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Design and optimization of a Chloride Molten Salt Fast Reactor

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Abstract – *This paper presents several studies for the design and optimization of a $3GW_{th}$ breeder Molten Salt Fast Reactor using chloride salts and the uranium cycle (Cl-MSFR). Results lead to a configuration of $45m^3$ of $NaCl\text{-}^{dep}UCl_3$-(spent UOX fuel TRUs)Cl_3 for the fuel salt, along with a fertile blanket of 80cm of width filled with $NaCl\text{-}^{dep}UCl_3$ in which a small amount of spent MOX fuel TRUs is dissolved for spent MOX reuse and proliferation resistance. A reprocessing scheme dedicated to the cleaning of these chloride salts is proposed. Finally, preliminary studies on Cl-MSFR deployment don't show any constraints on matter availability and present a good synergy with PWRs.*

I. INTRODUCTION

Current commercial reactors provide large amount of electricity at an affordable price. However, the technology it relies on, mainly light water reactors, faces difficulties. First, its durability is challenged for two reasons. It mostly relies on the only fissile material available in ores, ^{235}U that account for 0.7% of natural uranium. This could raise an issue of fuel availability should nuclear energy be massively developed to partially substitute fossil fuels. It also faces strong limitations for minor actinide conversion, that lead to a heavier cost of waste management. Second, following Fukushima accident, safety requirements were raised to levels such that cost and design complexity increased significantly. One way to tackle these challenges is to look at other reactor concepts.

Molten Salt Reactors are one of the six options selected by the Generation IV International Forum to focus studies on for the renewal of nuclear reactors [1], with the Molten Salt Fast Reactor as one of its concepts of reference. MSRs have the unique feature of a liquid fuel, that can also act as the coolant. The most studied MSFR, called reference MSFR in this paper, is a $3GW_{th}$ breeder using the Th/U cycle with LiF salt as solvent [2]. It shows promising performances such as an excellent safety on operation, due to large negative feedback coefficients. Nevertheless, it is interesting to study its possible alternative based on the only other nuclear fuel cycle, the U/Pu cycle that is currently used in the industry. This paper details the optimization of this reactor concept that uses chloride salts, so called Cl-MSFR, to close the uranium fuel cycle. Three aspects of the work are discussed here: the optimization of the fuel circuit (which comprises the core, the recirculation loops containing the heat exchangers, the expansion vessel, the fertile blanket and the neutronic protections), with static and depletion calculations; the reprocessing scheme associated to this MSFR version and thus dedicated to clean chloride salts; the impact of Cl-MSFR deployment in a nuclear fleet.

Following this introduction, part II details the methodology and tools, and part III presents the results and discusses them. As the fuel circuit optimization process makes use of the reprocessing scheme, part II and part III are divided in three sections in which reprocessing scheme is presented first, followed by the optimization of the fuel circuit and lastly by scenarios.

II. Methodology & tools

II.A. Reprocessing scheme

In an MSR, thanks to the liquid form of the fuel which is a molten salt, processing the fuel during reactor operation is possible with processes located on the same site as the reactor. The reprocessing scheme must address as much as reasonably achievable following constraints. Excess of gases accumulating from gaseous fission product formation should be extracted, to avoid pressure increase during operation. Non-soluble elements should be removed to avoid damages on pumps or structures. Impurities susceptible either to make soluble elements precipitate, to enhance corrosion processes, to reduce neutron economy (by parasitic captures or unwanted flux thermalization), or to change the thermal properties of the salt, should be extracted. Heavy nuclei must remain in the fuel. The conclusion is that any element produced in significant quantity (actinides and solvent excluded) should be removed.

The main form (saline, precipitate, gas) of each element can be postulated assuming a known and constant redox

potential of the salts using thermodynamics data [3]. Additionally, by performing a first depletion calculation using tools that will be presented in paragraph II.B, one can have an estimate of the produced mass of each element, to discard ones with a very low production rate. Relevant processes of extraction are then chosen.

II.B. Fuel circuit optimization

The optimization of the fuel circuit is an iterative process to account for constraints on neutronics, heat extraction, chemistry, proliferation resistance. Table I gives an overview of said constraints applied on the main parts of the circuit (BR stands for breeding ratio).

TABLE I

Constraints for optimization of the fuel circuit

Constraint source	Fuel salt	Fertile salt	Structure
Neutronics	Criticality Solvent transparency $BR_{fuel} < 1$	BR maximization	Neutron transparency, or opacity for protections
Heat extraction	Volume in HX		
Chemistry	Eutectic mix	Eutectic mix	Corrosion resistance
Proliferation resistance	Enrichment limitation	Pu vector quality	
Others	Inventory minimization Compacity	Inventory minimization	Compacity

The circulation of the fuel in heat exchangers will lead to the decay of a part of the precursors outside the core, thus reducing the delayed neutron fraction contributing to the chain reaction. Although no absolute threshold exists, it should be verified with coupling codes (neutronics + thermohydraulic) that the effective, circulating β noted $\beta_{eff,circ}$ is sufficient for the reactor not to be at risk of strong power fluctuations nor prompt criticality transients especially for overcooling incidents [4].

The choice of the solvent is first driven by the heavy nuclei to be used (either fissile or fertile). Transuranic elements have a relatively low solubility in fluoride salts while chloride salts can dissolve large amounts [5]. Other halogens are discarded as they are opaquer to neutrons and less attractive for processing. Additionally, chlorine must be enriched in ^{37}Cl to avoid the production of radioactive ^{36}Cl by (n, γ) reaction on ^{35}Cl. For alkali or alkaline earth metals, sodium is preferred for its low capture cross-section. This led to the NaCl-UCl_3-$(TRUs)Cl_3$ salt (TRUs standing for TRansUranic elements) being used for both fertile and fuel salt, differences between the two being the relative proportion of $(TRUs)Cl_3$ vs UCl_3 and the TRU vector.

Heat extraction and heavy nuclei inventory minimization constraints lead to finding the minimal salt volume in heat exchanger for which the nominal power can be extracted, provided that the breeding ratio of the fuel only is below 1 (otherwise reactivity will increase over time). To assess the heat extraction capability of a given configuration of the circuit, the in-house CNRS code SONGe, a multiparameter constraint-driven optimization code based on a genetic algorithm, is used. To be able to compare Cl-MSFR to reference MSFR, the same constraint on the electrical power required by pumps for heat extraction is taken.

Criticality is obtained by adjusting the $(TRUs)Cl_3$ proportion in the salt. In order to have an eutectic mix (to avoid preferential precipitation during cooling), TRUs are assumed to have the same chemical behavior as plutonium, and the ternary diagram of NaCl-UCl_3-$(TRUs)Cl_3$ (see Figure 1) is used [5] to adjust the proportion of NaCl-UCl_3.

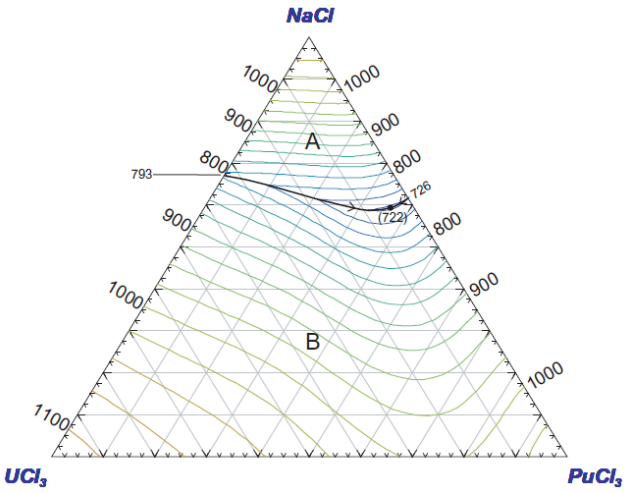

Fig. 1. Ternary diagram for NaCl-UCl_3-$(TRUs)Cl_3$ [5]

Static calculations are done using the transport code Serpent2 [6] using ENDF-B7.1 as the nuclear data library of reference. The feedback coefficients are obtained by calculating k_{eff} at different temperatures (600K, 900K, 1200K), fitting and taking the local derivative.

Depletion calculations considering the processing are made with the in-house CNRS code REM [7], that uses MCNP to solve the Boltzmann equation, and its own procedure to solve the Bateman equations to perform the evolution of all the matters during operation. Among other things, it makes possible mass balance computation and Pu vector quality tracking in the fertile salt.

II.C. Scenarios

To perform scenario studies, in-house CNRS ISF code is used [8]. Simplified descriptions of the reactors (electrical power, load factor, plant lifetime, ...) are provided, and inventories are tracked along time. Scenarios are driven by the power demand and reactors are ranked in terms of preference. Assuming n reactor types (PWR with UOX fuel, RNR-Na with MOX fuel, ...) ordered in a list where number n is the most wanted reactor, for each time step (usually a year), the process is as follow*:*

- the power demand is met by the current fleet: go to next time step;
- otherwise, try to start reactor n with available stockpiles;
- if required matters to start reactor n are not available, try to process them with factories (U enrichment facilities, MOX fabrication plant, ...);
- if reactor n cannot be started, try to start reactor n-1 and repeat;
- if no more reactors can be started and the demand is still not met, go to next time step.

In this study, scenarios are not meant to be precise but to assess deployment feasibility based on matter flows, and for general comparison purposes.

III. Results

III.A. Fuel reprocessing scheme

The principle diagram of the reprocessing scheme is presented in figure 2.

The main steps are following.

- Gas swipe and fuel tapping are performed in the vessel expansion in order to avoid in-core gas injection.
- Gases and volatile species are first sent towards cold traps to isolate the latters, then a part of the formers is processed with advanced techniques.
- Metallic particles are adsorbed on solid metallic traps. It is possible assuming that the fuel circuit is coated in a non-metallic material. Otherwise they are likely to be adsorbed in heat exchangers that present a significant area, ultimately leading to clogging and/or damages.
- Lanthanides can be extracted by electrolysis with separate compartments, but actinides must be extracted first because of the respective electrochemical potentials. Formers are then oxidized, they precipitate and are filtrated out. The latters are reinjected in the core.
- Advanced processing of fission products can be performed off-site, using aqueous chemistry. On-site chemical treatment can only use pyrochemical processes due to the high temperature and radiolysis risks for compounds at low temperature.

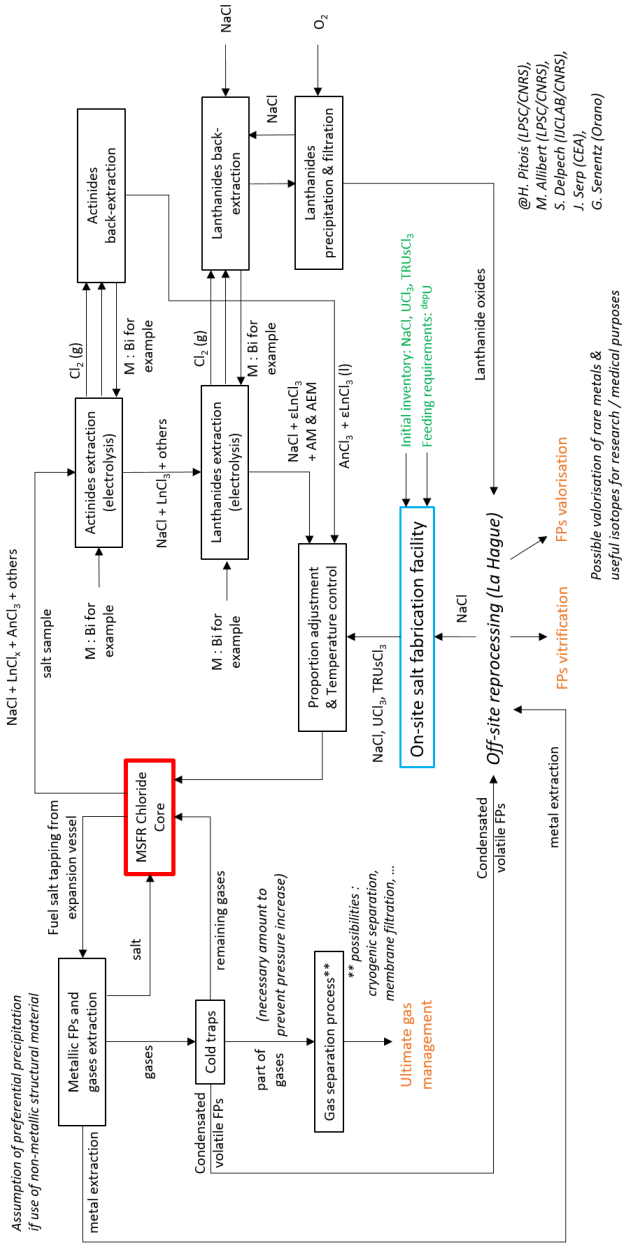

Fig. 2. Principle diagram for the Cl-MSFR reprocessing scheme

This reprocessing scheme is a first principle version built upon hypotheses on the chemical form of species that should be experimentally verified.

III.B. Fuel circuit optimization

For a better understanding of the following paragraphs, figure 3 presents the fuel circuit geometry used for static neutronic calculations.

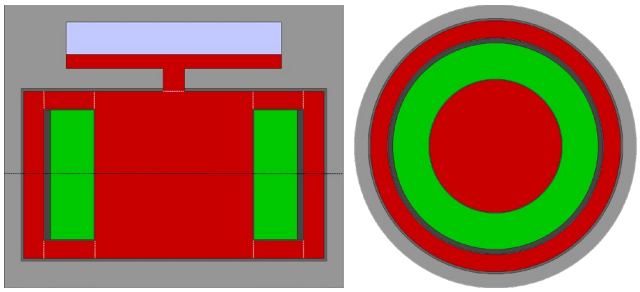

Fig. 3.: Model (Serpent2) of the fuel circuit for transport codes
Legend: fuel salt in red; fertile salt in green; protection/envelope in dark grey; structure in grey; inert gas in light purple.

The fuel is distributed in the core, the "heat exchangers" being the section behind sectors, plenums being sections in between core and fuel heat exchangers, and the expansion vessel above the core.

III. B. 1. Static calculations

Materials

An important question for molten salt reactor design is the choice of materials, especially structural materials in contact with the fuel salt. The first reason is that there is hardly any material qualified for this application. Hastelloy N, also known as INOR-8, was tested and validated during the exploitation of the MSRE [9]. It is also considered for the reference MSFR. But no material was ever validated for liquid fuels based on chloride salts. The second reason is, as mentioned in paragraph III.A, that the use of metallic alloys will lead to metallic fission product adsorption especially in heat exchangers.

For this study, we assume that the surface in contact with the fuel salt is coated with SiC. The structural material considered here is SS316H. Due to the difference in mechanical properties of the two materials, further work should be done to solve this issue; this is only a first choice to start the design optimization of the system in parallel to the development and experimental studies on materials

To protect heat exchangers from the neutron flux of the core, a 10cm layer of B_4C has been added. Additional studies considering precursor decays in the heat exchangers are being conducted to refine this choice.

Compositions and densities for SiC, SS316H and B_4C are taken from reference [10].

Finally, chlorine is considered to be enriched chlorine made of 99% of ^{37}Cl and 1% of ^{35}Cl to avoid the production of radioactive ^{36}Cl by (n,γ) reaction [11] and to decrease sulfur production [10].

Heat extraction

To assess the heat extraction capability of the system, we apply an iterative process using SONGe code and the neutron transport code Serpent2.

The total power is set at 3 GW_{th}. A total volume of fuel salt is defined as well as the proportion of salt in the central zone (core). Assuming a given total volume, the highest the proportion of salt in the core, the highest the $\beta_{eff,circ}$ the lowest the heat extraction capacity, so a compromise must be found. Studies on reference MSFR suggested that keeping a ratio close to 50% in-core -50% out-core is interesting so this hypothesis is kept here. The physical properties of the salt (heat capacity, density) varies when the composition is adjusted to reach criticality with Serpent2 for the configuration. This data is given to SONGe. While the electrical power required for the pumps to extract the heat is below the arbitrary chosen value of 2% of the total electrical power (~45% of 3 GW_{th}), a smaller total volume is taken for inventory minimization and the process repeats.

The result of this study shows an optimal for 45 m^3 of fuel salt: 20 m^3 in the core, 18 m^3 in heat exchangers and the remaining 7 m^3 in plenums. During the study it was found that the core volume should not exceed 20 m^3 to avoid excessive breeding in the fuel, leading to reactivity increase over time. It should also be added that the plenum volume represents a fuel that circulates in pipes of a given hydraulic diameter, optimized by SONGe but simplified for the geometry used by neutron transport codes.

This is a significantly larger volume of fuel when compared to the reference MSFR, mainly due to chloride salts being relatively worse as coolant.

Fertile blanket width

The fertile blanket width is the result of the compromise between BR maximization (towards large volume) and inventory minimization. Figure 4 presents the total capture rate on ^{238}U in the fertile salt versus the width.

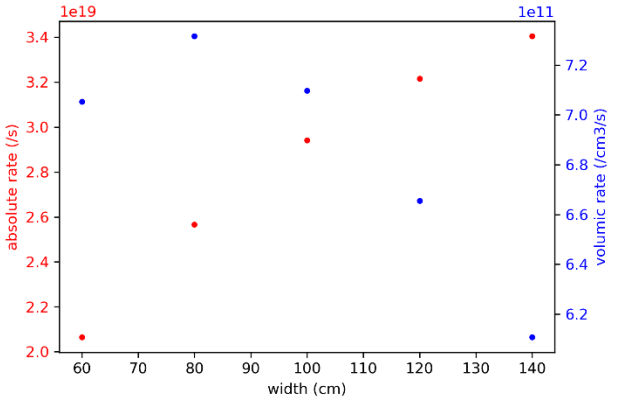

Fig. 4. Capture rate on ^{238}U in the fertile salt depending on fertile blanket width.

Looking at the volume rate of capture, there is an optimum width of 80 cm, for a volume of 21.7 m^3. This is again a larger volume when compared to the reference MSFR for which a fertile blanket width of 46 cm (7.3 m^3) is considered.

One way to reduce the inventory without diminishing too much the BR would be to replace a part of the fertile salt with a moderator, and aim to compensate the volume loss with the higher cross section given by the flux moderation. To maximize the impact of the moderator, several alternances of fertile salt and moderator could be arranged. Preliminary studies using MgO as the moderator were performed (properties of MgO taken from [10]). Figures 5 and 6 show the neutron spectrums along the radius at middle height for two configurations: the one without moderator, and one with 20 cm of fertile salt followed by 6 alternances of MgO and fertile salt layers of 5 cm each.

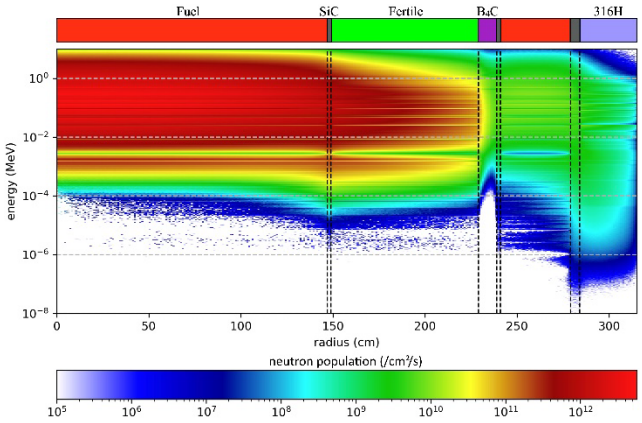

Fig. 5. Neutron spectrum without moderator for Cl-MSFR

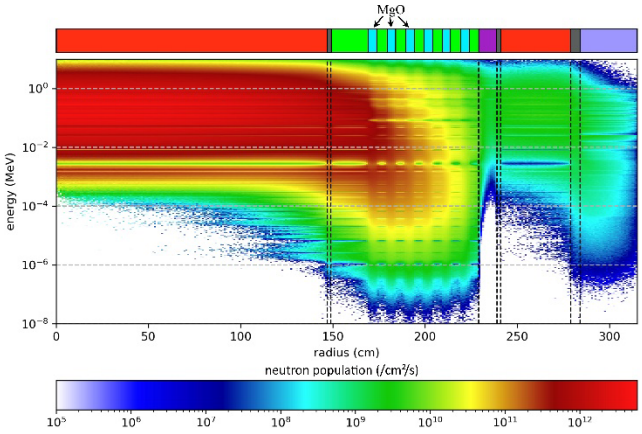

Fig. 6. Neutron spectrum with layers of moderator for Cl-MSFR

Figures 5 and 6 show the neutron spectrum maps, which correspond to the neutron spectrum along the ordinate as a function of the radius along the abscissa. These maps are estimated at middle height of the core for two configurations: the one without moderator, and one with 20 cm of fertile salt followed by 6 successive layers of MgO and fertile salt of 5 cm each. Due to the MgO layers, the neutron spectrum is thermalized (shifted to the bottom right in figure 6 compared to figure 5) and the thermal component is efficiently absorbed in the fertile regions in between the MgO layers. Table II gives the absolute and volume capture rates on ^{238}U for the two configurations, as well as the reactivity difference.

TABLE II

Capture rate on ^{238}U for the two configurations

	Without moderator	With layers of moderator
Δk_{eff}	/	+852(1) pcm
total capture rate on ^{238}U	2.45E+19	2.83E+19
volume capture rate on ^{238}U	1.13E+12	2.14E+12

The difference is significant: although the volume of fertile is reduced from 21.7 m^3 to 13.2 m^3, the total capture rate increased by 15%. To further finalize the optimization of the fertile blanket, more complex geometries are under implementation in the depletion calculation code used here. The first optimized configuration with 80 cm of fertile salt and no moderator is thus kept for the following.

Fuel salt initial composition

We seek for the Cl-MSFR to complement a nuclear fleet of LWR reactors (assuming PWRs here). Its main fissile sources should be spent UOX fuel TRUs and spent MOX fuel TRUs, associated with depleted uranium (0.2% of ^{235}U). To be more exhaustive, enriched uranium was also considered. The TRU vectors used for this study are taken from this reference [10] (p204, "PAU" and "PAM" for spent UOX and spent MOX fuel TRUs). The compositions to reach criticality are given in table III assuming the configuration of 45 m^3 of fuel salt and a fertile blanket width of 80 cm containing 0.5% of (TRUs)Cl_3 from spent MOX fuel (see paragraph *III.A 2.*).

TABLE III

Initial salt compositions adjusted for criticality

	NaCl (mol%)	UCl_3 (mol%)	(TRUs)Cl_3 (mol%)	density (g/cm^3)
Spent UOX fuel	65.7	28.4	5.9	3.20
Spent MOX fuel	65.1	26.8	8.1	3.22
^{enr}U (e = 15.8%)	67.0	33.0	0	3.15

Feedback coefficients

For MSFRs, two physical effects are important to stabilize the chain reaction in case of temperature modification, represented by two coefficients: the density coefficient (increase of neutron leakage with density decrease) and the Doppler effect (microscopic cross-section variation with temperature). The fuel being also the coolant, the density coefficient is also the void coefficient. Table IV presents the feedback coefficients (with statistical uncertainty) for the initial compositions of table III.

TABLE IV

Initial feedback coefficients for the different possible fuels

	Doppler (pcm/K)	Density (pcm/K)	Total (pcm/K)
Spent UOX fuel	-0.7 (0.2)	-20.8 (0.2)	-21.5 (0.4)
Spent MOX fuel	-0.4 (0.2)	-19.4 (0.2)	-19.8 (0.4)
^{enr}U (e = 15.8%)	-0.8 (0.2)	-20.9 (0.2)	-21.7 (0.4)

When compared to the reference MSFR for which both coefficients are close to -4 pcm/K [12], the Doppler effect is smaller in the case of the Cl-MSFR while the density effect is larger. The harder spectrum and differences in the fuel explain the former, while a larger dilatation coefficient as well as a harder spectrum explain the latter. Having a small Doppler coefficient could prove to be problematic during transients where the density effect cannot act as intended, such as during the filling of the reactor or in case of a strong reactivity insertion. Studies on this second point can be found in this reference [13].

The effective proportion of delayed neutrons β_{eff} without considering the circulation of precursors equals ~375pcm. Studies are undergoing for the computation of $\beta_{eff,circ}$.

Expansion vessel positioning

As the fuel volume will change with its temperature, a free volume is required to accommodate this variation, and also to collect the gaseous fission products. However, the expansion vessel should not have a significant contribution to the chain reaction, especially for the density feedback to be effective. The closer the extra volume is to the core, the higher its contribution is to the chain reaction as less neutrons leak out, so the smaller the feedback is. Additionally, the circulation will be reduced in this volume so the amount of fissions occurring there must be minimal. Finally, its exact geometry will have to be optimized using CFD codes to ensure a sufficient mixing [4] but it is outside this work. Here we assume the following hypotheses.

- The geometry is kept simple, being a cylinder of an arbitrary radius of 200 cm centered with the core.
- Its total volume can accommodate the full fuel salt volume increase, assuming an arbitrary maximum temperature of 1100K.
- At normal operation, it is filled with the fuel volume difference between the salt at operating temperature of 900K, and the salt at freezing temperature of 790K.
- One configuration is deemed acceptable if the fraction of total fission power in the vessel is below 1%.

Systematic studies using Serpent2 code lead to a minimum distance of 40 cm with this configuration. This distance could be reduced if necessary by adding a layer of absorbent material between the core and the expansion vessel. It is also possible to further increase the vessel volume should it be able to accommodate a fuel salt at a higher temperature without a significant impact on the neutronics, should this minimal distance be kept.

III.A 2. Depletion calculations with processing

Depletion calculations shown in this study are performed with the following hypotheses.

- k_{eff} is maintained at ~0.998 (1 - β_{eff}) using feeding of the fissile matter, with the same vector as the one used for initial composition.
- Heavy nuclei fraction is kept constant by feeding with fertile matter, depleted uranium here.
- Extractions are 100% efficient.
- Considering the challenge to isolate alkali and alkaline-earth elements from sodium, these elements are not extracted.

The first parameter to adjust is the processing flowrate for the fuel salt. This rate should be sufficient to avoid a significant shift of the mass of the fuel, taking into account the feeding of TRUs and depleted uranium to cover their loss by fission or capture. The figure 7 shows the evolution of masses of fission products. To avoid unnecessary constraints on the fuel treatment unit, the lowest rate satisfying the condition is chosen. However, considering a removal efficiency of 1 for all elements in the reprocessing, it should be interpreted as a minimum.

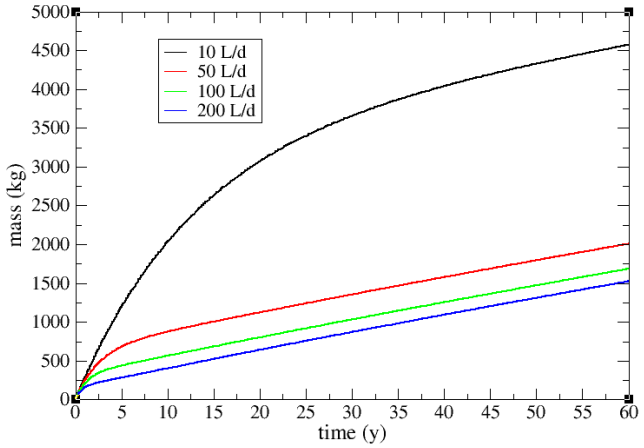

Fig. 7. Evolution of fission product masses in the fuel salt for different reprocessing flowrates.

Looking to masses, not much gain is obtained by increasing the flowrate above 50L/d. The mass increase is mainly driven by accumulation of alkali & alkaline-earth metals that are not extracted. At 60 years, barium, cesium, strontium accounts for respectively 44%, 14% and 10% of the 2010 kg of fission products in the fuel salt for this flowrate of 50L/d. Sodium represents about 90% of alkali & alkaline-earth metals after 60 years of operation. It could be considered to process the solvent entirely every 10 or 20 years to avoid chemistry issues. Nevertheless, this study suggests that a reprocessing flowrate of at least 50L/d for the fuel salt is required (100L/d seems more reasonable).

The second parameter to be optimized is the processing flowrate of the fertile salt. To avoid the production of weapon-grade plutonium and better complement PWRs, a small fraction of TRUs coming from spent MOX fuels is introduced. During operation, the Pu vector of the fertile salt will see its share of ^{239}Pu increase with the conversion of ^{238}U, and stabilizes at a value that depends on the reprocessing flowrate. Figure 8 shows the evolution of the ^{239}Pu share in the Pu vector of the fertile salt for different configurations.

The objective is to extract a plutonium that can be reused to produce "fresh" MOX fuel, while complying regulations about proliferation. Already existing spent UOX fuels of PWRs with a low burn-up can have up to ~60% of ^{239}Pu in their Pu vector, so it was decided to set this value as the maximum threshold. According to this criterion, two configurations are interesting here: either [0.5%mol (TRUs)Cl_3 \& 100L/d] or [1%mol (TRUs)Cl_3 \& 50L/d]. The choice depends then on the acceptable fission power in the fertile salt and the constraints on the fuel treatment unit. To minimize the fission power, the configuration [0.5%mol (TRUs)Cl_3 \& 100L/d] is more interesting as it reduces the fraction of total fission power produced in the fertile salt from 4.9% to 3.3%.

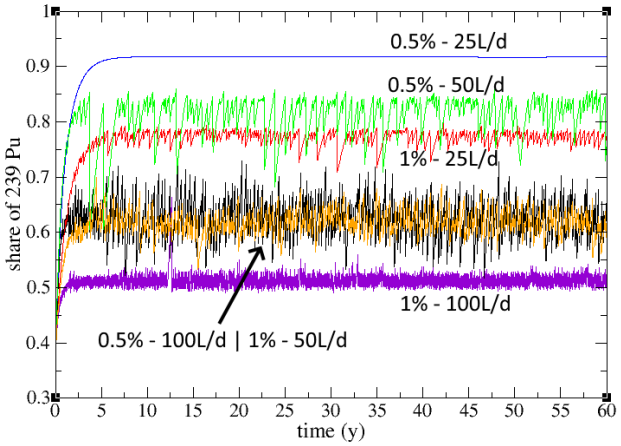

Fig. 8. Evolution of 239Pu share in the Pu vector of the fertile salt depending on the fraction of (TRUs)Cl_3 and the reprocessing flowrate

Finally, flows for actinides and elements of concern is given in table V, for a depletion calculation with this configuration: 45m^3 of fuel (started with spent UOX TRUs), 0.5 mol% of (TRUs)Cl_3 (spent MOX TRUs) in the fertile salt, and respective processing flowrates of 50L/d and 100L/d.

TABLE V

Flows of some elements of importance for one optimized configuration of Cl-MSFR

Element	input (fuel) (kg/y)	output (fuel) (kg/y)	input (fertile) (kg/y)	output (fertile) (kg/y)	net output (total) (kg/y)
U	1127.5	0	363.3	5.18	-1485.6
Np	1.8	0	43.0	45.6	0.74
Pu	30.4	0	438.6	736.2	267
Am	1.7	0	75.6	106.7	29.3
Cm	0.2	0	11.3	5.5	-6.04
^{36}Cl	0	0.2	0	0.15	0.35
S	0	0.3	0	0.3	0.6
total FPs	0	1098.4	0	35.8	1134.2

NB: ^{36}Cl is not extracted, it will be part of the final inventory and thus will have to be managed in the waste.

III.B. Scenarios

In order not to focus on any national fleet, a very general scenario will be chosen for the reference with an arbitrary year 0 as the beginning. The following hypotheses are made:

- Total power demand is set at 50GW$_e$ for 120 years.
- No more than 2 reactors can be deployed per year, per type.
- Use of MOX fuel is possible starting from year 20.

- MSFRs can be started at year 70.
- MSFRs produce 1.45 GW_e.
- To compensate for the lack of flexibility in fuel loading of reactors (one reactor = one fuel type), lifetime of reactors is halved (60y to 30y) to allow the possibility of fuel change for a given technology, such as switching from UOX fuel to MOX fuel.
- Considering the Pu vector of the salt extracted from the fertile blanket salt of Cl-MSFR, it is assimilated as "spent UOX TRUs"

Description of the reactors are provided in table VI. ^{dep}U and ^{rep}U respectively stand for depleted and reprocessed uranium, with respective proportions of ^{235}U being 0.2% and 1%. UOX is based on enriched uranium (4% of ^{235}U) obtained from ^{nat}U (natural uranium, 0.7% of ^{235}U).

TABLE VI

Description of simplified reactors for ISF code

Reactor	Input	Output
PWR 1000 UOX	UOX: 26t/y	TRUs: 0.3t/y FPs: 1t/y ^{rep}U: 24.7t/y
PWR 1000 MOX	MOX: 26t/y	TRUs: 0.3t/y FPs: 1t/y ^{rep}U: 24.7t/y
Cl-MSFR 1	spent UOX TRUs: 12.8t + 34kg/y spent MOX TRUs: 0.53t + 0.57t/y ^{dep}U : 95t+1.5t/y	TRUs (fuel): 13t TRUs (fertile): 0.9t/y + 0.51t FPs: 1.1t/y +2.1t ^{dep}U : 96t
Cl-MSFR 2	spent UOX TRUs: 12.8t + 34kg/y ^{dep}U : 61t+1.1t/y	TRUs (fuel): 13t FPs: 1.1t/y +2.1t ^{dep}U : 62t

Cl-MSFR 2 refers to the reactor without the fertile blanket, that recycles TRUs exiting Cl-MSFR 1 fuel.

Figures 9 and 10 present the evolution of the nuclear fleets for the reference scenario and the scenario with Cl-MSFR deployment.

A dozen of Cl-MSFR are started in the second scenario to replace some UOX fed PWRs, while the amount of MOX fed PWRs remains constant. No drop in the production is observed. Table VII gives the final stockpiles for the two scenarios.

The impact of Cl-MSFRs is visible: in this study with only 50 years of operation of these reactors, the use of ^{nat}U is reduced by 20%, and the stockpile of spent MOX TRUs is virtually consumed.

TABLE VII

Final stockpiles for the two deployment scenarios

Final stockpiles	Reference	With Cl-MSFRs
use of ^{nat}U	821t	663t
^{dep}U	777t	645t
Spent UOX TRUs	180t	199t
Spent MOX TRUs	261t	14t
Spent MSFR TRUs	/	162t
Total TRUs	441t	375t

Although the gain on the final stockpile of TRUs can seem small, the main benefit comes from the difference in the TRU vector quality between "Spent MSFR TRUs" and "Spent MOX TRUs". The latter contains more minor actinides (23% in Spent MOX fuel TRUs vs 4% in Spent MSFR fuel TRUs after 60y) which would mean more difficulties for disposal or reuse, and is currently considered as a waste.

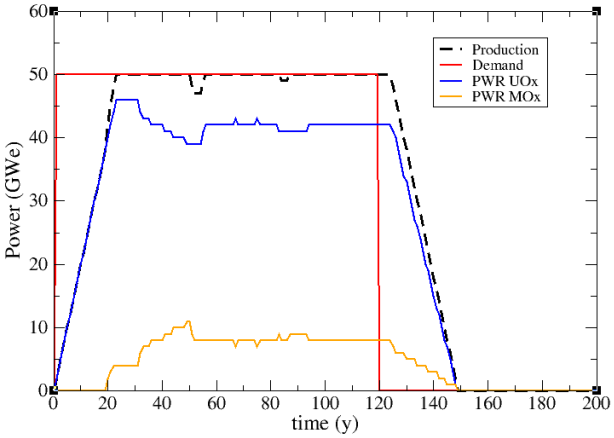

Fig. 9. Nuclear fleet evolution for the reference scenario

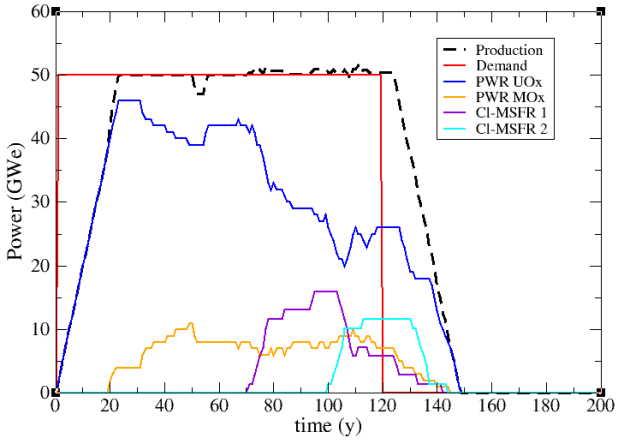

Fig. 10. Nuclear fleet evolution for the scenario with Cl-MSFRs

We have to remind that the scenarios presented here are simplified and aim at giving a global idea of the impact of MSR deployment. Yet it suggests that there is no issue

related to matter availability for Cl-MSFR to properly supplement PWRs. The possibility to use spent MOX fuel TRUs should be of interest to current nuclear waste managers. Nevertheless, the Cl-MSFR is optimized to use TRUs but not to incinerate them. For the sole purpose of reducing the stockpile of spent fuel TRUs, adding dedicated burner reactors [10] would be a preferred option.

IV. CONCLUSIONS

This paper presented some studies for the design and optimization of a molten salt fast breeder reactor using uranium cycle.

These studies suggest that for the generation of 3 GW_{th}, a fuel salt volume of $45m^3$ is indicated. To enhance the breeding ratio, a fertile blanket can be added. For an optimum use of the uranium, a width of 80 cm seems appropriate and as perspective is under further optimization with the use of a moderator.

A reprocessing flowrate of at least 50L/d for the fuel salt is required to avoid a significant mass shift. The alkali & alkaline-earth metal production is of importance as these elements cannot be easily extracted, with barium seeming especially concerning. The processing flowrate of the fertile salt depends on the amount of TRUs that are incorporated (should this solution for a better proliferation resistance be chosen) and the acceptable fission rate within it. This work indicates the addition of 0.5 mol % of (TRUs)Cl_3 from spent MOX fuel in the NaCl-UCl_3 , and an associated reprocessing flowrate of 100L/d.

Preliminary deployment studies showed no obstacle to the use of Cl-MSFR from a matter availability perspective. A good synergy with PWR reactors is possible thanks to the possibility of spent MOX fuel reuse in Cl-MSFR fertile blanket.

Finally, this study highlights two challenges related to Cl-MSFR. The first is related to the structural material, as no alloy was ever qualified to be used with chloride salt fuels. The second is correlated to the first, and is about the processing. The extraction of metallic fission products is not guaranteed as they are likely to be adsorbed in heat exchangers, especially if these exchangers are made of metallic alloys.

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NOMENCLATURE

$\beta_{eff,circ}$: effective delayed neutron fraction considering precursors circulation
FPs: Fission Products
HX: Heat eXchanger
MSFR: Molten Salt Fast Reactor
PWR: Pressurized Water Reactor
TRU: TRansUranic elements

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