# International Atomic Energy Agency: Dedicated Nuclear Databases

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**Abstract**

Well-characterised nuclear data are important to the confident development and exploitation of a wide range of nuclear programmes within Member States of the International Atomic Energy Agency (IAEA). Under these circumstances, highly focused databases are defined and assembled by recognised specialists by means of various data development projects organised by the IAEA Nuclear Data Section. Recent and on-going data development work is described that encompasses such applications as improved design and operational studies for nuclear power plant and associated facilities, nuclear medicine, analytical techniques, and basic nuclear physics research.

*KEYWORDS: cross-section standards, dosimetry, fusion, Th-U, nuclear databases*

## 1. Introduction

The production and maintenance of various forms of nuclear database are major features of the work undertaken under the auspices of the Nuclear Data Section (NDS) of the International Atomic Energy Agency (IAEA). NDS staff are always striving to identify the intermediate- and long-term data needs of a wide range of user communities that includes medical physicists, analytical scientists, design engineers for power reactors and fuel handling/reprocessing, and nuclear physicists undertaking basic research studies. A number of recently completed and on-going projects are described below that highlight the important function of the IAEA Nuclear Data Section in catalyzing and organising in a timely manner the evolution of specific nuclear databases of direct importance to users around the world. Consideration is given below to the following recently completed and on-going activities: fission reactor dosimetry – IRDF-2002; fusion studies – FENDL-2.1; Th-U fuel cycle; cross-section standards.

## 2. International Reactor Dosimetry File, IRDF-2002

The International Reactor Dosimetry File (IRDF-90, version 2) was released in 1993 [1], and has inevitably become dated. Therefore, during the course of 2002-04, a new file was prepared by a team of experts consulted by the IAEA NDS. The primary aim of this project was to prepare an updated, standardised and benchmarked cross-section library that contained recommended data for reactor dosimetry reactions suitable for service life assessments of reactor pressure vessels in nuclear power plants and other neutron metrology applications. This new library contains the best quality data available for reactor dosimetry calculations at the time of preparation (closing month of data collection was December 2003).

IRDF-2002 includes the following cross sections and other nuclear data:

(a) Multi-group data for metrology applications
- cross-section data for 66 neutron activation (and fission) reactions, and uncertainties in the form of covariance information;
- cross sections of three cover materials (B, Cd and Gd), without uncertainties;
- radiation damage cross sections:
  Fe dpa cross section (ASTM standard E693-1),
  dpa cross section for a special steel composition (Euratom),
  dpa cross sections for Cr and Ni (IRDF-90), for Si (ASTM standard E722-94), and for GaAs displacement (ASTM standard E722-94).

(b) Point-wise data
- all cross sections listed above, except radiation damage cross sections;
- total and elastic cross sections for all nuclides with neutron-capture and fission reactions in the library, including uncertainty information.

(c) Other nuclear data based on the ENSDF library [2]
- nuclear decay parameters for all reaction product nuclei of relevance;
- isotopic abundances for all target nuclides of interest.

Point-wise cross sections are given in ENDF-6 format, while multi-group data are provided in SAND II extended 640 energy group structure as two forms – ENDF-6 format (without decay data), and in a simplified form of ENDF-6 for backward compatibility with IRDF-90.

Analyses in the evolution of IRDF-2002 included checking the content and formats of the cross-section and uncertainty data in the files of interest (IRDF-90 [1], JENDL/D-99 [3], RRDF-98 [4], and the most recent evaluations in ENDF/B-VI (Release 8), JEFF-3.0 and CENDL-2 [5]). Cross-section data were characterized numerically by spectrum-averaging cross-section values for three theoretical spectrum functions (Maxwellian thermal, 1/E and Watt fission spectrum). Integral spectrum characteristics and uncertainty information were also compared and analysed. New cross-section evaluations with associated uncertainties were carried out at IPPE, Obninsk, for inclusion in IRDF-2002 [4]: $^{27}$Al(n,p), $^{58}$Ni(n,p), $^{103}$Rh (n,n'), $^{115}$In(n,n'), $^{139}$La(n,$\gamma$), $^{186}$W(n,$\gamma$), $^{204}$Pb(n,n') and $^{237}$Np(n,f). The contents of IRDF-2002 and the various data sources are listed in Table 1. Selection of the desired cross sections was based on the following criteria:

- comparison of the integral values of the candidate cross sections with the corresponding experimental data in the four standard neutron fields (thermal Maxwellian, 1/E slowing down, $^{252}$Cf and 14-MeV);
- quality of the uncertainty data;
- consistency of the data.

Table 1. Contents of IRDF-2002, and sources of the data.

| Reaction | Selected data source | Reaction | Selected data source |
|---|---|---|---|
| $^{6}$Li(n,t)$^{4}$He | IRDF-90[a] | $^{65}$Cu(n,2n)$^{64}$Cu | IRDF-90[a] |
| $^{10}$B(n,α)$^{7}$Li | IRDF-90 | $^{64}$Zn(n,p)$^{64}$Cu | IRDF-90 |
| $^{19}$F(n,2n)$^{18}$F | RRDF-98 (u) | $^{75}$As(n,2n)$^{74}$As | RRDF-98 (u) |
| $^{23}$Na(n,γ)$^{24}$Na[b] | IRDF-90[a] | $^{89}$Y(n,2n)$^{88}$Y | JENDL/D-99 |
| $^{23}$Na(n,2n)$^{22}$Na | JENDL/D-99 (u) | $^{90}$Zr(n,2n)$^{89}$Zr | IRDF-90 |
| $^{24}$Mg(n,p)$^{24}$Na | IRDF-90 | $^{93}$Nb(n,2n)$^{92}$Nb$^{m}$ | RRDF-98 |
| $^{27}$Al(n,p)$^{27}$Mg | RRDF-98 (n) | $^{93}$Nb(n,n′)$^{93}$Nb$^{m}$ | RRDF-98 |
| $^{27}$Al(n,α)$^{24}$Na | IRDF-90 | $^{93}$Nb(n,γ)$^{94}$Nb[b] | IRDF-90[a] |
| $^{31}$P(n,p)$^{31}$Si | IRDF-90 | $^{103}$Rh(n,n′)$^{103}$Rh$^{m}$ | RRDF-98 (n) |
| $^{32}$S(n,p)$^{32}$P | IRDF-90 | $^{109}$Ag(n,γ)$^{110}$Ag$^{m}$ | IRDF-90 |
| $^{45}$Sc(n,γ)$^{46}$Sc | IRDF-90 | $^{115}$In(n,2n)$^{114}$In$^{m}$ | IRDF-90[a] |
| $^{46}$Ti(n,2n)$^{45}$Ti | RRDF-98 (u) | $^{115}$In(n,n′)$^{115}$In$^{m}$ | RRDF-98 (n) |
| $^{46}$Ti(n,p)$^{46}$Sc | RRDF-98 (u) | $^{115}$In(n,γ)$^{116}$In$^{mb}$ | ENDF/B-VI |
| $^{47}$Ti(n,x)$^{46}$Sc[c] | RRDF-98 (u) | $^{127}$I(n,2n)$^{126}$I | IRDF-90 |
| $^{47}$Ti(n,p)$^{47}$Sc | IRDF-90 | $^{139}$La(n,γ)$^{140}$La | RRDF-98 (n) |
| $^{48}$Ti(n,x)$^{47}$Sc[c] | RRDF-98 (u) | $^{141}$Pr(n,2n)$^{140}$Pr | RRDF-98 (u) |
| $^{48}$Ti(n,p)$^{48}$Sc | RRDF-98 (u) | $^{169}$Tm(n,2n)$^{168}$Tm | JENDL/D-99 |
| $^{49}$Ti(n,x)$^{48}$Sc[c] | RRDF-98 (u) | $^{181}$Ta(n,γ)$^{182}$Ta[b] | JENDL/D-99 |
| $^{51}$V(n,α)$^{48}$Sc | RRDF-98 (u) | $^{186}$W(n,γ)$^{187}$W | RRDF-98 (n) |
| $^{52}$Cr(n,2n)$^{51}$Cr | IRDF-90 | $^{197}$Au(n,2n)$^{196}$Au | IRDF-90 |
| $^{55}$Mn(n,γ)$^{56}$Mn | IRDF-90[a] | $^{197}$Au(n,γ)$^{198}$Au | IRDF-90[a] |
| $^{54}$Fe(n,2n)$^{53}$Fe | RRDF-98 (u) | $^{199}$Hg(n,n′)$^{199}$Hg$^{m}$ | JENDL/D-99 (u) |
| $^{54}$Fe(n,α)$^{51}$Cr | RRDF-98 (u) | $^{204}$Pb(n,n′)$^{204}$Pb$^{m}$ | RRDF-98 (n) |
| $^{54}$Fe(n,p)$^{54}$Mn | IRDF-90[a] | $^{232}$Th(n,γ)$^{233}$Th [b] | IRDF-90 |
| $^{56}$Fe(n,p)$^{56}$Mn | RRDF-98 (u) | $^{232}$Th(n,f) | IRDF-90 |
| $^{58}$Fe(n,γ)$^{59}$Fe | JENDL/D-99 (u) | $^{235}$U(n,f) | IRDF-90 |
| $^{59}$Co(n,2n)$^{58}$Co | IRDF-90 | $^{238}$U(n,f) | JENDL/D-99 |
| $^{59}$Co(n,α)$^{56}$Mn | RRDF-98 (u) | $^{238}$U(n,γ)$^{239}$U | IRDF-90[a] |
| $^{59}$Co(n,γ)$^{60}$Co | IRDF-90[a] | $^{237}$Np(n,f) | RRDF-98 (n) |
| $^{58}$Ni(n,2n)$^{57}$Ni | JEFF 3.0 | $^{239}$Pu(n,f) | JENDL/D-99 |
| $^{58}$Ni(n,p)$^{58}$Co | RRDF-98 (n) | $^{241}$Am(n,f) | JENDL/D-99 |
| $^{60}$Ni(n,p)$^{60}$Co | ENDF/B-VI | $^{nat}$B(n,x)[d] | ENDF/B-VI |
| $^{63}$Cu(n,2n)$^{62}$Cu | ENDF/B-VI | $^{nat}$Cd(n,x)[d] | ENDF/B-VI |
| $^{63}$Cu(n,γ)$^{64}$Cu | IRDF-90[a] | $^{nat}$Gd(n,x)[d] | ENDF/B-VI |
| $^{63}$Cu(n,α)$^{60}$Co | RRDF-98 (u) | | |

[a] ENDF/B-VI Release 8.
[b] diagonal covariance matrix.
[c] (n,x) is the sum of the reactions (n,np) + (n,pn) + (n,d).
[d] cover material - no covariance information available.
(u) updated data; (n) new data.

Some shortcomings in the chosen cross-section data were noted, and should be resolved before any further comprehensive review of the library is undertaken. IRDF-2002 is distributed by the IAEA NDS, and can be downloaded from the IAEA NDS web site: http://www-nds.iaea.org/irdf2002/

## 3. FENDL-2.1

The Fusion Evaluated Nuclear Data Library (FENDL) has been validated and extensively tested for thermonuclear fusion applications, and used in the design development of ITER. FENDL consists of the following sub-libraries:

- FENDL/A-2.0 – neutron activation cross sections for 13006 reactions on 739 targets, ranging from $^{1}$H to $^{248}$Cm at incident energies up to 20 MeV (point-wise and processed data);
- FENDL/D-2.0 – decay type, decay energy and half-life for 1867 nuclides and isomers (point-wise and processed data);
- FENDL/DS-2.0 – use of IRDF-2002 is recommended (see Section 2);
- FENDL/C-2.0 – charged-particle cross sections for fusion reactions: $^{2}$H(d,n)$^{3}$He; $^{2}$H(d,p)$^{3}$H; $^{3}$He(d,p)$^{4}$He; $^{3}$H(t,2n)$^{4}$He and $^{3}$H(d,n)$^{4}$He (point-wise and processed data);
- FENDL/E-2.1 – basic nuclear data (neutron-nucleus interaction, including photon production and photon-atom interaction cross sections) for 71 materials (point-wise data); processed sub-libraries have also been prepared for use in discrete-ordinate and Monte-Carlo transport calculations (FENDL/MG-2.1 and FENDL/MC-2.1);
- collection of benchmarks for FENDL validation.

Significant effort was expended in 2004 to improve and up-date FENDL/E-2.0 (converted to FENDL/E-2.1) and associated sub-libraries, based on the detailed recommendations of an IAEA Consultants' Meeting held in Vienna, 10 – 12 November 2003 [6]. Isotopic evaluations for Cl, Ti, Mo and W increased the number of materials from the 57 of FENDL/E-2.0 to 71 in FENDL/E-2.1. Processing was performed by using NJOY-99.90 [7], and the resulting files are available in ACE and MATXS formats for MCNP and multi-group transport calculations, respectively. A new feature of the FENDL-2.1 package is the availability of the ACEDOP package which allows Döppler broadening of the cross sections in the ACE files, except for the energy region described by unresolved resonance representation in the original ENDF-formatted files [8].

The FENDL-2.1 package includes the following data files and information:

- FENDL/E-2.1 neutron and photo-atomic interaction data files;
- FENDL/MC-2.1 continuous energy data files for MCNP calculations;
- FENDL/MG-2.1 coupled neutron-photon multi-group data library for transport calculations;
- NJOY inputs for generation of FENDL-2.1;
- auxiliary programs and MSDOS/WINDOWS batch procedures used in the generation and verification of the FENDL-2.1 transport libraries;
- ACEDOP code package for Döppler broadening of ACE-formatted files;
- documentation (IAEA report INDC(NDS)-467 [9]).

The contents of FENDL/E-2.1 and the data sources are listed in Table 2. All of the work to produce the FENDL-2.1 package is described in Ref. [9], and further information and the various databases can be downloaded from: http://www-nds.iaea.org/fendl21/index.html

Table 2. Contents of FENDL/E-2.1, and sources of the data.

| No. | Material | FENDL/E-2.0 | FENDL/E-2.1 | Comments |
|---|---|---|---|---|
| 1 | 1-H-1 | ENDF/B-VI mod 1 | JENDL-3.3 | |
| 2 | 1-H-2 | BROND-2.1 | ENDF/B-VI.8 mod 4 | |
| 3 | 1-H-3 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 0 | No photon production |
| 4 | 2-He-3 | ENDF/B-VI mod 1 | JENDL-3.3 | Photon production added |
| 5 | 2-He-4 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | Minor revisions |
| 6 | 3-Li-6 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 0 | |
| 7 | 3-Li-7 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 0 | |
| 8 | 4-Be-9 | JENDL-FF | ENDF/B-VI.8 mod 8 | |
| 9 | 5-B-10 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 1 | |
| 10 | 5-B-11 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 2 | Minor revisions |
| 11 | 6-C-12 | JENDL-FF | JENDL-FF | MF = 14 corrected |
| 12 | 7-N-14 | JENDL-FF | JENDL-FF | |
| 13 | 7-N-15 | BROND-2.1 | BROND-2.1 | |
| 14 | 8-O-16 | JENDL-FF | ENDF/B-VI.8 mod 3 | |
| 15 | 9-F-19 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | Gamma spectra modified |
| 16 | 11-Na-23 | JENDL-3.1 | JENDL-3.3 | Covariance data |
| 17 | 12-Mg-nat | JENDL-3.1 | JENDL-3.2 | |
| 18 | 13-Al-27 | EFF-3.0 | JEFF-3.0 (EFF-3.0) | |
| 19 | 14-Si-28 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 2 | MF = 14 corrected |
| 20 | 14-Si-29 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 3 | |
| 21 | 14-Si-30 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 2 | MF = 14 corrected |
| 22 | 15-P-31 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 0 | |
| 23 | 16-S-nat | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | Gamma spectra modified |
| 24 | 17-Cl-35 | ENDF/B-VI mod 0 17-Cl-nat | ENDF/B-VI.8 mod 1 | MF = 14 corrected |
| 25 | 17-Cl-37 | | ENDF/B-VI.8 mod 1 | Minor correction |
| 26 | 19-K-nat | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | Gamma spectra modified |
| 27 | 20-Ca-nat | JENDL-3.1 | JENDL-3.2 | MF = 14 corrected |
| 28 | 22-Ti-46 | JENDL-3.1 22-Ti-nat | JENDL-3.3 | |
| 29 | 22-Ti-47 | | JENDL-3.3 | |
| 30 | 22-Ti-48 | | JENDL-3.3 | |
| 31 | 22-Ti-49 | | JENDL-3.3 | |
| 32 | 22-Ti-50 | | JENDL-3.3 | |
| 33 | 23-V-nat | JENDL-FF | JENDL-3.3 | FENDL/E-2.0: 23-V-51 |
| 34 | 24-Cr-50 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 35 | 24-Cr-52 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 4 | Gamma spectra modified |
| 36 | 24-Cr-53 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 4 | Gamma spectra modified |
| 37 | 24-Cr-54 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 38 | 25-Mn-55 | ENDF/B-VI mod 1 | JENDL-3.3 | |
| 39 | 26-Fe-54 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 40 | 26-Fe-56 | EFF-3.0 | JEFF-3.0 (EFF-3.1) | |

Table 2. Contents of FENDL/E-2.1, and sources of the data (continued).

| No. | Material | FENDL/E-2.0 | FENDL/E-2.1 | Comments |
|---|---|---|---|---|
| 41 | 26-Fe-57 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 4 | Gamma spectra modified |
| 42 | 26-Fe-58 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 4 | Gamma spectra modified |
| 43 | 27-Co-59 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 1 | |
| 44 | 28-Ni-58 | ENDF/B-VI mod 1 | JEFF-3.0(EFF-3.0) | |
| 45 | 28-Ni-60 | ENDF/B-VI mod 1 | JEFF-3.0(EFF-3.0) | MF = 12 corrected MT51-91 |
| 46 | 28-Ni-61 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 47 | 28-Ni-62 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 48 | 28-Ni-64 | ENDF/B-VI mod 1 | ENDF/B-VI.8 mod 4 | Gamma spectra modified |
| 49 | 29-Cu-63 | ENDF/B-VI mod 2 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 50 | 29-Cu-65 | ENDF/B-VI mod 2 | ENDF/B-VI.8 mod 5 | Gamma spectra modified |
| 51 | 31-Ga-nat | JENDL-3.2 | JENDL-3.2 | No photon production data |
| 52 | 40-Zr-nat | JENDL-FF | JENDL-FF | |
| 53 | 41-Nb-93 | JENDL-FF | JENDL-FF | MF = 12 corrected MT51-91 |
| 54 | 42-Mo-92 | JENDL-FF 42-Mo-nat | JENDL-3.3 | |
| 55 | 42-Mo-94 | | JENDL-3.3 | |
| 56 | 42-Mo-95 | | JENDL-3.3 | |
| 57 | 42-Mo-96 | | JENDL-3.3 | |
| 58 | 42-Mo-97 | | JENDL-3.3 | |
| 59 | 42-Mo-98 | | JENDL-3.3 | |
| 60 | 42-Mo-100 | | JENDL-3.3 | |
| 61 | 50-Sn-nat | BROND-2.1 | BROND-2.1 | |
| 62 | 73-Ta-181 | JENDL-3.1 | JENDL-3.3 | |
| 63 | 74-W-182 | JENDL-FF 74-W-nat | ENDF/B-VI.8 mod 2 | |
| 64 | 74-W-183 | | ENDF/B-VI.8 mod 2 | |
| 65 | 74-W-184 | | ENDF/B-VI.8 mod 2 | |
| 66 | 74-W-186 | | ENDF/B-VI.8 mod 2 | |
| 67 | 79-Au-197 | ENDF/B-VI mod 1 | ENDF/B-VI mod 1 | |
| 68 | 82-Pb-206 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | |
| 69 | 82-Pb-207 | ENDF/B-VI mod 1 | ENDF/B-VI mod 1 | |
| 70 | 82-Pb-208 | ENDF/B-VI mod 0 | ENDF/B-VI.8 mod 1 | |
| 71 | 83-Bi-209 | JENDL-3.1 | ENDF/B-VI.8 mod 2 | |

## 4. Th-U fuel cycle

A majority of the past developments in nuclear technology have been towards uranium-plutonium thermal and fast reactors in order to improve the utilization of sources of natural uranium. Nevertheless, in recent years, new concepts for nuclear power production have been investigated to explore other means of satisfying the need for increased inherent safety, reducing the risk of fissile material proliferation, and addressing the problem of long-term radioactive waste disposal. The thorium-based nuclear fuel cycle offers many advantages:

- $^{232}$Th neutron capture yields $^{233}$U – highly efficient fuel that can be adopted to create the concept of a thermal-breeder reactor based on thorium fuel;
- long-lived higher actinides are the main source of long-term radioactive waste, and

their build-up is much smaller in thorium than uranium fuel;
- thorium fuel is more proliferation-resistant due to the resulting highly-radioactive constituents that can not be easily separated from the fuel by chemical means;
- world reserves of thorium are much larger than uranium reserves.

As a consequence of the factors listed above, there is a rising interest in innovative fuel cycle concepts based on thorium. Unfortunately, due to the previous lack of interest in the thorium fuel cycle, the quality of nuclear data for the relevant materials is significantly lower than for comparable materials in the uranium and mixed-oxide fuel cycles [10, 11]. Important experimental measurements of the cross sections of materials relevant to the Th-U fuel cycle have been reported recently – these data need to be evaluated, verified and validated on the basis of integral benchmarks.

There has long been a need to improve nuclear data for the Th-U fuel cycle, and an IAEA Coordinated Research Project was initiated in 2002 to undertake the necessary work:

- neutron cross-section data for $^{232}$Th, $^{231, 233}$Pa and $^{232, 233, 234, 236}$U;
- critical assessment of available experimental information, and renormalization to standard cross sections, if necessary;
- evaluation of experimental data, derivation of resonance parameters (when relevant), and completion of data by means of nuclear model calculations to produce a suitably comprehensive database in ENDF-6 format;
- verification of the formatted data;
- processing of the data into application libraries for validation against benchmark test cases.

This programme was extended to include the production of covariance data for some of the nuclides ($^{232}$Th and $^{231, 233}$Pa) to cover both the resonance and fast regions. Covariance information for the resonance region of $^{232}$Th was derived by Leal [12], while covariances for the fast neutron region of $^{232}$Th were independently calculated by three different groups for subsequent study and assessment. The results for the diagonal elements of the covariance matrix agree within 20%, while comparisons of the non-diagonal elements are still on-going. Preliminary covariance data are also available for the $^{231, 233}$Pa cross sections. This work is outlined on IAEA NDS web site: http://www-nds.iaea.org/Th-U/

## 5. Cross-section standards

Previous evaluations of the neutron cross-section standards were completed in 1987, and disseminated as NEANDC/INDC [13] and ENDF/B-VI [14] standards. A number of points emerged from these activities in the 1980s that are noteworthy:

- evaluations were based on R-matrix model fits of experimental data for reactions leading to the formation of $^{7}$Li and $^{11}$B compound nuclei;
- evaluations were based on the generalised non-model least-squares fit for the heavy-nuclide standards;
- evaluations for H(n,n), $^{3}$He(n,p) and C(n,n) were based on independent R-matrix analyses;
- evaluated uncertainties were considered to be so low as to be unusable by many cross-section users (approximately a factor of 2 too low for the non-model heavy-

nuclide evaluations, and a factor of 4 to 10 too low for the R-matrix evaluations);

- strongly correlated discrepant data produced a systematic bias in the evaluated data relative to the expected "true" value (commonly referred to as "Peelle's Pertinent Puzzle (PPP)" [15]).

Under these particularly difficult circumstances, an IAEA Coordinated Research Project was launched in 2002 entitled "Improvement of the standard cross sections for light elements", that was designed to explore the significant uncertainty reduction observed in R-matrix model fits, improve the methodology for the determination of the covariance matrices of the uncertainties, and prepare new recommendations for the light-element cross sections. This programme was sensibly extended to the cross-sections standards for the heavy nuclides in 2003. A list of the neutron cross-section standards and the neutron energy ranges over which they have been evaluated are given in Table 3.

Table 3. Neutron cross-section standards.

| **Reaction** | **Neutron Energy Range** | |
|---|---|---|
| | **1987** | **2002-2006** |
| H(n,n) | 1 keV to 20 MeV | 1 keV to 20 MeV |
| $^{3}$He(n,p) | 0.0253 eV to 50 keV | 0.0253 eV to 50 keV (1987) |
| $^{6}$Li(n,t) | 0.0253 eV to 1 MeV | 0.0253 eV to 1 MeV |
| $^{10}$B(n,$\alpha$) | 0.0253 eV to 250 keV | 0.0253 eV to 1 MeV |
| $^{10}$B(n,$\alpha_1\gamma$) | 0.0253 eV to 250 keV | 0.0253 eV to 1 MeV |
| C(n,n) | up to 1.8 MeV | up to 1.8 MeV (1987) |
| Au(n,$\gamma$) | 0.0253 eV, and 0.2 to 2.5 MeV | 0.0253 eV, and 0.2 to 2.5 MeV |
| $^{235}$U(n,f) | 0.0253 eV, and 0.15 to 20 MeV | 0.0253 eV, and 0.15 to 200 MeV |
| $^{238}$U(n,f) | threshold to 20 MeV | 2 to 200 MeV |

The R-matrix model least-squares fit of the experimental data was used in the evaluation of the $^{6}$Li(n,t), $^{10}$B(n,$\alpha$) and $^{10}$B(n,$\alpha_1\gamma$) standard cross sections, while the non-model least-squares fit was applied to evaluations of the $^{197}$Au(n,$\gamma$), $^{235}$U(n,f) and $^{238}$U(n,f) reactions. More than 400 experimental data sets were used in the combined fits of all of these standards, including absolute, relative and shape cross-section measurements. Cross sections for other reactions and the results of integral measurements were also taken into account for the combined fits [16]. Great attention was paid to the problems that remained unsolved from the earlier evaluation exercise:

- ambiguities and reductions in the uncertainties of the R-matrix model fits;
- possible biases in the evaluated cross sections arising from PPP;
- analysis and treatment of discrepant experimental data.

The newly-recommended cross-section standards data exhibit general increases (particularly the fission cross sections for neutron energies above 14 MeV) that are supported by criticality benchmark experiments. Documentation is in preparation, and these data files can be downloaded from: http://www-nds.iaea.org/standards/

## 6. Conclusions

A number of well-planned and highly-focused initiatives to produce and improve specific nuclear data are supported by the IAEA Nuclear Data Section and their staff, as outlined above for a selected number of recent and on-going projects with a strong power-related theme. Other IAEA NDS projects of note include:

1. preparation of a Reference Input Parameter Library (RIPL-3) for nuclear model codes;
2. cross-section production data for therapeutic radioisotopes;
3. establishment of a reference database for neutron activation analysis;
4. establishment of a reference nuclear database for ion beam analysis;
5. updated decay data library for actinides;
6. atomic and molecular data for fusion plasma modelling;
7. atomic data for heavy-element impurities in fusion reactors.

One extremely important feature of all of these studies is the essential participation of experts from around the world. Each package of work has a definite beginning and end, with the primary aim of producing accessible data files for users in Member States of the IAEA – for further information see: http://www-nds.iaea.org

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