Analysis of sodium boiling initiated by unprotected loss of flow in European sodium fast reactor core with different subassembly designs

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ABSTRACT

In this paper, an Unprotected Loss of Flow (ULOF) assessment has been performed on the European Sodium Fast Reactor developed in the ESFR-SMART EU project. To conduct the analysis, a simplified 42 channel thermal-hydraulic model has been established in TRACE system code, using a point kinetics model accounting for various reactivity feedback effects. The assessment reveals the core behavior of a commercial size, 3600 MW, sodium fast reactor using a state-of-the-art low void effect reactor core design. The study focuses on the sodium boiling phenomenon and sodium reactivity feedback effect evolution during the accident with the reference subassembly (SA) design. Following this analysis, a study has been performed with a modified SA design. The boiling progression and phenomenology within the reference and the modified core have been compared, and the impact of the SA modification was described.

1. Introduction

Sodium-cooled Fast Reactors (SFR) are part of the six advanced reactor types chosen by the Generation IV International Forum as the most prominent advanced nuclear reactor technologies. These reactors have the potential to further increase safety, reduce nuclear waste, and be more economical compared to today’s conventional reactor types. Coming from the aforementioned potentials of this reactor type, in recent years, significant efforts have been made to improve the reactor design and to implement new design features leading to greater safety and simplicity of the reactor. As part of this effort, the European Sodium Fast Reactor Safety Measures Assessment and Research Tools (ESFR-SMART) European Union project was launched, having one of its primary goals to improve the reactor design to achieve better safety standards (Mikityuk et al., 2018).

As part of this project, different accidental conditions are assessed to evaluate the behavior of the reactor under critical circumstances. One of these accidental conditions is the Unprotected Loss Of Flow (ULOF) accident, which is the subject of the current paper. This accidental scenario is the most severe accident evaluated within the research project, providing an insight into what happens with the reactor in a beyond design basis accident, where the integrity of the reactor cannot be guaranteed by specifically designed safety features.

Although there is no specific safety feature operating to ensure the safe reactor shutdown under ULOF conditions, a state-of-the-art low void effect core is implemented, which is anticipated to stabilize the reactor behavior during the accident. This low void effect core has been considered in recent studies (Chenaud et al., 2013; Poplavsky, 2011), where a so-called sodium plenum is incorporated at the top of the fuel instead of the traditionally used fertile material region. This feature is meant to decrease the positive reactivity effect of the core voiding, due to sodium boiling, by increasing the neutron leakage of the core. This is achieved by the voiding of the plenum region, as a result of sodium boiling propagation upward from the fuel region of the subassembly (SA), and a corresponding decrease of neutron reflection towards the fissile fuel region. It has been previously demonstrated that such a design can result in a stabilized sodium boiling process, and subsequent power excursion can be evaded (Poplavsky et al., 2009; Raskach et al., 2021). The capability of this core design to keep the reactor stable during an unprotected accidental scenario is to be assessed in this study.

The paper is structured as follows: In section 2, the reactor core of the ESFR is described with its radial and axial SA zones. Section 3 describes the computational tool and the modeling approach utilized for the analysis, whereas in section 4, the results of the calculations are available online 18 October 2022

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presented. Finally, in section 5, the main conclusions of the work are summarized.

2. ESFR core description

The base design of the reactor core was inherited from the preceding CP-ESFR EU project (Buiron et al., 2013), to which further design improvements were applied within the ESFR-SMART project (Rineiski et al., 2018) for which the main specifications are given in Table 1. As a general description of the core, the fuel region can be divided into inner core (IC) and outer core (OC) sections with 218 and 288 SAs, respectively. The core was designed with a 6 batch fuel reloading pattern to smoothen the reactivity variations and the power peaking between the batches. The reactivity control is ensured by the 24 control and shutdown devices (CSD) and 12 diverse shutdown devices (DSD) present in the reactor core. An innovative safety measure of the new ESFR core is the corium guide tubes, which provide a clear path for the melted sodium heat up in all regions of the core. The resultant coolant flowrates within CG1 to CG5, in nominal conditions, and the number of SAs within each CG is presented in Table 2.

Positions of axial regions in SA of inner and outer zones are shown in Fig. 2. Apart from the fuel height difference and the orifice in the foot, all SAs are identical.

3. Modelling approach

3.1. The computational tool

The thermal–hydraulic behavior of the ESFR under ULOF accidental conditions was modeled with TRACE system code (U.S. R. Commission, 2011). The code was originally developed by the U.S. Nuclear Regulatory Commission (NRC) for the system analyses of light water-cooled reactors. In recent years, several modifications have been implemented at the Paul Scherrer Institut (PSI) (Mikituyuk et al., 2005) to extend the capabilities of the code simulating advanced fast reactors. In particular, for the current research, where sodium boiling is expected, the main modifications of the code are summarized below.

For the modeling of transient sodium boiling, the two-fluid model with new closure relations is used. The approach is based on the same main conservation equations as coded for two-phase water flow and modifies only the closure relations and equations-of-state. In particular, the thermal physical-two-phase properties of sodium are calculated according to (Fink and Leibowitz, 1995). The simplified two-phase sodium flow regime map used in the modified TRACE distinguishes single-phase liquid and single-phase vapor convection regimes with annular film boiling regime in-between as proposed in (Chenu, 2011). This simple approach to describe the various phases, compared to light water reactors, can be used due to the properties of sodium, namely the high heat conductivity, low specific heat and high vapor to liquid density ratio. These properties lead to the formation of large vapor bubbles quickly filling the SA channel following boiling onset. At typical fast liquid metal reactor conditions around atmospheric pressure, slug and annular flow regimes are prevailing for liquid metals, while bubbly flow is more characteristic under high pressure in light water reactors (Chenu, 2011).

In boiling condition, the condensation of vapor slugs periodically occurs and the flow regime therefore is switching between annular-slug and single-phase liquid, due to the recurrent chugging and re-entering of the cold liquid sodium from above the SAs.

The fraction of the perimeter wetted by liquid sodium is calculated by the following correlation for all regimes:

\[ \eta = \frac{1}{1 + 5 \times 10^5 \alpha (1 - \alpha)} \quad (1) \]

where \( \eta \) is the void fraction. This correlation predicts \( \eta \) equal to 1 for \( \alpha \) less or equal to about 0.95 and in the interval from 0.95 to 1 \( \eta \) is smoothly reducing to 0. The value of 0.957 was recommended as a dryout criterion in (Chenu, 2011). The \( \eta \) value therefore describes a transition from the annular film boiling regime to the single-phase vapor convection regime. The liquid-to-wall friction factor is calculated by the Rehme model (Rehme, 1972) for the wire-wrapped bundles and by the Churchill model (Churchill, 1977) for all other configurations, while the vapor-to-wall friction factor is calculated by the Churchill model (Churchill, 1977) for all configurations. In all cases, the friction factor is multiplied by the value of \( \eta \) for the liquid phase and by the value of \( 1 - \eta \) for the vapor phase. Liquid-to-wall and vapor-to-wall heat exchange coefficients are calculated according to the model proposed in (Mikituyuk, 2009) for the liquid phase and by the Dittus-Boelter model for the vapor phase. Interfacial area density is based on the geometrical representation of the annular film boiling pattern taking into account the value of \( \eta \) as used in the SABENA code (Ninokata and Okano, 1990) and recommended in (Chenu, 2011). Interfacial heat transfer and friction coefficients are calculated as in the SABENA code (Ninokata and Okano, 1990) and as recommended in (Chenu, 2011).

A validation has been performed in (Chenu, 2011) on incorporated sodium boiling model in TRACE considering annular flow as the dominant regime up to dryout and a smooth breakdown of the liquid film after dryout onset. Recently, validation of the models was performed using the KNS-37 sodium boiling experiment, which has been reported in (Perez-Martin et al., 2022).

3.2. TRACE model

The modeling of the ESFR-SMART core, to conduct the current analysis, was performed by utilizing a 42 one-dimensional channel model, shown in Fig. 3, which is a simplification compared to the previously used full reactor model for other accidental conditions (Bodi et al., 2022). This simplification was necessary due to the required resources to perform the simulation of the whole reactor model. Each of the 42 channels corresponds to a SA of the actual core, giving a 1/12th representation of the entire core, allowed by the symmetry of the design, modeling 18 IC and 24 OC SAs. Axially the IC and OC SA nodalization were identical, using 74 nodes of 0.05 m in length. The only alteration between the regions is the 5 fertile nodes and 15 fissile nodes for the IC channels and the 1 fertile node and 19 fissile nodes of the OC channels, shown in Table 3 and Fig. 2. The mesh size of 0.05 m for TRACE channel model was selected based on previous studies (Chenu, 2011; Sun et al., 2013).

The inlet flow supply is connected to an inlet plenum, representing the diagrid under the SAs. The inlet sodium flowrate is set as a boundary condition within the model, providing equivalent flowrate evolution to what was observed in the full reactor model (Bodi et al., 2022), where the flowrate was supplied by modeling the actual pump with their cost down curves. This simplification was considered to be reasonable, for

<table>
<thead>
<tr>
<th>Table 1</th>
<th>ESFR main specifications</th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power (MWh)</td>
<td>3600</td>
</tr>
<tr>
<td>Net electrical power (MWe)</td>
<td>1500</td>
</tr>
<tr>
<td>Global efficiency (%)</td>
<td>42</td>
</tr>
<tr>
<td>Core inlet / outlet temperatures, (°C)</td>
<td>395 / 545</td>
</tr>
<tr>
<td>Type of fuel</td>
<td>(U,Pu)202 / UO2</td>
</tr>
<tr>
<td>Fissile fuel content, (%)</td>
<td>18</td>
</tr>
<tr>
<td>Plant lifetime, (years)</td>
<td>60</td>
</tr>
<tr>
<td>Availability target, (%)</td>
<td>90</td>
</tr>
</tbody>
</table>
the given inlet core pressure, the primary system pressure drop and overall mild boiling pattern in the low-void core, no strong boiling-induced primary flowrate oscillations were expected in the ULOF conditions. This assumption is supported by (Alpy et al., 2016), where it was shown that the inlet flowrate oscillation is only a small fraction of the gradually reducing primary mass flow rate, especially considering that only the first ~ 100 s of the ULOF transient is simulated. Furthermore, as all SAs are modelled individually and a large number of SAs are not involved in boiling for the simulated period of time, the breakdown of the residual forced flow (which stays above 10 % of the nominal flow rate for the simulated period of time) is not expected. Thus, the total inlet flowrate is not affected noticeably by boiling, and strong local fluctuations of the flowrate are observed in boiling channels only. Each SA is connected to the inlet plenum through a valve component, establishing the right orifice of the SA to account for the different cooling groups of the core and to set up the right flowrate within each SA. At the top of the channel, another plenum component is situated, representing the hot pool of the reactor, which is connected to a pressure boundary condition of 0.2 MPa.

The sodium between the SAs (in the so-called inter-SA gap) is modeled using a 4 radial ring vessel component, where the first 2 rings correspond to the IC region and the outer 2 rings to the OC region. The inlet and outlet of the vessel are connected to the inlet and outlet plenums, respectively. By using this modeling approach, the radial pressure difference between the rings can be represented. In the first axial node of the vessel, the friction factor has been increased to model the high pressure drop present in the actual reactor between lower and upper plena, originating from the labyrinth seal at the foot of each SAs, through which the sodium enters the inter-SA gap, in order to achieve a reasonable mass flowrate through the inter-SA gap. The heat exchange between the sodium in the vessel component and the sodium in the SAs is achieved through heat structures elements at each axial node simulating the stainless steel SA wrapper.

### 3.3. Point kinetics model

The reactivity and power evolution during transient are calculated using the built-in point kinetics model of TRACE. The employed reactivity feedback coefficients were calculated using the Serpent Monte Carlo code on the End of Equilibrium Cycle (EOEC) core state (Baker et al., 2022). The reactivity feedback effects considered in the modeling are the Doppler, axial fuel expansion, control rod driveline (CRDL) expansion, sodium density, and void effect, presented in Table 4.

In the model, individual Doppler constants were used in four different regions of the reactor core, allowing a detailed spatial

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**Table 2**

<table>
<thead>
<tr>
<th>Cooling groups</th>
<th># of SA in CG</th>
<th>Sodium flowrate [kg/s]</th>
</tr>
</thead>
<tbody>
<tr>
<td>CG1</td>
<td>216</td>
<td>8802</td>
</tr>
<tr>
<td>CG2</td>
<td>111</td>
<td>4894</td>
</tr>
<tr>
<td>CG3</td>
<td>69</td>
<td>2558</td>
</tr>
<tr>
<td>CG4</td>
<td>60</td>
<td>1693</td>
</tr>
<tr>
<td>CG5</td>
<td>48</td>
<td>981</td>
</tr>
</tbody>
</table>

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**Fig. 1.** ESFR-SMART core cooling group scheme.
Fig. 2. Inner core and outer core SA axial region and heights.

Fig. 3. TRACE nodalization diagram of the ULOF 42 channel model.
representation of the reactivity effect related to fuel temperature change. The axial fuel expansion was modeled by calculating the expansion of the cladding, as an EOEC core, the cladding and fuel were considered to be in contact with each other, and the cladding temperature drives the effect. The CR position change in the core is calculated according to both the CRDL expansion, which is considered to be driven by the temperature evolution of the outlet plenum, in a simplified manner and variation of the fuel height due to the axial expansion of the fuel. 

The sodium density related effect is composed of two components. The first one is calculated based on the sodium density variation prior to boiling, and the second one relates to the contribution due to voiding. Traditionally, there is only one sodium reactivity coefficient used in point kinetics calculations for the whole range of sodium densities to obtain the reactivity effect during the transient. For more complex, low void fuel reactor core designs, this approach is not sufficient as the effect in some regions is strongly non-linear on density. In a previous study (Lemasson et al., 2013), a new method was developed to account for this non-linearity of the density feedback coefficient, which is especially relevant for the sodium plenum region. This approach was also implemented in the current research accounting for the evolution of the sodium feedback coefficient with the following equation:

\[ C_{Na} = C_{Na,\text{Density}} + \left( C_{Na,\text{Void}} - C_{Na,\text{Density}} \right) \times \text{Void signal} \tag{2} \]

where, \( C_{Na} \) is the overall sodium reactivity coefficient, \( C_{Na,\text{Density}} \) is the sodium density coefficient, \( C_{Na,\text{Void}} \) is the sodium void coefficient, and the \( \text{Void signal} \) is the void fraction within the node. The spatial distribution of every coefficient is given individually for each CG of the core, divided into 20 axial zones in the fuel region and 7 zones in the plenum region. Here, the fuel region refers to the fissile and fertile regions, together corresponding to the active core with a positive global contribution. The calculation is performed for every axial node in every SA with the corresponding coefficients, which are then summed together to a global sodium reactivity effect. In Table 4, the global fuel and plenum region void effects are given, whereas the sodium density effect is normalized to the percent of sodium density change. In the modeling, all reactivity effects are independent of each other. Therefore, effects such as deterioration of fuel Doppler effect due to sodium voiding are not being considered in the simulation. The error arising from this simplification is expected to be negligible due to few reasons. As the boiling onset occurs in the top of the fuel height, only a marginal influence on the total Doppler effect could be expected essentially from these upper fuel zones with a small contribution to the total effect. The boiling propagation to the core center that would affect the Doppler more noticeably is observed for a number of SAs only in the latter phase of the simulation. Furthermore, as the main feedback effect, driving the core power evolution following sodium boiling onset, is the sodium void effect, and the magnitude of the Doppler effect is relatively small, the potential reduction in Doppler would not affect the boiling progress significantly. Potentially, a decrease of the positive Doppler contribution during the simulated period of time would result in a slightly more negative total reactivity with overall compensation from other reactivity contributors, thus this approach seems to be overall conservative for the net reactivity evolution.

### 3.4. ULOF accident description

The initiating event of the accident is the loss of forced primary flow due to pump failure or loss of power supply followed by hypothetical failure of any safety system to shut down the reactor, thus the control and safety rods are not inserted into the reactor core. Under these conditions, the primary pumps of the reactor trip gradually with the primary mass flow halving time of 10 s. As the reactor shutdown systems fail, there is no sudden power drop in the system, but instead, the power evolution is governed by the inherent core reactivity feedback effects.

#### 3.5. ULOF analysis

The ULOF modeling is divided into two parts. First, a so-called reference case is used to model the accidental scenario to obtain results of the reactor behavior. This model corresponds to the description of the ESFR-SMART core design, provided in section 2, using the simplified 42 channel TRACE model. Whereas, within the second model, a new feature in the SA design has been introduced, described in (Sun et al., 2013; Mikityuk et al., 2013). The modification consists of a 4 cm diameter hole on each side of the hexagonal wrapper in 42nd axial node in the model, corresponding to the axial level just below the initial boiling region, which is top of the fuel region. The diameter of the wrapper opening was selected as a reasonable compromise between providing a large enough path for the sodium flow but without severely affecting the mechanical integrity of the SAs.

In accord with (Mikityuk et al., 2013), the wrapper opening aims to provide an accessible path for the coolant flow even if the sodium vapor production blocks the top of the channel. Hence, the cooling of the fuel region of the SA can be maintained, as schematically shown in Fig. 4, which is envisioned to improve the reactor behavior following the boiling onset.

The simulation is composed of two phases: 1) Up until 1000 s, a steady-state calculation is performed through which an equilibrium reactor state is to be reached, with nominal operational settings. 2) At 1000 s, the transient phase starts, initiated by inlet sodium mass flowrate decrease in accordance with the primary mass flow halving time, and the power evolution is calculated by the point kinetics solver.

### 4. Results

#### 4.1. Reference case

In Fig. 5, the power and mass flowrate normalized to their initial values are shown starting from the initiation of the transient phase. Therefore, from time 0 s, the substantial flowrate reduction is prominent since the forced convection of the primary system is lost. The blue curve...
indicates the power evolution, which exhibits a smooth reduction of the reactor power until ~ 50 s, from where the power starts to fluctuate with higher frequency resonances. These resonances, as it will be shown later, are related to the boiling of the sodium and subsequent quick condensation of the sodium bubbles. The simulation ends at 110 s, where the local clad melting temperature is being approached but still not reached (shown in Fig. 9), avoiding the simulation to abort and providing a consistent timeframe for the different simulation cases to be compared.

The right plot, in Fig. 5, with the given normalized power to flow ratio, is more representative for demonstrating the power removal capability and the thermal conditions in the core. Naturally, as the power to flow ratio increases, the reactor is approaching the limiting temperatures as the heat is not removed as readily as it is when higher coolant flow is present within the reactor core. In this sense, it is crucial to keep the power to flow ratio as stable and low as possible, increasing the grace time for any safety mechanism to be applied to keep the reactor under control. In the current simulation, until boiling of the sodium is reached, there is a steady but slow increase in the ratio, which halts at ~ 50 s, which is the point of boiling onset in the peak power SA. Starting from this moment, the transient is essentially driven by the balance of positive and negative contributions of sodium density effect. Initially, the power decreases due to an additional negative sodium voiding reactivity contributions at the plenum and at the top of the SA just below the sodium plenum. The voiding of the sodium plenum increases the neutron leakage out of the reactor core, which in return results in the reactivity decrease. Further transient progression is characterized by a high sodium vapor generation that slowly propagates downwards towards the mid-height of the fuel region of the SAs, which has an opposite effect on the reactivity of the core compared to the plenum voiding. This is mainly because the neutron spectrum is hardening with the lower amount of sodium, which in a fast reactor increases the reactivity. When the vapor propagates downwards in multiple SAs, a considerable positive reactivity is introduced, resulting in increased power generation, as it happens from ~ 90 s in the calculation.

The vapor propagation phenomenon described previously is visualized in Fig. 6 for the peak power SA located in the OC. The void fraction is plotted using grades of blue along the SA, demonstrating the vapor propagation along the axial height of the channel. The horizontal axis shows the elapsed time, starting from ~ 50 s corresponding to the boiling onset in the SA. The white color represents the liquid sodium in the channel, whereas the blue shows the sodium vapor. The bottom horizontal black line indicates the bottom of the fuel region, the middle is at the top of the fuel region, and the top line is at the top of the sodium plenum region. The initial vapor volume mainly occupies the top of the SA region, including the plenum, but with the passing of time, the vapor enters deeper into the fuel region until it is nearly entirely voided. Although the plot shows only the peak power SA, the same phenomenon happens increasingly for the other SAs, which eventually causes the power to increase rapidly.

Fig. 7 provides information about the reactivity evolution and its decompositon. The most prominent reactivity components after the boiling onset are related to the sodium effect. This effect was separated into the fuel and plenum regions, contributing with large positive and negative effects, respectively, while the blue line indicates the overall sodium effect in the core. At the transient initiation point, the effect is positive, as the decrease of sodium density in the fuel region has a stronger impact than the increased leakage due to density reduction in the plenum region. When the boiling starts, the overall positive sodium effect is mitigated as more vapor appears in the plenum regions. Nevertheless, in later stages, the positive effect starts to increase again as the fuel region is increasingly filled with vapor.

As for the Doppler effect, when the core power decreases, the
temperature of the fuel pellets is also decreasing, resulting in a positive feedback. This positive reactivity component decreases only when the sharp power peak occurs, resulting in an increase of fuel temperature from ~95 s. The control rod driveline (CRDL) reactivity is determined by the outlet sodium temperature evolution. The sodium temperature in the core and correspondingly the core outlet temperature are increasing, explained by the higher power to flow ratio shown in Fig. 5. This temperature increase results in a negative CRDL reactivity feedback as the control rods are inserted more into the core due to thermal expansion of their drivelines. The fuel expansion effect, driven by the increasing clad temperature, has a comparatively small effect on the generated power, decreasing the net reactivity.

In the following plot, in Fig. 8, a more detailed decomposition of the sodium effect is given as now the reactivity components are separated not only to plenum and core regions but also according to the different CGs of the reactor core. The summed total of the core and plenum region lines are identical to the corresponding reactivity components shown in Fig. 7.

Fig. 8 shows that CG2 has the fastest increase, first with the plenum sodium effect and then with the fuel region sodium effect. This is not surprising considering the fact that the highest power SAs are located within this group. Following the transient progression, the next group with the significant impact on the total core reactivity is the CG1, corresponding to the SAs in the IC region with high power outputs, which consists of nearly half of all SAs. Close to the end of the simulation, the previously observed sudden decrease of core power can be understood by looking at the sharp decline of the CG2 plenum reactivity component, driving down the reactor power. This phenomenon comes from the fact that the previous increase in power initiated sodium boiling in SAs, which did not boil previously. As it was shown before, the initial boiling phase is more pronounced in the plenum region until the vapor propagates down to the fuel region of the SA. Thus, nearly simultaneous involvement in boiling of a large group of SAs results in the introduction of the negative reactivity first. Later on, it can be expected that as the fuel region is also voided, the power will experience an even more significant increase than it was previously seen. The other cooling groups CG3, CG4, CG5, which are in the periphery of the core, are just starting to boil, and their effect is less pronounced, as well as less important in general as the number of SAs in CG at the core-periphery is smaller and both positive and negative contributions are weaker.

To observe the limiting temperature evolution within the peak power SA, Fig. 9 is plotted. The two most important temperatures, limiting the accidental scenario, are the fuel and cladding temperatures, of which the
inner surface values are plotted. Axially, the fuel node number 10 was chosen to be displayed as it is at the middle of the fissile fuel region having the highest power production. The sodium temperature is of little interest as it is already at the saturation point. Moreover, as it is a fast transient, the structural materials are not included either in the limiting temperature values as a more extended time period would be required for significant material creep. The Figure shows that the fuel temperature initially decreases, governed by the power reduction. This decreasing temperature phase ends soon after the boiling onset because the vapor produced in the peak power SA channel obstructs the sodium inflow, and therefore the cooling capability is declining. The vapor continues to fill the channel and propagates downwards, thus power production increases in the core, causing a jump in the fuel temperature by the end of the simulation. As for the cladding inner surface, the temperature evolution follows well the power to flow ratio growth up until the point of boiling onset, at which there is a jump in surface temperature followed by a high-frequency oscillating temperature phase. The sudden jump in the cladding temperature at ~ 60 s corresponds to the vapor propagation into the fuel region, reaching the middle fissile section, which reduces the heat transfer in the channel drastically. Finally, the further temperature increase at the end of the simulation is due to the substantial power increase similarly to the fuel temperature jump.

4.2. Results of the modified design SA

Based on the aforementioned results and the unfavorable behavior of the reactor after sodium boiling, displaying power increase by the end of the simulation that potentially leads to loss of the pin and SA structure integrity and further severe accident progression, a new simulation has been performed, where the SA wrapper design has been modified, as described in section 3.5.

The evolution of the core parameters and conditions are discussed hereafter for the improved SA design, as compared to those presented already for the reference case. First, the normalized power and mass flowrate are depicted in the left plot of Fig. 10. The mass flowrate is identical as it is a boundary condition of both simulations. On the other hand, the power evolution shows significant differences. Although, until the boiling onset, the power evolves the equivalent way for both reference and modified design cases, at ~ 60 s following a quick power decrease, there is a recovery and then a stable, slow reduction until the end of the simulation. The normalized power to flow ratio, shown in the right plot of the Figure, also reveals a promising tendency, as following the boiling onset, the value stays stable, not exceeding 4. Similar to the power evolution line, a decrease and a quick recovery at ~ 60 s are noticeable in this plot also.

To understand more the mechanism causing the dip and recovery of the power, Fig. 11 is plotted. The general structure of the Figure is the same as it was for the reference case, but here there is an extra horizontal red line, which corresponds to the elevation where the holes on the wrapper surface have been implemented. From the Figure, it is clear that at ~ 60 s after the start of the transient, there is a quick vapor generation within the peak power SA, where the vapor starts to propagate towards the fuel region. This vapor propagation quickly recovers, and as the bubbles are collapsing, the amount of vapor also reduces in the plenum region, which corresponds to the power increase in the reactor. This is followed by a more stable boiling pattern, which can be characterized as chugging boiling regime (Mambelli, 2018), creating the possibility of a stable decrease of the core power as the sodium vapor is retained only in the SA top region, including the plenum.

The reactivity decomposition, shown in Fig. 12, visualizes essential differences of the sodium reactivity effect evolution compared to the reference case. Soon after the boiling onset, the plenum contribution effectively counterbalances the fuel region boiling due to the applied design modifications. Therefore, the calculation shows close to zero overall sodium reactivity effect, which is an important safety accomplishment. As for the rest of the reactivity effects, both qualitatively and quantitatively comparable to the reference solution, except the Doppler effect, which does not include the decrease at the end of the simulation as the fuel temperature stays stable during the course of the calculation.

The distribution of the sodium effect between different CGs and axial regions is shown in Fig. 13. In this Figure, significant variation can be recognized compared to the reference case, which shows different boiling progress within the core. Initially, similarly to the reference figure, the peak power SA starts to boil in the CG2 region. This is followed by the bubble collapse and the stabilization of the boiling, reducing to a relatively low level of reactivity contribution from this CG. Simultaneously, within the CG1 region, both plenum and fuel sodium effects rapidly increase, dominating the overall sodium reactivity evolution. Overall, the plenum and fuel region contributions stay well balanced, having a close to zero total effect on the core. This shows that an increasing number of SAs reach the boiling onset, but the vapor propagation towards the fuel region is kept under control with the help of the holes in the SA wrappers. The changed flow pattern in the SAs gives greater importance to the CG1 SA units at least until the end of the simulated time period. Furthermore, the vapor generation in CG2 and CG3 are greatly reduced compared to the reference case, and they affect only marginally the sodium reactivity feedback effect. Should this stabilized boiling continue to spread radially towards the periphery region of the reactor, a further decrease in the reactivity could be foreseen.

Fig. 10. Normalized power/mass flowrate (on the left) and normalized power to flow ratio (on the right) vs time following the initiation of the transient.
because the leakage contribution is more significant in the periphery region SAs. In Fig. 14, the improvement in the limiting temperatures is shown. The parameters in the same SA and axial node are depicted as in the reference case analysis. When the fuel inner surface temperature is evaluated, there is a continuous decrease throughout the duration of the whole simulation, which is a definitive improvement on the reactor thermal state. Although the fuel temperature is improved, it has not been the primary concern but rather the melting point of the cladding, which was only around ~ 100 °C away. With the improved design, following a slow increase of temperature and a peak at ~ 60 s, it stabilizes at a nearly constant level. As the vapor does not propagate downwards towards the fuel, as it is blocked at the level of wrapper openings, a sufficient heat transfer from the fuel pins to the liquid sodium is ensured, and a sudden temperature increase, observed in the reference simulation, is avoided. This confirms the potential safety improvements which can be achieved by implementing the proposed design modifications.

### 4.3. Boiling comparison

To better understand the phenomenon behind the improved reactivity feedback effect evolution and more efficient power decrease during the accident progression in the core with the modified wrapper design, Figs. 15–17 have been plotted.

The expectation from the wrapper opening was that it creates a path for the sodium to flow through the fissile region even if the top of the SA is blocked by vapor as a result of boiling the coolant. In Fig. 15, the channel inlet and outlet liquid sodium velocities are depicted. On the left plot, which relates to the reference case, there is a fluctuating inlet flow velocity around 0 m/s during a few seconds time period at the end of the simulation, meaning that there is no real coolant flow into the SA. Periodically, the liquid sodium enters the channel when the condensation of a bubble occurs, and it is ejected from the channel when the evaporation increases. This inevitably leads, over time, to an increase of void fraction within the channel as the power generation continues. On the contrary, for the channel flow velocity on the right for the modified design, the liquid inlet flow velocity stays positive for the whole time period shown in the Figure, meaning the sodium inflow is maintained.
into the boiling channel. When, in the same plot, the liquid velocity at the channel outlet is assessed, it displays similar overall behavior as it was for the reference case, namely that the liquid is ejected and sucked into the channel as the sodium bubbles are increasing in size or being collapsed. This suggests that the sodium entering the channel leaves through the wrapper openings.

In Fig. 15, the sodium pressure evolution is plotted at four axial locations within the peak power SA for the reference case and the modified design case. The pressure plot reassures the chugging boiling pattern for both calculation cases and the bubble formation-driven flow velocity at the channel inlet for the reference case when the pressure peaks are compared to the inlet flow peaks in Fig. 14. Nevertheless, the size of the pressure peaks is about a factor of three times greater for the modified design case, suggesting a complete collapse of the sodium bubbles, in contrast to the partial decrease for the reference case. Therefore, by implementing the windows on the wrapper, the extra amount of cooling of the fuel, provided by the continuous coolant flow, allows complete bubble collapse, which is also shown in Fig. 6.

The state of the core can be characterized by the average void fraction in the plenum and fuel region over all channels, shown in Fig. 17, as the sodium reactivity feedback effect evolution is mainly governed by this parameter. The left plot in the Figure presents the void fraction for the reference core design, exhibiting a step increase in void fraction both

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**Fig. 14.** Fuel inner surface and cladding inner surface temperature evolution in the peak power channel at the fissile fuel middle node height.

**Fig. 15.** Liquid sodium velocity at the channel inlet and outlet for the reference case (left) and the modified design case with windows (right) within the peak power SA.

**Fig. 16.** Pressure evolution at the fissile top, sodium plenum, shielding, and channel top axial nodes for the reference case (left) and the modified design case with windows (right) within the peak power SA.
in the fuel and in the plenum regions. As more SAs reach the boiling onset condition, there is a continuous increase in the plenum voiding, but soon after the plenum boils, the sodium vapor propagates downwards into the fuel region, causing an exponential void increase in that region for the considered simulation time. When the image at the right is compared, which is the modified design case, to the reference condition, a stark difference to realize is the factor of two decrease in sodium void fraction. Besides, the void propagation throughout the core also differs significantly. At \( \sim 60 \text{ s} \), following a rapid increase in both the plenum and fuel void, it decreases close to zero, which relates to the initial power peaking SA boiling onset and the subsequent recovery and stabilization of the boiling process. Overall, the void propagation in the plenum is significantly stronger than it is in the fuel region, indicating a potentially safer and more stable boiling regime for the reactor. This leads to the conclusion that the boiling improvement seen in the power peaking channel is valid for the other channels of the reactor core also.

4.4. Sensitivity analysis

To assess the robustness of the calculation results and the sensitivity to certain modeling parameters, a few sensitivity cases have been analyzed. In Table 5, the modified modeling parameters are listed. These parameters were chosen based on their importance played in the sodium boiling phenomenon. Two separate groups of parameters can be distinguished: 1) Design modifications of the SAs, affecting the void propagation and 2) reactivity effect related parameters. For the current sodium boiling analysis the two most important reactivity effects are the Doppler effect and the sodium density reactivity effect. According to this, the chosen parameters have either direct effect on the reactivity evolution, such as plenum sodium void reactivity coefficient change or indirect effect, introduced by fuel gap conductance alteration, affecting the Doppler effect.

On the aforementioned sensitivity cases, no separate results description is given due to the fact that none of the changes modified qualitatively the boiling progressions within the reactor core. The fuel melting occurred for every of the cases, the maximum benefit achieved was only a few seconds of delay of the melting.

In this study, the specific sodium boiling modeling aspects included in TRACE by PSI have not been evaluated but rather emphasis was given to analyze the specific behavior of the ESFR-SMART core design. This is because such study has been performed in (Perez-Martin et al., 2022), based on the validation exercise of TRACE with a ULOF sodium boiling experiment.

5. Conclusions

In this paper, the ESFR-SMART core behavior under ULOF accidental conditions has been investigated with the TRACE system code, using a low void effect core design. In the analysis, special attention has been given to the sodium boiling reactivity effect and the coolant flow pattern evolution throughout the accident. The full reactor model in TRACE was simplified into a 42 channel reactor core model, accounting for 1/12th of the full core, exploiting the symmetry of the reactor. A ULOF accident simulation starting from an initial steady-state at nominal operating conditions is studied for the time period of 110 s. The simulations are performed for the reference design case and for the one with the modified SA design with the aim of increasing the stability of the reactor behavior during sodium boiling. The following conclusions are made based on the analysis:

- When the reference SA design was used in the analysis, the reactor power stays stable for \(~ 40 \text{ s}\) during transient, but later on, it starts to show instability and fluctuation. The sodium vapor initially fills the plenum above the fuel within the SA, counterbalancing the otherwise positive sodium reactivity effect. The anticipated stable chugging boiling is not reached with the originally proposed design as the vapor does not collapse entirely but starts to propagate downwards to the fuel region, providing strong positive reactivity insertion into the core. Although stable boiling condition was not reached, no power runaway has been observed throughout the accident for the simulated period of time, which can potentially provide enough grace for other safety measures, such as magnetic control rod release device, to shut down the reactor safely.

![Average void fraction along the fuel and plenum regions over all the modeled channels for the reference case (left) and the modified design case with windows (right).](image)

Table 5

<table>
<thead>
<tr>
<th>Sensitivity case</th>
<th>Change magnitude</th>
</tr>
</thead>
<tbody>
<tr>
<td>SA outlet design modification</td>
<td>The pin bundle geometry of shielding and reflector was changed into a sleeve geometry (Guidez and Prele, 2017)</td>
</tr>
<tr>
<td>SA flow gagging scheme modification</td>
<td>3 % of the CG1 sodium flow was redirected to the higher power CG2 zone</td>
</tr>
<tr>
<td>Fuel gap conductance increase</td>
<td>10 % increase</td>
</tr>
<tr>
<td>Fuel gap conductance decrease</td>
<td>10 % decrease</td>
</tr>
<tr>
<td>Plenum sodium void reactivity coefficient increase</td>
<td>20 % increase</td>
</tr>
<tr>
<td>Plenum sodium void reactivity coefficient increase + core sodium void reactivity coefficient decrease</td>
<td>20 % plenum increase + 20 % core decrease</td>
</tr>
</tbody>
</table>
As a modified case, a change has been implemented on the SA design. Namely, SA wrapper openings have been introduced in the model towards the inter-SA gap providing a flow path for the sodium even when at the top of the SA the path is blocked by sodium vapor. Using this design, stable chugging boiling regime was achieved starting from boiling onset, and the reactor behavior stays stable throughout the simulated timeframe. In this sense, the applied design modification could potentially help to reach the desired safety level for the reactor during a hypothetical ULOF accidental scenario. Nevertheless, before applying the assessed SA modification as the reference design option, its effect during nominal operation condition requires further study.

Declaration of Competing Interest

The authors declare that they have no known competing financial interests or personal relationships that could have appeared to influence the work reported in this paper.

Data availability

Data will be made available on request.

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