INFLUENCE OF THE 3-D PHENOMENA ON THE SAFETY PARAMETERS DURING A ULOF ACCIDENT IN THE MYRRHA REACTOR

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ABSTRACT

In a pool type liquid metal cooled reactor like MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications), the influence of 3-D thermal-hydraulic phenomena during Loss Of Flow (LOF) transients can potentially have an important impact on safety-relevant parameters.
3-D temperature distributions and local pressure gradients may affect the evolution of the coolant mass flow during the transition from forced to natural convection, with the possible generation of flow instabilities and dissipating flows. Furthermore, the presence of stagnant volumes may influence the characteristic propagation time of perturbations through the system.
The 1-D computational system codes used to perform reactor safety analyses were originally developed for loop type reactor designs that foresee the coolant flowing in pipes, with energy losses mainly due to wall friction. Therefore, this class of codes is not validated to simulate correctly the physics of the phenomena occurring in a pool type reactor.
The objective of this work is to assess the shortcomings of 1-D system codes in predicting the response of the MYRRHA reactor to a LOF event, identifying the 3-D safety-relevant phenomena that can have an influence on the transient evolution.
The adopted strategy is comparing the transient simulation results of the RELAP5 thermal-hydraulic system code with reference CFD simulations. An ANSYS-CFX coarse-mesh CFD model and a RELAP5 1-D model of the MYRRHA primary system were built to perform the analysis.
The scenario selected for this study is an envelope case for the Unprotected Loss Of Flow (ULOF) accident.

KEYWORDS
MYRRHA, Loss of flow, CFD, system code, 3-D phenomena

1. INTRODUCTION

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is an experimental Accelerator Driven System (ADS) currently being developed at SCK•CEN (the Belgian Nuclear Research Centre). It is conceived as a flexible fast spectrum irradiation facility, able to run in both subcritical and critical modes. In subcritical mode, MYRRHA will demonstrate the full ADS concept by coupling a proton accelerator, a spallation target and a subcritical reactor at a reasonable power level to allow operational feed-back for the study of efficient transmutation of high-level nuclear waste [1]. Being based on heavy liquid metal technology (namely LBE, Lead-Bismuth Eutectic) and capable of running in critical mode, MYRRHA will also play the role of European Technology Pilot Plant in the roadmap for the development of the Lead Fast Reactor, as indicated by the Sustainable Nuclear Energy Technology Platform (SNETP) [2]. Its application catalogue includes fuel development for innovative reactor systems, material developments for GEN-IV systems and fusion reactors, radio-isotope production, doped silicon production, and (thanks to the high power proton accelerator) fundamental science applications.

Performing the safety assessment of such an innovative system inevitably poses challenges. Since MYRRHA is a pool type reactor, the traditional thermal-hydraulic 1-D system codes used for deterministic safety analyses may prove unsuitable for studying certain accidental scenarios. One-dimensional system codes like RELAP [3] were originally conceived for loop type reactors (e.g. Pressurized Water Reactors), in which the coolant flows in ducts so that momentum losses are mainly due to wall friction effect. It is evident, however, that in some components of MYRRHA the flow is far from being 1-D: the upper and lower plena are the most evident examples. The objective of this study is therefore to perform an assessment of the shortcomings of the RELAP system code in predicting the response of the MYRRHA primary system to transients.

This work focuses on a very specific envelope case for the Unprotected Loss Of Flow (ULOF) accident (see Section 5). In the absence of reliable experimental data, the predictions of the system code can be validated by means of comparison against a Computational Fluid Dynamic (CFD) model of the reactor primary system, specifically developed for this purpose. An integral part of the proposed objective is identifying the 3-D phenomena that can have an influence on the transient evolution and determining their effective impact on the safety-relevant parameters (namely, the fuel cladding temperature). Special emphasis must be placed on the first seconds of the transient: as the LOF accident is easily detected, it is important to ascertain the evolution of the fuel cladding temperature during the time needed for the protection system to intervene (typically 2-3 seconds).

This study was carried out using the RELAP5-3D [3] system code (without the use of 3-D components available in that code) and the CFD code ANSYS CFX 15.0 [4].

2. DIFFERENCES IN THE MODELING APPROACH OF SYSTEM AND CFD CODES

In system codes like RELAP the coolant flow is confined in ducts, therefore the velocity field is uniform and oriented in one direction and the momentum losses are mainly due to the interaction between fluid and solid structures. The fluid-dynamic of an incompressible fluid can be approximated to a 1-D problem, because average parameters represent well the fluid flow characteristics. Regular 1-D coarse control volumes are used to discretize the reactor system domain, like the one showed in Fig. 1.

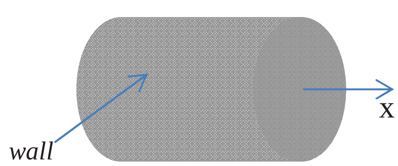


Figure 1. Typical RELAP5 control volume.

Considering the control volume in Figure 1, the RELAP 1-D momentum conservation equation [3] for an incompressible fluid can be simplified in:

$$\rho\frac{\partial U}{\partial t}+\rho U\frac{\partial U}{\partial x}=-\frac{\partial p}{\partial x}+\rho g-\underbrace{\boxed{\left(\frac{f}{D_h}+\frac{K}{\Delta x}\right)\rho U}}_{\textit{momentum diffusion towards the structure}} \quad (1)$$

where:

- U is the time-averaged velocity in the x-direction;
- f is the Darcy-Weisbach friction factor [3];
- D_h is the hydraulic diameter of the duct;
- K is the geometrical loss factor;
- Δx is the mesh size.

The momentum diffusion towards the structure is calculated using empirical correlations to estimate the Darcy-Weisbach friction factor and adding geometrical loss coefficients calculated by the modeler. This modeling approach cannot be valid to simulate non-confined flows, such as pool-flow, where the momentum losses are mainly due to 3-D convection and momentum diffusion between fluid threads. These additional terms of the momentum equation are correctly modeled by ANSYS-CFX. For an incompressible fluid moving preferentially in the x-direction the RANS equations [4] implemented in the CFX code can be simplified to:

$$\rho\frac{\partial U}{\partial t}+\rho U\frac{\partial U}{\partial x}=\underbrace{\boxed{-V\frac{\partial U}{\partial y}-W\frac{\partial U}{\partial z}}}_{\textit{momentum convection in the cross flow}}-\frac{\partial p}{\partial x}+\rho g+\underbrace{\boxed{\mu\left(\frac{\partial U^2}{\partial x^2}+\frac{\partial U^2}{\partial y^2}+\frac{\partial U^2}{\partial z^2}\right)-\rho\left(\frac{\partial\overline{u}^2}{\partial x}+\frac{\partial\overline{uv}}{\partial y}+\frac{\partial\overline{uw}}{\partial z}\right)}}_{\textit{momentum diffusion}} \quad (2)$$

where:

- V is the time-averaged velocity in the y-direction;
- W is the time-averaged velocity in the z-direction;
- $\mu\frac{\partial U^2}{\partial x_j^2}$ is the viscous stress term;
- $\rho\frac{\overline{\partial uu_j}}{\partial x_j}$ is the Reynolds turbulent stress term.

From the comparison of Equation (1) and Equation (2) it is clear that the RELAP 1-D modeling approach neglects three sources of momentum loss:

1) momentum convection in the cross-flow directions (y,z);
2) momentum diffusion due to viscous stresses;
3) momentum diffusion due to turbulent stresses.

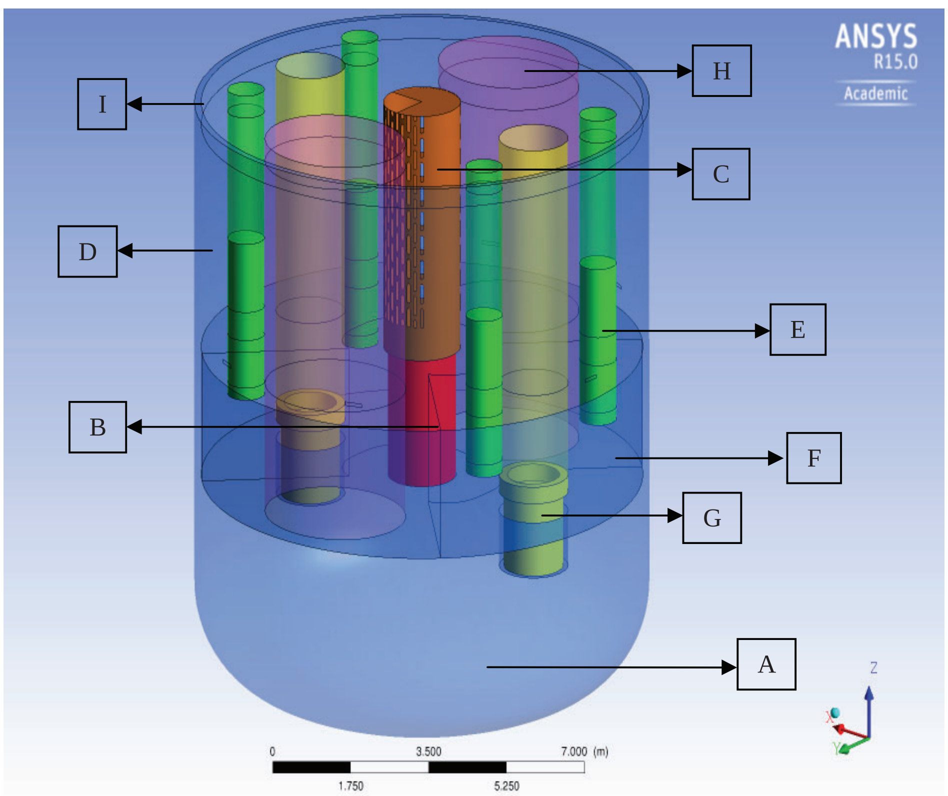
In the MYRRHA primary system we can identify two non-confined regions: the Upper Plenum (UP) and the Lower Plenum (LP) [1]: if the momentum losses in the plena are significant, RELAP5 overestimates the mass flow through the core and the Primary Heat Exchangers (PHXs). This can have important consequences on the predicted evolution of the safety parameters, because it influences the characteristic time response of the primary system and the heat exchange in the core and in the PHXs.

Hence, the proposed comparison between the 3-D CFX and 1-D RELAP models of the MYRRHA primary system aims at highlighting the possible impact of the momentum losses in the upper and lower plenum on the safety parameters.

3. ANSYS CFX MODEL IMPLEMENTATION

In pool type liquid metal cooled reactors, the reactor vessel houses the entire primary coolant (Lead-Bismuth Eutectic, LBE, in the case of MYRRHA) and many fundamental components that constitute the primary system. A detailed CFD model of the MYRRHA primary system is a challenging task which requires a very significant amount of modeling effort and computational resources (cf., for example, [5]). However, such a complex representation is out of the scope of the present work: a valuable insight into 3-D flow-paths and relevant thermal hydraulic phenomena can effectively be gained by means of a simplified, coarse-mesh CFD model which only takes into account the main components of the system and simplifies complex geometrical features with no substantial effect on the general flow patterns. Moreover, as the purpose of the study is performing a comparison against a RELAP model conceived to represent as closely as possible the reference CFD model, such simplifications facilitate the RELAP model development. As an added value, the coarse-mesh CFD model will constitute a flexible and relatively fast-running tool for performing conceptual design studies and analyzing other accidental events in which 3-D phenomena may play an important role (e.g. diaphragm break scenarios).

A picture of the CAD used for the CFD simulations, along with the specification of the considered components, is proposed in Figure 1. Some insight into the design of the core and primary system of MYRRHA is provided in reference [1]; although referred to a different design revision than the one considered in this work, the general description of the various components remains valid.



A – Lower Plenum
B – Core
C – Above Core Structure
D – Upper Plenum
E – Primary Heat Exchangers
F – Pump Boxes
G – Primary Pumps
H – In-Vessel Fuel Handling Machine
I – Annulus Structure

Figure 2. Simplified CFD model of the MYRRHA primary system.

The total number of mesh elements used for the computational grid is 1.23E+06; about 92% of the entire model volume is constituted by hexahedral elements. Mesh sensitivity analyses were carried out on stand-alone simulations of the main components, to make sure that a sufficient level of accuracy was achieved. Conjugate heat transfer through the main structures (e.g. the diaphragm which separates the hot plenum from the cold plenum) was taken into account by means of a thin material interface model [6], i.e. without explicit meshing of the actual thicknesses. All external boundaries of the model (i.e. reactor vessel wall and cover) were considered adiabatic, with the exception of a small opening on the top of the Above Core Structure (ACS, C in Figure 2), to avoid any (unrealistic) pressurization of the cover gas. The model is in fact two-phase, i.e. it reproduces both the LBE coolant and the cover gas above the free surface level. This is an essential feature for correctly simulating a LOF accident in the considered system, as the evolution of the free surface levels plays a decisive role at the beginning of transient (as explained in Section 6.2.1).

Porous media are commonly used in CFD, in order to model flows where the geometry is too complex to resolve with a grid (e.g. in rod or tube bundles). Their use becomes fundamental in a simplified and coarse-mesh model of MYRRHA, which cannot aim at the resolution of the flow field in the finest and densest structures of the primary system. However, the general consequence of the flow through these regions, i.e. the pressure drop due to friction, is correctly modeled by defining a porosity value and a resistance loss coefficient (calculated analytically on the basis of the characteristics of each specific region or component). The full porous model [4] has been used for the CFD implementation of the core, the four PHXs and the ACS.

3.1. Core Model

The core was modeled following a homogenization approach in radial zones. The maximum critical core was taken as a reference [7], considering 108 Fuel Assemblies (FA), 4 In-Pile Sections (IPS), 6 Control Rods (CR), 3 Safety Rods (SR), 42 inner dummies and 48 outer dummies (channels filled by LBE). As depicted in Figure 2, the CFD core layout was structured in 5 porous rings representing from the center: the inner FA, the combination IPS+CR+SR, the outer FA, the inner and the outer dummies.

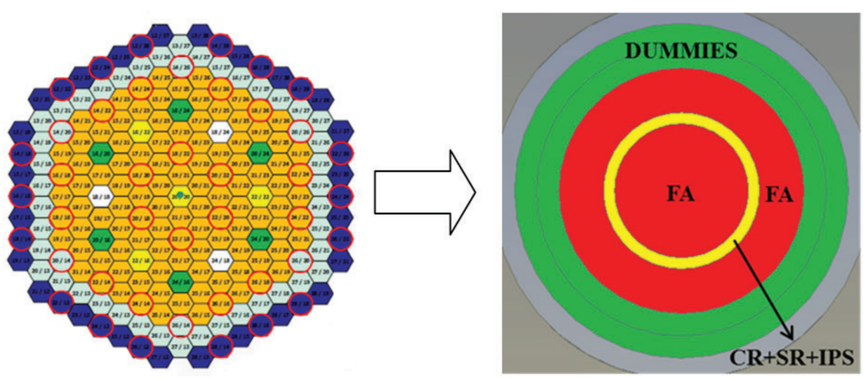


Figure 3. Maximum critical core layout (left) [7] and CFD implementation (right).

For the axial loss coefficient in the FA regions, the Rehme correlation for wire-wrapped fuel bundle was implemented [8]. The axial loss coefficient in the other two rings was kept constant and approximated on the basis of the expected mass flow rate distribution. To reflect real conditions, no radial flow was allowed in the core porous medium.

The heat source in the core was modeled as a volumetric heat source of 110 MW (envelope value for reactor power), with axial cosinusoidal profile and radial distribution calculated on the basis of the FA power peaking factors, obtained from neutronic design calculations. The heat-transfer correlation by Kazimi and Carelli [9] was implemented, as recommended in [10].

3.2. PHX and ACS Models

Homogeneous porous media were used to represent both the PHXs and the ACS. Stand-alone simulations of each component were performed in order to derive appropriate resistance loss coefficients, based on the expected mass flows and pressure drops.

The heat transfer between the primary and the secondary system in MYRRHA is represented by a variable heat sink located in the PHX porous media. The correlation developed by Ushakov for turbulent heat transfer over rod bundles [11] was implemented to evaluate the heat transfer coefficient on the primary coolant side.

3.3. Pump Model

Additional simplifications were adopted in the geometry of the pump internals: the rotor was not represented in the CFD model, but special care was devoted to reproducing (although in a simplified way) the flow restrictions along the internals, in order to introduce as few alterations to the flow field as possible. In order to reach steady-state conditions, the nominal mass flow during normal operation was simply imposed on the pump cross-section corresponding to the impeller location (i.e. in correspondence to the maximum flow restriction).

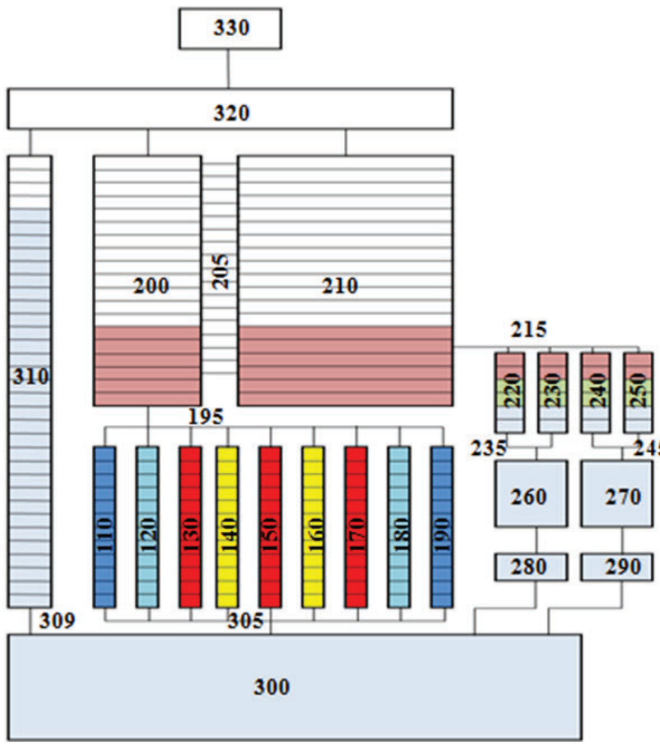
4. RELAP5-3D MODEL IMPLEMENTATION

The RELAP5-3D one dimensional model of the MYRRHA primary system is composed of 252 control volumes. Geometrical values are taken from the ANSYS-CFX model in order to limit as much as possible the influence of the input data on the comparative analysis. The nodalization developed for the MYRRHA primary system is shown in Figure 3. Effort has been devoted to keeping it as simple as possible (while maintaining the main geometrical features of the reference CFD model), in order to bring to evidence any potential discrepancy with respect to the CFD simulations. A description of the RELAP5-3D components can be found in [3].

From the cold plenum (single volume component 300) the LBE is distributed through the core channels by the multiple junction component 305. In accordance with the CFX discretization the core is modeled in several concentric channels. The heated pipe components are colored in red (components 130, 150 and 170). The other pipes represent the IPS, CR and SR channels (components 140 and 160), the inner dummies (components 120 and 180) and the outer dummies (components 110 and 190). The LBE streams exiting from the core mix in the ACS (component 200), connected to the core by the multiple junction component 195. The LBE spreads then radially into the UP (pipe component 210) across the barrel holes (multiple junction component 205). From the UP it flows down into the 4 PHXs (pipe components 220, 230, 240 and 250), through the inlet windows located at the top of the PHX shell (multiple junction component 215). After cooling down, it exits from the PHXs through the lower shell windows (multiple junction components 35 and 245) and it moves into the pump boxes (single volumes components). The LBE is then forced by the two primary pumps (pump components 280 and 290) into the cold plenum.

The pipe component 310 simulates both the cold stagnant LBE (Annulus) in-between the diaphragm and the reactor vessel and the LBE filling the In-Vessel Fuel Handling Machine (IVFHM) casings [1]. It is connected to the Lower Plenum by the junction component 309. The LBE fills the ACS (pipe component

200), the UP (pipe component 210) and the Annulus (pipe component 310) until the free surface level. Above the free surface level these components are filled with Argon. The top of these components is connected to the cover gas common space (component 320) and maintained at the atmospheric pressure by imposing a pressure boundary condition (component 330).



110, 190 - Outer Dummies
120, 180 - Inner Dummies
130, 170 - Outer FA
140, 160 - IPS+CR+SR
150 - Inner FA

200 - Above Core Structure
210 - Upper Plenum
220, 230, 240 ,250 - PHXs
260, 270 - Pump Boxes
280, 290 - Primary Pumps

300 - Lower Plenum
310 - IVFHM volume + Annulus
320 - Cover gas
330 – Pressure boundary condition

Figure 4. RELAP5-3D MYRRHA primary system nodalization.

The heat exchange between the LBE and the core, between the LBE and the PHXs and between the Annulus and the UP is simulated by coupling the components with thermal heat structures [3], in accordance with the ANSYS-CFX model.

In the confined flow regions (ACS, core and PHXs) the same correlations implemented in the ANSYS-CFX model are applied to calculate the Darcy friction factor and the heat transfer coefficient at the wall:

- the Rehme correlation [8] is used to calculate the friction factor in the core channels;
- the Kazimi-Carelli [12] correlation is used to calculate the LBE heat transfer coefficient at the pin wall;
- the Ushakov correlation [11] is applied to evaluate the heat transfer coefficient at the tube wall of the heat exchangers. The Kazimi-Carelli correlation implemented in RELAP5-3D, in fact, is not valid for the pitch-to-diameter ratio of the primary heat exchanger tubes. The parameters of the

correlation, which can be set by the user, have been tuned in order to reproduce the Ushakov correlation in the expected range of the Péclet number.

The secondary system is not modeled but a heat transfer coefficient and a bulk temperature are imposed on the heat structure simulating the PHX tubes.

Similarly to the ANSYS CFX implementation, the entire system is adiabatic, hence no heat transfer losses towards the reactor hall are considered.

5. UNPROTECTED LOSS OF FLOW: SCENARIO DEFINITION

The transient at study is a conservative envelope case for a ULOF accident: it postulates a simultaneous locked rotor condition on both primary pumps, with reactor power kept at its nominal value of 110 MW (i.e. unprotected conditions and no reactivity feedbacks considered). The probability of such an event is obviously very low (pertaining to the class of accidents excluded by design); nevertheless, it constitutes a convenient theoretical envelope case. If the cladding temperature remains at acceptable temperatures (i.e. below the failure criterion) over the time period needed for the protection system (scram) to intervene, studying more realistic but still conservative LOF scenarios (envisaging a coast-down period for the primary pumps) will not be necessary in the framework of the MYRRHA safety assessment.

In the CFX model, the locked rotor condition is simulated by switching off the interface flux parameter that imposes the nominal mass flow in the steady-state simulation and instantly replacing it by a pressure change condition across the same interface (i.e. the pump cross-section corresponding to the impeller location). Conservatively estimated loss factors (from a previous CFD simulation of the MYRRHA pump) for forward and reverse flow (K_f and K_r, respectively) were implemented to simulate the pressure drop across the primary pumps. Equivalent factors are applied in the RELAP5-3D model, to the pump inlet and outlet junctions. We define the described simulation as Case A.

An additional limiting case was studied, in which no pressure drops are imposed across the internals of the pumps and only the geometrical losses due to the variable flow cross section area (internally calculated by the CFX code) are taken into account. In this case, the contribution to the pressure drop given by the locked rotor is completely ignored. We define this as Case B.

Table I shows the loss coefficients used in the two cases, both for CFX and RELAP5-3D.

Table I. Pressure loss coefficients for the two transient simulations

SCENARIO	**CASE A**		**CASE B**	
CODE	**CFX**	**RELAP5-3D**	**CFX**	**RELAP5-3D**
K forward (K_f)	20	20	0	6.5
K reverse (K_r)	15	15	0	1.7

The values of the loss factors for Case B are not equal between the two codes: since RELAP5-3D cannot reproduce the losses due to the specific geometry of the pump internals, these pressure drops have to be modeled by means of additional friction coefficients. Empirical formulas from chapter 5 of [13] applied to the geometric characteristics of the MYRRHA pump were used to derive them.

6. RESULTS

6.1. Steady-state

The steady state results for full power operation conditions helped identifying possible sources of differences in the transient predictions between the two codes.

Table II lists the most relevant physical parameters evaluated by RELAP5-3D and CFX. The predictions of both codes show good agreement, falling within 2% of the expected values. We can conclude that the steady-state simulation represents a valid starting point for the comparative analysis of the ULOF transient.

Table II. Comparison of main thermodynamic values in steady-state

	RELAP5-3D	CFX	Expected
Lower Plenum average temperature	271 ˚C	270 ˚C	270 ˚C
Upper Plenum average temperature	324 ˚C	321 ˚C	325 ˚C
ΔT plena	53 ˚C	52 ˚C	55 ˚C
Average core exit temperature inner FA	405 ˚C	403 ˚C	408 ˚C
Average core exit temperature outer FA	347 ˚C	344 ˚C	349 ˚C
Maximum LBE core temperature	405 ˚C	405 ˚C	408 °C
Mass flow inner FA	2570 kg/s	2552 kg/s	2570 kg/s
Mass flow outer FA	5144 kg/s	5106 kg/s	5141 kg/s
Mass flow IPS+CR+SR	295 kg/s	300 kg/s	297 kg/s
Mass flow inner dummies	3081 kg/s	3085 kg/s	3089 kg/s
Mass flow outer dummies	2670 kg/s	2770 kg/s	2703 kg/s
Total mass flow	13788 kg/s	13813 kg/s	13800 kg/s

Non-uniform velocity and temperature distributions can be noticed in the CFX steady-state simulation, both in the ACS and in the UP. Looking at the UP temperature distribution depicted in Figure 4a, a vertical temperature stratification at the exit of the ACS can be remarked. This is due to the so-called 'plume effect': at the core outlet, the LBE exits from the various core regions at different temperatures; due to the high inertia, the coolant does not mix and follows separate paths in the ACS. The central hottest stream of LBE, with the highest inertia, enters the UP across the highest rows of holes in the barrel (the cylindrical structure that encloses the ACS, see Figure 1). The cold stream from the bypass channels (outer and inner dummies) has instead lower velocity and enters the plenum from the lowest rows of holes.

As it can be seen from the streamlines projections in Figure 4b, the LBE velocity field is not homogeneously distributed in the UP. A main pathway can be identified going from the barrel holes, around the IVFHM casings, to the diaphragm wall and then downwards, to the PHX inlet openings. There are two main stagnant zones: around the plane connecting the primary pump axes and in the space between the IVFHM casings and the diaphragm wall. The impact with the diaphragm slows down the coolant and enhances the mixing of the streamlines.

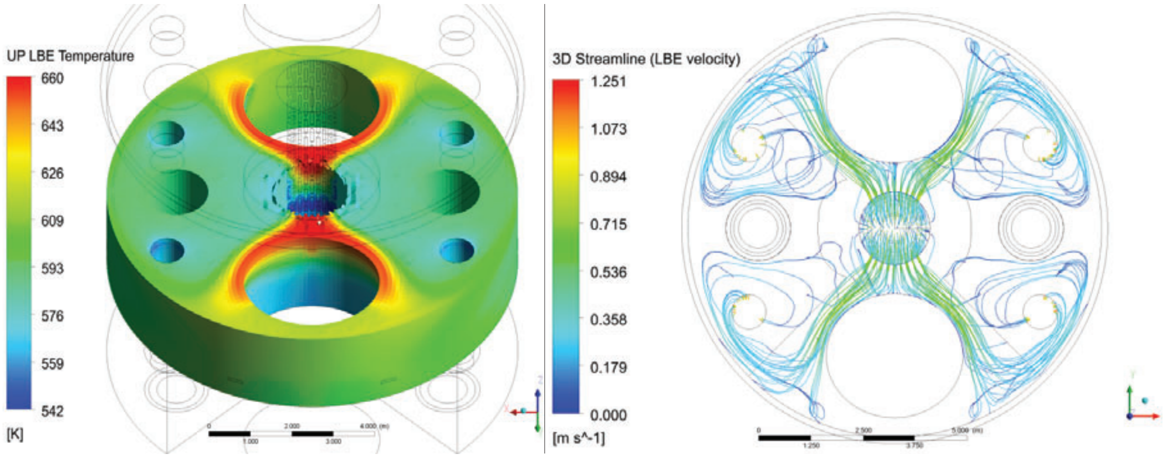


Figure 5. a) Steady-state UP temperature distribution. b) Steady-state streamlines projection.

The purpose of the transient analysis is therefore to ascertain if such 3-D phenomena (thermal stratification, stagnant volumes, recirculating flows) will influence the evolution of the ULOF event.

6.2. ULOF Transient

6.2.1. Preliminary considerations

In order to have a better understanding of transient simulation results, it is useful to refer to the sketch in Figure 6.

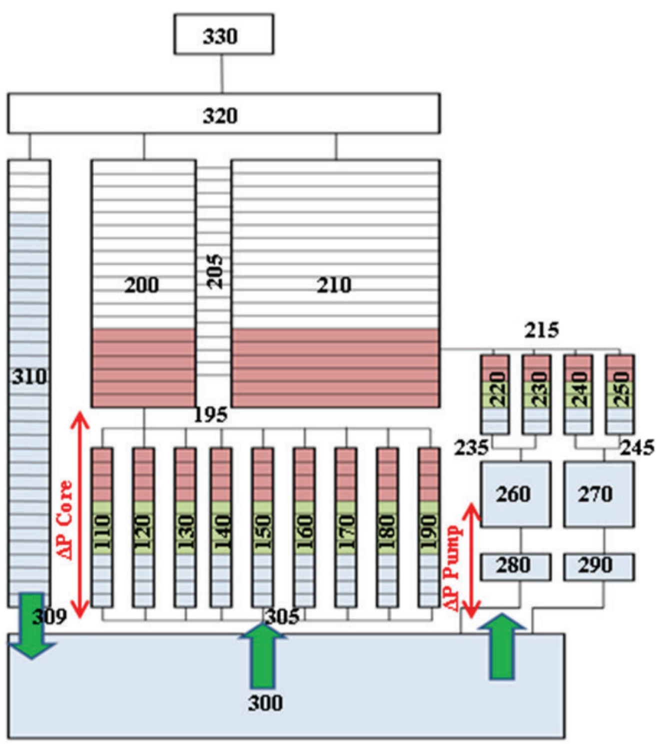


Figure 6. Parallel channels in MYRRHA reactor after the ULOF accident.

In case of locked rotor, the static pressure imposed by the 'cold leg' (component 310, simulating both the Annulus and the LBE inside the IVFHM casings) is no more counterbalanced by the pump head. Therefore, the cold LBE will flow downwards to the lower plenum (component 300) and then upwards, to the upper plenum (component 210), either via the core or via the alternative path across the pumps and the PHXs, until the levels in the two plena reach the equilibrium point.

If the pressure drops along the latter pathway are much lower than the ones along the FA channels, the coolant can bypass the core, with the subsequent risk of a loss of cooling and high cladding temperatures at which failure can occur. Furthermore, depending on the pressure drops in the system, the levels can oscillate around the equilibrium point or they can reach it monotonically. In case of level oscillations, it is possible to have a temporary stagnant flow in the core that may again jeopardize cooling. At the end of the transient, a new steady-state in natural circulation establishes in the primary system.

6.2.2 Result comparison

Figure 7 shows the evolution of the mass flows through the pump cross-section corresponding to the impeller location and through the central region of the core (inner FA) for Case A.

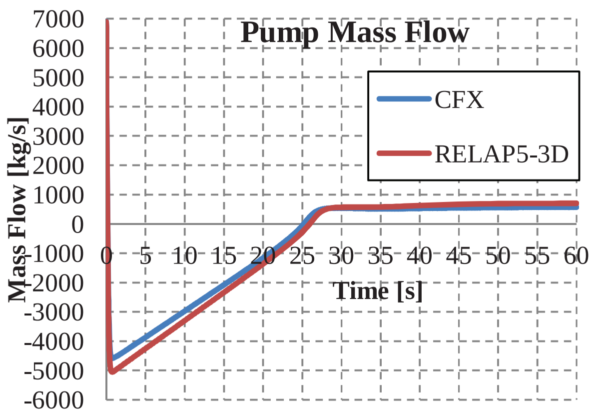


Figure 7a. Pump mass flow in Case A.

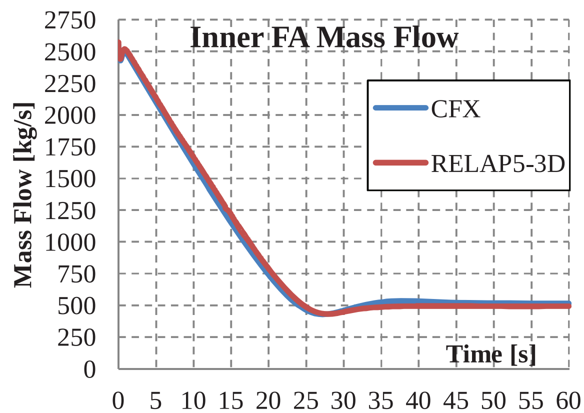


Figure 7b. Hottest core channel mass flow in Case A.

There are mainly two important effects that determine the evolution of the pump mass flow at the beginning of the transient: the inertia of the LBE and the static pressure exerted by the LBE in the cold plenum. Since the core represents a large pressure drop and the inertia of the LBE is almost negligible compared to the effect of the static pressure, there is an almost immediate flow inversion in the pumps (about 0.2 s after pump stop, see Figure 7a). Both codes reproduce very similar trends and predict very close mass flow values. The duration of the flow inversion phase in the pumps is also very similar: 26.0 s and 25.2 s are predicted respectively by RELAP5-3D and CFX (Figure 7a). After this phase, the transition from forced to natural convection ends and the mass flow stabilizes on the natural circulation value, with satisfactory agreement between the two codes.

The same considerations can be made for the mass flow evolution in the hottest core channel (Figure 7b). There is a limited undershooting in the mass flow with respect to its natural circulation value, but direct flow through the core is maintained throughout the duration of the transient.

The parameters maintain the same trend in Case B too, as showed in Figure 8.

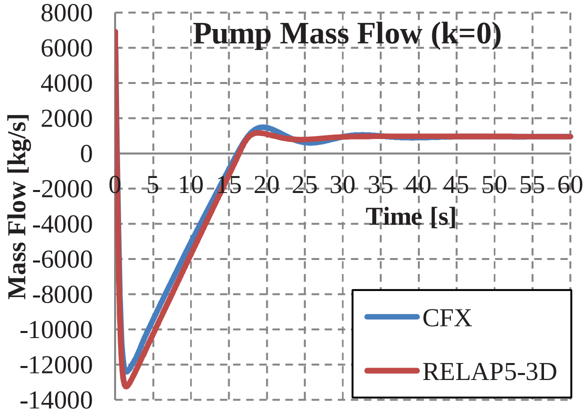


Figure 8a. Pump mass flow in Case B.

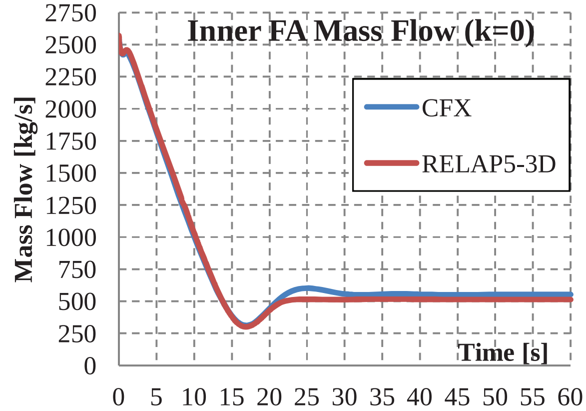


Figure 8b. Hottest core channel mass flow in Case B.

Because of the reduced pressure drops, the transient has a faster evolution (the natural circulation condition is anticipated by almost 10 s). This also explains the more pronounced oscillations in the mass flow values in the pumps and in the hottest core channel, expression of the decreased stability in the system.

Overall, the good match between the predictions of CFX and RELAP5-3D in transient conditions (even in Case B, where the pressure drops in the pump are limited to the geometrical losses) demonstrates that the momentum losses due to 3-D phenomena in the plena are negligible. The stagnant zone movement cannot be triggered because the velocities during the transient are always lower than the steady-state, thus no momentum is diffused from the main flow path to these regions.

6.2.3 Core coolability verification

The core coolability is verified in the time preceding the scram (supposed to occur in less than 5 seconds, considering the time needed for detection and actuation). The Peak Cladding Temperature (PCT) is the relevant safety parameter. Since the cladding temperature cannot be accurately estimated by the CFX porous medium model of the core, the maximum LBE temperature reached in the core during the first five seconds of transient is the selected parameter for our code-to-code comparison (Figure 9).

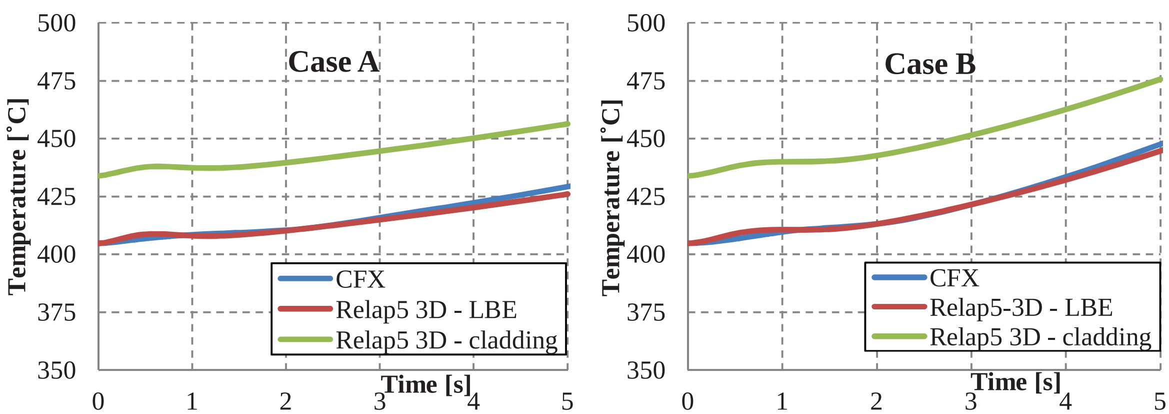


Figure 9. PCT and max LBE temperature evolution after ULOF accident in Case A and Case B.

The PCT predicted by RELAP5-3D is also plotted as a reference: it can be remarked that the temperature difference between the maximum LBE temperature and the PCT is always limited to less than 40°C for both Cases A and B. We recall here that Case B is the most conservative scenario, as a higher fraction of the total LBE mass flow bypasses the core, via the pumps.

The temperature evolution calculated by CFX and RELAP5-3D during the first seconds of the transient is almost superimposable. Crucially, the maximum LBE temperature in the core predicted by both codes remains below 450°C in both considered cases, with the peak cladding temperature about 30°C higher. No cladding failure should occur at such low temperatures. Therefore, even when the most (theoretical) bounding case (i.e. Case B) is considered, a sufficient grace time for the scram protection system to intervene is available and no cladding damage can occur.

7. CONCLUSIONS

Due to the peculiar characteristics of the MYRRHA reactor, assessing the suitability of the simulation tools adopted to determine the response of the system to postulated accidental events is an integral part of the safety assessment. In this framework, a comparative analysis to determine possible shortcomings of the RELAP5-3D system code in predicting the evolution of the safety parameters during the first phase of an ULOF event was carried out. A simplified, coarse-mesh CFD model of the MYRRHA primary system was used as a reference.

The main concern pertains to the use of 1-D system codes, originally developed for the safety analysis of loop type reactor designs, for transient evaluation of pool type reactors. In these applications, in fact, momentum loss terms such as turbulent or viscous stresses (which are neglected in RELAP-like system codes) may become important and substantially affect the evolution of safety parameters, like peak cladding temperature or the characteristic propagation time of perturbations in the system.

3-D phenomena such as thermal stratification, stagnant volumes and recirculating flows in the plena were identified in the steady-state CFD simulation, for full power operating conditions. This work aimed at assessing the effective impact of these phenomena on the ULOF transient, by comparing simulation results of two simplified models of the MYRRHA primary system, built with the RELAP5-3D system code (no 3-D components were used) and with the CFD code ANSYS CFX 15.0.

Results highlighted a remarkable agreement between the two codes for all the relevant parameters. It demonstrated that the momentum losses due to 3-D velocity and temperature distributions in the plena are negligible, so that most of the energy in the system is lost due to friction phenomena, which are correctly taken into account by RELAP. The suitability of RELAP5-3D to perform the safety evaluation of the defined envelope case for a ULOF transient (simultaneous locked rotor condition in both primary pumps) is therefore confirmed.

Lastly, a verification of the MYRRHA core cooling during the considered accident was carried out. The focus was set on the first seconds of the transient, in order to ascertain that the fuel cladding failure criterion is not exceeded during the time needed for the scram protection system to intervene. Even when the theoretical (unrealistic) envelope case for core cooling is considered (total absence of pressure drop induced by the locked rotor), the peak cladding temperature remains sufficiently low not to compromise the fuel cladding integrity. We conclude that, with the current design, the MYRRHA reactor can cope with any postulated LOF event during the time needed for detection and scram actuation (i.e. less than five seconds), with no damage to the fuel cladding.

GLOSSARY OF ABBREVIATIONS

ACS	Above Core Structure
CAD	Computer-Aided Design
CFD	Computational Fluid Dynamics
CR	Control Rod
FA	Fuel Assembly
IPS	In-Pile Section
IVFHM	In-Vessel Fuel Handling Machine
LBE	Lead-Bismuth Eutectic
LP	Lower Plenum
LOF	Loss of Flow
MYRRHA	Multi-purpose hYbrid Research Reactor for High-tech Applications
PCT	Peak Cladding Temperature
PHX	Primary Heat Exchanger
SR	Safety Rod
SNETP	Sustainable Nuclear Energy Technology Platform
ULOF	Unprotected Loss Of Flow
UP	Upper Plenum

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