JHR CORE FEATURES BASED ON TRIPOLI-4.9® AND CODE COMPARISON WITH MCNP-6

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ABSTRACT

TechnicAtome is in charge of both the design and building on behalf of CEA of the 100 MW Jules Horowitz Reactor (JHR). This modular Material Testing Reactor is under construction in southern France, with radioisotope production and material testing capabilities. The JHR fuel assembly is mainly composed of concentric plates and the reflector is mainly composed of beryllium. The design is completed and this paper is dealing with TRIPOLI-4.9® and MCNP-6 stochastic code comparison on some neutron and photon features of the reactor core. A Monte Carlo methodology has been developed to model the reactor with both codes. MCNP-6 and TRIPOLI-4.9® have led all the design technical studies and safety analysis.
The scientific interest in confronting several code results, added to the strategy of not depending on a single calculation line, leads to use these two codes. Both have the same capabilities on the JHR design. The latest up-to-date model used in MCNP-6 is well described implementing an innovative fine-tuned method for a good understanding of neutronic parameters on local structures. On the other hand, the TRIPOLI4.9® model comes from a tool which enables a simplification of datasets, especially for the future use of TRIPOLI4.9® in supporting operation calculations.
Both models enable to compare fluxes and heating in the inner part of the core. This paper proposes an evaluation of specific physical quantities in the experimental devices locations in core. Comparisons between TRIPOLI-4.9® and MCNP-6 are made along the paper. Methodology aspect is also provided, such as the normalization factor calculation and its components. Reactivity bias due to geometric simplification in TRIPOLI-4.9® is also explained in detail. Some neutron/photon fluxes and energy deposited distributions are also provided, resulting in a very good coherence between codes. Small discrepancies are explained by slight model differences between the designer geometry and the operator's. However it does not affect the overall experimental devices characterization.

KEYWORDS: JHR, TRIPOLI-4, MCNP-6

1. INTRODUCTION

Design and development of new research reactors is mainly aimed at qualifying structural materials, to characterize fuel behavior during nominal conditions or accident scenarios and to produce radioisotope for medicine application. In this scope, the Jules Horowitz Reactor (JHR) is intended to be a multipurpose Material Testing Reactor (MTR) which achieves the most important experimental capacity in Europe [1].

One application will be to validate material and fuel both for the current nuclear power plants of second and third generations and for the future ones offering high neutron flux, both in the thermal and fast range (around 5.10^{14} $n.cm^{-2}.s^{-1}$). High experimental capabilities will be placed in every part of the reactor. Experimental devices, like ADELINE, MADISON or MOLFI [2] for ^{99}Mo production made by CEA [3] can be loaded and irradiated in the reflector. JHR is designed to fulfill the flux and maximum heating requirement on such experimental devices.

HORUS [4] chained with MCNP-6 [5] are commonly used to compute neutronic physical quantities for thermalhydraulic and mechanical analysis [6]. Confronting two different calculation lines offers a scientific interest and leads to a more robust design. Consequently, TRIPOLI-4.9® [**Erreur ! Source du renvoi introuvable.**] is also used to perform code comparison and neutronic studies. The usage of both TRIPOLI4.9® and MCNP-6 codes, in addition with the deterministic scheme, has also been motivated by the wish to not depend on a single code. Both Monte-Carlo codes have the same capabilities for the JHR design.
Build-up of the JHR models is complex and has required several iterations between the 2 models. It should be pointed out that models are under constant evolution process and that the MCNP-6 model has recently been updated and refined. Moreover, the TRIPOLI4.9® model is generated by the GADGET tool which is included in HORUS. This generator has been created in order to simplify the structure of TRIPOLI4.9® datasets, especially for the future use of the code for supporting operation calculations.

The main goal of this paper is to compare the responses of the two models on a similar given situation. The results will point out both model similarities, and the potential updates that still remain to be included in order to reach the best synergy. This paper focuses on the main experimental location features in the core and neutronic methodology. Fluxes and energy deposition in fuel assemblies and experimental devices will be discussed. Some experiments have to be characterized using different neutronic tools in order to provide mechanical and thermalhydraulic studies with input for safety review. In this paper, TRIPOLI-4.9® is used to characterize the different distributions needed in design studies:

- Thermal, epithermal and fast neutron flux;
- Gamma deposited energy.

2. JULES HOROWITZ REACTOR

JHR is a 100 MW pool-type Material Testing Reactor cooled by light water. The core rack is a 60 cm height cylinder made of aluminum in which 37 drilled holes can host 34 fuel assemblies and 3 large irradiation devices. Every fuel assembly is composed of 8 cylindrical and concentric plates kept together with 3 stiffeners. A U_3-Si_2 fuel is considered within this study. A 3 cm height Al-B poisoned insert positioned 1 cm above the top of each plate aims at limiting the flux increase and the heat deposition in the upper side of the fissile zone. Seven small irradiation locations, accommodating isolated MICA [2], are placed in the center of the cylindrical fuel plates in order to reach high fast flux. Three larger experimental devices called clustered MICA replace a fuel assembly and contain each 3 isolated MICAs. The others fuel assemblies are filled with hafnium rods to handle reactor reactivity, providing both depletion compensation and safety shutdowns coverage. They are geometrically composed of two concentric hafnium tubes and an aluminum follower. The core is surrounded by an aluminum vessel (containing the primary circuit) and then by a reflector. The latter is mainly composed of beryllium elements, allowing a suitable thermal neutron flux for several materials tests and ^{99}Mo production locations.

In this paper and for this benchmark, the experimental configuration considers 12 ADELINE-type devices, containing a UO_2 1 % ^{235}U enriched fuel pin simulating high burn-up fuel for calculations. Neutrons coming from the inner core undergo more collisions in beryllium than in the water reflector, with less absorption and a lower energy decrease per collision. The core is considered with fresh fuel and with a 6 % mass content of boron at the top of fuel elements. The JHR neutronic model [8] is described on Fig.1 and Fig.2 and comes from MCNP-6 visualizations.

This work focuses on experimental devices located in the inner core in order to have both high and various ranges of fast flux:

- 7 isolated MICA devices are positioned within seven fuel assemblies centers, in various core slots: 2 within the first ring, 2 within the second and 3 close to the pressure vessel.
- 3 clustered MICA devices are positioned in each drilled hole.

The inner part of the core experimental devices has been chosen as described in Fig.2. An isolated MICA device is composed of three small cylindrical experimental samples made of NaK and housed in a stainless steel tube. The isolated MICA device is itself surrounded by various layers (cooling water, helium, aluminum…).

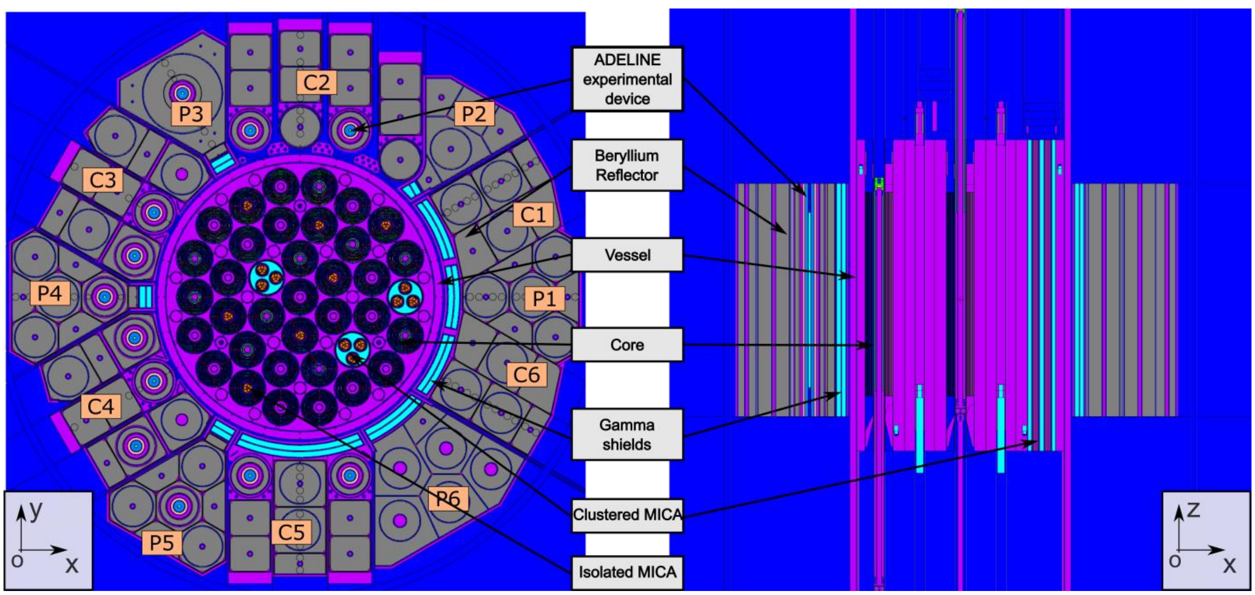

Figure 1. JHR general description (MCNP views).

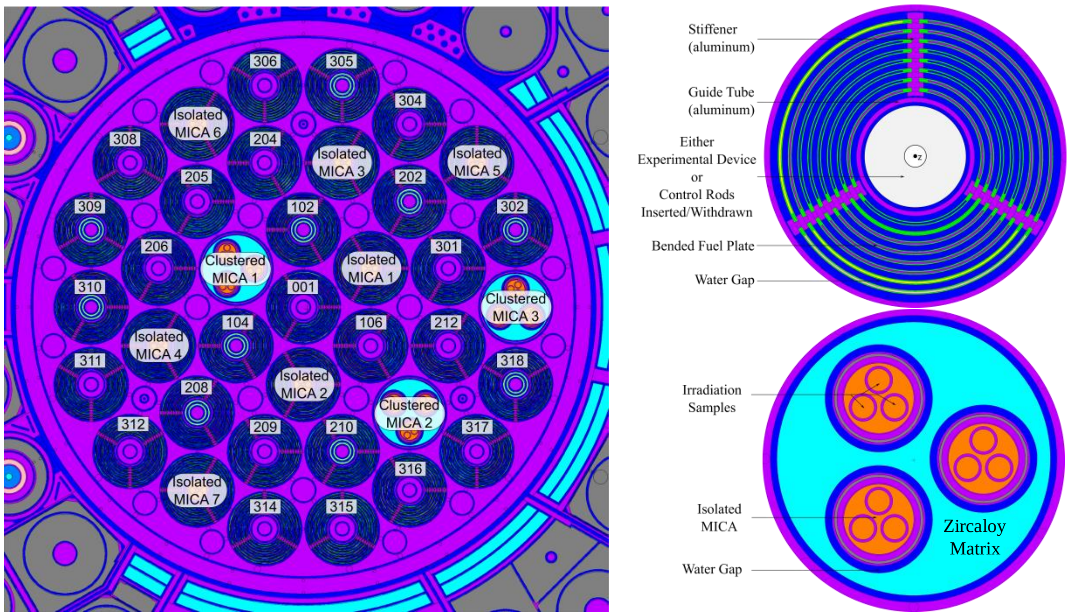

Figure 2. Core (left), fuel assembly (top right) and clustered MICA (bottom right) descriptions.

3. NEUTRONIC COMPUTATIONAL MODEL AND METHODOLOGY

Various neutronic calculation codes have been used at TechnicAtome for the JHR design: HORUS-3D/N (deterministic), MCNP-6 (Monte-Carlo) and TRIPOLI-4.9® (Monte-Carlo). The JHR design and more particularly its reflector require an efficient use of both deterministic and Monte-Carlo codes. The latter have an important advantage, enabling very fine geometry description.

TRIPOLI-4.9® has hereby been chosen for studying its capabilities in comparison with MCNP-6. The agreement between the 2 models enables to have 2 reference codes, which removes the risk relying on the dependence to a single code and guarantees JHR a robustness in studies (design and operation) when using Monte-Carlo codes.

Design of the JHR models (both TRIPOLI-4.9® and MCNP) is highly tuned and enables to have all the operator performance features. Moreover many updates of the reactor design have led to do a lot of modifications in the two models. The main goal of this paper is to compare the responses of the 2 models on a similar given situation. The results will show both model similarities, and the potential updates that still remain to reach the best synergy of the two models.

3.1. TRIPOLI-4 model

TRIPOLI-4 code is a three-dimensional, continuous energy computer code for particle transport. Based on the Monte-Carlo method, the code offers simulation of neutrons, photons, electrons, and positrons transport. It is used for determining some parameters needed for the JHR design as multiplication factor, power factors, control rod worth, both neutron and gamma heating… The nuclear data used in this paper are based on the European evaluation JEFF-3.1.1 [8] with the photonic library coming from Lawrence Livermore Nuclear Laboratory (EPDL-97) [10]. The latest version of TRIPOLI-4.9® enables to consider mesh (regular or irregular) independent of the real geometry, so-called "EXTENDED_MESH". One of its applications enables to generate, for example, flux and energy deposition (neutron and gamma) maps. The JHR geometry definition needs automation for being used easily by the nuclear operator. The GADGET [11] tool created by CEA allows to synthetize all the information to characterize the reactor state

(absorbers positions, number and characteristics of fuel assemblies, experimental devices description in the core and in the reflector…).
For this paper, a fictitious situation similar to other configurations studied with APOLLO2-MOC and with fresh fuel balance is considered. APOLLO2 being a 2D code, control rods are either inserted or outside the core. Thus, all compensation and operation absorbers are fully inserted, and all safety absorbers are fully extracted in order to match an APOLLO2 model while also being close to criticality. The neutronic model is described on Fig.3.

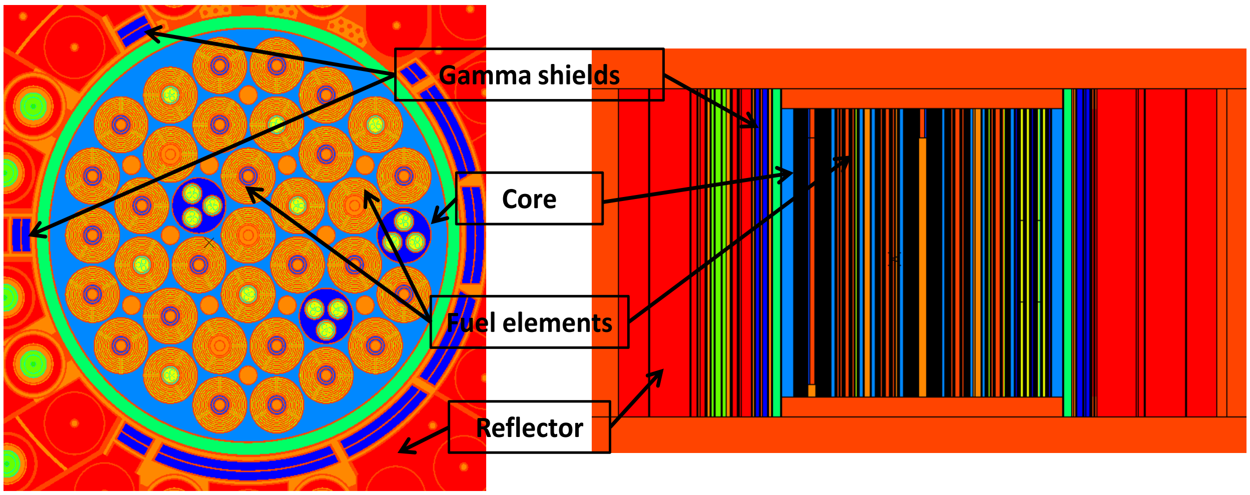

Figure 3. JHR general description (TRIPOLI-4® visualizations)

3.2. MCNP-6 model

MCNP-6 code has been used during the entire design process of JHR. Its main advantage is the ability to handle complicated geometries thanks to the universe option. Criticality calculation (using KCODE card) is performed and the "FMESH" option is used in this paper (similar to the "EXTENDED_MESH" option of TRIPOLI-4.9®). The same neutronic library is taken, based on JEFF-3.1.1. Photonic library based on LLNL evaluation [12] is an older version (92) than the one used in CEA but no major differences have been identified for nuclear reactor.

JHR geometry has been translated for MCNP-6 for the reflector area. Thus, details have been taken into account in the reflector and in the upper and lower part of the core, unlike in TRIPOLI-4.9®. Most physical quantities needed by mechanical, thermal and radiation protection studies have been obtained with MCNP-6.

3.3. Normalization methodology

TRIPOLI-4® and MCNP-6 are used in critical mode to determine the multiplication factor and the different physical quantities needed. In both codes scores are normalized to one source fission neutron. To translate these results in physical quantities (flux, deposited energy) for a given power, normalization factors are calculated with the following formula using core power, fission rate and energy deposited in the core:

$$f_n = \frac{P_{Core}}{C \cdot W_{fiss} \cdot \tau_f} \; with \; W_{fiss} = E^n_{fiss} + E^\gamma_{fiss} + E^\beta_{fiss} + E^\gamma_{FP}, \tag{1}$$

Where

- f_n is the flux normalization factor;
- P_{Core} is the nominal core power, corresponding to the energy released in the primary circuit;
- C is the eV-J conversion factor;
- τ_f is the total system fission rate;
- W_{fiss} is the energy produced per fission;
- E_{fiss}^{n} is the neutron heating deposited within the vessel (considered fission fragments, and both prompt and delayed neutrons contributions);
- E_{fiss}^{γ} is the prompt gamma heating deposited within the vessel;
- E_{fiss}^{β} is the prompt beta heating deposited;
- E_{FP}^{γ} is the delayed gamma heating deposited within the vessel coming from fission products.

In this study, all 5 values (τ_f, E_{fiss}^{n}, E_{fiss}^{γ}, E_{fiss}^{β} and E_{FP}^{γ}) needed for flux normalization factor determination has been estimated by TRIPOLI-4.9®. Fission rate is calculated with an "EXTENDED_MESH" containing the primary circuit. Energy fission contributors are calculated by the means of 3 simulations in which delayed contributions are either considered or not modeled:

- Simulation 1: standard simulation with consideration of both delayed beta and delayed gamma contributions in the fission Q-value;
- Simulation 2: option "ONLY PROMPT ENERGY" added in SIMULATION block: delayed beta and delayed gamma contributions are subtracted from the fission Q-value;
- Simulation 3: option "ONLY PROMPT GAMMA ENERGY" added in SIMULATION block: delayed beta energy is deposited locally, whereas delayed gamma contribution is subtracted from the fission Q-value.

The 4 energies contributors can be deduced from Table I and are summarized in Table II. The calculation of all contributors of fission energy is possible thanks to the latest development, enabling options "ONLY PROMPT GAMMA ENERGY" and "ONLY PROMPT ENERGY". In design studies, the delayed gamma contribution to heating was evaluated as proportionally (b_{dg}=36 %) to the prompt gamma heating as:

$$E_{FP}^{\gamma} = b_{dg}\, E_{tot}^{\gamma} = b_{dg}\left(E_{FP}^{\gamma} + E_{fiss}^{\gamma}\right) = \frac{b_{dg}}{1-b_{dg}}\, E_{fiss}^{\gamma}, \quad (2)$$

Table I. Results of the 3 normalization calculations

	Deposited energy [MeV/fission]	Simulation 1 (standard)	Simulation 2 (ONLY PROMPT ENERGY)	Simulation 3 (ONLY PROMPT GAMMA ENERGY)
In the core	Neutron	185.3	172.4	179.0
	Gamma	10.95	10.98	10.98
In the entire reactor	Neutron	186.0	173.1	179.7
	Gamma	13.03	13.03	13.03

With the results from Table I, the delayed gamma contribution can be directly evaluated with the difference of E_{fiss}^{γ} of simulation 1 and simulation 3. The beta contribution can be calculated with the difference of E_{fiss}^{n} of simulation 3 and simulation 2: 6.60 MeV confirms the 7 MeV hypotheses. The same exercise was made with MCNP-6 code, and the results are stored in Table II. Only E_{FP}^{γ} and E_{fiss}^{β} are not estimated.

Table II. Fission energy contributors (MeV/fission)

	TRIPOLI-4®	MCNP-6	Relative Bias (%)
τ_{fiss}	0.420	0.417	< 0.8
E_{fiss}^{n} (MeV)	172.4	172.7	< 0.2
E_{fiss}^{γ} (MeV)	10.95	10.88	< 0.7
E_{fiss}^{β} (MeV)	6.60	*7.00**	< 6
E_{FP}^{γ}(MeV)	6.34	*7.20**	*< 12*

**Values assumed by TechnicAtome*

Finally, with TRIPOLI-4® the flux renormalization factor is equal to $7.56 \cdot 10^{18}$. With MCNP-6 the equivalent value is $7.57 \cdot 10^{18}$, leading to a relative bias lower than 1 %. No nuclear data biases mainly coming from AMMON experiment [13] are considered in this paper but have been taken into account for design studies. Relative comparisons are thus done by using raw date (no normalization) for both codes: every absolute result taken is then normalized to one fission source.
It is noticeable that both MCNP-6 and TRIPOLI-4® lead to the same normalization factor.

4. RESULTS

4.1. Reactivity comparison

Reactivity is a key parameter to verify. Table III indicates reactivity and multiplication coefficient of the core configuration chosen for both TRIPOLI-4.9® and MCNP-6. The difference of 700 pcm can be mainly explained by the geometry modeling difference. Basically, the detailed description with MCNP-6 of all components above and below the core, including withdrawn control rods, has an impact of approximately ~100 pcm. This effect has been calculated with MCNP-6 by replacing all the structures by water in order to match the TRIPOLI-4.9® model. The remaining overestimation of ~600 pcm on the reactivity can also be explained by other geometry differences described in section 4.2 (neutron source carrier, peripheral devices and water channels).

	TRIPOLI-4® simplified model	MCNP-6 detailed model	MCNP-6 predicted values with TRIPOLI model like (without the reflector)
k_{eff}	1.0247	1.0174	~1.020
Reactivity (pcm)	2410	1710	~1970

Table III. Reactivity comparison with TRIPOLI-4® simplified model and MCNP-6 detailed model.

4.2. Neutron fluxes distribution in the core

Characterization of neutron fluxes is essential for the future experimental devices located within the core. Fine Cartesian meshes over the geometry have been used in order to obtain distributions over the full core. Spatial discretization is done by taking a Cartesian mesh with respectively 200, 200 and 8 meshes along the x, y and z axis (for a total of 320 000 meshes). Neutron fluxes (Fig.4) are taken for z = -5.0 cm, corresponding averagely to the maximum values. Fast flux follows a cosine curve with a maximum value right in the middle of the core. It is explained because all control rods are inserted except safety ones. Because maximum fission reaction rate occurs in the middle of the core, one can correlate the fast neutron flux to the prompt and delayed photon flux. Thermal neutron flux is important in the inner part of the core but also in the area close to the beryllium reflector, fulfilling one of its duties. Very low thermal flux values can be seen in the inner part of 23 fuel assemblies, due to the inserted control rods.

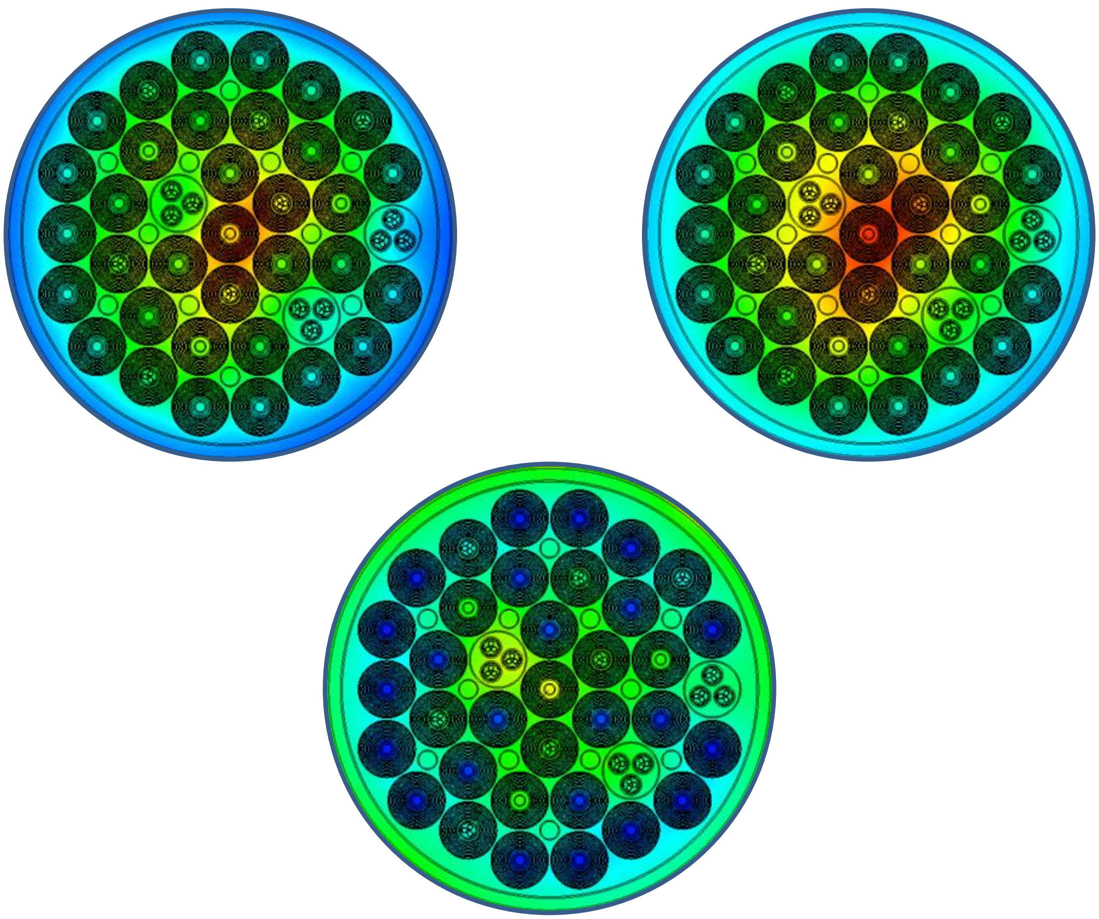

Figure 4. Neutron fluxes in n.cm^{-2}.s^{-1} (fast: up/left, epithermal: up/right, thermal: down) distributions over the full core

Comparisons with MCNP-6 have been performed and are reflected in Fig.**5**. Maximum stochastic uncertainty is 2 % at 2σ. No major differences can be found in the core area apart from "three visible holes" coming from the material definition and that have been fixed. This kind of result highlights the necessity of cross verification by using two independent models, codes and analysts in order to avoid mistakes. In this case, the difference is simply explained by the two small geometry differences. Firstly, MCNP-6 has more water rings in 16 locations whereas TRIPOLI-4.9® model has only aluminum (no cooling modeled), as a feature of the GADGET pre-processing tools. In this slot one can put extra samples for irradiation. Secondly, MCNP-6 has a source holder model in three areas with aluminum and water rings whereas TRIPOLI4.9® has stainless steel tubes with only one cooling water ring (see Fig. 6).

One can also notice a small difference between the aluminum vessel around the core and the reflector on Fig.6. Neutron fluxes in comparison are not exactly the same because MCNP-6 has a water gap larger than in TRIPOLI-4.9® model. Meanwhile, this gap was fixed for this paper by taking the same amount of water and changing the density. The latest distinct result is found in the upper right of the core, near a gamma shield. Once again, geometry is different in this point because the two solutions are possible. Those differences are shown here to see impact of the two possibilities.

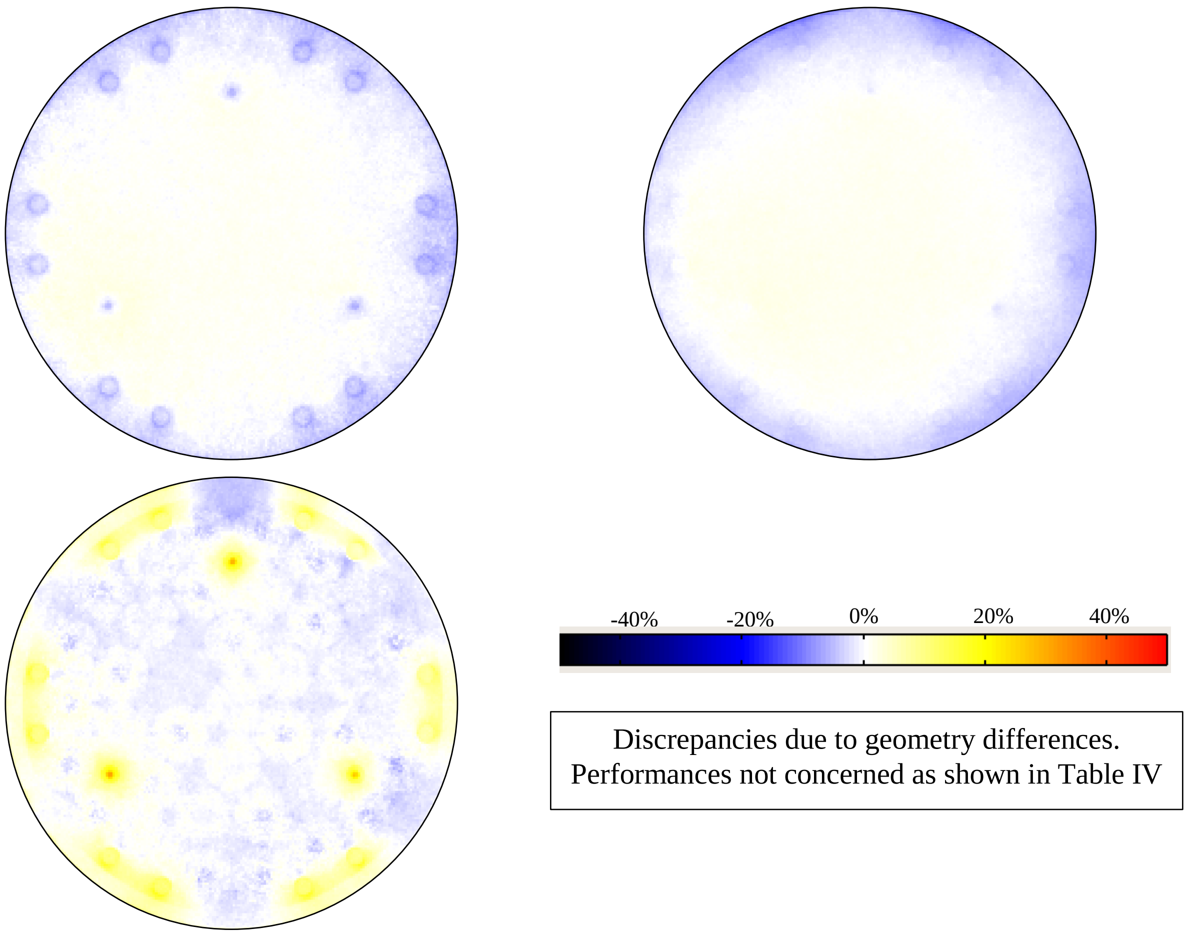

Figure 5. Relative neutron fluxes in % (fast: up/left, epithermal: up/right, thermal: down/left) comparison between TRIPOLI-4® simplified model as reference and MCNP-6 detailed model over the full core.

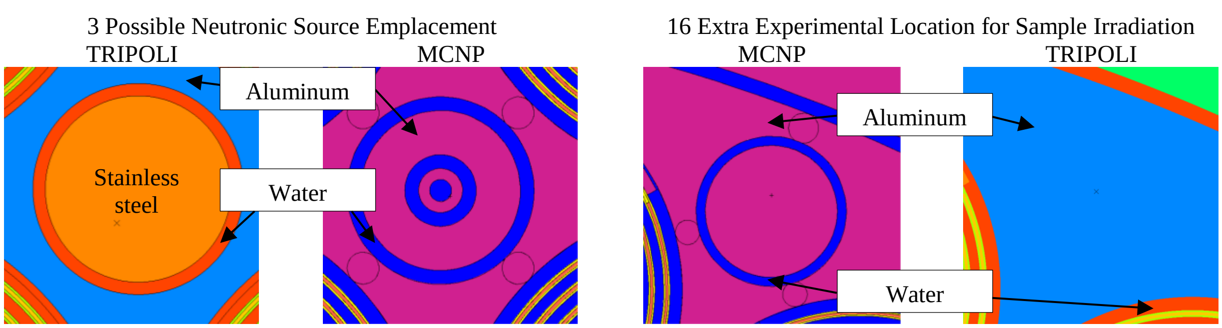

Figure 6. Close-up around geometry differences.

4.3. Photon fluxes and heating deposition distribution in the core

The experimental quality is as important as the neutrons flux levels in the future experimentations. It means both gamma fluxes and heating must be well-characterized and should be as low as possible. The same Cartesian mesh used for neutron fluxes enables to assess both gamma fluxes and heating deposited in the core (Fig.7). Photon distribution is similar to fast neutron distribution because maximum fission reaction rate occurs in the middle of the core. This highlights correlation between fast neutron flux and prompt photon flux. The distribution of deposited energy by gamma shows high levels of heating in the metallic structures, mainly hafnium (absorbers) and zirconium (clustered MICA). Furthermore, the gamma heating distribution follows the gamma flux distribution with more important values in the center than in the peripheral.

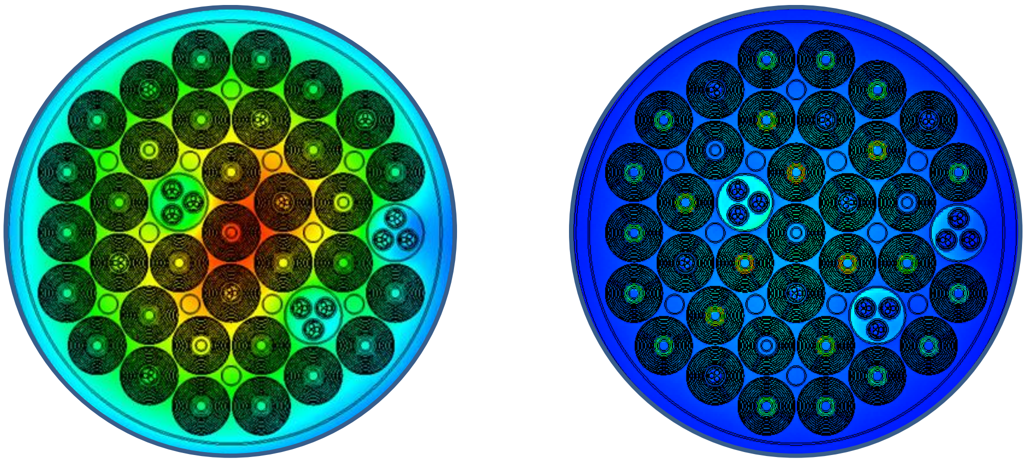

Figure 7. Photon flux in $\gamma.cm^{-2}.s^{-1}$ (left) and deposited energies in $MeV.cm^{-3}$ (right) distributions over the full core (TRIPOLI-4®)

Comparisons with MCNP-6 have been performed and are reflected in Fig. 8. All results are converged below 2 % at 2σ. Like the neutron fluxes, one can see the effect caused by slight geometry difference as seen in paragraph 4.2.

4.4. Flux and heating comparisons in experimental devices

Finally, the most important for future experimentations are the conditions inside the experimental devices. More precisely, both neutron fluxes (experimentation requirements) inside the experimental samples and gamma heating inside their clad (experimentation quality) are needed.

Table IV shows the good agreement between TRIPOLI-4.9® and MCNP-6 with a maximum absolute difference of 1.6 %, 1.6 % and 1.5 % respectively for thermal, epithermal and fast neutron fluxes. The small underestimation with TRIPOLI-4.9® can be explained by less leakage due to the absence of aluminum above and below the core. This effect is flattening the curve shape of the fast flux.

Finally both gamma energy deposited values in the experimental devices are similar with a maximum absolute difference of 2.4 % and 0.9 % respectively on the clustered MICA and the isolated MICA. The highest values can be explained by the Zircaloy in the clustered MICA surrounding the samples. Photon libraries or photon transport may explain such differences.

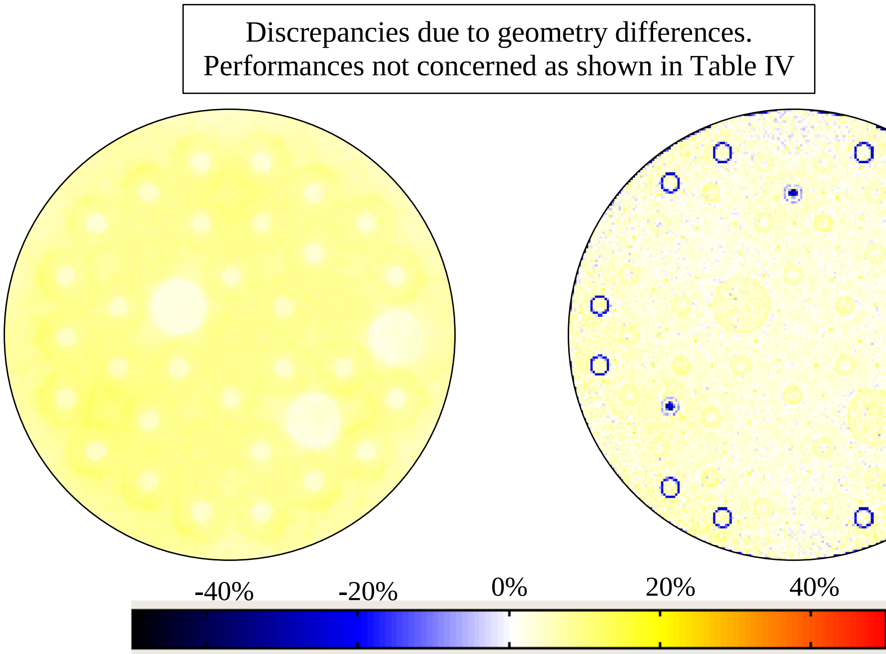

Figure 8. Relative photon flux relative comparison in % (left) and deposited energies comparison in % (right) between TRIPOLI-4® simplified model as reference and MCNP-6 detailed model over the full core.

Table IV. Flux and heating comparison in experimental devices. Fluxes are calculated on the NaK of the isolated MICA and energy depositions are calculated in the stainless steel around the NaK.

Assembly Number	$\varphi_{thermal}$ (%)	$\varphi_{epithermal}$ (%)	φ_{fast} (%)	E_{γ} (%)
Clustered MICA 1-A	1.2	-1.3	-0.6	-2.0
Clustered MICA 1-B	1.2	-1.1	-0.5	-2.4
Clustered MICA 1-C	1.0	-1.2	-0.5	-1.4
Clustered MICA 2-A	-1.6	-0.5	-0.6	-2.0
Clustered MICA 2-B	0.6	-0.9	-0.6	-1.3
Clustered MICA 2-C	0.3	-0.5	-0.3	-1.4
Clustered MICA 3-A	0.1	0.6	0.2	-2.0
Clustered MICA 3-B	0.8	0.0	0.2	-1.9
Clustered MICA 3-C	-1.2	1.2	1.5	-2.7
Isolated MICA 1	0.3	-1.2	-0.8	0.4
Isolated MICA 2	0.0	-1.2	-1.1	0.4
Isolated MICA 3	-0.3	-0.9	-1.0	0.6
Isolated MICA 4	-0.5	-1.6	-1.3	1.1
Isolated MICA 5	1.3	0.7	0.5	-0.8
Isolated MICA 6	0.5	0.2	-0.1	0.3
Isolated MICA 7	-0.4	-0.7	-1.0	0.9
Mean	0.2	-0.5	-0.4	-0.9

5. CONCLUSIONS

The JHR design has been mainly done by jointly using MCNP-6 and TRIPOLI-4.9®. This paper shows a benchmark on a virtual core composed of fresh fuel assemblies with all control rods withdraw, except the safety ones. MCNP-6, TRIPOLI-4.9® and HORUS-3D/N are commonly used by TechnicAtome and CEA for design and safety studies and this work shows comparisons on reactivity, fluxes and energy deposition within the JHR core. A bias can be seen on the reactivity, mainly explained by slight geometry differences, performed for simplicity reasons. The TRIPOLI-4.9® model, automatically generated by the GADGET tool (part of the HORUS package), is indeed aimed for operation calculations. Both neutron and photon fluxes are compared and are similar between the two models. Then, coherent and close results (< 2%) are found in the experimental device rigs for thermal and fast fluxes. Finally, this paper shows the overall good agreement of fluxes and energy deposition for the JHR, in a study case where fresh fuel is loaded.

Such consistency is the result of an iterative process between the designer and the reactor operator. Confronting two different calculation lines offers a scientific interest and leads to a more robust design. The usage of both TRIPOLI4.9® and MCNP-6 codes, in addition to the deterministic scheme, has also been motivated by the wish to not depend on a single code. In the end, both Monte-Carlo codes have the same capabilities for the JHR design.

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