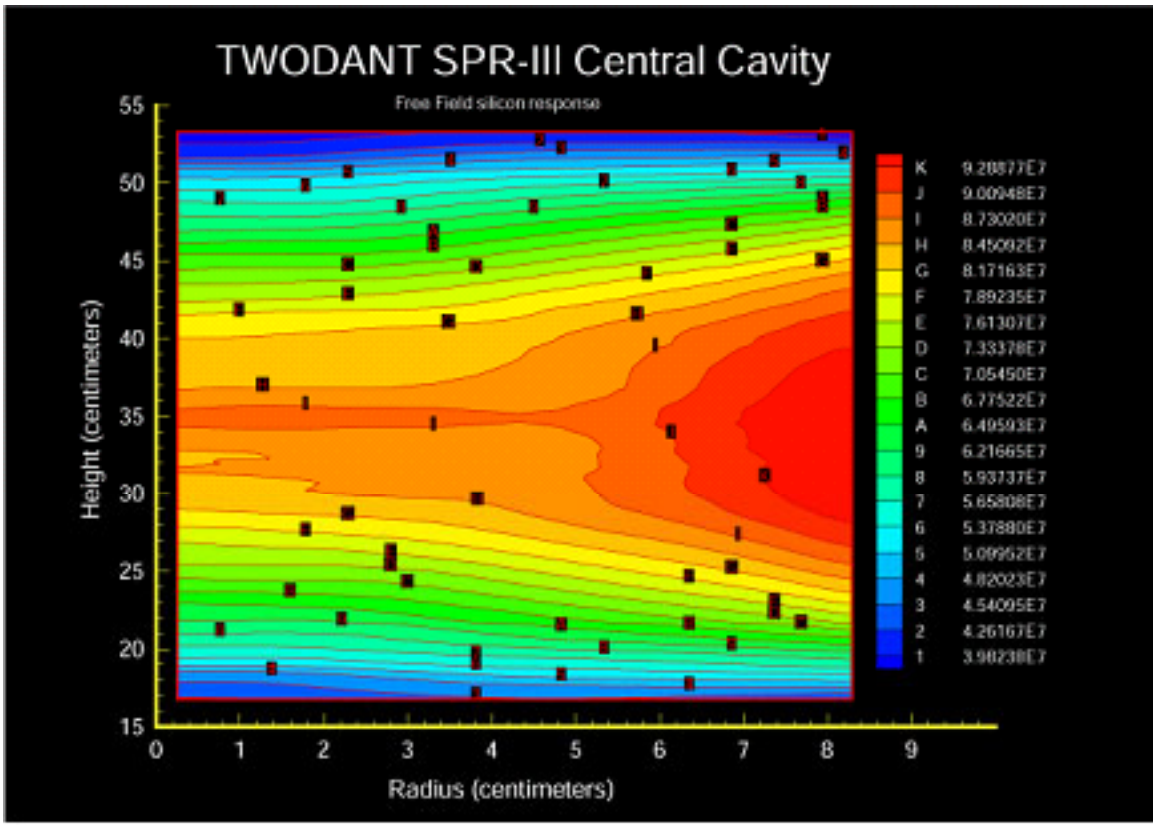


Figure 3: SPR-III Reactor with Internal Layered Test Object



Discrete ordinates modeling of SPR-III central cavity has also been done in order to estimate the uniformity of the field. Figure 4 shows the uniformity of the internal field a a function of the radius out from the central axis for the fast neutron 1-MeV(Si) response used for electronics testing. The field uniformity has also been computed for other responses and the fidelity of the calculation confirmed (for > 3-MeV neutrons) by mapping the field uniformity with the $^{32}S(n,p)$ dosimeters.

Figure 4: Uniformity of 1-MeV(Si) Damage Environment in SPR-II Central Cavity



III.iii Spectrum Adjustment/Unfold

The baseline SPR-III spectrum unfold was conducted with the SNL-SAND-II iterative code. The nuclear data was taken from the SNLRML dosimetry library [7].The input spectrum used in this approach can be arbitrary [9] and was adjusted to provide the best fit to the iterative unfold. The iterations in any calculation were limited in number to prevent the introduction of structural artifacts. Instead, the "trial" function was adjusted based on a comparison of artifacts with the cross section sensitivities. A smoothness criteria (in dn/dE space) was applied to the SAND-II unfolded spectrum. Conflicts between reactions were resolved by removing the less credible (more uncertain) reaction. After a final spectrum was determined, a Monte Carlo iteration was performed in order to derive an uncertainty and covariance in the spectrum. During the Monte Carlo iterated unfold the number of iterations was restricted to 5 while both the "trial" function and the activity uncertainties were sampled in a statistically valid manner (treating estimated *a priori* correlations in the "trial" function).

A Sandia modified version of the LSL code [8] was also used in the analysis. This spectrum adjustment produced a comparable spectrum and uncertainty. This modified LSL code has been changed to interface with the SNLRML dosimetry compendium and to use foil covers. This code was run using both 89 and 366 energy group representations. The 366-group representation was selected to include the energy break points from all of the representations of the reaction cross section covariance matrices. The much faster running 89-group calculations were used for parametric studies. A 640-group calculation was not done since it resulted in singularities in the matrix inversion operations..

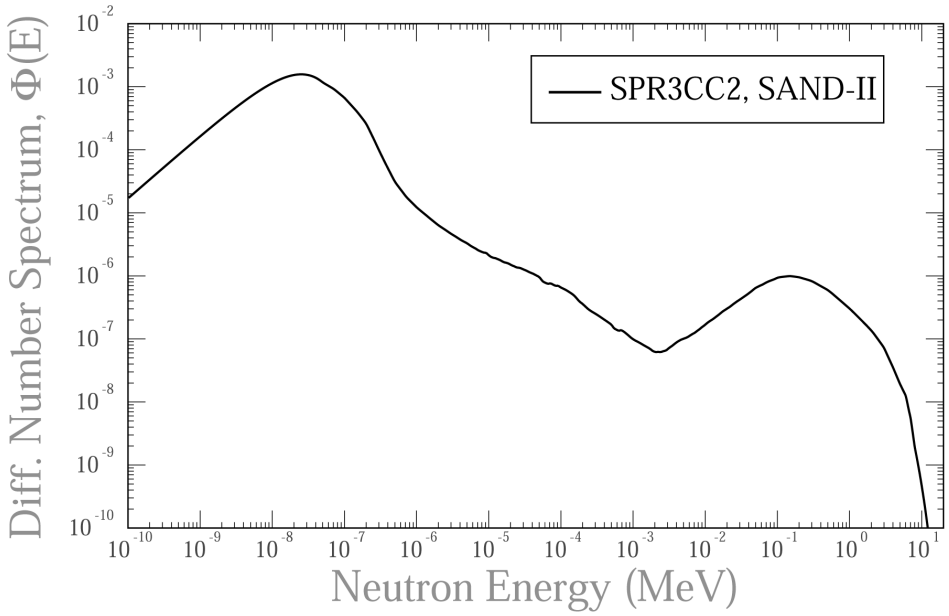


Figure 5: dn/dE Representation of Calculated Spectrum

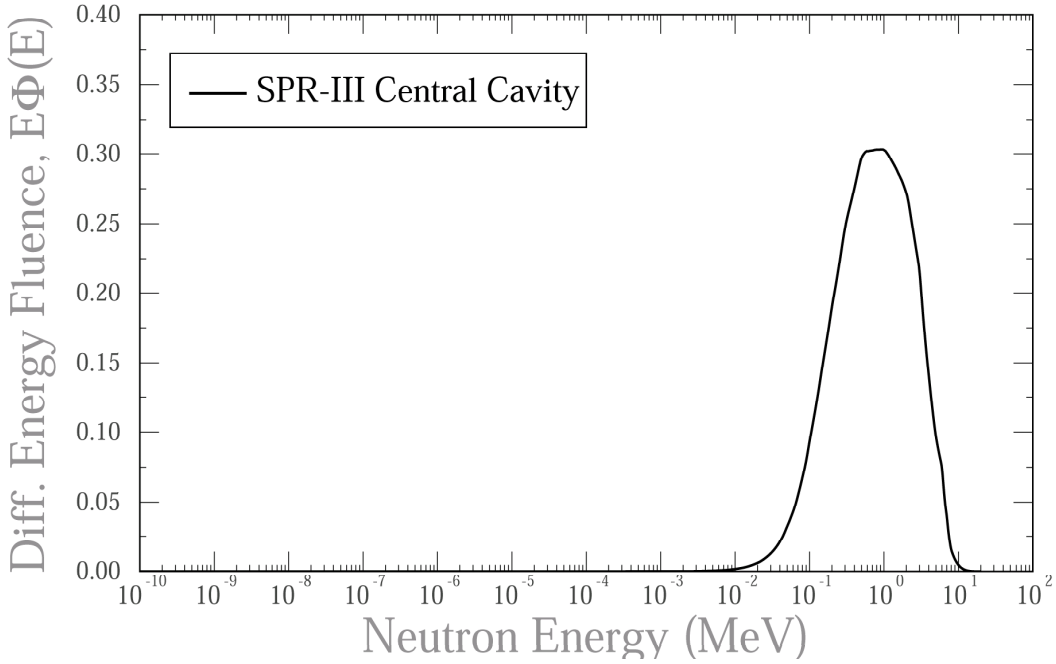


Figure 6: dE/dE Representation of Calculated Spectrum

Figure 5 shows the SAND-II unfolded neutron spectrum in a typical logarithmic fluence plot (notated as dn/dE or $\Phi(E)$). Figure 6 shows the calculated spectrum in a linear lethargy plot (often notated as dE/dE, $\Phi(\mu)$, or $E\Phi(E)$). In the lethargy plot (with linear y-axis and logarithmic energy x- axis or lethargy), equal areas under the curve correspond to equal neutron content.

In keeping with the goal of providing the highest quality neutron field characterization, a high fidelity treatment is used to model the influence by the dosimetry packageing on the measured foil activity. Responses from a 640 group calculation are used to account for the detailed response of the dosimetry covers and to account for self-shielding in resonance regions of the activation foil [10].

III.iv Model Validation

Several different types of test have been conducted to validate the spectrum characterization in the SPR-III central cavity. A transfer calibration of the 1-MeV(Si) damage to 2N2222A transistors verified the calculated damage ratio to within 5%. Comparisons of calculated k_{eff} for various reactor critical configurations (by definition a measured k_{eff} of unity) are shown in Table 2. Comparison of the measured and calculated worths for various reactor components is shown in Table 3. The good agreement of the calculated eigenvalues and reactivity worths with measurement attests to the fidelity of the reactor modeling.

Table 2: Summary of SPR-III MCNP Criticality Calculations

Reactor Configuration			k_{eff}	Rel. Err. (%)
Method	**Burst Element**	**Control element@**		
Rod Drop	Up*	6740, 6000, 6740	0.9994	0.06
	Down	6740, 6000, 6740	0.9917	0.07
	Down	5653, 6000, 5653	0.9993	0.06
	Down	5653, 6000, 5653 Saftey Block Down	0.9399	0.07
	Down	9000, 9000, 9000	0.9766	0.06
Asymetotic Period Measurement	Up*	6418, 6418, 6418	0.9993	0.05
	Up	5904, 5904, 5904	1.0047	0.05
	Down	5735, 5735, 5735	0.9997	0.05
	Down	5270, 5270, 5270	1.0052	0.05

@ Control element: position 0 = up, 9000 = down

* Starred configurations correspond to delayed critical configurations

Table 3: Comparison of the Calculated and Measured Reactivity Worths for Reactor Components

Reactor Configuration	Calculated		Measured	
	Δk($)	Δ	Δk($)	Δ
Burst element	1.20	0.20	1.114	0.005
Safety block	9.72	0.21	9.978	0.1
Partially inserted control element	3.57	0.19	3.487	0.019
Δ = relative error in percent				

IV. Comparison of IRDF-2002 Candidate Reactions

At this time there has been an initial "culling" of the potential cross section candidates but no final selection. Despite this lack of a "final" selection, the "reference" SPR-III central cavity neutron spectrum can still be examined to see the role that it may play in "validating" the final selection.

IV.i Variation Between Candidate Cross Sections

The first step in this process is to examine the overlap of the IRDF-2002 cross section candidates with the measured activities in the SPR-III neutron field. The variation between the spectrum-averaged cross sections for the candidate cross sections is compared to the uncertainty in the spectrum-averaged cross section in this field. Table 4 shows the reactions for which there are measured activities along with the cross sections. For each intersection of a measurement and a candidate cross section, the spectrum-averaged cross section along with the uncertainty is presented. The total uncertainty is composed of the combination (in quadrature since they are uncorrelated) of the uncertainty due to knowledge of the neutron spectrum and due to knowledge of the dosimetry cross section. To facilitate the examination of the usefulness of this neutron field in the "validation", both of these uncertainty components are identified.

Table 4: Spectrum-averaged Cross Sections for Candidate Eval. in the SPR-III Central Cavity

Reaction Number	Reaction/ Cover[1]	Candidate Xsec	Cross Section (mb)	5% - 95% Response Region (MeV)	Uncertainty (%)			Variation Between Candidates (%) (max - min)/min
					Xsec.	Spct.	Total	
1	Na23g	---	---	---	---	---	---	0
2	Na23g[Cd]	---	---	---	---	---	---	0
3	Na23g[Fis]	---	---	---	---	---	---	0
4	Mg24p[Cd]	IRDF90	7.38E-4	6.4 - 11.5	2.41	3.54	4.28	0.1
		JENDL/D-99	7.45E-4	6.4 - 11.5	1.29	3.54	3.77	
5	Al27a[Cd]	IRDF90	3.38E-4	6.35 - 12.0	1.41	3.60	3.87	0
		JENDL/D-99	3.38E-4	6.35 - 12.0	1.41	3.60	3.87	
6	Al27p[Cd]	IRDF90	1.99E-3	3.4 - 9.1	3.41	4.72	5.83	11
		JENDL-D99	2.21E-3	3.5 - 9.0	0.733	4.90	4.96	
7	S32p	IRDF90	3.25E-2	2.27 - 7.2	error	3.41	error	0

8	Sc45g[Cd]	IRDF90	1.05E-2	1.5E-2 - 1.5	7.27	4.97	8.81	0
9	Sc45g[Fis]	IRDF90	9.40E-3	3.E-2 - 1.7	8.00	5.14	9.51	0
10	Ti46p[Cd]	JENDL/D-99	5.77E-3	3.8 - 9.1	2.37	4.84	5.39	0.7
		RRDF-98	5.81E-3	3.8 - 9.3	3.23	4.71	5.71	
11	Ti48p[Cd]	IRDF90	1.33E-4	5.9 - 12.2	2.52	3.77	4.53	7.3
		JENDL/D-99	1.37E-4	5.7 - 12.2	2.04	3.76	4.28	
		RRDF-98	1.47E-4	5.8 - 12.1	5.34	3.73	6.52	
12	Ti47p[Cd]	IRDF90	9.11E-3	1.6 - 7.4	3.71	2.65	4.56	0
13	Mn55g[Cd]	IRDF90	5.50E-3	3.3E-4 - 2.1	10.6	5.27	11.8	0
		JENDL/D-99	5.50E-3	3.3E-4 - 2.1	10.6	5.27	11.8	
14	Mn55g[Fis]	IRDF90	4.687E-3	3.4E-2 - 2.3	9.7	4.5	10.7	1
		JENDL/D-99	4.687E-3	3.4E-2 - 2.3	9.7	4.5	10.7	
15	Fe54p[Cd]	IRDF90	3.99E-2	2.2 - 7.4	2.11	3.51	4.10	1
		JENDL/D-99	4.03E-2	2.28 - 7.4	0.981	3.58	3.71	
16	Fe56p[Cd]	RRDF-98	5.60E-4	5.5 - 11.1	2.78	4.41	5.20	2.8
		JEFF-3.0	5.45E-4	5.5 - 11.2	1.13	4.37	4.51	
17	Fe58g[Cd]	IRDF90	3.97E-3	1.9E-2 - 2.3	19.2	4.48	19.7	29
		JENDL/D-99	3.08E-3	1.1E-2 - 2.2	2.96	4.63	5.50	
18	Co59g[Cd]	IRDF90	1.14E-2	1.2E-4 - 2.2	1.90	5.07	5.41	0
19	Ni58p[Cd]	IRDF90	5.30E-2	1.9 - 7.3	2.19	3.17	3.85	2.7
		JENDL/D-99	5.25E-2	1.9 - 7.4	0.598	3.26	3.31	
		ENDF/B-VI	5.30E-2	1.9 - 7.3	2.41	3.17	3.99	
		JEFF-3.0	5.39E-2	1.85 - 7.3	3.41	3.17	4.66	
20	Cu63g[Cd]	IRDF90	1.62E-2	2.2E-2 - 2.4	10.7	4.29	11.5	0
		ENDF/B-VI	1.62E-2	2.2E-2 - 2.4	10.7	4.29	11.5	
21	Zn64p[Cd]	IRDF90	1.90E-2	2.4 - 7.3	4.86	3.56	6.02	0
22	Zr902[Cd]	IRDF90	4.72E-5	12.7 - 17.5	1.59	5.90	6.11	2.4
		JENDL/D-99	4.61E-5	12.7 - 17.5	0.562	5.89	5.92	
23	In115g[Cd]	---	---	---	---	---	---	0
24	In115n	IRDF90	1.02E-1	0.98 - 5.9	0	2.19	2.19	0
		JENDL/D-99	1.02E-1	0.98 - 5.9	2.17	2.19	3.08	
25	Au197g	IRDF90	0.176316	8.E-8 - 1.5	0.570	5.92	5.95	0
26	Au197g[Cd]	IRDF90	0.1499077	5.E-6 - 1.7	0.688	4.75	4.80	0
27	Np237f[Fis]	JENDL/D-99	0.8270836	0.54 - 5.3	0.262	2.84	2.85	0
28	Np237f[Cdna]	JENDL/D-99	0.8961254	0.5 - 5.2	0.259	2.82	2.83	0
29	U235f[Fis]	IRDF90	1.16	8.4E-2 - 4.1	0.310	3.19	3.21	0
30	U235f[Cdna]	IRDF90	1.263373	6.2E-2 - 4.0	0.307	3.19	3.20	0
31	U238f[Fis]	IRDF90	1.47E-1	1.4 - 6.6	0.53	2.34	2.40	1.4
		JENDL/D-99	1.49E-1	1.4 - 6.63	2.03	2.34	3.10	
32	U238f[Cdna]	IRDF90	1.62E-1	1.4 - 6.6	0.53	2.42	2.47	1.2
		JENDL/D-99	1.64E-1	1.4 - 6.6	2.07	2.4	3.18	
33	Pu239f[Fis]	IRDF90	1.5481	1.E-1 - 4.3	0.40	2.98	3.01	0.8
		JENDL/D-99	1.562	1.E-1 - 4.3	2.13	2.94	3.63	
34	Pu239f[Cdna]	IRDF90	1.680	8.7E-2 - 4.3	0.399	2.99	3.02	0.8
		JENDL/D-99	1.694	8.6E-2 - 4.2	2.15	2.95	3.65	

[1] Cross section identifier is the target isotope with a reaction symbol followed by a cover in square brackets. Reaction symbols include g=(n,γ),

p=(n,p), 2=(n,2n), a=(n,□). f=(n,f), n=(n,n'). Covers include [Cd]=Cadmium and [Fis]=10B-enriched boron ball with thick Cd, [Cdna] = Thick Cdmium.

Table 4 shows that only for the ^{27}Al(n,p), ^{48}Ti(n,p), and ^{58}Fe(n,γ) reactions does the variation between candidate cross sections exceed the uncertainty of the spectrum-averaged cross section in the reference field where both the reference field spectrum uncertainty and the dosimetry cross section uncertainty components are considered. The ^{48}Ti(n,p) reaction also appeared this way when examined in the ACRR reference field. The appearance of the ^{58}Fe(n,γ) reaction in this category is not surprising since it is well known that, even though this reaction has a good C/E in moderated neutron fields, it does have not have good C/E ratios in fast neutron fields.

IV.ii Normalized C/E Ratios in Reference Neutron Field

Since the SPR-III spectrum does not have time-of-flight spectrum measurement (as all standard benchmarks must have), an absolute calculated-to-experimental ratio (C/E) can not be formed. The SPR-III reactor exposures typically use the ^{58}Ni(n,p) reaction as an irradiation monitor, thus the spectrum adjustment was normalized to this measured activity. In the absence of a spectrum unfold for each combination of cross section candidates, we can form ratios of the individual activities to the ^{58}Ni(n,p) reference/monitor and then examine the C/E ratio of this metric. Table 5 shows the results of this approach.

Table 5: Ratio of Spectrum-averaged Cross Sections to Monitor Ni58p Reaction for Candidate Evaluations in the SPR-III Central Cavity

Reaction Number	Reaction/ Cover[1]	Candidate Xsec	Measured Ratio[2] (Bq/Bq)	Calc. Ratio (mb/mb)	C/E
1	Na23g	---	0.80976 +/- 2%	---	---
2	Na23g[Cd]	---	0.81924 +/- 2%	---	---
3	Na23g[Fis]	---	0.68133 +/- 2%	---	---
4	Mg24p[Cd]	IRDF90	1.303E-2 +/- 3%	1.39E-2 +/- 4.3%	1.067 +/- 5.2%
		JENDL/D-99		1.406E-2 +/- 3.8%	1.079 +/- 4.8%
5	Al27a[Cd]	IRDF90	6.437E-3 +/- 2%	6.226E-3 +/- 3.9%	0.967 +/- 4.4%
		JENDL/D-99		6.226E-3 +/- 3.9%	0.967 +/- 4.4%
6	Al27p[Cd]	IRDF90	3.278 +/- 3%	3.755E-2 +/- 5.8%	1.146 +/- 6.5%
		JENDL-D99		4.170E-2 +/- 5.0%	1.272 +/- 5.8%
7	S32p	IRDF90	6.018E-1 +/- 3%	6.132E-1 +/- err%	1.019 +/- err%
8	Sc45g[Cd]	IRDF90	1.939E-1 +/- 3.3%	1.981E-1 +/- 8.8%	1.022 +/- 9.4%
9	Sc45g[Fis]	IRDF90	1.721E-1 +/- 3.5%	1.774E-1 +/- 9.5%	1.031 +/- 10.1%
10	Ti46p[Cd]	JENDL/D-99	1.033E-1 +/- 3.4%	1.089E-1 +/- 5.4%	1.054 +/- 6.4%
		RRDF-98		1.095E-1 +/- 5.7%	1.060 +/- 6.6%
11	Ti48p[Cd]	IRDF90	2.6588E-3 +/- 1.2%	2.51E-3 +/- 4.5%	0.944 +/- 4.7%
		JENDL/D-99		2.585E-3 +/- 4.3%	0.972 +/- 4.5%
		RRDF-98		2.774E-3 +/- 6.5%	1.043 +/- 6.6%
12	Ti47p[Cd]	IRDF90	1.622E-1 +/- 2.7%	1.719E-1 +/- 4.6%	1.060 +/- 5.3%
13	Mn55g[Cd]	IRDF90	1.2253 +/- 2.6%	1.038E-1 +/- 11.8%	0.847 +/- 12.1%
		JENDL/D-99		1.038E-1 +/- 11.8%	0.847 +/- 12.1%
14	Mn55g[Fis]	IRDF90	8.437E-2 +/- 2.7%	8.843E-2 +/- 11%	1.05 +/- 11%

		JENDL/D-99		8.843E-1 +/- 11%	1.05 +/- 11%
15	Fe54p[Cd]	IRDF90	0.7354 +/- 3.2%	0.7528 +/- 4.1%	1.024 +/- 5.2%
		JENDL/D-99		0.7604 +/- 3.7%	1.034 +/- 4.9%
16	Fe56p[Cd]	RRDF-98	9.776E-3 +/- 2.4%	9.566E-3 +/- 5.2%	0.9785 +/- 3.3%
		JEFF-3.0		9.302E-3 +/- 4.5%	0.9515 +/- 4.5%
17	Fe58g[Cd]	IRDF90	6.163E-2 +/- 2.8%	7.491E-2 +/- 20.%	1.215 +/- 20%
		JENDL/D-99		5.81E-2 +/- 5.5%	0.943 +/- 6.2%
18	Co59g[Cd]	IRDF90	2.243E-1E-3 +/- 3%	2.151E-1 +/- 5.4%	0.959 +/-6.2 %
19	Ni58p[Cd]	IRDF90	1.0 +/- 3.1%	1.0 +/- 3.9%	1.0 +/- 5.0%
		JENDL/D-99		0.9906 +/- 3.3%	0.9906 +/- 4.5%
		ENDF/B-VI		1.0 +/- 4.0%	1.0 +/- 5.1%
		JEFF-3.0		1.017 +/- 4.7%	1.017 +/- 5.6%
20	Cu63g[Cd]	IRDF90	2.940E-1 +/- 2.2%	3.057E-1 +/- 11.5%	1.040 +/- 11.7%
		ENDF/B-VI		3.057E-1 +/- 11.5%	1.040 +/- 11.7%
21	Zn64p[Cd]	IRDF90	3.457E-1 +/- 2.2%	3.585E-1 +/-6.0 %	1.037 +/- 6.4%
22	Zr902[Cd]	IRDF90	8.884E-4E-2 +/- 3%	8.906E-4 +/- 6.1%	1.002 +/- 6.8%
		JENDL/D-99		8.698E-4 +/- 5.9%	0.9791 +/- 3.9%
23	In115g[Cd]	---	3.226 +/- 1.3%	---	---
24	In115n	IRDF90	1.9512 +/- 4.7%	1.9245 +/- 2.2%	0.986 +/- 5.2%
		JENDL/D-99		1.9245 +/- 3.1%	0.986 +/- 5.6%
25	Au197g	IRDF90	3.361 +/- 4.5%	3.3267 +/- 6.0%	0.990 +/- 7.5%
26	Au197g[Cd]	IRDF90	2.980 +/- 4.5%	2.8284 +/-4.8%	0.949 +/- 4.6%
27	Np237f[Fis]	JENDL/D-99	15.956 +/- 2.8%	15.605 +/- 2.9%	0.978 +/- 4.0%
28	Np237f[Cdna]	JENDL/D-99	16.658 +/- 2.8%	16.908 +/- 2.8%	1.015 +/- 4.0%
29	U235f[Fis]	IRDF90	20.20525 +/- 3.0%	21.887 +/- 3.2%	1.083 +/- 4.4%
30	U235f[Cdna]	IRDF90	23.691 +/- 3.0%	23.837 +/- 3.2%	1.006 +/- 4.4%
31	U238f[Fis]	IRDF90	3.000 +/- 3.2%	2.7736 +/- 2.4%	0.925 +/- 4.0%
		JENDL/D-99		2.811 +/- 3.1%	0.937 +/- 4.5%
32	U238f[Cdna]	IRDF90	3.1277 +/- 3.2%	3.057 +/- 2.5%	0.977 +/- 4.1%
		JENDL/D-99		3.094 +/- 3.2%	0.989 +/- 4.5%
33	Pu239f[Fis]	IRDF90	25.804 +/- 2.7%	29.209 +/- 3.0%	1.13 +/- 4.0%
		JENDL/D-99		29.472 +/- 3.6%	1.14 +/- 2.8%
34	Pu239f[Cdna]	IRDF90	30.143 +/- 2.7%	31.70 +/- 3.0%	1.05 +/- 4.0%
		JENDL/D-99		31.96 +/- 3.7%	1.06 +/- 4.6%

1 Cover nomenclature is identical to that used in Table 1

2 Uncertainty only includes that of main reaction cross section. The Ni58p/Cd /ENDF/B-VI cross section is treated as a reference with zero uncertainty

An examination of these C/E ratios in Table 3 clearly indicate that there are serious problems for the ^{239}Pu(n,f) reaction with the B_4C cover where the C/E deviation from unity by about three-sigma variation. This reaction is shaded. The problem may reside in 1) the spectrum characterization, 2) the candidate

dosimetry cross sections, or 3) the cross section processing used in this analysis - but this analysis clearly fails to validate the following reactions:

- ^{239}Pu(n,f) [Fis] - the consistency between the measurement and calculation was good (~5%) in the original spectrum unfold. The problem here is not clear. The difference is particularly troubling since the [Cd] covered ^{239}Pu(n,f) has good agreement. This will be referred to the facility for more investigation.

There were several reactions, ^{238}U(n,f)[Cd], ^{55}Mn(n,γ)[Cd], ^{58}Fe(n,γ)[Cd], and ^{27}Al(n,p)[Cd], that raised a caution. These reactions are also shaded.

- ^{27}Al(n,p)[Cd] - This reaction shows considerable spread in the evaluations and poor C/E with measurement. It is tempting to say that the IRDF90 evaluation should be used rather than the JENDL/D-99 based on these results - but one must remember that this is a validation step only and should not be used to change any selection - only to increase the inspection of the selection process.

- In any event, it appears that both evaluations underestimate the uncertainty in this cross section. That is enough to raise concern for users of this reaction.

  Note that the short half-life for this reaction makes it difficult to use. The SPR-III facility is being queried to ensure that they properly took into account the decay during irradiation for this measurement.

  ^{58}Fe(n,γ)[Cd] - This has always been a troubling reaction for fast burst reactors. It appears that the high energy portion of the cross section is routinely found to be inconsistent with measurements. In this case the IRDF90 agreement is acceptable only because the stated uncertainty is so large, ~20%. The JENDL/D-99 is in much better agreement but it has a much lower uncertainty and is also at the limits of its expected C/E uncertainty. The suggestion here is that the JENDL/D-99 evaluation is preferred based on the ACRR C/E metrics.

- ^{55}Mn(n,γ)[Cd] - This disagreement may be due to issues related to the Cd absorption cross section just above the cut-off energy. The SPR-III facility is being requested to gather data for a bare sensors to confirm this interpretation.

- ^{238}U(n,f)[Cd] - The source of this disagreement is unknown. Both the evaluation community and the facility should look into this.

IV.iii Spectrum Adjustment Using the Final IRDF-2002 Library

There are too many combinations of reactions to attempt to "validate" the cross sections through a spectrum adjustment for the SPR-III using all of the variations of candidate cross sections. However, once a final IRDF-2002 selection is made, we can perform a new spectrum adjustment using the selected dosimetry cross sections. We note again that the C/E values for this new adjustment should ONLY be used for the purpose of validating the final selection and not to alter the selection.

V. Conclusion

The well characterized and documented SPR-III Central Cavity neutron reference field with measured activity data for 34 dosimetry-quality reactions has been used to "validate" the final selection of

IRDF-2002 cross sections. In support of this "validation" uncertainty and covariance data were available for both the spectrum and the experimental activity measurements. Since the final cross section down-selection was not available at this time, the final "validation" testing was not performed.

Using this "trial" selection, the IRDF-2002 library candidates were found to be in good agreement with the previous SPR-III spectrum characterization and the measured activities except in the cases of the following reactions:

- ^{27}Al(n,p) - problem under investigation, this work suggests that the evaluated uncertainties are too small
- ^{58}Fe(n,γ) - Uncertainties in the high energy portion of this cross section persist. Both candidate cross section evaluations had C/E ratios near the 1-sigma limit. Additional work should be done on this reaction but the current IRDF-2002 candidate is acceptable at this time.

The users of the library should be warned about the potential underestimation of the uncertainties for the ^{27}Al(n,p) and ^{58}Fe(n,γ) reaction if they are using the IRDF-2002 library in fast fission fields, but all candidate IRDF-2002 reactions pass the validation check in the SPR-III fast fission neutron field.

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Damage Response Functions in the IRDF-2002 Library

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I. INTRODUCTION

It is useful to have commonly used response functions included in the IRDF-2002 library is a for that is formatted to be readily interfaced with neutron spectra unfolded using this dosimetry cross section library. To support this application, the IRDF-2002 library plans to include response functions for neutron displacement damage per atom (dpa) for iron, silicon, and gallium arsenide. The following sections detail the proposed response functions and provide attribution for the derivation of the response.

II. IRON DPA (LWR PRESSURE VESSEL DAMAGE)

ASTM standard E693, Standard Practice for Characterizing Neutron Exposures in Iron and Low Alloy Steels in Terms of Displacement Per Atom (DPA) [1] is the source the iron dpa response used in applications supporting the pressure vessel surveillance calculations performed in compliance with the U.S. Nuclear Regulatory Commission requirements. This E693 standard incorporates the ENDF/B-VI cross sections in the iron dpa exposure function and recommends the use of the NRT displacement formalism - as recommended at joint E10.02 and E10.05 meetings.

A considerable body of irradiated materials data has been reported using dpa cross sections based on the iron ENDF/B-IV [2, 3] cross section. Changes in the iron cross section [4], the recommendation to use the updated iron cross sections in radiation transport calculations of pressure vessel spectra [5], and the recent availability of ENDF/B-VI iron dpa cross section calculations [2,6,7] have resulted in the update of the recommended dpa cross section to reflect the ENDF/B-VI cross sections. Although the ENDF/B-VI based dpa cross section differs from the previously recommended ENDF/B-IV dpa cross section by about 60% in the energy region around 10 keV, by about 10% for energies between 100 keV and 2 MeV, and by a factor of 4 near 1 keV due to the opening of reaction channels in the cross section, the integral iron dpa values are much less sensitive to the change in cross sections. The update from ENDF/B-IV to ENDF/B-VI dpa rates when applied to the H. B. Robinson-2 pressurized water reactor result in "up to ~4% higher dpa rates in the region close to the pressure vessel outer surface" and in "slightly lower dpa rates ... close to the pressure vessel inner wall" [7,8]. Thus the update of the recommended dpa exposure parameter to reflect an iron cross section consistent with that used in the current radiation transport calculations is "not expected to introduce a bias in embrittlement data bases" [7] based on the change in the dpa cross section. Table 1 presents a comparison of the previous (E693-94) and the currently recommended dpa estimates for several neutron spectra.

$\sigma_d(E)$ is what is provided in the Table 1. The $\sigma_d(E>E_o)$ values were computed using an E_o of 10^{-10} eV and integrating over the given spectrum. Table 1 is included to illustrate the effect on the DPA cross sections resulting from the change from the ENDF/B-IV to ENDF/B-VI cross sections. The spectrum-average cross section values given are not recommended for other uses because of their sensitivity to the assumed spectrum representations and the lower energy integration limit.

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Table 1. Changes in Spectrum-Integrated DPA for Benchmark Neutron Spectra

Neutron Spectrum	Spectrum-averaged DPA cross section (barns) [1]		
	"Old" ENDF/B-IV-based E693 response	"Current" ENDF/B-VI-based E693 response	Difference [(Current - Old)/Old] (%)
ENDF/B-VI ^{235}U Thermal Fission	875.55	858.54	-1.9
Materials Dosimetry Reference Facility (MDRF)	345.03	343.58	-0.42
CFRMF [20, 23]	382.94	387.08	1.08
Intermediate-energy Standard Neutron Field (ISNF)	483.63	480.00	-0.75
Arkansas Nuclear ONE-1 (ANO) Cavity	134.40	139.44	3.75
ORNL Poolside Facility (PSF) T/4 position	242.14	238.33	-1.57
Oak Ridge Research Reactor (ORR)	291.68	288.86	-0.97
Yayoi	613.12	609.03	-0.67
BIGTEN	334.98	341.25	1.87
H.B. Robinson-2, in the vessel wall, close to the inner surface	219.43	218.81	-0.28
H.B. Robinson-2, ~1/4 T vessel wall	245.17	249.24	1.66
H.B. Robinson-2, ~3/4 T vessel wall	203.68	211.23	3.71

[1] The spectrum-average DPA values in this table were computed in a 640 SAND-II energy group representation and a lower integration bound of $E_o=10^{-10}$ eV.

The Table 1 entries and this discussion do not address the adequacy of the neutron spectrum representation. The adequacy of the neutron group structure used for the representation and calculation of the energy dependent variations in the neutron spectrum is an issue that must be addressed in conjunction with a re-evaluation of the spectra for these benchmark spectra.

This damage energy to displacement conversion procedure is consistent with ASTM Practice E521 and E821 recommendations on the treatment of radiation damage by charged particles. The values of the displacement cross section are based on ENDF/B-VI (revision 5) cross sections [4] as processed into dpa cross sections with the NJOY-97 code [9] using the Robinson analytic representation [10] of the Lindhard model of energy partition between atoms and electrons [11] and the Norgett-Robinson-Torrens (NRT) recommended conversion of damage energy to displacements [12] with an effective displacement threshold energy of E_d=40 eV and an atomic scattering correction factor of β=0.8. The NRT displacement equation defines the number of displacements, N_d, corresponding to a given damage energy, T_d, through the equation

$$N_d(T_d) = \begin{bmatrix} 0 & T_d < E_d \\ 1 & E_d \le T_d < 2E_d / \beta \\ \dfrac{\beta T_d}{2E_d} & 2E_d / \beta \le T_d < \infty \end{bmatrix} \qquad \textbf{(EQ 1)}$$

Note 1: The iron dpa cross section combines dpa from the individual ENDF/B-VI iron isotopic evaluations using the natural iron isotopic abundance values from Ref. [13]. The isotopic cross sections and relative abundance used were:

26-Fe-54, Mat=2625, Rev. 5, tape 140; rel. abundance = 5.9%
26-Fe-56, Mat=2631, Rev. 1, tape 123; rel. abundance = 91.72%
26-Fe-57, Mat=2634, Rev. 1, tape 123; rel. abundance = 2.1%
26-Fe-58, Mat=2637, Rev. 5, tape 140; rel. abundance = 0.28%

Version 97.45 of the NJOY97 code used in this analysis was modified to implement the NRT displacement threshold model.

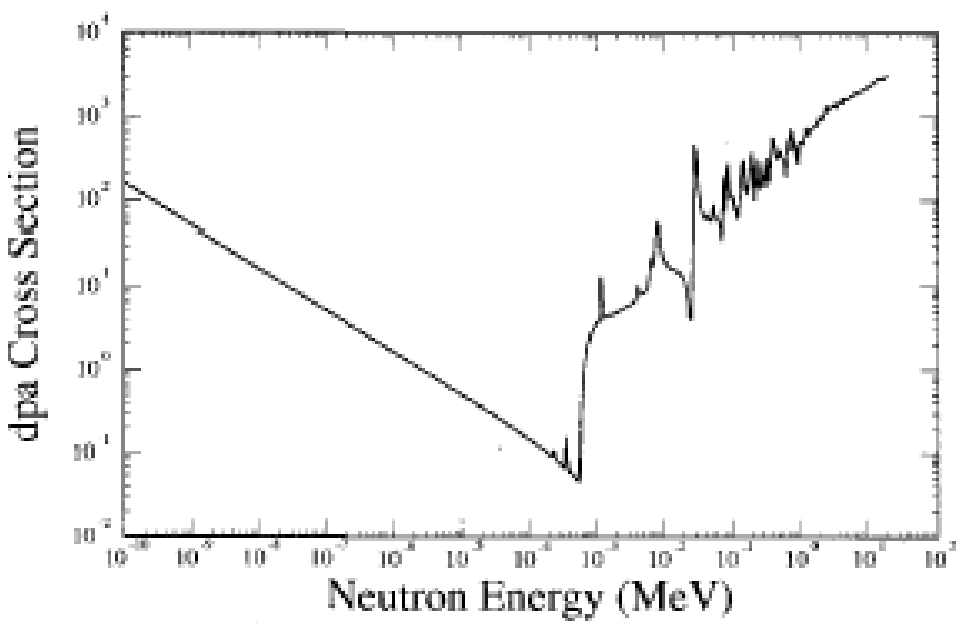

Figure 1: ENDF/B-VI-based Iron Displacement Cross Section

Table 2. END/B-VI-based Iron Displacement Cross Section

Bin	Eng* (MeV)	σ_d(barns)	Bin	Eng* (MeV)	σ_d(barns)	Bin	Eng* (MeV)	σ_d(barns)
1	0.100E-09	158.3543	2	0.1050E-09	154.6209	3	0.110E-09	151.1395
4	0.1150E-09	147.8895	5	0.120E-09	144.1054	6	0.1275E-09	139.9202
7	0.1350E-09	136.0860	8	0.1425E-09	132.5445	9	0.150E-09	128.7502
10	0.160E-09	124.7860	11	0.170E-09	121.1728	12	0.180E-09	117.8527
13	0.190E-09	114.8137	14	0.200E-09	111.9561	15	0.210E-09	109.3199
16	0.220E-09	106.8646	17	0.230E-09	104.5694	18	0.240E-09	101.8930
19	0.2550E-09	98.93331	20	0.270E-09	96.65981	21	0.280E-09	94.12717
22	0.300E-09	91.05218	23	0.320E-09	88.24872	24	0.340E-09	85.68787
25	0.360E-09	83.33912	26	0.380E-09	81.17265	27	0.400E-09	78.92472
28	0.4250E-09	76.63646	29	0.450E-09	74.53734	30	0.4750E-09	72.59930
31	0.500E-09	70.81827	32	0.5250E-09	69.14790	33	0.550E-09	67.59222
34	0.5750E-09	66.13822	35	0.600E-09	64.64189	36	0.630E-09	63.12039
37	0.660E-09	61.70157	38	0.690E-09	60.37332	39	0.720E-09	58.92732
40	0.760E-09	57.39681	41	0.800E-09	55.97892	42	0.840E-09	54.65984
43	0.880E-09	53.43220	44	0.920E-09	52.28703	45	0.960E-09	51.21545
46	0.100E-08	50.07727	47	0.1050E-08	48.89598	48	0.110E-08	47.79609
49	0.1150E-08	46.76870	50	0.120E-08	45.57125	51	0.1275E-08	44.25006
52	0.1350E-08	43.03653	53	0.1425E-08	41.91761	54	0.150E-08	40.71708
55	0.160E-08	39.46333	56	0.170E-08	38.32018	57	0.180E-08	37.26968
58	0.190E-08	36.30967	59	0.200E-08	35.40710	60	0.210E-08	34.57391
61	0.220E-08	33.79705	62	0.230E-08	33.06956	63	0.240E-08	32.22424
64	0.2550E-08	31.28942	65	0.270E-08	30.57002	66	0.280E-08	29.76999
67	0.300E-08	28.79791	68	0.320E-08	27.91048	69	0.340E-08	27.10139
70	0.360E-08	26.35879	71	0.380E-08	25.67357	72	0.400E-08	24.96309
73	0.4250E-08	24.23960	74	0.450E-08	23.57548	75	0.4750E-08	22.96268
76	0.500E-08	22.39920	77	0.5250E-08	21.87094	78	0.550E-08	21.37982
79	0.5750E-08	20.91994	80	0.600E-08	20.44705	81	0.630E-08	19.96509
82	0.660E-08	19.51724	83	0.690E-08	19.09670	84	0.720E-08	18.63984
85	0.760E-08	18.15581	86	0.800E-08	17.70708	87	0.840E-08	17.29049
88	0.880E-08	16.90205	89	0.920E-08	16.54074	90	0.960E-08	16.20166
91	0.100E-07	15.84242	92	0.1050E-07	15.46908	93	0.110E-07	15.12094
94	0.1150E-07	14.79594	95	0.120E-07	14.41855	96	0.1275E-07	14.00095
97	0.1350E-07	13.61661	98	0.1425E-07	13.26338	99	0.150E-07	12.88403
100	0.160E-07	12.48759	101	0.170E-07	12.12633	102	0.180E-07	11.79428
103	0.190E-07	11.49039	104	0.200E-07	11.20760	105	0.210E-07	10.94298
106	0.220E-07	10.69745	107	0.230E-07	10.46804	108	0.240E-07	10.20132
109	0.2550E-07	9.906717	110	0.270E-07	9.679449	111	0.280E-07	9.427035
112	0.300E-07	9.118745	113	0.320E-07	8.838819	114	0.340E-07	8.582926
115	0.360E-07	8.347962	116	0.380E-07	8.131618	117	0.400E-07	7.907534
118	0.4250E-07	7.678809	119	0.450E-07	7.468805	120	0.4750E-07	7.276812
121	0.500E-07	7.097598	122	0.5250E-07	6.930767	123	0.550E-07	6.775607
124	0.5750E-07	6.630701	125	0.600E-07	6.482083	126	0.630E-07	6.330435
127	0.660E-07	6.188963	128	0.690E-07	6.056631	129	0.720E-07	5.913148
130	0.760E-07	5.760298	131	0.800E-07	5.618617	132	0.840E-07	5.486900
133	0.880E-07	5.364337	134	0.920E-07	5.250029	135	0.960E-07	5.143171
136	0.100E-06	5.029254	137	0.1050E-06	4.911427	138	0.110E-06	4.801945
139	0.1150E-06	4.699714	140	0.120E-06	4.581585	141	0.1275E-06	4.450361
142	0.1350E-06	4.329614	143	0.1425E-06	4.218222	144	0.150E-06	4.099185
145	0.160E-06	3.974304	146	0.170E-06	3.860402	147	0.180E-06	3.756055
148	0.190E-06	3.660761	149	0.200E-06	3.571313	150	0.210E-06	3.488699
151	0.220E-06	3.411616	152	0.230E-06	3.339466	153	0.240E-06	3.256051
154	0.2550E-06	3.163276	155	0.270E-06	3.091628	156	0.280E-06	3.013147
157	0.300E-06	2.916945	158	0.320E-06	2.829403	159	0.340E-06	2.749808

160	0.360E-06	2.676750	161	0.380E-06	2.608835	162	0.400E-06	2.527930
163	0.4250E-06	2.440897	164	0.450E-06	2.374384	165	0.4750E-06	2.313606
166	0.500E-06	2.257462	167	0.5250E-06	2.205331	168	0.550E-06	2.154114
169	0.5750E-06	2.108733	170	0.600E-06	2.060134	171	0.630E-06	2.011735
172	0.660E-06	1.966045	173	0.690E-06	1.923586	174	0.720E-06	1.877395
175	0.760E-06	1.828596	176	0.800E-06	1.783008	177	0.840E-06	1.741254
178	0.880E-06	1.703417	179	0.920E-06	1.667792	180	0.960E-06	1.632082
181	0.100E-05	1.595754	182	0.1050E-05	1.558720	183	0.110E-05	1.522295
184	0.1150E-05	1.490200	185	0.120E-05	1.451554	186	0.1275E-05	1.409592
187	0.1350E-05	1.370247	188	0.1425E-05	1.334635	189	0.150E-05	1.296298
190	0.160E-05	1.256143	191	0.170E-05	1.219824	192	0.180E-05	1.187621
193	0.190E-05	1.156028	194	0.200E-05	1.126746	195	0.210E-05	1.099981
196	0.220E-05	1.075226	197	0.230E-05	1.051885	198	0.240E-05	1.026221
199	0.2550E-05	0.9965719	200	0.270E-05	0.9722222	201	0.280E-05	0.9476671
202	0.300E-05	0.9157118	203	0.320E-05	0.8876799	204	0.340E-05	0.8611951
205	0.360E-05	0.8377314	206	0.380E-05	0.8153836	207	0.400E-05	0.7928756
208	0.4250E-05	0.7695923	209	0.450E-05	0.7481711	210	0.4750E-05	0.7286609
211	0.500E-05	0.7107515	212	0.5250E-05	0.6941600	213	0.550E-05	0.6778895
214	0.5750E-05	0.6635014	215	0.600E-05	0.6480775	216	0.630E-05	0.6327240
217	0.660E-05	0.6182052	218	0.690E-05	0.6047435	219	0.720E-05	0.5900444
220	0.760E-05	0.5745604	221	0.800E-05	0.5600318	222	0.840E-05	0.5467011
223	0.880E-05	0.5345984	224	0.920E-05	0.5232391	225	0.960E-05	0.5118709
226	0.100E-04	0.5002917	227	0.1050E-04	0.4884902	228	0.110E-04	0.4769112
229	0.1150E-04	0.4666898	230	0.120E-04	0.4543560	231	0.1275E-04	0.4409762
232	0.1350E-04	0.4284223	233	0.1425E-04	0.4170648	234	0.150E-04	0.4048129
235	0.160E-04	0.3919477	236	0.170E-04	0.3802993	237	0.180E-04	0.3699498
238	0.190E-04	0.3598211	239	0.200E-04	0.3504475	240	0.210E-04	0.3418580
241	0.220E-04	0.3339029	242	0.230E-04	0.3263810	243	0.240E-04	0.3181047
244	0.2550E-04	0.3085413	245	0.270E-04	0.3006880	246	0.280E-04	0.2927655
247	0.300E-04	0.2824593	248	0.320E-04	0.2733916	249	0.340E-04	0.2648267
250	0.360E-04	0.2572175	251	0.380E-04	0.2499703	252	0.400E-04	0.2426542
253	0.4250E-04	0.2350771	254	0.450E-04	0.2280936	255	0.4750E-04	0.2217044
256	0.500E-04	0.2158471	257	0.5250E-04	0.2104036	258	0.550E-04	0.2050605
259	0.5750E-04	0.2003403	260	0.600E-04	0.1952700	261	0.630E-04	0.1902102
262	0.660E-04	0.1854221	263	0.690E-04	0.1809756	264	0.720E-04	0.1761094
265	0.760E-04	0.1709726	266	0.800E-04	0.1661474	267	0.840E-04	0.1616956
268	0.880E-04	0.1576549	269	0.920E-04	0.1538460	270	0.960E-04	0.1500470
271	0.100E-03	0.1461601	272	0.1050E-03	0.1421933	273	0.110E-03	0.1382971
274	0.1150E-03	0.1348528	275	0.120E-03	0.1306799	276	0.1275E-03	0.1261477
277	0.1350E-03	0.1218873	278	0.1425E-03	0.1180186	279	0.150E-03	0.1138330
280	0.160E-03	0.1094287	281	0.170E-03	0.1054279	282	0.180E-03	0.1018602
283	0.190E-03	0.9837523E-01	284	0.200E-03	0.9515446E-01	285	0.210E-03	0.9224781E-01
286	0.220E-03	0.1048459	287	0.230E-03	0.1028288	288	0.240E-03	0.8397242E-01
289	0.2550E-03	0.8057346E-01	290	0.270E-03	0.7782596E-01	291	0.280E-03	0.7506079E-01
292	0.300E-03	0.7147984E-01	293	0.320E-03	0.6842791E-01	294	0.340E-03	0.1581741
295	0.360E-03	0.7452445E-01	296	0.380E-03	0.6020231E-01	297	0.400E-03	0.5752645E-01
298	0.4250E-03	0.5484677E-01	299	0.450E-03	0.5239946E-01	300	0.4750E-03	0.5017326E-01
301	0.500E-03	0.4855164E-01	302	0.5250E-03	0.4750492E-01	303	0.550E-03	0.8634498E-01
304	0.5750E-03	0.3684567	305	0.600E-03	0.7230662	306	0.630E-03	1.085166
307	0.660E-03	1.433057	308	0.690E-03	1.753764	309	0.720E-03	2.100540
310	0.760E-03	2.428895	311	0.800E-03	2.722788	312	0.840E-03	2.998771
313	0.880E-03	3.258461	314	0.920E-03	3.465372	315	0.960E-03	3.618973
316	0.100E-02	3.787081	317	0.1050E-02	3.964271	318	0.110E-02	13.13789
319	0.1150E-02	10.69382	320	0.120E-02	4.262693	321	0.1275E-02	4.338387
322	0.1350E-02	4.435367	323	0.1425E-02	4.534225	324	0.150E-02	4.592575
325	0.160E-02	4.783180	326	0.170E-02	4.701623	327	0.180E-02	4.841220

328	0.190E-02	4.999009	329	0.200E-02	5.111335	330	0.210E-02	5.183993
331	0.220E-02	5.279069	332	0.230E-02	5.408910	333	0.240E-02	5.523736
334	0.2550E-02	5.695442	335	0.270E-02	5.826853	336	0.280E-02	5.975410
337	0.300E-02	6.173523	338	0.320E-02	6.395415	339	0.340E-02	6.658582
340	0.360E-02	7.183939	341	0.380E-02	10.01168	342	0.400E-02	9.194485
343	0.4250E-02	8.178266	344	0.450E-02	8.229657	345	0.4750E-02	8.508525
346	0.500E-02	8.910876	347	0.5250E-02	9.525395	348	0.550E-02	10.59739
349	0.5750E-02	13.20411	350	0.600E-02	21.17466	351	0.630E-02	19.67340
352	0.660E-02	18.59203	353	0.690E-02	24.36627	354	0.720E-02	39.42936
355	0.760E-02	56.38621	356	0.800E-02	48.66102	357	0.840E-02	35.11790
358	0.880E-02	27.24892	359	0.920E-02	23.52159	360	0.960E-02	20.66808
361	0.100E-01	19.98392	362	0.1050E-01	18.41869	363	0.110E-01	18.51923
364	0.1150E-01	16.80365	365	0.120E-01	16.40945	366	0.1275E-01	16.00651
367	0.1350E-01	16.06792	368	0.1425E-01	15.55876	369	0.150E-01	15.08970
370	0.160E-01	14.61809	371	0.170E-01	13.99511	372	0.180E-01	13.21823
373	0.190E-01	12.01959	374	0.200E-01	10.61530	375	0.210E-01	8.888706
376	0.220E-01	6.857686	377	0.230E-01	4.435299	378	0.240E-01	4.034283
379	0.2550E-01	51.58856	380	0.270E-01	462.4204	381	0.280E-01	374.2036
382	0.300E-01	138.8582	383	0.320E-01	91.99242	384	0.340E-01	75.21491
385	0.360E-01	66.92896	386	0.380E-01	62.40611	387	0.400E-01	65.56062
388	0.4250E-01	59.51893	389	0.450E-01	61.42897	390	0.4750E-01	55.61853
391	0.500E-01	80.54994	392	0.5250E-01	84.93746	393	0.550E-01	61.63473
394	0.5750E-01	61.83547	395	0.600E-01	61.39486	396	0.630E-01	51.44463
397	0.660E-01	41.05520	398	0.690E-01	35.18787	399	0.720E-01	195.2115
400	0.760E-01	72.97270	401	0.800E-01	139.3801	402	0.840E-01	276.0543
403	0.880E-01	138.4568	404	0.920E-01	104.0743	405	0.960E-01	113.7160
406	0.100	105.5489	407	0.1050	78.43095	408	0.110	78.06753
409	0.1150	60.44543	410	0.120	70.47430	411	0.1275	134.1587
412	0.1350	255.3630	413	0.1425	293.0161	414	0.150	133.0900
415	0.160	122.7596	416	0.170	156.0143	417	0.180	244.4100
418	0.190	370.7627	419	0.200	205.3339	420	0.210	101.0349
421	0.220	307.5111	422	0.230	146.9613	423	0.240	171.8077
424	0.2550	135.3387	425	0.270	282.5501	426	0.280	187.6945
427	0.300	142.0391	428	0.320	315.5791	429	0.340	166.5530
430	0.360	302.1184	431	0.380	545.7117	432	0.400	528.4010
433	0.4250	400.4412	434	0.450	309.7825	435	0.4750	351.4453
436	0.500	381.0001	437	0.5250	327.4008	438	0.550	316.9469
439	0.5750	264.2091	440	0.600	190.8042	441	0.630	255.5292
442	0.660	529.8745	443	0.690	389.0167	444	0.720	680.4310
445	0.760	627.3019	446	0.800	427.2328	447	0.840	461.2467
448	0.880	327.1398	449	0.920	278.1719	450	0.960	479.7678
451	1.00	498.4922	452	1.10	477.6893	453	1.20	707.1735
454	1.30	616.4640	455	1.40	716.4101	456	1.50	734.1659
457	1.60	758.8353	458	1.70	784.5580	459	1.80	819.3325
460	1.90	989.3265	461	2.00	928.7681	462	2.10	923.3754
463	2.20	969.4800	464	2.30	1079.716	465	2.40	1143.224
466	2.50	1348.954	467	2.60	1192.105	468	2.70	1263.538
469	2.80	1277.801	470	2.90	1271.578	471	3.00	1364.006
472	3.10	1387.724	473	3.20	1358.466	474	3.30	1346.851
475	3.40	1398.034	476	3.50	1334.743	477	3.60	1424.447
478	3.70	1440.708	479	3.80	1483.880	480	3.90	1464.868
481	4.00	1526.371	482	4.10	1560.443	483	4.20	1530.309
484	4.30	1582.916	485	4.40	1523.340	486	4.50	1626.724
487	4.60	1600.863	488	4.70	1617.455	489	4.80	1653.609
490	4.90	1642.833	491	5.00	1660.075	492	5.10	1682.554
493	5.20	1687.286	494	5.30	1716.469	495	5.40	1731.089

496	5.50	1729.785	497	5.60	1760.852	498	5.70	1760.538
499	5.80	1768.656	500	5.90	1792.105	501	6.00	1811.511
502	6.10	1800.940	503	6.20	1811.547	504	6.30	1859.081
505	6.40	1859.125	506	6.50	1879.237	507	6.60	1890.037
508	6.70	1891.762	509	6.80	1909.485	510	6.90	1909.076
511	7.00	1914.442	512	7.10	1936.414	513	7.20	1941.750
514	7.30	1966.663	515	7.40	1963.306	516	7.50	1986.044
517	7.60	1976.213	518	7.70	1989.243	519	7.80	2003.646
520	7.90	2006.771	521	8.00	2009.093	522	8.10	2013.259
523	8.20	2032.588	524	8.30	2064.755	525	8.40	2063.837
526	8.50	2061.365	527	8.60	2059.507	528	8.70	2072.344
529	8.80	2089.976	530	8.90	2107.525	531	9.00	2122.580
532	9.10	2135.077	533	9.20	2147.577	534	9.30	2160.074
535	9.40	2172.482	536	9.50	2185.892	537	9.60	2199.950
538	9.70	2213.918	539	9.80	2226.698	540	9.90	2238.281
541	10.0	2250.482	542	10.10	2263.392	543	10.20	2276.220
544	10.30	2288.971	545	10.40	2301.725	546	10.50	2313.910
547	10.60	2325.628	548	10.70	2337.342	549	10.80	2348.869
550	10.90	2360.301	551	11.0	2371.744	552	11.10	2383.112
553	11.20	2395.212	554	11.30	2407.952	555	11.40	2421.344
556	11.50	2434.325	557	11.60	2446.243	558	11.70	2458.956
559	11.80	2471.205	560	11.90	2482.247	561	12.0	2493.659
562	12.10	2506.016	563	12.20	2519.598	564	12.30	2534.971
565	12.40	2549.086	566	12.50	2562.977	567	12.60	2576.115
568	12.70	2586.936	569	12.80	2600.011	570	12.90	2615.468
571	13.0	2630.343	572	13.10	2644.455	573	13.20	2658.475
574	13.30	2672.218	575	13.40	2685.520	576	13.50	2698.683
577	13.60	2711.990	578	13.70	2725.313	579	13.80	2738.112
580	13.90	2750.418	581	14.0	2763.164	582	14.10	2775.980
583	14.20	2788.331	584	14.30	2800.214	585	14.40	2811.915
586	14.50	2824.208	587	14.60	2837.183	588	14.70	2849.781
589	14.80	2862.184	590	14.90	2874.421	591	15.0	2877.552
592	15.10	2871.084	593	15.20	2864.617	594	15.30	2858.147
595	15.40	2851.581	596	15.50	2844.839	597	15.60	2837.641
598	15.70	2830.538	599	15.80	2823.427	600	15.90	2816.329
601	16.0	2813.386	602	16.10	2814.819	603	16.20	2816.344
604	16.30	2817.782	605	16.40	2819.124	606	16.50	2819.920
607	16.60	2819.883	608	16.70	2819.846	609	16.80	2819.725
610	16.90	2819.631	611	17.0	2823.437	612	17.10	2831.455
613	17.20	2839.475	614	17.30	2847.220	615	17.40	2855.056
616	17.50	2862.956	617	17.60	2870.913	618	17.70	2878.960
619	17.80	2886.826	620	17.90	2894.594	621	18.0	2903.983
622	18.10	2914.913	623	18.20	2925.791	624	18.30	2936.340
625	18.40	2946.712	626	18.50	2956.448	627	18.60	2965.448
628	18.70	2974.450	629	18.80	2983.453	630	18.90	2992.455
631	19.0	2999.561	632	19.10	3004.583	633	19.20	3009.698
634	19.30	3014.721	635	19.40	3019.741	636	19.50	3025.406
637	19.60	3031.526	638	19.70	3037.737	639	19.80	3043.950
640	19.90	3050.161	641					
* Energies represent the lower bin boundary. The upper bin limit is 20.0 MeV								

III. SILICON DPA (ELECTRONICS DAMAGE)

The basis of the currently accepted protocol for the correlation of radiation damage effects in a semiconductor device with a neutron irradiation is through the displacement kerma produced in bulk silicon. This correlation assumes that volume (versus surface) effects are the dominant radiation damage mechanism. Experimental evidence indicates that displacement kerma is a valid measure of device performance degradation (for example, reduction in current gain) in bipolar transistors whose operation basically depends on volume mechanisms [14, 15]. For device types governed by surface phenomena (such as MOSFET devices), it is clear that this correlation is not valid. Surface-effect devices are more sensitive than are volume-effect devices to ionization radiation effects produced either by a neutron field or a mixed neutron-gamma field.

The current accepted methodology is to relate the damage caused by neutron irradiation to the corresponding damage from a neutron with a reference energy. The choice of the reference neutron energy for determining an equivalent fluence has been the subject of some controversy within the electronics hardness-testing community [16]. Some workers [17] have proposed that 1-MeV be used while others [18, 19] have suggested that 14-MeV is a more appropriate reference energy. The concept of 1-MeV equivalent fluence has gained broad acceptance in practice, and procedures for applying it to silicon are described in the ASTM E722 standard practice. A 1-MeV equivalent fluence in a given material can be defined for an irradiation by neutrons of any neutron spectrum. The neutron energy fluence, $\Phi(E)$, may be that determined from a neutron transport calculation, that determined from measurements, or that given in an environment specification document. The correlation of the damage from a specific neutron with energy E to the damage caused by the reference 1-MeV neutron is the 1-MeV(Si) damage response function.

The IRDF-2002 library response functions include results of calculations of silicon displacement kerma factors (displacement kerma per unit neutron fluence) as a function of neutron energy over the range from 10^{-10} to 20 MeV. The unit of the displacement kerma factor is megaelectron volt times millibarns (MeV-mbarn). Each factor can be multiplied by 3.435×10^{-13} to convert to rad(Si)–cm^2, or by 3.435×10^{-19} to convert to J-m^2/kg or Gy(Si)–m^2.

An average value of neutron displacement kerma factor near 1-MeV is difficult to determine because of sharp neutron cross-section resonances in that energy region. To avoid these difficulties, Namenson, Wolicki, and Messenger **(14)** fitted the function $AE(1 - \exp(-B/E))$ to various tabulations of $K_D(E)$ versus energy. The values of A and B obtained by a least squares fit yielded an average value at 1-MeV of 95±4 MeV- mbarn. Accordingly, the designated value used to convert a displacement kerma to a 1-MeV(Si) equivalence fluence has been set to be 95 MeV-mb.

The values for the silicon displacement kerma are determined by calculating the total kerma and then partitioning it into ionization and displacement fractions [20]. Because of the lack of adequate theory to partition the kerma and uncertainties in cross sections, the estimated uncertainty in the displacement kerma factor is about 10 % up to 3 MeV. Correlation of displacement kerma with measured damage in many neutron fields has been confirmed with integral uncertainties no larger than 10 % [15]. Figure 1 shows the neutron energy-dependent silicon displacement kerma. A 640-group representation is found in the body of the IRDF-2002 library.

Comparisons between the calculations with the SAND II unfolding code (using activation-foil input data), neutron transport codes, and experimental spectrometry data give an estimated uncertainty in the determination of $\Phi(E)$ of about 20 % over the energy region of interest [21]. No uncertainty in the calculation of 1-MeV equivalent fluence is attributable to the consistent use of these data. Therefore only the uncertainty in the determination of neutron spectrum need be considered in assigning an uncertainty to a spectrum-averaged 1-MeV equivalent fluence. Figure 2 shows a relative covariance matrix for the 1-MeV(Si) response that was derived by considering the uncertainty in the various cross section components [22].

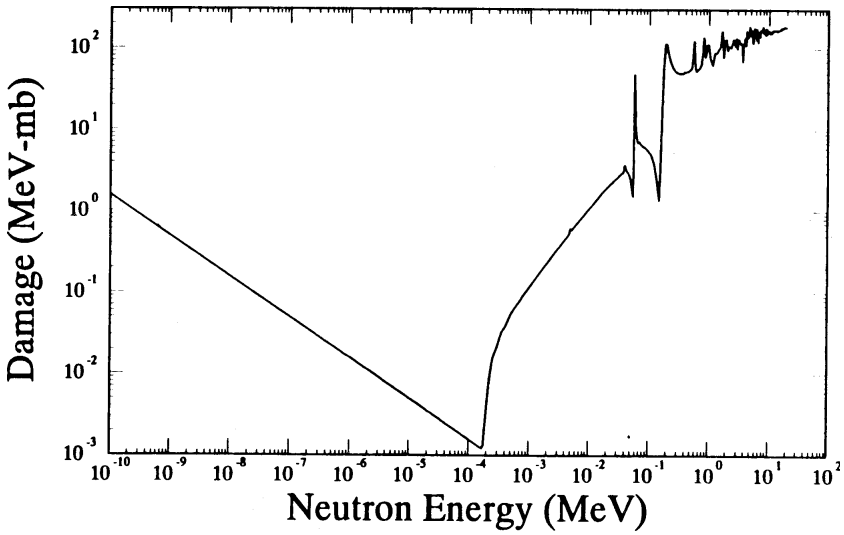

Figure 2: Energy-dependence of Silicon Displacement Damage Response Function

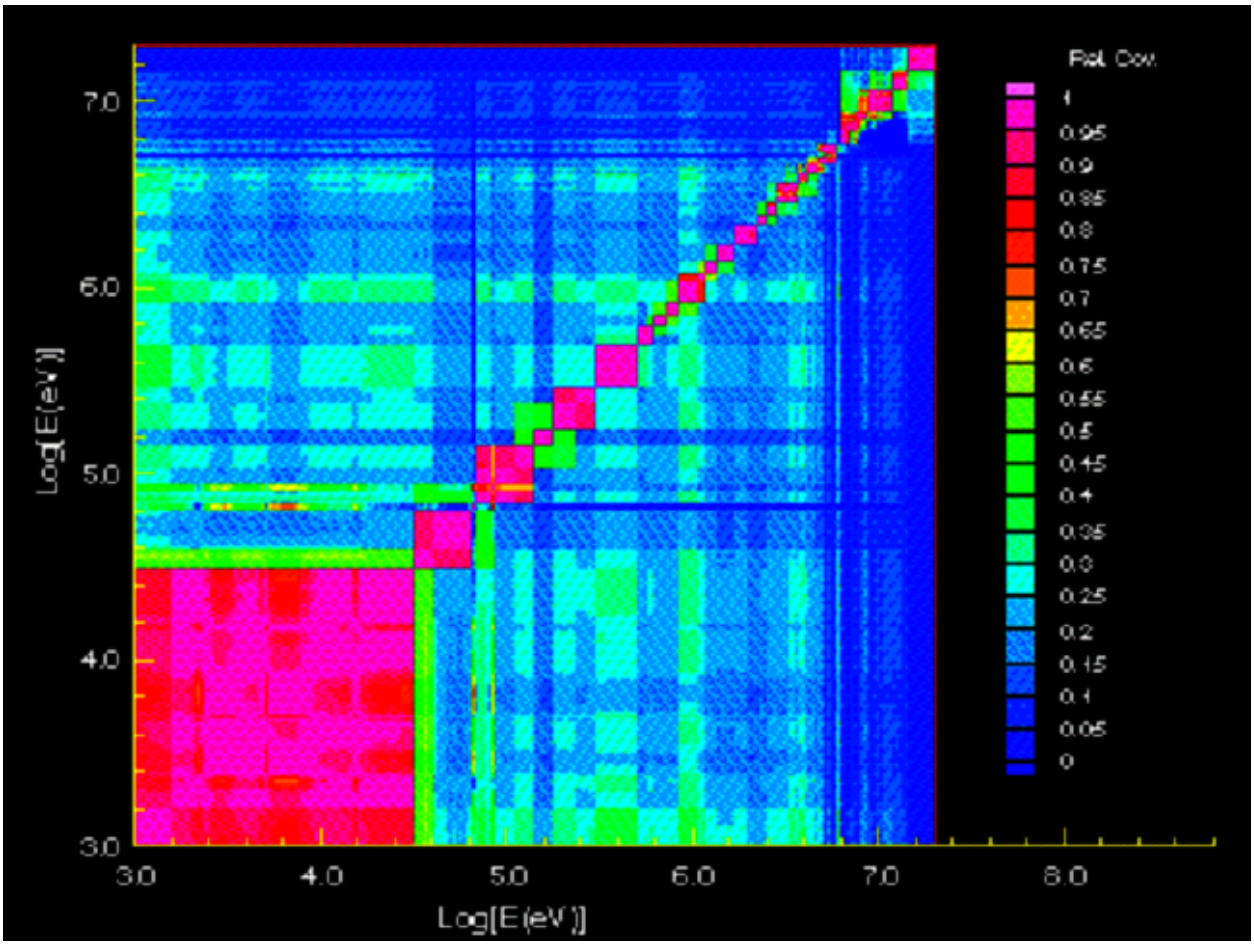

Figure 3: Relative Covariance Matrix for 1-MeV(Si) Damage Response Function

IV. GaAs DPA (ELECTRONICS DAMAGE)

The basis of the currently accepted protocol for the correlation of GaAs radiation damage effects in a semiconductor device with a neutron irradiation is through the displacement kerma produced in bulk gallium arsenide. This correlation assumes that displacement effects are the dominant radiation damage mechanism and that equal numbers of initially displaced atoms produce equal changes in device performance. Experimental evidence [23,24] indicates that displacement kerma is not a valid measure of changes in the fundamental properties (carrier concentration, mobility, and carrier lifetime) that determine device performance. The reason that displacement kerma does not correlate with property changes in gallium arsenide over the entire range of neutron energies of interest is attributed to variations in the defect production efficiency in displacement cascades of different sizes. This effect is also known to occur in other materials, including structural metals [25].

Despite these deficiencies, displacement kerma is still useful as an exposure parameter, analogous to the use of displacements per atom (dpa) for exposures of iron. When displacement kerma is used to compare property changes in gallium arsenide exposed to reactor neutrons in thermal and fast spectrum reactors, the discrepancies do not exceed ±10 % in reactors where careful comparisons have been made. When these reactor irradiations have been compared with accelerator irradiations with neutron energies of 3 and 14 MeV, however, much larger discrepancies have been observed [23,24].

Empirical efficiency factors that depend on the energies of the primary knock-on atoms (pka) have been proposed [23] in order to remove the discrepancies described above. Figures 4 and 5 show the shape of the empirical damage efficiency factor for GaAs. This damage efficiency function can be fit with an empirical function as shown in Figure 6. As in Ref [23], this PKA-energy damage efficiency factor is used in conjunction with a normalization factor of 2.2 in order to preserve the equivalence of the GaAs damage function and the displacement kerma for 1-MeV neutrons.

The choice of the specific energy for determining an equivalent fluence has been the subject of some controversy within the electronics hardness-testing community [16]. The concept of 1-MeV equivalent fluence has gained broad acceptance in practice, and procedures for applying it to gallium arsenide are adopted by the ASTM E722 standard. A 1-MeV equivalent fluence in a given material can be defined for an irradiation by neutrons of any neutron spectrum. The neutron energy fluence, $\Phi(E)$, may be that determined from a neutron transport calculation, that determined from measurements, or that given in an environment specification document.

Results of calculations of gallium arsenide displacement kerma factors (displacement kerma per unit neutron fluence) are shown in Fig. 5 as a function of neutron energy. The unit of the kerma factor is megaelectron volt times millibarns (MeV-mbarn). Each factor can be multiplied by 1.334×10^{-13} to convert to rad(GaAs)–cm^2 or by 1.334×10^{-19} to convert to J-m^2/kg or Gy(GaAs)–m^2.

An average value of neutron displacement kerma factor near 1 MeV is 70 MeV-mbarn. As is the case for silicon [16], the actual value chosen for the designated 1-MeV reference damage is arbitrary. What is important is that the whole radiation hardness community use the same value in setting hardness specification and in testing electronic parts. The damage function for gallium arsenide is normalized to the same value as the displacement kerma factor at 1 MeV. Accordingly, the designated value, adopted in ASTM E722, to be used to calculate a 1-MeV equivalent fluence in gallium arsenide is 70 MeV- mbarn.

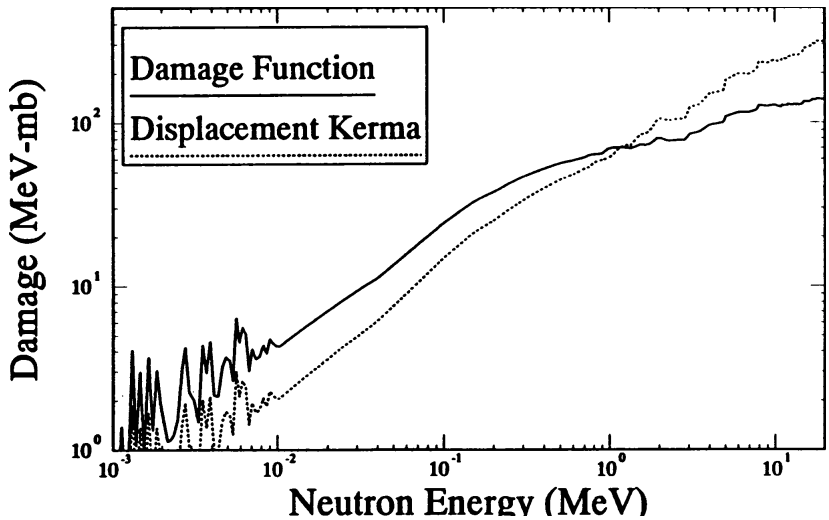

Figure 4: Energy-dependence of the GaAs Displacement and Damage Response Functions

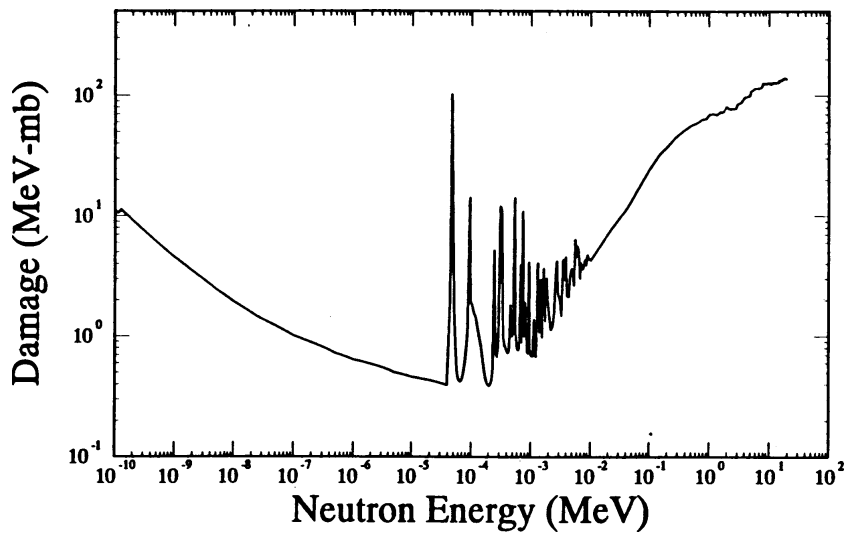

Figure 5: Energy-dependence of the GaAs Damage Function

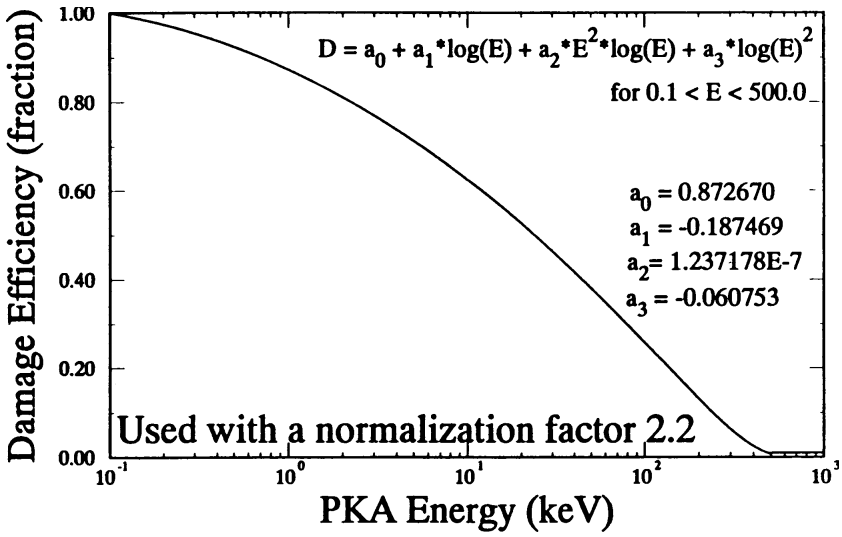

Figure 6: GaAs Damage Efficiency Curve

V. CONCLUSION

Community accepted damage functions for displacements in iron, silicon, and gallium arsenide have been described and documented. These response functions are included in the IRDF-2002 library as an aid to users doing neutron damage estimates for adjusted spectra.

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*Part of this work was supported by the United States Department of Energy under contract DE-AC04-94AL85000. Sandia is a multiprogram laboratory operated by Sandia Corporation, a Lockheed Martin Company, for the United States Department of Energy

IRDF-2002

Checking and Corrections made to the Files of Candidate Cross Sections, Graphic Representations

P. K. McLaughlin

IAEA-NDS, Vienna, Austria

Below are the details of the codes used for checking and processing the candidate files from the five libraries. The processed files were compared graphically using the latest available EXFOR data with ZVView.

Details of the codes used are given at the end of this write-up.

The files that were corrected or perhaps will need correction are also given further in this write-up.

ENDF/B-VI

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed pointwise files:

The original data file was first processed through the ENDF/B Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

RRDF-98

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the ENDF/B Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

JEFF-3.0

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed pointwise files:

The original data file was first processed through the ENDF/B Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

IRDF-90

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the Endf/b Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

If the original IRDF-90 files were from the ENDF/B-VI library then they were updated if more recent data was available.

Each of the candidate files has a "Readme.html" file which gives the source of the data and a report of any errors encountered and corrections made to the data.

JENDL/D-99

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from EXFOR database were taken from the Pointwise files:

In the case of the JENDL/D-99 files which are given as revised data above under "Current Status of Participants and their Contributions" section K. Shibata the data were taken from the given Pointwise files. For the other JENDL/D-99 candidate files recommended for IRDF-2002 the data was retrieved from the Pointwise files given with the full JENDL/D-99 Library.

CORRECTED FILES:

For IRDF-90

EMAX corrected for Ag109, Al27, Co59, Cr52, Cu63, I127, In115, La139, Mg24, Nb93, P31, S32, Sc45, Zn64, Zr90

THE MINIMUM INCIDENT ENERGY corrected for Al27(n,a), Au197(n,2n), Co59(n,2n), Cr52(n,2n), Cu63(n,2n), I127(n,2n), Mg24(n,p), Nb93(n,2n)/(n,inl), P31(n,p), Rh103(n,inl), Zn64(n,p), Zr90(n,2n)

LRP set equal to 1 for Co59,

ZAI SET TO 5.312700E+04 SEQUENCE NUMBER 2 for I127 Mf/MT= 2/151

For Ag109 ---- Original CNDC MF/MT does not match the MF/MT given for IRDF-90

For RRDF-98

EMAX corrected for Al27, As75, Co59, Nb93, Pb204, Pr141, Rh103, Ti46, Ti47, Ti48, Ti49, V51, W186, P31, S32, Sc45, Zn64, Zr90

THE MINIMUM INCIDENT ENERGY corrected for Cu63(n,2n),Nb93(n,2n)/(n,inl), P31(n,p), Rh103(n,inl), Zn64(n,p), Zr90(n,2n)

La-139 ---- Psyche failed to process anything in File 3 --- See Readme file on Web page

W-186 ---- Psyche failed to process anything in File 3 --- See Readme file on Web page

For JENDL/D-99

EMAX corrected for Tm169, Y89,

Fizcon gave many messages for File 33 for many materials in all libraries. No corrections were made to these files at this time.

IRDF-2002

Web page Description

This is the home page for McLaughlin on the Participants page:

P. McLAUGHLIN (Kevin)
IAEA-NDS, Vienna, Austria
P. McLAUGHLIN @iaea.org

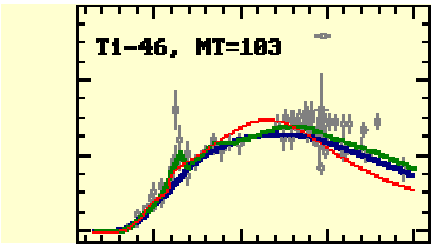
"Candidates reactions" from the different libraries have been checked (and corrected) using the codes STANEF, CHECKR, FIZCON, and PSYCHE, outputs have the extension ".che",".fiz" and ".psy" respectively for the latter 3 codes. (see: http://www-nds.iaea.org/ndspub/endf/utility/for more information on these codes).

The file "endfb.in" is the version of the file which has been processed through the codes "LINEAR", "RECENT", "SIGMA1", "FIXUP" and "GROUPIE". (see: http://www-nds.iaea.org/ndspub/endf/prepro/ for more information on these codes).

Candidates reactions, from **ENDF/B-VI, release 8** , from **JEFF-3.0**, from **IRDF-90**, from **JENDL/D-99,** and **RRDF-98**

Plots for Reactor Dosimetry Files

The evaluated curves are compared to experimental data retrieved from EXFOR (Sept. 2003).



WINENDF (Utility PC code to view and merge ENDF format files)

Pointwise Cross Sectiond for IRDF-2002 (will be placed here after final selection of the reactions)

Processed cross sections (620 groups) (will be placed here after final selection of the reactions)

This is the link "ENDF/B-VI, release 8" from the home page:

ENDF/B-VI

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the Endf/b Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

Proceed to the candidate files from ENDF/B-VI Rel.8

This is the link “JEFF-3.0” from the home page

JEFF-3.0

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the Endf/b Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

Proceed to the candidate files from JEFF-3.0

This is the link “IRDF-90” from the home page:

IRDF-90

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the Endf/b Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

If the original IRDF-90 files were from the ENDF/B-VI library then they were updated if more recent data was available.

Each of the candidate files has a "Readme.html" file which gives the source of the data and a report of any errors encountered and corrections made to the data.

Proceed to the candidate files from IRDF-90

This is the link “IRDF-90” from the home page:

JENDL/D-99

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from EXFOR database were taken from the Pointwise files:

In the case of the JENDL/D-99 files which are given as revised data above under "Current Status of Participants and their Contributions" section K. Shibata the data were taken from the given Pointwise files. For the other JENDL/D-99 candidate files recommended for IRDF-2002 the data was retrieved from the Pointwise files given with the full JENDL/D-99 Library.

Proceed to the candidate files from JENDL/D-99

This is the link “IRDF-90” from the home page:

RRDF-98

The data for the set of plots of evaluated curves for comparison with experimental data retrieved from the EXFOR database were taken from the processed Pointwise files:

The original data file was first processed through the Endf/b Utility codes STANEF, CHECKR, FIZCON and PSYCHE.

The pointwise files were created with the 2002 Pre-Processing codes LINEAR, RECENT, SIGMA1, FIXUP and GROUPIE. The processing listing is given for all the codes and the full output is given for the GROUPIE and FIXUP codes. The original input file is labelled "endfb.in".

Proceed to the candidate files from RRDF-98

As an example this is the link to the IRDF-90 candidate files from the “Proceed to the candidate files” on the IRDF-90 link.

Files

AG109/
AL27/
AU197/
B10/
CO59/
CR52/
CU63/
CU65/
FE54/
FE58/
I127/
IN115/
LI6/
MG24/
MN55/

NB93/
NI58/
P31/
PU239/
RH103/
S32/
SC45/
TH232/
TI46/
TI47/
TI48/
U235/
U238/
ZN64/
ZR90/

As an example this is the link to the Co-59 processed files “Co59/” link on the previous page.

Files
DICTIN.LST, 2 Kb, Thu Sep 25 16:13:52 2003
ENDFB.CHE, 2 Kb, Thu Sep 25 16:13:52 2003
ENDFB.FIZ, 3 Kb, Thu Sep 25 16:13:52 2003
ENDFB.IN, 416 Kb, Thu Sep 25 16:13:54 2003
ENDFB.PSY, 5 Kb, Thu Sep 25 16:13:54 2003
ENDFB_O.CHE, 2 Kb, Thu Sep 25 16:13:54 2003
ENDFB_O.FIZ, 3 Kb, Thu Sep 25 16:13:55 2003
FIXUP.LST, 9 Kb, Thu Sep 25 16:13:55 2003
FIXUP.OUT, 971 Kb, Thu Sep 25 16:13:58 2003
GROUPIE.LST, 4 Kb, Thu Sep 25 16:13:59 2003
GROUPIE.OUT, 96 Kb, Thu Sep 25 16:13:59 2003
LINEAR.LST, 4 Kb, Thu Sep 25 16:14:00 2003
README.HTML, 29 Kb, Tue Sep 30 17:28:49 2003
RECENT.LST, 20 Kb, Thu Sep 25 16:14:00 2003
SIGMA1.LST, 4 Kb, Thu Sep 25 16:14:01 2003
UNSHIELD.LST, 65 Kb, Thu Sep 25 16:14:01 2003

This is the link “Plots for Reactor Dosimetry files” from the home page.

Reactor Dosimetry Files in Pictures
Version 0.1-Development, Sept. 2003

Reactor Dosimetry Files are presented here as set of plots* of evaluated curves in comparison with experimental data retrieved from EXFOR database:
- Evaluations in comparison with experiments: **MAT-MT Matrix**

V.Zerkin, NDS, IAEA, Vienna-2003

Libraries:
A---- : **IRDF-90,** International Reactor Dosimetry File-90
-J--- : **JDOSM-99,** JENDL Dosimetry File, Japan-99
--R-- : **RRDF-98,** Russian Dosimetry File, Russia-98
---E- : **ENDF/B-VI,** ENDF/B-VI Rev.8
----F : **JEFF,** Jeff-3.3

n. Target	*MT:libs*	*MT:libs*	*MT:libs*	*MT:Libs*
1. 3-LI-6	105: A----			
2. 5-B-10	107: AJ---			
3. 9-F-19	16: -JR--			
4. 12-MG-24	103: AJ---			
5. 13-AL-27	103: AJR--	107: AJ---		
6. 15-P-31	103: AJ---			
7. 16-S-32	103: A----			
8. 21-SC-45	102: A----			
9. 22-TI-0	220: -J---	221: -J---	222: -J---	
10. 22-TI-46	16: -JR--	103: AJR--		
11. 22-TI-47	28: A-R--	103: A----		
12. 22-TI-48	28: AJR--	103: A-R--		
13. 22-TI-49	28: -JR--			
14. 23-V-51	107: --R--			
15. 24-CR-52	16: AJE-			
16. 25-MN-55	1: A----	102: AJ-		
17. 26-FE-54	16: --R--	103: AJ---	107: --R--	
18. 26-FE-56	103: --R-F			
19. 26-FE-58	1: AJ---	102: AJ---		
20. 27-CO-59	1: AJ---	16: AJ---	102: AJ---	107: AJR--
21. 28-NI-58	16: AJ--F-	103: AJREF		
22. 28-NI-60	103: ---EF			
23. 29-CU-63	1: AJ-E-	16: AJ-E-	102: AJ-E-	107: AJR--
24. 29-CU-65	16: AJ-E-			
25. 30-ZN-64	103: A----			
26. 33-AS-75	16: --R--			
27. 39-Y-89	16: -J---			

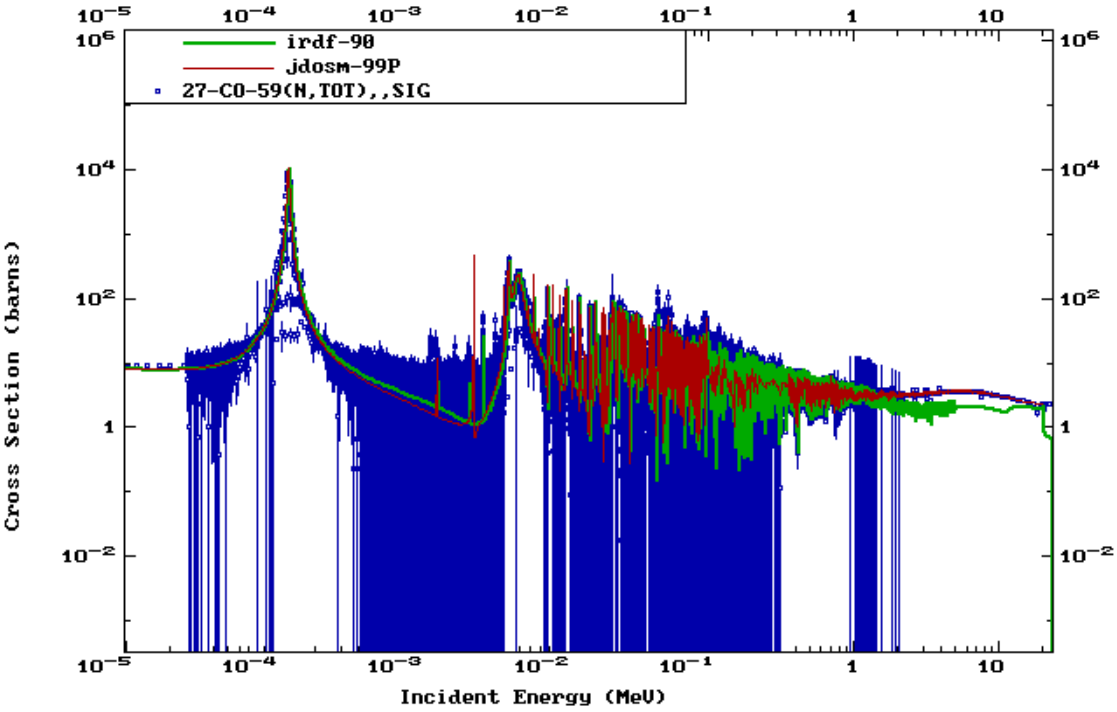
28. 40-ZR-90	16: AJ---			
29. 41-NB-93	1: A----	16: A-R--	51: A-R--	102: A----
30. 45-RH-103	51: A-R--			
31. 47-AG-109	1: A----	102: A----		
32. 49-IN-115	16: A----	51: AJR--		
33. 53-I-127	16: AJ---			
34. 57-LA-139	1: --R--	102: --R--		
35. 69-PR-141	16: --R--			
36. 69-TM-169	16: -J---			
37. 74-W-186	1: --R--	102: --R--		
38. 79-AU-197	1: A----	16: AJ---	102: A----	
39. 80-HG-199	51: -J---			
40. 82-PB-204	51: --R--			
41. 90-TH-232	1: A----	18: A----	102: A----	
42. 92-U-235	18: A----			
43. 92-U-238	1: A----	18: AJ---	102: A----	
44. 93-NP-237	18: -J---			
45. 94-PU-239	18: AJ---			
46. 95-AM-241	18: -J---			

V.Zerkin, NDS, IAEA, Vienna-2000

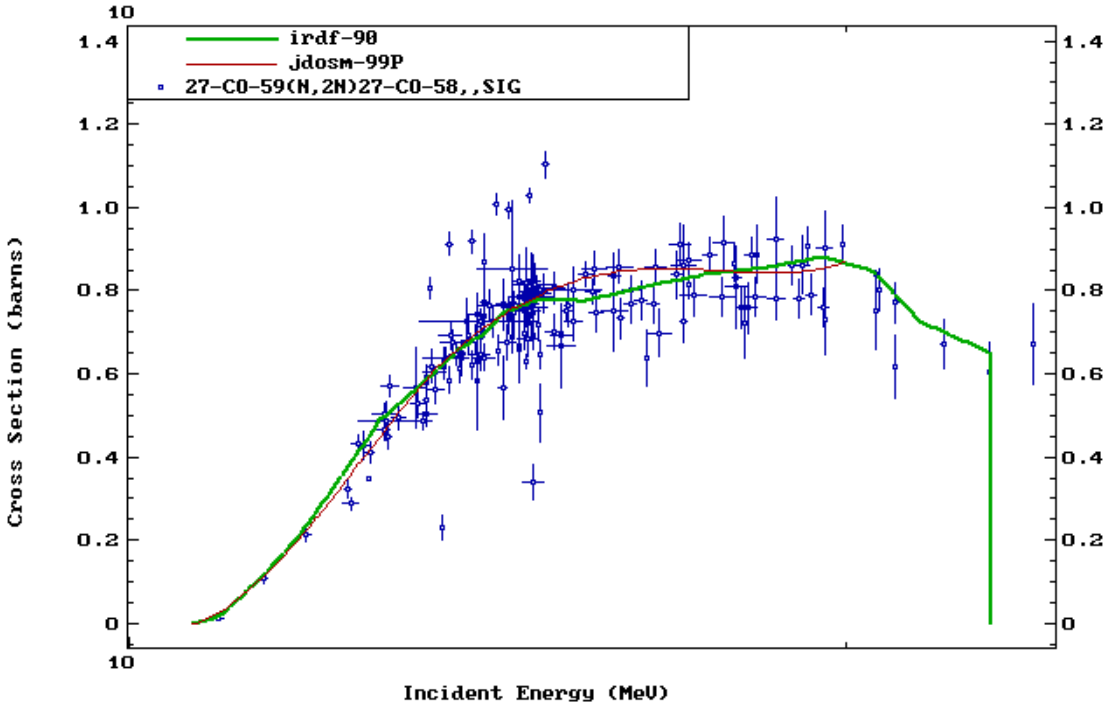
You are [ ] *visitor at this month*

This is the link “27-Co-59” from the previous page.

CO-59 MT=1
(size=2,901 Kb)



CO-59 MT=16
(size=27 Kb)



IRDF-2002

Checking and Processing Codes

Description

The ENDF/B Utility Codes

STANEF

STANEF is designed to perform bookkeeping operations on a data file containing one or more material evaluations in ENDF format. These operations include:

1. Creation or modification of a ``tape ID'' record,
2. Creation or update of the directory in MT=451,
3. Create or modify special hollerith ID records in MT=451 (ENDF-6 only),
4. Resequencing,
5. Conversion of integer and floating point fields to standard format,
6. Creation of a binary (ENDF alternate format) file.

CHECKR

CHECKR is a program for checking that an evaluated data file conforms to the ENDF format. It can recognize the difference between ENDF-6 and ENDF-5 formats and performs its tests accordingly. Integer control fields are checked to see that ENDF/B procedural limits on those fields are not violated. To the extent possible, fatal format errors are trapped to prevent unwanted termination of the program. Any file which passes through CHECKR without error messages fully conforms to the ENDF format. CHECKR will now operate on a file which has not been processed with STANEF. This has been done to facilitate processing by eliminating error messages in CHECKR which would be automatically fixed by STANEF which requires that the input file be in legal ENDF format.

FIZCON

FIZCON is a program for checking that an evaluated data file has valid data and conforms to recommended procedures. It can recognize the difference between ENDF-6 and ENDF-5 formats and performs its tests accordingly. Some of the tests performed include

1. data arrays are in increasing energy order,
2. resonance parameter widths add up to the total,

3. Q-values are reasonable and consistent,

4. no required sections are missing and all cover the proper energy range,

5. secondary distributions are normalized to 1.0,

6. energy conservation in decay spectra.

Optional tests can be performed to check that redundant cross sections such as the inelastic cross section has an energy grid which is the union of all its components and the the cross section values are the sum of the component values at each energy (SUMUP test). Also optionally, algorithms are used to check for possible incorrect entry of data values (Deviant Point test). It is assumed the the file being checked has passed the CHECKR program without any errors being detected.

PSYCHE

PSYCHE is a program for checking the physics content of an evaluated data file. It can recognize the difference between ENDF-6 and ENDF-5 formats and performs its tests accordingly. The present version checks for energy conservation for emitted neutrons and photons, checks Wick's limit for elastic scattering, analyzes resonance parameter statistics, calculates thermal cross sections and resonance integrals, exam- ines continuity across resonance region boundaries and checks ``Q" values against mass tables. It is assumed the the file being checked has passed the CHECKR program without any errors being detected.

The ENDF/B 2002 Pre-Processing Codes Utilty Codes

1) **LINEAR** - Linearize cross sections. ENDF/B allows cross sections to be represented as tables of data points using a number of different interpolation laws between tabulated points; in order to obtain accurate results it is important to interpret the data using these interpolation laws. The interpolation laws are very useful during evaluation, but can present problems when they are used in applications. The subsequent use of the data can be greatly simplified and the accuracy of results improved by first linearizing all of the cross sections, i.e., replace the original tabulated data points and interpolation law by a new table where one can use linearly interpolation between tabulated points to within any required accuracy.

2) **RECENT** - Add the contribution of resonances to the cross sections. ENDF/B allows cross sections to be represented as a contribution of resonance parameters and tabulated background corrections. This code will add the resonance contribution to the background cross sections in order to define the cross sections as linearly interpolable tables at 0 Kelvin (cold). Therefore subsequent codes need only deal with tabulated, linearly interpolable, 0 Kelvin cross sections.

3) **SIGMA1** - Doppler broaden cross sections to any temperature of interest in applications. As in the case of LINEAR and RECENT all cross sections read and written by this code are tabulated, linearly interpolable. All subsequent codes need not explicitly consider temperature effects and need only deal with tabulated, linearly interpolable cross sections at a given temperature.

4) **FIXUP** - Define all cross sections to be consistently exactly equal to the sum of their parts, make format corrections, and a number of other tests and corrections to the data, BEFORE the data is actually used in applications. It is extremely important for use in applications to guarantee that the cross

sections are exactly consistent. For example, the total cross section MUST to defined as equal to the sum of its parts at all energies that appear in one or more of the contributing parts. In addition it should be mentioned that the total will be equal to the sum of its parts at all energies (not just the energies at which the total is tabulated), only if all of the cross sections are linearly interpolable; this illustrates the importance of the steps described above in processing data through each of these codes. Note, if FIXUP's option to output all cross sections on a uniform energy grid is used, the FIXUP output is compatible for use as **NJOY** input.

5) **GROUPIE** - Calculates self-shielded, multigroup cross sections and multiband parameters. This code can be used as a simple and very economical means of obtaining multigroup cross sections, in the ENDF/B format, which can be used in many applications where only multigroup cross sections are required, e.g., dosimetry. For comparing data using **COMPLOT** this code can be used to reduce evaluations that have many resonances, to a form in which integral differences through the resonance region can be more easily seen.

Appendix 4: Proposed Outline for the Contents of IAEA TECDOC

International Reactor Dosimetry File – 2002
IRDF-2002

Table of Contents

(Proposal of E.M. Zsolnay)

1. INTRODUCTION
(Prerpared by R. Paviotti-Corcuera, IAEA, NDS, Vienna, Austria)

2. CONTENT OF THE LIBRARY

2.1. Cross section data and related uncertainty information
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)

2.2. Format of the library
(Prepared by IAEA NDS)

2.3. Isotopic abundances and decay constants
(Prepared by O. Bersillon, France) - shorter version of Section 6.

3. NEW RUSSIAN EVALUATIONS FOR IRDF-2002
(Prepared by K. Zolotarev, Obninsk, Russia)

4. SELECTION OF CROSS SECTIONS FOR IRDF-2002

4.1. Analysis and intercomparison of the data of different up-to-date national reactor dosimetry files and other new evaluations
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)

4.2. Selection of cross sections in the thermal and epithermal neutron energy region for the file IRDF-2002 and characterization of the selected data
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)

4.3. Response of activation reactions in the neutron field of spontaneous fission of ^{252}Cf
(Prepared by W. Mannhart, Physikalisch-Technische Bundesanstalt, Braunschweig, Germany)

4.4. Evaluation of Cross Sections for IRDF-2002 at 14 MeV
(*Prepared by L. R. Greenwood, Pacific Northwest National Laboratory, Richland, WA, USA)*

4.5. Selection of fast neutron (threshold) reaction cross sections for the file IRDF-2002 and characterization of the selected data
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)

5. RADIATION DAMAGE FILES AND COMPUTER CODES
(Prepared by L.R. Greenwood and P. Griffin, USA)

6. ISOTOPIC ABUNDANCES AND DECAY DATA FOR DOSIMETRY APPLICATIONS
(Prepared by O. Bersillon, France)

7. CROSS SECTION PLOTS
(Prepared by V. Zerkin, IAEA NDS)

Appendix 5: Contents of IAEA TECDOC as Discussed at Meeting

1. INTRODUCTION
(Prerpared by R. Paviotti-Corcuera, IAEA, NDS, Vienna, Austria)

2. CONTENT OF THE LIBRARY

2.1. Cross section data and related uncertainty information
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)
List of the cross sections in IRDF-2002, their sources and main characteristics
2.2. Format of the library
(Prepared by P. K. McLaughlin IAEA NDS)
Description of the format. Names: IRDF-2002P – point version,
IRDF-2002G – group version.
2.3. Isotopic abundances and decay constants
(Prepared by O. Bersillon, CEA/DAM Íle-de-France, Service de Physique Nucleare, France)
- short version of Section 7; text here, plots to Appendix 5

3. NEW RUSSIAN EVALUITIONS FOR IRDF-2002
(Prepared by K. Zolotarev, Institute of Physics and Power Engineeering, Obninsk, Russia) – shorter version, with complete documentation in Appendix 1

4. SELECTION OF CROSS SECTIONS FOR IRDF-2002

4.1. Analysis and intercomparison of the data of different up-to-date national reactor dosimetry files and other new evaluations
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)
Short summary of the results present in the previous Progress Reports

4.2. Selection of cross sections in the thermal and epithermal neutron energy region for the file IRDF-2002 and characterization of the selected data
(*Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary*
Short description of the selection procedure, results of comparison with experimental data in standard neutron fields, and suggestions

4.3. Response of activation reactions in the neutron field of spontaneous fission of 252Cf
(Prepared by W. Mannhart, Physikalisch-Technische Bundesanstalt, Braunschweig, Germany)
Intercomparison of calculated integral cross section data with experimental data in the standard neutron field of ^{252}Cf spontaneous fission; results, qualification of the data investigated

4.4. Evaluation of Cross Sections for IRDF-2002 at 14 MeV

(*Prepared by L. R. Greenwood, Pacific Northwest National Laboratory, Richland, WA, USA*
Results of comparison with experimental data - text here, plots to Appendix 2

4.5. Selection of fast neutron (threshold) reaction cross sections for the file IRDF-2002 and characterization of the selected data
(Prepared by E. M. Zsolnay, Institute of Nuclear Techniques, Budapest University of Technology and Economics, Budapest, Hungary)
Short description of the selection procedure, results and suggestions

5. CONSISTENCY TEST OF THE CROSS SECTION DATA IN THE ACRR REFERENCE NEUTRON FIELD
(Prepared by P. Griffin, Sandia National Laboratories, Albuquerque, USA)
Intercomparison of the calculated integral cross section data with experimental data in the reference neutron field, and results

6. RADIATION DAMAGE FILES AND COMPUTER CODES
(Prepared by L.R. Greenwood, Pacific Nortwest Laboratory, Richland, USA, and P. Griffin, Sandia National Laboratories, Albuquerque, USA)
Description of data and codes

7. ISOTOPIC ABUNDANCES AND DECAY DATA FOR DOSIMETRY APPLICATIONS
(Prepared by O. Bersillon, CEA/DAM Íle-de-France, Service de Physique Nucleare, France)
Description of the content of the nuclear data file, and most of the tables to Appendix 5

8. CROSS SECTION PLOTS
(Prepared by P. K. McLaughlin (V. Zerkin), IAEA NDS)
Short description of the plots and programs, and plots to Appendix 3

9. APPENDICES

Appendix 1: *K. Zolotarev* – Complete documentation of the new evaluations.
Appendix 2: *L. Greenwood* – 14 MeV cross sections plots
Appendix 3: *P. McLaughlin* – Plots of the cross sections in IRDF-2002
Appendix 4: *H. Nolthenius* – Covariance data of the cross sections in IRDF-2002
Appendix 5: *O. Bersillon* – Details of nuclear decay data (tables, plots)

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RIPL for FTP file transfer of RIPL;
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