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Preliminary safety calculations to improve the design of Molten Salt Fast Reactor

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ABSTRACT

Molten salt reactors are liquid fuel reactors so that they are flexible in operation but very different in the safety approach from solid fuel reactors. This study bears on the specific concept named Molten Salt Fast Reactor (MSFR). Since this new nuclear technology is in development, safety is an essential point to be considered all along the R&D studies. This paper presents the first step of the safety approach: the systematic description of the MSFR, limited here to the main systems surrounding the core. This systematic description is the basis on which we will be able to devise accidental scenarios. Thanks to the negative reactivity feedback coefficient, most accidental scenarios lead to reactor shut down. Because of the decay heat generated in the fuel salt, it must be cooled. After the description of the tools developed to calculate the residual heat, the different contributions are discussed in this study. The decay heat of fission products in the MSFR is evaluated to be low (3% of nominal power), mainly due to the reprocessing that transfers the fission products to the gas reprocessing unit. As a result, the contribution of the actinides is significant (0.5% of nominal power). The unprotected loss of heat sink transients are studied in this paper. It appears that slow transients are favorable (> 1min) to minimize the temperature increase of the fuel salt. This work will be the basis of further safety studies as well as an essential parameter for the design of the draining system.

Key Words: MSFR, molten salt reactor, nuclear safety, decay heat, delayed neutrons

1. INTRODUCTION

In the frame of the development of future energy resources and reducing nuclear waste, the specific molten salt reactor concept offers a large capability of operation. Previous studies led us to define the concept called Molten Salt Fast Reactor (MSFR), that is now one of the six concepts selected by the Generation IV International Forum (GIF), [3], for further study. The MSFR is to be operated in the Th/^{233}U fuel cycle with fluoride salts. Since ^{233}U does not exist in nature, the reactor can be started with the Plutonium and Minor Actinides produced in today's reactors as fissile material. Nevertheless, the reference configuration discussed here, is the reactor started directly with ^{233}U. The inventory converges to the same composition at equilibrium regardless of the initial fissile material, [2]. The sensitivity to the isotopic composition of the salt of the parameters presented in this paper should be evaluated in further studies. In the development of the MSFR design, we consider safety as an essential issue. For this reason we present here a systematic description of the systems in contact with the fuel salt. Indeed, in the case of a reactor shut down, the fuel salt continues to produce heat. Even when it is drained in the draining storage, all the components of the plant described here are in contact with heating fuel salt that can cause damage. In order to assess the design of each component, we need to study the residual heat produced by the fuel salt. The study of this safety parameter, described in section 3, is the main purpose of this paper. In section 4, we discuss additional heat issues due to transients.

2. MSFR REFERENCE CONFIGURATION

Since 1997, the study of the concept of a molten salt reactor was undertaken by CNRS, contributing to the development of the innovative GEN-IV reactors. These studies led to a new concept called the Molten Salt Fast Reactor or MSFR. Skipping over the historical development we will present here the reference MSFR with some important justifications.
As opposed to other molten salt reactors previously studied, the specificity of the MSFR is the removal of any solid moderator, (usually graphite), in the core. This choice is motivated by the study of parameters such as feedback coefficient, breeding ratio, graphite lifespan and ^{233}U initial inventory, as described in [5]. The result is a fast neutron spectrum, presented in [2]. We then proceeded to further develop the MSFR concept according to reactor safety guidelines, seeking both a high safety level and a high performance level. The problem is that precise safety guidelines are not technically neutral and those that are available are not adapted to a liquid fuel reactor. For this reason we are working on identifying the main accidents that can occur for this type of reactor, aware that only experience can finally define them. Those accidents will be the foundation for the Design Basis Accidents (DBA). Their control will be implemented within the design basis. To identify the accidents, we are using a risk analysis approach, an approach that is widely used in industry. The difficulty is to apply risk analysis to a concept that is still in development.
The first step of this approach is to describe the MSFR from a systematic point of view, i.e. to divide the plant into systems that interact with each other. It is important to identify the connections between the systems as only then will we be able to develop accidental scenarios.

2.1. Systematic Description Of The MSFR Concept

We present here only the systems which are in contact with the fuel salt during normal operation or during a DBA. During normal operation, the fuel salt circulates in the core and 16 external modules, so-called fuel loops. Each of them contains a pump, a heat exchanger and a bubbling system. We will describe each part of the plant and finally describe the connection between these systems. The systems are shown in figure 1, done by A3I *, our collaborator in design development.

- **Core**: As mentioned before, there is no solid moderator so that no structural elements are located in the core of the reactor. It contains only the fuel salt. The core is defined as the location where most fissions take place, including the injection zone at the bottom and the extraction zone at the top of the core. The reference concept, designed for a nominal power of 3 GW_{th}, corresponds to a heating in the core of ΔT= 150 K between the bottom and the top of the core. The core geometry was defined in the course of parametric studies seeking low neutron losses, low reflector irradiation and minimal fissile inventory, while maintaining a fuel salt volume in the heat exchangers large enough to ensure that salt cooling by ΔT= -150 K is feasible. The resulting core is a cylinder whose height is equivalent to its diameter, such that 1/2 of the entire salt volume is inside the core, the rest being located in the external fuel loops.

- **Fuel Salt**: The initial fuel salt is made of LiF (77.5%mol), with fissile and fertile heavy nuclei in it: ^{233}U (2.5%mol) and ^{232}Th (20%mol). The proportion of heavy nuclei corresponds to

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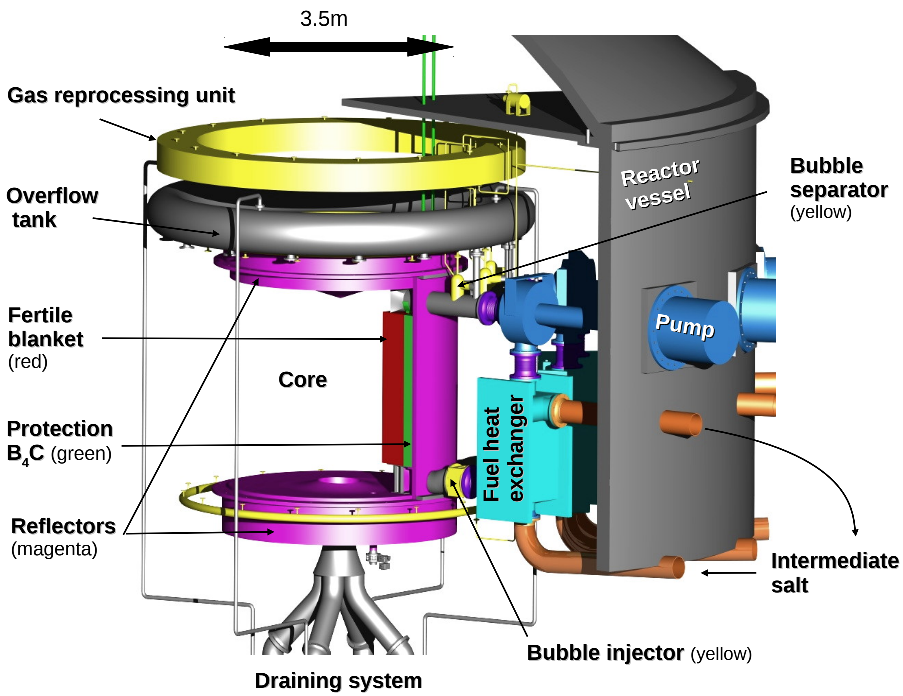

Figure 1. View of the MSFR systems in contact with the fuel salt, done by A3I

the eutectic point. The salt has the following characteristics: calorific capacity c_p = 1045 $J \cdot kg^{-1}K^{-1}$, thermal conductivity $\lambda = 0.510\ W \cdot m^{-1}K^{-1}$, density d= 4.1 and dilation coefficient of 0.8 $10^{-3}K^{-1}$ [2]. With the geometry described above, the total salt volume is $18m^3$. With a fusion temperature of 565°C, the mean operating temperature has been chosen at 700°C. The fission products created during operation are elements that are either soluble or insoluble in the salt. To maintain the physico-chemical and neutronic characteristics of the salt, it is necessary to clean the salt, i.e., to extract the fission products. Maintaining the physico-chemical properties of the salt because of safety considerations is clearly one of the main reasons for the reprocessing, along with improving the neutronic characteristics [6]. The fission products are extracted from the salt during reactor operation: the **bubbling system** extracts insoluble elements and the **pyrochemical reprocessing unit** extracts soluble elements. The behavior of the fuel salt is a crucial safety issue. A major safety parameter is the reactivity feedback coefficient, that is evaluated for the MSFR to be strongly negative [2]:

$$\left(\frac{dk}{dT}\right)_{Total} = \left(\frac{dk}{dT}\right)_{Doppler} + \left(\frac{dk}{dT}\right)_{Dilation} = -5 \pm 1pcm/K \qquad (1)$$

The feedback coefficient characterizes the behavior of the fuel salt in the event of neutronic transients. The salt's thermohydraulic behavior is closely coupled to its neutronic behavior, because the salt's circulating time (~4s) and the lifetime of the precursors of delayed neutrons (~10s) are of the same order of magnitude. The temperature of the salt depends strongly on the operation of the **pumps** and the cooling in the **heat exchangers**, so the transients should be calculated with a coupled neutronic-thermohydraulic tool.

- **Fertile Blanket**: The fertile blanket is necessary only to improve the breeding capabilities of the reactor. It contains the same type of salt but with 22.5%mol of Th and without any initial fissile material. Submitted to the neutron flux, the thorium produces ^{233}U, a fraction of which fissions so that some fission products are produced in the blanket. The small fission rate (∼7MW) and the captures on thorium in the blanket heat up the fertile salt (∼20MW). After some preliminary studies by INOPRO [†], partner of EVOL, a Euratom-Rosatom collaborative project, we concluded that natural convection cannot take place in the blanket and the heat produced cannot be evacuated by the fuel salt through the blanket walls. A fertile salt external cooling system will have to be implemented, so that the bubbling system and the salt extraction system designed for the fuel salt batch processing can be easily added for the blanket salt. The fertile blanket system as defined here includes the walls that enclose it. The wall between the fuel salt and the fertile salt is under stress. The possibility of its failure has to be taken into account.

- **Pyrochemical Reprocessing Unit**: Among the soluble fission products the lanthanides will be removed at a daily rate by pyrochemical reprocessing. This reprocessing unit is to be located on-site but outside the reactor vessel. The fuel salt reprocessing flow rate is very small (40l/day) and it will be done by batch. Since there is no direct connection between the pyrochemical reprocessing unit and the fuel salt system, we will not take this system into account in the rest of this paper. The fuel salt batches can be sampled on the external system of the fuel loop, near the **bubble separator**.

- **Upper and Lower Reflectors**: The lower and upper walls of the vessel are neutronic reflectors. The reflectors and other walls in contact with the fuel salt are assumed to be a NiCrW hastelloy, reference composition from [7]. The upper reflector is submitted to mechanical, thermal (the fuel salt's mean temperature in the upper plenum is 775°C with possible spatial and time dependent fluctuations) and radiation stresses. The high temperature seems to be the biggest challenge for the proposed composition so that the surface of the upper reflector should be provided with a **thermal protection**. The lower reflector is not under thermal stress. Its specificity is that it is coupled to the **draining system**.

- **Neutronic Protection**: To protect the heat exchangers from neutrons escaping the core, a neutronic protection made of B_4C is proposed. In fact, the residual neutron flux should be comparable to that due to the delayed neutrons emitted inside the exchangers, and the latter cannot be avoided.

- **Pump** (x16): Since natural convection is not sufficient to evacuate the nominal power, we need pumps to drive the fuel salt mass at a flow rate $Q \approx \frac{P_{nominal}}{c_p \cdot \Delta T} = 19t/s$. The pumps are a source of vibration. Since the circulation of the fuel salt is strongly coupled to the reactivity of the reactor, the impact of any deviation from the nominal behavior of the pump should be studied. For safety reasons the pumps should be provided with an inertia system.

- **Heat Exchanger** (x16): The heat exchangers are necessary not only for the energy transfer in view of electric power generation, but they are also an important factor in the control of the reactor. Indeed, any variation in the extracted power induces a transient. The power in the core tends to follow the extracted power as will be shown in section 4. A heat exchanger can influence the extracted power in two manners: first, the flow of the fuel salt in the heat

[†] www.inopro.com/

exchanger can be reduced due to some clogging; second, the temperature and the flow rate of the intermediate salt circuit may vary. Since in this paper we do not consider the systems beyond the heat exchangers, any initiator within the intermediate circuit or beyond liable to induce abnormal operation in the fuel salt circuit is handled as a heat exchanger failure. In further studies, we will associate different curves of decreasing power extraction to different types of initiators and study the ensuing transients.

- **Pipes**: The fuel salt flows between the fuel loop systems in pipes. The pipes in the upper part, that contain the fuel salt upstream from the heat exchangers, are under high thermal stress. To protect them from the heat, a thermal shield is implemented inside the pipes and they have an external **thermal protection**.

- **Bubble Injection** (x16): Some of the fission products created in the core are gaseous so that some bubbling occurs naturally in the core. This "natural bubbling" can also extract insoluble fission products from the salt. To increase the velocity of the bubbling extraction, gases are injected in the core at a flow rate of 7 liters/s.

- **Bubble Separator** (x16): The bubble separator is studied at LPSC in the frame of the FFFER project, [4], where the design of the separator is being developed. The bubble separator is connected to the gas reprocessing unit, that will not be detailed here. But since we are interested in the connection between the systems, it is important to note that the reprocessed gases are directed to the **bubble injection**.

- **Overflow Tank**:To compensate any temperature variations leading to volume variations of the fuel salt, an overflow tank is to be installed above the reactor core. The overflow tank, is tore shaped; it is connected to the fuel loop on the **upper pipes**. There is also a connection between the draining storage system and the overflow tank, in order to let gases from the **draining storage system** flow in the overflow tank in the event of fuel salt drainage.

- **Draining System**: As will be shown below, the draining system is a very important safety and operational system. In fact, for a planned shut down, the fuel salt will be evacuated by gravity under the reactor to be cooled passively. Any accidental deviation from nominal conditions leads to the drainage of the fuel salt into the draining storage system, whose design will ensure passive cooling. In order to have redundant safety systems, several drainage procedures (active and passive) will be defined. In all these instances, the salt will have to flow through the lower reflector. Draining pipes will also be installed between the bottom of the reactor vessel and the draining storage system. This will be useful in the event of a fuel loop rupture.

- **Reactor Vessel**: The core and the fuel loop systems described above are placed inside the reactor vessel which is otherwise filled with an inert gas maintained at T=400°C, as already implemented in the MSRE experimental reactor [8]. If a small amount of salt were to leak, it would immediately solidify, its fusion temperature being larger than 400°C.

Thermal protections will be installed on all the systems that are in contact with the fuel salt upstream from cooling in the heat exchangers. As this thermal protection is not yet well defined, it is not shown in figure 1. It will most probably include cooling by the intermediate salt.
A systematic description of the MSFR concept allows us to build accidental scenarios. To study

those scenarios, we need to know the reactor's behavior in special conditions. Some of the parameters that define the reactor's behavior are already fixed and presented here, others are being evaluated, in particular to take into account the current safety studies. A number of scenarios, such as pump or heat exchanger failures induce a reactor shut down and subsequent fuel salt drainage with or without external action. The reason for this is as follows. Both failures lead to the loss of the cooling, and so to a reactivity decrease thanks to the excellent safety coefficients of the MSFR [1]. After the reactor shut down, however, the fuel salt is still being heated by decay heat. For this reason the fuel salt is evacuated in the draining storage system, where it will be continuously cooled by means of a passive system. Since it is a very important safety issue, as we could unfortunately observe during the accident in Fukushima, the second part of this paper discusses the residual heat. This study will also allow us to design the draining system, both to determine the time within which the drainage must be completed, and to conceive the passive cooling system for the draining storage system. Another interesting issue is the failure of a single pump. The fuel salt located in the failing fuel loop would be stagnant and thus not cooled. The decay heat would heat the fuel salt, causing damage to the systems described above. For all these situations, we need to quantify the residual heat of the fuel salt. This calculation is described in the next section.

3. DECAY HEAT CALCULATIONS

The residual heat produced in the reactor after shut down is due to the presence of different radioactive materials in the core. Three main contributions can be identified:

⇒ **decay of the fission products**: the fission products are unstable and decay by emitting mainly γ rays or β particles;

⇒ **decay of the actinides**: the actinides created in the core through neutron captures are also unstable and decay by emitting different particles (i.e. γ, β, α);

⇒ **fissions due to the delayed neutrons**: some of the fission products emit neutrons as they decay and these neutrons may induce fissions even after the chain reaction is stopped.

In solid-fuel reactors such as PWRs, the contribution of the materials activated in the core has to be taken into account. In the MSFR configuration described in section 2, there is no solid material in the core, so this contribution can be ignored. At this point, it is interesting to discuss the role of the three main contributions to the residual heat in the core on the well-known example of the PWR, [9]. They have different time scales. The contribution of the delayed neutrons through the fissions they induce depends on the dynamic of the reactor shut down, or more precisely on the amount of antireactivity that is inserted. If the antireactivity is 4000 pcm or so, this source of residual heat is dominant at short time scales and disappears after 100 s. The actinides generally have a long lifetime, so a relatively weak activity. Their contribution is important only 30 years later. From the point of view of reactor safety we are interested in the range of time from some ten seconds up to some years. In that period of time, the main contribution is due to the decay of fission products. In the following we will present the developed tool used to calculate the decay heat in the MSFR.

3.1. Decay Heat Calculation Tool

Our numerical simulations of the reactor rely on the coupling of the MCNP neutron transport [10] with a home-made materials evolution code REM, [2]. This simulation tool takes into account the fissions, other nuclear reactions, the decay of isotopes, and the coupling with the reprocessing system. It gives the materials isotopic composition at any time during reactor operation. The newly developed tool for the decay heat calculations, called ECI (Isotopic Composition Evolution), takes an isotopic composition and, after constructing the decay chains, evaluates the energy generated by those decays. In this way, we can calculate the contributions of actinides and fission products to the residual heat. The nuclear reactions that can take place after reactor shut down are not taken into account by this tool. The fissions due to the delayed neutrons are calculated with another tool described in section 4.

3.2. Validation Of The ECI Tool

To validate the ECI tool, we calculate the decay heat of an elementary fission and compare it to reference data. We have chosen two nuclei, ^{233}U, because of its importance in the Thorium cycle and ^{235}U, because of the evaluations and the experiments that are available from several sources. To validate a calculation tool relatively to experiment, a deeper analysis is needed because the measured decay heat data is constrained by the experiment. For this reason, we present here a comparison of our results with other calculations, using the same fission yield database JEFF-3.1.1 for thermal neutrons. We evaluate the deviation of our results relative to DECROI calculations, a tool that has already been validated [11]. As shown in figure 2, our calculations with ECI for both nuclei fit the reference curves with an accuracy better than 2.5 %. Some of our points are underestimated while others are overestimated, but the integral of the curves fits within 1 %. Considering the uncertainties of the yield data (up to 30% for some nuclei) and of the decay energy data (sometimes there is no evaluation of the uncertainties), we can consider that our tool is validated.

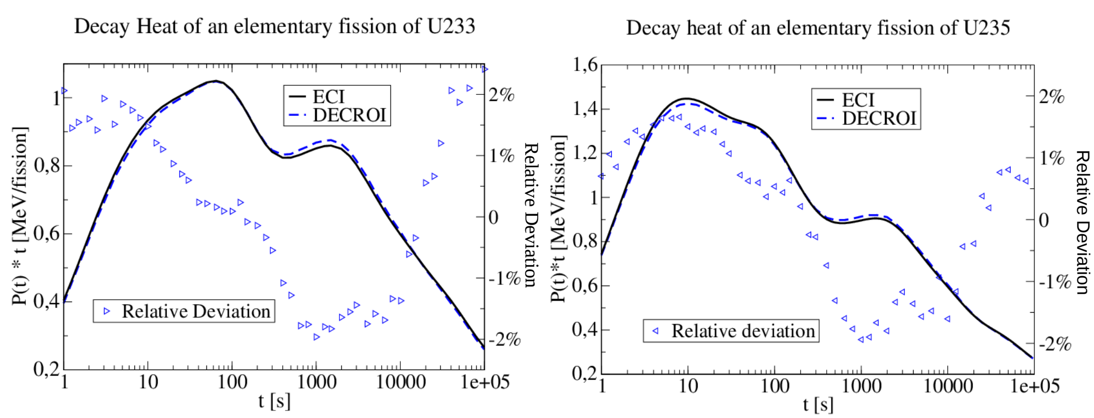

Figure 2. Comparison of ECI to DECROI calculations on the decay heat of an elementary fission of ^{233}U (left) and ^{235}U (right)

3.2.1. Comparison of the ^{235}U and ^{233}U elementary fissions

Now, we want to discuss the difference between an elementary ^{233}U and ^{235}U fission in the light of reactor safety. In the short term, 1s after the fission, the decay heat of the fission products of ^{235}U (0.74 MeV/s) is almost double that for ^{233}U (0.40 MeV/s) as shown in figure 2. A comparison of the decay heat of an elementary ^{233}U fission in a fast (0.46 MeV/s) and a thermal neutron spectrum (0.40 MeV/s), yields a difference of only 15%. We can thus conclude that the impact of the neutron spectrum on the residual heat is much smaller than that of the type of nucleus involved. The total energy emitted after the fissions studied confirms the above observations (E^{th}_{U233} = 10.25 MeV, E^{fast}_{U233} = 10.49 MeV and E^{th}_{U235} = 12.92 MeV). From the reactor safety point of view, the lower decay heat obtained with ^{233}U fissions as compared to ^{235}U fissions is an advantage.

3.3. Decay Heat In The MSFR

As already mentioned, the simulation of the reactor's evolution gives us the isotopic composition at any time during reactor operation. As discussed in [2], after 200 years of operation, the fuel salt composition is stable for almost all nuclei and we can consider it as the steady-state composition. It is the composition we use for this preliminary study. The decay heat produced by the steady-state

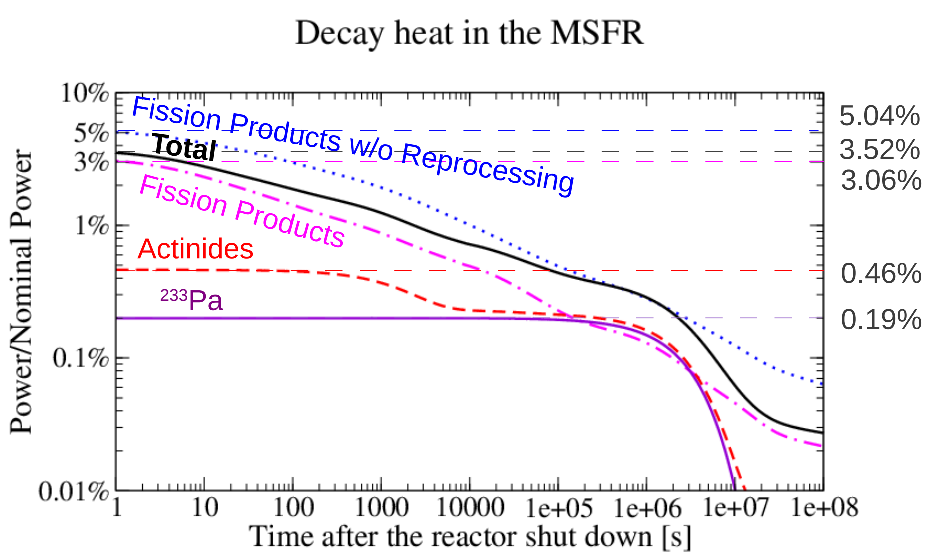

Figure 3. Residual heat in the MSFR, total: including fission products and actinides, fission products accumulated in the core, actinides and the decay heat of all fission products of ^{233}U fissions, neglecting all nuclear reactions after the fission and the reprocessing systems.

isotopic composition is displayed in figure 3 (black curve). We can separate the two contributions of this decay heat into those due to all nuclei with Z<70, corresponding mainly to fission products (dashed magenta curve) and with Z>80, corresponding to the actinides (dashed red curve). We can observe that, some hours after the reactor shut down, the contribution of the fission products is smaller than that of the actinides, as opposed to their contribution in the PWR [9]. This difference is due on the one hand to the ^{233}Pa decays (violet curve) and on the other hand to the reprocessing:

the fission products are extracted thus transferred from the core to the reprocessing system during reactor operation. The impact of this transfer is illustrated by comparing the contribution of the fission products in the steady state (dashed magenta curve) to the curve of accumulated fission products of ^{233}U (dotted blue curve). The latter corresponds to the heat we would have from the fission products without any reprocessing and any nuclear reaction on the fission products after $\sim$ 3 000 years of operation. This time was chosen because, if there is no reprocessing system, the accumulation of the fission products is not stable in 200 years but much later.
We conclude that the influence of the reprocessing on the decay heat is significant and leads to a low decay heat in the core and the fuel loops (3.5% compared to 6% in a PWR, [9]). We observe that an important part of the decay heat (1.98 % of nominal power) is located in the reprocessing units, mainly in the gas reprocessing unit, so that its safety should be studied separately. Finally, we have evaluated the decay heat in the core of the MSFR due to the fission products and the actinides. Another important contribution in the core is that of the fissions taking place just after the reactor shut down. This is discussed in the next section.

4. SHUT DOWN TRANSIENTS

Because of the strong coupling between the neutronics and the thermohydraulics, calculating the transients of a MSFR is quite challenging. However, it is interesting to develop a simplified tool to evaluate the transients. To take into account the dynamics due to the delayed neutrons, we developed a tool based on the point kinetics that can localize the precursors. For this, we define two lattices, a fixed and a moving one. With this spatial division, we define the global reactivity as the sum of the contribution to the reactivity of each individual cell, i.e. the reactivity weight of the cell. The coefficient $\left(\frac{dk}{dT}\right)_n$ represents the reactivity weight of fixed cell n. If the temperature decreases in the cell in the center of the core (maximum of the flux distribution), the feedback will be larger than for the cell on the bottom of the core. We can write $\frac{dk}{dT} = \sum_{n\subset Core}\left(\frac{dk}{dT}\right)_n = -5pcm/K$. Projecting the moving cells onto the fixed cells, we evaluate the physical quantities, that follow the equations:

Reactivity: $$\rho(t) = \sum_{n\subset Core} \rho_n(t) = \sum_{n\subset Core} \left(\frac{dk}{dT}\right)_n (T_n(t) - T_n^0) + I_n(t) \tag{2}$$

Power: $$\frac{\partial P}{\partial t} = \frac{\rho - \beta_{eff}}{l(1-\rho)} P + A \sum_{n\subset Core} \sum_i \lambda_i C_i^n \tag{3}$$

Precursor abundance of group *i*: $$\frac{\partial C_i^f}{\partial t} = \frac{\beta_i P_f}{l(1-\rho)A} - \lambda_i C_i^f \tag{4}$$

Temperature: $$\frac{\partial T_f}{\partial t} = \frac{P_f}{C_p d_f} \tag{5}$$

With:
f: Indicator for a moving cell, transporting the precursors; n: Indicator for a fixed cell; T_n^0: Mean temperature at the steady-state in cell n; $I(t)$: Reactivity insertion; β_i: Fraction of delayed neutrons of group i; $\beta_{eff} = \sum_i \beta_i \frac{\sum_{n\subset Core} C_i^n}{\sum_{n\subset Reactor} C_i^n}$: Total effective fraction of delayed neutrons, due to salt circulation; l: Mean lifetime of neutrons; C_i: Abundance of group i; λ_i: Decay rate of group i; A: Normalization factor; C_p: Specific heat; d: Salt density.

The heat exchanger is represented by the power extraction on the cells outside the core. We used this simplified model to calculate different transients.
Because of the strong negative reactivity feedback coefficient, a power excursion due to a reactivity insertion decreases rapidly [1]. In the MSFR, the reactivity reserves are very small: they are due to the precursors of delayed neutrons located outside the core during normal reactor operation, the ^{233}Pa decays, a wrong reinjection from the reprocessing unit. Those contributions do not exceed 300 pcm [1]. In this case a control rod would present the highest reactivity reserve and it was excluded from the MSFR concept in particular for this reason. A direct reactivity insertion seems not to be the typical accidental transient for this type of reactor. The loss of the cooling or of the salt circulation represent more interesting transients. In both cases we have an Unprotected Loss Of Heat Sink (ULOHS), the difference being in the evacuation or not of the delayed neutrons. For the definition of the transients, we consider that drainage is the MSFR's protection system. In this paper we discuss only the ULOHS transient in the case where the cooling is lost and the fuel circulation continues.

4.1. Decay Heat Due To The Delayed Neutrons

The transients due to diminishing extracted power while salt circulation is maintained are discussed in this section. Such a situation can occur, for example, in the event of a common cause failure of all intermediate salt pumps (station black-out and failure of emergency systems) and normal operation of the fuel salt pumps (no failure for the diesel generator).
In figure 4 on the left, we present 3 ULOHS transients lasting 1s, 30s and 60s, durations corre-

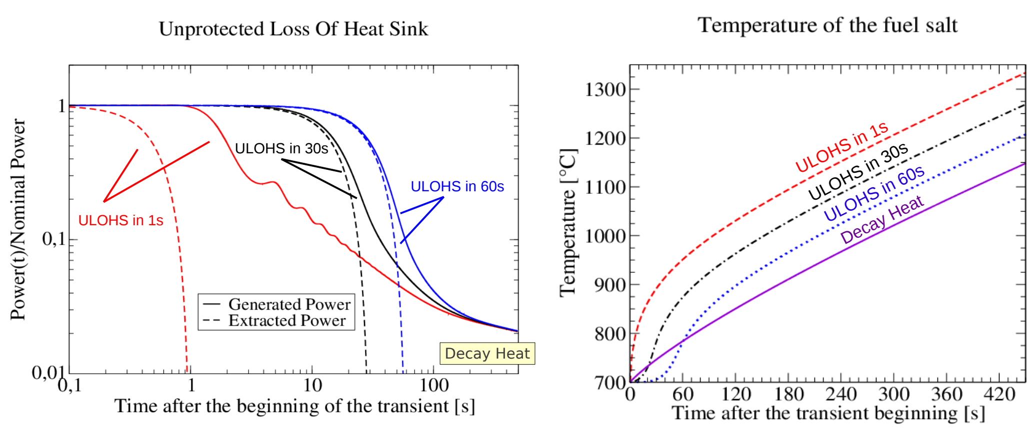

Figure 4. On the left: Transients after an Unprotected Loss Of Heat Sink in 1s (in red), 30s (in black) and 60s (in blue): the extracted power (dashed curves) and the power in the core (solid curves) including the fissions and the decay heat. On the right: Temperature of the fuel salt only due to the decay heat without fissions (violet solid curve), for unprotected loss of heat sink in 1s (red dashed curve), 30s (black dotdash curve) and 60s (blue dotted curve).

sponding to the inertia of the molten salts and the pumps of the intermediate circuit. It is the time in which the extracted power drop from nominal power to zero. For the 1s ULOHS transient, the power in the core shows oscillations, that are due to the delayed neutrons that are extracted and reinjected in the core because of the fuel salt circulation. For this transient, the loss of extracted power acts only on one part of the fuel salt, since the circulation time is $\sim$ 4s. For longer transients, the circulation of the fuel salt is fast enough to suppress those oscillations. Another advantage of the long transients is that, if the extracted power decreases slowly, the power in the core can follow its behavior. This information will be used to drive the reactor [1].

4.2. Temperature Of The Fuel Salt

In this paper we have studied the three main contributions of the residual heat in the core after a reactor shut down. The decay heat generated by the fission products and actinides is independent of the transients. In contrast, the heat due to delayed fissions depends strongly on the dynamic of the external parameters, such as the inertia of the pumps and other systems. In order to compare those two contributions for the MSFR, we will discuss the increase of the temperature of the fuel salt after a reactor shut down, displayed in figure 4 on the right. We consider that any heat generated, if it is not extracted, is stored in the fuel salt. We thus neglect heat losses through the surrounding structures, that depend on the precise design of the systems. Consequently, the real temperature increase will be slower.

Figure 4 shows the temperature increase due only to the decay heat (blue solid curve) and the above described transients. It is clear that the 1s fast transient (red dashed curve) is unfavorable, due to the fast temperature increase in 1 second and the global temperature increase that is almost 200°C higher than that due only to the decay heat. For the slow transients, at first the temperature is lower than that of the decay heat because power, including that of the decay heat, is still being extracted. Finally, for all three transients, the contribution of the delayed fissions leads to a larger temperature increase in the long term. Slow transients are thereby very favorable.

The residual heat study is the basis on which to specify the draining system. In figure 4, we can conclude that, if we set T=1200°C as the upper temperature limit for the surrounding structures in the core and the fuel loop, the drainage must occur before $\sim$7 minutes after the beginning of the transient. In view of avoiding fast transients, the inertia of the system should be maximized. The cooling of the draining storage system will be designed according to this evaluation of the residual heat. Finally, the impact of the stagnant heating fuel salt on each of the systems presented here in section 2.1 is under study, based on this evaluation.

5. CONCLUSIONS

The reference configuration of the MSFR concept, defined and presented in this paper, results from different parametric studies. To integrate safety into the design of the MSFR we are looking on possible improvements of this reference configuration. At present, we are working on identifying typical accidents for liquid fuel reactors. That implies a systematic description of the MSFR that will serve as the basis on which to develop accidental scenarios. They will be discussed and, in so far as possible, classified according to severity and a probability estimation. Thanks to the negative reactivity feedback coefficient, the main scenarios lead to a reactor shut down. In order to assess the behavior of the fuel salt after reactor shut down, we have developed and validated a tool to calculate the decay heat. Thanks to this, we conclude that the decay heat in the core and the fuel loops of the MSFR is relatively low (3.5% of nominal power) primarily thanks to the reprocessing system. The fission products that remain in the core contribute to the fuel salt heating up to 3% of nominal power. The gas reprocessing unit must handle the main part of the decay heat of the fission products as they are extracted from the core. The actinides also have an important contribution (0.5% of nominal power), that becomes dominant some hours after reactor shut down. With a tool based on point kinetics, we calculated loss of heat sink transients and studied their impact on the fuel salt temperature. The results of this study demonstrate the importance of the inertia of the systems. We conclude that slow transients (> 1 minute), thanks to a large system inertia, are advantageous and that, with them, the fuel salt temperature increase is slower. These residual heat calculations will be the basis for the design of the draining system, as drainage must occur for any reactor shut down, whether in normal or in accidental conditions. The impact of the stagnant heating fuel salt on the core and fuel loop systems will be studied as well.

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