A closed fuel cycle option using the MSFR concept with chloride salts and the U/Pu cycle

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Abstract

This paper proposes a concept of molten salt fast reactor (MSFR) using chloride salts and U/Pu cycle as solution to close the fuel cycle. It is making use of the available materials currently unused or even considered as waste: depleted uranium (^{dep}U) and transuranic elements (TRUs). Its fuel is a NaCl-$^{\text{dep}}UCl_3$-(TRUs)Cl_3 salt with an additional fertile blanket made of NaCl-$^{\text{dep}}UCl_3$-($^{ex-MOX}$Pu)Cl_3, to increase the breeding ratio of the system while improving the Pu isotopic vector of the spent MOx fuel. TRUs can come from various sources: spent MOx and/or spent UOx fuels, breeded plutonium; or the reactor can start with enriched uranium instead of TRUs. To take full advantages of the liquid fuel features, MSFRs are associated with a reprocessing unit allowing for on-site refuelling and reprocessing without stopping the reactor. Principle diagrams of reprocessing for the chloride MSFR are proposed. Impacts of reprocessing flowrates on neutronics and mass management are discussed. Results of neutronic depletion calculations for mass flowrates of input and output matters are presented. Finally, preliminary studies of French deployment scenarios will be proposed, to assess the possibilities of adding the reactor to the current fleet of PWRs from a resource perspective.

1 Introduction

Human societies (mainly developed countries) were able to raise their level of confort and life quality with a continuous increase of the use of fossile fuels (coal, oil, gas) since the beginning of the industrial era. To tackle the climat change caused by greenhouse gas emissions coming from fossile fuel combustion, the energy we use should be provided by low-carbon sources. Among them, nuclear energy could have an important contribution. The current commercial nuclear fleet accounts for only 4% of the world primary energy consumption [1]. For nuclear energy to play a role worldwide, its capacity should be increased significantly [2].

The current fleet consists of 440 reactors. 437 of them use ^{235}U as the main fuel, so 0.7% of the natural uranium. Relying only on this material as fissile matter could be problematic on the long run. Existing nuclear fuel cycles are either open, or semi-open (partial reuse of plutonium to stabilise its production [3]). Current water reactors cannot use the minor actinides that are produced during the reactor operation. Those minor actinides have a high level of radioactivity and a long life-span (half-life up to 10^5 years). Different options (geological repository, Accelerator-Driven Systems, fast reactors) are investigated for the disposal of matters considered as waste.

This paper proposes a solution to close the fuel cycle: a concept of Molten Salt Fast Reactor (MSFR) using chloride salts and the U/Pu cycle. In section 2 the chloride MSFR (MSFR-Cl) will be presented, then several starting configurations will be shown in section 3. Designs of the reprocessing units will be proposed in section 4. The fuel composition evolution, studied with the REM neutronic depletion code available at CNRS [4][5], will be given in section 5. Finally, preliminary results on French deployment scenarios will be discussed in section 6.

2 Presentation of the chloride MSFR

Since 2000, the Generation 4 International Forum (GIF) [6] has been defining criteria that new reactors should meet in order to select the promising technologies to focus studies on. In following years, six concepts were designated, among them Molten Salt Reactors (MSRs). In 2008, the GIF selected the Molten Salt Fast Reactor (MSFR) as the concept of reference for MSRs.

The MSFR is a reactor using a liquid circulating fuel also acting as a coolant. The liquid form provides very interesting advantages:

- the reactor has an excellent intrinsic safety with a large negative density feedback;
- the fuel composition can be modified during the operation, so continuous removal of undesired species (fission products (FPs), impurities) and refuelling are possible without stopping the reactor.

The first advantage allows the reactor to be operated without control rods: the reactor power naturally adjusts to the power demand [7].

The reference MSFR [8] is a 3 GW_{th} optimised as a breeder using Th/U cycle. The fertile element ^{232}Th and the fissile element ^{233}U are dissolved as fluorides in LiF. It is compact, with 18m^3 of fuel (half in a central volume, half in recirculating loops). It shows very promising performances. One of the few downsides is the low solubility of trivalent elements such as plutonium and minor actinides in the solvent. This last point is the main driver to study an alternative version of the reference MSFR, the MSFR-Cl.

The MSFR-Cl under study here uses NaCl-$^{dep}UCl_3$-(TRUs)Cl_3 as fuel salt (^{dep}U for depleted uranium), transuranic elements (TRUs) encompassing plutonium and minor actinides. Different possible compositions for the fuel will be presented in section 3. For the reactor to breed, the volume of the fuel should be large enough such that the fraction of fertile material that captures overcomes the fissile fraction that disappears; and/or a fertile blanket can be added, rich in fertile material dedicated to be transmuted to fissile material. The latter option will be discussed here as it prevents the necessity to extract actinides from the fuel to remove the excessive fissile matter. The former option shall be investigated in a future work. The fuel volume is a compromise between the ability to extract the heat (towards large volumes) and the required initial fissile inventory (which we would like to minimize), as well as compacity and keeping the breeding ratio of the core under 1 (towards small volumes). Both neutronics codes (Serpent2, REM) and an in-house multicriteria optimisation code (called SONGe) for the fuel circuit were used. Results suggest that a fuel volume of 60m^3 is a good compromise.

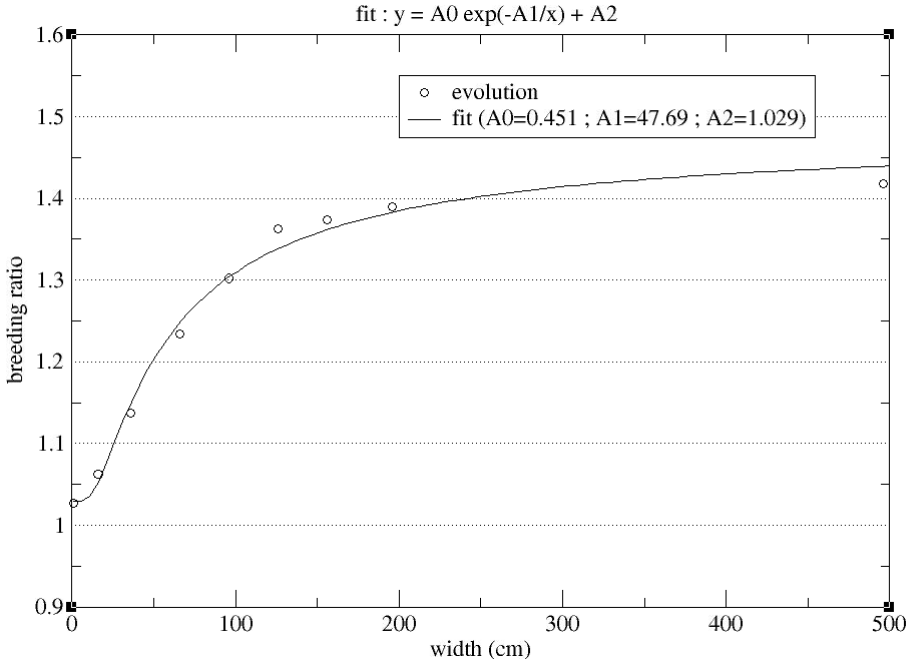


Figure 1: Breeding ratio at steady state depending on the blanket width for a 80m^3 fuel volume reactor

The fertile blanket has the following composition: NaCl-$^{dep}UCl_3$-$PuCl_3$. The plutonium here is present in a very small fraction and comes from spent MOx fuel. Its share of ^{239}Pu is relatively low (about 35%) and contains large amounts of fertile isotopes. During operation, the share of ^{239}Pu will increase, but we design the system so that the plutonium in the blanket never becomes of weapon-grade quality (93% of ^{239}Pu): it is a

measure to resist against proliferation. Different tests using the in-house CNRS REM code [4][5] that couples the neutron transport with depletion calculations to optimise the fertile blanket width were performed. Figure 1 presents the evolution of the breeding ratio depending on the blanket width. For this study, a width of 96cm (with 2x2cm of structure material) was chosen as a good compromise between the volume of the blanket (that will have to be processed) and the breeding ratio. The blanket volume here is 36.7m^3.
N.B: one could see that for a width close to 0, so "core-only" configuration, the breeding ratio is larger than 1. That is due to the fuel volume, chosen at 80m^3 for this study but the trend is similar for the optimised volume of 60m^3 selected after several studies.

As mentioned above, a layer of 10cm of B_4C acts as a neutron protection for the heat exchangers. This width is not yet optimised, recent investigations suggest that it should be increased. Finally, the structural material used here is the hastelloy N, a Ni-based alloy designed for high temperature molten salts and with a good resistance against irradiation.
For an overall view, the geometry of the reactor used for neutronics (in Serpent2 or MCNP codes) is shown in figure 2.

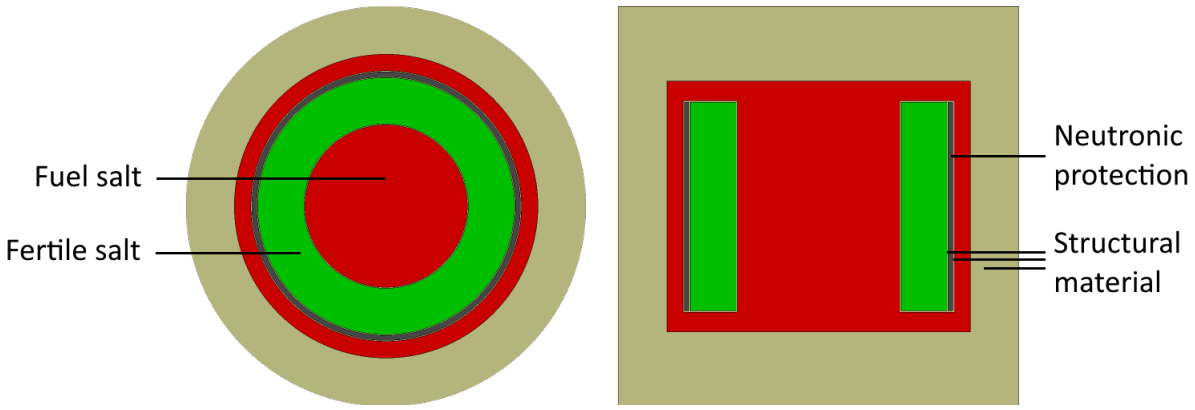


Figure 2: Geometry of the MSFR-Cl for neutronic transport codes (Serpent2 plot)

3 Fuel circuit configurations

The MSFR is a flexible concept that allows various compositions for the fuel salt. In this case, one would wish to start the reactor with already available matters. The fissile matters will come from: spent UOx fuel (5 years of cooling time to reduce the decay heat to facilitate chemical processing) with all its TRU elements, spent MOx fuel (15 years of cooling time) with all its TRU elements, enriched uranium, and Pu coming from the fertile blanket.
The method used to adjust the initial composition of the fuel is the following. There are two main constraints on the fuel salt: it has to be critical, and the mix should be eutectical (to avoid some issues such as preferential precipitation in case of local cold spot). With the hypothesis that TRUs are considered as Pu from the chemical perspective, by choosing the proportion of $(TRU)Cl_3$ to reach criticality, the proportions of NaCl and UCl_3 are directly given by the ternary phase diagram of the salt. This diagram is presented in figure 3, on which the four starting fuel mixes that are considered here are indicated by red dots.

The relative proportions as well as masses to start the reactor (60m^3 of fuel salt + 36.7m^3 of fertile salt) are given in table 1 with the following order: <fuel salt mass> + <fertile salt mass>. The fertile salt containing 3% of Pu from spent MOx fuel (15 years of cooling) (see section 5).

Table 1: Proportions and masses of solvent, fertile and fissile matters for initial compositions

Fuel	NaCl	UCl_3	$(TRU)Cl_3$		TRU mass	U mass	total mass
ex-UOx	65.5 %	28.1 %	6.4 %		16.4 + 4.7 t	71.0 + 48.4 t	170 + 104 t
ex-MOx (15y)	65.0 %	26.6 %	8.3 %		21.1 + 4.7 t	67.0 + 48.4 t	170 + 104 t
^{enr}U (e=15.9%)	67.0 %	33.0 %	0 %		/	84.6 + 48.4 t	167 + 104 t
Breeded Pu	65.8 %	28.6 %	5.6 %		14.3 + 4.7 t	72.6 + 48.4 t	170 + 104 t

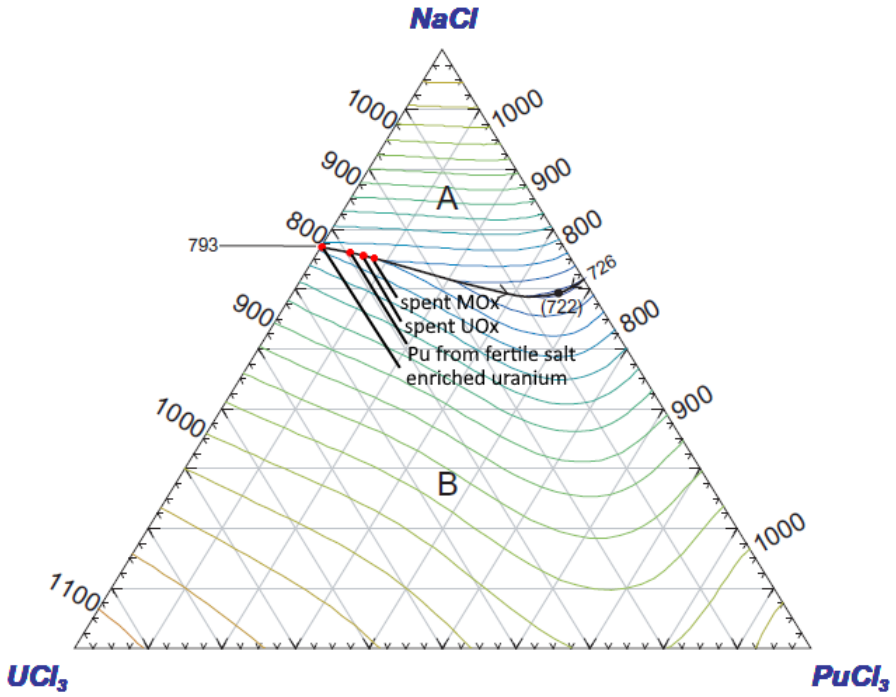


Figure 3: Ternary diagram of NaCl-UCl_3-$PuCl_3$ [9]

4 Reprocessing schemes

The reprocessing of the chloride MSFR can be done in three main ways:

- no on-site reprocessing, full off-site reprocessing at the end of the operation;
- full on-site online reprocessing;
- partial on-site reprocessing, with additional off-site treatment.

For MSRs, the first option is usually considered for SMRs (Small Modular Reactors), with the idea of simplified operating conditions; for this work, it will not be considered.
The second option is considered for the reference MSFR, and is also studied for the MSFR-Cl. It has several advantages such as the direct recovery of the enriched chlorine (reducing its global inventory) and possibly a safer transport scheme, as only lanthanides (Ln) in a stable form leave the site. Moreover, it does not require an initial excess of reactivity to compensate the burn-up as fresh fuel is continuously introduced, which is a significant asset from a safety perspective. It also has the drawback of a more complex on-site fuel treatment unit and often discards the water-based chemical processes, as molten salts entering the treatment unit are both highly radioactive and at high temperature.
The third possibility is seen as a good balance between fast heavy nucleus recovery and effective chemical treatment. The idea is to keep actinides within the site but send everything else in a dedicated off-site treatment facility that can process the salt using more conventional and slower processes. The on-site treatment is highly simplified, but it requires a larger enriched chlorine inventory.

4.1 Full on-site reprocessing

The principle diagram of this process is shown in figure 4. It is made of 3 parallel processes:

- cold traps for volatile FPs;
- a liquid metal filtration on which a sample of salt and gas filled with metallic products is regularly sent;
- a chemical extraction process in two steps, first to extract (An)s (due to their redox potential they should be extracted first), second to extract FPs, mainly (Ln)s.

Cold traps are considered for the following elements: zirconium, phosphorus, sulfur, arsenic, selenium, astatine. Among them, zirconium is of particular importance as it is produced in large quantities and is highly volatile as $ZrCl_4$.
The liquid metal filtration process is thought to operate as following. The injection of gas leads to the trapping of small metallic elements in bubbles, that burst at the surface of the free volume. A portion of the salt here, with the metallic deposit, is sent in contact with a liquid metal such as lead, that will adsorb the metallic elements that form the majority of light fission products (metals with a Z between 30 and 52).

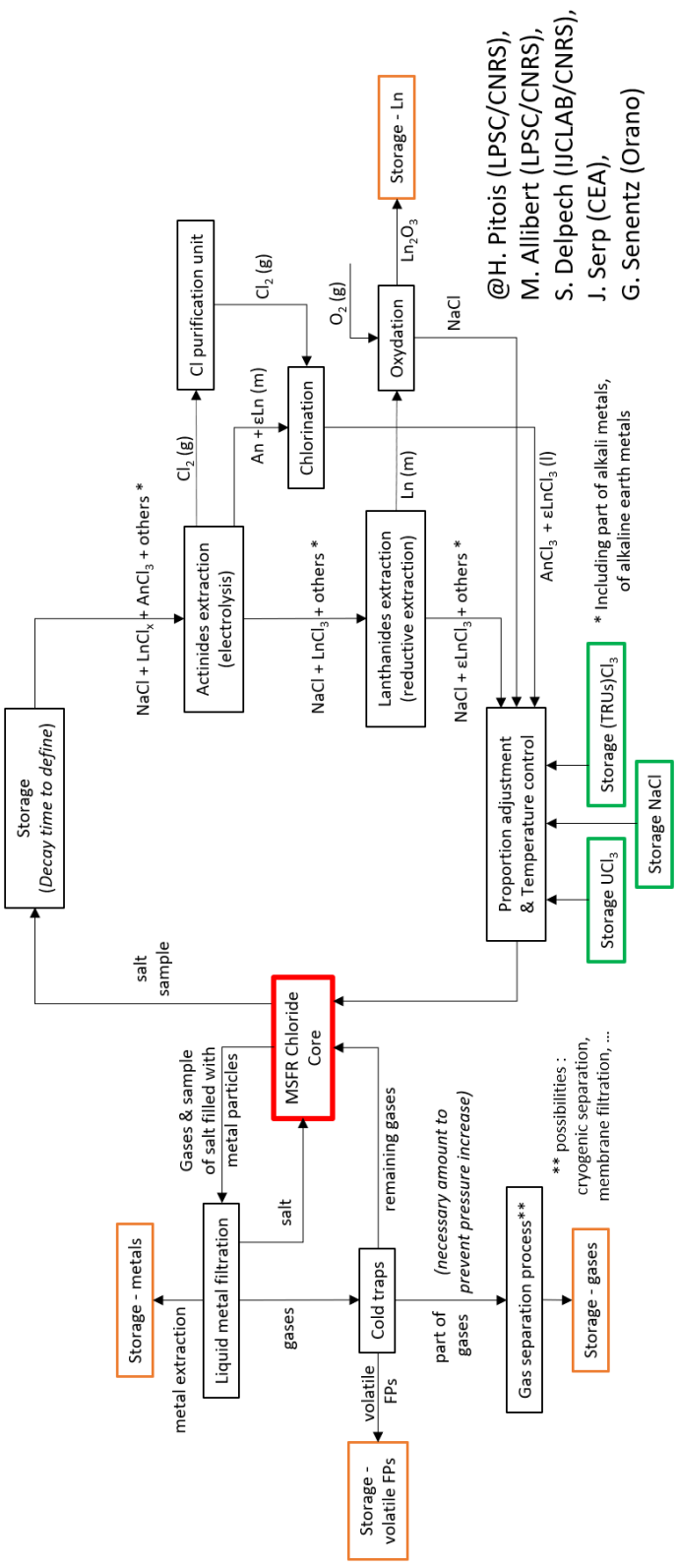


Figure 4: Full on-site reprocessing scheme

Chemical extractions aim to efficently remove the heavier FPs. These are the most delicate steps because one should prevent as much as possible the extraction of (An)s with (Ln)s in the second step, while extracting the latter with a sufficient efficiency. The task is difficult in chloride salts where the redox potentials of some (Ln)s are very close to the potentials of (An)s, and the sodium is a weaker reducing agent in comparison to lithium in fluoride salts. It is likely that several steps of extractions coud be necessary, thus increasing the amount of time of reprocessing as well as the amount of salt outside the reactor. The steps extract Ln and convert them into oxides are not well defined. They are various possibilities such as the oxidation of metal by gaseous oxygen

or water vapor hydrolysis of chlorides, with recycling of HCl to recover the enriched chlorine.

4.2 Partial on-site reprocessing & off-site treatment

This second method was recently considered for the MSFR-Cl. One principle diagram is proposed in figure 5.

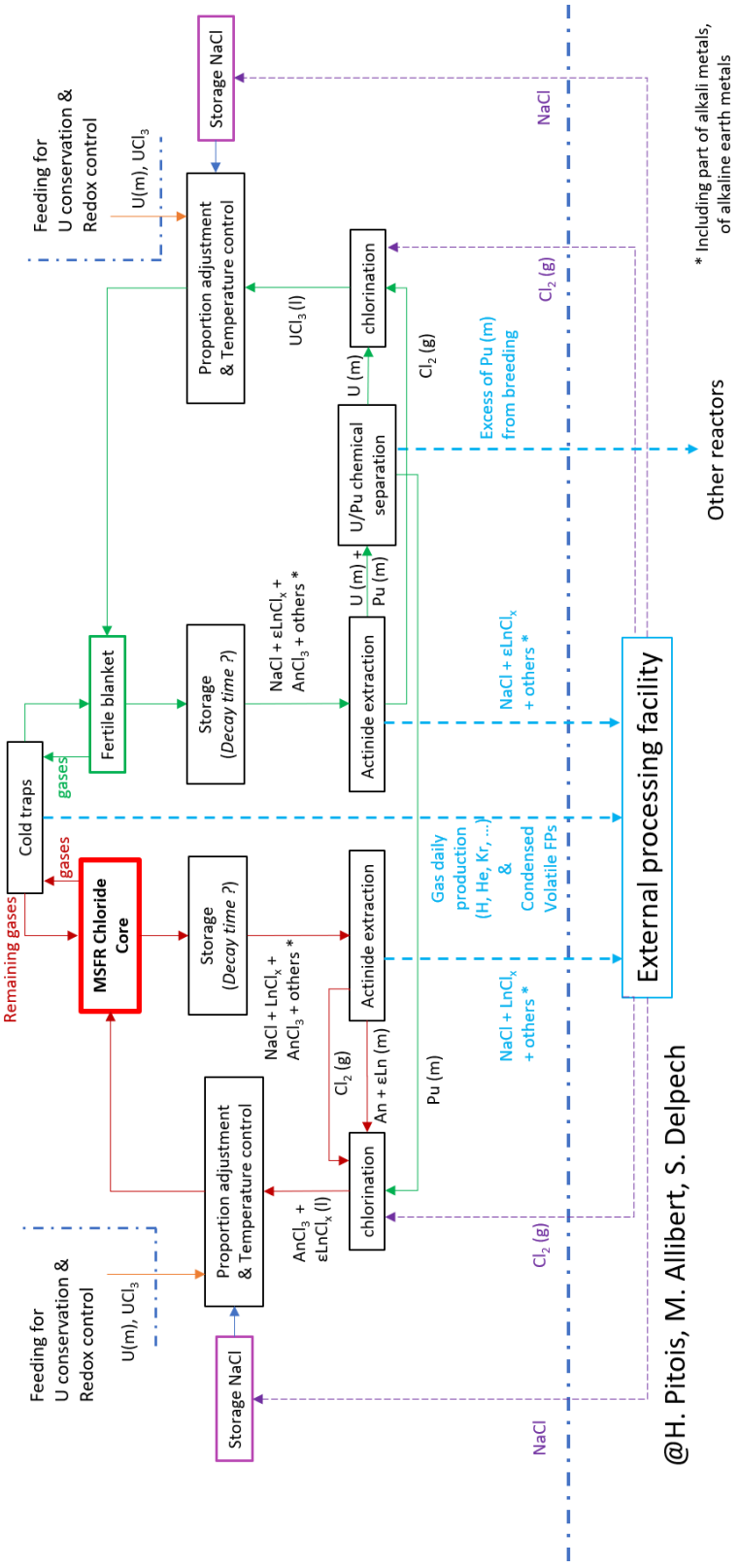


Figure 5: Partial on-site reprocessing

As previously mentioned, the on-site fuel treatment unit is highly simplified. Gases are extracted, and only (An)s are recovered while everything else is sent to an off-site processing facility. Said facility has for objectives

the extraction of FPs, the purification of the solvent especially from the alkali and alkaline earth metal other than sodium, and the recovery of enriched chlorine that should be sent back to the power plant as soon as possible to minimize its global inventory. The steps within the external processing facility are not detailed here as lots of processes can be foreseen and some work remains to select the most appropriate ones.

5 Neutronic computations with material evolutions

To assess the quantities of matters involved (inventory of heavy nuclei needed or amounts of each element produced during operation), one should perform neutronics depletion simulations of the reactor. Because of the specificity of feeding and extraction during operation allowed by the liquid fuel, dedicated codes must be used. The neutronic depletion calculations presented here were performed using the in-house REM [4][5], and ENDF-B7 nuclear data library. A first calculation without processing is presented, followed by calculations with advanced processing.

5.1 Neutronic depletion calculations without processing

This calculation is usually performed to assess the loss of reactivity through time if the reactor is not supplied with fissile matters; here it will be used to determine how much mass of each element is produced.
Using the spent-MOx-based fuel, table 2 records the masses (presented as mass flowrates) of fission products with a produced mass above 1kg (arbitrarily chosen for an easier reading) in the fuel salt after 10 years of reactor operation.

Table 2: Averaged mass flowrates (fuel salt) for 10 years of reactor operation without fuel processing

Z	q_m(kg/y)	Z	q_m(kg/y)	Z	q_m(kg/y)	Z	q_m(kg/y)	Z	q_m(kg/y)
2	1.83E+00	38	1.26E+01	46	7.26E+01	54	1.25E+02	62	2.78E+01
12	7.46E-01	39	6.80E+00	47	7.74E+00	55	1.01E+02	63	3.91E+00
16	1.90E-01	40	6.95E+01	48	1.48E+01	56	4.05E+01	64	3.58E+00
18	2.25E+00	41	2.28E-01	49	1.91E+00	57	3.08E+01	65	2.44E-01
34	1.48E+00	42	8.27E+01	50	1.70E+01	58	5.78E+01	66	2.72E-01
35	5.64E-01	43	2.07E+01	51	5.56E+00	59	2.72E+01	sum	9.89E+02
36	6.68E+00	44	8.04E+01	52	2.53E+01	60	9.19E+01		
37	6.15E+00	45	2.39E+01	53	1.40E+01	61	3.92E+00		

A periodic table with methods of removal is presented in figure 6. It was made in accordance with the options of full on-site or partial off-site reprocessing, the difference beeing that elements extractible using liquid metal will instead be removed off-site with other soluble FPs. Additionnaly, gases will be continuously extracted by a gas removal system. One should keep in mind that part of the radioactive noble gases will decay into alkali metals in this gas removal system.

Legend: cross: elements with a mass below 1g; orange triangle: gas/volatile FP removal system; purple square: liquid metal filtration; red square: An extraction (electrolysis). The elements without indication should be extractible during the second phase of the chemical extraction process.
N.B: one can see that Zr and O are marked with a red dot, signifying that they should be removed by the chlorination process after the An extraction.

5.2 Depletion calculations with full fuel treatment

Several depletion calculations are presented here for different fuels and different reprocessing rates.

As explained in section 2, the fertile blanket composition must be carefully monitored to avoid proliferation related issues. Thus, the first group of depletion calculations is focused on finding appropriate parameters to obtain a satisfactory Pu vector. It was decided to aim for a share of ^{239}Pu so that it can be reused in MOxed EPRs: around 60%. Figure 7 shows the evolution of the ^{239}Pu share in the fertile salt for various fractions of Pu and various An reprocessing flowrates (the fuel salt being spent-MOx-based fuel, but the fuel composition can be left aside of considerations here as it has a small impact on the blanket composition evolution).

The configuration with 3% of Pu and an An reprocessing flowrate of 10L/d offers the best ^{239}Pu share considering our requirements.

To stick with the objective of closing the fuel cycle, the following group of depletion calculations was conducted for MSFR-Cl fuelled with spent MOx TRUs, spent UOx TRUs, or Pu coming from the fertile blanket (with the idea that once the spent fuel stockpiles are low enough, MSFR-Cl operations should pursue

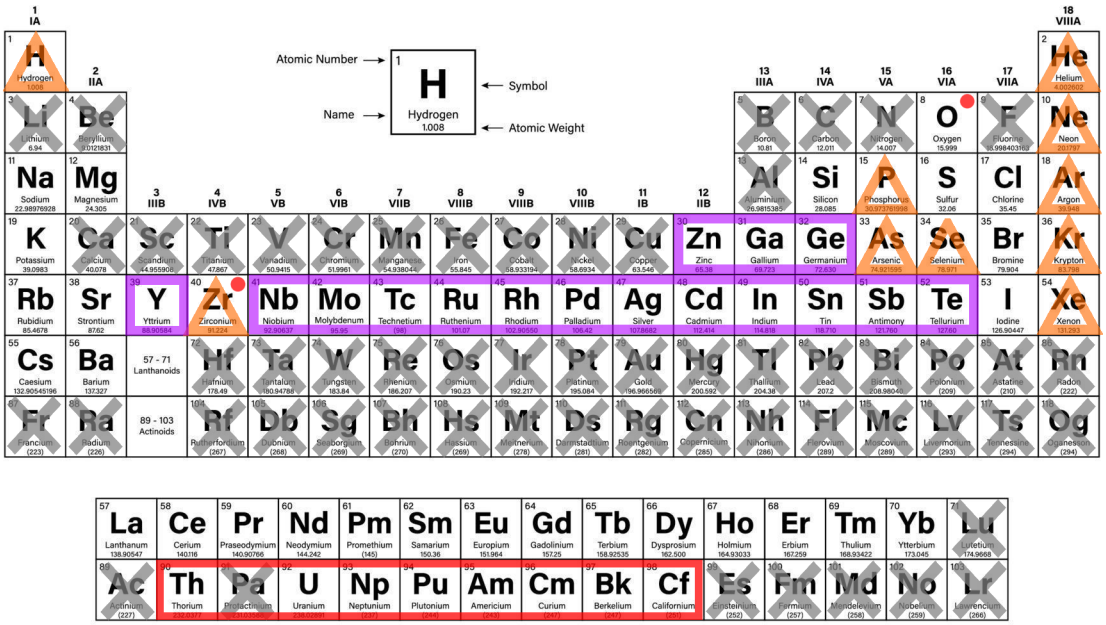


Figure 6: Periodic table for the management of elements in fuel and fertile salts

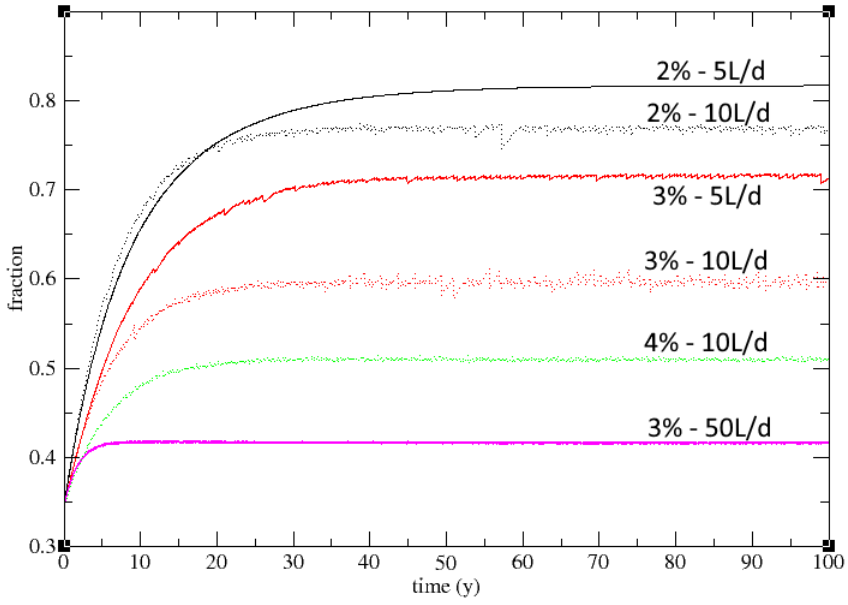


Figure 7: Evolutions of ^{239}Pu share in the fertile salt during reactor operation

with the breeded fuel). Respective reprocessing flowrates are: 100L/d for the fuel salt; 10L/d for An in fertile salt; 50L/d for FPs in fertile salt. Figure 8 presents the evolutions of An masses in the fuel salt.

Figure 9 shows the flowrates of elements that must be processed by the reprocessing unit every year, considering the MSFR-Cl fuelled with spent MOx TRUs. This is for both the fuel salt (reprocessing of 100L/d) and the fertile salt (reprocessing of 10L/d for (An)s and 50L/d for FPs). Table 3 gives the balance of TRUs after 50 years of operation of said configuration.

Several informations can be extracted from figure 9 and table 3.
First a significant amount of fission products are produced in gaseous form, with xenon (Z=54) being the most produced (about 150 kg/y). This highlights the importance of the gas removal system to maintain a low pressure (and reduce the source term).

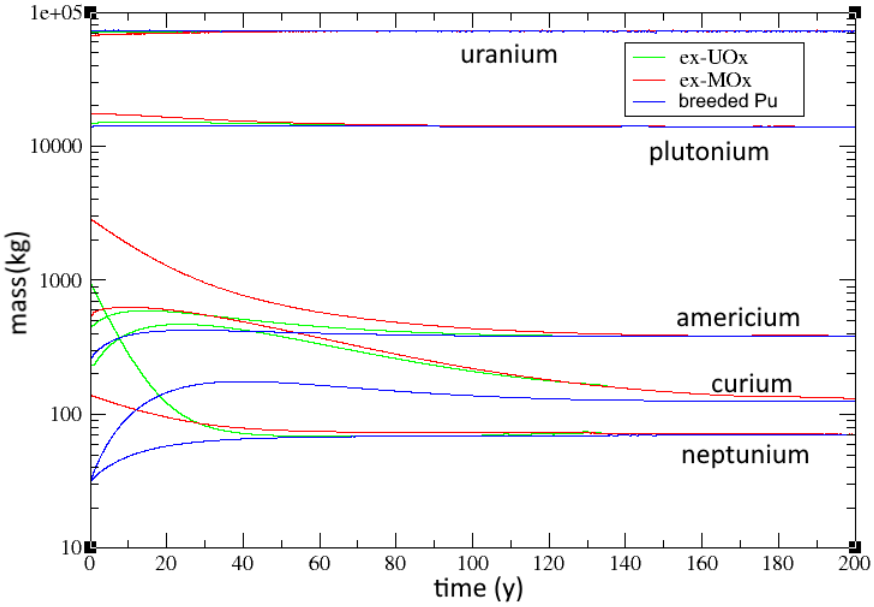


Figure 8: Evolutions of An masses in the fuel salt

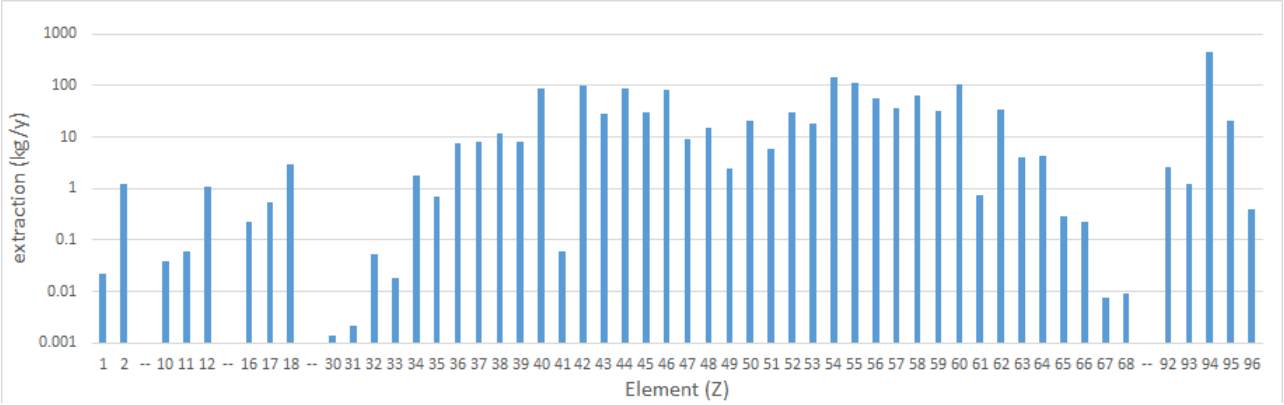


Figure 9: Total (fuel and fertile salts) extraction rates of elements for the MSFR-Cl fuelled with spent MOx TRUs (average on 50y)

Second, the net output of MAs appears positive while one would want to use the reactor to burn those elements. It is due to the chosen parameters for this configuration. They are extracted from the fertile blanket, but it is reasonable (although should be confirmed by further work) to assume that they could be reinjected in the core without a threat to the safety from a neutronic perspective, if they can be separated from the plutonium.
Third, one can see a net output of about 160 kg/y of plutonium. It is positive as expected from a breeder reactor, and the value makes sense from a chemical perspective. But it should be reminded that the extracted plutonium and the injected plutonium plutonium do not have the same importance from a neutronic perspective. While around 290 kg/y of low-quality Pu (35% of ^{239}Pu) are injected, around 450 kg/y of higher quality Pu (60% of ^{239}Pu) are extracted. This plutonium can be used as fresh fuel for MSFRs but could be also used to produce fresh MOx fuel for future EPRs for instance, adding value for the current industry.

mass (kg)	U	Np	Pu	Am	Cm
core feeding	$5.008\ 10^{4}$	$2.166\ 10^{-1}$	$2.876\ 10^{1}$	4.712	$8.224\ 10^{-1}$
fertile blanket feeding	$1.730\ 10^{4}$		$1.448\ 10^{4}$		
fertile blanket extraction	$1.298\ 10^{2}$	$6.044\ 10^{1}$	$2.260\ 10^{4}$	$1.040\ 10^{3}$	$1.972\ 10^{1}$
net output	$-6.726\ 10^{4}$	$6.022\ 10^{1}$	$8.093\ 10^{3}$	$1.035\ 10^{3}$	$1.890\ 10^{1}$
average net output per year	-1345	1.204	161.9	20.71	$3.780\ 10^{-1}$

Table 3: TRU balance after 50y of operation of the MSFR-Cl fuelled with spent MOx TRUs

6 French deployment studies including MSFRs-Cl

Scenarios have been performed for the French nuclear fleet to assess the capabilities of the MSFRs-Cl to efficiently use the currently available matters and reduce the amount of final wastes. The code used is the in-house code ISF (Innovative Scenarios for the Future) [2]. Those scenarios are made using a simplified description of the reactors and aim to provide relevant orders of magnitude, not precise figures.

6.1 Description of the scenario of reference

This scenario modelises as best as possible the past deployment of the French nuclear fleet. Then, it relies on the scenario called "N3" made by the national provider of electricity RTE [10], to reach 50 GW_e of installed power by 2050. It considers only EPRs to replace the current reactors. Hypothesis is made that by 2100, another technology will be chosen to take over nuclear fission, which is represented by a drop of the demand in 30 years. The idea here is to go up to the end of game of scenario to evaluate the remaining stockpiles of matters. In order to make use of most of the available materials, UOx-fuelled EPRs only start if MOx-fuelled EPRs cannot be deployed; uranium separated during fuel processing (^{rep}U for reprocessed) is used instead of ^{nat}U if ^{enr}U is required (there is no distinction here between ^{enr}U and re-enriched uranium). Table 4 gives the description of each reactor. ^{enr}U and UOx are used indifferently as only U mass is tracked.

Table 4: Characteristics of water reactors used in scenarios

Reactor	Power (MWe)	Load factor	Lifetime (y)	Mass input (t/y)	Mass output (t/y)	Starting time	Ending time
PWR 900 UOx	900	0.8	50	^{enr}U: 20	TRUs ex-UOx: 0.22 FPs: 0.68 ^{rep}U: 19.1	1977	1988
PWR 900 MOx	900	0.8	50	MOx: 20	TRUs ex-MOx: 0.22 FPs: 0.68 ^{rep}U: 19.1	1985	2000
PWR 1300 UOx	1300	0.8	60	^{enr}U: 28.89	TRUs ex-UOx: 0.32 FPs: 0.98 ^{rep}U: 27.59	1985	2020
EPR UOx	1650	0.8	30	^{enr}U: 36.67	TRUs ex-UOx: 0.40 FPs: 1.25 ^{rep}U: 35.02	2025	2099
EPR MOx	1650	0.8	30	^{enr}U: 36.67	TRUs ex-MOx: 0.40 FPs: 1.25 ^{rep}U: 35.02	2025	2099

One can see that the lifetime of EPRs (and some of the reactors presented in following paragraphs) is shorter than expected. It is a way to add flexibility in the model. The code does not offer the possibility to change the fuel once a reactor is deployed, so this short lifetime should be interpreted as a way to charge the reactor with another fuel after a given number of years if a better option if available (e.g. use of MOx instead of ^{enr}U), rather than closing the power plant and build a new reactor.

6.2 Description of the scenario with MSFRs-Cl

In this scenario we assume a deployment of MSFRs-Cl starting from 2050, in two possible configurations. In priority, spent MOx fuel-based MSFRs-Cl are started. If the stockpile of spent MOx fuel is not sufficient, spent UOx fuel-based MSFRs-Cl are deployed. Finally, MSFRs-Cl fuelled by the "high quality" plutonium (^{hq}Pu) from the fertile blanket are started. Description of MSFRs-Cl are provided in table 5.

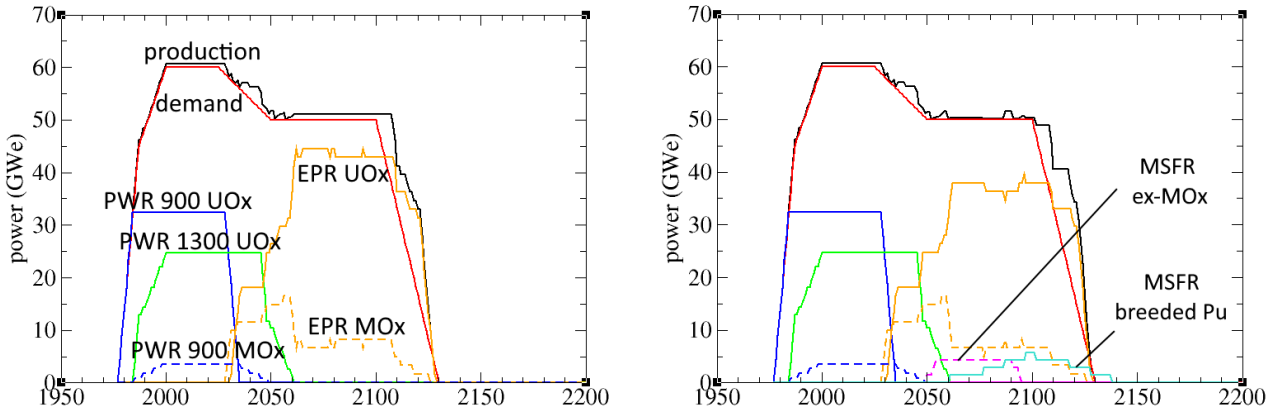
6.3 Results and comparisons

Figure 10 presents the evolution of nuclear fleets.

About fleets, there is no problem provided that the reactors can be built. For the scenario with MSFRs-Cl using this configuration, one can see that no MSFR fuelled with TRUs from spent UOx fuels is started and a relatively small amount of MSFRs are deployed. The limiting factor for MSFR deployment is the available amount of spent MOx plutonium required by the fertile blanket to avoid the production of weapon-grade plutonium. Futher studies are undergoing to reduce the amount of required spent MOx Pu to start MSFRs-Cl.

Table 5: Characteristics of MSFRs-Cl used in the scenario

Reactor	Power (MWe)	Load factor	Lifetime (y)	Mass input	Mass output	Starting time	Ending time
MSFR-Cl ex-UOx	1450	0.9	40	TRUs ex-UOx: 16.3t + 5.7kg/y ^{app}U: 119t + 1.36t/y Pu ex-MOx: 4.69t + 283kg/y	TRUs ex-MSFR: 15.6t FPs: 1.16t/y + 505kg ^{app}U: 120t ^{hq}Pu: 4.90t	2050	2099
MSFR-Cl ex-MOx	1450	0.9	40	TRUs ex-MOx: 21.1t + 0.69kg/y ^{app}U: 115t + 1.35t/y Pu ex-MOx: 4.69t + 290kg/y	TRUs ex-MSFR: 16.2t FPs: 1.16t/y + 505kg ^{app}U: 120t ^{hq}Pu: 474kg/y + 4.84t	2050	2099
MSFR-Cl breeded Pu	1450	0.9	40	breeded Pu: 14.3t + 3.04kg/y ^{app}U: 121t + 1.36t/y Pu ex-MOx: 4.69t + 284kg/y	TRUs ex-MSFR: 14.7t FPs: 1.16t/y + 504kg ^{app}U: 120t ^{hq}Pu: 476kg/y + 4.85t	2050	2099



(a) Reference (b) With MSFRs-Cl

Figure 10: Evolution of nuclear fleets

Table 6 gives some figures of interest to compare both scenarios. "Other TRUs" column refers to matters produced by MSFRs-Cl: ^{hq}Pu, TRUs from fuel salt at the end of a reactor operation, and extra Am and Cm resulting from the separation to obtain the plutonium from spent MOx.

Table 6: Final stockpiles for the two scenarios

Scenario	^{nat}U consummed	TRUs ex-UOx final mass	TRUs ex-MOx final mass	Other TRUs final mass	Total TRUs final mass
Reference	844 10^3t	234t	238t	0	471t
With MSFR-Cl	797 10^3t	215t	22t	244t	481t
Difference	-47 10^3t	-19t	-216t	244t	10t

There is a reduction by a factor of 10 for the spent MOx TRUs when MSFRs with this configuration of fertile blanket (3% of spent MOx plutonium) are used. In fact, it can be considered that this stock would be fully used, the remaining amount is due to the way ISF code manages the materials. It appears that in the end, the second scenario leads to a higher stock of total TRUs at the end. But this hides the significant differences between their natures, as TRUs from the final composition of MSFR fuel salts and the breeded plutonium are of much better quality from a neutronic perspective and can be directly used as fuel if more reactor operations are required.

One should also keep in mind that the longer MSFRs-Cl operate, the more visible differences become for ^{nat}U consumption. Although modest in the case of the French fleet, it would become of major importance if there is a worldwide significant deployment of nuclear reactors to move on from fossile power plants. The price of uranium aside, it is not impossible to reach the point where fissile matter supply can no longer keep up with

the demand, if we continue to use non-breeder reactors. Scenarios for the global demand are undergoing and will be the object of another work.
Another undergoing work aims to improve the synergy between MSFRs and EPRs. The idea would be to have the former converting spent MOx plutonium to high quality plutonium to reproduce MOx fuels.
Lastly, the MSFRs-Cl considered here are optimised to use TRUs but not to burn them. If one wishes to just get rid of spent fuels, it might be more interesting to look at alternative versions of the reactor: burners. More information on such reactors can be found in [11].

7 Conclusion

This paper proposes the use of MSFRs-Cl coupled to the U/Pu cycle in order to close the fuel cycle and make use of matters currently considered as wastes. The current configuration of the reactor can take as input TRUs from spent fuels including irradiated MOx that are currently left unused. It produces a plutonium that could either be reused as MOx or directly in the same reactor. First optimisations were performed to get optimum compromises for the breeding and initial inventories. Diagrams of principle for both full on-site reprocessing or partial on-site reprocessing coupled with an external processing facility are proposed. Preliminary scenarios suggest that MSFRs-Cl could be used along or take over current water-cooled reactors of the French fleet from the perspective of resources. Yet from an economical perspective, with current conditions of low uranium prices and moderated nuclear reactor development, the costs to bring MSFRs-Cl at the industrial level would probably be too high to be considered in France. However, their advantages would be more appraised for global scenarios where constraints on natural uranium and fissile matter availability would appear.

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References

[1] B.P Statistical Review of World Energy. *Statistical Review of World Energy 2021*. Tech. rep. B.P, 2021.

[2] Elsa Merle et al. "Scenarios for a worldwide deployment of nuclear power". In: *International Journal of Nuclear Governance, Economy and Ecology* 1.2 (2006), pp. 168–192.

[3] Shant Krikorian, IAEA Department of Nuclear Energy. *France's Efficiency in the Nuclear Fuel Cycle: What Can 'Oui' Learn?* 2019. URL: https://www.iaea.org/newscenter/news/frances-efficiency-in-the-nuclear-fuel-cycle-what-can-oui-learn (visited on 11/21/2021).

[4] Elsa Merle et al. "Simulation Tools and New Developments of the Molten Salt Fast Reactor". In: European Nuclear Conference ENC2010 Barcelona. 2010.

[5] Xavier Doligez et al. "Coupled study of the Molten Salt Fast Reactor core physics and its associated reprocessing unit". In: *Annales of Nuclear Energy* 64 (2014), pp. 430–440.

[6] Generation IV International Forum. *Technology roadmap update for generation IV nuclear energy systems*. Tech. rep. Generation IV International Forum, 2014.

[7] Axel Laureau et al. "Transient coupled calculations of the Molten Salt Fast Reactor using the Transient Fission Matrix approach". In: *Nuclear Engineering and Design* 316 (2017), pp. 112–124.

[8] Michel Allibert et al. "Handbook of Generation IV Nuclear Reactors". In: Woodhead Publishing Series in Energy, 2015. Chap. 7 - Molten Salt Fast Reactors.

[9] O. Beneš and R.J.M. Konings. "Thermodynamic evaluation of the NaCl–MgCl2–UCl3–PuCl3 system". en. In: *Journal of Nuclear Materials* 375.2 (Apr. 2008), pp. 202–208. ISSN: 00223115. DOI: 10.1016/j.jnucmat.2008.01.007. URL: https://linkinghub.elsevier.com/retrieve/pii/S002231150800010X (visited on 11/03/2021).

[10] RTE. *Futurs énergétiques 2050, Résumé exécutif*. Tech. rep. RTE, Oct. 2021.

[11] Lea Tillard, Laura Mesthiviers, Elisa Capelli, et al. "Estimation of the vitrified canister production for PWR fleets integrating MSRs with different fuel cycle strategies". In: GLOBAL 2022 Conference, Reims, France. 2022.