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IDENTIFICATION AND STUDY OF INCIDENTAL AND ACCIDENTAL SCENARIOS FOR THE MOLTEN SALT FAST REACTOR

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ABSTRACT

Molten Salt Reactors (MSRs) are very promising in terms of load-following capabilities, flexibility of fuel composition and they present interesting safety features. Among them, the Molten Salt Fast Reactor (MSFR) was retained by the Generation IV International Forum and is currently studied in the frame of the SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) project. With its liquid circulating fuel and its fast neutron spectrum, the MSFR calls for a new safety approach and the development of adapted numerical tools. In this frame, the system code PANDAS (Precursors Advection and Neutronic Diffusion System Code) has been developed for the study at system level of MSFR incidental and accidental transients. In addition, a safety approach suitable for MSRs has been developed in the SAMOFAR project and is being applied to the MSFR. The identification of the Postulated Initiating Events (PIEs) and the evaluation of their occurrence frequency and consequences is a major step of the safety assessment methodology. In this paper, some of the PIEs identified for the MSFR are presented and the associated unprotected transients are evaluated thanks to the PANDAS system code. The method is illustrated with fuel over-cooling scenarios and loss of heat extraction scenarios. The study of these transients allowed evaluating the consequences of the PIEs and the need to implement additional provisions in the design.

Key Words: Molten Salt Fast Reactor (MSFR), transient study, system code, Postulated Initiating Events (PIEs)

1. INTRODUCTION

Thanks to their liquid fuel, Molten Salt Reactors (MSRs) are very promising in terms of load-following capabilities, flexibility of fuel composition, etc. and they present interesting safety features. The Molten Salt Fast Reactor (MSFR) concept, initially developed by the French CNRS, was selected by the Generation IV International Forum [1, 2]. This reactor, still at the conceptual design stage, is currently studied in the frame of the SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) European project. Its reference design, as defined at the beginning of the SAMOFAR project, is described in section 2. The MSFR, with its peculiar characteristics such as its liquid circulating fluid playing also the role of the coolant, calls for a new safety approach and the development of adapted numerical tools. Therefore, system codes have been developed and are used in the frame of the MSFR operation and safety studies [3]. Among them, the PANDAS system code is dedicated to the study, at system level, of MSFR incidental and accidental scenarios. It is presented in section 3. In addition, a safety approach suitable for MSRs has been developed in the SAMOFAR project [4] and is being applied to the MSFR. Section 4 gives a few results of the safety analysis that includes the identification of the MSFR incident and accident initiating events and the computation of associated transients with the PANDAS system code.

2. MSFR CONCEPT

The reference reactor is a 3 GW thermal power reactor with a fast neutron spectrum. This breeder reactor (with a breeding ratio of 1.1 [5]) is operated in the thorium fuel cycle. As presented in Fig. 1, it includes three closed circuits involved in power generation (the fuel circuit, the intermediate circuit and the power conversion circuit -or balance of plant-) and an open circuit acting as heat sink. The fuel circuit is defined as the circuit containing the molten fuel salt during power generation. The selected fuel salt is a binary fluoride salt with 77.5 mol% of lithium fluoride; the remaining 22.5 mol% are a mix of heavy nuclei fluorides including fissile and fertile matters. The fluids of the intermediate and conversion circuits have not been selected yet but several options are studied in the frame of the SAMOFAR project. The main options are summarized in Table I. The plant also includes several protection systems. For instance, an emergency draining system (EDS) is foreseen to drain the fuel in case of in-core emergency [6]. As shown in Fig. 1, this system is located under the core to allow a passive gravitational draining.

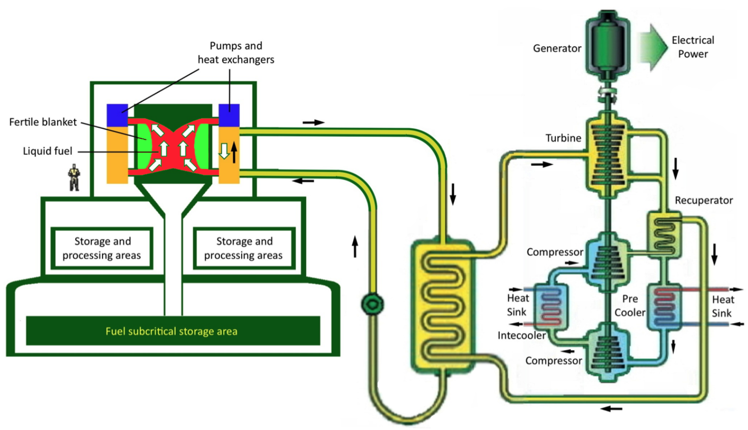

Figure 1. General representation of the MSFR system

In the fuel circuit, the fuel salt volume (18 m^3) is distributed half in the core and half in the 16 cooling sectors that are arranged circumferentially around the core vessel. The fuel flows upward in the core cavity, where it heats up due to the nuclear reactions, and downward in the sectors, where it is cooled down. The travel time of the fuel salt in the fuel circuit is about 3.9 seconds. The thermodynamic properties of the fuel salt are given in Table II; they are used for the safety studies presented in section 4. The mean fuel salt temperature is 700 °C, with a difference of about 100 °C between the cold leg and the hot leg. The structures of the fuel circuit are made of the hastelloy N Nickel based alloy [7]. They could be covered with a thermal protection layer, for instance in SiC, to improve their resistance to the high fuel temperature.

The design specificities of the MSFR impact the neutronics and the safety characteristics of the reactor. The MSFR has a negative thermal feedback coefficient, around -8 pcm/K [8], coming half from the density effect and half from the Doppler effect. It acts rapidly since the heat is produced directly in the coolant. Then, the fuel circulation drifts the delayed neutron precursors in low importance areas, reducing the effective fraction of delayed neutrons and contributing to the

Table I. MSFR main fluids composition

Fuel salt - initial composition	Option 1: LiF-ThF$_4$-^{233}UF$_4$ (77,5-19,9-2,6 mol %) Option 2: LiF-ThF$_4$-UF$_4$-(Pu-AM)F$_3$ (77,5-6,6-12,3-3,6 mol %)
Intermediate salt	Option 1 : fluoroborate Option 2: FLiNaK Option 3: LiF-ZrF$_4$ Option 4: FLiBe
Energy conversion circuit fluid	Option 1: helium Option 2: supercritical water Option 3: supercritical $C0_2$

Table II. Thermodynamic properties of LiF-ThF$_4$ (78%-22%) [9]

Property	Formula/Value	Validity range $[K]$
Density ρ $[g.cm^{-3}]$	$4.983 - 8.82 \cdot 10^{-4} \cdot T_{(K)}$	[893-1123]
Kinematic viscosity ν $[m^2.s^{-1}]$	$5.54 \cdot 10^{-8} \exp\left(3689/T_{(K)}\right)$	[898-1119]
Thermal conductivity λ $[W.m^{-1}.K^{-1}]$	$0.928 + 8.397 \cdot 10^{-5} \cdot T_{(K)}$	[891-1020]
Heat capacity C_p $[J.kg^{-1}.K^{-1}]$	$\left(-1.111 + 0.00278 \cdot T_{(K)}\right) \cdot 10^3$	[867-907]
Melting point $[K]$	858 K	/

reactor fast behavior. These characteristics provide an intrinsically stable behavior of the reactor to reactivity insertions that will be observed in some of the transients studied in this paper.

3. SIMULATION TOOL

The system code PANDAS (Precursors Advection and Neutonic Diffusion System Code) has been developed for the study, at system level, of MSFR incidental and accidental transients involving neutronics and thermohydraulics coupling. It seeks to provide a good estimation of global parameters of the plant during transients involving several circuits of the MSFR with a low computational cost. The code takes into account the specificities of the MSFR such as the drift of the delayed neutron precursors and of the residual power precursors in the fuel circuit due to the advection of the nuclear fuel. The partial differential equations (PDEs) solved in the code are given in the following paragraphs.

Neutron density: A standard 1-group neutronic diffusion equation is solved in the core:

$$\frac{\partial n}{\partial t} = D\Delta(v_n n) + (1-\beta_0)\cdot\nu\Sigma_f v_n n - \Sigma_a v_n n + \sum_f \lambda_f p_f \tag{1}$$

where n is the neutron density, β_0 is the total delayed neutron fraction, v_n is the average neutron velocity, D is the diffusion coefficient, ν is the number of neutrons produced per fission, and Σ_a and Σ_f are respectively the absorption and the fission cross-sections. The dependency of the cross-sections with the fuel temperature and density has been computed thanks to Monte-Carlo simulations with the Serpent 2 [10] code and the JEFF 3.1.1. data base.

Neutron precursor density: The precursor density is computed in the whole fuel circuit by solving the PDE:

$$\frac{\partial p_f}{\partial t} = \beta_{0,f}\nu\Sigma_f v_n n - \lambda_f p_f - \overrightarrow{v}\cdot\overrightarrow{\nabla} p_f \tag{2}$$

with p_f the delayed neutron precursor concentration, λ_f the decay constant and $\beta_{0,f}$ the delayed neutron fraction of family f. The term of production ($\beta_{0,f}\nu\Sigma_f v_n n$) appears only in the core while the term of loss ($\lambda_f p_f$) and the term of advection ($\overrightarrow{v}\cdot\overrightarrow{\nabla} p_f$), that allows to take into account the advection of the precursors by the fuel, appear in the whole fuel circuit. 8 families of precursors are considered. The decay constants and the delayed neutron fractions values have been computed thanks to Monte-Carlo simulations with the Serpent 2 code and the JEFF 3.1.1. data base [11].

Residual power precursor density: Similarly to the neutron precursors, the decay heat precursors are transported in the fuel circuit. They are managed in the same way as the delayed neutrons precursors, as already proposed for the MSFR in [12], by solving the PDE:

$$\frac{\partial r_k}{\partial t} = \beta_{r,k} P_n - \lambda_{r,k} r_k - \overrightarrow{v}\cdot\overrightarrow{\nabla} r_k \tag{3}$$

In Eq. (3), r_k, $\lambda_{r,k}$ and $\beta_{r,k}$ are the concentration, decay constant and fraction (or yield) of the decay heat precursors of family k. P_n is the neutronic power density computed as $P_n = Q\nu\Sigma_f n v_n$. The associated production term $\beta_{r,k}P_n$ only appears in the core. The decay constant and the fraction of the decay heat precursor have been computed thanks to the residual power data calculated in [13].

Temperature: The temperature is solved by introducing the energy balance equation as follows:

$$\rho C_p \frac{\partial T}{\partial t} = P_{vol} - \rho C_p \overrightarrow{v}\cdot\overrightarrow{\nabla} T \tag{4}$$

ρ and C_p are respectively the salt density and specific heat while P_{vol} is a heat source term including the neutronic power and the residual power density. Eq. (4) is used both for the fuel salt and for the intermediate salt but the source term concerns only the fuel. Finally, the mass flows of the fuel circuit and of the intermediate circuit are not computed but directly imposed as constant.

The previous PDEs are discretized with the finite difference method in 1D and are solved thanks to an explicit Euler scheme. A centered scheme is used for the Eq. (1) while an upwind scheme is

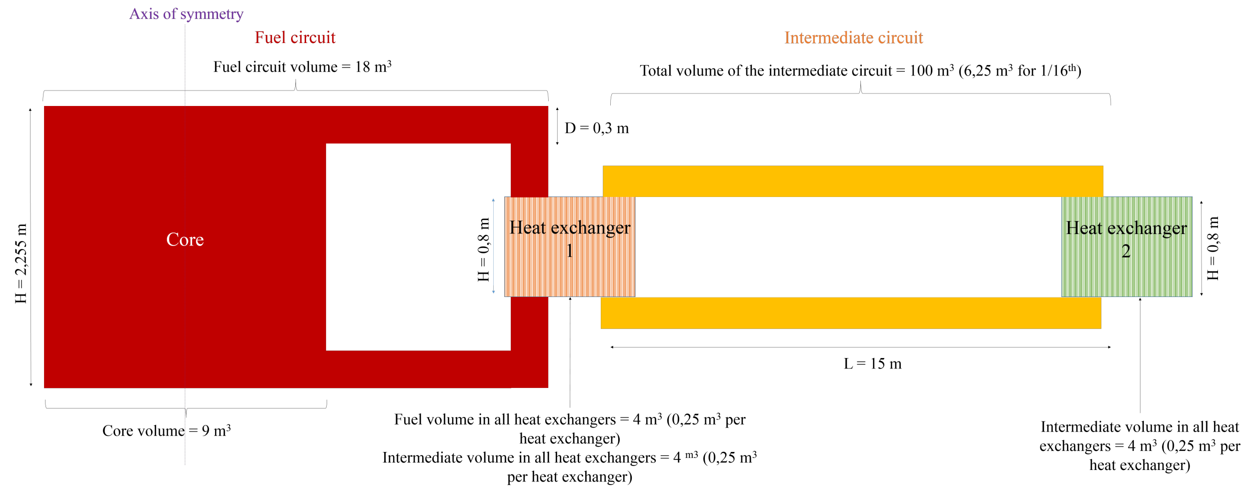

Figure 2. Graphical representation of the modelled geometry (not to scale)

considered for the Eqs. (2), (3) and (4). The geometry modelled is presented in Fig. 2 and includes the fuel circuit, the heat exchanger between the fuel and the intermediate circuit, the intermediate circuit and the heat exchanger between the intermediate and the energy conversion circuit. The total core volume is modelled while only one out of the 16 cooling sectors is modelled; the behavior of the fifteen other sectors is supposed identical to the one that is simulated. Cylindrical shapes are considered for the core and for all the pipes. Plate heat exchangers in hastelloy N are used for modelling both heat exchangers. The dimensions used are the results of optimization studies performed at CNRS with a static system code. Three modes of operation are available in the PANDAS code allowing to : 1) impose the power extracted in the first heat exchanger (the intermediate circuit is not modelled in this option), 2) impose the power extracted in the second heat exchanger or 3) impose the intermediate salt inlet temperature in the first heat exchanger. The PANDAS system code is under validation by benchmarking on the TFM-OpenFOAM code [8] for neutronic-thermohydraulic coupling in the fuel circuit.

4. RESULTS OF THE SAFETY STUDIES

In the frame of the SAMOFAR project, a safety approach suitable for Molten Salt Reactors has been developed. One of its main step is the elaboration of a list of Postulated Initiating Event (PIEs) to be studied in the later stages of the methodology. The elaboration of a list of PIEs has been undertaken for the MSFR and is described in paragraph 4.1. The consequences of some of these events have been quantified by studying the associated transients with the PANDAS system code as explained in paragraph 4.2.

4.1. Elaboration of a list of PIEs

The identification of the MSFR's initiating events (IEs) has been performed in normal operation conditions during power production, with a focus on the fuel circuit and the systems in direct interaction with it. To be as exhaustive as possible, the IEs of the MSFR have been identified with several risk analysis methods including top-down and bottom-up methods [14]. The IEs have been classified in families, depending on the phenomena involved, that are the following:

- Positive reactivity insertion
- Negative reactivity insertion
- Loss of fuel flow
- Increase of heat extraction / over-cooling
- Decrease of heat extraction / loss of heat sink
- Loss of fuel circuit tightness
- Loss of fuel composition/chemistry control
- Fuel circuit structures over-heating
- Loss of cooling of other systems containing radioactive materials
- Loss of containment of radioactive materials in other systems
- Mechanical degradation of the fuel circuit
- Loss of pressure control in fuel circuit
- Conversion circuit leak
- Loss of electric power supply

Then, inside each family, the IEs have been separated into categories, depending on the frequency of occurrence and the severity of the event. Considering the preliminary design stage, only three macro categories have been selected (incident, accident or limiting event) as described in Fig. 3. The classification has been performed on the basis of expert judgment and thanks to the existing operating feedback on MSRs available with the ORNL reports (such as [15]). Among all the IEs identified, only the most representative events of each family and category have been selected as PIEs to be studied in the next steps of the safety analysis. In this way, it is possible to focus the safety studies on the most relevant cases in terms of consequences and occurrence frequencies. As an example, Tables III and IV show an extract of the PIEs list for the families "increase of heat extraction" (or over-cooling) and "loss of heat extraction" respectively. The consequences of some of these events are studied in the next sub-section.

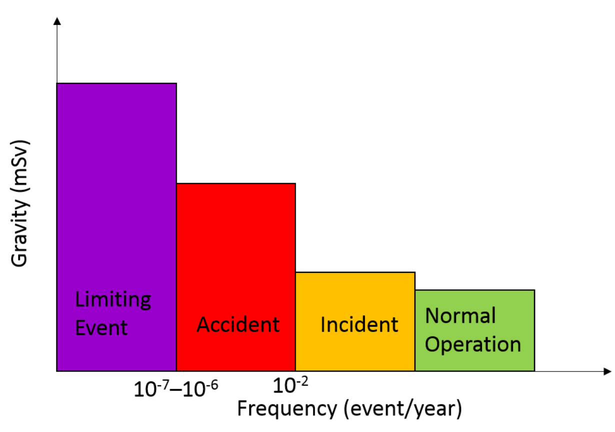

Figure 3. Farmer diagram proposed for the classification of the MSFR's IEs

Table III. List of the PIEs of the family "Increase of heat extraction"

Category	PIEs
Incidents	Over-working of one or several (up to all) fuel circuit pumps Overworking of one or several (up to all) intermediate circuit pumps Over-cooling at conversion circuit level
Accidents	Over-cooling at low power

Table IV. List of the PIEs of the family "loss of heat extraction"

Category	PIEs
Incidents	Loss of heat extraction at conversion circuit level Unwanted closure of a valve/gate in the intermediate circuit Failure/shut down of one or several (up to all) intermediate pumps Loss of main heat sink
Accidents	Inadvertent opening of a draining valve of the intermediate circuit Leakage of the intermediate salt (outside core vessel) Rupture/blockage of one or several (up to all) intermediate circuit pump Obstruction/blockage of the intermediate circuit
Limiting events	Complete loss of the intermediate salt

4.2. Study of transients scenarios

To implement appropriate provisions in the design, the study of unprotected transient scenarios is crucial in order to evaluate the consequences of the PIEs. In this scope, various transient studies have been performed for the MSFR; some of them are available in [8, 16]. The studies presented below have been performed with the PANDAS system code and includes over-cooling transients and loss of heat sink transients.

4.2.1. Over-cooling transient

Because of the negative thermal feedback coefficients of the reactor, a cooling of the fuel involves a positive reactivity insertion. This phenomenon is used for the driving of the MSFR, but can be problematic in case of an inadvertent over-cooling. Two over-cooling transients are presented in Fig. 4. They are obtained by reducing the temperature of the intermediate salt entering the fuel-intermediate heat exchanger. To simulate bounding cases, the temperature is instantaneously reduced, at 0,01 s, to 657 K, which is the freezing point of the fluoroborate, here considered as the intermediate salt.

The first case studied (Fig. 4 on the left) is an over-cooling at nominal power. It can be associated to the event "over-cooling at conversion circuit level" listed as incident in the previous sub-section.

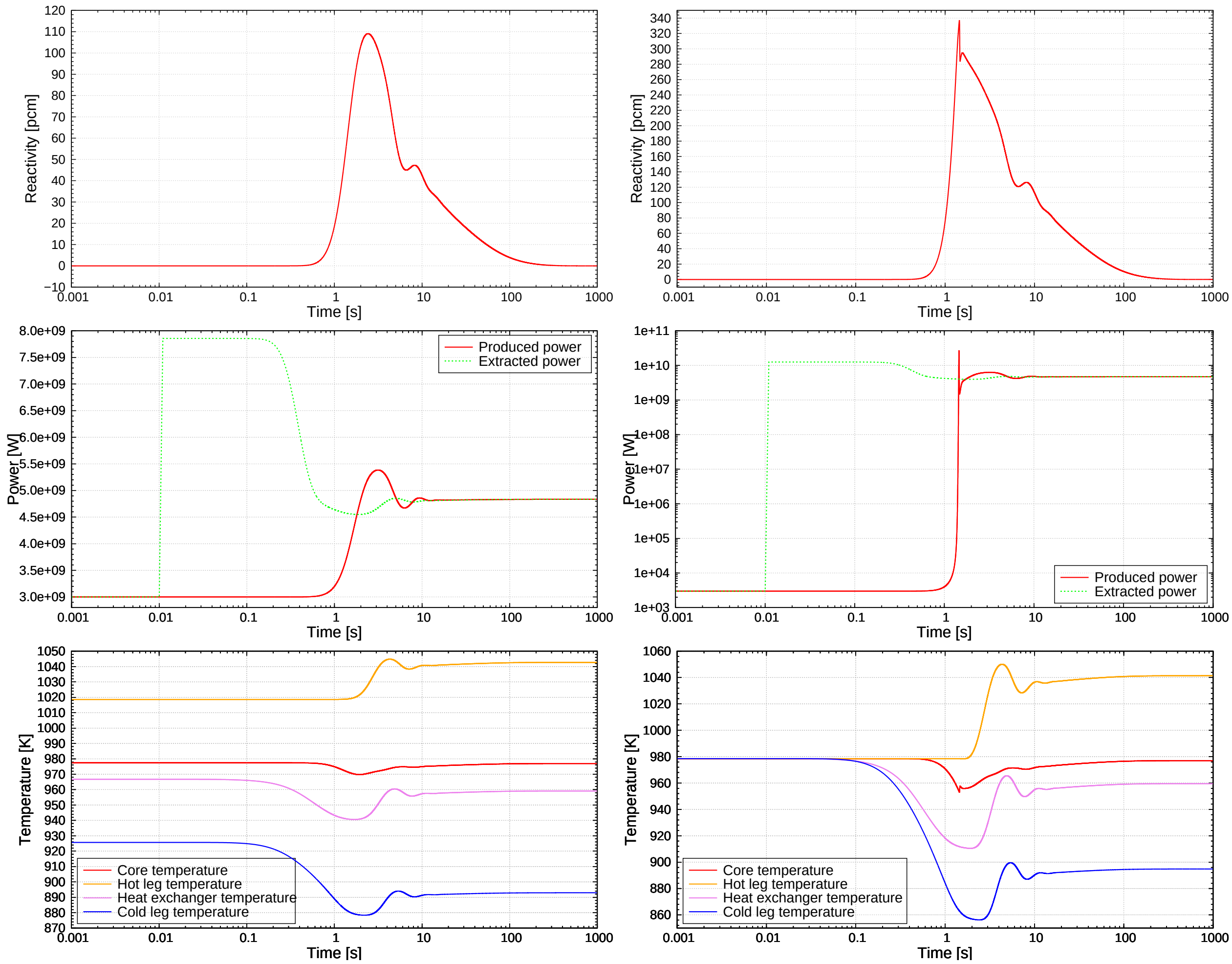

Figure 4. Evolution of the reactivity, power and temperature during an over-cooling transient at nominal power (case 1 -left) and at low power (case 2 - right)

The initial inlet temperature of the intermediate salt in the heat exchanger between the fuel and the intermediate circuits is 782 K and corresponds to a steady state at 3 GW. The reduction of the intermediate salt temperature implies an increase of the power extracted at the heat exchangers and a decrease of the fuel temperature leaving the heat exchangers. When the cold fuel arrives in the core, it causes a small reactivity insertion and thus a slight power increase (with a maximum at 5.4 GW). Thanks to the feedback reactions the reactor stabilizes at a new equilibrium with a power of 4.8 GW. The oscillations that can be observed on the curves have a period of 3.9 seconds and are due to the fuel circulation.

The phenomena involving reactivity insertions, such as over-coolings, are more constraining when the reactor is at low power because the coupling between neutronics and thermohydraulics is weaker. Thus, an over-cooling at low power has been studied for the second case (Fig. 4 on the right). It can be associated to the event "over-cooling at low power" listed as accident in the previous sub-section. The initial inlet temperature of the intermediate salt in the heat exchanger between the fuel and the intermediate circuits is 978 K and corresponds to a steady state at 3 kW. In this case, the reactivity insertion is high enough to reach prompt criticality. It causes a power peak around 27 GW. Thanks to the feedback reactions, the reactor finally stabilizes to the same equilibrium as in case 1.

The risks identified for over-cooling transients are mainly a too high temperature in the hot leg or a too low temperature in the cold leg with a possibility of fuel solidification in the heat exchangers. The fuel melting point is 858 K. In the simulated transients, the temperature in the hot leg stays below 1045 K for case 1 and do not exceed 1050 K for case 2, which is acceptable for the materials of the fuel circuit. In case 1, the fuel temperature in the cold leg stays above the melting point, with a margin of about 20 K, but in case 2, it reaches 856 K. These computations were useful to highlight the need to develop systems or procedures to manage the solidified salt in the fuel circuit. More precise computations should be performed to evaluate the local temperature of the fuel and the solidification phenomena on the walls of the heat exchangers.

4.2.2. Loss of heat sink transients

Two transients of loss of heat extraction are presented in Fig. 5. In both cases, the extracted power is reduced from 100% of nominal power to 0% linearly in 1s. They are bounding cases because of the speed of the transient and of the adiabatic final state.

The first case studied is a loss of heat extraction at the level of the heat exchanger between the fuel and the intermediate circuits (Fig. 5 on the left). It is associated to the limiting event "complete loss of the intermediate salt". The loss of heat extraction causes a temperature increase. Due to the negative thermal feedback reaction, it leads to the shutdown of the fission chain reaction and of the neutronic power. The temperature increases rapidly due to the fissions generated by delayed neutrons (at the beginning of the simulation) and the residual heat (during the entire simulation). It exceeds 1200°C 17 minutes after the beginning of the transient. According to the provisions implemented in the design, the heating of the fuel triggers the emergency draining of the fuel to the emergency draining tank where it is passively cooled down. To protect the fuel circuit structures, it is preferable to ensure that the draining of the fuel is fast enough (less than 17 minutes with the adiabatic hypothesis).

The second case is a loss of heat extraction at the level of the heat exchanger between the intermediate and the energy circuits (Fig. 5 on the right). It can be associated to the incident "loss of heat extraction at conversion circuit level". The intermediate circuit, with a salt volume of about 100 m^3, provides a greater thermal inertia and the mean temperature of the fuel reaches 1200°C after more than two hours. Thus, a longer grace period is available before the emergency draining of the fuel. The implementation in the design of an emergency heat removal system for the intermediate circuit would be favorable in this situation, as it would enable to keep and cool the fuel in the core.

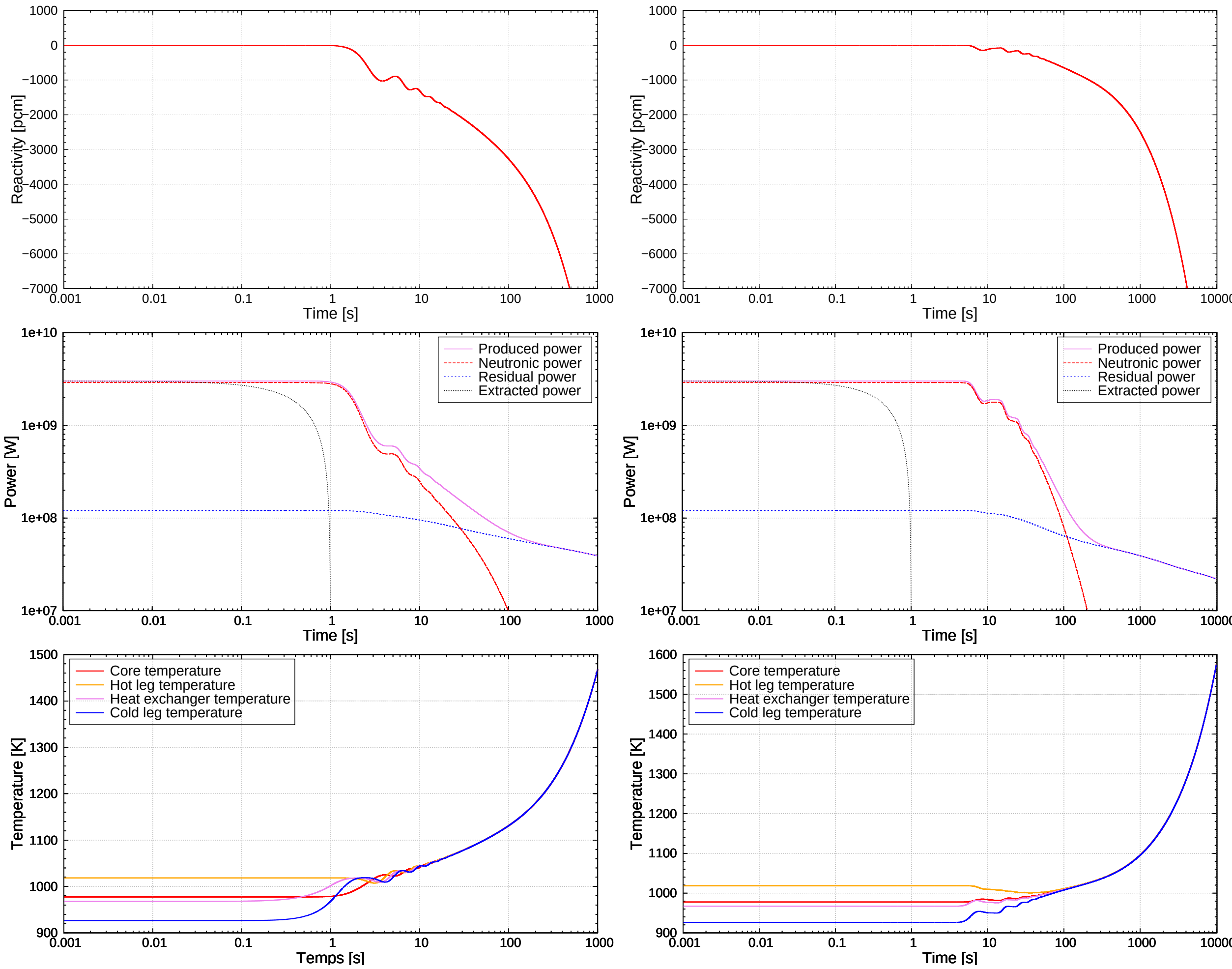

Figure 5. Evolution of the reactivity, power and temperature during a decrease of heat extraction from 100% of nominal power to 0% at fuel-intermediate heat exchanger (case 1 - left) or at intermediate-conversion heat exchanger (case 2 - right)

5. CONCLUSIONS

An identification and a classification of the PIEs of the MSFR have been proposed. The identification of events has been performed for the fuel circuit and the systems in interaction with the fuel circuit during power production. This identification should now be extended to the whole plant and for all operation modes including maintenance, start up and shut down. In addition, the classification in category of frequencies/consequences should be refined with the evolution of the design of the plant. Some of the identified events have been used as an input for deterministic safety studies. In particular, unprotected transients have been computed with the new PANDAS system code. This system code is based on simplified 1D models for the fuel and for the intermediate circuit. It solves the neutron diffusion equation and takes into account the specificities of the MSFR such as the fuel movement that impacts the transport of the temperature, of the neutron precursors and of the residual power precursors. The results of the unprotected transient computations confirm a satisfactory behavior of the reactor. These studies have proved to be helpful to evaluate the consequences of the identified PIEs and estimate the grace periods and the eventual needs of additional provisions in the design. In particular, systems or procedures should be developed to manage the solidified fuel salt and the importance of the systems foreseen to evacuate the fuel residual heat, such as the emergency draining system for the fuel and the emergency cooling system for the intermediate salt, was highlighted. In the future, more precise computations will be performed to evaluate the local temperature of the fuel in sensible areas, such as the heat exchangers, during the same incidental and accidental transients.

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