NEUTRON FLUX MEASUREMENTS WITH THRESHOLD DETECTORS

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OUTLINE

Motivation

Physical background

- Neutron flux, reaction rate, activation
- Typical neutron spectrum of a thermal reactor
- Typical cross section of different reactions

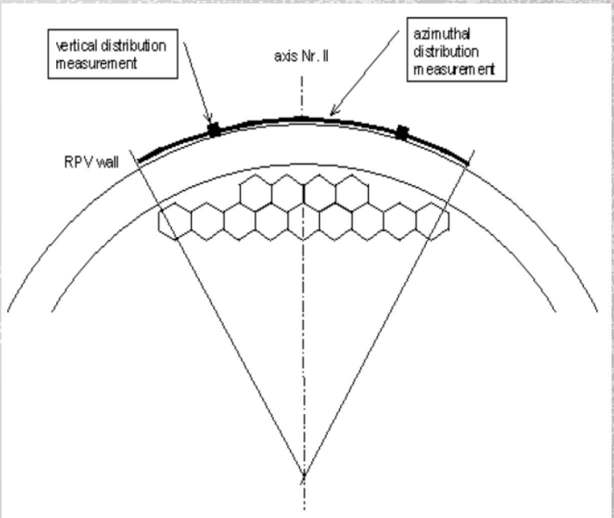
Measurement

Determination of the neutron spectrum

- Spectrum unfolding
- Information about the correlation matrices

MOTIVATION

Why is it essential to map the <u>neutron spectrum</u> in a nuclear reactor zone?

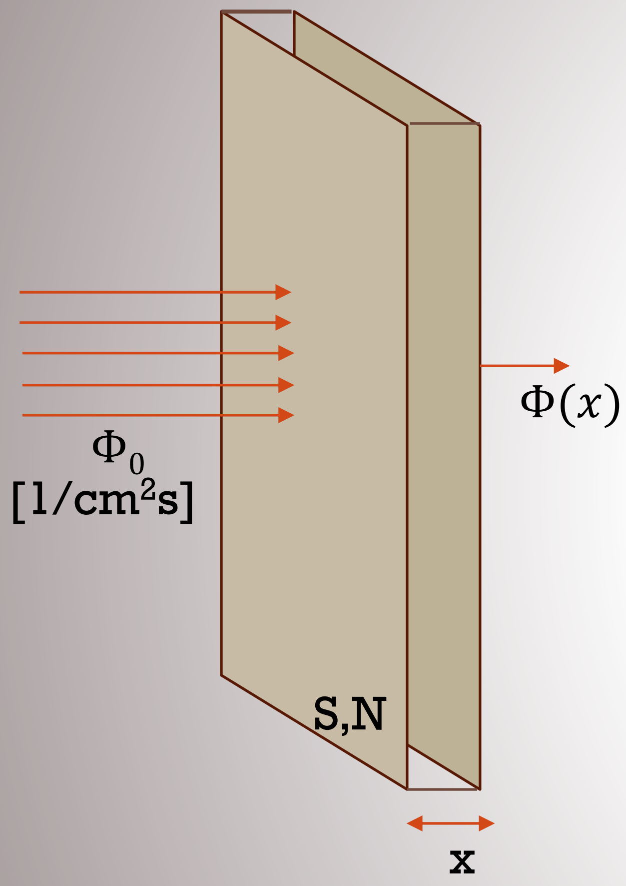


[2]

- **Damage** assessment of the vessel:
  - High-energy neutrons may alter the structure of the **steel vessel** by the activation of Fe nuclei
- Experimental nuclear physics:
  - Checking the validity of **neutron transport** codes inside the zone

PHYSICAL BACKGROUND

- Neutron flux (Φ)
- Reaction cross section (σ)
- Reaction rate (R)
- Activation ($A_I(t)$)



S: surface
N: number of target nuclei

- **σ**: refers to the probability that a certain reaction takes place

$$\sigma = \frac{R}{\Phi N}$$

- with **R** being the number of reactions taking place per one second

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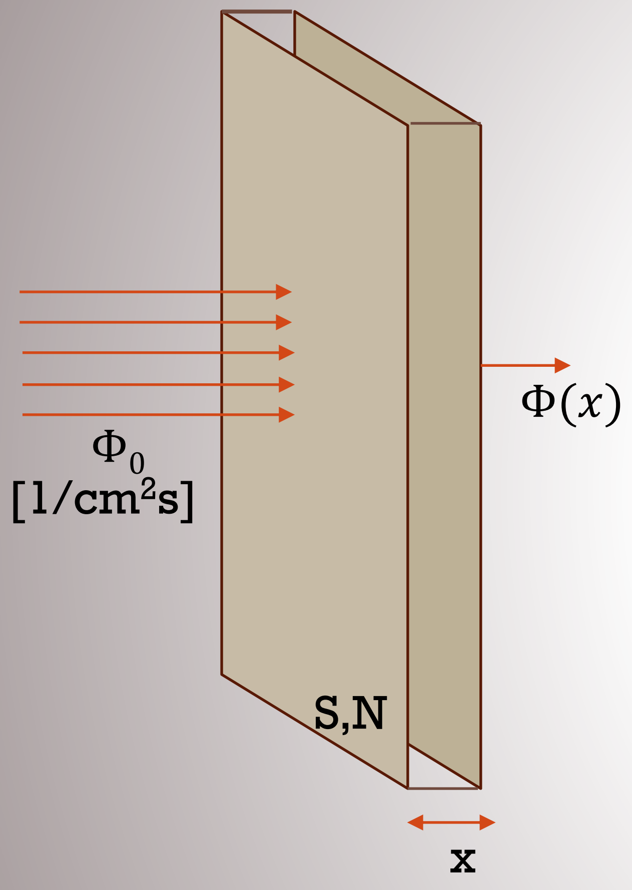

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The cross section depends on the energy of the incident neutron!

$$\boldsymbol{\sigma(E) = \frac{R}{\Phi(E) N}}$$

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- Reaction cross section (σ)
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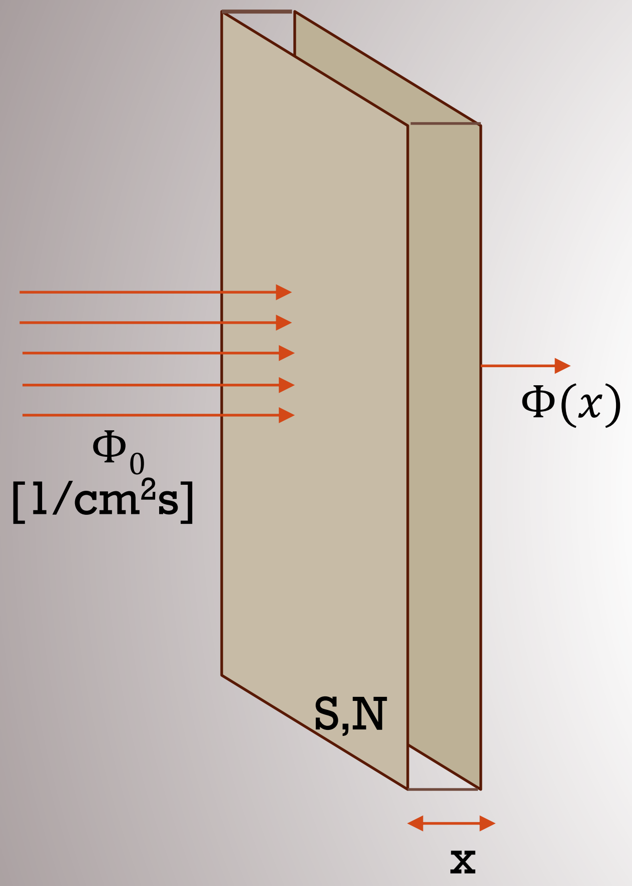


S: surface
N: number of target nuclei

The cross section depends on the energy of the incident neutron!

$$\sigma(E) = \frac{R}{\Phi(E)N}$$

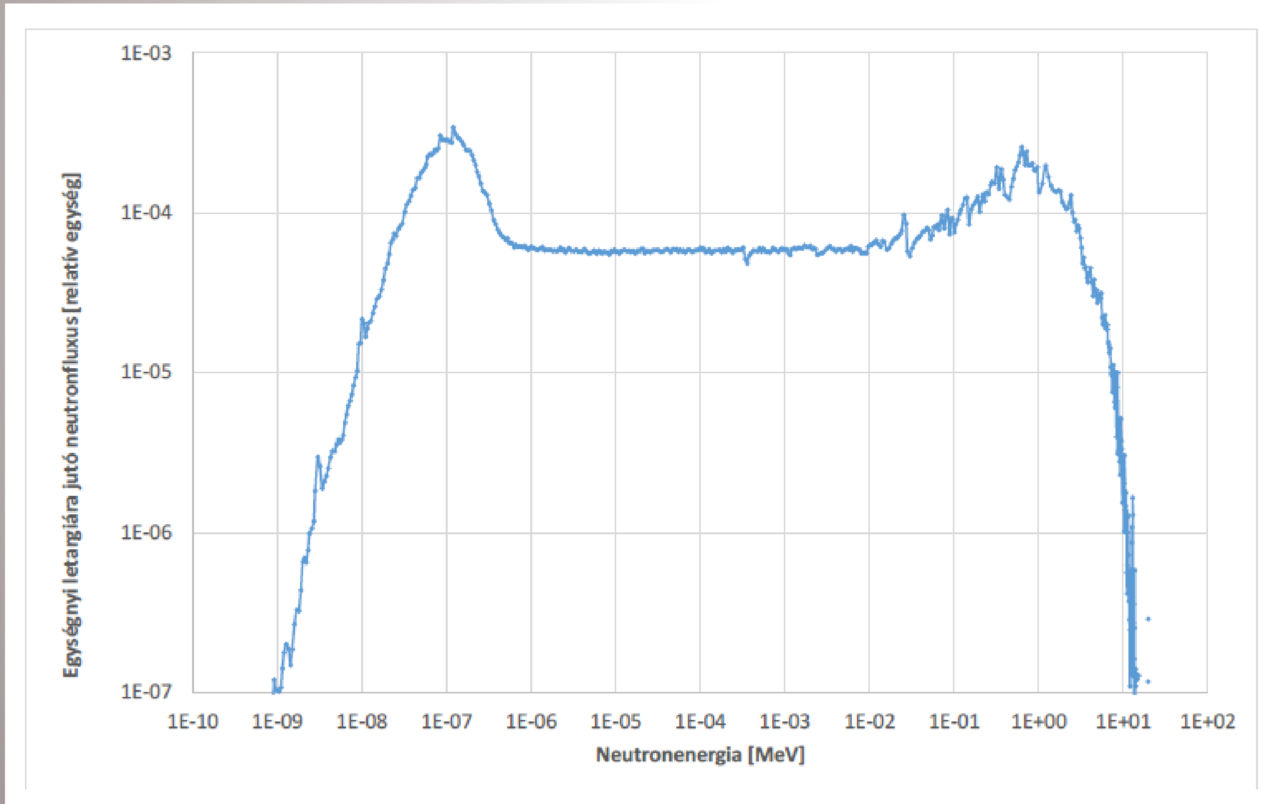
- the activity of the irradiated nuclei:
  - $A_I(t)$ gets saturated rapidly when the **half-life** of the irradiated nucleus is **too short**
    $A_I(t \to \infty) \to R$ (maximal)

WHAT DO WE MEASURE? –

$$\frac{d\Phi(E)}{dE};\ \sigma(E);\ R$$

The reaction rate

Neutron **spectrum** in a VVER-440 reactor: $\frac{d\Phi}{dE}$



Neutron flux over unit of lethargy -> measure of neutron slowing

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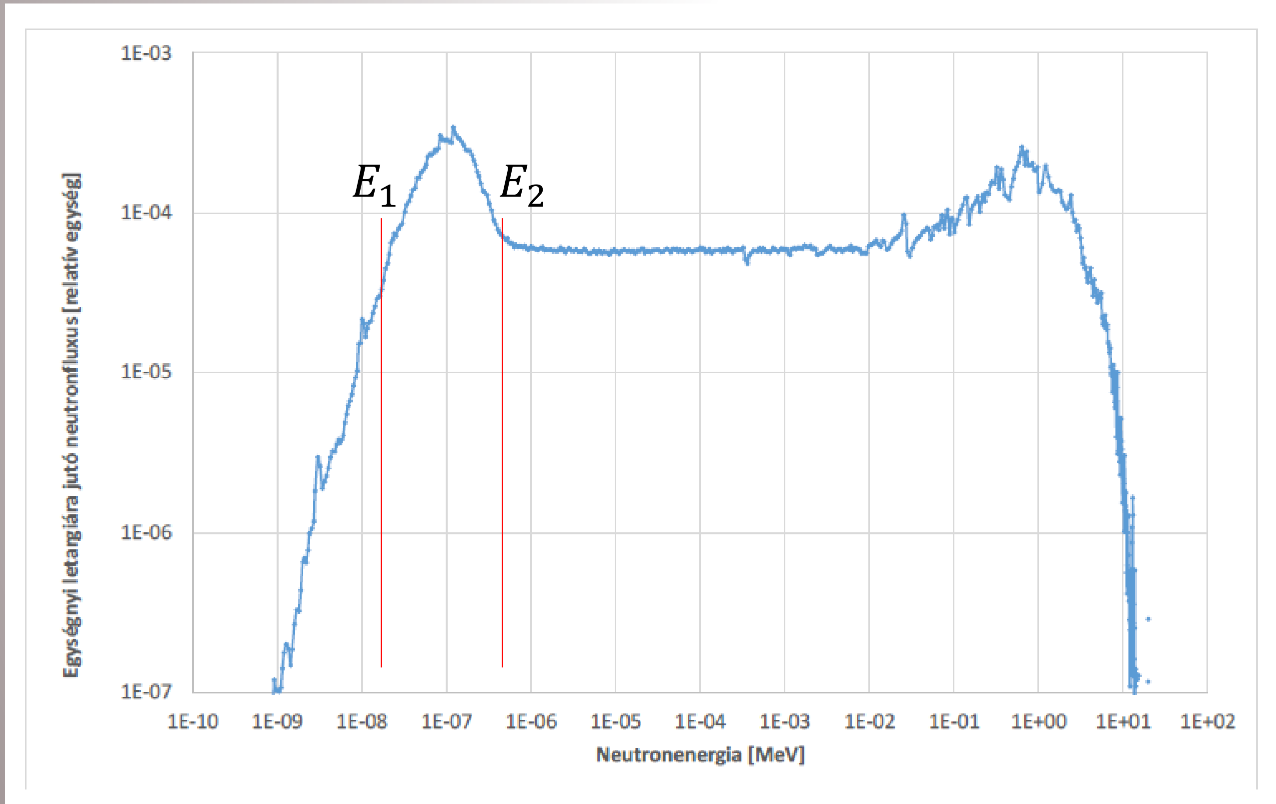
The reaction rate

Neutron **spectrum** in a VVER-440 reactor: $\frac{d\Phi}{dE}$



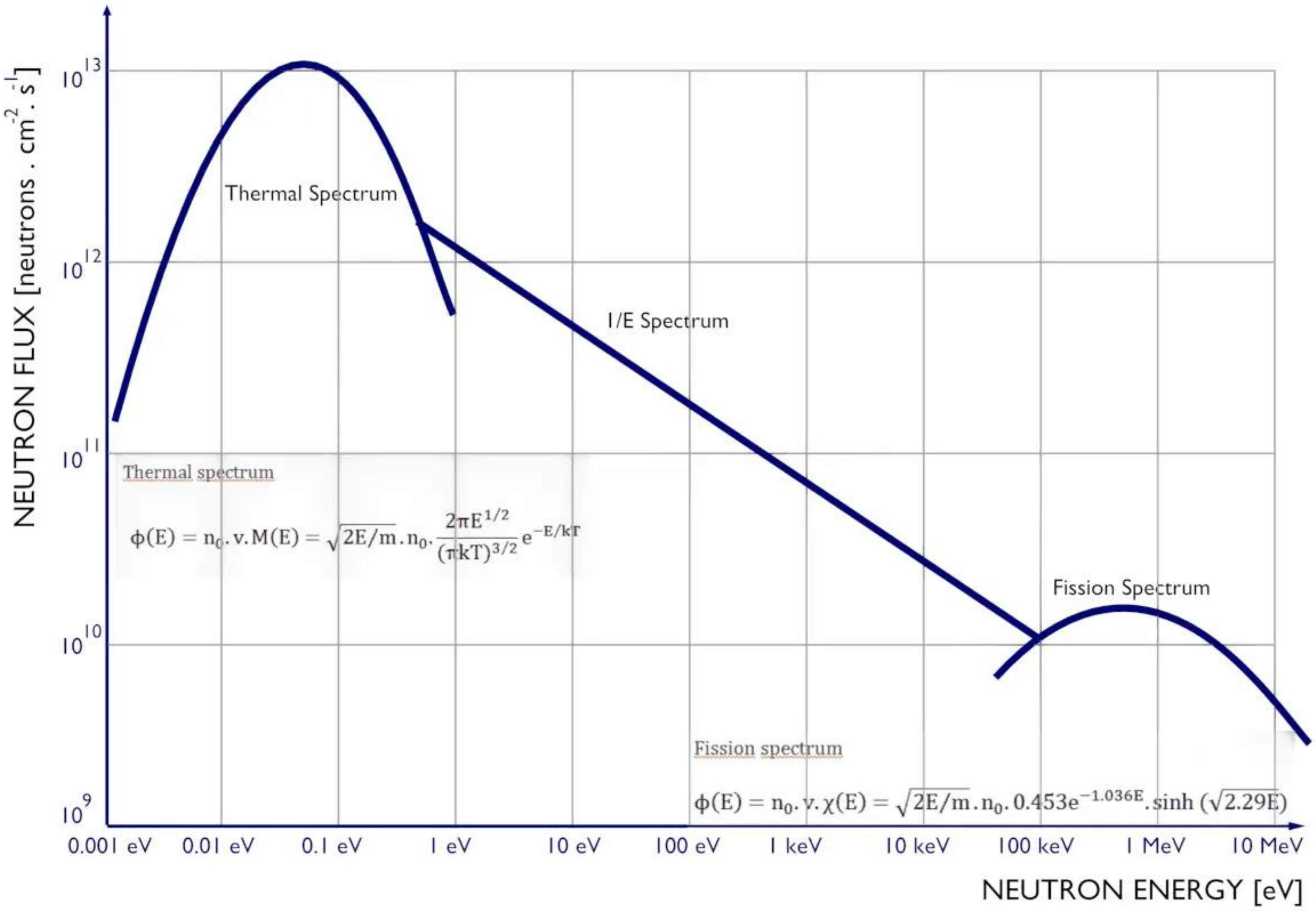
Neutron flux over unit of lethargy -> measure of neutron slowing

$$dR = \frac{d\Phi(E)}{dE} N\sigma(E)dE$$

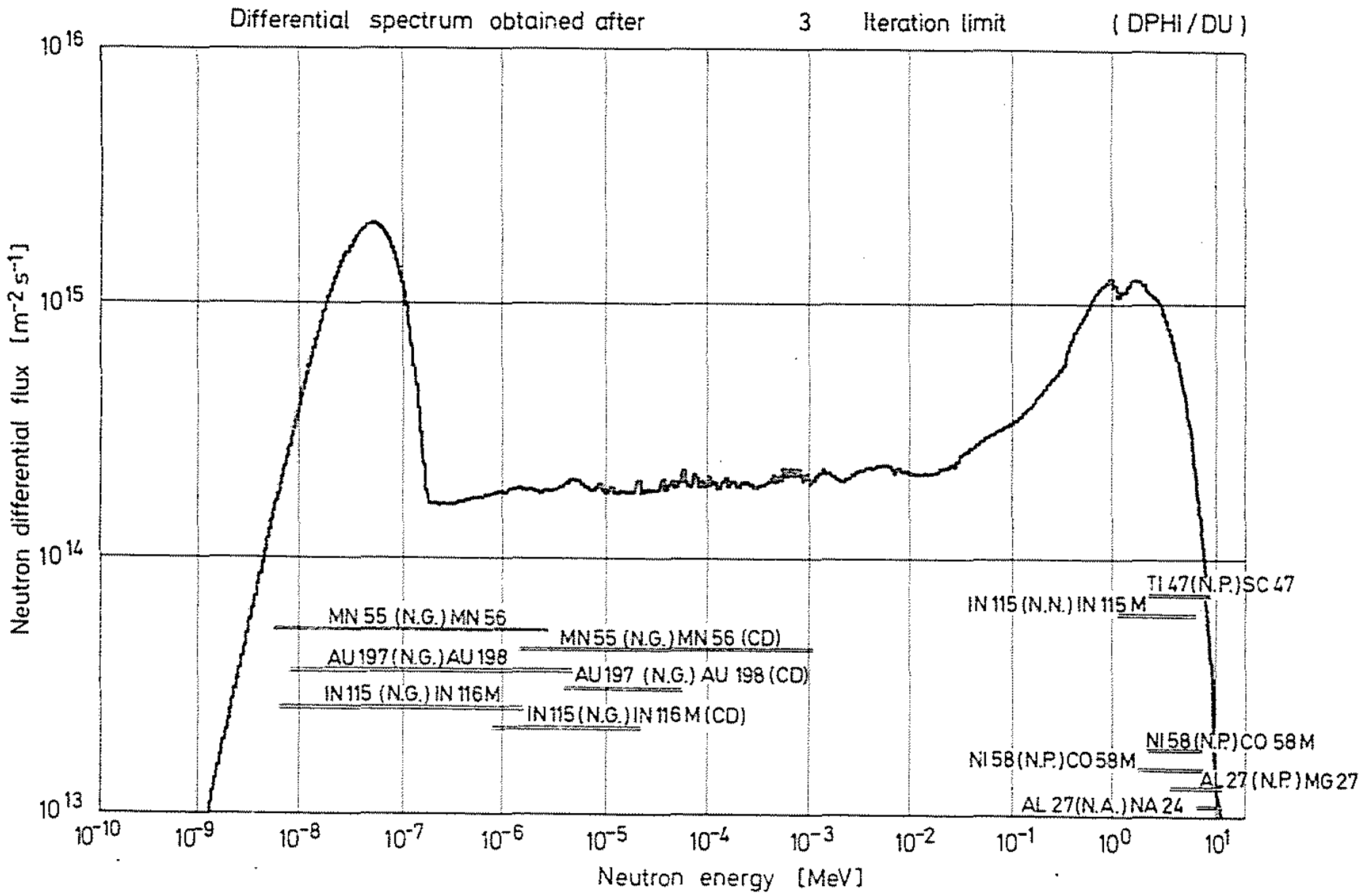
$$R = N\int_{E_1}^{E_2} \frac{d\Phi(E)}{dE}\sigma(E)dE$$

measure and calculate the reaction rate over an energy interval

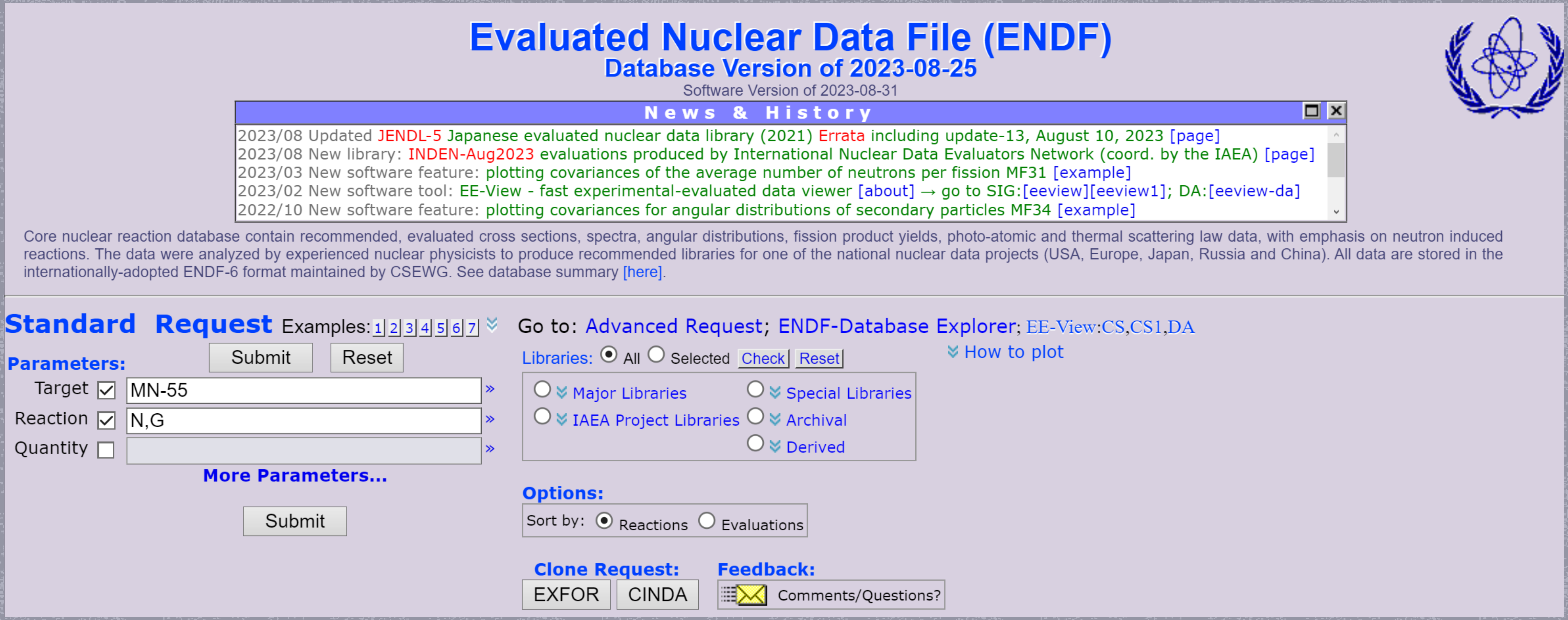
Typical energy regions of the neutron flux spectrum in thermal reactors [2]



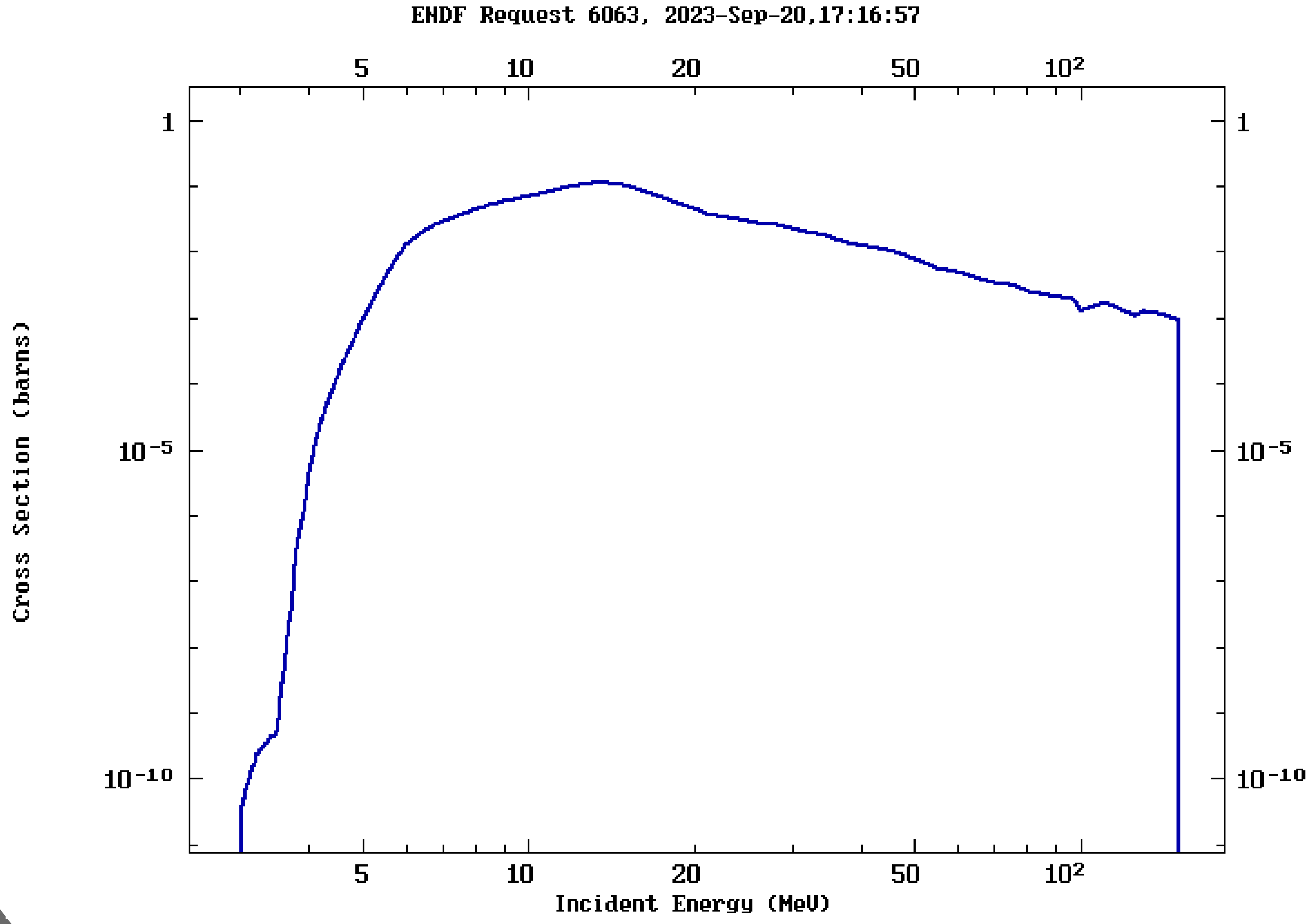
Neutron spectrum of a light-water reactor [3]



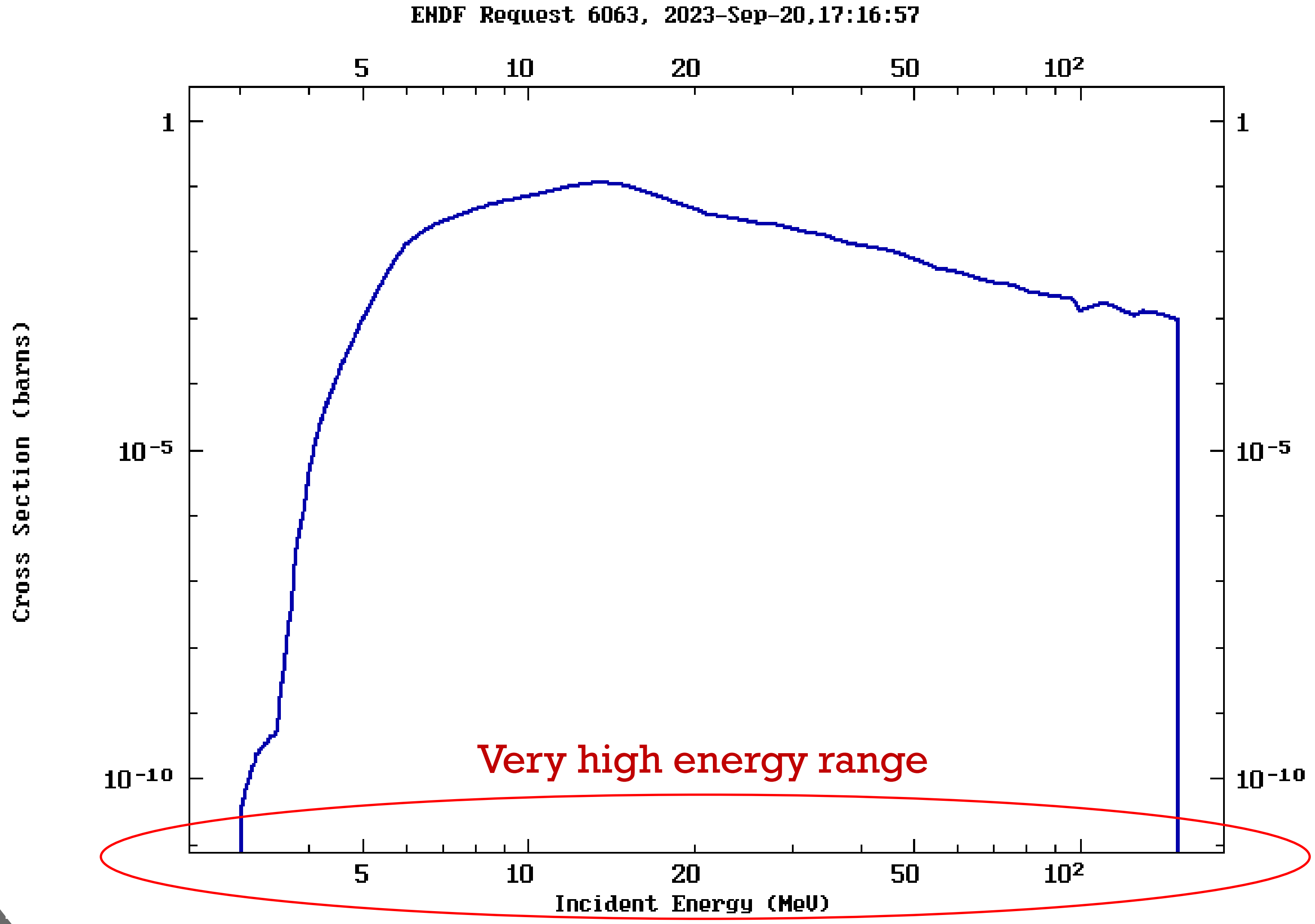
Cross sections are downloaded from:



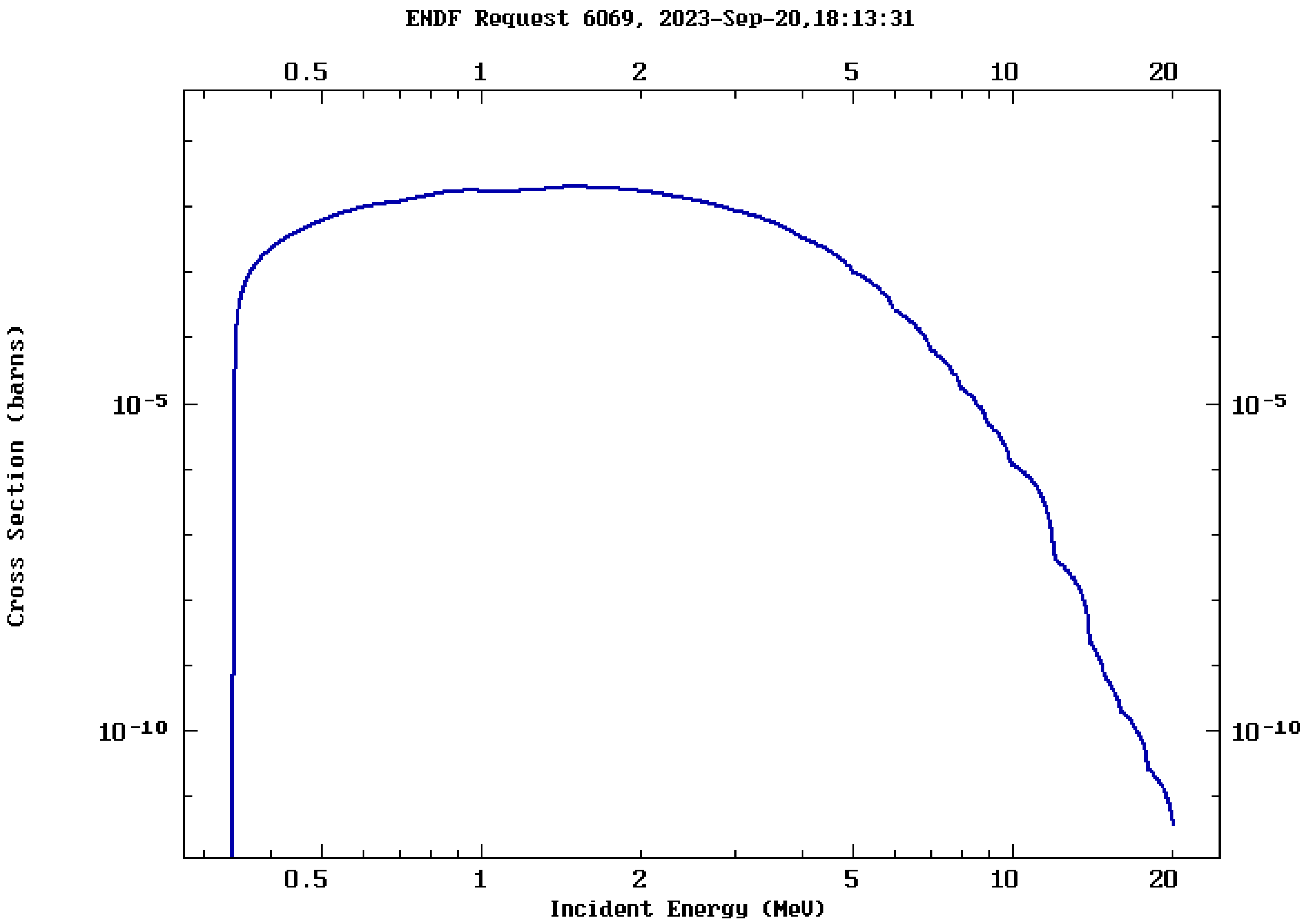
Energy-dependent cross section of the reaction: $^{56}Fe(n,\gamma)^{56}Mn^*$



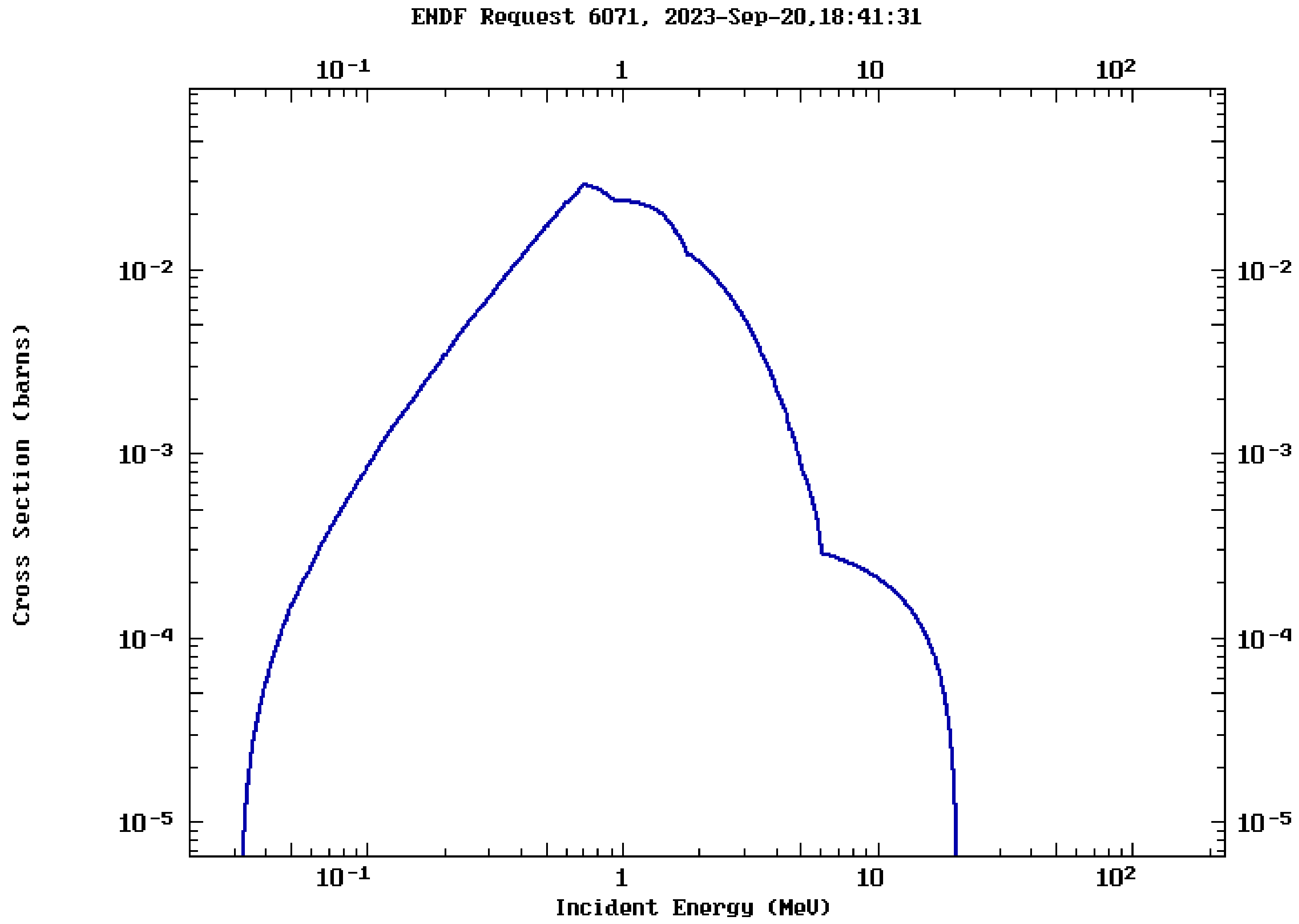
Energy-dependent cross section of the reaction: $^{56}Fe(n,\gamma)^{56}Mn^{*}$



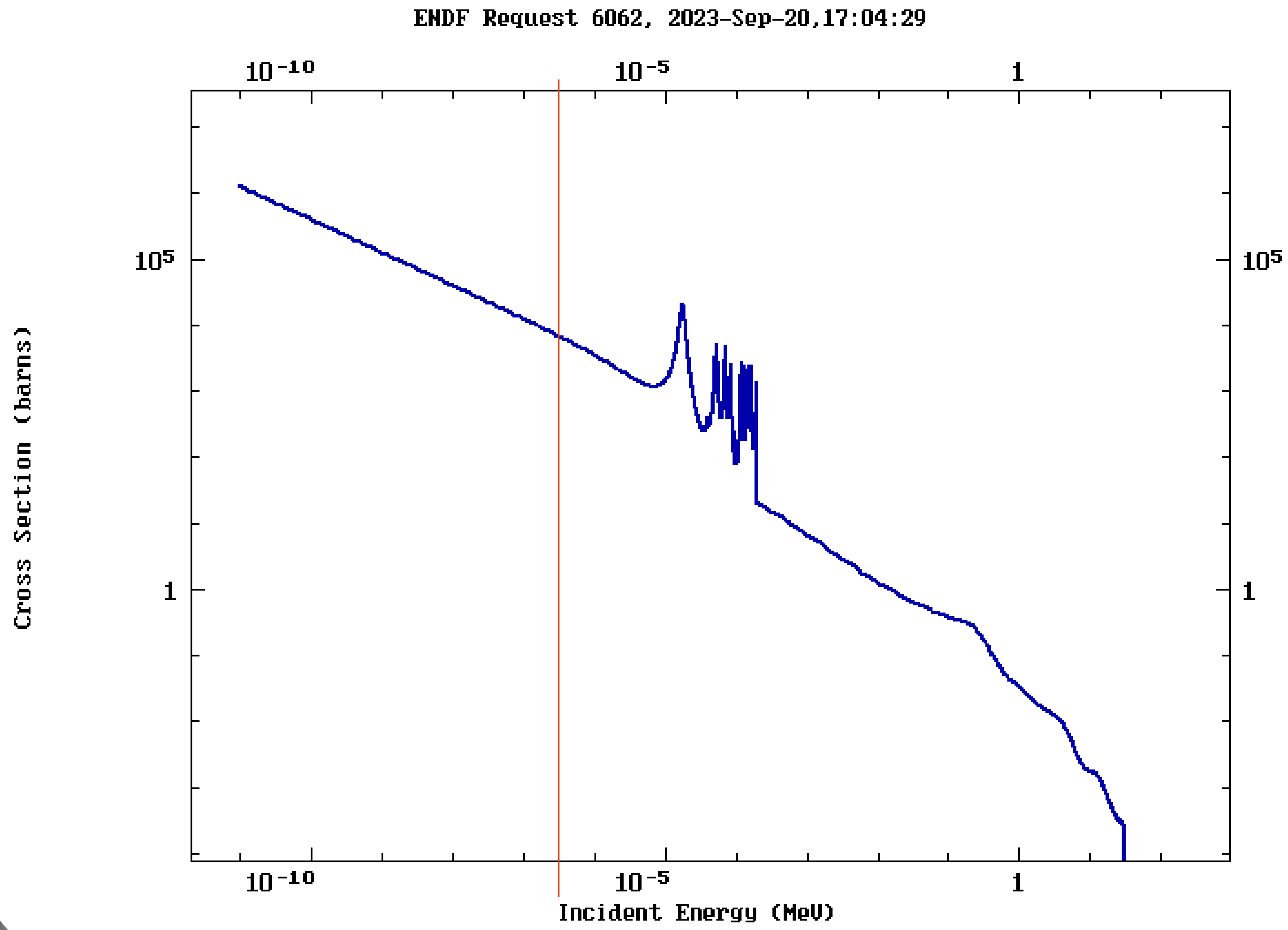
Energy-dependent cross section of the reaction: $^{115}In(n,n')^{115}In^*$



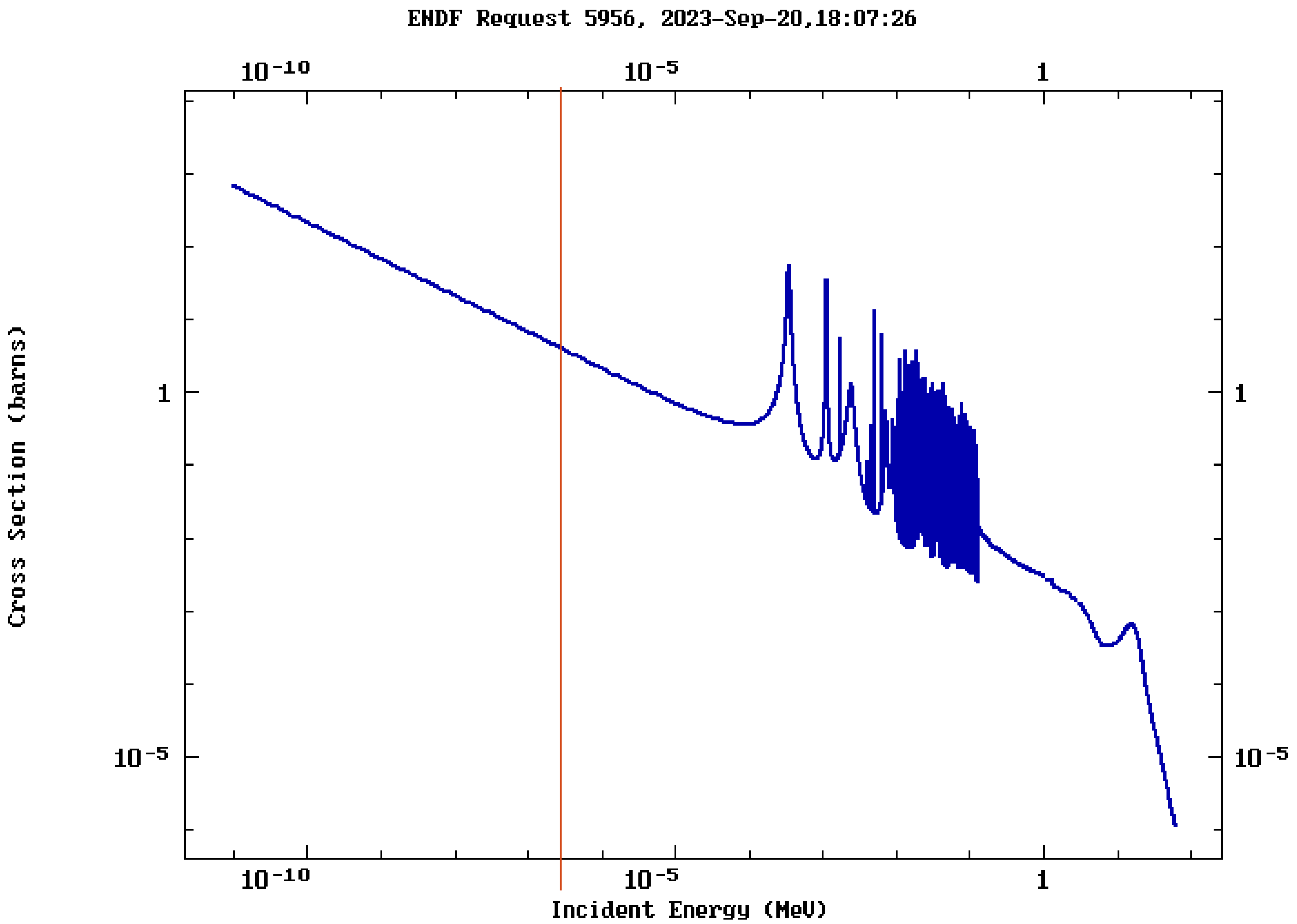
Energy-dependent cross section of the reaction: $^{93}Nb(n,n')^{93}Nb^*$



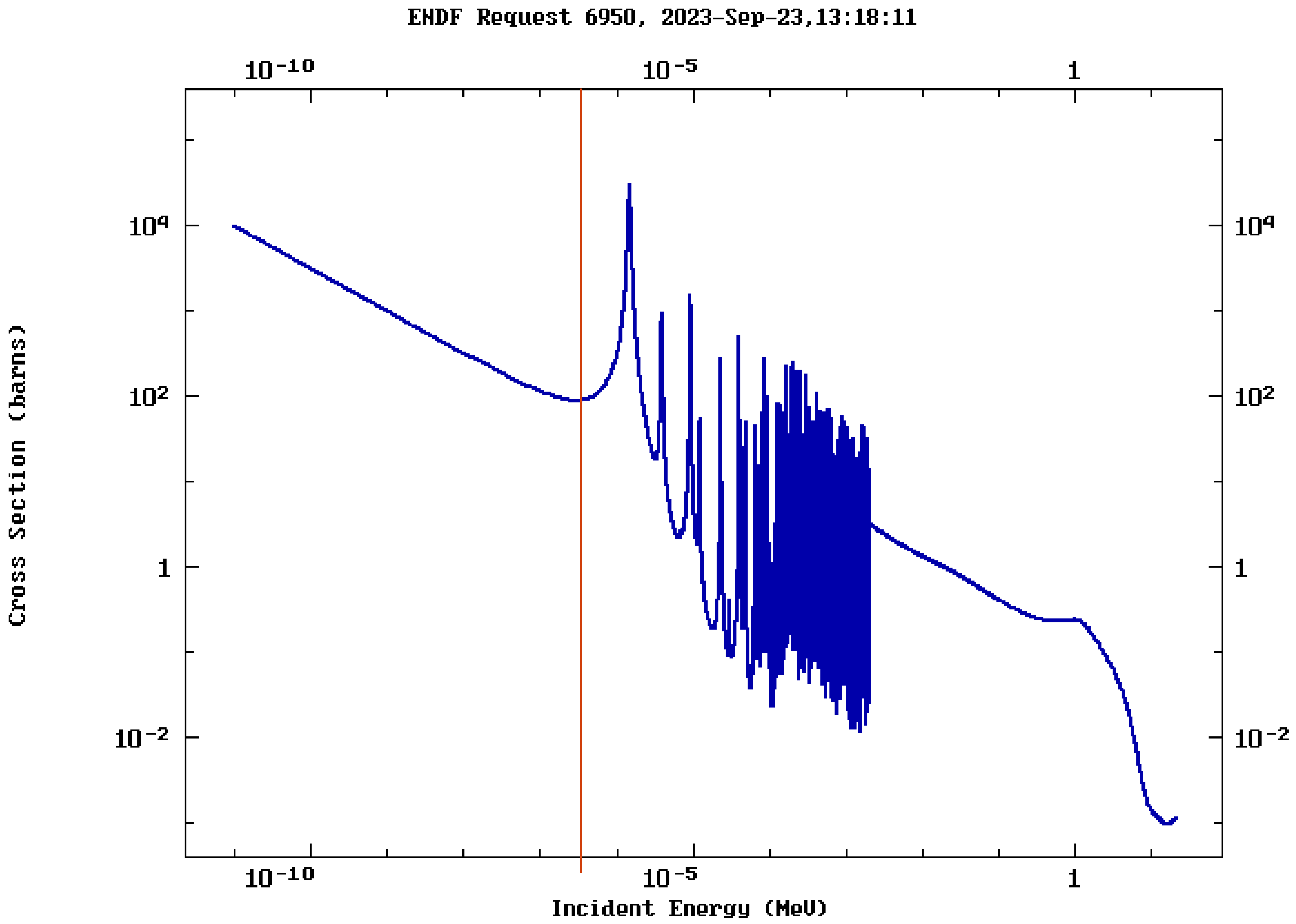
Energy-dependent cross section of the reaction: $^{198}Au(n,\gamma)^{199}Au$



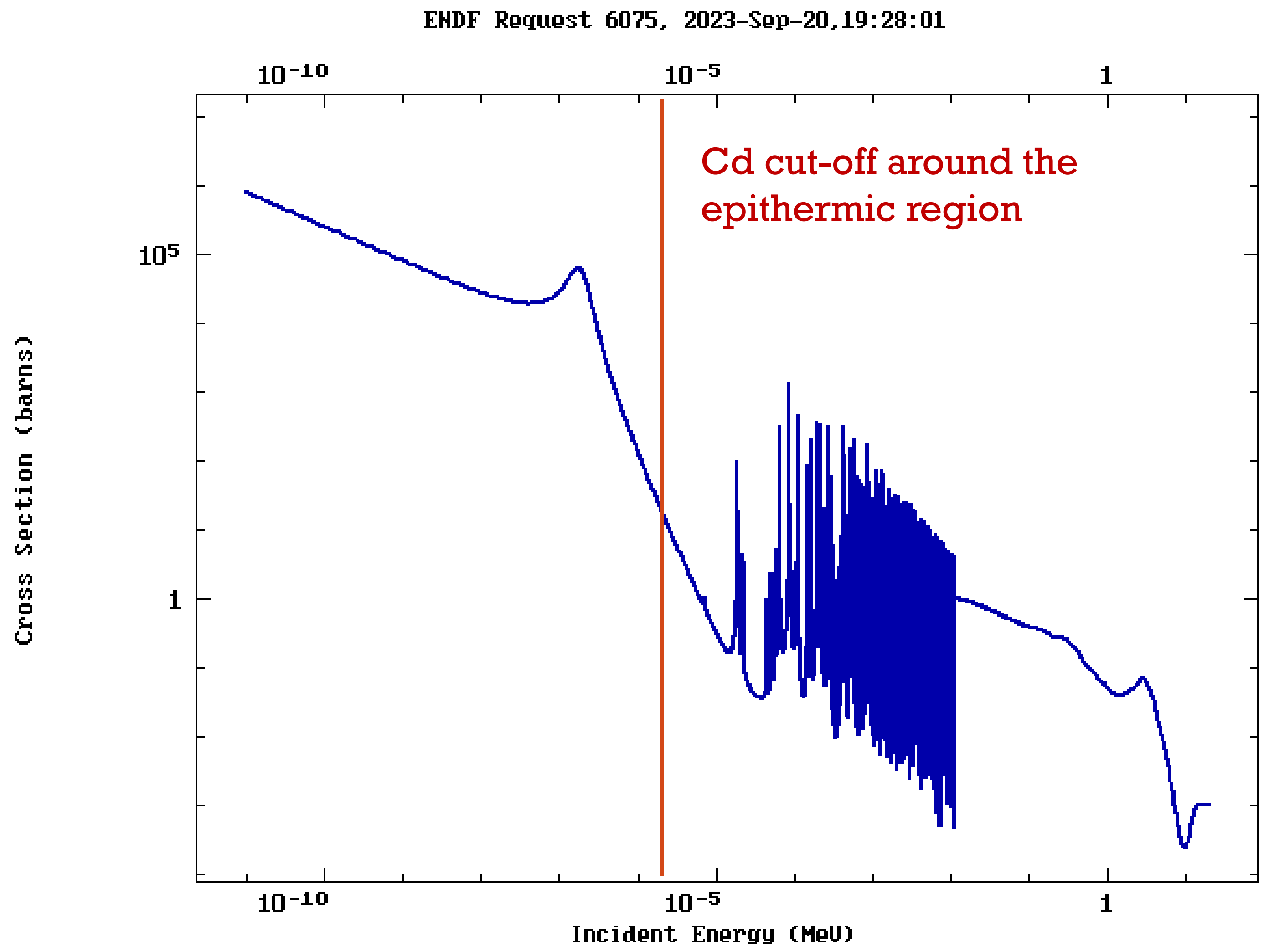
Energy-dependent cross section of the reaction: $^{55}Mn(n,\gamma)^{56}Mn$



Energy-dependent cross section of the reaction: $^{115}In(n,\gamma)^{115}In^*$

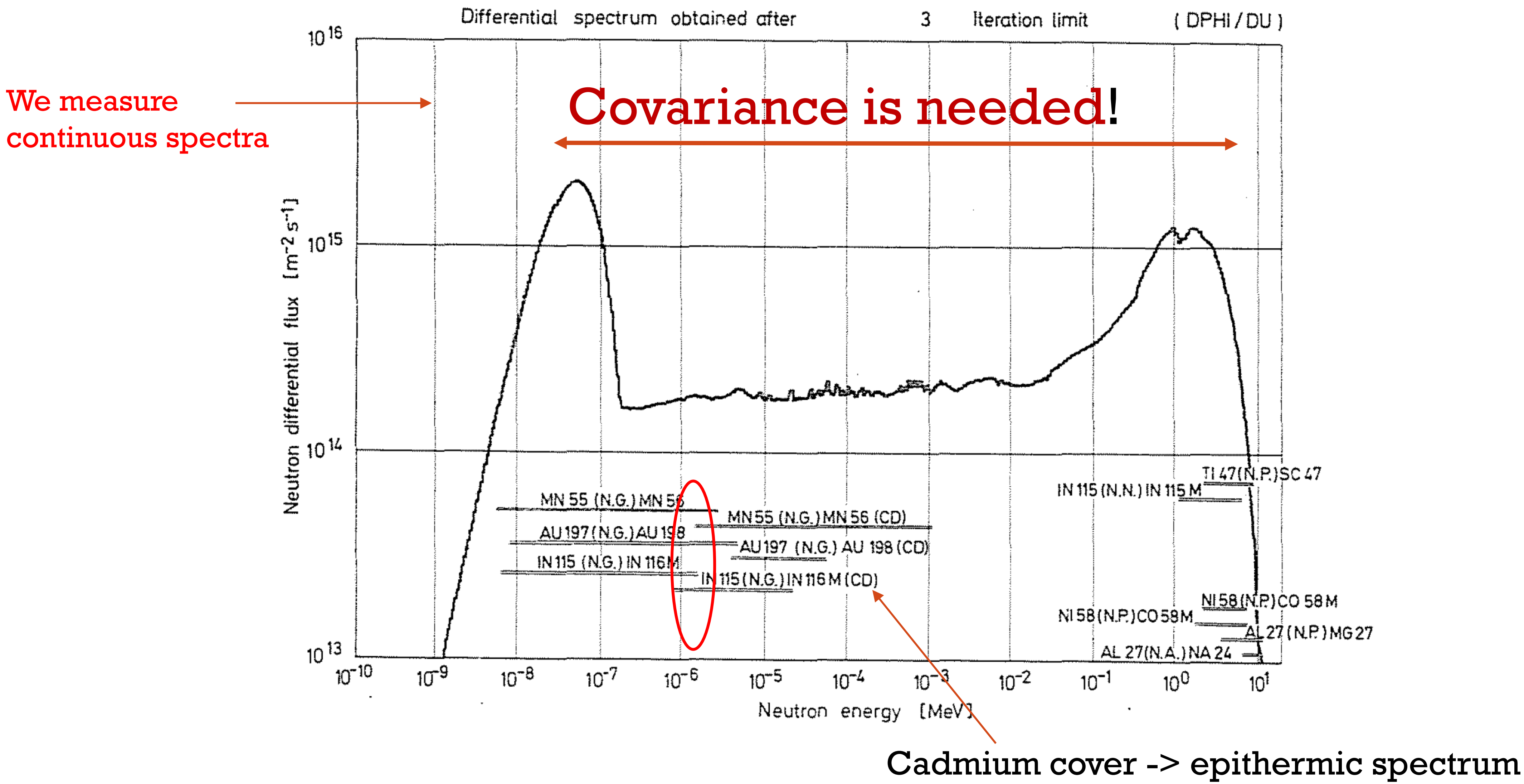


Energy-dependent cross section of the reaction: $^{113}Cd(n,\gamma)^{114}Cd$



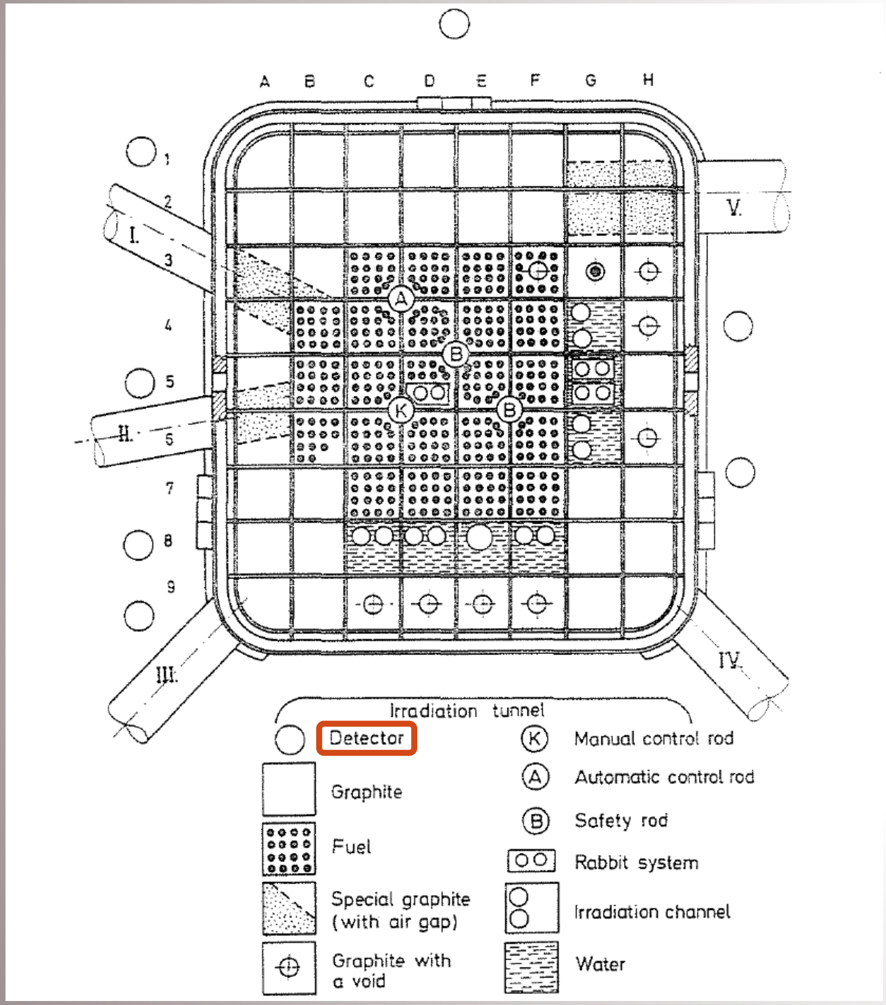
used to separate thermal and epitermal fluxes

Neutron spectrum of a light-water reactor [3]



ABOUT THE MEASUREMENT

- Reactor core, reactor operating at 10 kW [3]



Reaction	Half-life
$^{164}Dy(n, \gamma)^{165}Dy^{*}$	139.2 min
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$^{27}Al(n, p)^{27}Mg$	9.46 min
$^{27}Al(n, \alpha)^{24}Na$	15.03 h
$^{45}Sc(n, \gamma)^{46}Sc^{*}$	84.0 d
$^{56}Fe(n, p)^{56}Mn^{*}$	2.587 h
$^{58}Ni(n, p)^{58m}Co$	71.3 d
$^{58}Ni(n, p)^{58m}Co$	71.3 d
$^{115}In(n, \gamma)^{116m}In$	53.34 min
$^{115}In(n, \gamma)^{116m}In$	53.34 min
$^{55}Mn(n, \gamma)^{56}Mn$	155.2 min
$^{55}Mn(n, \gamma)^{56}Mn$	155.2 min
$^{197}Au(n, \gamma)^{198}Au$	2.695 d
$^{197}Au(n,\gamma)^{198}Au$	2.695 d
$^{115}In(n, n')^{115m}In$	4.5 h
$^{47}Ti(n, p)^{47}Sc$	80.4 h
$^{48}Ti(n, p)^{48}Sc$	44.1 h

Typical monitor substances

The reaction products are β-dacaying or γ-decaying isotopes

- HPGe semi-conductor detectors
- Sets of irradiated thin foils covered by Al or Cd (**neutron monitors**)
- Detecting γ-lines

ABOUT THE MEASUREMENT

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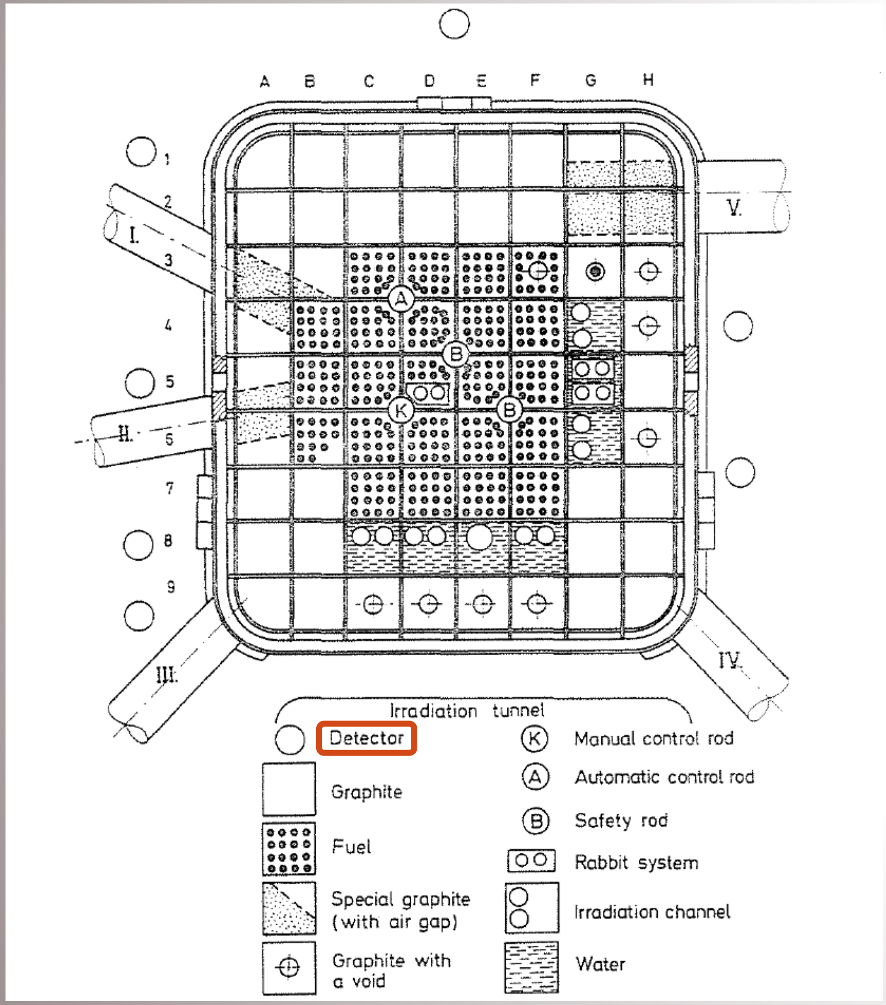


Reaction	Half-life	A_{sat}/target atom [Bq]
$^{164}Dy(n,\gamma)^{165}Dy$*	139.2 min	6.35 E-10
$^{164}Dy(n,\gamma)^{165}Dy$*	139.2 min	6.32 E-12
$^{27}Al(n,p)^{27}Mg$	9.46 min	9.01 E-16
$^{27}Al(n,\alpha)^{24}Na$	15.03 h	1.60 E-16
$^{45}Sc(n,\gamma)^{46}Sc$*	84.0 d	3.45 E-12
$^{56}Fe(n,p)^{56}Mn$*	2.587 h	1.35 E-17
$^{58}Ni(n,p)^{58m}Co$	71.3 d	2.47 E-14
$^{58}Ni(n,p)^{58m}Co$	71.3 d	2.48 E-14
$^{115}In(n,\gamma)^{116m}In$	53.34 min	1.75 E-11
$^{115}In(n,\gamma)^{116m}In$	53.34 min	1.22 E-11
$^{55}Mn(n,\gamma)^{56}Mn$	155.2 min	5.73 E-12
$^{55}Mn(n,\gamma)^{56}Mn$	155.2 min	2.79 E-13
$^{197}Au(n,\gamma)^{198}Au$	2.695 d	5.86 E-11
$^{197}Au(n,\gamma)^{198}Au$	2.695 d	3.22 E-11
$^{115}In(n,n')^{115m}In$	4.5 h	4.56 E-14
$^{47}Ti(n,p)^{47}Sc$	80.4 h	4.29 E-15
$^{48}Ti(n,p)^{48}Sc$	44.1 h	3.92 E-17

Typical monitor substances (also Cu, Ag, Co)

- HPGe semi-conductor detectors
- Sets of irradiated thin foils covered by Al or Cd (**neutron monitors**)
- Measure the activity values of the foil detectors
- The use of Cd enables us to extract the epithermic spectrum information

Spectrum unfolding with SAND II or SANDBP

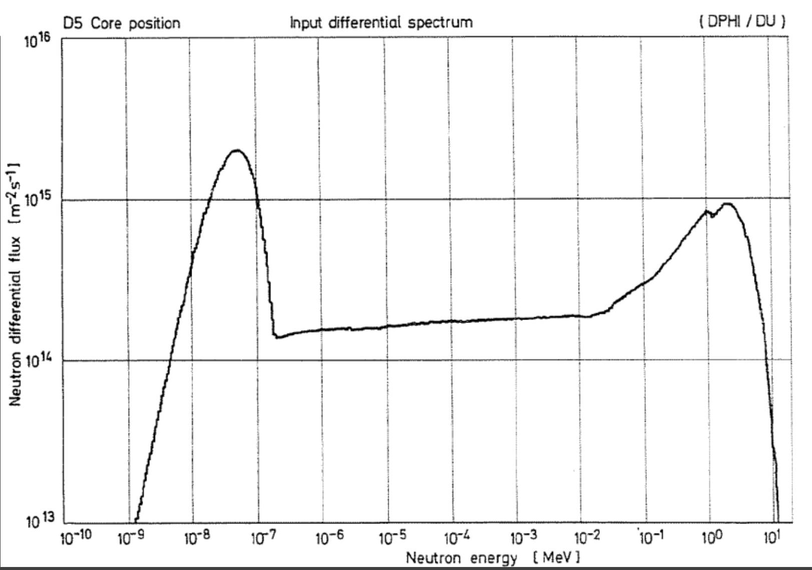
- Input
  - **Measure** the **activity** of the neutron monitors (per target atom)
  - **Use** the relevant **cross section** information from the IRDFF library
  - **Select** an initial approximation (**input**) **spectrum** (guess)
  - Estimate **relative errors** for the acitivities and cross sections (weighting with errors ->faster convergence)
- **Iterative adjusments of the input spectrum** in 640 energy intervals by fitting the calculated activities to the measured ones (with a given error limit, calculating DEV)
- Output
  - **Best-fit neutron flux density spectrum + covariance and correlation matrices (with Monte Carlo runs)**

DETERMINATION OF THE NEUTRON SPECTRUM

$$A_i = \int_0^\infty \sigma_i(E)\Phi(E)\,dE \qquad i = 1(1)n$$

saturation activity of the ith foil detector

The spectrum unfolding method

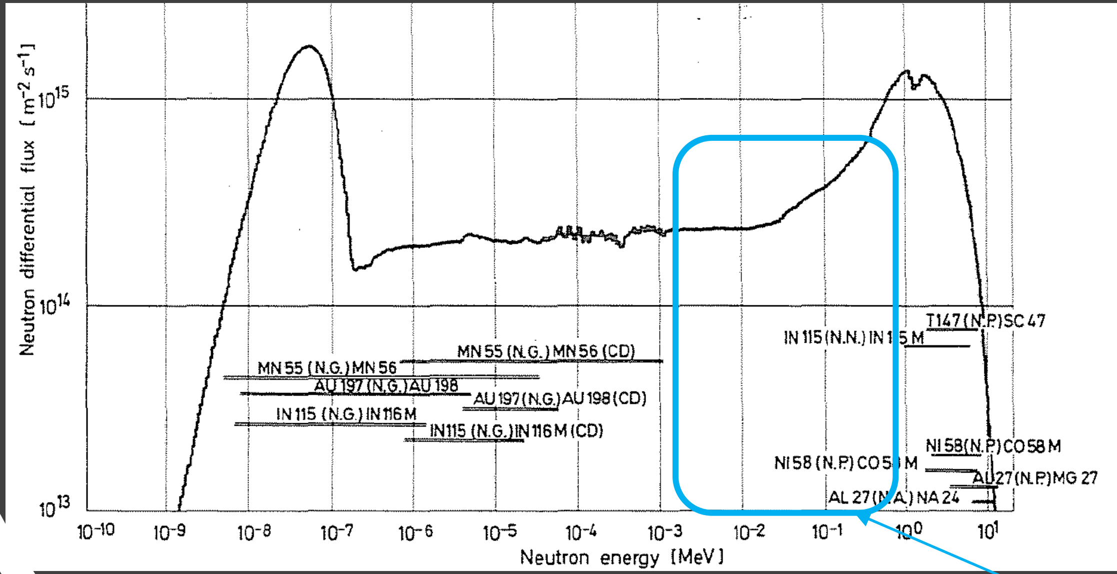


The **input spectrum** extrapolated to 640 energy intervals (j) [3]

$$\Phi_j^{k+1} = \Phi_j^k \exp\left(C_j^k\right)$$

$$C_j^k = \frac{\sum_{i=1}^{n} W_{ij}^k \ln\left(A_i / A_i^k\right)}{\sum_{i=1}^{n} W_{ij}^k}$$

$$W_{ij}^k = \frac{A_{ij}^k}{A_i^k}\left(\frac{1}{(\delta A_i)^u} \times \frac{1}{(\delta \sigma_{ij})^v}\right)$$



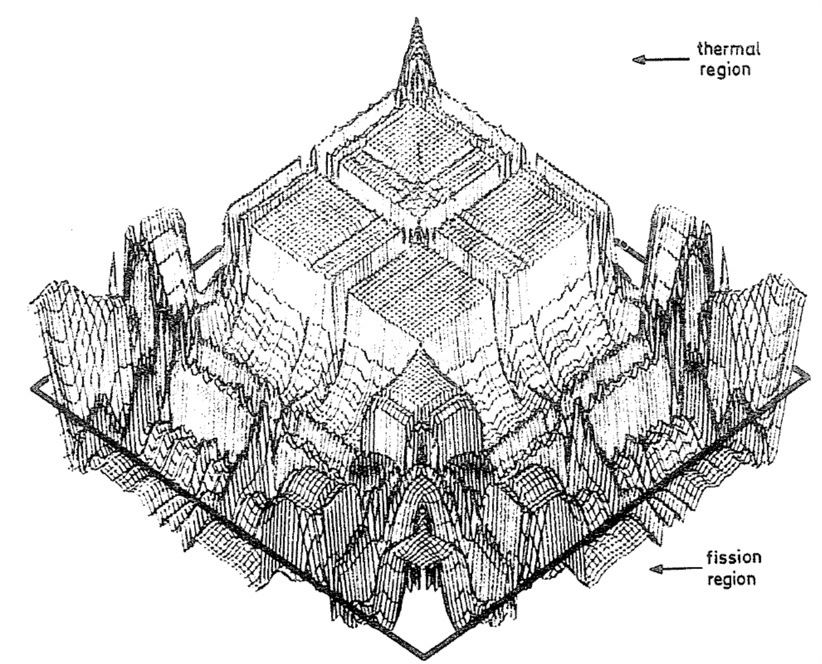
The **solution spectrum** obtained with

- error weighting (u,v)
- up to 3 iterations (k=3)
- 17 activation detectors (i) [3]

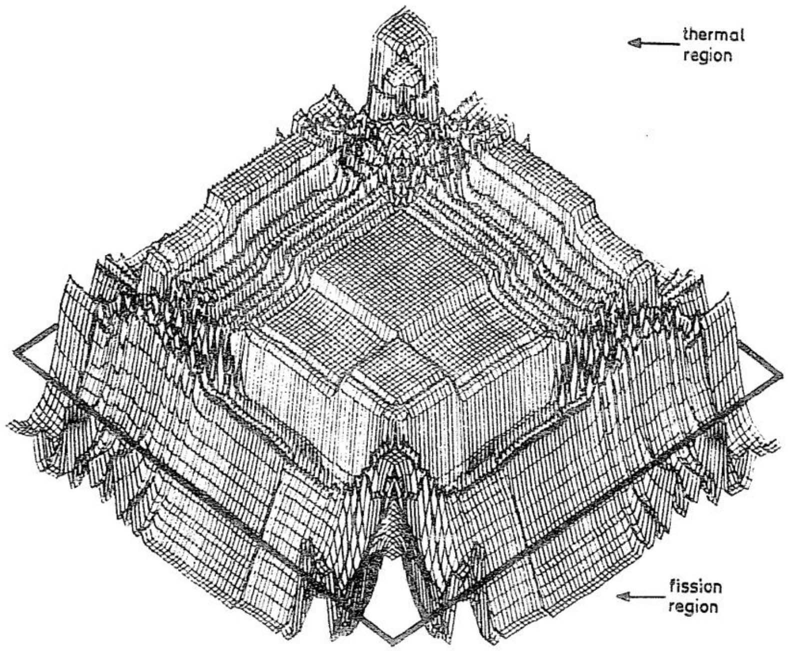
not covered by the response of the detector set!

DETERMINATION OF THE NEUTRON SPECTRUM

- The correlation matrices for the flux density values in all the neutron energy regions [1]



1.



2.

strong correlation is observed
(most energy regions are highly covered by the responses)

1. Correlation matrix without error weighting

1. Correlation matrix with higher error weighting

↓

smoother

References

[1] Zsolnay Éva, Czifrus Szabolcs, Kis Dániel Péter: A PAKSI ATOMERM 2. SZÁMÚ REAKTORBLOKKJÁN A REAKTORTARTÁLY KÜLS FELÜLETÉNÉL A 28. REAKTORKAMPÁNY SORÁN BESUGÁRZOTT NEUTRONMONITOROK VÁLASZÁNAK KIÉRTÉKELÉSE (2013)

[2] *https://www.nuclear-power.com/nuclear-power/reactor-physics/nuclear-engineering-fundamentals/neutron-nuclear-reactions/neutron-flux-spectra/*

[3] É. M. Zsolnay and E. J. Szondi: *Neutron spectrum determination by multiple foil activation method* (1982)