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A STARTING PROCEDURE FOR THE MSFR: APPROACH TO CRITICALITY AND INCIDENT ANALYSIS

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Molten salt reactors are liquid-fueled reactors so that they are flexible in terms of operation (load-following capabilities…) or design (core geometry, fuel composition, specific power level…) choices, but they are very different in terms of design and safety approach compared to solid-fueled reactors. Such reactors call for a new definition of their operating procedures. Dedicated studies are performed in the frame of the European SAMOFAR (Safety Assessment of Molten Salt Fast Reactors) project of Horizon2020. This paper focuses on the behavior of the MSFR fuel circuit in interaction with the intermediate circuit. It is devoted to the start-up procedure of the MSFR, including the description of the proposed procedure, a presentation of the approach to criticality and the reactivity measurement during the filling of the core, and finally studies of accident scenarios (overcooling, reactivity insertion) at low power during the divergence step of the MSFR, highlighting the very good behavior of the reactor to such abnormal transients.

I. INTRODUCTION

The MSFR[1,2], as a liquid-fueled reactor where the fuel also acts as the coolant and is circulating, calls for a new definition of its operating procedures taking into account the characteristics of such a system. Regarding the neutronics characteristics, the negative feedback coefficient of the MSFR, around -8 pcm/K coming half from the density effect and half from the Doppler effect, provides intrinsic reactor stability. Unlike with solid-fueled reactors, the negative feedback coefficient acts very rapidly since the heat is produced directly in the coolant, the fuel salt itself being cooled in the heat exchangers. Also in such a circulating-fuel system, the fraction of delayed neutrons is reduced because the fuel motion drift the delayed neutron precursors in low importance areas. The calculation of this important reactor kinetics parameter thus requires special tools[3,4]. More globally the modeling of such reactors requires specific treatments to take into account all the phenomena associated to the liquid fuel circulation.

Finally a MSFR design characteristics also impacts strongly the reactor operation: no control rod is foreseen in the core, the reactor being driven by the heat extraction.

This definition of the operating conditions of the MSFR is one of the main tasks to be accomplished in the frame of the European SAMOFAR (Safety Assessment of the Molten Salt Fast Reactor) project of Horizon2020. This definition will rely on a system code under development and on physical studies presented in this paper for the start-up procedure. These preliminary operation procedures are based on discussions, advice from experts and calculations with a coupled simulation tool for precise transient studies as well as simplified neutronics point-kinetics calculations to identify procedures. After a section dedicated to the description of the MSFR concept and of the startup procedure, the simulation tool used for the present study are presented in section III. The approach to criticality is then detailed in section IV through the reactivity measurement and prediction during the filling step of the core. Finally, relying on coupled neutronics-thermalhydraulics calculations, the behavior of the reactor is studied for two abnormal transients (over-cooling and accidental reactivity insertion) that may occur during the divergence step, at low power (section V).

II. DESCRIPTION OF THE CONCEPT AND THE STARTUP PROCEDURE

II.A. Molten Salt Fast Reactor Concept

Since 15 years, the National Centre for Scientific Research (CNRS, Grenoble-France) has focused R&D efforts on the development of a new molten salt reactor concept called the Molten Salt Fast Reactor (MSFR) selected by the Generation-IV International Forum (GIF) due to its promising design and safety features[1]. The reference MSFR design is a 3000 MWth reactor with a total fuel salt volume of 18 m^3, operated at a mean fuel salt temperature of 700°C (Ref. 2). The fuel salt is composed of a molten lithium fluoride salt containing the heavy nuclei: the fertile matter being ^{232}Th and the fissile matter that can be used being $^{233}UF_4$ and/or $^{enr}UF_4$ and/or (Pu-MA)F_3. The present studies have been done with a fuel salt composition corresponding to the beginning of life of a

^{233}U-started version of the MSFR: 77.5mol% of ^{7}LiF-20mol% of ThF_4-2.5mol% of $^{233}UF_4$.

Conceptual design activities are currently underway so as to ascertain whether MSFR systems can satisfy the goals of Generation-IV reactors in terms of sustainability (Th breeder), non-proliferation (integrated fuel cycle, multi-recycling of actinides), resource saving (closed Th/U fuel cycle, no uranium enrichment), safety (e.g. as far as regard the following MSFR characteristics: no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

The MSFR system includes three different circuits: the fuel circuit, the intermediate circuit and the power conversion circuit. This paper focuses on the fuel circuit behavior in interaction with the intermediate circuit.

I.A.1. Fuel Circuit Description

The fuel circuit, defined as the circuit containing the fuel salt during power generation, includes the core cavity and the recirculation or cooling loops or sectors to extract the heat produced. The fuel salt volume is distributed half in the core ($9m^3$) and half in recirculation out of the core.

To prevent the risks of leakage in this circuit, an innovative segmented geometry of the fuel circuit (see Fig.1) has been proposed in the frame of the SAMOFAR project.

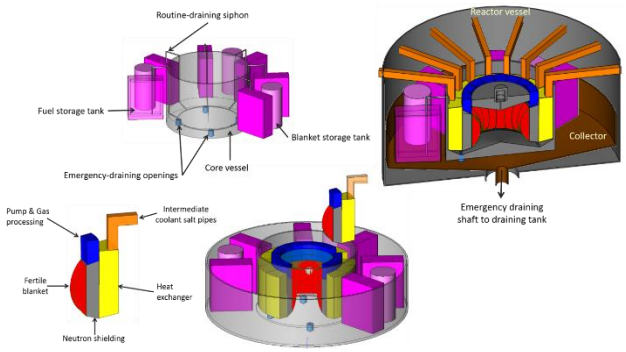

Fig. 1. Segmented geometry of the fuel circuit of the MSFR.

The core is enclosed in a vessel with an open top. The vessel serves as the container for the fuel salt. Its thick bottom (reflector) comprises openings for the fuel salt draining. The siphons for routine draining and core filling are placed on the sides of the vessel. The 16 cooling sectors are arranged circumferentially around the vessel. Each sector comprises: a heat exchanger, a circulation pump, a bubble injector and a gas separator, a blanket salt tank, and cooling equipment (using the intermediate fluid). The sectors are inserted from the top of the vessel and a sector may be replaced in case of failure of one component. The sectors are connected to the intermediate fluid circuits, for example 4 circuits, each feeding 4 sectors so as to still cool the core if one of the intermediate circuits fails.

I.A.2. Nominal Operation

During nominal operation, the fuel salt is sucked upward by the pumps in each sector, injected in the intermediate exchangers, then driven back into the bottom of the core vessel. The overall circulation time is approximately 4 seconds so as to limit the speed in the intermediate exchangers and thus the pressure drop in the exchangers, while limiting the in core temperature gradient.

II.B. Description of the startup procedure

The startup procedure is composed of two steps: first a filling of the core with the fuel salt, and once the core is filled the pumps are started to circulate the fuel so that the heat extraction is launched. These two steps are described in this section.

II.B.1. Step 1: Filling Step - Approach to Criticality and Reactivity Measurement

After preheating the intermediate circuit and the core vessel to a temperature at which the intermediate fluid is liquid (350 to 450°C depending on the fluid), the intermediate fluid is brought to temperature and put in slow circulation in order to heat the core vessel walls. To avoid overheating this circuit, the temperature is kept at a value slightly higher than the fuel salt melting temperature (585°C), say 620°C.

The fertile blanket is filled at a temperature of 620°C and its cooling by the intermediate circuit is started. The axial walls of the core vessel are then at the same temperature and core filling can begin.

The fuel salt, at a temperature of 620°C, is injected in the core through the central orifice that serves for regular fuel transfers during nominal operation. In the presence of any residual power, the fuel salt temperature increment will have to remain within limits. The duration of the filling procedure and the minimal delay after fuel salt draining are thus related. If the filling procedure is slow (several hours), the residual power must be small in order that the fuel salt temperature in the core not excessively overshoot the nominal operating temperature.

A neutron source located in the lower reflector will be used to monitor the reactivity level of the core all along the filling procedure. More precisely, the neutron flux from this source will be measured in the upper reflector, allowing the calculation of the neutron multiplication factor by the fuel salt during the core filling as detailed in section IV.A. While the core level is low, the fuel salt will absorb the neutrons and thus attenuate the source. It will then multiply the neutron source more and more as filling progresses. When the core is half full, the reactivity that will be reached at the end of the filling procedure can be predicted by this multiplication measurement within a few hundred pcm. If the expected final reactivity is too different from its nominal value, the filling procedure must be interrupted and the core must be drained. Otherwise, the

filling procedure can go on, with possible composition adjustments of the injected fuel salt.

At any rate, since the fuel salt temperature is below the nominal temperature and because of the feedback coefficient, criticality will be reached before the end of the filling procedure. Fissions will heat the fuel salt, thus reducing the reactivity. Because the filling procedure is slow, the multiplication factor will remain at unity and the temperature will increase as core filling proceeds. Note that, in this situation, the feedback coefficient is not the same as during normal operation. Indeed, only the Doppler is fully active while the density coefficient plays only partially since the amount of fuel salt in the core is not yet determined by the temperature. Only the geometry is slightly impacted (salt level depends on the temperature). Observe that all the delayed neutron precursors remain in the core so that β_{eff} (the effective fraction of delayed neutrons) is larger than during nominal operation.

The intermediate circuit being operational, heat will be continuously extracted from the core, proportionally to the temperature difference between the intermediate fluid and the fuel salt. This keeps the temperature of the walls close to their nominal operation temperature. The heat extraction is compensated by fissions and a temperature gradient that sets in between the fuel salt and the cooled surfaces.

As filling completes, the fuel salt temperature will not be exactly equal to the nominal operating temperature so that the reactivity has to be adjusted. The safety controls done at mid-filling ensure that this departure from the norm will not be large (a few hundred pcm, corresponding to a few tens of °C since the density coefficient now plays its role fully).

If the reactivity is too small, the temperature reached is lower than the nominal temperature so that the nominal power will not be reachable: the temperature difference between the intermediate fluid and the fuel salt will remain too small to allow nominal power evacuation at the intermediate exchangers. In this situation, fissile matter will have to be added via the sampling-injection mechanism of the salt cleaning process. This can take several days but does not put at risk the proper operation of the reactor.

If the reactivity is too large, the temperature reached will be too high and bubbles for example may be injected to compensate this reactivity (about 150 pcm per %volume of gas). Subsequently, again via the sampling-injection mechanism for the salt processing, some fissile matter will have to be removed. Likewise, this operation can last several days. Observe that the correct operation of the reactor is subject to the proper operation of the bubble-control injection mechanism.

II.B.2. Step 2: Divergence and Power Ramp Up Step

In any case, the power can be ramped up by starting the fuel salt circulation and increasing the power extracted by the intermediate circuit. At the same time, the reactivity adjustment can begin, using one of the levers described below. Observe that starting the fuel salt circulation implies that delayed neutron precursors will be dragged outside the central cavity and, as a consequence, the reactivity will decrease, by about 200 pcm, so that the fuel salt temperature falls by about 25 to 30°C.

A slow reactivity and thus criticality temperature variation can be performed by an adjustment of the fuel composition, one kilogram of ^{233}U corresponding to 9.5 ± 0.2 pcm.
A fast reactivity change can be obtained by increasing the gas injection rate in the core cavity when it is full of fuel salt: part of the fuel salt is then evacuated out of the core, resulting in a reactivity decrease of 150 pcm for 1% of gas volume in the core.
Finally, for a given fuel composition, another way to modify the reactivity of the reactor is to change the fuel temperature either by increasing or decreasing the heat extraction in the heat exchangers.

III. DESCRIPTION OF THE SIMULATION CODES FOR TRANSIENT STUDIES

Two simulation tools have been used in this paper: the first one is a neutronics modeling of a simplified geometry of the core, based on the point-kinetics model. The second tool is a coupling of neutronics and thermalhydraulics with the Transient Fission Matrix (TFM) Monte-Carlo based approach[5,6] for the neutronics part and the OpenFOAM CFD code for the thermalhydraulics part.

III.A. Point-kinetics model

A classical Point-Kinetics model (named PK in the following) has been used, consisting in factorizing the neutronic flux to separate the spatial and the energetic evolution on one side, and the time evolution on the other side. The underlying hypotheses are a uniform energy deposition in the core, together with an instantaneous propagation and extraction of the heat produced. This corresponds to Eq. (1) to (4).

$$\rho(t) = \frac{d\rho}{dT}\,[T(t) - T_0] + \mathrm{I}(t) \qquad (1)$$

$$\frac{\partial N}{\partial t}(t) = \frac{\rho(t) - \beta_{eff}}{l_{eff}\,(1-\rho(t))}\,N(t) + \sum_i \lambda_i C_i(t) + S(t) \qquad (2)$$

$$\frac{\partial C_i}{\partial t}(t) = \frac{\beta_{eff}^i \cdot N(t)}{l_{eff}\,(1-\rho(t))} - \lambda_i\, C_i(t) \qquad (3)$$

$$\frac{\partial T}{\partial t}(t) = \frac{P(t)-P_0}{C_P\, d} \quad (4) \quad \text{and} \quad P(t) = \frac{1}{\nu}\frac{N(t)E_{fission}}{l_{eff}(1-\rho(t))} \quad (5)$$

with t the time duration since the beginning of the transient, ρ the reactivity, dρ/dT the global thermal feedback coefficient, T the mean fuel salt temperature, T_0 the mean fuel salt temperature in nominal conditions, P the power density in the core, P_0 the nominal power density extracted, l_{eff} the effective prompt neutron lifetime (taken

equal to 1.064μs), C_P the fuel heat capacity in $J/kg/K$ $\left(-1111 + 2{,}78T_{(K)}\right)$, d the fuel salt density in kg/m^3 $\left(4983{,}56 - 0{,}882T_{(K)}\right)$, E_{fiss} the energy of one fission (in J) equal to 180 MeV, S an external volumic neutron source, and C_i and λ_i respectively the density and the decay constant of precursor family i. I(t) is the reactivity insertion. Finally β_{eff}^i are the effective fractions of delayed neutrons of the eight precursor families i and have been evaluated by the coupled calculations of the precise system with the TFM-OpenFOAM code described in the next section. The global effective fraction of delayed neutrons $\beta_{eff} = \sum_i \beta_{eff}^i$ is equal to 309 pcm.

The core geometry used here is a simple cylinder and not the torus shape of the reference MSFR visible in Fig. 1. As mentioned, the heat extraction is considered as instantaneous.

One has to notice that a specific PK method has to be used usually to model the transients of a molten salt reactor with a liquid circulating fuel, to account for the motion of the precursors. This is not necessary in the present study since the power production has not started and thus the fuel is not yet circulating in the out of the core in the cooling sectors. However the present model does not take into account the natural convection of the fuel inside the core, on the contrary to the model presented below. Finally the calculation time for one transient is of some minutes.

III.B. 3D neutronics-thermalhydraulics coupled code: TFM-OpenFOAM

In the Transient Fission Matrix (TFM) approach described in Refs 3, 4 and 5, fission matrices and average time transport matrices are used to perform neutron kinetics. These matrices characterize the neutron propagation spatial and temporal behavior of the system. They are computed with a single Monte Carlo calculation per core configuration, done prior to the transient calculation. Four fission matrices G are estimated during the same Monte Carlo calculation to take into account the prompt and the delayed neutrons, thus depending on the neutron spectrum (prompt χ_p or delayed χ_d) and the neutron multiplicity (prompt ν_p or delayed ν_d). Thanks to an interpolation of the matrices performed on the fly to follow the system evolution no more Monte Carlo calculation are required during the transient calculation. An interpolation model is thus implemented in the TFM approach.

This interpolation model uses perturbed versions of the matrices. Each matrix is calculated for 3 distinct core configurations: the reference configuration, the configuration with a modified fuel density, and a configuration with a modified fuel temperature (Doppler effect). The interpolation model described in Ref. 3 requires the calculation of the absorption matrices and is based on these two physical considerations: the fission rate depends on the fission position (line of the matrix), and the absorption per neutron emitted are conserved (column of the matrix). Finally, the interpolation on the fly of the fission matrices during the transient evolution is done as follows:

- Knowing the temperature field provided by the thermalhydraulics calculation, the matrix interpolation is performed on each line of the matrices using a linear interpolation for the density and a logarithmic dependency for the Doppler effect.
- Then each column is normalized to guaranty the absorption conservation.

The kinetics equations solved for the prompt neutrons ($\boldsymbol{N}_p$) and precursors of delayed neutrons of each family *i* ($\boldsymbol{P}_f$) are the following:

$$\frac{d\boldsymbol{N}_p}{dt} = \underline{\underline{G_{\chi_p \nu_p}}} \frac{1}{l_{eff}} \boldsymbol{N}_p + \underline{\underline{G_{\chi_d \nu_p}}} \sum_i \lambda_i \boldsymbol{P}_i - \frac{1}{l_{eff}} \boldsymbol{N}_p$$
$$\frac{d\boldsymbol{P}_i}{dt} = \frac{\beta_i}{\beta_0} \left(\underline{\underline{G_{\chi_p \nu_d}}} \frac{1}{l_{eff}} \boldsymbol{N}_p + \underline{\underline{G_{\chi_d \nu_d}}} \sum_i \lambda_i \boldsymbol{P}_i \right) - \lambda_i \boldsymbol{P}_i \quad (5)$$

with $\boldsymbol{l_{eff}}$ the effective prompt lifetime calculated with the time matrix and $\frac{\boldsymbol{\beta_i}}{\boldsymbol{\beta_0}}$ the fraction of delayed neutrons of family *i*. The G matrices are updated at each time step using the interpolation model, based on the fuel salt density and temperature distribution calculated by the thermalhydraulics part of the code.

The fluid flow distribution, the energy, and the delayed neutron precursor distribution are calculated using the OpenFOAM[7] CFD calculation code. It solves the Reynods Average Navier Stokes equation using the k-ε realizable turbulence model[8] that provides good results on this kind of flow configuration with detachment of boundary layer.

The equations of the TFM approach are directly implemented in OpenFOAM in order to simplify the fields exchange between neutronics and thermalhydraulics. The coupling scheme used in this study consists in updating at each time step the reactivity and the flux shape in the core using the interpolation model. Note that the effect of the precursor decay distribution shape on the neutronics is correctly taken into account.

The calculation time of a transient is of the order of one day.

III.C. Comparison of the TFM-OpenFOAM and PK codes on a reactivity insertion transient

To compare the two codes available, a benchmark has been performed on a linear reactivity insertion of 10pcm per second during 75s between the filling and the

divergence step, when the 18m^3 of fuel salt are loaded in the core but no nuclear power is produced. The results are displayed in Fig. 2 (margin to criticality), 3 (produced power) and 4 (mean fuel salt temperature in the core).

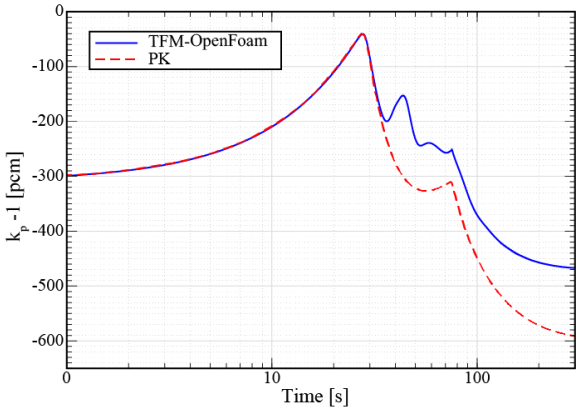

Fig. 2. Evolution of the margin to criticality evaluated with the TFM-OpenFOAM code (blue curve) and with the PK model (red dashed curve)

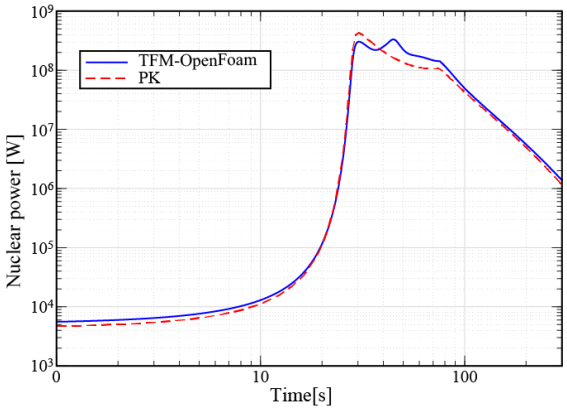

Fig. 3. Evolution of the produced power evaluated with the TFM-OpenFOAM code (blue curve) and with the PK model (red dashed curve)

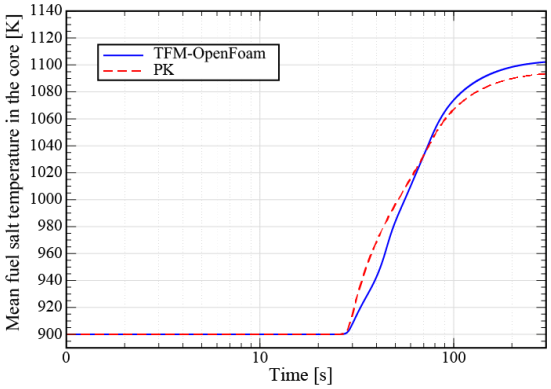

Fig. 4. Evolution of the mean fuel salt temperature in the core evaluated with the TFM-OpenFOAM code (blue curve) and with the PK model (red dashed curve)

The initial power of 1kW considered corresponds to the spontaneous fissions and the (α,n) reactions of the ^{233}U present in the fuel salt.

For the TFM-OpenFOAM calculation, the velocity flow is blocked in the heat exchanger, corresponding to a situation where the reactor is almost full but the circulation is not possible. Thus the heating by the fissions is limited to the fuel salt contained in the core cavity.

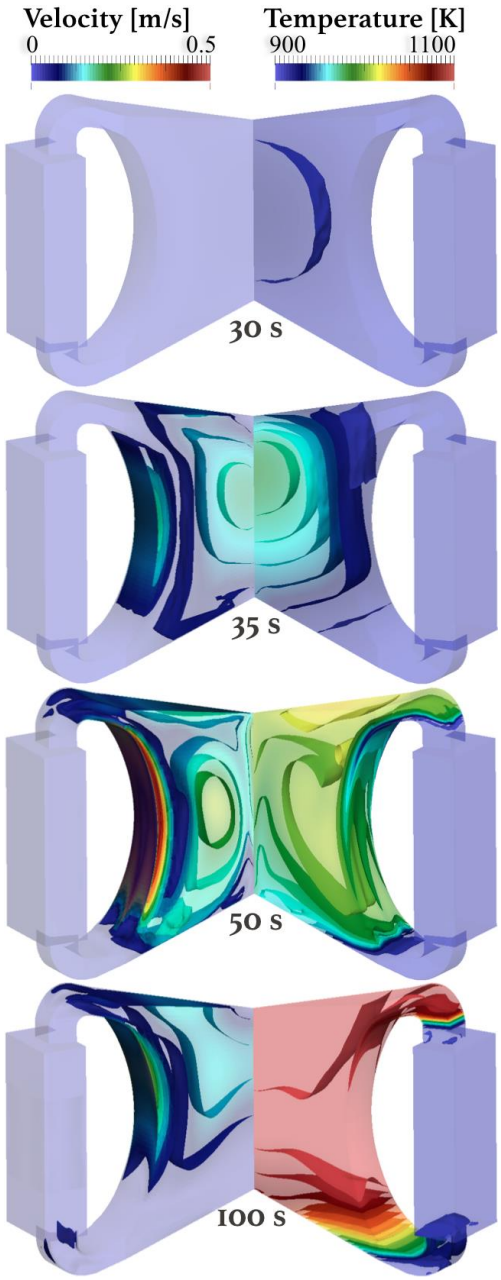

Fig. 5. Evolution during the reactivity insertion of the velocity (left) and temperature (right) distributions with the corresponding eigenvalues, calculated with the TFM-OpenFOAM code.

The only feedback coefficient considered is the Doppler effect in both simulations since this happens during the filling of the core and thus the density dilatation does not lead to a reactivity reduction (further explanations in section V.B).

A very good agreement is obtained on the three variables up to the maximum of reactivity near 30s after the beginning of the transient: there is no feedback from the thermalhydraulics to the neutronics yet because the amount of energy produced is limited (Fig. 5 - top) and then the convection is negligible. The heating of the liquid fuel, maximum in the center of the core volume, leads then to natural convection of the fuel inside the cavity: the fuel salt rise in the center of the cavity and go down near to the wall. This convective phenomenon, illustrated in Fig. 5, is taken into account in the TFM-OpenFOAM code only, which can be observed on the reactivity evolution after 35s. The 3D homogenization explains the discrepancy between the two calculations for the second part of the transient, resulting in a difference of 7 K on the fuel salt mean temperature in the core. The PK model may thus be used for preliminary studies of the first part of such transients since such a simulation tool allows very fast calculations, including the case of non-linear reactivity insertions as these corresponding to the filling of the core (see section IV.B).

IV. APPROACH TO CRITICALITY

IV.A. Reactivity measurement and prediction

This section focuses on two topics important to define the filling procedure and thus the approach to criticality of the MSFR. On a first hand, many factors introduce an uncertainty on the final multiplication factor that will be reached at the end of the filling step of the reactor, especially during the first startup of the reactor. This paper presents an evaluation of the impact of the uncertainties on the fuel salt composition and on the cross section database, which may be large. On the other hand and directly linked to these uncertainties that occur in all nuclear reactors, a measurement of the reactivity all along the reactor filling has to be defined. For this, we have considered a source of 2.5 MeV neutrons, source located in the lower reflector and whose neutron flux is measured in the upper reflector. The fuel salt will amplify this flux. The idea is that the evolution of this flux as a function of the fuel salt volume filled in the reactor will allow the determination of the final multiplication factor when the reactor is partially filled. It will then be possible to implement countermeasures to adjust this final multiplication factor.

IV.A.1. Methodology

This preliminary study is based on Monte Carlo neutronic calculations of the MSFR core only (9 m^3) based on a simple ortho-cylindrical geometry, the objective being to validate the principle of the measurement. The core is radially surrounded by the fertile blanket containing the fertile salt (77.5 ^{7}LiF-22.5ThF_4).

A set of 100 cross sections for ^{233}U and 100 cross sections for ^{232}Th has been used (see ftp://ftp.nrg.eu/pub/www/talys/tendl2014/random.html).

Two quantities have been estimated for each set of cross sections (^{233}U, ^{232}Th):

- The multiplication factor for a fuel salt volume of 1, 2 to 18 m^3 in the reactor. These factors are evaluated using the MCNP neutronic code with 100 inactive cycles followed by active 1400 cycles of 10 000 neutrons, resulting in a statistical uncertainty of around 30 pcm.
- The neutron flux in the upper reflector for a fuel salt volume of 0, 1, 2 to X m^3 in the reactor, where X is the larger volume (upper integer value) for which le reactor is sub-critical. The statistical uncertainty provided by the MCNP code for this neutron flux is around 0.3%.

IV.A.2. Prediction of the final multiplication factor

For each MCNP simulation, the neutron flux in the upper reflector F(V) is determined as a function of V, the fuel salt volume in the core. This flux is compared to the flux F(0) when the core is empty, this last one being due only to the external neutron source located in the lower reflector:

$$R(V) = \frac{F(V)}{F(0)}$$

This ratio R(V) depends on the multiplication factor that will be reached when the reactor is filled. This dependence can be written by the following empirical law:

$$R(V) = A(V)\tan\left(\frac{\pi}{2}\frac{k(V,\{\sigma\})}{k_M(V)}\right)^{1+V/C} - B$$
$$A(V) = \frac{1}{V^2} + \alpha_1 e^{-V/\alpha_2}$$
$$k_M(V) = 1 + D^2\frac{V_M^2 - V^2}{V_M^2 V^2} \qquad (6)$$

where:

- $k(V,\{\sigma\})$ is the multiplication factor for a core filling of V m^3 and the set of cross sections $\{\sigma\}$;
- V_M is the total fuel salt volume (18 m^3) ;
- $k_M(V)$ is the multiplication factor reached when the reactor is filled if criticality is reached for a filling for a fuel salt volume of V m^3.

When the reactor reaches criticality, the neutron flux measured in the upper reflector is diverging, leading to a divergence of R(V) for multiplication factor $k_M(V)$ value. The tangent function used in the empirical formula aims at reproducing this divergence.

As presented in Fig. 6, a fit has been done for the 5 parameters of (Eq.1), the corresponding values are listed in Tab. I.

TABLE I. Parameter values for the fit of R(V)

Parameter	Value
α_1	0.47587
α_2[m^3]	2.31949
B	0.07269
C[m^3]	39.8380
D[m^3]	2.71699

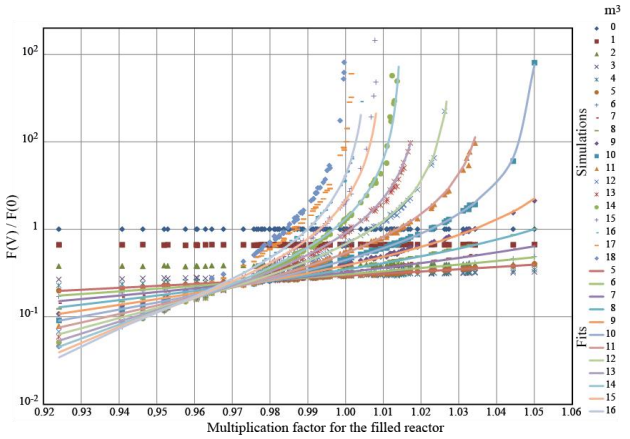

Fig. 6. Fit of R(V): the symbols represent the values of R(V) calculated as a function of the multiplication factor when the reactor is full. Each vertical symbol alignment corresponds to the same set of cross sections of ^{233}U and ^{232}Th. The plain curves are the fit of R(V) evolution as a function of the cross section set.

The dispersion ΔR of the points around the fit averages 5.6%. It is due to cross section effects and is thus representative of the uncertainty on the estimation of the multiplication factor at the end of the reactor filling. This dispersion is estimated for each volume V as presented in Tab. II.

TABLE II. Value of the dispersion ΔR as a function of the fuel salt volume V

V [m^3]	5	6	7	8	9	10
ΔR [%]	1.60	2.31	2.66	2.78	2.92	4.08
V [m^3]	11	12	13	14	15	16
ΔR [%]	4.85	4.38	7.22	12.5	15.2	6.70

The uncertainty of the multiplication factor at the end of the filling is then given by:

$$\Delta k = \frac{\partial k}{\partial R}\Delta R = \left(\frac{\partial R}{\partial k}\right)^{-1}\Delta R$$

When the multiplication factor is equal to 1 at the end of the filling, $\partial R/\partial k$ is given by:

$$\frac{\partial R}{\partial k}(V) = \frac{\pi A(1+CV)}{2k_M}\frac{\tan(\pi/(2k_M))^{CV}}{\cos(\pi/2k_M)^2}$$

Finally the distribution obtained for Δk as a function of the fuel salt volume takes the form:

$$\Delta k = \Delta k_0\left(1-\frac{V}{V_M}\right)^2$$

with Δk_0 = 1889 pcm.

As mentioned at the end of section II.B.2, reactivity corrections up to some hundreds pcm may be done either quickly by changing the fuel salt temperature, or more slowly by modifying the composition of the fuel salt injected in the reactor. We thus aim at predicting the final multiplication factor with a precision Δk of 250 pcm.

Finally we deduce from the Δk distribution above that such a precision on the multiplication factor prediction can be reached when the reactor contains at least a fuel salt volume of 11.5 m^3. A complementary study has to be performed including also an internal neutron source in the fuel salt due to the neutron production by the spontaneous fissions and the (α,n) reactions.

IV.B. PK study of an excess of reactivity insertion during the filling step

This section is dedicated to a preliminary study of the MSFR behavior when criticality is reached before the end of the reactor is filled and the filling still goes on. It is based on the PK model presented in section III.A.

The filling from 1 to 18 m^3 is distributed half in the core and half in the recirculation sectors The reactivity variation is evaluated with the following empirical law, adjusted on MCNP calculations:

$$k(V) = A\,\mathrm{th}(BV) + CBV$$

with V the volume of fuel salt injected and:

$$A = \frac{1-CBV_{k=1}}{\mathrm{th}(BV_{k=1})}$$
$$B = aA^2 + bA + c$$
$$a = -0.032048$$
$$b = 0.017759$$
$$c = 0.227166$$
$$C = 0.00946559$$

The core reactivity as a function of the fuel salt volume injected is expressed as:

$$\rho(V) = \frac{A\tanh BV + CBV - 1}{\tanh BV + CBV}$$

We study here the accidental transient due to a fuel salt injection in the core when the reactor is critical before the end of the filling. One has to notice that the corresponding reactivity injection as a function of the

volume is not linear here, on the contrary to the benchmark presented in section III.C, and as for this benchmark, only the Doppler feedback effect is efficient here.

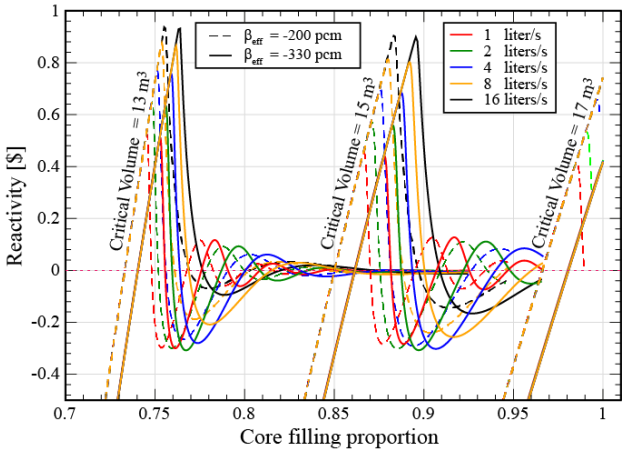

Fig. 8. Evolution of the reactivity in $ during the filling as a function of the criticality volume and of the filling rate. Two values have been considered for β_{eff}, the effective fraction of delayed neutrons, to study the impact of this parameter.

A sensitivity study has been done for different cross section sets, injection rates and values of the effective fraction of delayed neutrons. Fig. 8 presents the reactivity evolution for these different cases. Even for the smaller values of the criticality volume (13 m^3) and β_{eff} (200pcm), the reactivity stays below 1$. The reactivity increase is all the larger that the filling rate is high. Indeed during a slow filling, the feedback effects are able to compensate the reactivity addition during the injection.

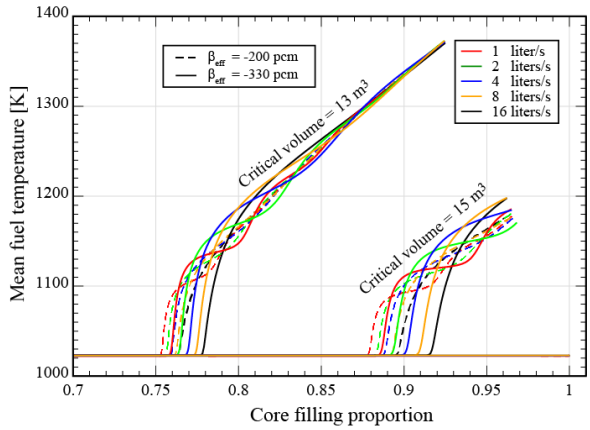

Fig. 9. Evolution of the mean fuel temperature in the core cavity during the filling a function of the criticality volume and of the filling rate, for two values of the effective fraction of delayed neutrons β_{eff}.

The corresponding evolutions of the mean fuel temperature are shown in Fig. 9. As expected, the maximum temperature reached for a criticality volume of 13 m^3 is too high. One has to remind that these simulations are based on a pure neutronic PK model with adiabatic hypotheses thus without any cooling by the walls or any thermalhydraulics effects as natural convection, these results are thus pessimistic.

In the case of a slow filling, the temperature increase will last around 700s, leaving the time to drain the fuel in the emergency draining system if necessary to protect the materials, which is not the case for the fastest filling considered here of 16 l/s corresponding to a heating duration of around 30s. Regarding this kind of abnormal transients, a slow filling is safer. The time to fill the core will be a compromise between the longest time to allow feedback effect stabilization and the shortest time to avoid a too large internal heating of the fuel salt due to the residual power (not simulated here), since the salt can be cooled by the heat exchangers as long as the reactor is not fully filled and thus the fuel circulation by the pumps is not possible.

V. ABNORMAL TRANSIENT STUDIES DURING THE DIVERGENCE STEP

Two cases will be presented in this section, based on calculations performed with the coupled 3D neutronics-thermalhydraulics code TFM-OpenFOAM details in section II.2. The over-cooling incident and its consequences on the reactor, already published in Ref. 9, are briefly described. Then the most likely scenario of a reactivity insertion at low power is presented and analyzed.

V.A. Over-Cooling Incident

The ramp up procedure may lead to an over-cooling incident, equivalent to a reactivity insertion due to the negative feedback coefficients, leading to a prompt critical situation. In case of a sudden increase of the heat extraction, the temperature of the fuel salt arriving in the core decreases and the reactivity increases while the power level is very low and thus the thermal feedback effects are not efficient.

Such an accident has been analyzed in Ref. 9. A parametric study of an over-cooling incident from 1 kW to 3 GW is illustrated in Fig. 10. A variation of the intermediate fluid temperature with a time constant from 0s (no inertia) to 128s has been considered to take into account the inertia of the intermediate circuit. The prompt critical situation is avoided for time constants larger than 32 s which is realistic. A very important point for the safety analysis of such reactors has to be noticed here: there is no cliff edge effect in the reactor behavior when it reaches prompt criticality. The reactivity and thus power and temperature increase is absorbed by the system as an expansion of the liquid fuel salt without any core damages,

such an effect being well known and used in systems like the SIRENE reactor[10,11].

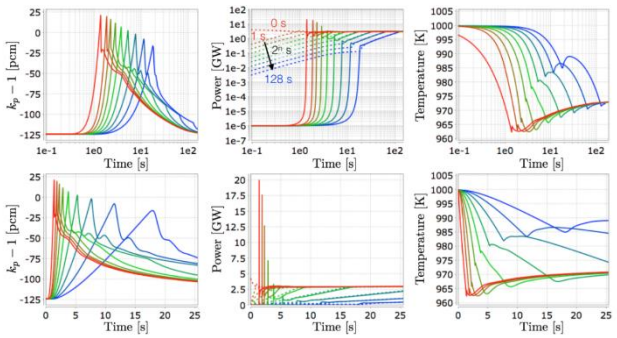

Fig. 10. Time evolution of the margin to prompt criticality, of the power and of the mean fuel salt temperature for a 1kW to 3GW over-cooling incident with a time constant of the intermediate fluid from 0 to 128s.

V.B. Reactivity Insertion Incident

Unwanted reactivity insertions at low power may occur during the filling procedure. A study of the effect of nonlinear reactivity variations and volume variations during the filling step have been presented in section IV.B. We propose here to study the effect of a reactivity insertion between the filling and the divergence steps, when the reactor is almost full but still without any fuel circulation out of the core cavity, in the heat exchangers. In this configuration the density effect may be not effective, because this effect assumes that the fluid dilatation reduces the amount of fissile mater in the core. If the core is not full, even if a fuel dilatation occurs in the reactor, the total amount of fissile matter stays in the core cavity. The following scenarios have thus been calculated for both "Doppler" and "Doppler+density" feedback configurations. In order to study different reactivity insertion rates, the amount of 1000pcm (corresponding to a bounding case of the reactivity margins of the system) have been inserted in 10, 33, 100, 333 and 1000 seconds. Note that the initial margin to the prompt criticality is smaller than the effective fraction of delayed neutrons: the reactor is initially subcritical.

Fig. 11 presents the time evolution of the margin to prompt criticality, of the nuclear power and of the maximum fuel salt temperature during the reactivity insertion, with and without the density feedback effect.

We can see that even with an injection rate of 33 pcm/s, the reactor remains below the prompt criticality. Depending on the total feedback coefficient, the final maximum temperature differs. Without the density feedback, the final temperature reached is around 1100 K instead of 1000 K. This maximum temperature provides a useful information, as illustrated in Fig. 5 (case at 100s - bottom): the temperature is almost uniform in the upper part of the core, and it corresponds to the temperature seen by the structural materials.

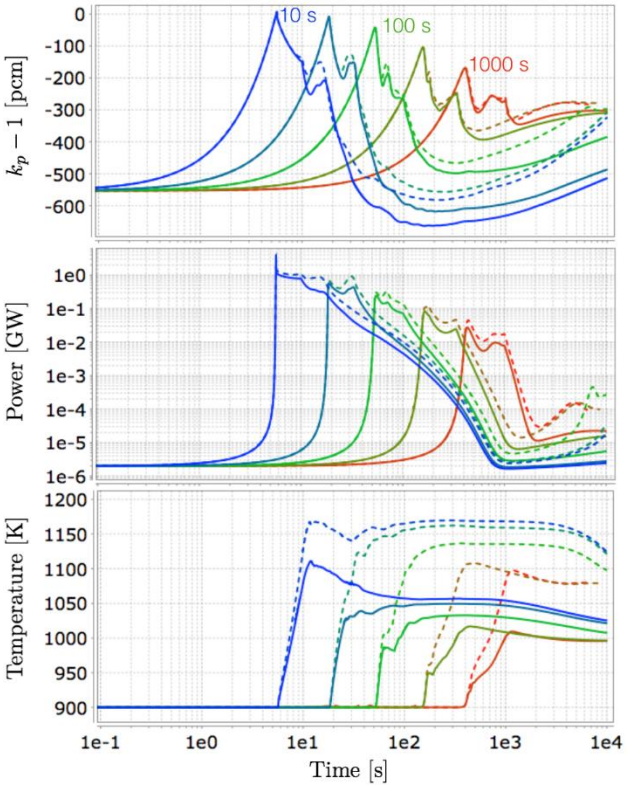

Fig. 11. Time evolution of the margin to prompt criticality, of the produced power and of the maximum fuel salt temperature for a reactivity insertion of 1000 pcm with an injection time of 10, 33, 100, 333 and 1000 seconds, with the Doppler+density feedbacks (plain line) and Doppler only (dashed line).

If the reactivity insertion rate is high, the energy released is more important that the amount required to make the reactor sub-critical and the temperature is higher. For a long time scale, the slow thermal diffusion between the bottom of the core and the center of the core where the neutron importance is higher induces a reactivity increase.

This study highlights that the reactor behavior during a reactivity insertion at low power is very good, even with a high insertion rate. The temperature variation is directly proportional to the amount of reactivity inserted. And if the reactor becomes prompt critical, the excess of energy released during the power peak is very limited. Finally, we can note that the Doppler effect alone is sufficient to stabilize the reactor if the core cavity is not full and the density effect is not efficient.

IV. CONCLUSIONS

Molten salt reactors with a liquid circulating fuel, like the MSFR concept developed initially at CNRS and now at the center of the SAMOFAR European project, are very

different in terms of design and safety approach compared to solid-fueled reactors. The definition of their operation procedures, directly linked to the development of the system design and of the safety analysis, is one of the objectives of the SAMOFAR project. Studies of the startup procedure of the MSFR have been presented in this paper, based on one hand on simple and very fast point kinetics calculations for some systematic studies and on the other hand on a code developed initially for such liquid-fueled reactors and coupling the TFM Monte Carlo based approach to the OpenFOAM CFD code to have precise 3D calculations of the reactor in a very reasonable computing time. These studies show that the reactor behavior during abnormal transients is very good and that it is feasible to start the reactor without control rods while controlling the reactivity.

Among the perspectives, a system code dedicated to the MSFR study and optimization is under development in the frame of the SAMOFAR project. Such a simulator will be used to define more precisely the operation procedures of this kind of reactors and will allow their optimization in terms of system efficiency and safety.

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