



FISSION CROSS SECTIONS

PRACTICAL SESSION FIESTA 2024

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OUTLINE OF PRACTICAL SESSION

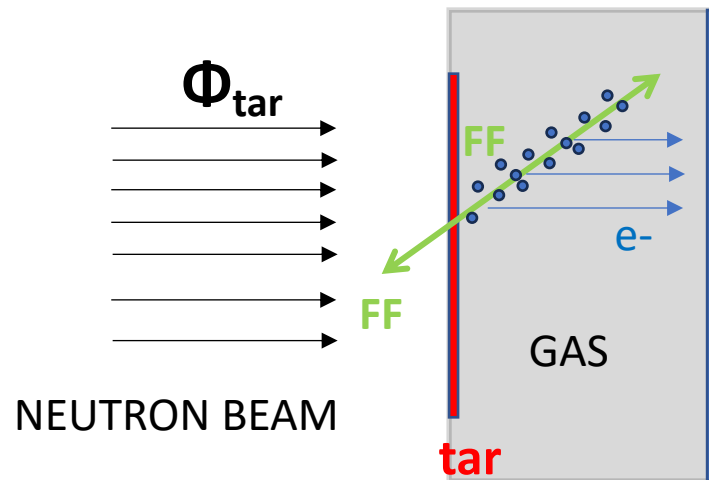
GOAL:

- 1) To **extract n-induced fission cross section (cs- « σ »)** from real data. **(EX. 1)**
- 2) Understand the importance of the **reference reaction chosen** **(EX. 2,3)**
- 3) Understand the differences of the **different techniques** (white neutron beam, monoenergetic neutron beam) **(EX. 2,3)**

Implementation of course in **Jupyter Notebook** (many thanks to **Thanos Stamatopoulos** (LANL))

GENERAL REMARKS / ASSUMPTIONS :

- 1) Detection of **fission fragments (FF)** with **gaseous detector**
- 2) Thin disks of the target under study (**tar**)



$$\sigma(E) = \frac{RR_{tar}(E)}{\Phi_{tar}(E) N_{tar}}$$

↓

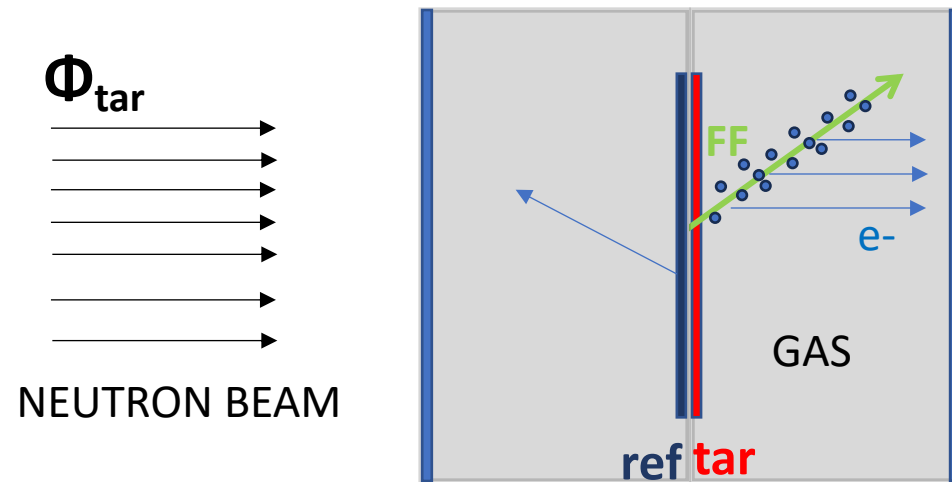
$$\sigma(E) = \frac{Y_{tar,corrected}(E)}{\Phi_{tar}(E) N_{tar}}$$

: No of FF detected, with the necessary correction factors

GENERAL REMARKS / ASSUMPTIONS :

3) Measurement of Φ_{tar} : **WITH REFERENCE TARGET (ref)**,

(very well-known n-induced cs – “primary standard”, “secondary standard”, “reference”)



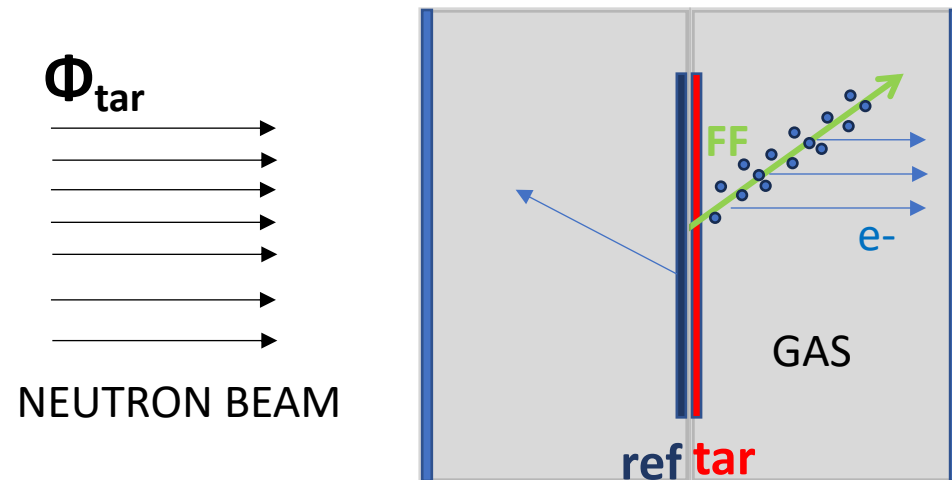
$$\sigma(E) = \frac{Y_{tar,corrected}(E)}{\Phi_{tar}(E) N_{tar}}$$

$$\sigma_{ref}(E) = \frac{Y_{ref,corrected}(E)}{\Phi_{ref}(E) N_{ref}}$$

GENERAL REMARKS / ASSUMPTIONS :

3) Measurement of Φ_{tar} : **WITH REFERENCE TARGET (ref)**,

(very well-known n-induced cs – “primary standard”, “secondary standard”, “reference”)



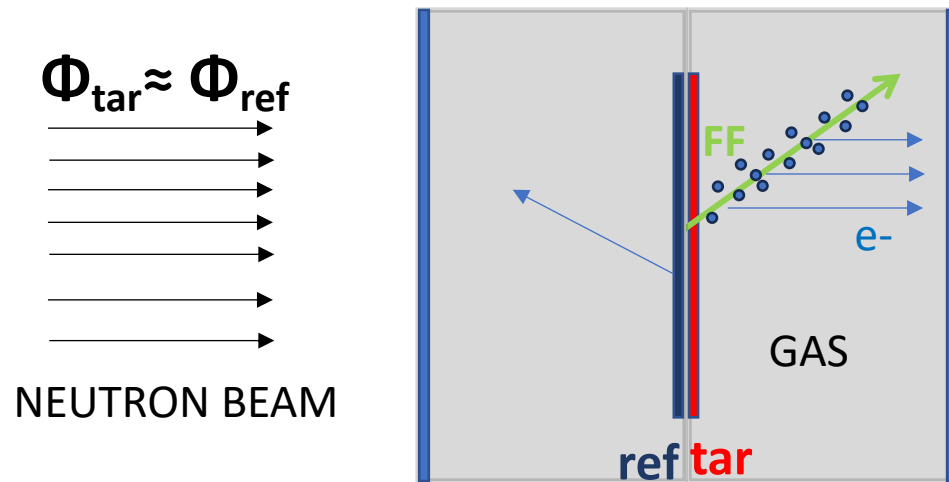
$$\sigma(E) = \frac{Y_{tar,corrected}(E)}{\Phi_{tar}(E) N_{tar}} \quad \sigma_{ref}(E) = \frac{Y_{ref,corrected}(E)}{\Phi_{ref}(E) N_{ref}}$$

$$\sigma(E) = \frac{Y_{tar,corrected}(E)}{Y_{ref,corrected}(E)} \frac{\Phi_{ref}(E) N_{ref}}{\Phi_{tar}(E) N_{tar}} \sigma_{ref}(E)$$

GENERAL REMARKS / ASSUMPTIONS :

3) Measurement of Φ_{tar} : **WITH REFERENCE TARGET (ref)**,

(very well-known n-induced cs – “primary standard”, “secondary standard”, “reference”)



$$\sigma(E) = \frac{Y_{\text{tar,corrected}}(E)}{\Phi_{\text{tar}}(E) N_{\text{tar}}} \quad \sigma_{\text{ref}}(E) = \frac{Y_{\text{ref,corrected}}(E)}{\Phi_{\text{ref}}(E) N_{\text{ref}}}$$

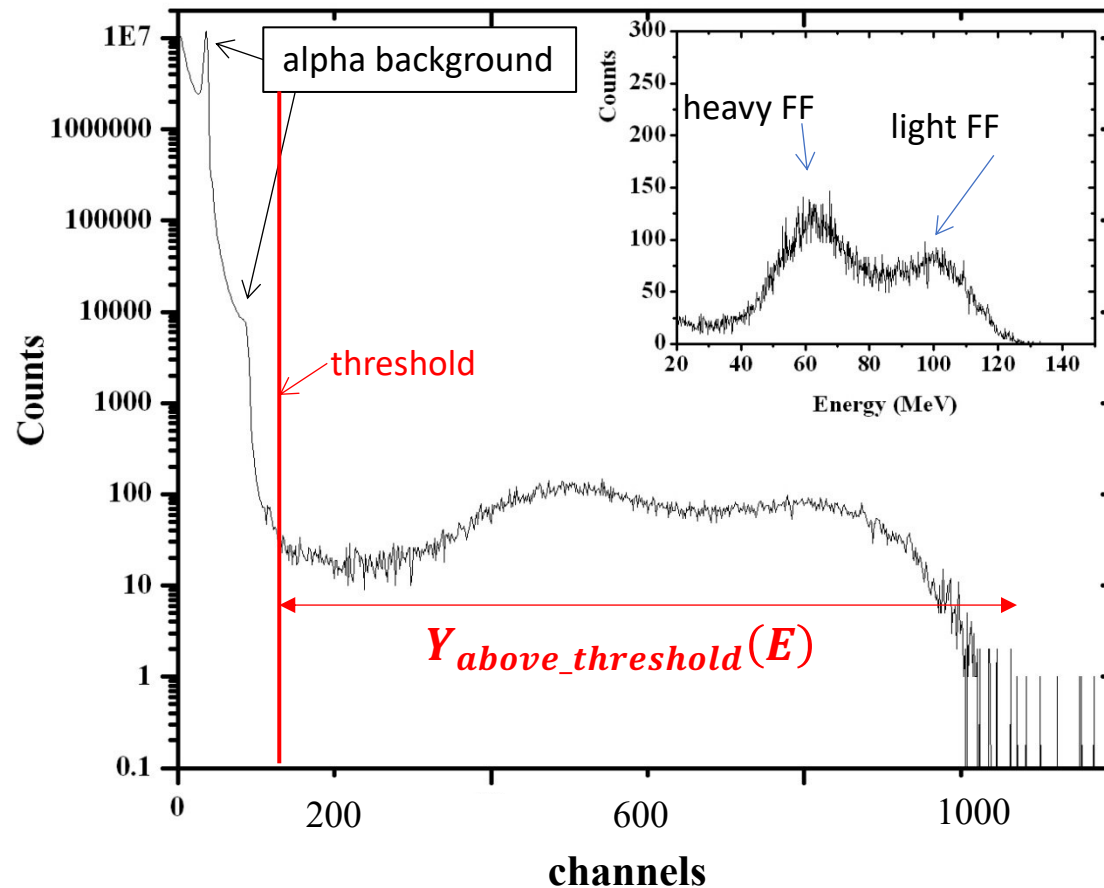
$$\sigma(E) = \frac{Y_{\text{tar,corrected}}(E)}{Y_{\text{ref,corrected}}(E)} \frac{\Phi_{\text{ref}}(E) N_{\text{ref}}}{\Phi_{\text{tar}}(E) N_{\text{tar}}} \sigma_{\text{ref}}(E) \Rightarrow \Phi_{\text{tar}} \approx \Phi_{\text{ref}}$$

$$\sigma(E) = \frac{Y_{\text{tar,corrected}}(E)}{Y_{\text{ref,corrected}}(E)} \frac{N_{\text{ref}}}{N_{\text{tar}}} \sigma_{\text{ref}}(E)$$

$Y_{\text{corrected}}(E)$: FF losses due to *absorption in the tar/ref, detection and analysis procedure (dead time/pile-up/ more specific corrections not taken into account at this course)*.

EXERCISE 1: n-induced fission cs («σ») at given energy E $^{232}\text{Th}(n,f)$ (reference: $^{238}\text{U}(n,f)$ reaction)

1a) Application of properly chosen threshold in the Counting spectra



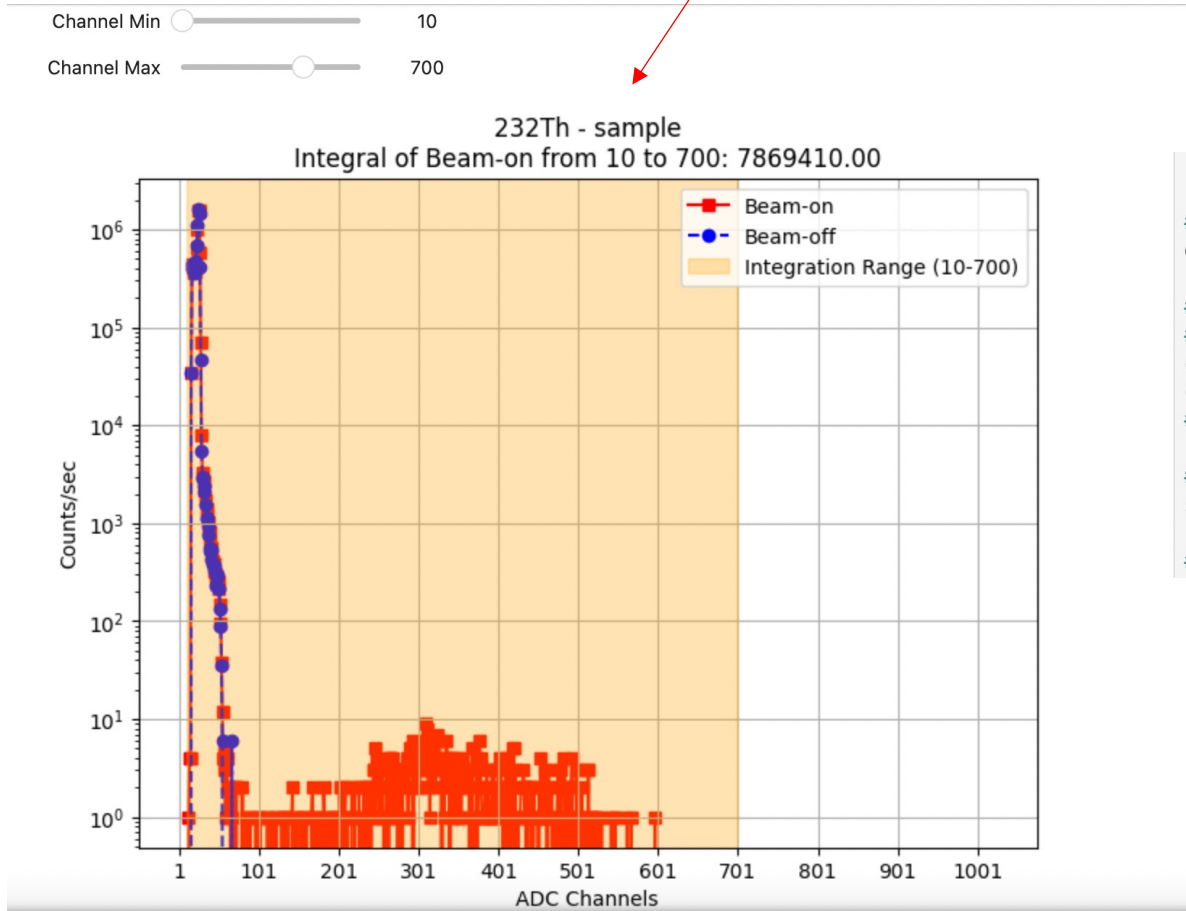
$$\sigma(E) = \frac{Y_{\text{tar,corrected}}(E)}{Y_{\text{ref,corrected}}(E)} \frac{N_{\text{ref}}}{N_{\text{tar}}} \sigma_{\text{ref}}$$

Choose **threshold** (BEAM-OFF)
to cut the alpha background
and integrate above threshold
 $\Rightarrow Y_{\text{above_threshold}}(E)$

integral_target

integral_reference

1a) integral_target and integral_reference



channel_min = chosen threshold

continue

```
# Convert the parsed data into a pandas DataFrame
df = pd.DataFrame(data, columns=['ch', 'beamOFF', 'beamON']) # ADC channels, beam off, beam on

# HERE IS WHERE STUDENTS PLAY FOR REFERENCE TARGET #####
# Define the channel range for integration (example: channels 10 to 700)
channel_min =  # <----- Choose your pulse height threshold
channel_max =  # <-----
#####

# Select the data for integration within the channel range
selected_data = df[(df['ch'] >= channel_min) & (df['ch'] <= channel_max)]

# Calculate the integral (sum) of the beamON_tar column over the selected range
```

[6]: integral_reference

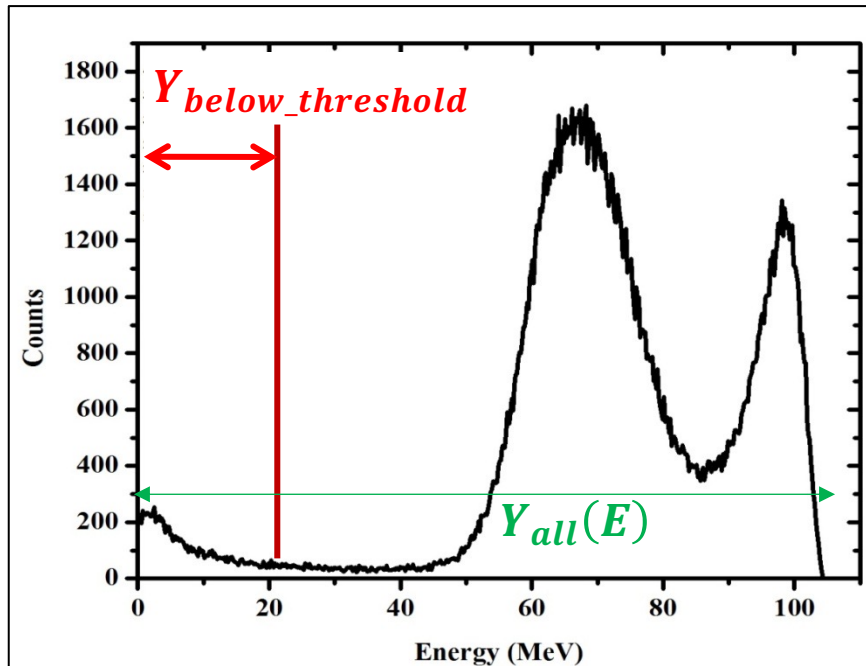
[6]: np.float64()

[2]: integral_target

[2]: np.float64()

EXERCISE 1: n-induced fission cs (« σ ») at given energy E $^{232}\text{Th}(n,f)$ (reference: $^{238}\text{U}(n,f)$ reaction)

1b) Correction of subthreshold fission fragment counts with **Monte Carlo simulations** (given)



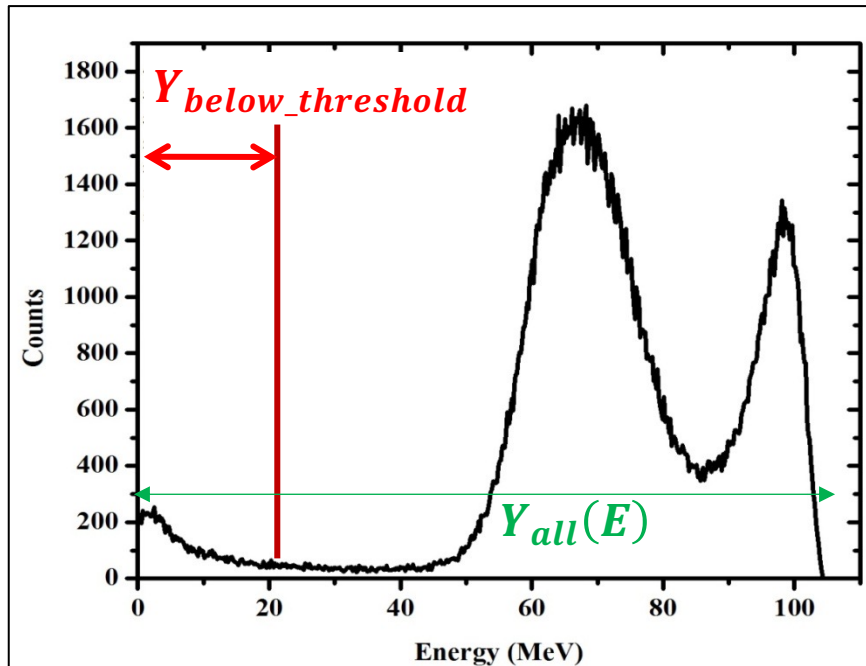
A) Threshold (ADC channels) -> Threshold (Energy):
From calibration (given):

```
#Convert threshold channel to deposited energy (CALIBRATION)
#Calibration given:
#232Th:
#Energy_232Th = 0.1959 * channel - 0.481 (MeV) <----- PUT HERE YOUR CHOSEN CHANNEL FOR 232Th
Energy_232Th = 0.1959 *  - 0.481

#238U:
#Energy_238U = 0.1859 * channel - 2.7895 (MeV) <----- PUT HERE YOUR CHOSEN CHANNEL FOR 238U
Energy_238U = 0.1859 *  - 2.7895
```

EXERCISE 1: n-induced fission cs («σ») at given energy E $^{232}\text{Th}(n,f)$ (reference: $^{238}\text{U}(n,f)$ reaction)

1b) Correction of subthreshold fission fragment counts with Monte Carlo simulations (given)



A) Threshold (ADC channels) -> Threshold (Energy):
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#232Th:
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#238U:
#Energy_238U = 0.1859 * channel - 2.7895 (MeV) <----- PUT HERE YOUR CHOSEN CHANNEL FOR 238U
Energy_238U = 0.1859 *  - 2.7895
```

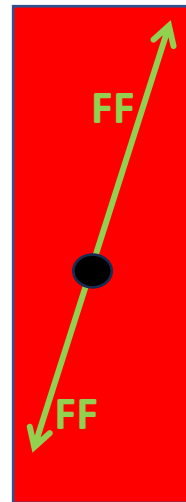
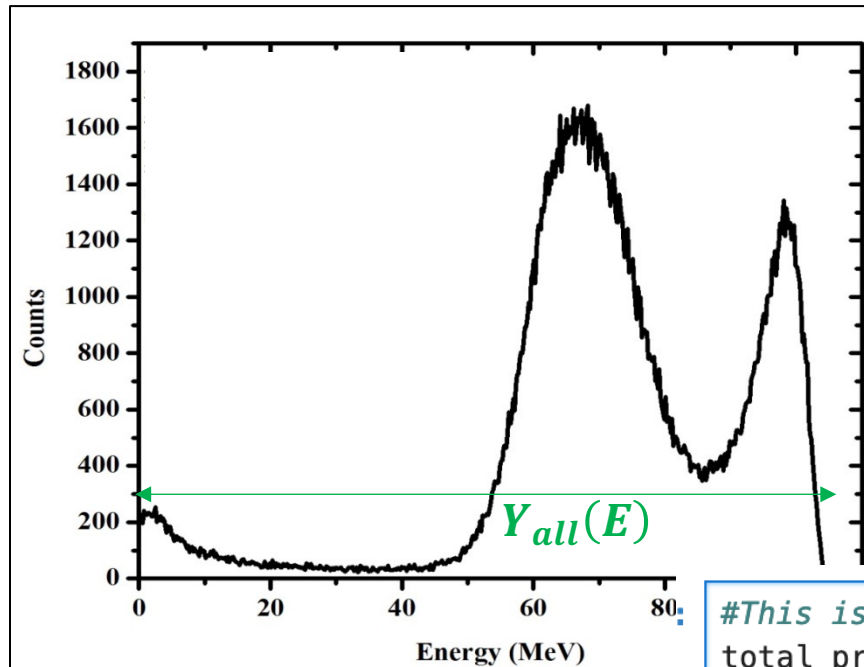
B) Estimate $Y_{below_threshold}$ from simulated spectra:

```
# Define the threshold for integration
threshold_232Th =  #<----- Put here your pulse height threshold in energy for 232Th
threshold_238U =  #<----- Put here your pulse height threshold in energy for 238U
```

$$Y_{subThr,corrected}(E) = Y_{above_threshold}(E) * (1 + Y_{below_threshold} / Y_{all}(E))$$

EXERCISE 1: n-induced fission cs («σ») at given energy E $^{232}\text{Th}(n,f)$ (reference: $^{238}\text{U}(n,f)$ reaction)

1c) Calculation of efficiency (from the same **Monte Carlo simulations**): FF counts absorbed in the actinide target



$$Y_{corrected}(E) = Y_{subThr,corrected}(E) / \text{eff}$$

(for tar and ref)

#This is the total number of primaries generated in the Monte Carlo simulations:

```
total primaries = 5.e6
```

```
eff_232Th = integral_total_232Th / total primaries
```

```
eff_238U = integral_total_238U / total primaries
```

EXERCISE 1: n-induced fission cs («σ») at given energy E ²³²Th(n,f) (reference: ²³⁸U(n,f) reaction)

1d) Extraction of the final cross section value (finally..... 😊)

```
# Calculate the corrected counts for 232Th
counts_232Th = integral_target*(1.+ratio_232Th)

#Calculate the corrected counts for 238U
counts_238U = integral_reference*(1.+ratio_238U)

#This is the cross section of 238U in b
sig_238U = 1.318

#This is the mass of 232Th (atoms)
mass_232Th = 9.31e+18

#This is the mass of 238U (atoms)
mass_238U = 2.06e+19

#####
#This is the efficiency of 238U (calculated in (1c))
eff_238U =  #<----- put the correct value here

#This is the efficiency of 232Th (calculated in (1c))
eff_232Th =  #<----- put the correct value here
#####

#This is the 232Th cross section from our experiment
sig_232Th = (counts_232Th/counts_238U)*(mass_238U/mass_232Th)*(eff_238U/eff_232Th)
```

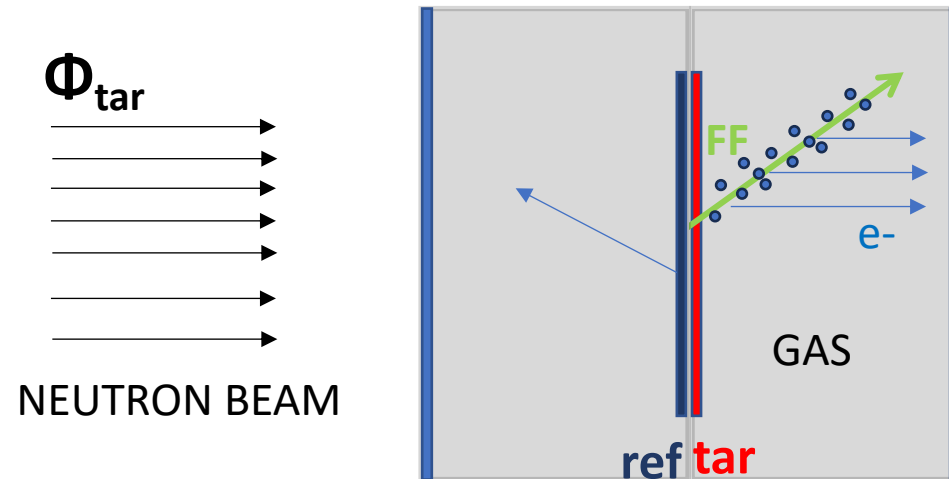
$$\sigma(E) = \frac{Y_{tar,corrected}(E) N_{ref}}{Y_{ref,corrected}(E) N_{tar}} \sigma_{ref}(E)$$

What is the value you obtain???

Uncertainties???

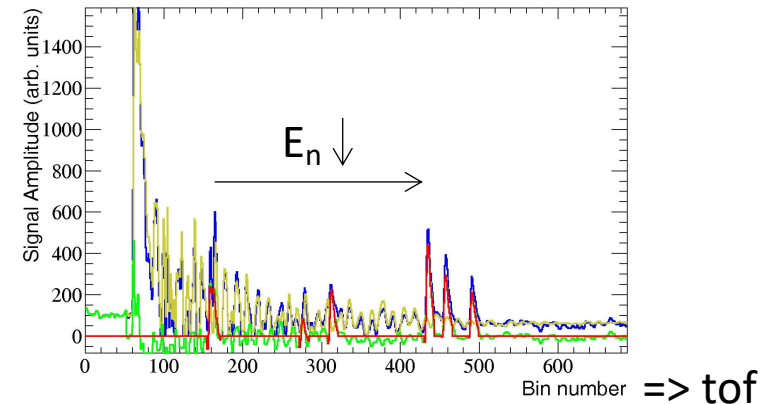
EXERCISE 2: Fission cross section measurements with the TOF technique

White pulsed neutron beam => Various neutron energies (E_n)



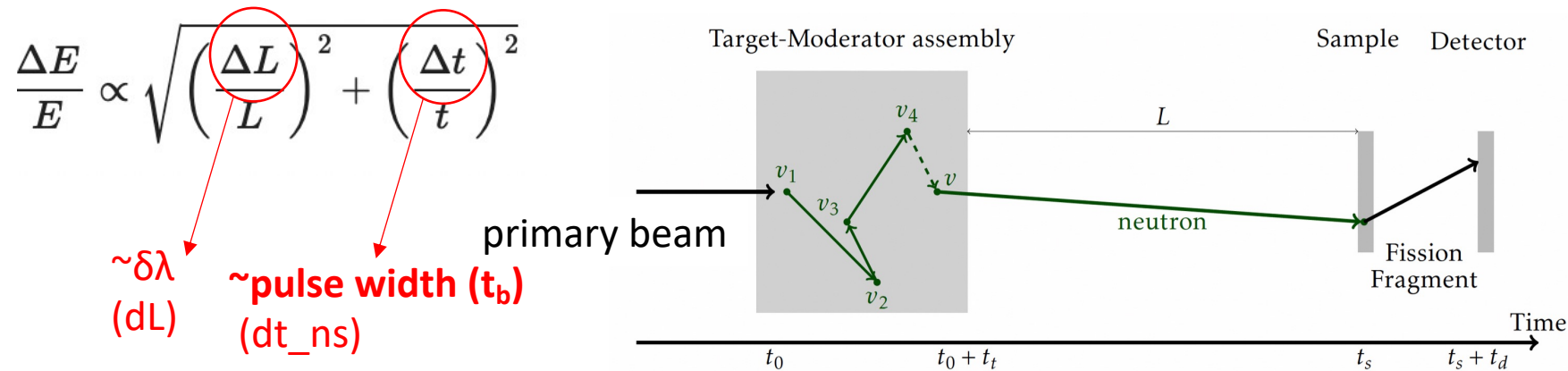
$$\sigma(E_n) = \frac{Y_{tar,corrected}(E_n)}{Y_{ref,corrected}(E_n)} \frac{\Phi_{ref}(E_n)}{\Phi_{tar}(E_n)} \frac{N_{ref}}{N_{tar}} \sigma_{ref}(E_n)$$

$$E_n = m_n c^2 \left(\frac{1}{\sqrt{1 - \beta^2}} - 1 \right), \quad \beta = \frac{v_n}{c} = \frac{L}{c \cdot \text{tof}}$$



EXERCISE 2: Fission cross section measurements with the TOF technique

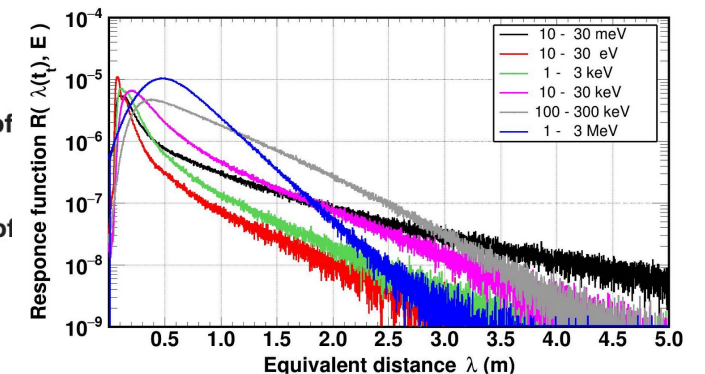
2a) The neutron energy resolution with the TOF technique and how it affects results



2a) i) Let's assume your facility has two beam lines, one with $L = 15$ m and one with $L = 150$ m. Where would you choose to perform a measurement to achieve the best possible neutron energy resolution?

2a) ii) Calculate the energy resolution at the 150 m flight path assuming the facility provides a proton beam with a pulse width of 150 ns FWHM.

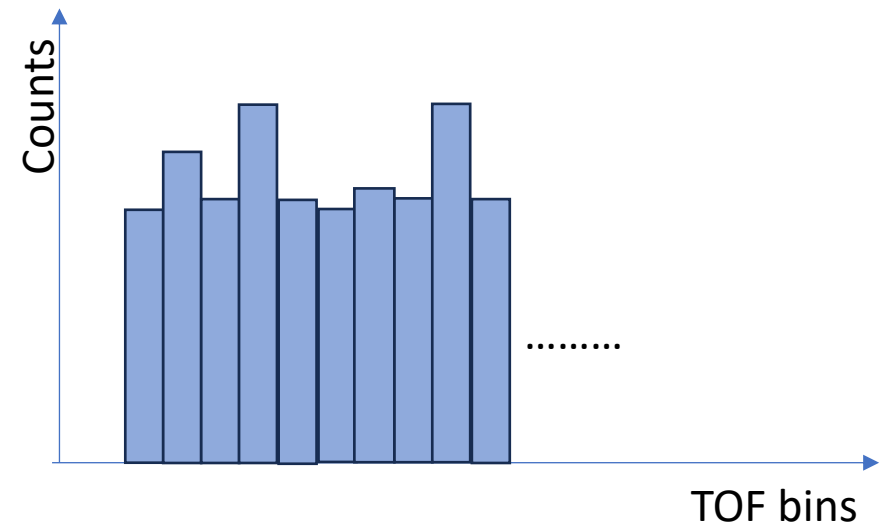
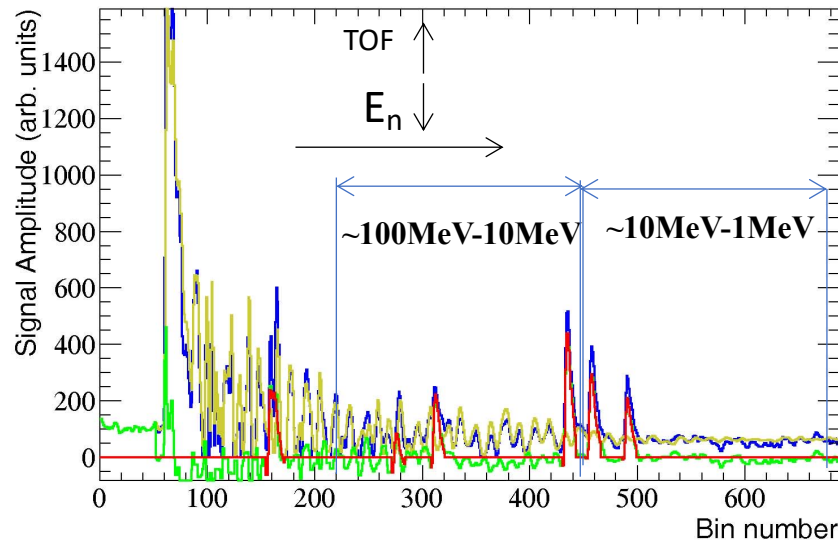
2a) iii) Now assume, you are in another facility and the longest flight path is only 30 m. However, this facility has a pulse width of 1 ns and a transport length 10 times smaller than the previous facility. Where would you choose to perform your experiment, if neutron energy resolution at 1 keV was the determining factor?



Decide $\delta\lambda$ from simulations for th-1keV and put pulse width (typically 1-10 ns)

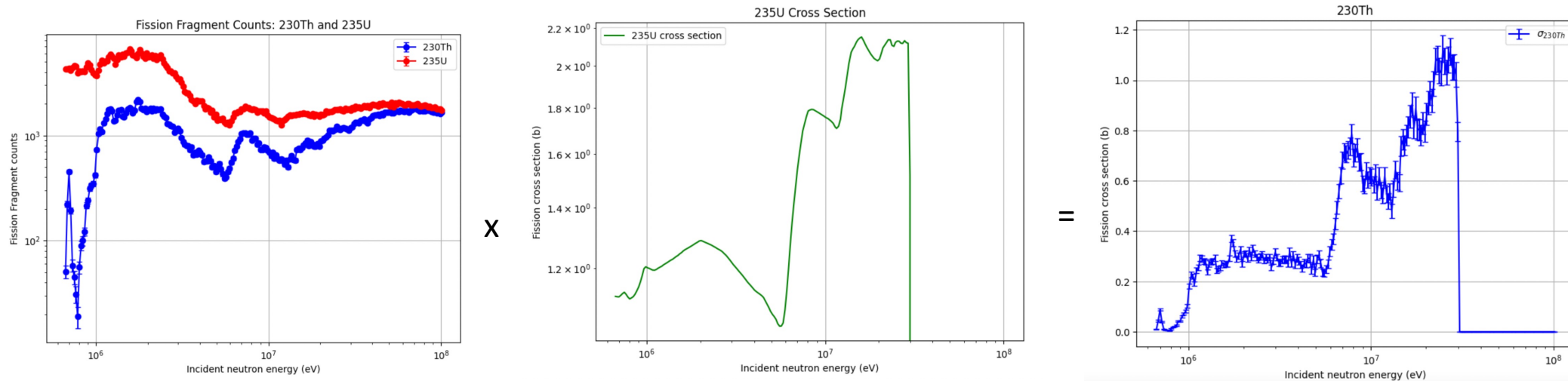
EXERCISE 2: Fission cross section measurements with the TOF technique

2b) Cross section calculation over a wide energy range $^{230}\text{Th}(n,f)$ at $600\text{keV} < E_n < 1\text{ GeV}$
(reference: $^{235}\text{U}(n,f)$ reaction)



- 2b) i)** Check the corrected Fission Fragment Counts for $600\text{keV} < E_n < 1\text{ GeV}$ for ^{230}Th and ^{235}U and the corresponding reference cross section ($^{235}\text{U}(n,f)$).
- 2b) ii)** Cross section calculation with the above corrected Fission Fragment Counts at the energy range $600\text{keV} < E_n < 1\text{ GeV}$ with selected **Binning**: 100 bins/energy decade (PLOT)

EXERCISE 2: Fission cross section measurements with the TOF technique



$$\sigma(E_n) = \frac{Y_{tar,corrected}(E) N_{ref}}{Y_{ref,corrected}(E) N_{tar}} \sigma_{ref}(E_n)$$

- bin-per-bin
- σ_{ref} **interpolation** at bin center

Which factors of σ change over energy???

EXERCISE 2: Fission cross section measurements with the TOF technique

2c: Sensitivity on choice of reference reaction

Effect of two different reference reactions, $^{235}\text{U}(\text{n},\text{f})$ and $^{10}\text{B}(\text{n},\text{a})$ in the extraction of $\sigma(^{243}\text{Am}(\text{n},\text{f}))$.
Which one is better to use (PLOT & DISCUSSION)

$$\sigma(E_n) = \frac{Y_{tar,corrected}(E) N_{ref}}{Y_{ref,corrected}(E) N_{tar}} \sigma_{ref}(E_n)$$

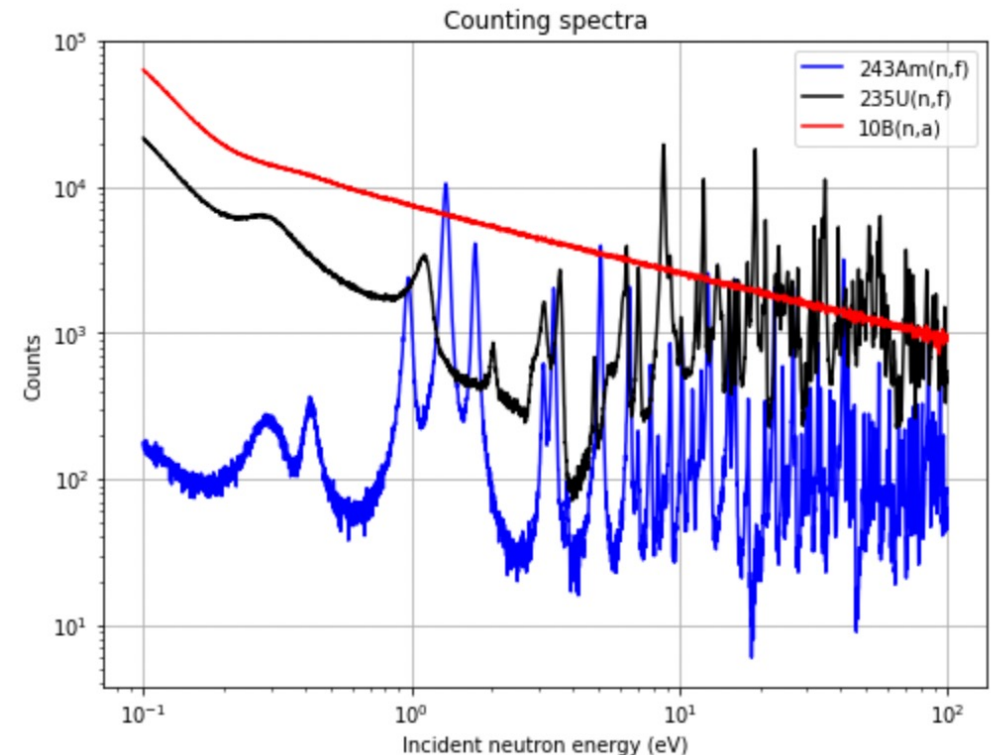
EXERCISE 2: Fission cross section measurements with the TOF technique

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Which one is better to use (PLOT & DISCUSSION)

$$\sigma(E_n) = \frac{Y_{tar,corrected}(E) N_{ref}}{Y_{ref,corrected}(E) N_{tar}} \sigma_{ref}(E_n)$$

2c) i) Comment on the ratios calculated below for the $^{243}\text{Am}(\text{n},\text{f})$ counts with the two different reference reactions.
WHAT WOULD YOU EXPECT????
WHAT DO YOU GET??????

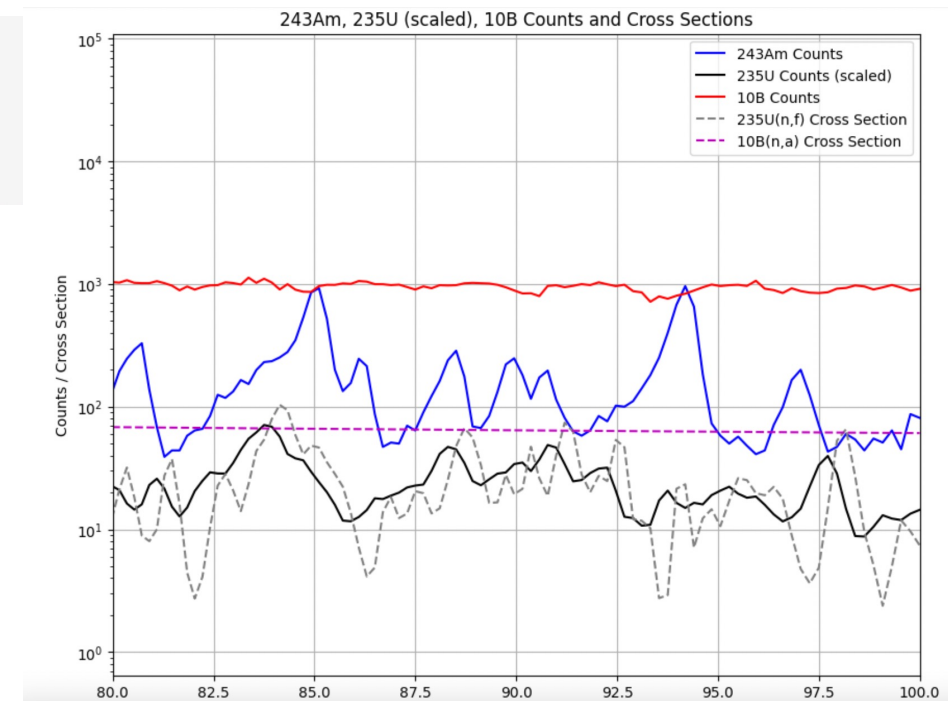


EXERCISE 2: Fission cross section measurements with the TOF technique

2c ii) Zoom in different energy regions to see how the structures of the Target and Reference reactions evolve
(TOF to energy conversion)

```
#####  
plt.xlim(0.01, 30)  
#plt.xlim(80, 100) <----- YOU CAN ZOOM WITH THIS COMMAND  
#####
```

$$\sigma(E_n) = \frac{Y_{tar,corrected}(E) N_{ref}}{Y_{ref,corrected}(E) N_{tar}} \sigma_{ref}(E_n)$$



EXERCISE 2: Fission cross section measurements with the TOF technique

2c iii) Effect of small neutron energy uncertainty in the fission cross section results
Energy uncertainty due to uncertainties in resolution function.

```
#####  
# Apply a Gaussian offset with a REALISTIC ENERGY RESOLUTION from previous exercise standard deviation to the energy  
np.random.seed(42) # For reproducibility  
gaussian_offset = np.random.normal(0, len(df['x'])) # Gaussian offset <-----  
#####
```

From your results, which reference reaction would you prefer to measure the cross section of the $^{243}\text{Am}(n,f)$ reaction at the energy range 1-100 eV (Resolved Resonance Region) ?

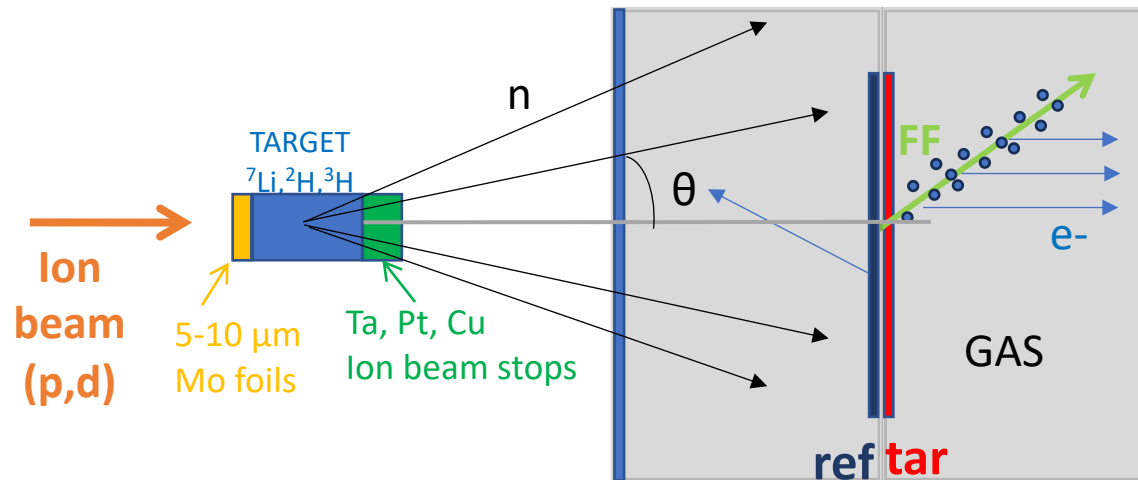
$^{235}\text{U}(n,f)$ or $^{10}\text{B}(n,\alpha)$?????????

2d) Choice of the proper binning (JUST PLOT, ZOOM AND DISCUSS):

```
#####  
#PLAY WITH ENERGY REGION LIMITS (eV) TO APPRECIATE THE EFFECT OF THE BINNING  
plt.xlim(0.5, 100) # <---- (1)  
plt.xlim(0.5, 3) # <---- (2)  
plt.xlim(35, 45) # <---- (3)  
plt.xlim(90, 100) # <---- (4)  
#####
```

EXERCISE 3: Fission cross section measurements with Quasi-monoenergetic beams

- Neutron producing reaction: ${}^3\text{H}(\text{d},\text{n}){}^4\text{He}$
- E_n : ~ 16.5 MeV



$$\sigma(E) = \frac{Y_{tar,corrected}(E) \Phi_{ref}(E) N_{ref}}{Y_{ref,corrected}(E) \Phi_{tar}(E) N_{tar}} \sigma_{ref}(E)$$

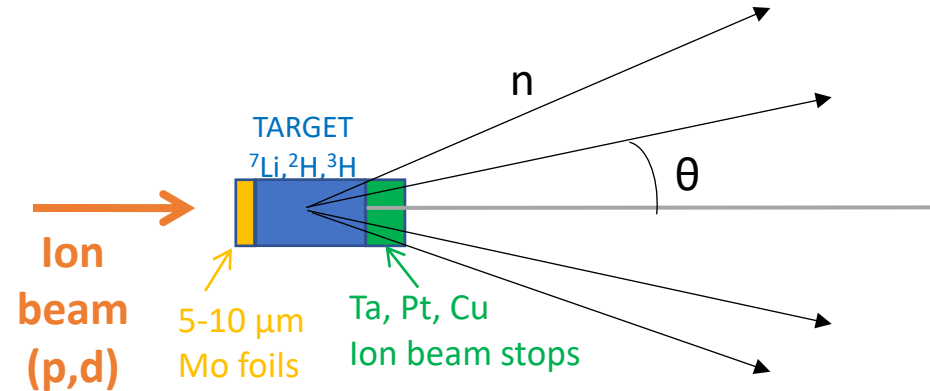
EXERCISE 3: Fission cross section measurements with Quasi-monoenergetic beams

Φ at *quasi*-monoenergetic neutron beams :

A) Main neutron beam - width :

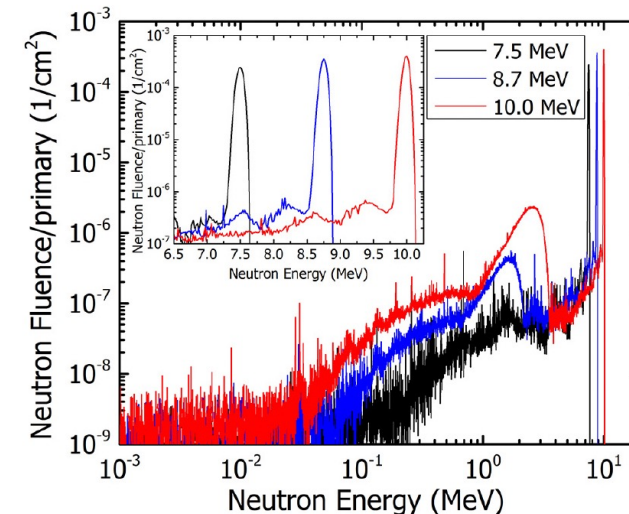
- 1) Energy straggling of the **ion beam**.
- 2) Differential cross section and kinematics:

Typically $\theta=5-10^\circ \Rightarrow \sim$ monoenergetic and isotropic



B) "Parasitic neutrons" :

- 1) Parasitic ion-induced reactions with structural materials of the target and the beam line (Mo, Ti, Ta, Cu, O, C....)
- 2) Deuteron breakup ($^2\text{H}(d,np)$) reaction for $E_d \geq 4.45$ MeV
- 3) Neutron scattering in the surrounding materials



EXERCISE 3: Fission cross section measurements with Quasi-monoenergetic beams

3a) Plot the main neutron peak (Monte Carlo simulation)

3b) Fit the main peak, find the mean neutron energy and extract an energy uncertainty: VERY IMPORTANT FOR FINAL RESULTS GIVEN AT EXFOR.

3c) Estimate the effect of lower energy parasitic neutrons at the fission cross section results for three different reactions: $^{232}\text{Th}(n,f)$ (fertile), $^{235}\text{U}(n,f)$ (fissile) and $^{238}\text{U}(n,f)$ (fertile): PLOT AND DISCUSS THE SENSITIVITY TO LOWER ENERGY NEUTRONS

3d) Estimate the fission reaction rate for the three different reactions: $^{232}\text{Th}(n,f)$, $^{235}\text{U}(n,f)$ and $^{238}\text{U}(n,f)$, taking into account the lower energy parasitic neutron "tail".

3e) Estimate the ratio (parasitic fission reaction rate)/(total reaction rate) for the three different reactions: $^{232}\text{Th}(n,f)$, $^{235}\text{U}(n,f)$ and $^{238}\text{U}(n,f)$, and decide which reference reaction ($^{235}\text{U}(n,f)$ or $^{238}\text{U}(n,f)$) you would use to measure the $^{232}\text{Th}(n,f)$ minimising the uncertainty from the fission events created from the parasitic neutron tail.

ANY IDEAS???

```
#####  
user_defined_energy_max = 15.4 # <----- Hard-coded user-defined maximum energy for partial integral  
#####
```

Congratulations!!

You estimated your first neutron induced fission cross sections both for monoenergetic and white neutron beams!!!

- 1) Systematic uncertainties???
- 2) Important factors to consider when designing a new measurement?
- 3) Now you should give your new data to EXFOR:
 - a) **absolute cross section data** AND the **ratio**
 - b) **statistical uncertainties AND the different systematic uncertainties** SEPARATELY,
 - c) **total uncertainty** with the propagation (with covariance).
 - d) The maximum info relevant to experiment useful to subsequent evaluation.

an EXFOR entry....

example of reported systematic uncertainties :

10B/235U/230Th x7 / 238U in Micromegas detectors.

Target	ID	Mass (mg)	Activity (MBq)	# of nuclei(E+19)
235U			280.0(13)E-06	
230Th	#3	4.546(8)	3.468(6)	1.190(21)
230Th	#4	4.053(13)	3.092(10)	1.061(34)
230Th	#5	2.249(9)	1.716(7)	0.5889(24)
230Th	#6	2.464(8)	1.880(6)	0.6541(21)
230Th	#7	4.118(16)	3.142(12)	1.078(41)
230Th	#8	4.848(13)	3.698(10)	1.269(34)
230Th	#9	4.441(9)	3.388(7)	1.163(24)
238U			179.5(9)E-06	

EN-MIN EV	EN-MAX EV	DATA NO-DIM	1 ERR-S NO-DIM	1 DATA B	2 ERR-S B
6.61455e+5	6.62217e+5	1.09352e-3	7.11759e-4	1.22404e-3	7.96716e-4
6.62217e+5	6.62979e+5	3.59050e-3	1.38969e-3	4.01857e-3	1.55538e-3
6.62979e+5	6.63743e+5	3.21307e-3	1.75505e-3	3.59630e-3	1.96438e-3
6.63743e+5	6.64508e+5	3.12776e-3	1.19931e-3	3.50040e-3	1.34219e-3
6.64508e+5	6.65273e+5	3.75183e-3	1.18004e-3	4.19900e-3	1.32068e-3
6.65273e+5	6.66040e+5	4.04050e-3	1.82634e-3	4.52154e-3	2.04378e-3
6.66040e+5	6.66807e+5	4.49477e-3	3.73704e-3	5.02972e-3	4.18182e-3
6.66807e+5	6.67575e+5	3.88863e-3	1.18115e-3	4.35130e-3	1.32168e-3
6.67575e+5	6.68344e+5	2.98367e-3	8.56934e-4	3.33826e-3	9.58774e-4
6.68344e+5	6.69114e+5	2.92310e-3	1.78787e-3	3.27065e-3	2.00045e-3
6.69114e+5	6.69885e+5	4.45243e-3	2.03719e-3	4.98127e-3	2.27916e-3
6.69885e+5	6.70656e+5	2.31942e-3	1.21449e-3	2.59484e-3	1.35871e-3
6.70656e+5	6.71429e+5	6.18492e-3	1.80067e-3	6.91917e-3	2.01444e-3
6.71429e+5	6.72202e+5	4.07756e-3	1.40796e-3	4.56152e-3	1.57507e-3
6.72202e+5	6.72977e+5	3.50112e-3	1.04792e-3	3.91625e-3	1.17217e-3
6.72977e+5	6.73752e+5	4.66969e-3	1.90001e-3	5.22367e-3	2.12541e-3
6.73752e+5	6.74528e+5	4.11658e-3	2.59082e-3	4.60448e-3	2.89788e-3
6.74528e+5	6.75305e+5	6.28822e-3	1.57658e-3	7.03278e-3	1.76326e-3
6.75305e+5	6.76083e+5	3.17999e-3	9.13355e-4	3.55643e-3	1.02148e-3
6.76083e+5	6.76862e+5	6.77362e-3	1.98400e-3	7.57532e-3	2.21882e-3
6.76862e+5	6.77642e+5	5.74632e-3	1.79912e-3	6.42634e-3	2.01202e-3
6.77642e+5	6.78422e+5	2.55076e-3	1.01814e-3	2.85234e-3	1.13851e-3
6.78422e+5	6.79204e+5	6.04240e-3	2.38555e-3	6.75671e-3	2.66757e-3
6.79204e+5	6.79986e+5	6.68723e-3	2.11132e-3	7.47708e-3	2.36070e-3
6.79986e+5	6.80769e+5	8.92851e-3	4.46651e-3	9.98295e-3	4.99399e-3
6.80769e+5	6.81554e+5	4.21810e-3	2.10917e-3	4.71618e-3	2.35822e-3
6.81554e+5	6.82339e+5	9.31440e-3	1.81686e-3	1.04133e-2	2.03122e-3
6.82339e+5	6.83125e+5	6.97792e-3	1.58455e-3	7.80113e-3	1.77149e-3
6.83125e+5	6.83912e+5	8.84705e-3	2.07038e-3	9.88992e-3	2.31443e-3
6.83912e+5	6.84699e+5	6.17144e-3	2.09744e-3	6.89834e-3	2.34448e-3
6.84699e+5	6.85488e+5	1.68059e-2	3.63603e-3	1.87851e-2	4.06425e-3
6.85488e+5	6.86278e+5	7.05014e-3	2.37389e-3	7.88043e-3	2.65345e-3
6.86278e+5	6.87068e+5	9.26765e-3	2.54381e-3	1.03583e-2	2.84316e-3
6.87068e+5	6.87860e+5	1.24875e-2	2.96969e-3	1.39559e-2	3.31890e-3
6.87860e+5	6.88652e+5	1.08457e-2	3.53388e-3	1.21211e-2	3.94942e-3
6.88652e+5	6.89446e+5	1.04189e-2	1.90425e-3	1.16440e-2	2.12816e-3
6.89446e+5	6.90240e+5	1.11217e-2	2.44391e-3	1.24276e-2	2.73086e-3
6.90240e+5	6.91035e+5	1.45150e-2	4.38134e-3	1.62194e-2	4.89580e-3
6.91035e+5	6.91831e+5	1.87576e-2	4.03942e-3	2.09585e-2	4.51339e-3
6.91831e+5	6.92628e+5	2.18032e-2	3.39014e-3	2.43616e-2	3.78794e-3
6.92628e+5	6.93426e+5	1.14185e-2	3.92467e-3	1.27575e-2	4.38489e-3