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Development of CRONO: A System Code for Safety-Informed Design of Liquid Metal-Cooled Fast Reactors

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ABSTRACT

The design of innovative liquid metal-cooled fast reactors (LMFRs) is driven by the need to comply with high safety and reliability standards, while maintaining economic competitiveness through compactness and modularity. Owing to the complexity and limited accessibility of most existing system-level codes, ENEA is developing CRONO: a flexible system code to support safety-informed core design, i.e., a design-oriented system code. The code couples neutron kinetics, solved with the point kinetics approximation, with simplified thermal-hydraulic models of the core and the reactor coolant system components. Emphasis is placed on modularity and computational efficiency. This paper presents the governing equations for each component, the overall code architecture, and the adopted numerical solution strategy. A preliminary benchmark against the system codes RELAP5 and SIM-LFR and the fuel performance code TEMIDE has been performed for an unprotected transient of overpower scenario, showing good agreement in predicting reactor power and core temperature evolution during the transient. The developed tool is intended to support design via system-level accidental transients studies, stability analyses and sensitivity and uncertainty evaluations of LMFRs.

Keywords: Liquid Metal-Cooled Fast Reactors, Reactor Dynamics, Feedback, Design-Oriented Code

1. INTRODUCTION

Generation IV nuclear reactors have gained significant attention over the past decades since they embody the ambitions of enhancing sustainability through higher and more efficient fuel utilization, minimizing nuclear waste, and improving safety and reliability through stringent standards and passive safety features, while maintaining economic competitiveness. Among these, Liquid-Metal-cooled Fast Reactors (LMFRs), and particularly Lead-cooled Fast Reactors (LFRs), are considered one of the most promising technologies because of their fast neutron spectrum, closed fuel cycle, operation at atmospheric pressure, and high outlet temperature. The design of such reactors is also driven by the need to comply with demanding safety and reliability requirements. The adopted philosophy is that of built-in safety, guiding concept definition and design development through safety-related insights that actively influence the evolving configuration. To this end, the reactor response to selected transients and accidental scenarios is typically analyzed using system codes capable of modeling most of the relevant core phenomena.

In this framework, several system codes have been adapted for the analysis of LMFRs, such as RELAP5 [1], SIM-LFR [2], CATHARE [3], and SPECTRA [4]. However, these codes are often highly complex and require detailed design data that are not available in the early design stages. Furthermore, many of them are proprietary, limiting access to the source code and hindering development of tailored functionalities. In addition, their high computational cost makes them more suitable for safety assessment than for

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safety-informed design. Consequently, there is a need for faster and less detailed computational tools, as exemplified by the BELLA code [5]. The CRONO code (from the Italian acronym *Cinetica del Reattore per l'Ottimizzazione della Neutronica in Operazione*, which means "reactor kinetics for neutronics optimization in operation"), is therefore being developed, and it is conceived as a flexible and modular tool for the analysis of accidental transients, linear stability behavior, and sensitivity and uncertainty analyses of critical and subcritical LMFR systems.

In this paper the concept and structure of the code is presented, with a focus on the physical models implemented for each component. To qualify as a Design-Oriented Code (DOC), a very low computational cost must be achieved while maintaining reliable and consistent results. For this reason, the neutronic, thermal, and thermal-hydraulic models are all formulated in zero dimensions, with the exception of the core thermal analysis, for which a multi-point description is adopted. Accordingly, Point-Kinetics (PK) is coupled with lumped-parameter representations of the core, steam generator, and pump. The code is written in Python, taking advantage of its object-oriented capabilities and simplicity for future developments. The structure is highly modular, allowing the various components to be combined within a single reactor model.

The remainder of this paper is organized as follows. Section 2 describes the models implemented for each reactor component, with particular emphasis on the system of ordinary differential equations solved in each case. Section 3 presents a preliminary benchmark against RELAP5 and SIM-LFR for an Unprotected Transient of Overpower (UTOP) in the ALFRED [6] reactor. Finally, Section 4 discusses the obtained results and outlines future developments.

2. METHODOLOGY

The present methodology was developed aiming at taking the most profit from the implementation of the code within a modular Python package. Following a monolithic approach, the dynamic models for all components are formally coupled and assembled into a large set of Ordinary Differential Equations (ODEs). These equations are solved implicitly in time, using either a built-in *scipy* function or a custom numerical solver. The subsequent subsections present the specific models developed for each system component.

2.1. Core - Neutronics

As anticipated, PK was selected for the simulation of neutron kinetics since it is computationally efficient and can satisfactorily describe the dynamic behavior of neutron flux and precursor concentrations through a zero-dimensional representation of the core. The governing differential equations are presented below [7]:

$$\begin{cases} \dfrac{dP(t)}{dt} = \dfrac{\rho_{ins}(t) + \rho_{fb}(t) - \beta}{\Lambda} P(t) + \sum_{i=1}^{R} \lambda_i \tilde{C}_i(t) + \tilde{S}(t). \\ \dfrac{d\tilde{C}_i(t)}{dt} = -\lambda_i \tilde{C}_i(t) + \dfrac{\beta_i}{\Lambda} P(t) \quad i = 1, \ldots, R. \end{cases} \tag{1}$$

Here, $P(t)$ represents the total core power ($P(t=0) = P_0$), while ρ_{ins} and ρ_{fb} are the inserted and feedback reactivities, respectively. The first ODE exhibits non-linearity due to the dependence of the feedback reactivity ρ_{fb} on core temperatures, which in turn are coupled to the core power $P(t)$. This relationship establishes a strong coupling between the neutronics model and the core thermal-hydraulic model, detailed in Section 2.2.

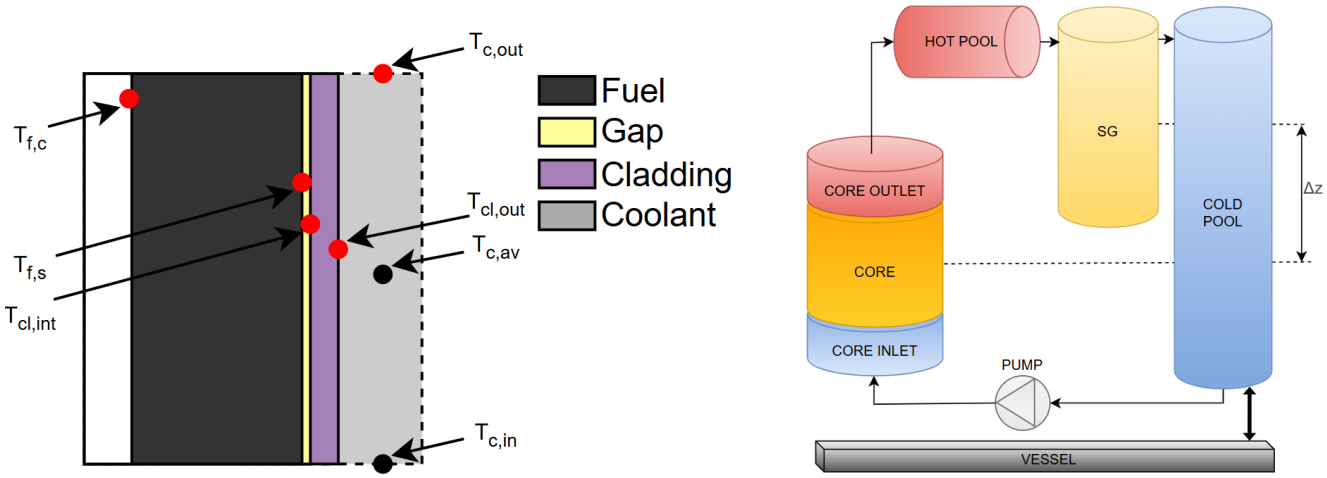
The feedback reactivity is computed through the integration of its differential form, as defined by the sum of contributions from various temperature-dependent effects:

$$d\rho_{fb} = \frac{K_D}{T_f} dT_f + \sum_i \alpha_i dT_i \tag{2}$$

where T_f is the fuel temperature, K_D is the Doppler constant, and α_i are the linear feedback coefficients, each associated with its specific driver temperature T_i.

2.2. Core - Thermal-hydraulics

For the core thermal-hydraulic analysis, multiple 0-D models were investigated and implemented. The model selected, which provides the most accurate transient response, is a core heat transfer model adapted to a 5-node average-pin discretization. The pin's radial geometry is spatially discretized into five thermal nodes to correctly capture the thermal inertia inherent to the radial heat propagation from the fuel to the coolant. In this approach, the entire core is approximated as a single average pin responsible for generating the total core power. The five thermal nodes are located at the center and surface of the fuel, at the inner and outer surfaces of the cladding, and at the coolant channel, as shown in Figure 1a.



(a) Five-nodes radial discretization of the average fuel pin for the core thermal-hydraulics model.

(b) Lumped-parameter schematic of the Reactor Coolant System.

Figure 1. Schematic representation of the reactor thermal-hydraulic modeling elements.

The governing system of ODEs, derived from the energy balance for each node, is presented below:

$$\begin{cases} \frac{m_f C_f(T_{f,av})}{2}\frac{dT_{f,c}}{dt} = \frac{P(t)}{2} - \frac{T_{f,c}-T_{f,s}}{2R_f(T_{f,av})} \\ \frac{m_f C_f(T_{f,av})}{2}\frac{dT_{f,s}}{dt} = \frac{P(t)}{2} + \frac{T_{f,c}-T_{f,s}}{2R_f(T_{f,av})} - \frac{T_{f,s}-T_{cl,int}}{R_g(T_{g,av})} \\ \frac{m_{cl} C_{cl}(T_{cl,av})}{2}\frac{dT_{cl,int}}{dt} = \frac{T_{f,s}-T_{cl,int}}{R_g(T_{g,av})} - \frac{T_{cl,int}-T_{cl,out}}{R_{cl}(T_{cl,av})} \\ \frac{m_{cl} C_{cl}(T_{cl,av})}{2}\frac{dT_{cl,out}}{dt} = \frac{T_{cl,int}-T_{cl,out}}{R_{cl}(T_{cl,av})} - \frac{T_{cl,out}-T_{c,av}}{R_c(T_{c,av})} \\ m_c C_c(T_{c,av})\frac{dT_{c,out}}{dt} = \frac{T_{cl,out}-T_{c,av}}{R_c(T_{c,av})} - \dot{m} C_c(T_{c,av})(T_{c,out}-T_{c,in}). \end{cases} \tag{3}$$

The temperature variables are shown in Figure 1a. The symbol m_i indicates the total mass of the i-th material in the active core, while $\dot{m}$ is the coolant mass flow rate in the core.

The fuel is discretized into two volumes on which the energy balance equation is volumetrically integrated. To better capture thermal inertia, each volume accounts for half of fuel mass and, consistently, half of the total thermal power generated (Eqs. 1 and 2). Analogously, for the cladding.

The material properties, such as specific heat capacity (C_i) and thermal resistance (R_i), are temperature-dependent, following established correlations. This dependency, particularly the thermal conductivity of the MOX fuel hidden within R_f, introduces non-linearity to the system.

The previous set of equations models the behaviour of the average pin, providing the average temperatures in the core, which affects the neutronic feedback. However, given that CRONO is mandated to provide safety-critical informations, it is also imperative to determine the maximum fuel and cladding temperatures within the core. This requirement necessitates the solution of an analogous set of thermal-hydraulic equations. This parallel set incorporates appropriate radial and axial peaking factors for the power and a distribution factor for the mass flow rate to model the localized conditions of the hottest channel and pin.

Given the narrow geometry of the gap, a notable error in the thermal resistance arises if only the cold gap thickness is considered. To account for this, a simple gap conductance model was implemented. This model computes the gap resistance R_g by taking into account the initial gap closure (caused by swelling, densification and relocation), the relative thermal expansion of fuel and cladding, the radiative heat transfer, the gas mixture composition and the roughness of the surfaces:

$$R_g = \frac{1}{\left(\dfrac{k_{mix}(T_{g,av})}{t(T_{f,av}, T_{cl,av})} + h_{rad}(T_{f,s}, T_{cl,int}) \right) (2\pi r_f h_{act} N_{pins})} \tag{4}$$

where k_{mix} is the thermal conductivity of the gas mixture and t is the actual thickness of the gap. For all other thermal resistances, expansion effects were not incorporated, as their impact is considered negligible.

2.3. Reactor coolant system

The reactor coolant system is modeled using a lumped-parameter approach, whereby the coolant flow is discretized into five interconnected control volumes. This simplification allows to capture the thermal inertia of the coolant within the system, which is essential for accurate transient feedback simulation, while maintaining computational efficiency. These five volumes - core outlet, hot pool, steam generator (SG), cold pool and core inlet - are schematically represented in Figure 1b.

The set of governing ODEs for the temperature change in each volume is based on the energy balance:

$$\begin{cases} m_{CO}\, C_c(T_{CO,av}) \dfrac{dT_{CO}}{dt} = \dot{m}\, C_c(T_{CO,av})(T_{c,out} - T_{CO}) \\ m_{HP}\, C_c(T_{HP,av}) \dfrac{dT_{HP}}{dt} = \dot{m}\, C_c(T_{HP,av})(T_{CO} - T_{HP}) \\ m_{SG}\, C_c(T_{SG,av}) \dfrac{dT_{SG}}{dt} = \dot{m}\, C_c(T_{SG,av})(T_{HP} - T_{SG}) - P_{sec} \\ m_{CP}\, C_c(T_{CP,av}) \dfrac{dT_{CP}}{dt} = \dot{m}\, C_c(T_{CP,av})(T_{SG} - T_{CP}) \\ m_{CI}\, C_c(T_{CI,av}) \dfrac{dT_{CI}}{dt} = \dot{m}\, C_c(T_{CI,av})(T_{CP} - T_{CI}) \end{cases} \tag{5}$$

In this system, $\dot{m}$ is the coolant mass flow rate within the reactor coolant system (currently assumed equal in each control volume) and C_c is the temperature-dependent specific heat capacity of the coolant. The masses m_{CO} and m_{CI} represent the coolant located in the plena immediately above and below the core active region, respectively. The symbols m_{HP} and m_{CP} represent the coolant mass which flows from the core outlet to the SG inlet (hot pool) and from the SG outlet to the core inlet (cold pool), respectively, while m_{SG} accounts for the coolant mass within the primary side of the SG. P_{sec} is the power exchanged with the secondary system, which is treated as a given input parameter for the simulation. As a closure relationship, the core inlet temperature $T_{c,in}$ used in the core thermal-hydraulic model (Section 2.2) is set equal to the temperature T_{CI} of the core inlet volume ($T_{c,in} = T_{CI}$).

The dynamics of the mass flow rate $\dot{m}$ is governed by the conservation of momentum, which accounts for inertial terms, concentrated and distributed pressure losses across the core and steam generator, buoyancy effects (natural circulation), and pump head. At this stage of implementation, the relative head between the coolant in the hot and cold pools is neglected, which justifies the omission of the hydrostatic term in the momentum conservation equation, consistent with the use of a single differential equation for the whole primary mass flow rate. The resulting differential equation yields:

$$\left(\sum_i \frac{L_i}{A_i}\right)\frac{d\dot{m}}{dt} = \rho g\, H_{pump}(t) + (\rho_c - \rho_h)\, g\, \Delta z - \sum_j^{conc} K_j \frac{\dot{m}^2}{2\rho A_j^2} - \sum_k^{distr} f_k \frac{L_k}{D_{h,k}} \frac{\dot{m}^2}{2\rho A_k^2}. \tag{6}$$

where ρ is the coolant density - with subscripts c and h referring to the cold and hot pool conditions, respectively -, H_{pump} is the pump head and Δz is the difference in elevation between the core and the SG, as represented in Figure 1b. In the event of a pump trip, the inertia of the pump and its components must be considered. To model this effect, the pump head decay is modeled exponentially, based on a characteristic time.

The thermal-hydraulic components and the coupled neutronics model described thus far represent the minimum set of systems required to accurately and efficiently simulate and analyze the safety-relevant accidental transients of interest. Specifically, this modeling scope is designed to effectively capture the reactor's inherent response during demanding Design Basis Accidents (DBA) and challenging Design Extension Conditions (DEC), such as Unprotected Loss of Offsite-Power (ULOOP) and Unprotected Transient Over-Power (UTOP) events [6].

3. RESULTS

This section presents a preliminary benchmark of the CRONO system code against RELAP5 and SIM-LFR for an UTOP event in the ALFRED reactor [6, 8]. The primary objective is to assess the reliability of CRONO in predicting core power evolution and the maximum fuel and cladding temperatures, which are safety-critical parameters. Furthermore, the fuel performance code TEMIDE [9] is employed for an additional comparison of the computed temperature evolutions.

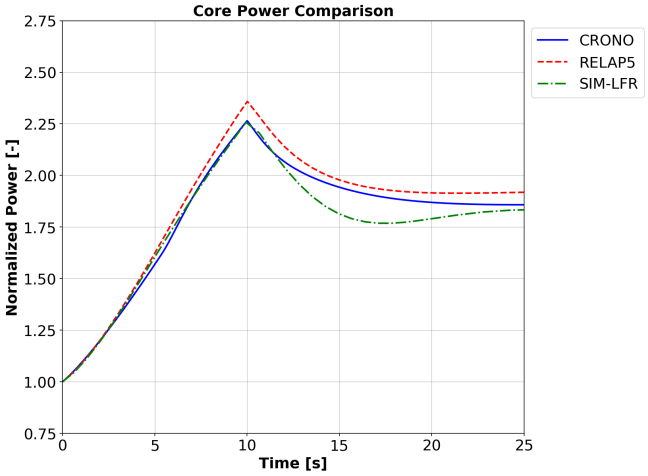
3.1. Benchmark against RELAP5 and SIM-LFR

The CRONO simulation adopted the following input parameters and assumptions to ensure consistency with the other two system codes:

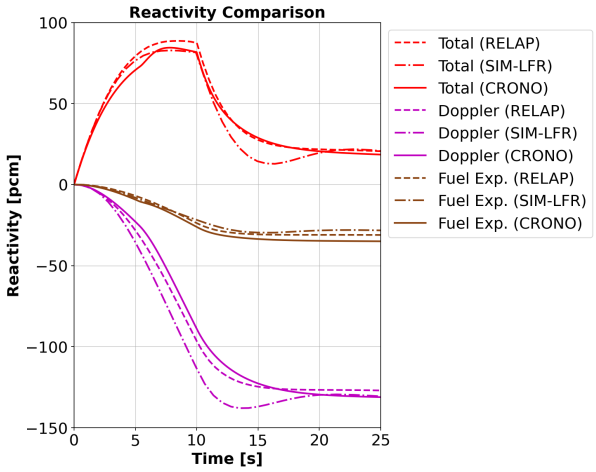
- fuel-cladding linked at 75%;
- gap conductance model activated;
- End-of-Life (EoL) state of the fuel;
- precursors in equilibrium with the nominal power.

To ensure consistency with previous simulations, the UTOP is induced by a linear insertion of 250 pcm of reactivity in 10 s. Due to the lack of input parameters related to the design of the reactor coolant

boundary, apart from the core, all the results are displayed in the first 25 s of the transient, since it is the range of time where the core thermal inertia plays the major role. The evolution of the normalized reactor power for all three system codes is presented in Figure 2a. CRONO adequately reproduces the overall power behavior, showing closer agreement with RELAP5 calculation and a maximum relative difference with the two codes of 5%. The predicted power peak, approximately 2.26 times the nominal power, coincides with the one predicted by SIM-LFR. The total reactivity and the dominant feedback components - the Doppler effect and fuel axial expansion - are compared in Figure 2b. CRONO accurately captures the total reactivity evolution. Also in this case, the predicted reactivity is very close to the one predicted by RELAP5. The peak core temperatures are displayed in Figure 3, where an overall



(a) Normalized power.



(b) Total, Doppler and fuel expansion reactivities.

Figure 2. Comparison between CRONO, RELAP5 and SIM-LFR on the power and reactivities evolution.

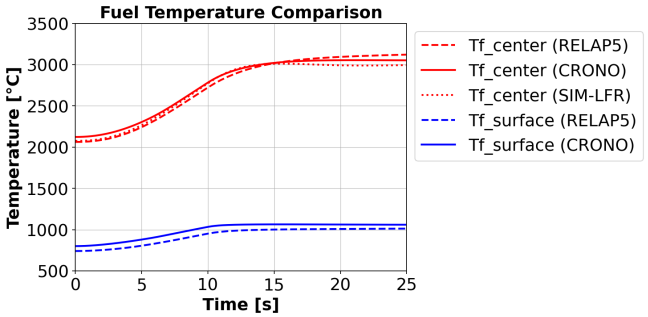
good agreement with the other system codes can be appreciated. In CRONO, the gap conductance in the hot pin is slightly underestimated at the beginning of the transient, resulting in a shift-up of the fuel temperature curves (Figure 3a), but the inertias and the behaviour of the gap during the transient are satisfactorily reproduced. Regarding the cladding temperatures (Figure 3b), a strong similarity is observed with RELAP5 results. Conversely, a notable difference in the thermal inertia of the fuel-cladding heat exchange is apparent between the CRONO and SIM-LFR predictions. This is caused by the different models for the gap conductance, which results in a lower value for SIM-LFR case and so in a slower heat transfer from the fuel to the cladding. In this transient, the cladding temperature primarily influences the fuel axial expansion feedback due to the partial mechanical linkage between fuel and cladding; the direct feedback from cladding thermal expansion is negligible. CRONO also correctly predicted the maximum fuel and cladding temperatures in the core during the transient, which is a fundamental output from the safety viewpoint.

3.2. Benchmark against TEMIDE

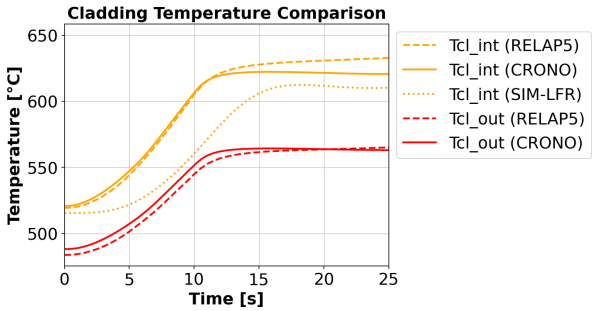
A further comparison is performed against the TEMIDE fuel performance code, specifically targeting the behavior of the hottest fuel pin subjected to the identical UTOP transient conditions.

To ensure modeling consistency, TEMIDE simulation uses the thermal-hydraulic output from CRONO as its boundary conditions. Also, the fuel relocation and cracking models within TEMIDE are deactivated since not present in CRONO, but their effects are taken into account in the initial thickness of the gap in both codes.

The fuel temperature results, displayed in Figure 4a, show a very good agreement between the two



(a) Center and surface fuel temperatures.

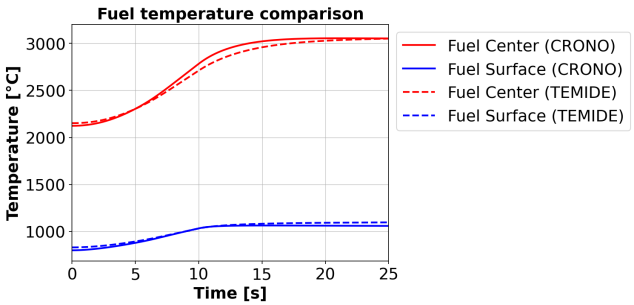


(b) Inner and outer cladding temperatures.

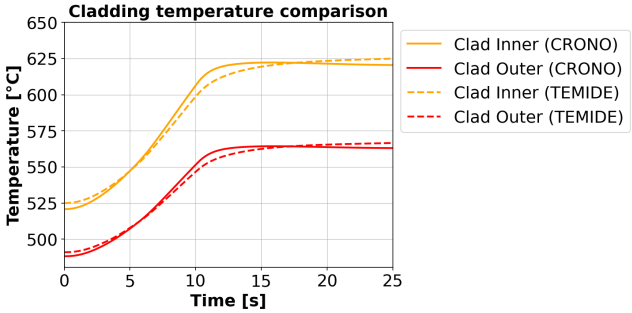
Figure 3. Comparison between CRONO, RELAP5 and SIM-LFR on the fuel and cladding temperatures.

codes. TEMIDE incorporates a more sophisticated treatment of physical phenomena like oxygen diffusion and the non-homogeneous power distribution resulting from thermal expansion, which are the sources of the small differences with respect to CRONO. Despite these added complexities, the inter-code deviation remains satisfactory.

Regarding the cladding temperatures (Figure 4b), the initial values are highly similar, confirming the consistency of the models for cladding-coolant heat exchange and cladding thermal conductance, but also the temperature profiles during the transient are very close to each other, meaning that the gap conductance model in CRONO is capable of capturing correctly the heat transfer from the fuel to the cladding.



(a) Center and surface fuel temperatures.



(b) Inner and outer cladding temperatures.

Figure 4. Comparison between CRONO and TEMIDE on the fuel and cladding temperatures.

4. DISCUSSION AND CONCLUSIONS

The results of this study successfully confirm the good predictive reliability of the CRONO code in analyzing the UTOP scenario for the ALFRED reactor, thereby establishing a preliminary benchmarking for its models. The alignment in power evolution and core temperature profiles with established system codes (RELAP5 and SIM-LFR) suggests the fundamental robustness of the implemented thermal-hydraulic and neutronic coupling.

Nonetheless, CRONO's ability to accurately predict the maximum peak fuel and cladding temperatures, as benchmarked against the system codes RELAP5 and SIM-LFR, demonstrates its capacity to provide essential safety-critical information. This achievement supports CRONO's role as a Design-Oriented Code. Its high computational performance - about 0.5 s of runtime for a 200 s transient - enables it to support safety-by-design during early conceptualization stages.

The primary limitation of the current study lies in its time-domain restriction to the first 25 seconds of the transient with main focus on the core. Therefore, future work should focus on the benchmark of other design-relevant transients where interactions of the core with other components are more relevant, such as the ULOF, ULOHS and ULOOP. These advancements will enable the comprehensive use of CRONO for linear stability studies and for rigorous sensitivity and uncertainty evaluations in LMFRs.

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