

The Molten Salt Reactor (MSR) R&D Status and Perspectives in Europe

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SUMMARY and KEY MESSAGES

Since several years, the R&D on MSR is focused on fast spectrum concepts (MSFR) which have been recognized as long term alternatives to solid-fuelled fast neutron reactors with attractive features (very negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle...). MSFR designs are available for breeding and for plutonium and minor actinides transmutation and recycling. They are robust reference configurations with significant improvement compared to the MSBR concept developed in the 60's in USA.

Important progress has been achieved in critical areas of MSFR R&D, particularly in the determination of liquid salt properties of reference compositions, the definition and assessment of the fuel salt clean-up scheme, and qualification of high performance materials (corrosion resistance).

A network on MSR R&D has been active in Europe from 2001 to 2008 with financial support from the Framework Programmes of the European Commission. In parallel, ISTC has provided another efficient way of collaboration between Russian research organizations, EU partners and non-EU partners. This has also been a favourable framework for the organisation of training sessions with participation of both EU and Russian scientists.

The coordinated effort on MSR R&D in the EU and Russia should be continued in the future. In 2009, a new MSR proposal has been submitted to the 3rd call of the 7th Framework Program as a joint Euratom-Rosatom project. This EVOL project is under contract negotiations after a positive expert evaluation, in the frame of the Euratom collaboration with Rosatom.

The EU (Euratom), France and USA participate in the Generation IV MSR Steering Committee. Although the EU and USA interests are focused on different baseline concepts (MSFR and AHTR, respectively), large commonalities in basic R&D areas (e.g. liquid salt technology, materials) exist and the Generation IV framework is useful to optimize the R&D effort.

1 INTRODUCTION

In a Molten Salt Reactor (MSR), the fuel is dissolved in a fluoride salt coolant. The technology was partly developed in the 1950's and 1960's in USA (ORNL). Compared with solid-fuelled reactors, MSR systems have lower fissile inventories, the absence of radiation damage that can limit fuel burn up, the possibility of continuous fission-product removal, the avoidance of the expense of fabricating fuel elements, the possibility of adding makeup fuel as needed, which precludes the need for providing excess reactivity, and a homogeneous isotopic composition of fuel in the reactor. These and other characteristics may enable MSRs to have potentially unique capabilities and competitive economics for actinide burning and extending fuel resources.

Prior MSR designs were mainly considered as thermal-neutron-spectrum graphite-moderated concepts. Recent MSR developments in Europe on TMSR-NM [6-12] and MOSART concepts [19,22,23] address advanced large power units without graphite in the core and a fast neutron spectrum. The Molten Salt Fast Reactor (MSFR) combines the generic assets of fast neutron reactors (waste minimization and extended resource utilization) to those relating to molten salt fluorides as fluid fuels and coolants (favourable thermal-hydraulic properties, high boiling temperature, optical transparency). In addition, the MSFR, without solid moderator in the core, exhibits large negative temperature and void reactivity coefficients, a unique safety characteristic not found in solid-fuel fast reactors.

Apart from MSR systems, other advanced reactor concepts are being studied, which use the liquid salt technology, essentially as a primary coolant for the Advanced High-Temperature Reactor (AHTR) [3] or intermediate coolant, as an alternative to secondary sodium, for Sodium Fast Reactors (SFR).

More generally, the development of higher temperature salts as coolants would open new nuclear and non-nuclear applications. These salts also are being considered for hydrogen production concepts, oil refineries, and shale oil processing facilities amongst other applications [1,2].

The renewal and diversification of interests have led to a shift of the R&D orientations and objectives initially promoted in the original Generation IV roadmap issued in 2002, so as to encompass in a consistent body the different applications envisioned today for fuel and coolant salts. The status of MSR in Generation IV is the subject of other papers [4,5].

The present paper concentrates on MSFR concepts which are receiving most attention in the EU context. It shows the main R&D achievements and some remaining issues to be addressed in such essential areas as (a) reactor conceptual design, (b) molten salt properties, (c) fuel salt clean-up scheme and (d) high temperature materials (section 2). The status and perspectives of MSR R&D in Europe are then discussed (section 3).

2 R&D ACHIEVEMENTS AND REMAINING ISSUES

Significant progress has been achieved in critical areas of MSFR R&D. In the European context and related collaborations with Russia, the essential facts are the following:

1. First fast neutron MSFR pre-conceptual designs for breeding and for plutonium and minor actinides transmutation and recycling are in progress [6-12,19,22,23].
2. Salt selection for different applications is stabilized, the needs of complementary data have been clarified (Euratom 6th FWP [13] and ISTC-1606 [21]).

3. A strongly improved (versus MSBR) fuel salt clean-up scheme has been developed [11,15,16].
4. Advanced high performance materials for MSFR designs have been proposed (Ni-Mo and Ni-W-Cr alloys). The latter are alternates, for higher temperature operation ($> 750^{\circ}\text{C}$), to available Ni-Mo alloys optimized for fuel salt temperatures $< 750^{\circ}\text{C}$ [17].
5. Better understanding has been obtained of the transmutation capabilities, dynamics and safety related parameters, for fertile and fertile-free MSR fuel concepts [19].

Those topics are the subject of the following sub-sections.

2.1 Pre-conceptual design of Th-U MSFR

The favorable experience gained by ORNL in the USA from the 8 MWt MSRE test reactor, operated from 1965-1969, led to a reference Molten Salt Breeder Reactor (MSBR) design development with a thermal spectrum and thorium fuel cycle [14]. Based on this concept, an innovative thorium-fuelled MSFR has been proposed [6,7]. This concept resulted from extensive parametric studies in which various core arrangements, reprocessing performances and fuel salt compositions were investigated. The main feature of the MSFR (Molten Salt Fast Reactor) concept is the removal of the graphite moderator from the core (graphite-free core). In terms of the fuel cycle, two basic options have been investigated, the ^{233}U -started MSFR and the TRU-started MSFR.

Reactor description

Realistic drawings showing the main MSFR components and their arrangement in the vessel have been elaborated. Figure 1 displays a schematic drawing of a vertical section of the Th-U MSFR while Table 1 presents some characteristics of the reactor.

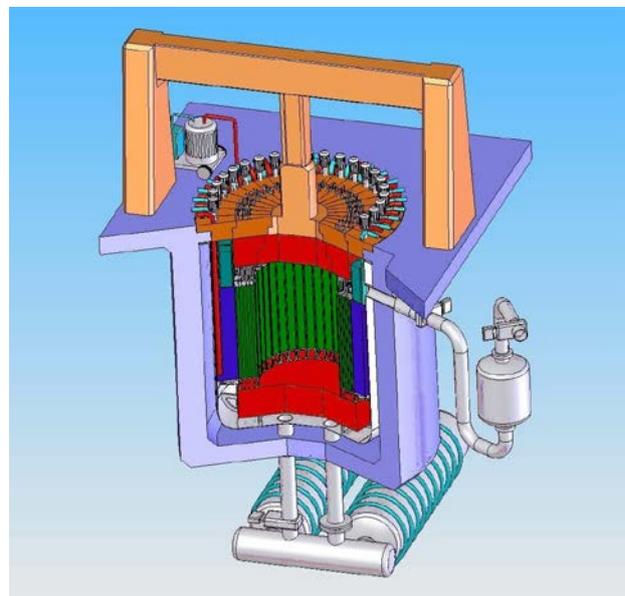
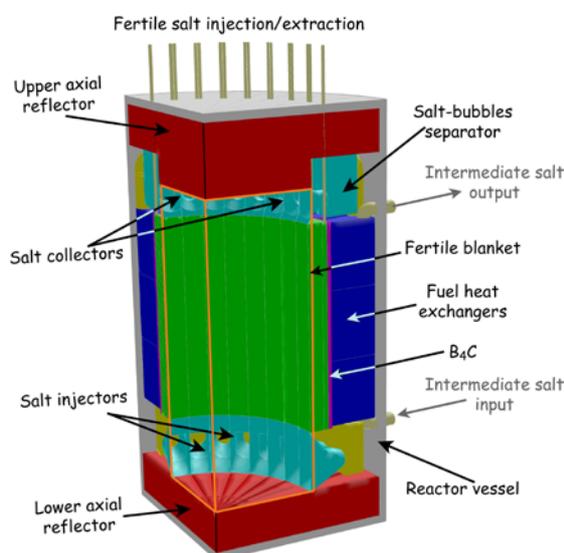


Figure 1: Th-U MSFR pre-conceptual design

The core is a single cylinder (diameter equal to height) where nuclear reactions take place within the flowing fuel salt. It is made of three volumes: the active core, the upper plenum and the lower plenum. The fuel salt is a binary salt, composed of LiF enriched in ^7Li (99.999%) and heavy nuclei (HN) amongst which the fissile element, ^{233}U or Pu. The $(\text{HN})\text{F}_4$ proportion is set at 22.5 mol% (eutectic point), corresponding to a melting temperature of

550°C. The choice of this fuel salt composition relies on many systematic studies (influence of chemical reprocessing on neutronic behavior, burning capabilities, deterministic safety level, deployment capabilities) [8-11]. This salt composition leads to a fast neutron spectrum in the core.

Table 1: Properties and characteristics of the MSFR

Thermal power (MWt)	3000				
Fuel molten salt composition (mol%)	LiF-ThF ₄ - ²³³ UF ₄ or LiF-ThF ₄ -(Pu-MA)F ₃ with LiF = 77.5 mol%				
Fertile blanket molten salt composition (mol%)	LiF-ThF ₄ (77.5-22.5)				
Melting point (°C)	550				
Operating temperature (°C)	700-800				
Initial inventory (kg)	²³³ U-started MSFR		TRU-started MSFR		
	Th	²³³ U	Th	Actinide	
	38300	5060	30600	Pu	11200
				Np	800
				Am	680
Cm				115	
Density (g/cm ³)	4.1				
Dilatation coefficient (/°C)	10 ⁻³				
Core dimensions (m)	Radius: 1.15 Height: 2.30				
Fuel salt volume (m ³)	18 9 out of the core 9 in the core				
Blanket salt volume (m ³)	8				
Thorium consumption (ton/year)	1.112				
²³³ U production (kg/year)	93 (²³³ U-started MSFR) 188 during 20 years then 93 (TRU-started MSFR)				
Breeding ratio (²³³ U-started MSFR)	1.085				

The outer core structures and heat exchangers are protected by thick reflectors designed to absorb more than 80% of the escaping neutron flux. These reflectors are themselves surrounded by a 10 cm thick neutronic protection of B₄C absorbing remaining neutrons. Axial reflectors are made of nickel-based alloys. The radial reflector consists in a fertile blanket (50 cm thick) filled with a fertile salt of LiF-ThF₄ with 22.5 mol% ²³²Th.

The level of deterministic safety reached by the concept is excellent since the feedback coefficients of the MSFR are negative in both ²³³U and TRU starting modes [8,11]. The total feedback coefficient is equal to -6 pcm/°C when the equilibrium state of the reactor has been reached and the density coefficient, which for MSRs can also be viewed as a void coefficient, is also largely negative at about -3 pcm/°C.

Fuel cycle aspects

The evolution of a typical fuel salt composition for a ²³³U-started MSFR and TRU-started MSFR in operation is shown in Figure 2. The inventories of U and Th in the two reactors

become equivalent after about 40 years in operation. At this time, more than 85% of the initial TRU inventory is burnt.

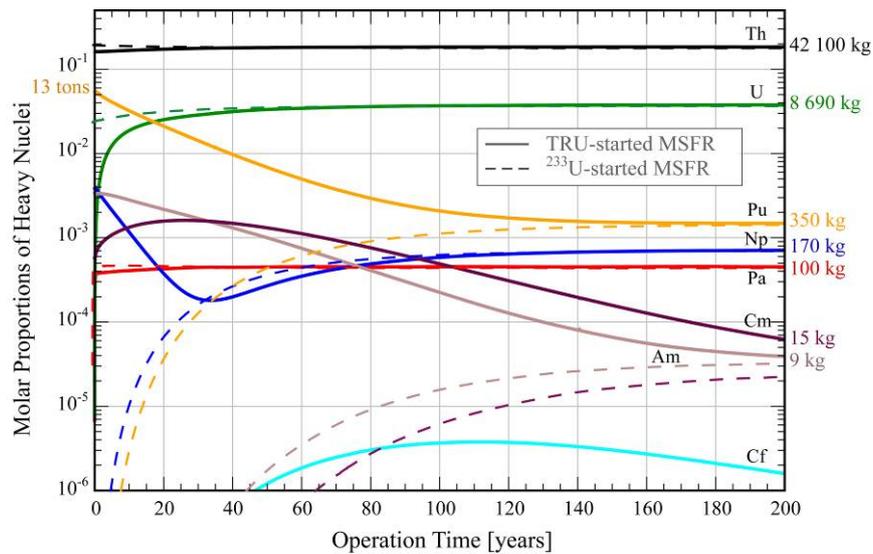


Figure 2: Evolution up to equilibrium of the heavy nuclei inventory for the ²³³U-started MSFR (dashed lines) and for the TRU-started MSFR (solid lines)

A good indicator of the deployment capability is the doubling time, defined by the operation time leading to the ²³³U inventory of a new reactor of the same type through breeding. For a ²³³U-MSFR, the annual ²³³U production is 120 kg which corresponds to 50 years doubling time per reactor [8]. Starting a MSFR from Generation II or III reactors spent fuel is more favourable and yields 35 years doubling time. Indeed, the presence of other fissile elements decreases the consumption of ²³³U and improves the deployment capability of the concept.

Studies of the Th-U MSFR also indicated that good breeding ratios could be obtained, but high power densities would be required to avoid excessive fissile inventories. Adequate power densities (see Table 1) appeared difficult to achieve without going to novel heat removal methods.

2.2 Salt selection for different applications

Reference MSR salt compositions

Selection of the salt composition strongly depends on the specific design application: liquid fuel (MSR burner or breeder), primary coolant (AHTR) or secondary coolant and heat transport fluid. Potential salt systems have been critically reviewed in the frame of the ALISIA project in the Euratom 6th FWP [13]. Reference compositions have been proposed or confirmed for MSR breeder and burner concepts (Table 2).

⁷LiF-ThF₄ (78:22 or even 71:29 in mol%) is the reference fuel solvent composition for thorium MSFR. The neutronic analysis of the MSFR has demonstrated the feasibility of the concept, but it must still be clarified whether the physico-chemical properties (melting temperatures, solubility for the actinide trifluorides, density, expansivity, viscosity, thermal conductivity, heat capacity) of this salt fuelled by significant amount of UF₄ (2-4% of the total heavy nuclei in the moderated and 12-18% in the fast systems) or AnF₃ (up to 25% of the total heavy nuclei in the fast concept) are consistent with safe operation of the reactor and fuel salt clean-up unit. To tune these properties, addition of other components is possible. The most obvious is BeF₂ but there is an incentive to keep the content of this material low (e.g. 71LiF-2BeF₂-27ThF₄ or 75LiF-5BeF₂-20ThF₄ in mol%) or even zero. Alternatives are NaF

and possibly CaF_2 . Therefore, the ${}^7\text{LiF-NaF-ThF}_4$ system must be further analyzed, whereas scoping studies of the ${}^7\text{LiF-CaF}_2\text{-ThF}_4$ system are required to assess the feasibility of this composition, including suitability for fuel salt processing.

Table 2: Fuel and coolant salts for different MSR applications

Reactor type	Neutron spectrum	Application	Carrier salt	Fuel system
MSR-Breeder	Thermal	Fuel	${}^7\text{LiF-BeF}_2$	${}^7\text{LiF-BeF}_2\text{-ThF}_4\text{-UF}_4$
	Fast	Fuel	${}^7\text{LiF-ThF}_4$	${}^7\text{LiF-ThF}_4\text{-UF}_4$
				${}^7\text{LiF-ThF}_4\text{-PuF}_3$
MSR-Breeder	Thermal/Fast	Secondary coolant	NaF-NaBF_4	
MSR-Burner	Fast	Fuel	LiF-NaF	$\text{LiF-(NaF)-AnF}_4\text{-AnF}_3$
			LiF-(NaF)-BeF_2	$\text{LiF-(NaF)-BeF}_2\text{-AnF}_4\text{-AnF}_3$
			LiF-NaF-ThF_4	

The molten salt actinide burner is a fast spectrum reactor too. The carrier salt for this application must have high enough solubility for the actinide trifluorides (Pu, Am, Cm etc.). Its solubility is maximum in pure LiF or NaF and decreases with addition of BeF_2 and ThF_4 . Decrease is more significant for BeF_2 addition, because PuF_3 is not soluble in pure liquid BeF_2 [24]. Adequate solubility can be achieved using ${}^7\text{LiF-(NaF)-BeF}_2$ with decreased Be content as solvent for the TRU from LWR spent fuel, and for minor actinide enriched fuels with ${}^7\text{LiF-NaF}$ or LiF-NaF-KF system. An interesting alternative is the use of plutonium and minor actinides as start-up for the thorium cycle in the MSR, leading to ${}^7\text{LiF-NaF-ThF}_4$ carrier salt.

In summary, it is clear that the ${}^7\text{LiF-(NaF)-AnF}_4\text{-AnF}_3$ salt (where An represent actinides) is the key system to be further investigated in parallel to the ${}^7\text{LiF-(NaF)-BeF}_2\text{-AnF}_4\text{-AnF}_3$ system. Optimisation of the fractions of the components is still needed with respect to the above-mentioned- physico-chemical properties, corrosion behaviour in the Ni-Mo alloys and fuel salt processing.

First priority needs in salt properties determination

In general, one can group the properties in (a) melting temperature, (b) actinide solubility, and (c) physico-chemical properties:

- (a) The phase transition behavior of most binary and ternary salts is reasonably well known, and reliable thermodynamic models have been developed to predict melting points of ternary and quaternary mixtures [25]. Exceptions are salts containing significant amount of TRUs. In this field, a significant effort is needed in the near future. The main needs are thus (i) measurements on plutonium and other actinide trifluorides (ii) verification measurements in ternary and quaternary systems to check and improve thermodynamic models.
- (b) Adequate solubility of fuel is a key issue for MSFR both burner and breeder concepts, but only a limited number of studies on PuF_3 solubility in solvent systems selected exist and none on the solubility of $\text{NpF}_4/\text{NpF}_3$, AmF_3 or CmF_3 . Solubility determinations in the systems with proper redox potential are therefore of prime importance. A prerequisite for such measurements is the availability of the pure actinide tri- and tetrafluorides in significant quantities, for which dedicated synthesis facilities are needed.

(c) Physico-chemical properties (density, viscosity, heat capacity, thermal conductivity) are poorly known for most of the salts that have been identified for the MSFR development. An exception is the ${}^7\text{LiF}\text{-BeF}_2\text{-ThF}_4\text{-UF}_4$ system that was extensively studied in the 1960s. Of these properties, the density (or molar volume) follows ideal behavior and can be easily obtained from the pure compounds. This is not true for the other properties, but there is generally a lack of data. Systematic experimental studies are needed here, especially for compositions with actinides. Especially thermal conductivity is poorly known, and also theoretical models are poorly developed. In this field, the benefits from atomistic calculations must be explored, in combination with local structure determination techniques (NMR, Raman, EXAFS) to create the fundamental basis for the understanding of the molten salt properties.

Complementary data on molten Li,Na,Be/F mixtures fuelled by PuF_3 have been measured in the ISTC-1606 project conducted in Russia [21]. Missing or uncertain data for molten salt mixtures containing increased (compared to prior studies) amounts of Th, U and TRU fluorides have been identified (phase behaviour, TRU solubility, thermal conductivity, expansivity). They will be acquired in the EU (JRC-ITU) and Russia (ISTC-3749 project).

2.3 Improved fuel salt clean-up scheme

The salt processing scheme relies on both on-line and batch processes in order to satisfy the constraints for a smooth reactor operation while minimizing losses to waste streams. ORNL experiments have provided some data mainly for the on-line gaseous fission product extraction process.

Acquisition of fundamental data for the separation processes is needed especially for the actinide-lanthanide separation. The extraction of lanthanides has to be done because of the low solubility of these elements and neutronic captures that decrease the reactivity balance.

The progress made in core design in the last two years has opened the door for the definition of an improved fuel salt reprocessing scheme (cf. section 2.1) with a realistic fuel clean-up rate (40 l/day) and minimized losses to wastes [8,11].

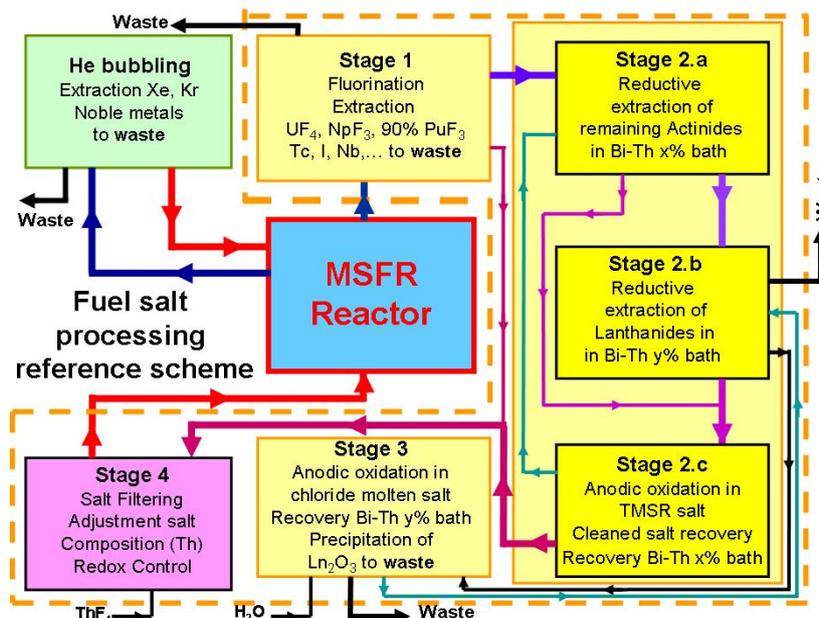


Figure 3: MSFR reference fuel salt processing flow sheet

The proposed reference processing scheme is shown in Figure 3. The first step (green box) involves an on-line gaseous extraction with helium bubbling to remove gaseous fission products, Xe and Kr, and a part of the noble metals. On the other hand, a batch fuel process separates the actinides which are returned to the reactor salt from the harmful fission products (mostly lanthanides). The fuel clean-up rate has been set at 40 liters per day, corresponding to the processing of 100 kg heavy nuclei per day. This value is almost two orders of magnitude less than the reference MSBR scheme.

The reference scheme depicted in Figure 3 involves 4 stages for the batch on-site fuel processing. The peculiarity of the concept appears in stages 2 and 3 by combining chemical and electrochemical methods for the extraction and the back extraction of actinides and lanthanides. This choice leads to fuel processing without effluent volume variation and the fuel processing balance is reduced to only one reaction: $2\text{LnF}_3 + 3\text{H}_2\text{O}(\text{g}) = \text{Ln}_2\text{O}_3 + 6\text{HF}(\text{g})$.

Critical steps of the new fuel clean-up scheme are addressed and will be experimentally assessed in new facilities (Figures 4 and 5). The design and construction of a molten salt loop to study both He bubbling efficiency and material corrosion attack has been initiated. An efficient technique for actinide/lanthanide separation is under qualification [16].

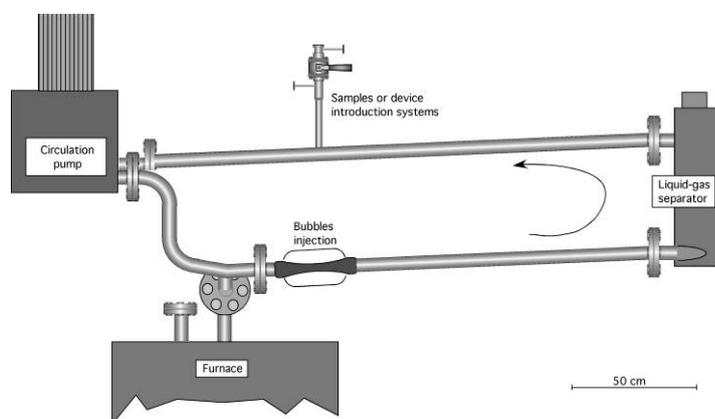


Figure 4: Simplified view of FFFER (*Forced Fluoride Flow for Experimental Research*) facility under construction. The LiF-KF-NaF loop is dedicated to bubbling studies (CNRS, France)



Figure 5: Glove box dedicated to ThF₄-based molten salts studies, particularly electrochemical steps of the reprocessing (CNRS, France)

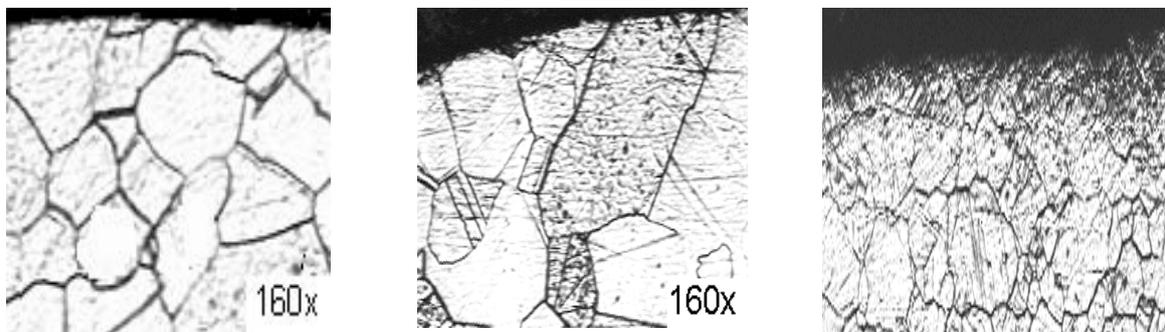
2.4 High performance materials

The success of a MSR is strongly dependent on the compatibility of the container materials with the molten salts used in primary and secondary circuits. The alloys must have satisfactory resistance to surface cracking when under stress in contact with the fuel salt in reactor environment, satisfactory corrosion resistance in fuel and coolant salts, and satisfactory mechanical properties over the long-term under reactor conditions. Design of practicable MSR system demands the selection of salt constituents such as LiF, NaF, BeF₂, UF₄, ThF₄, PuF₃ etc., that are not appreciably reduced by available structural metals and alloys whose components Fe, Ni and Cr can be in near equilibrium with the salt.

Early ORNL studies led to the development of a nickel-base alloy, Hastelloy N, for use with fluoride salts. The alloy contained 16% molybdenum for strengthening, and chromium sufficient to moderate oxidation rate in air and not enough to lead to high corrosion rates in salt. This alloy was the sole structural material used in the Li,Be,Zr,U/F MSRE and contributed significantly to the success of the experiment. However, two problems were noted with Hastelloy N, which needed further attention before more advanced reactors could be

built. First, it was found that Hastelloy N was embrittled by helium produced from ^{10}B and directly from nickel by a two-step reaction. This type of radiation embrittlement is common to most iron and nickel base alloys. Commercial heats of 2% Ti modified Hastelloy N with carbide forming elements were found to have minimum creep ductilities that were greater than 4% at 750°C after irradiation, which appear to be adequate. The second problem arose from the fission product tellurium diffusing a short distance into the metal along the grain boundaries and embrittling the boundaries. In material tests with Te, a modified Hastelloy N containing 2% Nb was found to be entirely free from cracks, but the test conditions were not sufficiently long or diversified to show that the alloy totally resists embrittlement.

In RRC-KI (Russia), the Ni-based alloy Hastelloy N modified (or HN80M) was also chosen as a reference. About 70 differently alloyed specimens of HN80M were tested. New findings shift the emphasis from alloy modified with titanium and rare earths to those modified with Nb, Al and Mn (Figure 6). The results of combined investigation [17,21] of mechanical, corrosion and radiation properties for various HN80M type alloys at temperatures up to 750°C permitted to suggest the Al-modified alloy named HN80MTY (or EK-50) as an optimum container material both for the Th-U MSFR breeder and MOSART burner systems. Nevertheless the weldability of the alloy deserves an improvement.



(a) HN80MTY with 1% Al (b) HN80M-VI with 1% Nb (c) XH80MT alloyed by Mn

Figure 6: Material specimens of three HN80M modified type alloys used in ISTC-1606 Na,Li,Be/F corrosion facilities with PuF_3 and Te additions. Specimens of these modified HN80M type alloys from corrosion loop hot leg exposed to Na,Li,Be/F (1200 hrs) at maximum temperatures 700°C showed maximum uniform corrosion rate $< 5 \mu\text{m/yr}$ [17,21].

Certainly, some alternative approaches are possible and could be of interest. Ni-W-Cr alloys have been recently proposed for their high potential to corrosion resistance for very high temperature operation ($> 750^\circ\text{C}$) [13]. Temperatures higher than 850°C would require the use of new solutions such as refractory alloys or graphite.

Further, alloy development should involve irradiation, generalized corrosion by fuel and coolant salt, tellurium exposure, mechanical property, and fabrication tests at MSFR operating parameters to finalize the composition for scale-up. Included in further evaluation should be also the assessment of (1) effects dealing with the use of new solvent Li,Na,Be/F, Li,Th/F or Li,Be,Th/F systems, (2) increased fuel salt outlet temperatures $> 750^\circ\text{C}$ and (3) lower salt redox potentials that must be maintained to avoid intergranular cracking for Ni-based alloys.

2.5 Criticality tests for the assessment of MSFR fuel and core behavior

The SPHINX (SPent Hot fuel Incinerator by Neutron fluX) project was originally defined as a suitable experimental basis at representative scale for the demonstration of MSR-burner feasibility [18]. It relies on the utilization of the zero power experimental reactor LR-0 being operated in the Nuclear Research Institute Řež (NRI), Czech Republic. This full scale

physical model of the PWR cores was modified in order to allow the measurement of all the neutronic characteristics of the MSR burner and/or breeder blanket, at first by room temperature and in future stage by conditions close to operational. (Figure 7).

Because two baseline concepts (MSFR, AHTR) are now considered in Generation IV, a corresponding broadening of the SPHINX project was discussed and formally adopted at the end of 2008. The LR-0 will thus be used for the validation of AHTR neutronics models (reactivity coefficients...) in the frame of collaboration between the Czech Republic (NRI) and USA (University of Berkeley).

Two versions of elementary blocks EROS 7 and EROS 8 (Figure 8), as simplified models of the AHTR core module, have been designed and manufactured. During December 2008, the critical tests of both those elementary blocks were performed. The simplified models are completely ready for complex testing of experimental and measuring methods for detailed neutron field distribution and principle neutronic characteristics prediction.



Figure 7: LR-0 zero power critical test facility

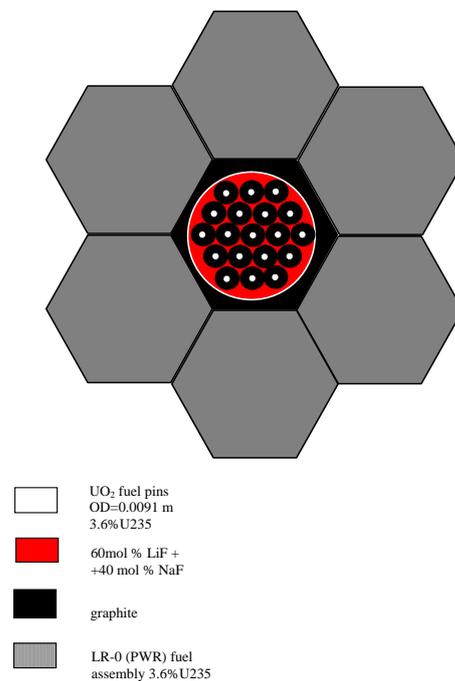


Figure 8: The EROS 8 scheme with simplified model of the elementary block of the AHTR core while inserted into the driving core of LR-0

2.6 MSR as an advanced option for effective incineration of radioactive waste

General description of fast spectrum burner concept

Recent European theoretical and experimental studies examined and demonstrated the feasibility of the Molten Salt Actinide Recycler & Transmuter (MOSART) system to reduce long-lived waste toxicity and to efficiently produce electricity in a closed fuel cycle. These activities were part of the ISTC-1606 project, the Euratom 6th FP ALISIA project and IAEA Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste” [19,22,23].

MOSART is a single-stream fast-spectrum MSR fueled with different compositions of plutonium and minor actinides from LWR spent nuclear fuel (SNF) without the use of

uranium or thorium. The start-up and feed material scenarios include plutonium and minor actinides from PWR enriched uranium (scenario 1) or mixed-oxide SNF (scenario 2). The fuel salt mixture is 15LiF-58NaF-27BeF₂ (mol%) with a melting point of 479°C or 73LiF-27BeF₂ mixture with a melting point of 530°C. The lithium is enriched to 99.99% ⁷Li. Depending upon the feed material, the salt at equilibrium contains 1.05 to 1.3 mol% of actinide and lanthanide trifluorides. Transient to equilibrium in the 2400-MWt MOSART core requires about 10 years. Masses of plutonium and minor actinides in the primary circuit at equilibrium for scenario 1 and scenario 2 are respectively about 7.3 and 9.3 t. The specific salt was chosen, in part, because of its high solubility (> 2 mol% at 600°C) for actinide and lanthanide trifluoride – a requirement for TRU burning.

The 2400-MW(t) MOSART system has a homogeneous core with intermediate-to-fast neutron spectrum. The core diameter is 3.4 m with a core height of 3.6 m. The core has 0.2-meter radial and axial graphite reflectors. The specific power is ~43 W/cm³, with an effective neutron flux of about 10¹⁵ n.cm⁻².s⁻¹. The salt inlet temperature is 600°C with an average outlet temperature of 720°C. The core inlet structure is designed to assure (1) that the maximum temperature of the solid reflectors is controlled to assure reasonable lifetimes and (2) no reverse or stagnate flow. The mechanical design of the balance of the nuclear system includes four pumps operating in parallel with an out-of-core circulation time of ~4 s. Heat is transferred from the primary salt to the power conversion system using a secondary NaF-NaBF₄ salt with a composition of 8–92 mol%.

The salt is sparged for removal of tritium, xenon, and krypton. The salt processing system removes the soluble fission products (rare-earth trifluorides) with an average residence time in the reactor of 300 effective full-power days. More rapid processing would reduce parasitic neutron losses but it would also increase the losses of actinides to the waste stream. Fuel salt clean up steps of Li,Be/F MOSART system are based on the method of reductive extraction in liquid bismuth. Reductive extraction of actinides from molten Li,Na,Be/F salt into liquid bismuth with their subsequent re-extraction into purified salt flow is the most acceptable method of actinide recycling. In order to purify molten Li,Na,Be/F salt from lanthanides, it will be necessary to use distillation and co-crystallization processes. Application of lanthanides co-crystallization with cerium trifluoride can sufficiently decrease salt solvent amount, which is to be processed by high-temperature distillation.

One of the attractive features of the MOSART system is the variety of fuel cycle feasible scenarios that can be implemented without its core modification. For example, the MOSART start-up and feed material scenarios can include compositions enriched by minor actinides (Am, Cm) and depleted by plutonium. In this case, for reasons of high solubility of actinides trifluorides in RRC-KI, the preferred solvent for MOSART concept fuelled by Am and Cm enriched compositions has been molten LiF-NaF-KF mixture with the lithium enriched to 99.995% in the ⁷Li isotope. The calculated maximal solubility of AnF₃ in the matrix of LiF-NaF-KF (43.9-14.2-41.9) at 600°C has been found as 19.3 mol% [13].

For the MOSART system, the option also exists to add a thorium-containing salt Li,Be,Th/F blanket to produce ²³³U fissile material to feed core in two-fluid MOSART design.

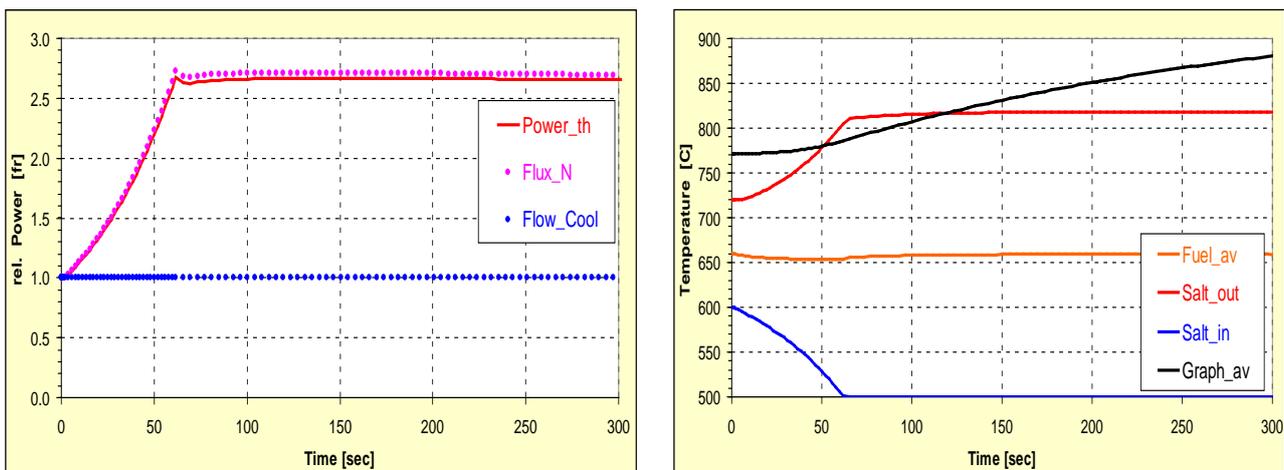
Analysis of burner concept in accidental conditions

In the frame of the IAEA Coordinated Research Project (CRP) on “Studies of Advanced Reactor Technology Options for Effective Incineration of Radioactive Waste”, different systems were considered. Among them, critical MSR fertile and fertile-free fuel concepts were investigated in their transmutation capabilities, their safety related parameters (reactivity

coefficients, effective delayed neutron fraction, etc.) and transient analyses for simulating relevant hypothetical accidents as well as fuel cycle reprocessing implications. Focus was placed on transient analyses of single stream Li,Na,Be/F MOSART fast spectrum system fuelled with compositions of plutonium plus minor actinide trifluorides (AnF_3) from UOX LWR spent fuel without U-Th support [19].

The transient studies have demonstrated that the MOSART design is an inherently stable reactor design on account of its large, negative fuel temperature coefficient ($-4.125 \text{ pcm}/^\circ\text{C}$) in combination with its negative graphite reflector reactivity coefficient ($-0.04 \text{ pcm}/^\circ\text{C}$). The MOSART reactor is expected not to be seriously challenged by the major, unprotected transients such as ULOF, ULOH, overcooling, or even UTOP.

In the case of the **overcooling transient** (Figure 9), the decreasing core inlet temperature leads to a decrease in the average fuel temperature whereas the fuel outlet temperature increases from 720°C to $\sim 830^\circ\text{C}$. Since the reactivity coefficient of the fuel is negative, a positive reactivity is inserted into the reactor leading to a power rise of a factor 2.7 about 60 sec into the transient. Due to the gradual increase in the temperature of the bulk graphite, additional negative reactivity is inserted into the core, leading to a levelling off of the power level at a factor of 2.7. Correspondingly, the core outlet temperature remains constant at $\sim 820^\circ\text{C}$. The mechanical integrity of the hot loop must be carefully monitored on account of potential long-term exposure of vessel and loop components exceeding temperatures of 800°C unless rectifying countermeasures are activated at some reasonable time into this transient. The reactor design is however inherently stable under this transient condition.



(a) Power transient

(b) Temperature transient

Figure 9: Simulation of Unprotected Over-Cooling Transient in MOSART system

While a substantial R&D effort would be required to commercialize MOSART, there are no killing unresolved issues in the needed technology. The major technical uncertainties in the conceptual design are in the area of tritium confinement, fuel salt processing and behavior of some fission products.

3 WHAT FUTURE FOR MSR R&D IN EUROPE

A European network on MSR R&D has been active from 2001 (MOST [20], 5th FWP) until 2008 (ALISIA [13], 6th FWP). It must be emphasized that Russia (RRC-KI) actively participated in MOST as an observer and in ALISIA as a full partner.

Figure 10 shows the continuity of the MSR network activity in the EU, with or without financial support by Euratom.

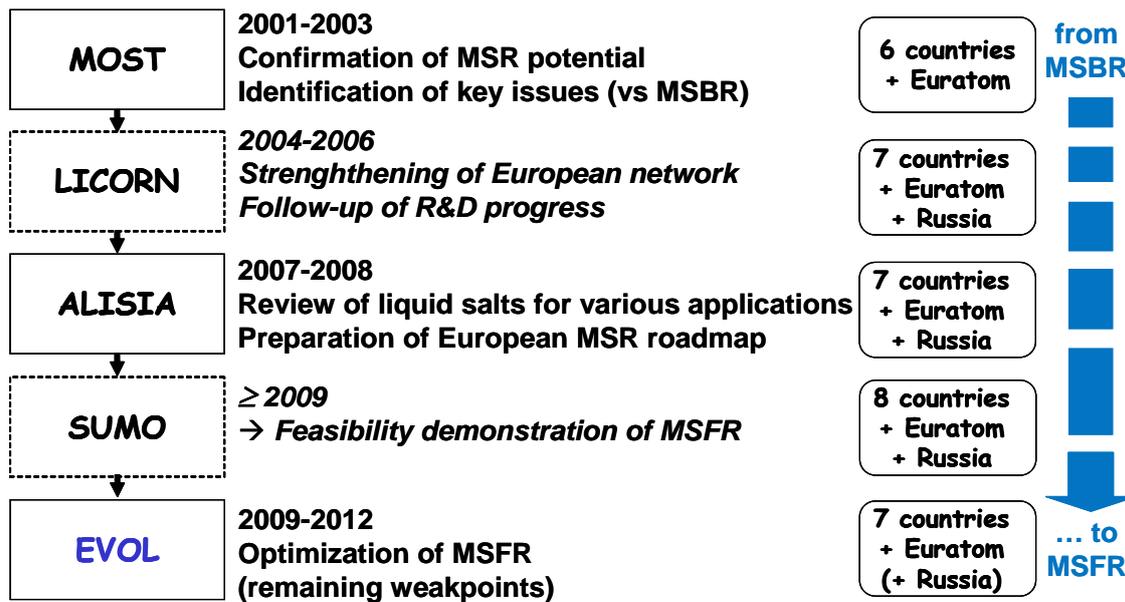


Figure 10: The MSR European R&D line

A MSR proposal (SUMO) was submitted in October 2008 in answer to the 2nd call of Euratom 7th FWP, but not accepted. In 2009, a new MSR proposal has been submitted to the 3rd call of the 7th Framework Program as a joint Euratom-Rosatom project. This EVOL project (acronym for Evaluation and Viability of Liquid Fuel Fast Reactor Systems) is under contract negotiations after a positive expert evaluation, in the frame of the Euratom collaboration with Rosatom, which is negotiating the complementary MARS (acronym for Minor Actinides Recycling in Molten Salt) project of Russian research organisations. Their common objective is to propose a conceptual design of Molten Salt Fast-neutron spectrum Reactor (MSFR) by 2012 as the best system configuration resulting from physical, chemical and material studies, and this for the reactor core, the reprocessing unit and the wastes conditioning. It is intended to deepen the demonstration that the MSFR system can satisfy the goals of Generation IV in terms of sustainability (Th breeder), non proliferation (integrated fuel cycle, multi-recycling of actinides), resources savings (closed Th/U fuel cycle, no uranium enrichment), safety (no reactivity reserve, strongly negative feedback coefficient) and waste management (actinide burner).

Since 2001, ISTC has provided another efficient way of collaboration between Russian research organizations, EU partners and non-EU partners. ISTC projects have offered the opportunity to benefit from the large expertise and facilities existing in Russia on molten salt technology (notably including pyrochemistry) and MSR concepts. The test facilities and experimental database created in the frame of ISTC-1606, from 2001 to 2007, are almost unique today for the assessment of liquid salt technologies [21]. Furthermore, two training sessions were held in the frame of ISTC-1606, in France (February 2006) and Russia (November 2006), on modern experimental and analytical methods for study of actinide-containing molten salts properties with participation of both EU and Russian scientists.

This effort has been continued in the frame of the ISTC-3749 project that formally started February 2009 for 3-4 years. ISTC-3749 receives official support by France (CEA, CNRS, EDF), Germany (FZK), Czech Republic (NRI), USA (ORNL), Canada (U. of Carleton), EC (JRC-ITU) and IAEA. The main mission of ISTC-3749 project is to test and select molten salts and metallic structural materials, which will operate successfully under the conditions of promising nuclear energy systems. Main emphasis will be focused on the study of the key properties for fluoride mixtures including as main constituents LiF, NaF, BeF₂, ThF₄, as well

as BaF₂ and CaF₂, as well as structural Ni-based materials compatibility with the fluoride mixtures selected.

Molten Salt Reactors were not explicitly considered in the Strategic Research Agenda (SRA) of the European Sustainable Nuclear Energy Technology Platform (SNE-TP). However, early 2009, it was suggested to append the SRA by mentioning that thorium fuelled molten salt reactors may represent an alternative to solid-fuelled fast spectrum reactors as sustainable nuclear energy systems in the long term, although they are still far from developments and demonstrations towards industrial applications. Consequently, a SRA annex dedicated to the MSR systems will be submitted in the very next future.

4 CONCLUSIONS

Since 2005, R&D on MSR is focused on fast spectrum concepts (MSFR) which have been recognized as long term alternatives to solid-fuelled fast neutron reactors with attractive features (very negative feedback coefficients, smaller fissile inventory, easy in-service inspection, simplified fuel cycle...). MSFR designs are available for breeding and for plutonium and minor actinides transmutation and recycling. They are robust reference configurations (with significant improvement compared to MSBR), allowing to concentrate on specific R&D issues. One of these, the definition of an appropriate safety approach, has been poorly addressed so far.

A European network on MSR has been active from 2001 (MOST project) until 2008 (ALISIA project) with financial support by Euratom. In 2009, a new MSR proposal (EVOL project) has been submitted to the 3rd call of the 7th Framework Program as a joint Euratom-Rosatom project. This project has received a positive expert evaluation.

In parallel, since 2001, ISTC has provided another efficient way of collaboration between Russian research organizations, EU partners and non-EU partners, thus giving the opportunity to benefit from the large expertise and facilities existing in Russia on liquid salt technology, including experiments with active materials. ISTC-1606 was terminated in 2007 but a new project, ISTC-3749, was formally started February 2009. ISTC and Euratom FWP projects on MSR have been conducted in a coordinated manner.

The EU (Euratom), France and USA participate in the Generation IV MSR Steering Committee. Although the EU and USA interests are focused on different baseline concepts (MSFR and AHTR, respectively), large commonalities in basic R&D areas (liquid salt technology, materials) exist and the Generation IV framework is useful to optimize the R&D effort.

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