

DE LA RECHERCHE À L'INDUSTRIE

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THE APOLLO3[®] SCIENTIFIC TOOL FOR SFR NEUTRONIC CHARACTERIZATION: CURRENT ACHIEVEMENTS AND PERSPECTIVES

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■ Introduction

■ Motivations

- ASTRID core description
- Neutronic Challenges in the ASTRID core design
- APOLLO3-SFR reference calculation scheme

■ Verification, Validation and Uncertainty Quantification (VV&UQ)

- General VV&QU philosophy
- APOLLO3 Verification & Validation
- Nuclear Data Verification & Validation

■ Conclusions and Perspectives

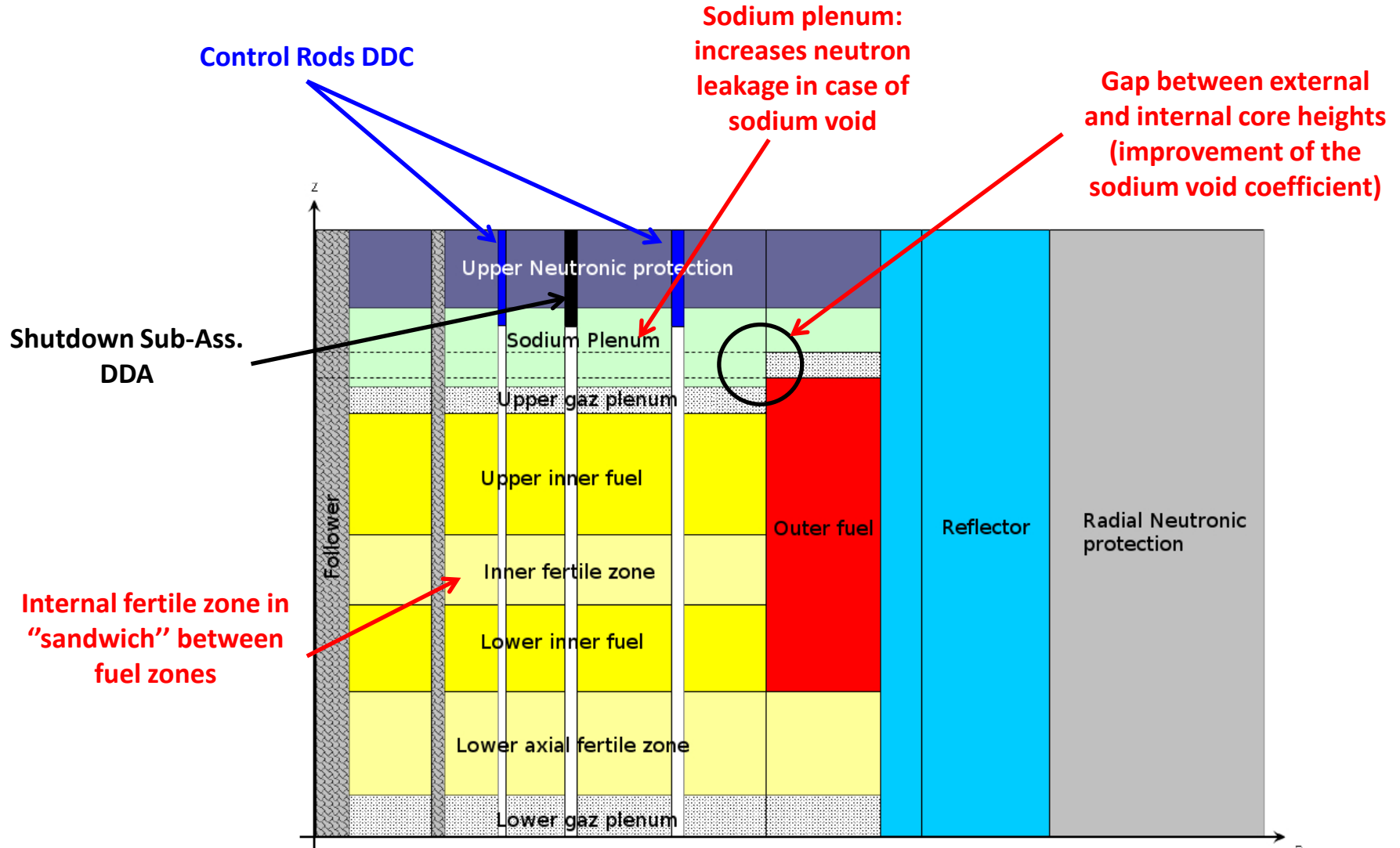
INTRODUCTION

- **Safety requirements are the driving force for the development of a new generation of simulation tools.** New laws on safety rules are requiring that current data and appropriate methods in use should be associated to uncertainties and a validity domain.
- **Novel reactors** either 3rd+ generation PWR cores and 4th generation fast neutron reactor **exhibit advanced features.**
- These advanced features require **new generations of simulation tools** and **APOLLO3[®]-SFR** is one of them.
- **ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration)** is a Sodium Fast Reactor design that will be France's Flagship 4th Generation Reactor.
- **Safety dossier will be built on best estimate results plus uncertainties** defined upon a series of research and developments which will be described in this presentation

MOTIVATIONS

- ❑ **ASTRID core description**
- ❑ **Neutronic Challenges in the ASTRID core design**
- ❑ **APOLLO3-SFR reference calculation scheme**

THE ASTRID CORE: (LOW SODIUM VOID CORE: CFV)



MOTIVATIONS

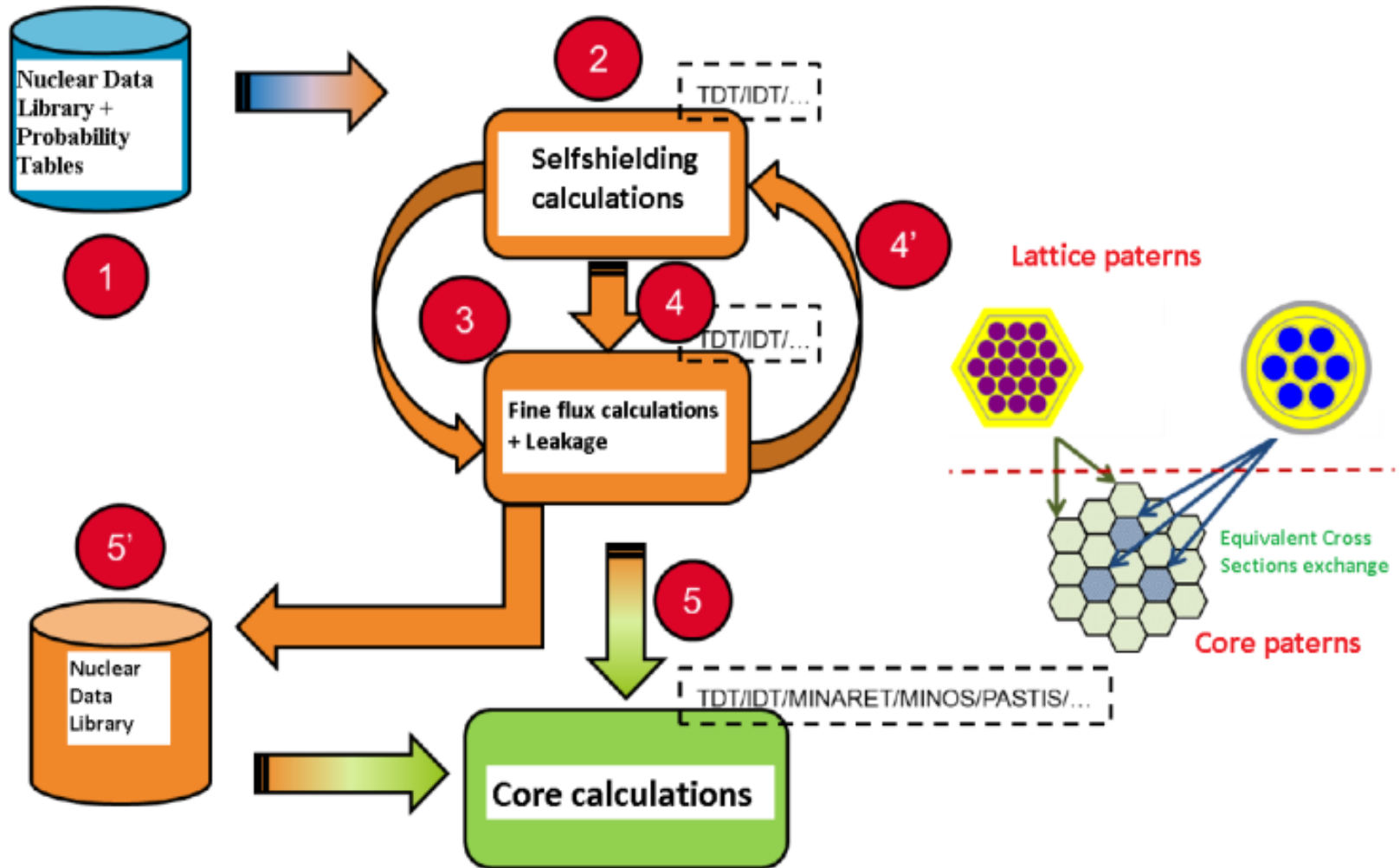
- ❑ **ASTRID core description**
- ❑ **Neutronic Challenges in the ASTRID core design**
- ❑ **APOLLO3-SFR reference calculation scheme**

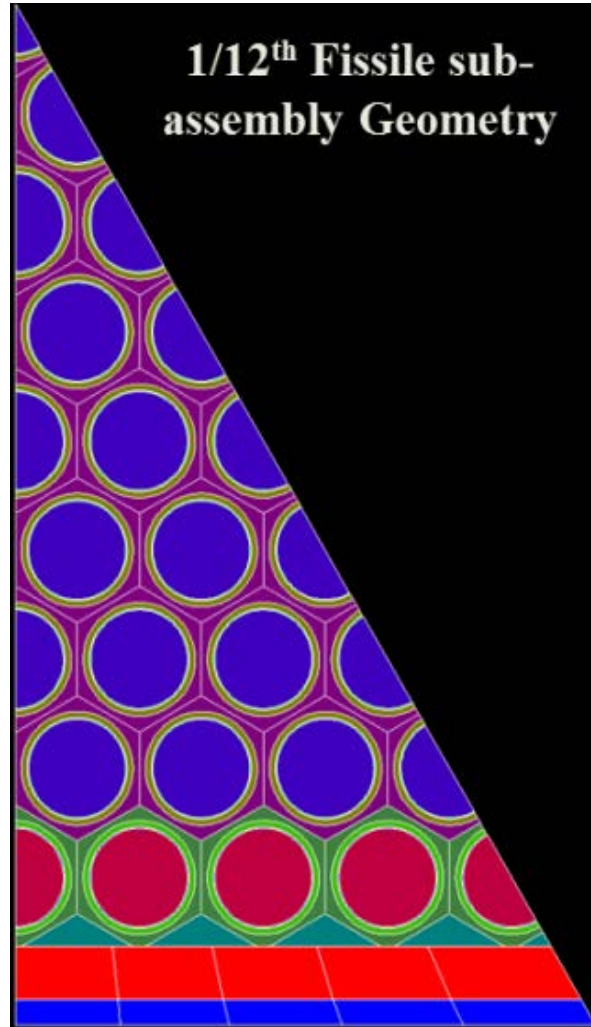
- **ASTRID's CFV core contains numerous radial and axial heterogeneities (in order to obtain a negative void coefficient) and interfaces which are challenging for deterministic calculation tools.**
- **Hence accurate modeling are required (3D simulations, parallel processing) like those being elaborated within the APOLLO® platform.**
 - several solvers (MOC, Sn, Pn...)
 - Innovative numerical methods: 2D & 3D-MOC, exact Pij, self-shielding methods (sub-group, Tone), MINARET
- **The APOLLO3-SFR package built with APOLLO3® solvers defines reference calculation schemes associated with a nuclear data library to calculate all neutronic parameters (critical masses, sodium void, Doppler coefficient, β_{eff} , etc...) together with certified biases and uncertainties derived from the VV&UQ process.**

→ Development and validation of a new code package, called APOLLO3-SFR (for Sodium Fast Reactor)

MOTIVATIONS

- ❑ **ASTRID core description**
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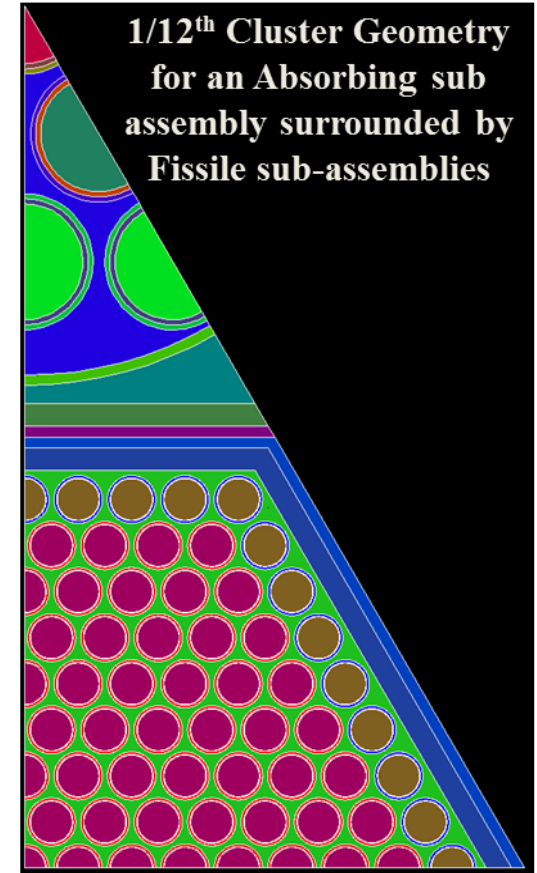
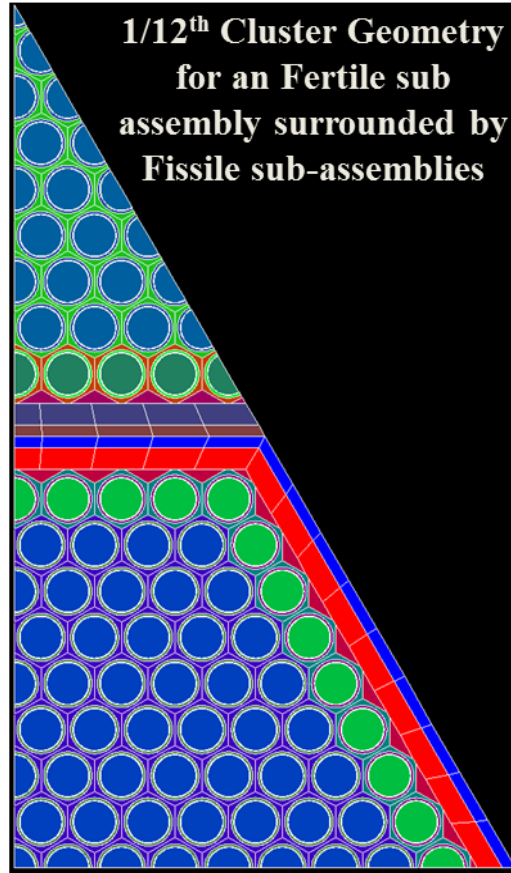
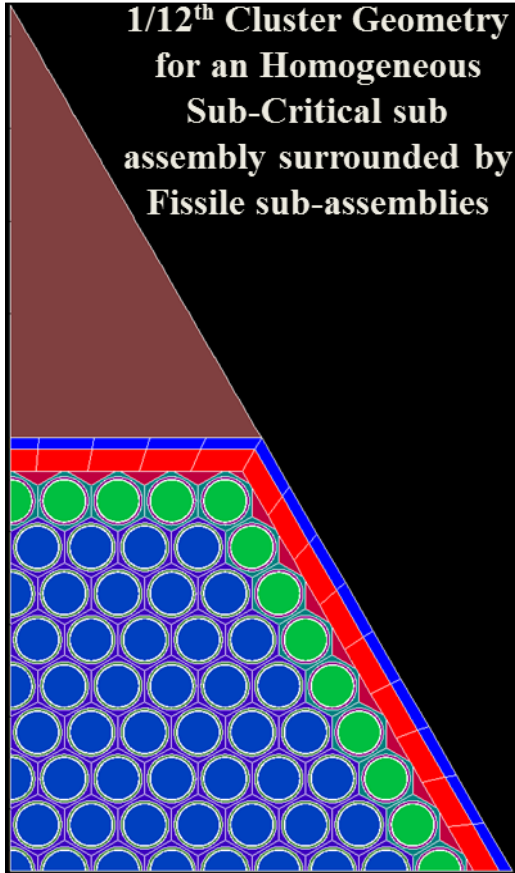
Calculation performed with the method of characteristics (TDT-MOC solver).

B1 heterogeneous algorithms implemented to better represent neutron leakage.

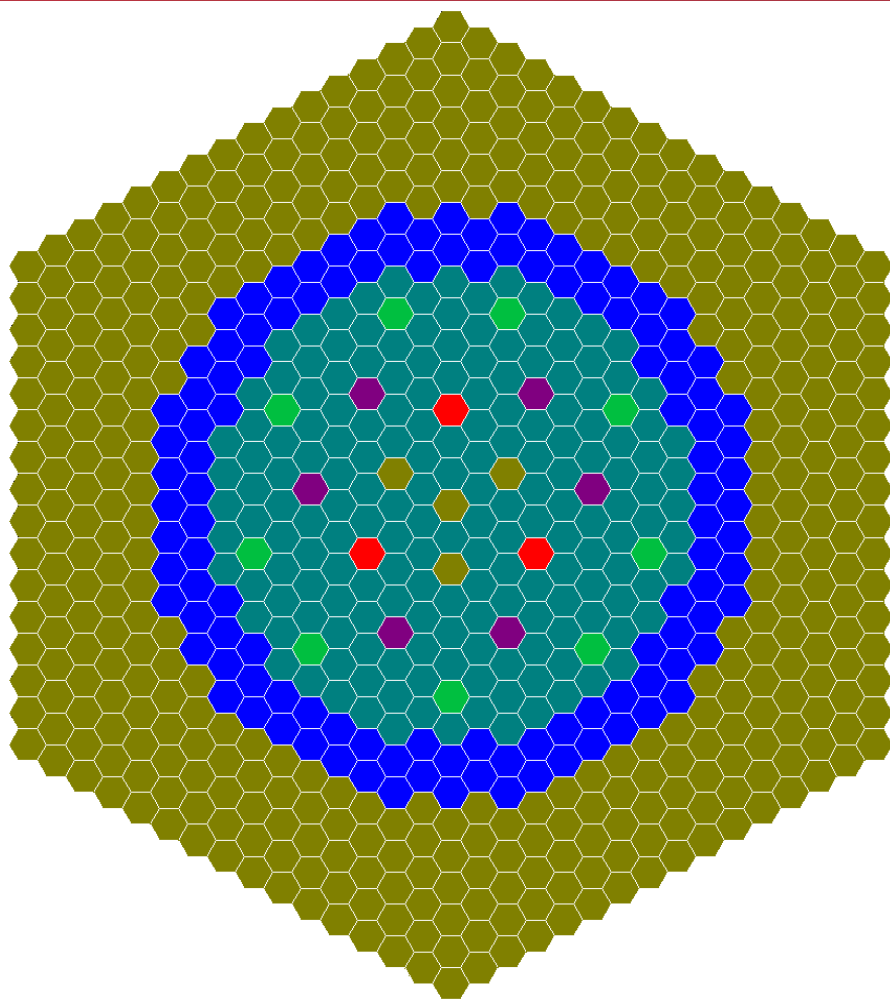
Symmetry properties of the sub-assemblies (description of only 1/12th of the full sub-assembly)

Fuel sub-assemblies are described with reflective boundary conditions.

This TDT-MOC flux solver is coupled with the subgroup method using the TDT-CPM solver (exact P_{ij}) with the same geometry.



Sub-critical sub-assemblies (fertile, control rod and structural media) are modeled by clusters



- Internal Fuel Sub-Assembly (C1)
- External Fuel Sub-Assembly (C2)
- Reflector Sub-Ass. + Followers Sub-Ass. (SPNL)
- Internal Control Rod Sub-Ass. (DDC)
- External Control Rod Sub-Ass. (DDC)
- Shutdown Sub-Ass. (DDA)

Core calculations are performed with the 3D Sn core solver MINARET using an unstructured conforming triangular spatial mesh (Discontinuous Galerkin Finite Elements)

Verification, Validation and Uncertainty Quantification (VV&UQ)

- ❑ General VV&QU philosophy
- ❑ APOLLO3 Verification & Validation
- ❑ Nuclear Data Verification & Validation



Verification

- Correct algorithms and numerical operations ;
- Fix programming errors and bugs ;
- Ensure non-regression.

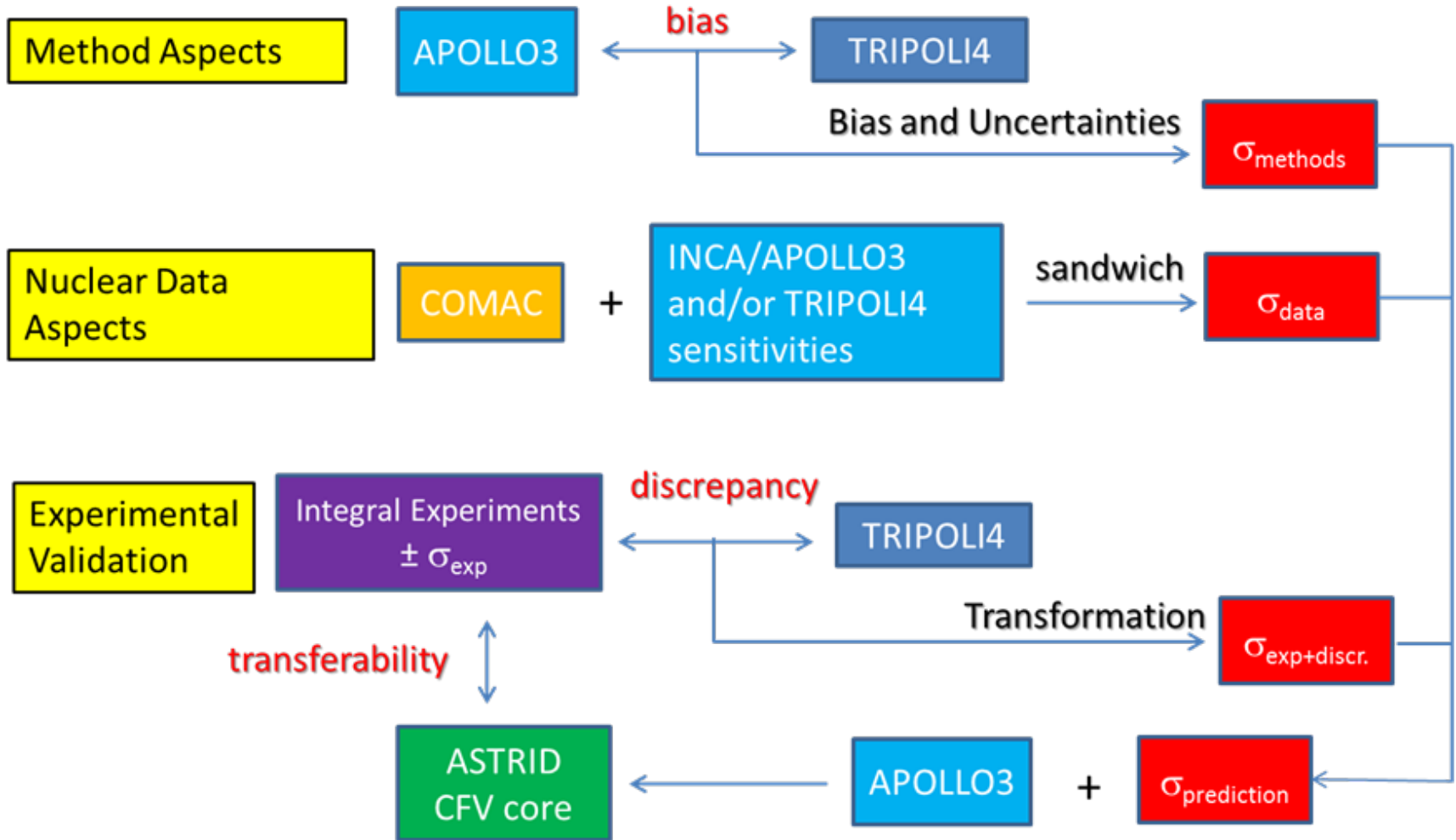
Numerical Validation

- Ensure capacity of representing physical phenomena ;
- Validation of main functionalities ;
- Optimization of calculation routes.

Experimental Validation

- Comparison to measurements ;
- **Resulting error is the combination of model biases and nuclear data uncertainties.**

BIAS AND UNCERTAINTIES FROM VV&UQ



Verification, Validation and Uncertainty Quantification (VV&UQ)

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New methodology for V&V (and associated mathematical background) with:

- 1/ **Evaluation of model biases** (independently from nuclear data uncertainties) ;
- 2/ Associate biases to **each relevant APOLLO3® model approximation** used in assembly/core calculation routes (i.e. error propagation from assembly to core) ;
- 3/ **Estimate model biases changes** by correlating output and input parameters ;
- 4/ Apply it to the assembly/core calculation routes of **all kind of reactor cores** (SFR/LWR/MTR) – The **physics** (and **geometries**) of these cores are **different**.

Calculation Step	Functionality	Value
Sub-assembly Calculation	Scattering Anisotropy	P1
	Fission Spectrum incident energy macrogroup	4
	Energy Grid	1968 Groups
	Self-shielding	Sub-group method
	Flux Solver	TDT-MOC (Method of Characteristics)
	Leakage Treatment	B1 Heterogeneous
Core Calculation	Energy Grid	33 Groups
	Flux Solver	MINARET (S_N)

Assembly bias

	Reactivity Bias (pcm)	
Assembly	Core 1	Core 2
Global	+ 10	+ 17
Self-Shielding	+ 14	+36
Nuclear Data Treatment	- 1	- 11
TDT-MOC	- 2	- 7

Bias for different core situations

Situation	Nominal	Rod inserted	Voided	Rod Efficiency	Void Effect
<i>Reactivity Biases (pcm)</i>	<i>+ 75</i>	<i>+ 87</i>	<i>+ 143</i>	<i>+ 13 (0,5 %)</i>	<i>+ 69 (8,7 %)</i>

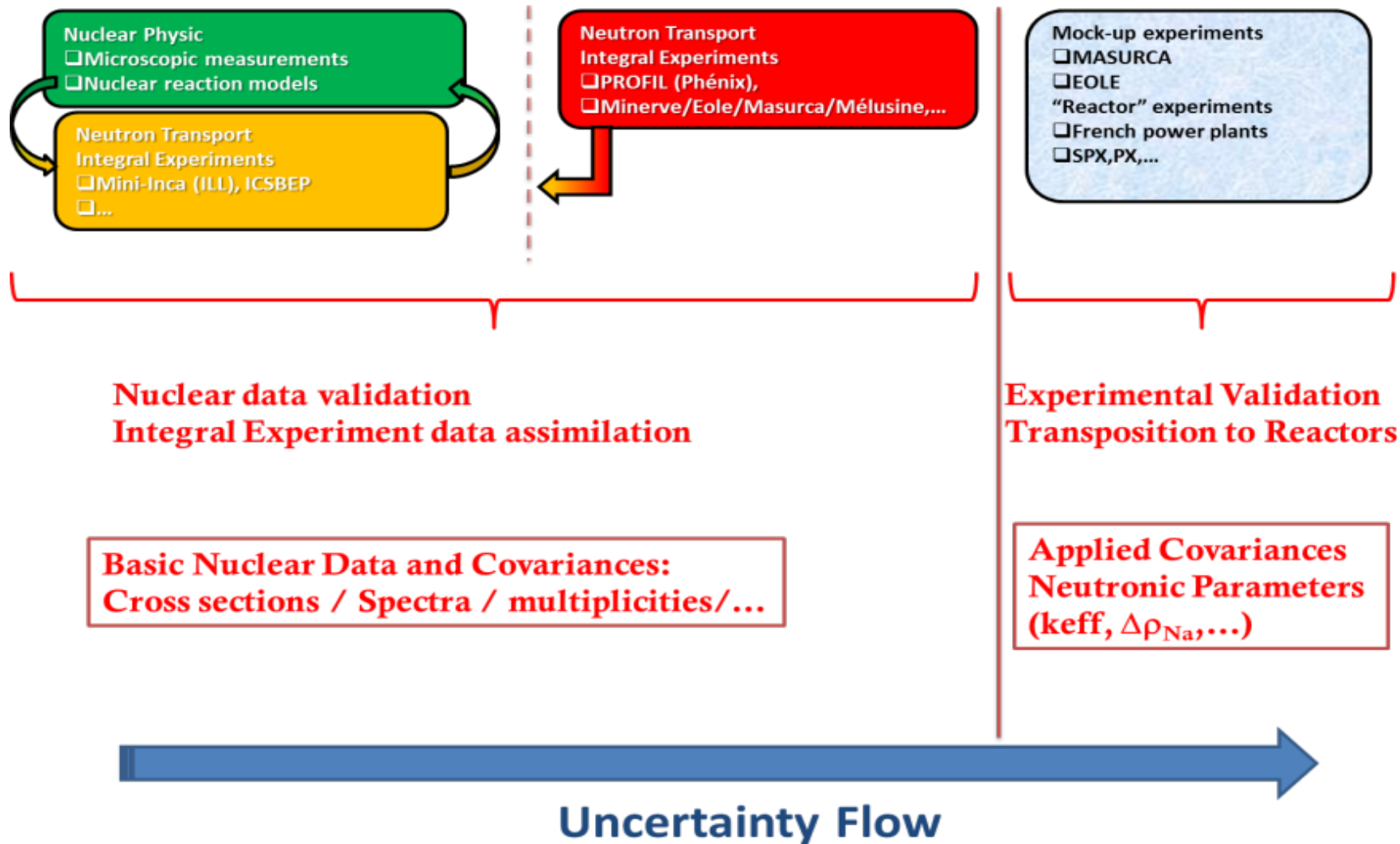
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COMAC V1 is the covariance file based on JEFF3.2 evaluation processes. It uses nuclear models and differential measurements.

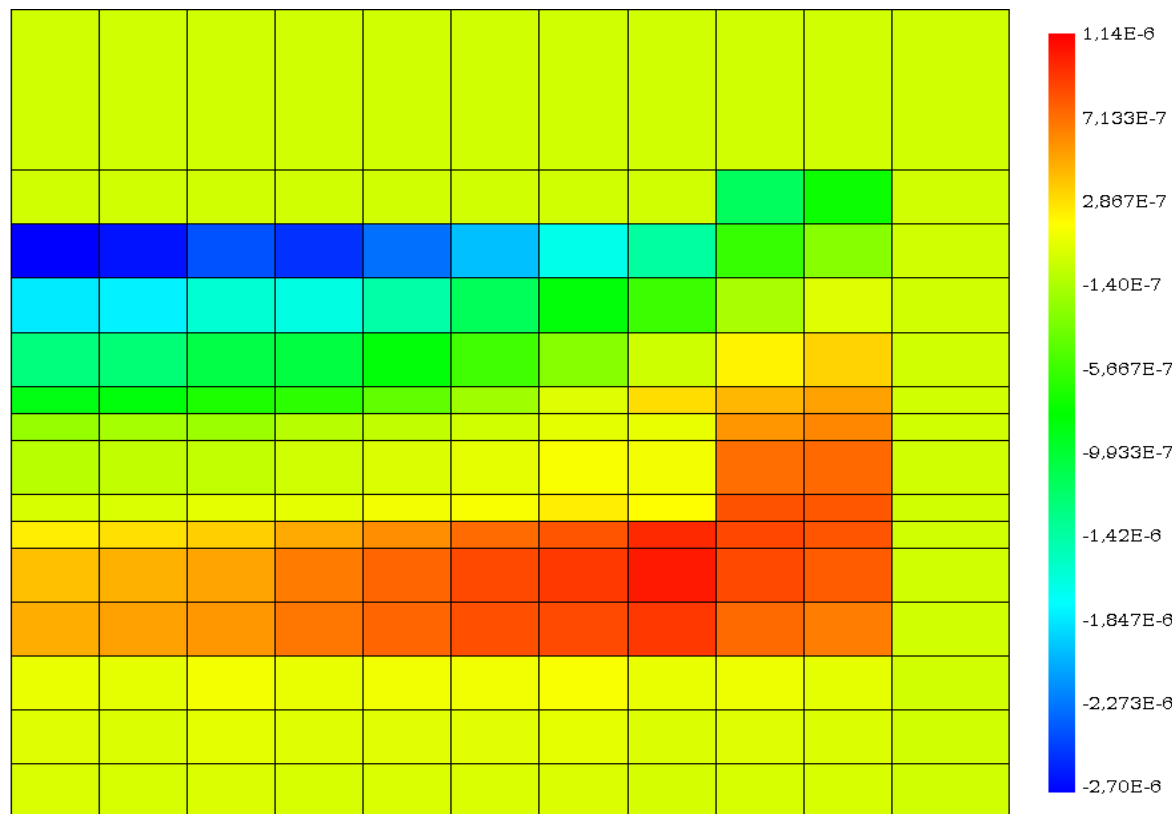
The propagation of these uncertainties to the ASTRID core characteristics is performed using the so-called sandwich formula using sensitivities & covariance matrix.

ASTRID core characteristic	Nuclear Data Uncertainties (1σ) (COMAC-V1)
keff (pcm)	804
Sodium Void (\$)	0.28
β_{eff}	3.4%
Doppler Effect	2.7%
Rod Worth	2.5%
Ratio $P_{\text{max}}/P_{\text{tot}}$	2.5%



Component	Void Effect	Central Component	Leakage component
$\Delta\rho_{Na}$ (pcm,\$)	-732 (-1.9 \$)	879 (2.3\$)	-1611 (-4.3\$)

The ASTRID core achieves negative sodium void reactivity effect through the existence of a large leakage component



The leakage component introduces a change of the fission sources which is significant especially in the upper and lower parts of the core region.

Experimental validation required with the GENESIS programme (at BFS and in MASURCA)

Absolute difference between reference and void volumic fission sources for the leakage component

CONCLUSIONS & PERSPECTIVES

- ❑ **ASTRID** (Advanced Sodium Technological Reactor for Industrial Demonstration) is a Sodium Fast Reactor design that will be **France's Flagship 4th Generation Reactor**.
- ❑ The **APOLLO3-SFR** package built with APOLLO3® solvers defines **reference calculation schemes** associated with a nuclear data library together **with certified biases and uncertainties** derived from the VV&UQ process. This VV&UQ process incorporates numerical validation, a-priori uncertainties based on nuclear data covariances as well as experimental validation mainly from MASURCA, a fast mock-up reactor, located at CEA Cadarache.
- ❑ **Innovative numerical scheme** based on the use of the sub-group method in a fine energy mesh together with a **2D method of characteristics** for calculating the heterogeneous angular fluxes (real and imaginary with the B1 heterogeneous method) enables a refine core sub-assembly calculation. Cross sections are smeared and condensed on the flux moments for use in the **Sn 3D core calculations**. V&V have shown the **high level of accuracy** obtained.
- ❑ A future programme called **GENESIS will be performed in support to the prototype ASTRID to validate the CFV core specificities** such as sodium void reactivity, control rod worth, power map distribution. A part of the GENESIS experimental program contains integral experiment underway at the BFS facility.

THANKS FOR YOUR ATTENTION

ANY QUESTIONS ?