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Launching the Thorium Fuel Cycle with the Molten Salt Fast Reactor

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Abstract – *Starting from the Molten Salt Breeder Reactor project of Oak-Ridge, an innovative concept called Molten Salt Fast Reactor or MSFR based on a fast neutron spectrum has been proposed, resulting from extensive parametric studies in which various core arrangements, reprocessing performances and salt compositions were investigated to adapt the reactor in the framework of the deployment of a thorium based reactor fleet on a worldwide scale. In the MSFR, the liquid fuel processing is part of the reactor where a small side stream of the molten salt is processed for fission product removal and then returned to the reactor. Because of this characteristic, the MSFR can operate with widely varying fuel compositions. Thanks to this fuel composition flexibility, the MSFR concept may use as initial fissile load, ^{233}U or uranium or also the transuranic elements currently produced by light water reactors. This paper addresses the characteristics of these different launching modes of the MSFR and the Thorium fuel cycle, in terms of safety, proliferation, breeding, and deployment capacities of these reactor configurations. To illustrate the deployment capacities of the MSFR concept, a French nuclear deployment scenario is finally presented, demonstrating that launching the Thorium fuel cycle is easily feasible while closing the current fuel cycle and optimizing the long-term waste management.*

I. INTRODUCTION

The Generation-IV International Forum (GIF) for the development of new nuclear energy systems has established a set of goals as research directions for nuclear systems[1]: enhanced safety and reliability, reduced waste generation, effective use of uranium or thorium ores, resistance to proliferation, improved economic competitiveness. Molten Salt Reactors (MSRs) are one of the systems retained in 2002 by this forum.

The CNRS has been involved in molten salt reactor studies since 1997. Starting from the Molten Salt Breeder Reactor project[2] of Oak-Ridge, an innovative concept called Molten Salt Fast Reactor or MSFR[3-9] has been proposed, resulting from extensive parametric studies in which various core arrangements, reprocessing performances and salt compositions were investigated to adapt the reactor in the framework of the deployment of a thorium based reactor fleet on a worldwide scale. The primary feature of the MSFR concept is the removal of the graphite moderator from the core (graphite-free core), resulting in a breeder reactor with a fast neutron spectrum and operated in the Thorium fuel cycle, as described in Section II of this paper. The MSFR has been recognized as a long term alternative to solid fueled fast neutron systems with a unique potential (excellent safety coefficients, smaller fissile inventory, no need for criticality reserve, simplified fuel cycle…) and has thus been officially selected for further studies by the Generation IV International Forum since 2008[10,11,12].

In the MSFR, the liquid fuel processing is part of the reactor where a small stream of the molten salt is set aside to be processed for fission product removal and then returned to the reactor. This is fundamentally different from a solid fuel reactor where separate facilities produce the solid fuel and process the Spent Nuclear Fuel. Because of this design characteristic, the MSFR can operate with widely varying fuel compositions. Thanks to this fuel composition flexibility, the MSFR concept may use as its initial fissile load, ^{233}U or enriched (between 5% and 30%) natural uranium or also the transuranic (TRU) elements currently produced by PWRs in the world. The characteristics (initial fissile inventory, safety parameters, and deployment capabilities) of each of these MSFR starting modes are detailed in section III, while the transition between the second and third generation reactors to the Thorium cycle is illustrated in Section IV by

considering the deployment capacities of a MSFR park in the French context.

II. MOLTEN SALT FAST REACTOR CONCEPT

II.A. Reactor Geometry

The standard MSFR is a 3000 MWth reactor with a total fuel salt volume of 18 m^3, operated between 650 and 750°C. As shown in Fig. 1, the core of the MSFR is a single cylinder (with the diameter equal to the height) where nuclear reactions occur within the flowing fuel salt.

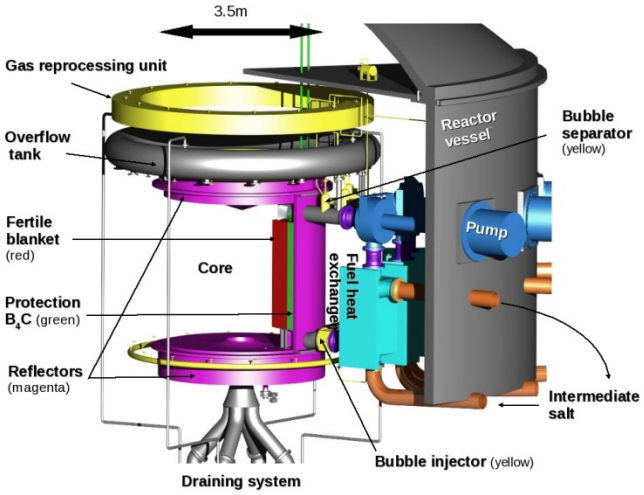

Fig. 1: Pre-design of the fuel salt circuit of the MSFR

MSFR simulations have been performed using a binary fluoride salt, composed of LiF enriched in ^7Li to 99.995 % and a heavy nuclei (HN) mixture initially composed of fertile thorium and fissile matter, either ^{233}U, $^{enriched}U$ or Pu. The $(HN)F_4$ proportion is set at 22.5 mole % (eutectic point), corresponding to a melting temperature of 565°C. The choice of this fuel salt composition rests on many systematic studies (influence of the chemical reprocessing on neutronic behavior, burning capabilities, deterministic safety level, deployment capabilities)[7,13]. This salt composition leads to a fast neutron spectrum in the core, as shown in Fig. 2 where the fast neutron spectrum of the simulated reference MSFR is compared to the spectra of 2 solid-fuel reactors: a Na-cooled Fast Neutron Reactor (FNR-Na) and a thermal Pressurized Water Reactor (PWR). The large Na capture cross-section appears clearly on the red curve at 2.8 keV, while the inelastic cross-section of fluorine is characteristic of the green curve between 0.1 and 1 MeV.

The external core structures and the fuel heat exchangers are protected by thick reflectors made of nickel-based alloys, which have been designed to absorb more than 90% of the escaping neutron flux. These reflectors are themselves surrounded by a 20cm thick layer of B_4C, which provides protection from the remaining neutrons. The radial reflector includes a fertile blanket (50 cm thick - green area in Fig. 1) to increase the breeding ratio. This blanket is filled with a fertile salt of LiF-ThF_4 with initially 22.5%- mole of ^{232}Th.

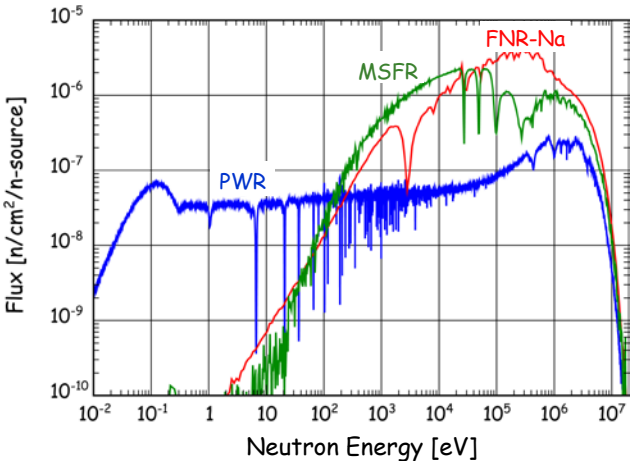

Fig. 2: Fast neutron spectra of the reference MSFR and of a Na-cooled fast neutron reactor (FNR-Na) compared to the thermalized spectrum of a pressurized water reactor (PWR)

The fuel salt flows upward in the core until it reaches an extraction area which leads to salt-bubble separators through salt collectors (see description of the gaseous extraction system of fission products in section II.B). The salt then flows downward in the fuel heat exchangers and the pumps before finally re-entering the bottom of the core through injectors. The fuel salt runs through the total cycle in around 3-4 seconds, depending on the salt flow velocity. The total fuel salt volume is distributed half in the core and half in the external fuel circuit (salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes). This external fuel circuit comprises 16 identical modules distributed around the core, outside the fertile blanket.

Finally the normal way to quickly and easily stop the nuclear reaction for the MSFR shutdown will be to drain the fuel circuit in tanks located under the core.

II.B. Reprocessing scheme

The on-site salt management of the MSFR combines a salt control unit, an online gaseous extraction system and an offline lanthanide extraction component by pyrochemistry[14,15,16]. This salt reprocessing scheme is presented in Fig. 3. The only continuous salt chemistry process is the gaseous extraction system. It consists first in injecting helium bubbles in the lower part of the core to trap the non-soluble fission products (noble metals) dispersed in the flowing liquid and also gaseous fission products. A liquid/gas phase separation is then performed on the salt flowing out of the core to extract gaseous species and dragged condensed particles.

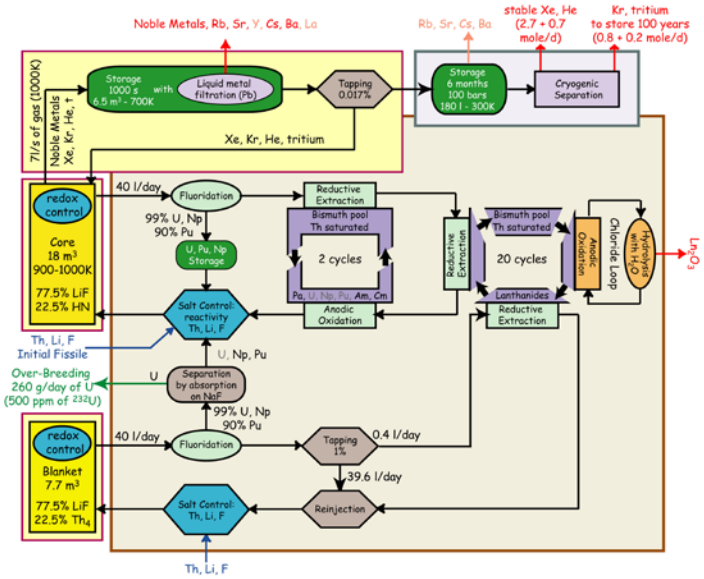

Fig. 3: Overall scheme of the fuel salt management including the online gaseous extraction (top) and the offline reprocessing unit (bottom) – The yellow boxes surrounded by a red line are enclosed within the reactor vessel

Following this "physical" process of purification, a small part of the gas is withdrawn in order to let the fission products decay and the remaining part of gas is sent back to the lower part of the core. An experimental forced flow loop of fluoride salt, the FFFER (Forced Fluoride Flow for Experimental Research) facility[17], is under construction at LPSC Grenoble and will be operated between 500 and 700°C, with a LiF-NaF-KF salt. One of its objectives is to evaluate the efficiency of this bubbling process in a fluoride salt, by reproducing the gases extraction of the MSFR at a 1/10th scale in a simulant salt at high temperature.

The salt properties and composition are monitored through the online chemistry control and adjustment unit. A fraction of salt (40 liters for the calculations of this paper) is periodically withdrawn and reprocessed offline in order to extract the lanthanides before it is sent back into the core. In this separate, batch reprocessing unit, 99% of Uranium (including ^{233}U) and Neptunium and 90% of Plutonium are extracted by fluorination and directly and immediately reintroduced in the core. The remaining actinides are then quickly extracted together with Protactinium and also sent back to the core. In the last step, the lanthanides are separated from the salt through a second reductive extraction and sent to waste disposal. The remaining salt is sent back to the fuel salt.

II.C. Simulation tools and methods

Our numerical simulations rely on the coupling of the MCNP neutron transport[18] with a home-made materials evolution code REM[16,19,20,21].

The probabilistic MCNP code evaluates the neutron flux and the reaction rates in all the parts (called cells) of the simulated system. This requires a precise description of the geometry and the characteristics of all materials involved (temperature, density, elements, isotopes, proportions), together with the interaction cross-sections of each isotope constituting the reactor.

These calculations are static, for a given and fixed state of the system. Following the reactor operation all along its life also requires simulating the temporal evolution of the system. The neutronics code thus has to be coupled with an evolution code, as displayed in Fig. 4.

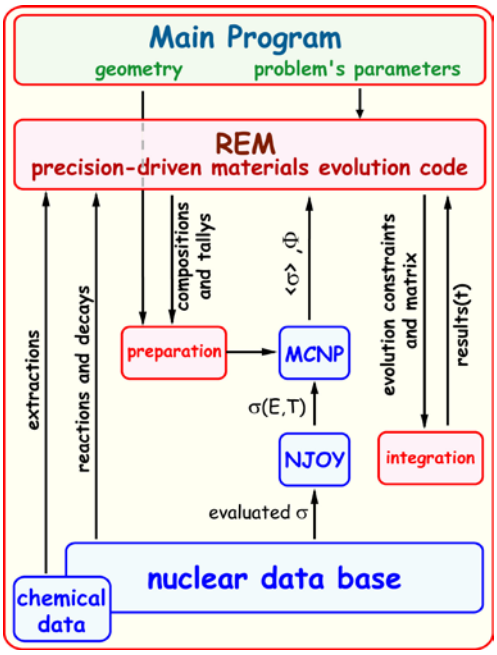

Fig. 4: Coupling scheme of the MCNP neutron transport code with the in-house materials evolution code REM

The evolution code REM solves the Bateman equations to compute the evolution of the materials composition isotope by isotope within the cells as a function of the nuclear reactions and decays occurring in the system and of external parameters like reprocessing or fuel adjustment. These last parameters are implemented through specific removal constants equivalent to decay constants. Our simulations consider several hundreds of nuclei (heavy nuclei, fission products, structural materials…) with their interactions and radioactive decays.

The simulations of reactor evolution take into account the input parameters (power released, criticality level, chemistry...), by continuously adjusting the materials composition and thus the neutron flux of the system, via multiple interactions between the neutronic and the evolution tools. The REM code is indeed a precision-driven code, i.e. it has been designed to determine the reactor evolution while controlling the precision of the results at each step of this evolution. The resolution of the

Bateman equations is constrained by several variables to keep the reactor's simulated physical parameters constant during the evolution. These include the total power (with a one percent or so precision) and the reactivity (with a huge precision of some tens pcm, much smaller than the computational uncertainty of this parameter under MCNP). The numerical integration of the Bateman equations is done using a Runge-Kutta method.

III. STARTING MODES OF THE MSFR

A fission nuclear reactor requires fissile matter to produce power. Generation 2 or 3 reactors (PWR, CANDU, EPR…) being under breeder systems, i.e. using more fissile matter than they produce, they need to be regularly re-fueled with fissile matter all along their operation time. On the contrary, breeder generation-4 reactors (SFR, MSFR, GFR…) require only one initial fissile matter load. They then produce at least the fissile matter they need to be operated all along their lifespan. Moreover, concerning molten salt reactors, only one fissile load is mandatory and not 2 as for solid-fueled reactors (one fissile load used in the reactor and one in the reprocessing/fuel manufacturing procedure) since no fuel re-fabrication is necessary and the fuel salt composition is controlled on-line without stopping reactor operation.

The only natural fissile matter on earth is ^{235}U (0.72% of natural uranium), which can be used directly as enriched uranium in breeder reactors for their initial fissile load, or which can be loaded on the side in generation 2 or 3 reactors to produce either ^{239}Pu by irradiating ^{238}U, or ^{233}U by irradiating ^{232}Th. To deploy the Thorium fuel cycle in MSFRs, we have thus investigated the following solutions:

1. Producing ^{233}U in the fertile blanket of other reactors (SFR…) or by irradiating ^{232}Th in an ADS for example, to start the MSFR directly with this ^{233}U as initial fissile matter. Once an initial park of the MSFRs based on the Th-^{233}U cycle is launched, ^{233}U will also be produced in MSFRs which are breeder reactors, allowing the deployment of such ^{233}U-started MSFRs in a second time period even if no ^{233}U is produced elsewhere.
2. Using as initial fissile matter the plutonium produced in current PWRs or in future EPRs or, even better, the mix of transuranic elements (TRU) produced by these Generation 2-3 reactors.
3. Starting the MSFR with enriched uranium as initial fissile matter, with an enrichment ratio lower than 20% due to proliferation resistance issues.
4. A mix of the previous starting modes. For example, ^{233}U may be produced by using special devices containing Thorium and Pu-MOx in current PWRs or in future EPRs.

Typical configurations of the MSFR corresponding to the different starting modes are detailed in the following paragraphs. The geometry of these MSFR configurations is identical to that presented in section II.A.

The fertile salt is always composed of LiF-ThF_4 with 22.5 mole% of heavy nuclei, just as the fuel salt made of LiF-$(HN)F_4$ with 22.5 mole% of heavy nuclei among which Thorium as fertile matter.

III.A. ^{233}U started-MSFR

The characteristics of the reference MSFR configuration started directly with ^{233}U as initial fissile matter are given in Tab. 1.

TABLE 1
Characteristics of the reference ^{233}U-started MSFR

Thermal/electric power	3000MW_{th} / 1500MW_e
Specific power (W_{th}/cm^3)	330
Fuel salt composition (mol%)	LiF (77.5%) - ThF_4 (20%) - $^{233}UF_4$ (2.5%)
Fertile Blanket Molten salt composition (mol%)	LiF (77.5%) -ThF_4 (22.5%)
Operating temperatures input/output (°C)	650 / 750
Initial heavy nuclei inventory per GWe	Th: 25.6 tons ^{233}U: 3.26 tons
Density (g/cm^3) [22]	4.1
Dilatation coefficient (/K) [22]	10^{-3}
Batch reprocessing rate	40 l of fuel salt / day
Th consumption (kg/year/GWe)	740
^{233}U production (kg/year/GWe)	63
Breeding ratio	1.08

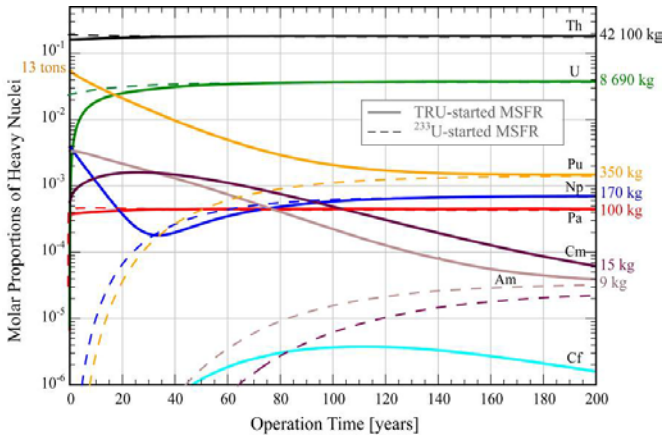

Figure 5: Time evolution up to equilibrium of the heavy nuclei inventory for the ^{233}U-started MSFR (dashed lines) and for the TRU-started MSFR (solid lines)

Its initial heavy nuclei inventory per GWe comprises 3.26 tons of ^{233}U and 25.6 tons of ^{232}Th. Fig. 5 (dashed lines) illustrates the evolution of the heavy nuclei

inventoried in the fuel salt all along the operation of this reactor, up to equilibrium. The proportion of minor actinides in the salt remains low: around one percent at equilibrium.

Regarding safety issues, the feedback coefficient of this configuration, equal to -5 pcm/K, is largely negative and remains stable during the reactor lifespan[7].

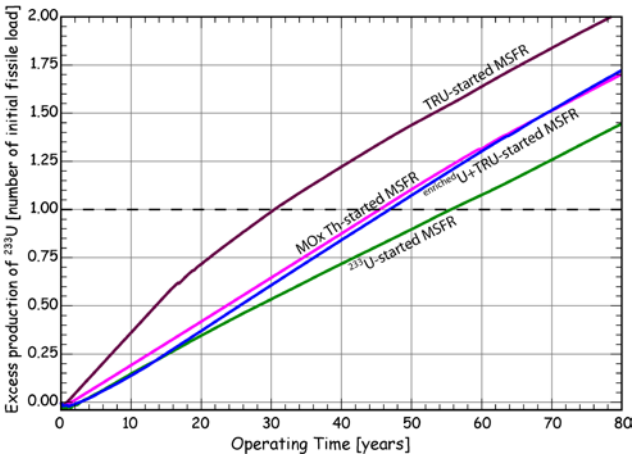

Figure 6: Excess production of ^{233}U for the different starting modes of the MSFR all along the reactor lifespan, in number of initial fissile load produced

Finally, when considering the deployment capabilities of such a MSFR, this configuration corresponds to a breeder reactor producing 95kg of ^{233}U in excess per year, corresponding to a reactor doubling time of 56 years as shown in Fig. 6. As previously indicated, these values are obtained with a simulated batch reprocessing rate of 40 liters of fuel salt per day, corresponding to a reprocessing of the whole core in 450 days. The breeding ratio, and thus this reactor doubling time, may be controlled through the reprocessing rate of the reactor as detailed in Table 2. Because of the fast neutron spectrum, the fission product capture cross-sections are small, so that the neutronic characteristics of the reactor as well as the breeding ratio are only slightly sensitive to the fission product extraction.

TABLE 2

Influence of the reprocessing rate on the breeding capabilities of the ^{233}U-started MSFR

Whole core reprocessing time	Breeding ratio	Excess ^{233}U produced per year	Reactor doubling time
225 days	1.09	103 kg	49 years
450 days	1.08	95 kg	56 years
900 days	1.065	78 kg	67 years

III.B. MSFR started with transuranic elements

Plutonium and the minor actinides (neptunium, americium and curium) produced in Generation 2-3 reactors may also be used as initial fissile matter in a MSFR. This would also allow to close the current fuel cycle while launching the Thorium fuel cycle. The mix of minor actinides used as initial fissile load in these calculations is detailed in Table 3 (third column). It corresponds to a UOX fuel after one use in a PWR without multi-recycling, for a burnup of 60 GWd/ton and after five years of storage[23]. The evolution of the heavy nuclei inventories for a TRU-started MSFR is displayed in fig. 5 (solid lines).

The utilization of TRU elements to start the reactor increases the initial amounts of minor actinides compared to the ^{233}U-started MSFR. But at equilibrium, the fuel salt compositions of TRU-started and ^{233}U-started MSFRs are identical, these TRU being converted into ^{233}U. Th, Pa, U and Np reach their equilibrium composition quickly, while some tens years are necessary to burn 90% of the Pu and a hundred years for Am and Cm. The Cm in core inventory reaches a maximum of 390 kg (among which 265 kg of ^{244}Cm) after 26 years of operation.

TABLE 3

Initial heavy nuclei inventories per GWe of the different starting modes of the MSFR

Starting mode	^{233}U [kg]	TRU (Pu UOx) [kg]	Th Pu-MOx [kg]	^{enr}U + TRU [kg]
Th 232	25 553	20 396	18 301	10 135
Pa 231			20	
U 232			1	
U 233	3 260		2 308	
U 234			317	
U 235			45	1 735
U 236			13	
U 238				11 758
Np 237		531	54	335
Pu 238		229	315	144
Pu 239		3 902	1 390	2 464
Pu 240		1 835	2 643	1 159
Pu 241		917	297	579
Pu 242		577	1 389	364
Am 241		291	1 423	184
Am 243		164	354	104
Cm 244		69	54	44
Cm 245		6		4

The deployment capacities of this MSFR configuration are better than those of the ^{233}U-started MSFR with a production of 120 kg of ^{233}U in excess per year during the first 30 years of operation, corresponding to a reactor doubling time of 30 years as shown in Fig. 6.

In the case of this TRU-started MSFR configuration, we have to pay a special attention to solubility issues of valence-3 elements (lanthanides and plutonium) in the fuel salt. The molar proportion of Pu in this configuration is

5.5% at the beginning (see Fig. 5), corresponding to 12 970 kg. 0.2% of valence-3 elements are progressively added due to the accumulation of lanthanides in the MSFR for a reprocessing rate of 40l per day as described in section II.B. Partial solubility data are available from experimental measurements performed at BARC in the 1970's and are listed in Tab. 4, their uncertainties being unknown. These values of Pu solubility in a fluoride salt, even though neither complete nor precise, are the only ones available. New measurements are necessary to verify and complete them.

TABLE 4

Plutonium solubility in $\%PuF_3$ for different $LiF\text{-}ThF_4$ compositions[24]

$LiF\text{-}ThF_4\%$	550°C	650°C	750°C
65 - 35	2.11	4.12	7.08
70 - 30	2.69	4.91	7.61
75 - 25	2.75	4.79	7.31
80 - 20	2.85	4.97	7.71

According to these data, the initial Pu proportion of the TRU-started MSFR reaches the solubility limit given for a $LiF\text{-}ThF_4$ salt. A solution may be to operate the reactor at a higher temperature, around 700°C, during the first ten years. Another possibility is to limit the Pu initial proportion in the fuel salt and to add small amounts of ^{233}U or $^{enriched}U$ to reach criticality. These starting modes combining TRU elements and either ^{233}U or enriched U as initial fissile load of the MSFR are described in the next two paragraphs.

III.C. MSFR started with enriched Uranium and TRU

Optimization studies[25] have highlighted an interesting configuration of the MSFR started with TRU elements and enriched uranium, after considering solubility limits, proliferation resistance, initial fissile inventory and breeding capacities of the reactor. The plutonium concentration has been fixed to 2/3 of the estimated solubility limit at 650°C. The initial fuel salt of this reference configuration also contains 35% of ThF_4 and uranium enriched to 13%. The calculated evolution of the actinide composition of this fuel salt during reactor operation is displayed in Fig. 7, compared to the TRU-started MSFR presented in paragraph III.B.

In this MSFR configuration, the initial Pu concentration is equal to 3%, thus largely below the solubility limit, and remains stable during 20 years before decreasing slowly, due to its production from the ^{238}U initially inserted in the core. The maximal amount of Cm is here equal to only 230 kg (with 155 kg of ^{244}Cm). This starting mode thus leads to lower TRU concentrations but they stay longer in the fuel salt.

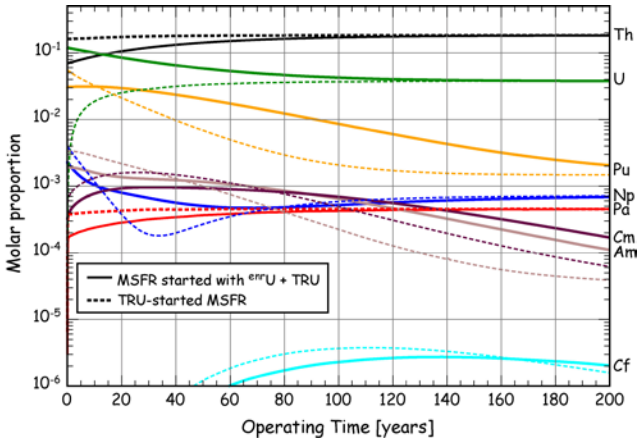

Figure 7: Time evolution up to equilibrium of the heavy nuclei inventory for the MSFR started with enriched Uranium and TRU elements (solid lines) compared to the TRU-started MSFR (dashed lines)

The deployment capacities of this MSFR configuration lie between the ^{233}U-started MSFR and the TRU-started MSFR, with a reactor doubling time of 45 years (see figure 6).

III.D. MSFR started with a mix of ^{233}U and TRU

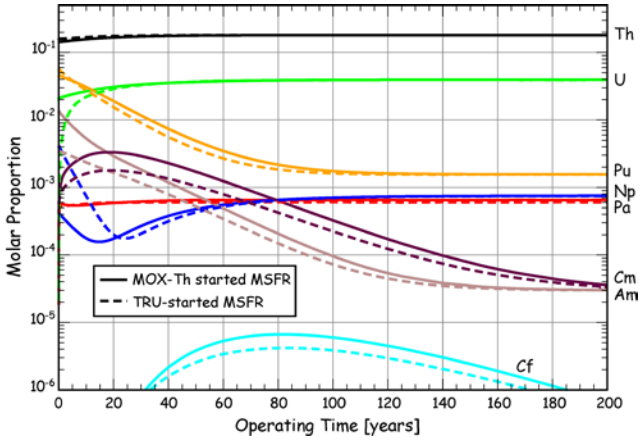

Figure 8: Time evolution up to equilibrium of the heavy nuclei inventory for the MSFR started with ^{233}U and TRU (solid lines) compared to the TRU-started MSFR (dashed lines)

In this case, the initial fissile load corresponds to the production of ^{233}U by using a Th-Pu MOx fuel in an EPR[26]. As detailed in Table 3 (fourth column), it results in a mix of different isotopes of Uranium, mainly ^{233}U, together with TRU elements. The evolution of the heavy nuclei inventories for this MSFR is displayed in fig. 8 (solid lines). Thanks to the initial addition of ^{233}U, the molar proportion of Pu in this configuration reaches only 4.5%, i.e. in theory below the solubility limit. The maximal amounts of Am, Cm and Cf are higher here compared to the previous TRU-started MSFR, due to the use of MOX fuel. The two configurations are identical after around 20 years, except for Cm and Cf.

The deployment capacities of this MSFR configuration are identical to those of the MSFR started with enriched U

and TRU elements, i.e. a reactor doubling time of 45 years for the reprocessing rate considered here.

If UOx fuel from Generation 2 or 3 reactors is still available, another way to start an MSFR is to use this fuel mixed with ^{233}U produced by breeder MSFRs based on another starting mode. This also allows a complete closure of the current fuel cycle. Such MSFRs are quite similar to the MSFRs presented at the beginning of this paragraph, in terms of heavy nuclei inventories and deployment capacities.

III.E. Incinerator version of the MSFR

Ultimately when fission based electricity production will be replaced by a novel technology (fusion for instance) all the actinides inside reactors will become discardable wastes. The possibility of eventually shutting down the running reactor parks has to be studied in so-called end-of-game scenarios, the heavy nuclei management being the key issue. If minor actinide losses during reprocessing are less than 0.1% and if the whole fuel salt volume is reprocessed between 1 and 5 years, then the actinides in-core inventory is larger than the losses for at least 1000 to 5000 years. Concerning long-term radiotoxicity issues, finding ways to further reduce the final HN inventory is thus more important than improving the reprocessing efficiency.

TABLE 5

Initial heavy nuclei inventories before/after incineration

HN inventory [kg]	9.4 MSFR	Incinerator after 60 years of operation	Burning rate
U	72 751	6 407	11.5
Np	1 381	506	2.8
Pu	2 768	1 530	1.8
Am	72	39	1.8
Cm	33	64	0.5
Total	77 005	8 550	9.1

We have studied a molten salt reactor used as incinerator to burn (and thus reduce) the final HN inventories of MSFR in 60 years. This MSR incinerator is identical to the MSFR in terms of system geometry and power production, and differs in the fuel salt composition and the removal of the fertile blanket.

We have considered a fuel salt made of 46.5% ^{7}LiF, 11.5% NaF, 41.7% KF and $(HN)F_4$ whose melting point is sufficiently low even with a small HN proportion (since there is no Th) in the salt and allowing a neutron spectrum that is not too thermalized. The initial heavy nuclei load evaluated to reach criticality is equal to 685 kg of transthoric elements (transTh) contained in the final heavy nuclei inventories of the MSFRs presented in the previous paragraphs. The incinerator is also fueled with these transTh final inventories discharged from the MSFRs to maintain reactivity all along the reactor operation, leading to the incineration of 9.4 final HN inventories of a MSFR as detailed in Table 5. The total burning rate of transTh elements is equal to 9.1, leading to a reduction by one order of magnitude of the long-term radiotoxicity in the period of 10^3 to 10^6 years (see figure 9), mainly thanks to the destruction of the ^{233}U stockpile.

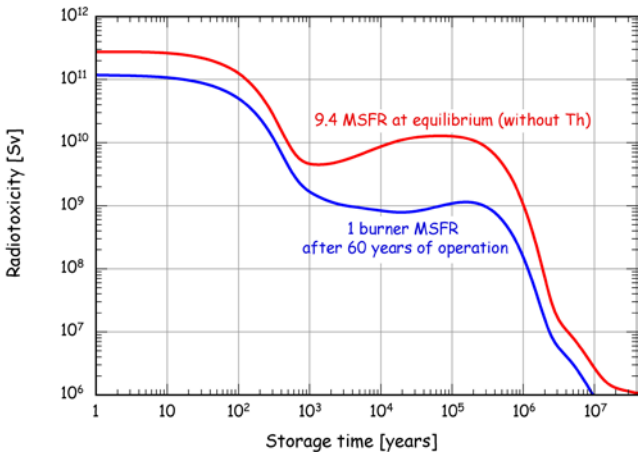

Figure 9: Time evolution of the radiotoxicity due to final heavy nuclei inventories of MSFR with and without a final incineration

IV. DEPLOYMENT SCENARIOS

A standing question is whether a park of MSFRs can be deployed given the absence of naturally available ^{233}U, both at the national, European and worldwide scales. In this paper, we will illustrate the deployment capacities of the MSFR at the French national scale.

The deployment scenarios of a park of nuclear reactors also led to an estimation of the production of heavy nuclei induced by the deployment of such a reactor park. We aim at evaluating the complexity of the management of these heavy nuclei stockpiles, as well as their radio-toxicity.

To illustrate the deployment capacities of the MSFR concept, we present here the following French scenario, displayed in figure 10: we have considered that the natural uranium resources available were large enough to require generation 4 reactors in 2070 only. The deployment scenario starts with the historical French nuclear deployment based on light water reactors (PWRs followed by EPRs)[23,27]. From 2040, some Generation 3 reactors will be fuelled with Pu-Uox in a Thorium matrix to reduce the minor actinide production and to prepare the launching of the Thorium fuel cycle in MSFRs. This park of Generation 3 reactors will then be progressively replaced by MSFRs using this Th-Pu MOx fuel from the last Generation 3 reactors, as described at the beginning of paragraph III.D. The deployment is finally completed with MSFRs started with a mix of ^{233}U produced in the previous MSFRs and the

remaining stockpiles of Pu-UOx and Pu-MOx irradiated in the light water reactors.

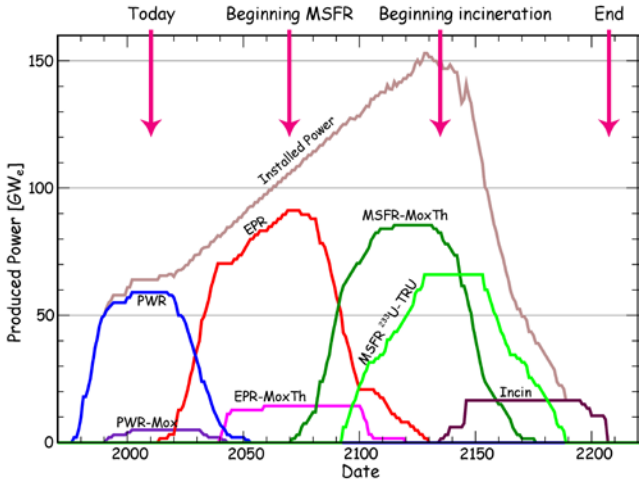

Figure 10: French nuclear power deployment based on PWRs, EPRs and MSFRs

Assuming a decision made in the first half of the XXIIth century to progressively and definitely stop fission nuclear energy production, this scenario ends with the introduction of incinerators (as detailed in paragraph III.E) to optimize the end-of-game scenario and to further reduce the TRU final inventories of the MSFRs after their shut-down.

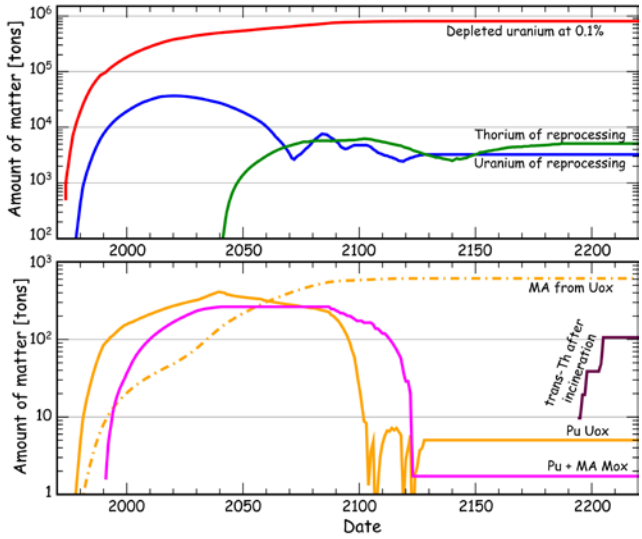

Figure 11: Evolution of the stockpiles of actinides during the scenario deployment

The final stockpiles of radioactive elements other than the fission products to be managed after the end of this nuclear fission deployment are the following, as presented in figure 11:

- Depleted uranium at 0.1%: 803 700 tons
- Uranium from reprocessing (minimized by the scenario management): 3 250 tons
- Irradiated Thorium: 5 100 tons
- Irradiated Uox fuel (minimized by the scenario management) represented in figure 11 by its Pu content (named 'Pu-Uox'): 5 tons of Pu standing for 450 tons of irradiated Uox
- Irradiated Mox fuel (minimized by the scenario management) represented in figure 11 by its Pu content (named 'Pu+MA Mox'): 0.76 tons standing for 12.4 tons of irradiated Mox
- Minor actinides separated from the Pu when the latter is used as Mox fuel in light water reactors, and vitrified (named 'MA from Uox'): 612 tons
- Final inventories of the incinerators: 106 tons

The evolution of the radiotoxicity corresponding to the final radioactive stockpiles of this scenario is displayed in figure 12, where it appears that the short-term radiotoxicity (the first tens of years) is dominated by the fission products (FP) while the long-term radiotoxicity (10^3 to 10^6 years) is mainly due to the vitrified minor actinides produced in light water reactors and not re-used in Mox fuel.

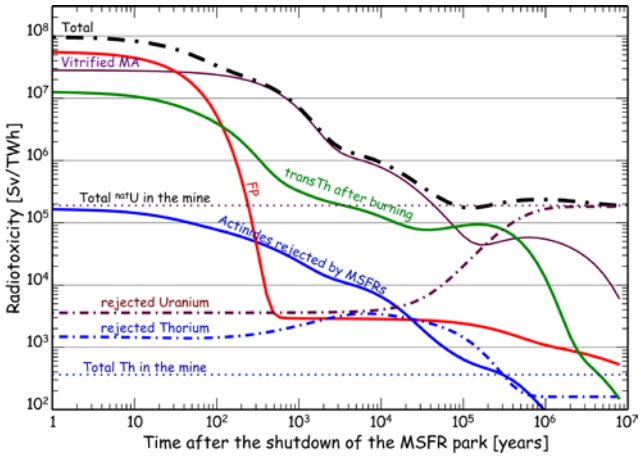

Figure 12: Time evolution of the various contributions to the radiotoxicity of the final radioactive stockpiles

V. CONCLUSIONS

In the frame of a major re-evaluation of the molten salt reactor (MSR) concept, and starting from the Molten Salt Breeder Reactor project at Oak-Ridge, we have performed parametric studies in terms of safety coefficients, reprocessing requirements and breeding capabilities. Our recent studies have highlighted the MSR configurations operated with a fast neutron spectrum in the Thorium fuel cycle, the Molten Salt Fast Reactor (MSFR), as robust and very promising. It has been selected for further studies by the MSR steering committee of the Generation IV International Forum.

The standard MSFR is a 3000 MW_{th} reactor with a total fuel salt volume of 18 m^3, operated between 650 and 750°C. In the MSFR, the liquid fuel processing is part of the reactor where a small side stream of the molten salt is processed for fission product removal and then returned to

the reactor. Because of this characteristic, the MSFR can thus operate with widely varying fuel compositions. Thanks to this fuel composition flexibility, the MSFR concept may use as initial fissile load, ^{233}U or enriched uranium or also the transuranic elements currently produced by light water reactors.

Our studies show that the MSFR configurations corresponding to various starting modes of the reactor are all characterized by excellent safety coefficients and have the same very good deployment capacities. Optimizing the specific power in the MSFR configuration started directly with ^{233}U as initial fissile matter has allowed a reduction of the initial fissile inventory down to 3 metric tons per GWe. The MSFR is characterized by a low proportion of minor actinides in the salt (around one percent at equilibrium) and by its excellent safety coefficients (-5 pcm/°C).

The TRU-started MSFR is able to efficiently convert the plutonium and minor actinides from generation 2-3 reactors in ^{233}U while improving the deployment capabilities of the MSFR concept. Its only drawback lies in its high initial plutonium concentration above its estimated solubility limit. To overcome this limitation while still using TRU elements in the initial fissile load of the MSFR to close the current fuel cycle, we have proposed two optimized solutions: mixing the TRU elements at a lower concentration (around 3 to 4 mol%) with either natural uranium with an enrichment ratio of 13% or ^{233}U produced in other reactors.

Finally the French nuclear deployment scenario presented here illustrates that launching the Thorium fuel cycle in Molten Salt Fast Reactors is feasible and efficient while closing the current fuel cycle and optimizing the long-term wastes management.

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