IMORN-31

Presentation: Coupled neutronic-thermohydraulic coupled simulations for neutronic noise studies in molten salt reactors

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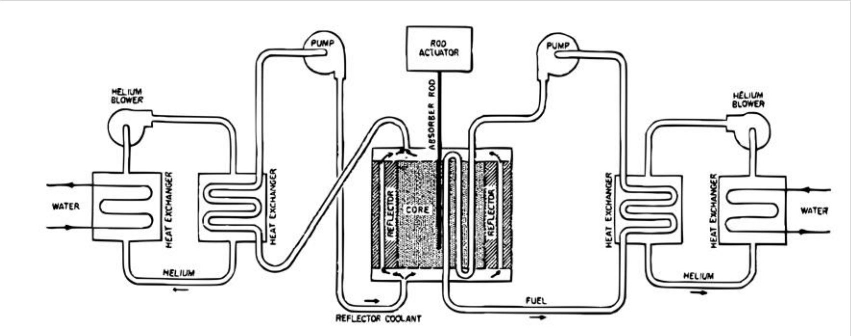
Contents

Part 1: Introduction

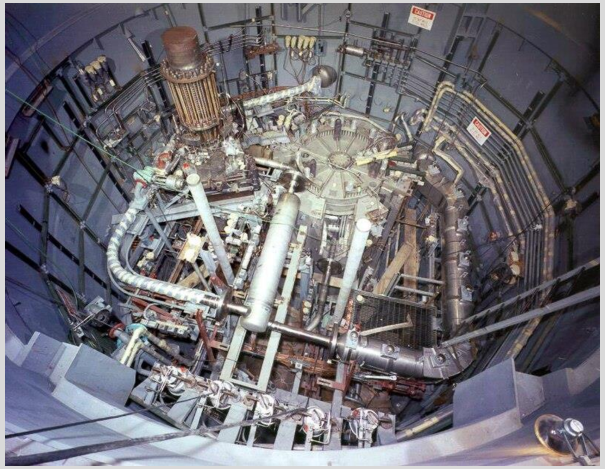
The molten salt reactors

Molten Salt Reactors (MSR) considered:

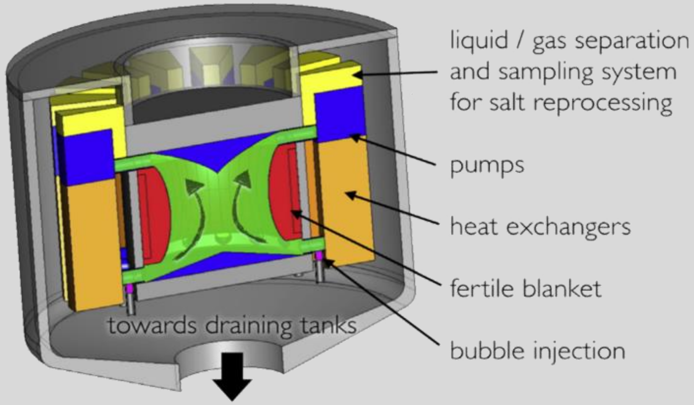
- Liquid fuel
- The fuel is also circulating and serves as coolant (as in most of the concepts)



Scheme of the aircraft reactor experiment (ORNL 1954)



Molten salt reactor experiment (ORNL 1964 – 1969)

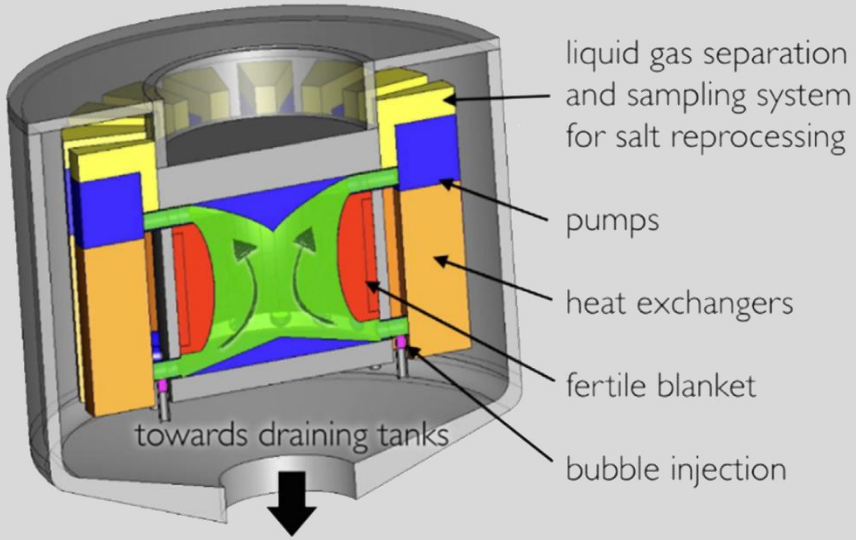


Scheme of the molten salt fast reactor (2000 – nowadays)

The molten salt fast reactor (MSFR)

This concept is the base design used for most of this work. The characteristics of its reference version are*:

- $3GW_{th}$ nominal power
- no internal structures (very turbulent flow inside the core)
- fast neutronic spectrum
- breeder reactor using the $^{232}Th/^{233}U$ cycle



Scheme of the molten salt fast reactor

* *Michel Allibert, Sylvie Delpech, Delphine Gerardin, Daniel Heuer, Axel Laureau, Elsa Merle, Chapter 7 - Homogeneous Molten Salt Reactors (MSRs): The Molten Salt Fast Reactor (MSFR) concept Editor(s): Igor L. Pioro, In Woodhead Publishing Series in Energy, Handbook of Generation IV Nuclear Reactors (Second Edition), 2023, Pages 231-257, ISBN 9780128205884*

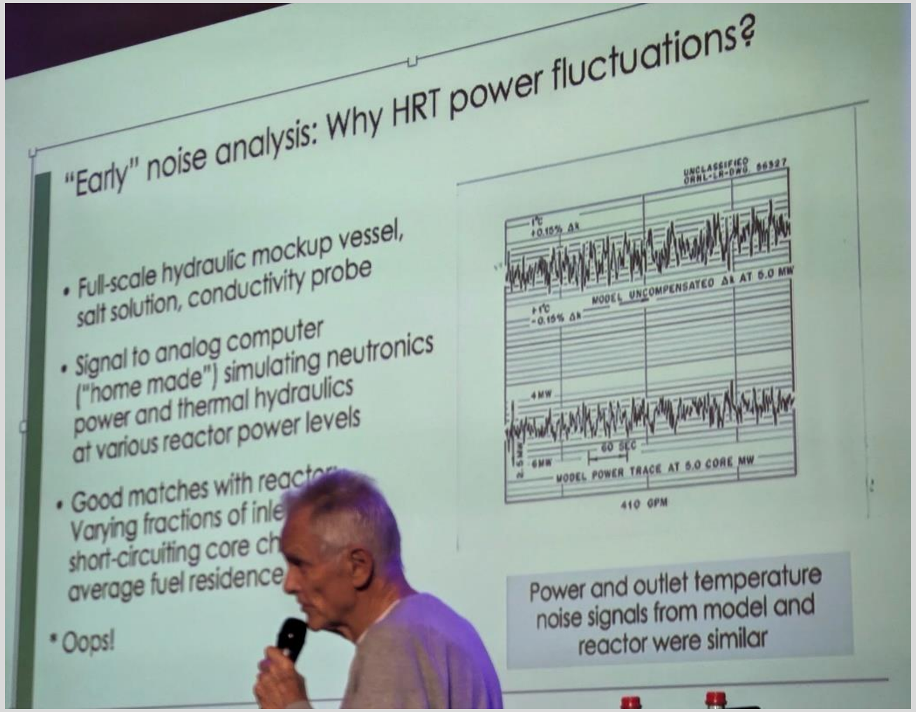
Link with neutronic noise

Objective:

- Study the impact of turbulence on neutronic noise:
  - → Transport of delayed neutron precursors
  - → Local variations of salt properties

Means:

- Create simulation code that can solve the coupled physics (neutronics and thermohydraulics) and chain a noise simulation code



Syd Ball, MSRE operator, presenting measurements on the impact of bubbling on neutron noise in the MSRE (French MSR bootcamp, October 2023).

Part 2:
Neutronic-thermohydraulic coupling

Neutronic-thermohydraulic coupling

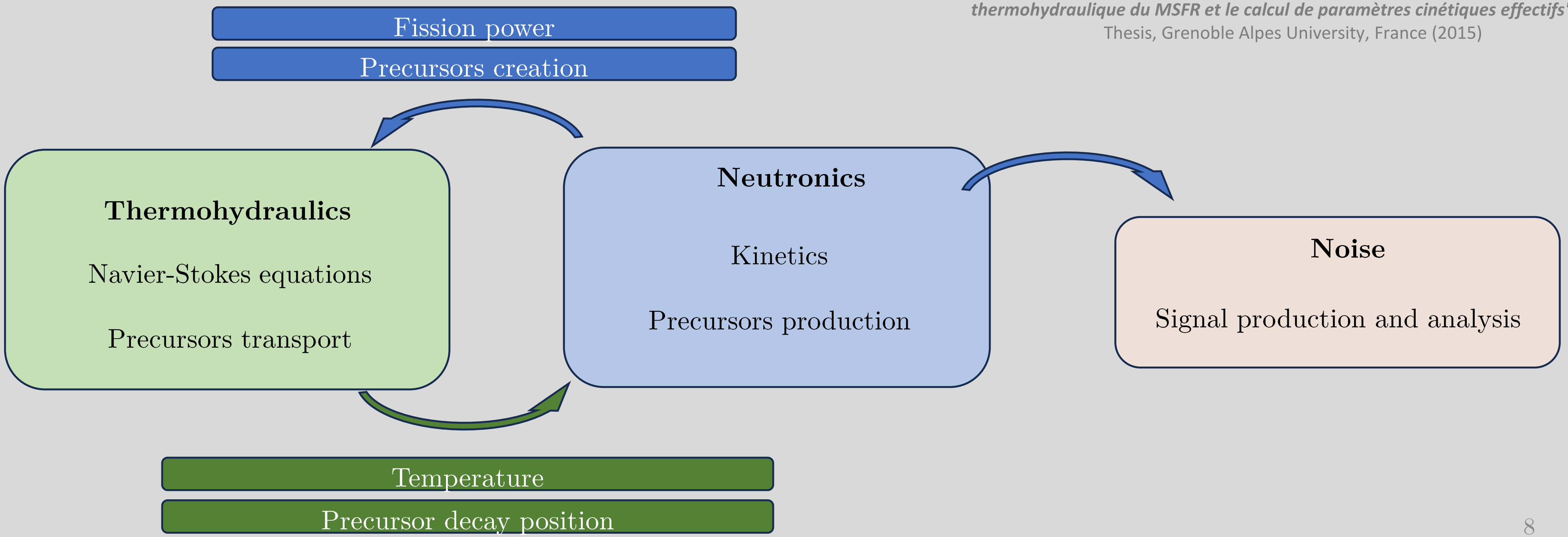
- To compute **fluid flow, temperature field and delayed neutron precursors transport** by the fuel salt:
  → Use of computational fluid dynamics (CFD) solver. **Fluent** has been chosen for this work

- To compute **fission power and delayed neutron production**:
  → Use of the **transient fission matrix (TFM)** approach, based on a Monte Carlo simulation

- To create the **noise signals**:
  → Use of the **TFM** approaach to create **neutron showers**

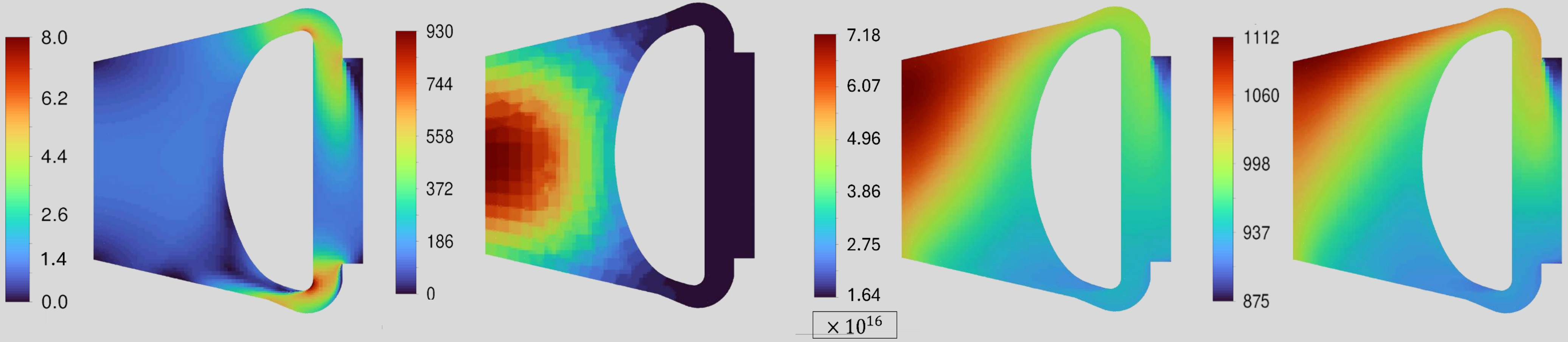
A. Laureau, M. Aufiero, P. Rubiolo, E. Merle-Lucotte, D. Heuer, "**Transient Fission Matrix: kinetic calculation and kinetic parameters β_{eff} and Λ_{eff} calculation**", Annals of Nuclear Energy, volume 85, p. 1035–1044 (2015)

Axel LAUREAU, ***"Développement de modèles neutroniques pour le couplage thermohydraulique du MSFR et le calcul de paramètres cinétiques effectifs"***, PhD Thesis, Grenoble Alpes University, France (2015)



Coupled computations

- Steady state is reached by imposing total power precursor quantity
- Reynolds Average Navier-Stokes (RANS) with $k-\epsilon$ realizable model for thermohydraulics
- Point kinetics (TFM) for neutronics
- Reactivity insertions are being tested (transient simulations)



Velocity magnitude $[m.s^{-1}]$

Volumetric power $[MW.m^{-3}]$

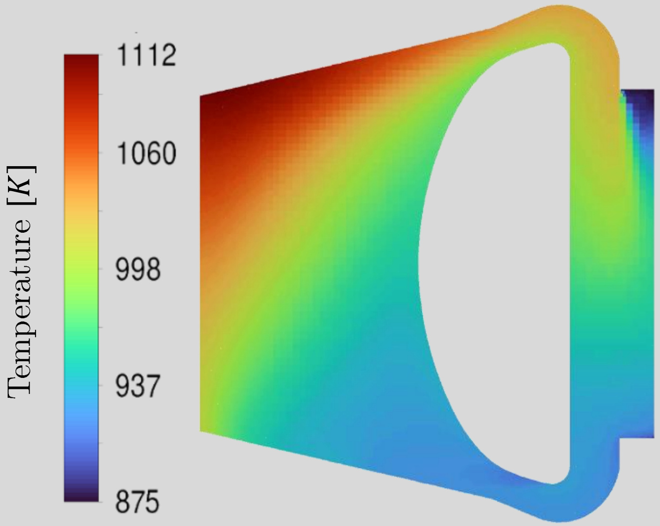
Volumetric delayed neutron emission rate $[m^{-3}.s^{-1}]$

Temperature $[K]$

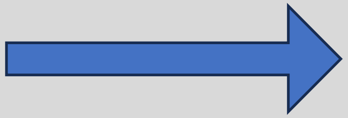
CFD: RANS versus DES

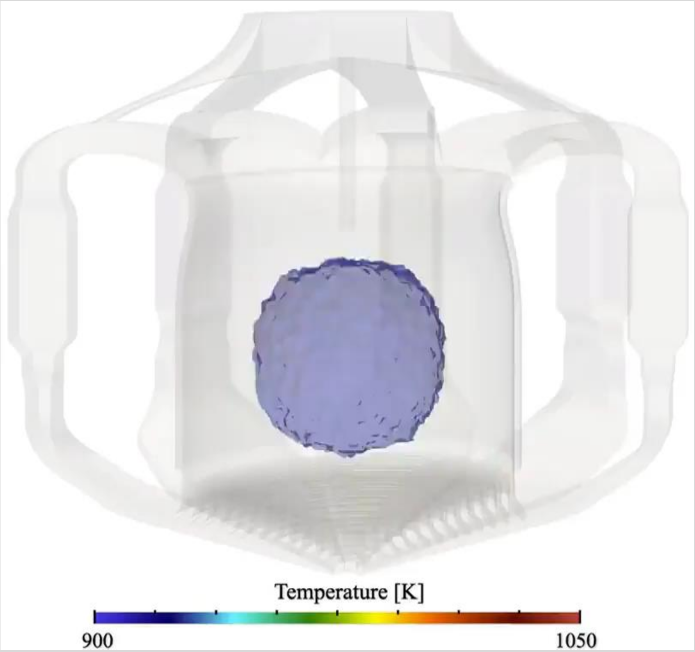
- Move from RANS to Detached Eddy Simulations (DES). It allows for computation of intermediate scale of turbulence, instead of modelling them.

- Study the influence of turbulence on local parameters and the final impact on noise measurement



Temperature field obtained with RANS simulation



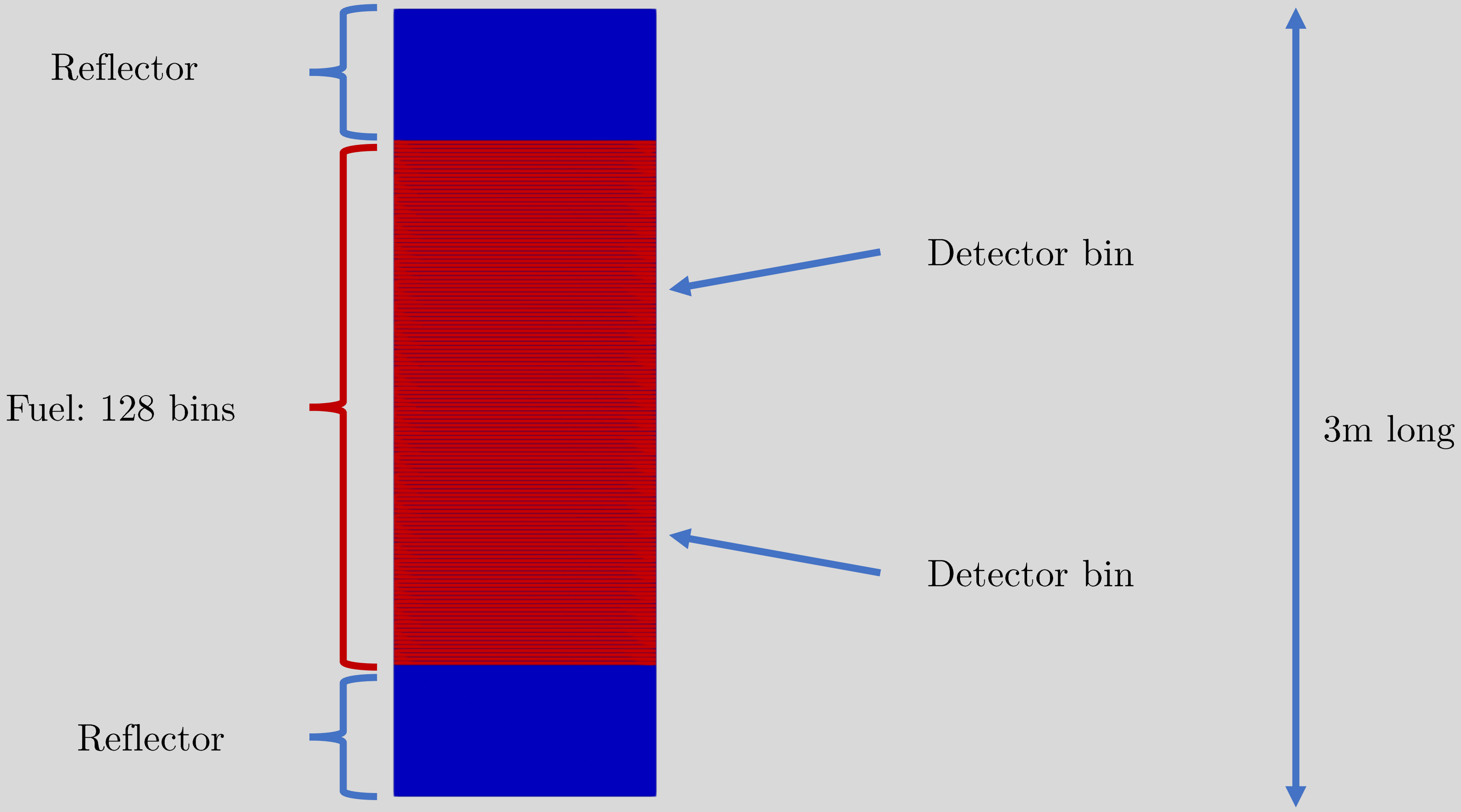


Temperature field obtained with DES simulation

(*A. Laureau, A. Bellè, M. Allibert, D. Heuer, E. Merle, A. Pautz,* ***"Unmoderated molten salt reactors design optimisation for power stability"****, Annals of Nuclear Energy, 177, p. 109265 (2022)*)

Part 3:
Neutronic noise

Illustration case: 1D fast reactor

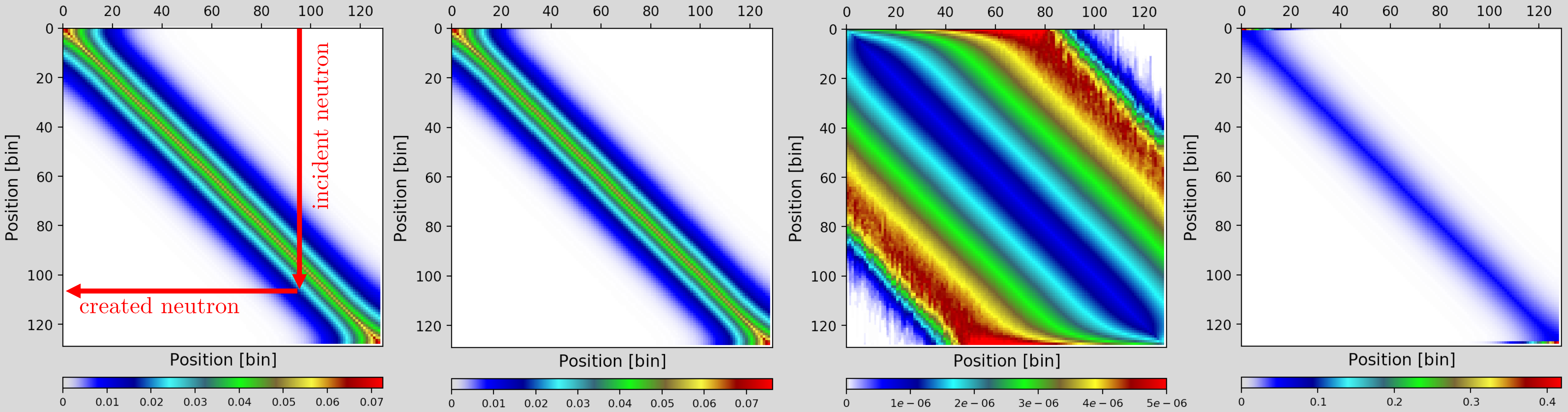


Scheme of the 1D reactor used as illustration case

3.1 Transient fission matrix approach

Transient fission matrices

- Matrices are pre-computed with Monte Carlo calculations (with OpenMC code)
- They contain the probability of absorption of one neutron born in bin **j** in all other bins **i**
- There is one matrix per interaction and per neutron type (delayed / prompt) and one for transport time
- The scattering is not resolved and only the fissions and captures are computed
- This model was initially developed for neutronic-thermohydraulics coupling and their application to noise simulation is tested in this work



$\underline{\underline{K}}_{\chi_p \to \nu_p}$ $\underline{\underline{K}}_{\chi_d \to \nu_p}$ $\underline{\underline{T}}_{p \to p}$ $\underline{\underline{A}}_d$

How to get kinetic parameters from matrices

$\underline{\underline{K}}_{tot} = \begin{pmatrix} \underline{\underline{K}}_{pp} & \underline{\underline{K}}_{dp} \\ \underline{\underline{K}}_{pd} & \underline{\underline{K}}_{dd} \end{pmatrix}$ is the complete transport operator:

→ Eigen value is the multiplication coefficient

→ Eigen vector is the steady state neutron distribution

$$\underline{\underline{K}}_{tot}\ \underline{V}_{tot} = k_{eff}\ \underline{V}_{tot} \quad \text{with} \quad \underline{V}_{tot} = (\underline{V}_p, \underline{V}_d)$$

It is also possible to define the importance of neutrons by making the same computation with the transposed operator

$$\underline{\underline{K}}^*_{tot}\ \underline{V}^*_{tot} = k_{eff}\ \underline{V}^*_{tot}$$

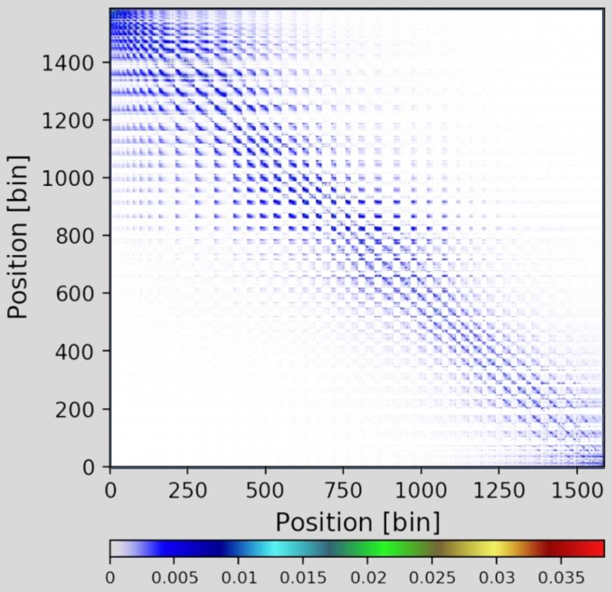
Other kinetic parameters and their effective values (adjoint flux weighted):

Delayed neutron fraction

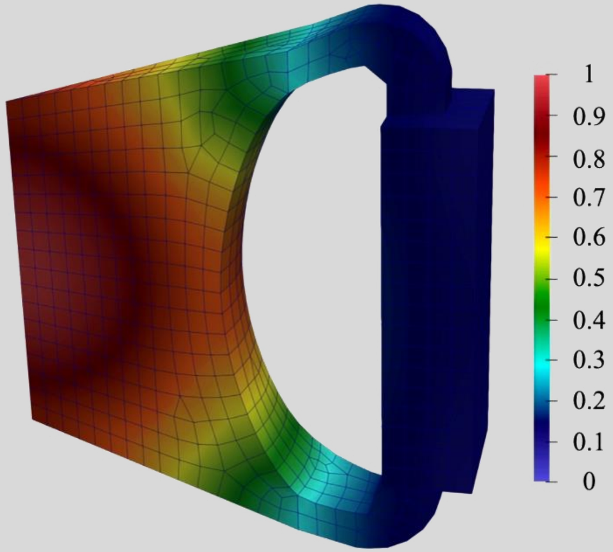
$$\beta_0 = \frac{\Sigma_i (\underline{V}_d)_i}{\Sigma_i (\underline{V}_{tot})_i} \quad \text{and} \quad \beta_{eff} = \frac{\underline{V}^*_d . V_d}{\underline{V}^*_{tot} . V_{tot}}$$

Fission to fission time

$$l = \frac{\Sigma_i \left(\underline{\underline{K}}_{pp}\ \underline{\underline{T}}_{pp}\ \underline{V}_p\right)_i}{\Sigma_i \left(\underline{\underline{K}}_{pp}\ \underline{V}_p\right)_i} \quad \text{and} \quad l_{eff} = \frac{\underline{V}^*_p . \left(\underline{\underline{K}}_{pp}\ \underline{\underline{T}}_{pp} V_p\right)}{\underline{V}^*_p . \left(\underline{\underline{K}}_{pp} V_p\right)}$$



$K_{\chi_p \to \nu_p}$

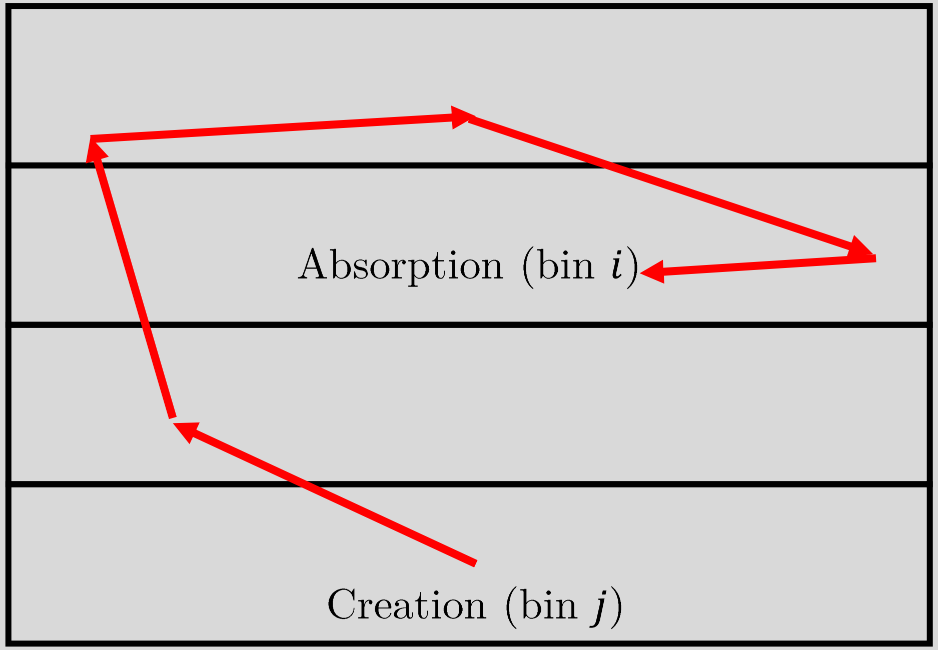


Eigenvector [a.u.]

Transient fission matrices vs Monte Carlo

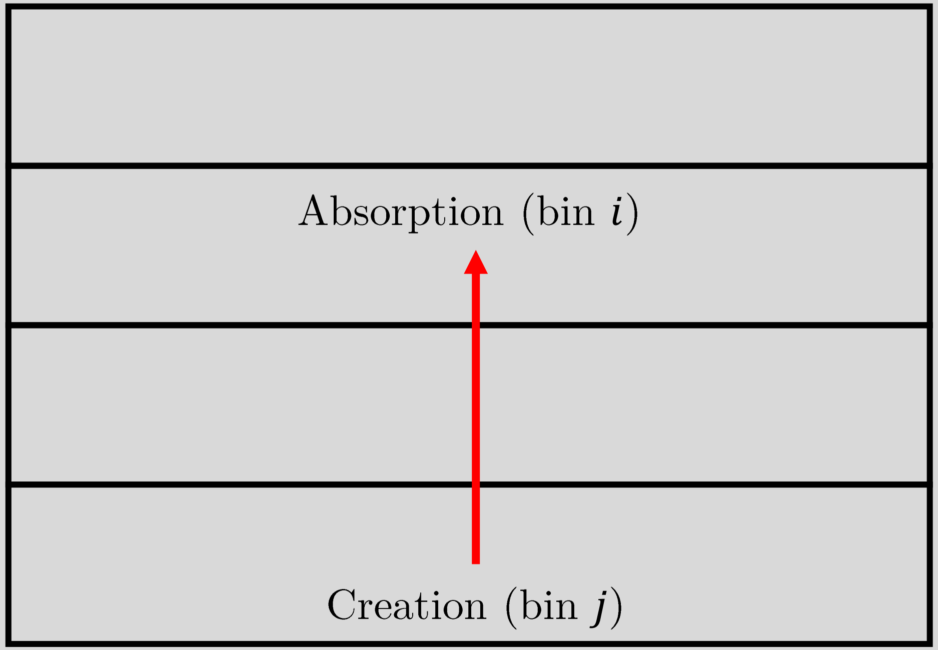
- **TFM** is a **macroscopic approach**, making it fast for producing neutron showers
- Reminder: only **absorptions and fissions**, **once** per neutron

Classic Monte Carlo simulation from j to i



Neutron transport requires multiple samples (distance, reaction…) up to neutron absorption

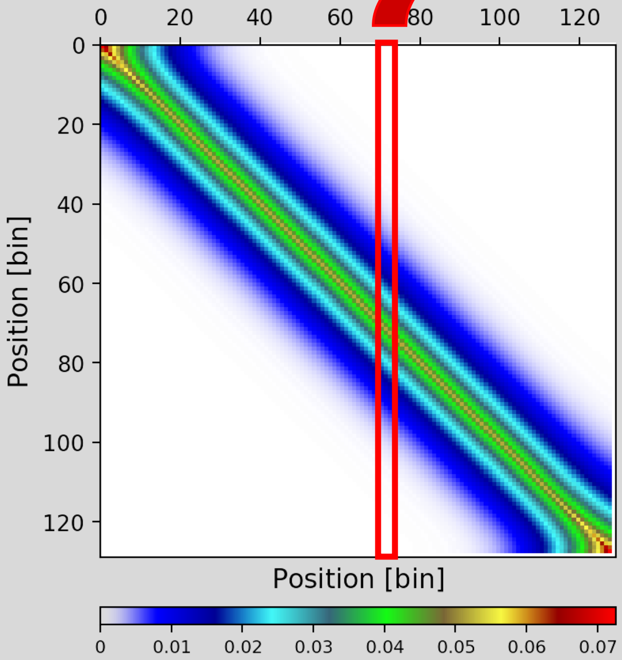
TFM based Monte Carlo Simulation from j to i



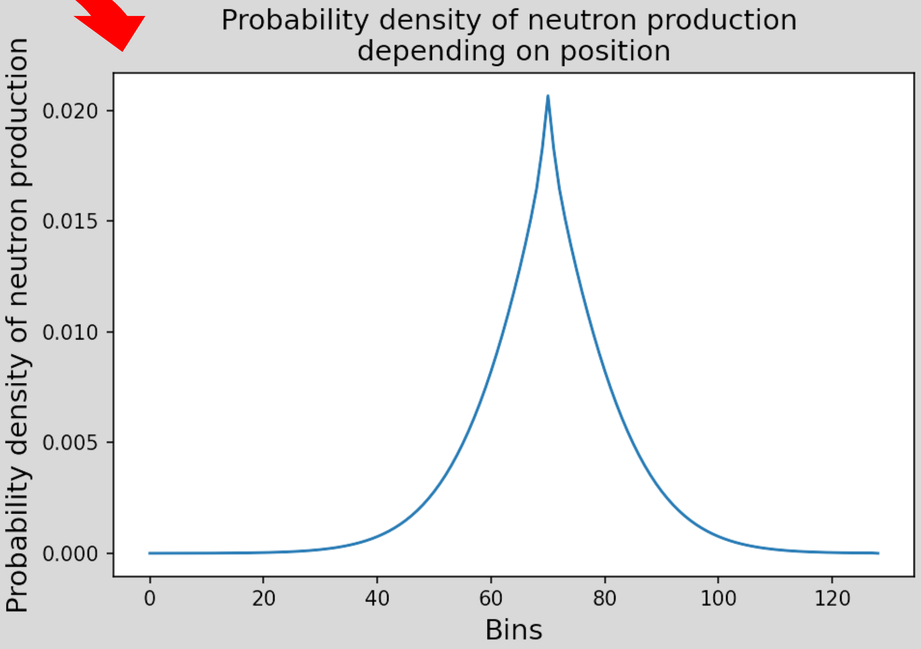
We only sample one destination bin and if a fission or a capture occurs

How to sample?

Extract the column from the matrix where the neutron is emitted

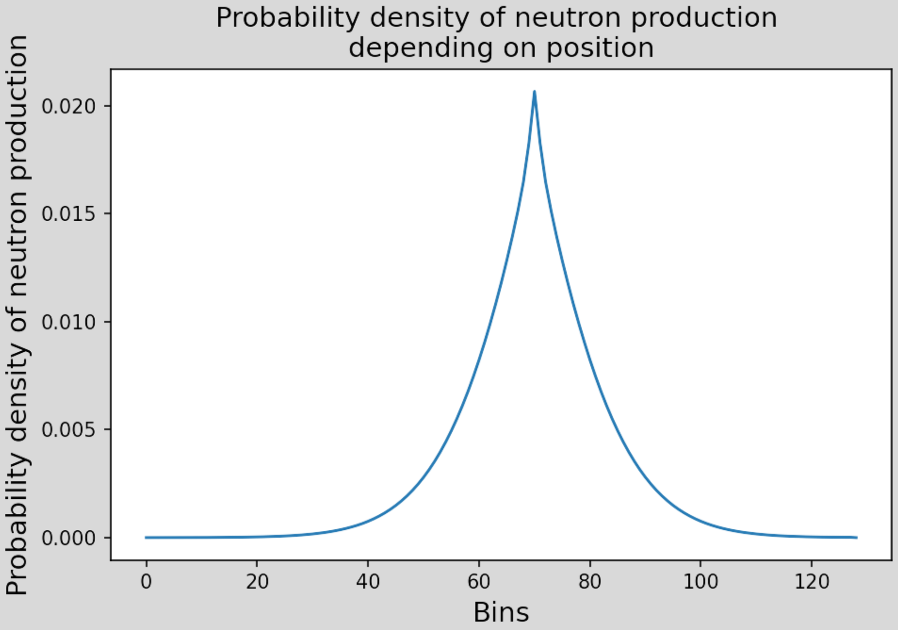


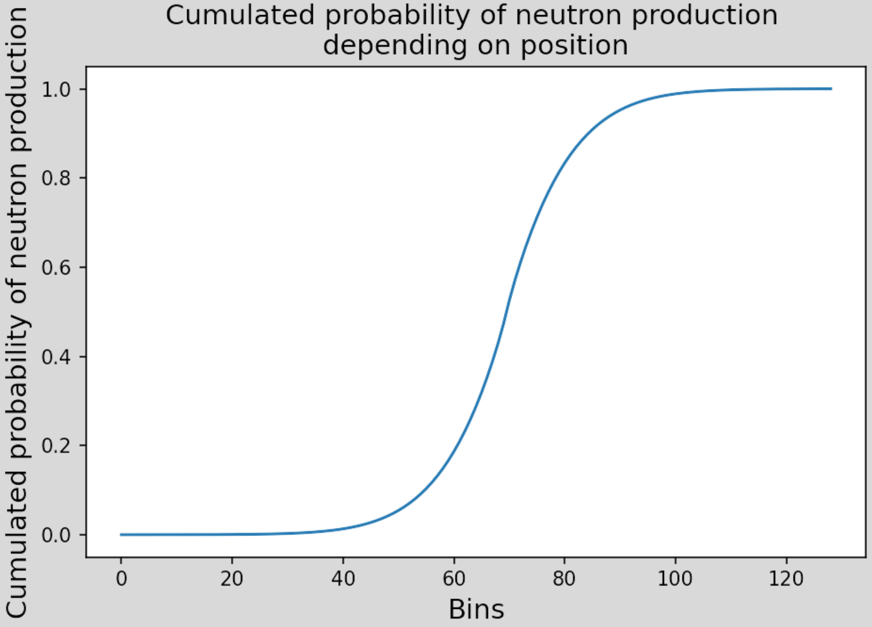
$\underline{\underline{K}}_{\chi_p \rightarrow \nu_p}$



How to sample?

Make a cumulated probability density function from the matrix column





How to sample?

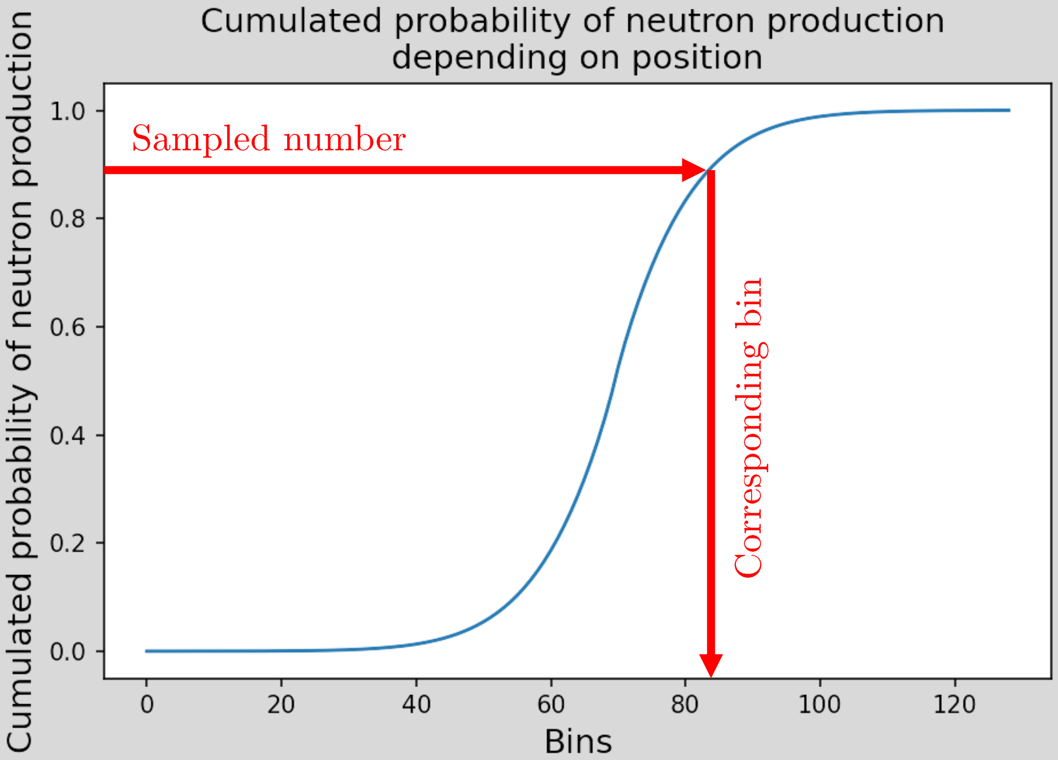
The cumulated density function is obtained from the matrix column j to choose one destination cell

Pick a random number in $[0, 1]$ and find which bin i corresponds in the cumulated density function

Pick another random number in $[0, 1]$. If the result is lower than $\frac{\left(\underline{\underline{K}}\, p\right)_{i \leftarrow j}}{\left(\underline{\underline{A}}\, p\right)_{i \leftarrow j}}$, it is a fission!

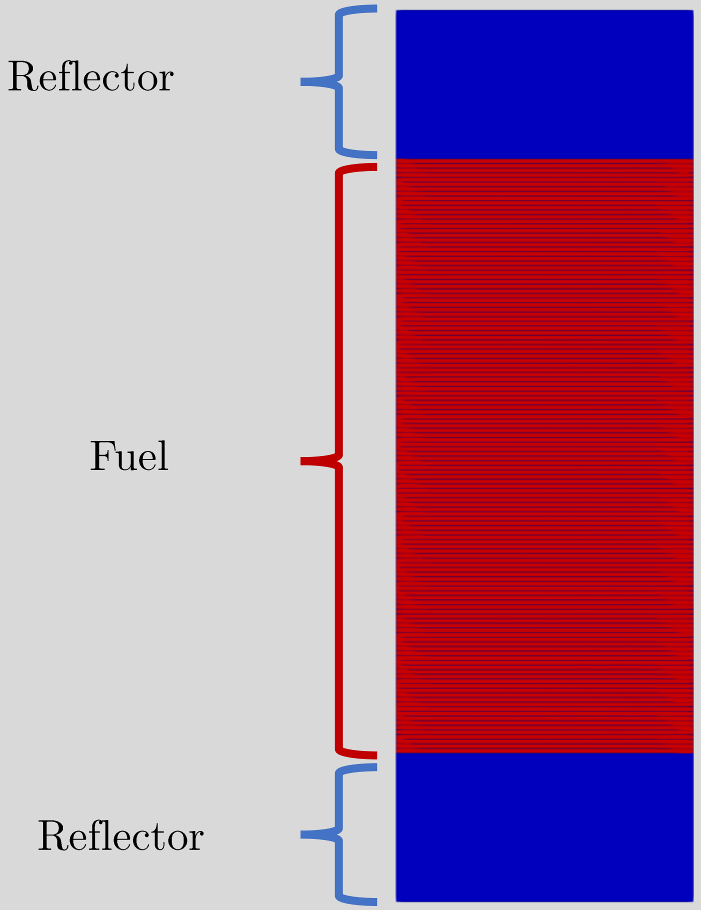
Pick between 2 or 3 neutrons to produce (avg $= \nu_i$)

Repeat from last fission position for all created neutrons

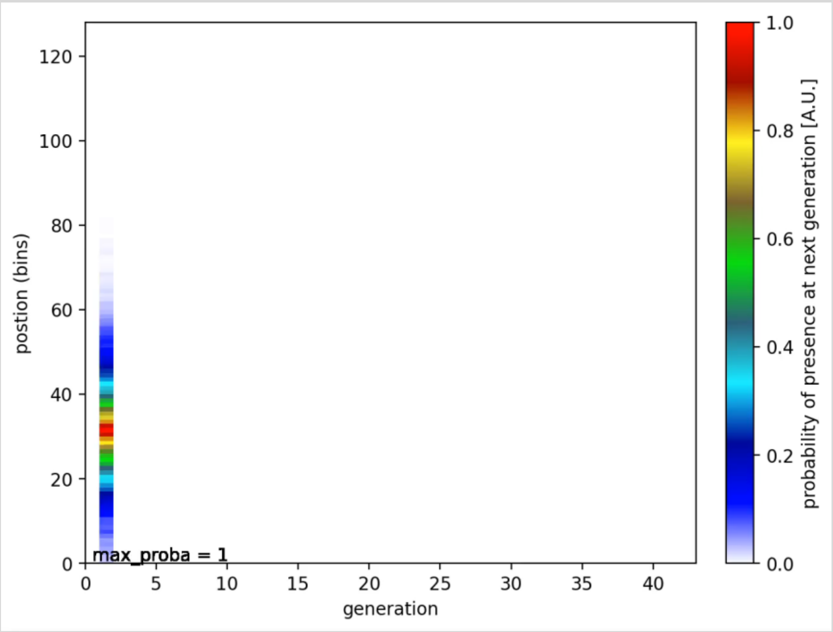


3.2 Results and bias identification

Example of neutron shower calculation



1D reactor used for illustration case

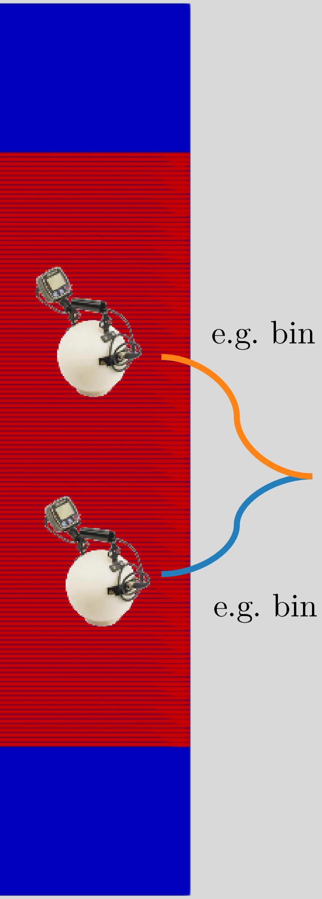


Neutron shower inside the 1D reactor:
Position of neutrons depending on generation

Time signal and cross-correlation (1D reactor)

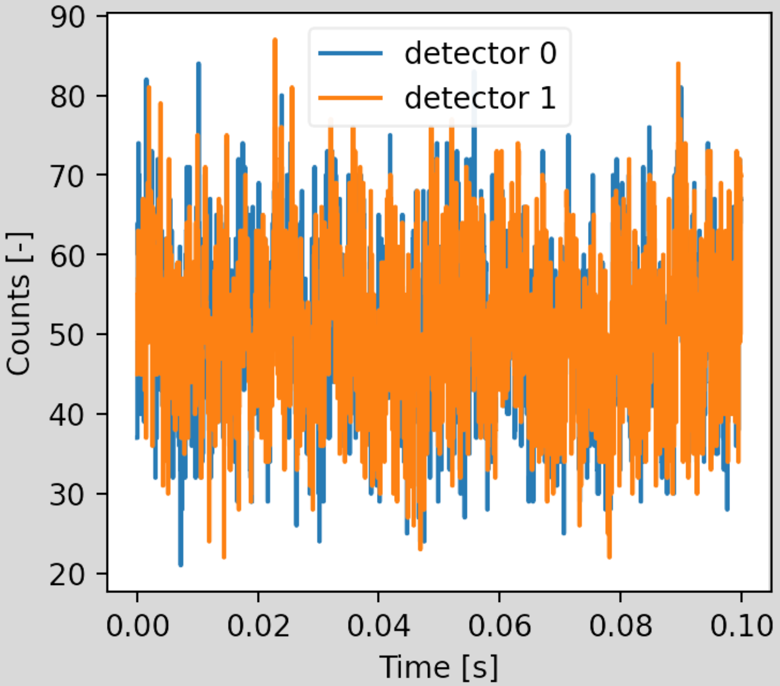
- α-Rossi method used here: the time signal is transformed with cross-correlation between 2 detectors, then the cross-correlation is fitted with function

$$N(t) = Ae^{-\alpha t} + c$$



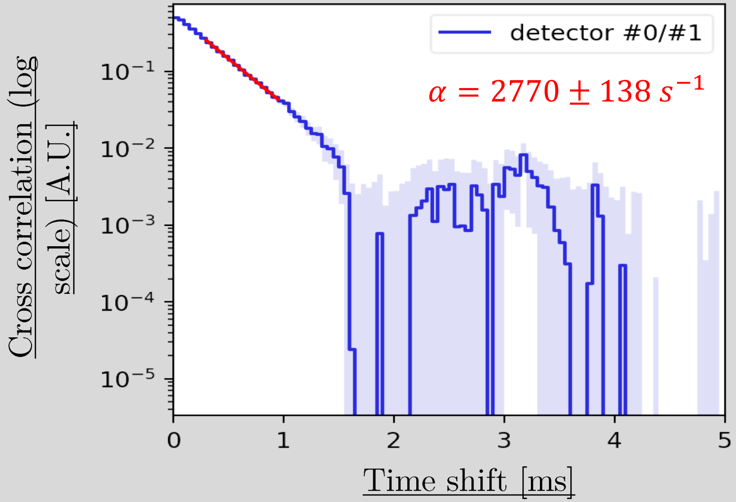
e.g. bin 88

e.g. bin 40



Temporal signal with 10^6 detections/second and detector efficiency = 1% Dwell time = 5.10^{-5} sec

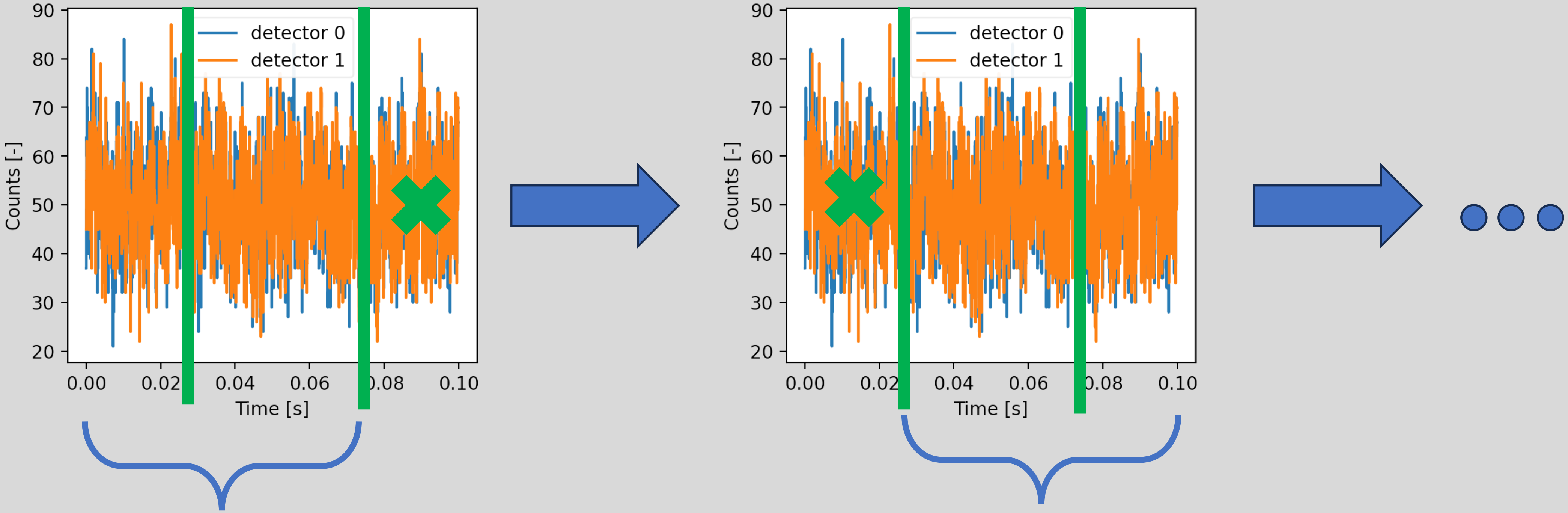
- Cross-correlation of the time signal
- Fitting
- Jackknife



Cross correlation of the time signal. The uncertainty is measured with the jackknife technique

The jackknife technique

To determine the uncertainty on the α-Rossi with only one time measurement, a jackknife technique is used (typically, 32 intervals are used)



→ cross-correlation → α_1

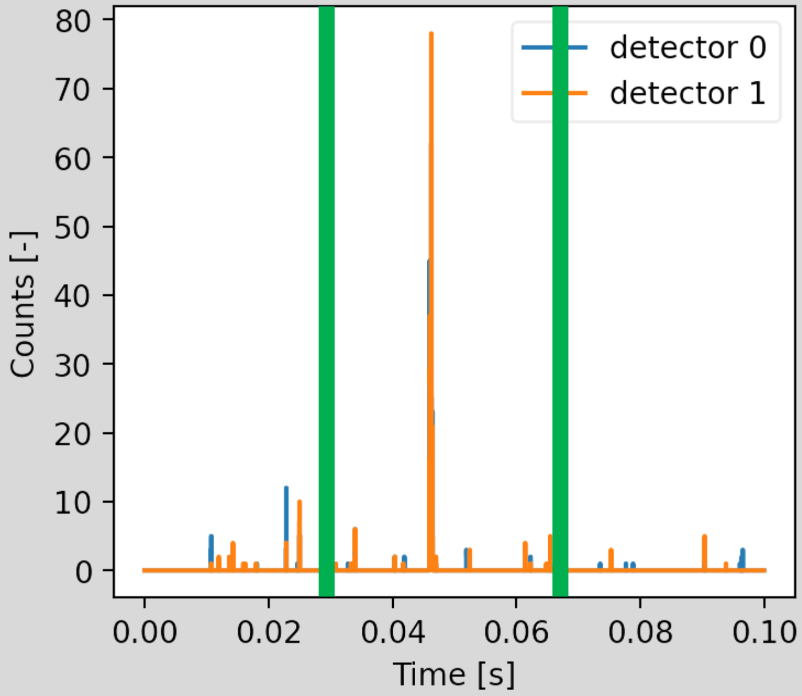
→ cross-correlation → α_2

$$\alpha = \frac{\sum \alpha_i}{n}$$

$$\sigma(\alpha) = \frac{1}{n-1} \sum (\alpha_i - \alpha)^2$$

The jackknife technique

Issues can occur if the count rate is too low and the detector efficiency is high:

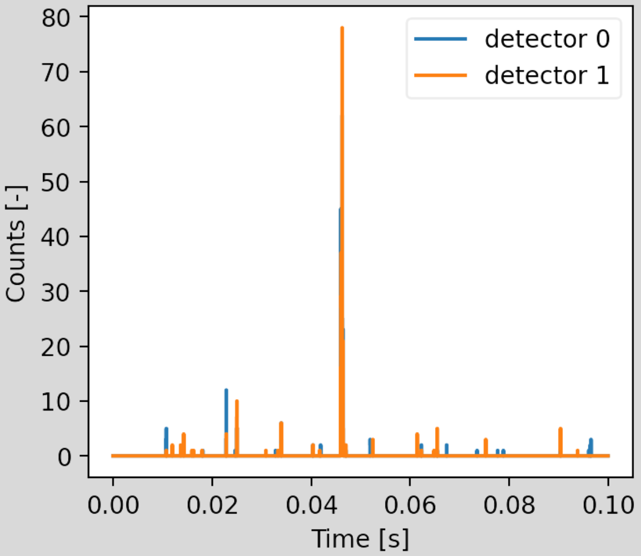


In this case the value of α_i will wary a lot, yielding poor results

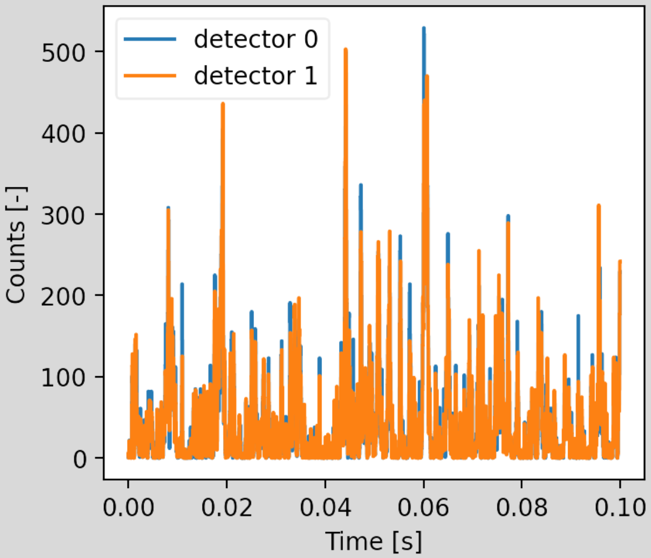
Influence of simulation ("experiment") configuration

→ Reactor power and detector efficiency can be easily modified

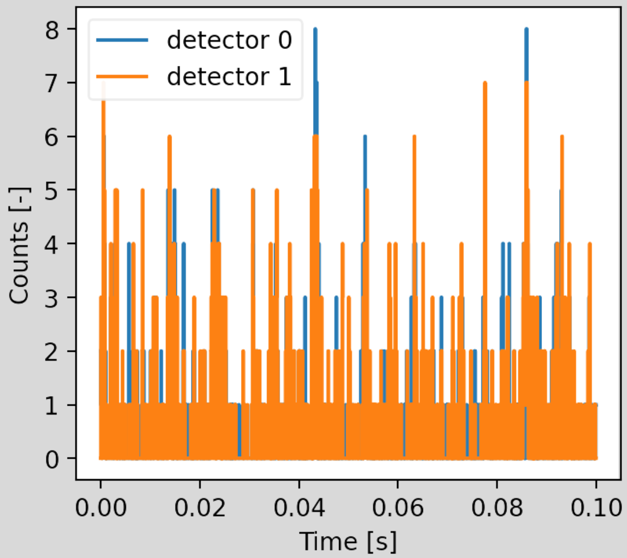
Detections per second = 10^4
Detector efficiency = 100%



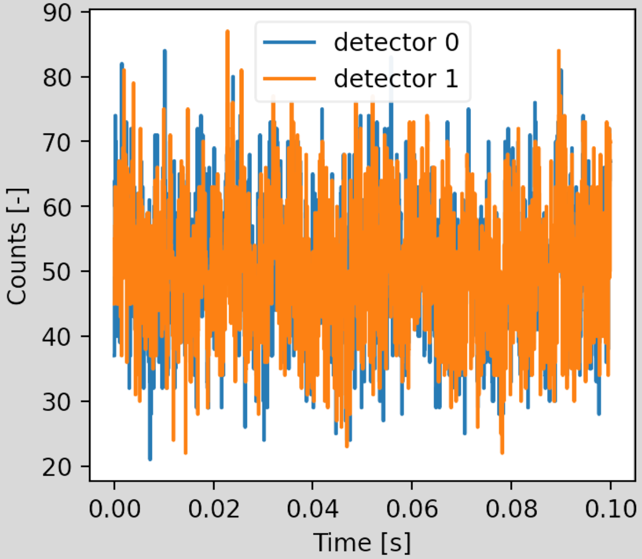
Detections per second = 10^6
Detector efficiency = 100%



Detections per second = 10^4
Detector efficiency = 1%



Detections per second = 10^6
Detector efficiency = 1%



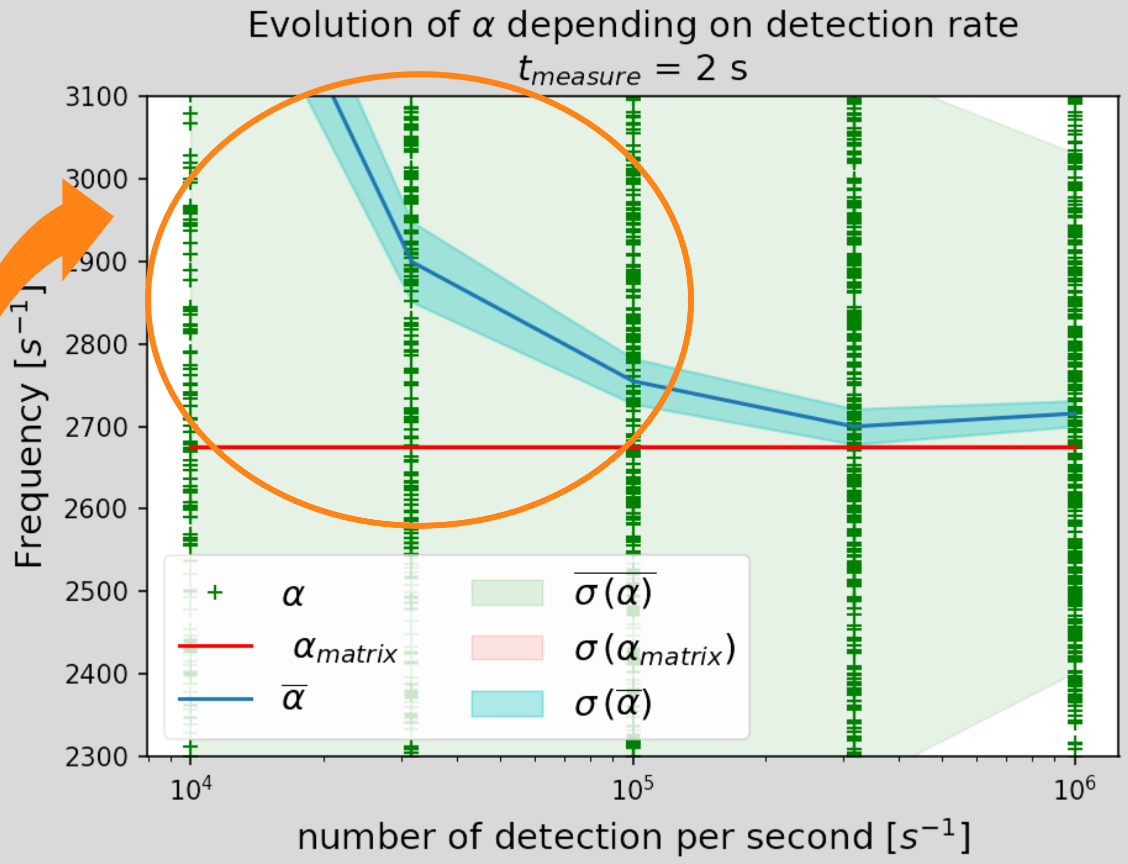
High detector efficiency $\Rightarrow$ more time correlated detections

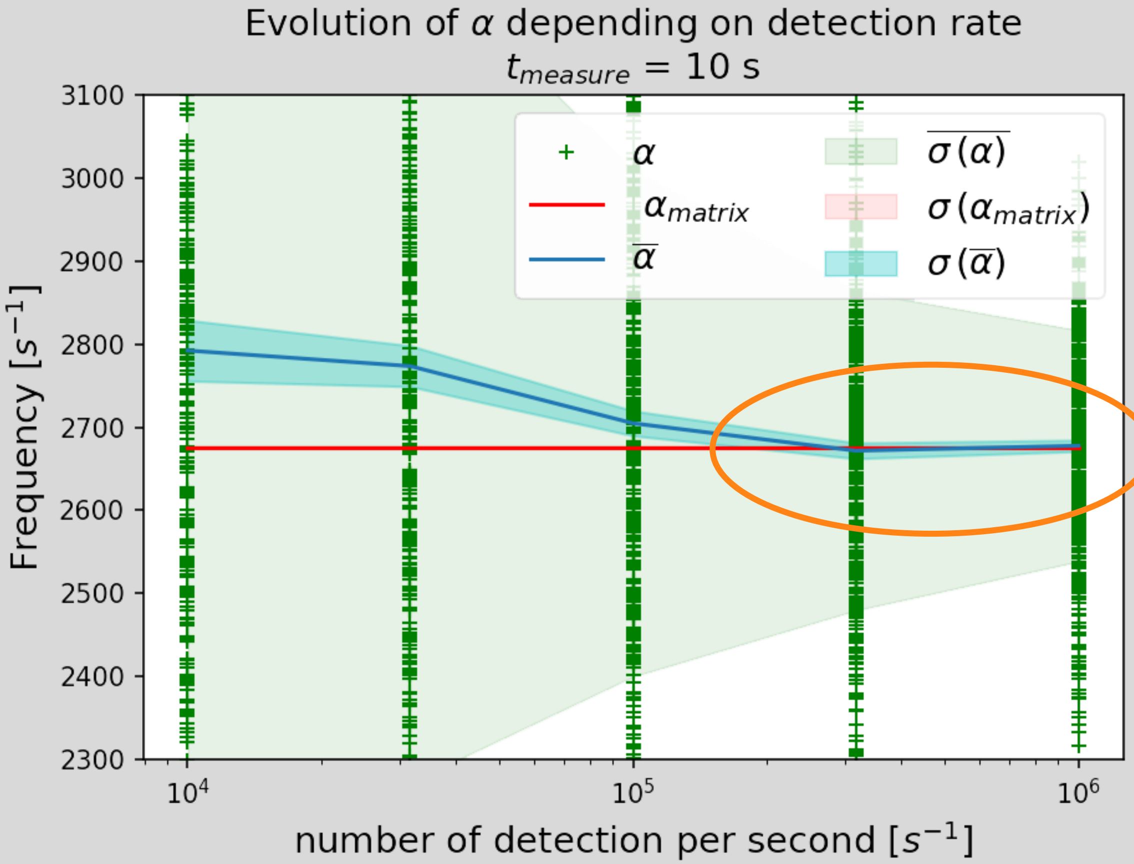
Low detector efficiency $\Rightarrow$ higher baseline

→ Realistic signal requires low detector efficiency

→ Typical physical time 2-10 seconds (up to few minutes)

Influence of simulation ("experiment") configuration: Impact on the α-Rossi

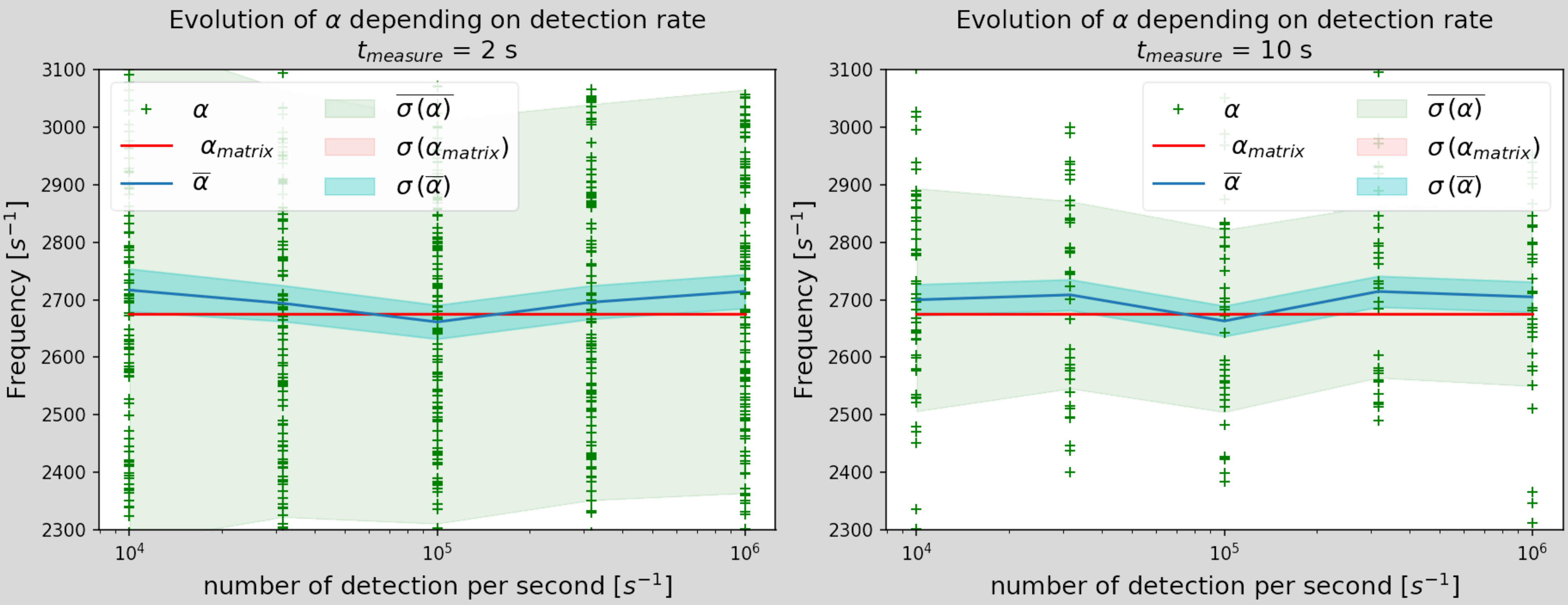




Detector efficiency = 100% → lot of singular events → poor results with low count rates and low measure times

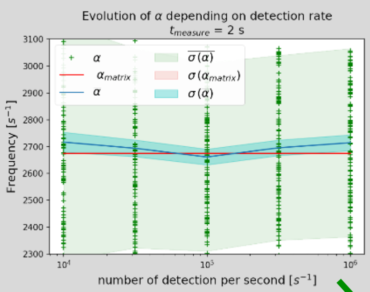
Good results requires a longer simulated time and high count rate, especially if the detector efficiency is high

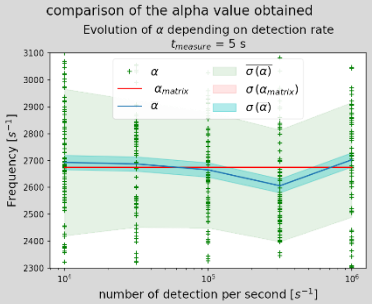
Influence of simulation ("experiment") configuration: Impact on the α-Rossi

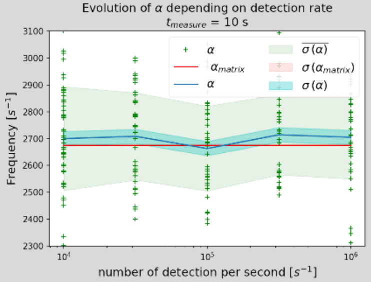


With lower detector efficiency (1%), the **results are compatible with the expected value**. To confirm this conclusion, more simulations need to be carried out to reduce further the uncertainty on the "measurement". Unfortunately, this requires a lot of time (1% detector efficiency → thousands of simulations and ~15 minutes per simulation on 160 threads)

Residue matrices

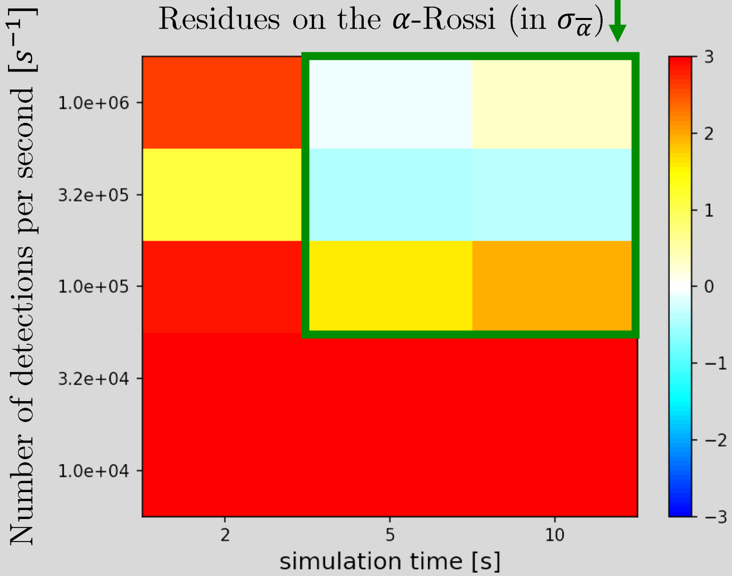


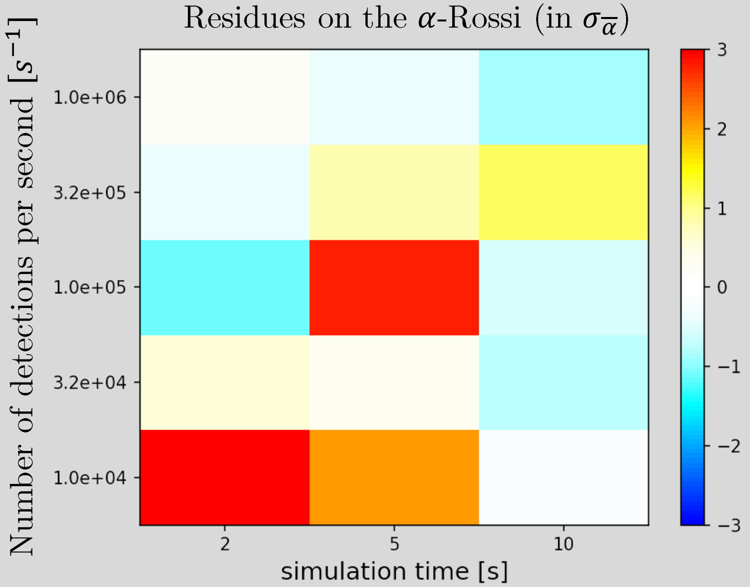


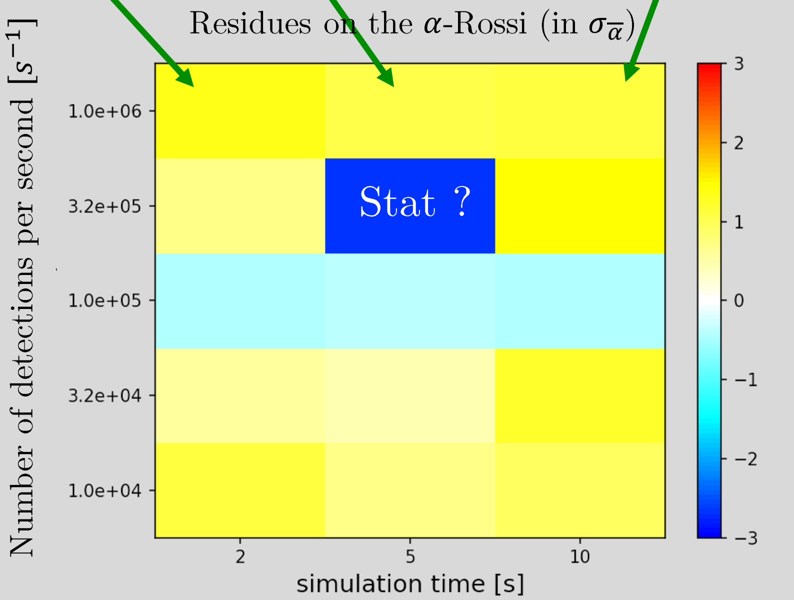


Requires higher count rates and time simulated

The results are correctly distributed







Residues $\left(\frac{|\overline{\alpha} - \alpha_{matrix}|}{\sigma(\overline{\alpha})}\right)$ with detector efficiency = 100%

Residues with detector efficiency = 10%

Residues with detector efficiency = 1%

Part 4:
Conclusions and prospects

Conclusions and prospects

- The physics of MSRs involves a strong coupling between thermohydraulics and neutronics
- A model to simulate neutronic noise using transient fission matrices is being tested to determine if the results are biased or not
- These studies are first carried out on a one-dimensional reactor and allow for intensive testing
- Further developments on the code are studied to improve its speed and to implement other analysis methods like Power Spectral Density (PSD)
- Studies about spatialisation on the CROCUS reactor of EPFL have been carried out by Axel Laureau and will be presented in a paper at the SNA + MC 2024 conference (Paris, end of October)