Recommendations for a demonstrator of Molten Salt Fast Reactor

E. Merle-Lucotte, D. Heuer, M. Allibert, M. Brovchenko, V. Ghetta, P. Rubiolo, A. Laureau

LPSC-IN2P3-CNRS / UJF / Grenoble INP, Grenoble, France

Abstract. There is currently a renewed interest in molten salt reactors, due to recent conceptual developments on fast neutron spectrum molten salt reactors (MSFRs) using fluoride salts. This concept, operated in the Thorium fuel cycle, may be started either with ^{233}U, enriched U and/or TRU elements as initial fissile load. It has been recognized as a long term alternative to solid fuelled fast neutron systems with a unique potential (such as large negative temperature and void coefficients, lower fissile inventory, no initial criticality reserve, simplified fuel cycle, wastes reduction…) and is thus one of the reference reactors of the Generation IV International Forum. This paper will focus on recommendations to define a demonstrator representing the key points of the reference MSFR power reactor (3000 MW_{th}, fuel salt volume of 18 m^3). The MSFR demonstrator is designed to assess the technological choices of this innovative system (fuel salt, structural materials, fuel heat exchangers…). It seems finally possible to slightly modify such a demonstrator which could then be a self-breeder modular reactor

1. Introduction and presentation of the MSFR concept

Molten Salt Reactor (MSR) technology was studied in the 1950's and 1960's in USA (Oak Ridge National Laboratory), including two demonstration reactors. These Molten Salt Reactors were initially designed as thermal-neutron-spectrum graphite-moderated reactors. Since 2004, the National Centre for Scientific Research (CNRS, Grenoble-France) has been focused R&D efforts on the development of a new fast-spectrum reactor based on the MSR concept. The reference MSFR (Molten Salt Fast Reactor) combines then the generic assets of fast neutron reactors (reduced neutron absorptions in the fission products, waste minimization) with those related to molten salt fluorides as fluid fuel and coolant (low pressure, high boiling temperature and optical transparency) [1][2][3][4]. As opposed to thermal molten salt reactors, the specificity of the MSFR is the absence of any solid moderator (usually graphite) in the core. This concept has been recognized as a long term alternative to solid fuelled fast neutron systems with a unique potential (large negative temperature and void coefficients, lower fissile inventory, no initial criticality reserve, simplified fuel cycle, wastes reduction…) and has thus been selected as one of the reference reactors of the Generation IV International Forum [5][6][7].

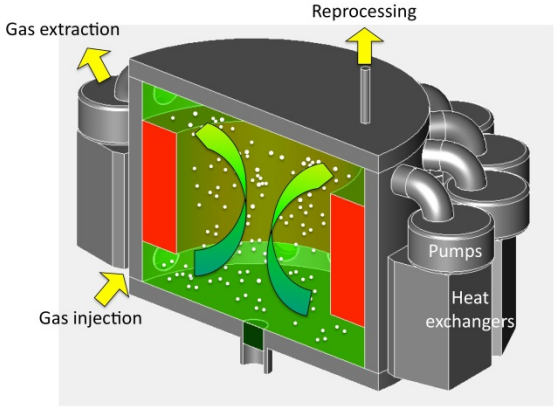

FIG. 1. Schematic conceptual MSFR design

The reference MSFR is a 3000 MWth reactor with a total fuel salt volume of 18 m^3, operated at a mean fuel temperature of 750°C. Figure 1 sketches the general component outlines for such a MSFR. The core consists of a circulating fluoride salt loaded with the fuel (note the absence of solid matter in core). The fuel salt composition currently being investigated is a binary fluoride salt with 77.5% of lithium fluoride enriched in ^7Li to 99.995 %; the other 22.5% are a mix of heavy nuclei fluorides. This proportion, set throughout the reactor evolution and corresponding to an eutectic point with a melting temperature of 565°C, leads to a fast neutron spectrum. This MSFR system thus combines the generic assets of fast neutron reactors (extended resource utilization, waste minimization) with those associated to a liquid-fuelled reactor. The characteristics of the MSFR are detailed in Table 1.

Table 1. Characteristics of the reference MSFR

Thermal power (MWth)	3000
Mean fuel salt temperature (°C)	750
Fuel salt temperature rise in the core (°C)	100
Fuel molten salt - Initial composition (mol%)	LiF-ThF_4-$^{233}UF_4$ or LiF-ThF_4-($^{enriched}U$+Pu+MA)F_3 with 77.5 % LiF
Fuel salt melting point (°C)	565
Fuel salt density (g/cm^3)	4.1
Fuel salt dilation coefficient (/°C) [8]	$8.82\ 10^{-4}$
Fertile blanket salt - Initial composition (mol%)	LiF-ThF_4 (77.5%-22.5%)
Breeding ratio (steady-state)	1.1
Total feedback coefficient (pcm/K)	-5
Core dimensions (m)	Radius: 1.1275 Height: 2.255
Fuel salt volume (m^3)	18 9 in the core 9 in the external circuits
Blanket salt volume (m^3)	7.3
Total fuel salt cycle in the fuel circuit	3.9 s

In the MSFR, the liquid fuel processing is performed during the reactor operation [9]. The on-site salt management of the MSFR combines a salt control unit, an online gaseous extraction system in the core (with gas injectors and bubble separators) and an offline lanthanide extraction component by pyrochemistry where a small stream (10 to 40 liters per day) of the molten salt is set aside to be processed for fission product removal and then returned to the reactor. This is fundamentally different from a solid fuel reactor where separate facilities produce the solid fuel and process the Spent Nuclear Fuel. Because of this design characteristic, the MSFR can operate with widely varying fuel compositions. Thanks to this fuel composition flexibility, the MSFR concept may use as its initial fissile load, ^{233}U or enriched natural uranium and/or also the transuranic (TRU) elements currently produced by PWRs in the world. The only natural fissile matter on earth is ^{235}U (0.72% of natural uranium), which can be used directly as enriched uranium in breeder reactors for their initial fissile load, or which can be loaded in fertile blankets in nuclear reactors to produce either ^{239}Pu by irradiating ^{238}U, or ^{233}U by irradiating ^{232}Th. To start a first fleet of MSFRs, we have thus investigated the following solutions [10]:

- Producing ^{233}U in the fertile blanket of other reactors (SFR…) or by irradiating ^{232}Th in an ADS for example, to start the MSFR directly with this ^{233}U as initial fissile matter. Once an initial park of the MSFRs based on the Th/^{233}U cycle is launched, ^{233}U will also be produced in MSFRs which are breeder reactors, allowing the deployment of such ^{233}U-started MSFRs in a second time period even if no ^{233}U is produced elsewhere.
- Using as initial fissile matter the plutonium produced in current PWRs or in future EPRs or, even better, the mix of transuranic elements (TRU) produced by these Generation 2-3 reactors.

The calculations show that the initial Pu proportion of a TRU-started MSFR reaches the solubility limit given for a LiF-ThF_4 salt. A solution to limit the Pu initial proportion in the fuel salt consists in adding small amounts of ^{233}U or $^{enriched}U$ to reach criticality (see below).

- Starting the MSFR with enriched uranium as initial fissile matter, with an enrichment ratio lower than 20% due to proliferation resistance issues.
- A mix of the previous starting modes. For example, ^{233}U may be produced by using special devices containing Thorium and Pu-MOx (called "MOx-Th" thereafter) in current PWRs or in future EPRs.

Table 2. Initial heavy nuclei inventories per GWe of the different MSFR starting modes

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

In order to initiate the work and the discussions on possible ranges for the reactor parameters, basic drawings have been developed from the preliminary conceptual design. Figure 2 illustrates one of the possible geometrical configurations (height/diameter ratio = 1). In these preliminary designs, the core of the MSFR is a single compact cylinder (2.25m high x 2.25m diameter) where the nuclear reactions occur within the liquid fluoride salt acting both as fuel and as coolant. The return circulation of the salt (from the top to the bottom) is divided into 16 loops located around the core, each loop contaning a pump and a fuel heat exchanger (also labelled IHX for Intermediate Heat Exchanger). The total fuel salt volume ($18m^3$) is distributed 50% of it in core, 5% in auxiliary volumes (overflow tank, spaces…) and 45% in the bubble separators, the pumps and pipes, and the heat exchangers [11][12]. Since here the fuel salt also plays the role of the coolant, one of the main constraints on the design of the fuel circuit of the MSFR is the ability to evacuate the heat generated while restraining the fuel salt volume mobilized for that task. The time circulation of the fuel salt in the whole fuel salt circuit is of the order of a few seconds.

The neutronic reflectors, made of NiCrW-based alloy [13] in our calculations, constitute the lower and upper walls of the core. The lower reflector is connected to a draining system: in case of a planned shut down or incident/accident leading to a temperature increase in the core, the fuel configuration may be changed passively by gravitational draining of the fuel salt in tanks located under the reactor where a passive cooling will be achieved.

The radial reflector includes a fertile blanket (50 cm thick - red area in Fig. 1 and 2) to increase the breeding ratio. This blanket is filled with a fertile salt of LiF-ThF_4 with initially 22.5mole % $^{232}ThF_4$. The blanket is surrounded by a 20cm thick layer of B_4C, which protects the external circuit from the remaining neutrons.

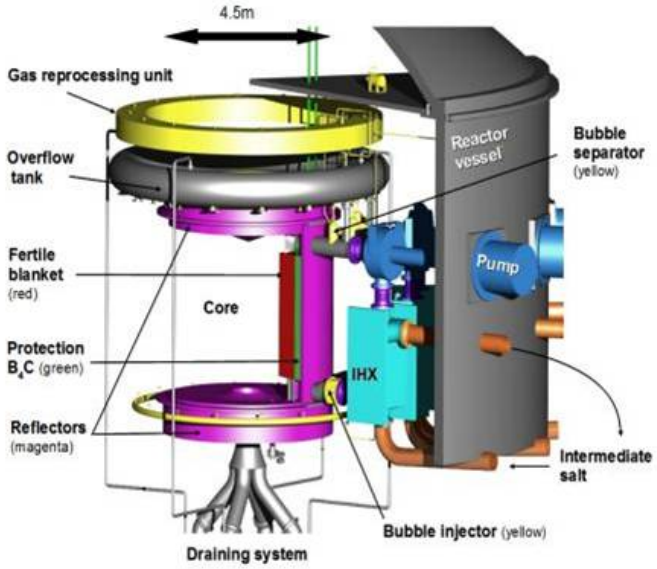

FIG. 2. Pre-design of the fuel salt circuit of the MSFR

2. Demonstration and Demonstrators of MSFR

2.1. What needs to be demonstrated

Given the nonconformity of the concept and the lack of current expertise, it is clearly not possible to move directly to a demonstration phase involving actinides. Prior to that, a variety of experimental setups have to be devised to validate insofar as possible the technological solutions, with non-radioactive or weakly radioactive materials, and demonstrate their viability to the safety authorities. The want of experts in the specific field of molten salts entails, in any case, a time lag due to the training of both the designers of the concept and the safety authority personnel.

In the frame of this R&D, the experimental setups can and should, in the initial phases of the demonstration, be limited in size and complexity so as to facilitate the necessary modifications that will be associated to the developments. Thus, at the early stages, the experiments may work with "simulant salts" and move to a fuel salt depleted in fissile elements (containing neither Pu nor $^{233}U/^{enriched}U$) only when it will become really necessary for the assessments considered. The devices that are to be planned later must go as far as the quantification of the characteristics so as to allow the validation of the mathematical models which will be used to define precisely the details of a demonstrator that integrates all the previously tested functions. Finally, experimental studies will have to involve a salt with induced fissions to demonstrate the viability, strictly speaking, of the concept.The section 2.4 will focus on this final demonstrator called "power demonstrator".

2.2. Phases of the demonstration and conception of the demonstrator

The demonstration phases comprise 3 degrees corresponding to 3 radioprotection levels: non radioactive (simulant salt), radioactive (^{nat}LiF-ThF_4-$^{depleted}UF_4$-$^{depleted}UF_3$), and radioactive with induced fissions. All technical innovations will have to be first tested with simulant salts and the move to radioactive systems should only prove that the innovations validated in non-radioactive devices are still valid. It is thus necessary to have access to equipments that have already demonstrated their operational aptitudes without the handicap of necessary radio-protection and with which the

exploration of wide operational domains (temperature, flow speeds, gas and suspended particle loads, sizing and particular geometries) will have established our understanding and at least partly validated the mathematical models.

2.3. Demonstrator with a simulant salt - test benches

The most common simulant fluoride is the "FLiNaK" eutectic (LiF-NaF-KF). Mixtures such as LiF-NaF-ZrF_4 or NaF-KF-ZrF_4, however, can be used to improve the viscosity and density. These salts are liquid beyond 450°C and they are thermally quite stable. They are also representative of the intermediate circuit salt (if this option is retained). The mastering of the following issues, some of which are interdependent, needs to be demonstrated or quantified:

- The handling of large amounts of molten salts.
- The design of the pumps, how well they perform, their reliability, how they withstand wear.
- The validation of the hydrodynamic flow models.
- The development of the instrumentation.
- The validation of molten salt heat exchanger components.
- The separation and extraction of dissolved gases and suspended particles.
- The validation of the flushing system components.

The FFFER (Forced Fluoride Flow for Experimental Research) facility [14], which is currently being built, constitutes a preparatory step towards this endeavor. It will be operated between 500 and 700°C, with a LiF-NaF-KF salt. One of its objectives is to evaluate the efficiency of this bubbling process in a fluoride salt, by reproducing the gases extraction of the MSFR at a 1/10th scale in a simulant salt at high temperature.

2.3.1. Radioactive demonstrator without induced fissions

The aim of these demonstrators is to take into account the specifics of the salt with actinides, in relation to the flow dynamics and to the salt chemistry. Some of the items are:

- A small corrosion demonstrator subjected to heat gradients.
- A small chemical processing demonstrator.
- A large scale hydrodynamic demonstrator, including the heat exchangers.

2.3.2. Radioactive demonstrator with induced fissions

The purpose of this demonstrator is:

- To obtain approval from the safety authorities, i.e. to answer beforehand any issues they might raise.
- To demonstrate that the reactor can be controlled (start-up, power changes, shut-down).
- To manage shut-downs and flushing in the specific conditions of the thermally active liquid.
- To generate gaseous and condensed fission products (that cannot be realistically simulated) and extract them.
- To validate the heat evacuation system in realistic conditions
- To irradiate the system as a whole and not just samples.

These items do not all have to be addressed with a single machine, that would accumulate the difficulties and the associated costs. The first two can be achieved with a small, very low specific power reactor (infrastructure economy). The other items require much larger power for the production of fission products and the production of representative irradiation damages (displacements per atom, He production, transmutations), as detailed in the next section.

2.4. Power demonstrator of the Molten Salt Fast Reactor

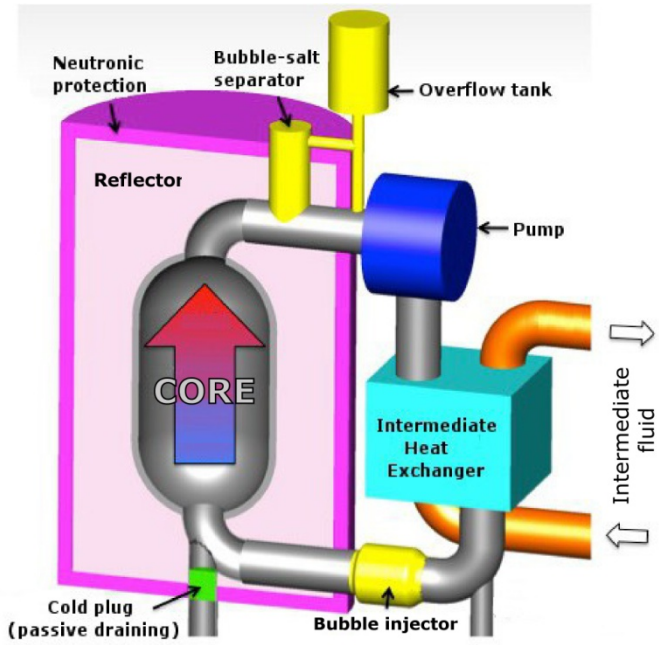

FIG. 3. Schematic view of the MSFR demonstrator with one of the 6 external loops

A 100MW_{th} or so demonstrator seems necessary to produce a sufficient amount of gaseous and metallic fission products to test the on-line gaseous fuel processing system. The salt volume involved (1.8 m^3) would be one tenth that of the reference reactor (18 m^3) with 60% of the salt in the core, to have a system representative of the reference MSFR in terms of heat exchangers (fluid velocity, thickness of the plates and gap between the plates - both on the side of the fuel salt and of the intermediate fluid), irradiation damages and neutronic behavior. It would be operated during only a few years. The batch reprocessing will thus not be mandatory but the system could provide, eventually, salt samples useful to test the off-line salt reprocessing (soluble fission product extraction). Since this demonstrator will be operated on a short period only, having a self-breeder system is not mandatory. Consequently, and to simplify the design, the fertile blanket will be replaced by a radial inert reflector identical to the axial reflectors, as illustrated in Figure 3. The return circulation of the salt (from the top to the bottom) is here divided into 6 loops located around the core, similar to the external loops of the reference MSFR.

Table 3. Characteristics of the MSFR power demonstrator

Thermal power (MWth)	100
Mean fuel salt temperature (°C)	725
Fuel salt temperature rise in the core (°C)	30
Fuel Molten salt initial composition (mol%)	LiF-ThF_4-$^{233}UF_4$ or LiF-ThF_4-(^{enriched}U+MOx-Th)F_3 with 77.5 % LiF
Fuel salt melting point (°C)	565
Fuel salt density (g/cm^3)	4.1
Core dimensions (m)	Radius: 0.556 Height: 1.112
Fuel Salt Volume (m^3)	1.8 1.08 in the core 0.72 in the external circuits
Total fuel salt cycle in the fuel circuit	3.5 s

Ideally, having a ^{233}U initial load of about 650kg available will be the simpliest way to start such a demonstrator. Failing that, starting with transuranic elements only as initial fissile load seems not possible here: the plutonium solubility constraint becomes more pressing than for the reference MSFR, because a reduction of the size of the reactor leads to increasing the fissile proportion in the salt by about one third. Start-up with ^{enriched}U mixed with transuranic elements has been studied, as presented in Figure 4 where the configuration at the top of the right axis is equivalent to the configuration started with TRU elements only (and located largely above the Pu solubility line as already mentionned). The configurations located at abscissa 0 correspond to an initial fuel salt composition without Thorium but with enough TRU elements to reach criticality. The blue line labeled "no TRU" at the bottom left represents different simulations of the MSFR started with enriched Uranium only, while the purple line labeled "^{enr}U at 100%" represents the results for MSFR configurations with pure ^{235}U mixed with Thorium and the amount of TRU elements necessary to reach criticality. The other solid line curves (labeled 10% to 30%) show the maximum concentrations in valence-III elements obtained during the reactor operation as a function of the (Th/Th+U) initial ratio for Uranium enrichment ratios from 10% to 30%.

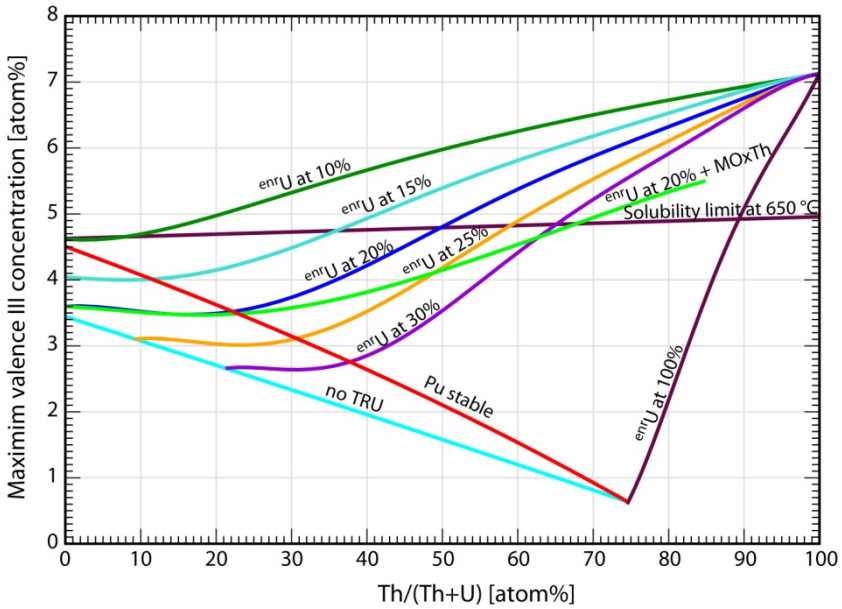

FIG. 4. Maximal concentration in valence-III elements in the initial fuel salt when starting with enriched Uranium and TRU elements, as a function of the Uranium enrichment and of the initial Th/(Th+U) ratio

In the area located on the left of the red line labeled "Pu stable", the maximal Pu concentration is reached after a few years of reactor operation, while it decreases during the entire reactor lifetime for the MSFR configurations located on the right of this line. For the configurations placed exactly on this line, the Pu concentration being stable during the first years of operation to then decrease and reach a Pu concentration equivalent to a ^{233}U-started MSFR at equilibrium. The more interesting configurations are thus placed on this line or on its right. Only the configurations located below the line "Solubility limit at 650°C" (650°C being the colder point of the salt in the fuel circuit) can be considered. Start-up with ^{enriched}U mixed with transuranic elements is then possible with enrichment between 15% and 20%.

Another option consists in mixing this enriched uranium with irradiated MOx-Th (contaning ^{233}U) that will have to be produced in today's reactors. One can see from this study (see light green curve labelled "^{enr}U at 20% + MOx-Th" in Figure 4) that such a demonstrator may easily be started with an initial fissile salt composed of Uranium enriched at 20% mixed with MOx-Th with a ratio of Th/(Th+U) between 20 to 65%. The larger the amount of Thorium is in the initial fuel salt, the better the breeding ratio of such a reactor will be.

2.5. From the MSFR demonstrator to a modular reactor

At the present time, it seems possible to slightly modify such a demonstrator which could then be a self-breeder modular reactor thanks to the addition of fertile blankets and a slow chemical reprocessing of the fuel salt ($1/10^{th}$ of the reprocessing rate of the reference MSFR only). It should be combined with a system to extract the uranium from the fertile blankets in order to provide the fissile matter feed-in. It is interesting to note that such a reactor could be operated at a power of up to $300MW_{th}$. This would then be a "modular" reactor that could be exploited commercially.

Table 4. Breeding capacities of different configurations of modular MSFR operated during 30 years

	No radial blanket and H/R=2	**No radial blanket and H/R=2**	**Radial blanket and H/R=2**	**Radial blanket and H/R=2**	**Radial blanket and H/R=3**	**Radial blanket and H/R=3**
Power [MW_{th}]	100	200	100	200	100	200
Initial ^{233}U load [kg]	654	654	667	667	677	677
Fuel reprocessing of 1l/day						
Feeding in ^{233}U [kg/an]	11.38	23.38	1.72	4.70	-0.07	0.98
Breeding ratio	-29.83%	-30.64%	-4.52%	-6.16%	***0.18%***	-1.29%
Total ^{233}U needed [kg]	1013.87	1388.37	738.83	835.16	715.05	754.25
Breeding ratio with an additional axial fertile blanket			***1.81%***	***-0.04%***		
Fuel reprocessing of 4l/day						
Feeding in ^{233}U [kg/an]	11.20	22.58	1. 48	3.58	-0.38	-0.26
Breeding ratio	-29.37%	-29.59%	-3.88%	-4.69%	***1.00%***	***0.34%***
Total ^{233}U needed [kg]	1001.86	1353.13	722.50	794.21	709.74	723.03
Breeding ratio with an additional axial fertile blanket			***2.49%***	***1.54%***		

Four parameters have been considered for the evaluation of the breeding capacities of such a reactor, as detailed in Table 4: the produced power (100 and 200 MW_{th}), the ratio of the height/radius (H/R) of the core, the addition (columns labeled 'Radial blanket') or not (columns labeled 'No radial blanket') of a radial fertile blanket around the core, and the impact of the reprocessing rate of the fuel salt between 1 and 4 liters per day. The results written in bold italic in Table 4, whose breeding ratio is higher or equal to zero, represent the configurations suitable for a breeder reactor. It is to be noted that the trends deduced from the calculations are correct but that the numerical results reported here are impacted by uncertainties on the nuclear data (mainly on the capture over fission ratio of ^{233}U) leading to +/- 2 to 3% margins on the breeding ratios.

With a reprocessing rate of 1 liter of fuel salt per day, only the configuration producing 100 MW_{th} with a ratio H/R equal to three and including a radial fertile blanket corresponds to an iso-breeder reactor. With a reprocessing rate of 4 liters of fuel salt per day, the two configurations with an elongated core (H/R=3) are breeder.

The addition of an axial fertile blanket above the core on top of the radial fertile blanket has been checked to improve the breeding capacities of the modular reactor. As presented in Table 4, this solution leads to breeder configurations even for a core whose height/radius ratio equal to 2 (height=diameter). A molten salt modular reactor could be operated at a power of $300MW_{th}$ in a breeder mode only with such an axial blanket.

ACKNOWLEDGEMENTS

The authors wish to thank the PACEN (Programme sur l'Aval du Cycle et l'Energie Nucléaire) and NEEDS (Nucléaire : Energie, Environnement, Déchets, Société) programs of the French National Centre for Scientific Research (CNRS), the IN2P3 department of CNRS, and the European program EVOL (Evaluation and Viability of Liquid Fuel Fast Reactor System) of FP7 for their support. We are also very thankful to Elisabeth Huffer for her help during the translation of this paper.

REFERENCES

[1] MATHIEU L., HEUER D. et al, The Thorium Molten Salt Reactor: Moving on from the MSBR, Prog in Nucl En, 48, 664-679 (2006).

[2] FORSBERG C.W.et al, "Liquid Salt Applications and Molten Salt Reactors", Revue Générale du Nucléaire N° 4/2007, 63 (2007).

[3] MATHIEU L., HEUER D., MERLE-LUCOTTE E. et al., Possible Configurations for the Thorium Molten Salt Reactor and Advantages of the Fast Non-Moderated Version, Nucl. Sc. and Eng., 161, 78-89 (2009).

[4] MERLE-LUCOTTE E., HEUER D. et al, Optimizing the Burning Efficiency and the Deployment Capacities of the Molten Salt Fast Reactor, Contribution 9149, Global 2009 - The Nuclear Fuel Cycle: Sustainable Options & Industrial Perspectives, Paris, France (2009).

[5] US DOE Nuclear Energy Research Advisory Committee and the Generation IV International Forum, A Technology Roadmap for Generation IV Nuclear Energy Systems, GIF-002-00 (2002).

[6] RENAULT C. et al., The Molten Salt Reactor (MSR) in Generation IV: Overview and Perspectives, http://www.gen-4.org/GIF/About/documents/30-Session2-8-Renault.pdf, Proceedings of the GIF Symposium 2009, Paris, France (2009).

[7] BOUSSIER H. et al, The Molten Salt Reactor in Generation IV: Overview and Perspectives, Proceedings of the Generation4 International Forum Symposium, San Diego, USA (2012).

[8] IGNATIEV V., FEYNBERG O., MERZLYAKOV A. et al, Progress in Development of MOSART Concept with Th Support, Proceedings of ICAPP 2012, Paper 12394 Chicago, USA (2012).

[9] DELPECH S. MERLE-LUCOTTE E. et al, Reactor physics and reprocessing scheme for innovative molten salt reactor system, J. of Fluorine Chemistry, Volume 130, Issue 1, p. 11-17 (2009).

[10] MERLE-LUCOTTE E. HEUER D. et al, Launching the Thorium Fuel Cycle with the Molten Salt Fast Reactor, Contribution 11190, Actes de la conférence International Congress on Advances in Nuclear Power Plants (ICAPP), Nice, France (2011).

[11] BROVCHENKO M. et al, Preliminary safety calculations to improve the design of Molten Salt Fast Reactor, PHYSOR 2012 Advances in Reactor Physics Linking Research, Industry, and Education, Knoxville, Tennessee, USA (2012).

[12] MERLE-LUCOTTE E., HEUER D. et al, Preliminary Design Assessment of the Molten Salt Fast Reactor, Contribution A0053, Proceedings of the European Nuclear Conference ENC2012, Manchester, UK (2012).

[13] DELPECH S. MERLE-LUCOTTE E. AUGE T. HEUER D., MSFR: Material issues and the effect of chemistry control, 2009 Generation4 International Forum Symposium, Paris, France (2009).

[14] GHETTA V. et al., Boucle en convection forcée pour l'étude du nettoyage en ligne de caloporteurs de type sel fondu, Contribution 0574, Proceedings of the Materiaux 2010 Conference, Nantes, France - In french (2010).