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F Martin, A Bergeron, G Campioni, Y Gorsse, N Greiner, et al.. Coupled Neutronics-Thermal-hydraulics Modelling of a Molten Salt Reactor: the Aircraft Reactor Experiment. M&C 2023 - The International Conference on Mathematics and Computational Methods Applied to Nuclear Science and Engineering, ANS - CNS, Aug 2023, Niagara Falls, Ontario, Canada, Canada. hal-04484208

HAL Id: hal-04484208

<https://hal.science/hal-04484208>

Submitted on 29 Feb 2024

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Coupled Neutronics–Thermal-hydraulics Modelling of a Molten Salt Reactor : the Aircraft Reactor Experiment

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ABSTRACT

The CEA multiphysics tool combining the deterministic neutronics code APOLLO3[®] and the CFD platform TRUST/TrioCFD is used to model the first ever built molten salt nuclear reactor, the Aircraft Reactor Experiment. A neutronics and thermal-hydraulics models are created and coupled. Steady-state simulations are performed, in order to reproduce experiments realised on the ARE. Simulations results and experimental data are compared as a way of validating the multiphysics tool. Some differences are observed, which are probably due to some simplifications in the numerical model.

KEYWORDS: MSR, multiphysics, ARE

1. INTRODUCTION

Molten Salt Reactors (MSRs) are a type of nuclear reactor where the fuel is a liquid salt, which acts also as the primary coolant. They present promising features : they operate at high temperatures and ambient pressure, present strong temperature feedbacks, and can sustain quick power variations. These reactors have been recognised as a GEN-IV reactor type [1].

The strong coupling between neutronics and thermal-hydraulics in MSRs — through both the fuel density feedback and the transport of delayed neutrons precursors — has required the development of specific multiphysics codes for MSRs studies. In CEA, a multiphysics tool combining the deterministic neutronics code APOLLO3[®] [2] — and notably its S_N - transport solver MINARET — and the CFD platform TRUST/TrioCFD [3] was developed. One step of the V&V roadmap for this coupled tool is the modelling of historical MSRs and the comparison between simulations results and experimental data.

The first ever built MSR was the Aircraft Reactor Experiment (ARE) at Oak Ridge National Laboratory in 1954 [4]. The initial goal of this project was to build a nuclear-propelled aircraft [5], so the compactness and excellent load-following capacity of a molten salt reactor made it a natural candidate. The ARE went critical in November 1954, and functioned for 221 hours before final shutdown. Many experimental data were gathered during its operation, and compiled in detailed reports [6–8].

In this paper, we briefly present the ARE (Sec 2), and our modelling of this reactor in APOLLO3[®] and TrioCFD (Sec. 3). Finally we present some compared results for steady-state simulations (Sec. 4).

2. DESCRIPTION OF THE AIRCRAFT REACTOR EXPERIMENT

The ARE core is contained in a metallic cylinder (120 cm in diameter and height). This cylinder is filled with hexagonal prisms made of beryllium oxide (Fig. 2 top). These BeO blocks act as the neutron moderator. The blocks in the center of the core have a 3.175 cm diameter circular hole in their center, and inconel tubes are inserted in these holes. The fuel salt flows from the top to the bottom of the core within the inconel tubes. The salt flow is divided into six parallel circuits, each of which passes 11 times through the core in a serpentine pattern, for a total of 66 fuel assemblies in parallel (Fig. 2 bottom). The six circuits then merge in the bottom salt collector, and the salt is cooled via two fuel-to-helium heat exchangers before returning to the core (Fig. 1).

The BeO blocks on the outer rings of the core have a 1.27 cm diameter circular hole, through which liquid sodium flows. The sodium acts as the moderator coolant. It flows from the bottom to the top of the core through the outer holes and the BeO porosities, and is then cooled via two sodium-to-helium heat exchangers (Fig. 1).

The fuel salt is a $\text{NaF} - \text{ZrF}_4 - \text{UF}_4$ mixture (53% - 41% - 6% mol). The uranium is 93.4% enriched in ^{235}U .

Four rods are present, one regulating rod (stainless steel) and three shim rods (B_4C) that offset the initial reactivity excess and allow for an emergency shutdown (Fig. 1).

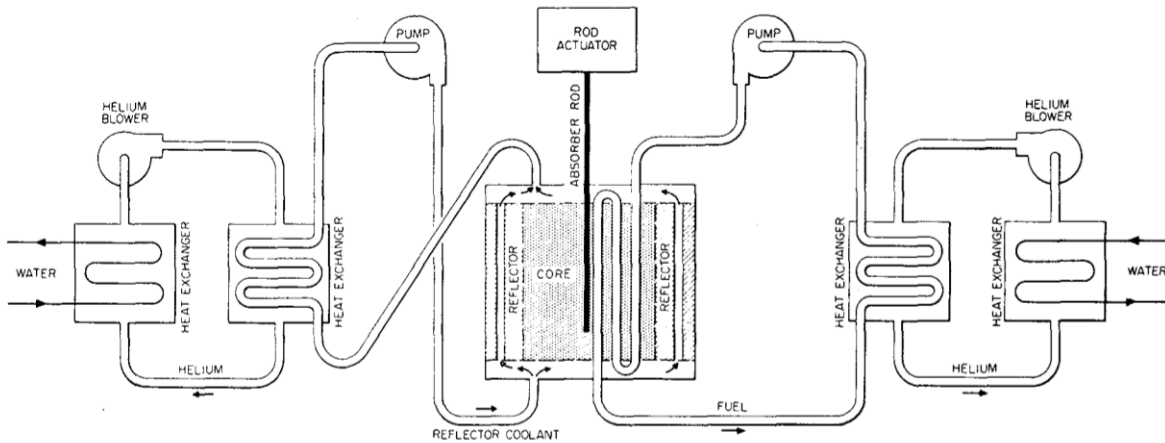


Figure 1: Schematic diagram of the ARE [6].

Multiphysics Modelling of the Aircraft Reactor Experiment

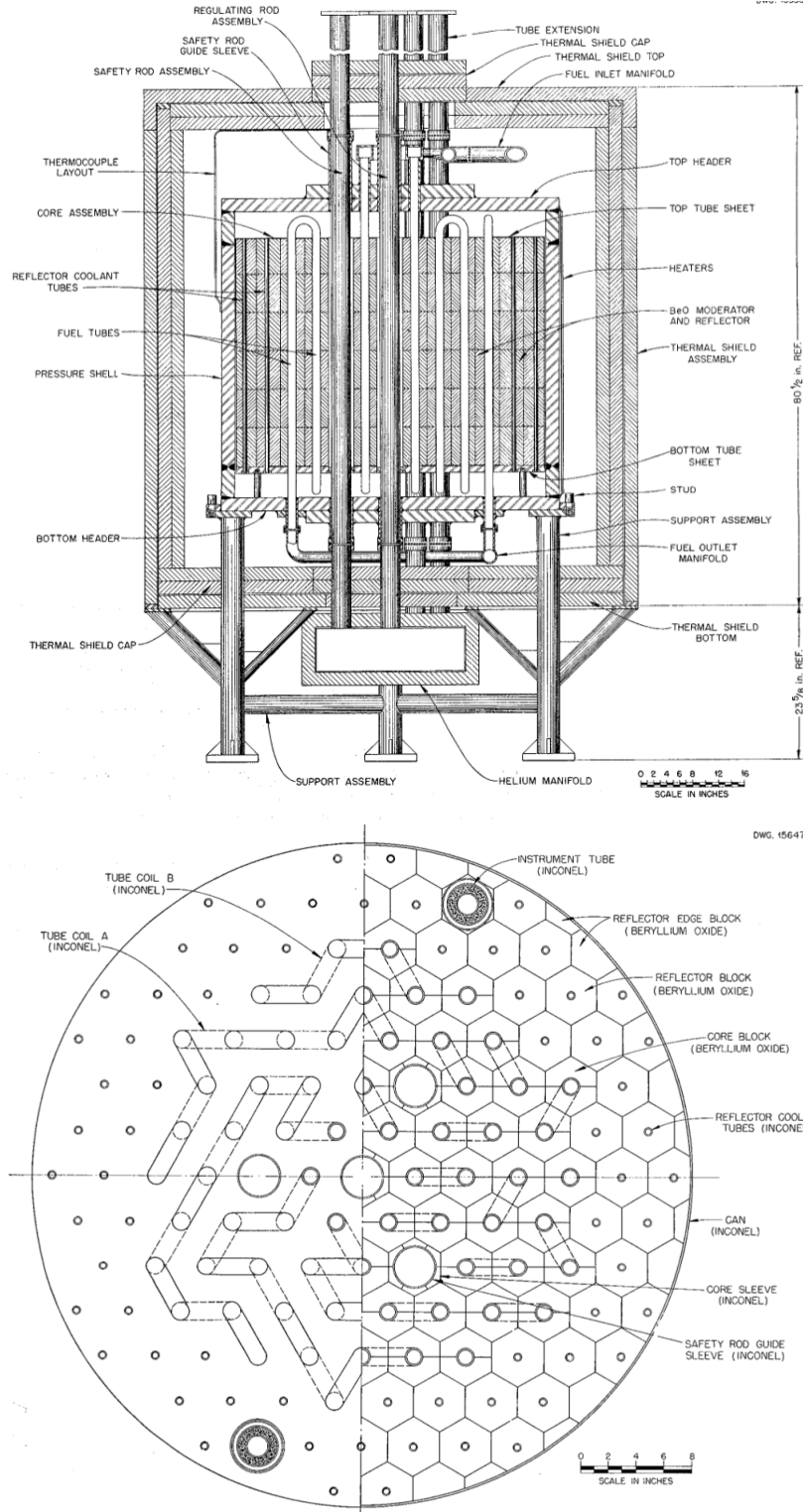


Figure 2: Elevation and plan sections of the ARE core [6].

3. MODELLING OF THE ARE

3.1. Resolved equations

For a molten salt reactor, the neutronics problem is different than for a solid-fuel reactor, because of the transport of the delayed neutron precursors. The neutronics problem is described in equation 1 :

$$\frac{1}{|v|} \frac{\partial \psi}{\partial t} = (-\mathcal{L}(T) + \mathcal{H}(T) + \mathcal{F}(T) + \mathcal{Q}(T, C_i))\psi \quad (1)$$

$$S_{C_i} = \int_{E_{min}}^{E_{max}} \int_{\mathbb{S}_2} \beta_i \nu \Sigma_f(T) \psi \, dE' \, d\omega'$$

With ψ the neutron flux, v the neutron speed, $\mathcal{L}, \mathcal{H}, \mathcal{F}, \mathcal{Q}$ the advection, scattering, prompt fission and delayed fission operators [9], which depend on the temperature field T and the precursor concentration fields C_i , given by the thermal-hydraulics calculation ; S_{C_i} is the precursor source field for the i -th precursor group, β_i the delayed neutron fraction for this group, ν the average number of neutrons created each fission, and Σ_f the macroscopic, temperature-dependant fission cross-section. The power density field is derived from the neutron flux and then used by the thermal-hydraulics code.

The thermal-hydraulic problem for the fuel salt is described in equation 2. The salt is assumed incompressible and Newtonian. Furthermore, the modelled flow is one-dimensional and laminar :

$$\frac{\partial u}{\partial x} = 0$$

$$\frac{\partial u}{\partial t} + u \frac{\partial u}{\partial x} = -\frac{1}{\rho} \frac{\partial P_h}{\partial x} + \mu \frac{\partial^2 u}{\partial x^2} + g(1 - \beta_{th})(T - T_0) + \frac{S_m}{\rho} \quad (2)$$

$$\frac{\partial T}{\partial t} + u \frac{\partial T}{\partial x} = a \frac{\partial^2 T}{\partial x^2} + \frac{S_{th}}{\rho c_p} + S_{exch}$$

$$\frac{\partial C_i}{\partial t} + u \frac{\partial C_i}{\partial x} = D \frac{\partial^2 C_i}{\partial x^2} + S_{C_i} - \lambda_i C_i$$

With u the salt velocity, P_h the hydraulic pressure, ρ the density, μ the dynamic viscosity, g the gravity acceleration, β_{th} the thermal expansion coefficient, T the temperature, a the thermal diffusivity, c_p the specific heat capacity, D the mass diffusivity, C_i the concentration field for the i -th precursor group, and λ_i the decay constant for this group.

For the momentum equation, buoyancy effects are taken into account through the Boussinesq approximation with a reference temperature T_0 ; the source term S_m corresponds to a regular pressure drop in the inconel tubes. For the energy equation, the source term S_{th} corresponds to the power density given by the neutronics calculation, and S_{exch} is a source term corresponding to the thermal exchanges with the other materials — sodium and solid. For the concentration equations — one for each precursor group — the source terms S_{C_i} corresponds to the precursor's source fields given by the neutronics calculation ; the radioactive decay of precursors is taken into account through the term $\lambda_i C_i$.

The sodium flow is modelled in a similar way : the regular pressure drop term is different, there is no thermal power source, and there is no concentration equation.

The solid moderator is modelled by a thermal conduction equation only, with thermal exchanges with the fuel salt and the sodium.

3.2. Neutronics Modelling

APOLLO3[®] uses a two-steps numerical scheme to solve the neutron flux equation (the Boltzmann equation) of the ARE core. In the first step, called the lattice calculation, it solves the equation on a two-dimensional, detailed geometry of a third of the core, and computes the macroscopic, self-shielded cross-sections for different temperatures (Fig. 3 top). Only a third of the core is used because the ARE presents a rotational symmetry by thirds. The boundaries conditions are symmetrical, or void on the outside of the third. The temperature-parameterised cross-sections are then regrouped in homogenised materials (called assemblies). In the second step, called the core calculation, we use these assemblies to build a three-dimensionnal, full core geometry (Fig. 3 bottom). The Boltzmann equation is then solved on this large, homogenised geometry, with vacuum boundary conditions. The lattice calculation is performed outside of the coupled calculation, only the core calculation is coupled to TrioCFD. For the core calculation, the energy is discretised into 20 groups, the angles into 80 directions (S_8 method), and the precursors into 8 groups.

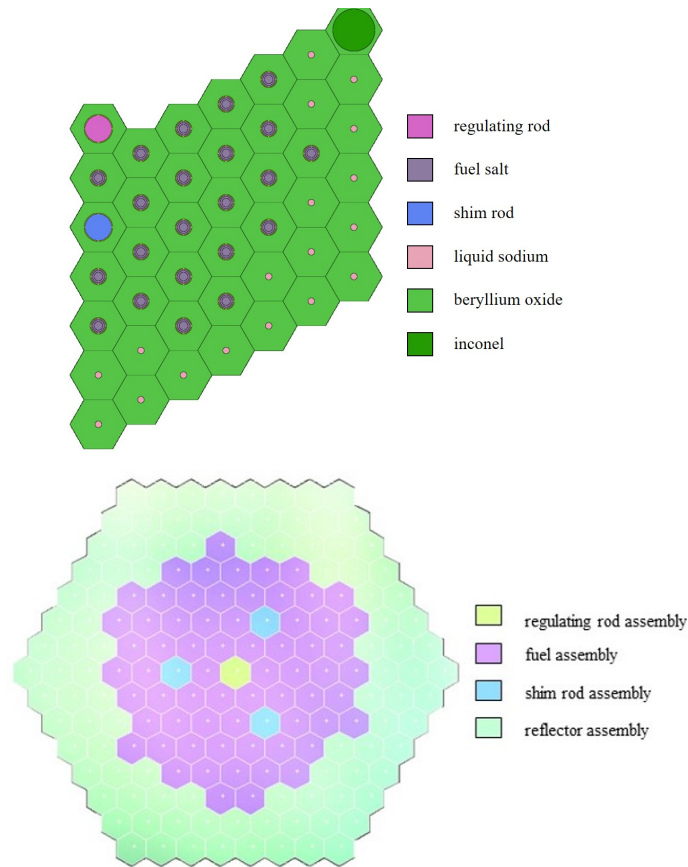


Figure 3: Geometries used for the lattice (top) and core (bottom) calculations.

For the ARE modelling, the assemblies used are a fuel assembly containing the fuel salt, the inconel tube, and the BeO block around the tube ; a reflector assembly containing the liquid sodium and the BeO block around it ; two different rods assemblies — one for the regulating rod, one for the three shim rods. All the assemblies cross-sections vary with the fuel temperature, the sodium temperature and the BeO temperature.

Because it is difficult to model the bends of the fuel tubes at the top and bottom of the core in APOLLO3[®], they are replaced by a homogeneous "paste" containing liquid sodium and fuel salt in the same proportions as with the bends. This dilution of the fuel in the bends is expected to lower the fission rate in these areas, and thus the total reactivity.

3.3. Thermal-hydraulics Modelling

Three different domains are used for the ARE thermal-hydraulics modelling : the fuel salt, the sodium, and the solid domain. The meshes for the three domains can be seen on Fig. 4. The solid domain is divided into two different zones corresponding to the BeO blocks and the inconel tubes, with different physical properties. The salt and sodium velocities are imposed at the inlet, the flows are supposed laminar. Adiabatic boundary conditions are used for the solid domain. Considering the long time the fuel salt spends out of the core — more than 30 s — we assume a null precursor concentration at the inlet. The properties of the different domains are summarised in Table I.

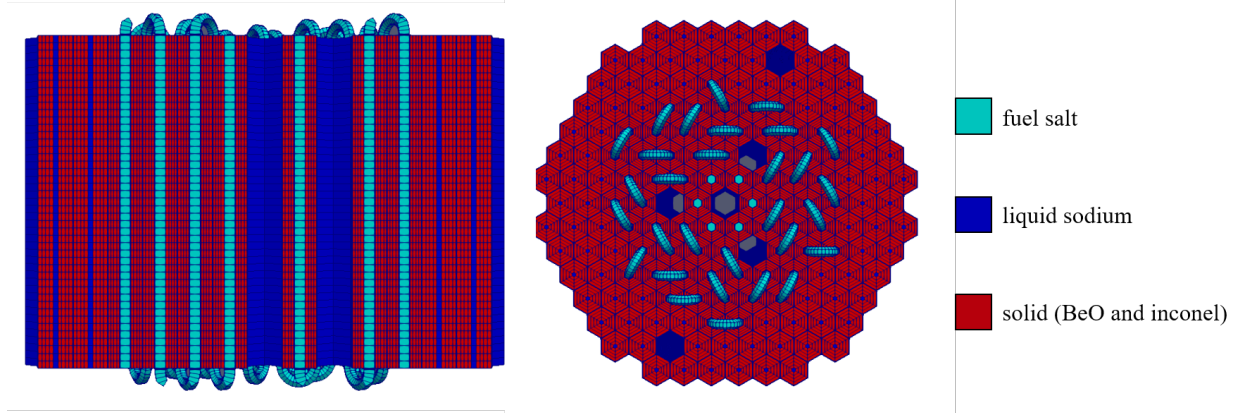


Figure 4: Elevation section and top view of the ARE meshing in TrioCFD .

Table I: Properties of the different materials in the thermal-hydraulic model.

Property	Unit	Material	Value
ρ	$kg.m^{-3}$	salt	3345
		sodium	738
		BeO	2800
		inconel	8510
c_p	$J.K^{-1}.kg^{-1}$	salt	1089
		sodium	875
		BeO	2010
		inconel	456
λ	$W.K^{-1}$	salt	2.6
		sodium	26.62
		BeO	1.3
		inconel	21.6
μ	$Pa.s$	salt	$1.25 \cdot 10^{-2}$
		sodium	$1.46 \cdot 10^{-4}$
β_{th}	-	salt	$3.01 \cdot 10^{-4}$
		sodium	$2.60 \cdot 10^{-4}$

3.4. Coupling

APOLLO3[®] is the first code in the numerical scheme. It computes the neutron flux, the power density field and the precursor source fields for an initial homogeneous temperature and a null precursor concentration. The power density field and the precursor source fields are sent to TrioCFD, which computes the salt and sodium velocities, the salt precursor concentrations, and the temperature fields for the three domains. The temperatures are then sent to APOLLO3[®] to update the macroscopic cross-sections, and the precursor concentration fields are used as a source term for the Boltzmann equation. Because the meshes for each code are different, the field exchanges and the remappings are handled by a python library called C3PO [10], developed in CEA, which uses the MEDCoupling library [11]. A functional diagram of the coupling scheme is presented in Fig. 5.

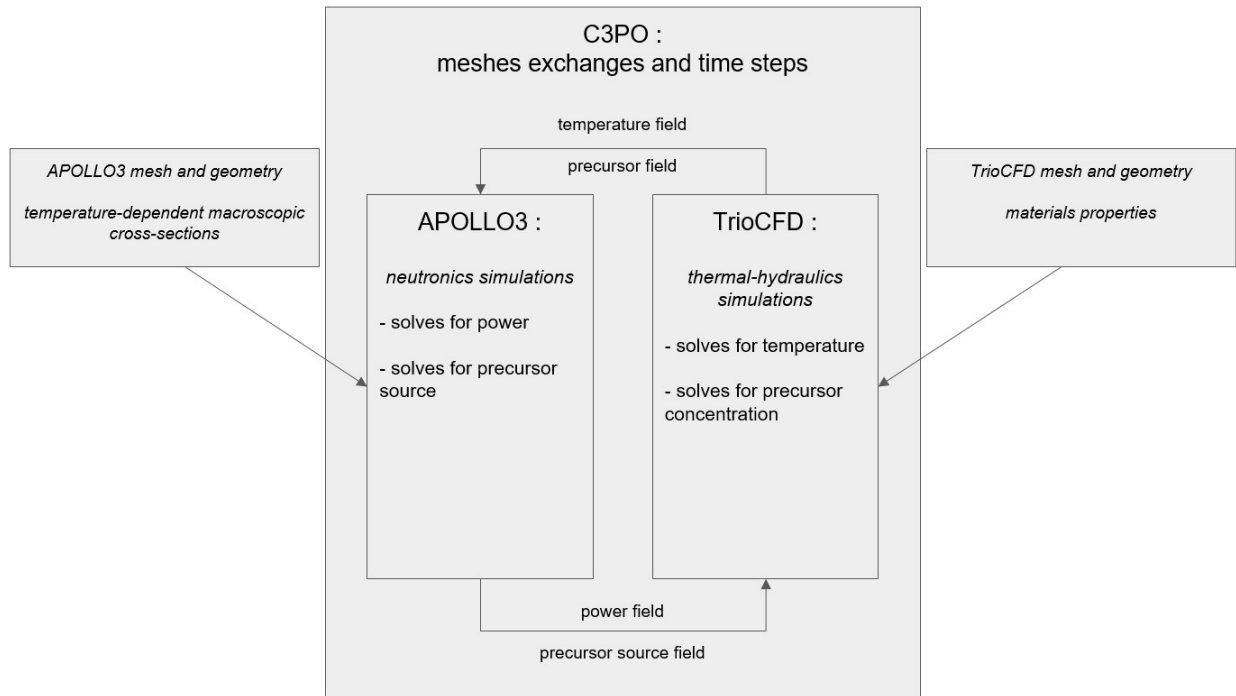


Figure 5: Diagram of the coupled code.

Because the self-shielded cross-sections for the salt depends on the three temperatures, and not just the salt temperature, and TrioCFD sends a null moderator temperature for the salt domain to APOLLO3[®] — there is no moderator in the salt domain — an interpolation is performed where the moderator temperature in a salt cell is equal to the mean of the surrounding moderator cells. The sodium temperature in a salt cell stays null, because the impact of the sodium temperature on the self-shielded salt cross-sections is negligible.

For steady-state calculations, each code reaches convergence before exchanging fields with the other code, until the difference between the exchanged fields at step n and $n - 1$ are below a threshold value.

4. RESULTS

4.1. Steady-state Calculations

For steady-state calculations, we study the ARE model in its nominal conditions (Exp. H-8 [6]). The total power is imposed at 2.1 MW, the fuel salt inlet temperature at 928 K, the salt flow at $3.10^{-3} \text{ m}^3.\text{s}^{-1}$, the sodium inlet temperature at 937 K, and the sodium flow at $10^{-2} \text{ m}^3.\text{s}^{-1}$. We then measure the fuel mean and outlet temperatures, as well as the sodium outlet temperature, and compare them to the experimental data. Results are shown in Table II.

Table II: Compared values for nominal operating conditions.

Measure	Experimental Value	Simulation Value
T_{fuel}^{mean} (K)	1102	1105
T_{fuel}^{out} (K)	1014	1020
T_{sodium}^{out} (K)	998	950

The important discrepancies for the sodium temperatures can be explained by our simplified sodium flow model. In `TriO`CFD, the liquid sodium flows only through the reflector holes and between the BeO blocks, while in the actual experiment it also flows through the BeO porosities, resulting in a much slower flow and a much higher outlet temperature. This can also explain the slightly higher salt temperature in the simulation : less heat is absorbed by the sodium, so it is absorbed by the salt instead. These sodium discrepancies can be partially corrected by reducing the sodium velocity in the numerical model.

4.2. Delayed neutron fraction variation

In order to measure a neutronics parameter, we choose to measure the variation of the effective delayed neutron fraction β_{eff} with the fuel flow (Exp. L-7 [6]). Because the precursors are transported out of the core by the salt, β_{eff} decreases when the fuel velocity increases. The L-7 experiment was realised at a low power (1W), so there is no thermal model in the `TriO`CFD model, only the fuel velocity and precursors concentrations are solved at a fixed temperature of 985 K. The neutronics power in `APOLLO3`[®] is imposed at 1W. Results are shown in Table III.

Table III: Compared values for the effective delayed neutron fraction variation.

Fuel flow ($\text{m}^3.\text{s}^{-1}$)	experimental $\Delta\beta_{eff}$ (%)	simulated $\Delta\beta_{eff}$ (%)
3.10^{-3}	0	0
0	0.398	0.357

The discrepancies could have two causes : the simplified, laminar and one-dimensional salt flow in the numerical model ; or the homogeneous "paste" in the bends at the top and bottom of the core. A 40 pcm difference is nonetheless satisfactory at this stage, and contributes to the validation of the code.

5. CONCLUSIONS

A numerical model of the ARE was created in APOLLO3®-TrioCFD, and first steady-states calculations have been performed in order to reproduce experiments made on the reactor. The overall results are quite satisfactory, given the simplifications made in the numerical model. Future work should focus on offsetting some simplifications made — in particular regarding the sodium flow — as well as performing transient simulations in order to measure temperature feedback coefficients.

ACKNOWLEDGEMENTS

APOLLO3® is a registered trademark of CEA.

F. Martin, A. Bergeron, G. Campioni, Y. Gorsse and N. Greiner thank EDF and Framatome for partial financial support.

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