Preliminary Transient Analysis of SEALER

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Abstract. SEALER (SwEdish Advanced LEad Reactor) is a small (8 MW_t) lead-cooled fast reactor operating on 19.9 % enriched UO_2 fuel, designed for commercial power production for off-grid consumers. The reactor features a low power density, and a very small temperature gradient over the core to reduce the degrading process of structural materials. The safety approach for SEALER relies essentially on passive and inherent characteristics, such as a negative temperature reactivity feedbacks, natural convection and heat radiation, the primary safety goal being that under no circumstances shall sheltering or evacuation of the public be necessary.

In this contribution, the results of preliminary SEALER transient analyses including the coupled primary and secondary systems are discussed. Calculations were carried out using BELLA, an in-house code *ad hoc* developed for dynamic simulation of lead-cooled fast reactors, based on a lumped-parameter approach to solve the coupled-physics governing equations.

As major outcomes of this study, it was concluded that, under the postulated accident conditions, adequate safety margins are provided against fuel melting and cladding failure, favored by an overall negative power feedback coefficient.

Key Words: Lead-cooled Fast Reactor (LFR), Small Modular Reactor (SMR), SEALER, safety.

1. Introduction

SEALER (SwEdish Advanced LEad Reactor) is a small 8 MW_t (3 MW_e) lead-cooled, fast spectrum reactor (LFR), currently under development by LeadCold. The reactor is intended for commercial power production in remote, off-grid sites. The expected lifetime of the SEALER core is 30 years, due to the fact that remote siting locations would complicate the periodical fresh fuel delivery, fuel reloading and spent fuel transportation. Therefore, the reactor is intended to function as a nuclear battery, meaning that no fuel reload will take place.

The choice of lead as coolant is determined by the request to ensure a combination of inherent safety, and a negligible release of radiologically significant nuclides in case of a core disruptive accident. Lead provides excellent radionuclide chemical retention and a high natural circulation potential, which allow designing a reactor with passive and inherent accident prevention and mitigation features. The major safety goal of SEALER is to eliminate the need for public sheltering or evacuation in case of the worst conceivable accident.

Transient simulations discussed in this paper were performed using the dynamics code BELLA. BELLA is a lumped parameter (0-D) system code developed in-house by LeadCold for transient analyses of SEALER. The code solves coupled neutron kinetic and thermal-hydraulic equations which allows investigating the time-dependent behavior of integral feedback effects and parameters (e.g. thermal power, temperatures, mass flow rates, etc.) important for system design and safety. Indeed, the intended use of BELLA is mostly for scoping analyses in support of safety-informed decision making. Development of the code

was motivated by relatively restricted options to apply modifications to the currently available system codes for fast reactor studies. When dealing with LFRs, though, such modifications are necessary to account for important physics and design-related features; the objective of BELLA is consequently to provide a simple, easily modifiable, and yet adequate in-house LFR dynamics simulator alternative (or, at least, complementary) to well-established system codes.

In this work BELLA was employed to simulate a set of safety-related transients in SEALER, the reference configuration of which is briefly described in Section 2. In Section 3 the modeling approach in adopted in BELLA is discussed, while the main results are presented in Section 4 and discussed in Section 5.

2. SEALER Configuration

SEALER is a 8 MW_t pool-type reactor cooled by liquid lead. The core consists of 19 fuel assemblies containing 91 fuel pins each. The UO_2 fuel is enriched to 19.9 % to allow for 30 years operation without refueling.

The distinguishing design feature of SEALER, as compared to other LBE- or lead-cooled fast reactors, is a low power density combined with a very small temperature gradient over the core. By keeping the maximum fuel cladding temperature below 450 °C and the core inlet temperature above 390 °C, the rate of degrading processes such as corrosion and embrittlement can be reduced to a minimum. By combining this narrow temperature window with adequate barriers for corrosion, it becomes possible to achieve a 30-years life time of the fuel cladding.

The primary system geometry is shown in FIG. 1. The lead coolant enters the reactor core from the “cold pool” to be collected in the “hot leg” and delivered to 8 Steam Generators (SGs). After passing through the SGs, the coolant flows downwards in the annular “cold leg” before returning to the “cold pool”. In order to maintain a small temperature gradient over the core, the current configuration of SEALER, relies on the use of eight pumps during normal operation. For emergency operation, natural circulation cooling is sufficient. The SGs transfer heat from the primary system to the secondary system, where superheated steam is produced and supplied to the turbo-generator. The main parameters relevant for simulations described in this paper are summarized in TABLE I. Further details on SEALER can be found in [1].

3. Modeling Approach in BELLA

The general approaches and assumptions implemented in BELLA were previously described in refs. [2,3]. Therefore, in this paper the focus is put on additional features recently implemented in the code, as well as on the assumptions specific to the modeling of the SEALER reactor. The current version of BELLA includes core neutronics, decay heat production, primary system thermal-hydraulics and heat transfer from the primary system to the secondary system in the SGs.

The primary system is divided into six regions/components: core, hot leg, SG, cold leg, cold pool and reactor vessel (see FIG. 1). Mass flow paths and energy flow paths in the calculation model are marked by the blue and red arrows respectively. Core bypass flow is neglected. Lumped parameter (0-D) conservation equations are solved for each region. In particular, the reactor core region model includes point-kinetics, reactivity feedbacks, decay heat production and fuel-to-coolant heat transfer. The point-kinetics model considers eight delayed neutron precursor groups. The effective neutron generation time and the delayed neutron fraction specific to SEALER UO_2 core are displayed in TABLE I together with the reference reactivity

coefficients. Reactivity feedbacks due to the Doppler effect, axial and radial expansions, and coolant density are considered. In addition, external reactivity due to control assembly movement is treated as a user-specified input parameter. Calculation is performed assuming constant reactivity coefficients, linear temperature dependence for axial, radial and coolant density feedbacks, and logarithmic dependence for the Doppler feedback.

A decay heat model was implemented in BELLA assuming that all decay heat precursors are produced due to U-235 fission. Such approximation results in a rough estimation of the total decay power, which is nevertheless considered sufficient for the current scoping studies. The decay heat equations are analogous to the delayed neutron precursor ones: the normalized precursor concentration multiplied by power, or "delayed power concentration" *H,* is calculated as:

$$\frac{dH_j}{dt} = \frac{E_j n}{Q} - \lambda_j H_j \tag{3.1}$$

where j = 1,2, … 23 represents the decay heat precursor groups, E_j and λ_j are the delayed power fraction and the decay constant for the respective group, given in [4]. Q is the energy per fission, and n is the normalized neutron population given by the solution of the point-kinetics equations. Infinite reactor operation at nominal power is assumed for calculating the initial precursor concentration at the beginning of the simulations.

The reactor total thermal power, including decay heat, is calculated as:

$$P = P_0 \left(n \left(1 - \sum_j H_j \right) + \sum_j H_j \right) \tag{3.2}$$

where P_0 is the nominal thermal power.

TABLE I: MAIN SEALER SIMULATION PARAMETERS

Parameter	**Value**
Nominal core thermal power	8000 kW
System power density ($P_{core}/V_{coolant}$)	300 kW/m^3
Average linear core power	4.2 kW/m
Height of the fuel column	1100 mm
Effective neutron generation time	212 ns
Effective delayed neutron fraction	716.8 pcm
Doppler constant	-335.0 pcm
Axial expansion reactivity coefficient	-0.33 pcm/K
Radial expansion reactivity coefficient	-0.40 pcm/K
Coolant reactivity coefficient (global)	-1.30 pcm/K
Nominal ΔT core	42 °C
Nominal primary coolant mass flow rate	1300 kg/s
Nominal secondary coolant mass flow rate	5.5 kg/s
Secondary coolant inlet temperature	330 °C
Reactor vessel mass	3380 kg
Emissivities (vessel, pit)	0.85, 0.20
Reactor pit temperature	90 °C

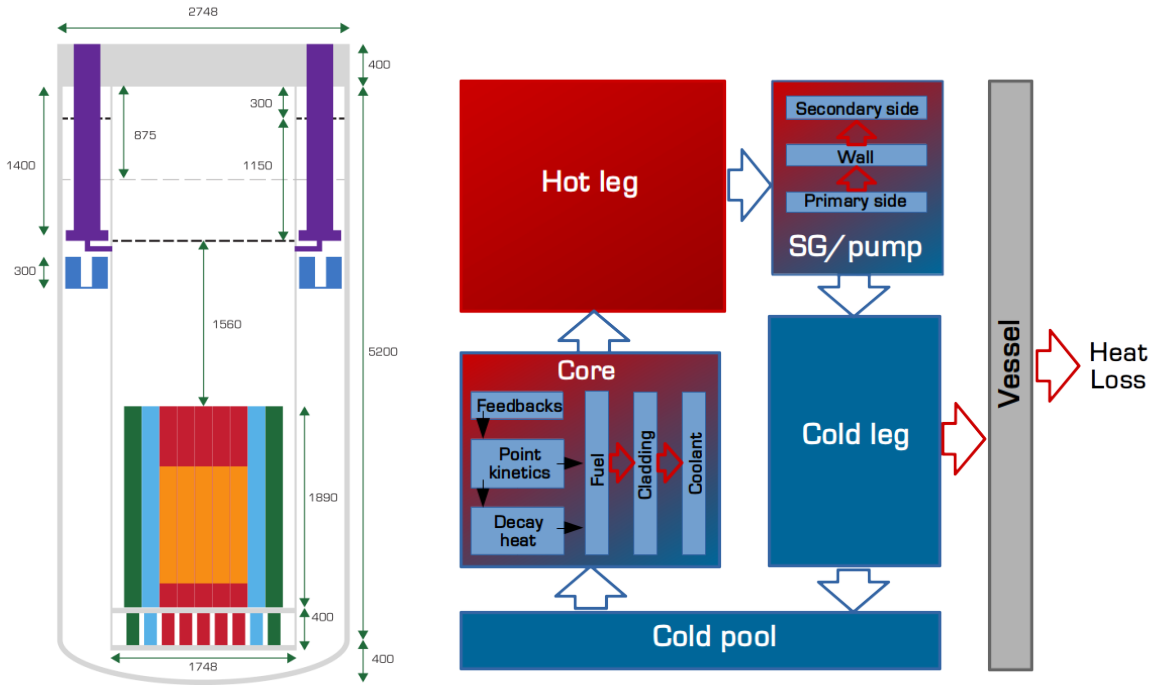

FIG. 1. SEALER primary system (left) and corresponding model in BELLA (right)

Fuel centerline, fuel outer, cladding inner, cladding outer and core outlet temperatures are calculated by the core heat transfer model. Single, lumped energy balance equation is solved for each temperature. Such model represents the simplified, averaged radial heat transfer in the fuel rods, axial heat transfer being neglected. Heat transfer through the fuel and cladding is assumed to occur only by conduction. Global conductive heat transfer coefficient in the fuel (*f*) is consequently calculated as:

$$h_f = 4\pi\lambda_f N_p H_f \tag{3.3}$$

and for the fuel-cladding gap (*g*) and cladding (*c*) as:

$$h_{g(c)} = \frac{2\pi\lambda_{g(c)} N_p H_f}{\ln\left(\frac{r_{out}}{r_{in}}\right)} \tag{3.4}$$

Here N_p and H_f denote the number of fuel pins and the height of the fuel column respectively, while λ denotes the thermal conductivity. Convective heat transfer is assumed at the cladding-coolant interface. The global core heat transfer coefficient is calculated as:

$$h_{Pb} = A_c \frac{\lambda_{Pb,core} \mathrm{Nu}_{Pb,core}}{d_{h,core}} \tag{3.5}$$

where A_c is the total surface area of the cladding and $d_{h,core}$ is the hydraulic diameter of the core. The Nusselt number is calculated from the Mikityuk correlation [5].

Heat transfer in the SG is modeled either by imposing a constant temperature drop, or a constant SG power, or a user-defined SG power function, or using a three-region moving boundary model. The moving boundary model is formulated based on mass and energy balance equations for the sub-cooled, two-phase and super-heated regions on the SG secondary side. Region lengths obtained from the water side solution are used for the wall and primary side solutions. Formulation of the water side and the wall equations is discussed by Jensen and Tummescheit [6]. The SG lead side model is based on energy balance for the sub-cooled, two-phase and super-heat regions, defined by the heat transfer conditions on the

secondary side. Formulation of the lead side equations is done assuming incompressible flow. The respective discussion is available in [7]. Enthalpy and mass flow rate boundary conditions are imposed on the water side inlet, while the outlet mass flow rate is calculated based on secondary side pressure p_{SG}:

$$\dot{m}_{SG,S,o} = C_v \sqrt{p_{SG} - p_c} \tag{3.6}$$

where C_v is an adjustable coefficient and p_c is downstream pressure lower than p_{SG}.

Heat losses from the primary system constitute an important safety feature in SEALER. Namely, the design intent is having sufficient losses through the reactor vessel walls to remove the residual power without resulting in unacceptably high system temperatures. The prediction of such losses is included in BELLA by modeling the heat transfer to/from the vessel component. Heat is assumed to be transferred to the vessel from the cold leg due to convection:

$$Q_{CL,V} = h_{CL}(T_{Pb,CL} - T_V) \tag{3.7}$$

where the heat transfer coefficient h_{CL} is calculated using the Nusselt number calculated from the Subbotin-Seban-Shimazaki correlation [5]. Heat is lost from the reactor vessel to the surrounding concrete pit assuming radiative heat transfer:

$$Q_{V,P} = \frac{A_{V,o}\sigma(T_V^4 - T_P^4)}{\frac{1}{\epsilon_V} + \frac{1}{\epsilon_P} - 1} \tag{3.8}$$

Here $A_{V,o}$ is the outer wall area of the reactor vessel, σ is the Stefan-Boltzmann constant, T_V and T_P are the absolute temperatures of the vessel and the concrete pit wall, and ε_V and ε_P are the thermal emissivities of the vessel and the pit respectively, summarized in TABLE I. The pit wall temperature is assumed to be constant, considering the surroundings as an infinite heat sink.

The vessel temperature is calculated by implementing an energy balance between the heat source (eq. 3.7) and sink (eq. 3.8):

$$m_V c_{p,V} \frac{dT_V}{dt} = Q_{CL,V} - Q_{V,C} \tag{3.9}$$

where m_v and $c_{p,v}$ are the mass and the specific heat capacity of the reactor vessel.

The model equations are implemented in Python programming language. The solution to the primary side thermal-hydraulics, neutron kinetics and decay heat equations is obtained using the LSODA solver. The SG equations are solved using the Adams-Bashforth/Adams-Moulton second order predictor-corrector method.

4. Transient Simulations

A set of transients, covering the consequences of most initiating faults, was simulated in un-protected mode, corresponding to a failure while inserting the shut-down elements to achieve sub-criticality. The system dynamics following (*i*) reactivity insertion leading to Unprotected Transient OverPower (UTOP), (*ii*) pump failure resulting in a Loss Of Flow (ULOF), and (*iii*) steam generator failure causing a Loss of Heat Sink (ULOHS) were simulated. In addition, the effects due to changes of boundary conditions on the secondary side, namely decrease of feed-water (*iv*) temperature and (*v*) flow rate were investigated.

4.1. UTOP Transient

The UTOP transient was simulated by a step-wise positive 0.5 β_{eff} reactivity insertion at time t = 200 s, which corresponds to an inadvertent withdrawal of a control assembly. Two cases

were compared: using the three-region SG model (curves marked with 'WSG') and assuming a constant 42 ºC temperature drop over the SG. The time-dependent evolution of the main variables of interest is summarized in FIG. 2.

Following the reactivity insertion, the core power rapidly peaks to approximately 5.8 times nominal. This causes an increase in fuel centerline temperature which, in turn, brings a fast negative reactivity insertion due to the Doppler and axial feedbacks, which, combined with negative coolant and radial feedbacks (due to increased lead temperatures), result in a total negative reactivity at time $t \approx 300$ s. The negative reactivity drives the reactor power down, until an equilibrium between the produced and the extracted power is reached. The use of the SG model results in a higher power transferred to the secondary side and, consequently, colder SG outlet temperature which yields the core thermal power to stabilize at a higher value as compared to the initial state. The SG becomes mostly super-heated, while the lengths of the two-phase and sub-cooled regions decrease.

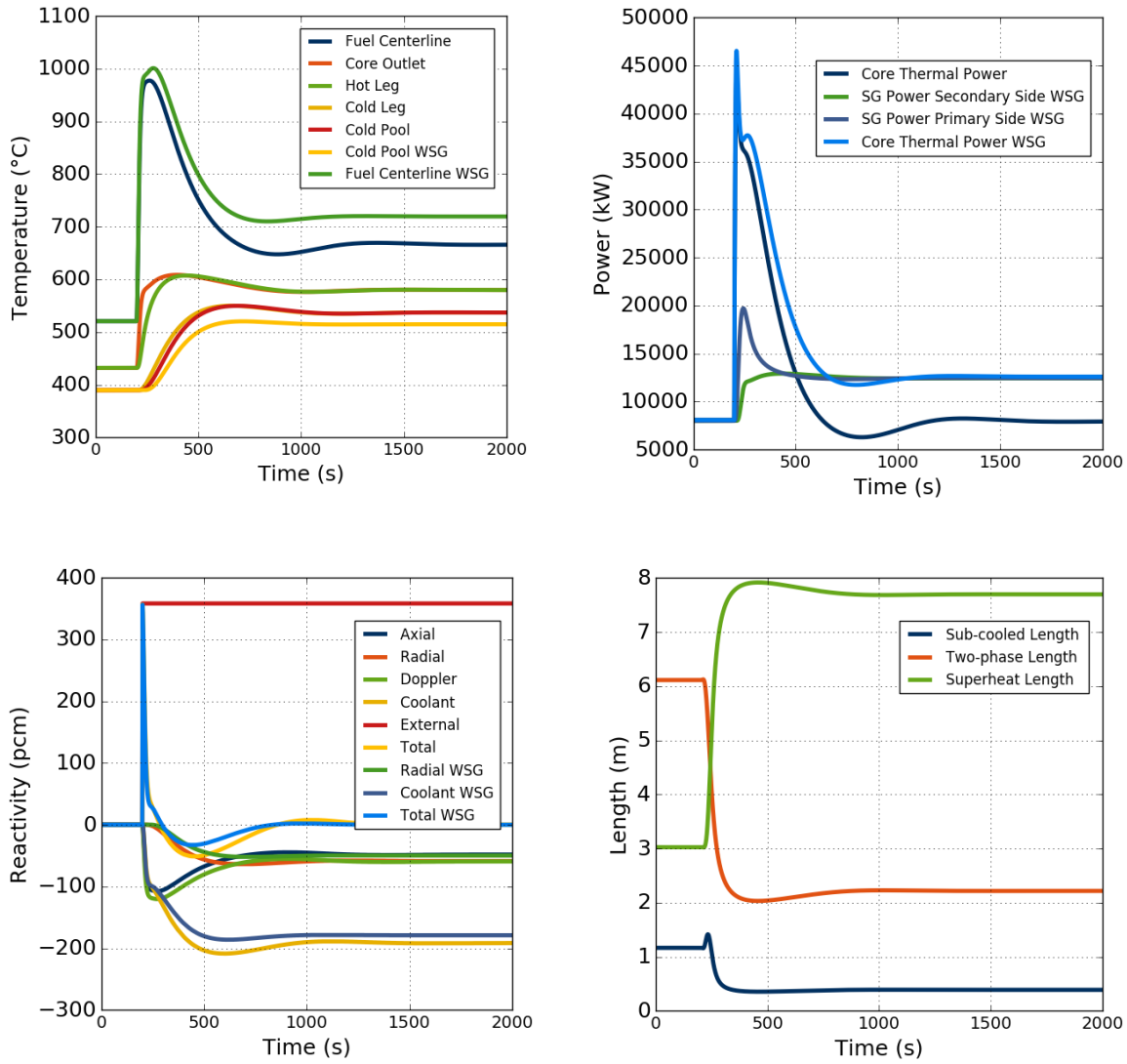

FIG. 2. System temperatures (top-left), power evolution (top-right), reactivity (bottom-left) and SG region lengths (bottom-right) during UTOP transient

4.2. ULOF Transient

The ULOF transient was simulated by exponentially decreasing the pump head at time t = 1000 s, which corresponds to the loss of electrical power to all 8 primary coolant pumps. The calculation was performed assuming a constant 42 ºC temperature drop over the SG. The time-depend evolution of the main variables of interest is summarized in FIG. 3.

The coolant mass flow rate drops as a result of pump head reduction. Degraded heat removal causes core temperatures and lead temperature at the core outlet to peak in the initial phase of the transient. Increase in temperatures results in a negative total reactivity insertion and the core power drops as a result. The largest negative reactivity insertion is due to the coolant reactivity feedback. Fuel axial and Doppler reactivity feedbacks lead to a positive reactivity insertion, due to the decreased fuel temperature compared to the initial steady state. Natural circulation flow establishes after approximately 8000 s, with the core power settling to 7 % nominal.

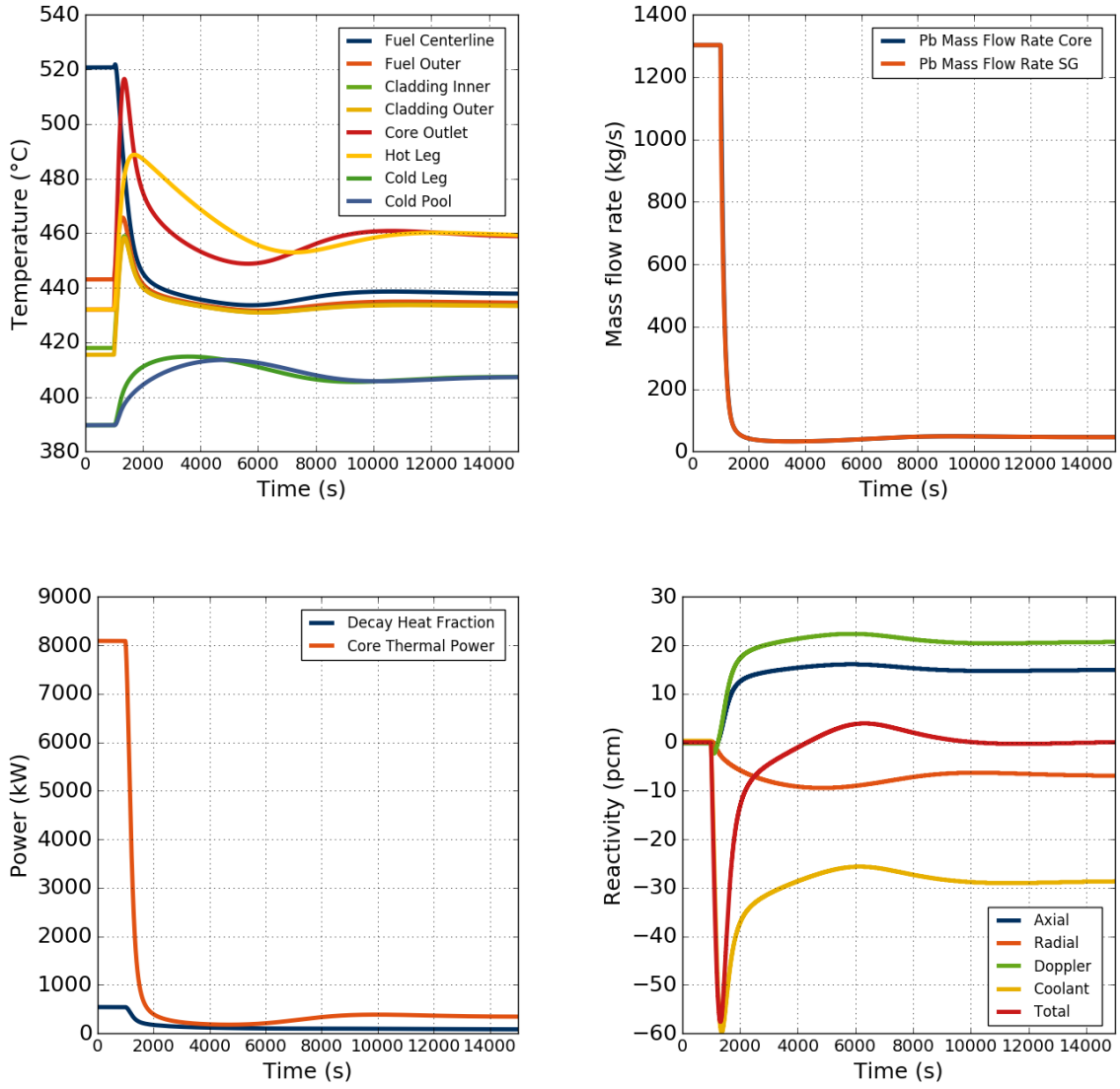

FIG. 3. System temperatures (top-left), Pb mass flow rate (top-right), power (bottom-left) and reactivity (bottom-right) during ULOF transient

4.3. ULOHS Transient

The ULOHS transient was simulated by a step-wise elimination of the heat transfer through the heat exchanger at t = 1000 s. This corresponds to an exaggeration of the secondary system malfunction scenario, for example due to a loss of feed-water supply. The time-depend evolution of the main variables of interest is summarized in FIG. 4.

The reduction of heat extraction causes the average lead temperatures to increase promptly after the transient initiation. This results in a negative total reactivity insertion driven by coolant and radial reactivity feedbacks. The core thermal power drops as a result, causing fuel and lead temperatures to decrease, due to the heat loss through the reactor vessel. Decreasing temperatures result in a positive reactivity insertion by the axial and Doppler feedbacks, and a decreasing negative insertion by the coolant and radial feedbacks. The total reactivity becomes positive at time $t \approx 35000$ s and the system becomes super-critical as a result. This in turn results in temperature increase, again yielding a negative reactivity insertion and sub-critical state.

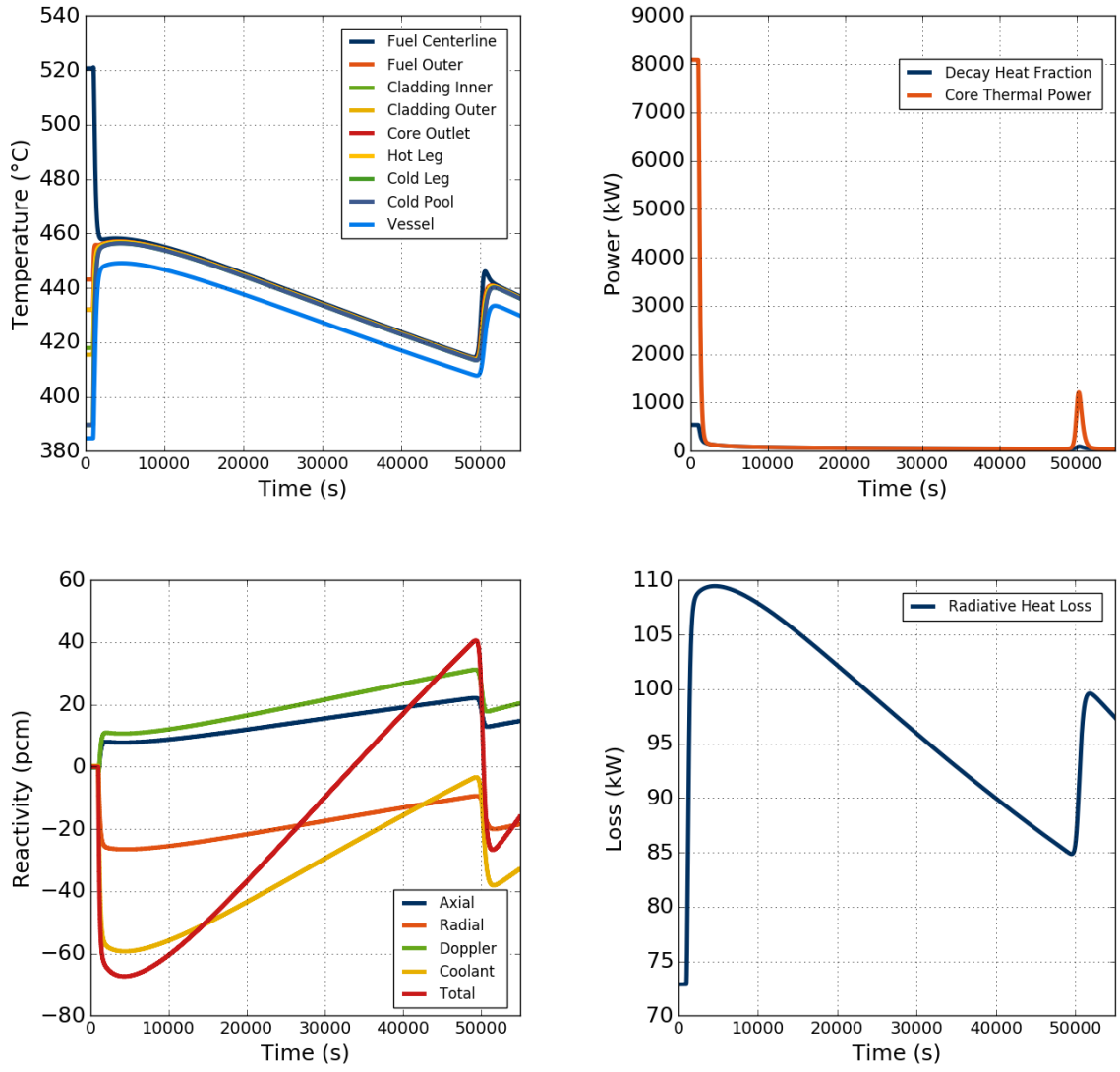

FIG. 4. System temperatures (top-left), power (top-right), reactivity (bottom-left) and radiative heat loss from the reactor vessel (bottom-right) during ULOHS transient

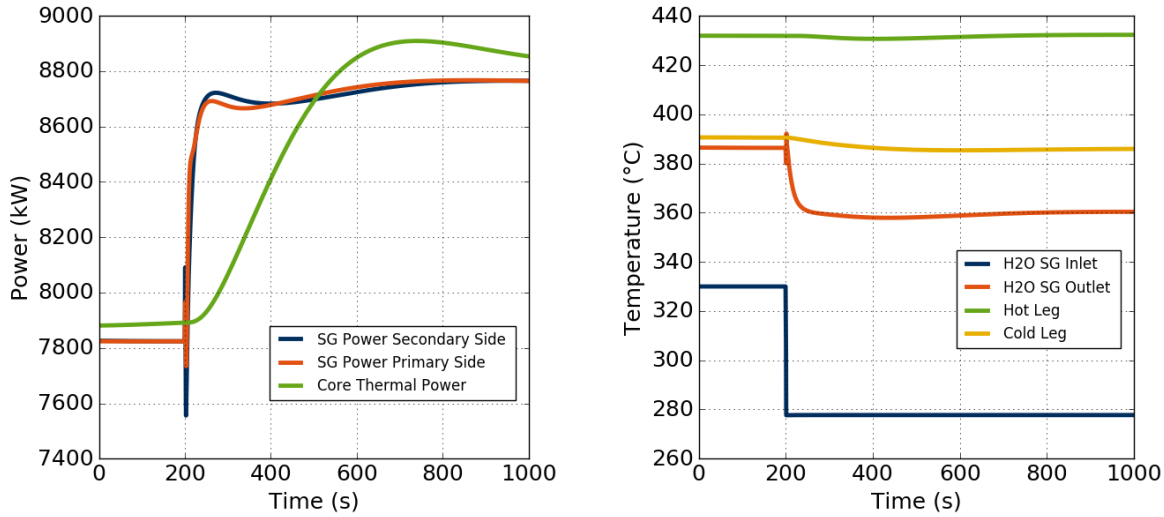

FIG. 5. SG and core power evolution (left) and temperature evolution (right) during feed-water temperature reduction transient

4.5. Secondary Side Transients

Perturbations on the SG secondary side boundary conditions were introduced to investigate the primary system response due to changes in the secondary system nominal conditions. These include (i) reducing the feed-water temperature (due to e.g. feed-water pre-heating malfunction) and (ii) reducing the feed-water flow rate (due to e.g. malfunction of feed-water pumps).

The time-depend behavior of system power, temperatures, and secondary side mass flow rate after the feed-water temperature is reduced by 50 ºC and feed-water mass flow rate is reduced

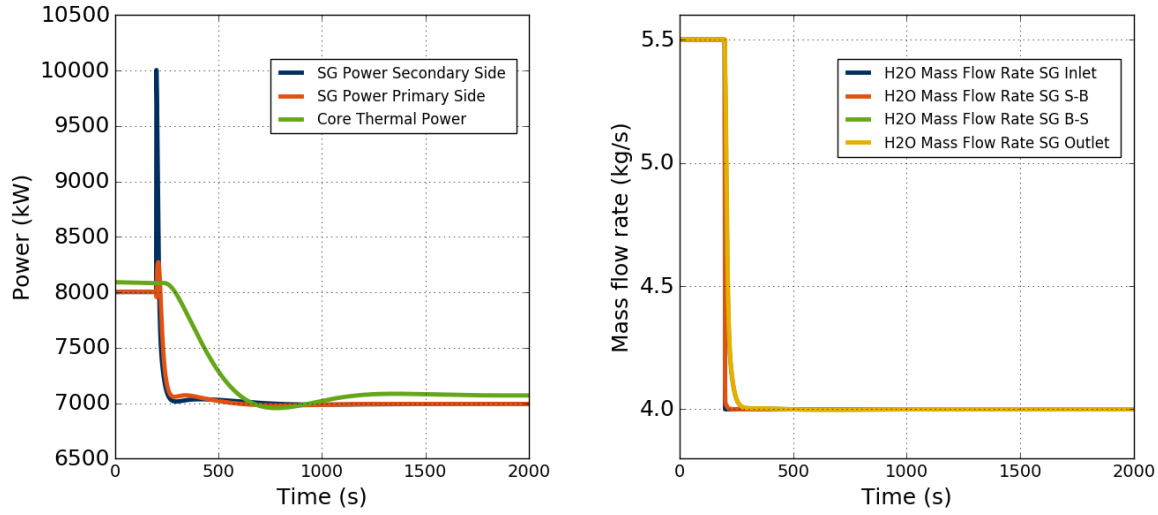

FIG. 6. SG and core power evolution (left) and mass flow evolution (right) during feed-water mass flow reduction transient

by 1.5 kg/s at t = 200 s, is shown in FIG. 5 and FIG. 6. respectively. The temperature reduction results in a higher power exchanged at the SG and consequently colder SG outlet temperature on the primary side, while the reduction of mass flow rate results in a lower SG power, and higher SG outlet temperature on the primary side. Reactivity changes due to coolant and radial feedbacks result in a core power increase and decrease, respectively, until a new balance between the power produced and the power removed is achieved.

5. Conclusions

A preliminary study of UTOP, ULOF, ULOHS, feed-water temperature, and feed-water flow rate decrease transients was performed using BELLA, an in-house lumped parameter transient code specifically developed for LFR dynamics and safety analyses.

The results show, that, in case of UTOP, ULOF and ULOHS scenarios, significant margins to coolant boiling (1749 ºC), fuel melting (2865 ºC) and cladding rapid creep failure (930 ºC) are retained, favored by an overall negative power feedback coefficient. As demonstrated by the ULOHS simulation, positive reactivity insertion due to decreasing coolant temperatures may act as a self-protection mechanism against coolant freezing. Variations in the SG secondary side boundary conditions result in coolant temperature-reactivity feedback-driven core power change, until a new balance between the produced and the extracted power is achieved. Detailed analysis of the system behavior, including the effects of local phenomena, is foreseen to confirm the preliminary findings.

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