Proceedings of ICAPP '06
Reno, NV USA, June 4-8, 2006
Paper 6132

Fast Thorium Molten Salt Reactors started with Plutonium

E. Merle-Lucotte, D. Heuer, C. Le Brun, L. Mathieu*, R. Brissot, E. Liatard, O. Meplan, A. Nuttin
LPSC/IN2P3/CNRS, 53, avenue des Martyrs, F-38026 Grenoble Cedex, France
Tel: (33)-4-76-28-41-50, Fax: (33)-4-76-28-40-04, Email:merle@lpsc.in2p3.fr
* *now at CENBG,, Le Haut-Vigneau, BP 120, F-33175 Gradignan Cedex, France*

Abstract – *One of the pending questions concerning Molten Salt Reactors based on the $^{232}Th/^{233}U$ fuel cycle is the supply of the fissile matter, and as a consequence the deployment possibilities of a fleet of Molten Salt Reactors, since ^{233}U does not exist on earth and is not yet produced in the current operating reactors. A solution may consist in producing ^{233}U in special devices containing Thorium, in Pressurized Water or Fast Neutrons Reactors. Two alternatives to produce ^{233}U are examined here: directly in standard Molten Salt Reactors started with Plutonium as fissile matter and then operated in the $Th/^{233}U$ cycle; or in dedicated Molten Salt Reactors started and fed with Plutonium as fissile matter and Thorium as fertile matter. The idea is to design a critical reactor able to burn the Plutonium and the minor actinides presently produced in PWRs, and consequently to convert this Plutonium into ^{233}U. A particular reactor configuration is used, called 'unique channel' configuration in which there is no moderator in the core, leading to a quasi fast neutron spectrum, allowing Plutonium to be used as fissile matter. The conversion capacities of such Molten Salt Reactors are excellent. For Molten Salt Reactors only started with Plutonium, the assets of the Thorium fuel cycle turn out to be quickly recovered and the reactor's characteristics turn out to be equivalent to Molten Salt Reactors operated with ^{233}U only. Using a combination of Molten Salt Reactors started or operated with Plutonium and of Molten Salt Reactors started with ^{233}U, the deployment capabilities of these reactors fully satisfy the condition of sustainability.*

I. INTRODUCTION

It has been demonstrated recently[1,2,3] that the Molten Salt Reactors (MSRs), which are one of the systems selected by the Generation IV forum, may be operated in simplified and safe conditions in the $Th/^{233}U$ fuel cycle and with fluoride salts, whether in a moderated or a fast neutron spectrum. These MSRs are called Thorium Molten Salt Reactors (TMSRs). Their main specific advantages come from their liquid fuel. The amounts of fissile and fertile matter can be adjusted without unloading the core, avoiding any initial reactivity reserve. Similarly, the Fission Products which poison the core can be taken out efficiently. This is true for instance for gases such as Xenon, which are easily removed with helium bubbling.

In Section II, we first present the motivations for a fast neutron TMSR. We show that such a reactor can operate with the relevant properties, in terms of breeding ratio, feedback coefficients, trans-uranian production and material steadiness to irradiation. The last paragraph of this section is devoted to studies addressing the problem of fissile inventory, the only drawback of this reactor type. Two methods are presented to minimize this inventory.

One of the pending questions concerning MSRs is thus the supply of the fissile matter, and as a consequence the deployment possibilities of a fleet of MSRs, since ^{233}U does not exist on earth and is not yet produced in the current operating reactors. In Sections III and IV, we detail two ways to produce ^{233}U: directly in standard MSRs started with Plutonium as fissile matter and then operated in the $Th/^{233}U$ cycle; or in dedicated MSRs started and fed with Plutonium as fissile matter and Thorium as fertile matter. Our idea is to design a reactor able to burn the Plutonium and the minor actinides produced in currently operating reactors, and consequently to convert this Plutonium into ^{233}U. We analyze the characteristics of such reactors, in terms of deterministic safety parameters, fissile matter inventory, salt reprocessing scheme, trans-uranian production, and overall conversion capabilities

This work is based on the coupling of a neutron transport code called MCNP[4] with a materials evolution code. 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 within 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 rest on a precise description of the geometry and consider several hundreds of nuclei with their interactions and radioactive decay; they allow a thorough interpretation of the results.

II. FAST NEUTRON MOLTEN SALT REACTOR BASED ON THE $^{232}Th/^{233}U$ FUEL CYCLE

II.A. Description of the General TMSR

The general concept of the Thorium Molten Salt Reactor (TMSR) is a 2500 MWth (1 GWe) graphite moderated reactor based on the $^{232}Th/^{233}U$ fuel cycle. Its operating temperature is 630 °C and its thermodynamic efficiency is 40 %. 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. One third of the 20 m^3 of fuel salt circulates in external circuits and, as a consequence, outside of the neutron flux. The salt used is a binary salt, LiF - (Heavy Nuclei)F_4, whose (HN)F_4 proportion is set at 22 mole % (eutectic point), corresponding to a melting temperature of 565°C. The ^{233}U proportion in HN is about 3 %. The salt density at 630°C is set at 4.3 with a dilatation coefficient[5] of 10^{-3}/°C.

A graphite radial blanket containing a fertile salt surrounds the core so as to improve the system's regeneration capability. The properties of the blanket are such that it stops approximately 80 % of the neutrons, thus protecting external structures from irradiation while improving regeneration. We assume that helium bubbling in the salt circuit is able to extract the gaseous Fission Products (FP) and the noble metals within 30 seconds. For the first presented study, we consider a delayed reprocessing of the total salt volume over a 6 month period with a complete extraction of the FPs and of the Trans-Uranians (TRU). We assume that the ^{233}U produced in the blanket is also 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 very thermalized neutron spectrum and a relatively fast spectrum. The core size is adjusted to keep the whole salt volume constant. Fig. 1 shows the influence of the channel radius on the neutronic behavior. 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 behaviors 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 positive feedback coefficient, while a fast spectrum implies a high breeding ratio but large fissile inventory. Studies have been carried out on these three fields of research to improve these configurations and to find relevant solutions[2,3]. In this paper we will focus only on the fast neutron spectrum since it appears to be the most promising and simple path of investigation. Moreover, a fast neutron spectrum opens the possibility of using fissile matter such as Plutonium, when is it hardly achievable with a thermal neutron spectrum[6].

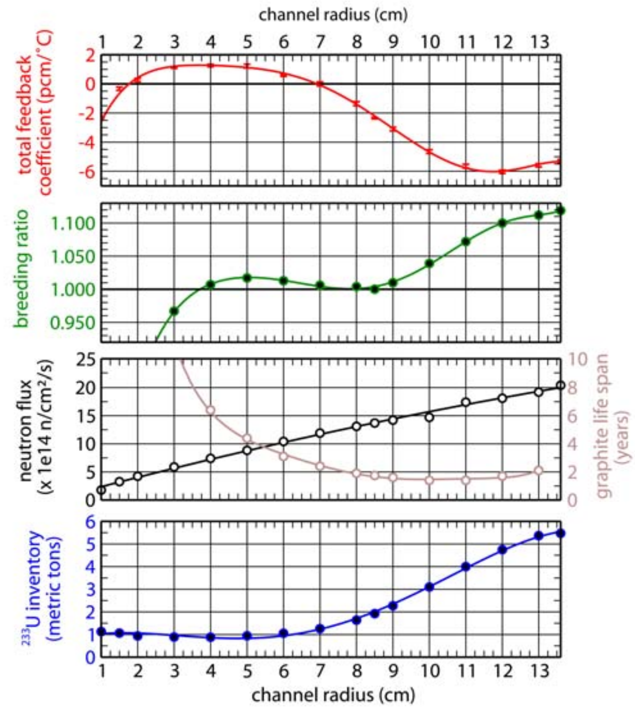

Fig. 1. Influence of the channel radius on the TMSR's behavior

II.B. Description of the fast neutron TMSR

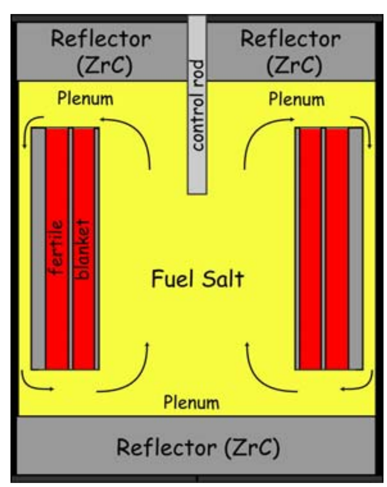

Fig. 2. Vertical section of the fast TMSR core

When the hexagons are fully filled with fuel salt, there is no more graphite within the core. The reactor is then composed of a single big salt channel. In such a configuration there is no graphite irradiation problem, since there is none of it inside the high neutron flux. The graphite blanket structure is much less irradiated and, as a result, has a longer life span. We now present the characteristics of this configuration.

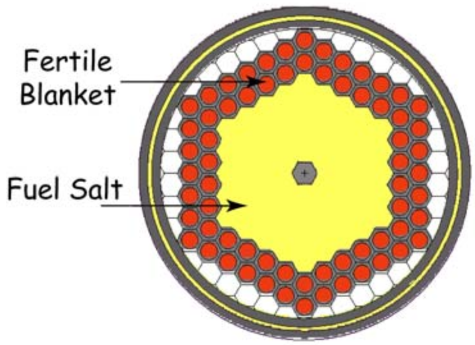

Fig. 3. Horizontal section of the fast TMSR core

Vertical and horizontal sections of this configuration are presented on Figs. 2 and 3. The salt channel, 1.25 m radius and 2.60 m height, is surrounded by a thorium blanket and by two axial reflectors. These reflectors are made of ZrO_2 or ZrC (same neutronic behavior) in order to avoid the use of a moderator material. An efficient bubbling system extracts gazeous FPs and noble metals in 30 s, and a reprocessing unit slowly removes other FPs and TRUs of the fuel salt in 6 months.

The feedback coefficient can be broken down into three sub-coefficients related to the different components of the core presented above:

$$\left.\frac{dk}{dT}\right)_{total} = \left.\frac{dk}{dT}\right)_{salt\_heating} + \left.\frac{dk}{dT}\right)_{salt\_dilatation} + \left.\frac{dk}{dT}\right)_{graphite\_heating}$$

The third sub-coefficient is negligible in our case since there is no graphite moderator in core. The contributions of the heating of the salt and of its dilatation are of the same order of magnitude, as shown in Table 1.

The total feedback coefficient for this TMSR configuration is equal to -5.37 pcm/°C.

The uncertainties on the coefficients come first from statistical errors which are precisely estimated and also from systematic errors that are not quantified, like the evaluation of the cross-sections for example. Only the statistical uncertainties are given in Table 2. The systematic uncertainties are not precisely known but are quite large, around 1 to 2 pcm/ °C. The safety level of such a MSR is excellent, as presented in reference[7].

TABLE 1

Total feedback coefficient for the TMSR and break down in sub-coefficients

Total Coefficient	Salt Heating (Doppler)	Salt Dilatation
-5.37 ± 0.04 pcm/°C	-3.14 ± 0.04 pcm/°C	-2.02 ± 0.04 pcm/°C

Fig. 4 shows the evolution of the uranium inventory during 100 years of operation. The sequential formation of ^{234}U and heavier uranium isotopes is clearly visible. The initial fissile inventory, 5.5 metric tons, necessary to start such a reactor is comparatively large. Even if this system is over-breeder, with a breeding ratio equal to 1.12 (thanks to the thorium blanket and the 6 month reprocessing of the FPs), this large initial inventory is the main drawback of this very interesting TMSR configuration, leading to deployment capabilities of such reactors that are not satisfactory in terms of sustainability. Solutions for the minimization of this initial fissile inventory are explored in the next paragraph.

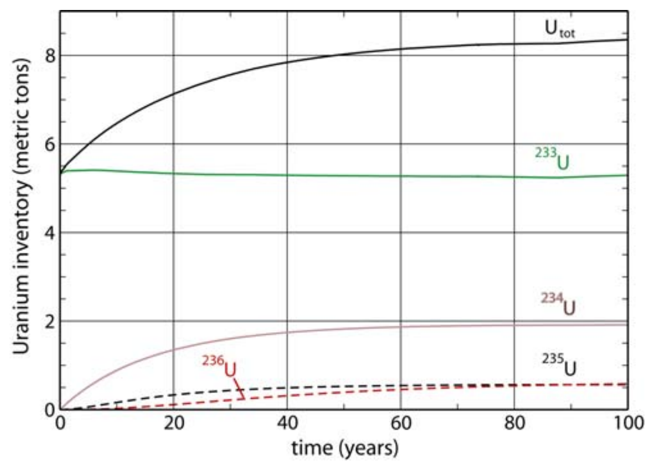

Fig. 4. Inventory evolution of uranium and its isotopes for the single channel configuration.

II.C. Minimization of the initial fissile inventory of fast TMSRs

The idea is to de-correlate the fuel function and the coolant function of the salt, leading to a reduced proportion of heavy nuclei in the salt.

This can be achieved in two ways:

- By introducing Beryllium in the fuel salt to lower its eutectic point. The salt considered is then $LiF-BeF_2-(HN)F_4$ at the same operating temperature as before.
- By operating at a higher temperature (around 1000°C), still using a binary fluoride salt $LiF-(HN)F_4$.

We present in the following of this section the results of the first solution, the Beryllium addition in the fuel salt.

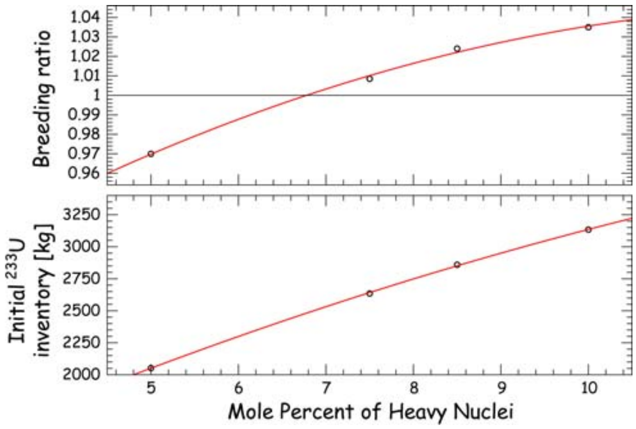

Fig. 5. Breeding ratio and initial fissile inventory of the fast TMSR as a function of the heavy nuclei proportion

Fig. 5 shows the variation of the breeding ratio and of the fissile inventory of the fast TMSR for different proportions of heavy nuclei in the fuel salt. We want a reference reactor configuration that ensures breeding (i.e. with a breeding ratio greater or equal to 1) and that can be started with as small an initial fissile inventory as possible. With less than 7% of heavy nuclei, breeding is out of reach and, with more than 10%, the fissile inventory is too large.

The reference configuration we have chosen is the TMSR with an 8.5% proportion of heavy nuclei in the salt. The heavy nuclei mole percentage being fixed, the remaining 91,5% are divided in Lithium (around 4/5) and Beryllium (around 1/5). We have checked that the Beryllium mole percentage in the salt has no influence on the neutronic behavior of the reactor.

To keep the same heavy nuclei flux in the reprocessing unit as presented in paragraph II.A, we consider here a delayed reprocessing of the total salt volume over a 3 month period.

The open question is now to determine the best way to produce the necessary initial ^{233}U inventory of these TMSRs in order to verify their deployment capabilities, and more generally to study how to launch a fleet of molten salt reactors.

To produce ^{233}U, a solution may consist in the use of special devices containing Thorium[8], in Pressurized Water (PWR) or Fast Neutron (FNR) Reactors. As already mentioned, we have considered here the production of ^{233}U directly in MSRs using an already available fissile matter such as ^{235}U or Plutonium. The use of ^{235}U to start/operate a MSR requires enrichment levels higher than some 30%, which is not realistic in terms of proliferation resistance. Consequently it is not discussed here. We have analyzed two types of MSR using Plutonium as fissile matter: MSRs operated with Plutonium and Thorium (section III) and MSRs only started with Plutonium and then based on the ^{232}Th/^{233}U fuel cycle (section IV). In both cases, we have combined fast reference TMSRs with the MSRs producing ^{233}U, to evaluate the deployment abilities of a fleet of Molten Salt Reactors.

III. FAST NEUTRON MOLTEN SALT REACTOR OPERATED WITH PLUTONIUM

III.A. Description of the reactor

We first describe the best way to quickly produce large amounts of ^{233}U. This could be done by using specific MSRs. The idea is to spare all ^{233}U produced and thus not to use it as fissile matter in the MSR, but to extract it as soon as possible. The use of a liquid fuel makes this quick Uranium extraction possible during reactor operation.

TABLE 2

Proportions of trans-uranians in UOX fuel after one use in PWR without multi-recycling (burnup of 60 GWd/ton) and after five years of storage[9]

Element	Proportion in the mix
Np 237	6.3 %
Pu 238	2.7 %
Pu 239	45.9 %
Pu 240	21.5 %
Pu 241	10.7 %
Pu 242	6.7 %
Am 241	3.4 %
Am 243	1.9 %
Cm 244	0.8 %
Cm 245	0.1 %

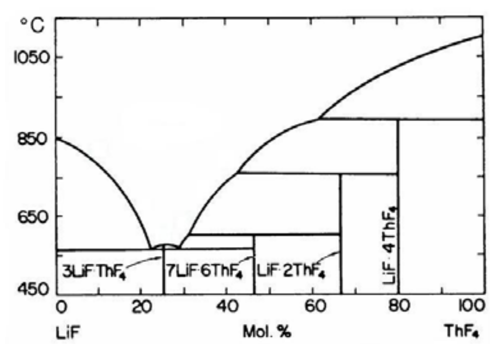

Fig. 6. LiF - ThF$_4$ phase diagram

This has two consequences for this MSR configuration:

- It has to be started and operated with Plutonium. In fact, to be more realistic, we have started and fed this MSR with a mix of Pu, Np, Am and Cm (Table 2) corresponding to the trans-uranians of UOX fuel after one use in a standard PWR and five years of storage.

- Consequently, this Pu-operated MSR has to be based on a fast neutron spectrum, and thus must contain a large percentage of heavy nuclei in the salt. We have chosen 28% of heavy nuclei, which is the second eutectic point of this LiF-(HN)F_4 salt as shown on Fig. 6.

III.B. Characteristics of the configuration

The initial fissile (Pu) inventory necessary to start such a reactor is 13 metric tons, and the Thorium initial inventory is 43 metric tons. The corresponding ^{233}U produced during the reactor's operation is displayed on Fig. 7 (red curve).

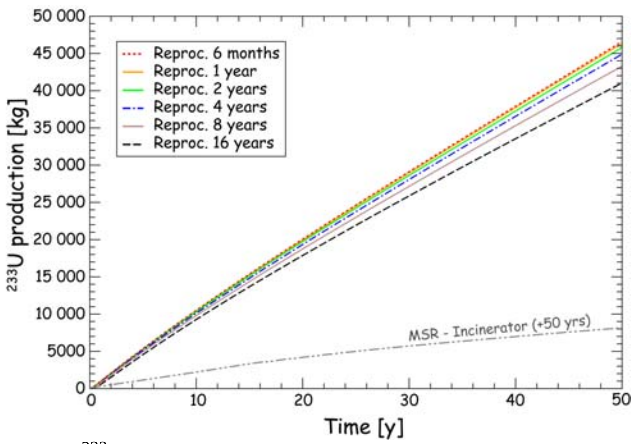

Fig. 7. ^{233}U production for various Pu-operated MSR fuel reprocessing times and for the MSR-incinerator

To minimize the wastes rejected during the reprocessing, we have studied the effect on the ^{233}U production of a slower reprocessing of the fission products and trans-uranians (Fig. 7) , ^{233}U being still extracted in a six month period. With a full core reprocessing in 16 years instead of six months, the ^{233}U production decreases by around 14% only, while the waste rejection is reduced by a factor 32. We thus considered a 16-year reprocessing for these MSRs fed with Plutonium.

Concerning safety, the total feedback coefficient for this MSR configuration is equal to -2.5 pcm/°C, it does not vary with the reprocessing time. As systematic uncertainties lie around 1 to 2 pcm/°C, this value is not negative enough to ensure a high safety level[7].

III.C. Evaluation and minimization of the wastes produced

Concerning the waste production, the amount of leakage occurring during the reprocessing is negligible compared to the final inventory, thanks to the 16-years reprocessing described in paragraph III. B. This trans-uranian inventory after 50 years of operation is detailed in Table 3 (4th column), the trans-uranians being not extracted during the reactor operation. The corresponding trans-uranian inventory of the reference TMSR (paragraph II. C) is also listed in Table 3 (first column).

The final inventory in trans-uranians of the MSR fed with Plutonium is large enough to cause trouble in terms of radiotoxicity. A simple solution is illustrated in the last column of Table 3: after 50 years of operation as Pu to ^{233}U converter, the Pu supply is stopped and the MSR continues operating with a ^{232}Th supply only, acting as a trans-uranian incinerator. This helps to reduce significantly the trans-uranian inventory, even if the inventories of the heaviest trans-uranians continue to increase. Thanks to the low quantities implied, this does not pose a huge problem.

TABLE 3

Trans-Uranian inventory/feeding (in kg) in the reference TMSR with 8.5% of heavy nuclei (at 60 years) and in the MSR with 28% of heavy nuclei fed with Plutonium

	TMSR Invent. 60 yrs	MSR-Pu Initial invent.	MSR-Pu Feeding during 50 yrs	MSR-Pu Invent. 50 yrs	MSR-Inciner Invent. + 50 yrs
^{237}Np	108.2	955.9	3430	614	48.84
^{238}Pu	149.3	423.6	1520	1751	351
^{239}Pu	57.14	7257	26040	4958	215.7
^{240}Pu	43.95	3400	12200	9619	2258
^{241}Pu	15.72	1703	6113	1940	421.3
^{242}Pu	10.73	1074	3854	3341	1206
^{241}Am	2.425	523.4	1878	1112	242.8
^{242m}Am	0.093	0	0	61.1	16.51
^{243}Am	3.700	296.8	1965	1083	449.8
^{242}Cm	0.312	0	0	52.5	12.11
^{243}Cm	0.063	0.97	3.46	7.68	2.164
^{244}Cm	4.045	126.6	454.3	962	506.5
^{245}Cm	1.500	11.73	42.11	292.1	185.8
^{246}Cm	0.610	1.67	5.981	108.2	161.9
^{247}Cm	0.154	0	0	12.91	28.61
^{248}Cm	0.038	0	0	3.06	12.94
^{249}Bk	0.003	0	0	0.052	0.243
^{249}Cf	0.006	0	0	0.197	1.125
^{250}Cf	0.004	0	0	0.059	0.417
^{251}Cf	0.002	0	0	0.008	0.090
^{252}Cf	0.0002	0	0	0.0004	0.005

IV. FAST NEUTRON MOLTEN SALT REACTOR STARTED WITH PLUTONIUM

IV.A. Description of the reactor

The idea in this section is to have a reactor able to burn Pu but without loosing the advantages of the ^{232}Th/^{233}U fuel cycle. We have thus simulated MSRs started only with Plutonium and then operated in the $^{232}Th/^{233}U$ fuel cycle. More precisely, as stated in paragraph III.A, these MSRs are started with the same mix of Pu, Np, Am and Cm listed in Table 2. They have a low heavy nuclei proportion in their fuel, as discussed in paragraph II.C. More precisely, the fuel salt considered is composed of $LiF\text{-}BeF_2\text{-}(HN)F_4$ with 8.5% of heavy nuclei.

IV.B. Characteristics of the configuration

The initial Plutonium inventory of this MSR is equal to 7010 kg. The ^{233}U production is displayed on Fig. 8 (solid lines), together with the corresponding ^{233}U production in the reference TMSR (dashed lines). This production is far better in the Pu-started MSR, improved by a factor 1.4.

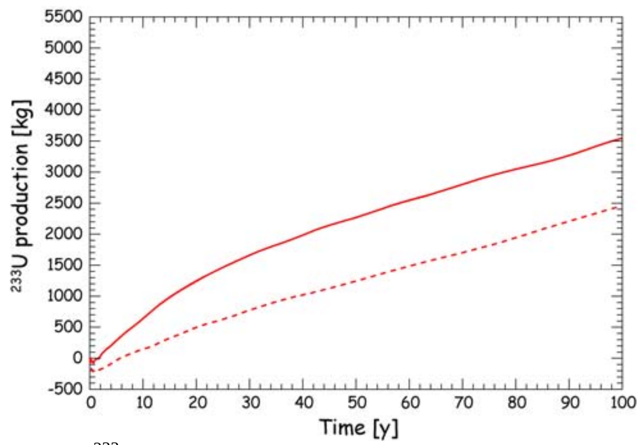

Fig. 8. ^{233}U production for the TMSR (dashed lines) and for the MSR started with Plutonium (solid lines)

Concerning safety, the total feedback coefficient for this MSR configuration lies around -9 pcm/°C. The contributions of the heating of the salt and of its dilatation are of the same order of magnitude. This very negative feedback coefficient ensures an excellent deterministic safety level[7].

IV.C Evaluation of the wastes produced

In terms of trans-uranian inventory, the MSRs started with Plutonium turn out to be nearly equivalent to Thorium Molten Salt Reactors operated with ^{233}U only after around forty years, as shown in Table 4 and Fig. 9.

As for the Pu-fed MSR (paragraph III.C), the inventories of the heaviest trans-uranians are higher for the MSR started with Plutonium than for the TMSR. Again, thanks to the low quantities implied, this does not pose a huge problem.

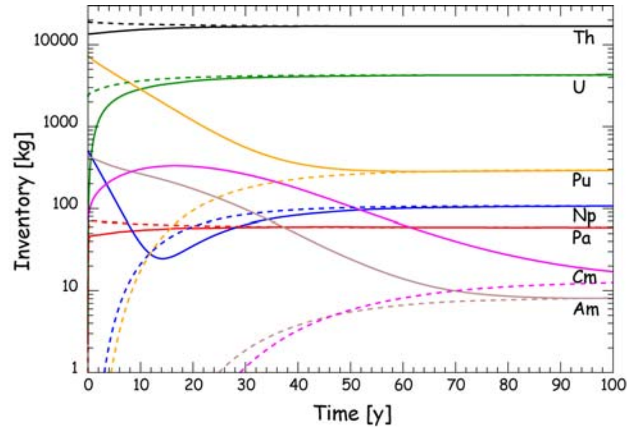

Fig. 9. Heavy nuclei inventory for the TMSR (dashed lines) and for the MSR started with Plutonium (solid lines)

TABLE 4

Trans-Uranian inventory (kg) in the TMSR with 8.5% of heavy nuclei and in the MSR started with Plutonium, initially and after 60 years of operation

	TMSR Invent. after 60 yrs (kg)	MSR-Pu Initial invent. (kg)	MSR-Pu Invent. after 60 yrs (kg)
^{237}Np	108.2	537.5	101.8
^{238}Pu	149.3	238.2	139.1
^{239}Pu	57.14	4080	52.85
^{240}Pu	43.95	1911	51.10
^{241}Pu	15.72	957.9	18.97
^{242}Pu	10.73	604.0	33.00
^{241}Am	2.425	294.3	3.237
^{242m}Am	0.093	0	0.129
^{243}Am	3.700	166.9	15.14
^{242}Cm	0.312	0	0.425
^{243}Cm	0.063	0.546	0.094
^{244}Cm	4.045	71.19	27.78
^{245}Cm	1.500	6.61	12.80
^{246}Cm	0.610	0.932	23.60
^{247}Cm	0.154	0	10.53
^{248}Cm	0.038	0	5.120
^{249}Bk	0.003	0	0.498
^{249}Cf	0.006	0	1.080
^{250}Cf	0.004	0	0.886
^{251}Cf	0.002	0	0.511
^{252}Cf	0	0	0.088

IV.D Worldwide deployment scenarios

The deployment scenarios described below rest on the following nuclear power progression[10]: starting at zero in 1970, nuclear power production rises to 340 GWe.y (GigaWatt electric-year) in 1985, to 400 GWe.y in 2000. Nuclear power remains stable from 2000 to 2015, then increases at the rate of 6.2% per year until 2050, achieving an eightfold increase by 2050; it then slowly increases by 1.1% per year until 2100. Extrapolating up to 2100 allows us to verify that the deployment scenarios are lasting.

TABLE 5

Characteristics of the FNRs considered

	Liquid metal coolant FNR
Output capacity	1.0 GWe
First operating date	2025
Reactor lifespan	50 yrs
Pu amount (per load)	6 tons
Loading periodicity	5 yrs
Number of loads	2
Breeding (per reactor-yr):	300 kg of Pu

We have simulated the deployment of several reactor technologies and examined how well they satisfy the anticipated energy demand:

- The first scenario involves light water reactors[10] (LWRs) and fast neutron breeder reactors (FNRs) (see Table 5).
- The second scenario involves light water reactors, MSRs producing ^{233}U and started with Pu (paragraph IV.A), and reference TMSRs (paragraph II.C).
- The third involves light water reactors, reference MSRs producing ^{233}U and operated with Pu (paragraph III.A), and reference TMSRs.

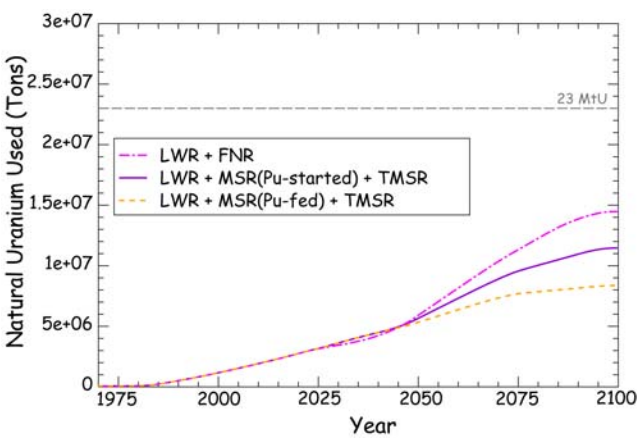

Fig. 10. Integrated natural uranium resources required by the three deployment scenarios

Our aim in this paragraph is to explore the potential for worldwide nuclear power deployment and its limitations. In this view, we first pay particular attention to the availability of uranium 235, the only natural fissile element, which is, as a consequence, the major constraining factor in the frame of sustainable development. Secondly, we evaluate the possibility of eventually shutting down the reactor fleets started, by taking in consideration the heavy nuclei produced, whose handling is tricky.

In terms of natural resources availability, as shown on Fig. 10 which represents the integrated amount of natural uranium[a] required by the nuclear industry, the best scenario for a sustainable use of the natural uranium reserves is the third one, based on LWRs, TMSRs and MSRs operated with Plutonium. The second scenario is also better than the first one (with FNRs) in terms of natural resource sustainability.

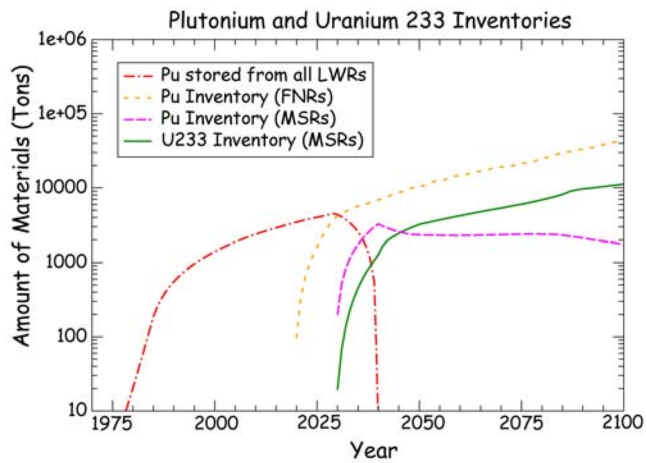

Fig. 11. Inventories in Plutonium and ^{233}U for two deployment scenarios: with LWRs and FNRs, and with LWRs, MSRs started with Pu and TMSRs

In terms of fissile matter inventories (Fig. 11), only the first two scenarios (the scenario with LWRs and FNRs, and the scenario with LWRs, MSRs started with Pu and TMSRs) are displayed, since the results of the third scenario are equivalent to the results presented here for the second scenario. A small number of MSRs operated with Pu are indeed necessary since they produce around twenty times more ^{233}U than the MSRs only started with Pu (Fig. 7 and Fig. 8). For a worldwide deployment, this compensates their individual larger trans-uranian inventory.

The scenarios based on MSRs allow a significant reduction of Plutonium inventories, down to around 1700 metric tons in 2100, instead of 45000 tons for the scenario based on FNRs, as displayed in Fig 11. The ^{233}U inventory in the scenarios with MSRs, replacing the Pu inventory for FNRs, lies around 10000 tons only, corresponding to a total amount of Uranium of 17000 tons. The role of the MSRs in the second and third scenarios is thus to close the

[a] We consider in our scenarios the optimistic value of 23 MtU for the total available natural uranium resource[10].

U-Pu fuel cycle and the amounts of plutonium and minor actinides produced are significantly smaller than in the scenario with FNRs. As a result, waste management is made simpler and easier to implement. Nuclear power deployment in this case is sustainable and efficient, the use of fissile matter and the production of wastes are optimized.

IV.E Further improvements

An interesting path of investigation to improve the MSRs consists in a reactor operating temperature increase. We have considered the same MSR configurations based on a LiF-BeF_2-$(HN)F_4$ salt (with HN equal to 8.5%) operating at 1030 °C, both for a TMSR and for a MSR started with Plutonium. At this temperature, the thermodynamic efficiency is assumed to increase from 40 % to 60 %. This has an incidence on the power of the reactor: a 2500 MWth reactor operating at 1030 °C produce 1500 MWe instead of 1000 MWe when operating at 630 °C. The characteristics (feedback coefficient, inventories) of these reactors are comparable to the results presented in paragraphs II.C and IV.B, except for the power production which is 1.5 times higher and for the breeding ratio which is slightly smaller. As a consequence, a worldwide deployment scenario based on these TMSRs and MSRs operated with Plutonium, and PWRs, is equivalent to the very good third scenario detailed in paragraph IV.D in terms of natural resource sustainability, but this time with reactors having an excellent level of deterministic safety.

The major problem of this solution is that common structure materials cannot withstand such a temperature increase. However, new promising solutions based on carbon (carbon-carbon, carbon fiber, carbides...) could help solve this problem. If these technologies are not implemented, then this solution will have to be ignored.

V. CONCLUSIONS

We have first presented a very interesting Molten Salt Reactor configuration called fast Thorium Molten Salt Reactor (TMSR); it is based on the ^{232}Th/^{233}U fuel cycle and a fast neutron spectrum. We have detailed the characteristics of this reference TMSR configuration. A solution has been proposed to minimize the initial fissile inventory of a fast TMSR down to 2.8 metric tons only. We have then tried to give an answer to a major question on MSRs, concerning the supply of the fissile matter (^{233}U), and as a consequence the deployment possibilities of a fleet of MSRs.

We have examined two new ways of producing ^{233}U: directly in standard MSRs started with Plutonium as fissile matter and then operated in the Th/^{233}U cycle; or in dedicated MSRs started and fed with Plutonium as fissile matter and Thorium as fertile matter. Our idea was to design a critical reactor able to burn the Plutonium presently produced in PWRs, and consequently to convert this Plutonium into ^{233}U.

We have analyzed the characteristics of such reactors, in terms of deterministic safety parameters, fissile matter inventory, salt reprocessing scheme, waste production… We have demonstrated that the conversion capacities of both MSR types are excellent.

In the case of MSRs started and fed with Plutonium, we have shown that the ^{233}U production is very large and allows the more sustainable worldwide deployment. We have also studied the large radiotoxicity and trans-uranian production, which seem to be the main drawbacks of this concept. We have proposed a solution that consists in extending the reactor's operation as trans-uranian incinerator. The last negative point for these reactors is their rather low deterministic safety level, which could result in lower economical advantages.

For MSRs only started with Plutonium, we have checked that the assets of the Thorium fuel cycle are quickly recovered and that the reactor's characteristics turn out to be equivalent to MSRs operated with ^{233}U only, except for the ^{233}U production capacity which is much better. Moreover, by combining the reference TMSR configuration presented at the beginning of this paper with this MSR started or operated with Plutonium, we have shown that the worldwide deployment of such a fleet of MSRs is possible and sustainable.

ACKNOWLEDGMENTS

We are thankful to Elisabeth Huffer for her help during the translation of this paper.

REFERENCES

1. L. Mathieu et al, "Proposition for a Very Simple Thorium Molten Salt Reactor", *Global Conference*, Tsukuba, Japan (2005).

2. L. Mathieu, D. Heuer et al, "The Thorium Molten Salt Reactor: Moving on from the MSBR", sub. to *Prog in Nucl Ener* (2006).

3. L. Mathieu, "Cycle Thorium et Réacteurs à Sel Fondu : Exploration du champ des Paramètres et des Contraintes définissant le Thorium Molten Salt Reactor", PhD thesis, Institut National Polytechnique de Grenoble, France (2005) (in French).

4. J.F.Briesmeister, "MCNP4B-A General Monte Carlo N Particle Transport Code", Los Alamos Lab. report LA-12625-M (1997).

5. I. Victor, E. Walle et al, "Density of Molten Salt Reactor Fuel Salts", *Nureth Conference,* Avignon, France (2005).

6. A. Nuttin et al, "Potential of Thorium Molten Salt Reactors", *Prog. in Nucl. En.,* vol 46, p. 77-99 (2005).

7. E. Merle-Lucotte, D. Heuer, L. Mathieu, C. Le Brun, "Molten Salt Reactor: Deterministic Safety Evaluation", *European Nuclear Conference*, Versailles, France (2005).

8. E. Merle-Lucotte, L. Mathieu, D. Heuer, J.-M. Loiseaux et al, "Molten Salt Reactors and Possible Scenarios for Future Nuclear Power Deployment", *Proceedings of the Physor 2004 Conference, The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*, American Nuclear Society (Ed.), 1-12 (2004).

9. C. de Saint Jean, M. Delpech, J. Tommasi, G. Youinou, P. Bourdot, "Scénarios CNE : réacteurs classiques, caractérisation à l'équilibre", CEA report DER/SPRC/LEDC/99-448 (2000) (in French).

10. E. Merle-Lucotte, D. Heuer, C. Le Brun and J.-M. Loiseaux, "Scenarios for a Worldwide Deployment of Nuclear Power - Full Report", LPSC Internal Report LPSC 05-72 (2005).