ENEA C.R. Frascati

Nuclear Data Base Relevant to Fusion Reactor Technology

Mario Pillon
Association ENEA-Euratom
UTS Fusione
ENEA CR Frascati, Italy

pillon@frascati.enea.it

Erice school on Fusion Reactor Technology
26July 1 August 2004

Introduction

Fusion reactions for near term power reactors

Deuterium+Tritium $\Rightarrow$ Alpha (3.5 MeV)+Neutron (14.1 MeV)

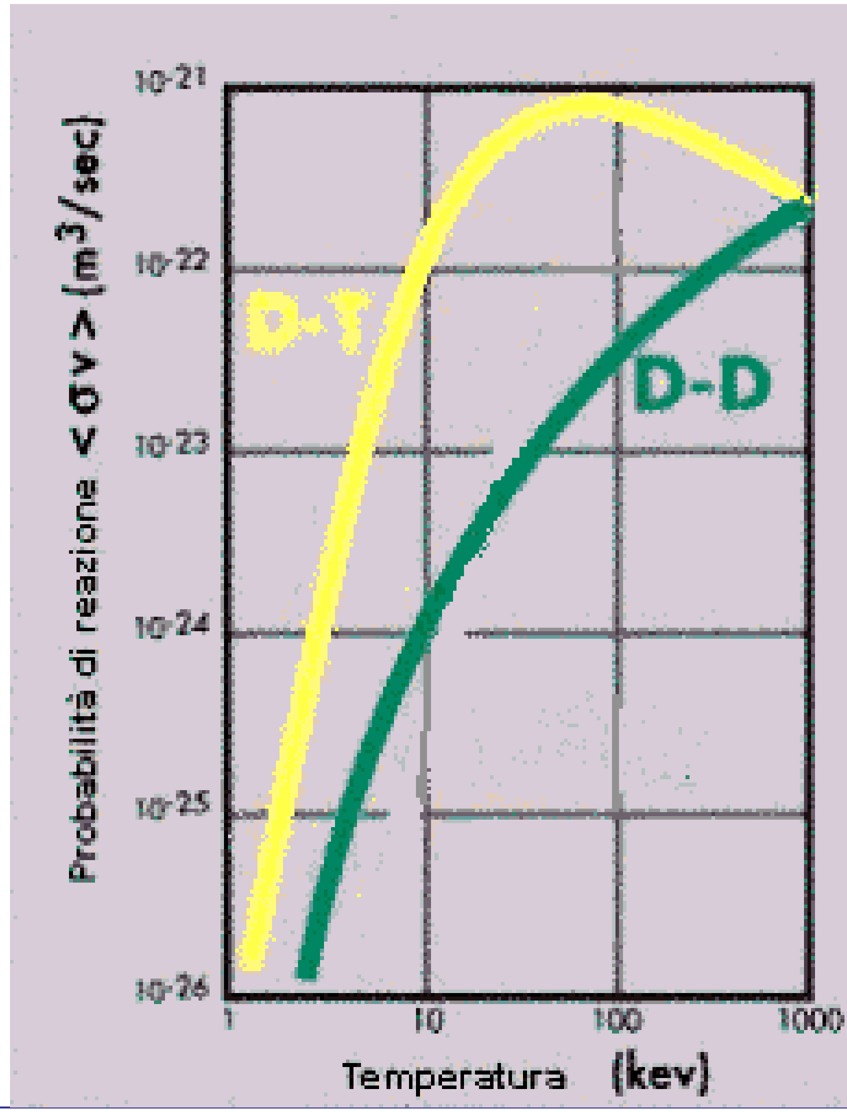
Other secondary reactions

Deuterium+Deuterium $\Rightarrow$ ^{3}Helium(0.82 MeV)+Neutron(2.45MeV)

Deuterium+Deuterium $\Rightarrow$ Tritium(1.01 MeV)+Proton(3.02 MeV)

Both these fusion reactions produce neutrons

Fusion reaction cross-sections



Introduction cont.

In a fusion reactor neutrons take away 80% of the produced energy. They will be adsorbed in the blanket surrounding the plasma core. The blanket must contains lithium (Li) which breed Tritium throughout the reactions:

$$\mathbf{Li^7 + n = He^4 + T + n^* - 2.5\ MeV}$$
$$\mathbf{Li^6 + n = He^4 + T + 4.86\ MeV}$$

(n^* = escaping low energy neutron)

Natural Lithium (92.5% Li^7, 7.5% Li^6) is abundant in the rocks (30 wppm) and is also present in the oceans.

Lithium Blanket contributes to slow down (reduce energy) the neutrons.

Some numbers

1 MW of D-T fusion power produces 3.67E17 neutrons

A neutron flux of 4.46E+13 n/(cm^{2}s) = heat flux of 1MW/m^2

1 DPA in SS corresponds to a 14 MeV fluence =1.4E+20 n/cm^2

A natural lithium blanket has an energy multiplication.

A 14 MeV neutron if fully moderated and adsorbed in a natural lithium blanket releases a total energy of 16.26 MeV (0.35 MeV is gamma energy).

The maximum theoretical Tritium Breeding Ratio (TBR) of natural lithium is 2.

more numbers

400 MW of fusion power (e.g. ITER) using 40 MW of NBI auxiliary heating (40A 1 MV) corresponds to the following neutron production from a D-T plasma:

1.4×10^{20} n/s with energy of 14.1 MeV from maxwellian plasma

$\approx 1 \times 10^{18}$ n/s with energy of 2.5 MeV from maxwellian plasma

3×10^{16} n/s energy 14.2-16.7 MeV from D-T beam-plasma

2.8×10^{15} n/s energy 2.5-4.1 MeV from D-D beam-plasma

9.6×10^{12} n/s energy 14.3-19.9 MeV from T-D beam-plasma

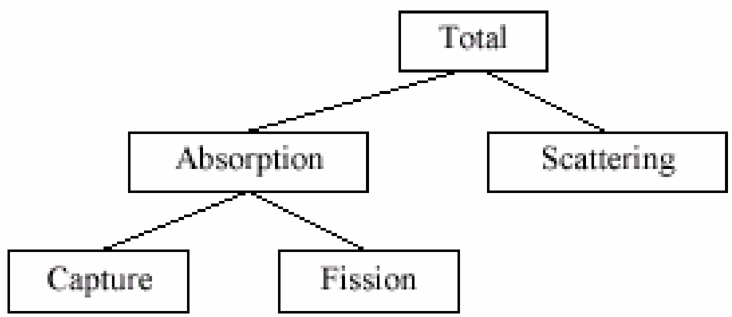
NEUTRON CROSS SECTIONS

The *microscopic cross section* (σ) is a property of a given nuclide; σ is the probability per nucleus that a neutron in the beam will interact with the nucleus; this probability is expressed in terms of an equivalent area that the neutron "sees." The *macroscopic cross section* (Σ) takes into account the number of those nuclides present

$$\Sigma = N\sigma \ [cm^{-1}]$$

The mean free path is *mfp* $=\lambda = 1/\Sigma$. The microscopic cross section is measured in units of barns (b): 1 barn equals 10^{-24} $cm^2 = 10^{-28}$ m^2.

Cross Section Hierarchy



$$\sigma_t = \sigma_s + \sigma_a = \sigma_s + (\sigma_c + \sigma_f) \qquad where\ \sigma_c \approx \sigma_\gamma$$

$$\Sigma_t = \Sigma_s + \Sigma_a = \Sigma_s + (\Sigma_c + \Sigma_f)$$

NEUTRON CROSS SECTIONS cont.

For a mixture of isotopes and elements, Σ's add. For example

$$\Sigma_a^{H_2O} = \Sigma_a^{H} + \Sigma_a^{O} = N_H\, \sigma_a^H + N_O\, \sigma_a^O$$
$$= 2\, N_{H_2O}\, \sigma_a^H + N_{H_2O}\, \sigma_a^O = N_{H_2O}\, (2\, \sigma_a^H + \sigma_a^O)$$

<u>1/v Law</u>

For very low neutron energies, many adsorption cross sections are 1/v due to the fact the nuclear force between the target nucleus and the neutron needs long time to interact.

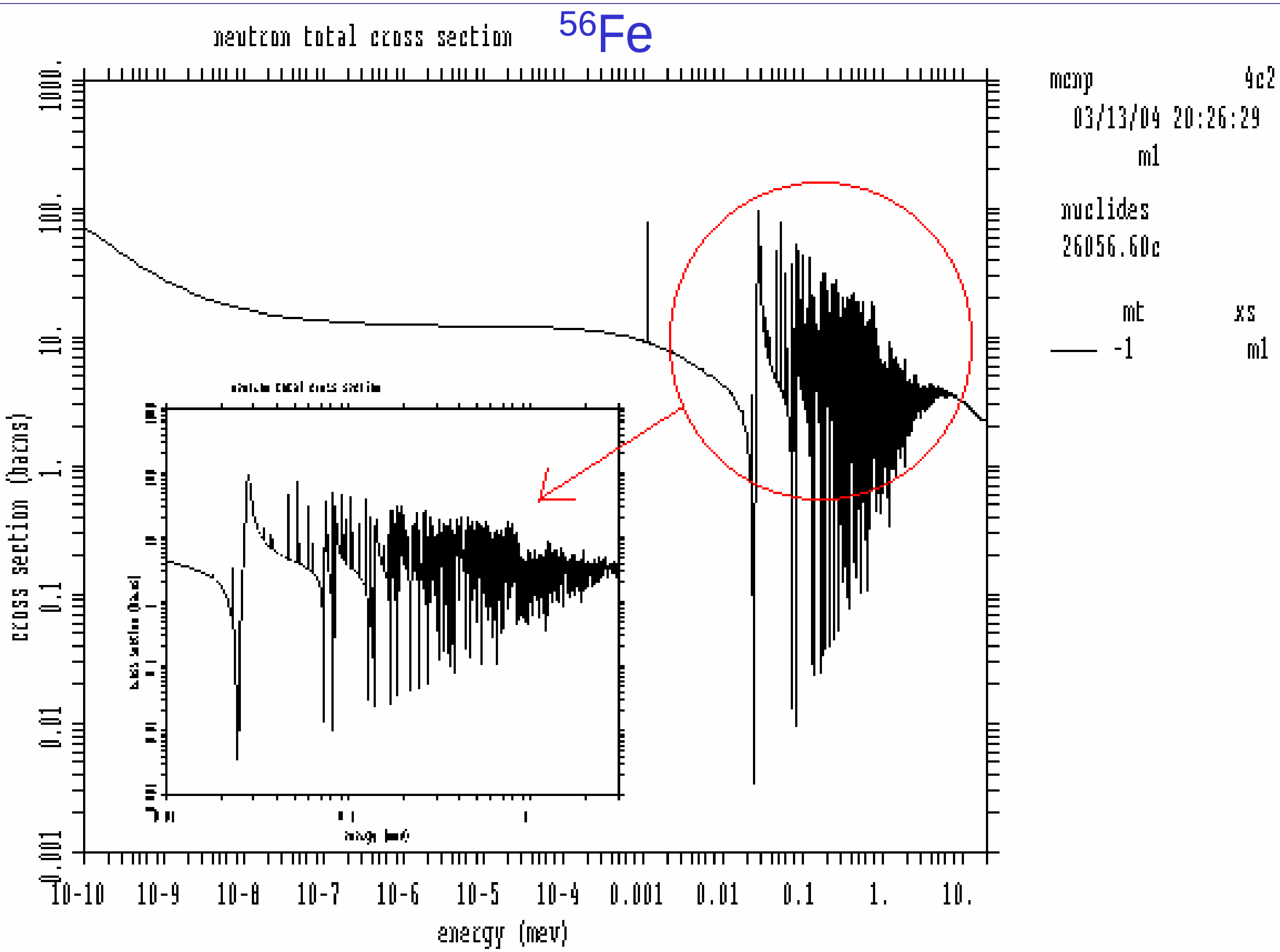
$$\sigma_a \propto \frac{1}{v} \propto \frac{1}{\sqrt{E}} \propto \frac{1}{\sqrt{T}}$$

Energy dependence of cross sections

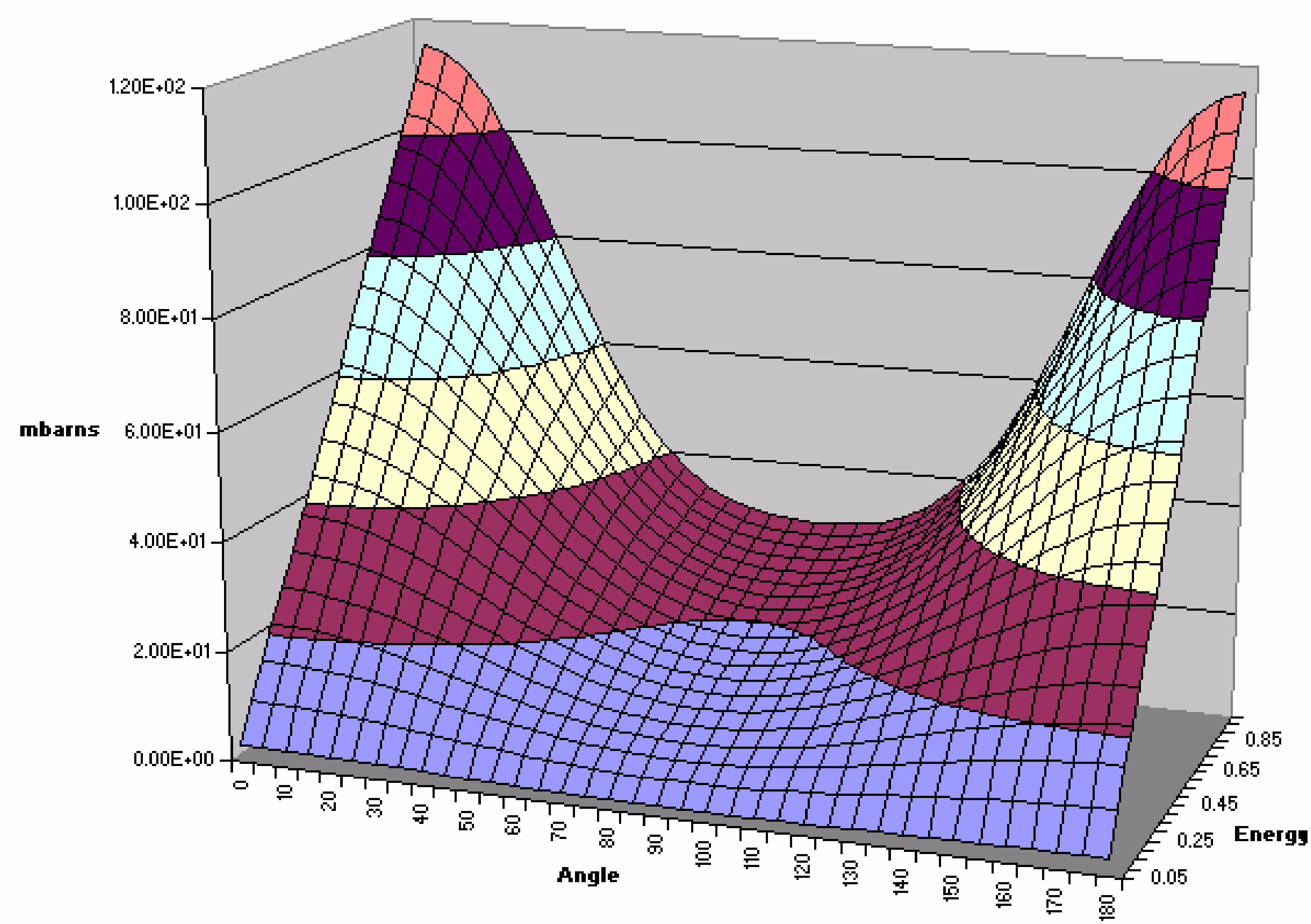
σ_s is independent of thermal energy (and temperature), σ_a (σ_t and σ_c) are energy dependent

$$\frac{\sigma_a(E)}{\sigma_{a0}} = \frac{v_0}{v(E)} = \sqrt{\frac{E_0}{E}} = \sqrt{\frac{T_0}{T}}$$

NEUTRON CROSS SECTIONS cont.



Cross sections are often energy-angle dependent

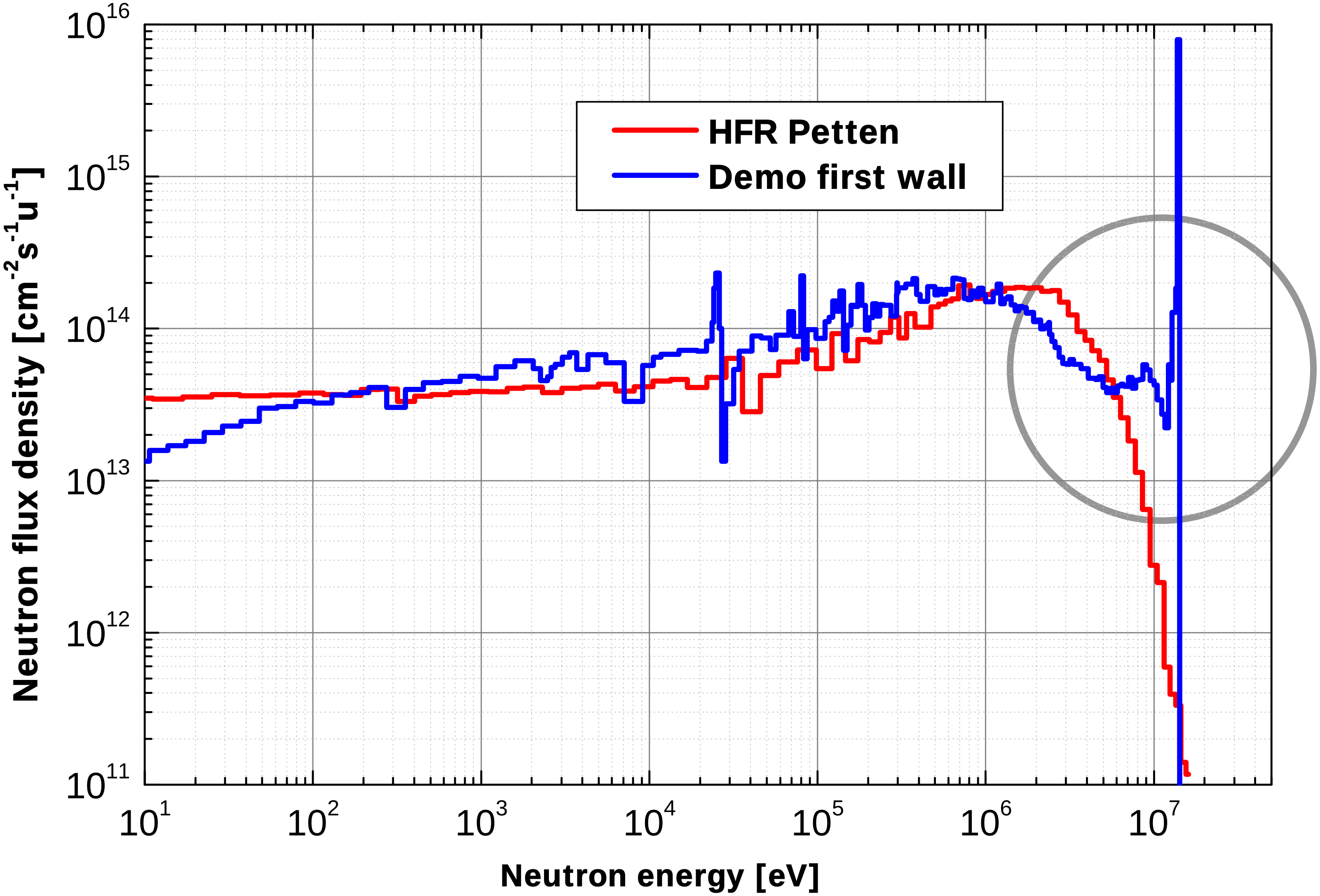


NEUTRON CROSS SECTIONS cont.

C.R. Frascati

TABLE OF CONTENTS

Neutron flux spectra: fusion vs. fission



M. Pillon, Erice school on Fusion Reactor Tecnology,26July-1 August, 2004

C.R. Frascati

Definition

- ”Neutronics”

  Neutron (and photon) transport phenomena

  – Simulation of transport in the matter

- ”Nuclear Data”

  – Nuclear cross-sections to describe interactions neutrons $\Leftrightarrow$ nuclei

    $\Rightarrow$ neutron spectra, nuclear responses, activation

- ”Fusion Technology Applications”

  – Nuclear design of fusion reactor systems

    $\Rightarrow$ 14 or 2.5 MeV neutrons (continuous spectra)

Introduction

"Status and Perspectives"

- "Status"
  - What do we have achieved ?
  - What were the objectives ?
- "Perspectives"
  - Where to go ?
  - New objectives ?
  - What needs to be done ?

Fusion Reactor Design Issues

- Blanket
  - Tritium breeding, power generation, (shielding)

  ⇒ *assure tritium self-sufficiency, provide nuclear heating data for thermal-hydraulic layout*
- Shield
  - Attenuate radiation to tolerable level

  ⇒ *assure sufficient protection of super-conducting magnet*

  ⇒ *helium gas production in steel structure (≤ 1 appm)*
- Safety & Environment, Maintenance
  - Material activation (Big difference respect to fission!)

  ⇒ *minimise activation inventory with regard to short-term and long-term hazard potential*

  ⇒ *maintenance service during reactor shutdown (dose level)*

C.R. Frascati

Environmental Issues Associated with Fusion Power Plants

- Generation of Environmentally Undesirable Materials with Fusion neutrons

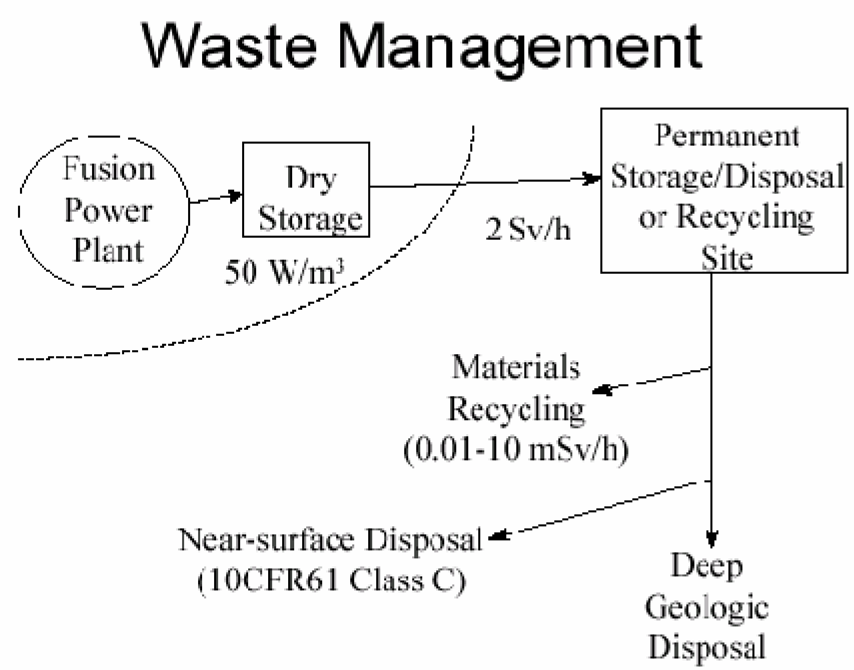
- Fusion Neutron Induced Long-lived Radioactivity (High Energy and Fluence)

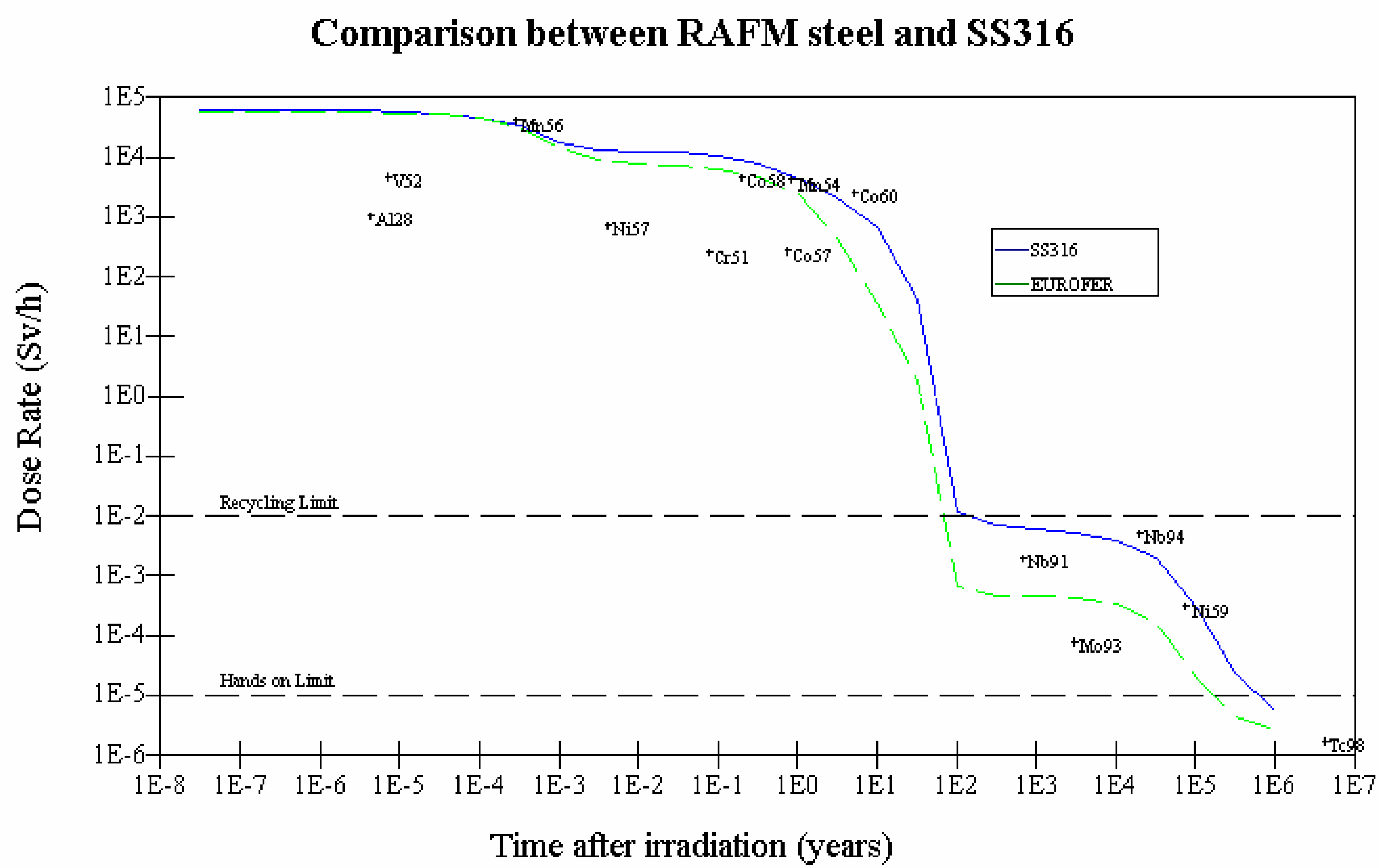
- Half-lives from 418 Y (Ag-108^{m}) to 720000 Y (Al-26)

- 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

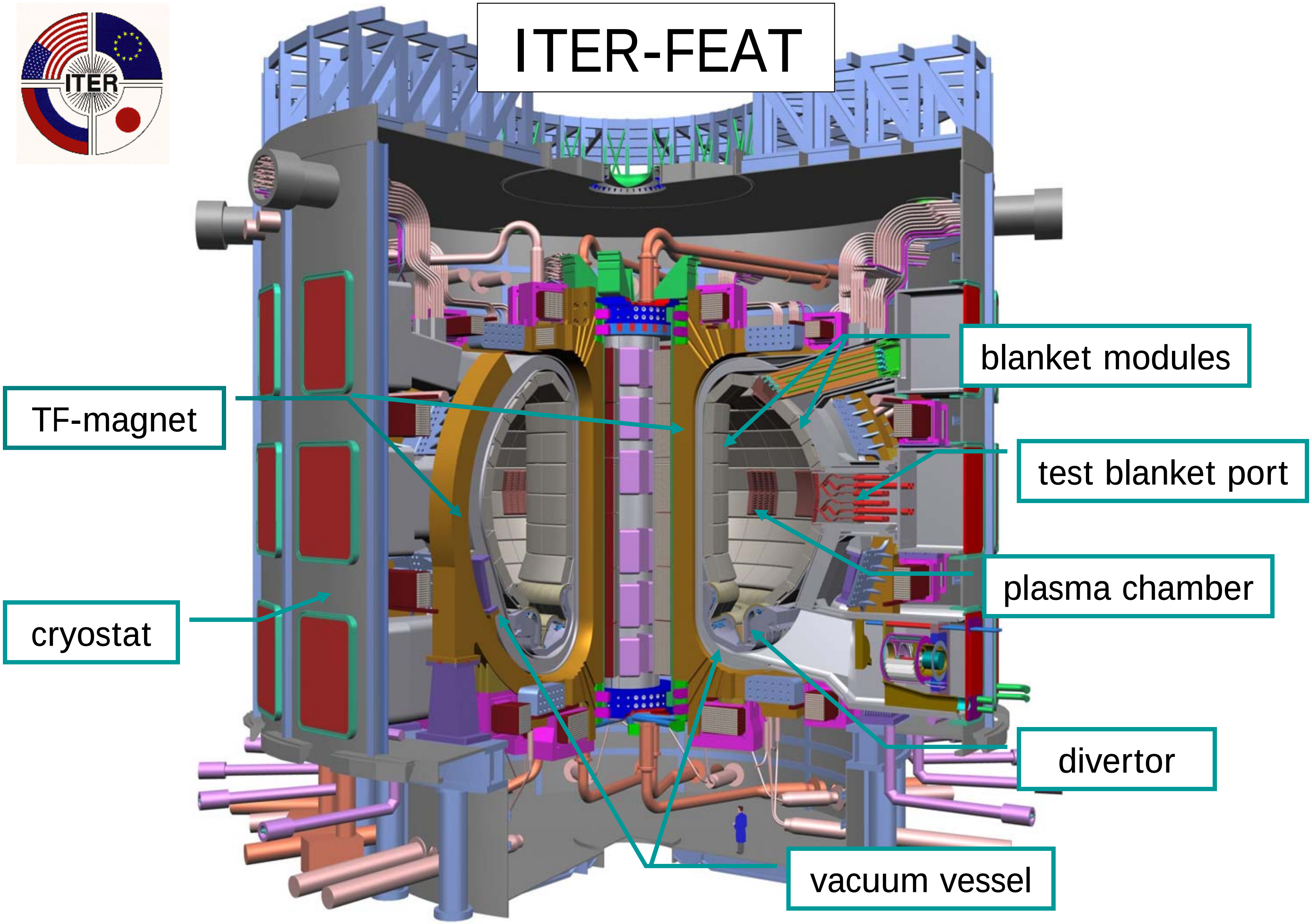
- Selection of Reduced Activation Fusion Materials (RAFM)– Reduction of Activation Level of Fusion Waste
- Recycle of Fusion Power Plant Component and Materials – Reduction of Fusion Waste Quantities





Comparison between RAFM steel and SS316
Dose Rate (Sv/h)
Time after irradiation (years)
1E5
1E4
1E3
1E2
1E1
1E0
1E-1
1E-2
1E-3
1E-4
1E-5
1E-6
1E-8
1E-7
1E-6
1E-5
1E-4
1E-3
1E-2
1E-1
1E0
1E1
1E2
1E3
1E4
1E5
1E6
1E7
Recycling Limit
Hands on Limit
SS316
EUROFER
Mn56
V52
Al28
Ni57
Cr51
Co58
Mn54
Co60
Co57
Nb94
Nb91
Ni59
Mo93
Tc98

ITER-FEAT


blanket modules
TF-magnet
test blanket port
plasma chamber
cryostat
divertor
vacuum vessel

Fusion Reactor Design

- Relies on data provided by nuclear design calculations
- Qualified computational simulations are required
  - $\Rightarrow$ Appropriate computational methods, tools (codes) and data (nuclear cross-sections)
- Qualification through integral benchmark experiments.

Fusion Nuclear Data

- Neutron transport data

  σ_a (E), σ_{tot}(E) *E: neutron energy*

  σ_{nem}(E,E',μ) *μ=cos (θ), θ=scattering angle*

  Angular distributions ("SAD") for elastic scattering; energy-angle distributions (" DDX") for inelastic reactions: (n,n'γ), (n,xn),...

- Photon transport data

  γ - production cross-sections and spectra

  γ - interaction cross-sections

- Response data

  tritium production, energy deposition (heating), gas production, activation & transmutation, radiation damage

$\Rightarrow$ Complete data libraries are required (!)

C.R. Frascati

Fusion Reactor Materials

Breeder materials	Li, Pb-Li, Li_4SiO_4, $LiAlO_2$, Li_2ZrO_3, Li_2O, Li_2TiO_3, Flibe, Sn-Li
Neutron multiplier	Be, Pb
Structural materials	SS-316, MANET, Eurofer,F82H, V-5Ti, SiC, Cr-alloys, ... [Fe, Cr, Mn, Ni, Mo, Cu, Co, Nb, W, Ta, V, Si, C, ...]
Cooling materials	He, H_2O, Pb-17Li, Li, Flibe, Sn-Li
Other materials (shielding, magnet, insulator, etc.)	B_4C, Nb_3Sn, W, Al_2O_3, MgO

C.R. Frascati

Fusion Nuclear Data Libraries

- European Fusion/Activation File (EFF/EAF)
  - Developed as part of EU fusion technology programme in co-operation with JEF (Joint European File) project
- JENDL-FF (Japanese Evaluated Nuclear Data Library - Fusion File)
  - Fusion data library based on JENDL-3.3
- ENDF/B-VI (US Evaluated Nuclear Data File)
  - General purpose data library suitable for fusion applications
- FENDL Fusion Evaluated Nuclear Data Library
  - Developed for ITER under co-ordination of IAEA/NDS

⇒ *Working libraries for discrete ordinates codes (DORT,TORT-multigroup data),Monte Carlo codes (MCNP-continuous energy representation) & inventory codes (FISPACT-one group spectra collapsed data)*

EU Fusion Nuclear Data **(present available)**

- EFF-3 general purpose data files
  - ^{7}Li, ^{9}Be, ^{27}Al, ^{28}Si, ^{nat}V, ^{52}Cr, ^{56}Fe, ^{58}Ni , ^{60}Ni; (Mo, ^{nat}Pb EFF-2/-1)
  - Complete evaluations for neutron cross-sections from 10^{-5} eV to 20 MeV in ENDF-6 data format
  - Multi-group and Monte Carlo (ACE) working libraries available
  - Validated through extensive benchmarking
  - Integrated to Joint European Fission and Fusion File JEFF-3
- European Activation File EAF-2003
  - 774 target nuclides from Z=1 (hydrogen) to 100 (fermium)
  - 12617 neutron cross-sections from 10^{-5} eV to 20 MeV.
  - 1917 nuclides decay data information
  - Validated for important materials through integral activation experiments. 171 reactions have been validated.

EU Fusion Nuclear Data (E> 20 MeV) (present available)

- Intermediate Energy Activation File IEAF-2001
  - Developed as part of IFMIF project
  - 679 target nuclides from Z=1 (hydrogen) to 84 (polonium)
  - $\approx$ 134.000 neutron-induced reactions from 10^{-5} eV to 150 MeV
  - $E \leq$ 20 MeV: EAF-99 activation cross-sections
  - Multi-group data library available for activation calculations with ALARA (P. Wilson, UW); not compatible with FISPACT code
- Intermediate Energy General Purpose Data Files
  - Developed as part of IFMIF project (ENDF-6 data format)
  - ^{1}H, ^{23}Na, ^{39}K, ^{28}Si, ^{51}V, ^{52}Cr, ^{56}Fe (50 MeV); $^{6,7}Li$, ^{9}Be (150 MeV)
  - Processed data files for use by MCNP available

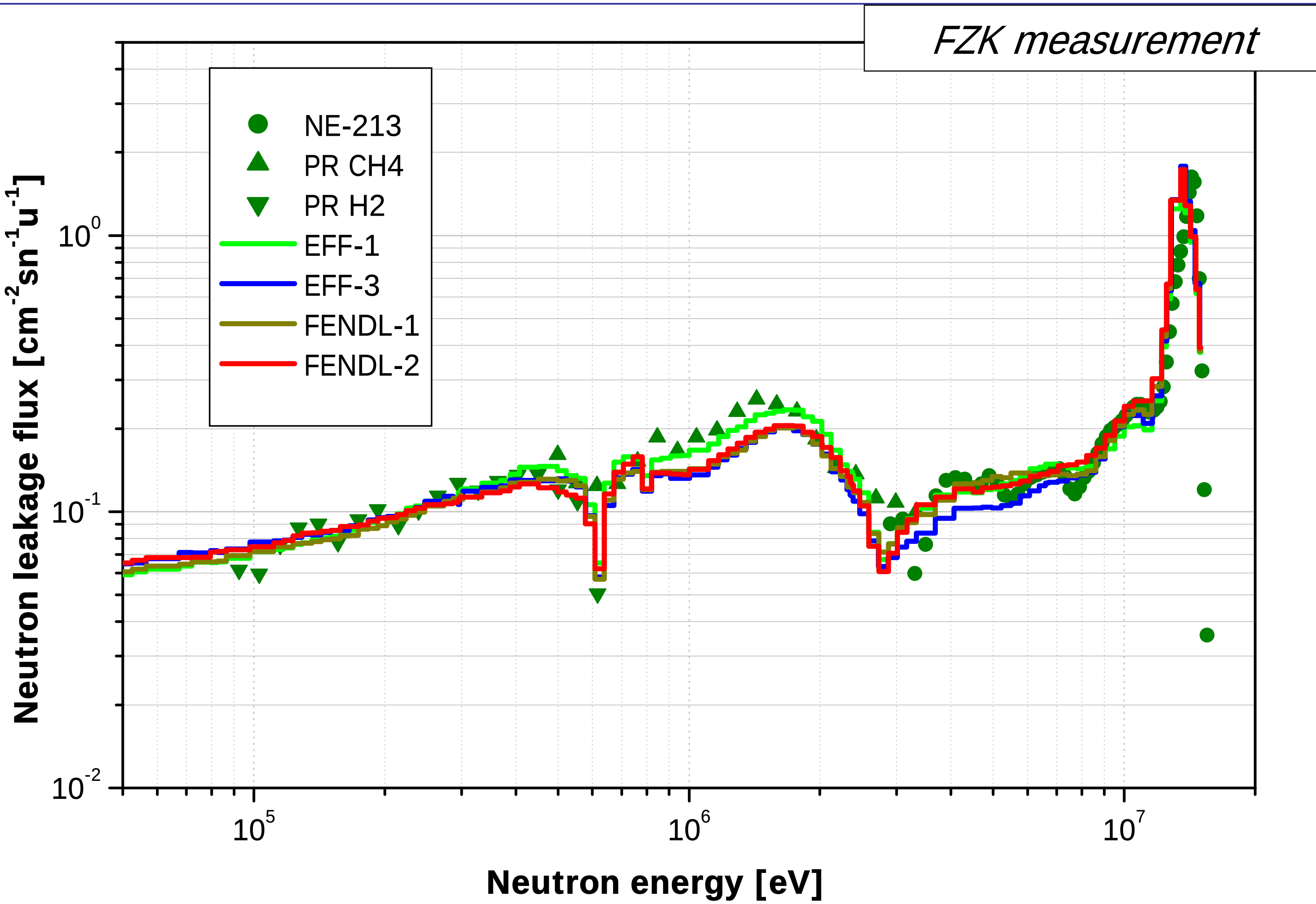
Nuclear Data Testing

- Integral experiments & their computational analyses
  - ⇒ *Assure reliable results of nuclear design calculations*
- Benchmark experiments for nuclear data testing
  - ⇒ *Qualification of nuclear data*
  - – 14 MeV neutron transport experiments (simple geometry, one material)
  - – Activation experiments (irradiation of foils)
- Design-oriented mock-up experiments
  - – Validation of nuclear design calculations
  - – Assessment of calculation uncertainties

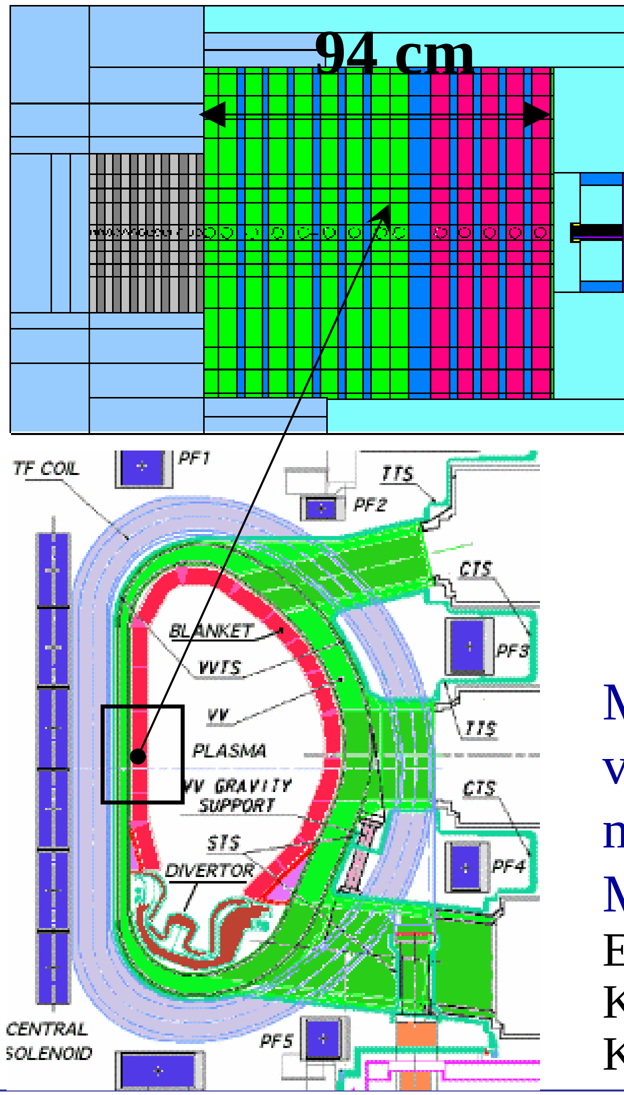
EU Validation Experiments

- 14 MeV neutron transport benchmark experiments
  - *Transmission experiments on Be spheres (KANT), iron slab (TUD)*
  - *FNG experiments on stainless steel assembly, ITER bulk shield & streaming mock-up (steel/water), SiC assembly.Tungsten assembly*
  - *FNG shut-down dose rate experiment using ITER shield mock-up.*
- Integral activation experiments
  - *FNG, TUD (SNEG-13), FZK (cyclotron)*
  - *Eurofer, SS-316, F82H, MANET, Fe, V, V-alloys, Cu,CuCrZr, W, Al, SiC, Li_4OSi_4.,Yttrium, Scandium, Samarium, Gadolinium,Dysprosium, Lutetium,Hafnium,Nickel,Molybdenum,Niobium*
- IFMIF validation experiments
  - *Activation experiments on SS-316, F82H, V and V-4Ti-4Cr using d-Li reaction (FZK cyclotron, 52 MeV)*
  - *Transport benchmark experiment on iron slab using ^3He-D_20 reaction (NPI Rez cyclotron, 40 MeV)*
  - *D-Li source simulation experiments (FZK, NPI Rez)*

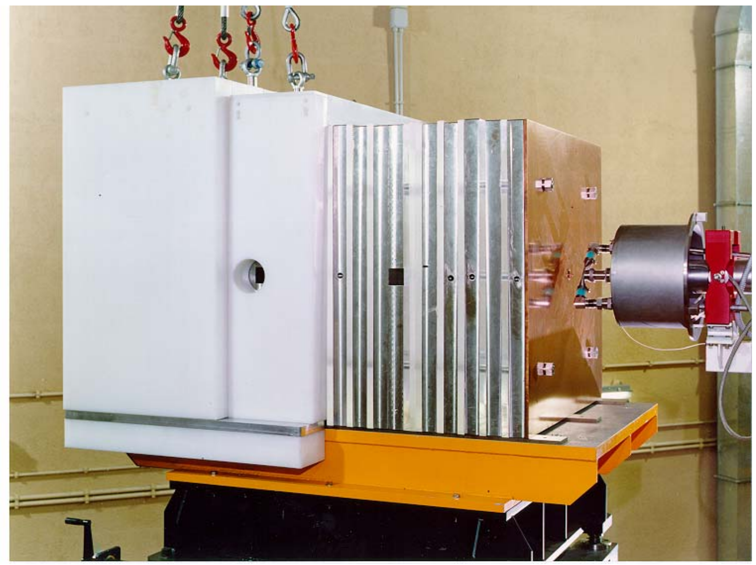
KANT Beryllium spherical shell (5/22 cm)



BULK SHIELD EXPERIMENT for ITER



Objective: verify design shielding calculations for ITER



Mock-up of first wall, shielding blanket and vacuum vessel (stainless steel+water), SC magnet (inboard) irradiated at the Frascati 14 MeV Neutron Generator (FNG)

ENEA- Frascati, TU Dresden, CEA Cadarache, FZK Karlsruhe, Josef Stefan Institute Lubljana, Kurchatov Institute Moscow

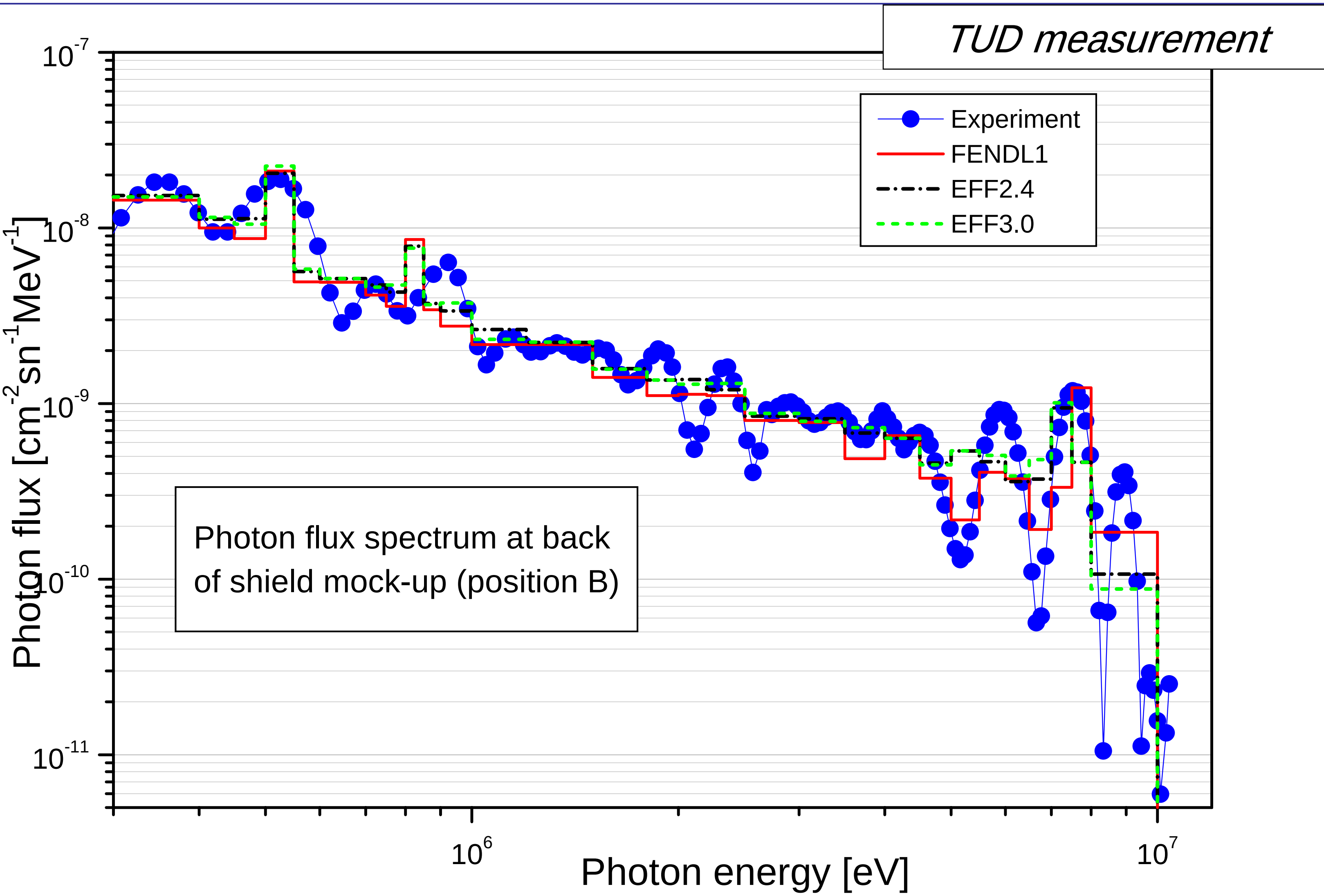
ITER Bulk Shield Experiment (FNG)

Neutron flux integrals at the back of vacuum vessel mock-up

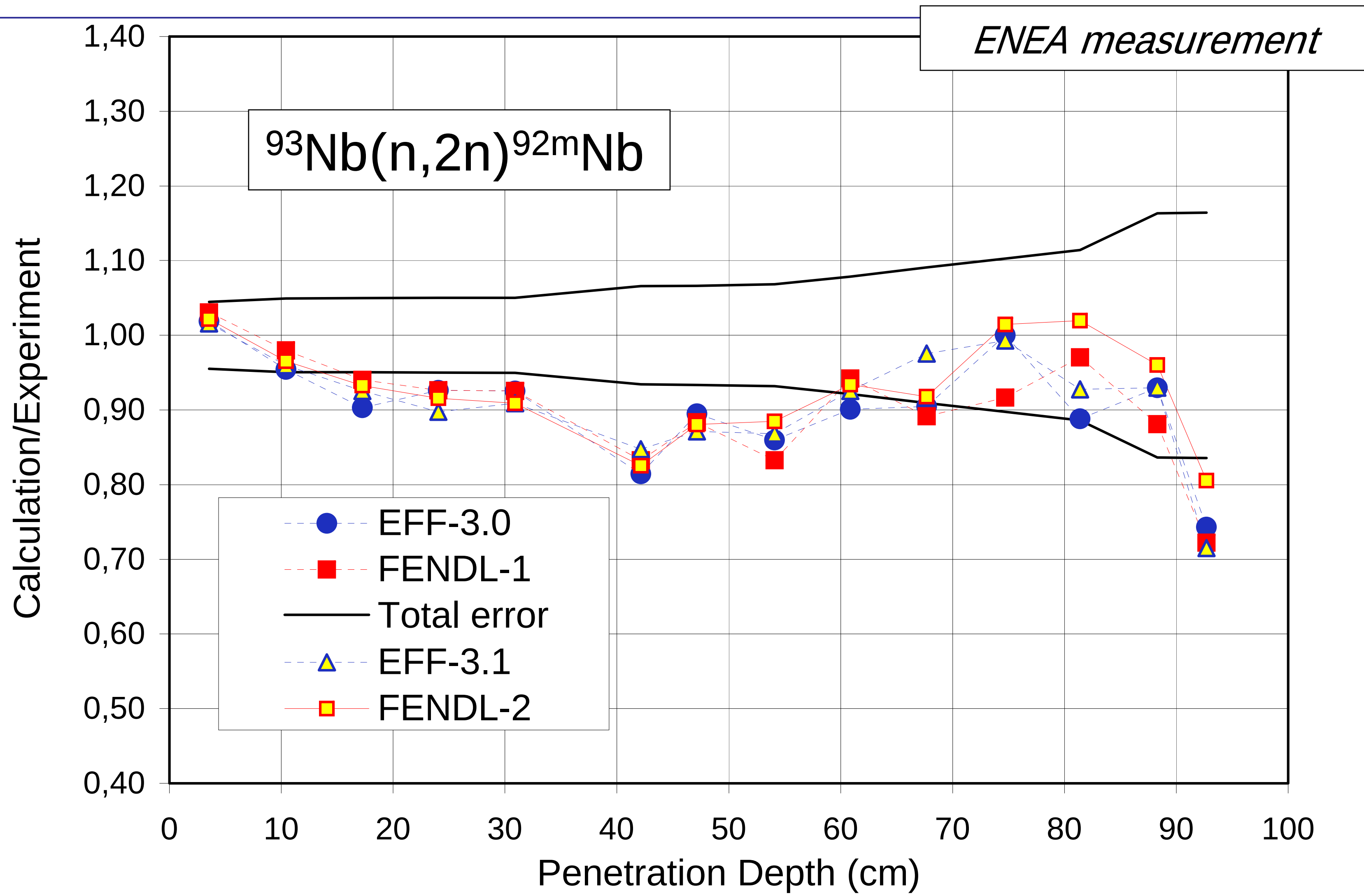
Energy interval [MeV]	Experiment(*) $[cm^{-2} \cdot sn^{-1}]$	Calculation/Experiment (C/E) EFF-3.1	FENDL-1	FENDL-2	Stat. Error [%]
0.1 – 1	$(8.78 \pm 0.89) \cdot 10^{-9}$	0.76	0.68	0.71	0.08
1 – 5	$(2.37 \pm 0.13) \cdot 10^{-9}$	0.88	0.78	0.77	0.09
5- 10	$(2.69 \pm 0.14) \cdot 10^{-10}$	1.02	1.00	0.88	1.5
E > 10 MeV	$(5.79 \pm 0.15) \cdot 10^{-10}$	0.86	0.81	0.80	1.5
E > 0.1 MeV	$1.20 \cdot 10^{-8} \pm 7.5\%$	0.79	0.71	0.73	0.7

(*) TUD measurement

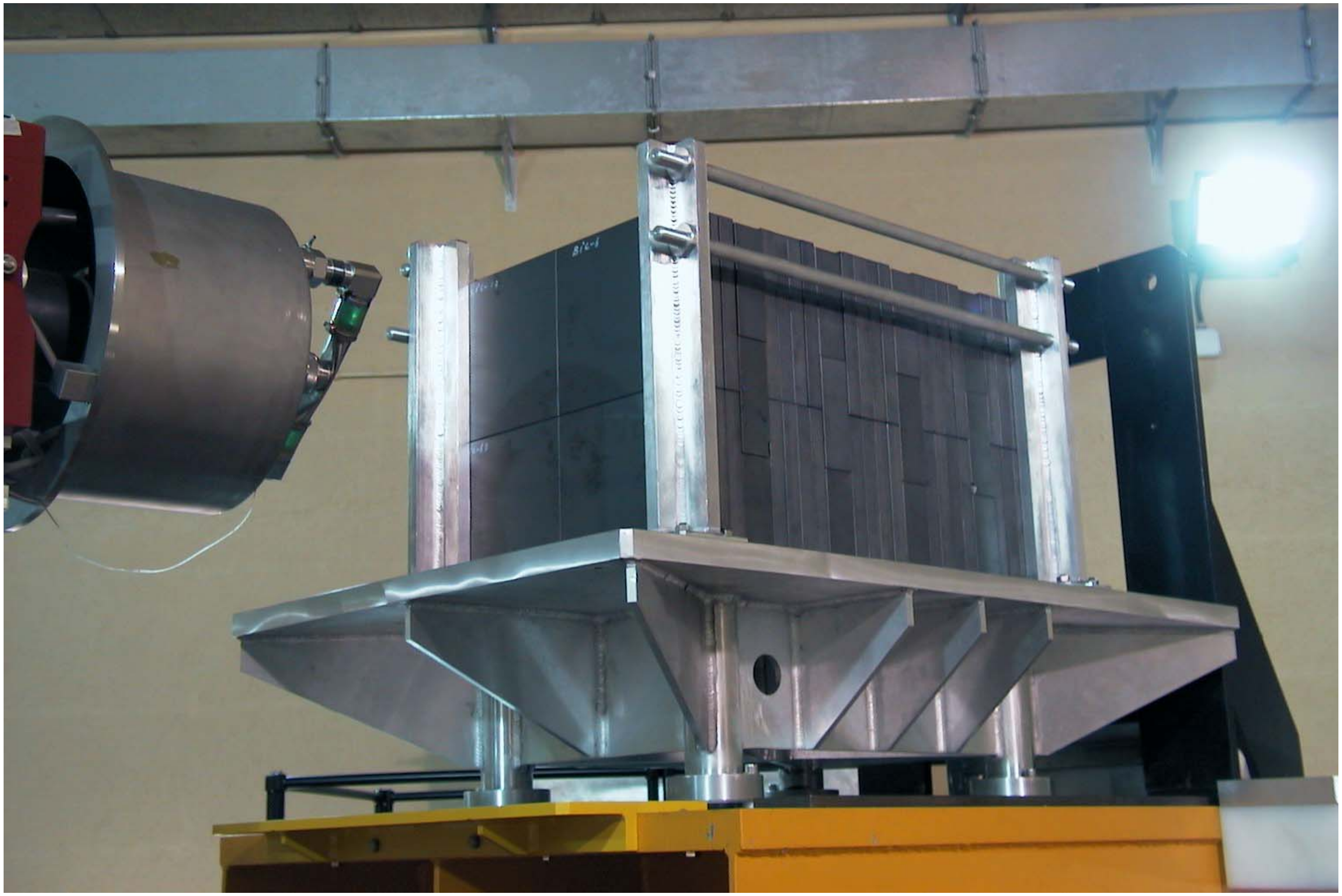
ITER Bulk Shield Experiment (FNG)



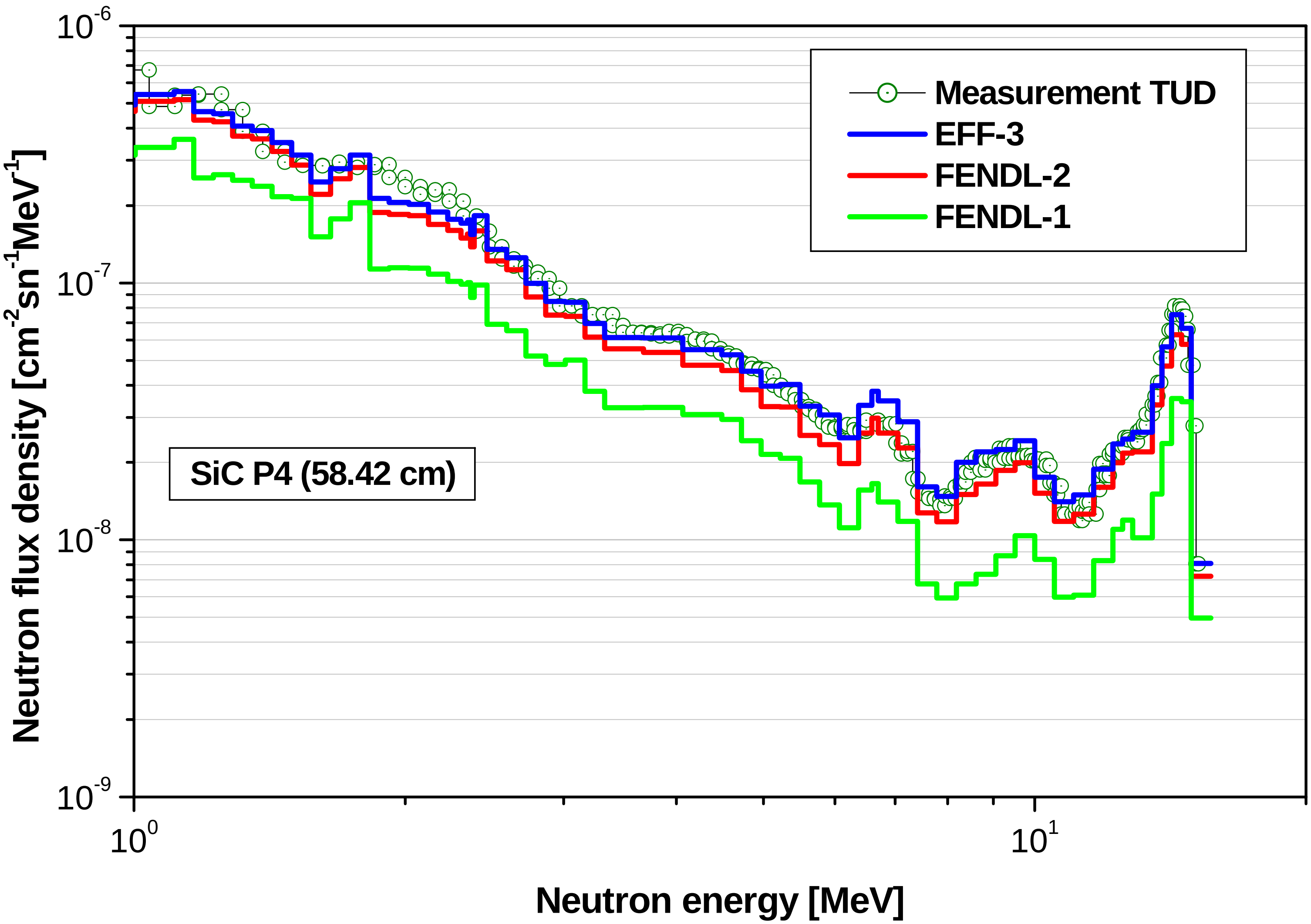
ITER Bulk Shield Experiment (FNG)



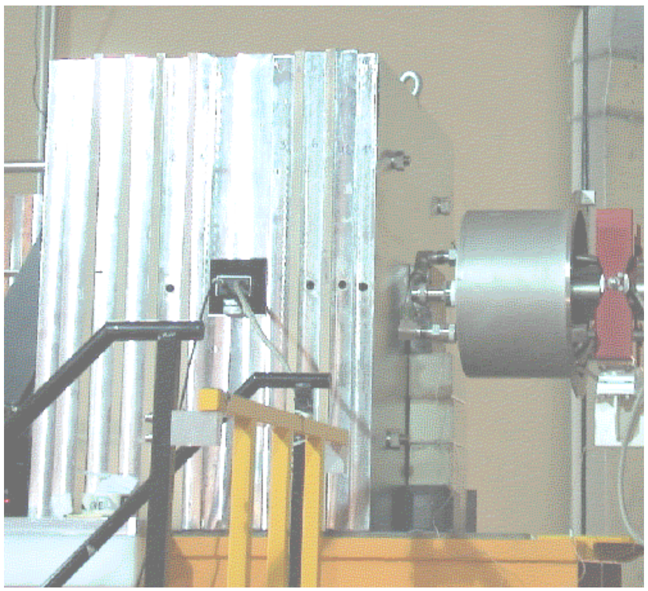
SiC transmission experiment (FNG)

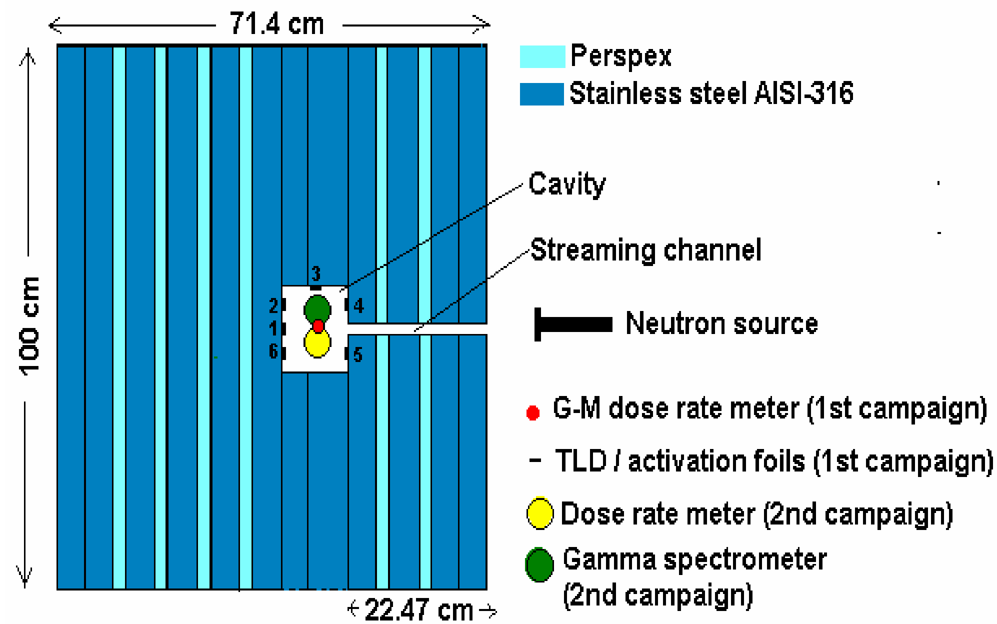


SiC transmission experiment (FNG)



SHUTDOWN DOSERATE EXPERIMENT





Objective: Verify shut down dose rate calculation for ITER out-vessel

SHUTDOWN DOSERATE EXPERIMENT cont.

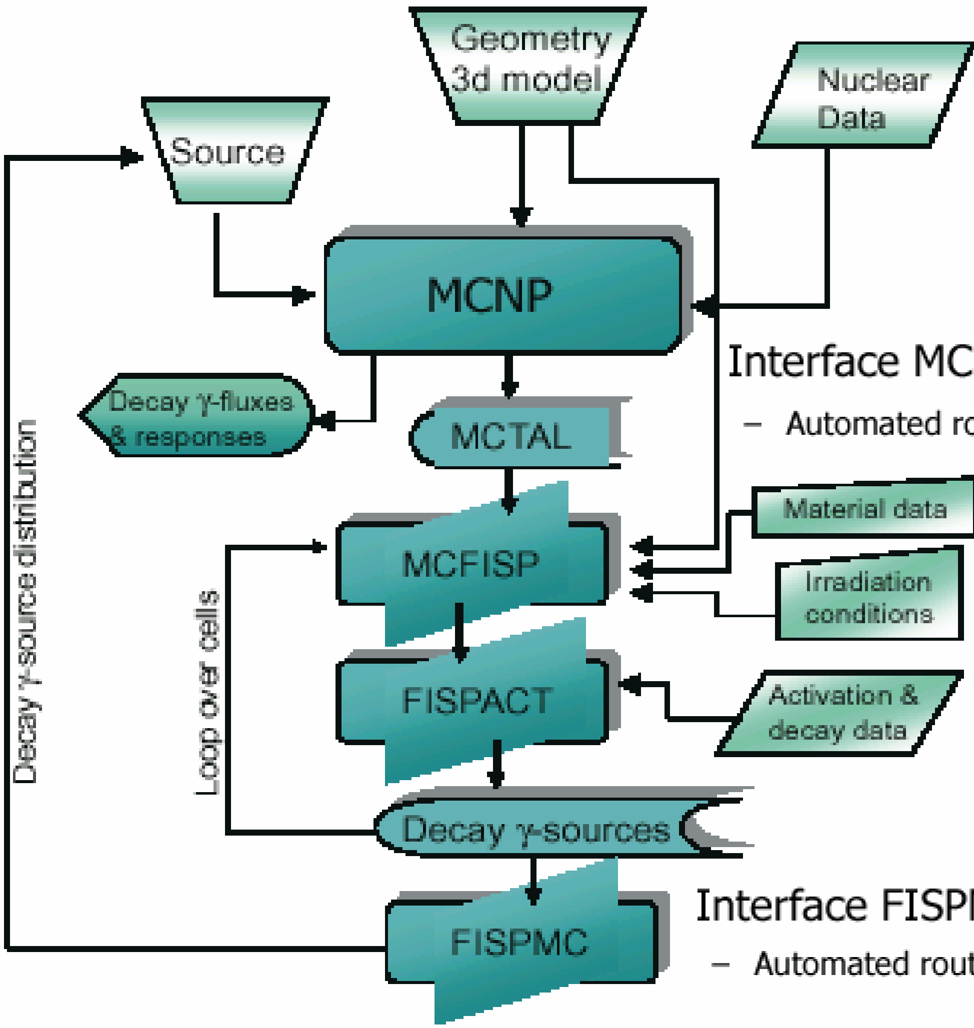
The γ-radiation originates from radionuclides produced by neutrons in the structural and coolant materials of the reactor during the operation. The calculation of the γ-dose rate for given positions comprises, in principle, a three-step procedure:

a) determination of the spectral neutron flux in the materials with transport code and data as MCNP [1] and FENDL[2],
b) calculation of the radioactivity induced by the neutrons as a function of irradiation and decay time with inventory code as FISPACT [3] and corresponding activation and decay data libraries,
c) γ-transport calculation from the activated materials to the position of interest and conversion of the flux to dose rate.

This is a very long calculation procedure for a complicate devices like ITER or DEMO!

RIGOROUS SHUTDOWN DOSERATE CALCULATION SCHEME (2 STEPS METHOD)

FZK development



Interface MCFISP: MCNP-FISPACT

– Automated routing of neutron flux spectra distribution

Interface FISPMC: MCNP-FISPACT

– Automated routing of decay gamma source distribution

C.R. Frascati

RIGOROUS SHUTDOWN DOSERATE CALCULATION SCHEME cont.

MCNP source routine for sampling decay gamma source distribution

- one single source sampling volume (!)

  sampling of point within given range [r_{min}, r_{max}, ϕ_{min}, ϕ_{max} z_{min}, z_{max}]
- check for source cell of sampled point
- weight assignment according to decay gamma source intensity

DIRECT MCNP BASED 1 STEP METHOD

ITER & ENEA TEAM development

- Replacement of prompt photons by decay gammas in MCNP transport calculation

  ⇒ *One single MCNP run for neutron & decay photon transport calculation*

- Ad hoc modified MCNP data files
  - Replacement of prompt photon production data (yields & spectra) by decay gamma data
  - Replacement of transport with activation cross-section data

- No activation calculation for decay gamma sources
  - Adjustment factors required for decay gamma source to assess correct decay gamma dose rate

  ⇒ *Depending on radio-nuclide, irradiation history and cooling times.*

Two steps method R2S (FZK)

MCNP

Neutron flux

FISPACT

Gamma sources in the cells

MCNP

Gamma doses

One step method D1S (ENEA - ITER)

Modified **MCNP**

Prompt gamma doses

Time correction factors

Gamma doses

C.R. Frascati

R2S and D1S limits comparison

R2S

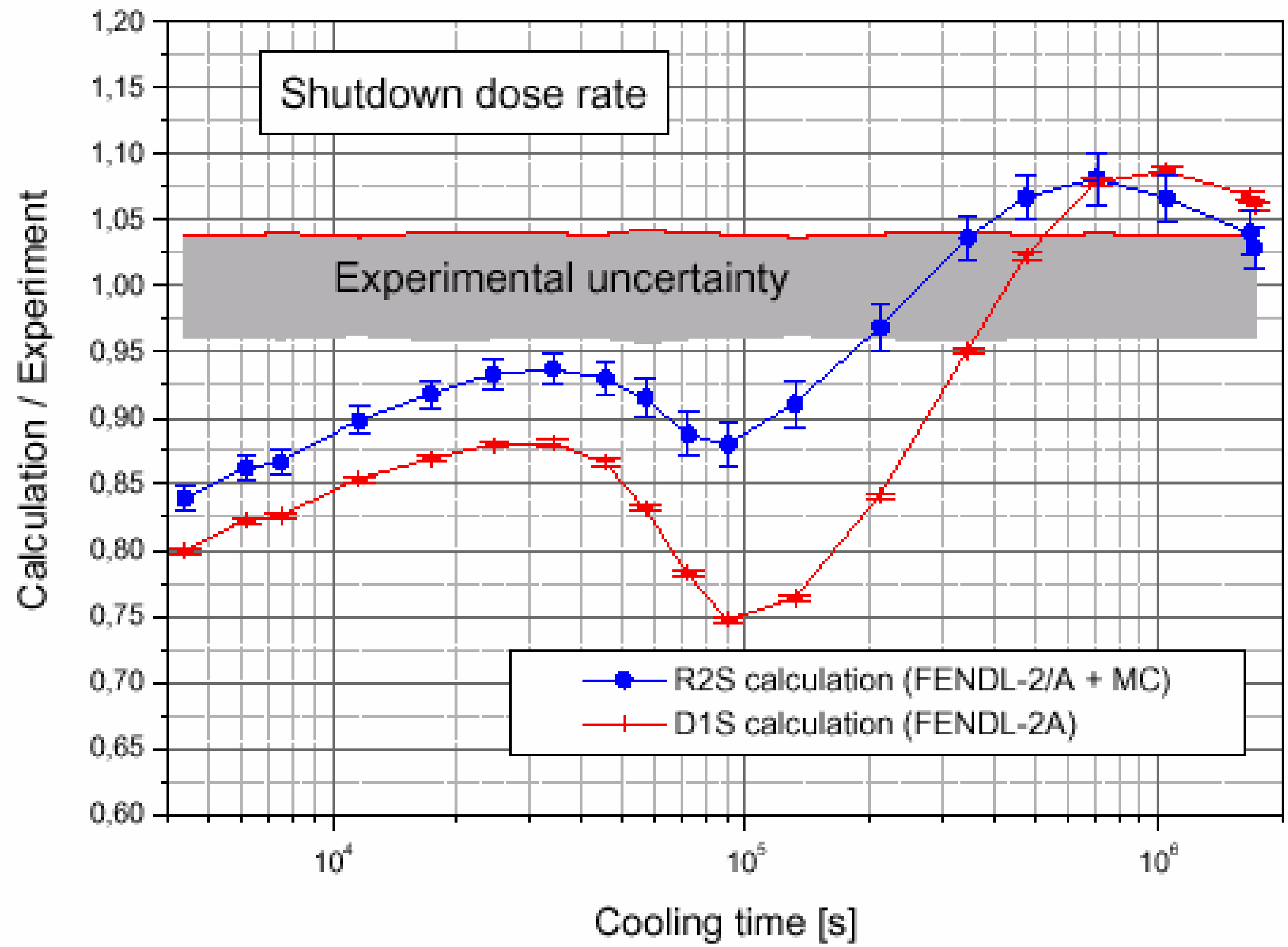
- Lost of energy information in activation cross section
- Use of cell average gamma sources

D1S

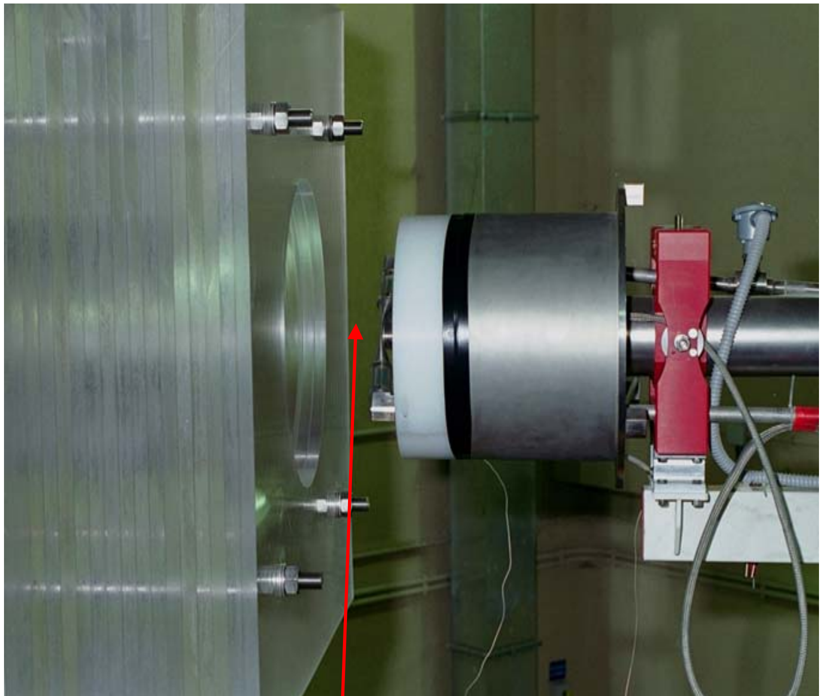
- Limited only to 1 step reactions
- Require an a priori choice of the dominant radionuclides produced (problem dependent)

ITER Shutdown Dose Rate Experiment (FNG)

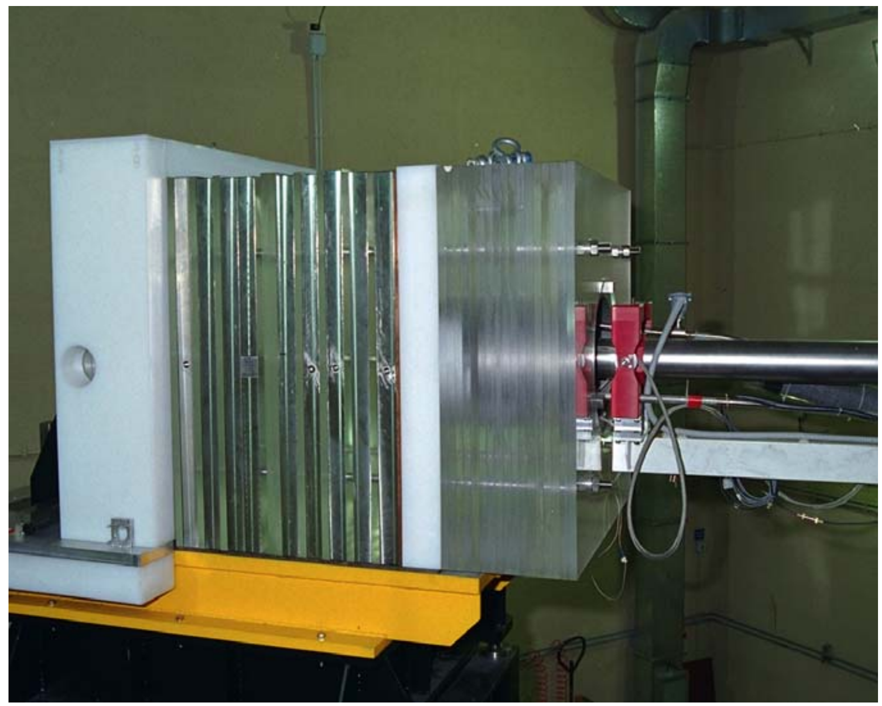
TUD measurement



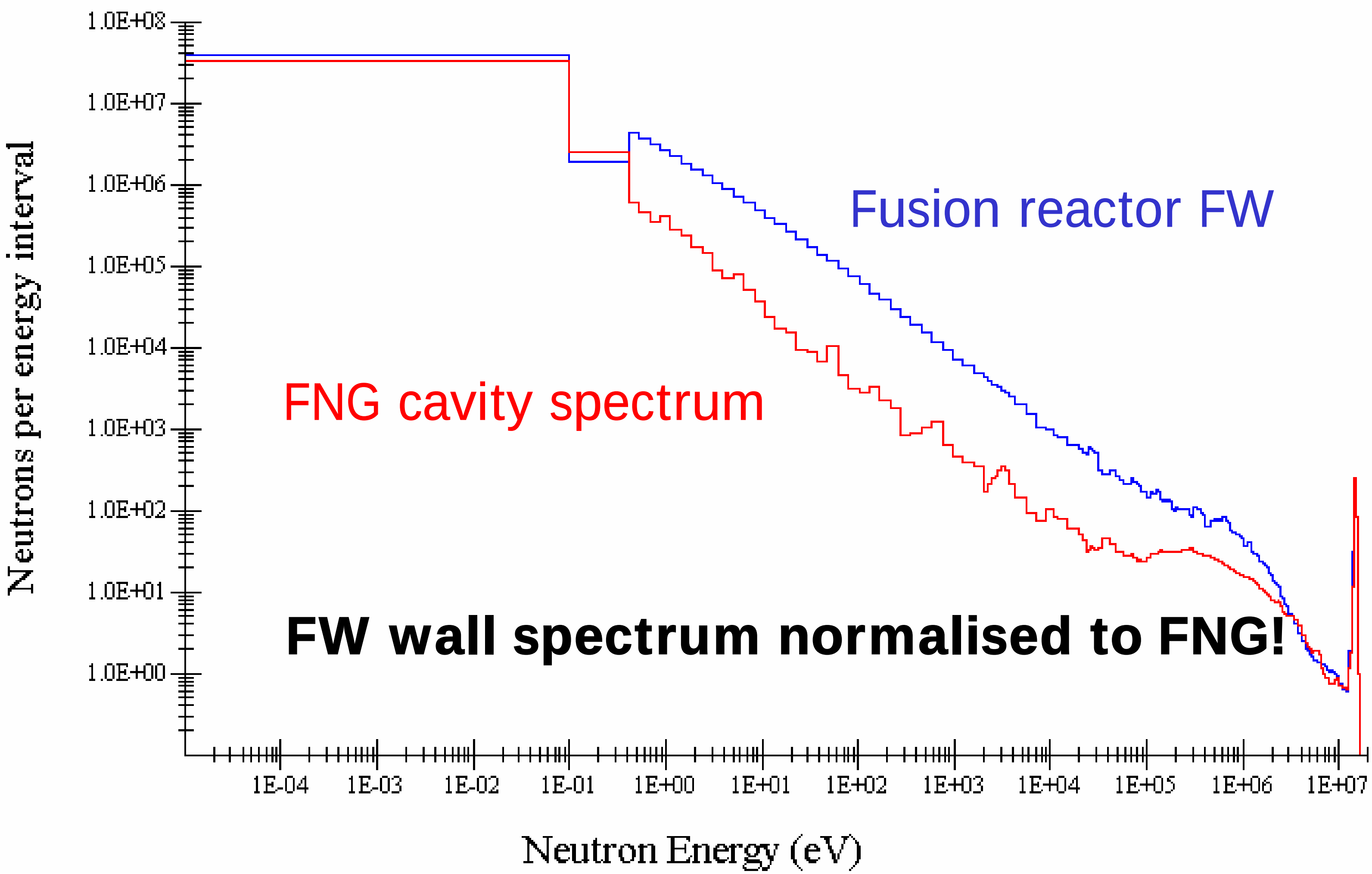
Integral Activation Experiments (FNG)



Samples irradiation position

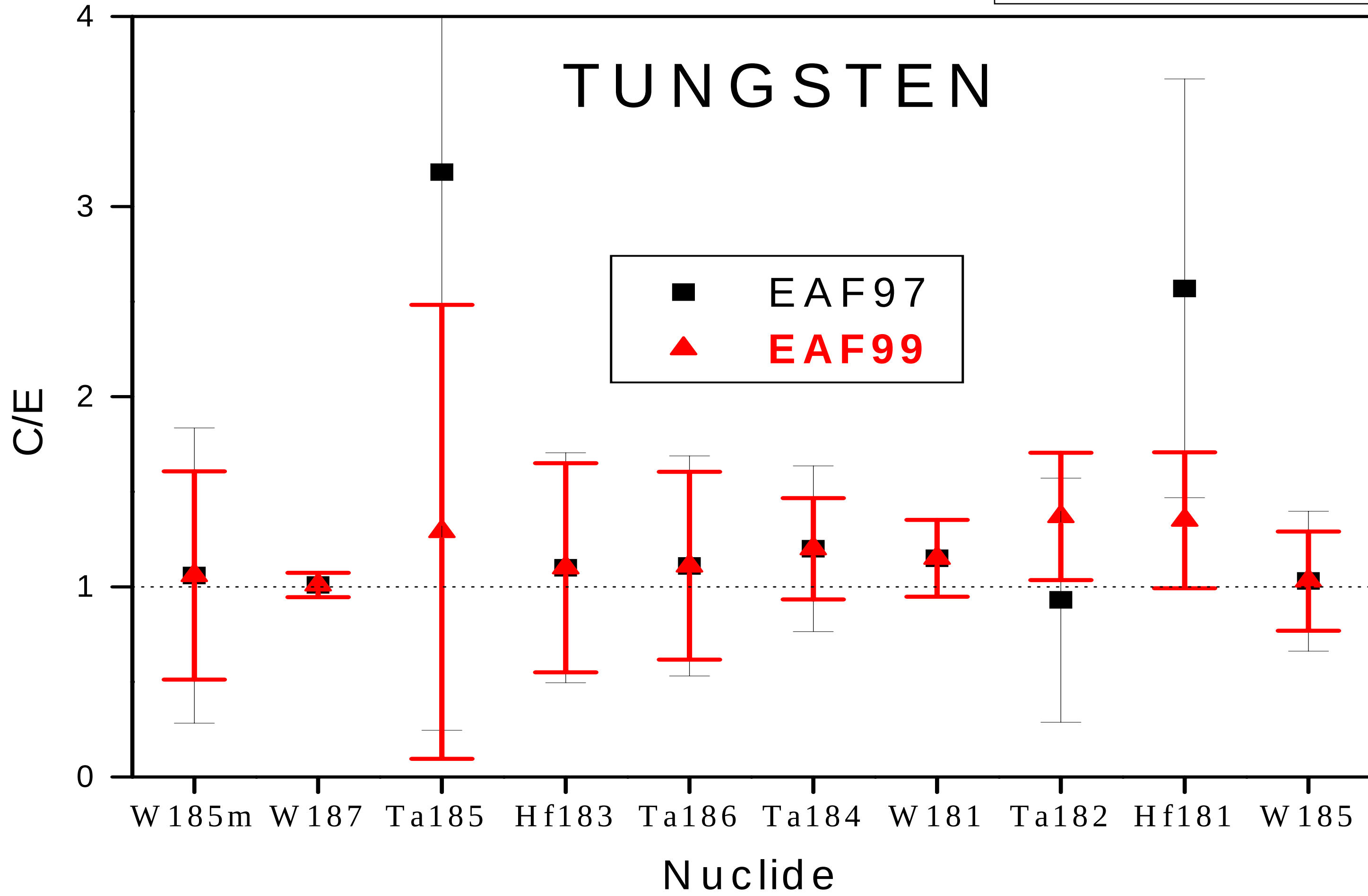


Integral Activation Experiments cont



Integral Activation Experiments (FNG)

ENEA measurement

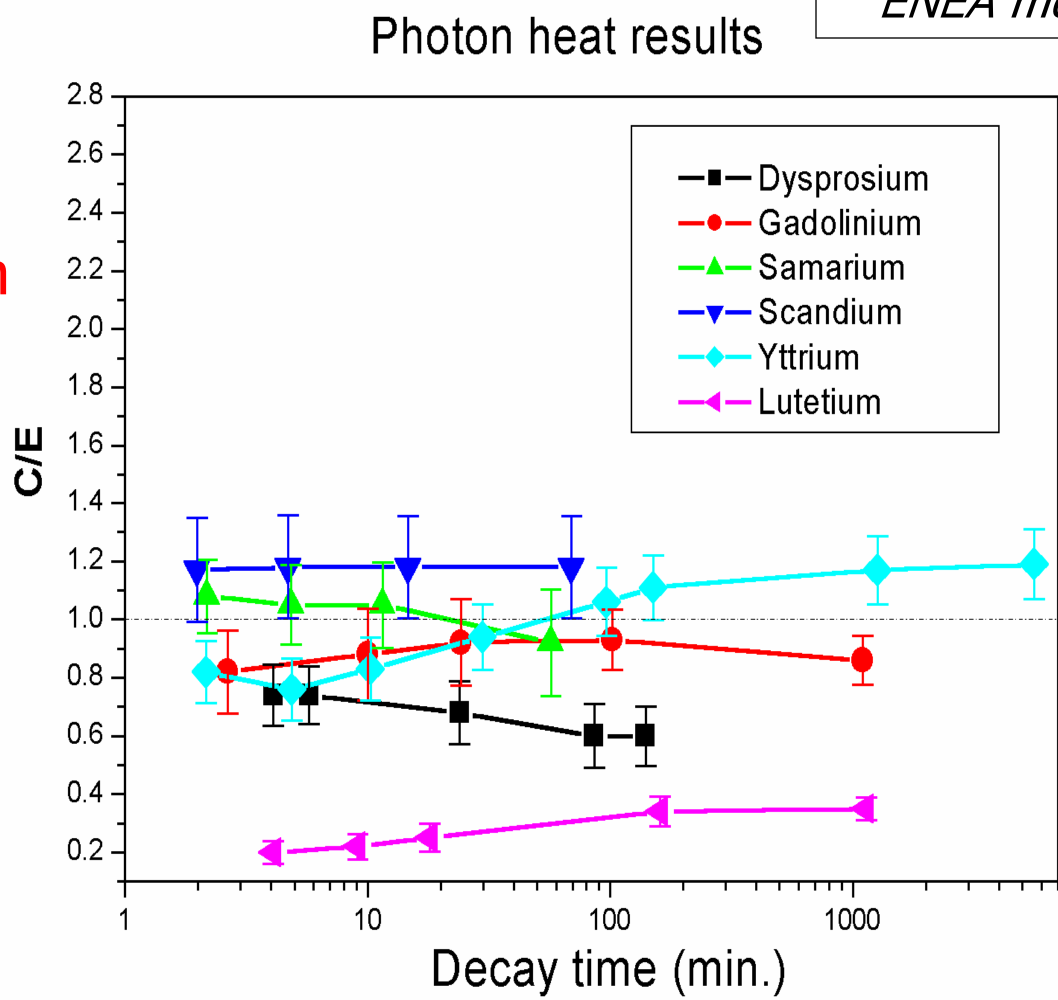


Integral Decay Heat (FNG)

ENEA
C.R. Frascati

ENEA measurement

EAF-2001
comparison

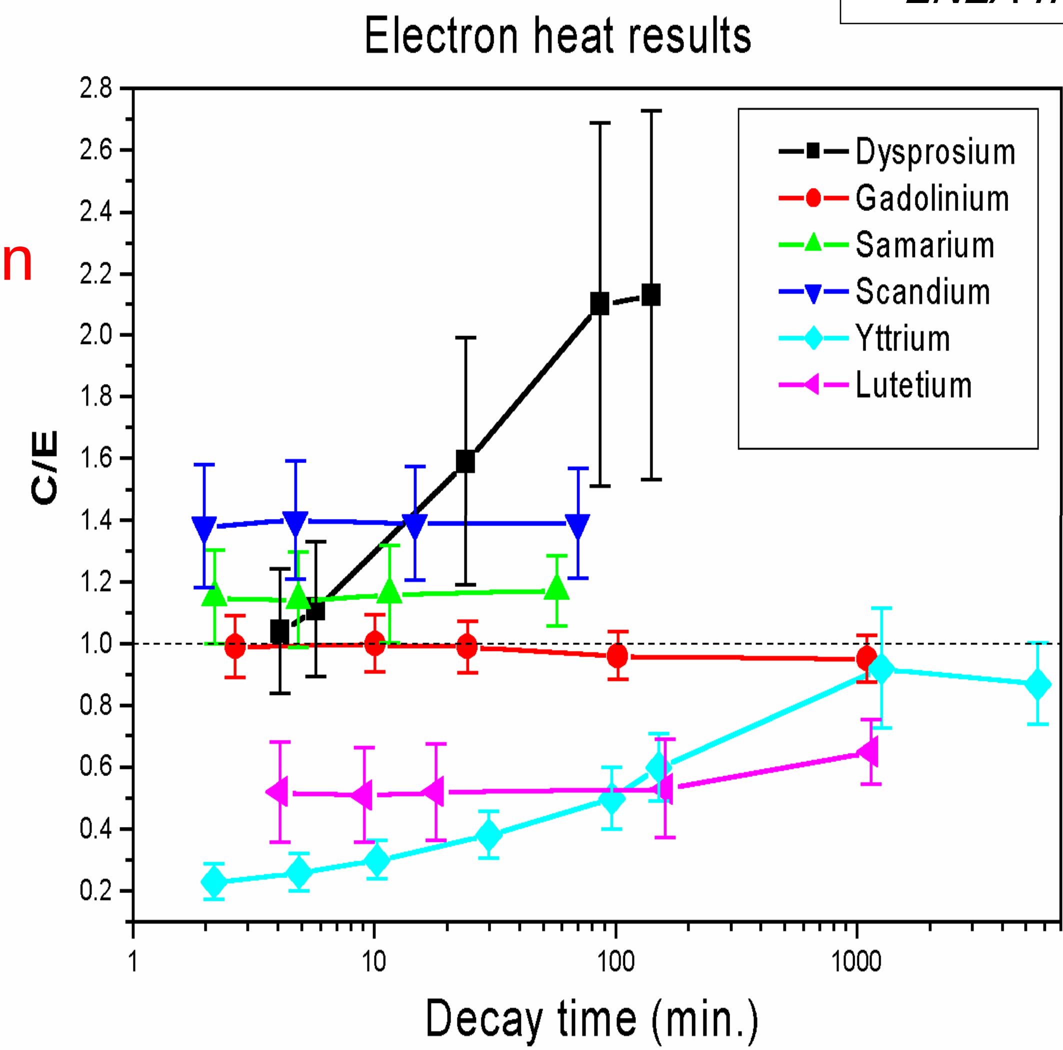


Integral Decay Heat (cont.)

C.R. Frascati

ENEA measurement

EAF-2001 comparison



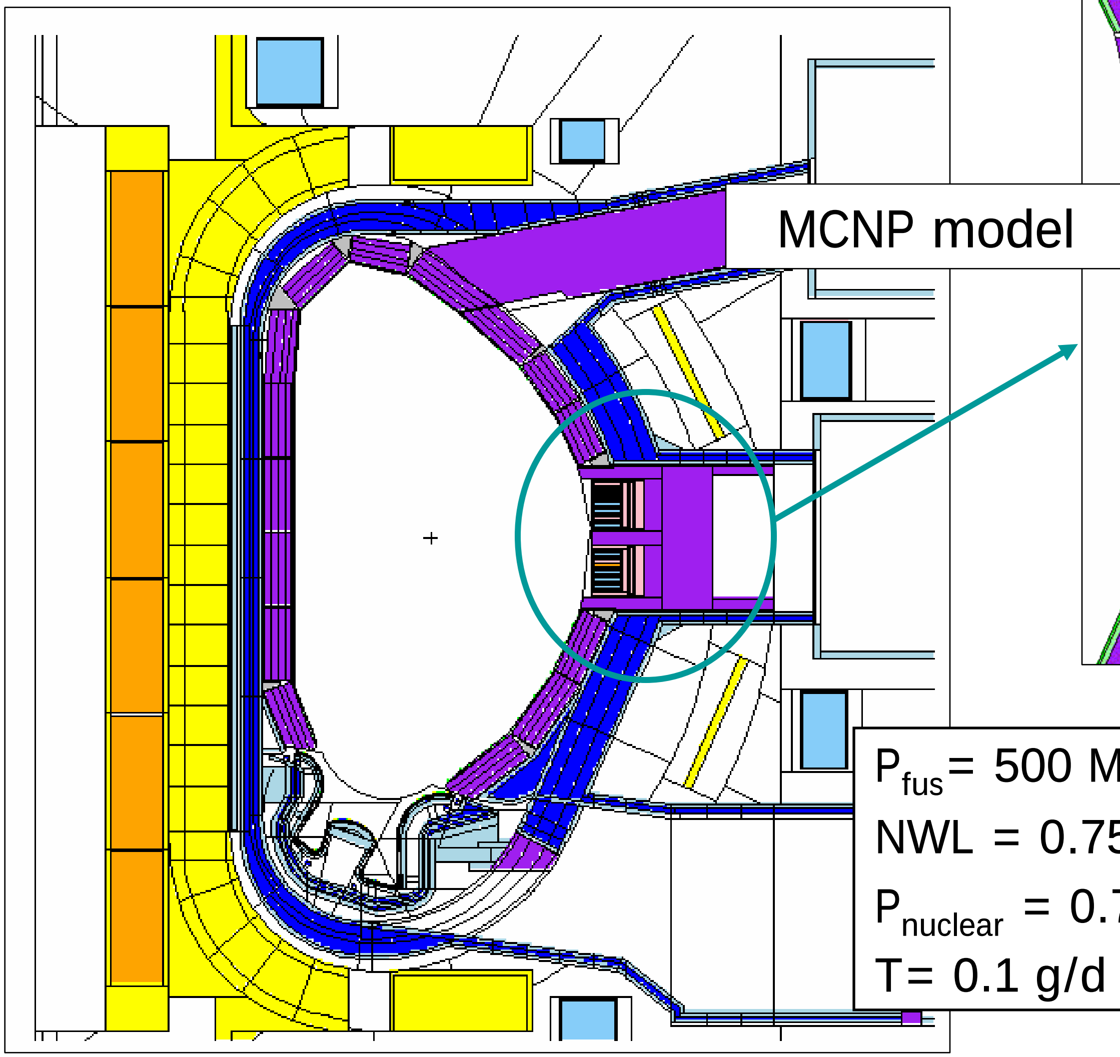
Fusion Nuclear Data Perspectives

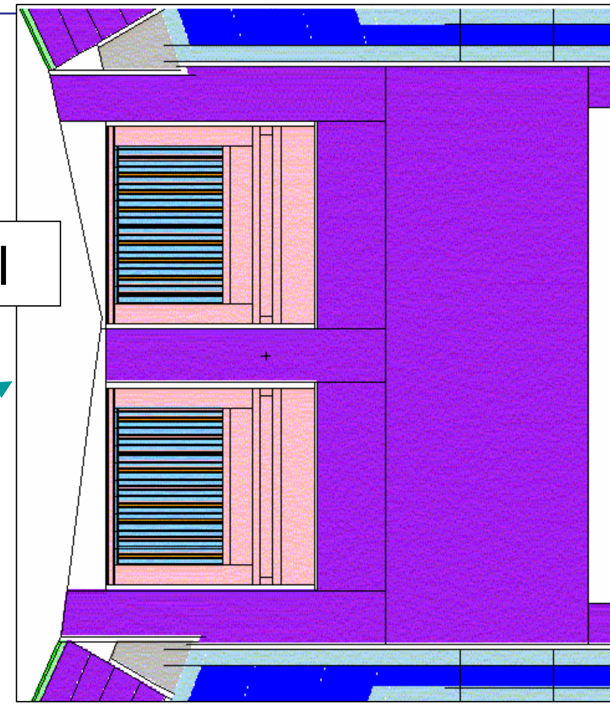
- EU fusion technology strategy
  - Long-term orientation towards fusion power plant
  - Short-term focus on ITER & IFMIF
- EU Framework Programme 6 (2003-2006)
  - Design of ITER Test Blanket Modules (TBM)
  - IFMIF intense neutron source
- Nuclear Data Related Activities
  - Fusion Nuclear Data related R&D work integral part of fusion technology programme, materials task area
  - Strategy paper basis for Nuclear Data programme in FP6 and beyond (EFF-DOC 828, April 2002)
    - $\Rightarrow$ *Required effort on data evaluation & computational tools*
    - $\Rightarrow$ *Need for integral benchmark experiments*

ITER TBM Design

- Objectives for TBM testing in ITER
  - Demonstrate tritium breeding performance and on-line tritium recovery
  - Demonstrate high-grade heat extraction & integral performance of blanket system
  - Validate computational tools and data for complex geometrical configuration $\Rightarrow C \pm \delta C$ *vs.* $E \pm \delta E$
- TBM design and fabrication of mock-ups in FP6
- Based on EU blanket concepts
  - Helium-cooled pebble bed (HCPB) blanket
  - Water or helium cooled lithium lead (W/HCLL) blanket

HCPB Test Blanket Module in ITER





P_{fus} = 500 MW ($\approx 1.78 \cdot 10^{20}\ s^{-1}$)

NWL = 0.75 MW/m^2

$P_{nuclear}$ = 0.7 MW (TBM)

T= 0.1 g/d ($\approx 1.4 \cdot 10^{-3}$ at. /sn^{-1})

TBM Design Related Activities in FP6

Data Evaluations and Computational Tools

- Nuclear data for TBM design and ITER
  - Review of available cross sections and co-variance data for major TBM materials such as Li, Be, Al, Si, Ti, O, Pb, Fe, Cr, W, Ta.
  - New and/or updated data evaluations where needed.
  - Focus on co-variance data evaluation
  - Processing and benchmarking.
- Monte Carlo TBM sensitivity/uncertainty analysis
  - Development of algorithms for track length estimator and implementation into MCNP/MCSEN

C.R. Frascati

TBM Design Related Activities in FP6

Benchmarking and Integral Experiments

- TBM neutronics experiment
  - HCPB or W/HCLL neutronics mock-up
    - ⇒ Neutron generator(s), possibly JET
  - Measurement & analysis of tritium generation, nuclear heating, neutron and γ-spectra, activation & afterheat
  - Sensitivity/uncertainty analysis
- Integral activation experiments
  - Validation experiments for selected TBM materials
  - Activation & decay heat measurements

IFMIF Intense Neutron Source

- Dedicated to high fluence irradiations of fusion reactor materials at fusion relevant conditions.
- Li(d,xn) reaction to produce high energy neutrons
  - E_d=40 MeV, $E_{n,\,max} \leq$ 55 MeV, source intensity $\approx$1.1 10^{17} s^{-1}
- IEA Task Agreement (EU, J,US, RF)
  - Conceptual Design Activity (CDA) 1995-96
  - Conceptual Design Evaluation (CDE) 1997-99
  - Key Evaluation Technology Phase (KEP) 2000 – 2002
- Engineering Design & Validation Activity (EVEDA) to be started in 2004

IFMIF Nuclear Design

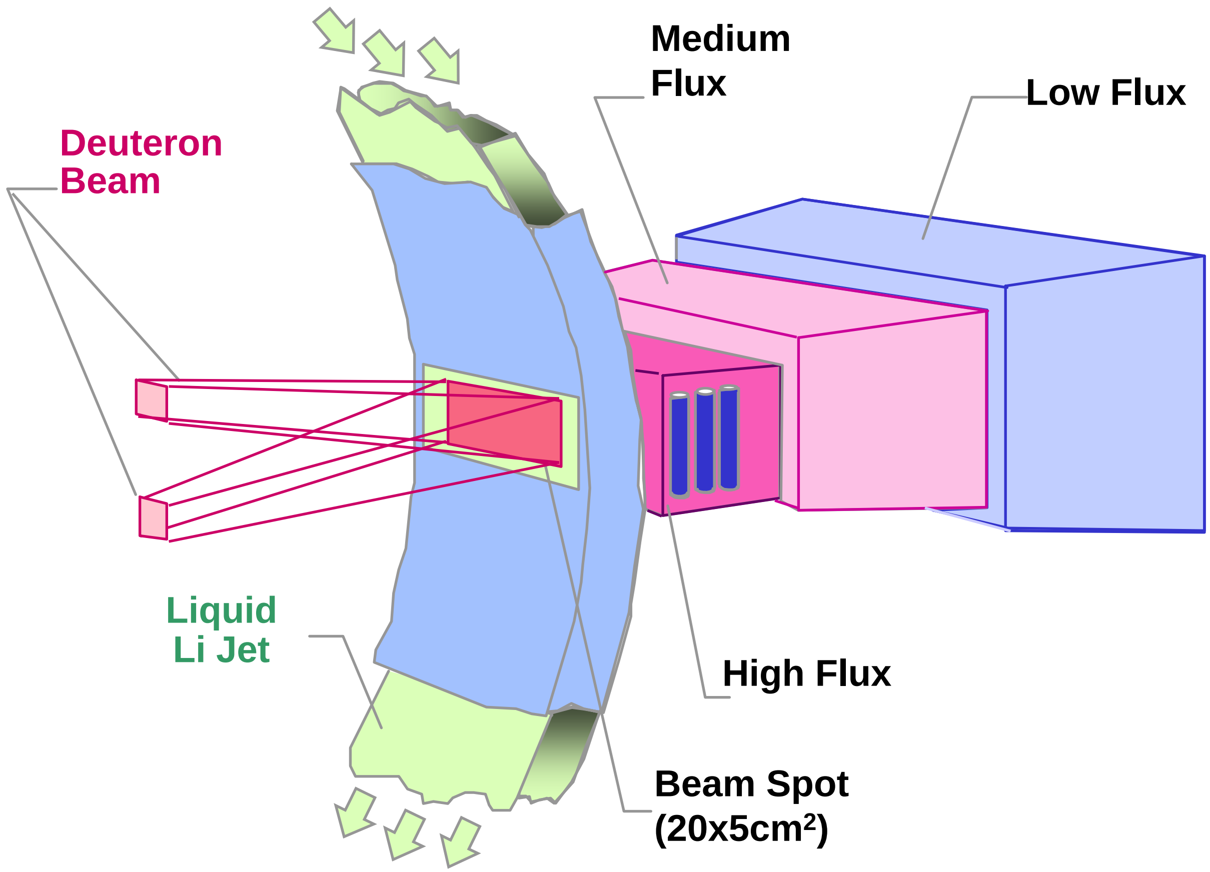
Key role of neutronics & nuclear data in establishing IFMIF as neutron source for fusion material testing

- Suitability as neutron source for fusion reactor material test irradiation
  - to be proven by means of neutronic calculations

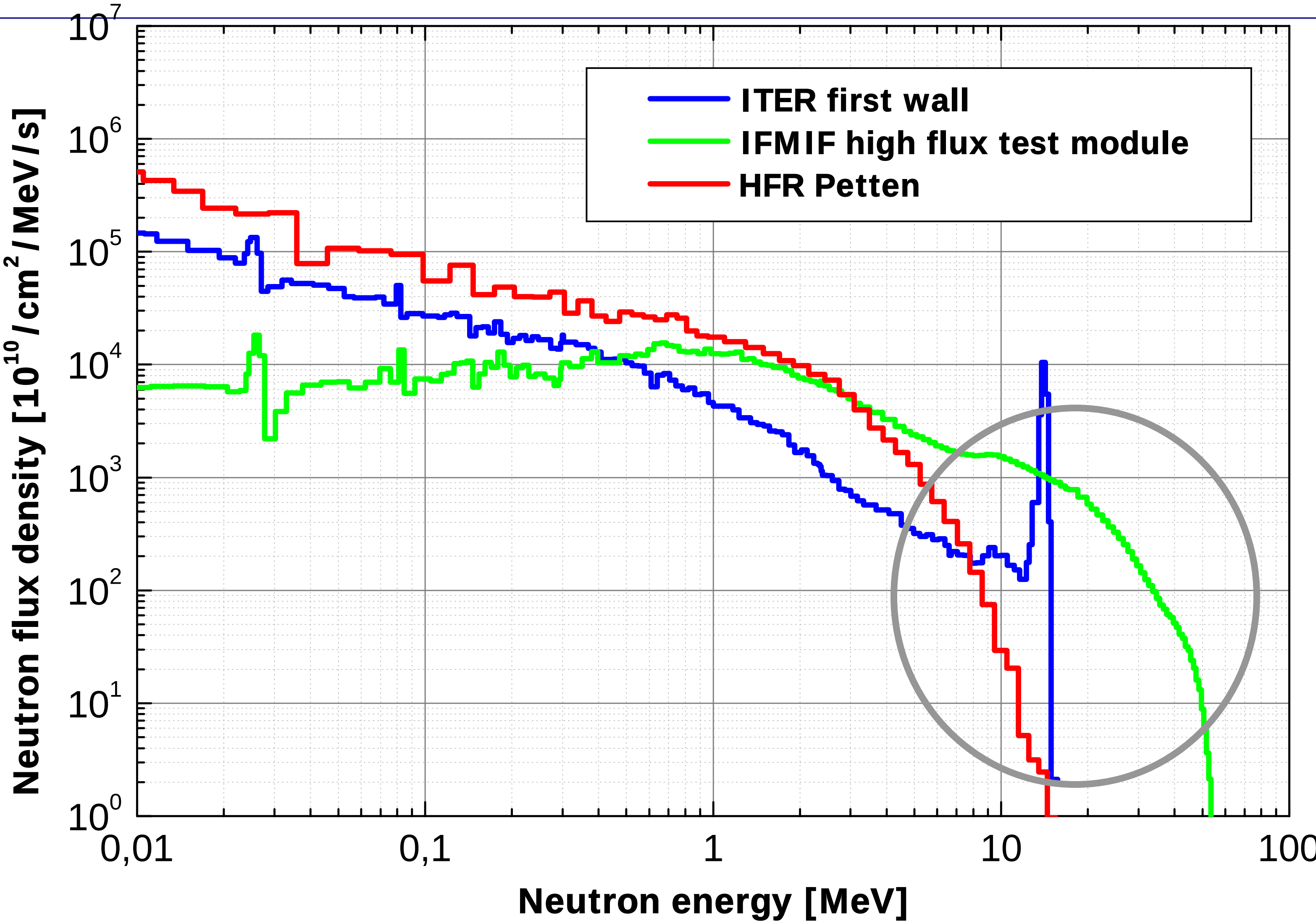
- Technical lay-out of test modules & sub-systems
  - relies on data of neutronic design calculations

$\Rightarrow$ *IFMIF design goal: irradiation test volume 0.5 L with at least 20 dpa per full power year.*

IFMIF Intense Neutron Source



Neutron flux spectra: IFMIF/fusion/fission



IFMIF Related Activities in FP6

Extension of nuclear data libraries to E_n> 20 MeV

- General Purpose Nuclear Data File (ENDF) for Transport Calculations up to 150 MeV
  - New and/or updated complete evaluation of high priority data
  - To be adapted to existing EFF/JEFF evaluations below 20 MeV
  - Other data evaluations from other sources and model calculations.

  ⇒ *Complete general-purpose 150 MeV data library to be prepared within FP6 (!)*

- Activation Data File(s)
  - Extension of EAF-2003 data library to 55 MeV covering cross-section and decay data.

    NB. 55 MeV sufficient for IFMIF activation analysis; allows individual reaction channels to be handled in traditional way by FISPACT
  - IEAF-2001 (150 MeV) available for activation analysis of IFMIF and other high energy neutron sources by using ALARA code

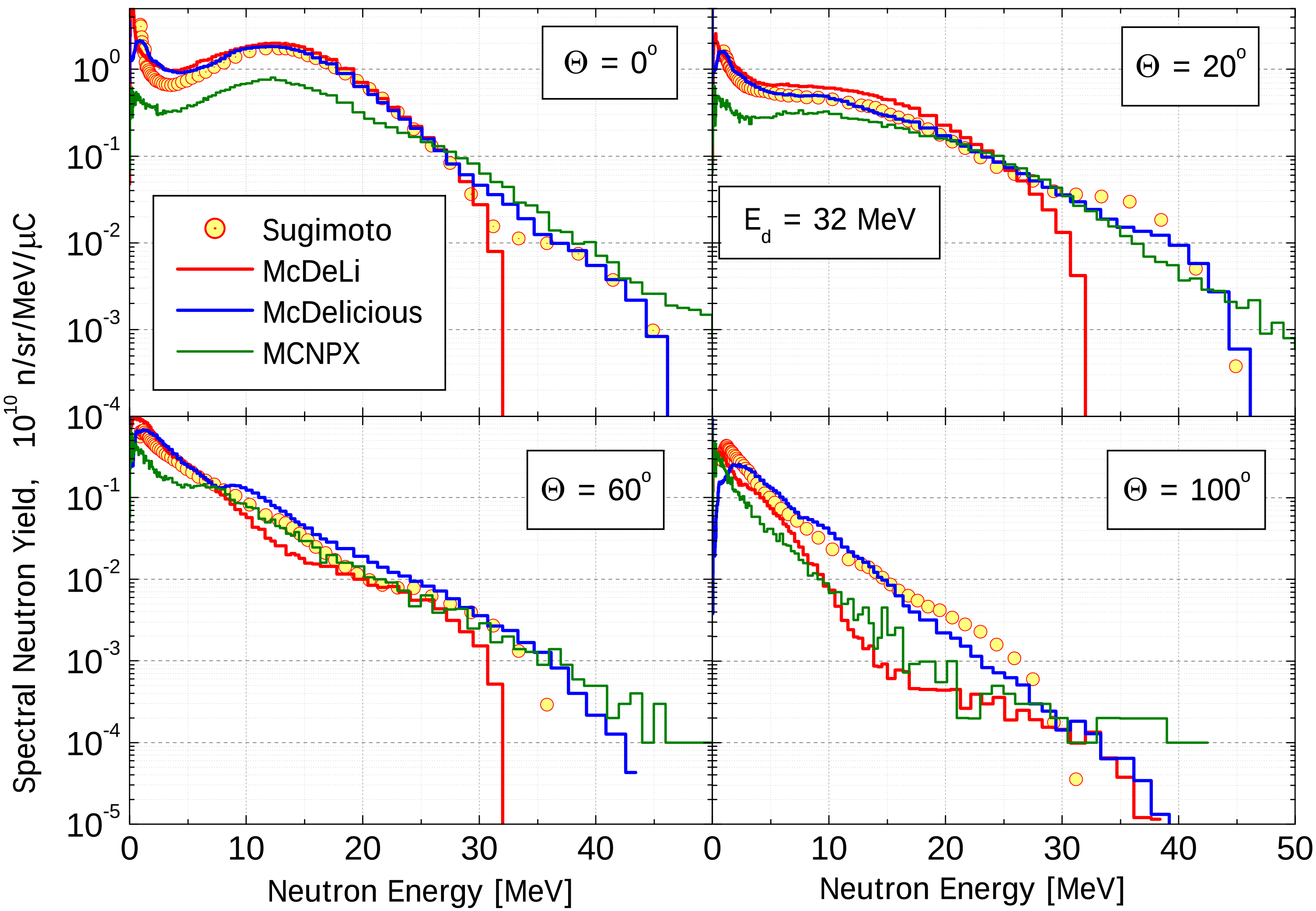
IFMIF Related Activities in FP6

Benchmarking and Integral Experiments

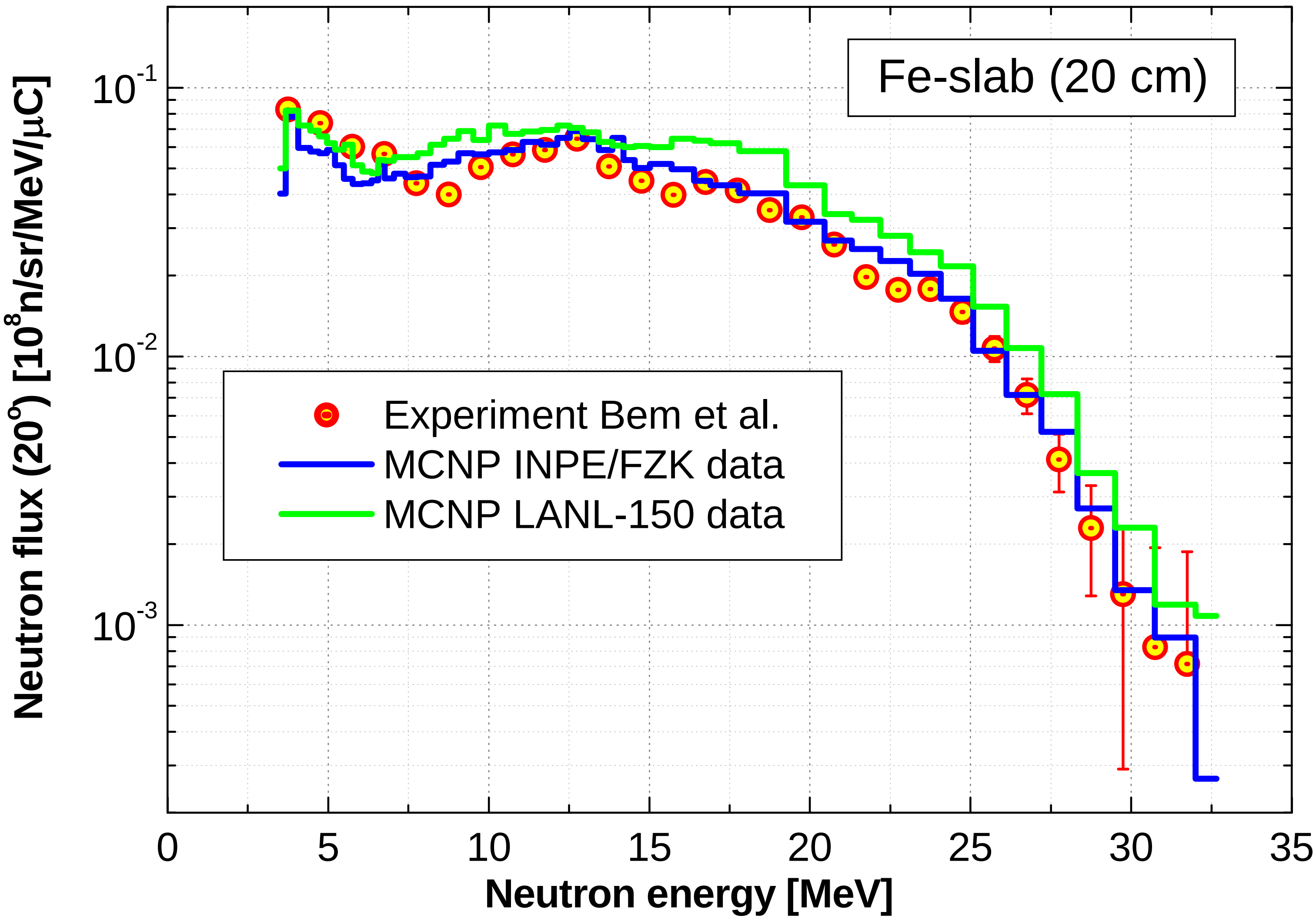
- D-Li neutron source term simulation
  - Validation of "M^{c}DeLicious" approach based on new experimental thick Li target data

    Sampling of D-Li source neutrons in Monte Carlo calculation based on d+ ^{6,7}Li data
  - -- Up-dating of ^{6,7}Li + d data evaluation
- Integral Activation Experiments
  - Validation of activation cross-sections in IFMIF-like neutron spectrum
  - *NPI Rez cyclotron, ENEA cyclotron (?)*
- Transport Benchmark Experiments
  - Validation of transport data in IFMIF-like neutron spectrum

D-Li thick target neutron yield spectra



NPI Rez Neutron Transport Benchmark



Conclusions & Outlook

Neutronics and Nuclear Data for Fusion Technology

- " Status"
  - Computational methods & tools well established
  - Qualified nuclear data libraries developed
- " Perspectives"
  - Continuous effort for adapting tools & data to changing user needs, thereby improving and maintaining their quality.
  - IFMIF Intense Neutron Source
    - Extension of nuclear data libraries to E_n> 20 MeV
    - Validation experiments
  - ITER Test Blanket Module(s)
    - Uncertainty analysis: MC tools & co-variance data
    - Neutronics mock-up & activation experiments