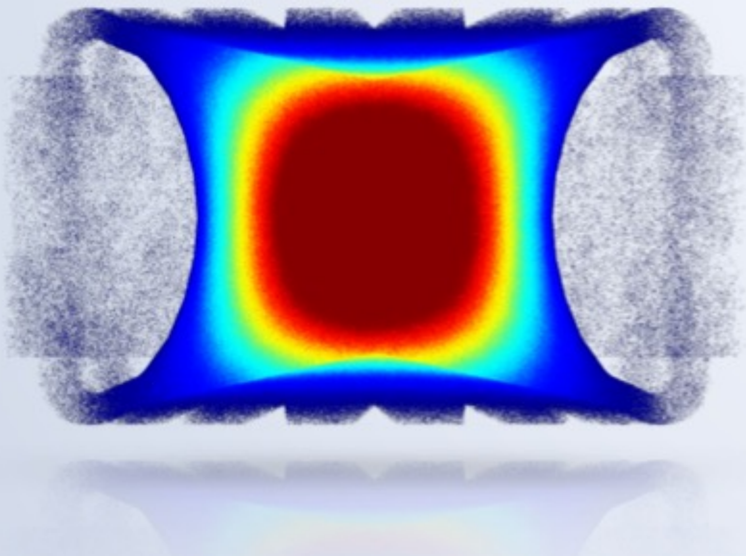
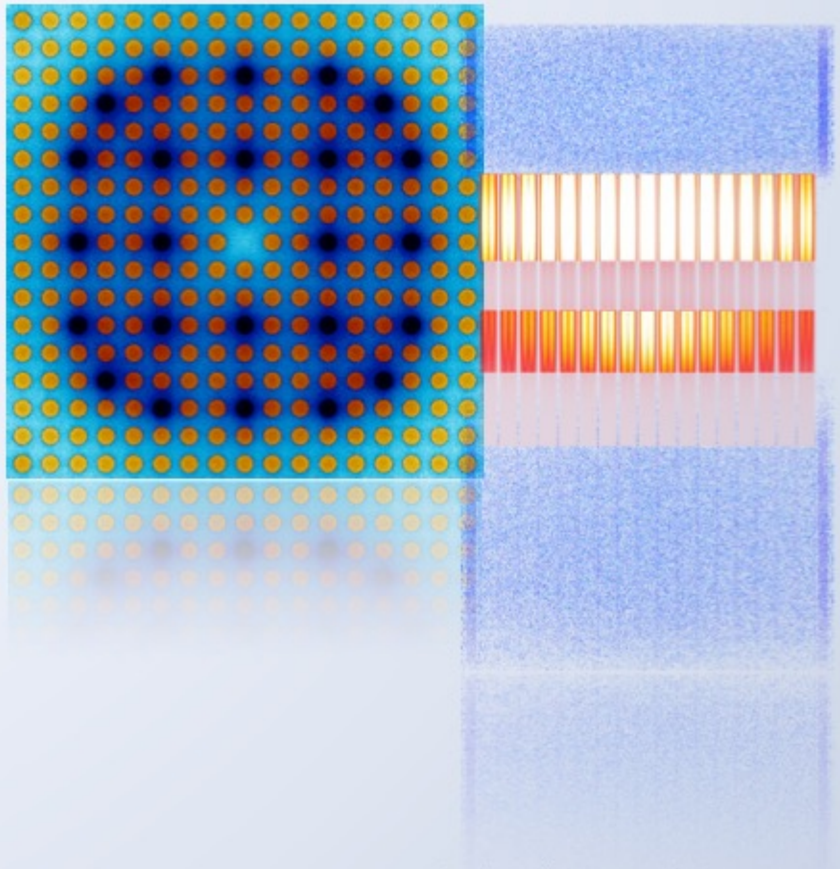
IPHC SEMINAR

SIMULATION TOOLS AND MODELS BASED ON HYBRID MONTE CARLO APPROACHES FOR SAFETY STUDIES AND REACTOR PHYSICS





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LPSC-IN2P3-CNRS,
Grenoble-INP

24 FÉVRIER 2017

OUTLINE

I. FRAME
   - NUCLEAR REACTORS

INTRODUCTION

II. PHYSICS PRESENTATION
   - THERMALHYDRAULICS
   - NEUTRONICS

TOOLBOX

III. PHYSICAL MODELS DEVELOPED
   - OBJECTIVES
   - NEUTRON SHOWER APPROACH
   - TRANSIENT FISSION MATRIX APPROACH
   - REACTOR SPECIFICITIES

DEVELOPMENT OF INNOVATIVE NEUTRONICS MODELS

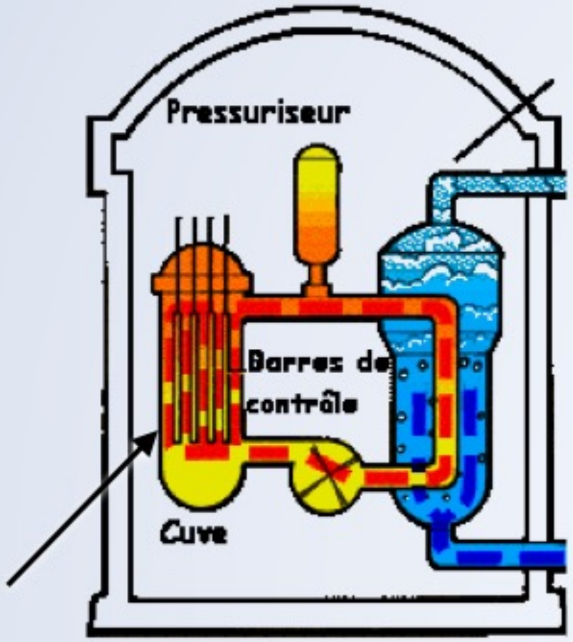
IV. COUPLING & SAFETY STUDY
   - GENERAL STRATEGY
   - STEADY STATE CHARACTERIZATION
   - MSFR SAFETY STUDIES
   - ASTRID ACCIDENTAL STUDY

SYSTEM STUDIES AND OPTIMIZATION

Different generations of reactors…

Generations II & III - present

Pressurized Water Reactor
- EPR - …

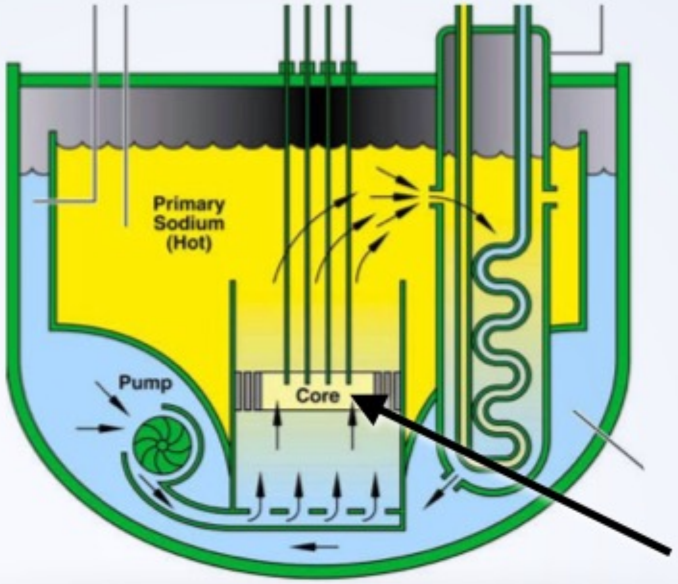


core

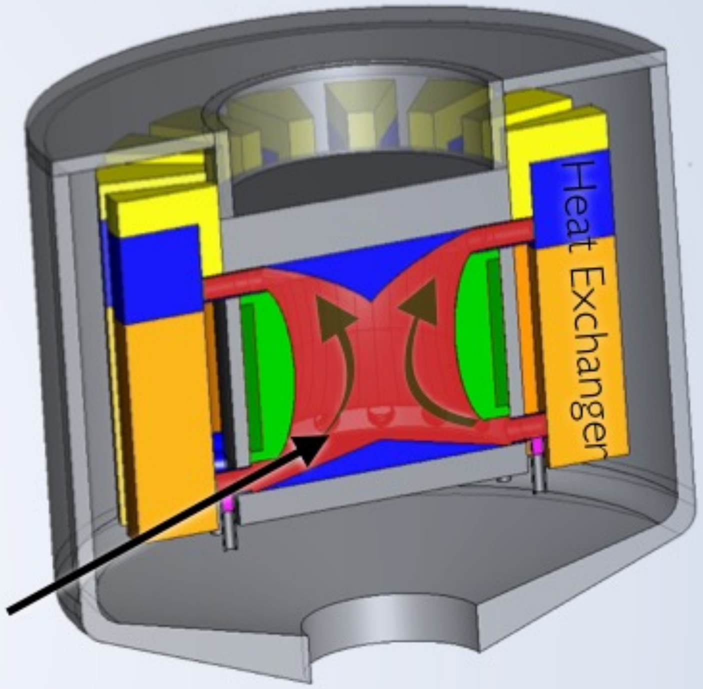
optimization of the concept

Generation IV - future

Sodium Fast Reactor:
ASTRID

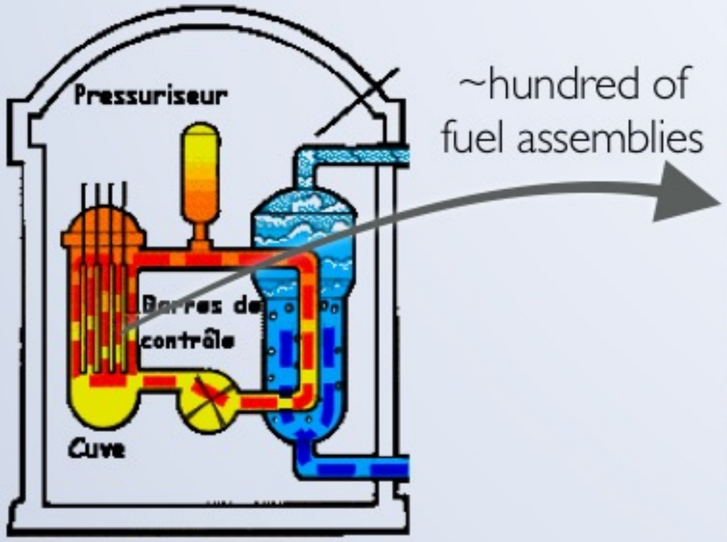


Molten Salt Reactor:
MSFR

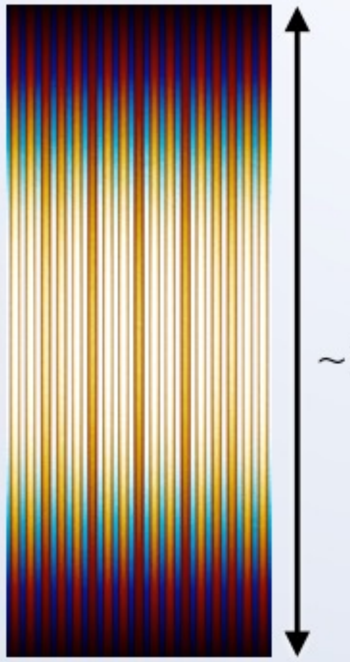


core

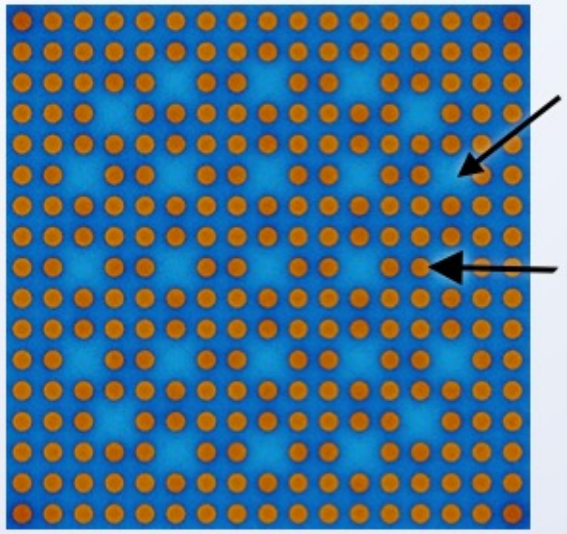
innovation



~hundred of fuel assemblies



Axial view



water (/control rod)

fuel pin

Radial view

Water acts as:

- **moderator**: neutrons “thermalized” by the hydrogen of H_2O
- **coolant**: transport the produced energy

NUCLEAR REACTORS

Different generations of reactors…

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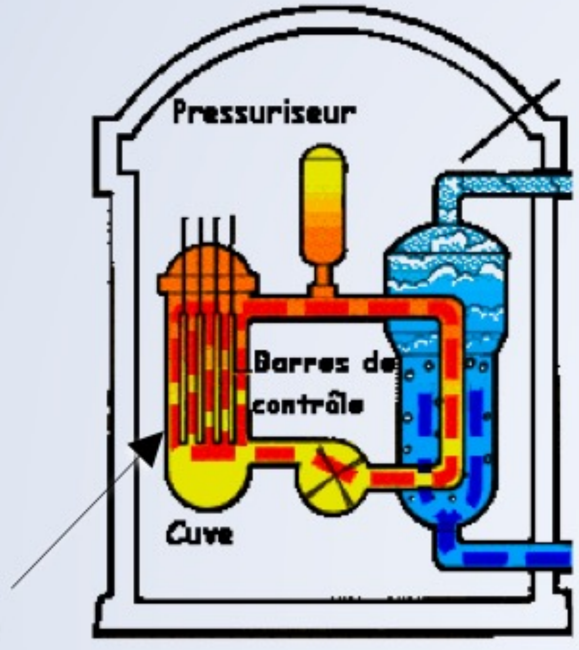
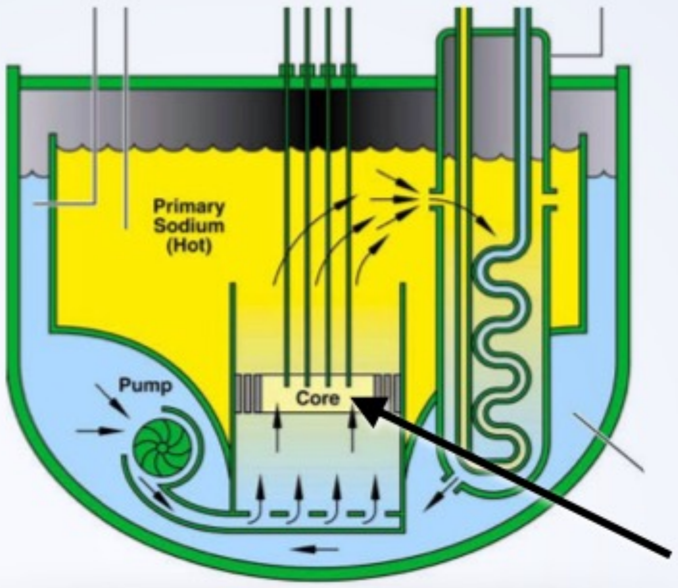
Pressurized Water Reactor
- EPR - …



optimization of the concept

Generation IV - future

Sodium Fast Reactor:
ASTRID



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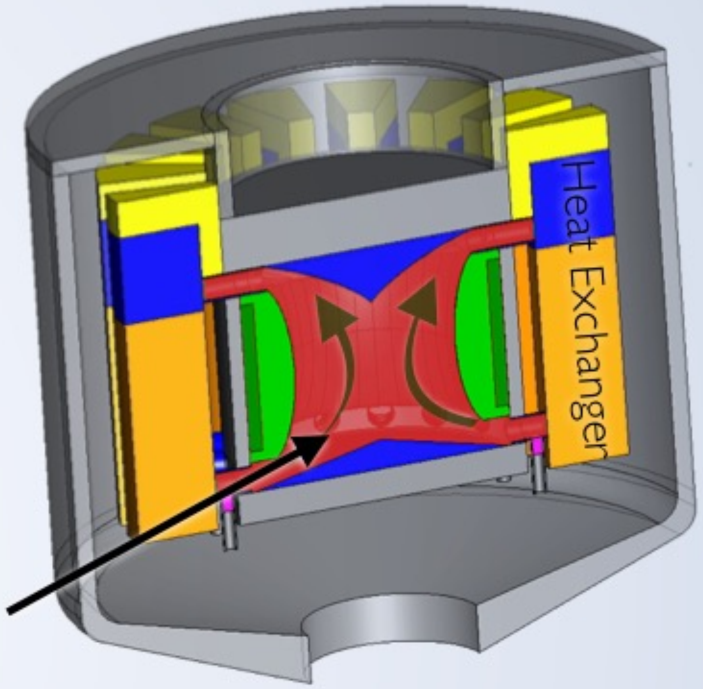
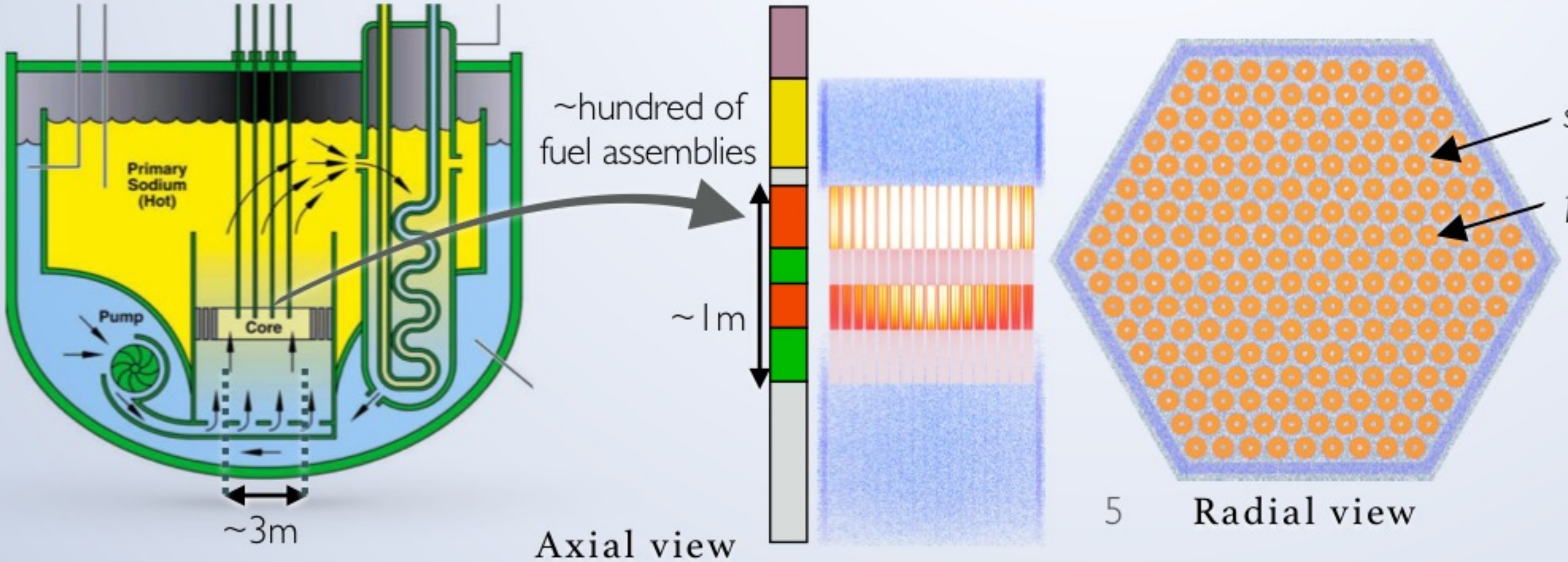
MSFR



core

innovation

ASTRID: Advanced Sodium Technological Reactor for Industrial Demonstration



Sodium is:

- **transparent** to neutrons
- the **coolant**: transport the produced energy

Different generations of reactors…

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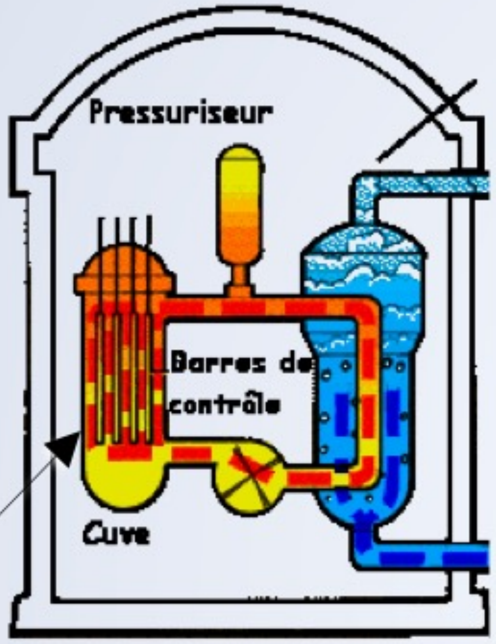
Pressurized Water Reactor
- EPR - …

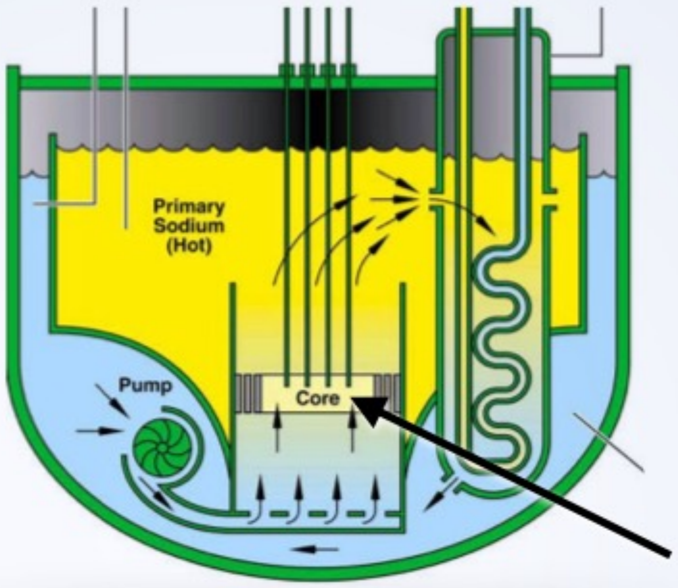


core

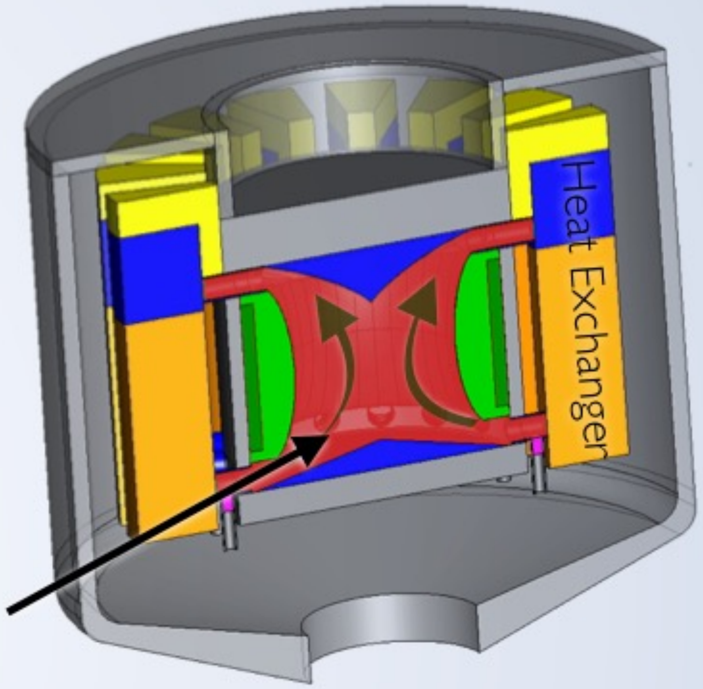
optimization of the concept

Generation IV - future

Sodium Fast Reactor:
ASTRID



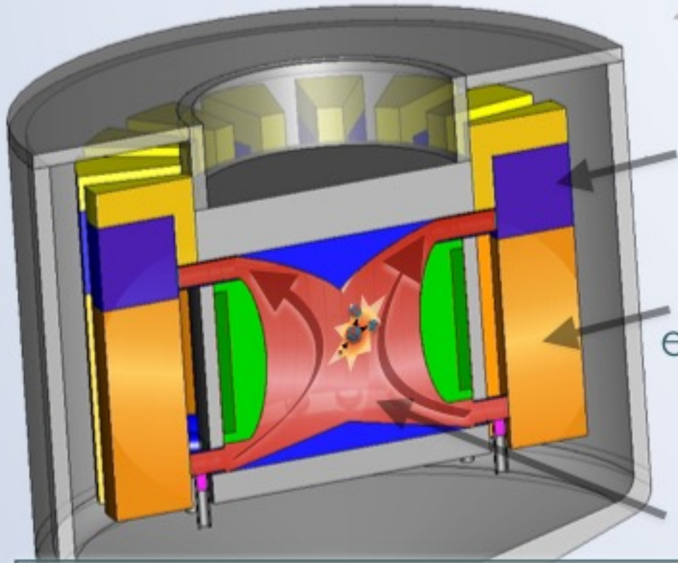
Molten Salt Reactor:
MSFR



core

innovation

MSFR: Molten Salt Fast Reactor

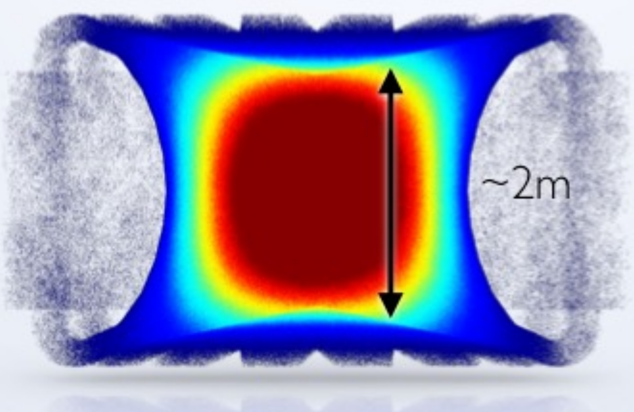


Pump

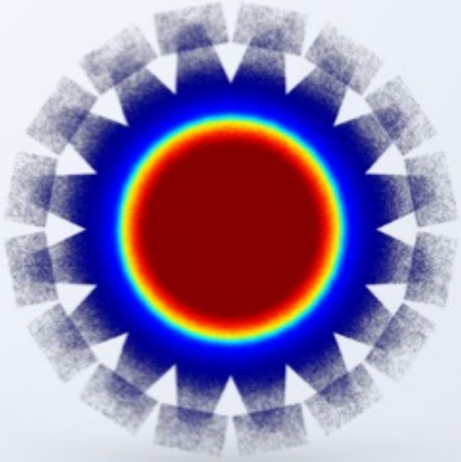
Heat exchanger

Fuel

Liquid circulating fuel, circulation time ~4 s, kinematic viscosity ~water



Axial view

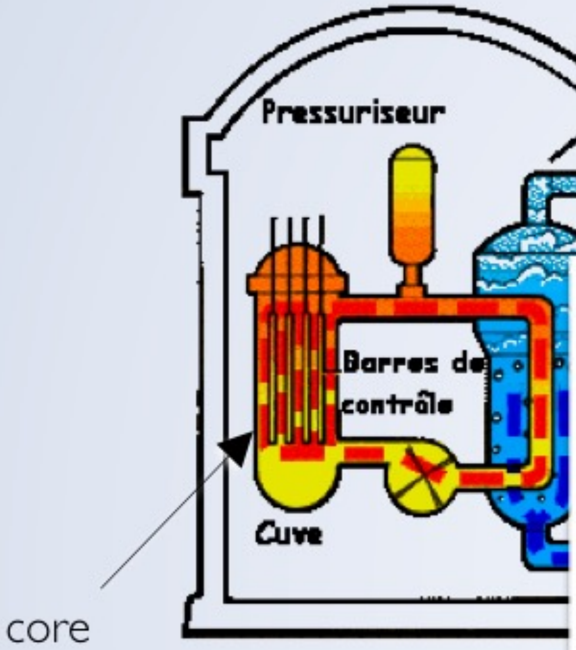


Radial view

Molten salt acts as:
- **fuel**: fissile matter in the salt
- **coolant**: transport the produced energy

Different generations of reactors…

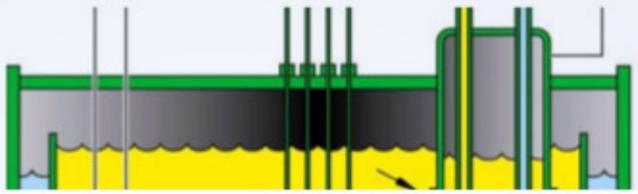
Generations II & III - present
Pressurized Water Reactor
- EPR - …



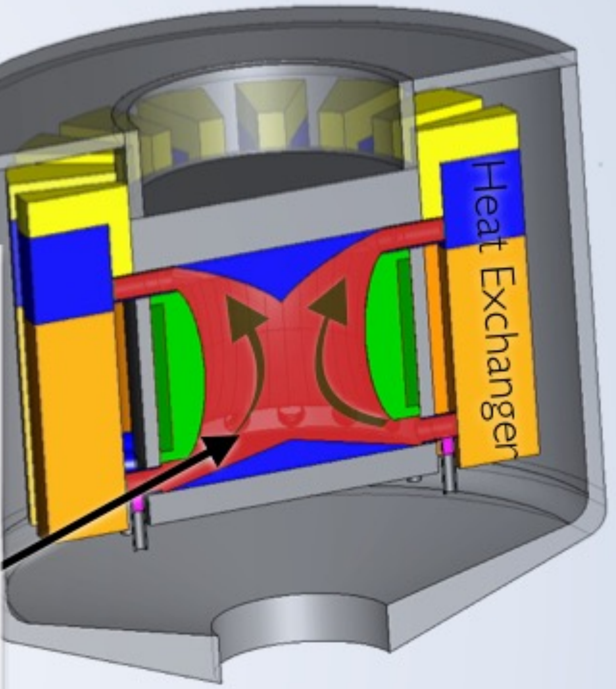
optimization of concept

Generation IV - future

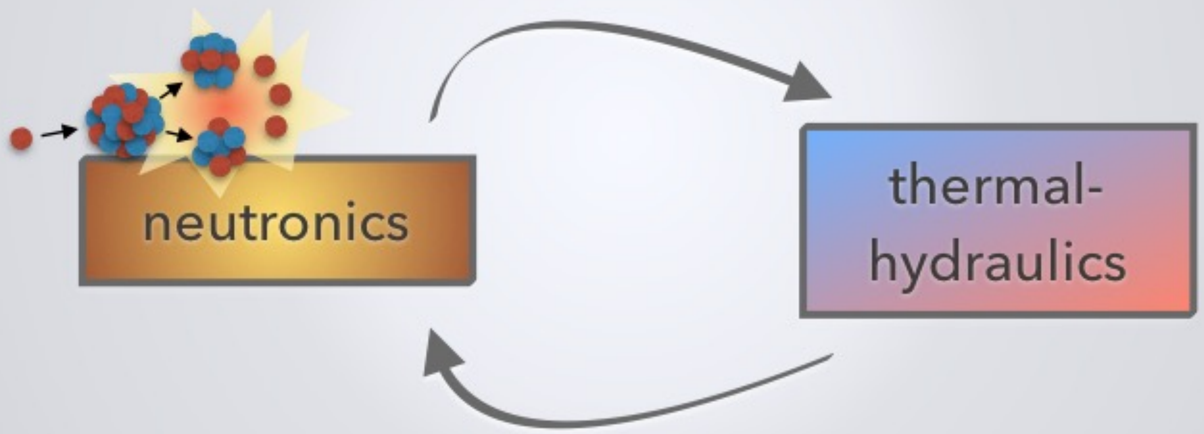
Sodium Fast Reactor:
ASTRID



Molten Salt Reactor:
MSFR



Understanding the core physics
requires multi-physics tools



Different physics
- Neutron energy
  ~25meV (thermal) up to ~3MeV (fast)
- Fuel
  composition ^{233}U, ^{235}U, ^{239}Pu
  materials water, sodium, salt
  liquid/solid

different physical approaches to study these systems

Research objectives:
- Safety
- Power variation: driveability
  electric mix, renewables
- Sustainability
  fuel recycling
  waste management

Outline

I. Frame
   - Nuclear reactors

Introduction

II. Physics presentation
   - Thermalhydraulics
   - Neutronics

Toolbox

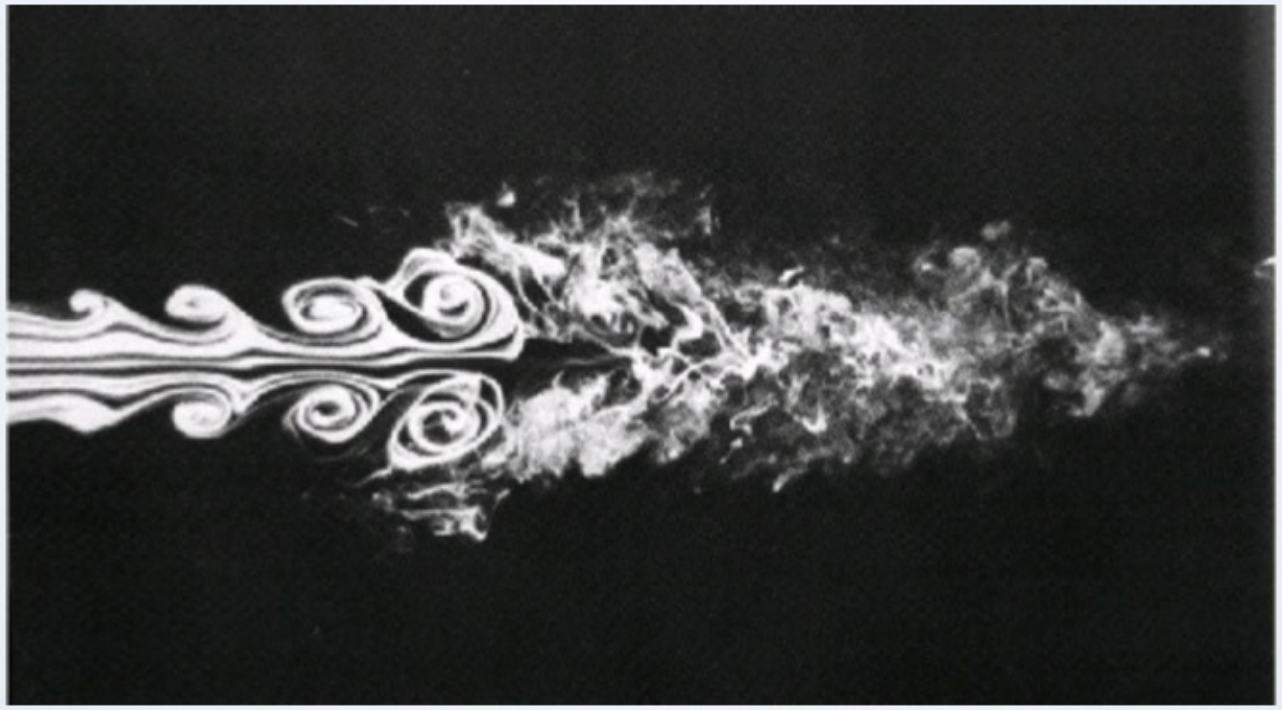
III. Physical models developed
   - Objectives
   - Neutron shower approach
   - Transient Fission Matrix approach
   - Reactor specificities

Development of innovative neutronics models

IV. Coupling & safety study
   - General strategy
   - Steady state characterization
   - MSFR safety studies
   - ASTRID accidental study

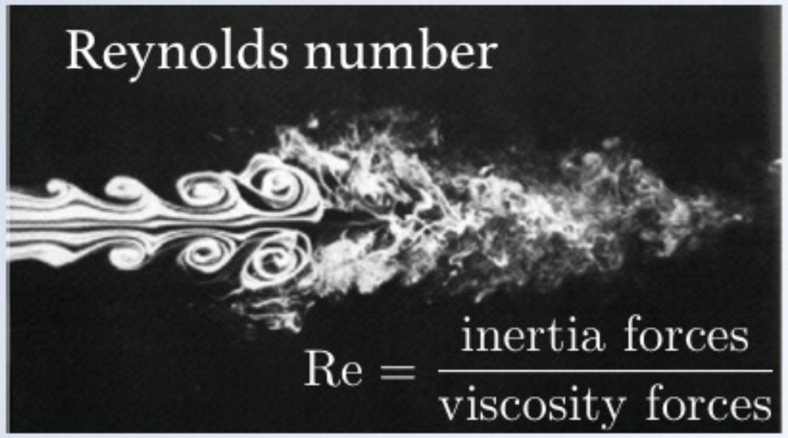
System studies and optimization

TOOLBOX PART I:
THERMALHYDRAULICS



Equations

Discipline based on 3 conservation equations: mass, momentum, energy



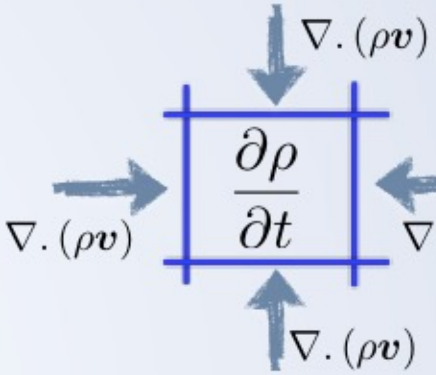
Reynolds number

$$\mathrm{Re} = \frac{\text{inertia forces}}{\text{viscosity forces}}$$

density

velocity

$$\frac{\partial\rho}{\partial t} + \nabla.\left(\rho\boldsymbol{v}\right) = 0$$



no matter created

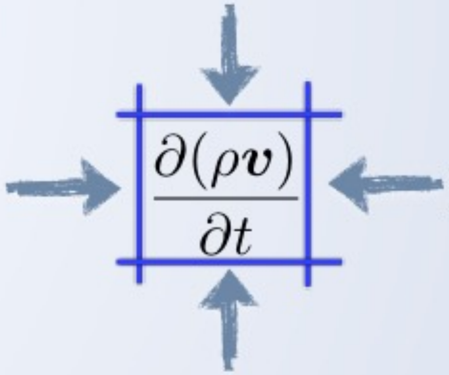
quantity of motion

constraint tensor

$$\frac{\partial(\rho\boldsymbol{v})}{\partial t} + \nabla.\left(\rho\boldsymbol{v}\otimes\boldsymbol{v}\right) = \nabla.\left(\underline{\underline{\sigma}}\right) + \rho\boldsymbol{f}_{\mathbf{ext}}$$

highly non-linear

external forces



$\Sigma\boldsymbol{F} = m\boldsymbol{a}$

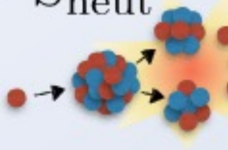
re-written for a fluid

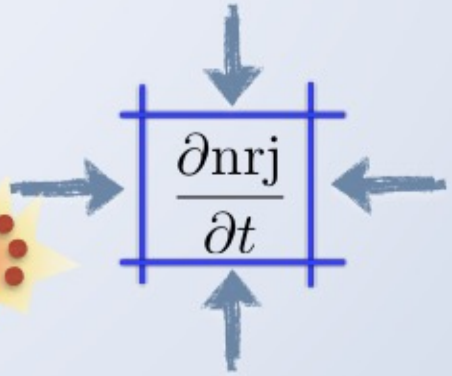
internal and kinetic energy

work done by forces

$$\frac{\partial\left[\rho\left(e+\frac{\boldsymbol{v}^2}{2}\right)\right]}{\partial t} + \nabla.\left(\rho\left(e+\frac{\boldsymbol{v}^2}{2}\right)\boldsymbol{v}\right) = \nabla.\left(\boldsymbol{v}.\underline{\underline{\sigma}}\right) + \rho\boldsymbol{f}_{\mathbf{ext}}.\boldsymbol{v} - \nabla.\left(\boldsymbol{q}\right) + \mathrm{S}_{\mathrm{neut}}$$

heat transfer



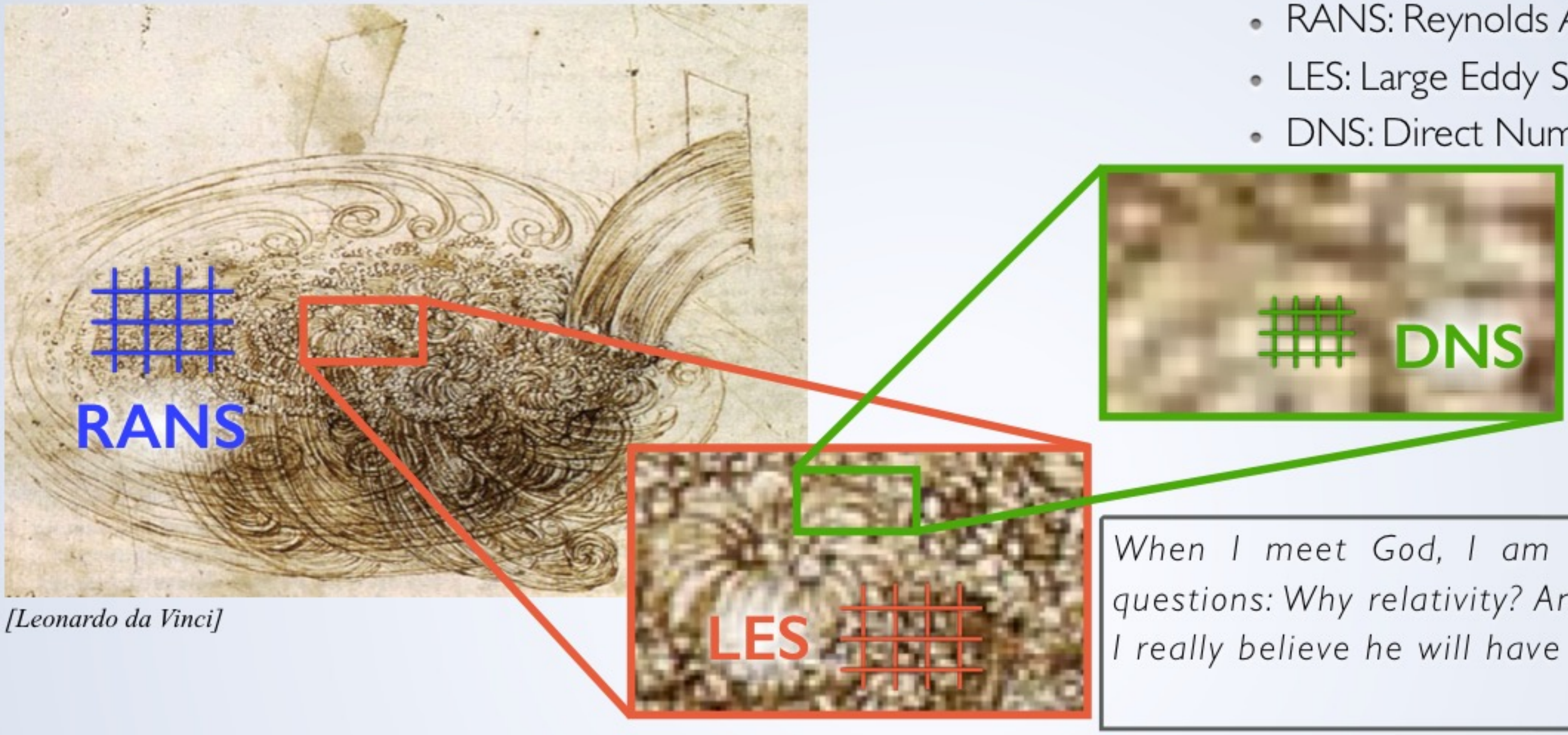


work done by forces

Resolution method

Familly code: CFD - Computational Fluid Dynamics

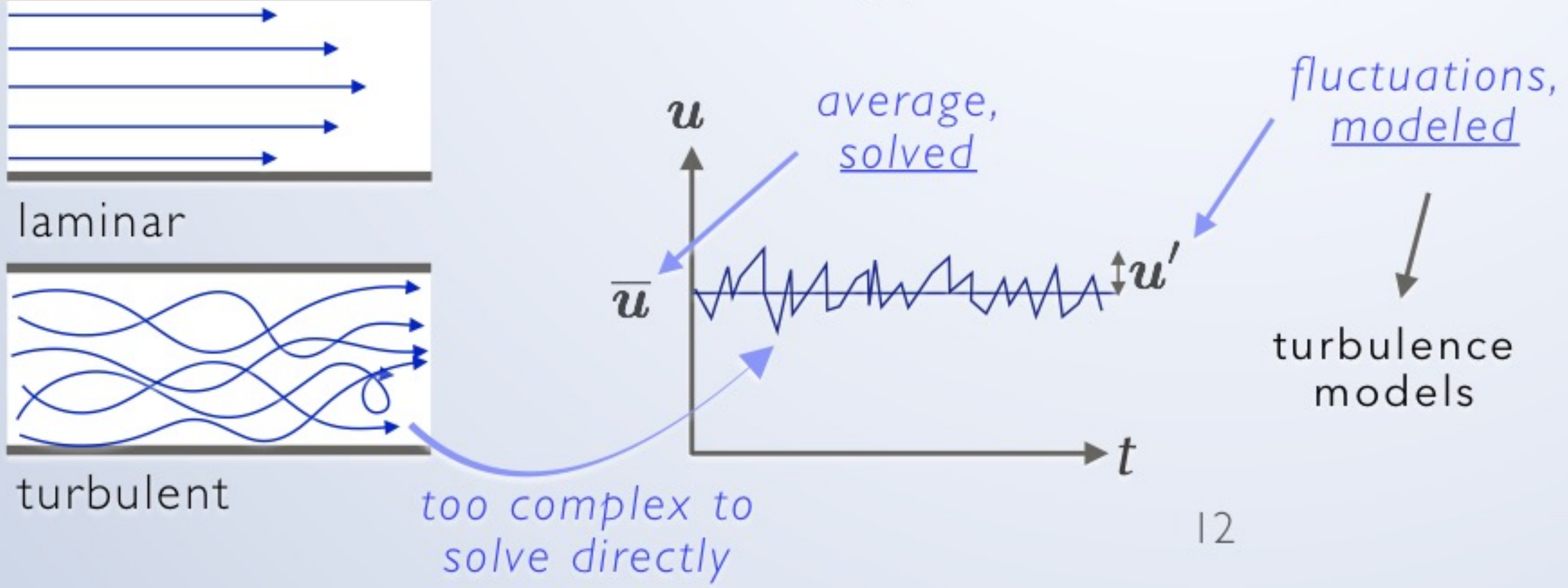
- RANS: Reynolds Average Navier Stokes
- LES: Large Eddy Simulation
- DNS: Direct Numerical Simulation



[Leonardo da Vinci]

> *When I meet God, I am going to ask him two questions: Why relativity? And why turbulence? I really believe he will have an answer for the first.*
>
> Werner Heisenberg

RANS approach



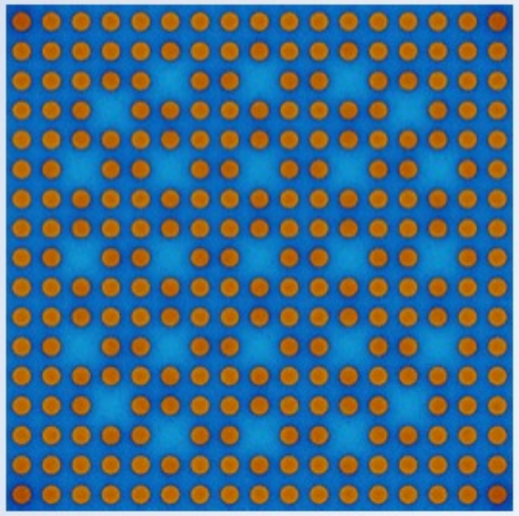
Calculation code used:

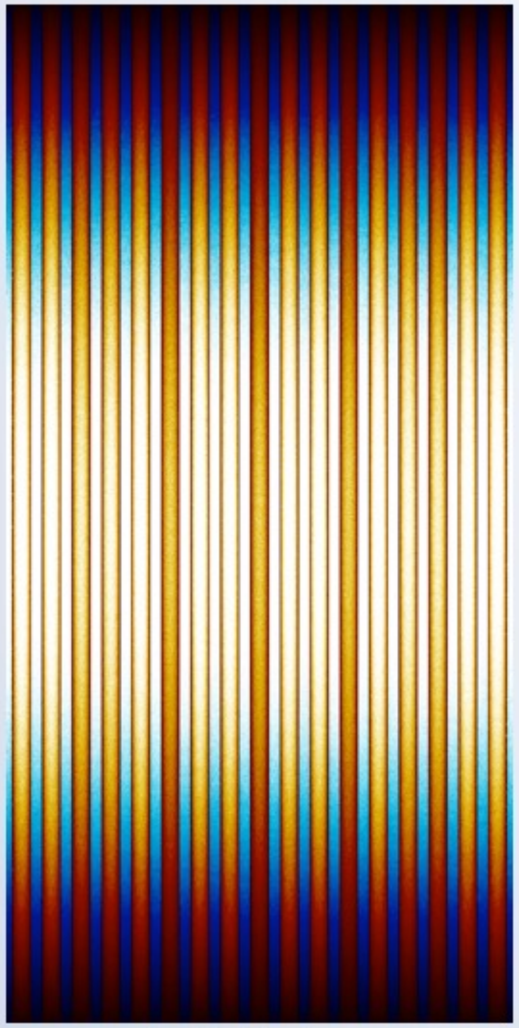
OpenFOAM

- finite volume library in C++
- opensource toolbox
- many models available
- multi-physics

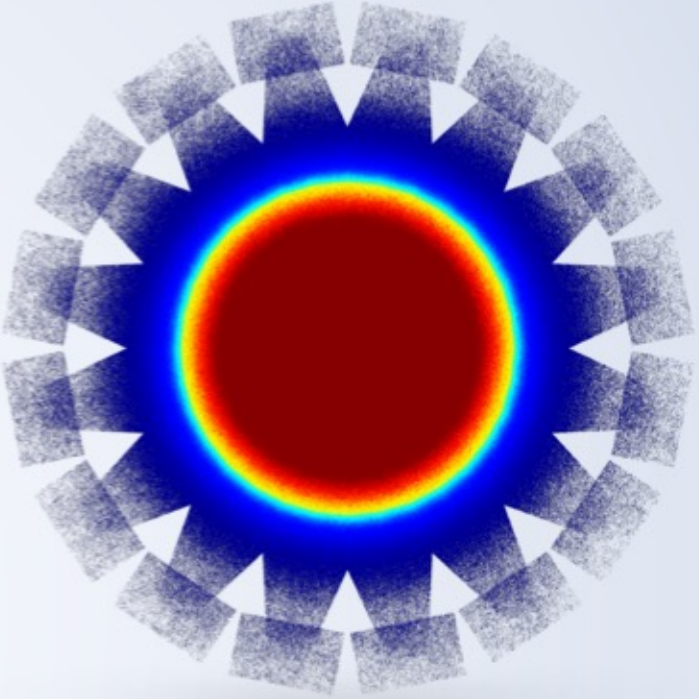
TOOLBOX PART II : NEUTRONICS

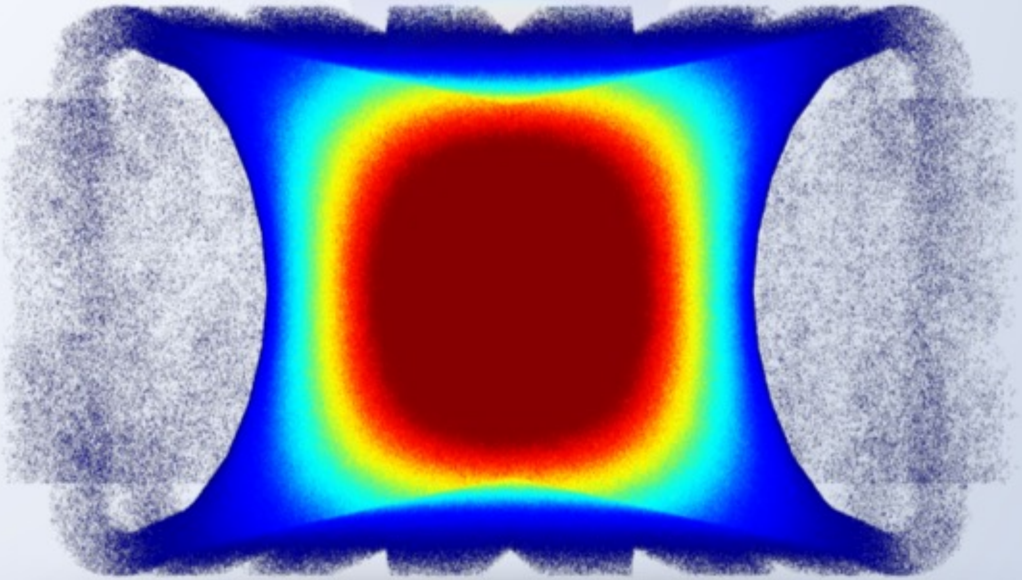
Pressurized Water Reactor


Radial view

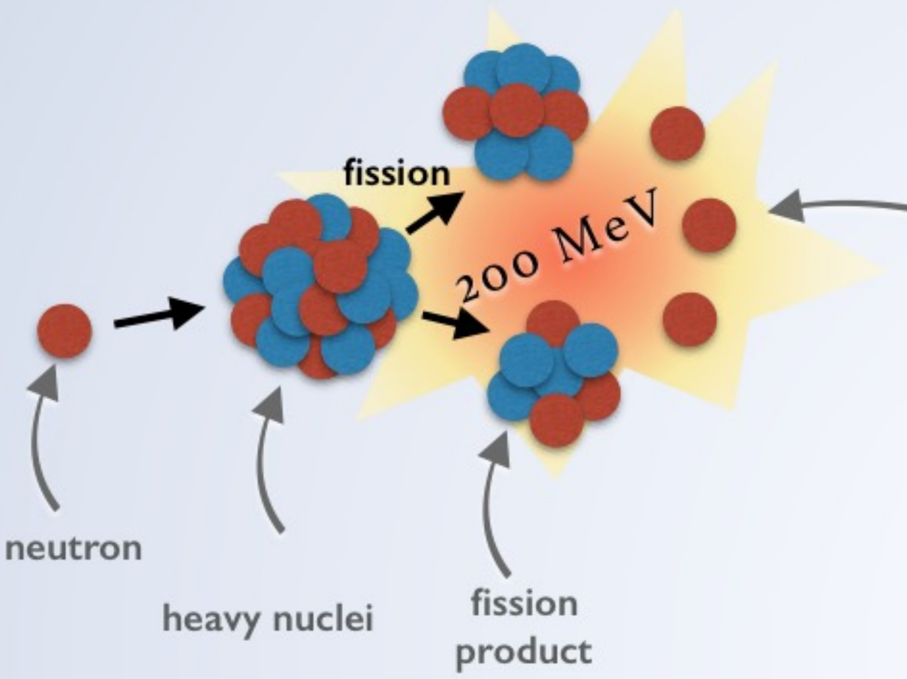

Axial view

Molten Salt Fast Reactor

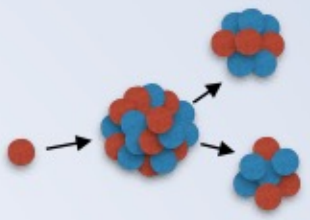

Radial view


Axial view

Chain reaction



$\nu \simeq 2.5$ neutrons



- 1 for the chain reaction
- 1 for fuel breeding
- 0.5 lost

$^{232}\text{Th} + ● \rightarrow {}^{233}\text{U}$

$^{238}\text{U} + ● \rightarrow {}^{239}\text{Pu}$

« fertile » « fissile »

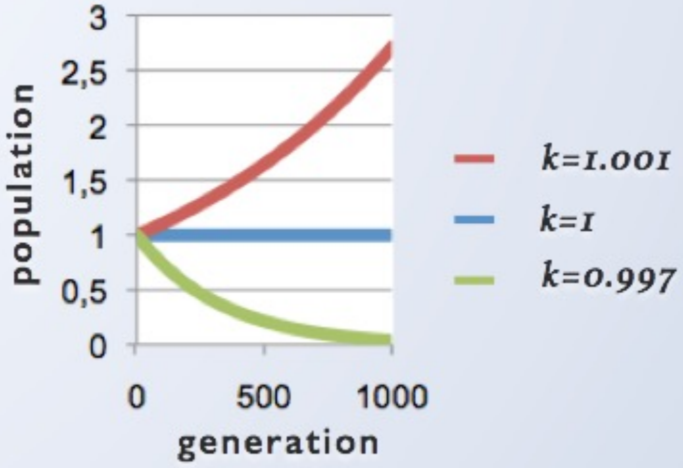
multiplication factor:

$$k = \frac{\textbf{number of neutrons at generation n+1}}{\textbf{number of neutrons at generation n}}$$

reactivity :

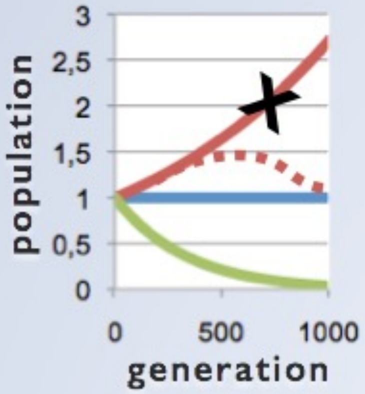
$$\rho = \frac{k-1}{k}$$ in pcm (10^{-5})

"critical" : ρ=0



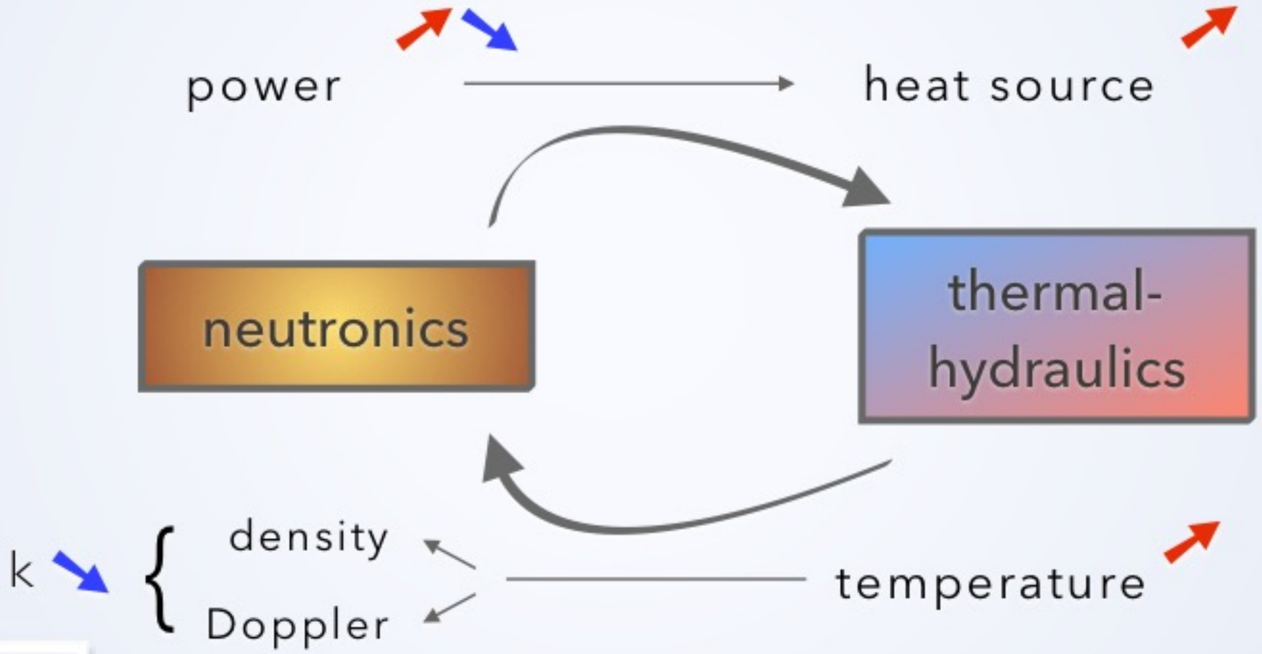
"critical" : k=1

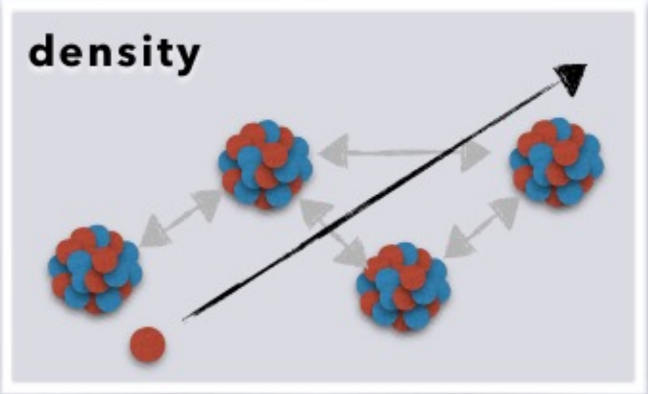
Feedback - chain reaction stabilization

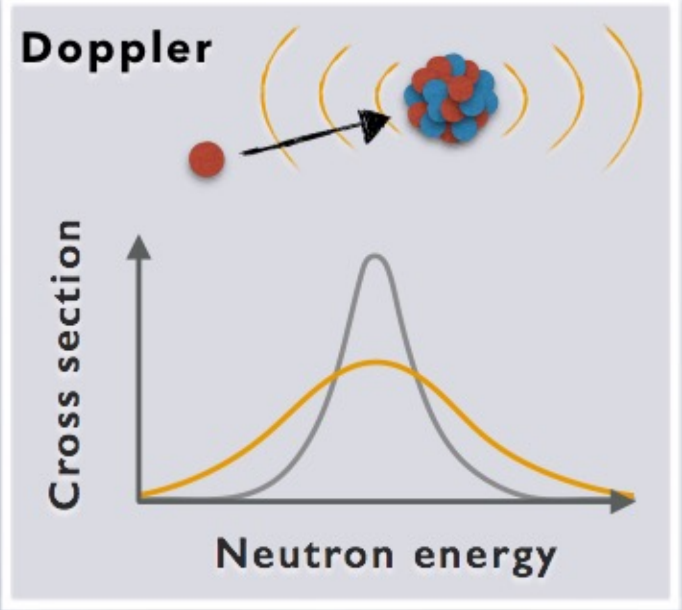


In order to maintain the chain reaction, it is important to make it self-stabilizing

A modification of the environment impacts the neutron behavior:





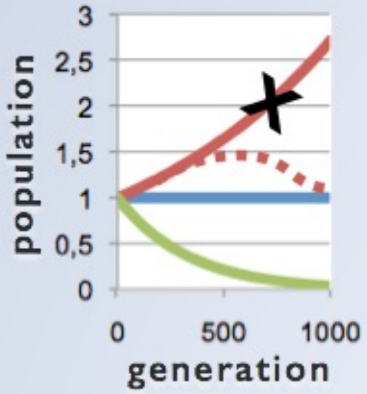


During the reactor conception, a variation of *k* negatively correlated to the temperature **T** stabilises the system

$$\text{MSFR:} \quad \left(\frac{\mathrm{dk}}{\mathrm{dT}}\right)_{\text{total}} \sim -8 \ \mathrm{pcm/K}$$

NEUTRONICS

Delayed neutrons



For a reactor with a neutron mean lifetime of ~ 10 μs

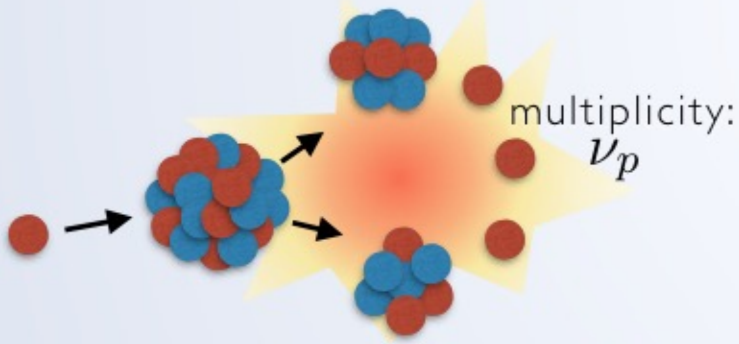
if $\rho = \dfrac{k-1}{k} = 10\ pcm\ (0.01\%)$, during 1 second the neutron population is multiplied by 22000

Fraction of delayed neutrons: β

$\nu \simeq 2.5$ produced neutrons

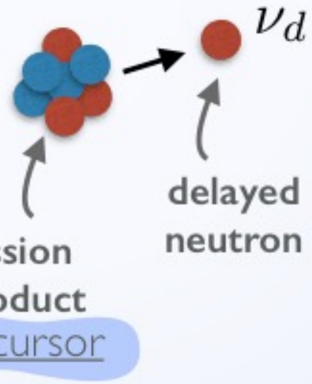
prompt

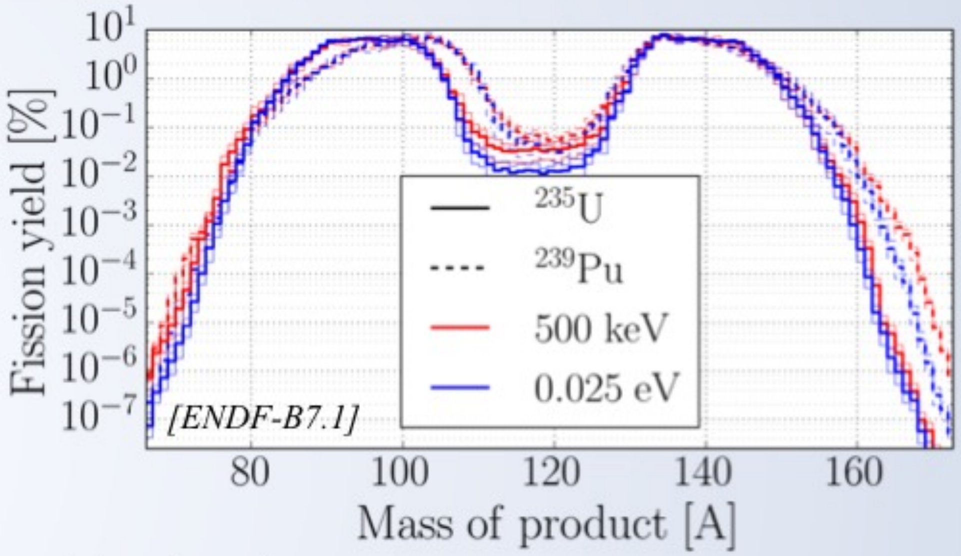
immediately



delayed

after ~ 1s





delayed neutron precursor families

$T_{1/2}$ from 0.1s to 1min

❗ linked to the fuel ❗

finally: $k = k_p + \beta k$ ($k_p \rightarrow \mu s$, $\beta k \rightarrow s$)

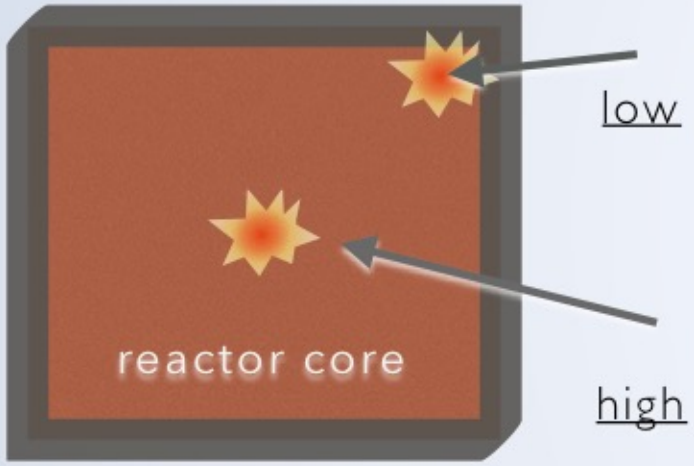
$$\beta = \frac{\nu_d}{\nu} \simeq 0.3 \text{ up to } 0.7\%$$

during a transient... $k_p - 1$
- → *used to characterize the reactor*
- → *margin to prompt criticality*
- → *violent behavior if* **>0** *- prompt critical*

Neutron importance

$k = 1 = k_p + \beta$ **... But... all the neutrons are not equivalent!**

- Depending on the emission postion, the effect is different

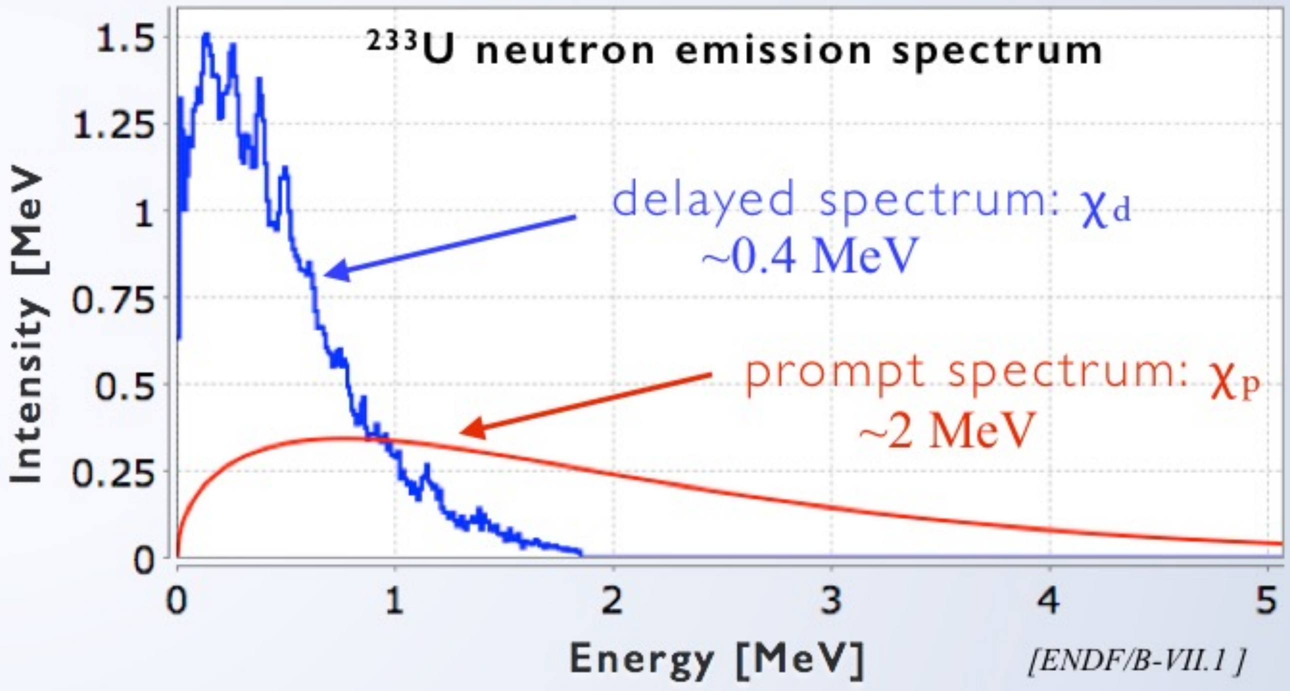


peripheral emission: low probability to produce a new fission

centered emission : high probability to produce a new fission

→ areas with different *importances*

- Different emission energies



[ENDF/B-VII.1]

we define β_{eff}, the *effective* fraction of delayed neutrons, that takes into account these effects

At equilibrium: $\cancel{1 = k_p + \beta} \longrightarrow 1 = k_p + \beta_{eff}$

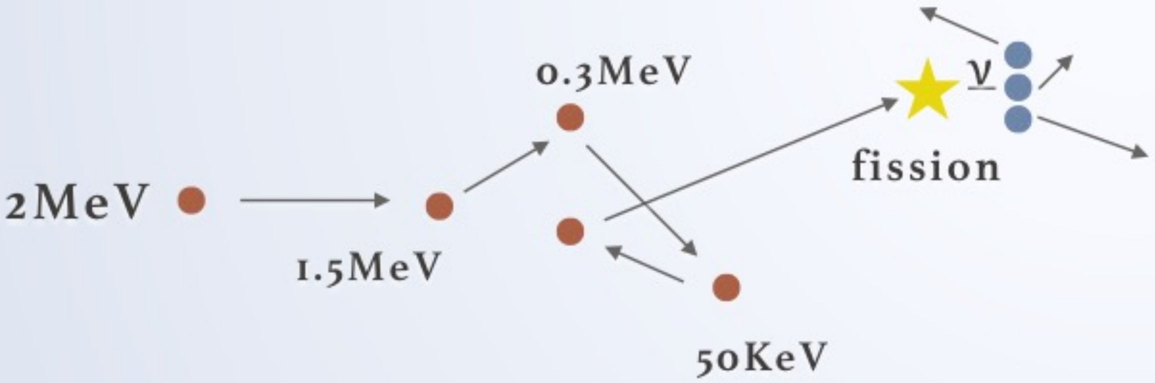
Resolution method

- Deterministic approach:
  Neutron behavior modeled by a transport equation (Boltzmann) on the angular neutron density: $\varphi(x,y,z,E,\Omega)$ in neutrons/cm²/s/eV/sr

fast / numerical biases (discretisation & shema)

- Stochastic approach:
  Monte Carlo: track punctual particles (position, energy, angle), « Ballistic » transport with associate laws (cross sections)

slow / reference result



calculation codes used:

MCNP	SERPENT
• reference	• recent
• well known	• sources available

a large number of neutrons is simulated to obtain a general behavior and then estimate the neutron flux

our issue: the delayed neutron precursors may be moving...

What is the impact on the safety? Codes not adapted

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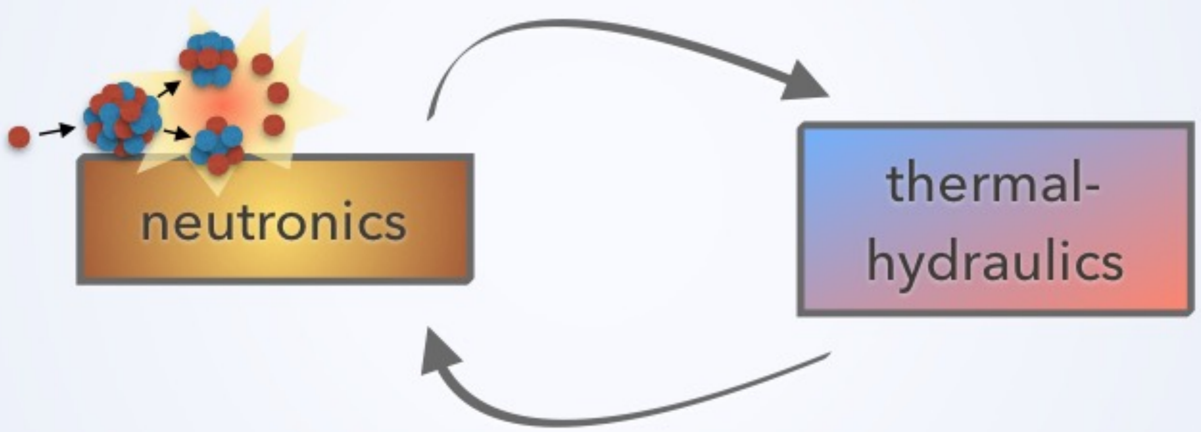
SYSTEM STUDIES AND OPTIMIZATION

PHYSICAL MODELS DEVELOPED

Need to take into account different physics interacting together!
Numerical core thematics



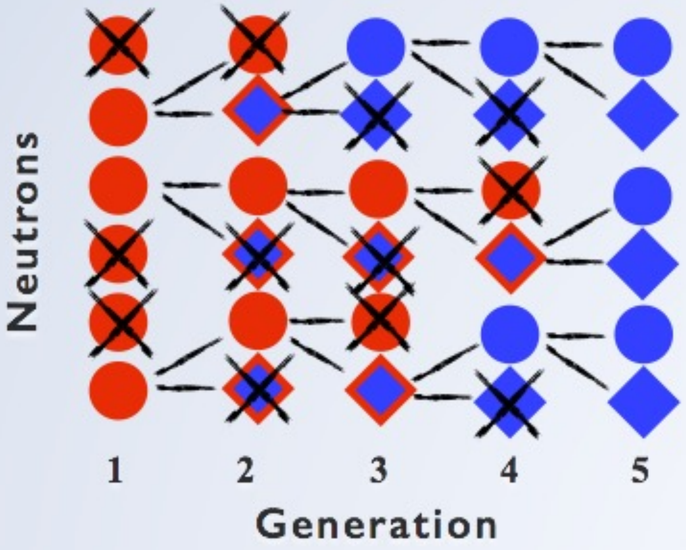
- neutronics - thermalhydaulics coupling
- adaptable to various nuclear systems
- accurate ...
- ... with a reasonable calculation time (~day)

need for new neutronics model adapted to this objective:

hybrid approach
Monte Carlo / deterministic

NEUTRON SHOWER APPROACH

How can we obtain a correct representation of the neutron flux?



● prompt neutron

◆ delayed neutron

prompt shower

shower induced by delayed neutrons

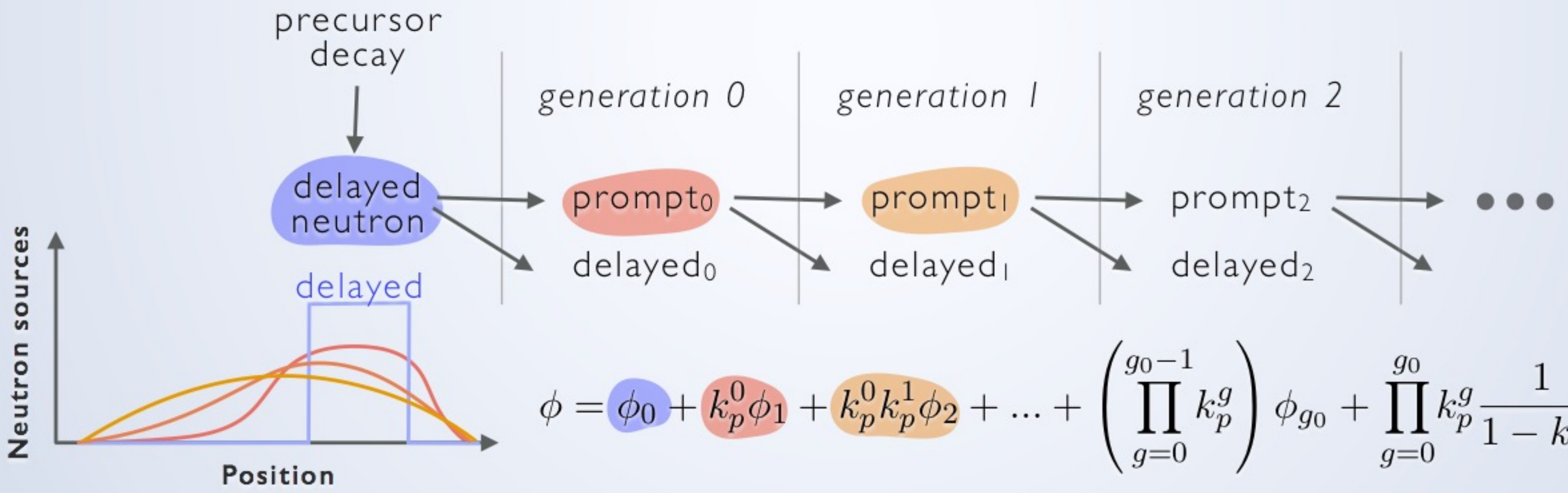
$\boxed{k=1}$

k_p=0.5 et β=0.5

not realistic / didactic case

In a critical reactor, all the neutrons come from a delayed neutron

Neutron shower approach:



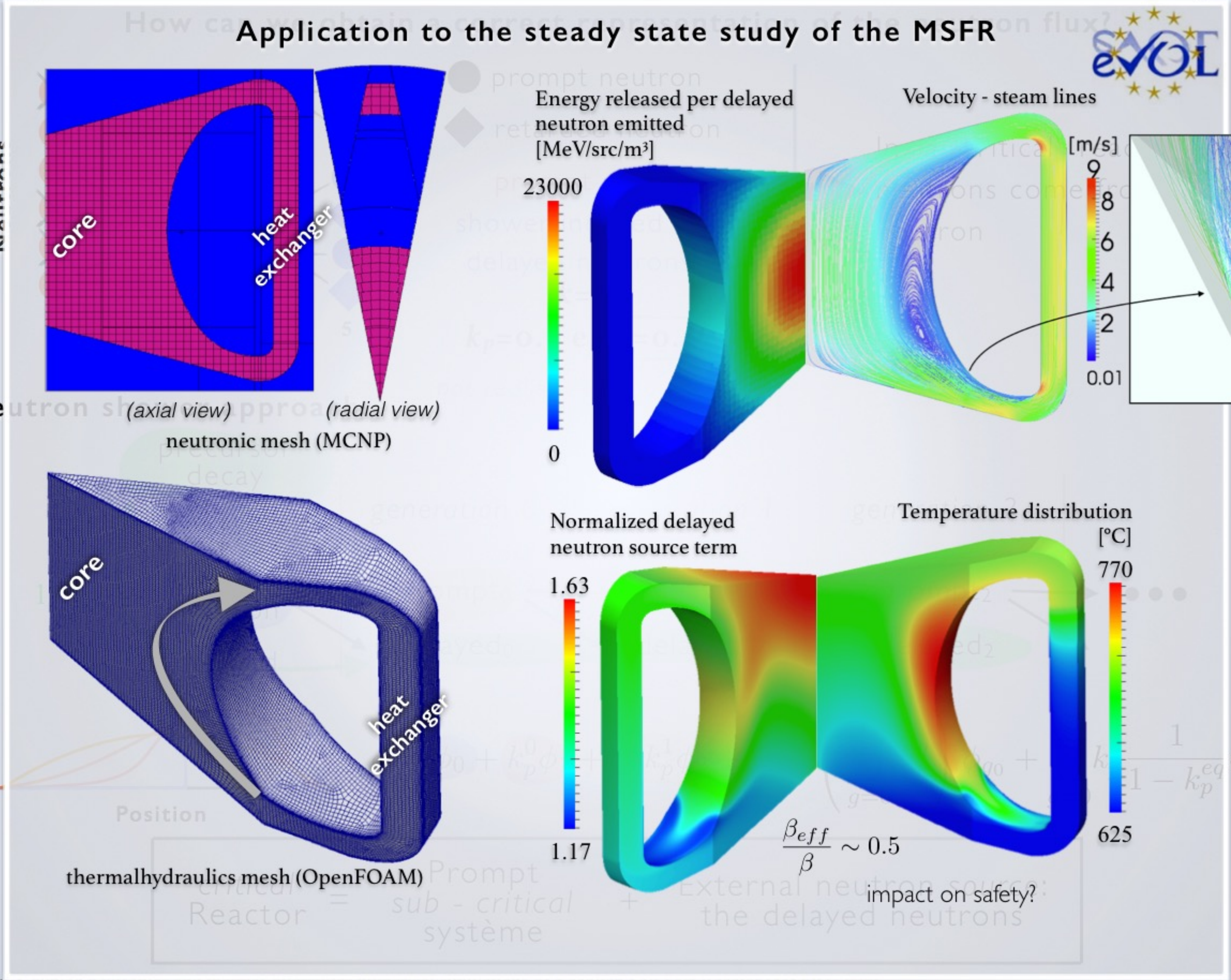
$$\phi = \phi_0 + k_p^0\phi_1 + k_p^0 k_p^1\phi_2 + \ldots + \left(\prod_{g=0}^{g_0-1} k_p^g\right)\phi_{g_0} + \prod_{g=0}^{g_0} k_p^g \frac{1}{1-k_p^{eq}}\phi_{eq}$$

critical Reactor = Prompt *sub - critical* system + External neutron *source*: the delayed neutrons

[A. Laureau *et al.* Coupled neutronics and thermal-hydraulics numerical simulations of a Molten Salt Fast Reactor (MSFR). *Joint International Conference on Supercomputing in Nuclear Applications+ Monte Carlo (SNA & MC 2013), Paris, France, 2013*]

NEUTRON SHOWER APPROACH

Coupling to the termalhydaulics



Application to the steady state study of the MSFR

$\frac{\beta_{eff}}{\beta} \sim 0.5$

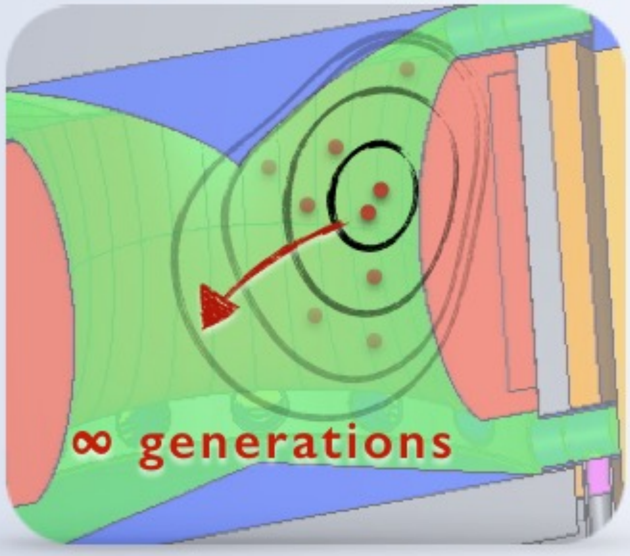
impact on safety?

[A. Laureau et al. Coupled neutronics and thermal-hydraulics numerical simulations of a Molten Salt Fast Reactor (MSFR). Joint International Conference on Supercomputing in Nuclear Applications+ Monte Carlo (SNA & MC 2013), Paris, France, 2013]

Transient Fission Matrix approach

What about transient (*time dependent*) study?

Neutron shower



global propagation of the neutrons

accurate image ot the whole reactor at steady state:
physical study of the design / core optimisation

Transient Fission Matrix (TFM)

Green function: spatial & temporal transport

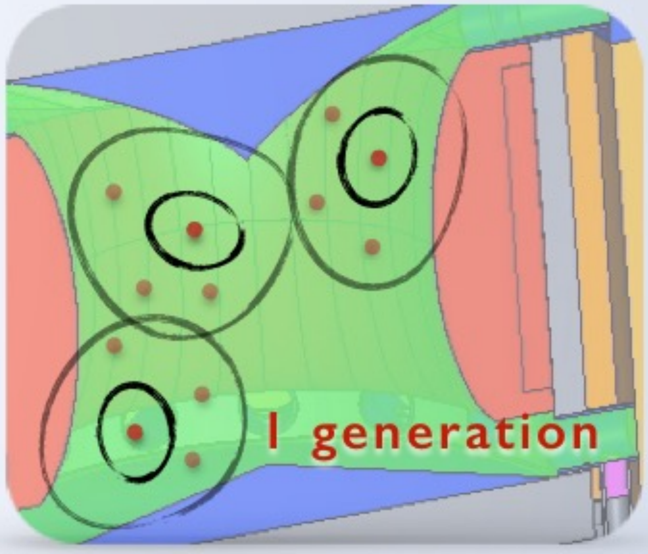
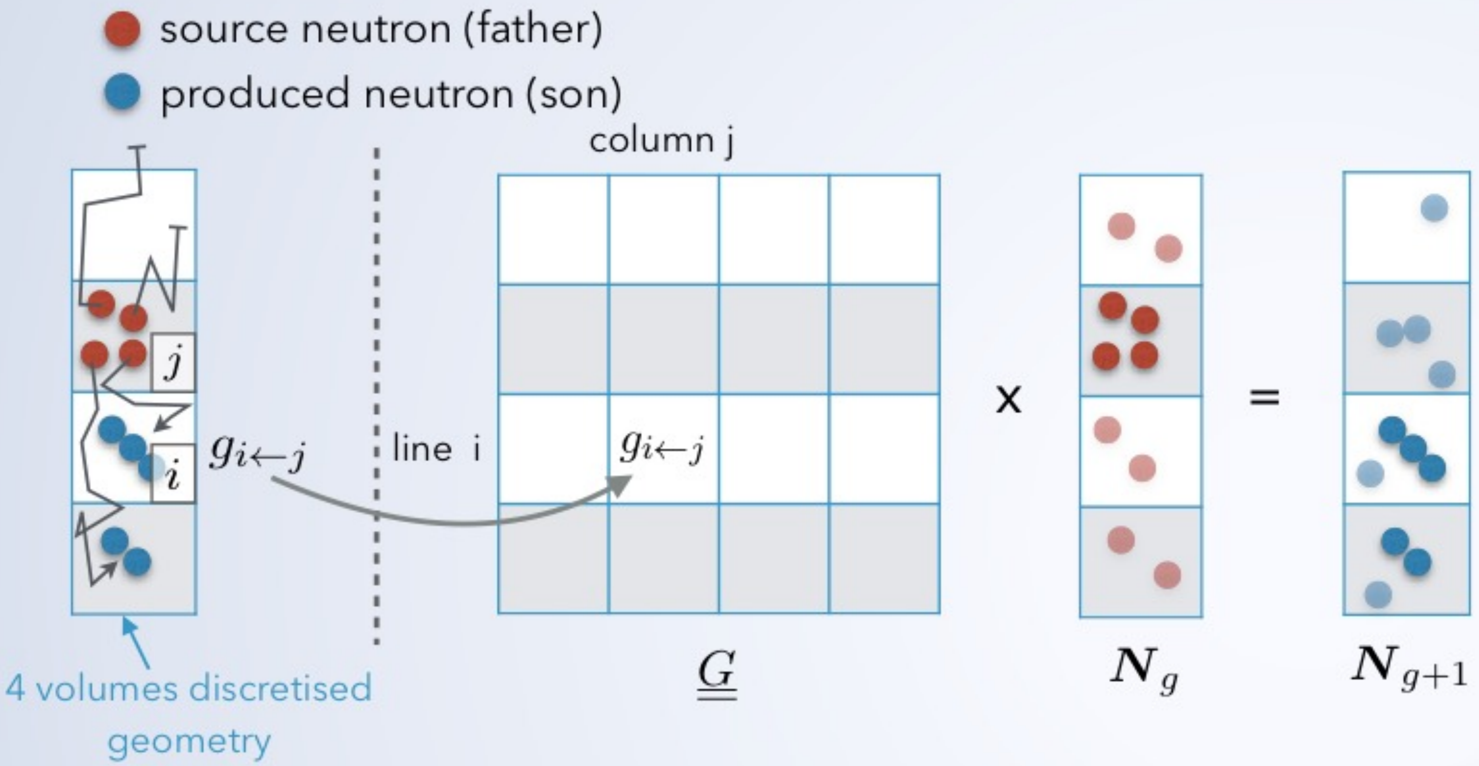
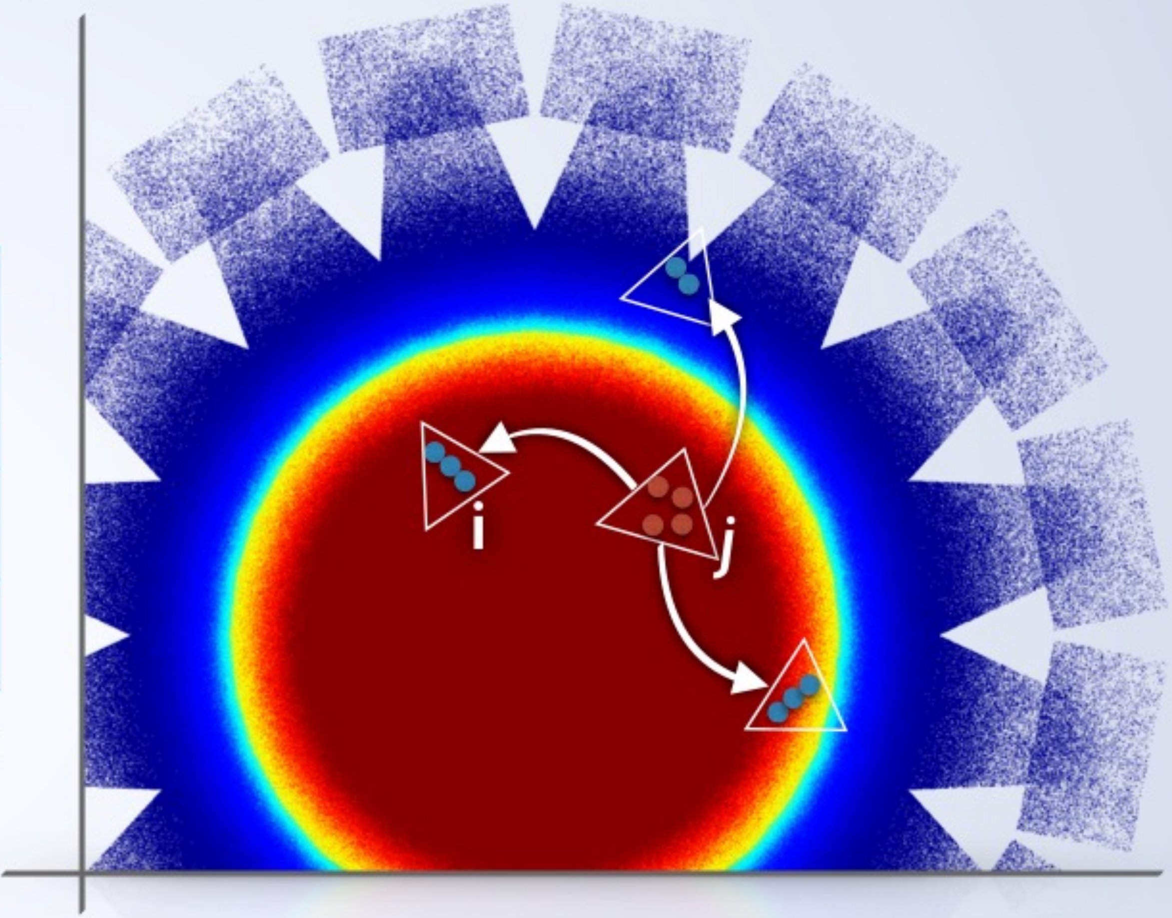


image of the local neutron ***tranport****:*
transient study / reactor behavior
(normal and accidental)

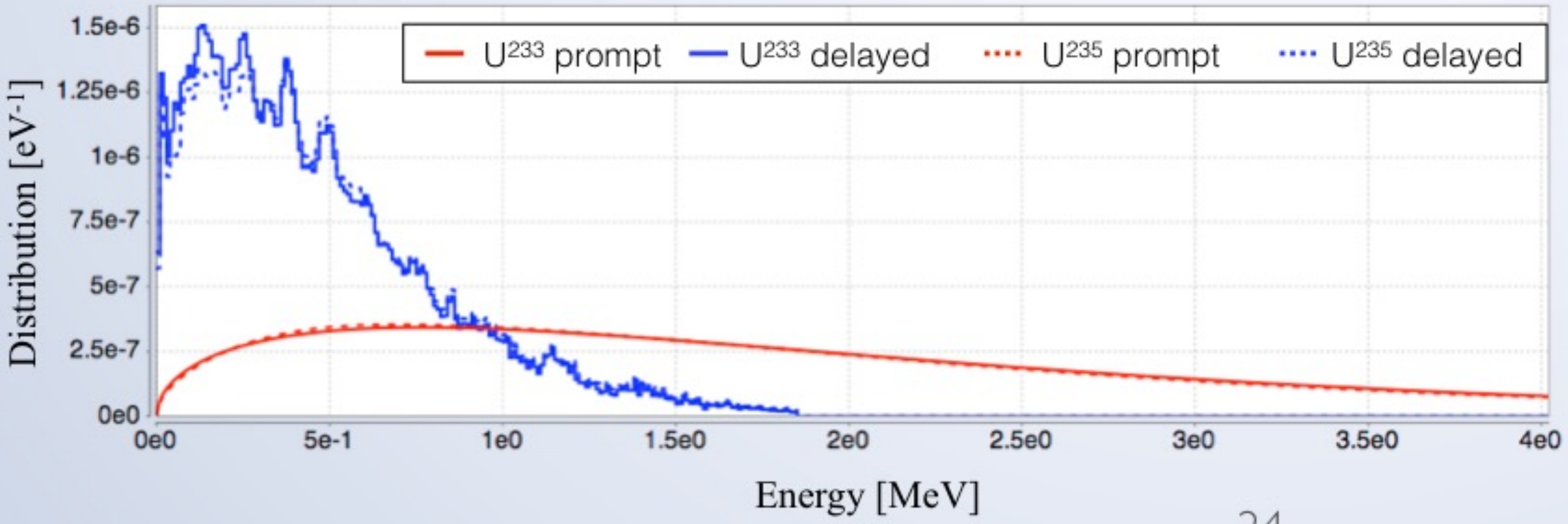
TRANSIENT FISSION MATRIX APPROACH

OVERALL PRINCIPLE: CHARACTERIZE THE SYSTEM RESPONSE TO A NEUTRON PULSE ON ONE GENERATION (GREEN FUNCTION)





But… neutrons can be prompt or delayed!



four fission matrices are required

$\underset{p \to p}{\underline{\underline{G_{\chi_p\nu_p}}}}$, $\underset{p \to d}{\underline{\underline{G_{\chi_p\nu_d}}}}$, $\underset{d \to p}{\underline{\underline{G_{\chi_d\nu_p}}}}$ and $\underset{d \to d}{\underline{\underline{G_{\chi_d\nu_d}}}}$

new!

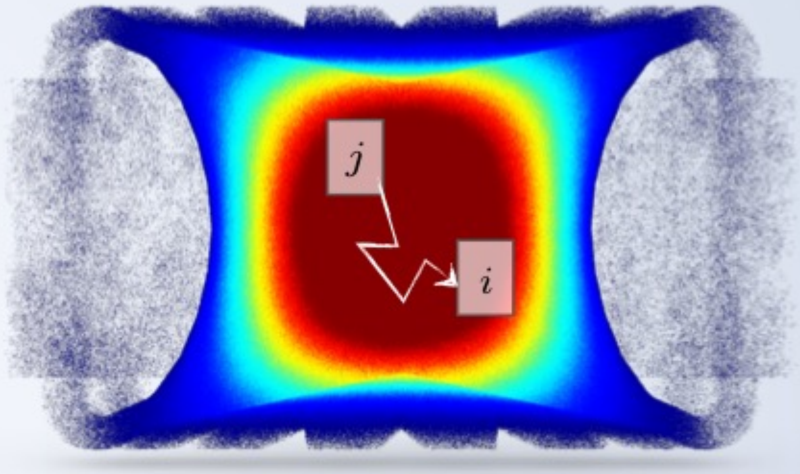
+1 time matrix for the local propagation time: $\underline{\underline{T_{\chi_p\nu_p}}}$

Estimated using Monte Carlo calculation codes

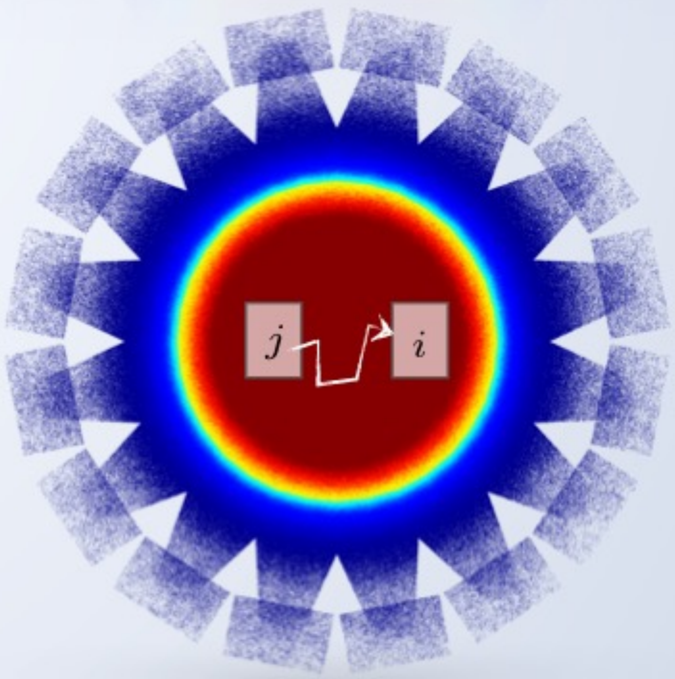
Monte Carlo implementation

Implementation in a Monte Carlo calculation code

- In a criticality calculation, for each source neutron:
  - label if "prompt" or "delayed"
  - associate the birth position *j*
- At each interaction:
  - get the interaction position *i*
  - score the fission rate and the neutron lifetime



- All the fission/time matrices are estimated in a single calculation
- This implementation is possible thanks to the easy source modification in SERPENT
- The calculation code SERPENT is able to use a « CAD » complex geometry



TRANSIENT FISSION MATRIX APPROACH

Mathematic properties

Mathematic properties

equilibrium prompt neutron source distribution

Eigen vector & Eigen value: $\underline{\underline{G}}_{\chi_p\nu_p}\cdot \longrightarrow \underline{\underline{G}}_{\chi_p\nu_p}\boldsymbol{N}_p = k_p\boldsymbol{N}_p$

multiplication factor

Source importance map: *(for effective parameter calculation)*

- The adjoint flux is the solution of the adjoint transport equation. It corresponds to the neutron transport where the time is inverted.
- The transpose matrix ~~i←*j*~~: i→*j* corresponds to the backward source transport!

$$\underline{\underline{G}}^{tr}_{\chi_p\nu_p}\boldsymbol{N}^*_p = k_p\boldsymbol{N}^*_p$$

importance map of the neutron source
(= how many neutrons come from this position)

With delayed neutrons:

- We use a global matrix including both prompt and delayed neutrons

$$\underline{\underline{G}}_{all} = \begin{array}{c} \begin{array}{cc} \mathrm{p} \to \mathrm{p} & \mathrm{d} \to \mathrm{p} \end{array} \\ \begin{pmatrix} \underline{\underline{G}}_{\chi_p\nu_p} & \underline{\underline{G}}_{\chi_d\nu_p} \\ \underline{\underline{G}}_{\chi_p\nu_d} & \underline{\underline{G}}_{\chi_d\nu_d} \end{pmatrix} \\ \begin{array}{cc} \mathrm{p} \to \mathrm{d} & \mathrm{d} \to \mathrm{d} \end{array} \end{array} \begin{array}{l} \nearrow \\ \searrow \end{array} \begin{array}{l} \underline{\underline{G}}_{all}(\boldsymbol{N}_{p,eq}\ \boldsymbol{N}_{d,eq}) = k(\boldsymbol{N}_{p,eq}\ \boldsymbol{N}_{d,eq}) \\ \underline{\underline{G}}^{tr}_{all}(\boldsymbol{N}^*_{p,eq}\ \boldsymbol{N}^*_{d,eq}) = k(\boldsymbol{N}^*_{p,eq}\ \boldsymbol{N}^*_{d,eq}) \end{array}$$

Effective parameter calculation

Estimation of effective kinetic parameters

Effective fraction of delayed neutrons β_{eff}

delayed

$$\beta_{eff} = \frac{\boldsymbol{N}^*_{d,eq}\boldsymbol{N}_{d,eq}}{\boldsymbol{N}^*_{d,eq}\boldsymbol{N}_{d,eq} + \boldsymbol{N}^*_{p,eq}\boldsymbol{N}_{p,eq}}$$

importance weighting

delayed

prompt

lifetime

Effective prompt generation time $\Lambda_{eff} = \frac{l_{eff}}{k}$

average j→i transport time

The effective prompt lifetime l_{eff} (*global*) is extracted from the fission to fission propagation time matrix $\underline{\underline{T_{\chi_p\nu_p}}}$ (*local*)

average time × production × sources

$$l_{eff} = \frac{\boldsymbol{N}^*_p\left(\underline{\underline{T_{\chi_p\nu_p}}} \cdot \underline{\underline{G_{\chi_p\nu_p}}}\right)\boldsymbol{N}_p}{\boldsymbol{N}^*_p\,\underline{\underline{G_{\chi_p\nu_p}}}\boldsymbol{N}_p}$$

production × sources

TRANSIENT FISSION MATRIX APPROACH

Kinetics equations

Kinetic model v1:

- discretize the matrices through time
- matrix-vector products = time propagation

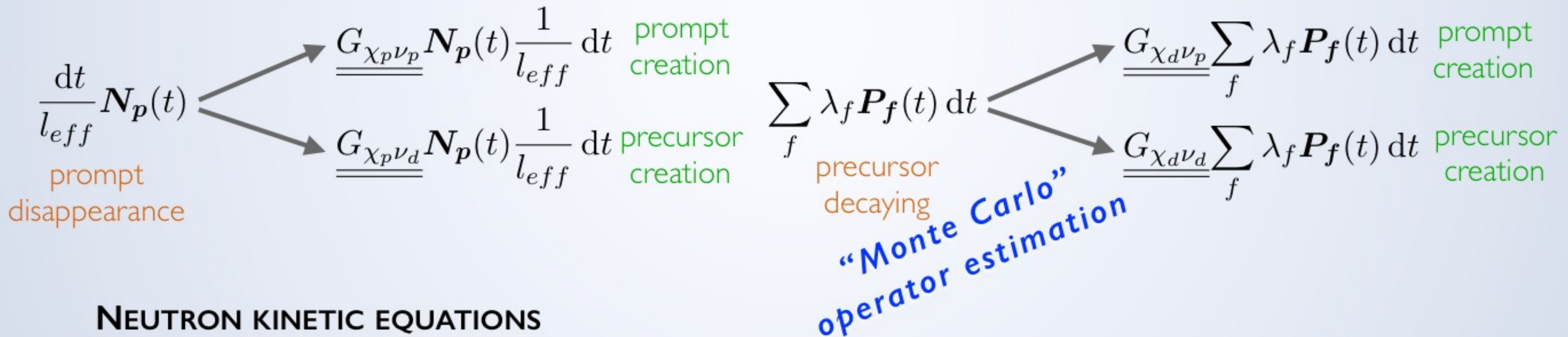
valid for any prompt distribution

Kinetic model v2:

- use the average time response matrix $\underline{\underline{T_{\chi_p\nu_p}}}$

assumes that the prompt distribution is near to equilibrium

Prompt neutron distribution $\boldsymbol{N_p(t)}$ associated to the disappearing time l_{eff} and the delayed neutrons $\boldsymbol{P_f}$ to their decay constant λ_f
During $\mathrm{d}t$:



$\frac{\mathrm{d}t}{l_{eff}}\boldsymbol{N_p}(t)$ prompt disappearance

$\rightarrow \underline{\underline{G_{\chi_p\nu_p}}}\boldsymbol{N_p}(t)\frac{1}{l_{eff}}\,\mathrm{d}t$ prompt creation

$\rightarrow \underline{\underline{G_{\chi_p\nu_d}}}\boldsymbol{N_p}(t)\frac{1}{l_{eff}}\,\mathrm{d}t$ precursor creation

$\sum_f \lambda_f \boldsymbol{P_f}(t)\,\mathrm{d}t$ precursor decaying

$\rightarrow \underline{\underline{G_{\chi_d\nu_p}}}\sum_f \lambda_f \boldsymbol{P_f}(t)\,\mathrm{d}t$ prompt creation

$\rightarrow \underline{\underline{G_{\chi_d\nu_d}}}\sum_f \lambda_f \boldsymbol{P_f}(t)\,\mathrm{d}t$ precursor creation

"Monte Carlo" operator estimation

NEUTRON KINETIC EQUATIONS

Prompt neutrons

$$\frac{\mathrm{d}\boldsymbol{N_p}}{\mathrm{d}t}(t) = \underline{\underline{G_{\chi_p\nu_p}}}\boldsymbol{N_p}(t)\frac{1}{l_{eff}} + \underline{\underline{G_{\chi_d\nu_p}}}\sum_f \lambda_f \boldsymbol{P_f}(t) - \frac{1}{l_{eff}}\boldsymbol{N_p}(t)$$

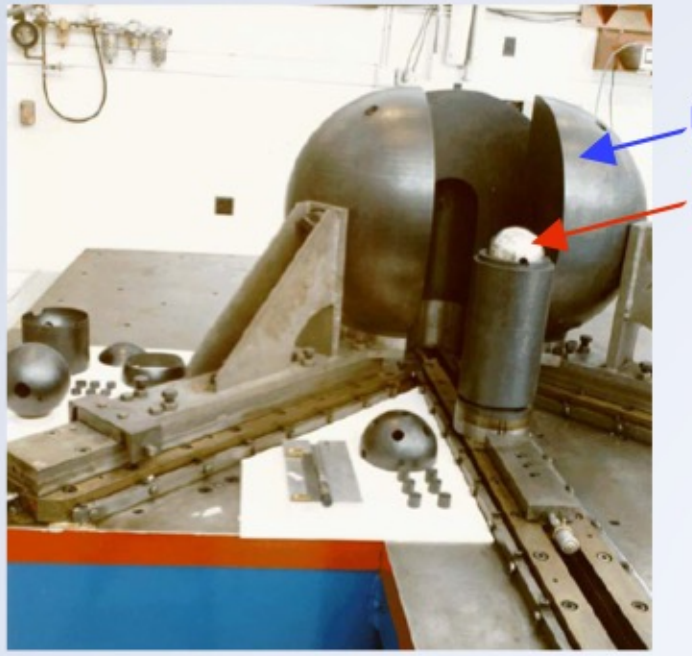
Precursor familly *f*

$$\frac{\mathrm{d}\boldsymbol{P_f}}{\mathrm{d}t}(t) = \frac{\beta_f}{\beta_0}\left[\underline{\underline{G_{\chi_p\nu_d}}}\boldsymbol{N_p}(t)\frac{1}{l_{eff}} + \underline{\underline{G_{\chi_d\nu_d}}}\sum_f \lambda_f \boldsymbol{P_f}(t)\right] - \lambda_f \boldsymbol{P_f}(t)$$

"deterministic" integration

TRANSIENT FISSION MATRIX APPROACH

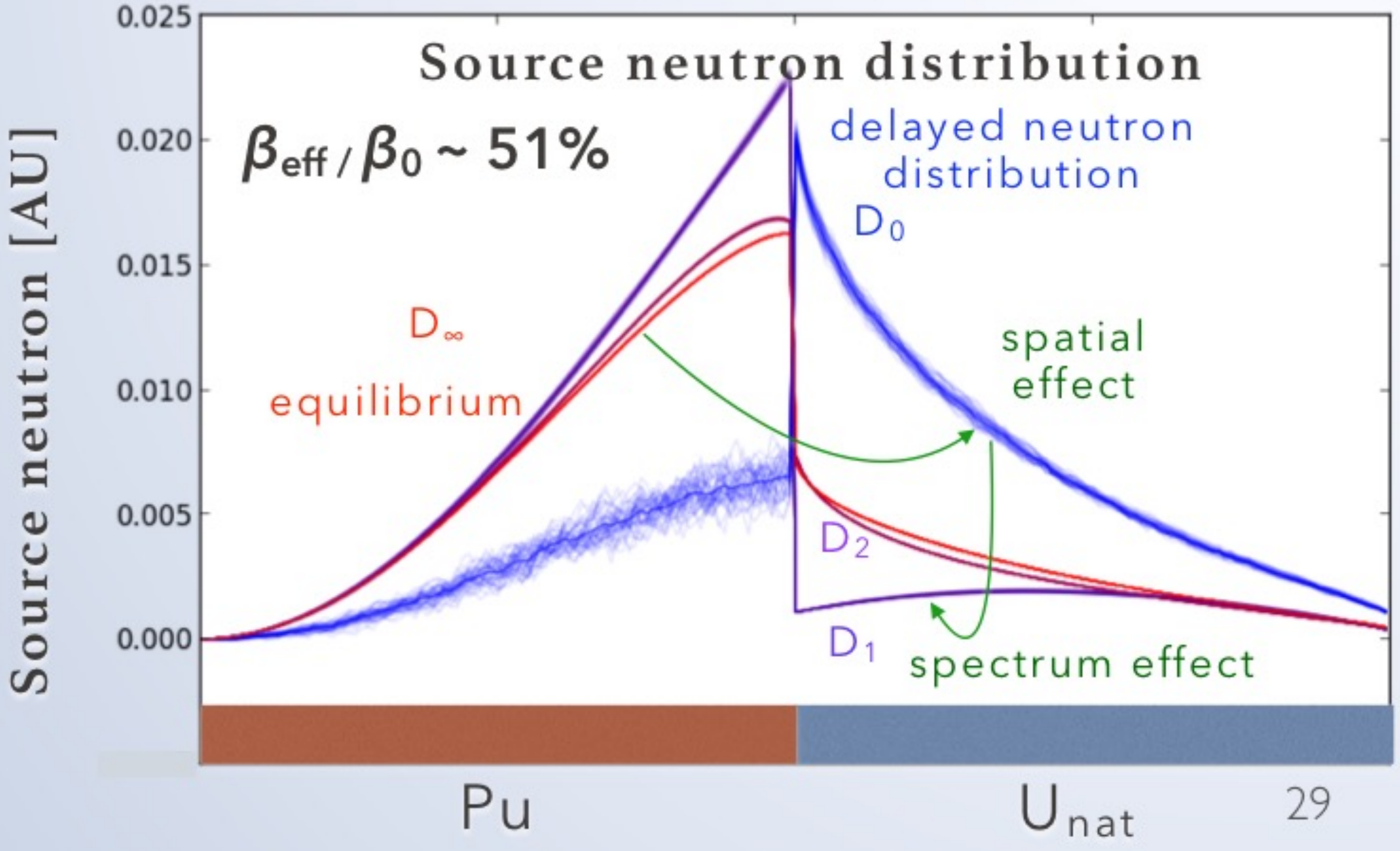
Validation - Flattop experiment

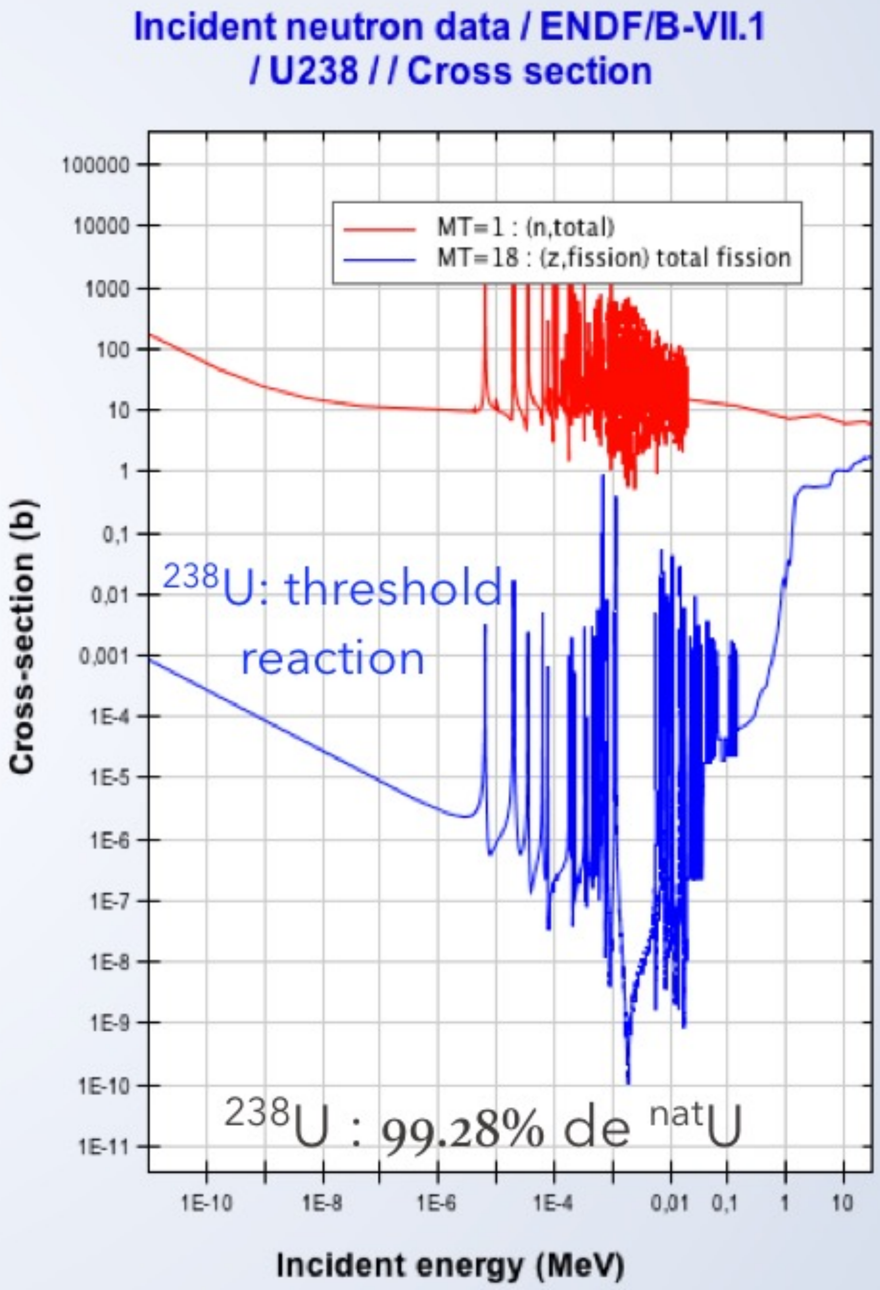


U_{nat} « fertile »

Pu « fissile »

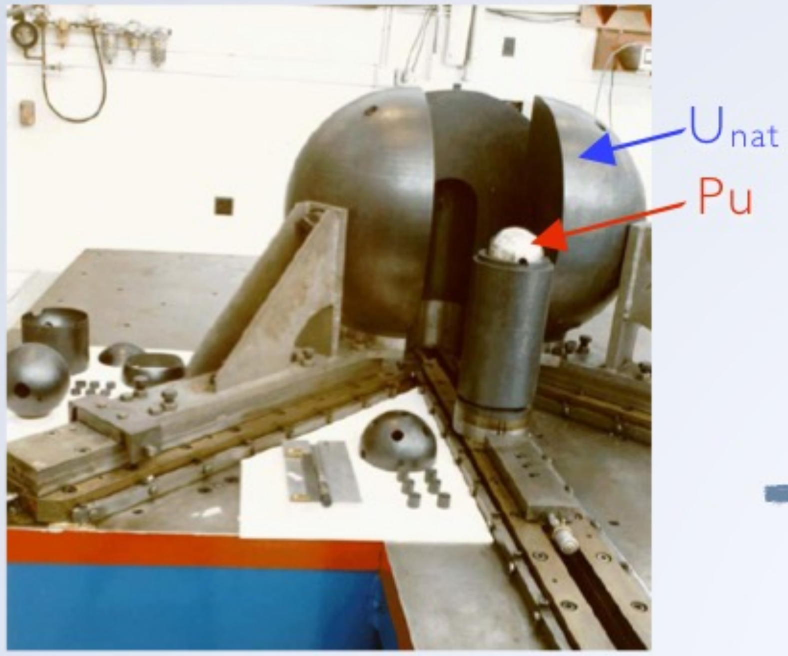
FLATTOP EXPERIMENT - LANL
REFERENCE : EXPERIMENTAL + IFP





TRANSIENT FISSION MATRIX APPROACH

Validation - Flattop experiment



FLATTOP EXPERIMENT - LANL
REFERENCE : EXPERIMENTAL + DIRECT MONTE CARLO

Observable: α_{Rossi}

- critical system : $k=k_{\mathrm{p}}+\beta_{\mathrm{eff}}=1$
- low power (few precursors)
- neutron pulse injection in the system

➡ neutron kinetics lead by the prompt neutrons:

- at each generation Λ_{eff},
- the neutron population decreases of $-\beta_{eff}$,

neutron shower decays according to:

$$exp(\boxed{-\frac{\beta_{eff}}{\Lambda_{eff}}}\cdot t) \longrightarrow \alpha_{Rossi} = -\frac{\beta_{eff}}{\Lambda_{eff}}$$

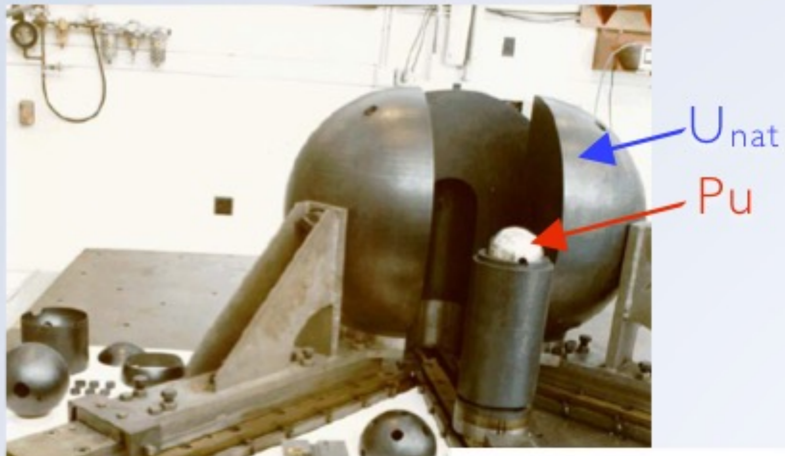
Method	β_{eff}	Λ_{eff}	$-\frac{\beta_{\mathrm{eff}}}{\Lambda_{\mathrm{eff}}} = \alpha_{\mathrm{Rossi}}$
Experiment	-	-	$-0.214 \pm 0.005\ \mu\mathrm{s}^{-1}$
Reference calculation	276.9 ± 0.3 pcm	13.246 ± 0.003 ns	$-0.2091 \pm 0.0003\ \mu\mathrm{s}^{-1}$
TFM / reference	1.001 ± 0.002	1.0005 ± 0.0004	1.000 ± 0.002

- Very good agreement between TFM and reference calculations (no biais) …
- … and with experimental measurements!

[*A. Laureau et al.* Transient Fission Matrix: kinetic calculation and kinetic parameters βeff and Λeff calculation, *Annals of Nuclear Energy, volume 85, p. 1035–1044 (2015)*]

TRANSIENT FISSION MATRIX APPROACH

Validation - Flattop experiment



Observable: α_{Rossi}

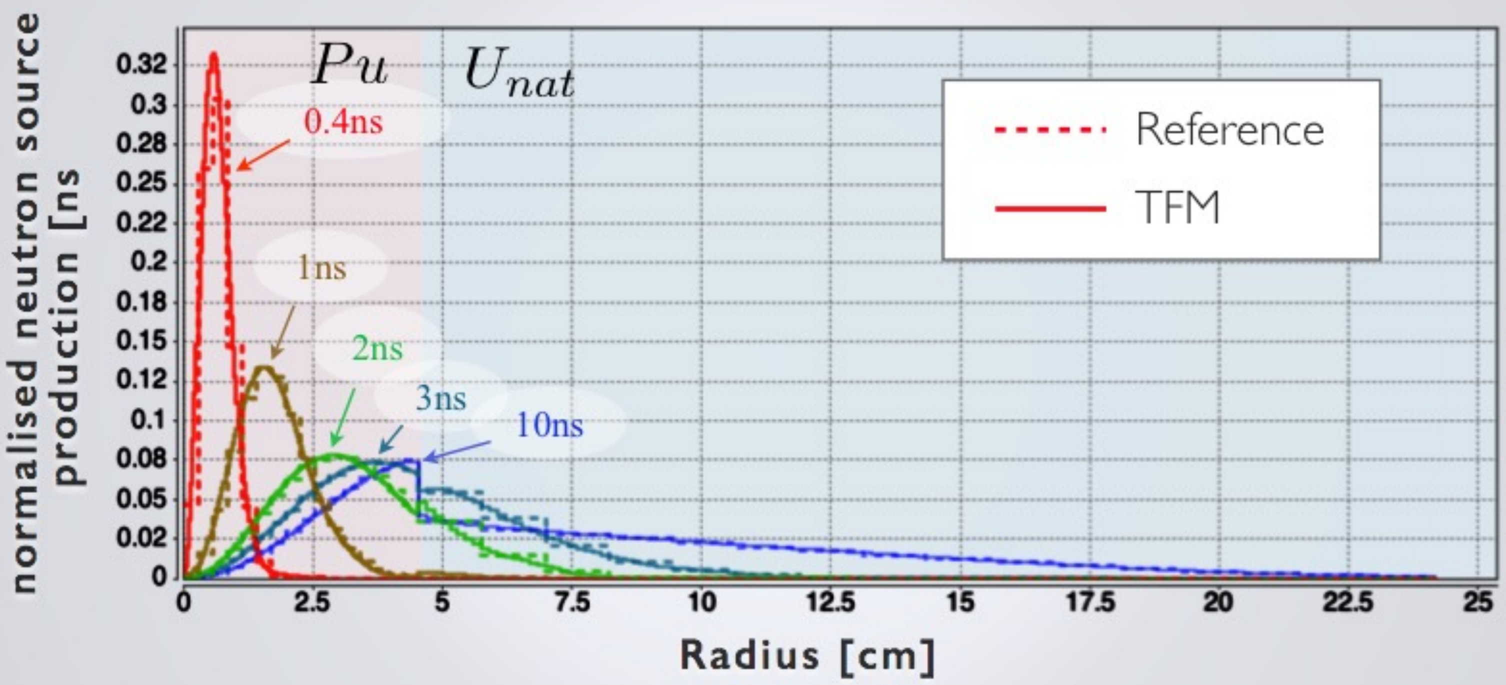
- critical system : k=k_p+β_{eff}=1
- low power (few precursors)
- neutron pulse injection in the system

:

$- \frac{\beta_{eff}}{\Lambda_{eff}}$

FLATTOP EXPERIMENT
REFERENCE : EXPERIMENT
MONTE CARL

Neutron prompt shower: Spatial evolution



Very good agreement on the spatial distribution

Method			ssi
Experiment			μs^{-1}
Reference cal			3 μs^{-1}
TFM / reference	1.001 ± 0.002	1.0005 ± 0.0004	1.000 ± 0.002

- Very good agreement between TFM and reference calculations (no biais) …
- … and with experimental measurements!

[*A. Laureau et al.* Transient Fission Matrix: kinetic calculation and kinetic parameters βeff and Λeff calculation, *Annals of Nuclear Energy, volume 85, p. 1035–1044 (2015)*]

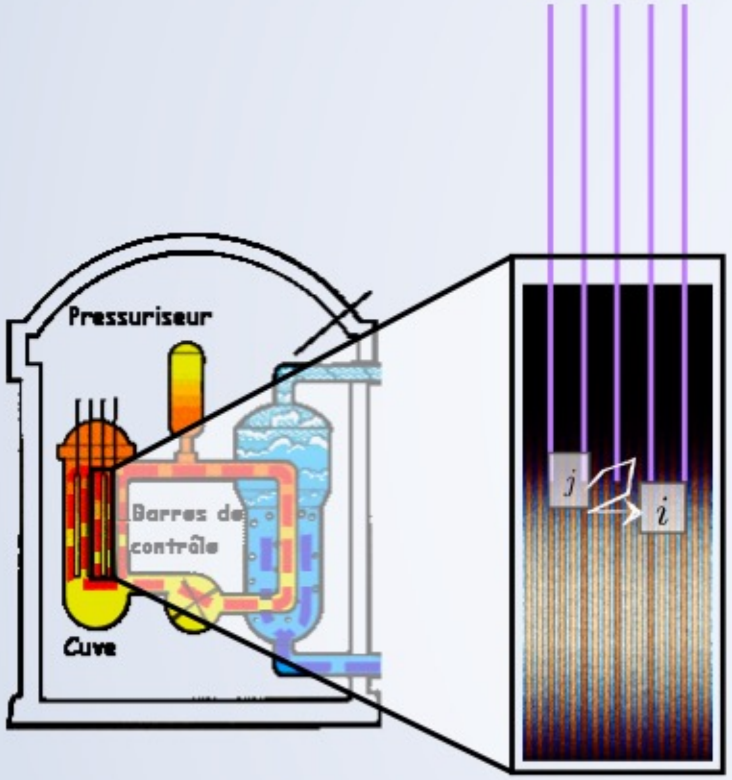
TRANSIENT FISSION MATRIX APPROACH

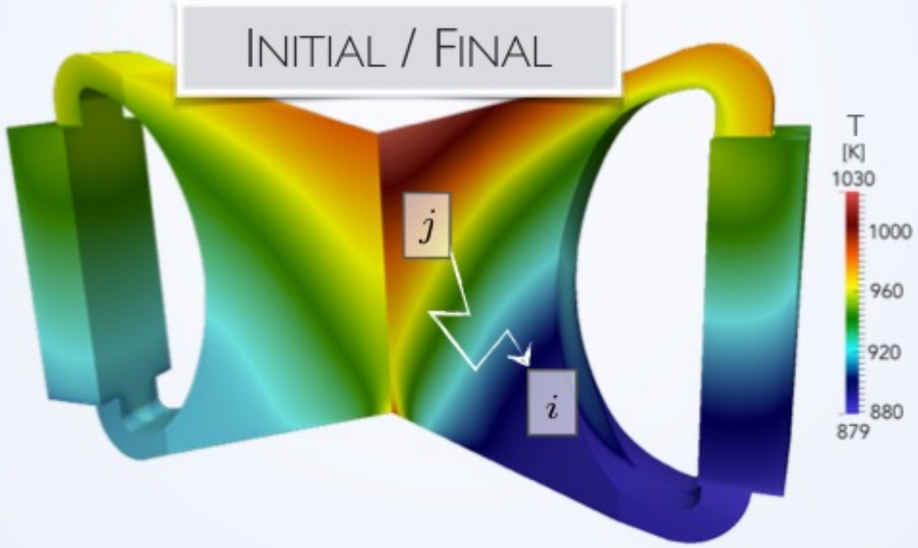
Fission matrix interpolation

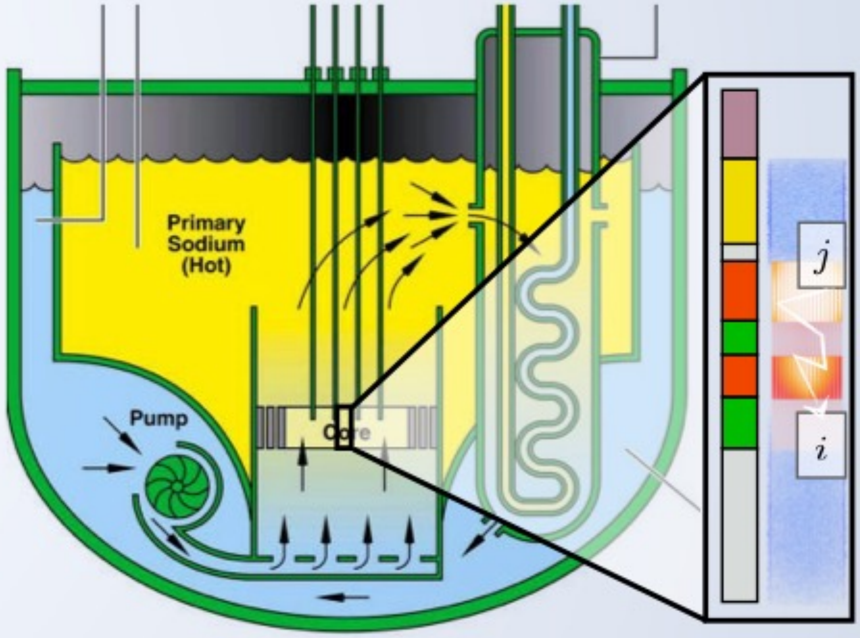
ONE MORE THING…
THE SYSTEM IS CHANGING DURING A TRANSIENT…

During transient coupled calculations, composition-geometry-temperature will evolve …

How can we take into account this phenomenon without new Monte Carlo calculations ?







Pressurized Water Reactor:

- Small migration area
- Heterogeneous

MSFR:

- Large migration area
- Homogeneous

Sodium Fast Reactor:

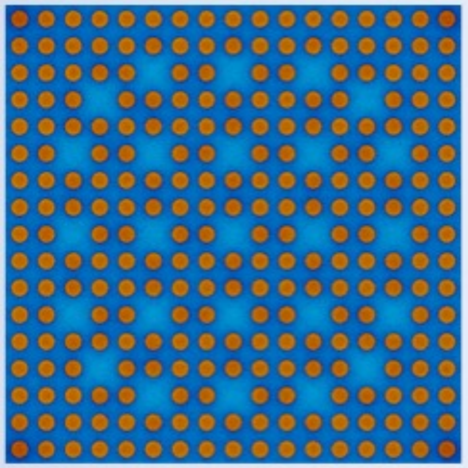
- Large migration area
- Heterogeneous

TRANSIENT FISSION MATRIX APPROACH

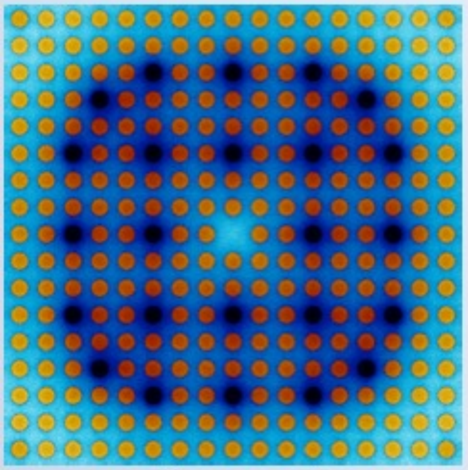
PWR interpolation

Fission matrices interpolation: absorber rods in a PWR assembly - international benchmark Minicore (EDF collaboration)

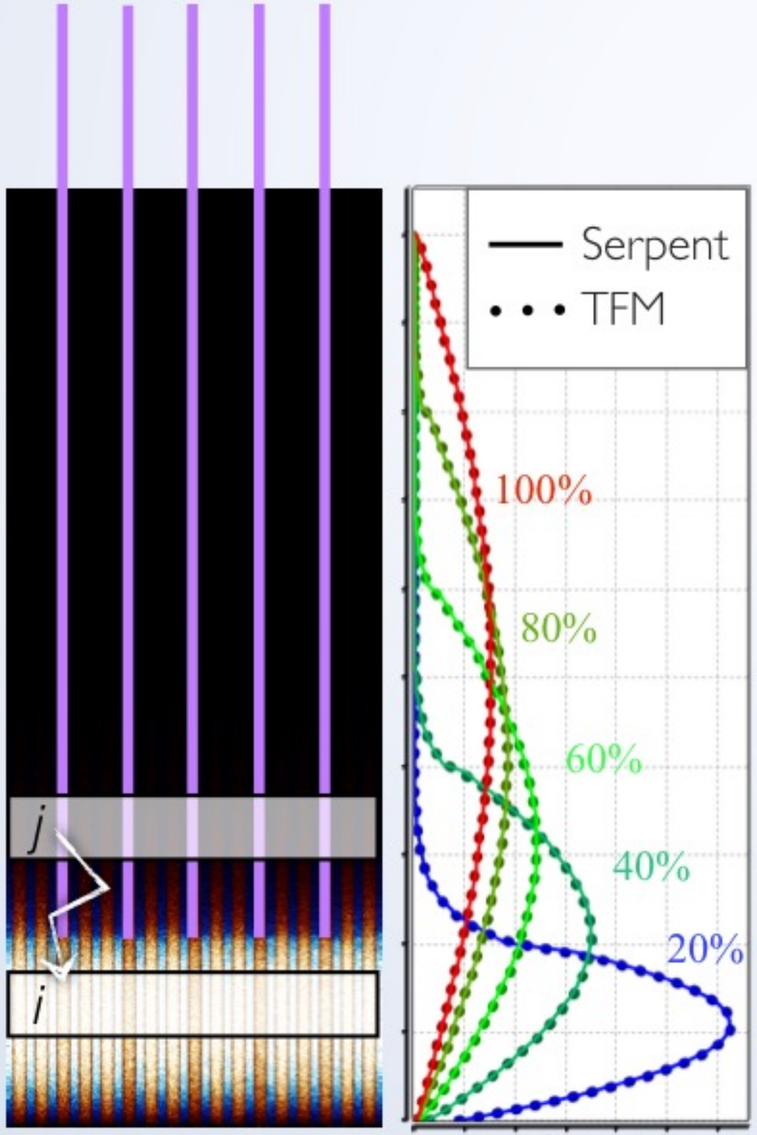
blue - thermal flux
red - power



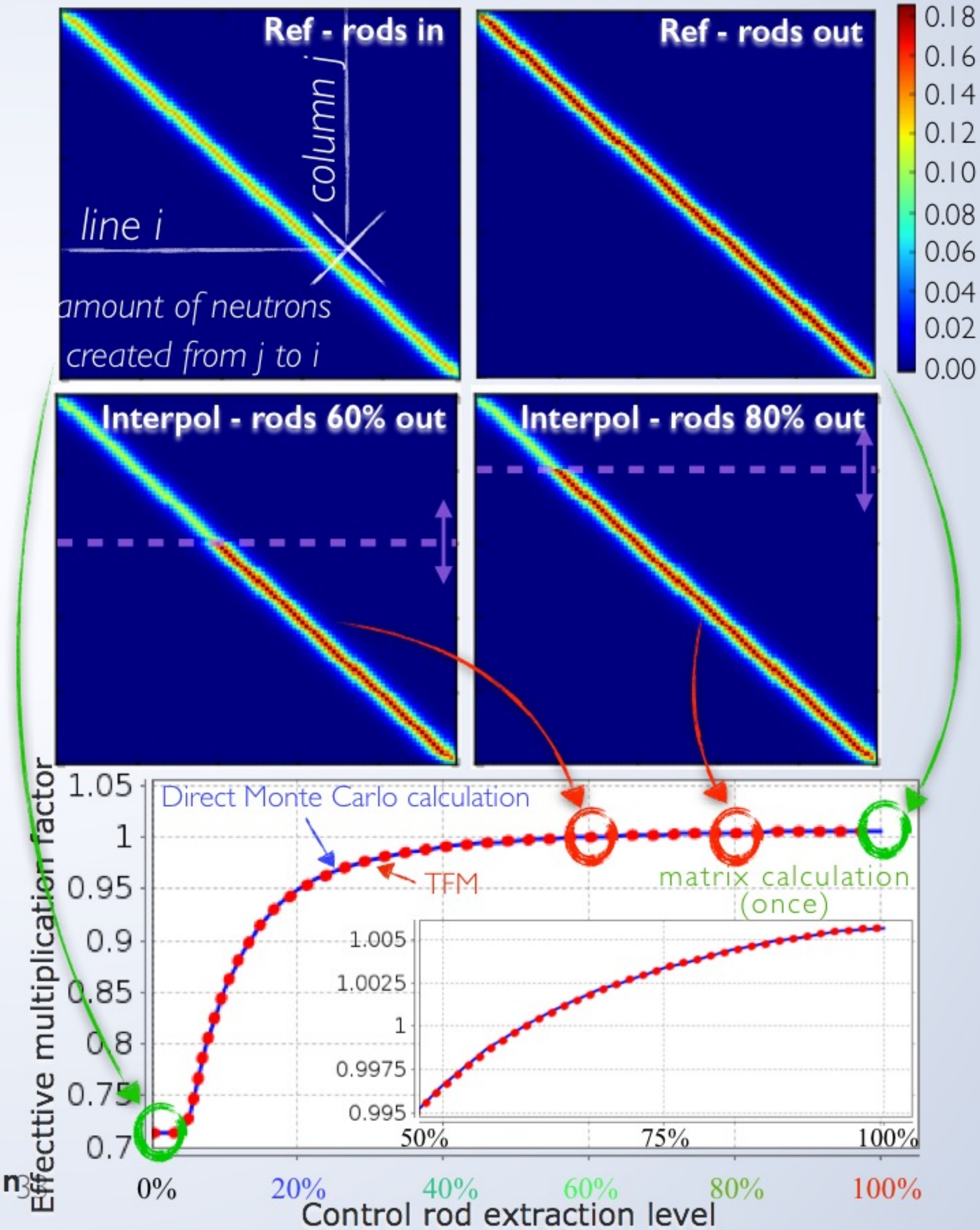
assemby - rods in



assemby - rods out



source distribution (normalised)



TRANSIENT FISSION MATRIX APPROACH

PWR interpolation

Fission matrices interpolation: absorber rods in a PWR assembly - international benchmark Minicore (EDF collaboration)

blue - thermal flux
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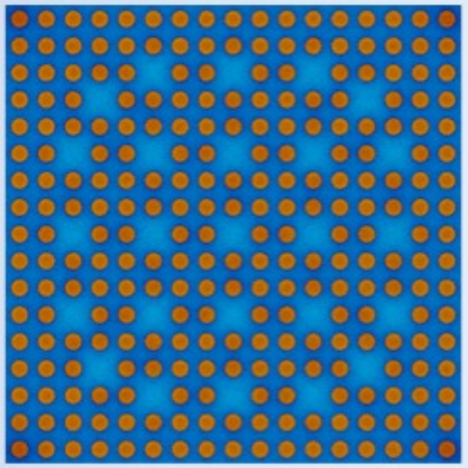
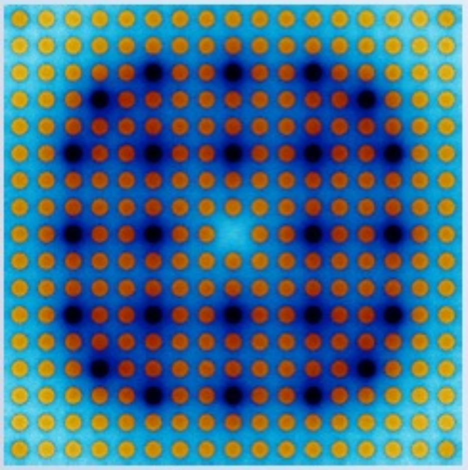
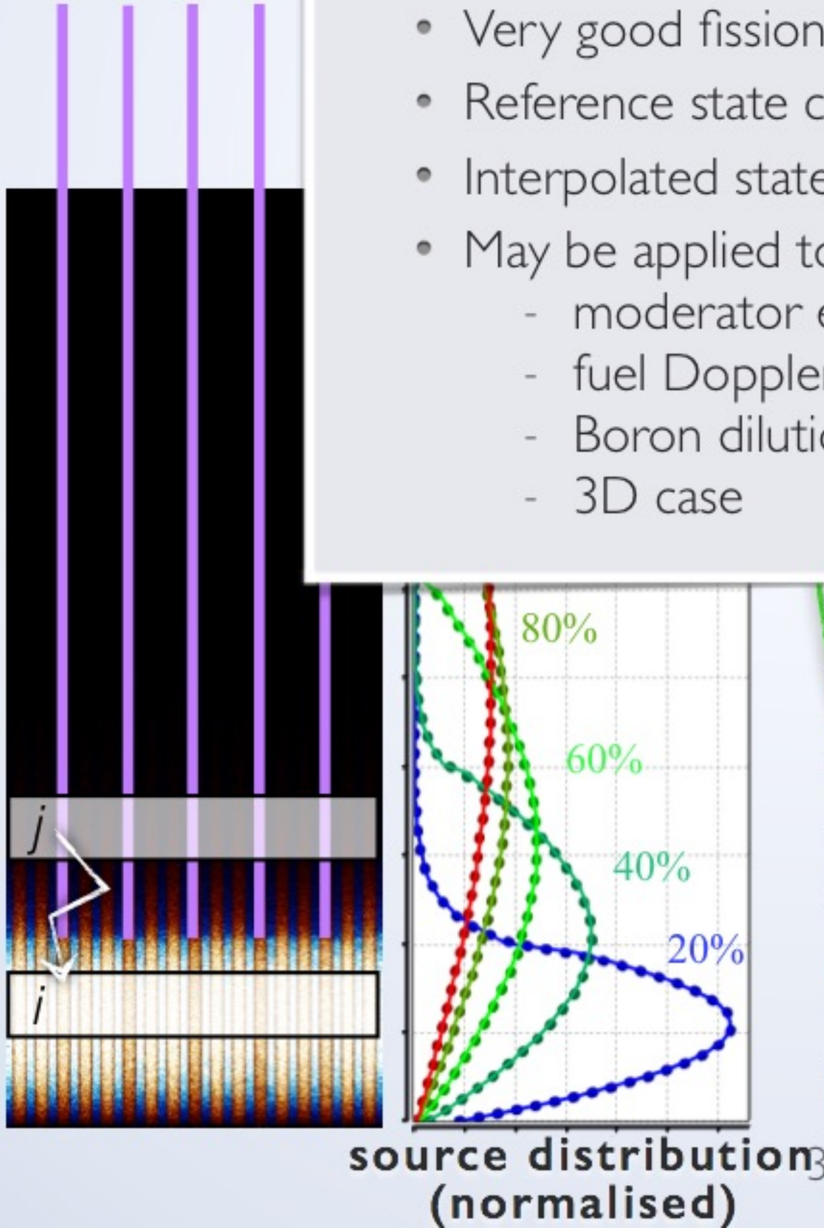


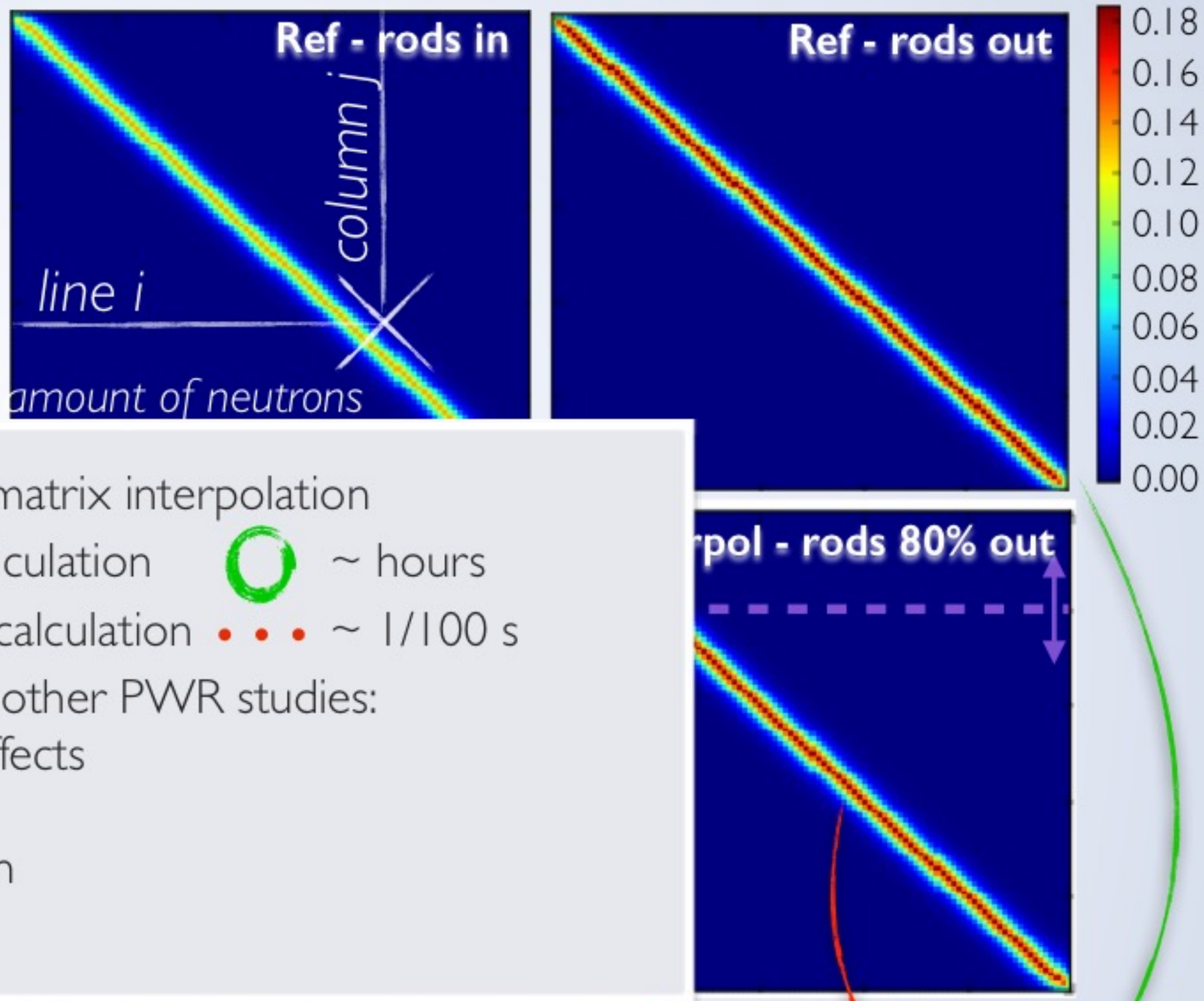
assemby - rods in

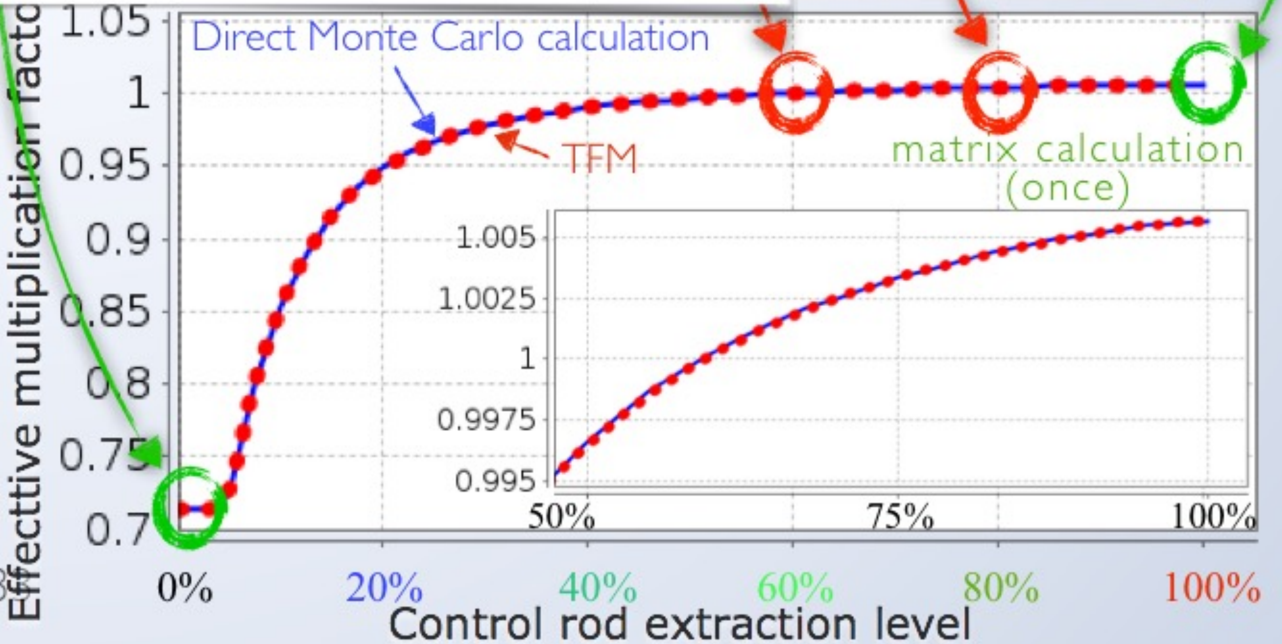


assemby - rods out

- Very good fission matrix interpolation
- Reference state calculation ○ ~ hours
- Interpolated state calculation • • • ~ 1/100 s
- May be applied to other PWR studies:
  - moderator effects
  - fuel Doppler
  - Boron dilution
  - 3D case



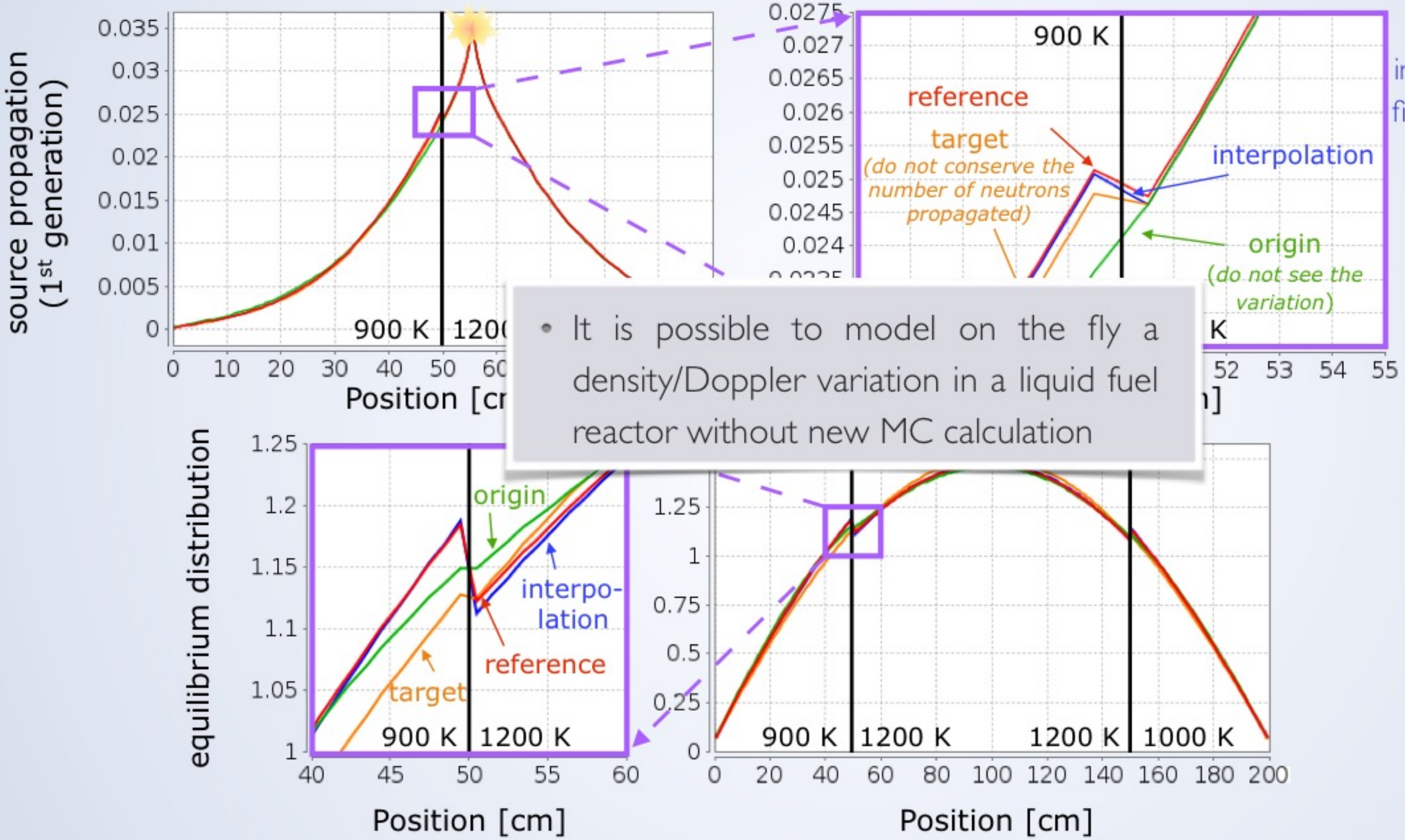




MSFR interpolation

Test case: 200 cm 1D reactor with the MSFR fuel salt composition

Two pieces of information: temperature in the origin cell (*j*), and in the target cell (*i*)



- It is possible to model on the fly a density/Doppler variation in a liquid fuel reactor without new MC calculation

interpolation: use the "target" fission rate normalised by the "origin" absorption rate

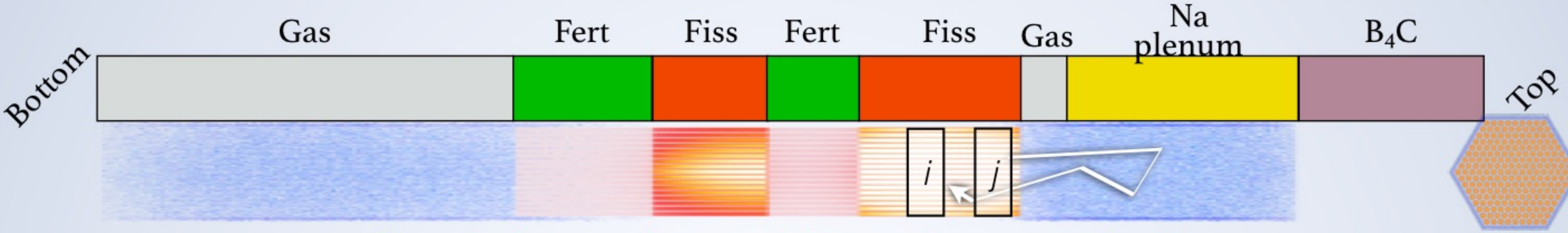
requires to estimate the absorption matrix

Method	Reference	Origin (T_j)	Target (T_i)	Intepolation
Variation (pcm)	-1174 ± 2	-1358 ± 2	-1354 ± 2	-1222 ± 2
Difference	-	+15.6%	+15.4%	+4.1 %

TRANSIENT FISSION MATRIX APPROACH

ASTRID interpolation

Sodium Fast Reactor ASTRID representative assembly - correlated sampling technique



Need to know the sensibility to the crossed cells

interaction probability modification ⇔ neutron weight modification

↳ for each distance/reaction sampling, the neutron weight is modified

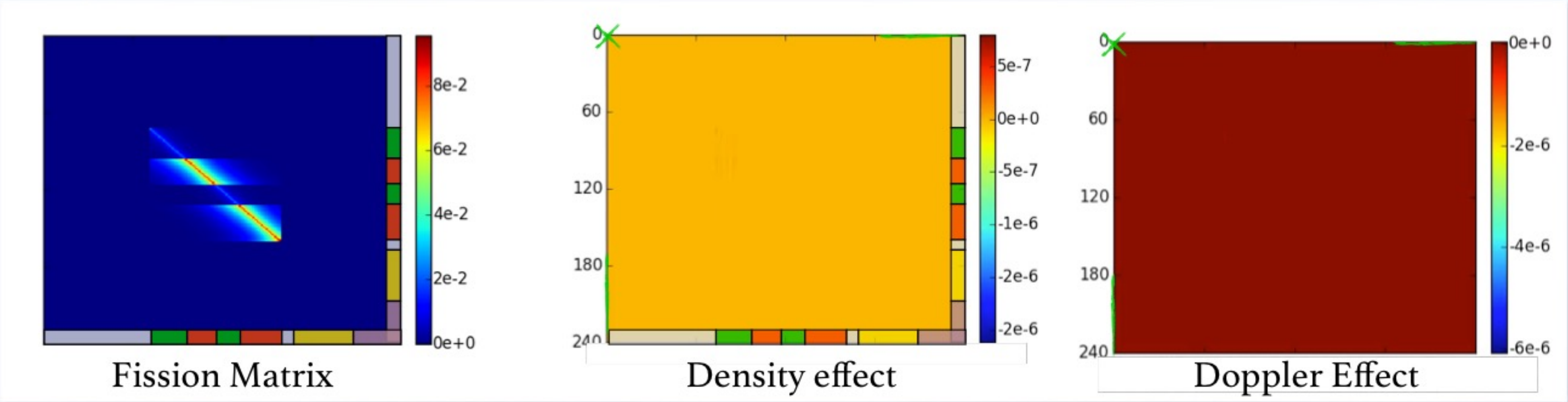
$$\frac{\Sigma_{\rm tot}^{\rm pert}\exp(-d\cdot\Sigma_{\rm tot}^{\rm pert})}{\Sigma_{\rm tot}\exp(-d\cdot\Sigma_{\rm tot})}$$

distance *d*

reaction *n,r*

$$\frac{\Sigma_{\rm n,r}\cdot\Sigma_{\rm tot}^{\rm pert}}{\Sigma_{\rm tot}\cdot\Sigma_{\rm n,r}^{\rm pert}}$$

additional variation matrices:
j → *i* … assuming a perturbation in *k*!



Fission Matrix | Density effect | Doppler Effect

[*A. Laureau et al.* Local correlated sampling Monte Carlo calculations in the TFM neutronics approach for spatial and point kinetics applications. *accepted to **EPJ-N** (2017)]*

[A. Laureau et al. Fission Matrix Interpolation for the TFM Approach Based on a Local Correlated Sampling Technique for Fast Spectrum Heterogeneous Reactors. *proceeding accepted for the International Conference M&C, Jeju, Korea (2017)]*

TRANSIENT FISSION MATRIX APPROACH

ASTRID interpolation

Reference matrix

Interpolated matrix of "any" perturbation distribution

$$\underline{\underline{G_{\chi_x\nu_x}}}(\Delta\rho_{\mathrm{sodium}}(\mathrm{k})) = \underline{\underline{\mathrm{G}_{\chi_\mathrm{x}\nu_\mathrm{x}}}} + \sum_{\mathrm{k}} \underline{\underline{\widetilde{\mathrm{G}}^{\mathrm{den\ k}}_{\chi_\mathrm{x}\nu_\mathrm{x}}}} \cdot \Delta\rho_{\mathrm{sodium}}(\mathrm{k})$$

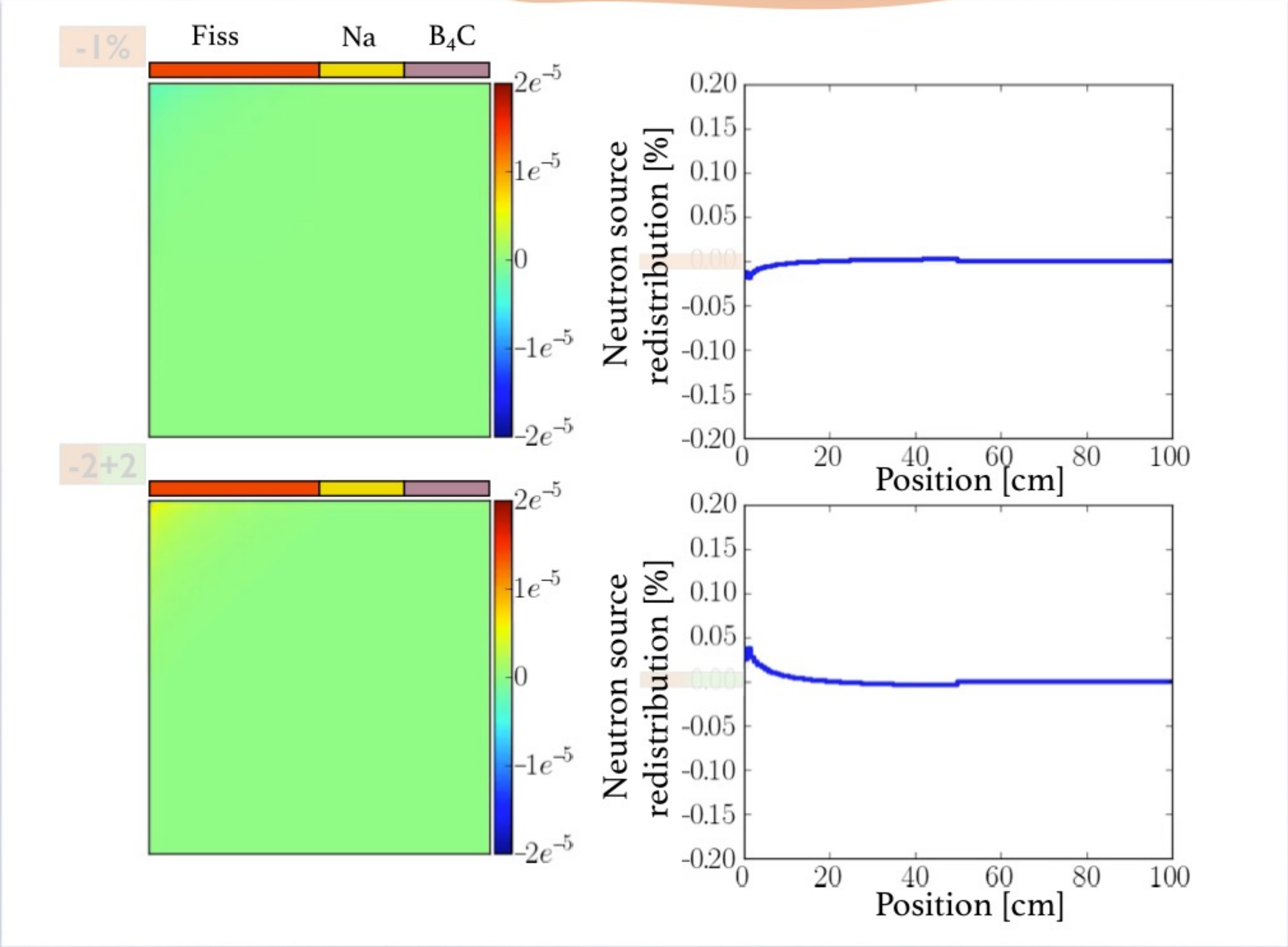
$$\underline{\underline{G_{\chi_x\nu_x}}}(\underline{T}(\mathrm{k})) = \underline{\underline{\mathrm{G}_{\chi_\mathrm{x}\nu_\mathrm{x}}}} + \sum_{\mathrm{k}} \underline{\underline{\widetilde{\mathrm{G}}^{\mathrm{dop\ k}}_{\chi_\mathrm{x}\nu_\mathrm{x}}}} \frac{\log(T(\mathrm{k})/\mathrm{T}_{\mathrm{ref}}(\mathrm{k}))}{\log((\mathrm{T}_{\mathrm{ref}}(\mathrm{k})+300)/\mathrm{T}_{\mathrm{ref}}(\mathrm{k}))}$$

Sum of local contributions

density - linear

Doppler - logarithmic

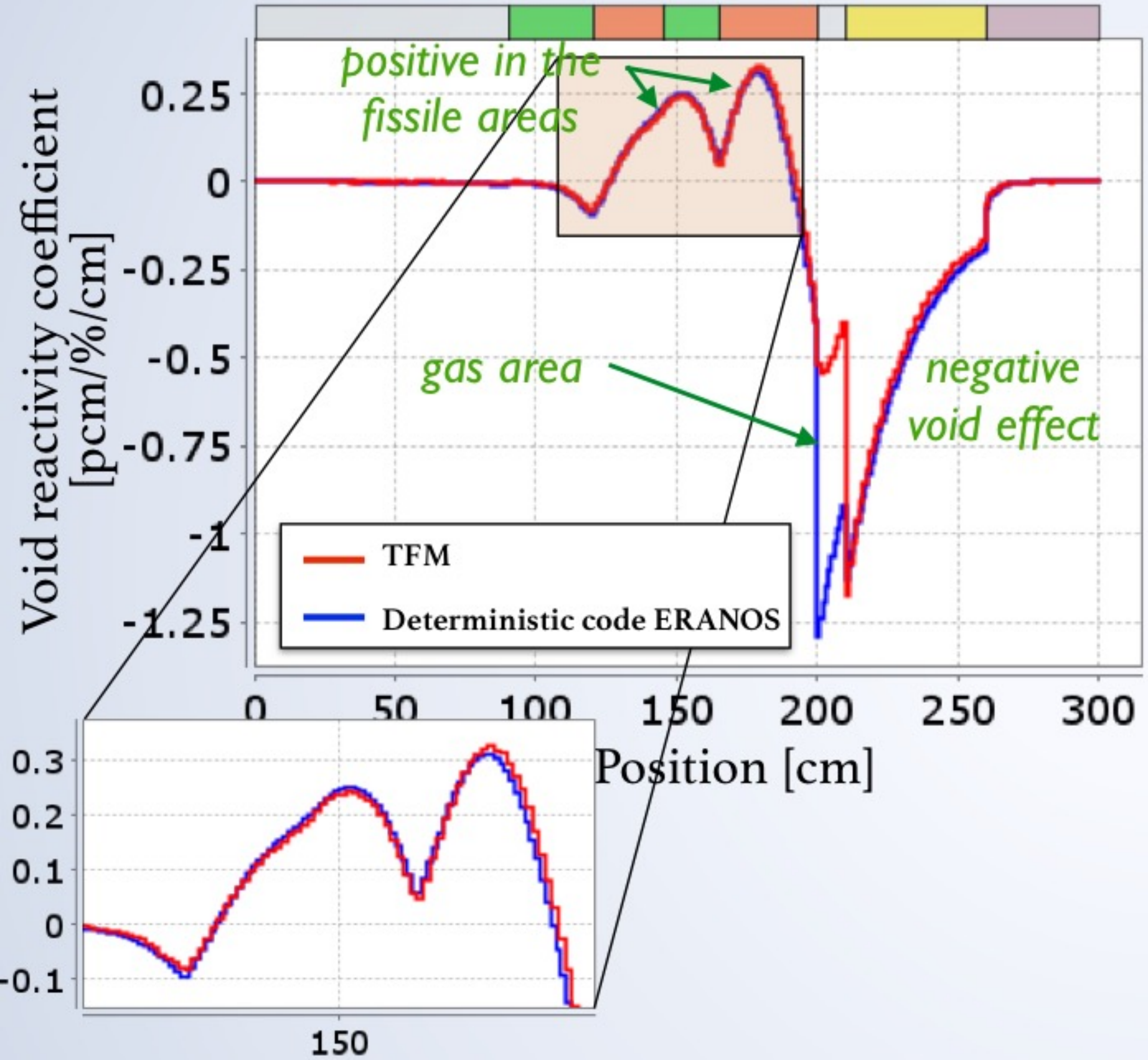
Reconstruction on the fly of any kind of perturbation in the reactor

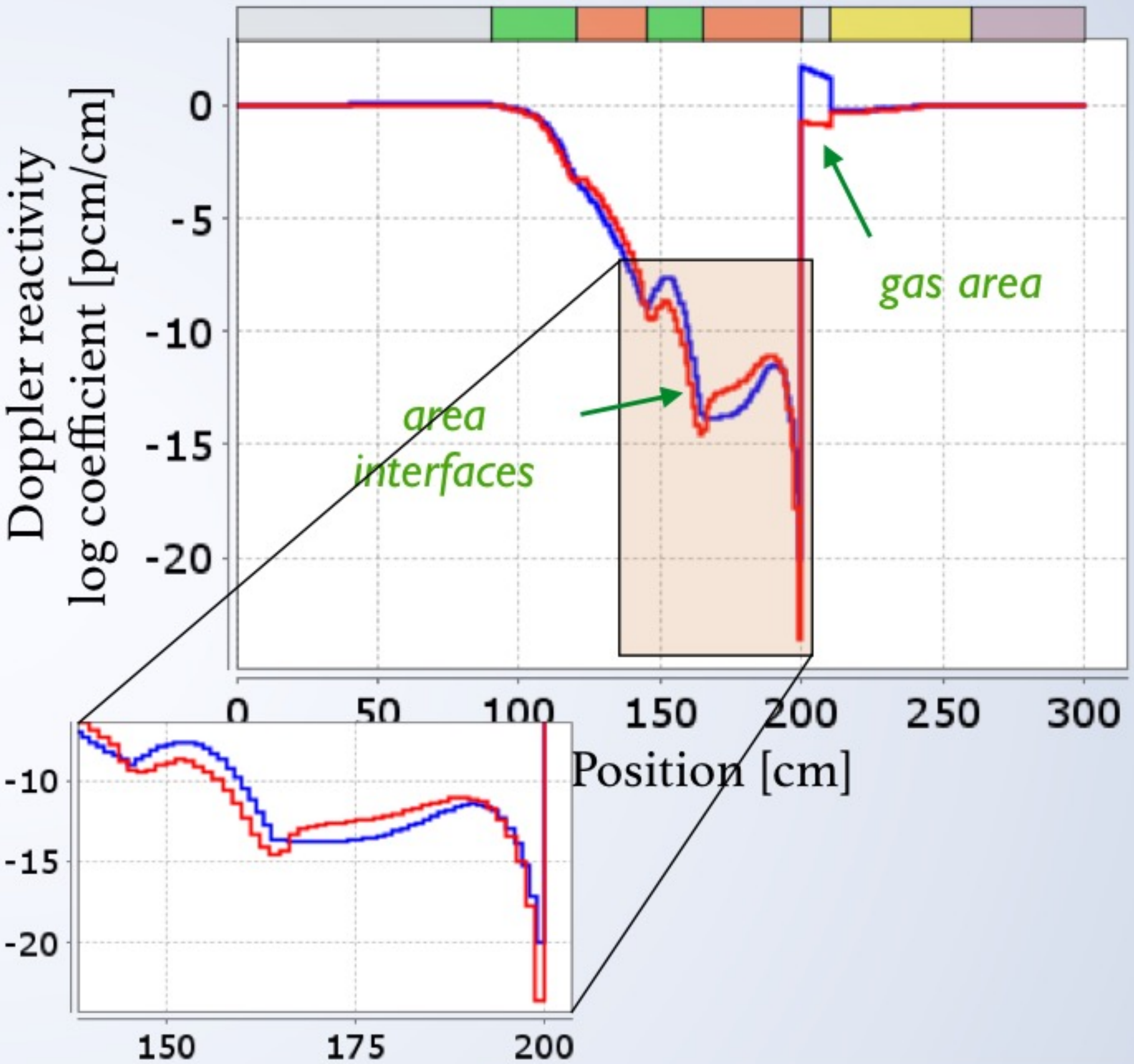


TRANSIENT FISSION MATRIX APPROACH

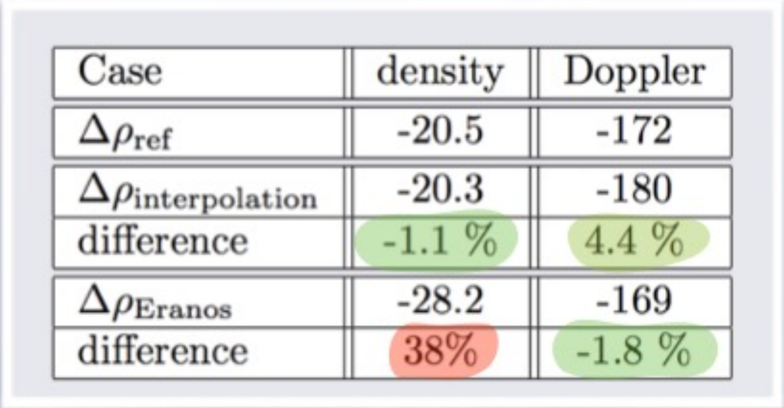
ASTRID interpolation

Local eigenvalue = point kinetic feedback coefficient





- Estimation of local feedback parameters using Monte Carlo (reference)
- Good global agreement with ERANOS
- Quantification of the impact of some hypotheses of deterministic calculation schemes



Case	density	Doppler
$\Delta\rho_{\mathrm{ref}}$	-20.5	-172
$\Delta\rho_{\mathrm{interpolation}}$	-20.3	-180
difference	-1.1 %	4.4 %
$\Delta\rho_{\mathrm{Eranos}}$	-28.2	-169
difference	38%	-1.8 %

OUTLINE

I. FRAME
  - NUCLEAR REACTORS

INTRODUCTION

II. PHYSICS PRESENTATION
  - THERMALHYDRAULICS
  - NEUTRONICS

TOOLBOX

III. PHYSICAL MODELS DEVELOPED
  - OBJECTIVES
  - NEUTRON SHOWER APPROACH
  - TRANSIENT FISSION MATRIX APPROACH
  - REACTOR SPECIFICITIES

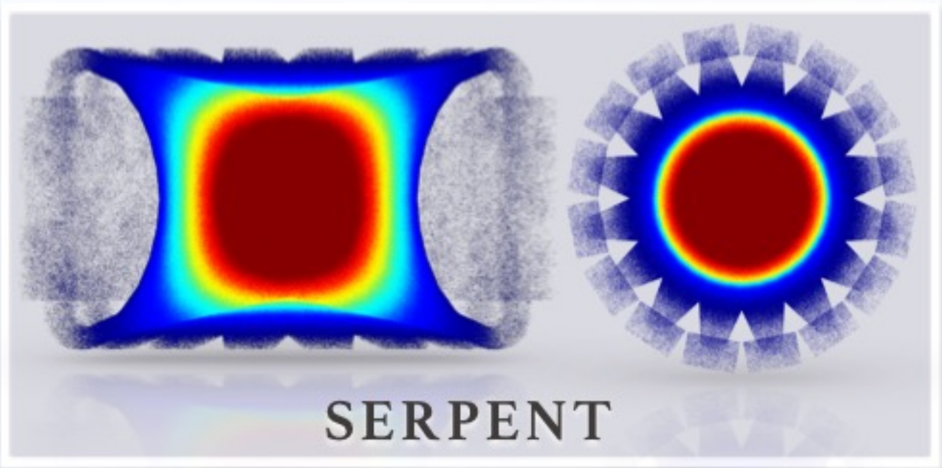
DEVELOPMENT OF INNOVATIVE NEUTRONICS MODELS

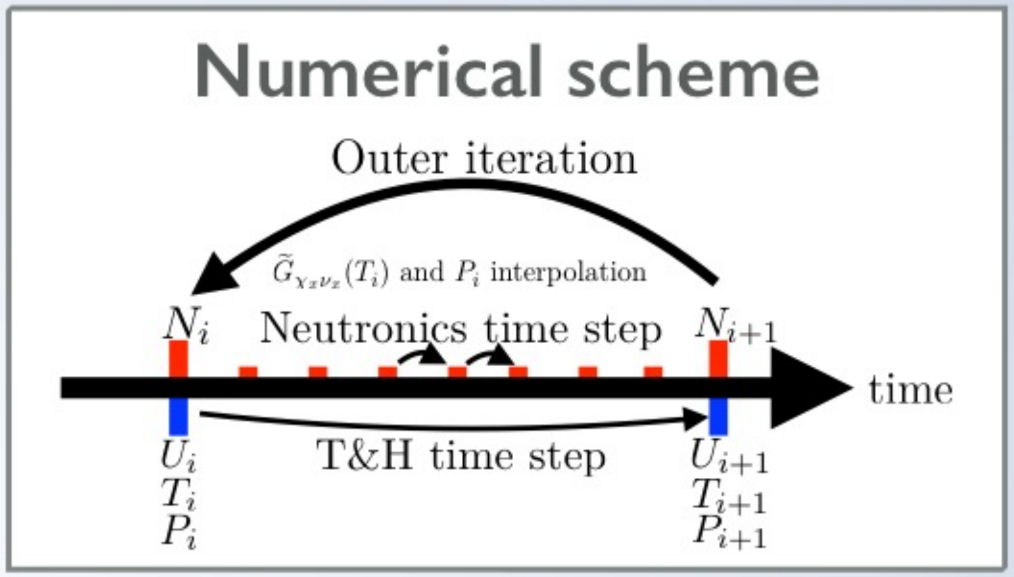
IV. COUPLING & SAFETY STUDY
  - GENERAL STRATEGY
  - STEADY STATE CHARACTERIZATION
  - MSFR SAFETY STUDIES
  - ASTRID ACCIDENTAL STUDY

SYSTEM STUDIES AND OPTIMIZATION

COUPLING & SAFETY STUDY

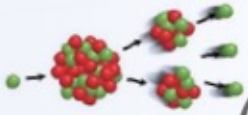
General strategy





matrices calculation

TFM



power → temperature, buoyancy effect, precursor production

neutronics

thermal-hydraulics

CFD

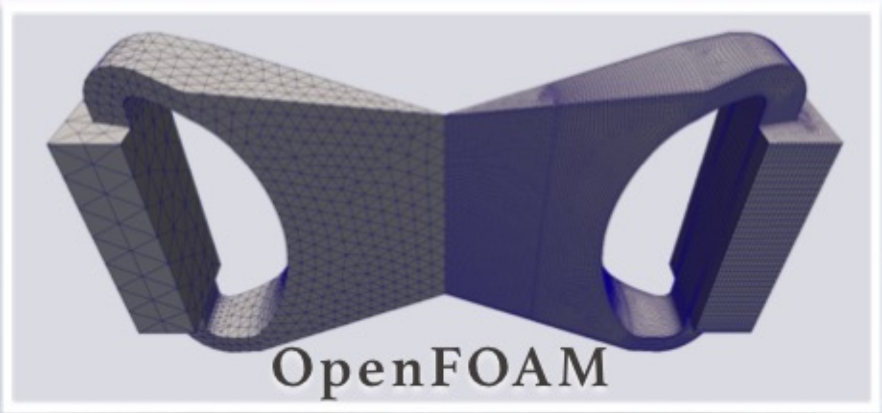
k-ε realizable turbulence model

delayed neutron source
Doppler
density

temperature
precursor decay position

directly implemented and performed in

performed by



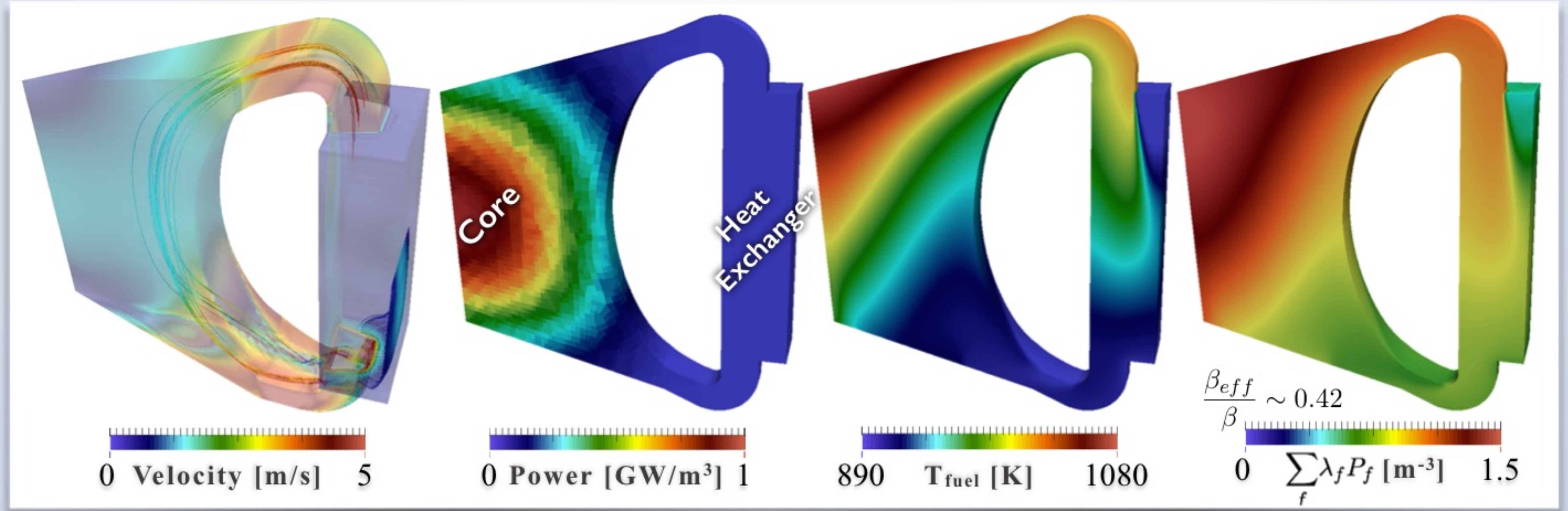
[*A. Laureau et al* Coupled neutronics and thermal-hydraulics transient calculations based on a fission matrix approach : application to the Molten Salt Fast Reactor. *Joint International Conference M&C+ SNA+ MC 2015, Nashville, USA*]

COUPLING & SAFETY STUDY

Steady state characterization

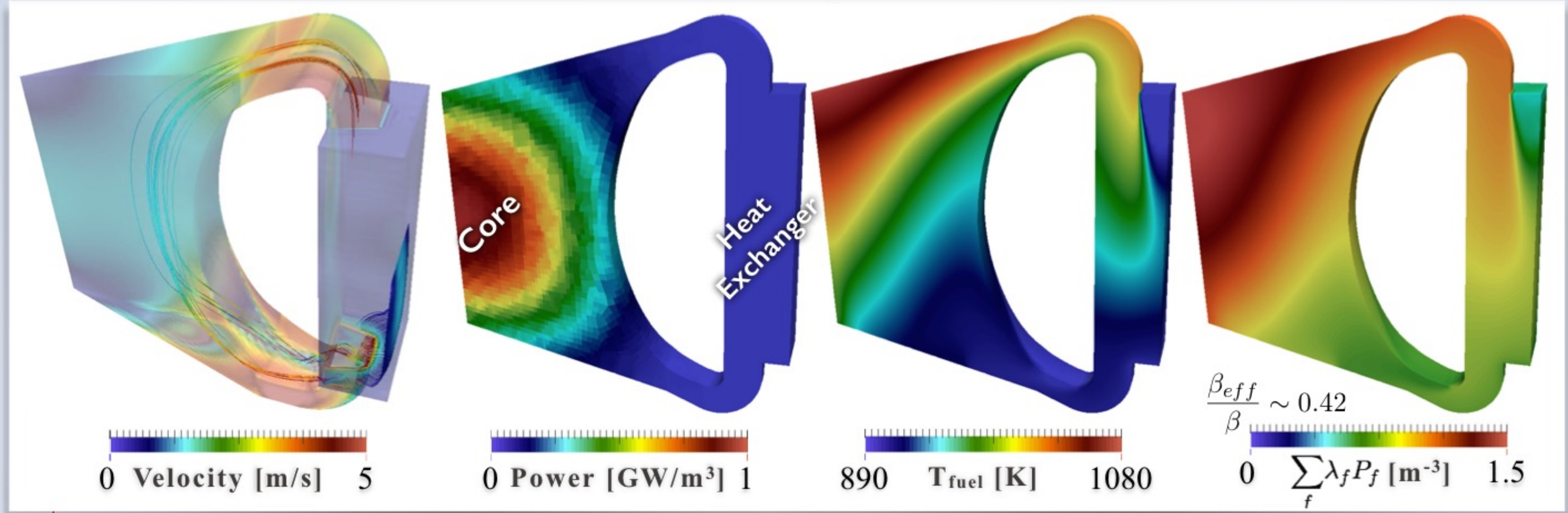
Step 1 - equilibrium solution:

- fixed known parameters
  3GW, average temperature 973 K, circulation time 4s, …
- iterations over problem parameters
  temperature distribution, pump power, …



COUPLING & SAFETY STUDY

Steady state characterization



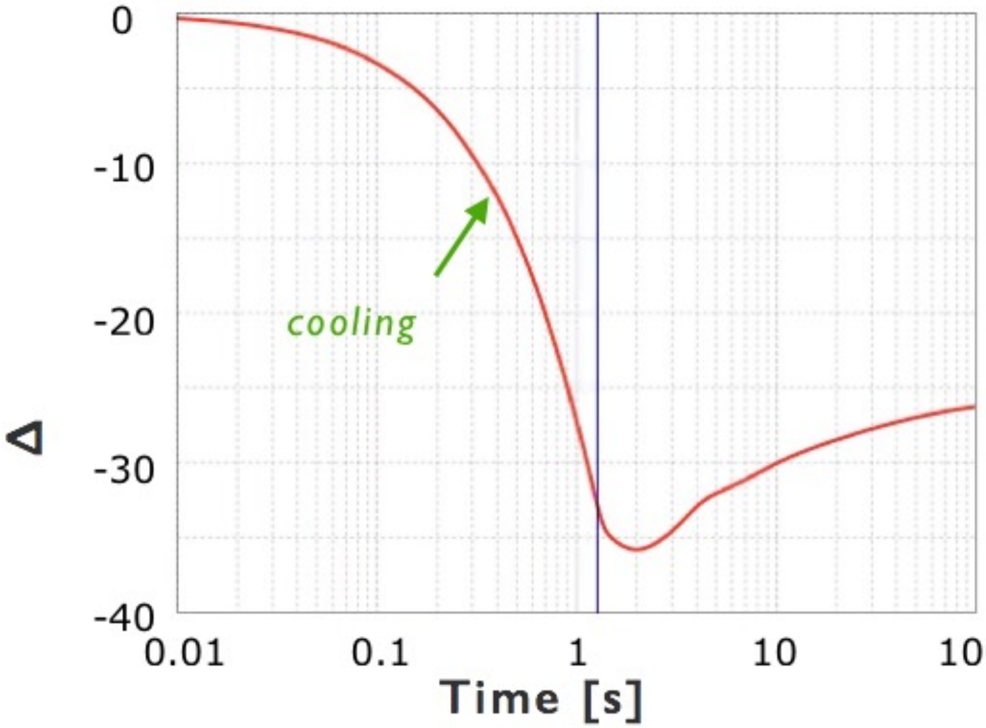
initialise

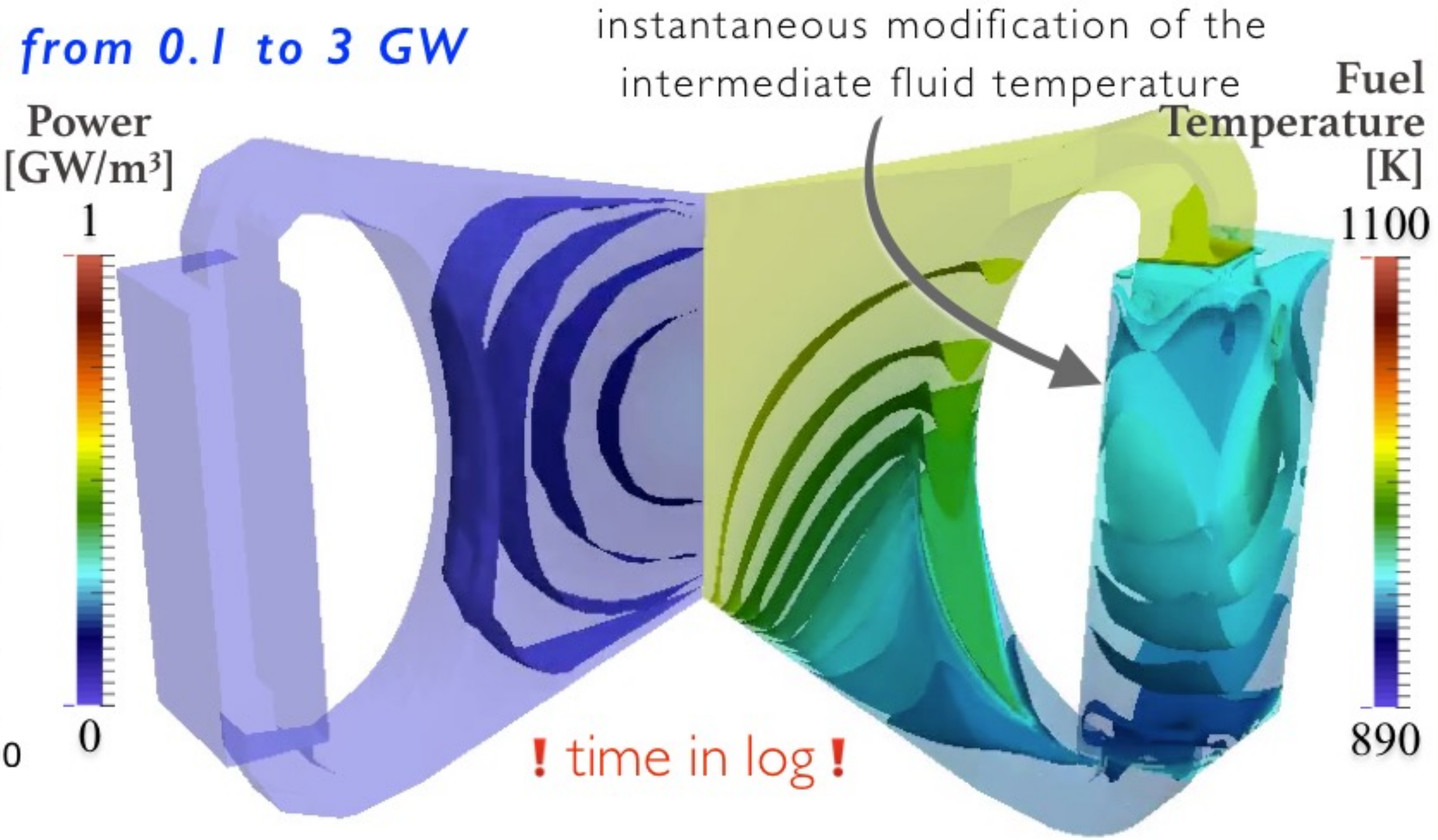
Step 2 - transient studies:

- load following (normal operation)
  power extraction modification via the intermediate fluid temperature
- overcooling (accident)
  fast modification of the intermediate temperature
- reactivity insertion (accident)
  "artificial" increase of the multiplication factor

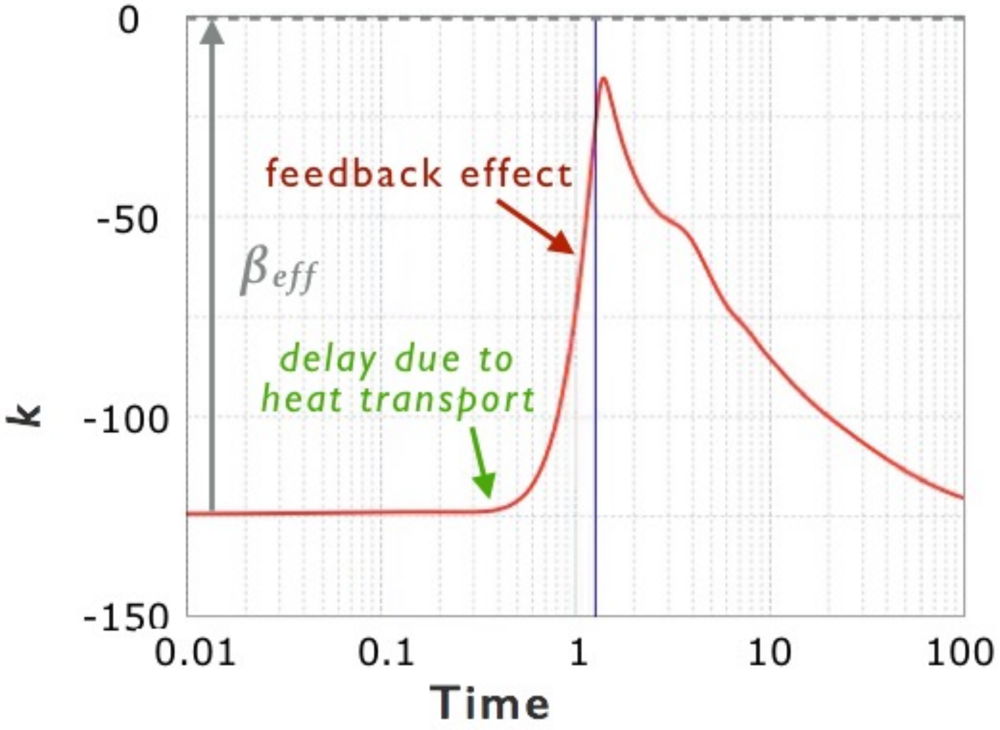
Overcooling accident of the MSFR

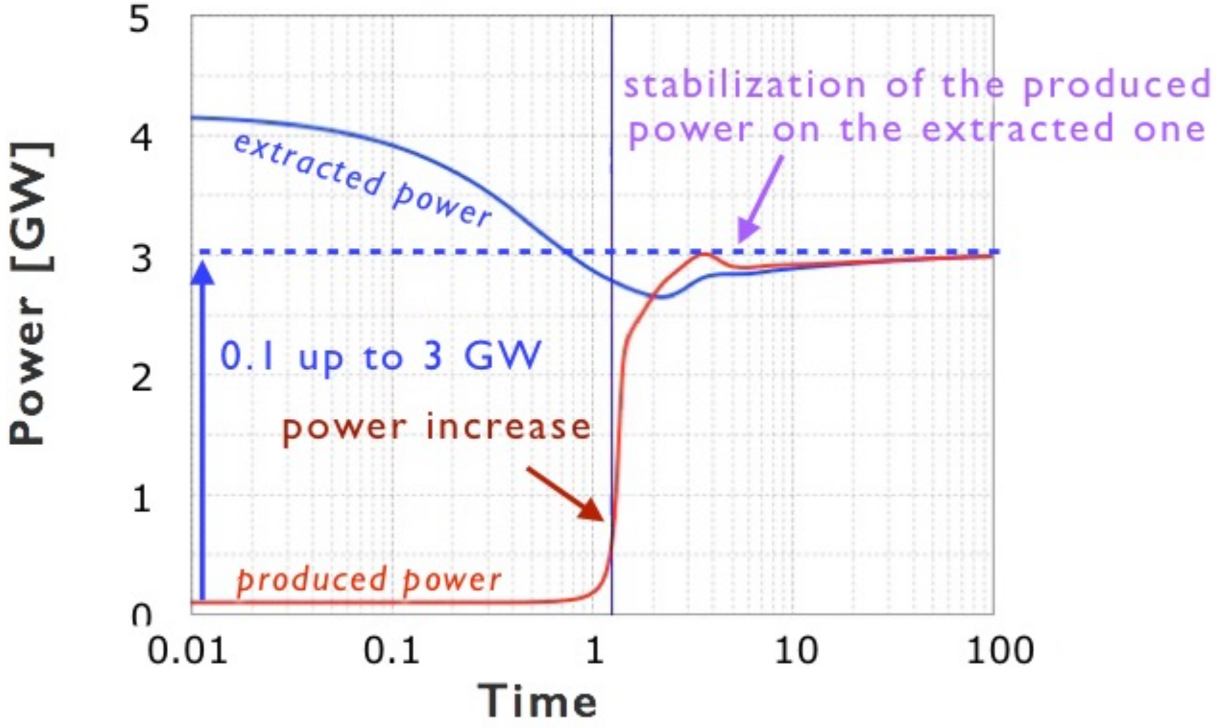
Overcooling: Instantaneous variation from 0.1 to 3 GW





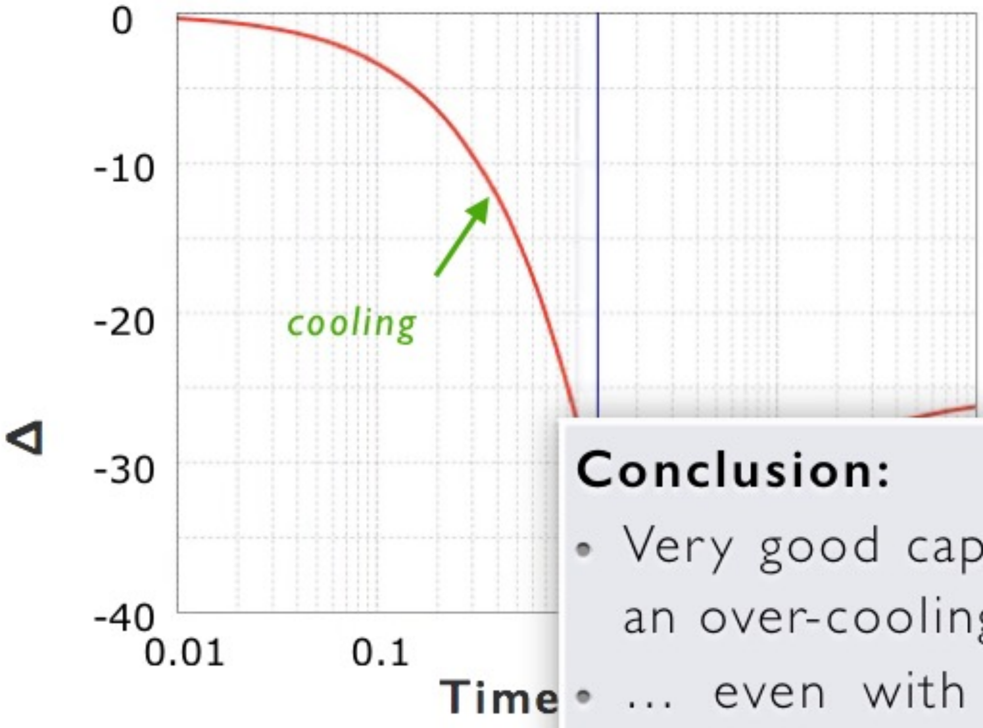
! time in log !

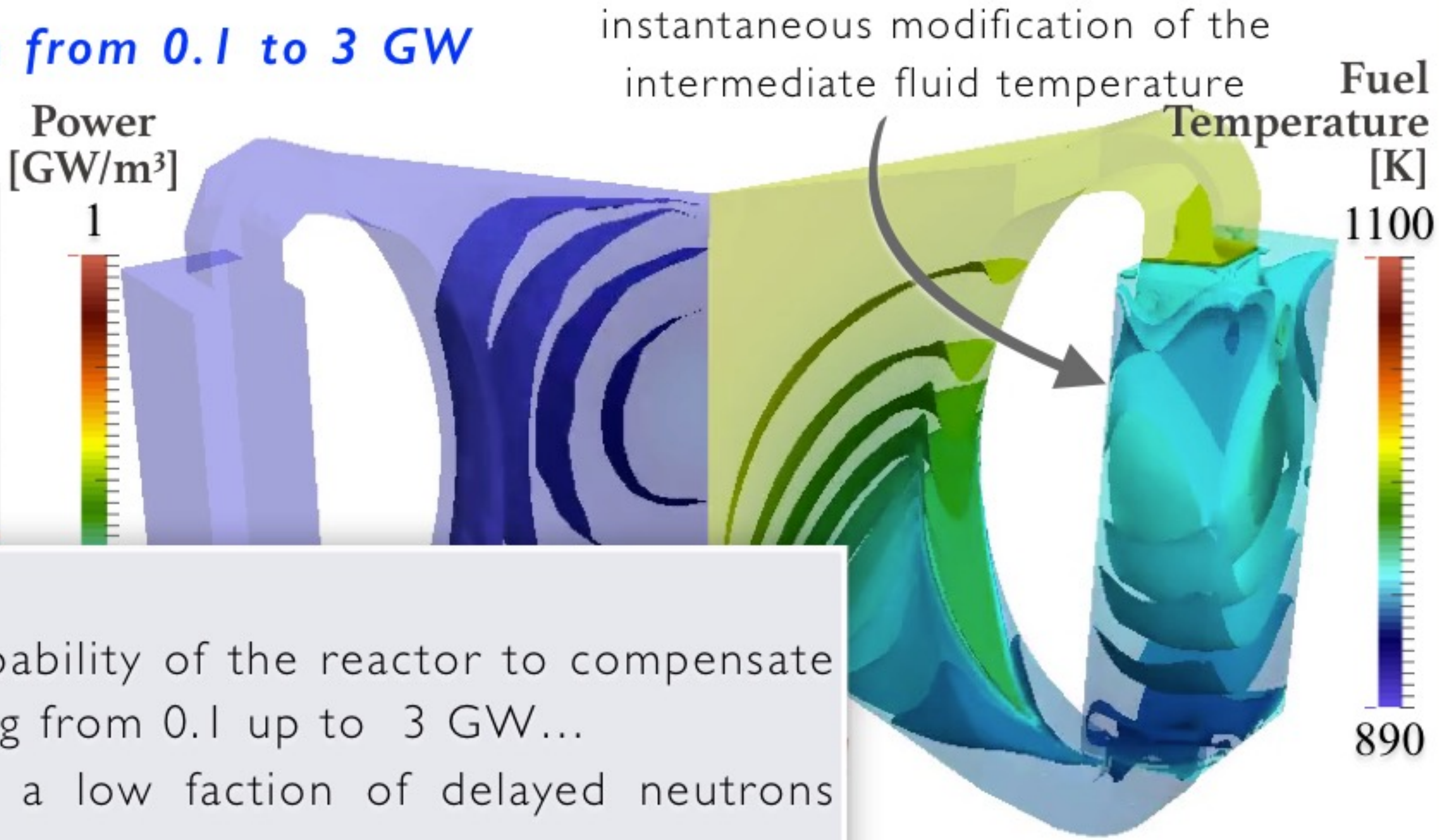


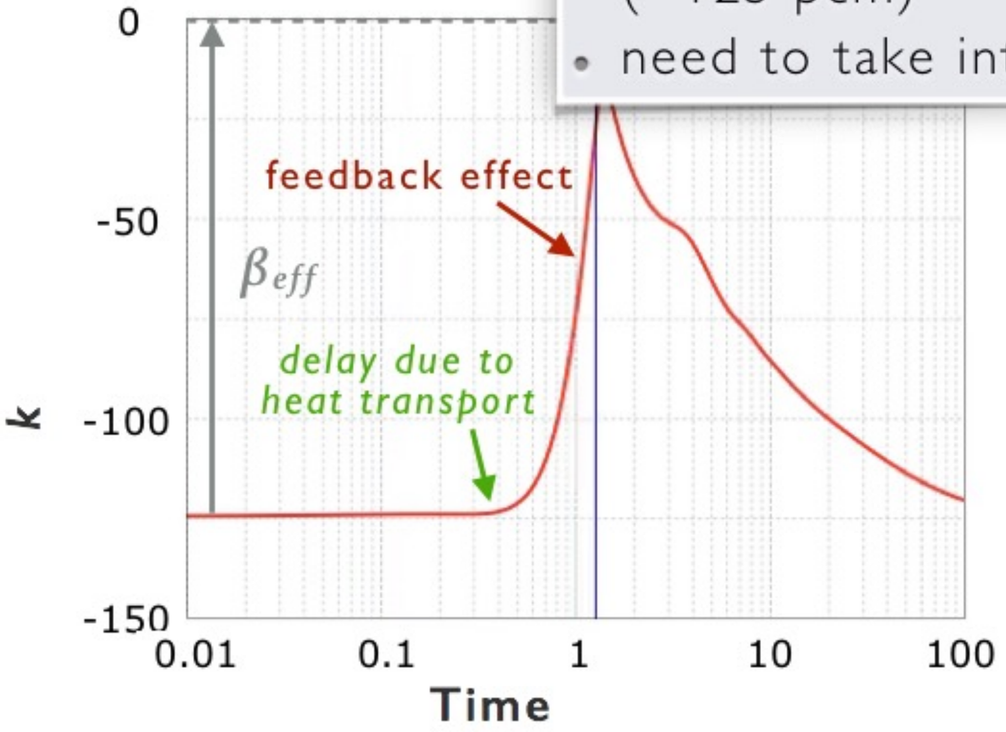


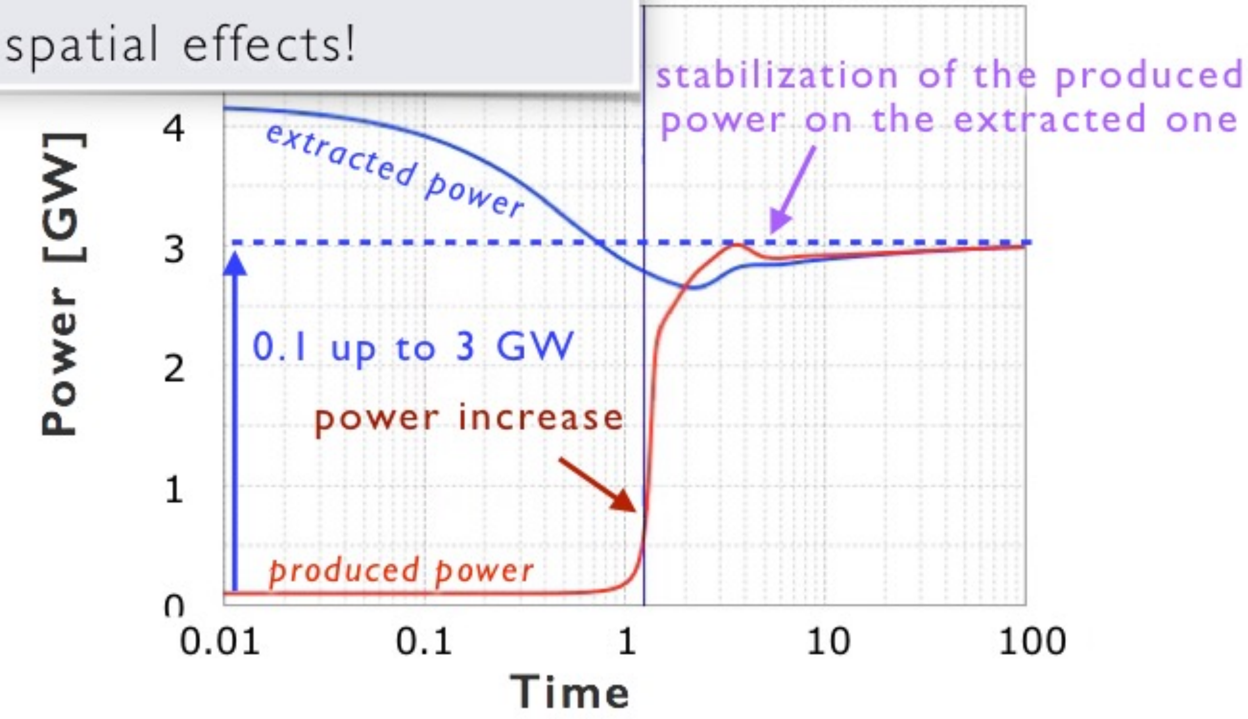
Overcooling accident of the MSFR

Overcooling: Instantaneous variation from 0.1 to 3 GW







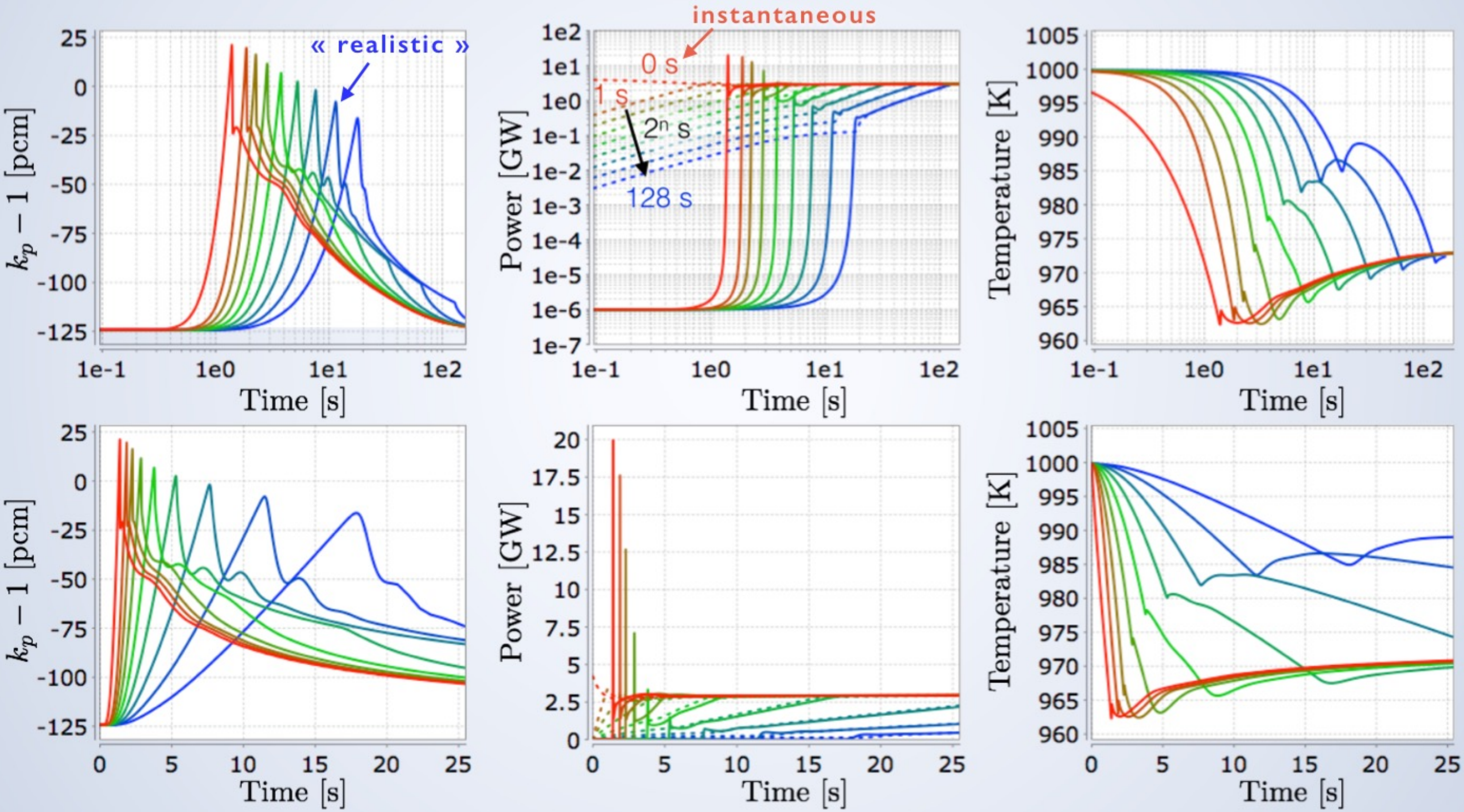


Conclusion:

- Very good capability of the reactor to compensate an over-cooling from 0.1 up to 3 GW…
- … even with a low faction of delayed neutrons (~125 pcm)
- need to take into account spatial effects!

Overcooling accident of the MSFR

Parametric study (realistic overcooling with inertia in the intermediate system)



[A. Laureau, et al. Transient coupled calculations of the Molten Salt Fast Reactor using the Transient Fission Matrix approach, *accepted in Nuclear Engineering and Design (2017)]*

Overcooling accident of the MSFR

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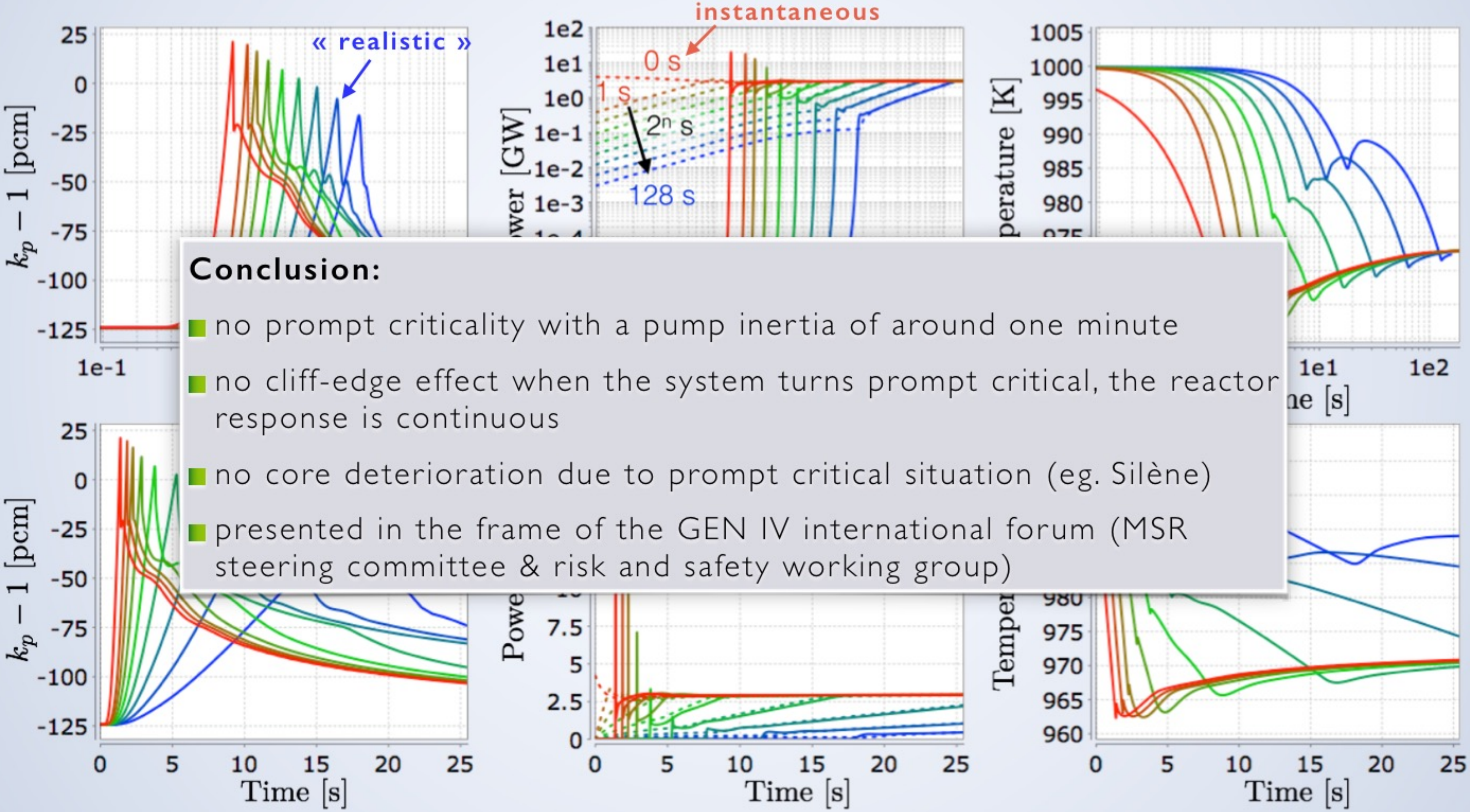


Conclusion:

- no prompt criticality with a pump inertia of around one minute
- no cliff-edge effect when the system turns prompt critical, the reactor response is continuous
- no core deterioration due to prompt critical situation (eg. Silène)
- presented in the frame of the GEN IV international forum (MSR steering committee & risk and safety working group)

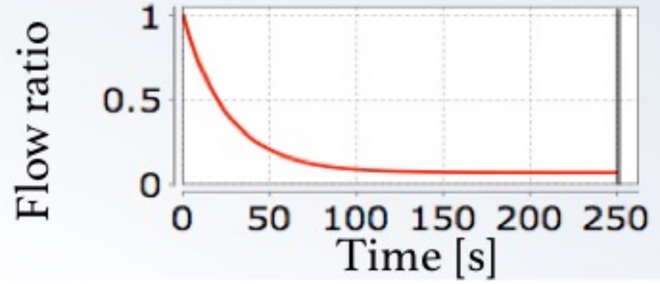
[A. Laureau, et al. Transient coupled calculations of the Molten Salt Fast Reactor using the Transient Fission Matrix approach, *accepted in Nuclear Engineering and Design (2017)]*

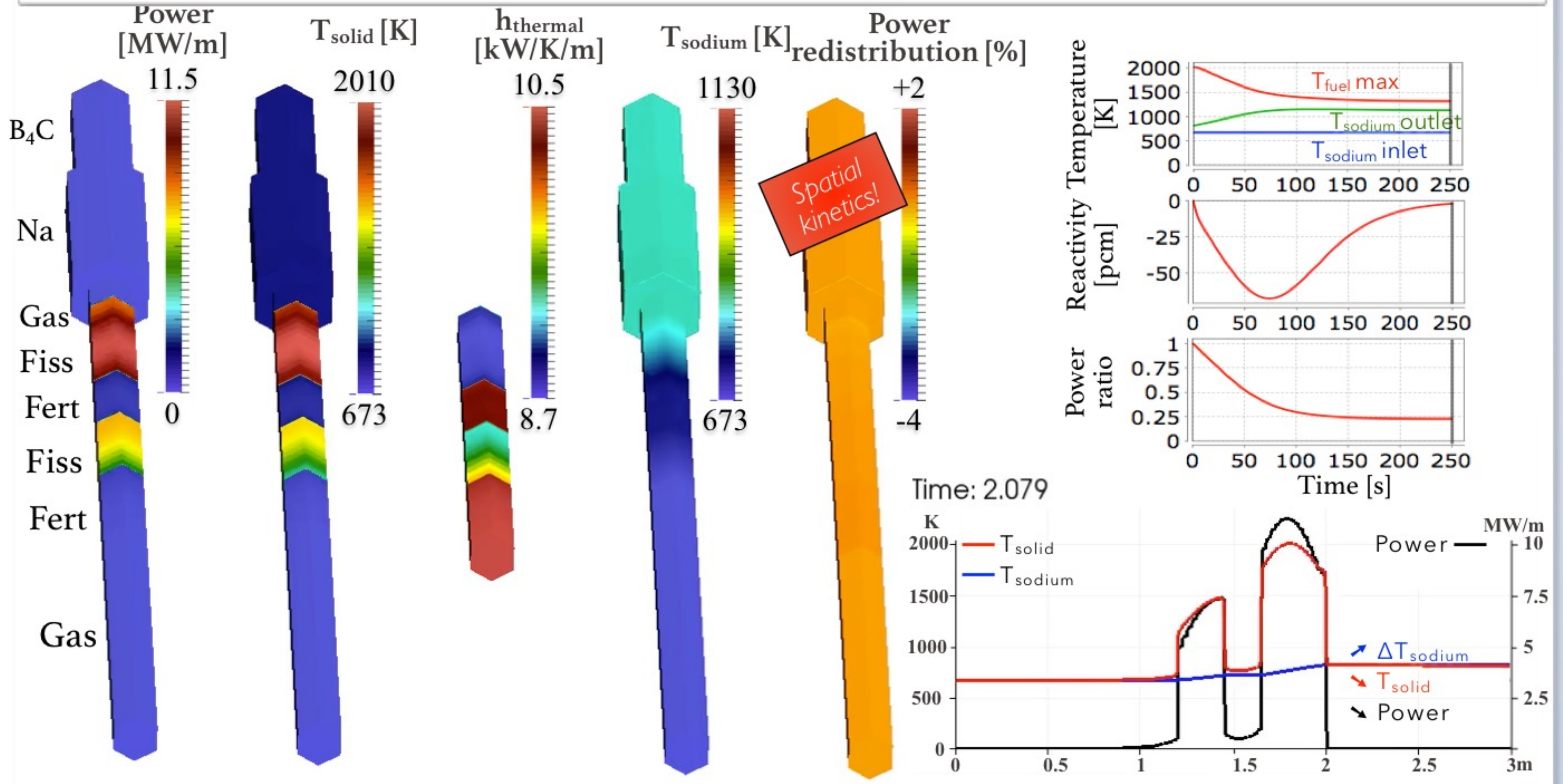
Unprotected loss of flow on ASTRID

Context: benchmark with IAEC (Israël) on the influence of the spatial kinetics on accidental transients in SFR

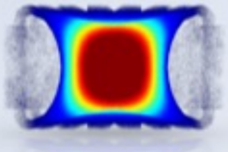
Scenario:

- beginning of cycle
- full power
- pump loss at t=0 and $T_{1/2}$=20 s
- minimum flow 7% (no boiling)

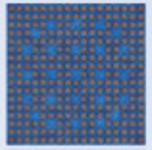




[A. Laureau, Y. Lederer et al. Transient coupled neutronics-thermalhydraulics study of ULOF accidents in sodium fast reactors using spatial kinetics: comparison of the TFM Monte Carlo and SN approaches. *proceeding accepted to the International Conference ICAPP'2017, Kyoto, Japan (2017)]*



CONCLUSIONS & PERSPECTIVES



ELABORATION OF INNOVATIVE SIMULATION TOOLS

Transient Fission Matrix (TFM): neutronic approach characterizing the spatial & temporal response

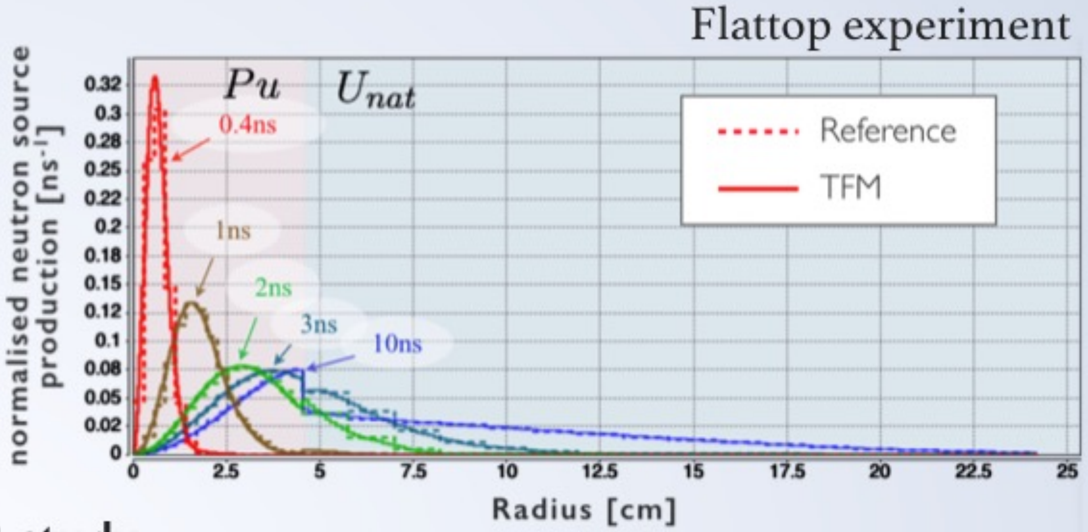
- different models depending on the studied phenomena
- from the nano-second up to minutes
- effective parameters calculation
- coupling to the thermalhydraulics
- reasonable calculation time
  ~ day on a office computer
- Monte Carlo "spirit" and precision

Power Water Reactor (EdF reactor) study

- modeling of phenomena with large impact of flux distribution
- kinetic study with control rod motion

Sodium cooled reactor ASTID study

- international benchmark on kinetic for safety study
- decoupling phenomena on large core
- identify biases in traditional calculation schemes
- improve simulation tools



MSFR study

- interesting reactor to develop new models
- promizing concept
  - selected by the GEN IV international forum
- researches on the safety and drivability
  - normal operation (load following)
  - accidents (over-cooling, reactivity insertion)
- development of a system code « fissions to electrons »
  TFM-OpenFOAM used as reference

NEEDS nucléaire énergie • environnement • déchets • société

SAMOFAR