International Atomic Energy Agency

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I N D C INTERNATIONAL NUCLEAR DATA COMMITTEE

Summary Report of an IAEA Consultants' Meeting

"Maintain FENDL Library for Fusion Applications"

FENDL-2 Library for Fusion Applications – Status and Future Developments

IAEA Headquarters
Vienna, Austria
10 – 12 November 2003

Prepared by
R. Forrest, UKAEA Culham Division
and
A.Trkov, IAEA Nuclear Data Section

November 2003

IAEA NUCLEAR DATA SECTION, WAGRAMER STRASSE 5, A-1400 VIENNA

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Abstract

The discussions and conclusions of the meeting to "Maintain FENDL library for Fusion Applications" are summarized in this report. A presentation was made by each of the participants, followed by a review of FENDL-2: evaluations and recommendations, and discussions on the special purpose libraries and processed files, with relevant further action thereon being determined.

November 2003

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Introduction

The objective of the meeting was to determine and assess possible improvements to the FENDL-2 library, and agree on any recommended activities well in advance of the next INDC meeting in May 2004.

The meeting was opened by Mr. A Nichols (IAEA), who welcomed the participants (Appendix 1).

Mr. U. Fischer was elected chairman of the meeting, with Mr. R. Forrest being elected rapporteur.

The agenda of the Meeting was adopted as presented (Appendix 2).

Presentations

The presentations made during the meeting are attached to this summary report (Appendix 3).

Mr. Trkov gave a presentation on the current status of the FENDL-2 libraries. It was noted that the libraries are static and act as a reference but, with ITER approaching a new phase, there is a need to consider an update. Mr. Trkov noted that a new dosimetry library is to be released by the IAEA (IRDF-2002) and this should be taken into consideration when discussing an update to the FENDL-2 dosimetry library. Similarly, Mr. Forrest noted that the most recent European activation library (EAF-2003) is available.

With regard to processed files it was noted that the ACE library for FENDL-2.0 is available. Assuming that an updated library is recommended, the question was raised whether it is necessary to reprocess the whole ACE library to include the new MCNP features, such as probability tables, and whether for the multi-group library finer group structures need to be added. A decision on this matter was deferred to later in the meeting (q.v. Processed files).

It was felt that the ability to do MCNP calculations with temperature-dependent cross section data would be useful, whereas work to enable the existing ACE files to be Doppler broadened is already underway. It was noted that in order for this transformation to be successful, the original file must contain enough data points.

Mr. Cheng raised the question whether it was necessary to have both dosimetry and activation files and, in this regard, it was noted that there is a different level of detail, including uncertainty, in the files. Furthermore, there is no guarantee that the same data source will be used in the activation and dosimetry files.

Mr. Ignatyuk asked whether new materials should be introduced into the general purpose library and it was agreed that this would be discussed later in the meeting (q.v. Review of FENDL-2).

A presentation was given by Mr. Fischer on FENDL benchmarking conducted since the last FENDL consultants meeting in October 1998. Included therein were the major findings for Be-9, Fe-56, Ni-58 60, Cr-52, Si-28, C-12 and W and these were used in the subsequent discussions.

A presentation was given by Mr. Nishitani on the re-analysis of integral testing for FENDL-2 and JENDL-3.3 covering the elements O, Fe, Cu, V, W, Li, Al, Si and C. A brief overview of the new features of JENDL-3.3 was also presented.

Mr. Forrest gave a presentation on the development of European activation libraries since FENDL-2-A, including new work on data above 20 MeV for IFMIF applications. It was mentioned that the library EAF-2003 is now available and that the next version to 60 MeV will be available in about twelve months time.

A presentation was given by Mr. Cheng on the application of activation data to waste management studies. It was concluded that hands-on recycling would be an option for reduced activation ferritic martensitic steel structures in a fusion power plant.

Mr. Ignatyuk gave a presentation on some new evaluations for BROND-3 (Zr, Pb isotopes and important fission products), which generally contains data up to 20 MeV, but some files for minor actinides go to 150 MeV. It is expected that BROND-3 will be available in 2005. It was pointed out that the Russian dosimetry file is a major contributor to IRDF-2002.

Mr. Fischer presented the new data needs as found in the European technology programme, focussing on the ITER test blanket modules and IFMIF test facility needs (EFF-DOC-852).

Review of FENDL-2: Evaluations and Recommendations

The current status of each of the evaluations in the FENDL-2 general purpose library was reviewed based on the EFF and JENDL benchmarking and the new data needs, as shown in EFF-DOC-852 and the contents list of FENDL-2.

The general recommendations arising from the discussions are listed below:-

- The update should be of a relatively minor nature with only major omissions or inaccuracies corrected.
- The update will be termed FENDL-2.1 and should be produced on a time scale of about one year.
- Generally the upper energy range should remain at 20 MeV, but evaluations with higher energy for IFMIF applications will be accepted where available.
- The requirement for covariance data and isotope evaluations are major factors guiding the need for updating FENDL-2.0.
- A major update to FENDL, named FENDL-3, should be considered when the anticipated new regional evaluated libraries become available in about 2005.

Recommendations for updates to the FENDL-2.0 materials are given below. All changes made should be checked - format, processing, applicability and benchmarking if available - to confirm the suitability of the new data file.

H-1 Leave as it is, but consider the new preliminary standards file or JENDL-3.3, which includes covariances, as an alternative.

H-2 Replace BROND-2/JENDL-FF with JENDL-3.3.

H-3 Leave as it is.

Li-6 Leave as it is, but consider the new preliminary standards file so as to include covariance data.

Li-7 Leave as it is.

Be-9 JEFF-3.0 contains the most recent evaluation but there are possible problems with the split (n,2n) when processing with NJOY. Processing has been done at NEA data bank, but this needs to be checked. It was noted that (n,γ) is missing in JEFF-3.0, and this will be added by the JEFF team. The benchmarking results between JEFF-3.0 and FENDL-2 are similar, the availability of covariance data being the deciding factor for the change. Recent experiments by the Japan Atomic Energy Research Institute (JAERI) on breeder assemblies containing Be show severe overestimation of the tritium production. The Forschungszentrum Karlsruhe GmbH (FZK) will confirm that benchmarking is no worse than for current data.

B-10, 11 Leave as it is.

C-12 Replace JENDL-FF/ENDF/B-VI with JENDL-3.3. FZK will confirm that benchmarking is no worse than for current data.

N-14, 15 Leave as it is.

O-16 Update to more recent ENDF/B-VI to include MF=6 and covariance data. JAERI will confirm that benchmarking is no worse than for current data.

F-19 Leave as it is.

Na-23 Replace JENDL-3.1 with JENDL-3.3 because it includes covariance data.

Mg isotopes FENDL-2 is an elemental evaluation, but no isotopic evaluations are available, so leave as it is.

Al-27 Leave as it is.

Si-28 Consider as replacements EFF-3.0 or ENDF/B-VI.8. FZK will confirm that benchmarking is no worse than for current data.

Si-29, 30 Update to more recent ENDF/B-VI.8. FZK will confirm that benchmarking is no worse than for current data.

P-31 Leave as it is.

S, Cl, K, Ca FENDL-2 contains elemental evaluations, but no isotopic evaluations are available so leave as they are.

Ti isotopes FENDL-2 is an elemental evaluation and thus needs to be revised to give isotopic evaluation. Neither JENDL-3.3 nor ENDF/B-VI is complete but JENDL-3.3 is the best choice. JAERI will confirm that benchmarking is no worse than for current data.

V-51 Replace JENDL-FF with JENDL-3.3, this includes covariance data. JAERI will confirm that benchmarking is no worse than for current data.

Cr-52 No change of data source, however, neutron emission spectrum at 14 MeV should be looked at again to remove a longstanding error. FZK will confirm that benchmarking is no worse than for current data.

Cr-50,53,54 Leave as it is.

Mn-55 Replace JENDL-3.1 with JENDL-3.3 because of covariance file. JAERI will confirm that benchmarking is no worse than for current data.

Fe-56 Update to EFF-3.1 as this is now used in JEFF-3.0 and extensively benchmarked. New integral data performed at Ohio University should be included in SINBAD so that new FENDL file can be tested against it. FZK will confirm that benchmarking is no worse than for current data.

Fe-54, 57, 58 Leave as it is.

Co-59 Leave as it is.

Ni-58, 60 ENDF/B-VI should be replaced by EFF-3.0 because of improved covariance file and additional high resolution data. FZK will confirm that benchmarking is no worse than for current data.

Ni-61, 62, 64 Leave as it is.

Cu-63, 65 Leave as it is.

Ga Leave as it is.
Zr FENDL-2 is an elemental evaluation, but no isotopic data is yet available (BROND-3 is the only candidate for future update); leave as it is.

Nb-93 Leave as it is.

Mo isotopes FENDL-2 is an elemental evaluation and needs to be revised to give isotopic data. Consideration should be given to replacement with JENDL-3.3. JAERI will confirm that benchmarking is no worse than for current data.

Sn isotopes FENDL-2 is an elemental evaluation, but no isotopic data is yet available. Leave as it is.

W isotopes FENDL-2 is an elemental evaluation, thus needs to be revised to give isotopic data. Consideration should be given to replacement with JENDL-3.3. FZK and JAERI will confirm that benchmarking is no worse than for current data.

Ta-181 Consideration should be given to replacement with JENDL-3.3 if different from current data. It is recommend that an integral benchmark experiment be carried out.

Au-197 Leave as it is.

Pb Leave as it is.

Bi-209 Replace JENDL-3.1 with ENDF/B-VI.8 because of the availability of covariance data.

In summary, the following materials need, or may need, to be revised: H-1, H-2, Li-6, Be-9, C-12, O-16, Na-23, Si isotopes, Ti isotopes, V-51, Cr-52, Mn-55, Fe-56, Ni-58, 60, Mo isotopes, W isotopes, Ta-181 and Bi-209. It is anticipated that these revisions can be completed by the end of 2004.

Special purpose libraries

The current status of each of the special purpose libraries was discussed. The recommendations arising therefrom are listed below:

- The existing FENDL-2 file for dosimetry will not be updated. The use of IRDF-2002 is recommended instead.

- The existing FENDL-2 file for activation will not be updated. The use of EAF-2003 is recommended instead. The cross section library will be available in both EAF and ENDF format (the same as used for JEFF-3.0) through the IAEA NDS. It should be noted that this includes the decay data library.

- No changes are recommended to the FENDL-2 charged particle library.

Processed files

Processed files (ACE and multi-group) for FENDL-2.0 are available to users and an updated library will require similar files. In principle, it should be easy to rerun NJOY to generate ACE files suitable for MCNP. There are options to either re-process all files or only re-process the new materials to include the new MNCP features. Following discussion, it was recommended that:

- As a first priority, new materials will be processed using the probability table feature.
- As a second priority, task existing materials will also be re-processed with the new feature.

Mr. Trkov described how ACE files can be Doppler broadened with the SIGACE code. It was agreed that this code will be made available to users (by download from the FENDL web page) in order that files at arbitrary temperature can be generated.

The group files were discussed, however, the possibility of additional group structures was not considered attractive, with the Vitamin-J structure a standard for many applications. Consequently only new materials need to be re-processed. This re-processing will be undertaken by the IAEA NDS, and the processed files then made available to the users by download and CD-ROM.

APPENDIX 1

International Atomic Energy Agency

Consultants Meeting

"Maintain FENDL library for fusion applications"

FENDL-2 Library for Fusion Applications – Status and Future Developments

10 – 12 November 2003

IAEA Headquarters, Vienna, Austria
Meeting Room A0418

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APPENDIX 2

Consultants Meeting

"Maintain FENDL library for fusion applications"
FENDL-2 Library for Fusion Applications – Status and Future Developments

IAEA Headquarters, Vienna, Austria
10 – 12 November 2003
Meeting Room A 0418

Agenda

Monday 10 November

08:30 - 09:30 Registration (at Gate 1, IAEA Headquarters)
09:30 - 10:30 Opening session
Welcoming address by A.L. Nichols, Head, Nuclear Data Section
Election of chairman and rapporteur
Adoption of the Agenda
10:30 - 11:00 Coffee break, Administrative and Financial Matters
11:00 - 11:30 A. Trkov: Current status of the FENDL-2 Library
(evaluated files, Ace and group libraries, temperature
dependence and SIGACE code)
11:30 - 12:30 U. Fischer: Identified deficiencies of FENDL-2
(followed by extended discussion)

12:30 - 14:00 Lunch

14:00 - 15:30 Presentations by participants (20 min+10min discussion)
15:30 - 16:00 Coffee break
16:00 - 17:30 Presentations by participants (20 min+10min discussion)
17:30 Reception

Tuesday 11 November

09:00 - 10:30 Selection of basic evaluated data files
Is the current list of materials adequate?
Is there a need to replace individual files and which?
10:30 - 11:00 Coffee break

11:00 - 12:30 Processing of Ace files
(Are any refinements needed with respect to input options
thinning tolerances, gamma production damage and heating
data?)
Doppler broadening of Ace files and SIGACE code.

12:30 - 14:00 Lunch

14:00 - 15:30 Processing of group files
Is the current group structure adequate?
Are the data processing tools adequate?
What refinements can we do?
15:30 - 16:00 Coffee break
16:00 - 18:00 Summarizing discussion

Wednesday 12 November

09:00 - 10:30 Drafting of the summary report and formulation of conclusions
10:30 - 11:00 Coffee break
11:00 - 12:30 Review of the summary report

12:30 - 14:00 Lunch

14:00 - 15:30 Final review and approval of the summary report
15:30 Closing of the meeting

APPENDIX 3

PRESENTATIONS BY PARTICIPANTS

3.1. In Search of an Environmentally Attractive Fusion Power Plant; *E.T. Cheng*

3.2. FENDL-2 Benchmarking – Overview of Recent Results; *U. Fischer*

3.3. Needs for Nuclear Data Development and Improvement in the Fusion Technology Program in 2003-2006 and Beyond; *P. Batistoni, U. Fischer and R.A. Forrest*

3.4. Nuclear Data for Design Analyses of the Test Blanket Modules in ITER: Review and Recommendations for EFF/JEFF Evaluations; *U. Fischer*

3.5. Activation libraries since FENDL/A-2, *R. Forrest*

3.6. Comments on FENDL-2 from ITER-IT Naka Joint Work Site; *T. Nishitani (on behalf of Hiro Iida)*

3.7. Outline of Evaluations for JENDL-3.3 - fusion related issues; *T. Nishitani (on behalf of Keiichi Shibata and Nuclear Data Center, JAERI)*

3.8. Re-Analysis of Integral Tests using MCNP-4c with JENDL 3.3, -3.2 and FENDL2 for Fusion Related Materials; *T. Nishitani, F. Maekawa and M. Wada*

3.9. FENDL-2 library for Fusion Applications – Status and Future Developments, *Andrej Trkov*

Appendix 3.1.

In Search of an Environmentally Attractive Fusion Power Plant

E.T. Cheng

In Search of an Environmentally Attractive Fusion Power Plant

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IAEA Consultants Meeting on “Maintaining FENDL Library for Fusion Applications”

IAEA Headquarters, Vienna, Austria
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Intrinsic Attractiveness of Fusion Energy

- Its Fuel Cycle Does Not Depend on Fissionable Materials
- It does not Generate Radioactive Fusion Reaction Products

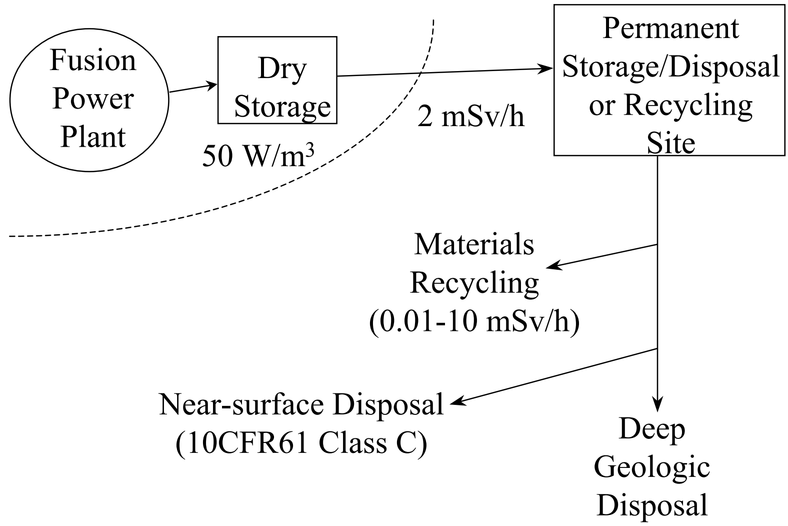
Environmental Issues Associated with Fusion Power Plants

- Generation of Environmentally Undesirable Materials with Fusion Neutrons
- Fusion Neutron Induced Long-lived Radioactivity (High Energy and Fluence)
  - Half-lives from 418 y (Ag108m) to 720,000 y (Al26)
  - Making Fusion Waste Difficult to Justify as Low Level Waste (10CFR61) because the Radioactivity would not Decay Away in 500 Years

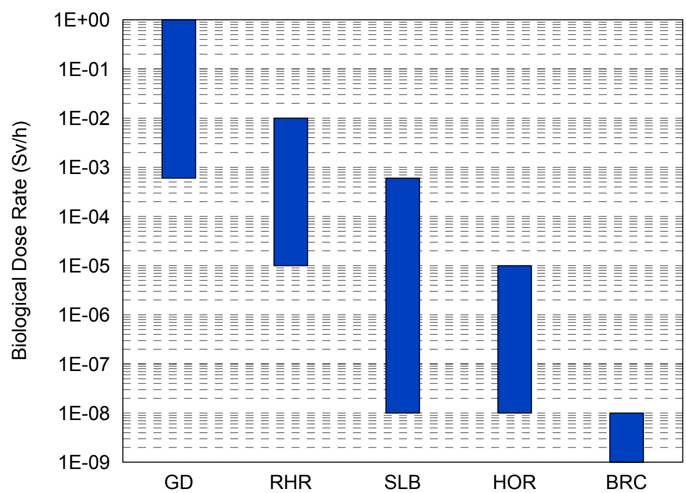
Mitigating Long-lived Fusion Waste
(Envisioned Through Decades of Studies)

- Selection of Reduced Activation Fusion Materials – Reduction of Activation Level of Fusion Waste
- Recycle of Fusion Power Plant Components and Materials – Reduction of Fusion Waste Quantities

Waste Management



Waste Classification and Dose Rate



Objective of Study

- Demonstration in Calculations a Fusion Power Plant that All Discharged Reactor Components Could be Recycled by Hands-on Handling Mechanism: Contact Dose Rate ≤ 10 μSv/h After Reasonable Cooling Times

Fusion Power Extracting Components Adapted to ARIES-RS (with Beryllium Neutron Multiplier)

Component	Outboard	Inboard
First Wall (3 mm/2.5 FPY)	RAFM Steel	RAFM Steel
Blanket-1	60%MS+30%Be+10%RAFM Steel (0.2 m/2.5 FPY)*	60%MS+30%Be+10%RAFM Steel (0.2 m/2.5 FPY)*
Blanket-2	90%MS+10%RAFM Steel (0.3 m/7.5 FPY)	-
Replaceable Shield	10%MS+ 90%RAFM Steel (70 mm/7.5 FPY)	10%MS+ 90%RAFM Steel (0.2 m/7.5 FPY)
High Temperature Shield	5%MS+4%W+ 76%304SS+ 15%RAFM Steel (0.28 m/40 FPY)	5%MS+4%W+ 76%304SS+ 15%RAFM Steel (0.26 m/40 FPY)

*Without beryllium neutron multiplier, Blanket-1 becomes 90%MS+10%RAFM Steel.

Fusion Blanket Concepts Adapted to ARIES-RS (Other Lifetime Components)

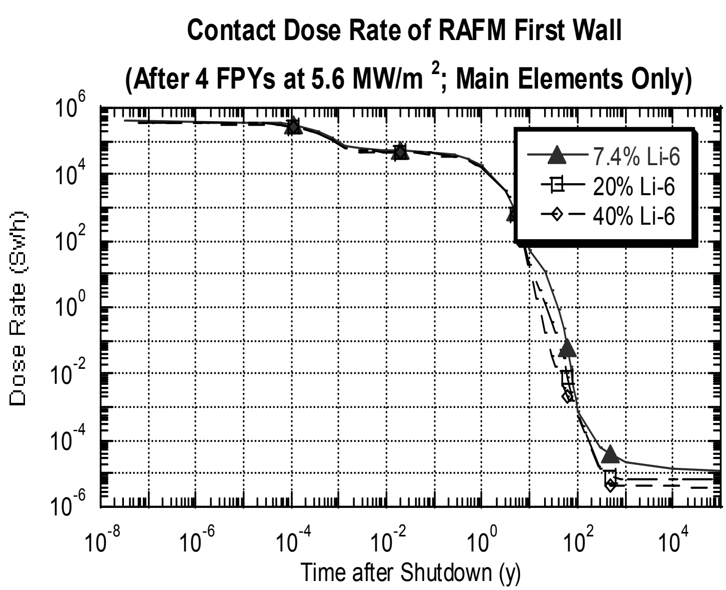
Component	Inboard	Outboard
Low Temperature Shield	5%He+32.3%WC+ 47.7%B_4C+ 15%304SS (0.28 m)	5%He+7%B+ 88%SS304 (0.4 m)
Vacuum Vessel	5%He+24%WC+ 36%B_4C+35%SS304 (0.2 m)	5%He+7%B+ 88%SS3304 (0.3 m)
Cryostat	18%SS316+82%Void (0.24 m)	18%SS316+82%Void (0.24 m)
Magnet	SS316+Cu+NbTi+ LHe+Ins. (0.5 m)	SS316+Cu+NbTi+ LHe+Ins. (0.5 m)

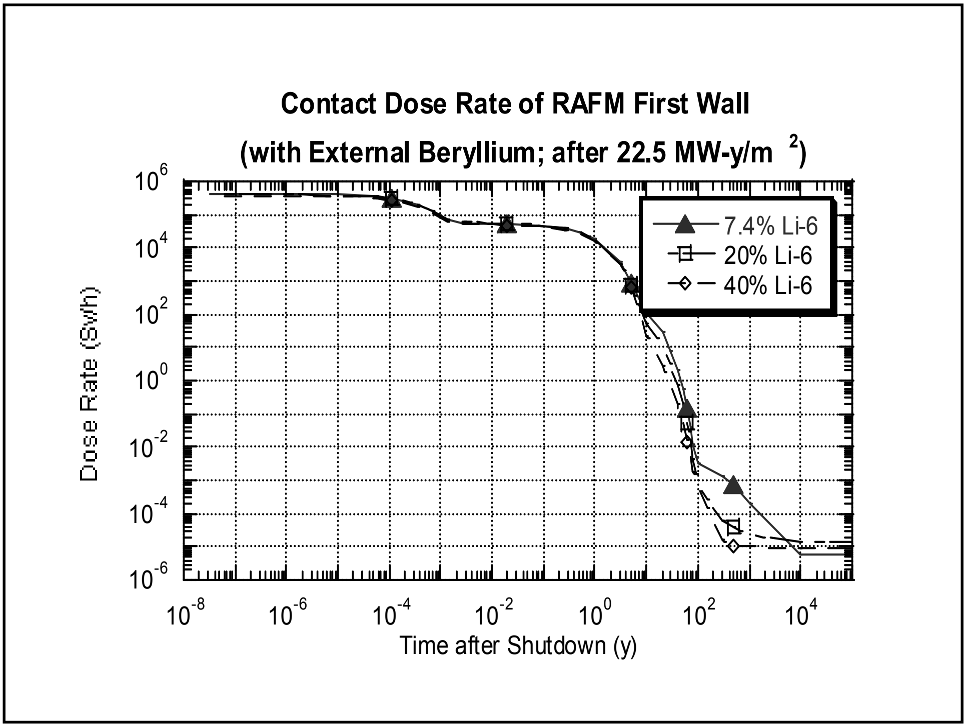
Methods of Calculations

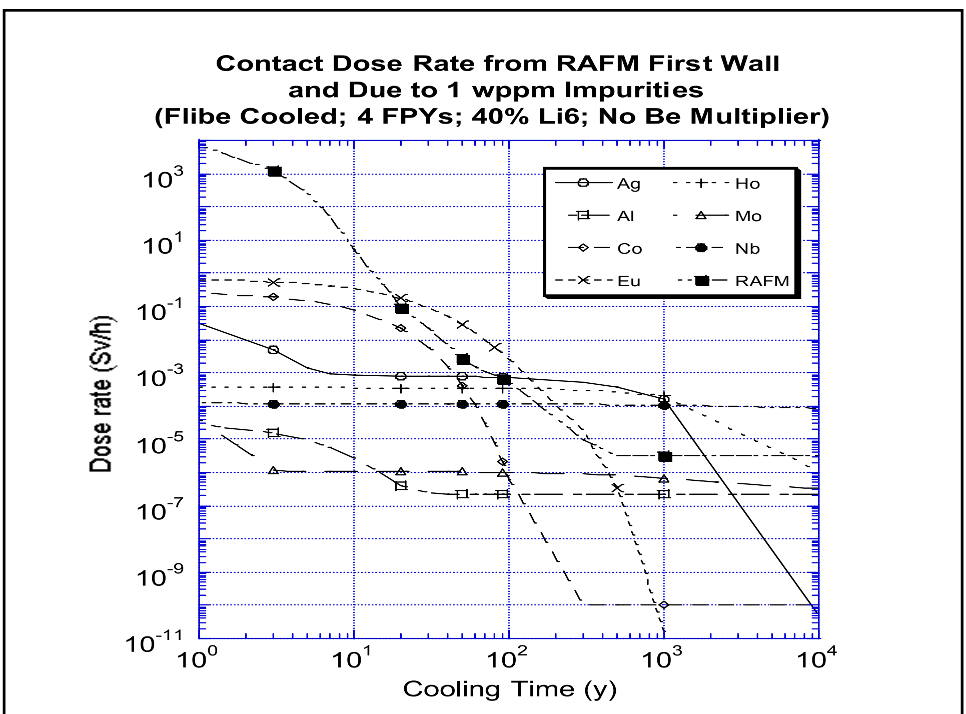
- Transport Code: ANISN, One-dimensional Neutronics Model
- Nuclear Data Libraries: FENDL Multigroup and Activation Data Libraries
- Activation Calculation Code: REAC3
- Accuracy of Calculations is Vital to the Conclusions to be Drawn from these Calculations. Nuclear Data Plays the Most Important Role in these Assessments.

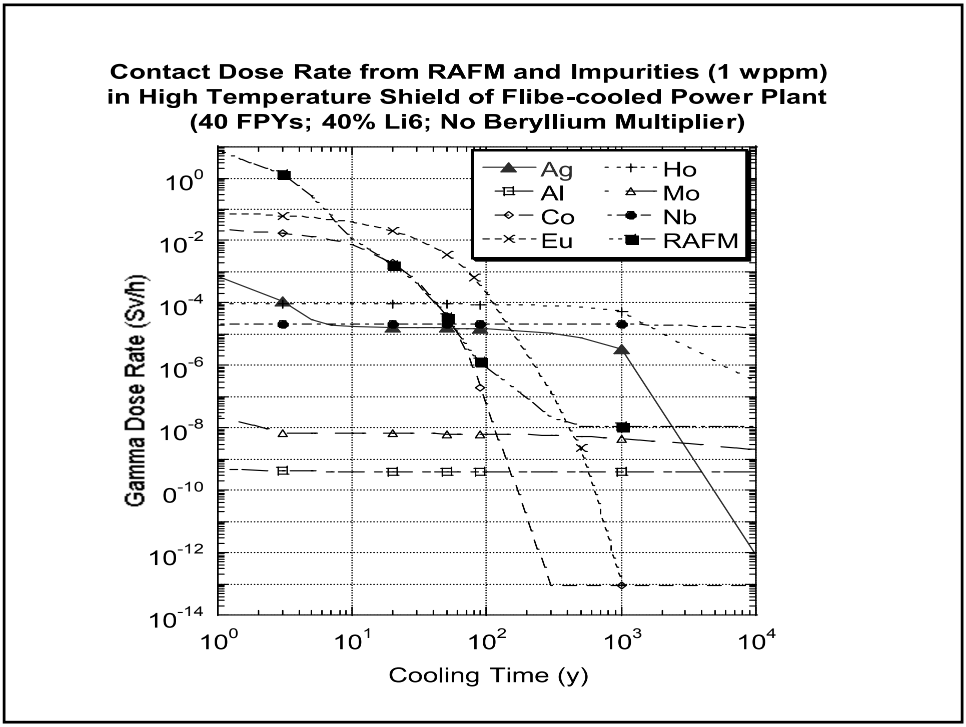
Compositions of RAFM Steel

- Main Alloying Elements:12%Cr, 3%W, 0.4%Ti, 0.25%Y_2O_3 (ODSFS), and balance of Fe, all by wt. % (S. Zinkle, ORNL, March 2002, private communication).
- Deleterious Impurities: Ag (< 0.005), Al (0.5), Co (<0.02), Eu (0.01), Ho (0.01), Mo (1), Nb (<0.02)…all in wppm (R. Klueh, et al., J. Nucl. Mats., 280 (2000) 253.

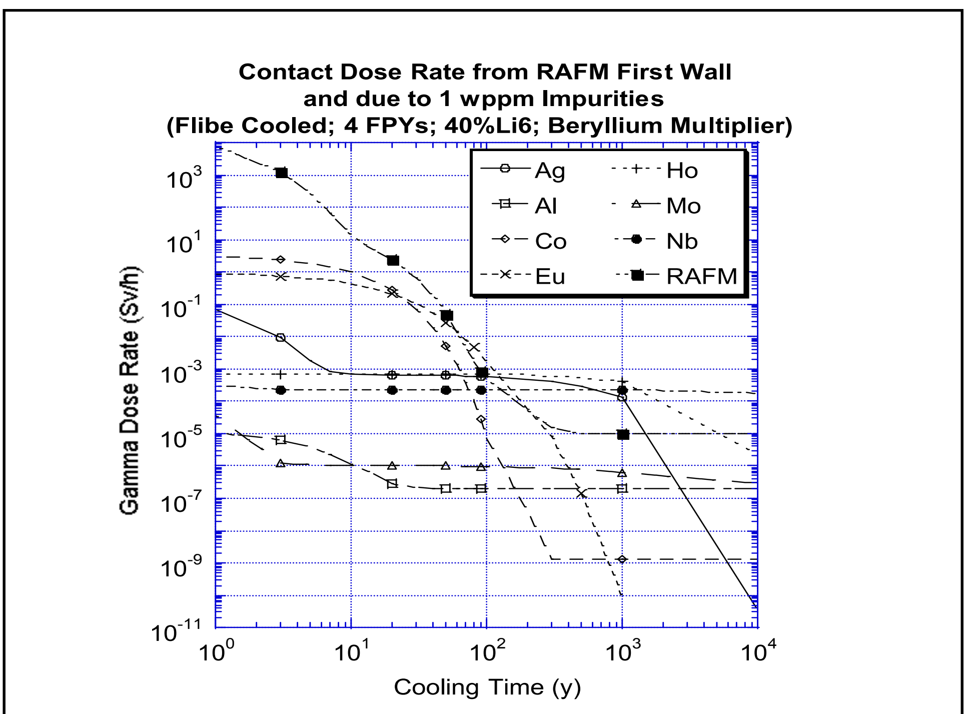



Contact Dose Rate of RAFM First Wall
(with External Beryllium; after 22.5 MW-y/m 2)
Dose Rate (Sv/h)
Time after Shutdown (y)
7.4% Li-6
20% Li-6
40% Li-6

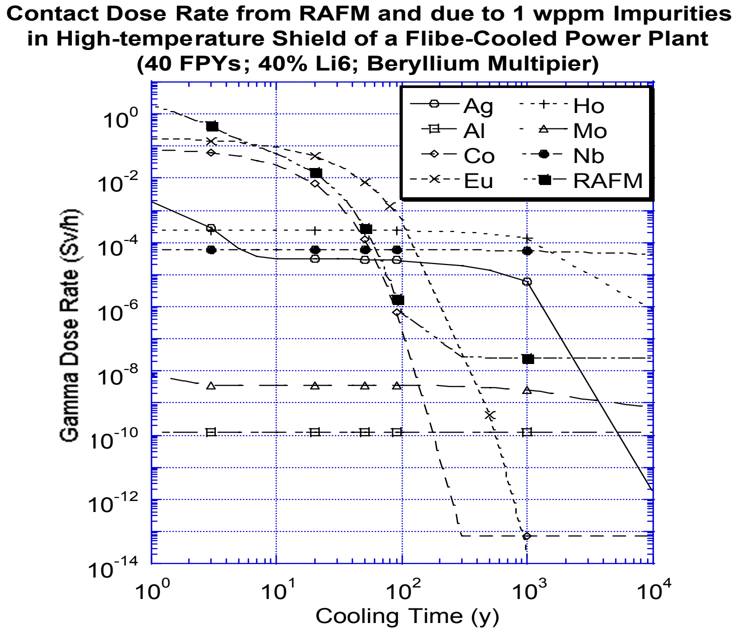

Contact Dose Rate from RAFM First Wall
and Due to 1 wppm Impurities
(Flibe Cooled; 4 FPYs; 40% Li6; No Be Multiplier)
Dose rate (Sv/h)
Cooling Time (y)
Ag
Ho
Al
Mo
Co
Nb
Eu
RAFM



Contact Dose Rate from RAFM and Impurities (1 wppm)
in High Temperature Shield of Flibe-cooled Power Plant
(40 FPYs; 40% Li6; No Beryllium Multiplier)
Ag
Al
Co
Eu
Ho
Mo
Nb
RAFM
Gamma Dose Rate (Sv/h)
Cooling Time (y)



Contact Dose Rate from RAFM First Wall
and due to 1 wppm Impurities
(Flibe Cooled; 4 FPYs; 40%Li6; Beryllium Multiplier)
Ag
Al
Co
Eu
Ho
Mo
Nb
RAFM
Gamma Dose Rate (Sv/h)
Cooling Time (y)



Contact Dose Rates due to Main Alloying Elements in RAFM Steel

- Medium-lived radionuclides are mainly Co60 (half-life 5.27 y) which is generated from Fe58(n,γ)Fe59 →Co59(n,γ) reactions, and Hf178n (31 y) which is primarily due to the W182(n,n'α) reaction.
- Long-lived radionuclides are Re186m (2x105 y) which is due to W186(n,γ)W187→Re187(n,2n) and W184(n,γ)W185→Re185(n,γ) reactions, and Mn53 (3.7x106 y) from Fe54(n,2n)Fe53→Mn53 reaction.
- It takes about 500 years for medium-lived activity to diminish until the long-lived radionuclides to dominate activity level.

Contact Dose Rates due to Main Alloying Elements in RAFM Steel Cont.)

- The first wall activity could be about 10 μSv/h after 500 years of cooling
- For the high-temperature shield component, however, it would take only 100 years or less for the contact dose rate to drop below 10 μSv/h level
- The power plant with a beryllium neutron multiplier shows an advantage in shortening the needed cooling times to allow the activity to decay below 10 μSv/h dose rate level. These cooling times are 300 and 70 years, respectively, for the first wall and the high-temperature shield components compared to 500 and 100 years for the power plant without a beryllium neutron multiplier

Contact Dose Rates due to Impurity Elements in RAFM Steel (1)

- Ag
  - There is no medium-lived radionuclide observed.
  - The long-lived radionuclide is Ag108m (481 y), which is generated from Ag107(n,γ) and Ag109(n,2n) reactions.
  - The first wall contact dose rate due to 1 wppm of Ag is about 1 mSv/h, while the high-temperature shield contact dose rate is a few tens μSv/h for the first few hundred years after discharge.
  - The power plant with a beryllium neutron multiplier shows a slightly lower dose rate for both components.

Contact Dose Rates due to Impurity Elements in RAFM Steel (2)

- Al
  - There is no medium-lived radionuclide observed.
  - The long-lived radionuclide is Al26, which is generated from the Al27(n,2n) reaction.
  - The contact dose rates due to 1 wppm Al in the first wall component is about 0.2 μSv/h for both power plants.
  - The contact dose rate in the high-temperature shield component is about 0.5 pSv/h or lower, a level much lower than the natural radiation background.

Contact Dose Rates due to Impurity Elements in RAFM Steel (3)

- Co
  - The medium-lived radionuclide is Co60 (5.27 y), generated via the Co59(n,γ) reaction.
  - The long-lived, but very low activity, radionuclide is Fe60 (1.5x106 y), generated due to the (n,p) reaction with the radioactive Co60.
  - The dominance of Co60 is about 100 years for the first wall component and 50 years for the high-temperature shield component before the contact dose rates due to 1 wppm of Co drop below 10 μSv/h.
  - The power plant without the beryllium neutron multiplier has a slight advantage in having a lower Co60 activity due to the reduced neutron population in the blanket components.
  - The dose rate due to fe60 (main gamma rays are emitted from its decay daughter radionuclide Co60) at 300 years after discharge and thereafter is well below the natural background level.

Contact Dose Rates due to Impurity Elements in RAFM Steel (4)

- Eu
  - There is no long-lived radionuclide observed.
  - There are quite a few medium-lived radionuclides generated including: Eu152 (13.5 y), via the Eu151(n,γ) reaction, Eu154 (8.59 y), via the Eu153(n,γ) reaction, and Eu150 (36.9 y), via the Eu151(n,2n) reaction.
  - The 1 wppm of Eu shows a contact dose rate of 10 μSv/h in the first wall component at about 300 years after discharge for both power plants.
  - In the high-temperature shield, the dose rate could drop below 10 μSv/h at 100 years or less after discharge.

Contact Dose Rates due to Impurity Elements in RAFM Steel (5)

- Ho
  - The only significant radionuclide generated is the long-lived Ho166m (1,200 y) which is generated via the Ho165(n,γ) reaction
  - The contact dose rate levels due to 1 wppm of Ho are about 0.4 mSv/h and 0.1 mSv/h, respectively, at the first wall and high-temperature shield components of the power plant with no beryllium neutron multiplier.
  - The contact dose rates due to Ho in the power plant with beryllium neutron multiplier are about a factor of 2 higher than that without a neutron multiplier, and it is primarily because of the increased neutron population in the blanket components.

Contact Dose Rates due to Impurity Elements in RAFM Steel (6)

- Mo
  - There is one, but not significant, medium-lived radionuclide generated in Mo: namely Nb93m (16.1 y), induced due to the Mo93(n,p) reaction.
  - The long-lived radionuclides include Nb91 (680 y), due to the Mo92(n,2n)Mo91→Nb91 reactions, Mo93 (4,000 y), due to Mo94(n,2n) and Mo92(n,γ) reactions, and Nb94 (20,000 y), due to Mo94(n,p), and Mo95(n,n'p) reactions.
  - The contact dose rate levels due to 1 wppm Mo are 1 μSv/h and 10 pSv/h, respectively, at the first wall and high-temperature shield components.
  - The power plant with a beryllium neutron multiplier has a slight advantage in generating a lower long-lived radioactivity in the high-temperature shield component.

Contact Dose Rates due to Impurity Elements in RAFM Steel (7)

- Nb
  - There is no medium-lived radionuclide observed.
  - The only long-lived radionuclide generated is Nb94 (20,000 y), and it is via the Nb93(n,γ) reaction.
  - The contact dose rate levels due to 1 wppm of Nb are 0.1 mSv and 20 μSv/h, respectively, in the first wall and high-temperature shield components of the power plant without the beryllium neutron multiplier.
  - The contact dose rates in the power plant with the beryllium neutron multiplier are higher by a factor of 2, and it is primarily due to the higher neutron population in the blanket components.

Contact Dose Rates (mSv/h) from Main Alloying Elements of the Reduced Activation Ferritic Martensitic (RAFM) Steels and Deleterious Impurities (1 wppm) at several cooling times after lifetime Irradiation in Flibe-cooled Power Plants. All with 40%Li6 in Lithium: Case A: No Beryllium Neutron Multiplier; Case B: with Beryllium Neutron Multiplier).

Element (1 wppm)	Cooling Time (y)					
	10	30	50	100	300	500
RAFM: Main Alloying Elements						
A: First Wall	4,870	22	2.9	0.53	0.0092	0.0033
A: High-temp. Shield	11	0.43	0.033	$9.0x10^{-4}$	$2.1x10^{-5}$	$1.1x10^{-5}$
B: First Wall	13,500	644	48	0.51	0.015	0.01
B: High-temp. Shield	58	4.0	0.29	$6.0x10^{-4}$	$2.8x10^{-5}$	$2.5x10^{-5}$
Ag						
A: First Wall	0.84	0.81	0.79	0.72	0.52	0.37
A: High-temp. Shield	0.017	0.016	0.016	0.015	0.011	0.0076
B: First Wall	0.67	0.64	0.62	0.57	0.41	0.30
B: High-temp. Shield	0.032	0.031	0.03	0.027	0.020	0.014
Al						
A: First Wall	0.0027	$2.3x10^{-4}$	$2.2x10^{-4}$	$2.2x10^{-4}$	$2.2x10^{-4}$	$2.2x10^{-4}$
A: High-temp. Shield	$3.9x10^{-7}$	$3.9x10^{-7}$	$3.9x10^{-7}$	$3.9x10^{-7}$	$3.9x10^{-7}$	$3.9x10^{-7}$
B: First Wall	0.0012	$2.1x10^{-4}$	$2.1x10^{-4}$	$2.1x10^{-4}$	$2.1x10^{-4}$	$2.1x10^{-4}$
B: High-temp. Shield	$1.2x10^{-7}$	$1.2x10^{-7}$	$1.2x10^{-7}$	$1.2x10^{-7}$	$1.2x10^{-7}$	$1.2x10^{-7}$
Co						
A: First Wall	81	5.9	4.2	$5.9x10^{-4}$	$1.1x10^{-7}$	$1.1x10^{-7}$
A: High-temp. Shield	7.2	0.52	0.038	$5.2x10^{-5}$	$9.1x10^{-11}$	$9.1x10^{-11}$
B: First Wall	1000	72	5.2	0.0073	$1.3x10^{-6}$	$1.3x10^{-6}$
B: High-temp. Shield	25	1.8	0.13	$1.8x10^{-4}$	$7.5x10^{-11}$	$7.5x10^{-11}$

Contact Dose Rates (mSv/h) from Main Alloying Elements of the Reduced Activation Ferritic Martensitic (RAFM) Steels and Deleterious Impurities (1 wppm) at several cooling times after lifetime Irradiation in Flibe-cooled Power Plants. All with 40%Li6 in Lithium: Case A: No Beryllium Neutron Multiplier; Case B: with Beryllium Neutron Multiplier) (continued)

Element (1 wppm)	Cooling Time (y)					
	10	30	50	100	300	500
Eu						
A: First Wall	337	91	28	2.5	0.019	$3.3x10^{-4}$
A: High-temp. Shield	39	11	3.4	0.22	$1.3x10^{-4}$	$2.2x10^{-6}$
B: First Wall	435	104	27	1.7	0.0080	$1.4x10^{-4}$
B: High-temp. Shield	92	25	7.6	0.47	$3.9x10^{-5}$	$4.1x10^{-7}$
Ho						
A: First Wall	0.36	0.36	0.35	0.34	0.30	0.27
A: High-temp. Shield	0.094	0.093	0.092	0.090	0.080	0.071
B: First Wall	0.71	0.70	0.70	0.68	0.54	0.40
B: High-temp. Shield	0.25	0.25	0.24	0.24	0.21	0.19
Mo						
A: First Wall	0.0011	0.0011	0.0011	0.0010	$9.2x10^{-4}$	$8.2x10^{-4}$
A: High-temp. Shield	$6.5x10^{-6}$	$6.5x10^{-6}$	$6.4x10^{-6}$	$6.3x10^{-6}$	$5.7x10^{-6}$	$5.2x10^{-6}$
B: First Wall	0.001	0.001	0.001	$9.6x10^{-4}$	$8.6x10^{-4}$	$7.7x10^{-4}$
B: High-temp. Shield	$3.5x10^{-6}$	$3.5x10^{-6}$	$3.5x10^{-6}$	$3.5x10^{-6}$	$3.3x10^{-6}$	$3.1x10^{-6}$
Nb						
A: First Wall	0.11	0.11	0.11	0.11	0.11	0.11
A: High-temp. Shield	0.022	0.022	0.022	0.022	0.022	0.022
B: First Wall	0.23	0.23	0.23	0.23	0.23	0.23
B: High-temp. Shield	0.058	0.058	0.058	0.058	0.058	0.057

Contact Dose Rates (µSv/h) from Reduced Activation Ferritic Martensitic (RAFM) Steels Including the Best Educated Guessed Ultimate Impurities: First Wall in a Flibe-cooled Power Plant; 4 Full-power Years of Operation with 40%Li6 in lithium (Numbers in parentheses are percentages contributing to the respective sums)

Element (Impurity Level wppm)	No Beryllium Multiplier			With Beryllium Multiplier		
	Cooling Time (y)			Cooling Time (y)		
	100	300	500	100	300	500
RAFM: Main Alloying Elements	530 (94%)	9.2 (49%)	3.4 (31%)	510 (94%)	15 (52%)	10 (49%)
Ag (<0.005)	<3.6	<2.6 (14%)	<1.9 (17%)	<2.9	<2.1 (7%)	<1.5
Al (0.5)	0.11	0.11	0.11	0.1	0.1	0.1
Co (<0.02)	<0.012	2.2x10^{-6}	2.2x10^{-6}	<0.15	2.7x10^{-5}	2.7x10^{-5}
Eu (0.01)	25	0.19	0.0022	17	0.08	0.0014
Ho (0.01)	3.4	3.4 (18%)	2.7 (25%)	6.8	6.0 (21%)	5.4 (25%)
Mo (1)	1	0.92	0.82	0.96	0.86	0.77 (3.5%)
Nb (<0.02)	<2.3	<2.2 (12%)	<2.2 (20%)	<4.6	<4.6 (16%)	<4.5 (20%)
Sum	<570	<19	<11	<540	<29	<22

Contact Dose Rates (µSv/h) from Reduced Activation Ferritic Martensitic (RAFM) Steels Including the Best Educated Guessed Ultimate Impurities: High-temperature Shield in a Flibe-cooled Power Plant; 40 Full-power Years of Operation with 40%Li6 in lithium (Numbers in parentheses are percentages contributing to the respective sums)

Element (Impurity Level wppm)	No Beryllium Multiplier			With Beryllium Multiplier		
	Cooling Time (y)			Cooling Time (y)		
	100	300	500	100	300	500
RAFM: Main Alloying Elements	0.9 (20%)	0.021 (1.6%)	0.011 (0.9%)	0.69 (7.6%)	0.028 (0.8%)	0.025 (0.8%)
Ag (<0.005)	<0.073 (1.6%)	<0.053 (4%)	<0.038 (3%)	<0.14	<0.098	<0.07
Al (0.5)	0.0002	0.0002	0.0002	0.00006	0.00006	0.00006
Co (<0.02)	<0.001	1.9x10^{-9}	1.9x10^{-9}	<0.0036	1.5x10^{-9}	1.5x10^{-9}
Eu (0.01)	2.2 (49%)	0.0013	2.2x10^{-5}	4.7 (52%)	0.0004	4.6x10^{-6}
Ho (0.01)	0.9 (20%)	0.8 (61%)	0.71 (59%)	2.4 (26%)	2.1 (62%)	1.9 (61%)
Mo (1)	0.0065	0.0057	0.0052	0.0035	0.0033	0.0031
Nb (<0.02)	<0.43 (10%)	<0.43 (33%)	<0.43 (36%)	<1.2 (13%)	<1.2 (35%)	<1.1 (35%)
Sum	<4.5	<1.3	<1.2	<9.1	<3.4	<3.1

Conclusions (1)

- One of the mechanisms to lower the Induced long-term radioactivity in fusion power plants is by enriching lithium-6 in lithium to suppress the neutron fluxes.
- The other is using the reduced activation reactor materials, as generally accepted and is a basis for developing the reduced activation fusion material.
- Waiting for a longer cooling time than normally considered, such as waiting from 50 years to 100 years or longer after discharge, before taking action to handle the discharged reactor components is also sometimes effective in reducing the contact dose rate from the discharged components, because it allows the medium-lived radionuclides generated in the reactor materials to decay away.

Conclusions (2)

- Within the constraints of the nuclear data used and calculations performed, we may conclude:
  - The RAFM steel from the first wall component of flibe-cooled power plants with 40% lithium-6 enrichment in lithium was found to be within 1 mSv/h in contact dose rate after 100 years of cooling. The contact dose rate could be further reduced by longer cooling. It drops to 30 and 20 μSv/h or less, respectively, when the cooling times are 300 and 500 years after discharge.
  - The RAFM steel discharged from the high-temperature shield component would be allowed for hands-on recycling after 100 years of cooling. The contact dose rate at that time after discharge and thereafter is 10 μSv/h or less.

Appendix 3.2.

FENDL-2 Benchmarking – Overview of Recent Results

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FENDL-2 Benchmarking - Overview of Recent Results -

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Forschungszentrum Karlsruhe

FENDL-2 Consultants Meeting, IAEA, Vienna, November 9-12, 2003

Overview

Benchmark analyses conducted 1999-2003 in the frame of European Fusion File (EFF) Programme

Nuclide	Data evaluations	Experiments analysed
Be-9	EFF-3.0, -3.03, -3.05, EFF-1, FENDL-1,-2	FNS time-of-flight (TOF) experiment, KANT transmission experiment
Fe-56	EFF-3.0, -3.1, FENDL-1,-2	FNS TOF-experiment, FNG bulk shield experiment, IPPE Obninsk transmission experiment
Ni-58, -60	EFF-3.0, FENDL-1,-2	IPPE Obninsk and OKTAVIAN transmission experiment
Cr-52	EFF-3.01, -3.03, FENDL-1,-2	OKTAVIAN transmission experiment
Si-28	EFF-3.0, FENDL-1, -2, EFF-2.4	OKTAVIAN transmission experiment, FNG SiC transport experiment
C-12	EFF-2.3, FENDL-1, -2	FNG SiC transport experiment, FNS TOF-experiment on carbon
W	EFF-2.4, FENDL-1, -2	FNG tungsten experiment

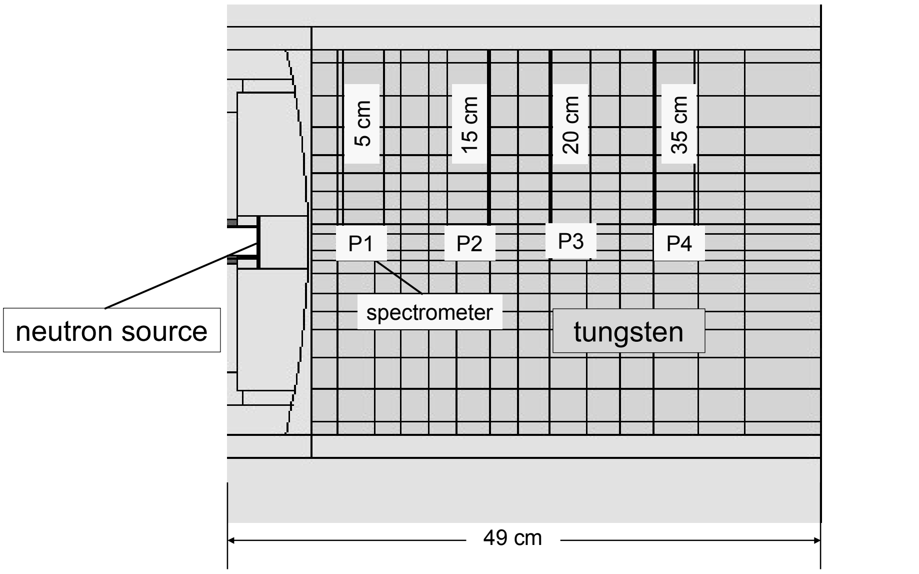
General Benchmark Methodology

- Process ENDF data file (NJOY/ACER)
- Compare processed cross-section data
  - FENDL-1, -2, EFF-2.4, EFF-3.0
  - $\sigma_{tot}(E)$, $\sigma_{el}(E)$, $\sigma_{abs}(E)$, $\sigma_{nem}(14.1\ \mathrm{MeV}, E')$
- Benchmark calculations (MCNP + ACE data)
  - Neutron & photon flux spectra
  - Compare data evaluations for calculated spectra
  - Compared to experimental results (spectra, flux integrals)
- Sensitivity analyses and uncertainty assessments
  - Monte Carlo method for point detector (R. Perels code MCSEN)
  - Identification of important reactions and energy ranges

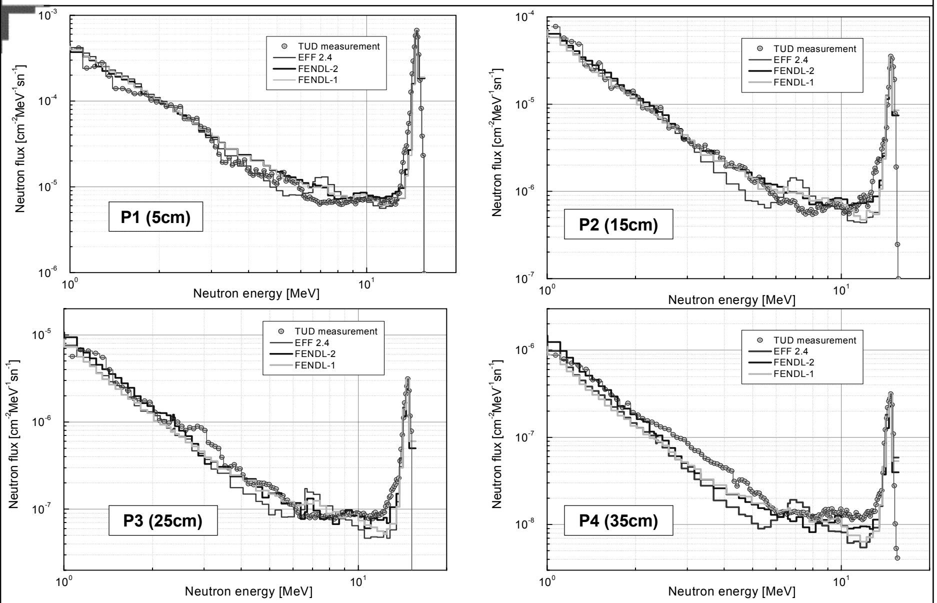
W benchmark experiment

- TUD Experiment on W at FNG
  - Measurement of neutron & photon flux spectra in W assembly using a NE 213 liquid-scintillation spectrometer
  - Spectra measured in four positions in W assembly
- Monte Carlo Transport Analysis
  - MCNP4C calculations for 3D model of W assembly & rack, spectrometer, neutron generator and experimental hall (FNG)
  - W data: EFF-2.4 (=JENDL-3.0), FENDL-1(=ENDF/B-VI.0), FENDL-2(=JENDL-FF)
  - Comparison of flux spectra & C/E-data
- Cross-section data checks
- Monte Carlo Sensitivity Analysis
  - Calculation of point detector sensitivities using MCSEN at four positions

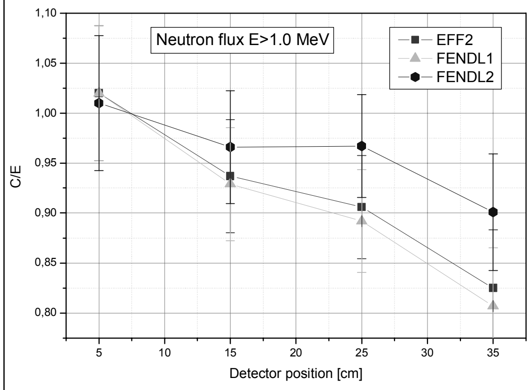
W assembly model

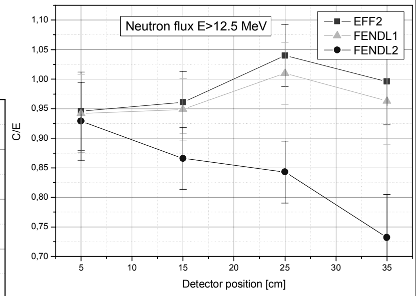


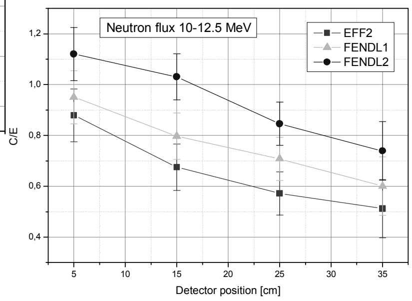
W transmission experiment (FNG)



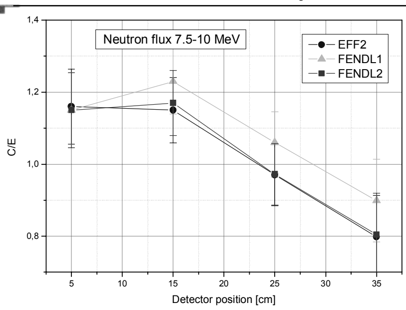
C/E comparison of neutron flux integrals

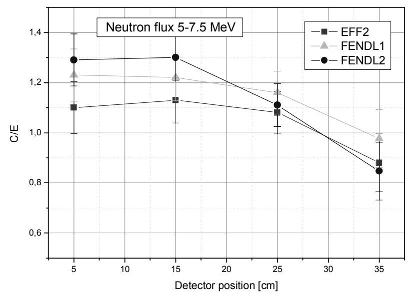


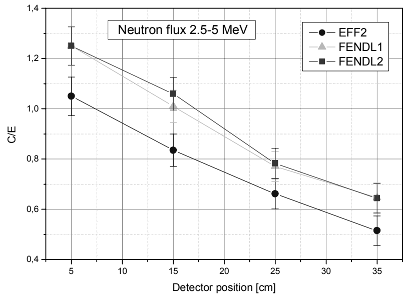


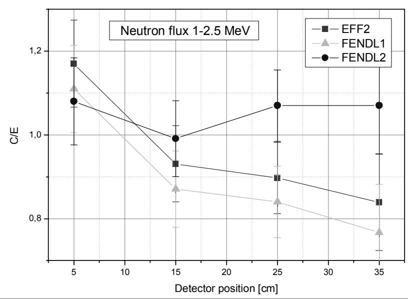


C/E comparison of neutron flux integrals

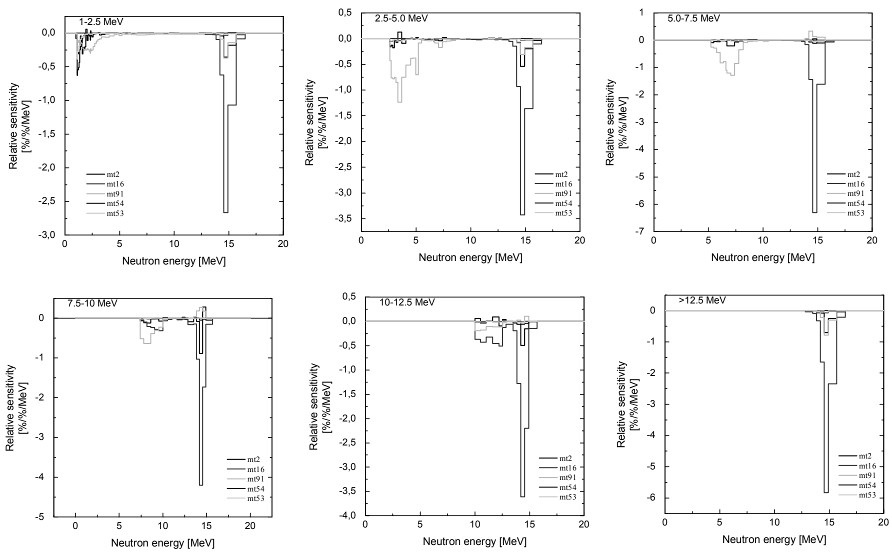








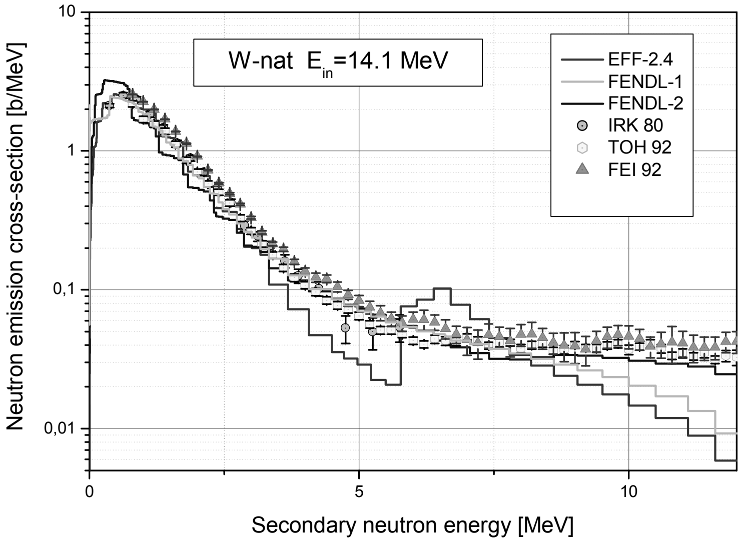
Sensitivity profiles of neutron fluxes at P4



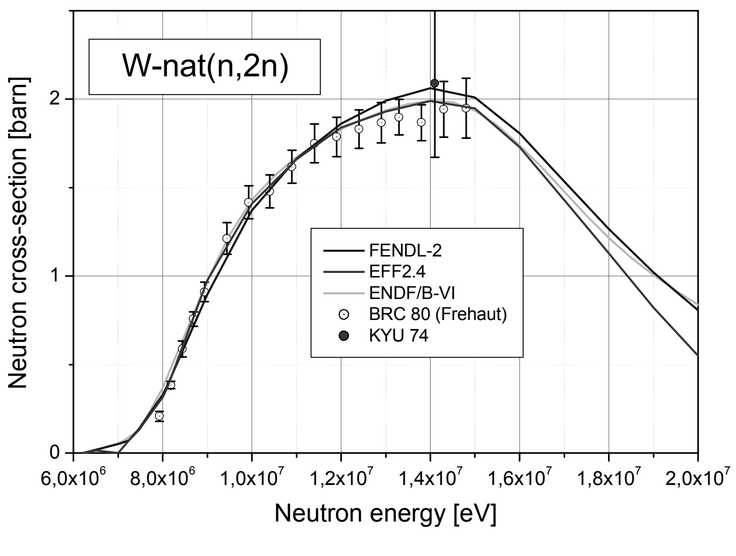
Integrated neutron flux sensitivities at P4

MT	1-2.5 MeV	2.5-5 MeV	5-7.5 MeV	7.5-10 MeV	10-12.5	>12.5 MeV
2	-0,572	-0,426	-0,207	-0,527	-0,217	-0,431
16	-1,724	-2,307	-3,583	-3,419	-3,845	-3,881
51-52	-0,049	-0,036	-0,034	-0,023	-0,019	-0,017
53	-0,144	-0,071	-0,053	-0,029	-0,036	-0,027
54	-0,129	-0,065	-0,065	-0,054	-0,039	-0,041
55	-0,117	-0,059	-0,065	-0,044	-0,040	-0,032
56-63	-0,201	-0,055	-0,029	-0,020	-0,013	-0,022
64-73	-0,228	-0,032	-0,005	-0,003	-0,001	-0,002
91	-0,770	-2,018	-1,841	-0,931	-0,318	-0,522
102+103+107	-0,051	-0,017	-0,008	-0,006	-0,007	-0,008

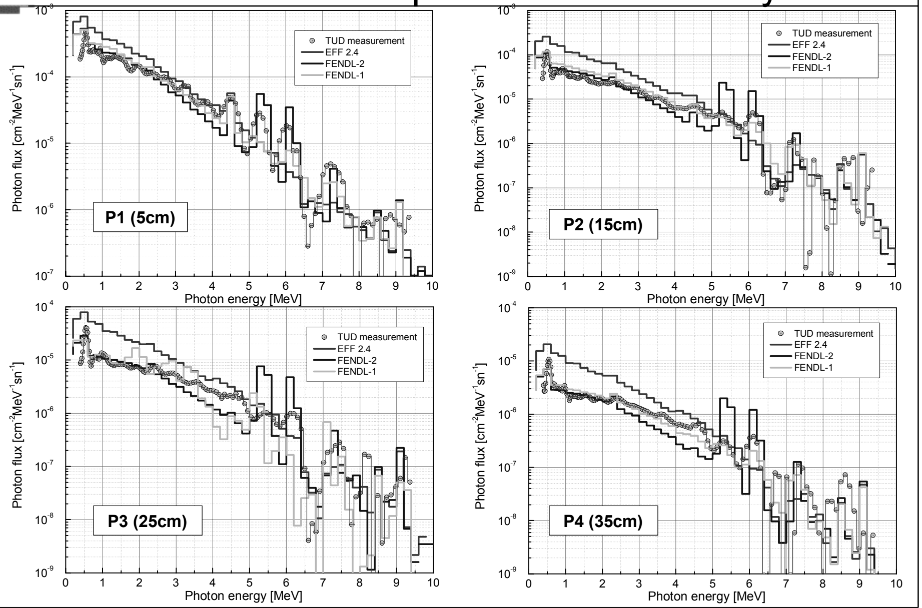
^{nat}W neutron emission cross-section



$^{nat}W(n,2n)$ cross-section



Photon flux spectra in W assembly



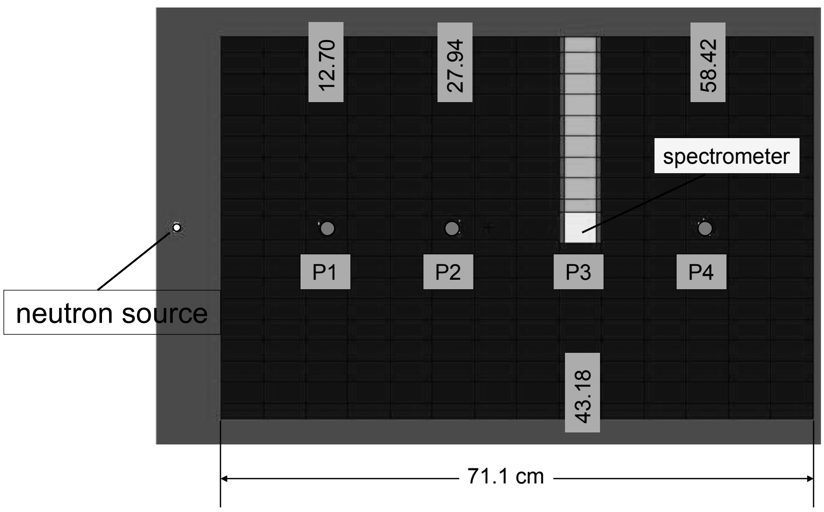
Conclusions of W benchmarking

- Fast (E>1 MeV) neutron flux in W assembly:
  - Reproduced within 20% by calculations (EFF2.4, FENDL-1,-2)
  - FENDL-2 shows best overall agreement, although flux above 12.5 MeV is underestimated by up to 30% (P4)
    - ⇒*Indicates too high (n,2n)-cross-section at 14 MeV*
- Gamma flux (E> 0.4 MeV) in W assembly:
  - Reproduced within 20% uncertainty by FENDL-2 calculations
  - Gross overestimation by EFF-2.4 (factor 2-3)
- Sensitivity analysis shows dominant effect of (n,2n) cross-section on neutron spectra in W assembly
- Need for new evaluation of W data for EFF-3
  - ⇒ Secondary energy distribution
  - ⇒ Gamma production cross-section
  - ⇒ Isotope data, MF=6 data

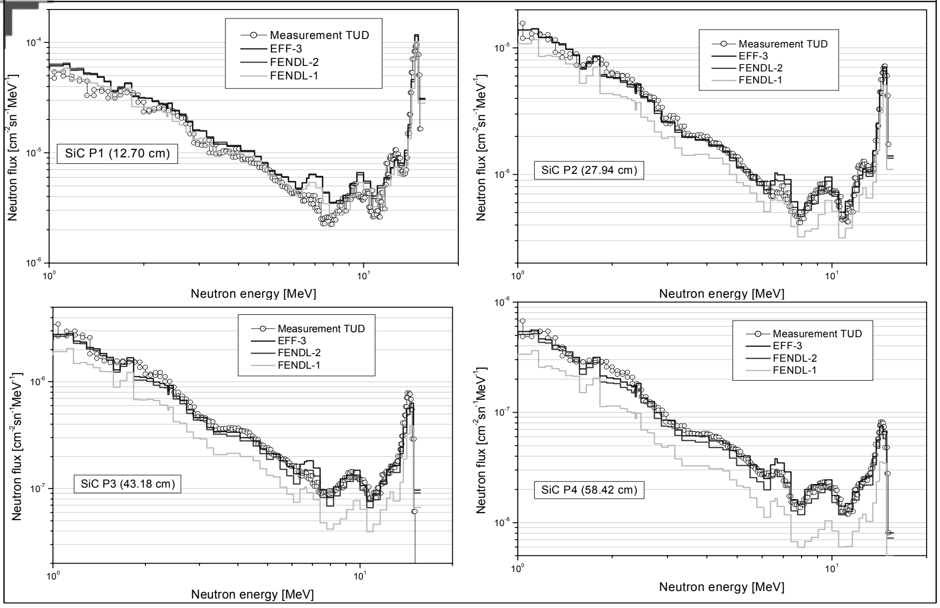
SiC benchmark experiment

- TUD Experiment on SiC at FNG
  - Measurement of neutron & photon flux spectra in SiC assembly using a NE 213 liquid-scintillation spectrometer *(K. Seidel, EFF-DOC-822)*
  - Spectra measured in four positions in SiC assembly
- Monte Carlo Transport Analysis
  - MCNP4C calculations for 3D model of Sic assembly & rack. spectrometer. neutron generator and experimental hall (FNG)
  - Si data: EFF-3.0, -2.4. FENDL-1, -2; C-data: EFF-2.4, FENDL-2
  - Comparison of flux spectra & C/E-data
- Monte Carlo Sensitivity & Uncertainty Analysis
  - Calculation of point detector sensitivities at four positions
  - Calculation of uncertainties using EFF-3 Si and EFF-2.4 C co-variances

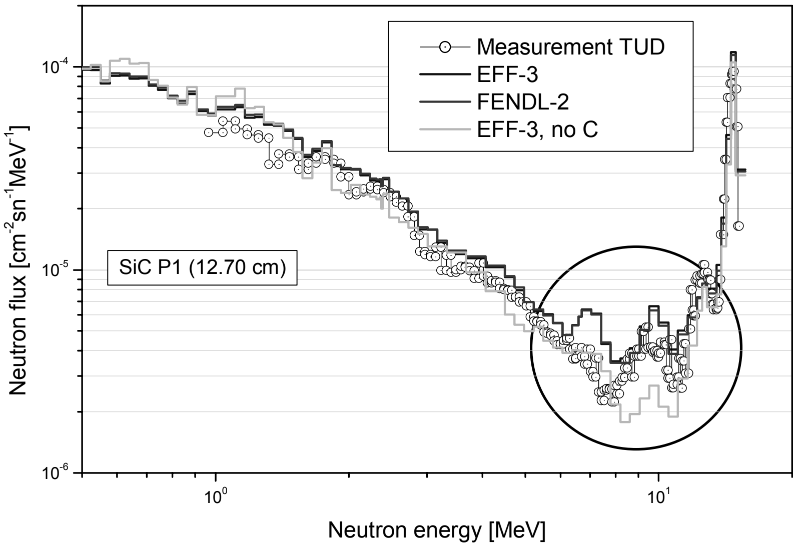
SiC assembly model



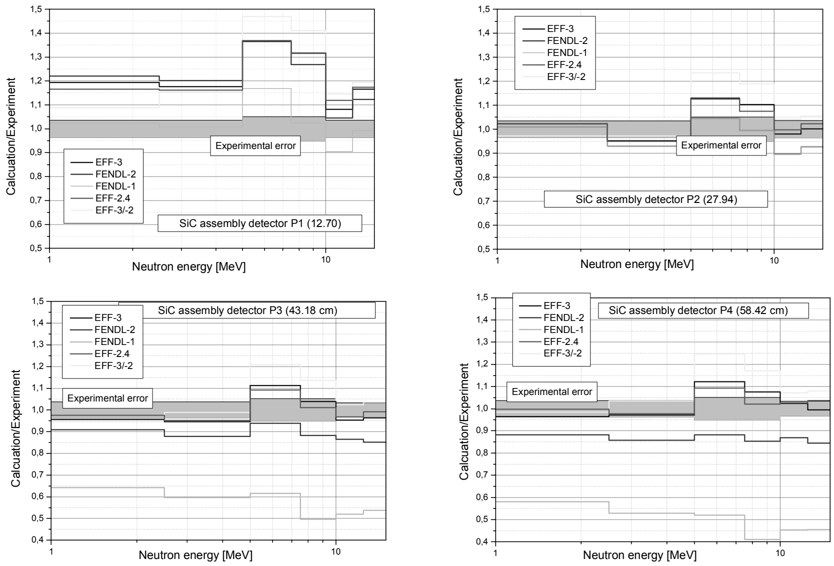
Neutron flux spectra in SiC assembly (FNG)



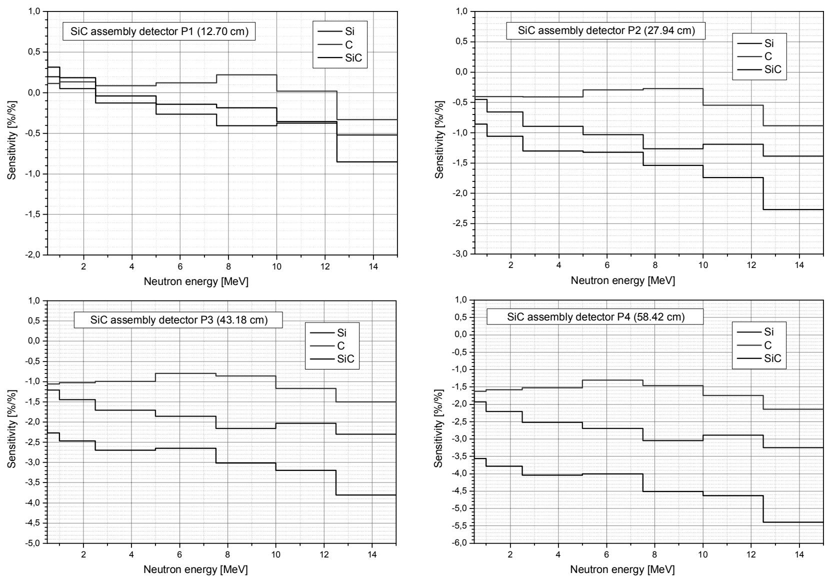
Effect of C on neutron spectrum in SiC at P1



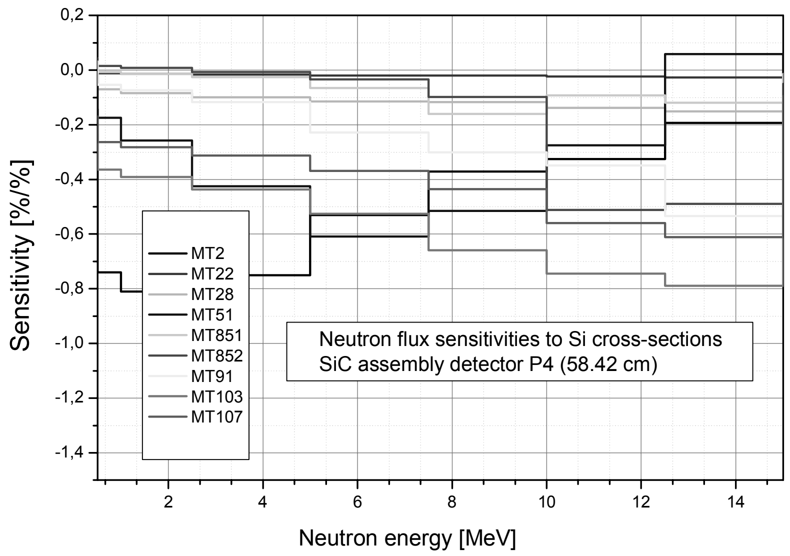
C/E comparison of neutron fluxes in SiC



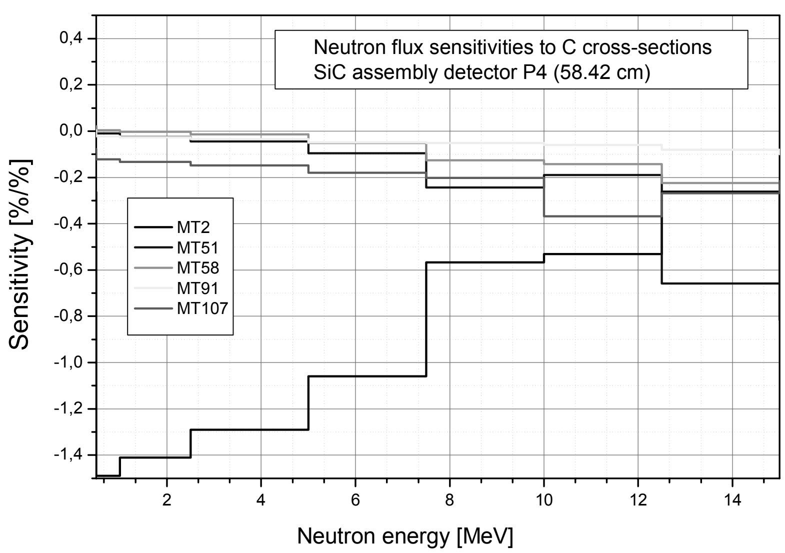
Sensitivities of neutron fluxes to Si and C data



Sensitivities to specific Si cross-sections at P4



Sensitivities to specific C cross-sections at P4



Uncertainties of neutron fluxes at P4

		Calculation uncertainties [%]					Experiment		Ex. + C.
E [MeV]	flux	MC	SiC	Si	C	Calc.	C/E(*)	[%]	Un. [%]
E<0.1	8.67E-06	0.96	1.62	1.28	0.99	1.88			
0.5	6.65E-07	0.6	2.87	1.96	2.09	2.93			
1	3.37E-07	0.6	3.29	2.24	2.41	3.34			
2.5	4.19E-07	0.32	4.02	2.76	2.92	4.03	0.99	3.70	5.48
5	1.74E-07	0.41	5.37	3.88	3.71	5.39	1.03	3.70	6.54
7.5	8.86E-08	0.33	6.55	4.69	4.57	6.56	1.25	5.07	8.29
10	5.34E-08	0.39	8.44	5.71	6.21	8.45	1.17	5.07	9.85
12.5	4.55E-08	0.34	12.2	8.09	9.13	12.20	1.07	3.33	12.65
E >12.5	1.11E-07	0.14	12.29	7.54	9.7	12.29	1.08	3.33	12.73
total	1.06E-05	0.79	1.62	1.27	1.01	1.80			

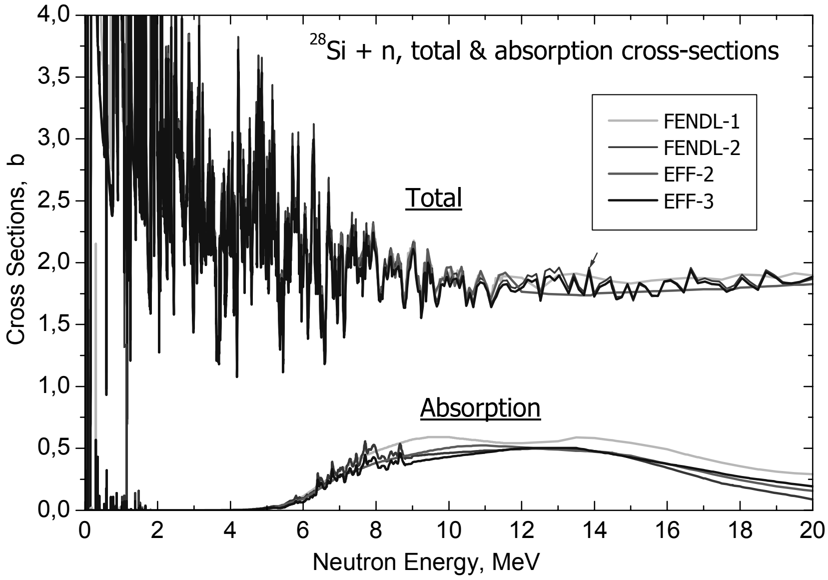
(*)EFF-3 Si and C EFF-2.4 data

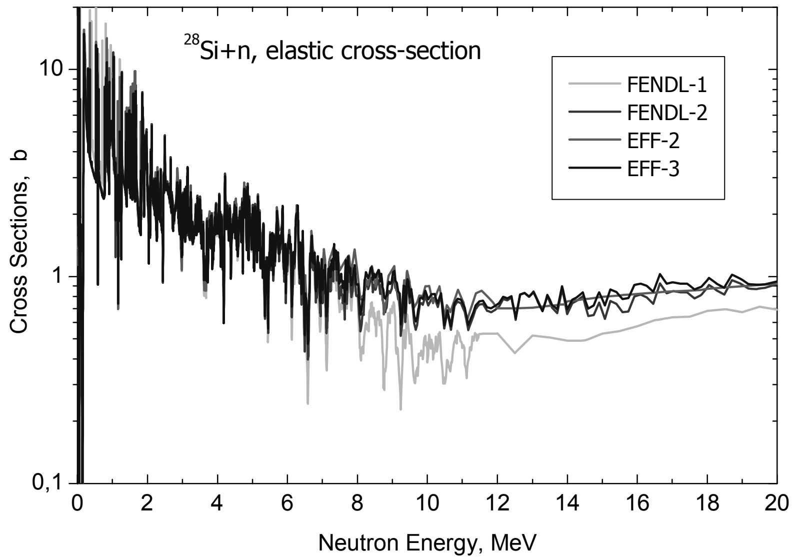
Conclusions of SiC benchmarking

- SiC transport experiment shows good quality of EFF Si data
  - EFF-3 and EFF-2.4 Si data reproduce well fast (E>1 MeV) neutron flux when used with FENDL-2 C (JENDL-FF) data.
- Si-data: significant improvement over FENDL-1 & FENDL-2
  - FENDL-1: Gross underestimation of fast neutron flux (Si data)
  - FENDL-2 tends to underestimate fast flux at deep penetrations
- C-data: FENDL-2 superior to EFF-2.4
  - EFF-2.4 C data (ENDF/B-VI): to be re-considered (inelastic scattering)
- Sensitivity to Si and C data
  - Fast flux more sensitive to Si than to C cross-sections
- Calculated Uncertainties
  - Dominated by inelastic scattering (Si, C) and absorption (Si) co-variances
  - With few exceptions calculated uncertainties larger than experimental ones
  - Total (experimental and calculated) uncertainty is in the range 5 to 12%

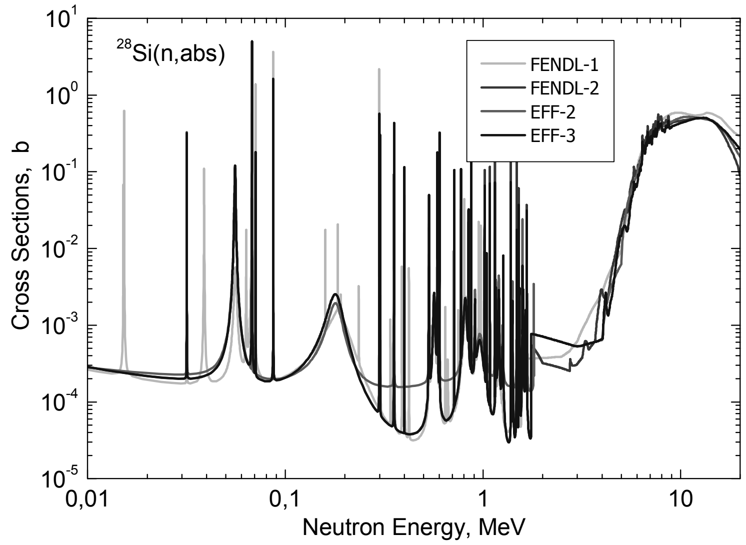
Si benchmarking

- New EFF-3 Si-28 data evaluation
  - H. Vonach, S. Tagesen, IRK Vienna & A. Trkov IAEA/NDS
  - Based largely on ENDF/B-VI.5 (MF=2,4,6,12,14)
  - MF=3 and MT=33 new
  - ENDF & ACE data files (November 2001)
- Data tests performed
  - Cross-section comparisons & checks (ACE data)
    - FENDL-1 (BROND), FENDL-2 (ENDF/B-VI), EFF-2.4, EFF-3, EXFOR data
    - $\sigma_{tot}(E)$, $\sigma_{el}(E)$, $\sigma_{abs}(E)$, $\sigma_{nem}(14.1\ \mathrm{MeV}, E')$
  - MCNP benchmark calculations (ACE data)
    - OKTAVIAN Si spherical assembly (Ø 60 cm, 20 cm shell thickness)
    - Comparison and C/E data Leakage spectra
    - Comparison of data evaluations & experimental results

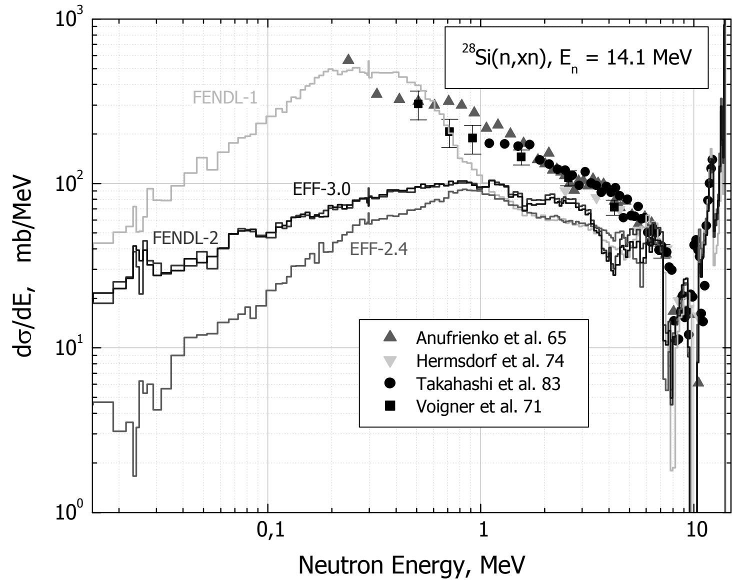




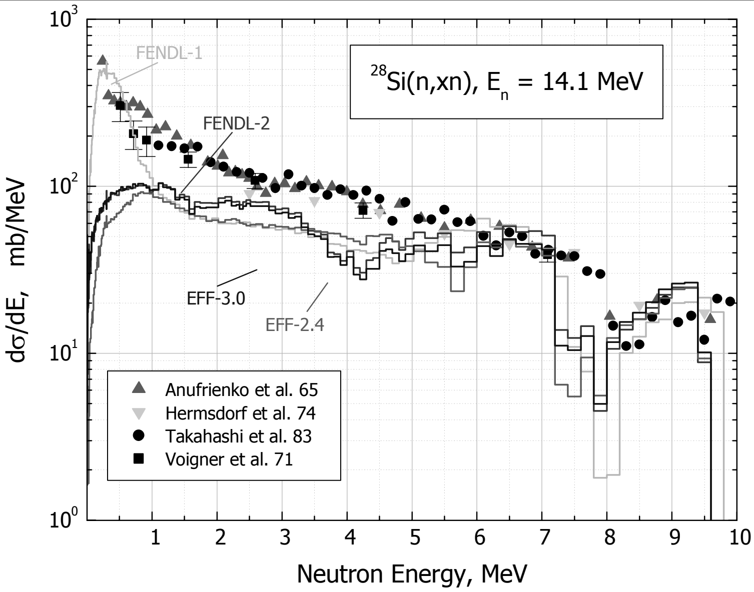
^{28}Si+n, elastic cross-section
FENDL-1
FENDL-2
EFF-2
EFF-3
Cross Sections, b
10
1
0,1
0
2
4
6
8
10
12
14
16
18
20
Neutron Energy, MeV



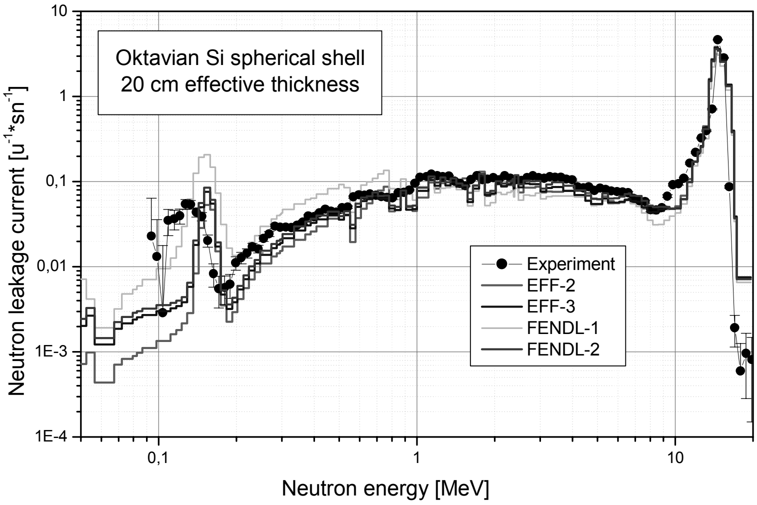
^{28}Si(n,abs)
FENDL-1
FENDL-2
EFF-2
EFF-3
Cross Sections, b
10^{1}
10^{0}
10^{-1}
10^{-2}
10^{-3}
10^{-4}
10^{-5}
0,01
0,1
1
10
Neutron Energy, MeV



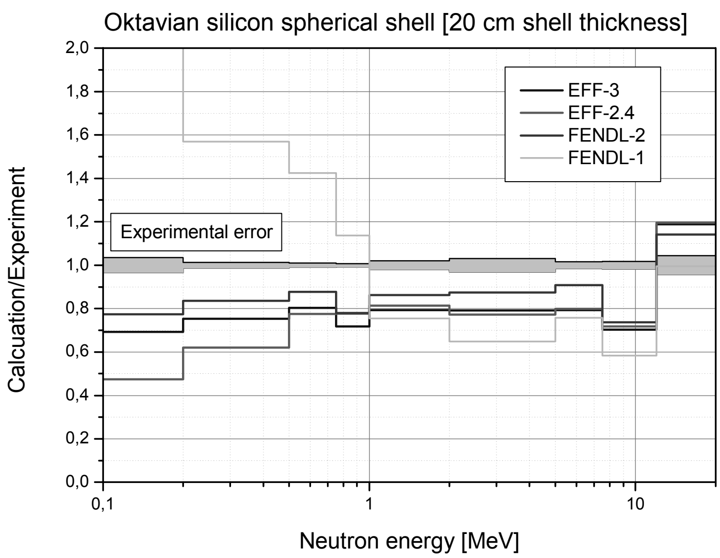
^{28}Si(n,xn), E_n = 14.1 MeV
FENDL-1
EFF-3.0
FENDL-2
EFF-2.4
Anufrienko et al. 65
Hermsdorf et al. 74
Takahashi et al. 83
Voigner et al. 71
dσ/dE, mb/MeV
Neutron Energy, MeV



^{28}Si(n,xn), E_n = 14.1 MeV
FENDL-1
FENDL-2
EFF-3.0
EFF-2.4
Anufrienko et al. 65
Hermsdorf et al. 74
Takahashi et al. 83
Voigner et al. 71
dσ/dE, mb/MeV
Neutron Energy, MeV



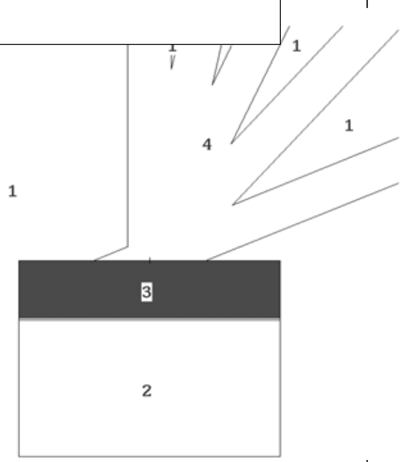
Calculation/Experiment Comparison



FNS TOF experiments on cylindrical C slabs

- Cylindrical C slab: radius 314 mm, thicknesses 50.6, 202.4, 404.8 mm
- TOF at 0, 12.2, 24.9, 41.8, 66.8 degree

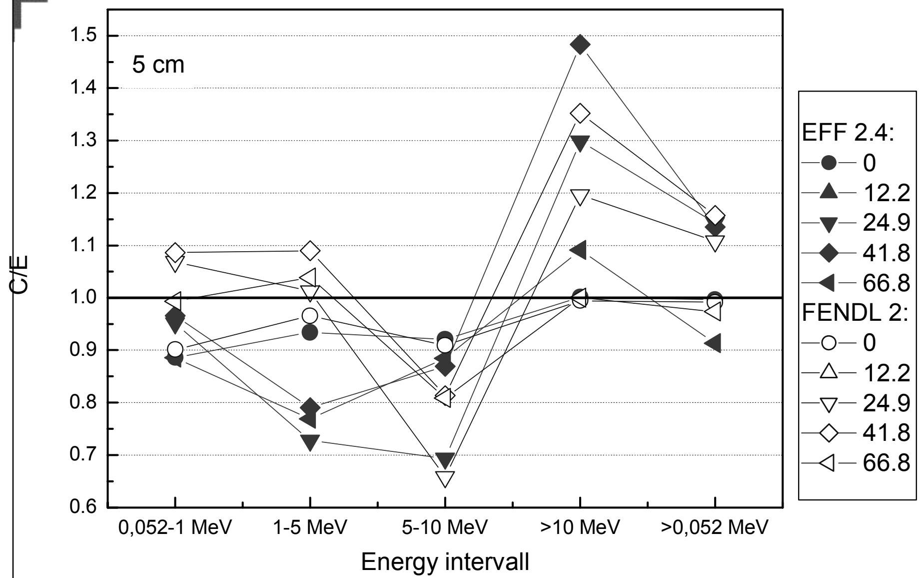
C data evaluations:
- EFF2.4: C-nat
  - ENDF/B-VI,Revised 1990
- FENDL2: C-12
  - JENDL-FF, revised 1993/1996 (FENDL2 modifications)



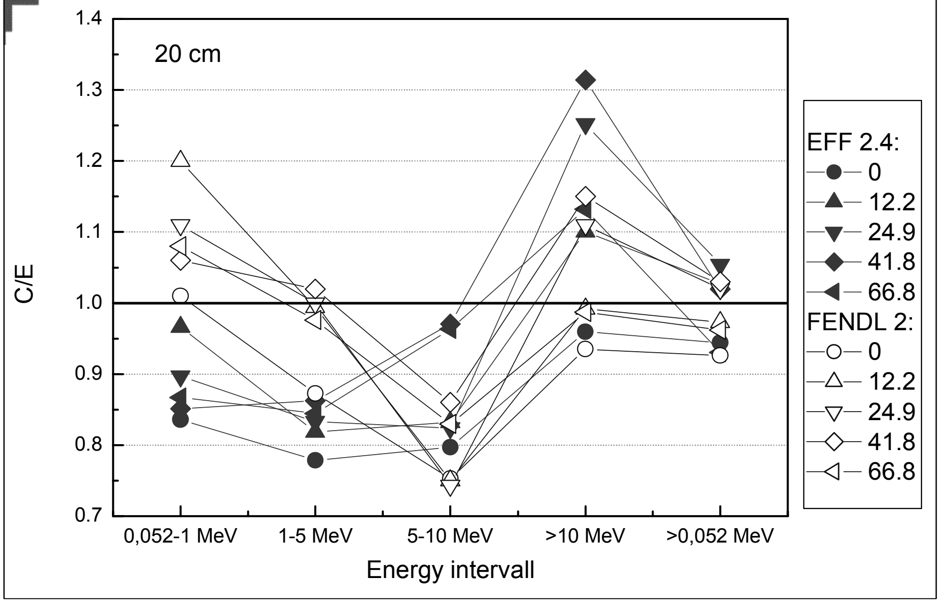
C/E of neutron flux integrals

		0 deg		12.2 deg		24.9 deg		41.8 deg		66.8 deg	
	E [MeV]	EFF2.4	FENDL 2	EFF2.4	FENDL 2	EFF2.4	FENDL 2	EFF2.4	FENDL 2	EFF2.4	FENDL 2
5 cm	0.052-1	0.89	0.90			0.95	1.07	0.97	1.09	0.89	0.99
	1-5	0.93	0.97			0.73	1.01	0.79	1.09	0.77	1.04
	5-10	0.92	0.91			0.69	0.66	0.87	0.81	0.88	0.81
	>10	1.00	0.99			1.30	1.20	1.48	1.35	1.09	1.00
	>0.052	1.00	0.99			1.14	1.11	1.13	1.16	0.91	0.97
20 cm	0.052-1	0.84	1.01	0.97	1.20	0.90	1.11	0.85	1.06	0.87	1.08
	1-5	0.78	0.87	0.82	0.99	0.83	1.00	0.86	1.02	0.84	0.98
	5-10	0.80	0.75	0.83	0.75	0.82	0.74	0.97	0.86	0.96	0.83
	>10	0.96	0.93	1.10	0.99	1.25	1.11	1.31	1.15	1.13	0.99
	>0.052	0.94	0.93	1.03	0.97	1.05	1.02	1.02	1.03	0.93	0.96
40 cm	0.052-1	0.88	0.99	0.88	0.99	0.98	1.10	0.73	0.82	1.10	1.23
	1-5	0.81	0.85	0.91	0.96	0.91	0.95	0.86	0.88	0.91	0.91
	5-10	0.84	0.76	0.92	0.80	0.95	0.82	1.03	0.87	1.06	0.87
	>10	0.99	0.94	1.15	1.00	1.18	1.01	1.22	1.04	0.99	0.84
	>0.052	0.96	0.92	1.04	0.95	1.03	0.96	0.95	0.90	0.98	0.94

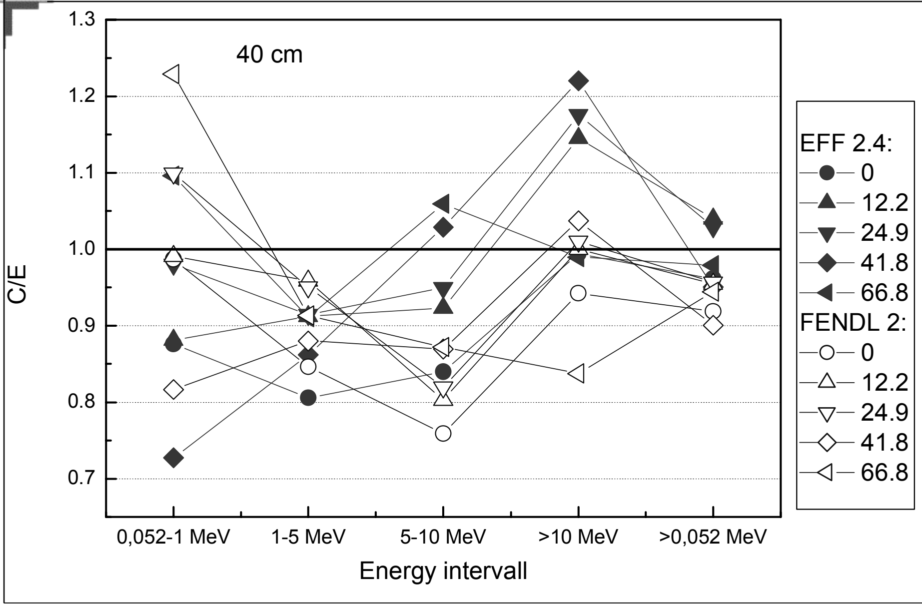
C/E of neutron flux integrals



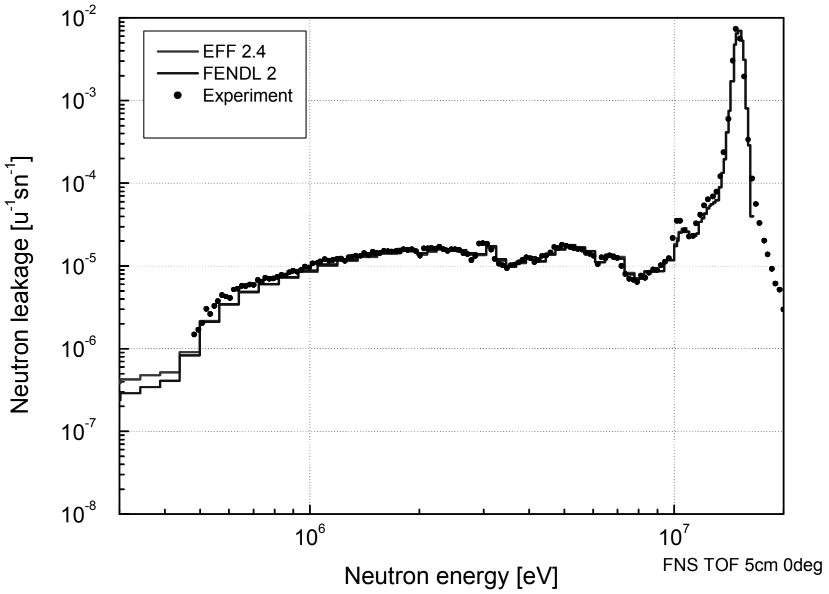
C/E of neutron flux integrals



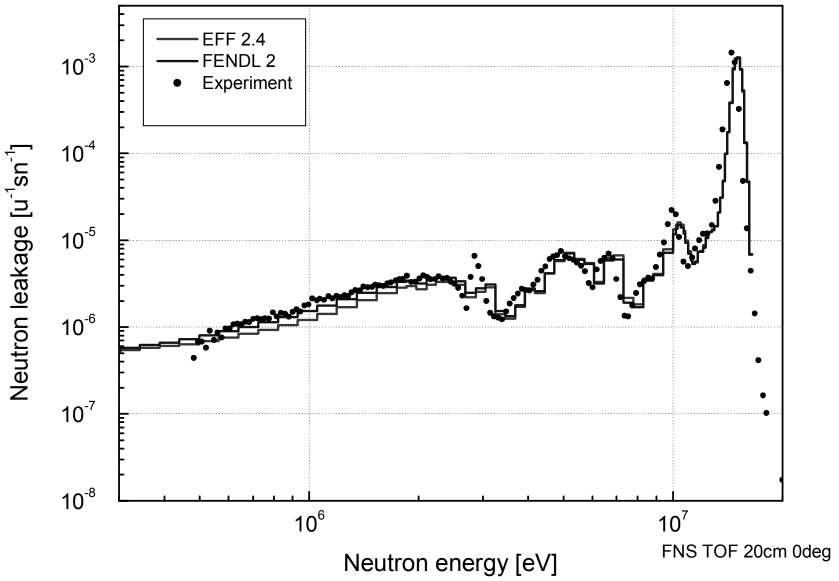
C/E of neutron flux integrals



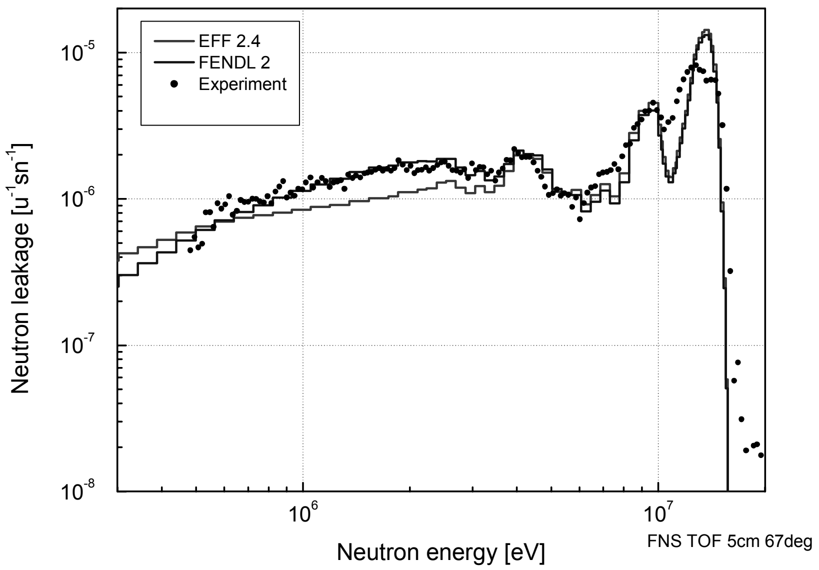
Angular neutron leakage spectra (FNS C, 5 cm 0°)



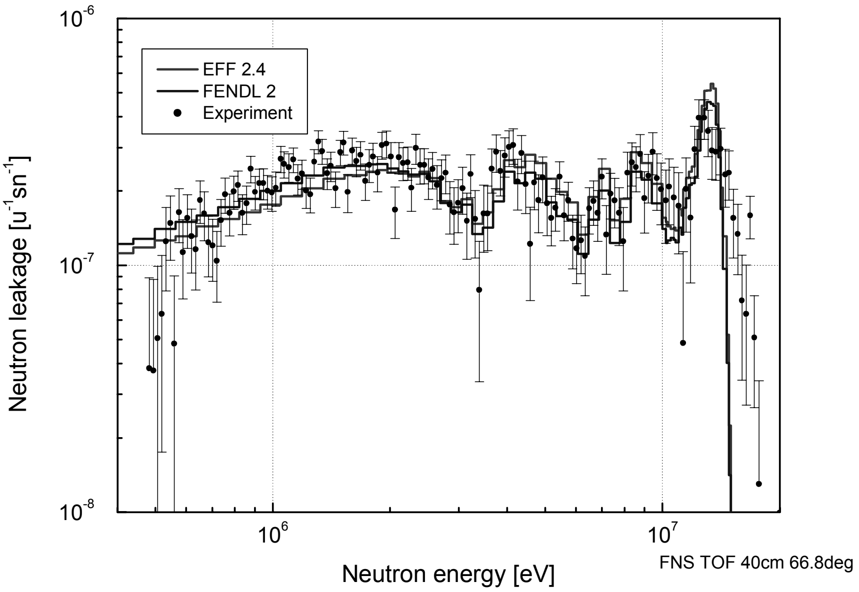
Angular neutron leakage spectra (FNS C, 20 cm 0°)



Angular neutron leakage spectra (FNS C, 40 cm 0°)



Angular neutron leakage spectra (FNS C, 40 cm 66.8°)



Conclusion of C benchmarking

- Comparison to results of SiC-Benchmark
  - General (large) overestimation at 5…10 MeV cannot be explained by deficiencies in C data
  - FENDL2 superior to EFF2.4 cannot be supported by benchmark analysis of C FNS/TOF experiment

Cr-52 benchmarking

- Process ENDF data file (NJOY/ACER)
- Compare processed cross-section data
  - FENDL-2 (ENDF/B-VI), EFF-2.4, EFF-3.01, 3.03 (new)
  - $\sigma_{tot}(E)$, $\sigma_{el}(E)$, $\sigma_{abs}(E)$, $\sigma_{nem}(E_{in}=14.1\text{ MeV}, E')$
- Benchmark calculations (MCNP + ACE data)
  - OKTAVIAN Cr sphere assembly (Ø 40 cm, 9.58 cm eff. thickness)
  - Leakage spectra
    - Compare data evaluations
    - Compare to experimental results

Revisions Cr-52 EFF-3.01 to 3.03

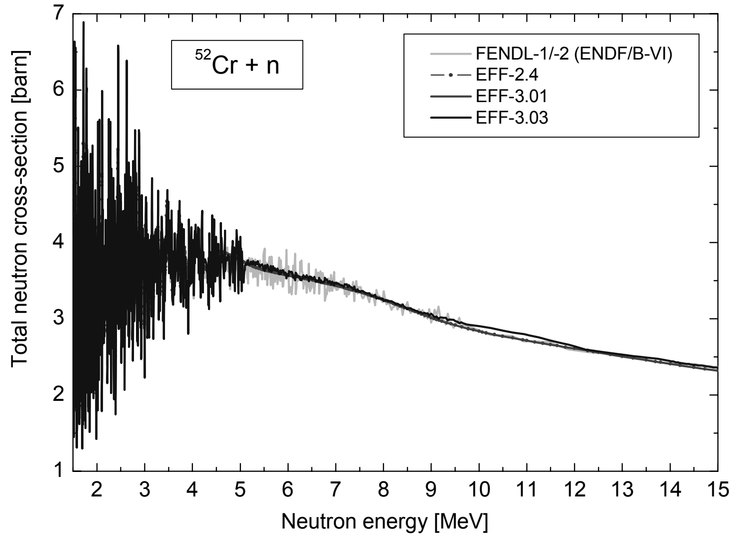
9/2001: New data added for MT 1, 2, 16 and 51, eval. threshold increased to 1.2 MeV to be consistent with new MF 2 evaluation by Olivier Bouland (2001)

11/2001: The fine structure was superimposed on the total cross section by A. Trkov. Contribution from minor isotopes was subtracted adopting ENDF/B-VI.7 cross sections.

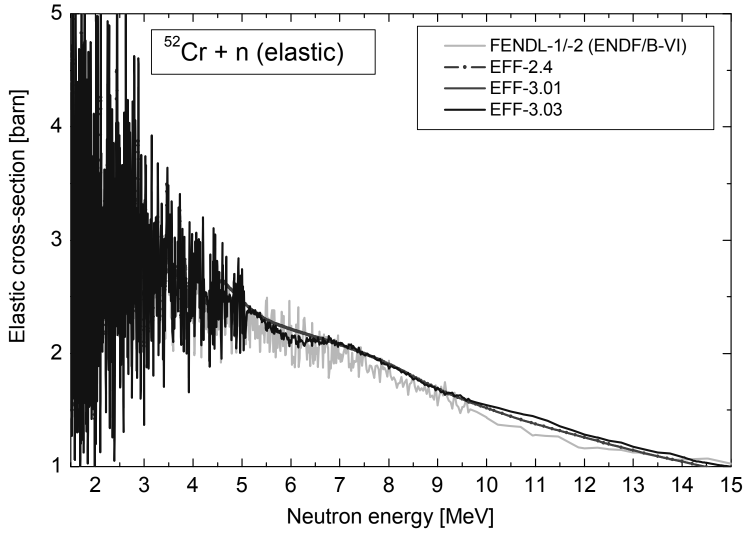
02/2002: (n,t) and (n,He3) cross sections from IEAF-2001 file were inserted together with corresponding covariance files evaluated by S. Tagesen and H. Vonach.

02/2003: MF2 and MF4MT2 replaced by new evaluation of O. Bouland; MF 6, 12, 14 and 15 replaced by ENDF-B 6.2

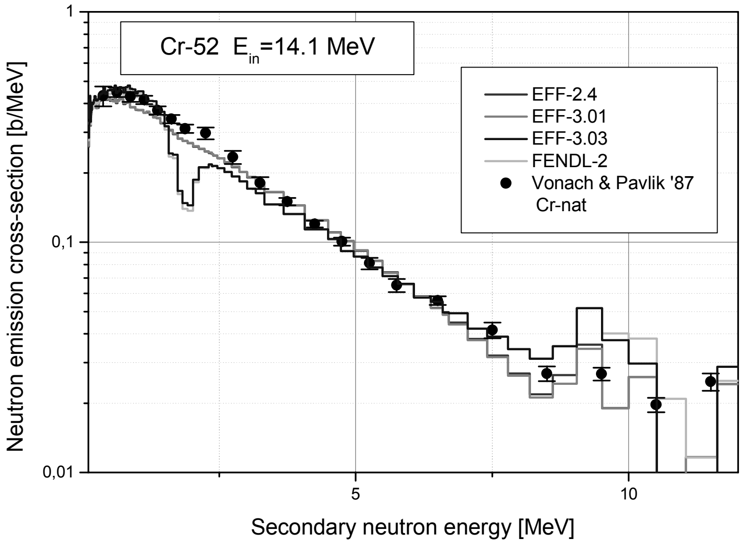
Total Neutron Cross-Sections



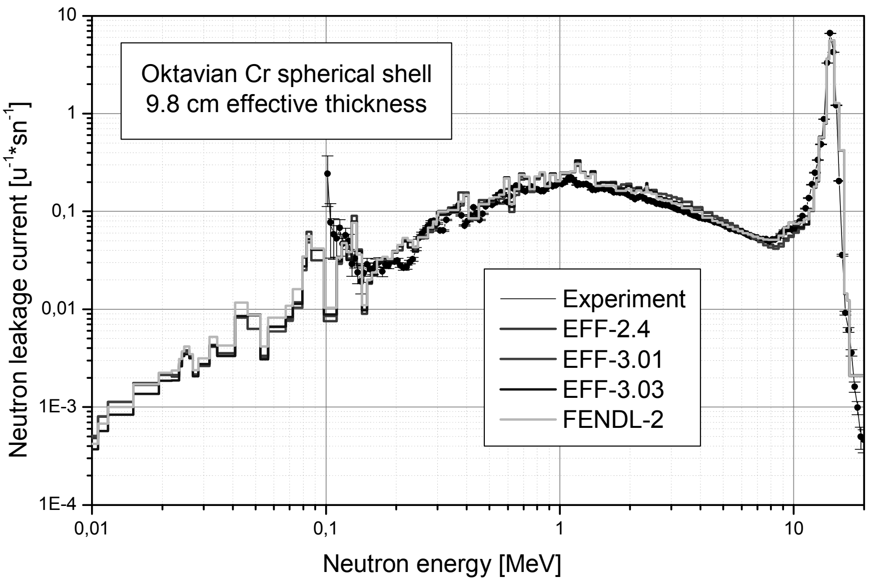
Elastic Neutron Scattering Cross-Sections



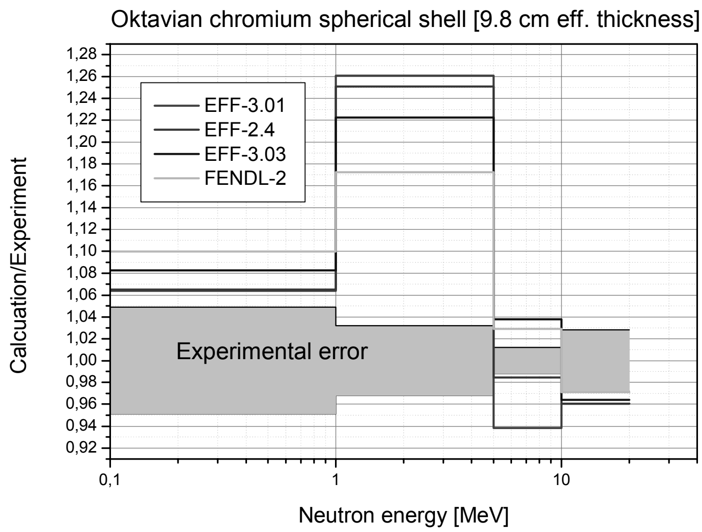
Neutron Emission Cross-Sections



Neutron Leakage Spectra



Neutron Flux Integrals: C/E Comparison



Cr-52 benchmark conclusions

- EFF- 3.01 vs -2.4
  - New resonance parameter evaluation up to 1.2 MeV
- EFF-3.03 vs. 3.01
  - Energy range above 1.2 MeV updated
  - Co-variance data included
- EFF-3.03 in agreement with FENDL-2 (ENDF/B-VI)
- Satisfactory agreement with OKTAVIAN experiment
- Neutron emission cross-section needs to be corrected in EFF-3.03 and FENDL-2 (MF=6 data)

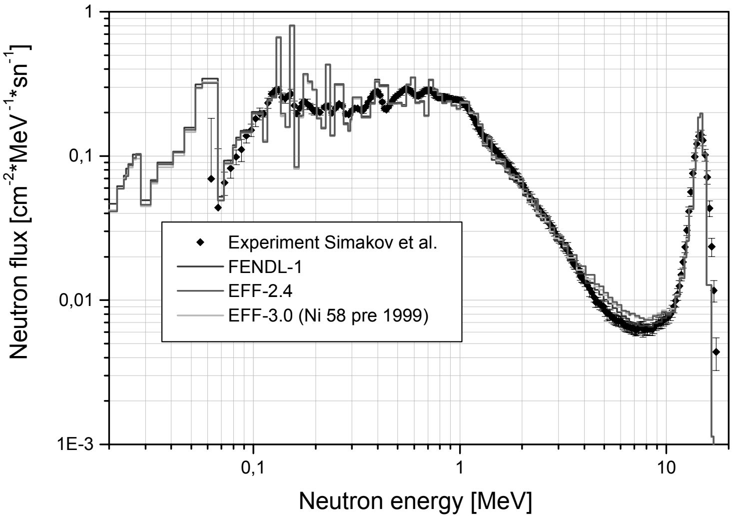
Ni –58, -60 benchmarking

IPPE transmission experiment on nickel spherical shell (r=4.5/12.0 cm)

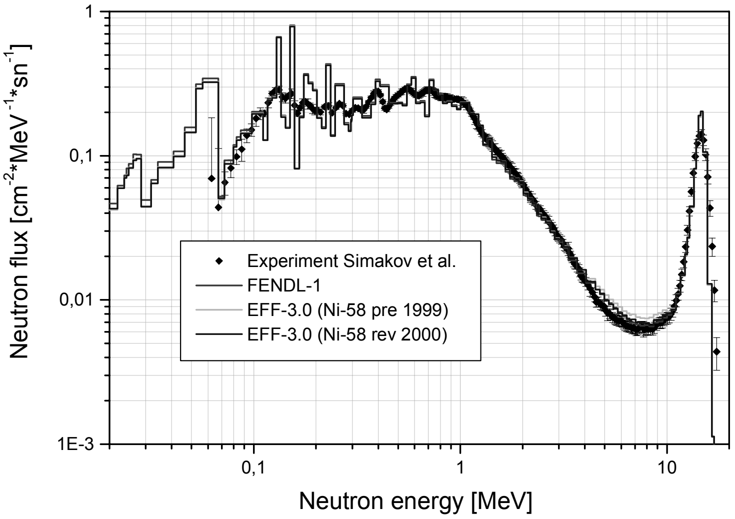
- Revised EFF-3 Ni-58 evaluation
  - Inelastic cross-section around 14 MeV reduced as a result of previous benchmark & sensitivity analyses
- New EFF-3 Ni-60 evaluation
  - Inclusion in benchmark analyses
- Revised Monte Carlo sensitivity & uncertainty calculations
  - Removal of minor bugs in MCsen affecting previous sensitivity results

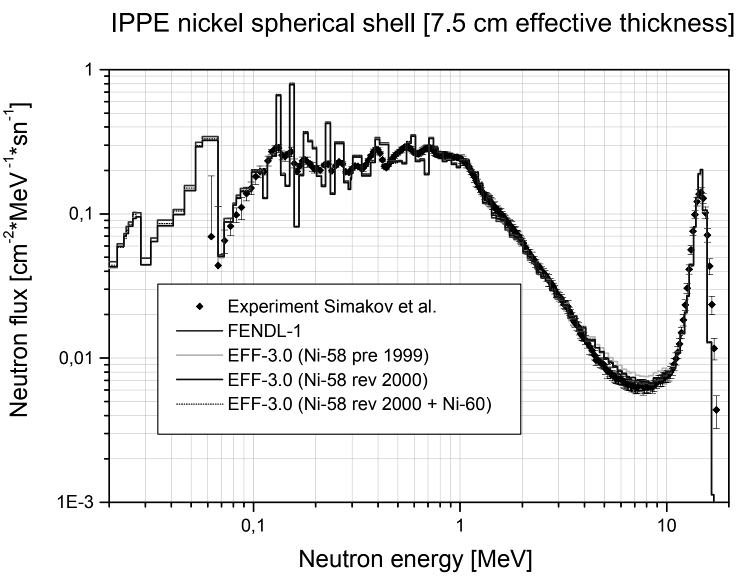
- MCNP calculation for neutron flux spectrum with point detector tally
  - Perel's version MCSEN for calculation of point detector sensitivities
- Data evaluations
  - FENDL-1 (ENDF/B-VI)
  - EFF-2.4
  - EFF-3.0 ^{58}Ni (previous & revised version)
  - EFF-3.0 ^{60}Ni (new)
- Sensitivity/uncertainty calculation in VITAMIN-J group structure for ^{58}Ni
  - Neutron flux detector tally in VITAMIN-J group structure
  - Co-variance data generated with NJOY

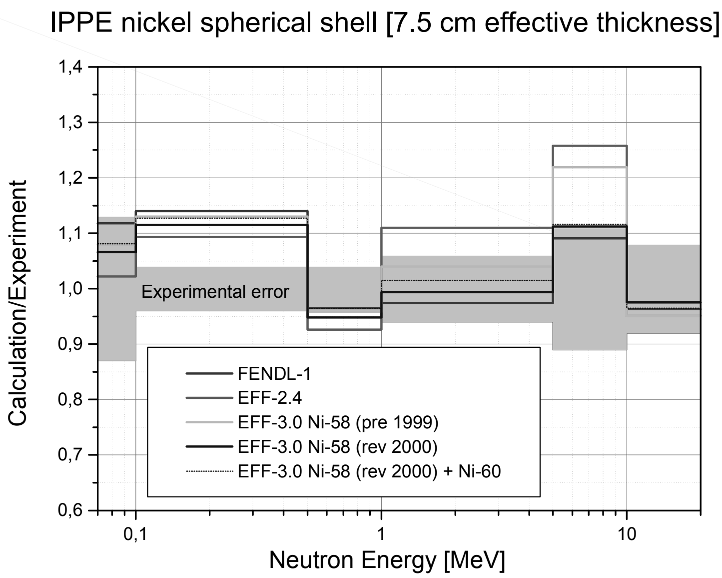
IPPE nickel spherical shell [7.5 cm effective thickness]



IPPE nickel spherical shell [7.5 cm effective thickness]




IPPE nickel spherical shell [7.5 cm effective thickness]
Neutron flux [cm-2*MeV-1*sn-1]
Neutron energy [MeV]
1
0,1
0,01
1E-3
0,1
1
10
Experiment Simakov et al.
FENDL-1
EFF-3.0 (Ni-58 pre 1999)
EFF-3.0 (Ni-58 rev 2000)
EFF-3.0 (Ni-58 rev 2000 + Ni-60)


IPPE nickel spherical shell [7.5 cm effective thickness]
Calculation/Experiment
Neutron Energy [MeV]
1,4
1,3
1,2
1,1
1,0
0,9
0,8
0,7
0,6
0,1
1
10
Experimental error
FENDL-1
EFF-2.4
EFF-3.0 Ni-58 (pre 1999)
EFF-3.0 Ni-58 (rev 2000)
EFF-3.0 Ni-58 (rev 2000) + Ni-60

IPPE nickel spherical shell [7.5 cm effective thickness]

Calculated uncertainties [%] of neutron flux integrals due to ^{58}Ni data uncertainties

^{58}Ni FENDL-1 (ENDF/B-VI)

E[MeV]	< 0.07	0.07-0.1	0.1-0.5	0.5-1	1-5	5-10	E> 10	total
Uncertainty [%]	3.01	2.90	2.45	2.52	2.76	6.58	3.82	1.76

^{58}Ni EFF-3.0

E[MeV]	< 0.07	0.07-0.1	0.1-0.5	0.5-1	1-5	5-10	E> 10	total
Uncertainty [%]	1.64	3.98	0.93	0.84	0.88	2.59	0.91	0.40

Ni benchmarking conclusions

- Revised ^{58}Ni EFF-3.0 evaluation
  - Good agreement with IPPE nickel transmission experiment
  - Results similar to FENDL-1
- ^{60}Ni EFF-3.0 evaluation
  - No significant effect on benchmark results
- Sensitivity/uncertainty calculations
  - Calculated uncertainties for EFF-3.0 considerably lower than for FENDL-1 data.

Beryllium benchmark analyses

KANT Beryllium spherical shell experiment (5/22 cm)

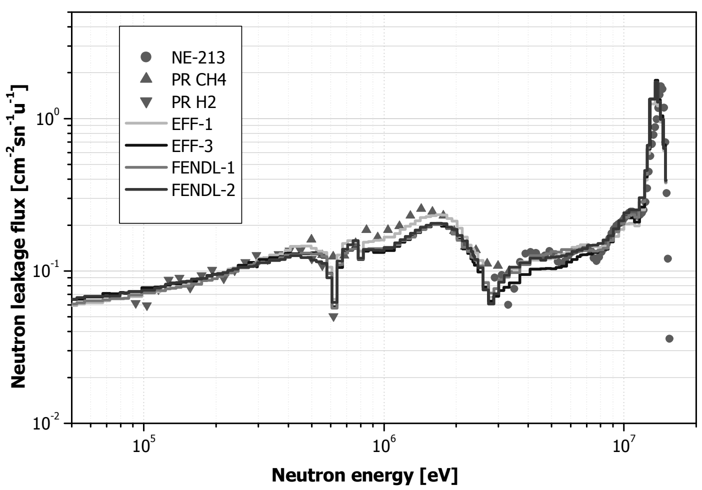
M = Multiplication factor   C/E = Calculation/Experiment

Experiment		Calculation				
			EFF-1	EFF-3	FENDL-1	FENDL-2
	M	M	1.696	1.684	1.684	1.709
Integrated spectrum	1.661 ± 7%	C/E	1.021	1.014	1.014	1.029
Bonner spheres	1.695 ± 7%	C/E	1.001	0.994	0.994	1.008

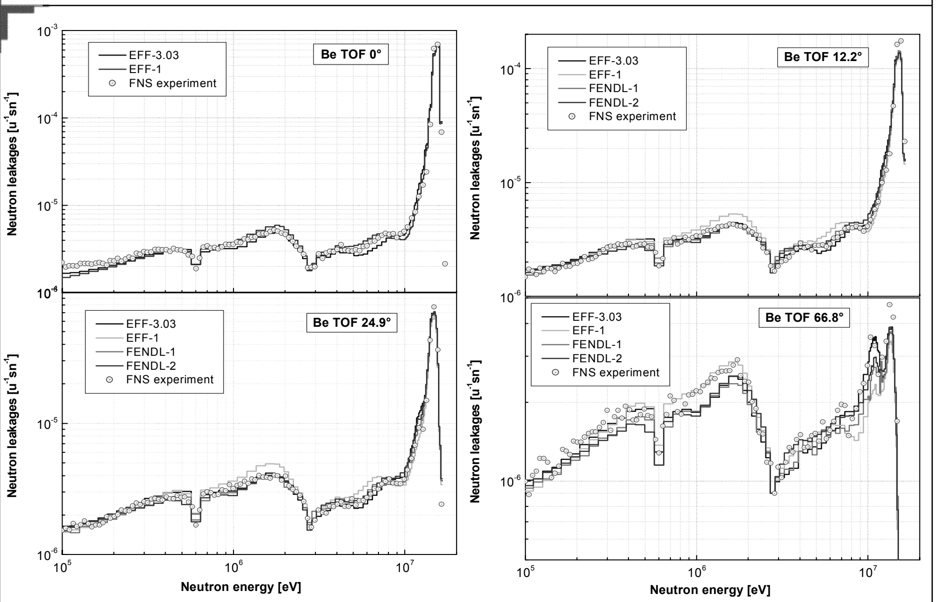
Uncertainties (in %) of the calculated neutron leakage due to uncertainties of the excitation function of the EFF-3 ^{9}Be data evaluation.

Energy [MeV]	< 0.1	0.1 - 1	1 – 2.5	2.5 - 6	6 – 10	10 – 13	> 13	Total
Uncertainty [%]	1.9	1.8	1.8	1.7	1.8	3.5	3.0	1.1

KANT Beryllium spherical shell (5/22 cm)

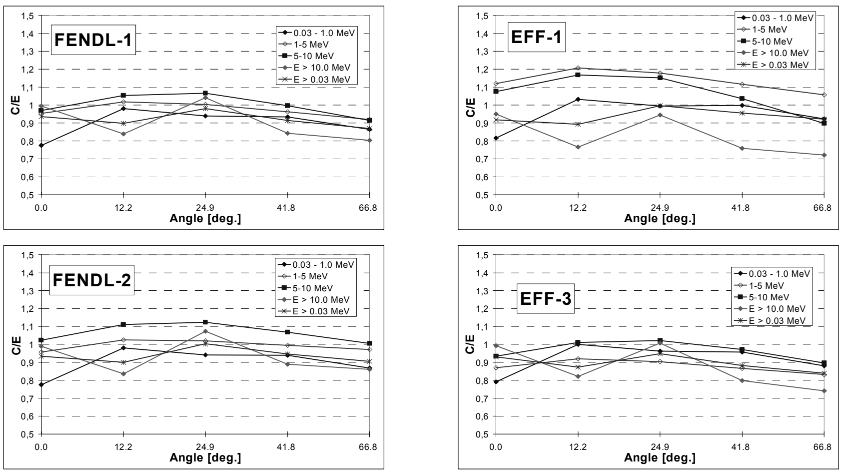


FNS TOF experiment on beryllium slab (15.24 cm)

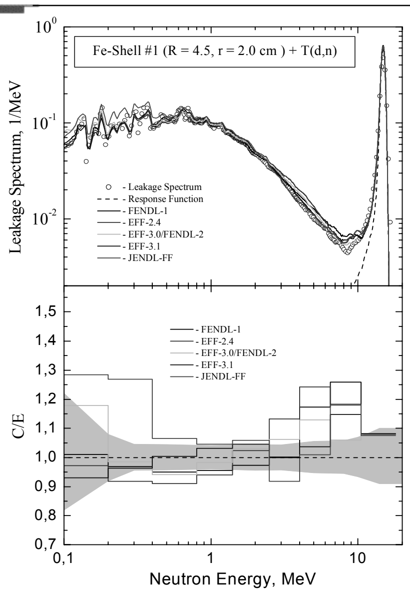


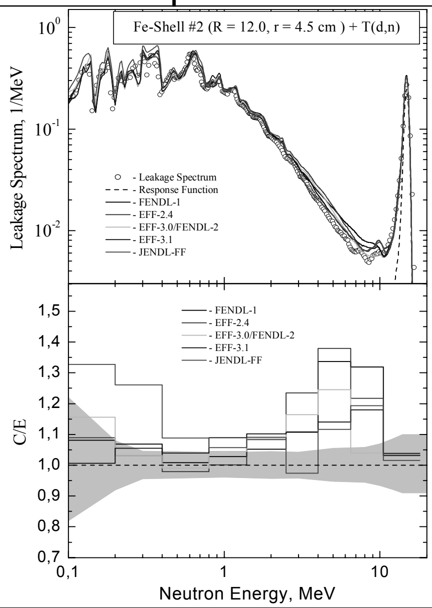
FNS/JAERI TOF Be slab experiment (t= 15.24 cm)

C/E comparison for neutron flux integrals

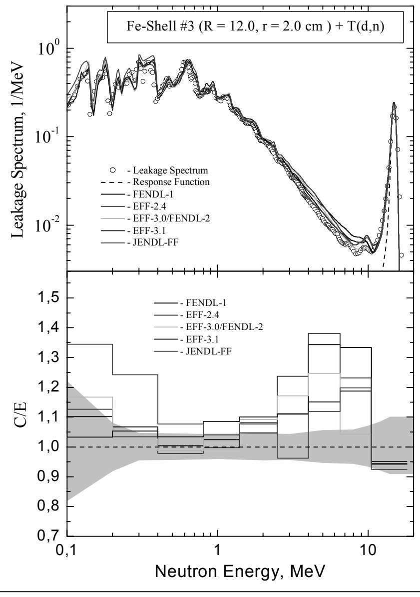


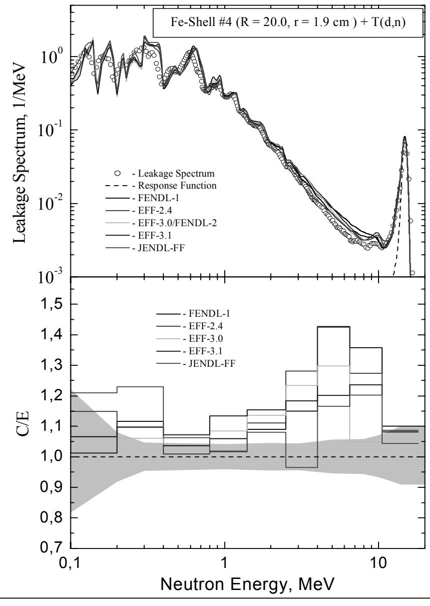
IPPE iron benchmark experiments



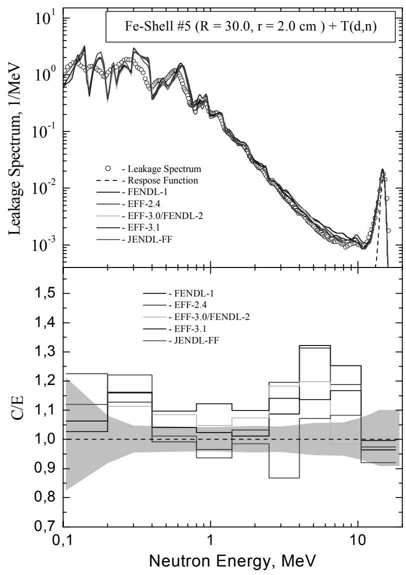


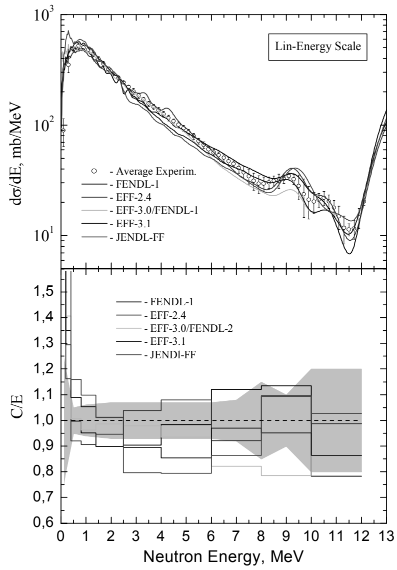
IPPE iron benchmark experiments



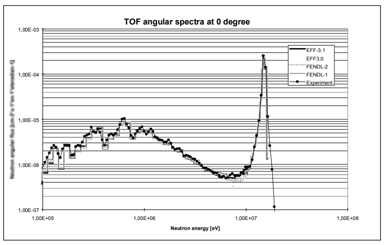


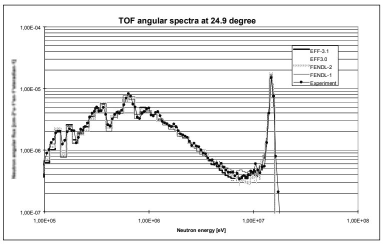
IPPE iron benchmark experiments

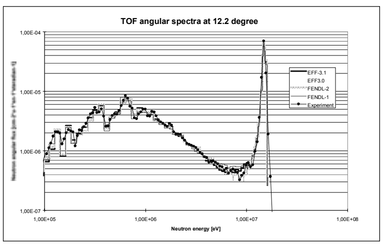


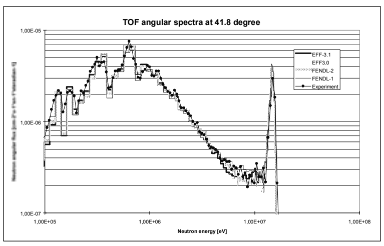


FNS/JAERI TOF iron slab experiment (t=20 cm)



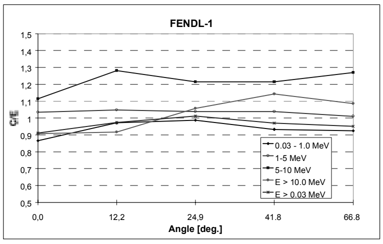


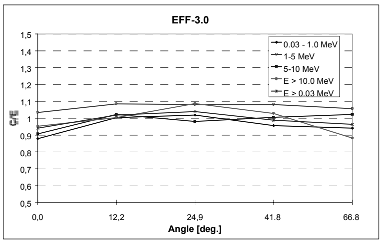


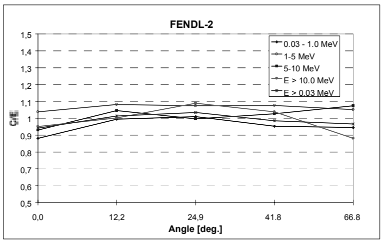


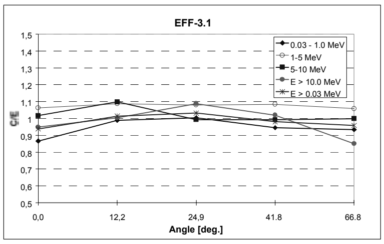
FNS/JAERI TOF iron slab experiment (t=20 cm)

C/E comparison for neutron flux integrals









Major Findings

Benchmark analyses conducted 1999-2003 in the frame of European Fusion File (EFF) Programme

Nuclide	Status	Recommendations
Be-9	Still discrepant results	Consider new EFF-3 evaluation; co-variance data
Fe-56	Satisfactory	No revisions required
Ni-58, -60	Satisfactory	Update of co-variance data evaluation required recommended (see EFF-3 evaluation)
Cr-52	Satisfactory	Neutron emission cross-section needs to be corrected
Si-28	Not satisfactory	Update required; see new EFF-3 evaluation
C-12	Satisfactory	No revisions required
W	Unsatisfactory	New/updated (isoptope) evaluation required

Appendix 3.3.

Needs for Nuclear Data Development and Improvement in the Fusion Technology Program in 2003-2006 and Beyond

P. Batistoni, U. Fischer and R.A. Forrest

NEEDS FOR NUCLEAR DATA DEVELOPMENT AND IMPROVEMENT IN THE FUSION TECHNOLOGY PROGRAM IN 2003-2006 AND BEYOND

P. Batistoni, U. Fischer and R. A. Forrest

1. Background

The present long-term strategy for the EU Fusion Technology Programme is based on the assumptions that ITER will be built in 2006 and will start operation in 2014, and that the IFMIF neutron source will be built in parallel with similar dates. Both facilities are needed to enable the design of DEMO and of power plants that can contribute to future electricity supplies.

Neutronics and nuclear data have an essential role to play in this development programme. A well-qualified nuclear database and validated computational tools are required for reliable neutronics and activation calculations and to provide assessments of the associated uncertainties. Within the EFF (European Fusion File) and EAF (European Activation File) projects, the EU is conducting a unique effort on nuclear data for fusion technology (FT) applications. This effort has led to the development of nuclear data libraries such as EFF-3 and EAF-2001 tailored to the ongoing and varying needs of the EU FT programme. A focused nuclear data programme is required for the coming decade to serve the needs of the FT programme with its long-term orientation towards fusion power plants (FPP) and short-term focus on ITER and IFMIF.

This paper addresses the needs for neutronics and nuclear data resulting from such a programme orientation, reviews the status achieved with the EFF/EAF project until the 5th EU framework programme (FP5) and outlines, on this basis, requirements for a future programme (for FP6 and beyond) so as to meet the FT programme objectives by utilizing the unique expertise available in the EU associations in the fields of nuclear data evaluation, processing, benchmarking, sensitivity/uncertainty analyses and nuclear experiments.

2. Nuclear data, computational tools and supporting experiments required for fusion design calculations

The nuclear design of fusion devices such as ITER, Demo and FPP rely on the results of neutronics calculations. These include neutron and photon transport calculations to provide the neutron/photon flux spectra which then form the basis for the calculation of nuclear responses of interest when convoluted with related nuclear data. Appropriate and qualified computational simulations are required to insure that the calculated nuclear responses are reliable. These in turn require appropriate computational methods and tools for simulating neutron transport, along with nuclear data both for the calculation of neutron transport and nuclear responses.

Neutron cross section data must be provided for the variety of nuclides constituting the materials to be used in a fusion device, including the breeders, neutron multipliers, coolants, shielding, magnets and insulators. Special emphasis must be put on high-quality data around 14 MeV. A major feature is the importance of inelastic neutron reactions, which require the use of double-differential cross section data to properly describe the energy-angle distributions of emitted secondary neutrons. As secondary photons, produced in neutron-induced reactions, contribute significantly to specific nuclear responses, it is required to include in the nuclear data libraries photon production and interaction data for use in coupled neutron-photon transport calculations. In addition, specific nuclear response data are required such as tritium production, kerma factors, gas production and radiation damage data. Activation and transmutation cross section data must be provided for the isotopes of all stable elements that may be present as impurities in materials. Also radionuclide targets require cross section data, as multistep reactions are also important in the high neutron fluxes. The decay data of all possible nuclides must be available to enable activation calculations to be carried out.

Computational tools based on the Monte Carlo simulation technique are required for neutronics design and sensitivity/uncertainty calculations of fusion devices that have a complex geometry such as ITER, Demo and FPP. Tools for activation and radiation damage calculations must be available with the related data and should be linked to the Monte Carlo tools for full 3D calculations.

Computational tools and data need to be validated to assure they give reliable results when applied in design calculations. This can be achieved through integral benchmark experiments where suitable material assemblies are irradiated with 14 MeV neutrons and nuclear responses of interest are measured and compared to calculations which closely simulate the experimental set-up. For validating activation data, small material samples have to be irradiated in a well-characterized neutron field.

3. Specific needs for ITER - TBM design

There are currently two EU breeder blanket concepts to be tested in ITER, the Helium-cooled Pebble Bed (HCPB) and the Water-cooled Lithium-Lead (WCLL) blanket, the latter possibly to be replaced by a Helium-cooled variant (HCLL). HCPB and W(H)CLL Test Blanket Modules (TBM) have to be designed within FP6 using DEMO relevant materials and technologies. The nuclear design of TBMs, the performance predictions including the achievable accuracy, as well as the design of the tests to be performed in ITER, including all safety aspects, are greatly affected by available nuclear data and nuclear computational tools.

Major objectives of the TBM tests in ITER from the neutronics point of view are to demonstrate the tritium breeding performance of the breeder blanket concepts and to check and validate the capability of the neutronics codes and data to predict the nuclear responses in the TBM with sufficiently high accuracy. This will allow the computational tools and data to be applied with confidence in design calculations for Demo and FPP.

The assessment of the output of nuclear tests on TBMs in ITER will be done in terms of the comparison (C/E ratios) between results of measured TBR, nuclear heating, decay heat and activation (E) in TBMs and the calculated predictions (C) of such quantities taking into account the uncertainties both on measurements and calculations. C/E deviations from unity within the total uncertainties will be regarded as experimental confirmation of numerical predictions, while larger deviations will lead to the conclusion of non-reliability of nuclear data or numerical tools. The value of the test output will depend on the quality of both the experimental and numerical tools used, i.e. on the narrowness of uncertainties.

Therefore, the efficient exploitation of TBMs tests on ITER requires that, the computational tools and nuclear data are made available within FP6. In particular, development work is necessary on Monte Carlo tools for sensitivity/uncertainty calculations and on the evaluation of covariance data for the nuclides present both in the TBMs and the surrounding components in ITER that affect the calculation for the TBM. In addition, integral experiments on TBM mock-ups irradiated with appropriate neutron spectra are required to validate, as much as possible, the computational tools and data prior to testing in ITER.

Such experiments test both the nuclear data and the neutronics tools and allow the uncertainty of the nuclear quantities to be determined. They provide valuable information to designers and help reduce safety factors and conservatisms when designing Demo or FPP concepts.

4. Specific needs for IFMIF neutronics

The International Fusion Material Irradiation Facility (IFMIF) uses the d-Li stripping reaction to produce neutrons for high fluence irradiations of FPP candidate materials. A flowing liquid lithium target is bombarded by a high current deuteron beam accelerated up to 40 MeV energy. The resultant neutron spectrum is fusion-relevant but includes a high-energy tail that extends up to 55 MeV. Neutronics and nuclear data play a key role in proving IFMIF's suitability as a neutron source for fusion-specific simulation irradiations. In addition, the data that are required for the technical layout of the test modules and facility sub-systems must be provided by neutronics calculations. These include the proof that IFMIF can meet its design goal with regard to the required irradiation test volume as well as the attainable annual fluence accumulation.

Dedicated computational tools and data are required for neutronics calculations of the IFMIF neutron source. These tools must be capable of simulating the transport of neutrons generated by Li(d,xn) reactions and of photons produced both in the lithium target and the material test assembly. Cross section data must be provided over the whole neutron energy range of IFMIF, which extends up to 55 MeV. Such data must be evaluated for a variety of nuclides important for neutron transport calculations. They must include all data types and reactions that are required to calculate the important nuclear responses such as nuclear heating, gas production and radiation damage. For activation calculations, on the other hand, a full set of data for all potential target nuclides must be available. To allow the preparation of working libraries for use with state-of-the-art transport and activation codes, complete data sets must be prepared in accordance with standard nuclear data format rules. The codes for neutronics and activation calculations must also be capable of handling the reaction channels that are open above the traditional 20 MeV energy domain.

Differential experiments are required to provide basic data for the d-Li reactions up to 40 MeV deuteron energy and neutron-induced cross section data up to 55 MeV neutron energy. Integral benchmark experiments are required to enable the validation of the cross section data for transport and activation calculations above 20 MeV. In addition, thick lithium target yield data are needed for checking and improving the d-Li neutron source term.

5. Status of computational tools, data and experiments (FP5 achievements)

The production of a complete data library for nuclear design calculations is a huge task that requires many steps including experimental measurements, theoretical evaluations, data processing, testing and benchmarking and, eventually, iterations to provide feedback and improvement of the data evaluations. The final product is a documented and validated data library containing a wide range of nuclides and quantities required for design applications such as reaction cross sections, secondary energy and angle distributions, gas productions, kerma factors, including uncertainty or covariance data. Europe has expertise in all these areas, and in addition the role of the NEA Data Bank in providing maintenance, quality assurance (QA) and co-ordination is essential for the successful development of the EFF/EAF nuclear data libraries.

The continuous effort has resulted in the following achievements during FP5:

- **The EFF-3 general purpose nuclear data library for neutron transport calculations.**
  - This includes original EFF-3 evaluations for ^{7}Li, ^{9}Be, ^{27}Al, ^{28}Si, ^{nat}V, ^{52}Cr, ^{56}Fe, ^{58}Ni and ^{60}Ni, and original EFF-2/-1 evaluations for the Mo isotopes and ^{nat}Pb. The remaining data evaluations are taken from other sources (ENDF/B-VI, JENDL-3.2, JEF-2.2). Only a few of the evaluations have covariance data.
  - Processed library for MCNP (ACE) and discrete ordinate (VITAMIN-J) calculations.

- **The European Activation File EAF-2001.**

Includes activation and transmutation cross sections for 766 target nuclides from Z=1 (hydrogen) to 100 (fermium) with neutron-induced reactions from 10^{-5} eV to 20 MeV. Partially validated for the important materials through integral activation experiments conducted within FP5.

- **The Intermediate Energy Activation File IEAF-2001.**

679 target nuclides from Z=1 (hydrogen) to 84 (polonium) with neutron-induced reactions from 10^{-5} eV to 150 MeV. Developed as part of IFMIF project within FP5. Based on results of model calculations above 20 MeV; validation required, not compatible with the FISPACT inventory code.

- **Intermediate Energy (50 MeV) general-purpose (ENDF) data files.**

Specific files for ^{1}H, ^{56}Fe, ^{23}Na, ^{39}K, ^{28}Si, ^{12}C, ^{52}Cr, ^{51}V up to 50 MeV incident neutron energy prepared as part of the IFMIF project for transport calculations. More intermediate energy data (150 MeV) files are available from other projects (APT, ADS).

- **Computational tools.**
  - Main tools for neutronics calculations are available from other sources or projects. Monte Carlo code MCNP (LANL) and the discrete ordinate code systems (DANTSYS, DOORS).
  - Specific tools for sensitivity calculations were developed as part of EFF project. SUSD3D for deterministic calculations and MCSEN for Monte Carlo sensitivity calculations of point detectors.
  - Monte Carlo transport code McDeLi/McDeLicious. This is an extension to MCNP to handle d-Li neutron source term for neutronics calculations, developed as part of the IFMIF project.
  - For activation calculations FISPACT was developed to support the EAF project. For IFMIF activation calculations with the IEAF-2001 library, the ALARA code (University of Wisconsin) is available.

- **Validation experiments**
  - Transport benchmark experiments. Integral 14 MeV neutron transport experiment on stainless steel assembly (AISI-316), ITER bulk shield, SiC assembly and the streaming mock-up (AISI-316/water).
  - Shut-down dose rate experiment using ITER streaming mock-up.
  - Activation experiments on Eurofer, SS-316, F82H, MANET, Fe, V, V-alloys, Cu, CuCrZr, W, Al, SiC, Li_4SiO_4.

6. Requirements for future programme on nuclear data

The requirements for a future programme on neutronics, nuclear data and experiments follow directly from the needs of the EU long-term fusion programme by comparing the computational tools, nuclear data and experiments already available with those required to satisfy these needs. Following this guideline, the effort has to focus on the following activities:

- **Extension of nuclear data libraries to neutron energies above 20 MeV**

The extension of the general-purpose files (EFF) for all the materials required for IFMIF is a large task and could not realistically be carried out in the time scale by the EU programme alone. It will be necessary to select available files from other sources and compile these into a form suitable for the EU programme. This will require some limited evaluation effort, but will be mostly concerned with compilation, testing, processing and benchmarking. The provision of uncertainty data for these higher energies will be more difficult and time consuming and should be added at a later stage.
The activation libraries (EAF) will similarly require extension to higher energies. In addition more data to describe the new reactions possible at these energies will be required. Calculational tools are available for these tasks within Europe, the main concern is that the extended library should not be totally based on theory, but should utilise existing experimental data. Benchmarking using integral data is needed at the traditional energy range using existing facilities and at higher energies using a cyclotron source. Higher energies will extend the range of possible radionuclides and hence the required decay data.

- **Tools and data for sensitivity and uncertainty assessments**

The assessment of uncertainties in any kind of complex geometry is a challenging task that can only be accomplished on a long-time scale. To this end it is necessary to develop computational sensitivity tools for Monte Carlo codes. Such techniques are currently available for handling point detector sensitivities for application in the analysis of benchmark experiments. In addition covariance data for all nuclides and nuclear reactions of interest in the design calculations are required. Again this is a large and demanding task that requires a strong effort on long-time scale for a successful completion. Covariance data are currently available only for a limited number of nuclide evaluations and reaction types. A large database of experimental cross sections with uncertainties is needed for preparing covariance data. Missing covariance data could be provided to some extent by theoretical assessments and model calculations.

- **Integral experiments**

Neutronic experiments on mock-ups of TBMs to check and validate the computational tools and nuclear data for design calculations, including the associated uncertainties are required. Such experiments have to be performed in suitable and well-characterized neutron fields available at the 14 MeV neutron generators in the EU associations. Measurements in such facilities provide the lowest possible experimental uncertainties that are required when checking with predicted calculational uncertainties. Neutronics experiments on TBM mock-ups at 14 MeV neutron generators could be followed by analogous experiments in JET during a DT experiment in FP6. This would provide a tokamak environment including a volume DT neutron source but with higher experimental uncertainties.
Activation experiments are required to validate the activation data file. In particular these are required for the activation cross section data above 20 MeV where only few experimental data are available and no validated data exist. Experiments have to be performed with facilities that can provide an IFMIF-like neutron spectrum that extends above 20 MeV. Further validation is also required for the traditional activation data below 20 MeV. Materials and elements not considered so far have to be investigated at 14 MeV neutron generators with the ultimate goal to arrive at a fully validated activation data library.

- **IFMIF D-Li neutron source term**

The validation of the D-Li neutron source term requires further experimental data on thick and thin Li targets, computational analyses of the experiments using McDeLicious calculations, and up-dating of the d-Li cross section evaluations.

7. Requirements for FP6 work programme on Nuclear Data

The following outline is guided by the objective to successfully conduct the envisaged FP6 fusion programme which focuses on the needs for ITER (TBM design) and IFMIF as well as providing the basis for a continued later programme which will focus on the needs for the construction and operation of ITER (TBM licensing, testing) and the design of Demo and FPP.

In detail the following activities are required within FP6:

- **Nuclear data for TBM design and ITER:**
  - Detailed analysis of the TBM designs using existing data libraries to identify important reactions and possible deficiencies in the current data evaluations.
  - Review of available cross sections and covariance data with focus on the TBM materials such as Li, Be, Al, Si, Ti, O, Pb, Fe, Cr, W, Ta.
  - Identification of most critical elements/isotopes using existing data files.
  - New and/or updated data evaluations where needed. Complete and validated data evaluations for isotopes are necessary.
  - Processing and benchmarking of new/updated evaluations.

- **Development and implementation of algorithms for the Monte Carlo** calculation of sensitivities and uncertainties of nuclear responses due to nuclear data variations and uncertainties in complex 3D geometry (extension of MCSEN code).

- **Extension of nuclear data libraries to neutron energies above 20 MeV**

General purpose (ENDF) data file for transport calculations

  - Review of cross section data evaluations available from the IFMIF project and other sources (LANL, JEFF-IE, JAERI).
  - Elaboration of a priority list for data evaluations to be performed within FP6.
  - Focus on detailed and complete evaluation of high priority data; these evaluations have to be adapted to the existing EFF/JEFF evaluations below 20 MeV; upper energy limit is 150 or 200 MeV (in accordance with JEFF-IE evaluations).
  - Other data evaluations (less priority) have to be taken from other sources (LANL, JEFF-IE, JAERI) and model calculations (TALYS, GNASH, ALICE).
  - Complete general-purpose 150 MeV data library must be prepared within FP6 (second priority data evaluations can be up-dated in a follow-up programme).

Activation data file for activation and transmutation calculations

  - Extension of the EAF-2001/3 libraries to 55 MeV covering cross section and decay data. In parallel, extension of the development tool (SAFEPAQ-II) and other codes (e.g. FISPACT) to handle the data up to 55MeV.
  - The upper energy limit of 55 MeV is sufficient for application to IFMIF activation analysis and allows individual reaction channels to be handled in the traditional way by FISPACT.
  - Validation of the Intermediate Activation File IEAF-2001 (150 MeV).

- The already existing IEAF-2001 activation data library, developed in the framework of the IFMIF project, is available for activation analyses of IFMIF as well as neutron sources with higher energies (e.g. spallation source). IEAF-2001 is not compliant with the FISPACT methodology. Validation of the data above 20 MeV is required. Results will be fed into the extension of the EAF-2003 data to 55 MeV.

- **Validation of the D-Li source term for IFMIF**

The validation of the D-Li neutron source term should be accomplished within FP6. In particular it is necessary to validate the McDeLicious approach on the basis of experimental thick Li target yield data. Up-dating of the underlying ^{6,7}Li + d data evaluations will be necessary making use of advanced evaluation methodologies and new experimental data.

- **Nuclear data improvement for materials required for divertor and first wall armour.** This would include Cu, W and Mo.

- **Benchmarking of the new, updated or extended nuclear data files** to check and validate the evaluation and the processing.

- **Integral activation experiments covering activation of materials** of relevance to TBM and IFMIF over as much of the energy range (0-55 MeV) as possible. This will cover materials not yet investigated such as Be, Pb, breeder ceramics and EUROFER-ODS. Data for as wide a range of impurities as possible are necessary as these are of great significance to activation of realistic materials. Experiments have to be performed with 14 MeV neutron generators (TBM materials) and with facilities that provide a neutron spectrum that extends above 20 MeV (IFMIF materials).

- **Integral benchmark experiments with 14 MeV neutron generators.**

Neutronics experiments have to be performed on mock-ups of TBMs (HCPB, HCLL) with the objective to check and validate the computational tools and nuclear data for design calculations including the associated uncertainties, as well as to develop high-precision measurement techniques not yet available. A follow-up neutronics experiment on a TBM mock-up in JET would be useful.

- **Integral benchmark experiments** are required with an IFMIF–like neutron source spectrum for the checking and validating the general-purpose data evaluations above 20 MeV. The focus has to be put on the materials of highest priority for IFMIF applications.

8. Summary

In conclusion, the results of the FP6 nuclear data work must be the construction of fully qualified data libraries (including uncertainty data) for all materials of importance to the TBMs and IFMIF design and the generation of computational tools for transport, activation and sensitivity/uncertainty analysis. This will also involve an experimental programme covering benchmarking, neutronics experiments and integral activation measurements.

Extension of the data evaluations to neutron energies above 20 MeV and preparation of covariance data are the highest priority tasks, in addition to the neutronics experiments to validate the data. By utilizing other data sources external to EFF/EAF it is judged that complete data libraries can be prepared within FP6 covering at least the energy range of the IFMIF neutron source. As the preparation of uncertainty data requires an even greater effort as well as the availability of sufficient experimental data, the effort should focus on the nuclides important for the TBM design and within FP6 should be limited to the traditional energy domain up to 20 MeV.

Appendix 3.4

Nuclear Data for Design Analyses of the Test Blanket Modules in ITER: Review and Recommendations for EFF/JEFF Evaluations

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EFF-DOC-852

Nuclear Data for Design Analyses of the Test Blanket Modules in ITER: Review and Recommendations for EFF/JEFF Evaluations

U. Fischer
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Forschungszentrum Karlsruhe

EFF/EAF Monitoring Meeting, NEA Data Bank, Paris, April 28-30, 2003

Overview

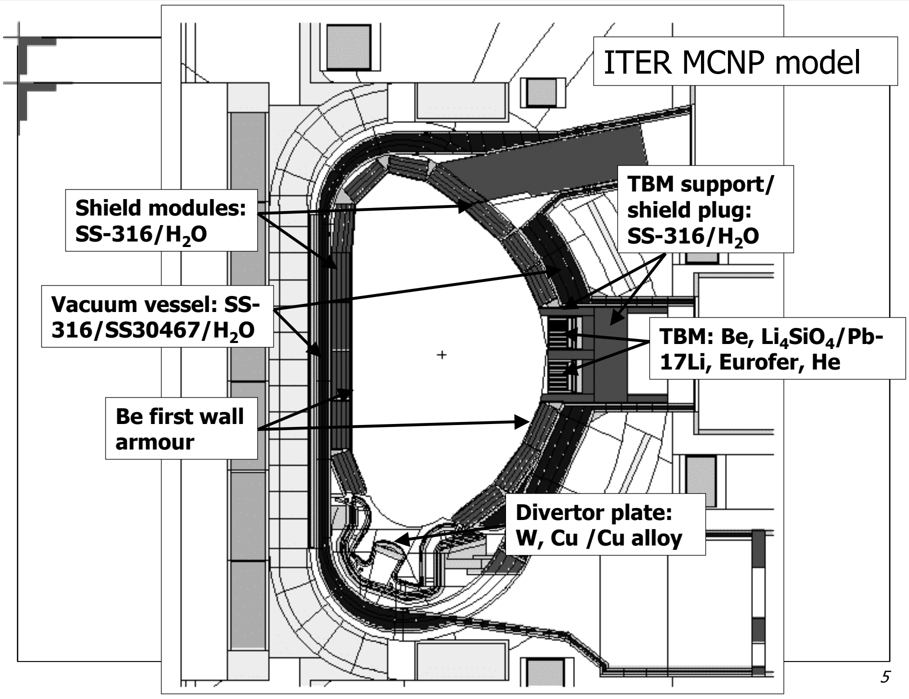
- ITER materials for nuclear analysis
  - Test Blanket Modules (TBM)
  - Shield modules, vacuum vessel, plasma facing components
  - Super-conducting magnet, minor importance materials
- Review of available nuclear data evaluations
  - EFF-3/JEFF-3.0 (EU)
  - FENDL-2.0, JENDL-3.3, ENDF/B-VI
  - MF=6 data, co-variances, γ-production
  - Benchmark analyses (data quality)
- Recommendations for evaluations
  - Priorities for EFF data evaluations in FP6
  - Update/revision/completion of data evaluations according to needs for TBM design
  - Extension for E > 20 MeV (IFMIF application)

Materials for ITER

- TBM materials
  - Be (multiplier, first wall armour)
  - Li_4SiO_4 (Li_2TiO_3), Pb-17Li
  - Eurofer (Fe, Cr, W, Ta, V, Mn, C, ….)
  - He (coolant)
- Plasma facing components
  - Be, W
  - Cu/Cu alloy (CuCrZr, CuAl25)
- Shield modules, vacuum vessel
  - SS-316L (N): Fe, Cr, Ni, Mo, Mn, C, N, ..
  - SS 30467 (2w% B): Fe, Cr, Ni, Mo, Mn, B, C, N, P, S, ..
  - H_2O

Materials for ITER (cont.)

- Super-conducting magnet
  - SC strand (Cu, Nb, Sn, Ta, Ti ,Cr)
  - Cu wire
  - SS-316LN (Fe, Cr, Ni, Mo, Mn, C, Si, P, S, N, …)
  - He, Incoloy
  - Insulator (Si, O, B, Al, H, C, ..)
- Bioshield
  - Concrete (LWR standard): O, Si, Ca, Al, Na, K, H, Fe, Mg, S
- Other materials
  - Ni alloys (bolts)
  - Ti alloy (module support cartridges)
  - Ceramics (electrical insulator): Al_2O_3/$MgAl_2O_4$



Eurofer Chemical Composition

Element	Specification [w%]	Element	Specification [w%]
C	0.090-0120	W	1.0-1.2
Mn	0.20-0.60	Ti	<0.01
P	<0.005	Cu	<0.005
S	<0.005	Nb	<0.001
Si	<0.05	Al	<0.01
Ni	<0.005	N	0.015-0.045
Cr	8.50-9.50	B	<0.001
Mo	<0.005	Co	<0.005
V	0.15-0.25	O	<0.01
Ta	0.05-0.09	Fe	balance

Element Priority List

Required elements for TBM design:

- Be, Pb, Li, Si, O, Fe, Cr, W, Ta, Cu, Ti, …

Requirements for data evaluations:

- MF=6 data for multiple particle emission reactions
  - Secondary particle & recoil spectra ($\Rightarrow$ kerma, dpa)
- Co-variance data for uncertainty analyses
  $\Rightarrow$ high priority for TBM design assessment
- γ- production data for heating calculations
- Isotope evaluations
- Qualified by benchmark analyses

Be-9

Be-9	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	EFF-3.05	EFF-3.05	JENDL-FF	JENDL-FF	VI.1
Co-variance data	1, 2, 3, 16, 103-105, 107, 875-890	1, 2, 3, 16, 103-105, 107, 875-890	no	no	no
MF=6 data	16, 107, 875-890	16, 107, 875-890	16	16	16, 600, 650, 700, 701, 800
γ-production data	6	6	12,14	12,14	12,14

Benchmarking	Results not satisfactory (Be spherical shell leakage spectra, tritium production in JAERI experiment)
Recommendation	Further benchmarking required for Be/breeder mock-ups experiments (JAERI, FNG); possibly revision/update required, extension to 150 MeV

Pb-nat, $^{204,206,207,208}Pb$

	EFF-3.0/2.4 Pb-nat	JEFF-3.0 $^{204,206\text{-}208}Pb$	FENDL-2.0 $^{206\text{-}208}Pb$	JENDL-3.3 $^{204,206\text{-}208}Pb$	ENDF/B-VI $^{206\text{-}208}Pb$
Origin	EFF-1	JENDL-3.2	ENDFB-VI	J-FF/3.3	VI.6 (150)
Co-variance data	no	no	1-4,16,17, 51-53,102	no	see FENDL
MF=6 data	16,17,22, 28, 91	no	16,17,91	16,17,22,28 91,203, 207	2,5,16,17,22,24,28,32,33,37, 41,51-91,600, 601-604, 649-653, 699-703, 749, 800-802, 849
γ-production data	12,14,15	12,14,15	12,14,15	12,14,15	6,12,14,15

Benchmarking	Available results (FNS-TOF leakage spectra: EFF-1, FENDL-2, JENDL-FF) satisfactory
Recommendation	EFF evaluation for isotopes required; co-variance data to be included; benchmarking to be repeated; extension to 150 MeV

Li-6

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	ENDF/B-V	ENDF-VI.3	ENDF-VI.1	J-FF/3.3	VI.1
Co-variance data	1,2,105	no	no	no	no
MF=6 data	no	no	no	no	no
γ-production data	12,14,15	12,14	12,14	12,14,15	12,14

Benchmarking	Available results (OKTAVIAN: Li, FNS-TOF: Li_2O) are very satisfactory including $^6Li(n,\alpha)T$ production rate (FENDL-2, JENDL-FF)
Recommendation	EFF evaluation needs to be updated to ENDF/B-VI standard including co-variance data; extension to 150 MeV

Li-7

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	EFF-2.4	EFF-2.4	ENDF-VI.0	JENDL-3.2	VI.0
Co-variance data	no		1,2,4,16,24, 25, 51-82, 102, 104, 851-859	no	1,2,4,16,24 25, 51-82, 102, 104, 851-859
MF=6 data	16, 24,25, 28,53,91		no	no	no
γ-production data	12,14		12,14	12,14	12,14

Benchmarking	Available results (OKTAVIAN: Li, FNS-TOF: Li_2O) are satisfactory including $^{7}Li(n,n'\alpha)T$ production rate (FENDL-2, JENDL-3.2)
Recommendation	EFF-2.4 evaluation needs to be benchmarked (!) & extended to 150 MeV

Si-28

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	EFF-3.0	EFF-3.0	ENDF-VI.0	JENDL-3.3	VI.6 (150)
Co-variance data	1-4, 16, 22,28, 51-67, 91, 102-107, 111, 600-613,649, 800-815, 849, 851, 852	1-4, 16, 22,28, 51-67, 91, 102-107, 111, 600-613,649, 800-815, 849, 851, 852	1-4,16,22,28, 51-67, 91,102-104,107, 600-613,649, 800-815,849	no	1-4,16,22,28, 51-67, 91,102-104,107, 600-613,649, 800-815,849
MF=6 data	16, 22,28, 51-67, 91, 600-613,649, 800-815, 849	16, 22,28, 51-67, 91, 600-613,649, 800-815, 849	16,22,28, 51-67, 91,102-104,107, 600-613,649, 800-815,849	16,22,28,91, 203,207	5, 16,22,28, 51-67, 91, 600-613,649, 800-815,849
γ-production data	12,14	12,14	12,14	12,13,14,15	6,12,14

Benchmarking	Available results (OKTAVIAN: Si, FNG: SiC) satisfactory; (EFF-3, FENDL-2); minor discrepancies in 5-7 MeV range not yet resolved
Recommendation	EFF evaluation to be extended to 150 MeV

$^{29,30}Si$

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	-	JENDL-3.2	ENDF-VI.0	JENDL-3.3	VI.6 (150)
Co-variance data	-	No	1-4,16,22,28, 51-64, 91, 102, 103,107, 600-615,649, 800-819,849	no	1-4,16,22,28, 51-64, 91,102, 103,107, 600-615,649, 800-819,849
MF=6 data	-	No	16,22,28, 51-64, 91, 600-615,649, 800-819,849	16,22,28,91, 203,207	5, 16,22,28, 51-64, 91, 600-615,649, 800-819,849
γ-production data	-	12,13,14,15	12,14	12,13,14,15	6,12,14

Benchmarking	Minor effects on neutron transport
Recommendation	$^{29,30}Si$ evaluations to be included in EFF-3.0 & extended to 150 MeV

O-16

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	ENDF/B-VI	ENDF/B-VI	JENDL-FF	JENDL-3.3	VI.6 (150)
Co-variance data	no	no	no	1,2,4,16,2228, 51-79, 91,102, 103,104,107; MF=34(2)	1,2,4,16,24 25, 51-82, 102, 104, 851-859
MF=6 data	no	no	no	no	5,16,22,23,28, 32,41,44,45, 91,108,112, 749
γ-production data	12,13,14	12,13,14	12,13,14,15	12,14,15	6,12,13,14

Benchmarking	Available results (FNS-TOF: Li_2O, liquid O) satisfactory with FENDL-2 and FENDL-1(=ENDFB/-VI) data
Recommendation	EFF to be updated for co-variance data & extended to 150 MeV

Fe-56

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	EFF-3.1	EFF-3.1	EFF-3.0	JENDL-3.3	VI.6 (150)
Co-variance data	1-4, 16, 22,28, 51-82, 91, 102-107, 600-613, 649, 800-810, 849-853 MF=34 (2)		1-4, 16, 22,28, 51-82, 91, 102- 107, 600-613, 649, 800-810, 849-853 MF=34 (2)	1, 2, 4, 16,22,28, 51-77,91, 102,103,107 MF=34 (2)	1-4,16,22,28, 51-75,91, 102-107
MF=6 data	16,22,28, 91, 649,849		16,22,28, 91, 649,849	16,22,28,91, 203,207	5, 16,22,28, 91, 103,107
γ-production data	6,12, 14		12,14,15	12,14,15	6,12,14,15

Benchmarking	Many analyses performed; results satisfactory
Recommendation	EFFevaluation to be extended to 150 MeV

$^{54,57,58}Fe$

	EFF-3.0/2.4	JEFF-3.0(*) $^{57,58}Fe$	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	JEF-2	JEF-2	ENDF/B-VI.1	JENDL-3.3	VI.6 (150)
Co-variance data	No	No	1-4,16,22,28, 51-..,91, 102, 103,(104),107	No	1-4,16,22,28, 51-..,91, 102, 103,(104),107
MF=6 data	No	No	16,22,28, 51-.., 91, 103,107	16,22,28,91, 203,207	5(*), 16,22,28, 51-.., 91, 103,107
γ-production data	12, 13, 14,15	12, 13, 14,15	12,14,15	12,14,15	6,12,14,15
		(*) ^{54}Fe is from ENDF/B-VI.3			(*) not for ^{58}Fe (E< 20 MeV)

Benchmarking	
Recommendation	Evaluations required for EFF (including co-variance & MF=6 data, up to 150 MeV)

Cr-52

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	EFF-3.03	EFF-2.4	ENDF-VI.1	JENDL-3.3	VI.6 (150)
Co-variance data	1-4, 16, 22,28, 51-60, 91, 102- 107, 851, 852 MF=34 (2)	No	1-4,16,22,28, 51-60,91, 102,103,107	1, 2, 4, 16,22,28, 51-62,91, 102,103,107 MF=34 (2)	1-4,16,22,28, 51-60,91, 102,103,107
MF=6 data	16,22,28, 51-60, 91, 103,107 (adopted from ENDF/B-VI)	5	16,22,28, 51-60, 91, 103,107	16,22,28,91, 203,207	5, 16,22,28, 51-60, 91, 103,107
γ-production data	6,12, 14, 15	12,14,15	12,14,15	12,14,15	6,12,14,15

Benchmarking	EFF-3.0/EFF-2.4 file was shown to be obsolete
Recommendation	Revised evaluation EFF-3.03 evaluation (IRK,CEA, March 2002) to be processed and benchmarked; to be extended to 150 MeV.

$^{50,53,54}Cr$

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	JEF-2	ENDF-VI.3	ENDF-VI.1	JENDL-3.3	VI.6 (150)
Co-variance data	No	1-4,16,22,28, 51-..,91, 102, (103),104,107	1-4,16,22,28, 51-..,91, 102, (193),104,107	No	1-4,16,22,28, 51-..,91, 102, (103),104,107
MF=6 data	No	16,22,28, 51-.., 91, 103,107	16,22,28, 51-.., 91, 103,107	16,22,28,91, 203,207	5, 16,22,28, 51-.., 91, 103,107
γ-production data	12, 13, 14, 15	12,14,15	12,14,15	12,14,15	6,12,14,15

Benchmarking	
Recommendation	Evaluations required for EFF (including co-variance & MF=6 data, up to 150 MeV)

W-nat, $^{182,183,184,186}W$

	EFF-3.0/2.4 W-nat	JEFF-3.0 $^{182-184,186}W$	FENDL-2.0 W-nat	JENDL-3.3 $^{182-184,186}W$	ENDF/B-VI $^{182-184,186}W$
Origin	JENDL-3.0	JENDL-3.2	JENDL-FF	JENDL-3.3	VI.6 (150)
Co-variance data	no	no	no	no	no
MF=6 data	no	no	16,17,22, 28,91,203, 204,207,219	16,17,22,28, 91,203,204, 207	5
γ-production data	12,13,14,15	**No (!)**	12,13,14,15	12,13,14,15	6,12,13,14,15

Benchmarking	Results (FNG W experiment) not satisfactory: fast flux underestimation; EFF: gross overestimation of γ-production
Recommendation	Re-evaluation required for EFF (isotopes including co-variance & MF=6 data, up to 150 MeV)

Ta-181

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	ENDF/B-V	JENDL-3.2	JENDL-3.1	JENDL-3.3	ENDF/B-V
Co-variance data	No	No	No	No	No
MF=6 data	No	No	No	No	No
γ-production data	12, 13, 14,15	12,14,15	12,14,15	12,14,15	12,13,14,15

Benchmarking	Nothing available
Recommendation	New evaluation required (up to 150 MeV)

$^{63,65}Cu$

	EFF-3.0/2.4	JEFF-3.0	FENDL-2.0	JENDL-3.3	ENDF/B-VI
Origin	ENDF/B-VI	ENDF/B-VI.3	ENDF-VI.2	JENDL-3.3	VI.6 (150)
Co-variance data	1-4, 16, 22,28, 51-72(63), 91, 102-104, 106, 107	1-4, 16, 22,28, 51-72(63), 91, 102-104, 106, 107	1-4, 16, 22,28, 51-72(63), 91, 102-104, 106, 107	No	1-4,16,22,28, 51-72(63),91, 102,103,107
MF=6 data	16,22,28, 51-72(63), 91, 103,107	16,22,28, 51-72(63), 91, 103,107	16,22,28, 51-72(63), 91, 103,107	16,22,28,32, 91,203,204, 207	5, 16,22,28, 51-72(63), 91, 103,107
γ-production data	12, 14, 15	12,14,15	12,14,15	12,14,15	6, 12,14,15

Benchmarking	Available benchmark results fairly good.
Recommendation	EFF evaluations to be extended to 150 MeV (ENDF/B-VI.6 ?)

Ti-nat, $^{46,47,48,49}Ti$

	EFF-3.0/2.4 ^{48}Ti	JEFF-3.0 Ti-nat	FENDL-2.0 Ti-nat	JENDL-3.3 $^{46\text{-}50}Ti$	ENDF/B-VI $^{46\text{-}50}Ti$, -nat
Origin	Ongoing EFF-3 evaluation (IRK Vienna)	JENDL-3.2	JENDL-3.1	JENDL-3.3	VI.1 ENDFV/B-V
Co-variance data	yes	no	no	*1,4,16,22,28 102,103,107 (^{48}Ti only !)*	28, 103 (46,47,^{48}Ti)
MF=6 data	yes	no	no	16,(17),22, 28,91,203, 207	no, also no MF=4,5 data for isotopes !
γ-production data	yes	12,13,14,15	12,13,14,15	12,14,15	12,13,14,15 (Ti-nat only)

Benchmarking	Available results (OLTAVIAN experiment; FENDL-1, -2, EFF-2 data not satisfactory) not satisfactory; overestimation of fast neutron flux.
Recommendation	Ongoing ^{48}Ti evaluation to be completed & benchmarked; evaluations to be extended to other isotopes (including co-variance & MF=6 data)

Required Isotope Data Evaluations

⇒ *TBM design*

Priority	Isotopes	Possible other sources
High		
	$^{204,206,207,208}Pb$	ENDF/B-VI
	$^{182,183,184,186}W$	ENDF/B-VI
Medium		
	^{181}Ta	No
	$^{29,30}Si$	ENDF/B-VI
	$^{54,57,58}Fe$	ENDF/B-VI
	$^{50,53,54}Cr$	ENDF/B-VI
Low		
	$^{63,65}Cu$	ENDF/B-VI
	$^{46,47,49}Ti$	JENDL-3.3

Required Co-variance Evaluations

⇒ *TBM design*

Priority	Isotopes	Possible other sources
High		
	^{6}Li	ENDF/B-V (obsolete)
	$^{206,207,208}Pb$	ENDF/B-VI
	^{16}O	ENDF/B-VI
	$^{182,183,184,186}W$	No
Medium		
	$^{29,30}Si$	ENDF/B-VI
	$^{54,57,58}Fe$	ENDF/B-VI
	$^{50,53,54}Cr$	ENDF/B-VI
	$^{63,65}Cu$	ENDF/B-VI
Low		
	^{7}Li	ENDF/B-VI
	^{181}Ta	No
	$^{46,47,49}Ti$	JENDL-3.3

Required 150 MeV Data Evaluations

⇒IFMIF

Priority	Isotopes	Available	Priority	Isotopes	Available
High			Medium		
	^{56}Fe	ENDF/B-VI.6, NRG, FZK/INPE (50)		$^{54,\ 57,58}$Fe	ENDF/B-VI.6
	^{52}Cr	ENDF/B-VI.6, FZK/INPE (50)		$^{50,\ 53,54}$Cr	ENDF/B-VI.6
	$^{182,183,\ 184,\ 186}$W	ENDF/B-VI.6		^{29,30}Si	ENDF/B-VI.6
	^{9}Be	FZK/INPE		$^{63,\ 65}$Cu	ENDF/B-VI.6
	^{6,7}Li	FZK/INPE		^{1}H	ENDF/B-VI.6
	^{28}Si	ENDF/B-VI.6, FZK/INPE (50)		^{181}Ta	-
	^{12}C	ENDF/B-VI.6, FZK/INPE (50)		+ many more	
	^{16}O	ENDF/B-VI.6, FZK/INPE (50)	Low		
	^{23}Na	FZK/INPE (50)		$^{46,\ 47,48,49}$Ti	
	^{39}K	FZK/INPE (50)		+ many more	.

Recommendations

TBM design

- Isotope evaluations:
  - Start with ^{206,207,208}Pb, $^{182,183,\ 184,\ 186}$W and take into account available ENDF/B-VI evaluations
  - Next: ^{181}Ta
- Co-variance data:
  - Start with ^{6}Li, ^{16}O, ^{206,207,208}Pb, $^{182,183,\ 184,\ 186}$W and take into account available ENDF/B-VI evaluations

IFMIF application

- 150 MeV evaluations
  - Detailed priority list to be elaborated (9/2003)
  - Tentative: start with ^{56}Fe, ^{52}Cr, $^{182,183,\ 184,\ 186}$W; Next: ^{9}Be, ^{6,7}Li, ^{28}Si, ^{16}O, ^{12}C
  - Take into account available 150 MeV evaluations & ensure consistency of E<20 MeV data with EFF/JEFF evaluations

Appendix 3.5.

Activation libraries since FENDL/A-2

R..A. Forrest

Activation libraries since FENDL/A-2

Robin Forrest
Euratom/UKAEA Fusion Association
Culham Science Centre

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Outline

- FENDL/A-2
- EAF-97, EAF-99
- SAFEPAQ-II
- EAF-2001, EAF-2003
- Validation of EAF by integral measurements
  - EAF-2001
  - EAF-2003
- IFMIF requirements
- IEAF-2001
- EAF-2004
- JEFF-3.0/A
- Future needs for FENDL/A

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FENDL/A-2

- Released in 1997
- 390 important reactions, relevant to ITER design, significant at short and long cooling times compiled by E. Cheng, R. Forrest, J. Kopecky and F. Mann
- Input taken from EAF-4.1, ADL-3 and JENDL/A-3.2
- Results of the IAEA CRP on Activation Cross Sections for the Generation of Long-Lived Radionuclides and FENDL/A-1.1 also considered
- Graphical intercomparisons prepared by NDS showing the various data sets and experimental data from EXFOR
- Selection of best data for important reactions + rest of library from EAF-4.1
- 13,006 reactions on 739 targets

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FENDL/A-2 (continued)

- Decay data library complementing FENDL/A-2 - both required for activation calculations
- Taken directly from EAF-4.1 decay data library
- Note that there is no ‘official’ FENDL-2 uncertainty library, but that the EAF-4.1 uncertainty library was distributed as an ‘unofficial’ FENDL library
- The point-wise file was available in both EAF format and in an ENDF format
- Processed data, for MCNP and in multi-group (Vitamin-J) format available for applications
- No modifications since issue, available for download from IAEA web site
- Acts as a reference library

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EAF-97

- Released in summer 1997
- 12,469 reactions on 766 targets
- Emphasis on actinide and capture data
- Erroneous interpolation laws in some reactions were noted
- EAF-97.1 (Jan 1998) corrected 353 reactions

- Note that further details of the history of activation libraries is given in the EAF-2003 cross section library report **UKAEA FUS 486**) download from http://fusion.org.uk/easy2003/

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EAF-99

- Released in beginning of 1999
- 12,468 reactions on 766 targets
- Emphasis on integral data (generated as part of European Fusion Technology programme)
- About 750 reactions significantly modified based on integral data and EXFOR data
- New data sources:
  - Russian dosimetry file (RRDF)
  - Helium production library (HEPRL-96)
  - Data from IRK Vienna and LANL
- Improvements in 1/*v* region for non-threshold reactions
- Improvements in uncertainty file

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Library production tools

Automated tools are essential to produced activation libraries

- SYMPAL (UNIX scripts + FORTRAN) used for EAF-0 to EAF-99
- SAFEPAQ (UNIX widgets + PV-WAVE) used as a front end for SYMPAL
- Work started on a new tool to replace SYMPAL/SAFEPAQ in 1998. Radical design:
  - Relational databases to hold all data
  - Windows application, built using Visual Basic (VB)
  - SQL queries in databases called from VB code
- EAF-2001 and subsequent libraries built using SAFEPAQ-II
- Note that EAF is a part of EASY

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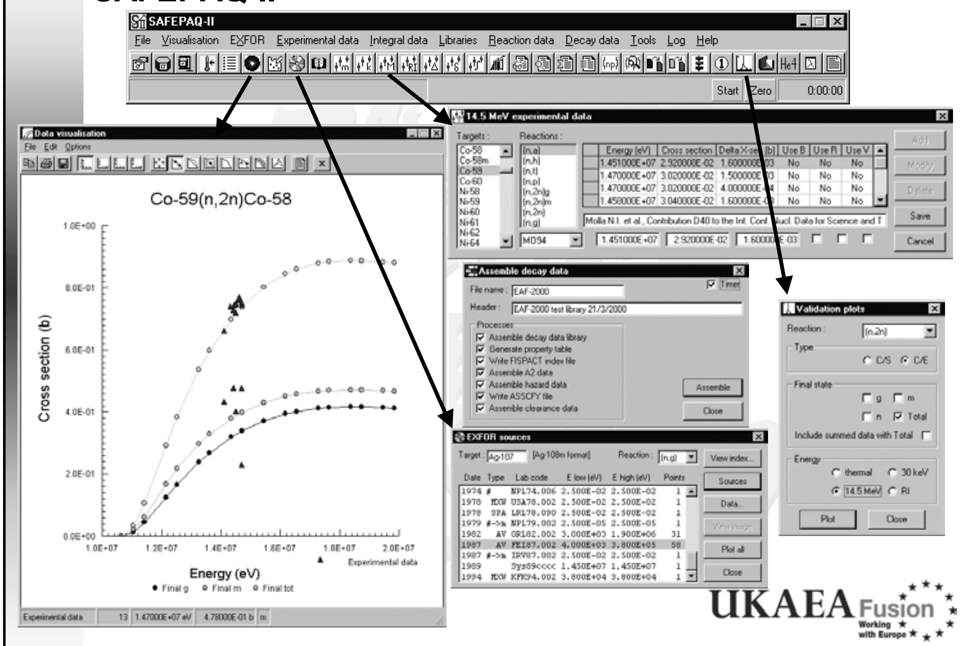
SAFEPAQ-II



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SAFEPAQ-II - summary

- Central idea - all data stored in relational databases
- Convert cross section source files → databases
- MASTER - a copy of data from original source
- PARAMETER - all experimental data (“know how”)
- EXFOR - selection from EXFOR CDs
- FINAL - modified version of MASTER
- Visualisation - central to comparison
- Modifications - the real “work”
- Validation
- Integral data
- Iteration
- Finishing off

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EAF-2001

- Released in March of 2001
- 12,470 reactions on 766 targets
- Emphasis on integral data comparisons
- New data sources:
  - IRK evaluations of Ni isotopes
  - Mengoni calculations on light nuclei
- Maintenance release distributed in December 2001

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EAF-2003

- Released December 2002 - current version
- 12,617 reactions on 774 targets
- New systematic for (n,n′α) at 14.5 MeV
- (n,γ) and (n,f) reactions improved by SRA where no available resonance data
- Model calculations of 33 (n,pα) reactions from TALYS
- New data sources:
  - CEA evaluations of Y isotopes and ^{197}Au
  - CENDL-2.1
  - Avrigeanu calculations for V, Co and Ni isotopes
  - ENDF-B/VI release 7
  - JENDL-3.3, JENDL-99D
  - TALYS-2
- Integral and differential validation

EASY-2003

EAF-2003

Nuclear data library

Cross sections

Uncertainties

Decay data

Biological hazards

EASY-2003

EAF-2003

FISPACT-2003

EASY-2003 User Interface

EASY-2003 documentation

FISPACT-2003

Inventory code calculates amounts of radionuclides

EASY-2003 User Interface

Allows FISPACT to be run

View and summarise output

Plot graphs

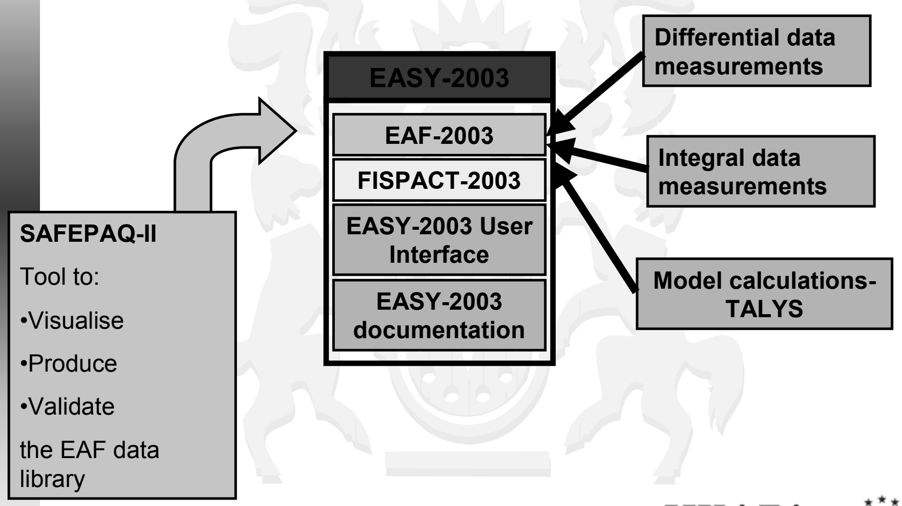
View all nuclear data

EASY-2003 documentation

6 UKAEA reports

On www & CD-Rom

EASY-2003



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Validation of EAF by integral measurements

- Large number of measurements made as part of EFDA programme
  - ENEA Frascati (FNG)
  - FZK
  - TUD (SNEG)
- Analysed with EASY and where possible an effective cross section can be extracted and entered into SAFEPAQ-II
- This is σ measured in a particular neutron spectrum (σ^E)
- At any stage in library development the EAF σ can be averaged in the same spectrum and σ^C found
- Ratio (C/E) indicates agreement
- For EASY-2001 65 reactions investigated, 35 validated, 25 recommended to be improved
- Report (UKAEA FUS 467) can be downloaded

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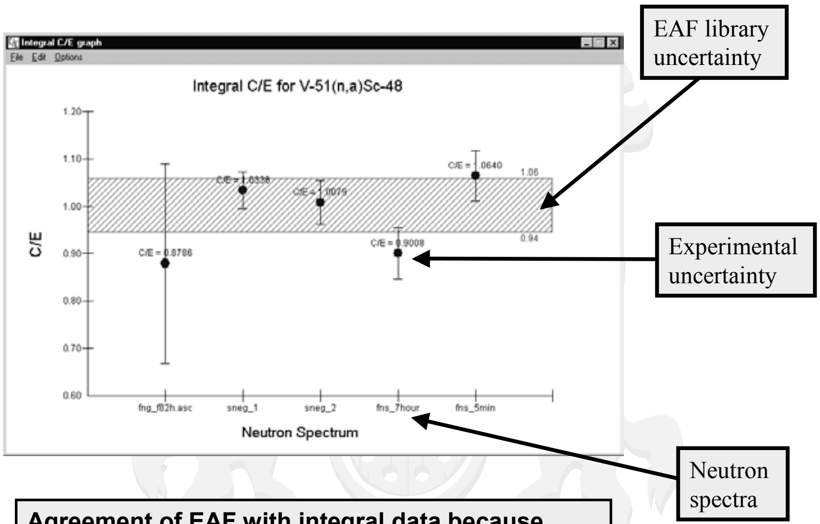
Validation of EAF-2001



Agreement of EAF with integral data because error bars overlap the error band

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Validation of EAF-2003

- Building on the EASY-2001 integral database
  - FNG and TUD measurements of CuCrZr
  - TUD measurements of W
  - FNG heat measurements
  - Sublet analysis of JAERI heat measurements (FNS)
  - ^{252}Cf spectrum measurements
- Effective cross sections for 287 reactions
- 171 reactions validated
- Report (UKAEA FUS 500) will be available on UKAEA web site soon
- A new feature built into SAFEPAQ-II that allows any library to be compared with the integral database - see later

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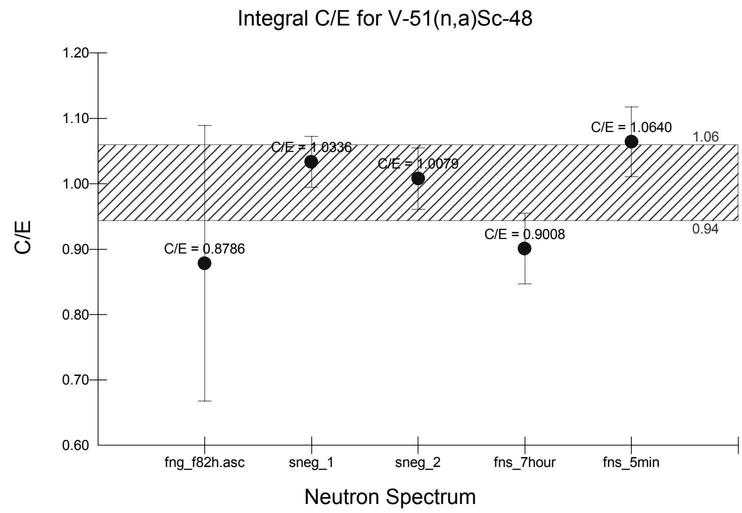
Integral data



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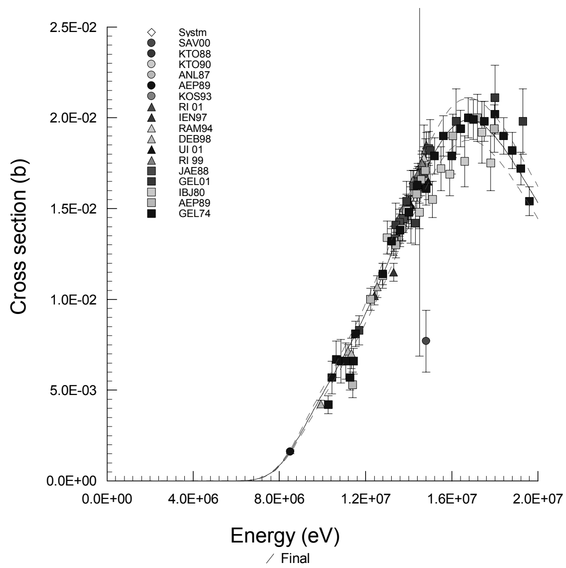
$^{51}V(n,\alpha)^{48}Sc$



Data in EAF-2003 agrees with both differential and integral measurements

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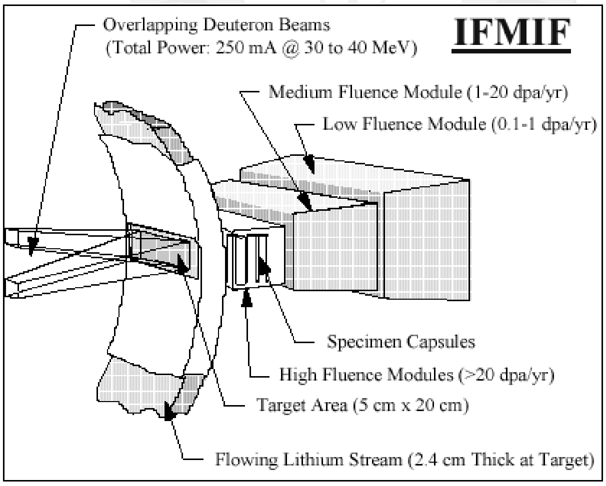
IFMIF requirements

- Deuteron beams on flowing Li target



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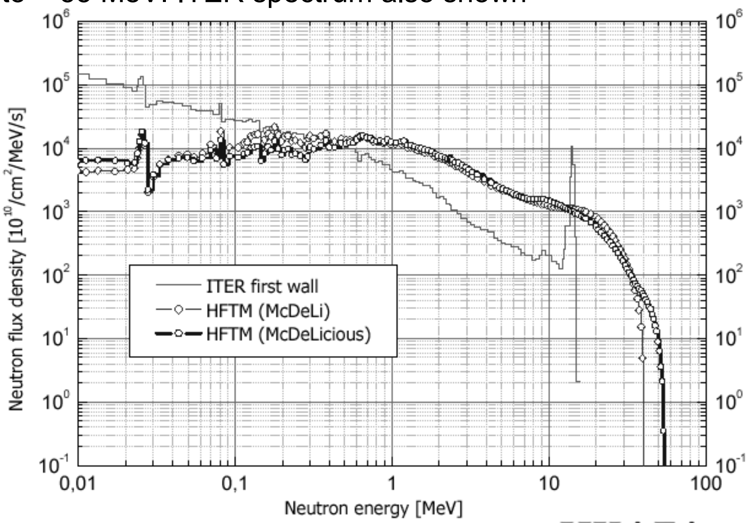
IFMIF (continued)

- Neutron spectra calculated by FZK shows a high energy tail to ~ 55 MeV. ITER spectrum also shown



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IFMIF requirements

- Activation calculations for IFMIF require
  - Cross section data for E > 20 MeV
  - Decay data on additional nuclides
  - Inventory code extended to use new data
- Initial approach by FZK resulted in
  - Production of IEAF-2001 (using total cross section + 'fission yield' giving production of all final nuclides)
  - Use of ALARA inventory code
  - Now extended to use a modified version of FISPACT
- Now EFDA has decided to extend EASY so allow high energy calculations
  - EAF-2004 (test) and EAF-2005
  - FISPACT extended

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IEAF-2001

- Produced by FZK using cross section calculations by Obninsk
- Covers energy range up to 150 MeV, H - Po
- Based on EAF-99 below 20 MeV
- ENDF format
  - Total cross section in MF=3
  - Isotope production yields in MF=6, MT=5
  - Reaction cross section is product of the two
- Number of reactions: 134,431 on 679 targets
- Group data in 256 groups = 175 (VITAMIN-J) + 1 MeV bins to 50 MeV + 2 MeV bins to 150 MeV
- Report (U. Fischer et al., FZK Interner Bericht IRS-Nr. 10.01 -FUSION-Nr. 179, 2001)
- Used for IFMIF calculations

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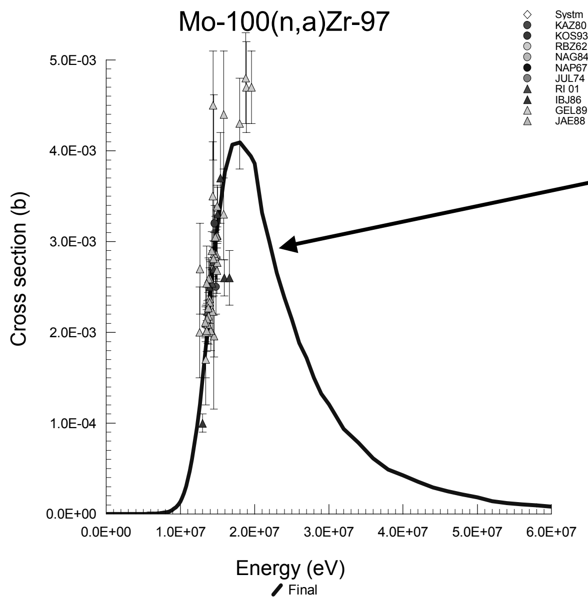
EAF-2004

- Energy range up to 60 MeV (sufficient for IFMIF)
- Uses same format as in previous EAF libraries, requiring extension of MT numbers
- Use of (a European) code system, TALYS (NRG and CEA) for data > 20 MeV
- Based on EAF-2003, all reactions are extended to 60 MeV (smooth join)
- New reaction classes (up to 8 particles emitted) extend from threshold - 60 MeV
- Number of reactions: 62,680 on 775 targets
- Requires increase in decay data library because of additional reactions
- 2,195 nuclides (increased from 1917)
- New group structures VITAMIN-J+ (211), TRIPOLI+ (351)

UKAEA Fusion Working with Europe

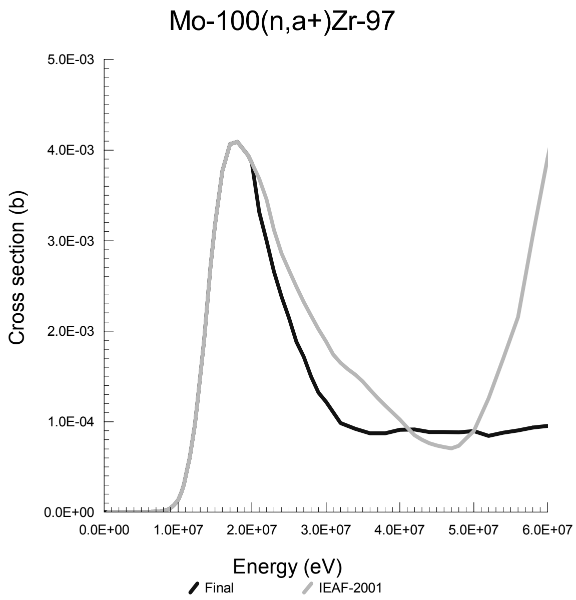
$^{100}Mo(n,\alpha)^{97}Zr$ - EAF-2004



Extended smoothly > 20 MeV

UKAEA Fusion Working with Europe

^{100}Mo(n,α+)^{97}Zr - EAF-2004 / IEAF-2001



UKAEA Fusion Working with Europe

JEFF-3 activation file

- EAF file uses the EAF format which is a slight modification to the ENDF standard (MF-3). This is to allow easy handling of isomeric states
- It is being converted to ENDF but much more complex:
  - MF-1 general info
  - MF-2 skeleton (radius)
  - MF-3 Total σ
  - MF-8 File pointer (MF-3/9 or MF-10)
  - MF-9 Isomeric branching ratio
  - MF-10 Partial cross section
- MF-3 or MF-10, no totals when partials exist
- MF-3/9 for non-thresholds, MF-10 for thresholds
- EAF-2003 $\Rightarrow$ JEFF-3/A

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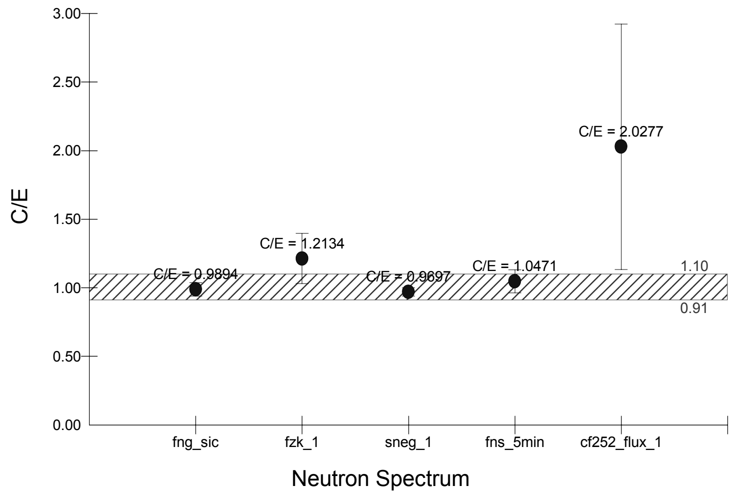
Future needs for FENDL/A

- Could keep FENDL/A-2 data for 390 important reactions, use EAF-2003 for remainder
- Uncertainty file required?
- High energy data important?
- What about decay data file?
- Format?
- Processed files?
- Validation?
  - FENDL/A-2 not validated using the integral database, but this could be possible
  - Tools exist in SAFEPAQ-II to plot C/E for FENDL/A-2

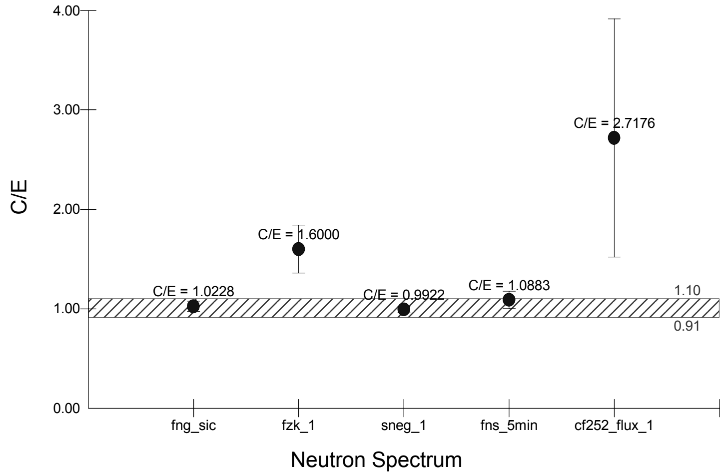
UKAEA Fusion Working with Europe

$^{29}Si(n,p)^{29}Al$ - EAF-2003



UKAEA Fusion Working with Europe

$^{29}Si(n,p)^{29}Al$ - FENDL/A-2



Fit is worse

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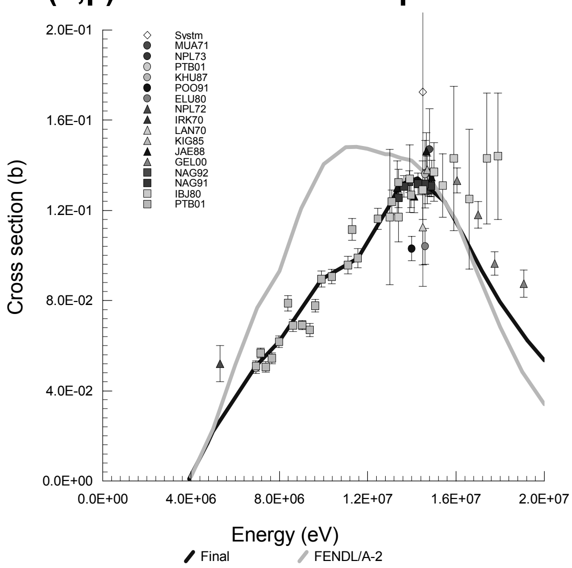
$^{29}Si(n,p)^{29}Al$ - one of the important reactions



EAF-2003 significantly better fit to differential data then FENDL/A-2

UKAEA Fusion Working with Europe

Appendix 3.6.

Comments on FENDL-2 from ITER-IT Naka Joint Work Site

T. Nishitani (on behalf of Hiro Iida)

Comments on FENDL-2 from ITER-IT Naka Joint Work Site

Presented byT. Nishitani

On behalf of Hiro IIDA,
ITER-International Team, Naka Joint Work Site

Consultants Meeting to "Maintain FENDL library for fusion applications"
10 -12 November 2003, IAEA Headquarters in Vienna, Austria.

Comments

(1) For neutron transport, FENDL2 looks to be enough.Even FENDL1 is not so bad and we saw practically no difference between the two in assessing neutron fluxes and heatings, etc.

(2) Dose rate calculation is different. At least with present available computer speed and capacity, we can not calculate accurate ITER dose rate with traditional method.

We need 1-step Monte Carlo method for the accurate calculation.

This method requires, detailed cross section data.

For example Mo data is available as a natural Mo in FENDL2, but not isotope wise.

The 1-step Monte Carlo requires, isotope wise data, namely data for Mo92, Mo94,Mo95,Mo96, Mo97, M098.

In this sense we need improvement of FENDL2.

Appendix 3.7.

Outline of Evaluations for JENDL-3.3 - fusion related issues

T. Nishitani (on behalf of Keiichi Shibata and Nuclear Data Center, JAERI)

Outline of Evaluations for JENDL-3.3 —— fusion related issues ——

Presented by Takeo NISHITANI
On behalf of
Keiichi SHIBATA and Nuclear Data Center
Japan Atomic Energy Research Institute

Consultants Meeting to "Maintain FENDL library for fusion applications"
10 -12 November 2003, IAEA Headquarters in Vienna, Austria.

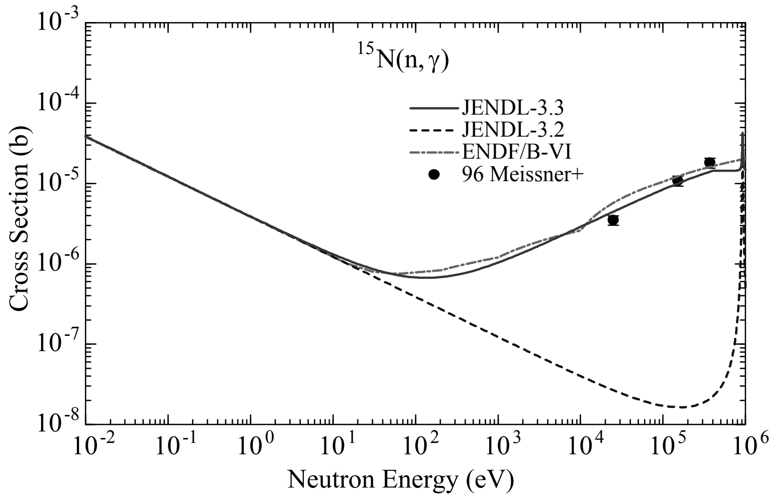
Problems of JENDL-3.2

- Neutron emission spectra for heavy nuclei
  - Incorrect shape
- Missing of direct and semi-direct capture cross sections in MeV region
- Inconsistency between elemental and isotopic data for medium-heavy nuclides

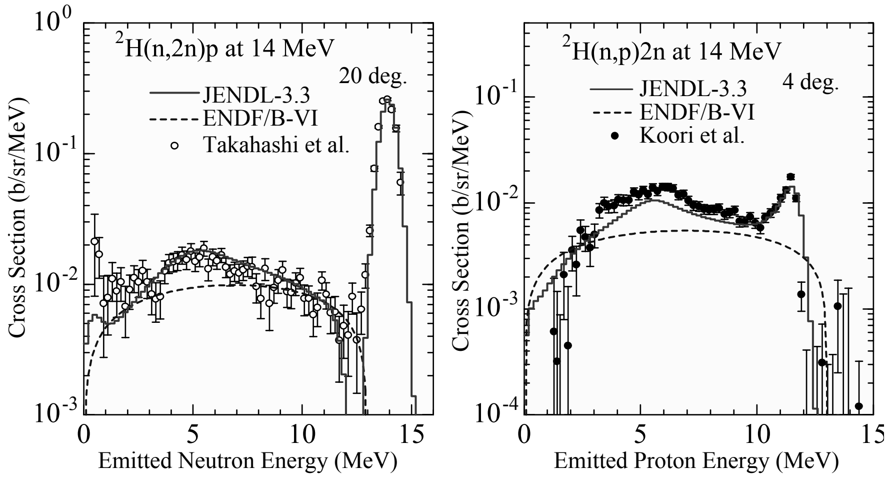
Evaluations for Light Nuclides

- Cross sections
  - ^{11}B Total, elastic
  - C Total, elastic
  - $^{14}N(n,p)$ 2200m/s 1.78 b 1.83 b
  - $^{15}N(n,\gamma)$ σ ~ 1/v + v
  - ^{16}O (n,2n), gamma-ray production
- Particle emission spectra
  - ^{2}H n,p spectra taken from JENDL/F-99
  - ^{9}Be (n,2n)2α revised above 16 MeV

Radiative Capture Cross Sections of ^{15}N



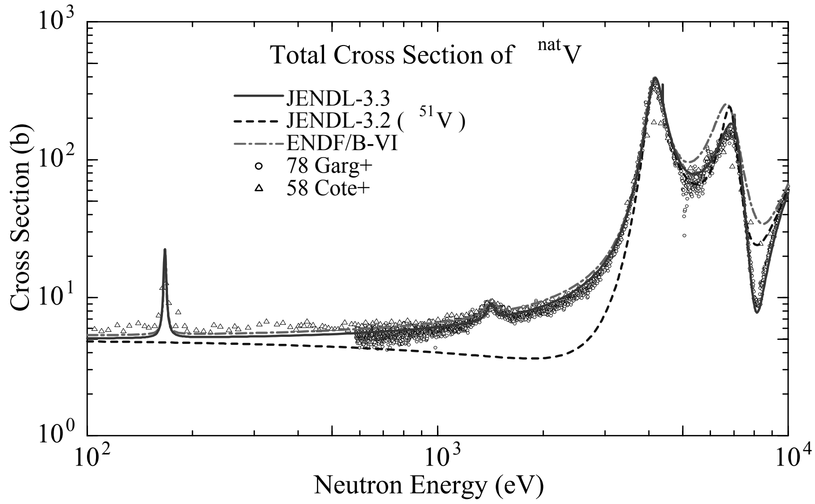
Neutron and Proton Spectra from ^{2}H(n,2n)p



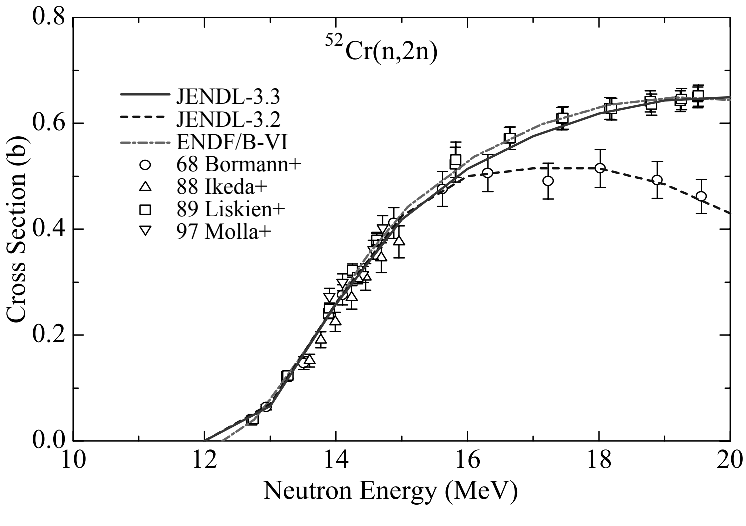
Evaluations for Medium-heavy Nuclides

- Resonance parameters
  - Ti, V, Cr, Fe, Co, Ni, W
- Isotopic evaluations
  - Mg, Si, S, Cl, K, Ca, Ti, Cr, Fe, Ni, Cu, Ga, Ge, Zr, Mo, Ag, Cd, Sb, Eu, Hf, W, Pb (no natural elements)
  - Consider measured elemental data, e.g., total cross section
- Threshold reactions nuclear model calc.
- Neutron spectra JENDL/F-99
  - Double differential form → $d^2\sigma(E_n',\theta_n)/dE_n'd\theta_n$
  - Interpolation INT=2 INT=22
- New evaluations Hg、Er

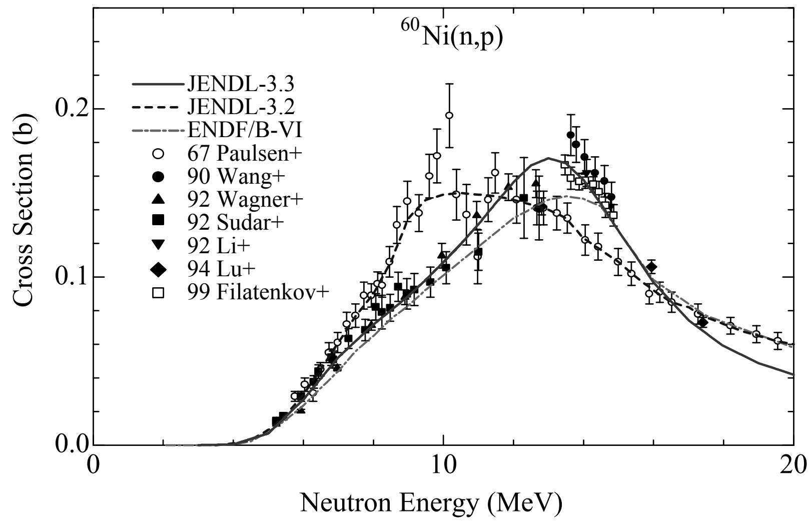
Total Cross Section of V



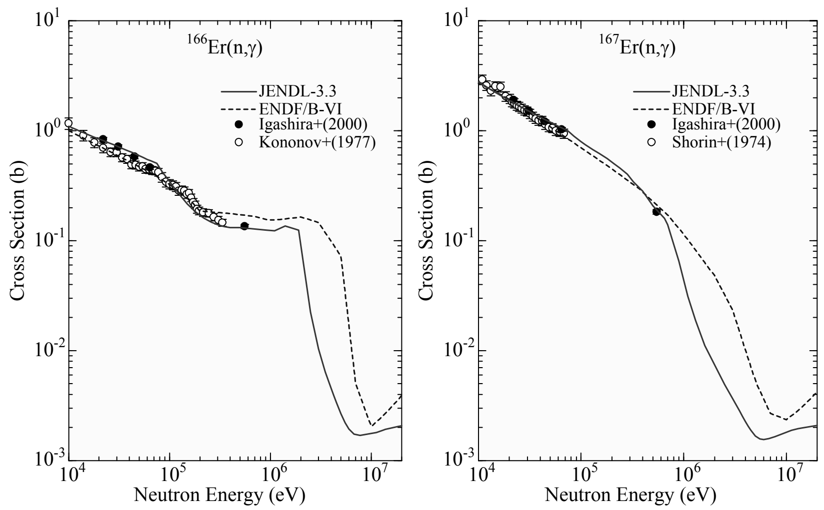
^{52}Cr (n,2n) Reaction Cross Section



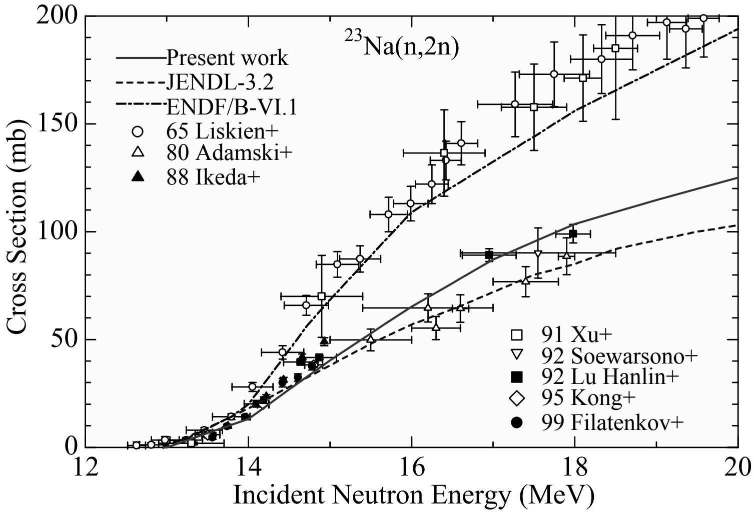
$^{60}Ni(n,p)$ Reaction Cross Section



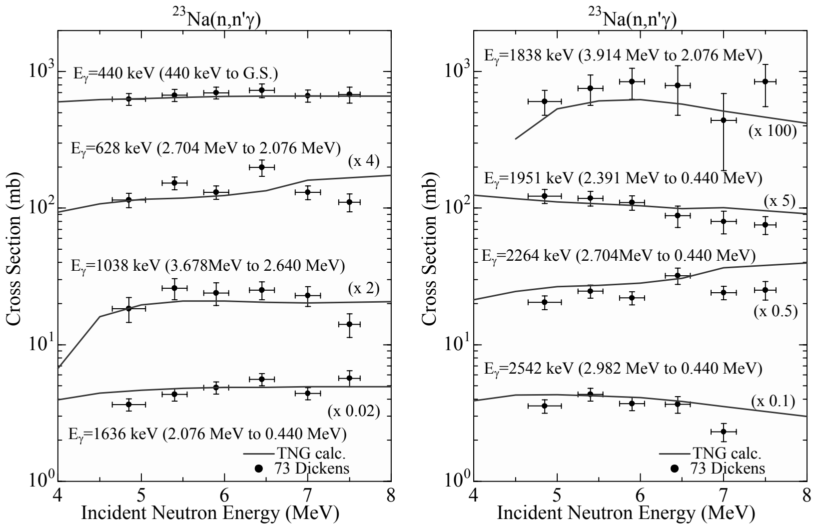
Radiative Capture Cross Sections of $^{166,167}Er$



^{23}Na(n,2n) Reaction Cross Section



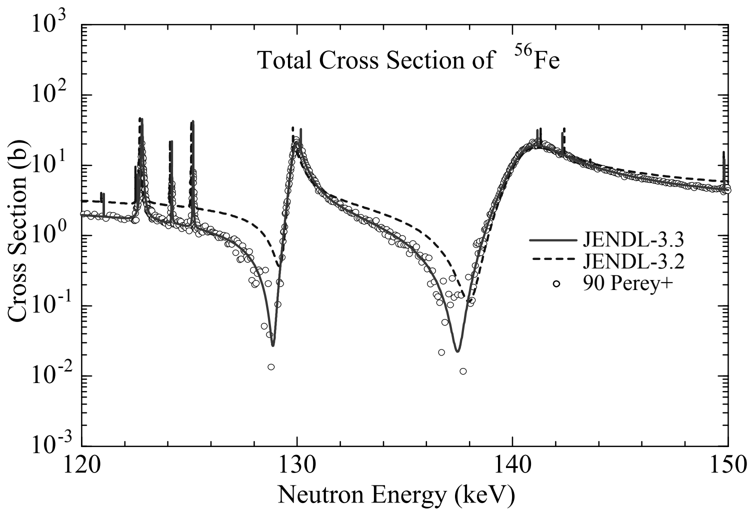
^{23}Na(n,n'γ) Cross Sections



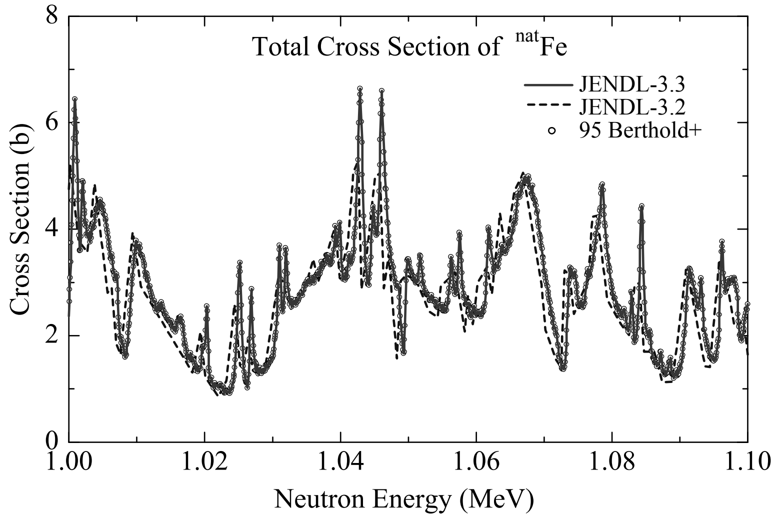
Revision of Iron Data

- Resonance parameters （Reich-Moore ）
  - ^{54}Fe ENDF/B-VI
  - ^{56}Fe JEF-2.2
  - $^{57,58}Fe$ JENDL-3.2
  - Upper boundary ^{54}Fe: 250 keV ⇒ 700 keV
    ^{56}Fe: 250 keV ⇒ 850 keV
- Total cross section
  - ^{54}Fe isotopic measurements（Geel、ORNL）
  - $^{57,58}Fe$ optical model calc. = JENDL-3.2
  - ^{56}Fe elemental data （Geel、ORNL、NIST） - other isotopic contributions

Resolved Resonances for ^{56}Fe



Total Cross Section of Elemental Iron



Evaluations for Minor Actinides

- Update of Np-237, Pu-236, -238, Am, Cm

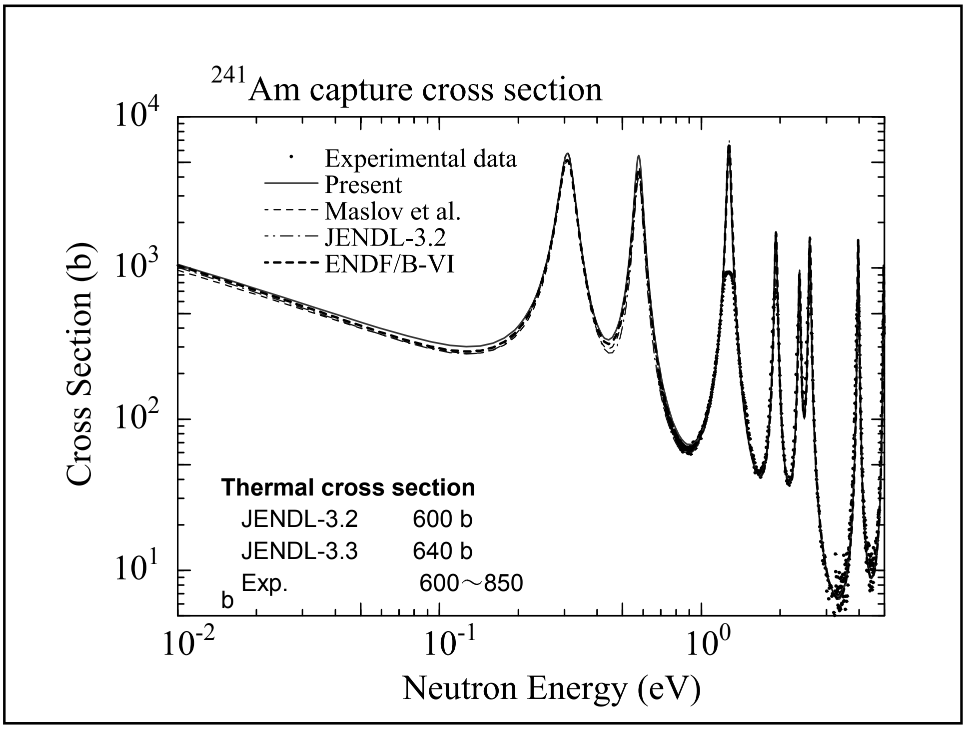
  Problems of JENDL-3.2

  no direct and pre-equilibrium processes

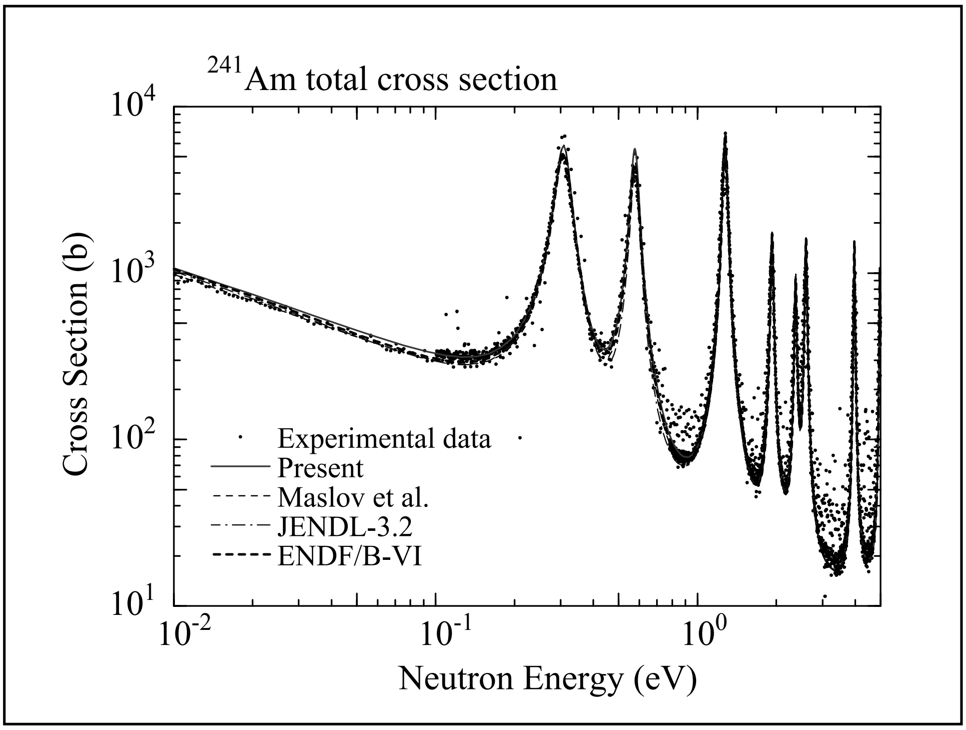
  ISTC Projects: Np-237, Pu-238, Am-241, -242g, -242m, -243, Cm-243, -245, -246

- New evaluations

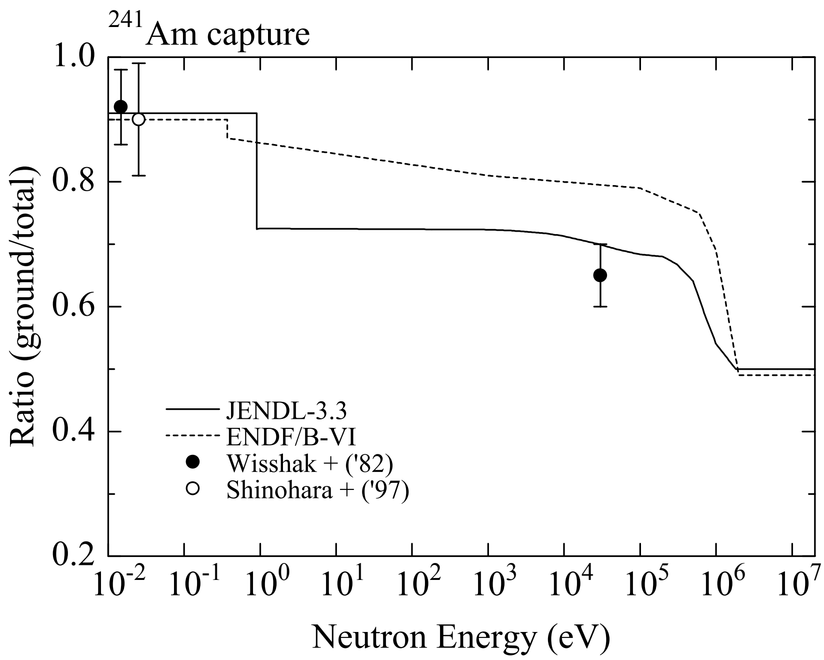
  Np-235, Pu-237, -244, -246, Cm-240, Bk-247



241Am capture cross section
Experimental data
Present
Maslov et al.
JENDL-3.2
ENDF/B-VI
Thermal cross section
JENDL-3.2 600 b
JENDL-3.3 640 b
Exp. 600～850 b
Cross Section (b)
Neutron Energy (eV)



241Am total cross section
Experimental data
Present
Maslov et al.
JENDL-3.2
ENDF/B-VI
Cross Section (b)
Neutron Energy (eV)

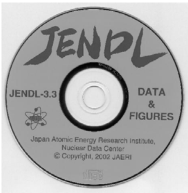


Covariance Data

- Uncertainties in nuclear data ⇨ uncertainties in design calculations
- JENDL-3.2 Covariance File （2002）
  - Measurements, model calculations
  - Cov. of RRP ⇨ cov. of measured average cross sections
- Covariances of fission cross sections for major actinides above 30 keV
  - Simultaneous evaluation for JENDL-3.3
- 20 nuclides in JENDL-3.3
  - ^{1}H、^{10}B、^{11}B、^{16}O、^{23}Na、^{48}Ti、V、^{52}Cr、^{55}Mn、^{56}Fe、^{59}Co、^{58}Ni、^{60}Ni、^{90}Zr、^{233}U、^{235}U、^{238}U、^{239}Pu、^{240}Pu、^{241}Pu

Concluding Remarks

- Evaluation and compilation for JENDL-3.3
  - Problems of JENDL-3.2 were resolved.
  - Gamma-ray production, DDX, covariance data
- The library was released on the 10th of May 2002.
  - http://wwwndc.tokai.jaeri.go.jp
  - CD-ROM, Graphs & Tables



Appendix 3.8.

Re-Analysis of Integral Tests using MCNP-4c with JENDL 3.3, -3.2 and FENDL2 for Fusion Related Materials

T. Nishitani, F. Maekawa and M. Wada

Re-Analysis of Integral Tests using MCNP-4c with JENDL 3.3, -3.2 and FENDL-2 for Fusion Related Materials

T. Nishitani, F. Maekawa and M. Wada

Japan Atomic Energy Research Institute,
Tokai-mura, Ibaraki 319-1195, Japan

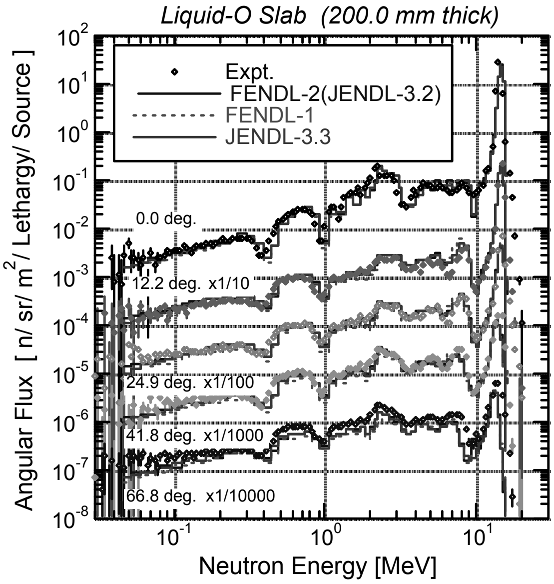
Consultants Meeting to "Maintain FENDL library for fusion applications"
10 -12 November 2003, IAEA Headquarters in Vienna, Austria.

Contents

- Oxygen (Angular neutron spectrum)
- Iron (Neutron spectrum)
- Copper (Neutron spectrum)
- Vanadium
  (Neutron and Secondary γ-ray spectra)
- Tungsten(Neutron spectrum)
- $LiAlO_3$ (Secondary γ-ray spectra)
- SiC (Neutron spectra)

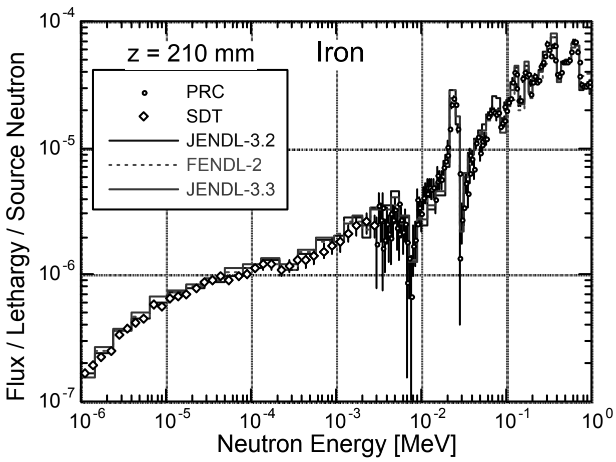
Oxygen (angular neutron spectrum)

- **There is not significant difference between JENDL-3.3 and FENDL-2 (JENDL-3.2)**



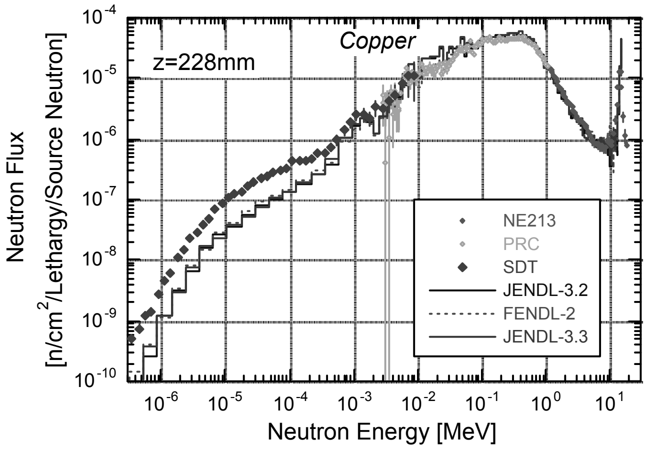
Iron (neutron spectrum)

- **Results of JENDL-3.3 agrees well to the experimental data compared with JENDL-3.2 and FENDL-2 in the energy range 24 keV - 1 MeV.**
- **JENDL-3.3 overestimates slightly in the energy range 1 - 1000 eV.**



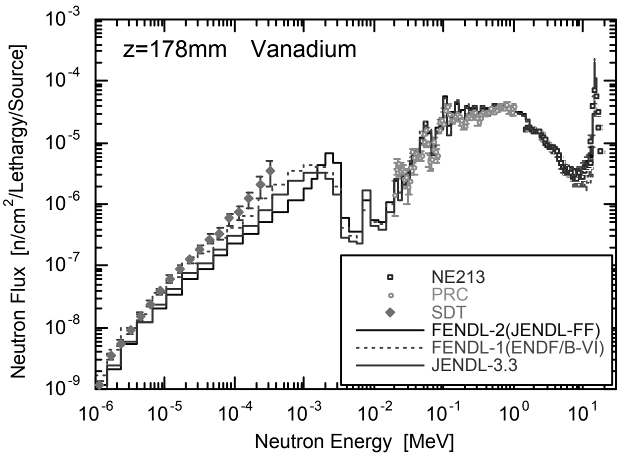
Copper (neutron spectrum)

- **JENDL-3.3 overestimates the experiment data in the energy range 100 - 500 KeV.**
- **JENDL-3.3 , JENDL-3.2 and FENDL-2 underestimate the experiment data measured with the slowing down technique below 1 keV.**



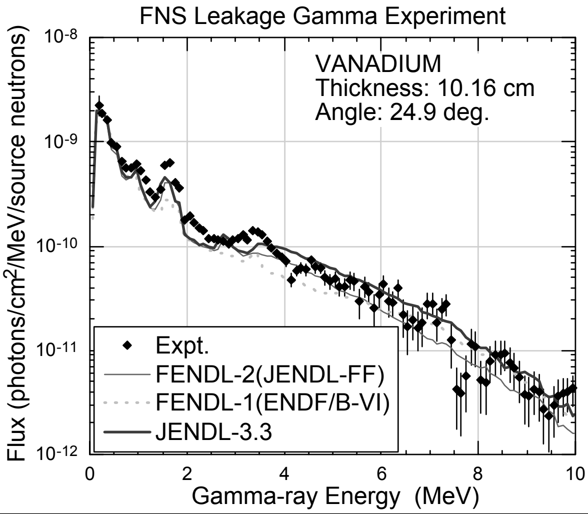
Vanadium (neutron spectrum)

- **A good agreement is obtained above 20 keV.**
- **JENDL-3.3 underestimates the neutron flux below 1 keV, but has improvement compared with FENDL-2.**



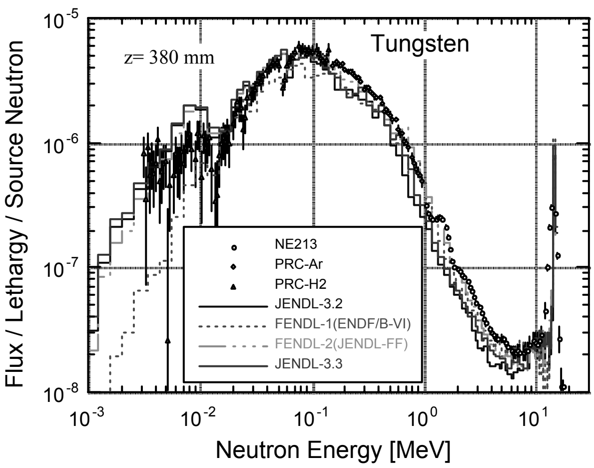
Vanadium (secondary γ-ray spectrum)

- **JENDL-3.3 agrees well to the experiment compared with JENDL-3.2 and FENDL-2 in the energy range 4 - 10 MeV.**



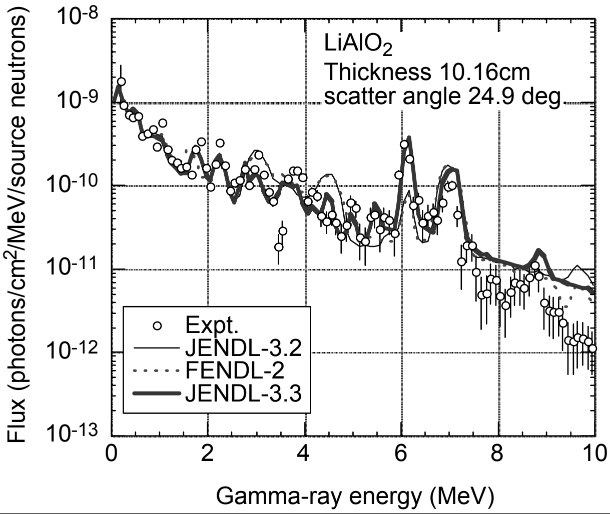
Tungsten (neutron spectrum)

- **JENDL-3.3 underestimates slightly the experiment data in the energy range above 150 keV, and has improvement compared with JENDL-3.2 and FENDL-2.**



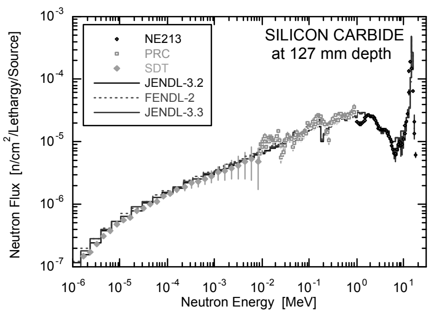
$LiAlO_3$ (secondary γ-ray spectrum)

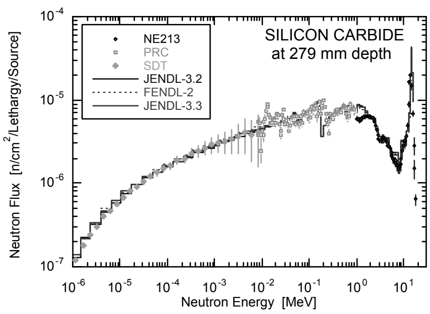
- **JENDL-3.3 and FENDL-2 agree well to the experiment data compared with JENDL-3.2.**



SiC (neutron spectra)

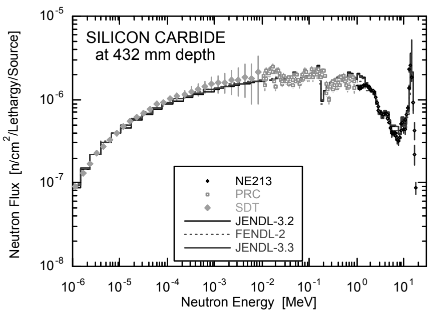
- **Generally JENDL-3.3, -3.2 and FENDL-2 agree well to the experiment data.**
- **These overestimate experiment data in the energy range 1-2 MeV and 4-7 MeV.**

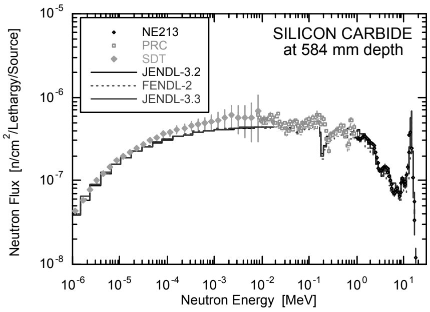




SiC (neutron spectra) II

- **JENDL-3.3, -3.2 and FENDL-2 underestimate the experiment data measured with Slowing Down Technique in the energy range 100 eV - 10 keV.**





Summary

- **Oxygen (Angular neutron spectrum)**
  There is not significant difference between JENDL-3.3 and FENDL-2 (JENDL-3.2).
- **Iron (Neutron spectrum)**
  JENDL-3.3 agrees well to the experiment compared with JENDL-3.2 and FENDL-2 in the energy range 24 keV - 1 MeV.
- **Copper (Neutron spectrum)**
  JENDL-3.3 overestimates the experiment data in the energy range 100 - 500 KeV.
- **Vanadium**
  JENDL-3.3 has improvement compared with FENDL-2 for neutron and secondary γ-ray spectra.
- **Tungsten(Neutron spectrum)**
  JENDL-3.3 has improvement compared with FENDL-2 above 150 keV.
- **$LiAlO_3$ (Secondary γ-ray spectra) and SiC (Neutron spectra)**
  There is not significant difference between JENDL-3.3 and FENDL-2.

Appendix 3.9.

FENDL-2 library for Fusion Applications – Status and Future Developments

Andrej Trkov

International Atomic Energy Agency

FENDL-2 Library for Fusion Applications – Status and Future Developments

Andrej Trkov

10-12 November 2003

IAEA Headquarters, Vienna, Austria

Topics for discussion

- Evaluated nuclear data files
  - Transport
  - Dosimetry
  - Activation
- Ace library for Monte Carlo
- Group library

Evaluated Nuclear Data Files

Options to consider:

- Start selection process from scratch?
- Remove known deficiencies in existing files and restrict selection process to new materials?

Ace library for Monte Carlo

Options:

- Use same processing options
- Upgrade input options including new features of MCNP-4C +

Doppler broadening of Ace files

- The Agency supported the development of SIGACE code for the purpose, alleviating the need for multi-temperature libraries.

Group Library

Options

- Use the same specifications?
- Refine group structure?

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Online: TELNET or FTP: iaeand.iaea.org
username: IAEANDS for interactive Nuclear Data Information System
usernames: ANONYMOUS for FTP file transfer;
FENDL2 for FTP file transfer of FENDL-2.0;
RIPL for FTP file transfer of RIPL.
NDSONL for FTP access to files sent to NDIS “open” area.
Web: http://www-nds.iaea.org