Chugging boiling in low-void SFR core: new phenomenology of unprotected loss of flow

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Abstract. Calculational analysis of the unprotected loss of flow (ULOF) accident in a Generation-IV SFR, featuring a low-void core design, shows that the chugging sodium boiling regime in the core could last for several hundred seconds during the accident. While in the case of the traditional positive-void SFR core the sodium boiling onset is almost immediately followed by the power run-away, fuel bundle overheating, melting and relocation (*i.e.* severe accident), the chugging boiling regime in the low-void SFR core could allow avoiding the power runaway and avoiding or at least significantly postponing the cladding overheating and melting caused by the permanent dryout. The low-void core design therefore could be classified as a new safety measure acting as a level of defence preventing the severe accidents. The knowledge in the area of the chugging regime of the sodium boiling is very limited and very few corresponding experiments were performed. The paper discusses possible phenomenology of the chugging boiling in the low-void SFR core and makes proposal for the new test facility. The goals of the new experiment are to better understand the conditions under which the chugging boiling regime establishes in the specific geometry of the low-void SFR core channel as well as the conditions under which the pressure pulses caused by the abrupt condensation can occur, and validate the system code for these conditions.

Key Words: Sodium fast reactor, ULOF transient, chugging

1. Introduction

The concept of Generation-IV sodium fast reactors (SFR) aims at improving the use of natural resources while reducing the amount of high-level radioactive waste, providing enhanced safety features and keeping resistance against proliferation.

The low (near-zero) void reactivity effect (*i.e.* the change of the multiplication factor when removing all sodium from the core) is an important safety goal of a number of Generation-IV SFR designs, *e.g.* French Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) [1, 2] or Russian BN-1200 reactor [3]. A number of innovative features are considered to reach this goal, in particular: 1) the flat active core, *i.e.* low height-to-diameter ratio to enhance the axial neutron leakage from the core; 2) a free space called sodium plenum (cavity) above the fuel bundle to scatter back (reflect) neutrons during normal operation; 3) absorber pins above the sodium plenum to reduce the neutron back-scattering (reflection) in case of the sodium plenum voiding.

One of the accident scenarios for which safe behaviour of the core should be demonstrated is an unprotected loss of flow (ULOF) accident. This paper briefly describes a typical low-void fuel subassembly design, discusses the ULOF phenomenology using published analyses and presents the first results of the analytical study on designing the new water test facility to mock up sodium boiling in the low-void core design geometry.

With the new experiments we aim at studying the conditions under which the chugging boiling regime may occur in the low-void SFR core channel geometry. We also aim at better understanding the conditions under which the condensation-induced pressure pulses can take place. The new knowledge will be used to validate the system codes for these conditions and to study the applicability of the new data for the sodium boiling. In particular, the ability of the system codes to correctly predict the chugging boiling as well as the pressure waves for the steam-water flow will evidence that the code predictions of the phenomena during the sodium two-phase flow (*e.g.* pressure spikes) have possible physical explanations and not just the result of numerical instabilities.

2. Low-void core concept

A typical configuration of the low-void fuel subassembly is illustrated in *FIG. 1* [4]. A sodium plenum (cavity) is located right above the top of the active fuel and a bundle of neutron absorbing elements is positioned above the sodium plenum. This particular characteristic allows turning the sodium void effect from positive to nearly zero[1]. Based on the sodium temperature distribution, the most effective leakages will be at the top of the core, where the density of sodium is the lowest and the margin to boiling is minimal. The sodium plenum acts as a reflector during the normal operation, improving the neutron economy by reflecting back the neutrons leaking towards the top of the core. In case of sodium boiling, the sodium plenum will be voided and the neutrons will tend to leak upwards along the axial direction being absorbed in the absorbing zone, which prevents the neutrons from returning back into the core. A similar fuel subassembly design complemented by a number of other heterogeneous features to enhance the axial neutron leakage is used in the ASTRID core (see *e.g.* [5] for more details).

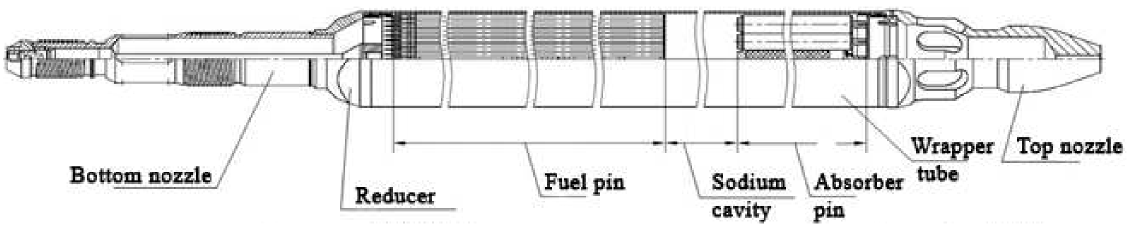


FIG. 1. Heterogeneous structure of the low-void fuel subassembly of BN-1200 SFR [4]

3. Unprotected Loss Of Flow (ULOF) accident

In the case of a traditional positive-void SFR core (*e.g.* Superphenix) the sodium boiling onset is quickly followed by the power run-away, fuel bundle overheating, melting and relocation (*i.e.* severe accident), see *e.g.* [6].

Calculational analysis of the ULOF accident in a Generation-IV SFR, featuring a low-void core design, shows that the cyclic sodium boiling regime in the core could last for a long time (several minutes) during the accident.

In [3] the analysis of the 1D DINROS code predicts neither a power excursion nor permanent dryout in the low-void core during the ULOF accident (see Figure 1, Option 1 in [3]). During the simulated 80 s of the accident, the reactor power keeps reducing. After the boiling onset, the power exhibits oscillations, caused by the reactivity oscillations due to cyclic voiding and filling of the sodium plenum.

[1] It should be noted that the optimization of the sodium void reactivity effect should be performed in combination with optimization of the Doppler effect, *e.g.* the use of the metallic fuel with the operating temperature closer to the coolant temperature compared to the oxide fuel could significantly reduce the positive Doppler effect component during ULOF (see Section 15.3.3 of [7]). The optimization of the Doppler effect however is outside the scope of the paper.

In [8] the ULOF accident in the low-void SFR core was analysed in more details using the 1D CATHARE 2 thermal-hydraulic code with point kinetics option. For the chosen (relatively slow) pump coastdown scenario during the simulated period of 2500 s neither power excursion nor permanent dryout were predicted after the sodium boiling onset. According to the conducted analysis after the boiling onset the inlet mass flowrate in the boiling subassemblies oscillates around the natural circulation level, remaining all the time positive, demonstrating the absence of a permanent channel blockage due to the formation of vapour bubbles, which would have caused the fuel pins dryout due to the downwards progression of the boiling front.

This behaviour could be explained by the presence of a hydraulic instability inducing a cyclic process along the transient: the chugging phenomenon. According to the classification in [9], the chugging phenomenon is defined as a compound relaxation instability, which foresees a periodic adjustment of metastable conditions during a boiling regime. Compound relaxation instabilities involve rather static phenomena coupled to each other. The outcome is the production of a repetitive behaviour, often but not necessarily periodic. It is important to point out that these repetitive cycles can be strongly irregular, so that each flow excursion can be considered as hydraulically independent from the others. Chugging is defined as the cyclic phenomenon characterized by periodic expulsion of coolant from a flow channel [9]. The expulsion of the coolant through the ends of the channel is usually performed by the bubbles of vapour formed during the boiling regime in rather small channels. The chugging cycle can be divided in three main phases: incubation, expulsion and re-entry of the fluid.

An important outcome in this study is the predicted vapour collapse and surface re-wetting occurring at the top of the fissile zone. This would mean that the chugging boiling regime in the low-void SFR core could allow avoiding the power runaway and avoiding or at least significantly postponing the cladding overheating and melting caused by the permanent dryout. The low-void core design therefore could be classified as a new safety measure acting as a level of defence preventing the severe accidents.

Since the knowledge in the area of the chugging regime of the sodium boiling is very limited and very few corresponding experiments were performed, it is important to study the sodium chugging phenomenon in order to verify the actual possibility of re-wetting and to assess the phenomena involved in case of collision between the two liquid fronts due to liquid re-entry from the upper plenum. For this purpose, a small-scale two-phase flow facility is in the phase of conceptual design. The preliminary study on the design of this water facility to mock up sodium boiling is presented in the next section.

4. A new water test facility

A small-scale two-phase flow water facility is being designed at Paul Scherrer Institut (PSI) in frame of the new Horizon-2020 ESFR-SMART project [10] in order to study the chugging boiling regime for the specific geometry of the low-void SFR core concept.

A preliminary study on the design of the facility was performed at PSI with the purpose of simulating the chugging cyclic process with the US NRC TRACE code in a simple configuration representing the low-void subassembly axial geometry (*FIG. 2*). The facility is filled with water at room temperature and atmospheric pressure and saturated steam is injected to study the conditions of the chugging boiling regime occurrence.

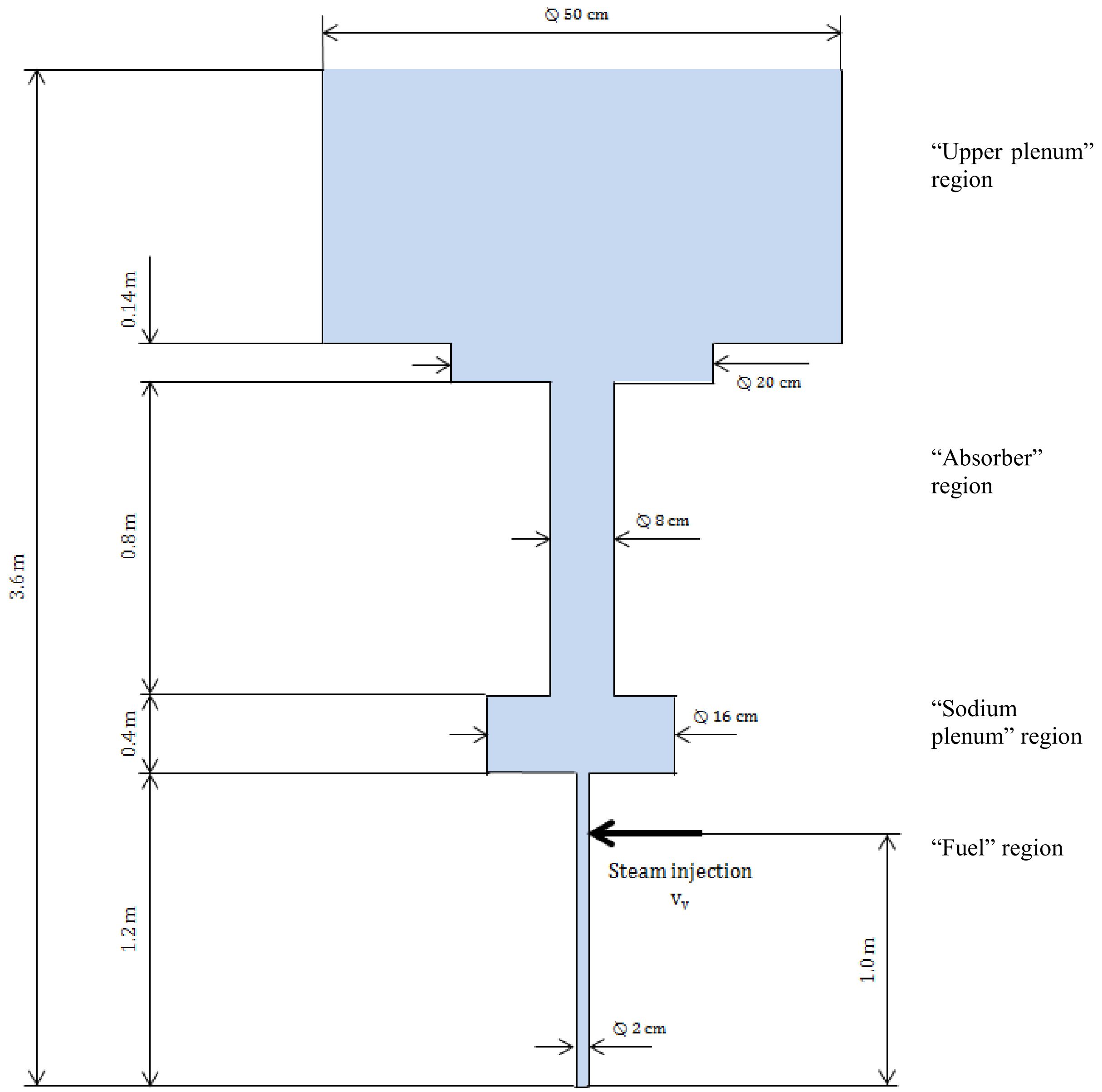


FIG. 2. Geometrical parameters of the model setup.

The configuration and the setup of the simulation conditions are chosen according to several assumptions:

- The test section is filled with water at 20ºC at 1 bar in order to reproduce the SFR subassembly conditions. The use of the water as a sodium simulant is justified by similar fluid dynamics properties of water and sodium. The water has already been revealed to be an accurate water simulator in other experiments [11].

- Saturated steam (100ºC) produced in a separate steam generator is injected with variable flowrates following a sinusoidal trend, in order to simulate upper fuel region boiling regime. Indeed, the vapour generation is not constant during the ULOF transient, as described by the neutronics/thermal hydraulics coupled calculation accounting for the reactivity feedbacks [3, 8]. When the vapour generated at the top of the fuel bundle enters the sodium plenum, a negative void feedback leads to the power decrease and to the drastic reduction of the amount of steam generated. On the other hand, when the vapour collapses in the sodium plenum, the power rises up again due to the positive reactivity insertion brought by the liquid re-entry. As a result, the vapour generation rate increases. The period of the oscillations is chosen 4 s in the presented analysis (*FIG. 3*) to be consistent with frequency of the power oscillations in [3]. The dependence of the boiling regime on the oscillation frequency will be one of the subjects of a sensitivity study.

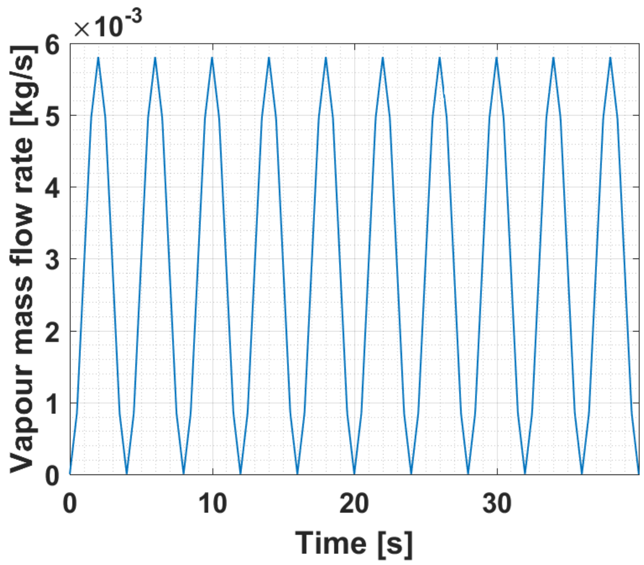
- Zero inlet flowrate was set as a boundary condition. This reveals to be a strong assumption because even during pump coastdown in ULOF transients a natural circulation flow in the subassembly occurs, helping the surface to be wetted. However, since the influence of liquid re-entering back from the "upper plenum" must be assessed, this assumption is satisfactory.

- The fuel rod bundle is "lumped" into a single 1D channel connected to the centre of the "sodium plenum".

- The absorber pin bundle is represented by a 1D single channel.

- In order to add one degree of freedom to the problem, the "sodium plenum" and the "upper plenum" are modelled as 2D (r-z) VESSEL components.

- The ratio between flow areas of the different sections is approximately reproduced. The objective of this study is to find the simplest reliable configuration allowing reproducing the conditions for chugging phenomenon in presence of sodium boiling. The dimensions of both "sodium plenum" and "absorber" regions are chosen in order to resemble the real geometrical parameters of the low-void subassembly. With the purpose of reproducing the flow cross section inside the fuel channel, a fuel pipe diameter of 2 cm is modelled. The geometry setup employed in this preliminary study is represented in *FIG. 2*.

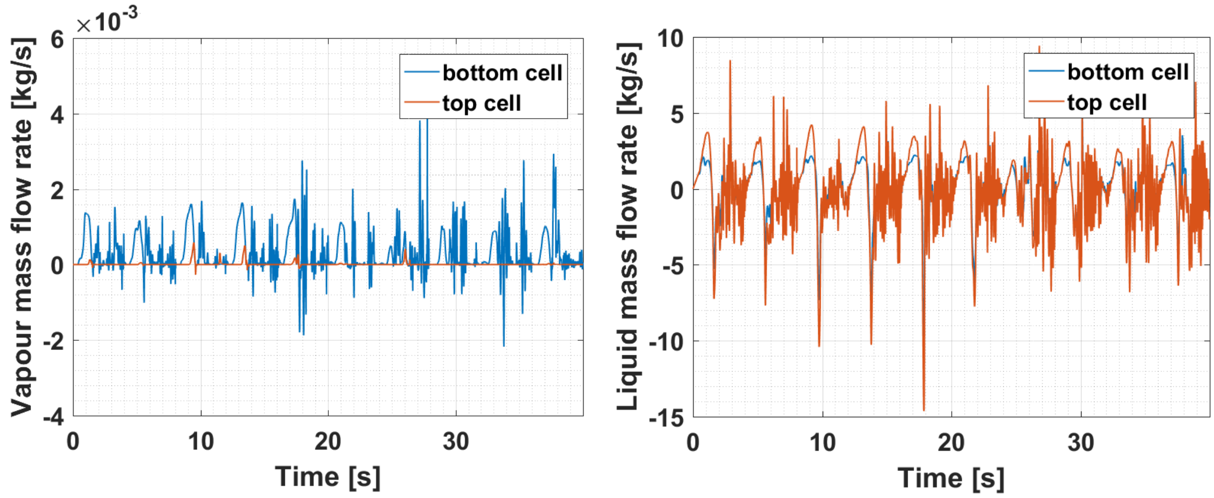


- *FIG. 3. Vapour mass flow injected into the "fuel" region*

The total simulation time is equal to 40 seconds. The results show how the cycling chugging process is simulated. The mass flowrates of vapour and liquid, respectively, in the "absorber" region are plotted in *FIG. 4a* and *FIG. 4b*. The steps of chugging phenomenon can be recognized. As the vapour is injected in the upper part of the "fuel" zone, the voided region tends to expand upwards toward the "sodium plenum" and the "absorber" regions. The steam bubbles rising in the channel either collapse due to the contact with cold water or reach the "upper plenum" region, departing from the boiling channel. Due to the consequent local pressure drop formed in the channel, the liquid flow changes direction: the liquid water from the "upper plenum" is sucked back into the "absorber" channel. As a result, the remaining vapour in the "absorber" region is pushed downwards by the liquid, reaching the plenum vessel. The liquid phase, accelerated by the pressure drop, reaches the plenum and here causes the abrupt condensation of the steam and the collapse of the vapour phase, as visible from the average void fraction in the "sodium plenum", plotted in *FIG. 5a*. It has to be highlighted that the periodical variation of the injection rate determines the overall behaviour of the system. Indeed, even if the trend of the physical variables presented is strongly irregular and continuously oscillating, the strongest peaks and dips occur in correspondence with the sinusoidal behaviour of the injection. For example, the void fraction in the "sodium plenum" (*FIG. 5a*) reaches the zero value correspondingly to the periodic minimum injection rate.

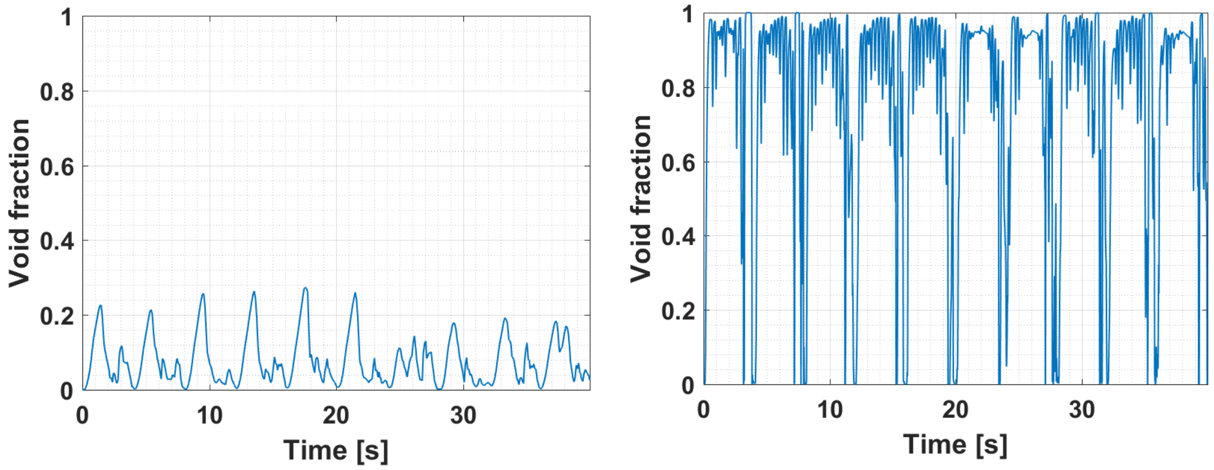
Another important outcome is the demonstration of the possibility that the "fuel" channel wall may be rewetted thanks to the chugging phenomenon. *FIG. 5b* represents the evolution of the void fraction at the top of the "fuel" region over time. It is important to notice how the re-wetting of the "fuel" zone occurs only at the moments of the minima of the injection rate: the re-entry of liquid into the plenum occurs also in the middle of the period, as visible in *FIG. 6*, where the liquid velocity assumes high negative values. However, Counter Current Flow Limitation (CCFL) conditions are established at the bottom of the "sodium plenum" while the vapour is injected at high rate and the liquid is not able to enter the fuel channel and re-wets the surface. On the other hand, when the vapour injection rate approaches the minimum, the CCFL conditions do not occur anymore and the liquid is able to re-enter even in the fuel channel. *FIG. 6* shows the liquid mass flowrate at the top of the "fuel" zone.

The investigation of the pressure behaviour at the bottom of the rig (*FIG. 7*) reveals surprising results. Pressure spikes up to more than 100 bars are generated in the system in correspondence to the vapour collapse times, namely of the minimum injection rates and zero voids. The explanation of this behaviour can be attributed to the phenomena occurring during liquid re-entry: the abrupt condensation and bubble collapse result in the two liquid fronts bumping into each other and the liquid velocities drop suddenly to zero. The kinetic energy of the two liquid fronts is turned into a considerable pressure pulse, which propagates downwards through the fuel channel. The phenomenon of the pressure pulse generation and propagation is known in the two-phase water flow as a condensation-induced water hammer. In general, a condensation induced water hammer occurs when steam is either allowed to enter into a pipe containing water or, inversely, when subcooled water is injected into a piping system containing steam (*i.e.* as a result of accumulator injection during a LOCA in LWRs). The abrupt condensation at steam/water interface leads to the propagation of significant pressure wave through the core structures, which may lead to considerable damage to the material. Regarding sodium, condensation induced hammers were never measured. However, the behaviour of sodium is similar to water. As a consequence, the propagation of "Sodium Hammer" pressure waves in space must be taken into serious consideration, since it can potentially impact core reactivity during the interaction with the fuel subassembly material.



a) b)

FIG. 4. Vapour (a) and liquid (b) mass flowrates in the "absorber" region.



a) b)

FIG. 5. Average void fraction in the "sodium plenum" region (a) and void fraction at the top of the "fuel" region (b).

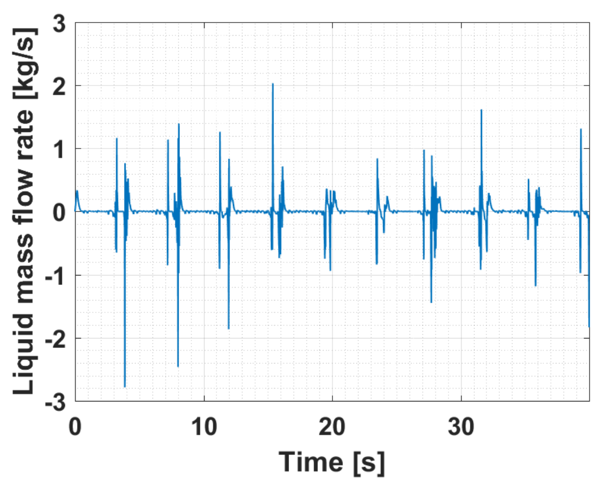


FIG. 6. Liquid mass flowrate at the top of the "fuel" region.

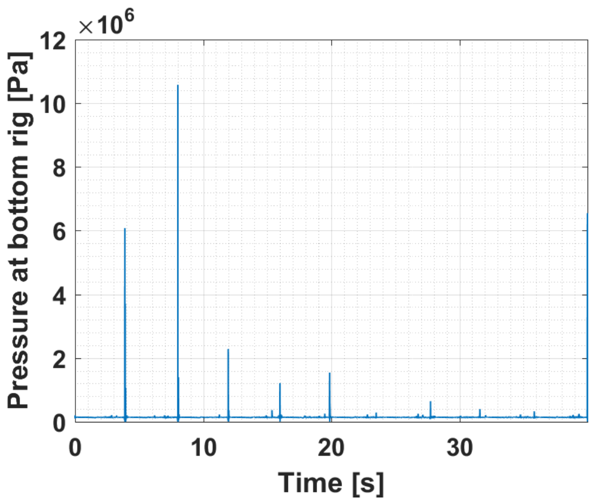


FIG. 7. Pressure at the bottom of the facility.

The TRACE code is able to simulate a water hammer and pressure wave propagation as demonstrated *e.g.* in [12]. Indeed, the TRACE equations include terms which are important for modelling condensation-induced water hammer (note that in the two-phase sodium simulation, the same equations are used as for the two-phase water flow [13]). A deep study of the chugging boiling phenomenology including pressure wave generation is the fundamental objective of the future small-scale two-phase flow facility. Indeed, the presence of possible "Sodium Hammers" potentially impacting the core reactivity must be assessed, since it can be related to important safety features such as the avoidance of permanent dryout during the ULOF accident.

5. Summary

The new phenomenology (compared to traditional SFR core designs) of the coupled thermal-hydraulic and neutronics behaviour of the low-void SFR core designs in the Unprotected Loss Of Flow event was found in a number of analytical studies. Inspired by these findings and by the limited amount of the relevant data, a calculational analysis using the US NRC TRACE code was performed in support of the conceptual design development of a water test facility to mock up sodium boiling in the low-void core subassembly during Unprotected Loss of Flow Accident. This analysis shows that during the course of the accident a hydraulic instability known as the chugging phenomenon can occur. The periodical departure of the vapour from the boiling channel and the re-entry of the relatively cold liquid sodium from the upper plenum can result in the vapour collapse and fuel clad surface re-wetting. Such periodical cooling regime may considerably postpone or even exclude the permanent dryout of the fuel rod claddings. Another finding is that the strong condensation induced pressure waves predicted by TRACE may influence the core reactivity via perturbation of the core geometry. These findings of the conducted analytical study motivate the further optimization and construction of the facility planned in the frame of the new Horizon-2020 ESFR-SMART project.

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