

Design-Related Studies for the Preliminary Safety Assessment of the Molten Salt Fast Reactor

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Abstract—Molten salt reactors are liquid fuel reactors so that they are flexible in operation, but they are very different from solid fuel reactors in the approach to safety. This study concentrates 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 research and development studies. After a short description of the MSFR systems, necessary to device accidental scenarios, this paper will focus on the decay heat evaluation of such a reactor. Among different contributions, the decay heat of fission products in the MSFR is evaluated to be low (3% of nominal power), mainly due to the reprocessing during the reactor operation. As a result, the contribution of the actinides is significant (0.5% of nominal power). However, the decay heat of the fission products is important, and among the different uncertainty sources, the fission yield uncertainties are pointed out. The unprotected loss of heat sink transients are studied in this paper. It appears that slow transients are favorable (>1 min) 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.

I. INTRODUCTION

In the framework 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), which is now one of the six concepts selected by the Generation IV International Forum¹ for further study. The MSFR is to be operated in the Th/²³³U fuel cycle with fluoride salts. Since ²³³U 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 ²³³U. 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 to be an essential issue. Indeed, in the case of a reactor shutdown, 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 fuel salt heating that can cause damage. 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 Sec. III, is the main purpose of this paper. In Sec. IV, we discuss additional heat issues due to transients.

II. MSFR REFERENCE CONFIGURATION

Since 1997, the study of the concept of a molten salt reactor was undertaken by the Centre National de la Recherche Scientifique (CNRS), contributing to the development of the innovative Generation IV reactors. These studies led to the new MSFR concept. Moving beyond

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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 Ref. 3. The result is a fast neutron spectrum, presented in Ref. 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 (DBAs). 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. For this we use XRisk, a tool developed in order to allow a systematic modeling and systematic risk analysis, described in Ref. 4. Thereby the difficulty is to apply the 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.

Here, we present only the systems that 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 in 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 Fig. 1, done by A3I,^a our collaborator in design development.

1. *Core*: As mentioned earlier, 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(thermal), corresponds to a heating in the core of $\Delta T = 100\text{ K}$ 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 $\Delta T = -100\text{ K}$ is feasible. The resulting core is a cylinder whose height is equivalent to its

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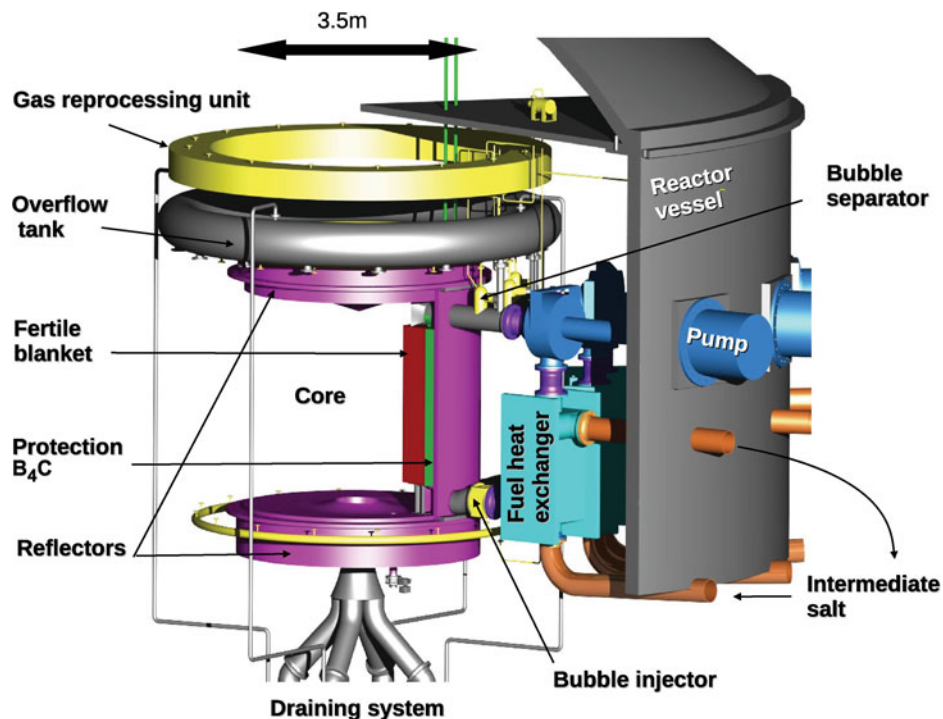


Fig. 1. View of the MSFR systems in contact with the fuel salt, done by A3I.

diameter, such that one-half of the entire salt volume is inside the core, the rest being located in the external fuel loops.

2. *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 the eutectic point. The salt has the following characteristics: calorific capacity $c_p = 1594 \text{ J kg}^{-1} \text{ K}^{-1}$; thermal conductivity $\lambda = 1.01 \text{ W m}^{-1} \text{ K}^{-1}$; density $d = 4.1 \text{ g/cm}^3$; and thermal expansion coefficient of $8.82 \times 10^{-4} \text{ K}^{-1}$ (Ref. 5). With the geometry described above, the total salt volume is 18 m^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 physicochemical and neutronic characteristics of the salt, it is necessary to clean the salt, i.e., to extract the fission products. Maintaining the physicochemical 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 neutronic behavior of the fuel salt is a crucial safety issue. A major safety parameter is the reactivity feedback coefficient, which is evaluated for the MSFR to be negative²:

$$\begin{aligned} \left(\frac{dk}{dT}\right)_{\text{Total}} &= \left(\frac{dk}{dT}\right)_{\text{Doppler}} + \left(\frac{dk}{dT}\right)_{\text{Density}} \\ &= -5 \pm 1 \text{ pcm/K} . \end{aligned} \quad (1)$$

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 ($\sim 4 \text{ s}$) and the lifetime of the precursors of delayed neutrons ($\sim 10 \text{ s}$) 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. Compared to pressurized water reactors (PWRs), the acting time of the feedback effect is very fast, on the order of 10^{-2} s (see Ref. 7).

3. *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.

4. *Pyrochemical Reprocessing Unit*: Among the soluble fission products, the lanthanides will be removed at a daily rate by pyrochemical reprocessing. This repro-

cessing unit is to be located on-site but outside the reactor vessel. The fuel salt reprocessing flow rate is very small (40 l/day), and it will be done by batch.

5. *Pump* (16 units): Since natural convection is not sufficient to evacuate the nominal power, we need pumps to drive the fuel salt mass at a flow rate $Q \approx P_{\text{nominal}}/c_p \cdot \Delta T = 19 \text{ t/s}$. 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.

6. *Heat Exchanger* (16 units): 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 Sec. IV. A heat exchanger can influence the extracted power in two manners: First, the flow of the fuel salt in the heat exchanger can be reduced because of some clogging; second, the temperature and the flow rate of the intermediate salt circuit may vary.

7. *Bubble Injection* (16 units): 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 to obtain a volumetric ratio of bubbles of 0.1% in the fuel salt.

8. *Bubble Separator* (16 units): The bubble separator is studied at Laboratoire de Physique Subatomique et de Cosmologie in the frame of the Forced Fluoride Flow for Experimental Research project,⁸ where the design of the separator is being developed. The bubble separator is connected to the gas reprocessing unit, which will not be detailed here. It is important to note that the reprocessed gases are directed to the bubble injection.

9. *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.

10. *Draining System*: As shown below, the draining system is a very important safety and operational system. In fact, for a planned shutdown, 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 criticality inability and passive cooling. To have redundant safety systems, several drainage procedures (active and passive) will be defined.

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 Fig. 1. It will most probably include cooling by the intermediate salt.

To study the accidental scenarios of the MSFR, 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 shutdown 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; see formula 1.

After the reactor shutdown, 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.

III. DECAY HEAT CALCULATIONS

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

1. *Decay of the fission products*: The fission products are unstable and decay by emitting mainly gamma rays or beta particles.
2. *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).
3. *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 (fuel cladding and

structural materials) has to be taken into account. In the MSFR configuration described in Sec. II, 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 (Ref. 9). They have different timescales. The contribution of the delayed neutrons through the fissions they induce depends on the dynamic of the reactor shutdown, or more precisely on the amount of negative reactivity that is provided by the shutdown system. If the negative reactivity is 4000 pcm or so, this source of residual heat is dominant at short timescales and disappears after 100 s. The actinides generally have a long lifetime, thus, 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 10 s 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.

III.A. Decay Heat Calculation Tool

Our numerical simulations of the reactor rely on the coupling of the MCNP neutron transport¹⁰ with a home-made materials evolution code² REM. 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.

A newly developed tool for the decay heat calculations, called isotopic composition evolution (ECI), 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 shutdown are not taken into account by the ECI tool. The fissions due to the delayed neutrons are calculated with another code described in Sec. IV.

III.B. 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: ²³³U, because of its importance in the thorium cycle, and ²³⁵U, because of the evaluations and the experiments that are available from several sources. To validate a calculation tool relative to experiment, a deeper analysis is needed because the measured decay heat data are 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

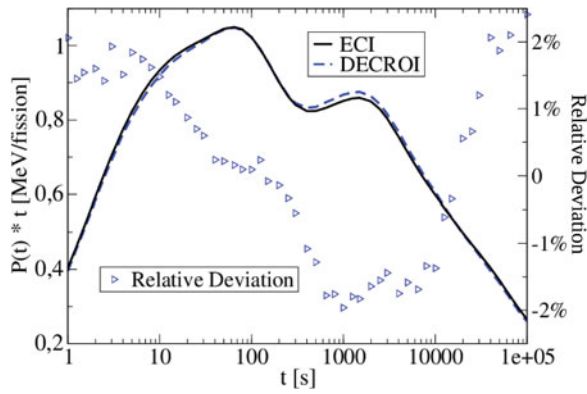


Fig. 2. Comparison of ECI to DECROI calculations on the decay heat of an elementary fission of ^{233}U .

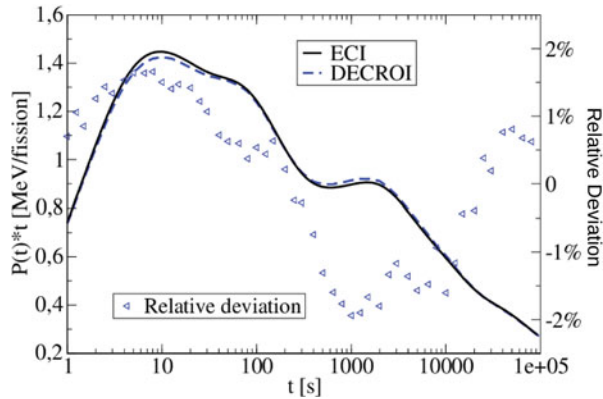


Fig. 3. Comparison of ECI to DECROI calculations on the decay heat of an elementary fission of ^{235}U .

Figs. 2 and 3, our calculations with ECI for both nuclei fit the reference curves with an accuracy $>2.5\%$. 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), which will be discussed in Sec. III.B.1, and of the decay energy data (sometimes there is no evaluation of the uncertainties), we can consider that our tool is validated.

III.B.1. Comparison of the ^{235}U and ^{233}U Elementary Fissions

This section focuses on the difference between an elementary ^{233}U and ^{235}U fission (see Figs. 2, 3, and 4), to understand its origin. For this, we present in Fig. 5 fission yields of those two nuclei. The shapes of the light peaks (mass $A \sim 95$) are slightly different while the heavy peaks ($A \sim 140$) are very close for ^{233}U and ^{235}U fission yields. The uncertainties presented in this figure are calculated using the evaluated data of JEFF-3.1.1, as rec-

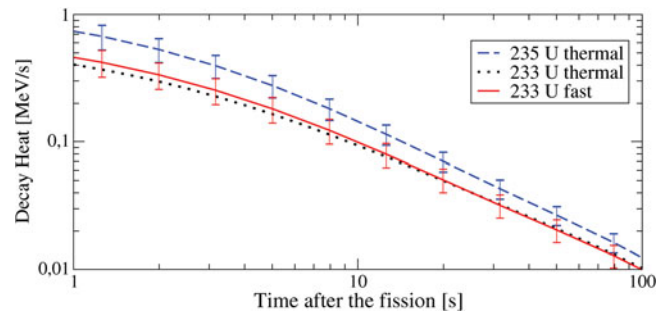


Fig. 4. Decay heat of an elementary fission of ^{233}U in thermal and fast neutron spectrum and of ^{235}U in thermal neutron spectrum, using fission yield database JEFF-3.1.1 and decay database JEFF-3.0.

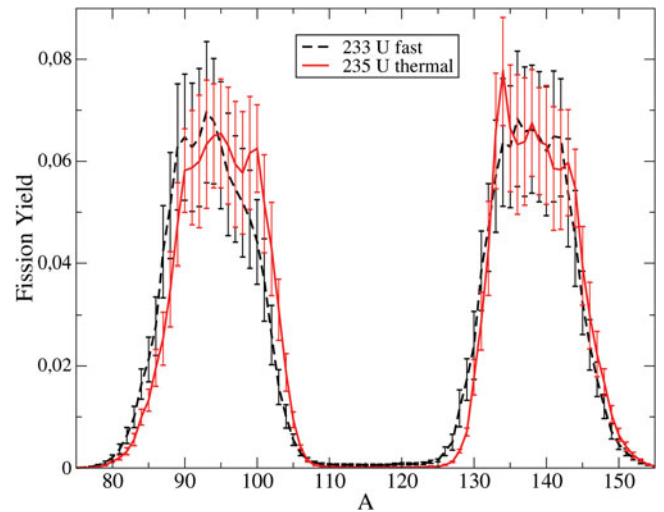


Fig. 5. Fission yields of ^{233}U for a mean neutron energy $E = 400$ keV and of ^{235}U for a mean neutron energy $E = 0.0253$ eV, using fission yield database JEFF-3.1.1.

ommended in Ref. 12, without taking into account any correlation between the isotopes, since no correlation was found in the evaluated data.

In Fig. 4, the decay heat is presented for ^{233}U in fast and thermal neutron spectrum and for ^{235}U in thermal neutron spectrum. In the short term, 1 s after the fission, the decay heat from the fission products of ^{235}U (0.74 MeV/s) is almost double that for ^{233}U (0.40 MeV/s) for thermal neutron spectrum. The uncertainties on the decay heat were evaluated through propagation of the uncertainties on the fission yield without any correlation. The uncertainties on the decay heat are then largely overestimated. However, the difference between the two nuclei seems significant. 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. This discrepancy comes from the fission yields of the two nuclei observed in Fig. 5. The total energy emitted after the fissions considered confirms the above observations ($E_{U233}^{th} = 10.25$ MeV, $E_{U233}^{fast} = 10.49$ MeV, and $E_{U235}^{th} = 12.92$ 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.

III.C. 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. We considered the steady-state composition of the fuel salt for this preliminary study,² which is the enveloping case for this study.

The decay heat produced by the steady-state isotopic composition is displayed in Fig. 6 (solid curve, labeled "Total"). We can separate the two contributions of this decay heat into those due to all nuclei with $Z < 70$, corresponding mainly to fission products (dotted-dashed curve) and with $Z > 80$, corresponding to the actinides (dashed curve). We can observe that, some hours after the reactor shutdown, the contribution of the fission products is smaller than that of the actinides, as opposed to their contribution in the PWR (Ref. 9). This difference is due on the one hand to the ^{233}Pa decays (solid curve, labeled " ^{233}Pa ") 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 (dotted-dashed curve) to the curve of accumulated fission products of ^{233}U (dotted 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 ~ 3000 years of opera-

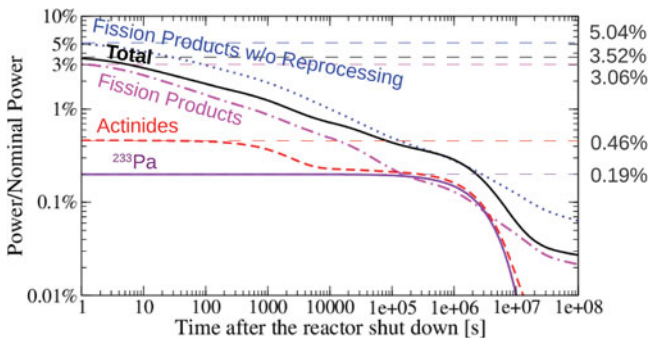


Fig. 6. Decay heat in the MSFR, total including fission products and actinides, fission products accumulated in the core, actinides and the decay heat of all fission products of ^{233}U fissions, neglecting all nuclear reactions after the fission and the reprocessing systems.

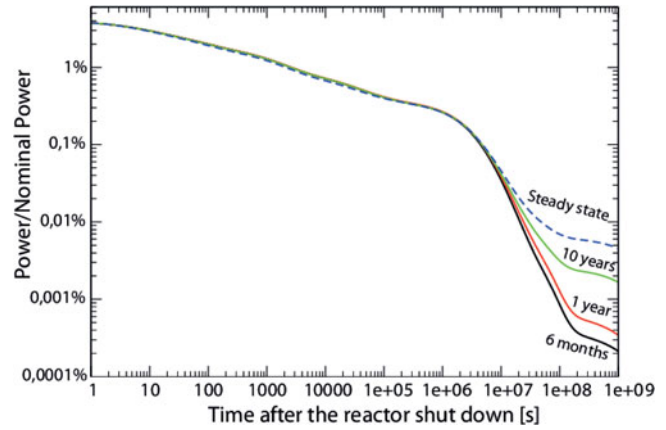


Fig. 7. Influence of the reactor operation time on the decay heat in the fuel salt.

tion. This time is needed to stabilize the accumulation of fission products without the reprocessing system.

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 (Ref. 9)]. We observe that an important part of the decay heat is located in the reprocessing units, mainly in the gas reprocessing unit, so that its safety will have to be studied separately.

The influence of the operation time on the decay heat has been evaluated, as shown in Fig. 7. We see that after a short time of operation of 6 months, there is no influence of the operation time on the decay heat in the short term ($t < \text{some years}$). In the longer term the difference observed is due to the accumulation of the actinides.

III.C.1. Uncertainty Sources due to Fission Products

As discussed in Sec. III.B.1, the uncertainties given by the evaluators for the fission yields are high for some nuclei, and the correlations are not given. Especially, the fission yield of ^{233}U for fast neutron spectrum is not well known. To discuss the importance of the knowledge of this data for the safety study, we focus on the main contributors to the decay heat of the steady-state composition. First, we will compare these contributors with those of an elementary fission. The isotopes that contribute to the decay heat of the steady-state composition and the fresh fuel are the same as for a fresh fuel. The difference between the steady-state composition and the fresh fuel is due to the buildup of the fission products in the fuel salt and to the reprocessing system, which extracts a part of the fission products before they decay in the fuel salt.

In Figs. 8 and 9, the sum of contributions to the decay heat of the elements as function of their mass A is presented. The contributions change over time. We present

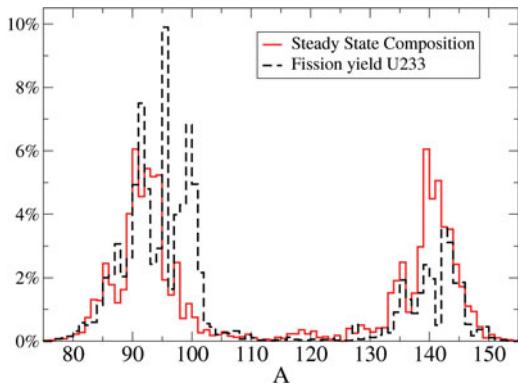


Fig. 8. Contribution on the decay heat of fission products as a function of mass A for the steady-state composition and the fresh fuel composition, corresponding to the elementary fission, 10 s after the fissions stop.

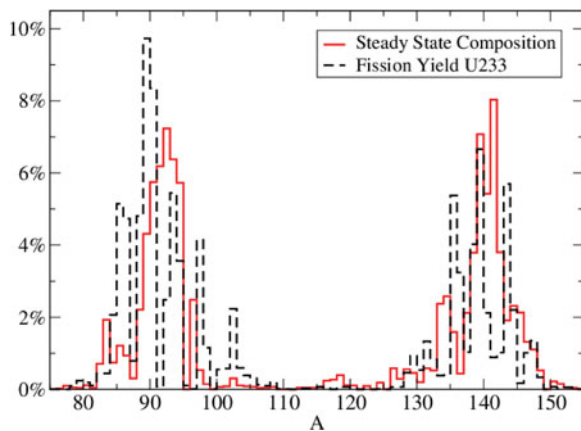


Fig. 9. Contribution on the decay heat of fission products as a function of mass A for the steady-state composition and the fresh fuel composition, corresponding to the elementary fission, 100 s after the fissions stop.

here the contributors for $t = 10$ s and $t = 100$ s after the fission for an elementary fission or after the reactor shutdown (no more chain reaction) for the steady-state composition. The main contributors of an elementary fission 10 s after the fission are the isotopes from the light peak, while the heavy peak becomes more important in the longer term. For the fission products contained in the fuel salt, the main contributors are partly different. The light peak is as important as the heavy peak even 10 s after the reactor shutdown. The difference between the dashed curve (100 s after the fission) and the solid curve (100 s after the chain reaction stop) of Fig. 9 is significant especially for the masses $A = 87$ and 136 , mainly due to ^{87}Br and ^{136m}I in the fission yield, and $A = 92$, 93 , 141 , and 142 , mainly due to ^{92}Y , ^{92}Sr , ^{93}Sr , ^{93}Y , ^{141}Ba , ^{141}La , ^{142}La , and ^{142}Ba in the steady-state composition.

To reduce the uncertainty on the decay heat of the fuel salt, the amount of these isotopes as well as the energy emitted by their decay should be well known. On the production of the isotopes, we have to point out three main sources of uncertainties: the fission yields, the reprocessing efficiency, and the neutron reaction cross sections of the fission products. The most important isotopes that contribute to the decay heat of the fission products in the MSFR are listed in Table I. In this table, the fifth and seventh columns represent respectively the production of the isotope directly from the fission and through the decay of isotopes that are produced by the fission. The last column contains the production of the isotope that comes from reaction and decay trees and goes through a neutron reaction. This contribution shows the importance of the neutron cross sections of the fission products on the decay heat evaluation, when required data are available (neutron cross-section database used is ENDF/B-VI).

The values from columns 5 and 7 compared to column 8 of Table I clearly point out the fission yield data as the main source of uncertainties. The heavy peak of ^{233}U fission yields with correlations for thermal neutrons is currently under evaluation.¹³ These new data could reduce the uncertainty on the production of the fission products created either directly by the fissions or through decays. The decay constant of very unstable isotopes may also have an uncertainty, but this one would be important only in the very short term ($t \leq 1$ s). The neutron cross-section data for the fission products are unfortunately not complete, so the real importance of these reactions cannot be evaluated. As presented in Sec. III.C, because of the bubbling system, the decay heat in the core is reduced. In other words, if an element has finally a smaller extraction efficiency, this can increase the decay heat in the core. The knowledge of these data is thus important for the safety study.

Finally, we have evaluated the decay heat in the core of the MSFR due to the fission products and the actinides. We have discussed the different sources of uncertainties on the main contribution of the decay heat due to the fission products. Another important contribution of the residual heat in the core is due to the fissions taking place just after the reactor shutdown. This is discussed in the next section.

IV. SHUTDOWN TRANSIENTS

Because of the strong coupling between the neutronics and the thermohydraulics, that was shown for example in Ref. 14, 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,

TABLE I
Main Isotopes that Contribute to the Decay Heat 100 s After the Chain Reaction Stop*

Isotope	$T_{1/2}$	Inventory (mol)	Decay Heat (%)	Production Rate (mol/day)			
				Fission	Uncertainty (%)	Decay	Neutron Reaction
^{142}La	91 min	0.070	5.8	0.10	32	0.70	$<10^{-3}$
^{94}Y	18.7 min	0.015	4.8	0.11	32	0.66	20^{-9}
^{140}La	1.7 days	1.693	4.7	0.004	34	0.68	30^{-3}
^{93}Sr	7.4 min	0.006	4.6	0.47	17	0.01	$<30^{-4}$
^{95}Y	10.3 min	0.008	4.1	0.32	21	0.45	10^{-9}
^{92}Y	3.5 h	0.175	3.3	0.004	36	0.80	70^{-4}
^{141}Ba	18.3 min	0.013	3.0	0.44	18	0.39	30^{-4}
^{92}Sr	2.7 h	0.132	2.9	0.24	26	0.47	$<30^{-4}$
^{93}Y	10.2 h	0.527	2.6	0.004	35	0.82	$<20^{-4}$
^{135}I	6.6 h	0.209	2.5	0.54	13	0.005	$<20^{-4}$
^{91}Sr	9.6 h	0.436	2.4	0.057	34	0.61	$<30^{-5}$
^{97}Zr	16.7 h	0.702	2.4	0.17	28	0.44	60^{-3}
^{134}I	52.5 min	0.016	2.2	0.24	16	0.15	$<40^{-4}$
^{142}Ba	10.6 min	0.007	2.2	0.60	12	0.17	$<10^{-3}$
^{90m}Rb	258 s	0.001	2.2	0.25	24	0.03	$<10^{-4}$
^{139}Cs	9.3 min	0.005	2.2	0.53	14	0.15	$<40^{-4}$
^{143}La	14.2 min	0.009	2.2	0.27	22	0.46	$<50^{-4}$
^{138}Cs	33.4 min	0.007	1.9	0.11	23	0.02	$<10^{-4}$
^{141}La	3.9 h	0.181	1.8	0.03	34	0.73	50^{-4}
^{91}Rb	58 s	0.001	1.8	0.51	15	0.25	$<30^{-4}$
Sum		4.2	60		5.0	—	<0.0153
All fission products		5170	100		26.8	—	0.236

*Main contributors to the decay heat of the fission products for the steady-state composition at $t = 100$ s after the fissions stop. Here are listed their production direct from fission with the uncertainty, indirect from fission so through decay, and the production due to a neutron reaction. The line "Sum" indicates the values for the sum of isotopes listed in this table, and the last line gives the values for all fission products. For these lines, the production rate through decay is left empty because the isotopes are coupled through decay, and the sum of their production through decay will be meaningless.

one fixed and one moving. With this spatial division, we define the total reactivity as the sum of the contribution to the reactivity of each individual cell, i.e., the reactivity weight of the cell. The coefficient α_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 evaluate this coefficient assuming $\alpha_n = (dk/dT)_n$. We can write $dk/dT = \sum_{n \in \text{Core}} (dk/dT)_n = -5$ pcm/K. Projecting the moving cells onto the fixed cells, we evaluate the physical quantities that follow these equations:

$$\begin{aligned} \text{Reactivity: } \rho(t) &= \sum_{n \in \text{Core}} \rho_n(t) \\ &= \sum_{n \in \text{Core}} \left(\frac{dk}{dT} \right)_n (T_n(t) - T_n^0) \\ &\quad + I_n(t), \end{aligned} \quad (2)$$

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

Precursor abundance of group i :

$$\frac{\partial C_i^f}{\partial t} = \frac{\beta_i P_f}{l(1 - \rho) A} - \lambda_i C_i^f, \quad (4)$$

$$\text{Temperature: } \frac{\partial T_f}{\partial t} = \frac{P_f}{C_p d_f}, \quad (5)$$

where

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
- $\beta_{eff} = \sum_i \beta_i (\sum_{n \in Core} C_i^n / \sum_{n \in Reactor} C_i^n)$: 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 feedback coefficient and the low reactivity margins² in the MSFR, a direct and involuntary reactivity insertion seems not to be a typical accidental transient for this type of reactor. The cooling of the fuel salt may be lost because of a fuel salt circulation loss or because of a heat sink loss coming from the intermediate circuit. These two types of transients have similar behaviors. Their only difference is in the contribution 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 unprotected-loss-of-heat-sink (ULOHS) transient in the case where the cooling is lost and the fuel circulation continues.

IV.A. Residual 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 blackout and failure of emergency systems) and normal operation of the fuel salt pumps (no failure for the diesel generator).

In Fig. 10, we present three ULOHS transients calculated with inertia values of 1, 30, and 60 s. This inertia of the molten salts and the pumps of the intermediate circuit corresponds to the time necessary to drop the extracted power from its nominal value to zero. For the 1-s ULOHS transient, the power in the core shows oscillations that are due to the precursors of the delayed neutrons that flow out and back in the core because of the fuel salt circulation. For this quick transient, the loss of extracted power acts only on a fraction of the fuel salt,

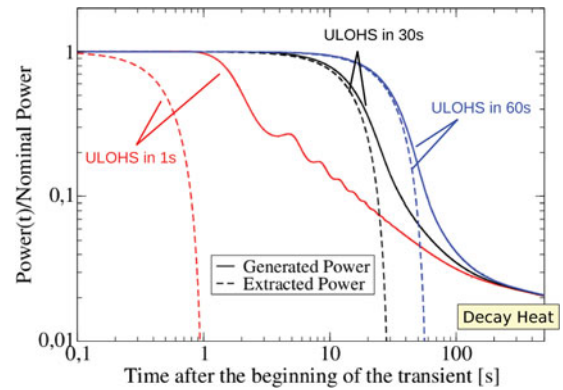


Fig. 10. Transients after an ULOHS in 1 s, 30 s, and 60 s: the extracted power (dashed curves) and the power in the core (solid curves) including the fissions and the decay heat.

since the circulation time of ~ 4 s is larger than the power drop. 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.⁷

IV.B. 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 shutdown. 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. To compare those two contributions for the MSFR, we will discuss the increase of the temperature of the fuel salt after a reactor shutdown, displayed in Fig. 10. We consider that any heat generated, if not extracted, is stored in the fuel salt. Thus, we 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 11 shows the temperature increase due only to the decay heat (long-dashed curve) and the above-described transients. It is clear that the 1-s fast transient (short-dashed curve) is unfavorable because of the fast temperature increase in 1 s 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.

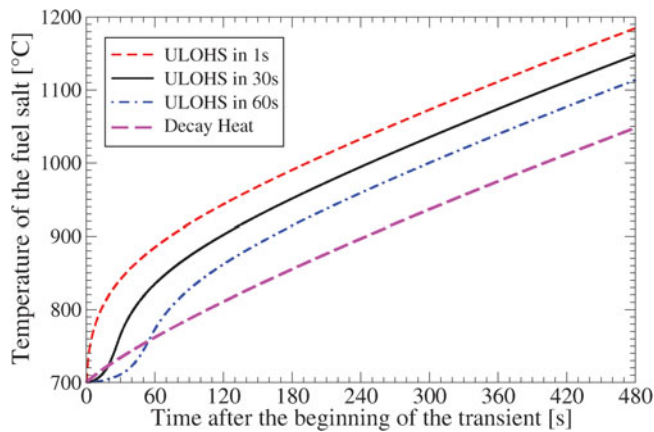


Fig. 11. Temperature of the fuel salt due only to the decay heat without fissions (long-dashed curve), for ULOHS in 1 s (short-dashed curve), 30 s (solid curve), and 60 s (dashed-dotted curve).

The residual heat study is the basis on which to specify the draining system. In Fig. 11, we can conclude that if we set $T = 1200^{\circ}\text{C}$ as the upper temperature limit^{15b} for the surrounding structures in the core and the fuel loop, the drainage must occur before ~ 8 min 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 in Sec. II is under study, based on this evaluation.

V. 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 at 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, insofar as possible, classified according to severity and a probability estimation. Thanks to the negative reactivity feedback coefficient, the main scenarios lead to a reactor shutdown. To assess the behavior of the fuel salt after reactor shutdown, 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

^bMaterial studies are currently being undertaken to determine the limit temperature.

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 shutdown. The main uncertainties on the decay heat are discussed, and the importance of the fission yields knowledge and their correlations are pointed out. 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 min), 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 shutdown whether in normal or in accidental conditions. The impact of the stagnant heating fuel salt on the core and the fuel loop systems will be studied as well.

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