Influence of the Processing and Salt Composition on the Thorium Molten Salt Reactor

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Abstract: *Molten Salt Reactors (MSRs) are one of the six systems retained by Generation IV as a candidate for the next generation of nuclear reactors: Molten Salt Reactor is a very attractive concept especially for the Thorium fuel cycle which allows nuclear energy production with a very low production of radiotoxic minor actinides, so that it has been selected by the Generation-IV International Forum. Its main characteristic is the strong coupling between neutronics and salt processing. Such nuclear reactors use a liquid fuel which is also the coolant. Elements produced during the reactor's operation, like fission products or transuranic elements, modify the neutronic balance of the reactor by capturing neutrons. As the fuel is liquid, partial processing of limited amount taken from circulating salt is possible, in order to remove the poisoning elements, without stopping reactor operation. In this article, we present a configuration, which we consider as a reference one for a Thorium Molten Salt Reactor (TMSR) and we study the influence of efficacy of different types of processing on the neutronic behaviour of this reactor. By considering both the possibilities in chemistry and the neutronic effects, our aim is to work out an efficient, reliable and realistic processing scheme.*

The processing includes in fact two components: an in-line bubbling system within the reactor which extracts the gaseous and metallic fission products quickly, and a slower external processing unit that extracts the other fission products. A salt volume equal to the core volume is thus cleaned in several months. We have studied the influence of different processing rates on the reactor's behaviour. This mainly affects the breeding ratio.

Properties of the salt are also crucial. We choose in our simulations of the TMSR a 78 mole % LiF – 22 mole % (Heavy Nuclei)F_4 salt for the fuel, but lower heavy nuclei (HN) proportions in the fuel salt are also examined, in order to minimize the Uranium-233 inventory in the reactor. The neutron spectrum is largely modified by the HN proportion and has a deep impact on the reactor behaviour. Our simulations evaluate the degradation of breeding ratio from more than 1 for the reference configuration down to 0.86 due to a decrease of the HN proportion in the fuel salt.

We conclude that the simplification of the salt processing which is addressed in this work improves the feasibility of the TMSR system.

Keywords: Molten Salt Reactor, Thorium fuel cycle, TMSR, Processing, Breeding

1. Introduction

The Generation-IV International Forum for the development of new nuclear energy systems has established a set of goals as research directions for nuclear systems: 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 by Generation IV. MSRs are based on a liquid fuel, so that their technology is fundamentally different from the solid fuel technologies currently in use and lead to specific advantages in terms of safety/reliability [ROS70]. Furthermore, this type of reactor is particularly well adapted to the Thorium fuel cycle (Th-^{233}U) which has the advantage of producing less minor actinides than the Uranium-Plutonium fuel cycle (^{238}U-^{239}Pu) [LEC01][NUT05][LEB07]. Moreover, while breeding in the U-Pu cycle can be obtained only with a fast neutron spectrum, in the Th-^{233}U fuel cycle it can, in principle, be obtained either in a thermal or in a fast neutron spectrum. In a thermal neutron spectrum, poisoning due to the fission products being worse than in a fast neutron spectrum, the rate at which fuel processing is performed can become a major issue. Because, in an MSR, the fuel is liquid, continuous extraction of the fission products is a possibility. Although MSRs can also be operated as burner reactors, they will be discussed in this paper only as energy producing critical systems.

This Molten Salt Reactor (MSR) concept was developed in the early 1950's at the Oak Ridge National Laboratory (ORNL). The point was to conceive reactors whose fuel would be liquid, serving both as fuel and as coolant. The primary advantage of this concept is to allow continuous adjustment of the fuel salt composition,

and thus ensure its operation over time with no reactivity reserve. Moreover, it is possible to reach very high temperatures and very high power densities without large internal pressures, therefore without severe constraints for the structural elements. The first experimental MSR stemmed from an American military program concerning plane propulsion which led to the Aircraft Reactor Experiment (ARE) [BRI57] in 1954. This 2.5 MWth reactor ran satisfactorily during 100 hours. The continuation of this work led to the commissioning of the Molten Salt Reactor Experiment (MSRE) [HAU70]. Generating 8 MW_{th} of power, the MSRE was operated with different fuels (^{235}U, ^{233}U, mixing Pu-^{233}U) from 1965 to 1969. The success of this reactor and the expertise gained during this experiment led, in the 1970s, to the elaboration of a power reactor project, the Molten Salt Breeder Reactor (MSBR) [BET70] [ROS72] [WHA70]. Even though the concept looked promising, the studies were stopped in 1976. These studies demonstrated that fuel breeding is possible with the Thorium fuel cycle in a thermal spectrum, provided very efficient and, as a consequence, constraining, on-line chemical processing of the salt is achieved. Over the past few years, the MSBR has been reassessed in the light of new calculating methods [LEC01] [NUT05] so as to create a new reactor concept that we call the Thorium Molten Salt Reactor (TMSR) [MAT05] [MAT06] and which is described in this paper.

The results presented in this paper are based on calculations, obtained thanks to the coupling of a neutron transport code called MCNP [BRI97] with an in-house materials evolution code REM [MAT05]. The former calculates the neutron flux and the reaction rates in all the cells while the latter solves the Bateman equations for the evolution of the materials composition in the cells. These calculations take into account the input parameters (power released, criticality level, chemistry), by adjusting the neutron flux or the materials composition of the core on a regular basis. Our calculations are based on a precise description of the geometry and consider several hundreds of nuclei with their interactions and radioactive decay; they allow fine interpretation of the results. All the data discussed in this paper result from the simulation of the reactor operation over 100 years.

2. The Thorium Molten Salt Reactor

2.1 Reactor Definition

Our standard system, called Thorium Molten Salt Reactor or TMSR is displayed on figure 1. It is a 1 GWe graphite moderated reactor. Its mean operating temperature is 630°C with a maximum variation of 100 degrees, and its thermodynamic efficiency is 40 %. The graphite matrix comprises a lattice of hexagonal elements with 15 cm sides. The total diameter of the matrix is 3.20 m. Its height is also 3.20 m. The density of such nuclear grade graphite is 1.86. The salt runs through the middle of each of the elements, in a channel whose radius is 8.5 cm. One third of the 20 m^3 of fuel salt circulates in external circuits and, as a consequence, outside of the neutron flux. A Thorium and Graphite radial blanket surrounds the core so as to improve the system's breeding capability. The properties of the blanket are such that it stops approximately 80 % of the neutrons, thus protecting external structures from irradiation while improving breeding. The processing of the whole blanket with extraction of the ^{233}U produced within is spread over a 6 month period.

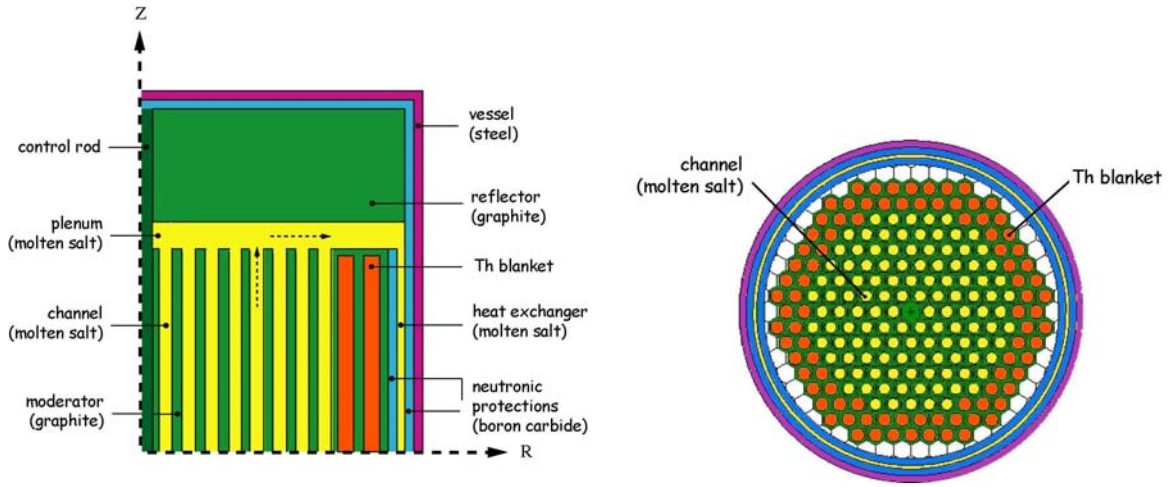

Figure 1: Vertical Cut of a quarter of the TMSR (left) and Horizontal Cut of the TMSR (right)

The salt used in the planned TMSR is based on the binary mixture LiF – $(HN)F_4$ (where HN stands for all Heavy Nuclei), the total proportion of $(HN)F_4$ being fixed to 22 mole %. That corresponds to a eutectic melting temperature of about 565 °C, lower than the operation temperature of 630°C chosen for our simulations. The salt density at 630°C is 4.3 with a dilatation coefficient of 10^{-3}/°C [WAL03]. The salt is composed initially by LiF, ThF_4 and $^{233}UF_4$, with an initial inventory of fissile material (^{233}U) of 1.9 metric tons.

2.2 Chemical Processing Scheme

The MSBR project was discontinued before the reactor's building. To reach the MSBR goal, which was to obtain as high a breeding ratio as possible, the on-line chemical processing unit was designed to process the entire salt volume within 10 days. This drastic condition involved very complex processing flowsheet [WAL03]. The high breeding ratio required in the case of the thermal spectrum of the MSBR, and which led to serious handicap for salt processing, is not essential in the faster neutron spectrum of our under consideration TMSR. As a consequence, nowadays, a fast on-line processing is no more necessary and simpler process may be sufficient. In this paper, we discuss the impact of a new simplified and adapted processing on the neutronic behaviour of the TMSR.

2.2.1 The Aims of the Processing

As detailed in reference [LEB07], a reactor based on the Th-^{233}U fuel cycle has a tight neutron balance, especially in a thermal to epithermal neutron spectrum. We have thus to minimize neutron losses. As a consequence, any element consuming neutrons by capture has to be evacuated if not necessary for the reactor operation. When fissions occur in the molten salt, Fission Products (FPs) and TRansUranic elements (TRU) are formed, as shown on figure 2, where the elements proportions have been calculated for the equilibrium state of the reactor. Quantitative distribution of fission products (elements with a charge Z between 28 and 70) shows that Zr, La, Ce, Pr and Nd have major contributions. For atomic number higher than 90, beside Thorium and Uranium which remain in a majority, the transuranic elements Protactinium (Pa) and Neptunium (Np) have important contributions too. All these elements impact the neutronic balance of the reactor by capturing neutrons. It implies that transmutation of ^{232}Th into ^{233}U is degraded.

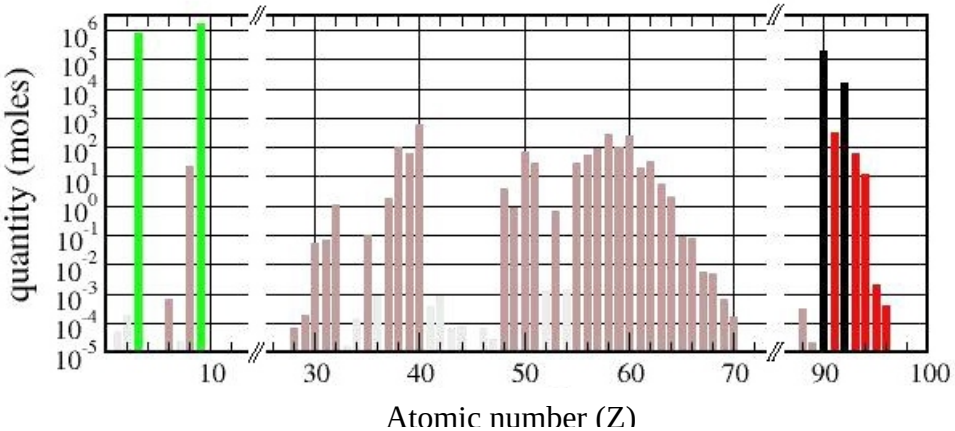

Figure 2: Composition of the TMSR at equilibrium. Fluorine and Lithium are in a majority for elements with a charge Z lower than 10. Quantitative distribution of fission products (elements with a charge Z between 28 and 70) shows that Zr, La, Ce, Pr and Nd have major contributions. For atomic number higher than 90, beside Thorium and Uranium which remain in a majority, the transuranic elements Protactinium and Neptunium have important contributions too.

A precise calculation of the neutron captures in our molten salt fuel reveals that the major part of the parasitic neutron absorptions is due to fission products and, in a second way, to transuranic elements. By studying the contribution of the fission products to the neutron capture, we can notice that some are more problematic than others. The left part of figure 3 and Table 1 reports the most poisoning isotopes. One can notice the predominance of the ^{149}Sm over all other fission products, followed by the ^{147}Pm, the ^{147}Nd, the ^{151}Sm, the ^{145}Nd and the ^{152}Sm. From a physico-chemical point of view, a most relevant way of considering the problem consists in gathering these fission products by element and not by isotope. The distribution of the poisoning fission products is presented by element on the right part of figure 3 and Table 1.

Samarium is still the most disturbing poison. The following ones, like Neodymium or Promethium, belong to the same chemical family, the lanthanides. All the elements of this family are drawn in the same colour to highlight their predominant contribution to the neutron captures. Among the other contributing fission products, Zirconium has a noticeable part, representing more than 50 % of the captures produced by other elements than lanthanides.

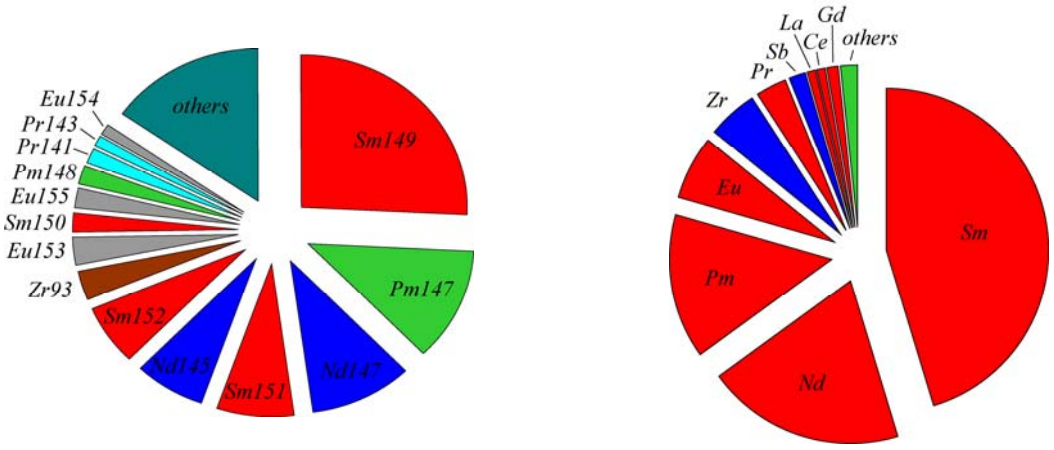

Figure 3: Repartition of the capture rate in the fission products by isotope (left) and by chemical element (right) in the core after Helium bubbling only

Table 1: Repartition of the capture rate in the fission products by isotope (left) and by chemical element (right) in the core after Helium bubbling only

Isotope	Proportion of the capture rate
Sm149	25,11%
Pm147	11,56%
Nd143	9,58%
Sm151	7,47%
Nd145	6,57%
Sm152	6,31%
Zr93	3,01%
Eu153	2,84%
Sm150	2,05%
Eu155	1,98%
Pm148	1,92%
Sm147	0,89%
Others	20,71%

Element	Proportion of the capture rate
Sm	45,39%
Nd	19,50%
Pm	14,54%
Eu	6,38%
Zr	4,96%
Pr	3,05%
Sb	1,67%
La	0,96%
Ce	0,89%
Gd	0,89%
Others	1,77%

2.2.2 The Slow Delayed Processing Procedure

The standard processing we consider is divided in two parts. The first one consists in a bubbling system in the reactor which extracts quickly the gaseous and metallic fission products. We assume that helium bubbling in the salt circuit is able to extract the gaseous fission products and the noble metals within 30 seconds. The second one is a slower and outside extraction of the other fission products. The schematic processing procedure is shown on Figure 4 [MAT05].

In the slow processing part, a salt volume equal to the core volume is cleaned in several months. This slow processing part comprises first a fluorination stage [CAR71] for Uranium removing and re-injection in the salt preparation and control unit. Thorium is also extracted in a following step and re-injected in the same unit. Outside of Uranium and Thorium elements, treatment is spread out according to optional batch processes: slow delayed processing of the salt with external storage of the Protactinium, complete extraction of the fission products and the transuranic elements.

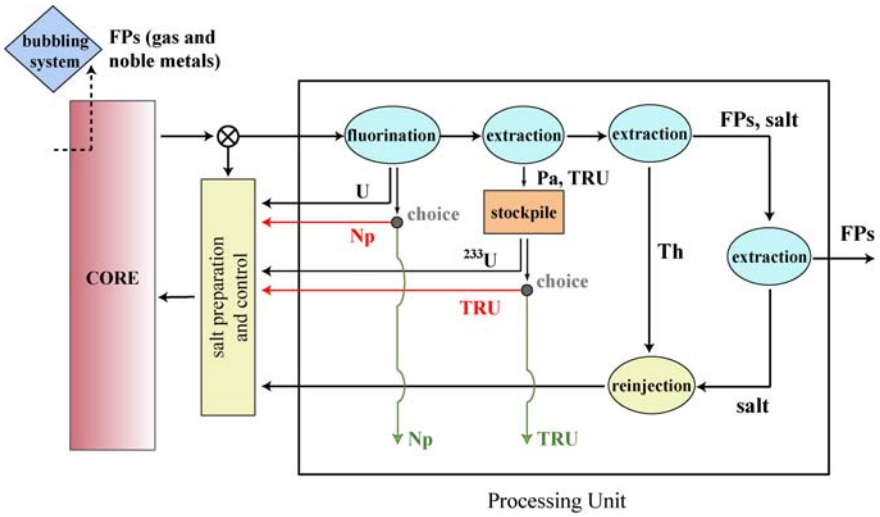

Figure 4: Processing Overview (FPs standing for Fission Products, TRUs for TRansUranic elements, Pa for Protactinium and Np for Neptunium) [MAT05].

Some of the stages shown in this general schematic flow sheet, such as Protactinium storage or extraction of transuranic elements, can be eliminated while maintaining the primary assets of the processing. The difficult part of the processing is Fission Product extraction (as listed in Figure 3) in the presence of Thorium. We exploit the idea, made possible by slow processing, of first extracting the Thorium, so as to avoid being handicapped by its presence in the extraction process of fission products. This method could not be applied in the MSBR because of the large Thorium flow involved, reaching several tons per day while it is only a few hundreds of kilograms per day in the case of a six month processing time.

If the time needed to process the core volume is equal to the time before re-injecting the salt, there is as much salt outside the core as inside it. Thus, up to 6 months can separate the extraction of the fuel salt and its re-injection in the core, after removal of the fission products. The fissile matter inventory is not increased, however, thanks to the possibility of extracting the uranium during a preliminary fluorination stage. In the case of slow processing, we assume that very good extraction efficiencies[1] will be possible owing to favourable conditions (small quantities and long time of treatment).

The impact of this processing type on the other parameters of the reactor is studied in the following section.

3. Influence of the Processing on the TMSR Neutronic Behaviour

In this section, we examine how the salt processing scheme of the TMSR impacts the neutronic behaviour of the reactor and in particular the system's fuel breeding capability. The parameter, which has the greatest influence, is the time spreading out of processing.

3.1 Fuel Breeding Capability

The Thorium fuel cycle uses the following reactions:

$$^{232}\mathrm{Th} + \mathrm{n} \rightarrow {}^{233}\mathrm{Th} \rightarrow {}^{233}\mathrm{Pa}\ \text{(half-life of 27 days)} \rightarrow {}^{233}\mathrm{U}$$

The breeding ratio expresses the balance between the creation of ^{233}U through neutron capture on ^{232}Th and the destruction of ^{233}U through fission or neutron capture. The breeding ratio in a critical reactor can thus be written:

$$BR = \frac{r_{c,^{232}Th} - r_{c,^{233}Pa}}{r_{f,^{233}U} + r_{c,^{233}U}} \qquad (1)$$

r_c and r_f being respectively the capture yield and the fission yield of the different isotopes.

[1] These efficiencies have been set to 1 in the calculations (their precise values having to be measured by further chemical studies). But a variation of this efficiency value is equivalent to a variation of the processing rate: an efficiency of 10% instead of 100% will simply result in a processing time 10 times larger.

A breeding ratio less than 1 implies that ^{233}U is consumed so that fissile matter must be fed into the core on a regular basis. In order to satisfy the breeding constraint, we try to achieve a breeding ratio at least equal to 1.

3.2 Impact of the Processing Time on the Breeding Ratio

In Table 2, we present the breeding ratios obtained at equilibrium for the reactor configuration previously described (§ 2.1), if various processing options are applied in addition to the bubbling step and ^{233}U recovery in the blanket : MSBR type treatment, our proposed slow treatment spread out on different duration and no treatment. The MSBR type processing is labelled “fast (10 days)” because of the rate at which the Protactinium is to be extracted. However, the extraction of the fission products is partial, making the real processing rate longer (equivalent to 50 days for the fission products, which induces most capture).

The best breeding ratio is, logically, obtained with the MSBR processing and the worst with no processing.

Table 2: Breeding ratio for several processing options

Processing duration	Breeding Ratio
Fast (10 days)	1.062
Slow (3 months)	1.024
Slow (6 months)	1.000
Slow (1 year)	0.986
Slow (2 years)	0.961
Bubbling only	0.562

The ^{233}U stockpile corresponding to the breeding ratios of Table 2 are displayed on figure 5 for the same respective processing options.

Varying the processing time from 3 months to 2 years induces about a 0.06 loss in the breeding ratio, while the ^{233}U stockpile goes from a “production” of 20 kg by day (over-breeder reactor) to a consumption of 40 kg by day (under-breeder reactor). The change in the breeding ratio is due mainly to the change in the capture rate of the fission products and, to a lesser degree, of the transuranic elements. On the contrary, with fast processing, 80% of the Protactinium is stored outside of the neutron flux instead of 30 % for a 3-months processing; that is the direct cause of the MSBR's good breeding ratio. Thus, unless it is extracted rapidly, the incidence of the Protactinium treatment on breeding is minor.

We now know the leeway afforded by the processing, since a doubling of the processing time induces a breeding ratio loss of about 0.02 and a variation of about 10 kg per year in the ^{233}U stockpile.

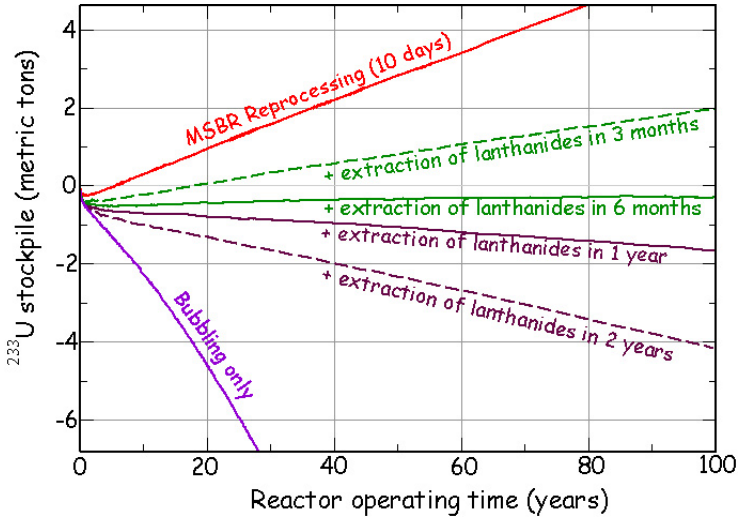

Figure 5: ^{233}U stockpile in the TMSR for different processing times, compared to the MSBR processing

3.3 Influence of the bubbling extraction on the waste management

3.3.1 Influence of the bubbling extraction on the processing of lanthanides

Every fission occurring in the fuel results in the production of two fission fragments away from the stability line. The corresponding nuclei are stabilized through successive β decays following an isobar (the mass number A remaining constant). These β decays are characterized by time constants that increase along with the decay chain. Every element has an isotope with a mean lifetime in the same order of the bubbling time. For these isotopes, the bubbling efficiency strongly impacts their extraction probability before they decay. This directly influences the salt composition, especially in the case of lanthanides which have the longer processing times (from a few days to a few months) and thus are particularly sensitive to the processing efficiencies. Other elements are concerned, as alkaline elements and alkaline-earth elements, which follow noble gases in the β decay chain, as well as elements following noble metals (like cadmium following silver).

3.3.2 Isotopic separation in the bubbling unit

We will consider the example of the Xe/Cs extraction (see Table 2) to illustrate this section. One has to specify that Xenon, which is a noble gas, could be extracted by the bubbling system, while Cesium could not.

Table 2: Mean lifetime of four isotopes present in the TMSR core and concerned by bubbling extraction

Isotope	^{135}Xe	^{137}Xe	^{135}Cs	^{137}Cs
Lifetime	9.14 h	3.82 mn	2.3 10^6 yrs	30.1 yrs

Consequently, for a bubbling time shorter than the ^{137}Xe lifetime, ^{135}Xe and ^{137}Xe will be extracted from the core by the bubbling system before decaying respectively in ^{135}Cs and ^{137}Cs. These decays will occur in the bubbling unit. In this first case, where the short extraction time is equivalent to a good bubbling efficiency, the amount of ^{137}Cs mixed with the lanthanides is negligible. With a bubbling time ranging from the ^{137}Xe lifetime to the ^{135}Xe lifetime, ^{135}Xe will be extracted before decaying, while ^{137}Xe will decay in ^{137}Cs which will remain in the core and will be extracted later during the processing of the fission products. In this case, ^{137}Cs could be stored separately from ^{135}Cs but mixed with the lanthanides.

As a result, different waste managements are allowed according to the bubbling capabilities, which have to be studied.

3.4 Impact of the Extracted Elements on the Breeding Ratio

We have simulated the effect of different processing options on the breeding capacities of the reactor, assuming an extraction of the lanthanides in six months. The results are displayed on figure 6, where the line labelled "Pa and TRU extracted – BR = 1.000" is exactly equivalent to the line labelled "+extraction of lanthanides in 6 months" on figure 5.

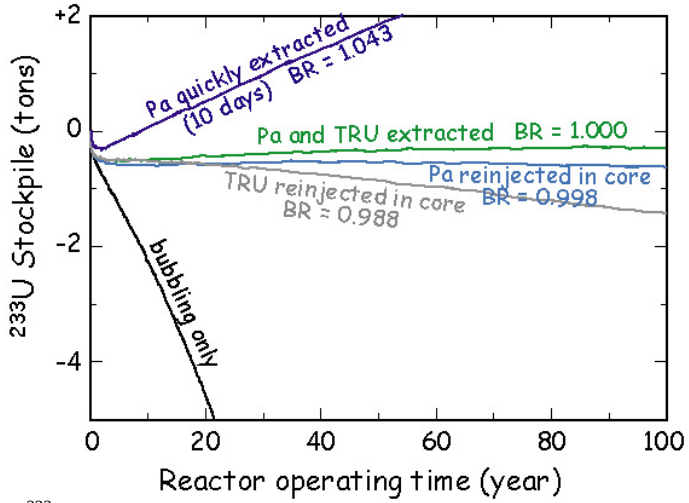

Figure 6: Production of ^{233}U simulated in the TMSR for different extraction options during processing. 'Pa' stands for the Protactinium element, 'TRU' for the transuranic elements, and 'BR' for the Breeding Ratio resulting from each extraction option.

As shown on figure 6, the extraction of the transuranic elements (TRU) moderately impacts the ^{233}U production ("Pa and TRU extracted" and "TRU reinjected in core" lines). Concerning the Protactinium extraction, two conclusions could be drawn out of the analyses: because of the slow processing presented in section 2.2, a non-extraction of the Protactinium (figure 6, "Pa reinjected in core" and "Pa and TRU extracted" lines) impacts only very slightly the ^{233}U production, while a quick extraction (similar to the extraction of the MSBR project, see "Pa quickly extracted" line on figure 6) allows the production of high amounts of ^{233}U.

4. Influence of the Salt Composition on the TMSR Neutronic Behaviour

Properties of the salt (density, viscosity, phase diagram) are also crucial. As presented in section 2.1, we planed to use in the TMSR a 78% LiF 22% $(HN)F_4$ salt for the fuel, but the use of lower heavy nuclei (HN) proportions is possible. In these cases, the meting points of these salt mixtures will be higher as it can be seen on the LiF-ThF_4-UF_4 ternary phases diagram [ORN59], which shows well this thermodynamic behaviour.
A reduction of the fissile inventory, i.e. of the initial quantity of ^{233}U necessary to start the reactor, leads to higher deployment abilities for the MSR. Such a reduction can be achieved by diminishing the volume of the reactor of course, but the heat evacuation is then less efficient, as the salt is the fuel as well as the coolant. To avoid thermodynamic problems, the idea consists consequently in reducing the proportion of HN in the salt and not in reducing the salt volume.
The reduction of the HN proportion entails an increase of the melting point of the salt. The reactor has to be designed at a higher temperature. The first and positive consequence of high temperature operation is to improve the thermodynamic efficiency of the reactor. The neutron spectrum is greatly modified with HN proportion and it has a deep impact on reactor behaviour. Not only the fissile inventory is affected, since safety is dependant of this proportion as are other characteristics like materials steadiness or breeding ratio. Our calculations again show that the most sensitive parameter is the system's fuel breeding capability. Table 3 summarizes the variations of the breeding ratio and of the ^{233}U inventory of the TMSR, operating at a temperature of 1030 °C, as a function of the HN proportion in the salt.

Table 3: Breeding ratio and ^{233}U initial inventory simulated for different heavy nuclei proportions

HN proportion	2 mole %	5 mole %	10 mole %	22 mole %
Breeding Ratio	0.868	0.951	0.984	1.026
^{233}U initial inventory	350 kg	520 kg	820 kg	1 630 kg
HN inventory	5.8 tons	13.5 tons	24.4 tons	45.3 tons

For the reference configuration of the TMSR presented in this study, the breeding ratio is only slightly deteriorated by a decrease of HN proportion in the salt, as shown in Table 3. Our calculations show that for other volumes or channel designs than the one of the paragraph 2.1, the breeding ratio is much more dependant of the HN proportion. The reduction of breeding ratio can reach a level of 0.03 while HN proportion is reduced by a factor 2. This is due to the higher proportion of fluoride in the salt, for which neutron captures are important.

5. Conclusions

The Molten Salt Reactor concept allows great potentialities owing to its running flexibility (proportion of heavy nuclei in the salt, liquid state of the fuel, core geometry, split modular salt treatment). The Thorium fuel cycle allows nuclear energy production with a very low formation of radiotoxic minor actinides, so that it has been selected by the Generation-IV International Forum.

We planed to use in the TMSR a 78 mole % LiF – 22 mole % $(HN)F_4$ salt for the fuel, but the use of lower Heavy Nuclei proportions is possible, so as to minimize the Uranium-233 inventory in the reactor. The neutron spectrum is greatly modified with HN proportion and it has a deep impact on the reactor behaviour. Our study enhances the degradation of breeding ratio, which results from a decrease of the HN proportion in the molten fuel salt.

In this paper, we presented a reference configuration called the Thorium Molten Salt Reactor (TMSR) and studied the influence of choices among different stages and time spreading out for salt processing. The processing time, which is one of the adjustable parameters, influences mainly the breeding ratio, which represents the ratio of Uranium-233 consumed over Thorium-232 converted into Uranium-233. The processing

time has a negligible impact on reactor safety, on the inventory of fissile materials or the life-time of the structure.

We conclude that the simplification of the salt processing presented in this study improves the feasibility of the MSR system. But the processing procedure has to be refined, the possibilities offered by chemistry and the neutronic impacts studied above have to be considered together.

References

[BET70] E.S. Bettis, R.C. Robertson, 1970, "The design and performance features of a single-fluid molten salt breeder reactor", Nuclear Applications and Technology, vol. 8, 190-207.

[BRI57] R.C. Briant and A.M. Weinberg, 1957, "Aircraft Nuclear Propulsion Reactor", Nuclear Science and Engineering, vol. 2, 795-853.

[BRI97] J.F. Briesmeister, 1997, "MCNP4B-A General Monte Carlo N Particle Transport Code", Los Alamos Laboratory report LA-12625-M.

[CAR71] W.H. Carr, L.J. King, F.G. Kitts, W.T. McDuffee, F.W. Miles, 1971, "Molten-Salt Fluoride Volatility Pilot Plant: Recovery of Enriched Uranium from Aluminium-Clad Fuel Elements", ORNL-4574.

[EDF77] EDF/DER, 1977, "Analyse critique du projet MSBR", internal report HT-12/24/77 (in French).

[HAU70] P.N. Haubenreich and J.R. Engel, 1970, "Experience with the Molten Salt Reactor Experiment", Nuclear Applications and Technology, vol. 8, 107-117.

[LEB07] C. Le Brun, 2007, "Molten Salts and Nuclear Energy Production", Journal of Nuclear Materials Volume 360, Issue 1, 1-5.

[LEC01] D. Lecarpentier, 2001, "Le concept AMSTER, aspects physiques et sûreté", PhD thesis, Conservatoire National des Arts et Métiers, Paris (in french).

[MAT05] L. Mathieu, 2005, "Potentialités des réacteurs à sels fondus en cycle Thorium dans les scénarios du futur : le concept de Thorium Molten Salt Reactor", PhD thesis, Institut National Polytechnique de Grenoble, France (in French).

[MAT06] L. Mathieu, D. Heuer, R. Brissot, C. Le Brun, E. Liatard, J.M. Loiseaux, O. Méplan, E. Merle-Lucotte, A. Nuttin, 2006, "The Thorium Molten Salt Reactor: Moving on from the MSBR", Prog in Nucl En vol 48, p 664-679.

[NUT05] A. Nuttin et al., 2005, "Potential of Thorium Molten Salt Reactors: Detailed Calculations and Concept Evolution With a View to Large Scale Energy Production", Progress in Nuclear Energy, Vol. 46, No.1, pp. 77-79.

[ORN59] R.E. Thoma, 1959, "Phase diagrams of nuclear reactor materials", ORNL-2548, p 119.

[ROS70] M.W. Rosenthal et al., 1970, "Molten Salt Reactors - History, Status, and Potential", Nuclear Applications and Technology, Vol. 8, pp. 107-117.

[ROS72] M. W. Rosenthal, P. N. Halbenreich and R. B. Briggs, 1972, "The Development Status of Molten-Salt Breeder Reactors," ORNL-4812, Oak Ridge National Laboratory.

[WAL03] E.Walle, J.Finne, G.Picard, S.Sanchez, O.Conocar, J.Lacquement, 2003, "Molten Salt Reactors: Chemistry of Fuel Salt and Fuel Salt Cleanup", Global, New Orleans, USA.

[WHA70] M.E. Whatley, L.E. McNeese, W.L. Carter, L.M. Ferris, E.L. Nicholson, 1970, "Engineering development of the MSBR fuel recycle", Nuclear Applications and Technology, vol. 8, 170-178.

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