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# IMPACT OF NEUTRONIC PARAMETERS IN A LOSS-OF-FLOW TRANSIENT SIMULATION OF A LOW VOID REACTIVITY SODIUM-COOLED NUCLEAR REACTOR

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## ABSTRACT

This paper presents the impact of the neutronic calculation methodology on the simulation of an ULOF transient scenario for a sodium-cooled CFV reactor concept. Several set of point kinetics (PK) parameters are first prepared with APOLLO3<sup>®</sup> according to different cross section preparation strategies and core solvers. The PK parameters are then furnished to a transient simulation tool (MACARENa) and their impact on the outcome of the ULOF is discussed.

KEYWORDS: Multiphysics, Point Kinetics, ULOF, CFV, APOLLO3<sup>®</sup>

## 1. INTRODUCTION

In order to reach the strict safety criteria expected for nuclear reactors, high fidelity simulation tools are required in all the disciplines involved in the reactor physics analysis (neutronics, system thermo-hydraulics and mechanics). When it comes to model the dynamic behavior of a system in incidental or accidental situations, bridges need to be built at the interfaces between the codes.

Despite all physical phenomena are tightly connected, the complexity of the problem (non-linearity, range of time scales, number of unknowns. . .) leads to choose pragmatic approaches in which the coupling information is collapsed into a set of lumped parameters. In particular, neutronic calculation methodologies are often based on point kinetics (PK) models in which the lumped parameters are pre-computed prior to the transient simulation using a “best-estimate” critical neutron transport calculation.

This paper investigates how the outcome of an Unprotected Loss of Flow (ULOF) scenario in a Sodium-cooled Fast Reactor (SFR) is affected by the choices made in the neutronic strategy used to compute the PK parameters. The case of application is a simplified model of a *Coeur a Faible Vidange* [1] (CFV) that presents a high level of spatial heterogeneities thus making the neutronic calculation task quite arduous. The long term objective is to define the level of accuracy required for neutronic calculations in more sophisticated multiphysics approaches (e.g. quasi-static approximation).

This paper is organized as follows: Sec. 2 presents the CFV model, the ULOF phenomenology as well as the neutronic and thermo-hydraulic calculation methods. Sec. 3 presents the sensitivity of the PK parameters to the neutronic calculation strategy and Sec. 4 their impact on the ULOF outcome. Finally, conclusions and perspectives for future work are given in Sec. 5.

## 2. PROBLEM SPECIFICATION

### 2.1. Presentation of the Benchmark

#### 2.1.1. The CFV fuel assembly

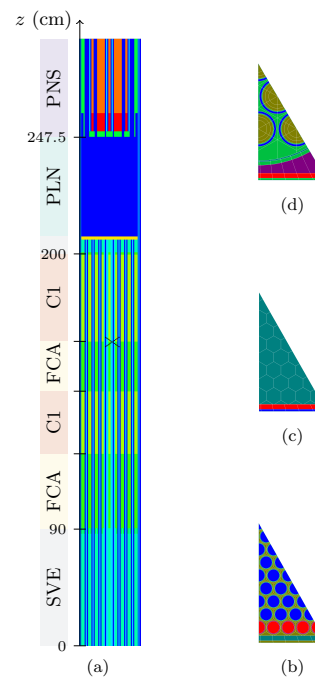
The benchmark chosen is a hexagonal sodium-cooled fuel assembly with CFV design [2] (see Fig. 1). The main features of the CFV assembly are the presence of a sodium plenum (PLN) located at the top of the assembly and the insertion of a fertile  $\text{UO}_2$  layer (FCA) in the middle of the active fissile  $(\text{U,Pu})\text{O}_2$  zone (C1). This heterogeneous design enhances the natural behavior of the core during unprotected transients and, in our particular case (not depleted fuel), it even ensures a negative sodium-void reactivity worth. Data concerning the assembly geometry and material composition can be found in references [3] and [4].

For the transient simulation, it will be assumed that the core is composed of identical such assemblies. Doing so, the question of the reflector modeling is neglected. Yet, this benchmark concentrates most of the difficulties linked to the neutronic modeling of innovative cores presenting a high level of geometrical heterogeneities.

#### 2.1.2. The ULOF transient

The ULOF transient is an accidental scenario characterized by an unexpected primary pumps trip without scram. This scenario leads to a quick decrease of the sodium cooling flow rate ( $Q$ ) in the core while the reactor is still in a power state.

As the flow rate decreases, the reactor power ( $P$ ) varies according to the reactivity feedback coefficient values. Usually, a core design is optimized in order to ensure a negative reactivity insertion during a transient (e.g. with the French KGH [5] or American ABC [6] integral approaches). In a



**Figure 1: CFV fuel assembly - a) axial layout, b) fuel pins mesh (1/12<sup>th</sup>), c) sodium plenum mesh, d) axial protection mesh.**

*Remark: C1 stands for fissile zone, FCA for fertile layer, PLN for sodium plenum, SVE for expansion tank and PNS for axial neutronic protection*

CFV for instance, this is mainly achieved through a negative sodium–void reactivity worth\*.

During the first phase of the ULOF scenario, the reactor power therefore decreases. The phenomenology of the transient is then driven by the evolution of the power over flow–rate ratio. In particular, if  $\frac{P}{Q}$  remains high then the heat cannot be extracted fast enough and the sodium may reach its boiling point.

When the boiling crisis starts, different outcomes may be observed depending on the value of the reactor power [7]. If  $P$  is low enough, the boiling front–line may remain stable near the core outlet and the heat might be extracted with natural convection. Else, dynamic instabilities arise from the strong coupling between neutronic and thermo–hydraulic phenomena [8]. If the bubbles propagate to the core center, the core integrity cannot be guaranteed.

## 2.2. Simulation Tools

### 2.2.1. MACARENa

Such an ULOF scenario has been modeled with MACARENa, which is a validated tool that has been developed at CEA for the analysis of SFR loss–of–flow transients [9]. On the thermo–hydraulic side, MACARENa relies on:

- a 0D model for the momentum conservation law of the coolant;
- a 2D radial–axial simplified model (one radial mesh per material) for the energy conservation law in order to compute the temperature fields;
- a 1D axial two–phase flow mixture model for boiling situations;
- a degradation model which enables to simulate core damage from overheating; no such phase will however be considered in this paper.

On the neutronic side, MACARENa relies on a PK approximation: the power spatial distribution is kept unchanged during the transient but its amplitude is allowed to vary. The PK model of MACARENa includes  $N_p = 8$  precursors families. The equations that are solved are:

$$\left\{ \begin{array}{l} \frac{d\bar{N}}{dt} + \frac{\bar{\beta} - \bar{\rho}}{\bar{\Lambda}} \bar{N} = \sum_{i=1}^8 \lambda_i \bar{C}_i \\ \frac{d\bar{C}_i}{dt} + \lambda_i \bar{C}_i = \frac{\bar{\beta}_i}{\bar{\Lambda}} \bar{N}, \quad i \in 1, 2, \dots, 8 \end{array} \right. \quad (1a)$$

$$\left\{ \begin{array}{l} \frac{d\bar{N}}{dt} + \frac{\bar{\beta} - \bar{\rho}}{\bar{\Lambda}} \bar{N} = \sum_{i=1}^8 \lambda_i \bar{C}_i \\ \frac{d\bar{C}_i}{dt} + \lambda_i \bar{C}_i = \frac{\bar{\beta}_i}{\bar{\Lambda}} \bar{N}, \quad i \in 1, 2, \dots, 8 \end{array} \right. \quad (1b)$$

where  $N$  is the total number of neutrons and  $C_i$  the precursors concentration in family  $i$ . Assuming that the residual power  $P_r$  (decay heat) remains proportional to the total power  $P$  during the transient ( $P_r \approx 7\%P$ ), it directly follows that  $P$  is proportional to  $N$ .

The parameters of the PK model are the reactivity  $\rho$ , the delayed neutron fraction  $\beta_i$  per family  $i$  and total  $\beta$ , the mean neutron generation time  $\Lambda$  and the precursors decay constants  $\lambda_i$ . In theory,

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\* At least for low burnups (beginning–of–cycle conditions).

these parameters should be updated during the transient as the neutron flux shape changes [10]<sup>†</sup>. Yet and except for the reactivity, these variations are neglected in the PK framework.

The only parameter that is allowed to change is the reactivity  $\rho$  which is quite sensitive to local perturbation of the reactor state (coolant density, fuel temperature. . .). In MACARENa, the reactivity variation is re-constructed using pre-computed local feedback coefficients. In this work, the physical phenomena that have been considered are the sodium–void and Doppler effects:

$$\bar{\rho}(t) = \bar{\rho}_0 + \int_{\text{fuel}} K_D(\mathbf{r}) \ln \left( \frac{T(\mathbf{r}, t)}{T(\mathbf{r}, t=0)} \right) d^3r + \int_{\text{coolant}} \alpha_{\text{Na}}(\mathbf{r}) \Delta d_{\text{Na}}(\mathbf{r}, t) d^3r \quad (2)$$

$K_D$  is the Doppler constant,  $T$  the fuel temperature,  $\alpha_{\text{Na}}$  the sodium–void reactivity coefficient and  $\Delta d_{\text{Na}}$  the relative variation of the coolant density.

### 2.2.2. APOLLO3<sup>®</sup>

APOLLO3<sup>®</sup> is a deterministic neutronic platform developed at CEA in France [11]. For this work, it has been used to produce the parameters required by the PK module of MACARENa (initial power shape  $\mathcal{P}$ , delayed neutron fractions  $\beta_i, \beta$ , mean neutron generation time  $\bar{\Lambda}$  and spatial reactivity feedback coefficients  $K_D, \alpha_{\text{Na}}$ ).

The initial power shape  $\mathcal{P} = \mathcal{P}(\psi_0)$  is computed as a function of the flux  $\psi_0$  solution of a critical neutron equation defined over the unperturbed reactor:

$$\mathcal{A}\psi_0 = \frac{1}{k}\mathcal{F}\psi_0 \quad (3)$$

where  $\mathcal{A}$  stands for advection and absorption operators while  $\mathcal{F}$  is the production operator. As for the delayed neutron fractions and mean neutron generation time, they are computed as:

$$\bar{\beta}_i = \frac{\langle \psi_0^\dagger \cdot \mathcal{F}_{d,i}\psi_0 \rangle}{\langle \psi_0^\dagger \cdot \mathcal{F}\psi_0 \rangle}, \quad \bar{\beta} = \sum_{i=1}^{N_p} \bar{\beta}_i, \quad \bar{\Lambda} = \frac{\langle \psi_0^\dagger \cdot \frac{1}{v}\psi_0 \rangle}{\langle \psi_0^\dagger \cdot \mathcal{F}\psi_0 \rangle} \quad (4)$$

where  $\langle . \rangle$  stands for the integral over phase-space (scalar product),  $\mathcal{F}_{d,i}$  is the delayed neutron production operator in family  $i$  and  $\psi_0^\dagger$  is the solution of the adjoint problem corresponding to Eq. 3. The integral reactivity feedback coefficients for a perturbation  $p$  (Doppler, sodium void) are finally computed in the perturbation theory framework as:

$$\delta\bar{\rho}_p = \frac{\langle \psi_0^\dagger \cdot \left( \frac{1}{k}(\delta\mathcal{F})_p - (\delta\mathcal{A})_p \right) \psi_p \rangle}{\langle \psi_0^\dagger \cdot \mathcal{F}\psi_p \rangle} \quad (5)$$

where  $(\delta\mathcal{F})_p, (\delta\mathcal{A})_p$  symbolize the operator perturbations and  $\psi_p$  is the perturbed flux. The perturbation formalism allows to split the integral in the numerator of Eq. (5) in order to retrieve the values of the local coefficients  $K_D(\mathbf{r}), \alpha_{\text{Na}}(\mathbf{r})$ . Under the assumption of small perturbations, these coefficients are assumed to be constant during the transient<sup>‡</sup>.

<sup>†</sup>Except for decay constants  $\lambda_i$  that are evaluated nuclear data and therefore do not depend on the flux shape

<sup>‡</sup>MACARENa includes an option to take into account the non-linearity of the sodium–void coefficient in the plenum.

This paper will not focus on the PK approximation but rather on the impact of the approximations that are made to solve Eq. 3. In particular, three strategies will be investigated:

- the reference calculation scheme implemented in APOLLO3<sup>®</sup> for SFR applications; at the core level the strategy relies on a 33 group transport calculation with a validated  $S_N$  solver (MINARET) ; at the lattice level, consistent effective cross sections are prepared over 3D fuel assembly geometries with a 2D/1D method and a flux angular moments homogenization technique [4];
- an approximated strategy in which the  $S_N$  method used to compute the flux at the core level is replaced by a  $SP_3$  method (MINOS solver);
- another approximated strategy in which cross sections are prepared over 2D geometries (instead of representative 3D patterns) and weighted with the standard flux–volume formulae (i.e. the angular information is neglected) [12].

### 3. ANALYSIS OF POINT KINETICS PARAMETERS

#### 3.1. Validation of APOLLO3<sup>®</sup> Reference Method

In this section, the reference calculation methodology of APOLLO3<sup>®</sup> is validated for the calculation of the PK parameters against independent Monte Carlo simulations performed with the TRIPOLI4<sup>®</sup> code [13]<sup>§</sup>. In particular, Tab. 1 presents a comparison of multiplication factors ( $k$ ), delayed neutron fractions ( $\beta$ ), mean neutron generation times ( $\Lambda$ ), integrated over space sodium–void reactivity worths for 1% and 99% sodium density perturbations ( $\Delta\rho_{\text{Na}}^{1\%}$  and  $\Delta\rho_{\text{Na}}^{99\%}$  with  $\Delta\rho_{\text{Na}} = \int_{\text{coolant}} \alpha_{\text{Na}}(\mathbf{r})d^3r$ ) as well as Doppler constants for a  $\Delta T = 1000$  K temperature variation in fissile and fertile materials ( $K_D^{\text{fiss}}$  and  $K_D^{\text{fert}}$ ).

**Table 1: Validation of integral neutronic parameters**

	$k$	$\beta$ (pcm)	$\Lambda$ (ns)	$\Delta\rho_{\text{Na}}^{1\%}$ (pcm)	$\Delta\rho_{\text{Na}}^{99\%}$ (\$)	$K_D^{\text{fiss}}$ (pcm)	$K_D^{\text{fert}}$ (pcm)
T4	1.07532	366.7	456.7	-6.5	-4.7	-599	-233
$\sigma$	$\pm 0.00001$	$\pm 1.1$	$\pm 0.5$	$\pm 0.3$	$\pm 0.005$	$\pm 4$	$\pm 4$
AP3	1.07523	367.1	461.9	-6.3	-4.9	-612	-243
(AP3-T4)/T4	-8 pcm	+0.1%	+1.1%	-1.6%	+4.0%	+2.2%	+4.3%

Remark: 1 pcm =  $10^{-5}$  and 1\$ = 366.7 pcm

The magnitude of the differences presented in Tab. 1 validates the reference calculation methodology implemented in APOLLO3<sup>®</sup> for the calculation of the PK parameters. The validation of the power distribution calculation can be found in reference [4].

<sup>§</sup>TRIPOLI4<sup>®</sup> includes an IFP method to compute  $\Lambda$  and  $\beta$  as well as a correlated sampling method to compute small reactivity perturbations.

## 3.2. Sensitivity to the Calculation Method

### 3.2.1. Integral perturbations

Tab. 2 now shows the sensitivity of the PK parameters to the neutronic calculation strategy (reference vs. approximated core solver vs. approximated cross sections). For a parameter  $p$ , the sensitivity is computed as  $S(p) = \frac{p_{\text{approx}} - p_{\text{ref}}}{p_{\text{ref}}}$ .

**Table 2: Sensitivity of integral neutronic parameters to the calculation method**

	$k$	$\beta$ (pcm)	$\Lambda$ (ns)	$\Delta\rho_{\text{Na}}^{1\%}$ (pcm)	$\Delta\rho_{\text{Na}}^{99\%}$ (\$)	$K_D^{\text{fiss}}$ (pcm)	$K_D^{\text{fert}}$ (pcm)
AP3 (reference)	1.07523	367.1	461.9	-6.3	-4.9	-612	-243
2D XS + Ref. solver $S(p)$	1.08107 +0.5%	366.6 -0.1%	446.3 -3.4%	-3.2 <b>-50%</b>	-3.2 <b>-35%</b>	-621 +1.5%	-225 <b>-7.4%</b>
Ref. XS + $SP_3$ solver $S(p)$	1.07410 -0.1%	/	/	-5.8 <b>-7.9%</b>	-5.0 +2%	-612 -0.03%	-244 +0.5%

The results show that PK parameters are quite sensitive to the cross section preparation methodology (2D vs. 3D cross sections). In particular, the drawback of 2D cross sections is that they are not representative of the core environment which is quite heterogeneous in the axial direction<sup>¶</sup>. On the other hand, the sensitivity to the flux calculation method at the core level ( $SP_3$  vs  $S_N$ ) is rather low. In all cases, the most sensitive parameter is the sodium reactivity worth (up to 50% difference on  $\Delta\rho_{\text{Na}}^{1\%}$  when cross sections are prepared with an inconsistent method).

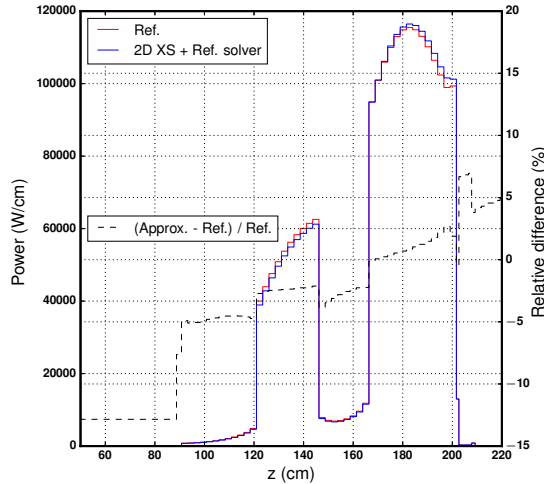
In the following and for the sake of simplicity, the discussions will be based on a comparison between the two sets of PK parameters that show the highest differences i.e. the reference set vs. the set produced with 2D cross sections. The set generated with the  $SP_3$  solver is dropped out in order to focus on these two “extreme” situations.

### 3.2.2. Spatial profiles

Fig. 2 now shows the power axial profiles computed with the reference and approximated (2D XS.) methodologies. While the profile shapes are similar, a fine analysis shows that significant differences are found between the two methodologies (5% difference at the top of the core and up to 12% difference in the lower fertile blanket). As explained in reference [4], this discrepancy comes from the inconsistency of the information that is used to perform the homogenization of cross sections in 2D models.

As for the reactivity feedback coefficients, the axial profiles (obtained from the decomposition of the integral in the numerator of Eq. 5) are presented in Fig. 3.

<sup>¶</sup>In CFV geometries, it has been shown that 3D angular information is required to perform the cross section homogenization (see Fig. 3 in reference [4])



**Figure 2: Comparison of power profiles**

The main difference between the reference and approximated methodologies is found for the sodium–void reactivity coefficient in the sodium plenum. As for the power profile, this discrepancy arises from the cross section homogenization with 2D weights. The Doppler profile is more robust to the neutronic calculation methodology.

### 3.2.3. Discussion

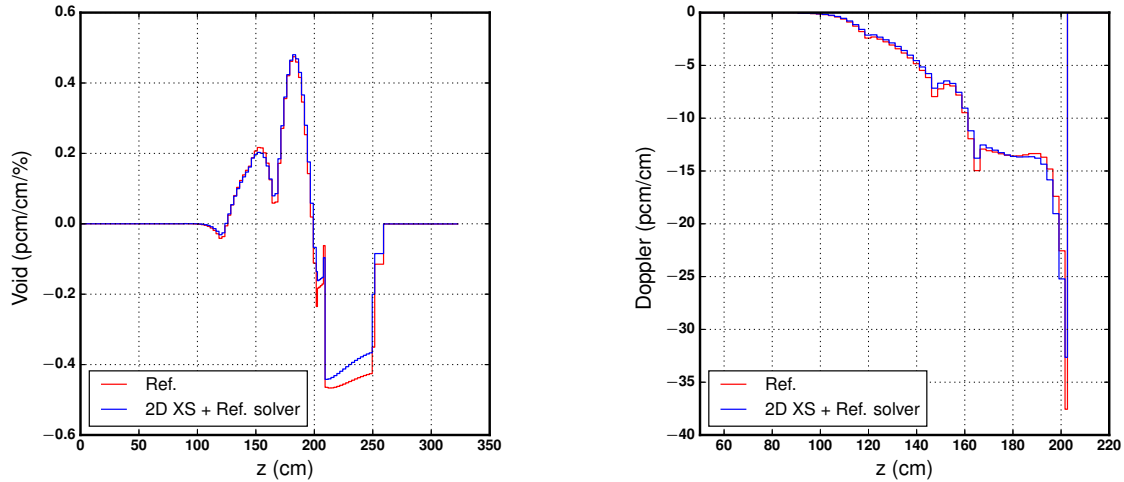
The PK parameters are quite sensitive to the quality of the cross sections that are used to solve Eq. (3). In particular, the sodium–void reactivity coefficient is the most sensitive parameter: depending on the methodology, up to 50% variation is found corresponding to  $\approx 2\%$  bias on the total void worth. Since this parameter is expected to drive the evolution of the reactivity during the ULOF scenario, it is likely then that significant differences will be observed on the outcome of such an accidental situation.

## 4. IMPACT ON TRANSIENT CALCULATION

In order to assess the veracity of this assumption, the impact of these two set of PK parameters (AP3 reference vs 2D XS approx) on the phenomenology of the ULOF scenario is investigated. Results are presented in Fig. 4 (evolution of the reactor power), Fig. 5 (reactivity) and Fig. 6 (evolution of temperatures).

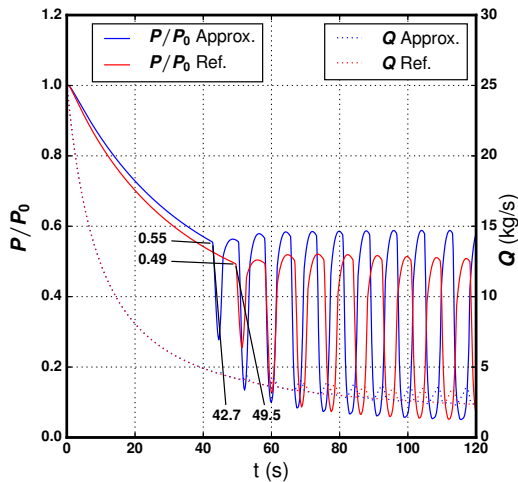
At time  $t = 0$ , the pumps of the primary circuit fail ( $\tau_{1/2} = 10$  s) resulting in an exponential loss of the coolant flow rate  $Q$ . During the first phase of the transient ( $t < 50$  s), the transient is monotonous: the reactivity insertion is negative (Fig. 5) and the power  $P$  decreases (Fig. 4).

Yet,  $\frac{P}{Q}$  remains high so the coolant finally reaches its boiling point at the top of the core. When the boiling crisis starts ( $t > 42.7$  s or  $t > 49.5$  s depending on the set of PK parameters), oscillations are observed on the reactivity, on the power, and on the temperatures. At this stage, the representa-

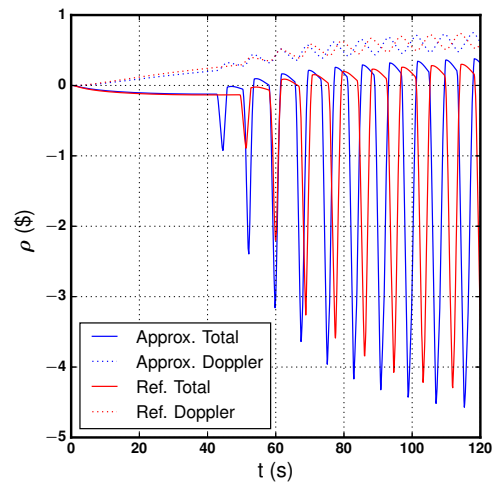


**Figure 3: Comparison of reactivity feedback coefficient axial profiles. Left: sodium-void reactivity worth  $\alpha_{Na}^{1\%}$ . Right: Doppler constant  $K_D$**

tivity of the benchmark (core composed of identical assemblies) is questionable because it implies that the boiling crisis occurs in all the assemblies whereas, in realistic scenari, the coolant reaches the boiling point only in a few positions. Thus, the oscillations that are observed are amplified in our model and cannot be trusted with confidence<sup>||</sup>. However, the first phase of the transient (before the boiling crisis) is quite representative of the phenomenology of an ULOF scenario in a CFV core [9]. Thus, conclusions can be drawn concerning the two PK parameter sets.

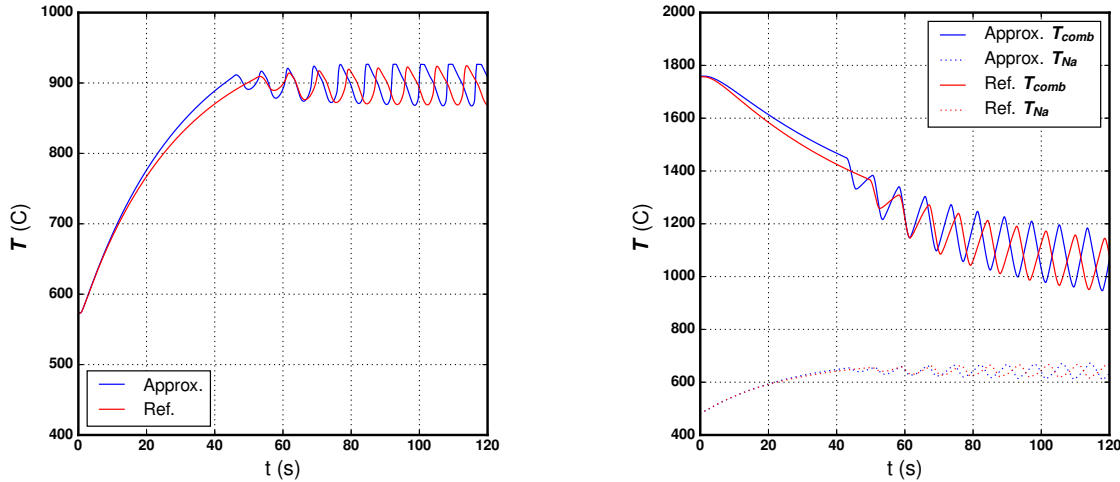


**Figure 4: Reactor power evolution**



**Figure 5: Reactivity evolution**

<sup>||</sup>In realistic situations, the amplitude of the reactivity oscillations ( $\approx 4\%$ ) is smaller. Besides, further investigations would be required to certify that these oscillations are “physical” and not “numerical”.



**Figure 6: Comparison of temperature evolution. Left: coolant temperature at core outlet. Right: fuel and coolant temperatures in the core (mid-height).**

As Fig. 4 shows, these two sets of coefficients lead to a similar phenomenology but to quantitatively different predictions for the state of the reactor when the boiling crisis starts: ( $t = 49.5$  s,  $P = 0.49P_0$ ) for the “reference” set and ( $t = 42.7$  s,  $P = 0.55P_0$ ) for the “approximated” set.

Yet, when these differences are confronted to the amplitude of the discrepancy of the PK parameters and in particular of the reactivity feedback coefficients (50% for the sodium–void reactivity coefficient, 7.4% for the Doppler constant), the conclusions need to be somehow mitigated.

## 5. CONCLUSIONS / PERSPECTIVES

This paper presents a first investigation of the impact of the neutronic calculation methodology for the evaluation of PK parameters to be used in a loss–of–flow transient simulation of a CFV core.

The methodology implemented in APOLLO3<sup>®</sup> is validated for the calculation of the PK parameters against reference Monte Carlo results proving its high accuracy. This reference methodology is then compared to simplified ones showing that the quality of the few group homogenized cross sections is decisive in the evaluation of the PK parameters. In particular, the sodium–void reactivity worth is the most sensitive coefficient (up to 50% difference).

These different sets of PK parameters are then supplied to the MACARENa system tool in order to simulate the ULOF itself. No significant differences is found on the transient phenomenology but the predictions for the beginning of the boiling crisis are quantitatively different. Yet, these differences are somehow limited compared to the large discrepancies found for the PK parameters.

From the safety point of view, a  $\approx 2\%$  bias on the total void worth (see Tab. 2) is not acceptable. Nevertheless, these results point out that, for transient simulations, there is no need to improve the PK neutronic calculation methodology beyond a certain threshold (e.g. 10/20% on PK parameters). Thus, the most significant progress margin is probably to be found in improving the neutronic /

thermo-hydraulic coupling model before upgrading the APOLLO3<sup>®</sup> strategy itself. Yet, this paper is a preliminary work and, even though independent confirmations have been obtained with the KGH approach, the conclusions still need to be strengthened with probabilistic sensitivity studies.

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