Passive Complementary Safety Devices for ASTRID severe accident prevention

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Abstract. Sodium-cooled Fast Reactor is one of the Generation IV reactor concepts. It has been selected to secure the nuclear fuel resources and to manage radioactive waste. In this context, the CEA (French Commission for Atomic Energy and Alternative Energy) with its partners is involved in a substantial effort on the ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) Project.
ASTRID core design is mainly guided by safety objectives. The first one is prevention of the core meltdown accident, at first through natural favourable behaviour of the core and of the reactor, and with the addition of passive complementary systems if natural behaviour is not sufficient for some transient cases. The second one is the mitigation of the severe accident to guarantee that core melting accidents do not lead to significant mechanical energy release.
The robust safety demonstration is supported by complementing ASTRID core with two types of Complementary Safety Devices dedicated to core damage prevention that would passively shut down the reactor. The first type is based on the Curie point use of electromagnetic devices that hold some specific ASTRID shutdown systems to address unprotected loss of heat sink transients (ULOHS).
Key Words: Passive complementary safety devices, ASTRID, unprotected loss of flow, unprotected loss of heat sink

1. Introduction

Fast Reactors have a unique capability as sustainable energy source in terms of both utilisation of fissile material for energy production and minimisation of the nuclear wastes, due to hard neutron spectrum. As a result of a screening review of candidate technologies and in the frame of the international forum Generation IV, Sodium-cooled Fast Reactors (SFRs) are among the selected systems to address the sustainability issues with a coherent set of innovative requirements. Guidelines for the definition of such innovative requirements are the Generation IV goals with significant improvements on economy, safety, environment, waste management and proliferation resistance as promising milestone towards a sustainable nuclear energy.
In terms of sustainability, Generation IV systems shall make the best use of Uranium resource, be able to multi-recycle Plutonium, and have the capability to perform transmutation of certain minor actinides. This calls for fast neutron reactors and a closed fuel cycle.
In terms of safety, improved and robust safety demonstration with regard to former fast reactors is expected: enhanced prevention of whole core melting accidents, exclusion of credible way energetic accident sequences, prevention and mitigation of risks due to sodium chemical reactivity, robustness to external hazards. The level of safety must be at least equivalent to Generation III reactors. Lessons learnt from Fukushima accident will also be taken into account.
In terms of economy, Generation IV systems shall be competitive, for the same overall performance, compared to other sources of energy at the time they will be put into operation.

This means a lot of efforts with regard to investment costs but also to availability and operation costs.
Proliferation resistance applies not only on reactor, but also to the whole fuel cycle.
A June 2006 French law on sustainable management of radioactive materials and wastes requests that, concerning transmutation of long-lived radioactive elements, studies and investigations shall be conducted, in order to provide by 2012 an assessment of the industrial prospects of those systems. Fast Reactor strategy was confirmed in May 2008 at Ministry level and in September 2010 an agreement was published between CEA and French Government in order to conduct design studies of ASTRID prototype and associated R&D facilities [1].
ASTRID is an acronym for Advanced Sodium Technological Reactor for Industrial Demonstration. The essential objective of the ASTRID prototype is to demonstrate advances on an industrial scale by testing innovative options in areas earmarked for improvement (in particular safety, operability and inspection and repair). ASTRID will also be capable of carrying out radioactive waste transmutation in order to demonstrate the industrial scale feasibility of this technique for reducing volume of end waste and lifetime of ultimate waste. Furthermore, ASTRID should be capable of carrying out experimental irradiation in the fast neutron spectrum.
ASTRID is a 1500 MWth reactor (about 600 MWe). Characteristics should be extrapolable to future high-powered industrial SFRs. Innovation and technological breakthroughs have been favoured, while maintaining risk at an acceptable level [2].
ASTRID core design is mainly guided by safety objectives. The first one is prevention of the core meltdown accident, at first through natural favourable behaviour of the core and of the reactor, and with the addition of passive Complementary Safety Devices if natural behaviour is not sufficient for some transient cases. The second one is the mitigation of the severe accident to guarantee that core melting accidents do not lead to significant mechanical energy release.

2. Core

ASTRID core design objectives are the following [3]:
- Natural favourable behaviour for ULOF (target criteria: no sodium boiling) and ULOHS (target criteria: limitation of the reactor temperature).
- Sodium void effect minimized (target criteria: negative sodium void effect).

The CFV (French acronym for Low Sodium Void Effect) core concept is based on a low sodium void effect. This core concept involves in the inner part a heterogeneous axial fuel column made of $UPuO_2$ pellets with a thick fertile plate in the central zone and characterized by an asymmetrical, crucible-shaped core with a sodium plenum above the fissile area (Figure 1). Feedback coefficients are optimized (in particular sodium void coefficient) to allow better natural behaviour in case of unprotected loss of flow transients, in particular no sodium boiling with enough margins. These characteristics allow the possibility to eliminate energetic severe accident scenarios for the CFV core in case of ULOF. The overall negative sodium void coefficient provides margin in case of global core boiling or sodium draining. The low core pressure drop is also favourable in case of rapid loss of flow transient or in case of natural convection flow. The weak loss of reactivity of the core during irradiation cycle is also favourable in case of the control rod withdrawal accident.
The CFV oxide core is composed of inner and outer core fuel subassemblies, and rows of reflectors. Additional rows are added for shielding. Locations are foreseen for failed and experimental subassemblies (S/As).

In the reference design of the reactivity control architecture of the ASTRID reactor, there are two shutdown systems composed of two different types of devices, diversified with respect to the absorbing assemblies (RBC main devices and RBD diversified devices) and the mechanisms. Disconnection in case of scram between the RBD rod and its drive mechanism would occur via an in-sodium electromagnet. This constitutes a diversification against common mode failure of insertion of RBC rods into the subassemblies that makes it possible to guarantee safe core shutdown in case of significant deformation of the reactor block that would be likely to block the RBC mobile rods in their wrappers. Both RBC and RBD systems are dedicated to power regulation, compensation for the reactivity change during the lifetime and normal or emergency shutdown.

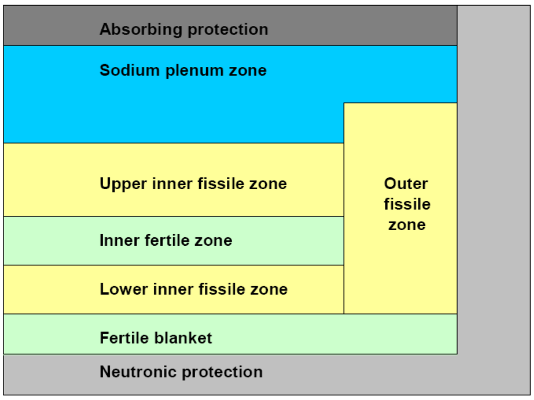

FIG. 1. CFV core.

3. Complementary safety devices

The CFV core has favourable kinetic properties. However, the need for additional safety devices that would complement ASTRID CFV design core natural behavior in case of ULOF and ULOHS transients emerged from the transient studies in order to meet temperature criteria on coolant, core and primary circuit structures [4]. Therefore, the implementation of complementary safety devices appears necessary to provide sufficient margin (Figure 2).
The first type is a hydraulically suspended absorber rod subassembly, called RBH, dedicated to ULOF transients; under normal operation, the absorber rod subassembly is hydraulically suspended above the core by the upward flow of the sodium coolant. Should an ULOF event and the associated drop in flow rate occur, this upward force would become insufficient, thus allowing the absorber insertion into the active core region by gravity.
The second type (RBD) is based on the Curie point use of electromagnetic devices that hold some specific ASTRID shutdown systems to address ULOHS transients.

3.1.RBH

This concept consists of a mobile absorber rod in a stationary hexagonal wrapper tube identical to the one of fuel subassemblies. At the normal operation condition, the absorber rod is hydraulically suspended above the core by the upward flow of the sodium coolant. Should a ULOF event and the associated drop in flow rate occur, this upward force could become insufficient, making the rod drop hydraulically actuated and allowing the absorber material to insert by gravity into the active core region.

TABLE I: RBH reference data

Actuation mass flow rate of the nominal value	45
Response time (s)	0
Drop time (s)	1
Negative reactivity (pcm)	742

3.2.RBD

Additionally to its function in the reactivity control system, RBD could fulfil the purpose of a passive complementary safety device using the Curie point characteristics of suitable material in its in-sodium electromagnetic system: since the magnetic force is abruptly lost when the alloy is heated up to its Curie point in case of core outlet temperature increase under ULOHS, the device detaches the absorbing part of the rod.

TABLE II: RBD reference data

Response time (s)	2
Drop time (s)	1
Negative reactivity (pcm)	2571 (for a total of 6829)
Actuation temperature (°C)	650

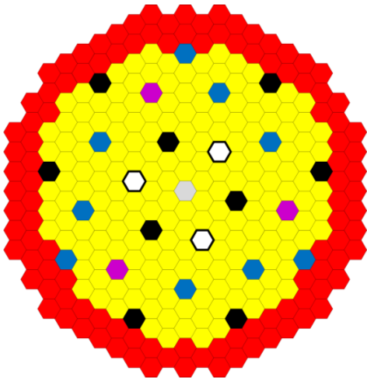

FIG. 2. Configuration of ASTRID core including RBH (fuel S/As in yellow and red; RBC and RBD S/As in black and blue; RBH in pink).

4. Thermal-hydraulic safety studies

4.1.Tools

CATHARE code

The CATHARE thermal-hydraulic system code has been developed and extensively validated in collaboration between CEA, EDF, IRSN and AREVA for the French Pressurized Water Reactors (PWR). The CATHARE code is the reference code in France for PWR safety analysis, but it has also been used for VVER, BWR, RBMK and experimental reactor applications. The code is based on a two-fluid model with six equations [5]. Mass, momentum, energy balances equations are written for the gas phase.
The CATHARE code is now the French reference code for SFR application. Sodium has been introduced in CATHARE as a new fluid and specific friction and heat transfer correlations have been implemented.
A large qualification effort has been performed for two-phase thermal-hydraulics. A similar work is required for single-phase thermal-hydraulics applied to Sodium-cooled Fast Reactors.
The calculation of the evolution of the core power is performed by the point kinetics model of CATHARE. Specific reactivity feedbacks in SFR are considered in the CATHARE code:

- Doppler effect induced by fuel temperature variation,
- void effect induced by sodium density change,
- fuel axial expansion,
- thermal expansion of core structures (clads and wrappers),

- control rods insertion in the core due to relative thermal expansion of different structures,
- modification of the compactness of the core (radial expansion) related to the grid plate expansion.

These reactivity feedback effects are taken into account in the CATHARE code through reactivity coefficients based on 3D neutronic computations.
The present calculations are carried out using the CATHARE2 version CATHARE v25_3_ mod5.1.

TrioCFD code
A an important challenge has involved the coupling of system and CFD codes to take into account 3D effects on the global system behavior during transient situations The TrioCFD Computational Fluid Dynamic (CFD) code developed at CEA has been progressively adapted to SFR concerns.

TrioMC code
Core design and safety studies require the calculation of the local pin cladding temperature, in order to ensure that design and safety criteria are met. To that end, a complete-core sub-channel code (TrioMC), has been developed at CEA in the framework of the TrioCFD code [6].

4.2.Computational thermal hydraulic schemes

For thermal-hydraulic transient calculations required to support the safety analysis of SFRs, the modeling and calculation schemes to be implemented are determined based on the studied physical phenomena, functionalities of the various codes used for these calculations and the parameters of interest in the different parts of the reactor [7].
The choice of computational thermal-hydraulics schemes also depends upon the objectives of the characterization of the transient e.g. directed at a single component or a single phenomenon or addressing the entire plant and upon the simulation time.

4.3.Unprotected Loss of Flow sequence due to the primary pump coast down without reactor scram while the secondary circuits remain operational for power removal

In this sequence, with the secondary circuits continuing to evacuate power, the outlet temperature of the intermediate heat exchangers remains nearly constant during the transient. The evolution of the flow of natural convection in the primary circuit should be less influenced by the stratification in the hot plenum. With the secondary circuits continuing to operate, we expect that the flow of natural convection in the primary circuit is rather high and is relatively little impacted by recirculations between subassemblies and with the inter-wrapper.
The transient study is based on a CATHARE2 calculation completed by post-processing using TrioMC. Without RBH, CATHARE2 calculations are stopped deliberately when sodium boiling occurs for time calculation constraints due to time step decreases.
Calculations use the reactivity feedback defined for each group of simulated fuel subassemblies, i.e. axial and radial dilatation of hexagonal tubes, axial and radial dilatation of clad, fuel dilatation, Doppler, sodium density and diagrid dilatation. The groups were defined according to power to flow ratio.
The ULOF transient results from the loss of alimentation of primary pumps (their inertia of 4900 kg.m^2 leads to the pump halving time of 10 seconds); the secondary circuit is staying at normal operation.

Decay Heat Removal systems are not activated in this simulation.
CATHARE2 calculations are performed taking into account the prevention rods.
On Figure 3 we can see that the total reactivity in the core decreases where different contributors to this decrease are detailed in Figure 4. It can be seen that the sodium voiding, fuel and control rod expansion are resulting in important negative reactivity insertion.
On Figure 5, the reactor power evolution is shown. It can be seen that as expected, the total reactor power decreases as an answer on reactivity evolution. However, the power decrease is not able to compensate the inlet sodium flow decrease. Thus, without RBH, the transient result in sodium boiling (around 40s). The CATHARE2 calculation ends once the clad melting occurs [8].
In CATHARE2 input deck nodalization, the fuel subassemblies were redistributed within 31 groups. The Figure 6 shows the evolution of temperature at outlet of fuel subassemblies, corresponding to the minimum and maximum outlet temperature within the groups. For ASTRID severe accident prevention, RBH achieve absence of sodium boiling.

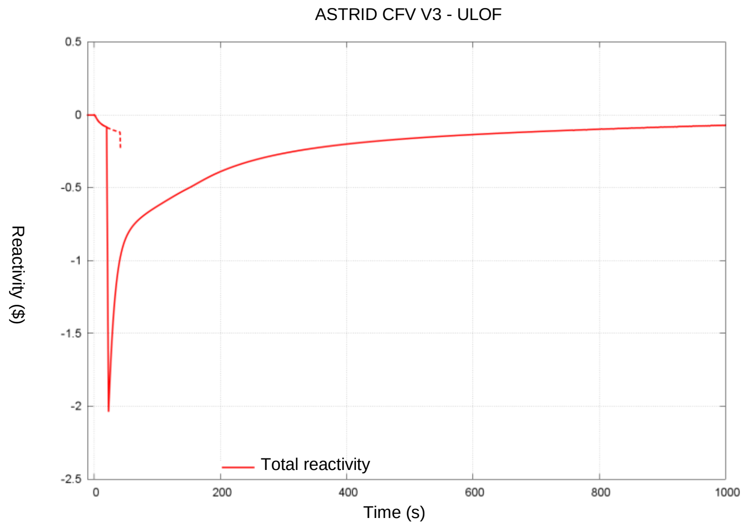

FIG. 3. Evolution of the total reactivity during ULOF (dashed line: without RBH).

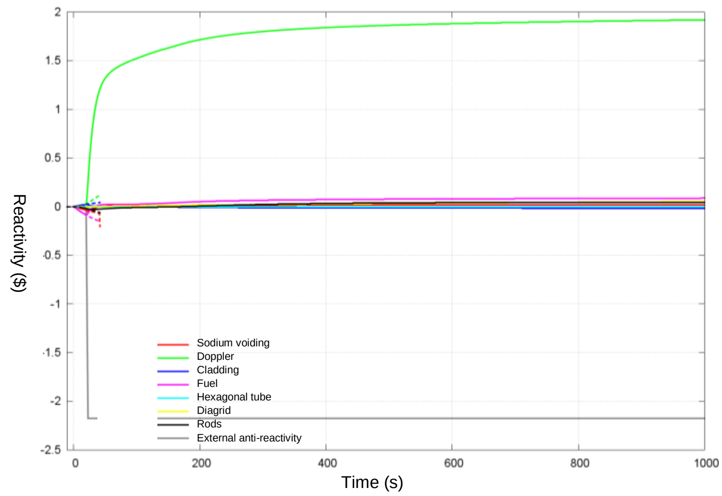

FIG. 4. Evolution of reactivity feedback during ULOF (dashed line: without RBH).

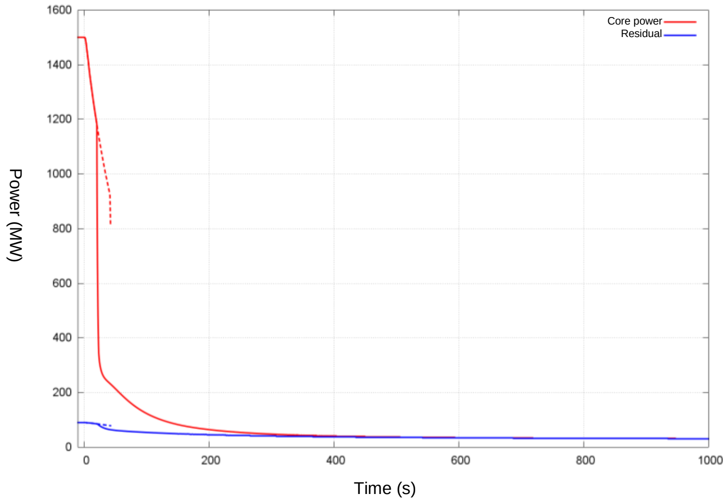

FIG. 5. Evolution of power during ULOF (dashed line: without RBH).

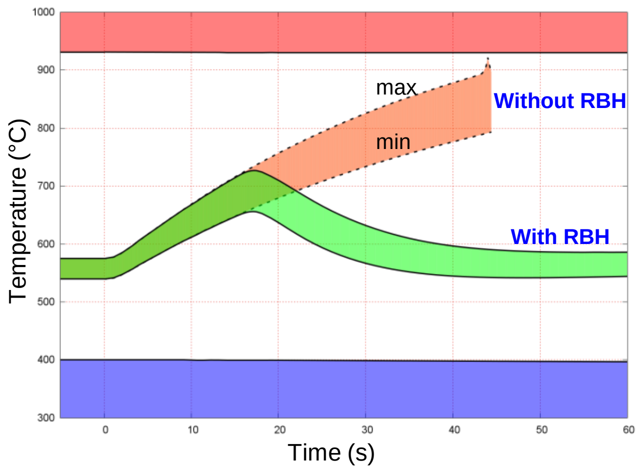

FIG. 6. Evolution of temperature during ULOF (dashed line: without RBH).

4.4.Unprotected Loss of Flow sequence due to total loss of supply station power

This sequence results in the coast down of the primary and secondary coolant pumps and steam generator dry out, without reactor scram.

The establishment of natural convection during this transient gives rise to several complex phenomena, which cannot be represented in CATHARE2. The thermal stratification in the plena in particular can only be represented with a CFD code. The coupling of CATHARE2/TrioCFD has been developed.

The natural convection appears in secondary loops during several hundred of seconds. Mass flow rate, greater than 5% after 500 seconds and around 1% after 1000 seconds, leads to the establishment of natural convection in the primary circuit. Temperatures (core outlet, local pin cladding) are lower (more than 100°C of margin) to criteria (Figure 7). RBH achieve absence of sodium boiling.

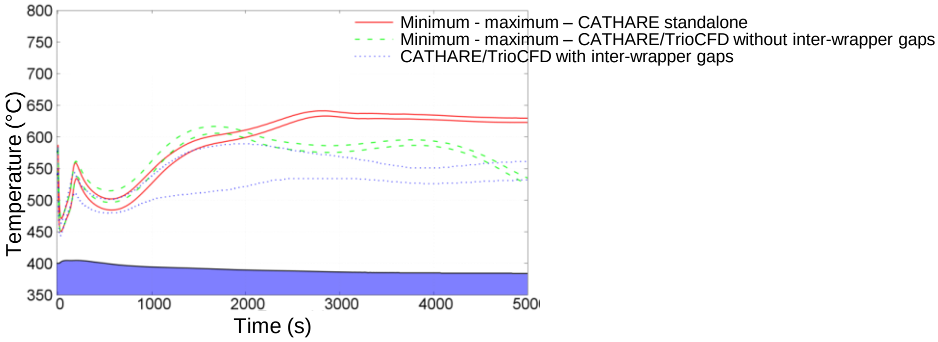

FIG. 7. Evolution of temperature during a total loss of supply station power.

4.5. Unprotected Loss of Heat Sink due to secondary pump coast down and steam generator dry out, without reactor scram

This accident leads to the increase of temperatures in all parts of the reactor. With the primary pumps continuing to operate during the transient, a standalone CATHARE2 calculation allows to determine the global dynamics (primary flow, core power, temperatures) of the primary circuit during this transient.
RBD actuation temperature is fixed at 650°C. RBD drop before the core inlet temperature reach 650°C, around 500 seconds after the begin of the transient allowing a negative reactivity of 7$ (Figure 9), the total reactor power decreases as an answer on reactivity evolution (Figure 8). Core outlet temperature reaches core inlet temperature (Figure 10). RBD achieve limitation of the reactor temperature.

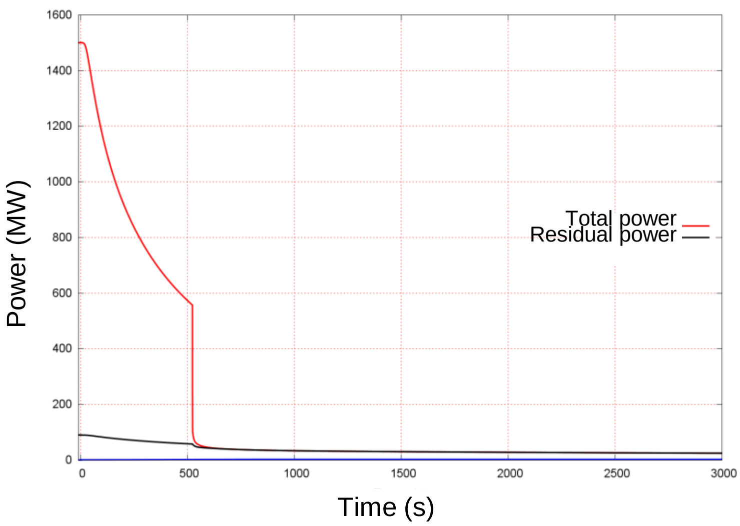

FIG. 8. Evolution of power during ULOHS with RBD.

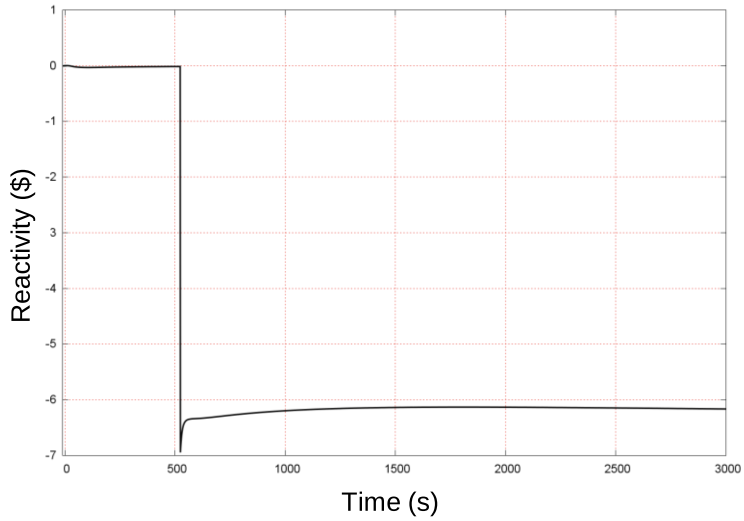

FIG. 9. Evolution of reactivity feedback during ULOHS with RBD.

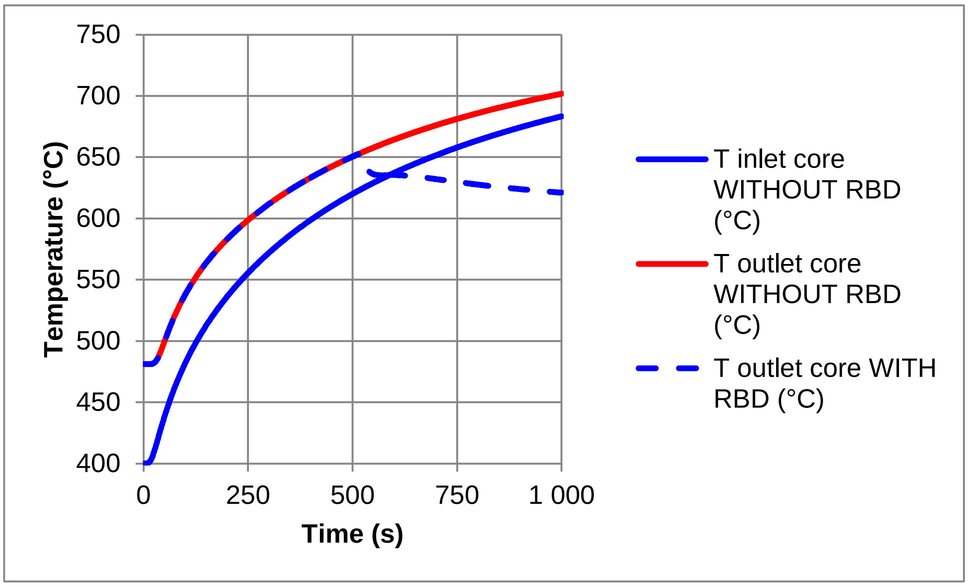

FIG. 10. Evolution of temperature during ULOHS.

5. Conclusions and perspectives

Following representative events of transients have been studied with CATHARE2 code completed by post-processing using TrioMC and CATHARE2/TrioCFD coupling:

- Two sequences of Unprotected Loss of Flow: the first type is characterized by a trip of all the primary pumps while the secondary coolant pumps remain operational to remove power; the second type due to a total loss of power supply results in the trip of both primary and secondary coolant pumps and steam generators drying out;
- A sequence of Unprotected Loss of Heat Sink: the secondary pumps are tripped and the steam generators dry out.

Two passive Complementary Safety Devices dedicated to core damage prevention passively shut down the reactor. The RBH type is a hydraulically suspended absorber rod subassembly dedicated to ULOF transients. The RBD type is based on the Curie point use of electromagnetic devices that hold shutdown systems to address ULOHS transients. RBH and

RBD, for ASTRID severe accident prevention, achieve absence of sodium boiling (ULOF) and limitation of the reactor temperature (ULOHS).
A methodology will be applied for transient studies to evaluate the uncertainty associated with a safety parameter.

6. Nomenclature

ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
CFD	Computational Fluid Dynamic
CFV	Cœur à Faible effet de Vide sodium (low void effect core)
PWR	Pressurized Water Reactors
RBC	control rods group
RBD	diversified rods group
RBH	concept of hydraulically suspended absorber rod subassembly
S/As	Sub-Assembly
SFR	Sodium-cooled Fast Reactor
ULOF	Unprotected Loss Of Flow
ULOHS	Unprotected Loss Of Heat Sink

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