A Dummy Core for V&V and Education & Training Purposes at TechnicAtome: In and Ex-Core Calculations

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Abstract. Core calculations, both deterministic and stochastic, constitute every-day activities in TechnicAtome (formerly AREVA TA) Core Physics Department for the design and operation of nuclear research facilities. Developments of calculation schemes but also methods are a key issue.
In order to enable objective comparisons in methodologies, TechnicAtome has designed a dummy core. This non-existing core provides a mixture of some common features found in research reactors throughout the world. Its characteristics enable TechnicAtome to validate and qualify both calculation and computational techniques on a single-of-a-kind core with significant heterogeneities, thus extending the validation to almost any kind of small light-water reactor.
In addition to comparing methodologies, this dummy core is also used for non-regression tests performed between incremental versions of home-made calculation schemes. It is finally also used for Verification and Validation purposes, when it comes to comparing different codes on one given object.
This paper presents the fictitious core itself, with its description, from the standard FAs (Fuel Assemblies), to the hafnium-plate controlled assemblies and the full core with its in and ex-core features. It then presents the main results in terms of reactivity and power distribution, both for a basic 2D infinite periodic assembly and for the full core.
The paper also describes a methodology developed to calculate ex-core detector responses. This method consists in the TRIPOLI4© "Green Function" feature to determine the importance of each assembly concerning detector signal. This knowledge then allows to instantly predict the detector response in case of different control rod configurations. Comparisons with full simulations are performed.
Finally, the paper illustrates the sketch of TRIPOLI4© and GEANT4 unification at geometry level with further outlooks.

1. Introduction

Core neutronic calculation calls for frequent and intensive Verification and Validation processes (V&V). For this key-issue purpose, as well as for methods and techniques development, TechnicAtome has designed a dummy core by mixing some features commonly found in existing Research Reactors (RR) throughout the world. Further features can also be added such as a reflector and experimental devices.

This dummy core has many goals:

- It enables objective comparisons between methodologies,
- Its significant heterogeneities enable the validation and qualification of both calculation and computational techniques, thus extending the validation to almost any kind of small light-water reactor,
- In addition to comparing methodologies, this dummy core is also used for non-regression tests performed between incremental versions of home-made calculation schemes,
- It is an easy Education and Training object,
- It is finally also used for V&V purposes, when it comes to comparing different codes on one given object.

This paper first presents this fictitious core and its main performances, benchmarked between two Monte-Carlo codes. It then presents an application on the development of a methodology to calculate ex-core detector responses with the TRIPOLI4© Green Function feature.

2. Dummy core description

In this section, the fictitious core itself is presented, with its description, from the standard FAs (Fuel Assemblies), to the hafnium-plate controlled assemblies and the full core with its in and ex-core features.

The fuel pattern consists of a succession of standard U_3Si_2 slabs composed of planar JHR-type fuel and contained in aluminum cladding (see FIG. 1). The choice has been made to take JHR (Jules Horowitz Reactor) plates and water gap thickness: 0.061 cm for fuel plate, 0.137 cm for cladding and a water gap of 0.2 cm. Other dimensions concerning the fuel pattern are represented in FIG. 1.

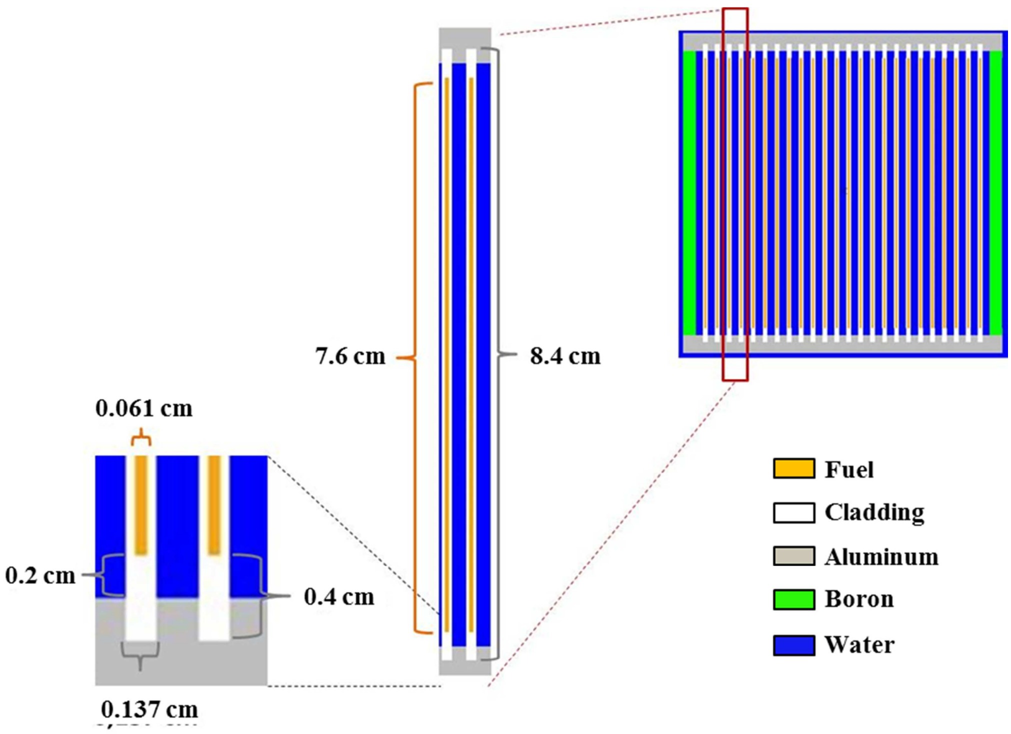


FIG. 1. Cladding and fuel geometry (fuel plate and assembly)

Each fuel plate is inserted in a 0.2 cm aluminum crimping groove allowing to place a cadmium wire, if required for reactivity control. Natural boron (1% mass fraction, in green in FIG. 2.) is chosen instead for simplicity and placed as a burnable poison in the external part of the aluminum rack to control the reactivity excess.

There are two types of fuel assemblies (FA): standard FA and hafnium-plate controlled assembly. Again, Hf is chosen for simplicity and is representative of RR control rods. The standard FA is made of 24 fuel plates while control FA only contains 18 plates (see FIG. 2). The 6 missing plates are replaced by two hafnium control rods, 6 mm wide. The dimensions of both assemblies are identical and are 9 x 9 cm in the xy plan, and the active part is 80 cm high. Furthermore, the fuel array grid is identical in both FAs.

Both FAs can then be stacked in a fuel lattice in order to build up a core. FIG. 3 and FIG. 4 show two examples of cores used in our education, training and V&V studies: a larger one (64 FAs) and a smaller one (32 FAs). The 64 FAs core will be considered in the following study.

FIG. 5 completes the description of the system by presenting the ex-core environment. This model is used in order to study detector responses outside and far from the core itself. The multi-layer structure is meant to be representative of a PWR, with successive light water, stainless steel and lead acting as radiation shielding.

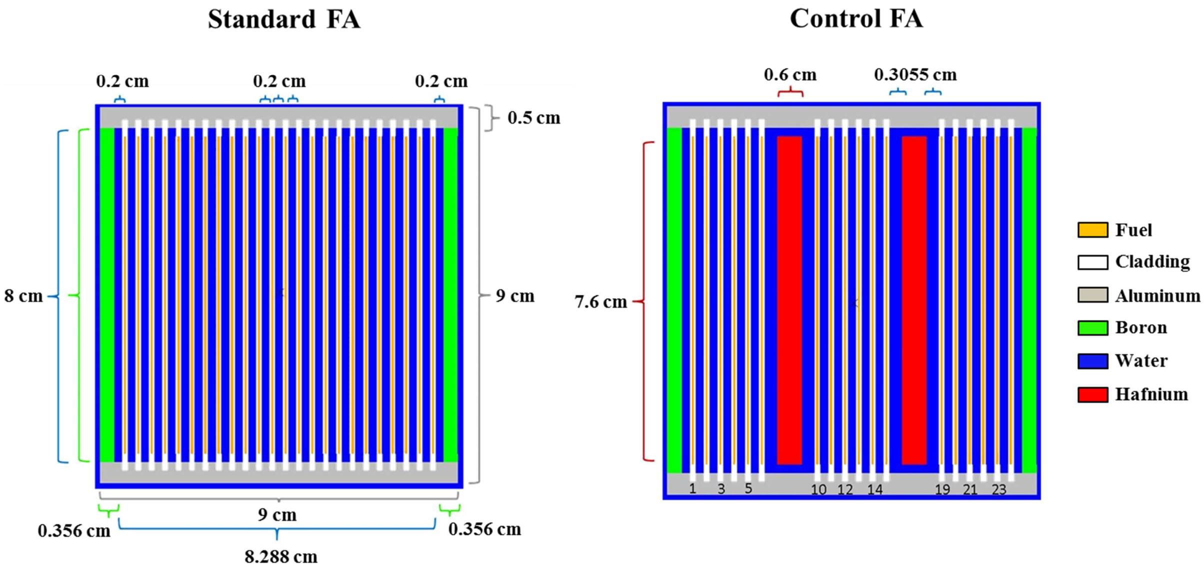


FIG. 2. Cross section geometry for both standard (left) and control assemblies (right)

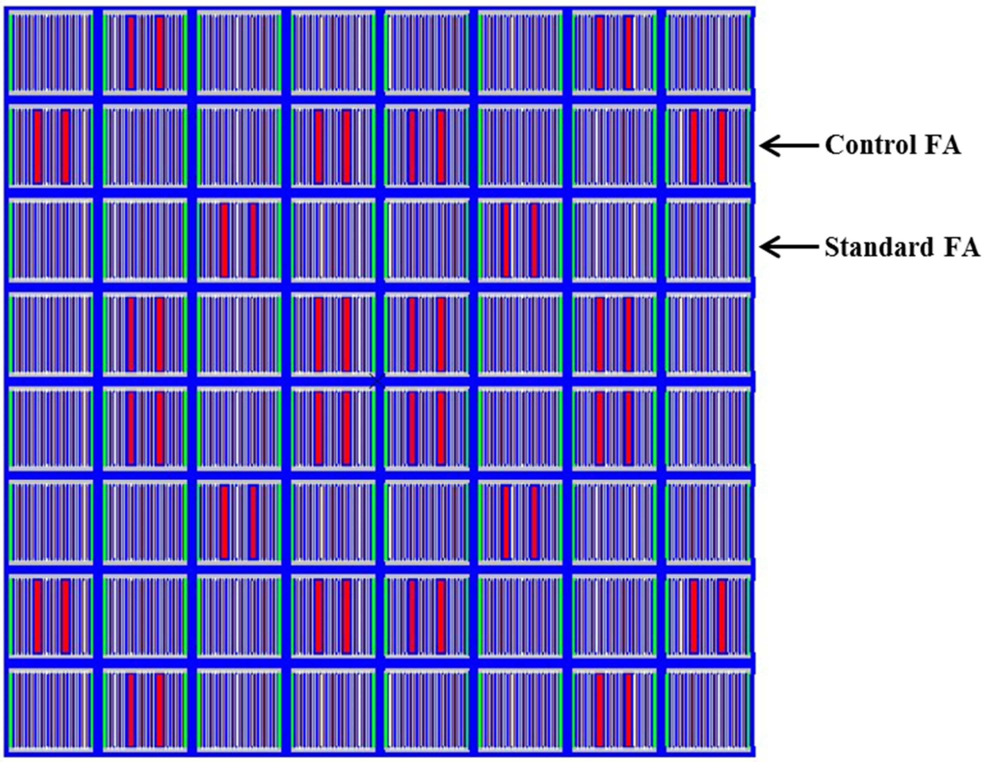


FIG. 3: Cross section of an entire 64 FA core (8×8 assemblies)

Outside the pressure vessel (see FIG. 5), boron thin layer detectors, such as those described in [1], are placed in order to detect leaking neutrons from the core. The response on these detectors is studied in section 4.

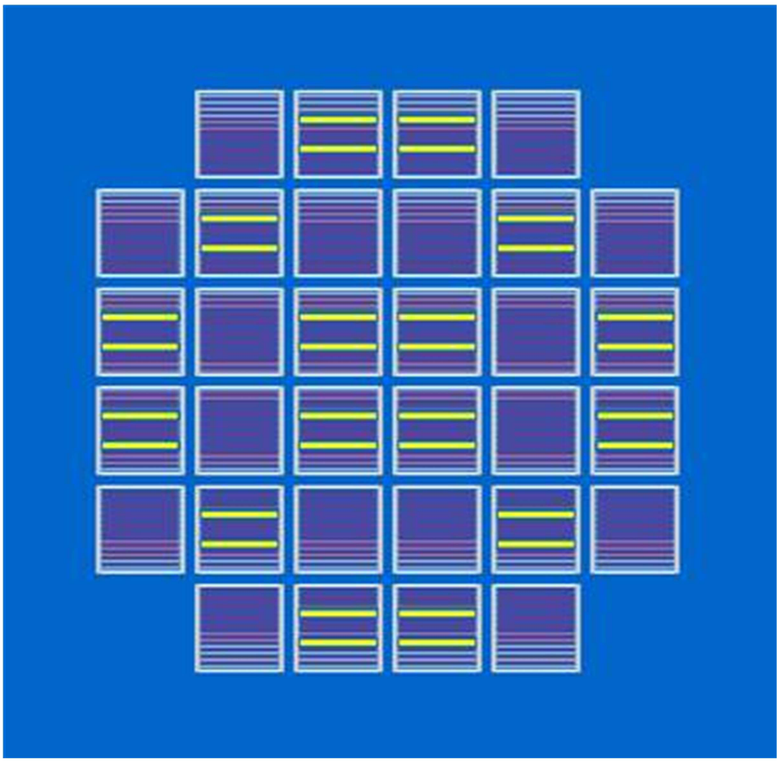


FIG. 4: Cross section of an entire 32 FA core

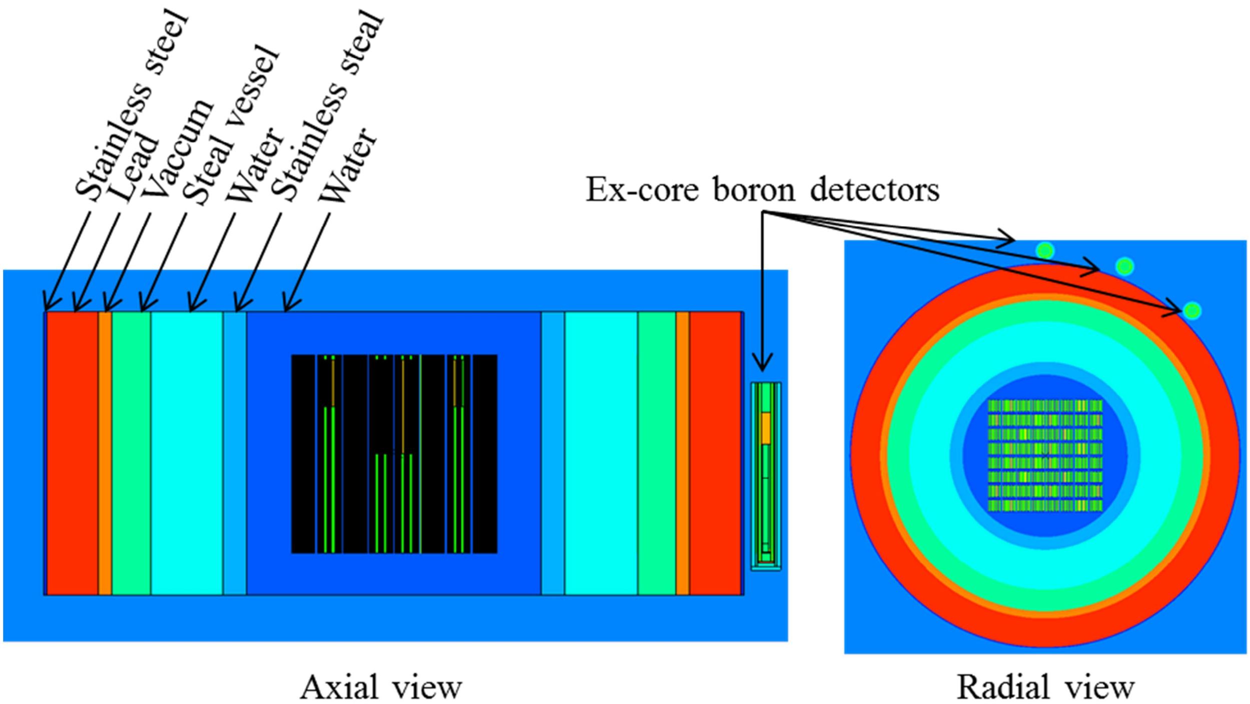


FIG. 5. Vertical and horizontal cross sections of the core in its environment

The neutronic performances of the assemblies can then either be studied in 2D infinite lattices such as in [2] or in a finite 3D core configuration such as in the next sections.

3. Main neutronic results concerning the fictitious core

This section presents a brief description of the 64 FAs core studied through its main neutronic performances. Results are discussed in terms of reactivity and both neutron flux and plate power distributions. Reference calculations use Monte-Carlo TRIPOLI4.10© code, developed by CEA [3]. Some configurations are also benchmarked with MCNP6 [4] for comparison. In both cases, geometry and material models, as well as output meshes (when used) are identical. All calculations are performed with JEFF3.1 libraries at 300K (or otherwise if stated) and simulate between 2.10^7 and 6.10^7 neutrons.

When required, reactivity worth is calculated using the following formula:

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

3.1. Multiplication factor for different configuration (control rods and soluble boron)

Four configurations are detailed below:

- All rods in: all hafnium control rods are inserted in the core at fuel bottom level
- All rods out: all hafnium control rods are in their uppermost level (fuel top)
- 1000 ppm of soluble boron is diluted in the water moderator, all rods out
- Criticality

In the latter, criticality is simulated by inserting some rods with a given pattern detailed in FIG. 6. Both fuel and Hf absorbers height being 80 cm, the figures result in absorber being inserted at z = 0, z = 30, z = 50 or z = 80 cm (see FIG. 6).

TABLE 1. K-eff values for TRIPOLI4© and MCNP6 calculations in different configurations for the 64 FAs core

Configuration	TRIPOLI4 k-eff	TRIPOLI4 σ	MCNP6 k-eff	MCNP6 σ	discrepancy (pcm)
All rods in	0.92149	*18*	0.92046	*19*	-121
All rods out	1.19105	*16*	1.19084	*19*	-15
All rods out + 1000 ppm boron	1.05583	*15*	1.05574	*20*	-8
Critical	1.00080	*15*	0.99881	*19*	-199

TABLE 2. Absorber efficiencies for TRIPOLI4© and MCNP6 for the 64 FAs core

Configuration	TRIPOLI4	MCNP6	discrepancy (%)
Control rods efficiency (pcm)	24560	24667	0.43%
Boron efficiency (pcm/ppm)	10.8	10.7	-0.06%

TABLE 1 shows k-eff values calculated both with TRIPOLI4.10 and MCNP6. TABLE 2, on the other hand, shows control rod and boron efficiencies which derive from TABLE 1. All values show excellent agreement, which verifies the TRIPOLI4© model for the next section. Rod and boron efficiencies bear orders of magnitude which are perfectly consistent with common light water reactors. Once again~~,~~ this fictitious core is representative.

3.2. Moderator temperature coefficient

TABLE 3. Temperature feedback for the 64 FAs core calculated with TRIPOLI4©

Configuration	k-eff	σ	Moderator temperature coefficient (pcm/K)
All rods in, 300K	0.92149	18	-6.5
All rods in, 324K	0.92016	18	

TABLE 3 provides the effect on reactivity of a 24 °C increase in light water (both the moderator and the reflector). Feedback is unsurprisingly negative, with a rather low value which is consistent with small light water reactors physics. Subsequent studies could reveal that this figure is the sum of two opposing effects: a stronger negative feedback for the water internal to the core, added to a smaller but positive contribution of the water in the reflector.

3.3. Flux and power distribution

Further studies of the 64 FAs core were performed in the critical state. Criticality was simulated with partial insertion of Hf absorbers, accordingly with FIG. 6.

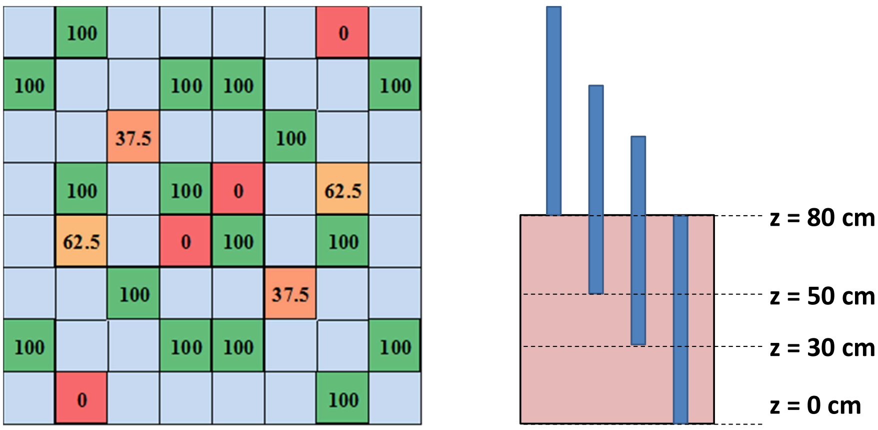


FIG. 6. Insertion percentage and corresponding altitude of each control rod leading to a critical state

Complete flux and power distributions are obtained with the EXTENDED_MESH feature in TRIPOLI4©, the equivalent of the FMESH feature in MCNP6. Both meshes are 5x5x800 mm^3 lattices.

Neutron fluxes are integrated over 3 energy groups:

- Thermal: 0 – 0.625 eV
- Epithermal: 0.625 eV – 1 MeV
- Fast: 1 – 20 MeV

FIG. 7 through FIG. 10 show the results, both absolute (TRIPOLI4©) and relative (MCNP6 vs. TRIPOLI4©). Flux behaviors are consistent with those of light water reactors. Thermal flux is low in the fuel plates because of U-235 absorption for fission. It increases in water channels and goes through an optimum a few cm inside the reflector.

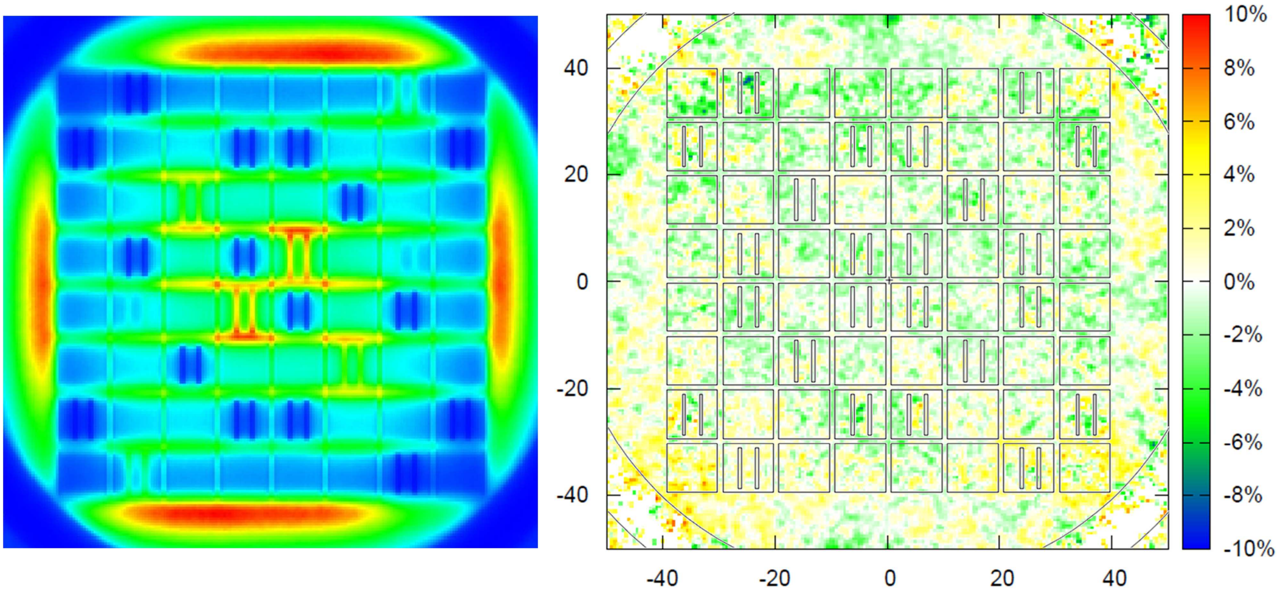


FIG. 7: Thermal flux in the critical state (TRIPOLI4©, left) and comparison MCNP6 vs. TRIPOLI4© (%, right)

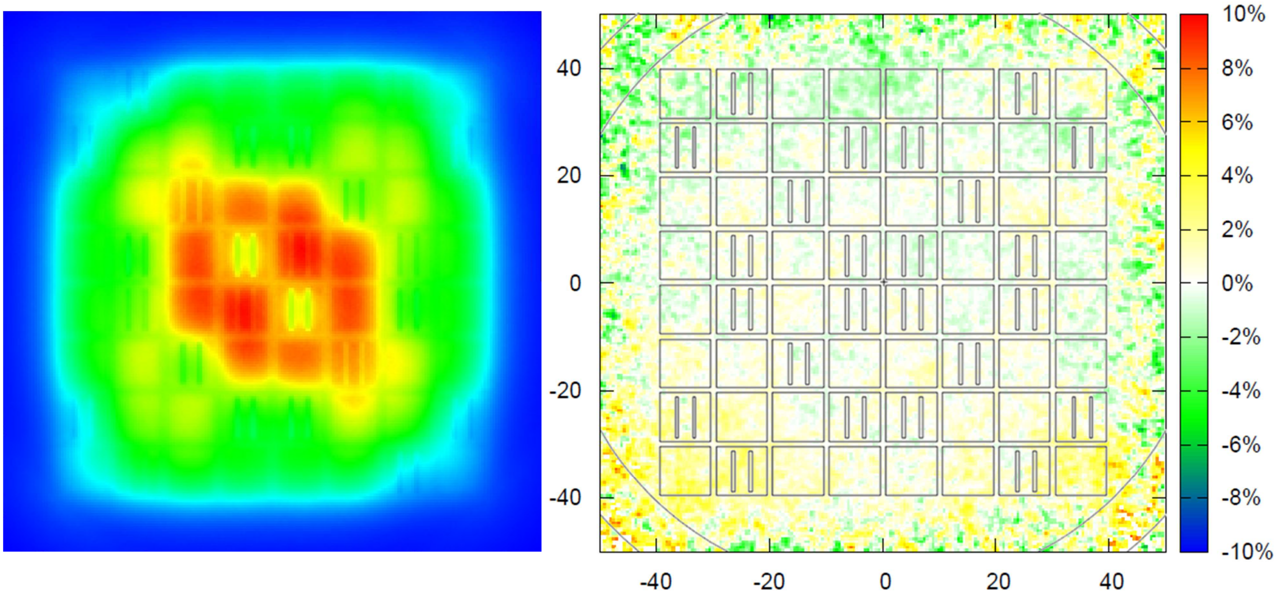


FIG. 8: Epithermal flux in the critical state (TRIPOLI4©, left) and comparison MCNP6 vs. TRIPOLI4© (%, right)

On the other hand, epithermal and especially fast flux tend to decrease with distance from the center of the core. Fast flux is extremely low in the reflector. Fission thermal power distribution more or less matches that of the fast flux.

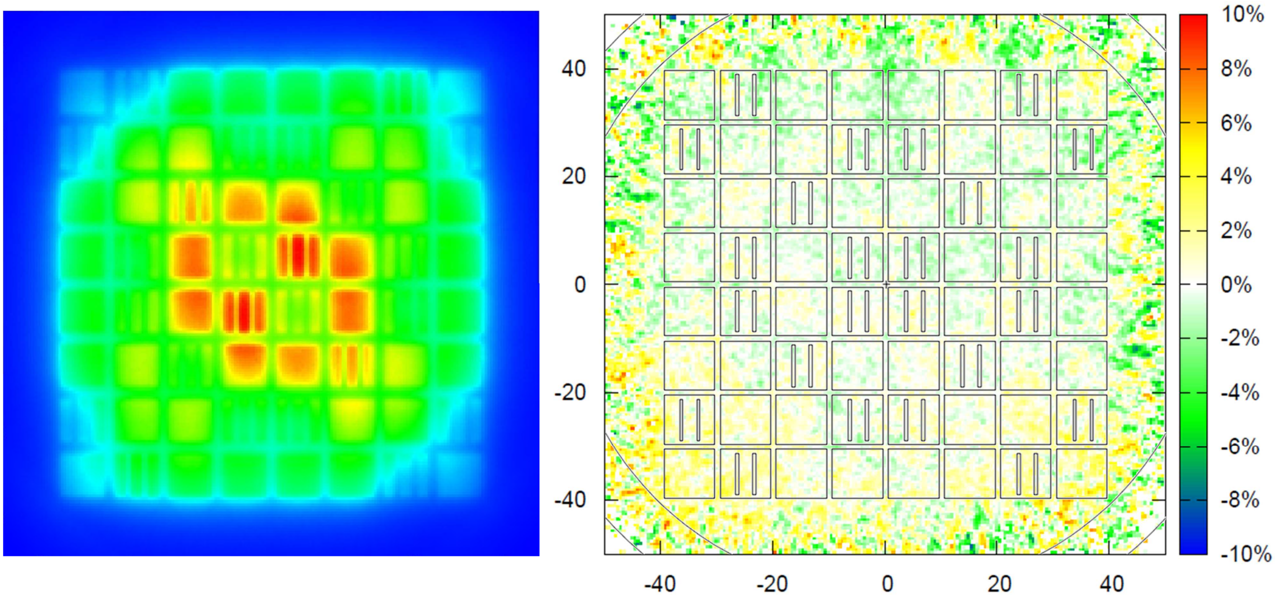


FIG. 9: Fast flux in the critical state (TRIPOLI4©, left) and comparison MCNP6 vs. TRIPOLI4© (%, right)

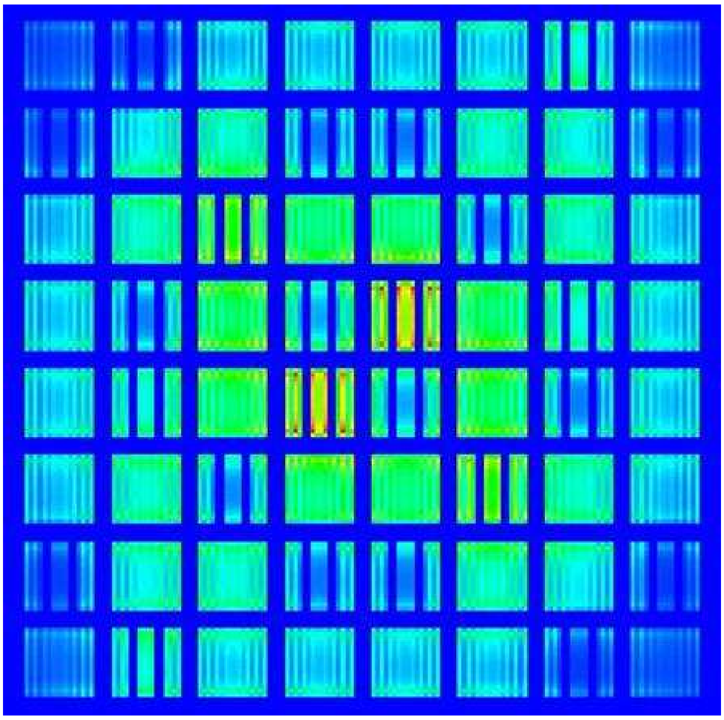


FIG. 10: Fission power distribution in the critical state (TRIPOLI4©)

It is clear through the comparative distributions that MCNP6 and TRIPOLI4© give remarkably identical results. Discrepancies observed consist of statistical noise and match the Monte-Carlo uncertainties of the absolute flux values and are thus non-significant.

4. Main Application of the fictitious core: Green functions

This section describes a methodology developed to calculate ex-core detector responses [5]. This method consists in the TRIPOLI4© "Green function" (GF) feature to determine the importance of each assembly with respect to detector signal. This knowledge then allows to instantly predict the detector response in case of different control rod configurations. Comparisons with full simulation are performed.

On a theory point of view, Green Functions are elementary δ (Dirac) answers to a given, possibly complex, differential equation. Knowledge of a δ answer enables to rebuild any solution by calculation methods. In TRIPOLI4©, GF are transfer functions that stack contributions of neutron sources to a given detector [3]. Once this preliminary calculation is performed, any other simulation is almost instantaneous which is highly noticeable on computing time. This feature then becomes interesting, for instance, in case of a subcritical approach problematic when it comes to determining the detector responses to neutron flux for both subcritical and critical states.

In order to determine detector signals, one first needs to calculate the fission source distribution in the critical or subcritical state. The second step consists in propagating neutrons with this source, to reach the external detectors. Depending on the environment, variance reduction might be required. The fictitious core modelled in this study, and the long distance between the core and the detectors (see FIG. 5) call for such techniques. The Green Functions, once generated, allow to cancel the second step (which otherwise requires significant calculation time) by an instantaneous calculation of detector signal corresponding to the source distribution established in the first step.

The following sections describe the generation of Importance Matrices and the calculation of detector responses.

4.1 Importance Matrices

An importance matrix consists of the sum of contributions of all the neutron sources to the counts scored by the detectors. This matrix can then be used to calculate the detector response in another control rod configuration. Two steps are required to create these matrices:

- **Neutron propagation from each mesh to the detectors**. This step requires to propagate a source distribution allowing to obtain statistically converged results of (n,α) reaction rates coming from each mesh (this calculation is done during the step 2). At the end of this step, a GF file (one per detector) contains information about initial and final states of every neutron having reached the detector.
- **GF files exploitation**. A second calculation only uses the GF files previously created. The (n,α) reaction rates resulting from each mesh is computed. A post treatment normalizes the reaction rate of each mesh with its respective source intensity.

FIG. 11 illustrates an example with the 64 FAs core. It is meshed in 16x16 radial cells, meaning that each FA is divided in four source meshes. The core is also meshed axially in this example, with 4 equivalent slabs along the z-axis to account for progressive insertion - or extraction - of control rods.
The figure on the left shows contributions (or weights) for each source to detector 1. The central figure does the same for detector 2. Finally, the figure on the right shows the

Importance Matrices for detector 3. It is clear in the latter that it is the FA in the upper-left corner that contributes mostly (but not exclusively) to the detector counts.

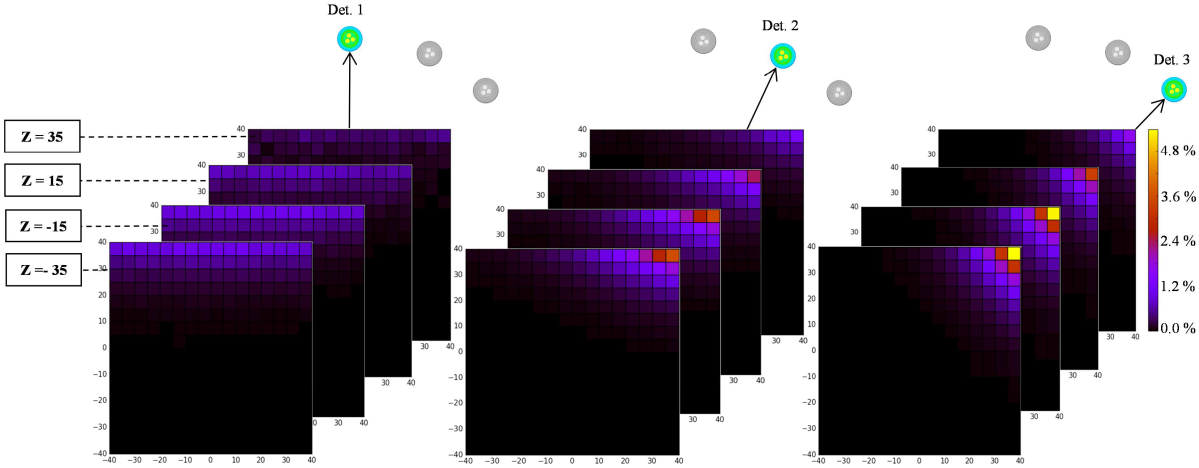


FIG. 11. Contribution (%) of each source mesh on detectors' response

Data analysis also confirms the contribution of fast neutrons. Fast neutrons that leak out of the core have a chance of reaching the vicinity of the detector before being thermalized and thus increasing significantly the odds of undergoing a (n,α) reaction in boron. Thermal neutrons inside the core, on the other hand, have almost no chance at all of going so far.

4.2 Detector responses calculation using Importance Matrices

Given the power **W** inside the core in a critical state, one can determine the detector signal.

$$\boldsymbol{W = \frac{S}{v} \times E_f \times C} \qquad (1)$$

with :

- **S** : number of fission neutrons
- **ν** : mean number of neutrons emitted per fission
- $\mathbf{E_f}$: mean energy release per fission (in eV)
- **C** : constant , C = 1 eV = 1.602 10^{-19} J

The detector signal estimation is based on the detector sensitivity ($\boldsymbol{D_s}$) given by the manufacturer. In our case, this sensitivity is 12 counts per second per unit of conventional flux (cps/$\phi_{0.025}$, flux at 0.025 eV). Therefore, the detector signal **A** (in cps) is:

$$\boldsymbol{A = D_s \times \Phi_{0.025}} \qquad (2)$$

However, the neutron spectrum interacting inside the detector by (n,α) reaction in the boron after having gone through all the shielding materials is not thermal but poly-energetic. The conventional flux inside the detector is calculated using the total (n,α) reaction rate ($\boldsymbol{\tau_{total}}$) into the boron. In TABLE 4, results are given per unit of power (equation (2)). Finally, the detector signal is given by the following formula (with $\boldsymbol{\Sigma_{0.025}}$ being the macroscopic (n,α) cross section in boron at 0.025 eV):

$$\boldsymbol{A = \frac{1}{W} \times D_s \times \frac{\tau_{total}}{\Sigma_{0.025}}} \qquad (3)$$

TABLE 4 provides an example of the use of Green Functions. Importance Matrices are calculated in the critical state of the core (see FIG. 6). A full calculation is performed in order to determine detector responses. During the same calculation with TRIPOLI4©, Importance Matrices are generated. All three detector responses determined with GF match and discrepancies are extremely low and consistent with Monte-Carlo uncertainties.

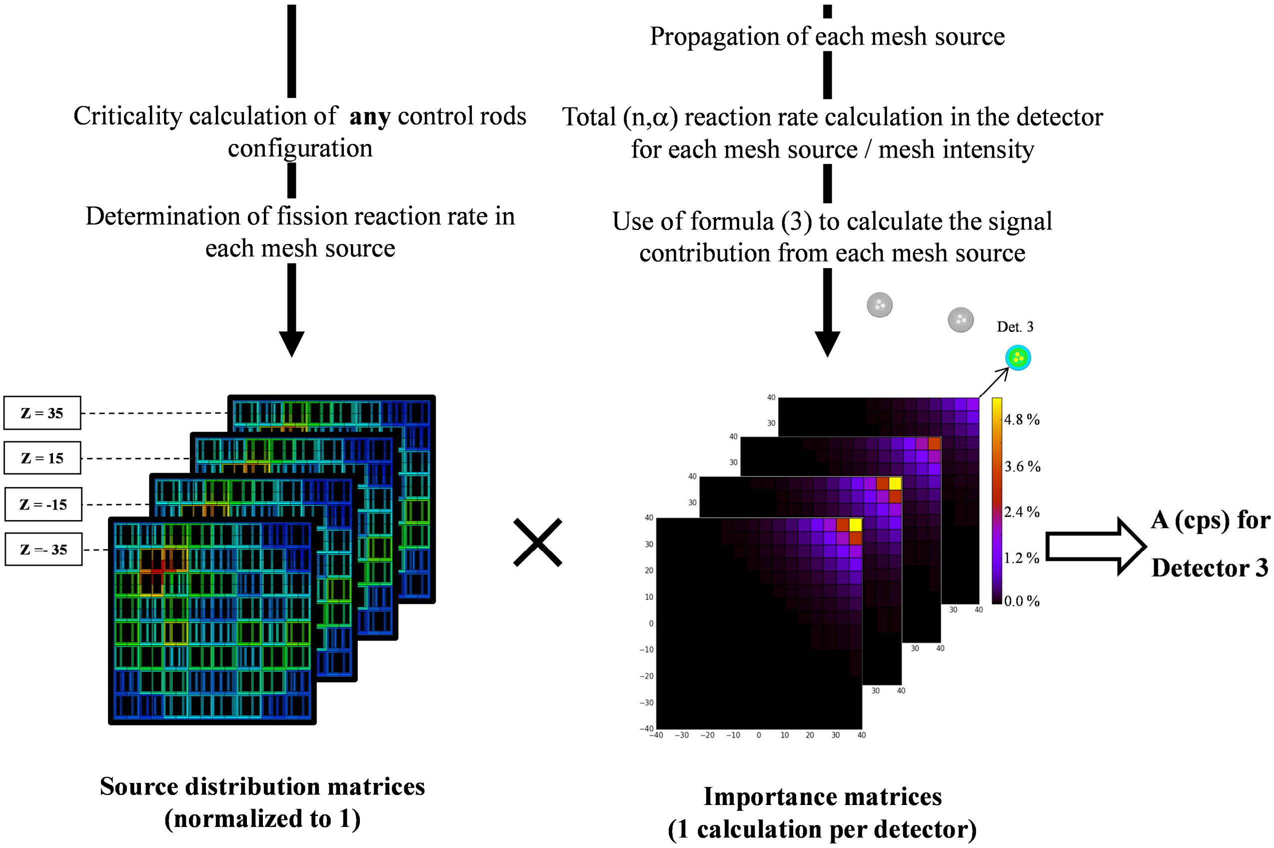
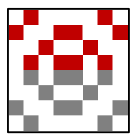


FIG. 12. Detector response calculation with Green Functions

TABLE 4. Detector responses in two different rod configurations, calculated directly or with Green Functions

Detector	Detector signal (cps) With GF	Detector signal (cps) Full simulation	Discrepancy (%)	Discrepancy uncertainty (%)	Configuration
Det. 1	74.4	73.8	0.94	0.71	Critical (see FIG. 6.)
Det. 2	59.7	59.6	0.21	0.68	
Det. 3	43.5	43.1	0.74	0.85	
Det. 1	112.7	112.8	-0.08	0.72	
Det. 2	94.7	93.9	0.81	0.69	
Det. 3	76.5	76.9	-0.41	0.83	

It is remarkable that GF enable to calculate detector responses in a completely different control rod configuration (see TABLE 4). Indeed, with half the rods inserted (all the ones in the North of the core) and inducing a severe flux slide, GF determined in the critical state still enable to perfectly and instantaneously calculate detector responses, thus saving precious computer time.

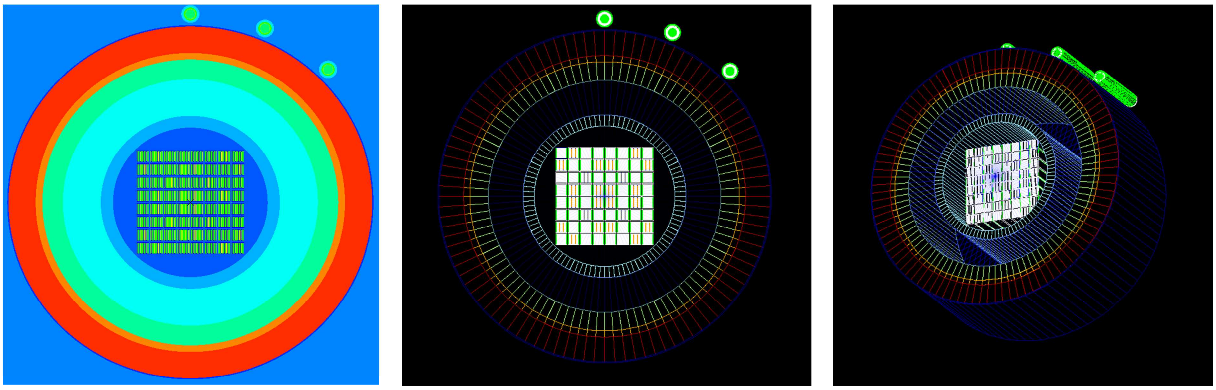
This is due to a feature already pointed out: only fast neutrons contribute to detector responses in our case and hafnium is mostly a thermal and epithermal absorber. Rod configuration has then a very limited effect on Importance Matrices.

4.3 Geometry unification: TRIPOLI4© and Geant4

For every neutronic or shielding study, it is important to compare the results between two codes in order to validate the model and the calculation of the main physical quantities. It requires to multiply the work by the number of codes to use. Generating the geometry and creating materials with the syntax of each code can be very time consuming. It is usual to have errors coming from differences on geometry or materials definitions and not from the simulation itself. Therefore, it is interesting to be able to create a geometry corresponding to several codes with a single data file.

In order to define the geometry, it is possible for TRIPOLI4© and Geant4 codes to both use Root software [7]. TRIPOLI4© can read binary files created with Root, while Geant4 is able to read GDML (Geometry Description Markup Language) files exporting with Root.

Thus, FIG 13 represents TRIPOLI4© and Geant4 visualizations performed with a single Root file (binary exportation for TRIPOLI4© and GDML exportation for Geant4).



TRIPOLI4© Geant4

FIG. 13. TRIPOLI4© and Geant4 dummy core geometry created with a unique Root file.

Further studies concerning calculation comparison of main physical quantities with these two codes will be done using this geometry unification.

5. Conclusions

In order to use a shared object for both V&V processes and methods and techniques development, TechnicAtome has designed a dummy core. This core is used for different aims such as methodology comparisons, computational techniques validation and qualification and non-regression tests.

Calculations were successfully benchmarked with MCNP6. This, along with the common orders of magnitude of calculated neutronic parameters, validates the representability of the mock FAs and cores designed by TechnicAtome for its internal education, training and V&V purposes.

An example of methodology development concerning the calculation of ex-core detector responses with the Green Function feature of TRIPOLI4© is shown. Green Functions allow determining the contribution of each source mesh on detector responses, and then creating Importance Matrices to instantly predict the signal in different control rod configurations.

Data analysis confirms the contribution of fast neutrons, and highlights the fact that detectors should be placed accordingly with their angular detection zone. Signal prediction in completely different control rod configurations, with Importance Matrices, is remarkable. More calculations in different configurations and for a subcritical would support the confirmation that Importance Matrices can predict detector signal.
Finally, after TRIPOLI4© and Geant4 unification at geometry level, the next step is to determine main neutronic quantities to validate this model.

6. References

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