N JOY / FISCHER-01

Experiences in Processing New High Energy Neutron Cross-Section Data with NJOY/ACER for the IFMIF Neutron Source

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Overview

I. IFMIF neutron source
II. Data evaluation
III. Data processing
IV. MCNP applications to IFMIF

IFMIF - International Fusion Materials Irradiation Facility

Mission

To provide an accelerator-based, deuterium-lithium (d-Li) neutron source to produce high energy neutrons at sufficient energy, intensity and irradiation volume to test samples of candidate materials up to a full lifetime of anticipated use in fusion energy reactors.

- Conceptual Design Activity (CDA) study 1995-96 to provide a reference design and a project basis

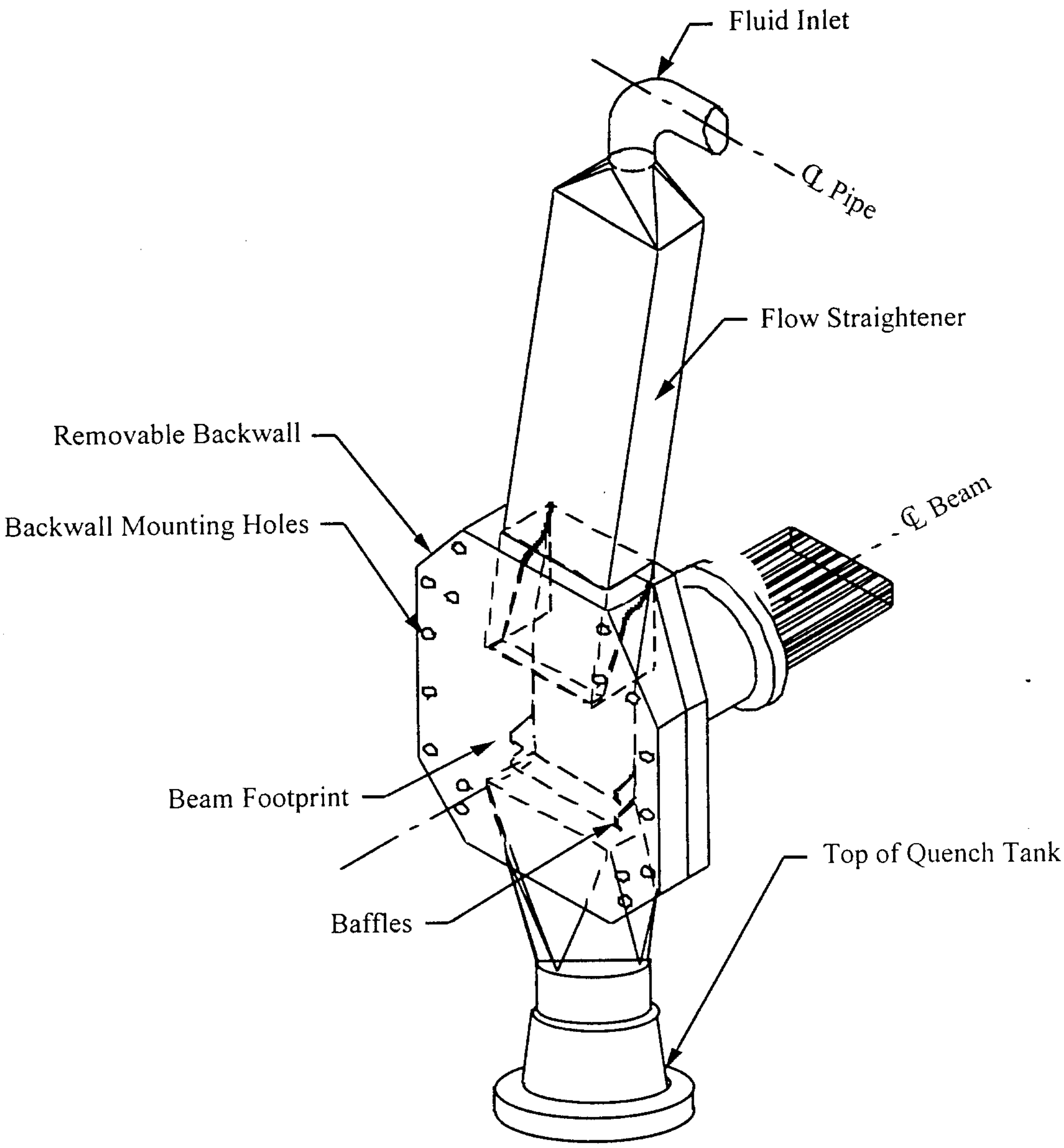
  Carried out 1995-96 in the framework of an IEA (International Energy Agency) task agreement by US, EU & Japan

  EU: Implemented in Fusion Technology Long-Term Programme

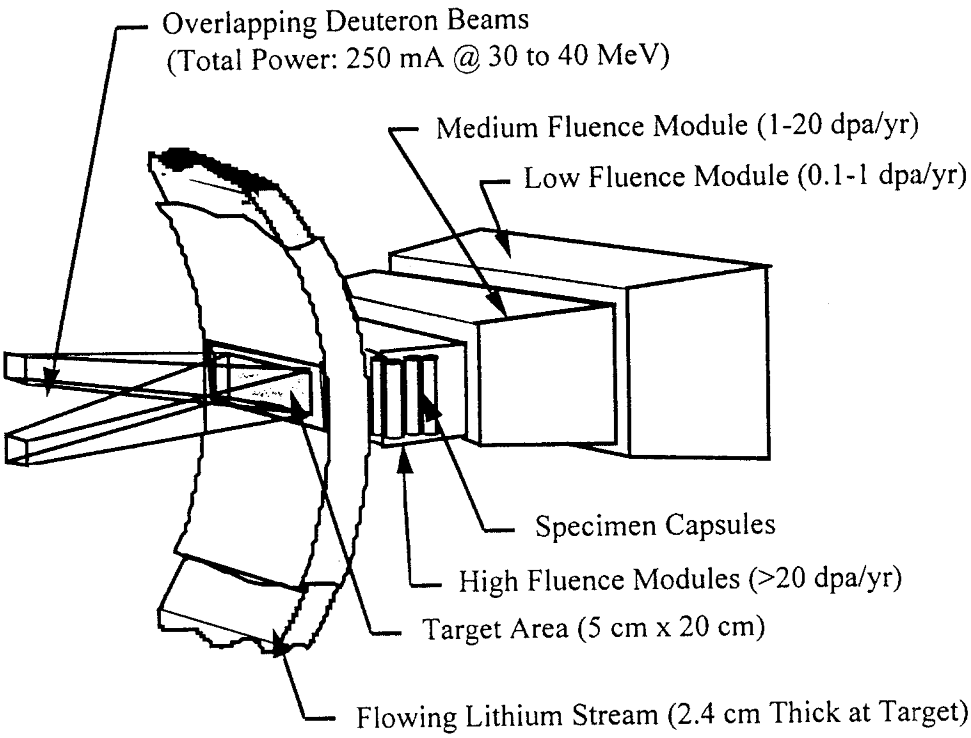
  Final CDA report issued by the end of '96

- Engineering Validation Phase (EVP) planned for 1997-99

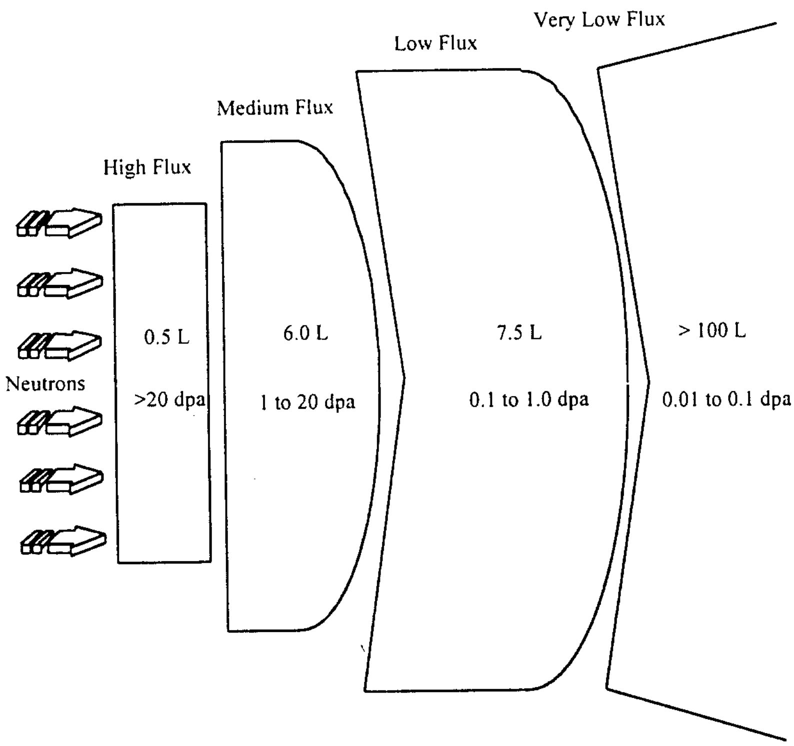
  Detailed engineering design of IFMIF



IFMIF target assembly with removable back wall



Interface of IFMIF facilities in Test Cell



Test Cell Configuration

Use of Li(d,xn)-reaction to produce high energy neutrons

- 40 MeV deuteron energy at 2 ×125 mA

  ⇒ Neutron production: ≅ $1.1\ 10^{17}\ s^{-1}$

  ⇒ Neutron flux density at first wall: ≅ $7.5\ 10^{14}\ cm^{-2}s^{-1}$

- Neutron spectrum extends to E ≥ 40 MeV

  ⇒ Need for cross-section data above 20 MeV

  ⇒ High priority: data for steel components and coolants for neutron transport calculations & nuclear responses (dpa, gas production, nuclear heating)

  ⇒ Data for major nuclides:

  i) ^{56}**Fe**, ^{23}**Na**, ^{39}**K**

  ii) ^{52}**Cr**, ^{51}**V**, ^{28}**Si**, ^{12}**C**

Nuclear data evaluation of high energy-cross sections up to 50 MeV neutron incidence energy

Co-operation of Forschungszentrum Karlsruhe and Institute of Nuclear Power Engineering , Obninsk

Comprehensive ongoing nuclear data evaluation programme started in 1995

Objective

Provide high energy neutron cross-section data needed for neutronics design analyses of IFMIF

Specifications

- Standard ENDF-6 nuclear data format using file6 data
- Transport & response nuclear data
- To be processed by NJOY/ACER for use with MCNP
- Priority list for nuclide evaluations:

  ^{56}Fe, ^{23}Na, ^{39}K, ^{52}Cr, ^{51}V, ^{28}Si, ^{12}C & minor stable isotopes of Fe, Cr

Data evaluation methodology

- Data evaluation above 20 MeV using modern nuclear reaction theory

  Geometry dependent hybrid exciton and evaporation models taking into account d, t, ^{3}He - and α - cluster emission

- Adoption of JENDL-3, ENDF/B-VI-data below 20 MeV

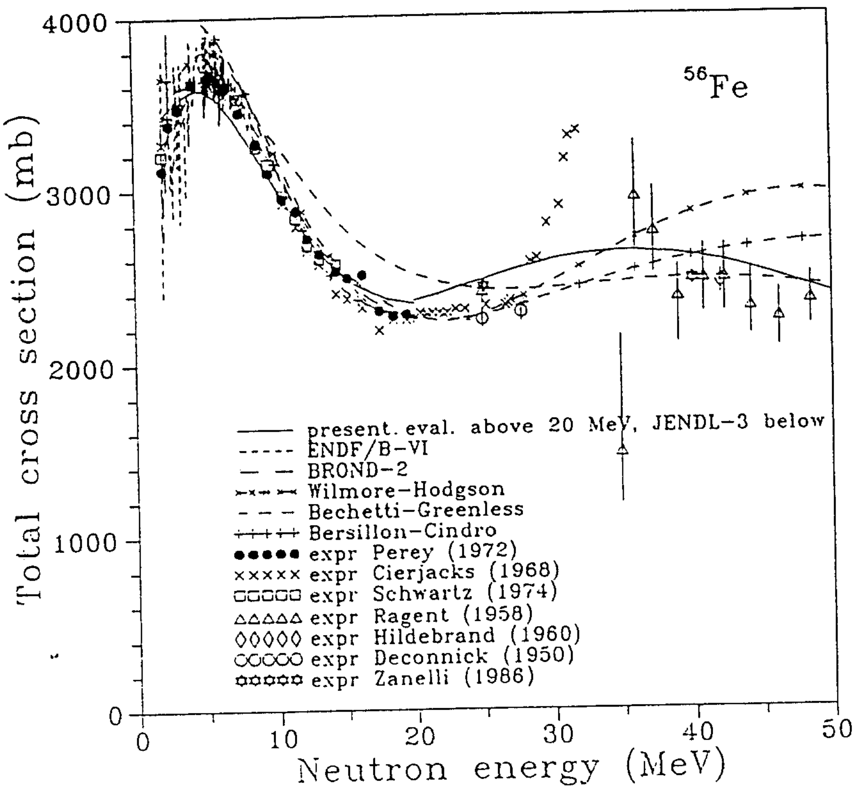
  For elastic & discrete inelastic scattering & (n,γ)-reactions.

- Test against experimental neutron and proton induced high energy reaction cross-section data

Present status

- Evaluated neutron cross-section data files for ^{56}Fe, ^{23}Na, ^{39}K, ^{52}Cr, ^{51}V, ^{28}Si, and ^{12}C
- ENDF-6 format with MT=5, MF=6 for neutron emission of many particle reactions
- Tabulated secondary energy-angle distributions (MF=6) in LAB-system (LAW=7)
- New ^{56}Fe data file (11/96) including γ - production cross-sections and spectra, recoil nucleus spectra, p - and α -spectra *(subsections of MT=5, MF=6)*

Yu. A. Korovin, A. Yu. Konobeyev, V .P. Lunev, P. E. Pereslavtsev, A. Yu. Stankovsky, U. Fischer, U. v. Möllendorff , M. Sokcic- Kostic, P. Wilson, D. Woll : **Evaluation and Test of High Energy Neutron Cross-Section Data for the IFMIF Intense Neutron Source,** 19th Symposium on Fusion Technology, Lisbon, Portugal, September 16-20, 1996.


4000
3000
2000
1000
0
0
10
20
30
40
50
Total cross section (mb)
Neutron energy (MeV)
^{56}Fe
present. eval. above 20 MeV, JENDL-3 below
ENDF/B-VI
BROND-2
Wilmore-Hodgson
Bechetti-Greenless
Bersillon-Cindro
expr Perey (1972)
expr Cierjacks (1968)
expr Schwartz (1974)
expr Ragent (1958)
expr Hildebrand (1960)
expr Deconnick (1950)
expr Zanelli (1986)

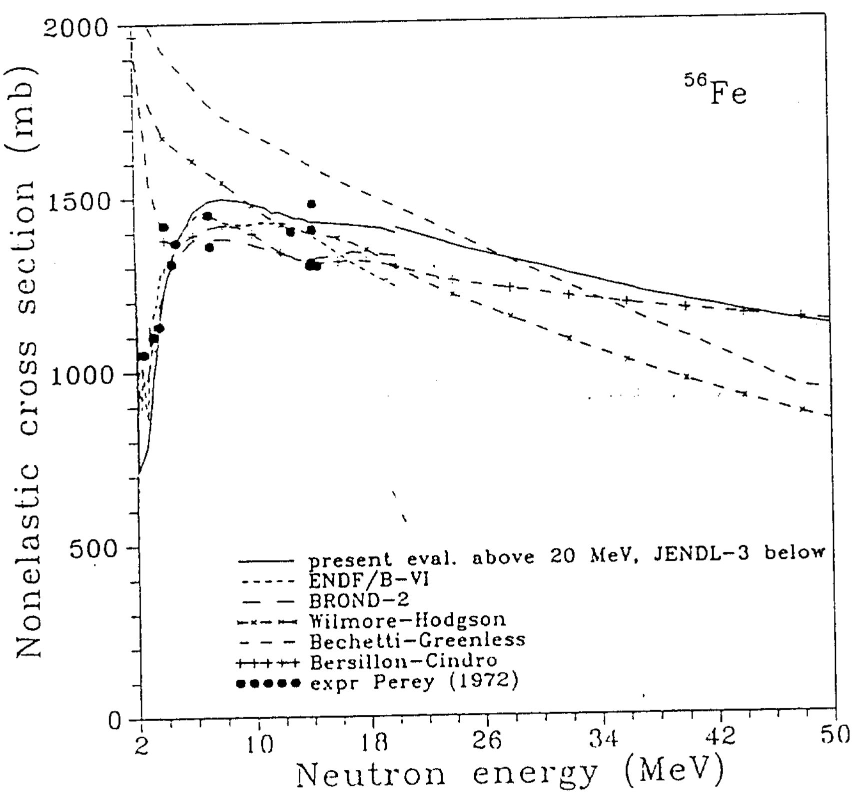
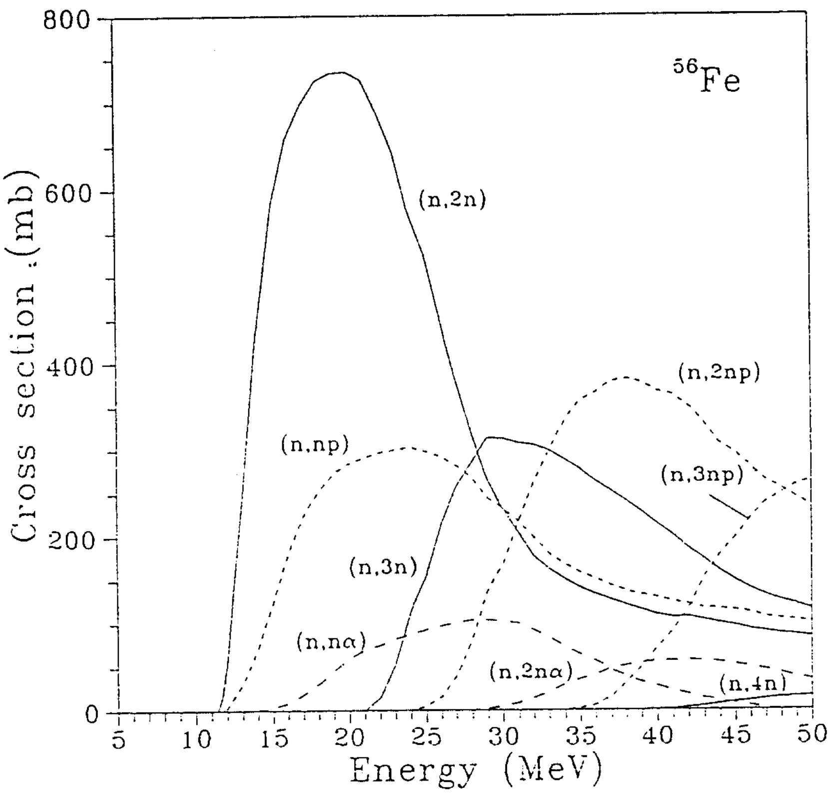

2000
1500
1000
500
0
2
10
18
26
34
42
50
Nonelastic cross section (mb)
Neutron energy (MeV)
^{56}Fe
present eval. above 20 MeV, JENDL-3 below
ENDF/B-VI
BROND-2
Wilmore-Hodgson
Bechetti-Greenless
Bersillon-Cindro
expr Perey (1972)

fig 1



^{56}Fe
(n,2n)
(n,2np)
(n,np)
(n,3np)
(n,3n)
(n,nα)
(n,2nα)
(n,4n)
Cross section (mb)
Energy (MeV)
0
200
400
600
800
5
10
15
20
25
30
35
40
45
50

Processing of high energy-cross section data for use with MCNP calculations

Processing with NJOY/ACER

- ENDF-6 formatted nuclear data
- No NJOY/ACER experiences with processing of high energy cross-section data

First trial: MT=201 (neutron production), MF=6 for ^{56}Fe

- ***Advantage***: MT=16, ... , 45 file 3 data can be kept
- ***Disadvantages***: Neutron yields included implicitly but MCNP needs yields explicitly, MT=201 not accepted by ACER

- ***Ad-hoc solution*** (12/96): Add MT=201 capability to ACER
  - File 3 data : minor modifications
  - File 6 data: major modifications
  - Yields to be calculated from file 3 data for MT=16, ... 45
  - File 6 data to be normalised
  - Works, but ...

    ... better to have yields explicitly on file

    ... be consistent with ENDF-6 format rules

$\Rightarrow$ ***Switch to MT=6, MF=6***

Final solution: MT=5 (n, anything), MF=6 data

Advantages:

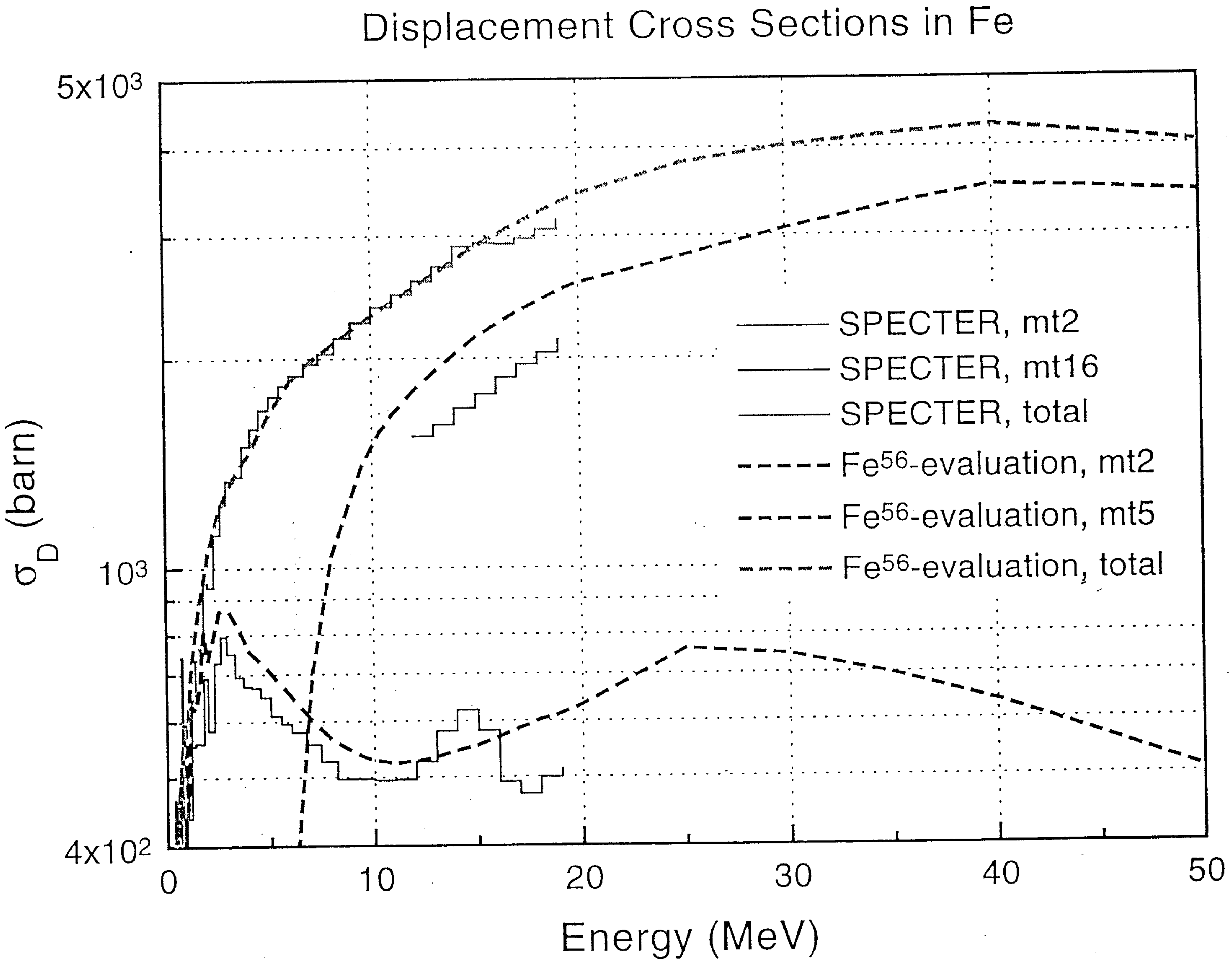
- Normalised energy-angle distributions
- Energy-dependent particle yields
- Subsections for different secondary particles (n, p, α, γ, recoil nuclei)
- Most suitable for subsequent use with MCNP
- Full consistency with ENDF-6 format rules
- Accepted by ACER without any modifications

Disadvantage:

Individual many-particle reactions included in MT=5 (e. g. MT=16, .. 45, 91) to be dropped from file 3

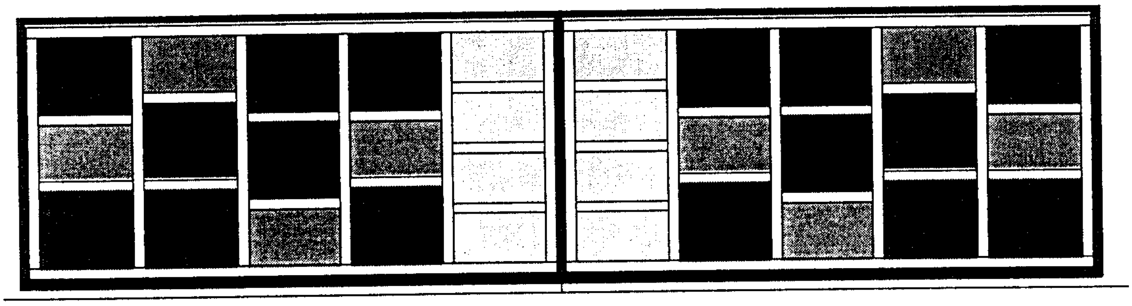
$\Rightarrow$ Add lumped gas production data to file 3 (MT= 203, ..., 207)

$\Rightarrow$ Put variety of individual reactions on activation file

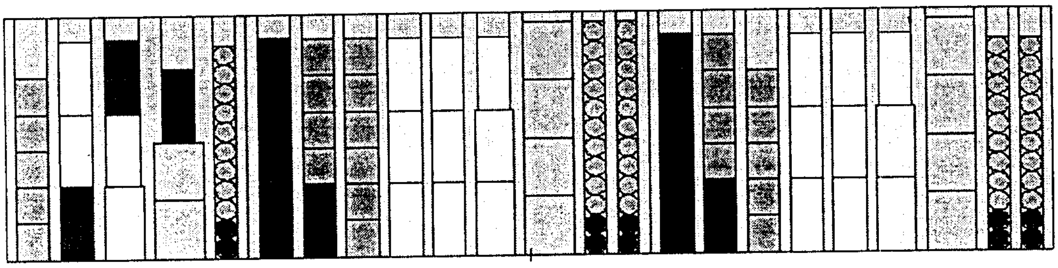


Displacement Cross Sections in Fe
5x10^3
10^3
4x10^2
σ_D (barn)
0
10
20
30
40
50
Energy (MeV)
SPECTER, mt2
SPECTER, mt16
SPECTER, total
Fe56-evaluation, mt2
Fe56-evaluation, mt5
Fe56-evaluation, total

Neutronics design analyses for IFMIF



Helium cooled high flux test module



NaK cooled high flux test module

3d MCNP-calculations

- Monte Carlo Li(d,n) neutron source function for MCNP
- Use of new high energy cross-section data
    - Flux & spectra distribution in test module specimens
    - DPA, gas production & nuclear heating
    - Neutron wall loading distribution at first wall of test module
    - Comparison to ITER/DEMO-conditions
    - Comparison to uncollided spectra calculations

⇒ *Comprehensive FZKA-report under preparation*