

Introduction to the Physics of Molten Salt Reactors

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Abstract

In the frame of a major re-evaluation of the molten salt reactor (MSR), we have developed a new concept called Thorium Molten Salt Reactor (TMSR), particularly well suited to fulfill the criteria chosen by the Generation IV forum. This reactor may be operated in simplified and safe conditions in the Th/²³³U fuel cycle with fluoride salts. Amongst all TMSR configurations, many studies have highlighted the configurations with no moderator in the core as simple and very promising. Since ²³³U does not exist on earth and is not being produced today, we aim at designing a critical MSR able to burn the Plutonium and the Minor Actinides produced in today's reactors, and consequently to convert this Plutonium into ²³³U. Thus, the current fuel cycle can be closed thanks to TMSRs started with transuranic elements on a Thorium base, i.e. started in the Th/Pu fuel cycle, similarly to fast neutron reactors operated in the U/Pu fuel cycle. We analyze the characteristics of these reactor configurations, in terms of fissile matter inventory, salt reprocessing, waste production and burning, and finally deployment capabilities. Using a simple kinetic-point model, we analyze the reactor's behaviour as the total reactivity margins are introduced in the core. We thus confirm, beyond the classical advantages of molten salt reactors, the satisfactory behaviour of the TMSR and the excellent level of deterministic safety which can be achieved in such reactors. We then illustrate how the reactor can be driven with no control rod, either by controlling the extracted power or by monitoring the operating temperature. Finally we stress the hardiness and the flexibility of this TMSR concept, allowing it to be adjustable without losing its advantages in the event of any technological stumbling block.

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 leading to specific advantages in terms of safety/reliability (Rosenthal et al. 1970). Furthermore, this type of reactor is particularly well adapted to the Thorium fuel cycle (Th-²³³U) which has the advantage of producing less minor actinides than the Uranium-Plutonium fuel cycle (²³⁸U-²³⁹Pu) (Lecarpentier 2001)(Nuttin 2002)(Nuttin et al. 2005). While breeding in the U-Pu cycle can be obtained only with a fast neutron spectrum, in the Th-²³³U fuel cycle it can, in principle, be obtained with a more or less moderated neutron spectrum. In a thermal neutron spectrum, poisoning due to the Fission Products (FP) being worse than in a fast neutron spectrum, the rate at which fuel reprocessing is performed can become a major issue. Because, in an MSR, the fuel is liquid, continuous extraction of the fission products is a possibility.

Within the general TMSR concept, various reactor configurations can be obtained by varying the graphite moderator ratio, leading to a neutron spectrum that can range from very thermalized to fast. The performances of these configurations in terms of safety coefficients, breeding ratios, graphite lifetime and fissile inventories are briefly outlined here. We then focus on the TMSR without any moderator in the core, this configuration presenting the most interesting characteristics, among which the capability to use the TRU produced in Light Water cooled Reactors (LWRs) as fissile matter mixed with a thorium base. MSRs are thus described in this paper as energy producing critical systems as well as actinide burners. Our studies of these reactors are described from a physics standpoint: influence of the chemical reprocessing on the neutronic behaviour, burning capabilities, deployment capabilities, and finally a complete deterministic safety analysis.

This work made use of the MCNP neutron transport code (Briesmeister 1997) coupled with an in-house materials evolution code REM (Nuttin 2002)(Mathieu 2005). The former evaluates 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 within the cells. These calculations take into account the input parameters (power released, criticality level, chemistry...), by continuously adjusting the neutron flux or the materials composition of the core. Our calculations rest on a precise description of the geometry and consider several hundreds of nuclei with their interactions and radioactive decay.

1 Molten Salt Reactors and Thorium Fuel Cycle

The 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) (Briant and Weinberg 1957) in 1954. This 2.5 MWth reactor ran satisfactorily during 100 hours. The continuation of this work led to the commissioning of the 8 MWth Molten Salt Reactor Experiment (MSRE) (Haubenreich and Engel 1970). This reactor, based on a lithium, beryllium and zirconium fluoride salt was operated with 30% ^{235}U enriched uranium from 1965 to 1968, with ^{233}U from 1968 to 1969 and finally with plutonium mixed with ^{233}U in 1969. The success of this reactor warranted the study of a thorium power breeder reactor coupled to a reprocessing unit for the on-line extraction of fission products: the Molten Salt Breeder Reactor (MSBR) (Bettis and Robertson 1970)(Whatley et al. 1970). Even though the concept looked promising, the studies were stopped in 1976.

Studies on this sort of reactor were resumed in the 1980's in Japan with the Thorims-NES (Furukawa et al. 1990) then the FUJI-AMSB projects (Furukawa et al. 2005), as well as in France with studies on the MSBR by EDF¹ and the CEA². In the 1990's, the concept was taken up again with a view to incinerating nuclear wastes in sub-critical reactors such as the TIER-1 project (Bowman 1998), proposed by C. Bowman to transmute the plutonium of pressurized water reactors, the CEA TASSE project (Berthou 2000), or the EDF AMSTER project (Lecarpentier 2001).

The CNRS³ has been involved in this type of reactor since 1999 through complete studies aiming to re-evaluate the MSBR concept (Nuttin 2002)(Nuttin et al. 2005). This revaluation led to at least two important results which questioned the MSBR concept. On one hand, the feedback coefficient was proved to be globally positive so that the reactor is intrinsically unstable, on the other hand, the reprocessing unit appears not only unrealistic from the operational standpoint, and probably too complex to ensure economic cost effectiveness. Taking this into account, we have undertaken a systematic study of MSRs to resolve these problems and to optimize the reactor in the framework of the deployment of a

¹ *Electricité de France* – French Electricity Company

² *Commissariat à l'Energie Atomique* - French Atomic Energy Commission

³ *Centre National de la Recherche Scientifique* – French Scientific Research Center

thorium based reactor fleet on a worldwide scale (Heuer et al. 2006)(Merle-Lucotte et al. 2004)(Merle-Lucotte et al. 2006).

2 General Thorium Molten Salt Reactor (TMSR) Concept

The general Thorium Molten Salt Reactor (TMSR) concept is a 2500 MWth (1 GWe) graphite moderated reactor based on the $^{232}\text{Th}/^{233}\text{U}$ fuel cycle. The graphite matrix comprises a lattice of hexagonal elements with 15 cm sides. The density of this nuclear grade graphite is set to 1.86. The salt runs through the middle of each of the elements. For the systematic studies presented in this section, the salt considered in the simulations was a binary salt, $\text{LiF} - (\text{Heavy Nuclei})\text{F}_4$, whose $(\text{HN})\text{F}_4$ proportion is set at 22 mole % (eutectic point), corresponding to a melting temperature of 565°C.

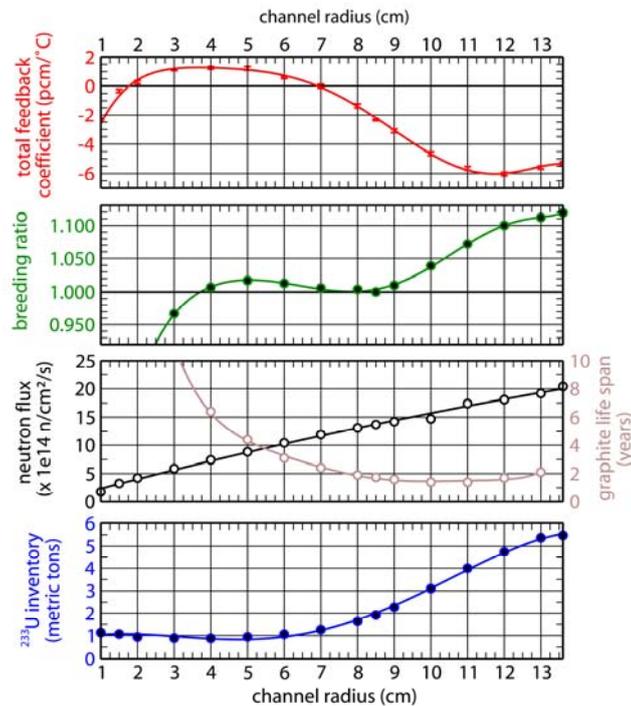


Fig. 2.1. Influence of the channel radius on the TMSR's behaviour. Note: There is no graphite lifespan point for the 13.6cm radius since there is no more graphite in the core.

A graphite radial blanket containing a fertile salt (LiF 78 mole% - ThF_4) surrounds the core to protect the external structures, so that approximately 80 % of the escaping neutrons are stopped, and the breeding is improved. Concerning

reprocessing (see paragraph 3.2 for details), we assume in our calculations that helium bubbling in the salt circuit is able to extract the gaseous fission products and the noble metals within 30 seconds. In this section, our calculations are based on a delayed reprocessing of the total salt volume over a 6 month period with complete extraction of the fission products and of the transuranic elements. The ^{233}U produced in the blanket is assumed to be extracted within a 6 month period.

The moderation ratio can be altered by changing the channel radius. This modifies the neutron spectrum of the core, placing it anywhere between a much thermalized neutron spectrum and a relatively fast spectrum. The core size is adjusted to keep the whole salt volume constant and equal to 20 m^3 . Fig. 2.1 shows the influence of the channel radius on the neutronic behaviour. In order to evaluate the performance of a reactor configuration, we check a number of parameters: total reactivity feedback coefficient, breeding ratio, graphite life span and initial fissile inventory.

A wide variety of neutronic behaviours is available by changing the moderation ratio. We define three types of configurations: thermal, epithermal and fast spectrum. Each one has advantages and drawbacks: a thermal spectrum leads to a low fissile inventory but slightly positive feedback coefficient due to the graphite, an epithermal spectrum like that of the MSBR requires efficient salt processing and leads to safety coefficients that are too limited while a fast spectrum implies a high breeding ratio and excellent safety coefficients but requires a large fissile inventory. Extensive studies have been carried out on these three fields of research to improve these configurations and to find relevant solutions (Mathieu 2005) (Mathieu et al. 2005) (Mathieu et al. 2006). In this paper, we focus on the configuration without any graphite moderator in the core, since it appears to be the most promising in terms of safety coefficients, reprocessing requirements, breeding and deployment capabilities.

3 The Non-Moderated Thorium Molten Salt Reactor

3.1. Core Description

As shown in Fig. 3.1, the core is now a single cylinder (1.25 m radius and 2.60 m height) where the nuclear reactions occur within the flowing salt. One third of the 20 m^3 of fuel salt circulates outside of the neutron flux in external circuits flowing through the pumps, the gaseous extraction system which removes the gaseous and non soluble fission products (see paragraph 3.2) and the heat

exchangers before coming back into the core. A fertile blanket surrounds the core, as already mentioned in section 2.

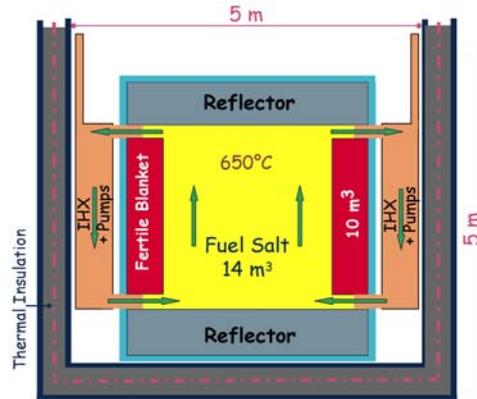


Fig. 3.1. To scale vertical representation of the TMSR, including pumps and intermediate heat exchangers (IHX)

3.2. Salt Reprocessing

Beyond the on-line control and adjustment of the salt composition and properties (redox potential measurement, reactivity, temperature...). the salt reprocessing is split in two steps. First, an on-line gaseous extraction system with helium bubbling is dedicated to the removal of gaseous fission products which are strong neutron poisons. As shown by ORNL data, this bubbling also extracts at least some of the noble metals and non soluble fission products. In our simulations, we assume that this helium bubbling is able to extract the gaseous fission products and the noble metals within 30 seconds, a less efficient extraction having little effect on core behaviour: up to an extraction time of a few days, the breeding ratio remains nearly constant. To optimise the extraction efficiency which depends on the interaction between the liquid salt, the metallic clusters and the gas bubbles, scientific investigations and dedicated measurements are necessary. Studies are also needed to determine how to separate the fission products from the gas, store them and purify the gas.

The second step in the reprocessing deals with the extraction of the other fission products, mainly lanthanides. A fraction of salt is periodically set aside to be reprocessed off-line (batch mode). The fissile matter (uranium) can be extracted quickly by fluorination and sent back into the core. The other actinides and lanthanides can be separated via several methods like electrolysis, reduction into metallic solvents, solid precipitation, or any other method developed in the frame of pyrochemistry reprocessing (Poignet and Fouletier 2007). Finally the

actinides are sent back into the reactor core to be burnt, while the lanthanides are stored. The performance of the reactor, in terms of breeding and deployment capabilities, depends directly on the rate at which this off-line reprocessing is done (see next paragraph).

3.3. Fuel Salt Composition

The core contains a fluoride fuel salt, composed of LiF enriched in ^7Li (99.999 %) and heavy nuclei (HN) amongst which the fissile element, ^{233}U or Pu. We have done parametric studies of this salt composition by varying the proportion of heavy nuclei in the fuel salt, and the reprocessing (Merle-Lucotte et al. ICAPP2007). This has an influence on neutron energy moderation, actinide solubility, and the initial fuel inventory. For HN proportions ranging from 20 to 30 mole%, a binary salt LiF-(HN)F₄ has been chosen whose melting point is around 570°C. For lower proportions of HN, the calculations have been done with a salt containing 80 mole% of LiF completed with BeF₂ (other possible components like NaF or CaF₂ for example are under study) to lower the eutectic point temperature and to allow operation at 630°C. The salt density ranges from 3.1 to 4.6 according to the HN proportion, with a dilatation coefficient of $10^{-3}/^\circ\text{C}$ (Ignatiev et al. 2005).

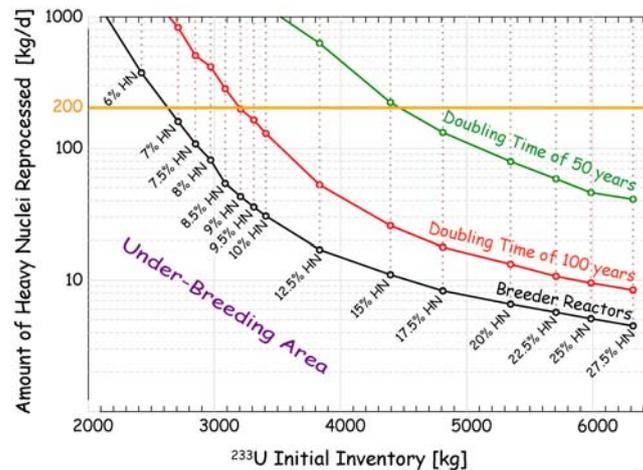


Fig. 3.2 Amount of heavy nuclei reprocessed per day versus initial fissile (^{233}U) inventory, for different heavy nuclei proportions in the fuel

The results of these parametric studies are presented in Fig. 3.2 where the reprocessing capacity required for a TMSR system is shown. The ordinate is the weight of heavy nuclei reprocessed per day while the abscissa corresponds to the initial fissile (^{233}U) inventory. Heavy nuclei proportions in the MSR fuel are indicated (x % mole HN). For each proportion of heavy nuclei in the salt, ranging from 6 mole % to 27.5 mole %, we evaluate by simulation the breeding ratio of

each reactor configuration as a function of the amount of heavy nuclei reprocessed per day. The black line separates breeding and non-breeding zones. In the breeding zone, the red and green lines indicate the reprocessing requirements which allow the generation of the ^{233}U needed for a new TMSR reactor with the same inventory over 100 years and 50 years respectively.

Under-breeder reactor configurations, which are located under the black line (bottom of the figure), will not be considered in the following since they do not allow sustainable reactor deployment.

The ^{233}U initial inventory ranges from 2400 kg for a HN proportion in the salt of 6 mole % to 6300 kg for a HN proportion of 27.5 mole %. This corresponds to a variation of the neutron spectrum from an epithermal to a fast spectrum, as shown in Fig. 3.3.

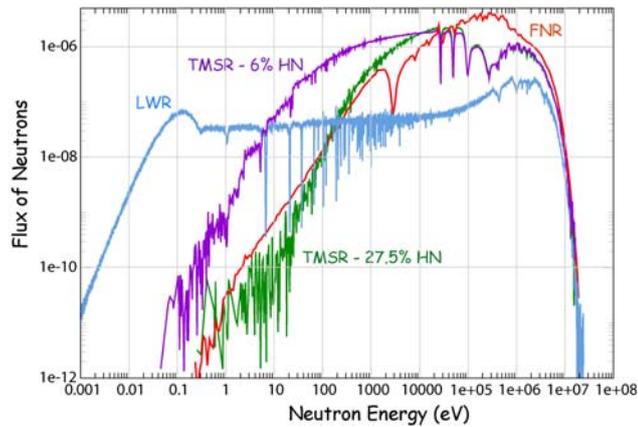


Fig. 3.3 Neutron spectrum of two TMSR configurations (6% and 27.5% of heavy nuclei in the salt) compared to the neutron spectrum in a Light Water cooled Reactor (LWR) and in a Fast Neutron Reactor (FNR)

We have chosen a medium configuration with 17.5 mole% of HN in the salt, that we will refer to as the 'typical TMSR' in this paper. For the results presented below, we have also selected a reasonable reprocessing rate of 200 kg of HN per day, indicated by the orange horizontal line in Fig. 3.2.

As already stated, TMSRs are operated in the Thorium fuel cycle, using either ^{233}U (^{233}U -started TMSR) or Pu (transuranic-started TMSR or TRU-started TMSR) as initial fissile matter. The initial load of the non-moderated TMSR can thus be a mixture of thorium and, for its fissile material, the transuranic elements (Pu, Np, Am and Cm) produced in the water moderated reactors fed at present with natural or slightly enriched uranium. Actually, to be more realistic, these TMSRs are started with the mix of 87.5% of Pu (^{238}Pu 2.7%, ^{239}Pu 45.9%, ^{240}Pu 21.5%, ^{241}Pu 10.7%, and ^{242}Pu 6.7%), 6.3% of Np, 5.3% of Am and 0.9% of Cm,

corresponding to the transuranic elements of an UOX fuel after one use in a standard LWR and five years of storage (de Saint Jean et al. 2000).

For the typical TMSR configuration (with 17.5 mole% of HN in the salt), an amount of 7300 kg of fissile elements is needed initially, corresponding to 4.5 mole % of Plutonium.

3.3 Heavy Nuclei Inventory and Burning Capabilities

Fig. 3.4 illustrates the evolution of a typical fuel salt composition (17.5% mole HN) all along the operation of this reactor, for the ^{233}U -started (solid lines) and for the transuranic-started (dashed lines) TMSR.

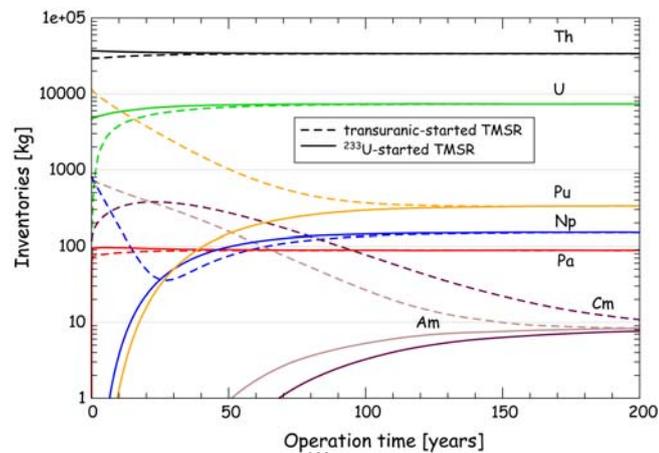


Fig. 3.4 Heavy nuclei inventory for the ^{233}U -started TMSR (solid lines) and for the transuranic-started TMSR (dashed lines) with 17.5 mole% of heavy nuclei in the salt

In terms of transuranic inventory, as shown in Fig. 3.4, the TMSRs started with transuranic elements tend towards TMSRs directly started and operated with ^{233}U after about forty years for a fuel salt with 17.5% of heavy nuclei, where more than 85% of the initial TRU inventories are burned. More generally, the assets of the Thorium fuel cycle are finally recovered for these transuranic-started TMSRs after 25 to 50 years for HN proportions ranging from 7 to 27.5 mole %, except for minor actinides where longer times are needed.

3.4 Deployment Capabilities

The deployment capabilities of a fleet of 4th generation reactors are directly linked to the breeding capabilities of each reactor since their initial fissile matter (^{233}U or Pu) does not exist on earth.

The interesting variable is thus the operating time necessary to produce one initial fissile (^{233}U) inventory, called the reactor doubling time. We have studied the amount of excess ^{233}U produced in each TMSR configuration. Transuranic-started TMSRs allow the extraction of significantly larger amounts of ^{233}U during their first 20 years of operation, thanks to the burning of TRUs which saves a part of the ^{233}U produced in the core. The higher deployment capabilities allowed by the use of TRUs in the transuranic-started TMSR are visible on the reactor doubling times, displayed in Fig. 3.5 (dashed line), where the configurations with HN proportions larger than 15% have the lower reactor doubling times, around 30-35 years, to be compared with the doubling times greater than 45-50 years reachable with the ^{233}U -started TMSRs.

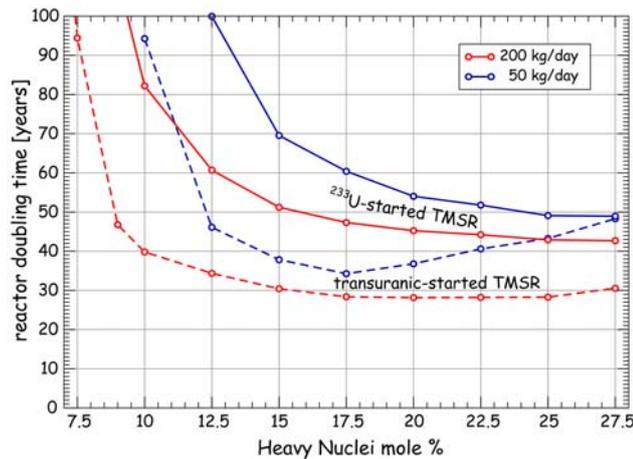


Fig. 3.5 Reactor doubling time of ^{233}U -started (solid lines) and transuranic-started TMSRs (dashed lines) for the different HN proportion configurations and for two off-line reprocessing schemes: 200 kg (red lines) and 50 kg (blue lines) of HN reprocessed per day.

The configuration with 17.5% of heavy nuclei in the salt appears particularly optimal in terms of deployment capabilities, especially with the faster reprocessing scheme and for the transuranic-started TMSRs. With these results, we have simulated the deployment of a fleet of such reactors at a worldwide and at a European scale and verified that large scale deployment is possible both with regard to the availability of fissile matter and to the control of fissile matter stockpiles and radioactive wastes (Merle-Lucotte et al. 2006) (Merle-Lucotte et al. ENC2007).

4 Deterministic Safety Study of the Non Moderated TMSR

4.1 Deterministic Safety Parameters of the TMSR

4.1.1 Fraction of Delayed Neutrons

The fraction of delayed neutrons, β , is very important in reactor control, it is a determining safety parameter. The total number of fission product atoms giving rise to delayed neutron emissions will depend on the fissile composition of the reactor and, to a lower extent, on the type of neutron spectrum. The value of β at equilibrium for the TMSR considered here has been estimated by simulation to 450 pcm.

Group	1	2	3	4	5	6	7
Precursor	⁸⁷ Br	¹³⁷ I	⁸⁸ Br	⁹³ Rb	¹³⁹ I	⁹¹ Br	⁹⁶ Rb
Half-Life	55.9 s	24.5 s	16.4 s	5.85 s	2.3 s	0.54 s	0.199 s
Abundances							
²³³ U (fast)	0.0788	0.1666	0.1153	0.1985	0.3522	0.0633	0.0253
²³³ U (thermal)	0.0787	0.1723	0.1355	0.1884	0.3435	0.0605	0.0211
²³⁵ U (fast)	0.0339	0.1458	0.0847	0.1665	0.4069	0.1278	0.0344
²³⁵ U (thermal)	0.0321	0.1616	0.0752	0.1815	0.3969	0.1257	0.0270
Mean Value	0.0742	0.1679	0.1209	0.1915	0.3533	0.0684	0.0240

Table 4.1 Abundances of seven delayed neutron precursors for two uranium isotopes

We have considered in our study seven precursor families listed in Table 4.1 (d'Angelo 2002) for two fissile nuclei, ²³³U and ²³⁵U. Regarding the neutron spectrum of the TMSR, we take a value between those indicated for precursor abundances in a fast and in a thermal spectrum. We have considered two fissile nuclei since, in the TMSR, 90% of the fissions are due to ²³³U and 10% to ²³⁵U. The final values used for the safety evaluation are summarized at the bottom of Table 4.1.

4.1.2 Feedback Coefficient

The feedback coefficient or temperature coefficient is the variation of the multiplication coefficient dk for a given variation dT in temperature of the whole or part of the core. This feedback coefficient has to be negative to ensure the

intrinsic stability of the reactor. The practical evaluation of a feedback coefficient is done as follows. The multiplication coefficient k is first computed for the core with the matter compositions at equilibrium for a temperature of 900 K. It is then re-calculated using the same compositions but at a different reactor temperature. In practice, the modifications concern the temperature of the salt itself, the temperature of the graphite moderator (if any), together with the density of the salt because of its dilatation (dilatation coefficient of $10^{-3} / ^\circ\text{C}$ in our case). Other temperature variations like those of the reflectors or the blanket are not considered since these materials have a very small contribution and heat up very slowly.

The feedback coefficient $\frac{dk}{dT}$ is traditionally broken down into three sub-coefficients related to the different components of the core presented above:

$$\left(\frac{dk}{dT}\right)_{Total} = \left(\frac{dk}{dT}\right)_{Salt_heating} + \left(\frac{dk}{dT}\right)_{Salt_dilatation} + \left(\frac{dk}{dT}\right)_{Moderator_heating}$$

The third sub-coefficient is negligible in our case since there is no moderator in core.

The uncertainties indicated in this paper are a quadratic combination of the statistical and systematic uncertainties on the determination of the sub-coefficients. The statistical errors are estimated precisely by simulation, the quantification of the systematic errors is more difficult. More precisely, concerning the systematic uncertainties on the contribution of salt heating, the cross-sections concerned are well known, inducing only negligible uncertainties. The uncertainties on the salt density and its dilatation lead to systematic errors less than 20% on the contribution of salt dilatation, the value used for the estimation of uncertainties in the next sections.

The total feedback coefficient at equilibrium is displayed in Fig. 3, together with its components, the contributions of the salt heating and salt density, as a function of the HN proportion in the salt. All these safety coefficients are significantly negative for all HN proportions, including the density coefficient which can be seen as a void coefficient. The total feedback coefficient is largely negative, ranging from -10 pcm/K to -5 pcm/K.

Concerning TMSRs started with transuranic elements, the total feedback coefficients after one year of operation are also displayed in Fig. 4.1 as a function of the HN proportion. They correspond to the initial safety behavior of the reactor, since the inventories after one year of operation are quite similar to the initial inventories, but we also take into account the effects of the fission products which are not present in the initial load. As shown in Fig. 4.1, the safety coefficients are initially slightly smaller in the case of the transuranic-started TMSRs but they remain largely negative, being equal at equilibrium to the safety coefficients of the ^{233}U -started TMSRs.

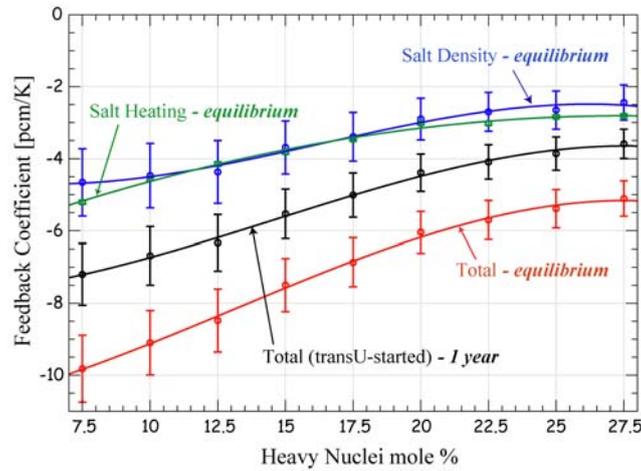


Fig. 4.1 Feedback Coefficients of the TMSR started both with ^{233}U or with transuranic elements at equilibrium and of the TMSR started with transuranic elements after one year of operation, as a function of the HN proportion in the fuel salt.

We have applied the safety analysis presented in the next paragraph on the case of a typical TMSR configuration, i.e. with a heavy nuclei proportion of 17.5% at equilibrium. Our calculations lead to a total feedback coefficient equal to $-7 (\pm 1)$ pcm/K at equilibrium, the coefficient of the corresponding configuration started with transuranic elements ranging from -5 pcm/K at the beginning of its operation to -7 pcm/K at equilibrium.

4.1.3 Reactivity Margins in the TMSR

Beyond the safety aspects, using a liquid fuel allows the adjustment of fertile and fissile matter without unloading the core, doing away with the need for any initial reactivity reserve, contrary to the case of a LWR where this reactivity reserve amounts to 10 000 pcm. Some reactivity margins may be introduced in the core of the TMSR involuntary through three possible perturbations: a direct insertion of reactivity, the loss of salt circulation, and finally the loss of the fertile blanket surrounding the core.

The impact of different kinds of reactivity insertions has to be studied. If the adjustment of fissile and fertile matters in the MSR is an important advantage, it may indeed present potential specific dangers which are evaluated below. We first consider the unintentional introduction of ^{233}U in core. The reactivity increase of the TMSR is equal to 13.6 ± 0.2 pcm per extra kilogram of ^{233}U (Mathieu 2005). For an iso-breeder reactor, the core needs to be fed 2.6 kg/day in ^{232}Th . If ^{233}U is added, instead of ^{232}Th , this would lead to an increase of the reactivity of 35 pcm/day. ^{233}U is also directly produced in core as follows:



The radioactive period of ${}^{233}\text{Pa}$ being 27 days, it decays rather quickly in core - around 2.55% per day - giving ${}^{233}\text{U}$. This corresponds to (1) a disappearance of neutron capturing matter and (2) an appearance of fissile matter. If the reactor were to stop operating, the first effect would increase the reactivity by 33 ± 11 pcm/day, the second effect by 27 ± 1 pcm/day, giving a total reactivity margin of 60 ± 11 pcm/day.

As detailed in section 4.1.1, the fraction of delayed neutrons for the whole salt at equilibrium is equal to 450 pcm. As two thirds of the salt is in the core and one third outside in the heat exchangers during normal operation, 2/3 of the delayed neutrons (300 pcm) are emitted within the core. If the salt circulation is stopped, all the delayed decays will occur in the reactor, since all the fission products will stay in core. This will represent an addition of 150 pcm to the multiplication coefficient in some ten seconds.

Two other problems have to be considered: the loss of coolant (mainly for reactors with solid fuel) and the loss of the fertile blanket. In our case, the draining of the core is equivalent not only to a loss of the coolant but also to a loss of fuel, leading to a decrease of the reactivity. Since the reactor is equipped with a draining system through stoppers made for example of solid salt and designed to melt at a given temperature, it was important to check that the salt evacuation will not worsen an accident, since it occurs mainly in abnormal situations.

Finally, our reactor is surrounded by a thorium blanket so as to optimize the breeding ratio of the system. The structure of this blanket is made of ZrC, containing a salt composed of LiF (72.5%) – ThF₄ (27.5%). This salt may be solid or liquid. The second solution simplifies the recovery of the fissile matter but the blanket could then be accidentally drained. Our simulations show that the extraction of the entire blanket leads to a decrease of reactivity. Because the ZrC structure of the blanket does not moderate neutrons, draining the blanket leads to increased neutron leakages and thus to a decrease of reactivity.

Origin	${}^{233}\text{U}$ Addition	${}^{233}\text{Pa}$ Decay	Salt Circulation	Core Draining	Blanket Draining	Reactivity Margins
Reactivity Margin	35 pcm	60 pcm	150 pcm	-	-	< 500 pcm
Time of Insertion	Some seconds	1 day	Some 10 s	-	-	> some seconds

Table 4.2 Reactivity margins of a typical TMSR configuration

Table 4.2 summarizes the reactivity margins, with their corresponding times of insertion.

4.2 Safety Evaluation of the TMSR

Using a single kinetic-point evaluation presented in the next paragraph, we analyze how the safety parameters presented above impact the system when the total reactivity margins are simultaneously introduced in the TMSR. Deterministic safety implies the definition of a viability domain, corresponding to the range of acceptable core parameters. As internal pressure is very low in a TMSR, only phenomena following a temperature increase of the fuel salt could endanger the reactor. As a consequence, the viability region is limited by the fuel solidification temperature with $T_{\min} = 800K$ as lower limit, and by the salt dissociation temperature with $T_{\max} = 1600K$ as upper limit.

4.2.1 The kinetic-point model

Transient simulations were carried out using a simple mathematical model which includes the following system of point reactor kinetics equations with seven groups of delayed neutrons:

$$\frac{\partial N}{\partial t} = \frac{\rho - \beta}{l} N + \sum_i \lambda_i C_i \quad (1)$$

$$\frac{\partial C_i}{\partial t} = \frac{\beta N}{l} - \lambda_i C_i \quad (2)$$

$$\frac{\partial T}{\partial t} = (P - P_o)/(C_p d) \quad (3)$$

N : neutron flux in the core,
 t : time since the beginning of the transient,
 ρ : reactivity,
 β : proportion of delayed neutrons (300 pcm in the TMSR),
 l : the mean time between two fissions (8.46 μ s here),
 λ_i : $\ln 2 / (t_{1/2})_i$ where $(t_{1/2})_i$ is the decay period of the delayed neutron precursors of the i^{th} -group,
 C_i : proportion of delayed neutron precursors of the i^{th} -group,
 T : mean core temperature ($T_o = 900K$),
 P : instantaneous power per cm^3 ,
 P_o : extracted power per cm^3 (125 W/cm^3 , constant),
 C_p : specific heat (1.05 $\text{J}/\text{g}/\text{K}$),
 d : density of the fuel salt (4.3 g/cm^3).

This kinetic-point model implies:

- A uniform distribution of the fissions within the core (correct in our case)
- No heat propagation

- No follow-through of the precursors of the delayed neutrons

The goal of this section is to give an idea of the reactor's global behavior during accidental transients due to reactivity insertion or heat evacuation loss. Further calculations taking in consideration heat propagation, a realistic distribution of the fissions and the propagation of the precursors would loosen the limits of the kinetic-point model.

4.2.2 Safety evaluation of the reference configuration

From Table 4.2, we can conclude that the insertion of the total reactivity margins of the TMSR corresponds to the addition of less than 1000 pcm in some minutes (around 100 seconds). The insertion of this upper limit of 1000 pcm, in 0.001 to 100 seconds, is thus displayed on figure 4.2 in terms of temperature, power and reactivity evolutions. The calculations have been done using a feedback coefficient equal to -5 pcm/K, close to the worst coefficient value of the typical configuration of the TMSR with a fuel salt containing 17.5 mole % of heavy nuclei.

As expected, the final temperature reached at equilibrium is equal to $T_o + \frac{1000 \text{ pcm}}{dK/dT} = 1100 \text{ K}$. We can conclude from Figure 4.2 first that the prompt criticality is reached for an insertion time shorter than one second, since the maximal reactor reactivity is higher than the fraction of delayed neutrons (equal to 300 pcm). If we detail for example the reactivity insertion in 0.001 second (figure 4.2, black curves), we first have to consider the evolution of the reactivity (figure 4.2, top) since we modify it directly. In this case, the whole 1000 pcm reactivity is added before the reactor begins to react. The reactivity insertion then leads to an increase of the reactor power (figure 4.2, middle) which reaches a maximum at 1 MW/cm³. Finally, the reactor temperature (figure 4.2, bottom) begins to increase after some milliseconds. A significant temperature increase triggers the reactor feedback, and as a consequence reactivity and power decrease. The heat in excess has finally to be evacuated to recover normal operating conditions.

However, even in the case of the prompt reactivity regime, a huge and dangerous increase of power and temperature could be avoided thanks to the reactor's safety parameters, as seen for the insertion time of 0.1 second. In this case (figure 4.1, green curves), the thermal feedback is fast enough to lower the reactivity back to zero before the end of the 1000 pcm reactivity insertion. A bounce can thus be observed, especially on the power evolution, when the remaining reactivity margins are inserted. A second feedback leads to the final reactivity and power decreases, and normal running conditions are reached again when the accumulated heat is evacuated.

For insertion times greater or equal to one second, and a fortiori in a more realistic case of 100 seconds, the reactor feedback is fast enough to avoid the prompt reactivity regime. Thus the reactor succeeds in absorbing at least a 1000 pcm reactivity insertion in one second and behaves safely.

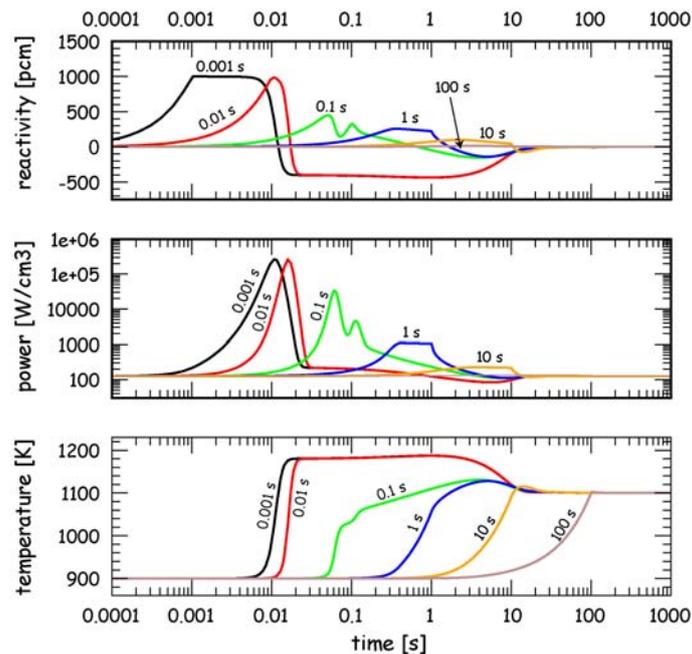


Fig. 4.2 Evolution of the reactivity, the reactor's power and the salt temperature following an insertion of reactivity of 1000 pcm in 0.001 to 100 seconds, the total feedback coefficient of the reactor being equal to -5 pcm/K

Concerning the validity domain of our results, our simulations are reliable for the longer insertion times and are pessimistic for the shorter insertion times, since we consider a uniform power distribution within the core and we do not take into account the time to reach the density equilibrium (equilibrium considered as immediate). Larger local powers are indeed reached within the core, leading to a faster feedback to reactivity insertion than shown in figure 4.2. For the insertion times between 0.01 and 10 seconds, things are more complex and precise calculations have to be done to confirm our results, mainly to take into account the follow-through of the precursors of the delayed neutrons.

4.2.3 Control of the reactor

The TMSR can be driven by the extracted power, and thus by the energy demand through the secondary circuit, as illustrated in Fig 4.3. As the energy demand increases, the power extracted becomes larger than the power produced

by fission in the core. As a consequence, the temperature of the fuel salt decreases, leading to a reactivity increase as shown in Fig 4.3 for the extracted powers greater than 100% (red and green curves corresponding respectively to an extracted power of 125% and 150% of the nominal extracted power). This logically brings on a larger production of power in the reactor. This produced power then temporarily exceeds the power demand, which restores the initial temperature and reactivity. The reactor finally returns to equilibrium at its nominal temperature, while producing the needed power.

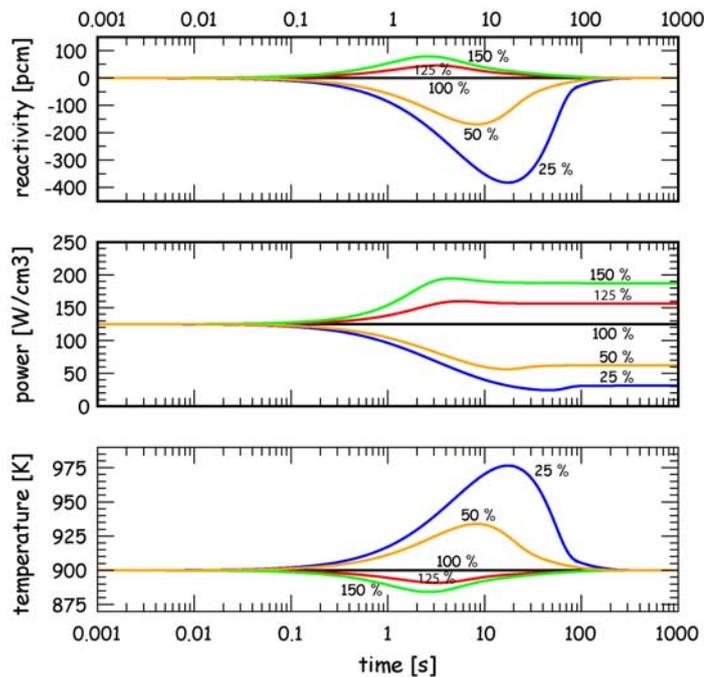


Fig. 4.3 Evolution of the reactivity, the reactor's power and the salt temperature as a function of the extracted power (in percent of the nominal extracted power), for a total feedback coefficient of the TMSR equal to -5 pcm/K

As shown on the other curves of Fig. 4.3 for extracted powers lower than 100% of the nominal reactor operation, the same goes in the case of a power demand decrease, this bringing on this time a rise of temperature, a drop of reactivity and thus a decrease of the produced power.

The reactor's reactivity can also be controlled by stabilizing its operating temperature. A decrease of temperature with a constant produced power reveals a drop of reactivity due to a lack of fissile matter in the core. Reversely, an increased operating temperature reveals a too large reactivity which has to be corrected by stopping the addition of fissile matter (in the case of an under-breeder reactor for example).

Conclusions

In the frame of a major re-evaluation of the MSR concept, we have first introduced the innovative concept of Thorium Molten Salt Reactor or TMSR.

We then focussed on the configurations without any graphite moderator in the core, since they appear to be really promising. More precisely, the non-moderated TMSR configurations with high proportions of heavy nuclei, leading to a rather fast neutron spectrum, present particularly interesting characteristics regarding their safety performance and their ability to be first loaded with transuranic elements produced by today's water cooled reactors. Finally, their rather large initial fissile inventory does not inhibit their capability for a fast deployment thanks to their very good ^{233}U breeding ratio during the burning of transuranic elements.

Using a simple kinetic-point model, we have analyzed the reactor's behaviour as the total reactivity margins are introduced in the core. For insertion times greater or equal to one second, and a fortiori in a more realistic case of 100 seconds, the reactor succeeds in absorbing the reactivity insertion and behaves safely. Beyond the classical advantages of molten salt reactors, our studies confirm the satisfactory behaviour of the TMSR and the excellent level of deterministic safety which can be achieved in such reactors. We have also illustrated how the reactor can be driven with no control rod, either by controlling the extracted power or by monitoring the operating temperature.

Under these conditions the TMSR appears to be a very appealing concept; our calculations do not indicate any major reprocessing constraint, allowing batch mode reprocessing in the vicinity of the reactor. The main additional studies needed to demonstrate the scientific feasibility of the concept deal with the on-line control of the salt composition and of its chemical and physical properties. Such studies are in progress in the frame of the French concerted research program 'Molten Salt Reactor' (PCR-RSF). Finally we want to point out the hardiness and the flexibility of this TMSR concept, allowing it to be adjustable without losing its advantages in the event of any technological stumbling block.

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