

## **MOLTEN SALTS AS POSSIBLE FUEL FLUIDS FOR TRU FUELLED SYSTEMS: ISTC #1606 APPROACH**

**Victor Ignatiev<sup>1</sup>, Raul Zakirov<sup>1</sup>, Konstantin Grebenkine<sup>2</sup>**

<sup>1</sup>RRC-Kurchatov Institute, 123182, Moscow, Russian Federation

<sup>2</sup>VNIITF, 456770, Snezhinsk, Russian Federation

### **Abstract**

The principle attraction of the molten salt reactor (MSR) technology is the use of fuel/fertile material flexibility (easy of fuel preparation and processing) for gaining additional profits as compared with solid materials. This approach presents important departures from traditional philosophy, applied in current nuclear power plants, and to some extent contradicts the straightforward interpretation of the defence-in-depth principal. Nevertheless we understand there may be potential to use MSR technology to support back end fuel cycle technologies in future commercial environment.

The paper aims at reviewing results of the work performed in Russia, relevant to the problems of MSR technology development. Also this contribution aims at evaluation of remaining uncertainties for molten salt burner concept implementation. Fuel properties & behaviour, container materials, and clean-up of fuels with emphasis on experiments will be of priority. Recommendations are made regarding the types of experimental studies needed on a way to implement molten salt technology to the back-end of the fuel cycle.

To better understand the potential and limitations of the molten salts as a fuel for reactor of incinerator type, Russian Institutes have submitted to the ISTC the Task #1606 *Experimental Study of Molten Salt Technology for Safe and Low Waste Treatment of Plutonium and Minor Actinides in Accelerator Driven and Critical Systems*. The project goals, technical approach and expected specific results are discussed.

## 1. Introduction

Last years important R&D efforts were placed worldwide to find the ways to reduce the long term radionuclide inventory resulting from the nuclear power generation. This approach calls for the introduction of some innovative technologies to overcome some of technical hurdles presented by traditional ones. In our understanding, introduction of the innovative reactor concept to back end of fuel cycle pursues the following goals:

- Reduced actinides total losses to waste.
- Low plutonium and minor actinides total inventory in the nuclear fuel cycle.
- Minimal  $^{235}\text{U}$  support.
- Minimal neutron captures outside actinides.

Within this context the general matrix describing major innovative reactor & fuel cycle options could be written as follows:

Dedicated TRU burners; Once-through fuel cycle
TRU-free fuel cycle system; Recycling of actinides

The use of the molten salts as the fuel material has been proposed for many different reactor types and applications. Molten salt fuelled reactor (MSR) concepts have been prepared for fast breeders and thermal reactors more particularly in the USA, Russian Federation, France and Japan. Though, molten salt nuclear fuel concept has been proven by the successful operation experimented in MSRE experimental reactor at ORNL [1,2], this approach has not been implemented in industry. The fuel chosen for the operation of MSRE and for subsequent reactors of this type was a mixture of  ${}^7\text{LiF}-\text{BeF}_2-\text{ZrF}_4(\text{ThF}_4)-\text{UF}_4$ .

In Russia, the MSR programme started in the mid-seventies. RRC-Kurchatov Institute (KI) was a basic organisation which supervised a collaboration (an expert group composed by) of specialised institutions. A reduction of activity appeared after 1986 due to Chernobyl accident as well as a general stagnation of nuclear power and nuclear industry. Then at the end of the eighties there was an increase of conceptual studies as a result of the interest to the inherent safe reactors of a new generation. An extensive review of MSR technology developments at RRC-KI through 1989 is given in the publication in reference [3].

Today's interest in MSR technology stems mainly from an increased fuel residence time in the system, reduced actinides mass flow rate and relatively low waste stream when purifying and reconstituting the fuel by pyrochemical methods. We could then expect that in the future the MSR technology could find a role in symbiosis with standard reactors in the management of TRUs and thorium utilisation. New MSB concept (a reactor of incinerator type) requires a reconsideration of prior MSR concept, including optimisation of the neutron spectra and power density in the core, selection of the salt composition and new approaches towards its behaviour control and clean-up.

Below, we try to give our understanding of key issues, remaining uncertainties, methods available and improvements to integrate P&T recycle in molten salts. Also, planned tests for three years within the ISTC Task #1606 *Experimental Study of Molten Salt Technology for Safe and Low Waste Treatment of Plutonium and Minor Actinides in Accelerator Driven and Critical Systems* will be discussed.

## 2. The fuel salt for MSB concept

Many chemical compounds can be prepared from several “major constituents”. Most of these, however, can be eliminated after elementary consideration of the fuel requirements. Consideration of nuclear properties leads one to prefer as diluents the fluorides of Be, Bi, <sup>7</sup>Li, Pb, Zr, Na, and Ca, in that order. Simple consideration of the stability of these fluorides towards reduction by structural metals eliminates, however, the bismuth fluorides (see Table 1).

Several fluorides salts satisfy the characteristic properties of the thermal stability, radiation resistance, low vapour pressure and manageable melting point. To achieve lower melting temperatures, two or more salts are combined to produce still lower melting mixtures.

Table 1. Thermodynamic properties of fluorides

Compound (solid state)	$-\Delta G_{f,1000}^{\circ}$ , kcal/mole	$-\Delta G_{f,298}^{\circ}$ , kcal/mole	$E_{298}^0$ , V (Me F <sub>2</sub> )
LiF	125	140	6.06
CaF <sub>2</sub>	253	278	6.03
NaF	112	130	5.60
BeF <sub>2</sub>	208	231	5.00
ZrF <sub>4</sub>	376	432	4.70
PbF <sub>2</sub>	124	148	3.20
BiF <sub>3</sub>	159	200	2.85
NiF <sub>2</sub>	123	147	3.20
UF <sub>3</sub> (UF <sub>4</sub> )	300 (380)	330 (430)	4.75
PuF <sub>3</sub> (PuF <sub>4</sub> )	320	360 (400)	5.20
ThF <sub>4</sub>	428	465	5.05
AmF <sub>3</sub>	325	365	5.30
CeF <sub>3</sub>	345	386	5.58
LaF <sub>3</sub>	348	389	5.63

Note, that ZrF<sub>4</sub>, as a part of basic solvent, was found to distil from melt and condense on cooler surfaces in the containment system [1]. Control of the ZrF<sub>4</sub> mass transport was considered too difficult to ensure, so the 2LiF-BeF<sub>2</sub> solvent system was chosen as basic option at ORNL and later at RRC-KI. Also, in order to minimise problems associated with chemical treatment of the fuel salt and associated reduction of the basic components, the priorities should be given to the system with lowest possible ZrF<sub>4</sub> and PbF<sub>2</sub> content. Note, that use of Zr and Pb, instead e.g. the sodium in the basic solvent will lead to the increased generation of the long-lived activation products in the system [4].

Trivalent plutonium and minor actinides are stable species in the various molten fluoride salts [5,6]. Tetravalent plutonium could transiently exist if the salt redox potential was high enough. But for practical purposes (stability of potential container material) salt redox potential should be low enough and corresponds to the stability area of Pu(III). PuF<sub>3</sub> solubility is maximum in pure LiF or NaF and decreases with addition of BeF<sub>2</sub> and ThF<sub>4</sub>. Decrease is more for BeF<sub>2</sub> addition, because the PuF<sub>3</sub> is not soluble in pure BeF<sub>2</sub>. The solubility of PuF<sub>3</sub> in LiF-BeF<sub>2</sub> and NaF-BeF<sub>2</sub> solvents is temperature and composition dependent and PuF<sub>3</sub> solubility seems to be minimal in the “neutral” melts. For last ones, it

reaches about 0.5% mole at a temperature of 600°C and increases to about 2.0% mole at 800°C. For some excess free fluoride solvent system Li, Be, Th/F (72-16-12% mole), studies indicated that solubility of PuF<sub>3</sub> increased from 0.7-0.8% mole at 510°C to 2.7-2.9% mole at 700°C.

The other TRU species are known to dissolve in Li, Be, Th/F solvent, but no quantitative definition of their solubility behaviour exists. Such definition must of course be obtained, but the generally close similarity in behaviour of the AnF<sub>3</sub> makes it most unlikely that solubility of this individual species could be a problem. As expected substitution of a small quantity of AnF<sub>3</sub> scarcely changes the phase behaviour of the solvent system.

The trifluoride species of AnF<sub>3</sub> and the rare earth's are known to form solid solutions so, that in effect, all the LnF<sub>3</sub> and AnF<sub>3</sub> act essentially as a single element. It is possible, but highly unlikely, that the combination of all trifluorides, might exceed this combined solubility at the temperature below the reactor inlet temperature. A few experiments must be performed to check this slight possibility. The solubilities of the AnO<sub>2</sub> in Li, Be, Th/F are low and well understood. Plutonium as PuF<sub>3</sub> shows little tendency to precipitate as oxide even in the presence of excess BeO and ThO<sub>2</sub>. The solubility of the oxides of Np, Am, Cm has not been examined. Some attention to this problem will be required.

The molten fluoride chemistry (solubility, redox chemistry, chemical activity, etc.) for the 2LiF-BeF<sub>2</sub> system is well established and can be applied with great confidence, if PuF<sub>3</sub> fuels are to be used in the 2LiF-BeF<sub>2</sub> solvent. The properties of the most developed 2LiF-BeF<sub>2</sub> solvent however, are not all near the optimum for MSB application (very limited PuF<sub>3</sub> solubility, high enough melting point, etc.). Alternative solvent composition which will meet the lower liquidus temperature and increased PuF<sub>3</sub> solubility may be chosen e.g. from Na, Li, Be/F system. However, new less understood solvents system must be considered carefully before application in order to avoid severe problems with process operation. For MSB's needs next more important is consideration of PuF<sub>3</sub> chemical behaviour in these solvent systems: PuF<sub>3</sub> solubility in Li, Be/F, Na, Be/F and Li, Na, Be/F solvent, oxide tolerances of such mixtures and redox effects of the fission products.

The specific physical properties which were measured within the Russian MSR program include density, heat capacity, heat of fusion, viscosity, thermal conductivity and electrical conductivity. Particular emphasis has been placed for U/Th fuelled reactor cores. Properties are in each case adequate to the proposal service. Many of properties required for the MSB concept development are estimates rather than measured values. In some cases, especially for alternative solvents, careful re-measurement of some properties (e.g. thermal conductivity) is reasonable and desirable.

### **3. Fission products clean-up**

For molten salt fuels, fission products could be grouped in the three broad classes: 1) the soluble at salt redox potential fission products, 2) the noble metals and 3) the noble gases. The MSR would manage the noble gas removal by sparging with helium. As it was mentioned before, the problem here is to prevent the xenon from entering the porous graphite moderator. For the noble metals the situation is not so good and more experimental efforts is required in order to control their agglomeration, adhesion to surfaces and transport in purge gas.

In MSRs, from which xenon and krypton are effectively removed, the most important fission products poisons are among lanthanides, which are soluble in the fuel. Also, in combination of all trifluorides, AnF<sub>3</sub> solubility in the melt is decreased by lanthanides accumulation. Since plutonium

and minor actinides must be removed from the fuel solvent before rare earth's fission products the MSR must contain a system that provides for removal of TRUs from the fuel salt and their reintroduction to the fresh or purified solvent.

The available thermodynamic data (calculated or measured) for An/Ln trifluorides include considerable uncertainties and dispersion. For example, estimations on An-Ln separation ability have shown the most favourable situation for fluoride melts in comparison with chloride ones [7]. The similar conclusion was obtained in paper [8]. From our point of view, the comparison of thermodynamic potentials given in paper [9], for actinides and lanthanides both in chloride and fluoride systems for the benefit of chlorides is not correct enough, as the values of Gibbs energy for fluorides of lanthanides are underestimated. All mentioned above specify a necessity for further measurements of thermodynamic potentials of Ln and An fluorides by a uniform technique (e.g. EMF method with solid fluorine ion conducting diaphragms).

A number of pyrochemical processes (reductive extraction, electrochemical deposition, precipitation by oxidation and their combinations) for removing the soluble fission products from the fluoride based salt have been explored during the last years. Studies of the full scale MSB fuel salt chemical processing system are not as far advanced, but small scale experiments lead to optimism, that a practicable system can be developed.

### 3.1 Reductive extraction

Selective extraction from molten fluoride mixtures into liquid metals have been studied in details for essentially all pertinent elements [2]. This method of processing is optimal from technological perspective. It allows to realise on-line processing of fuel composition using simple design of extractors. The process capacity is rather high, and can be easily enlarged by intermixing. Obviously, use of the metal transfer system essentially simplify and accelerates the process, but several stages are required to reach desired recovery.

Rare earth removal unit based on Bi-Li reductive extraction flow-sheet developed in ORNL for LiF-BeF<sub>2</sub> solvent system could provide negligible losses of TRU (about 10<sup>-4</sup>) by use of several counter current stages. The separation factors  $\Theta$  of AnF<sub>n</sub> and LnF<sub>n</sub> are approximately equal to 10<sup>3</sup> for the Li, Be/F and Li, Na,K/F solvents. These values are very convenient for the lanthanide's separation by the reductive extraction. However, when the melt is complicated by the addition of large quantities of ThF<sub>4</sub> the situation becomes considerably less favourable. Under the conditions used U and Zr are the most easily reduced of the species shown above. U and Pa should be easily separated under the proper conditions, Pu-Pa separation is possible.

Note the following drawbacks of reductive extraction as applied to MSB:

- Less favourable scheme of An and Ln separation due to decreased difference in thermodynamic potentials of An and Ln (alloys with liquid metal).
- Materials compatibility pose substantial problems.
- Poor separation of thorium from rare earths for fluoride system; it can be made by use of LiCl.
- Changes in fuel composition because of the significant amount of lithium required; rare earths are removed in separate contactors in order to minimise the amount of Li required.

### 3.2 Precipitation of oxides

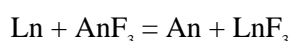
Although the metal-transfer process appears to give the best fuel salt purification in case of processing system with relatively short cycle (10-30 days), there are other possibilities for rare earth removal that are perhaps worth keeping in mind. If a bismuth containing system proves expensive or if unseen engineering difficulties (e.g. material required) develop, the other methods may be applicable, especially at longer processing cycle times. If treatment of a MSB fuel on a cycle-time of 100 days or more is practicable, such an oxide precipitation might be used for periodic removal of rare-earths. In experiments [10] a successful attempt was made to precipitate mixed uranium, plutonium, minor actinides and rare earths from LiF-NaF molten salt solution by fluor-oxide exchange with other oxides (for example CaO, Al<sub>2</sub>O<sub>3</sub>, etc.) at temperatures 700-800°C. The rare earths concentration in the molten salt solution was about 5-10 mole%. It was found the following order of precipitation in the system: U-Pu-Am-Ln-Ca. Essentially all U and TRU were recovered from the molten salt till to rest concentration  $5 \times 10^{-4}\%$ , when rare earths still concentrated in solution.

Main advantage of a method of processing the fuel composition by a sequential sedimentation of its components by oxides, non-soluble in fluoride melts, is the simplicity of the equipment for processing unit and more acceptable corrosion of structural materials in comparison with reductive extraction. Note, that recovery of oxide precipitates from a molten salt need further development.

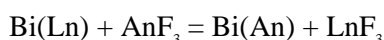
### 3.3 Electrochemistry

Main advantage of electrochemical methods is a possibility for the fuel clean up without introduction to the melt of additional reagents, which could change salt composition and influence its chemical behaviour and properties. Some processing flow-sheets with electrochemical deposition on solid electrodes, at first of all TRU elements, and after that the fission products, by the same way, or as alternative, for example, by oxides precipitation are possible. The reintroduction TRU in molten salt could be carried out electrochemically, or by simple dissolution, for example, of HF use.

In principle, An/Ln separation on solid electrodes could be more attractive in comparison with liquid electrodes. For the first case the overall reaction is the following:



and the latter:



In first case equilibrium in the separating process is carried out at the maximum value of a difference of thermodynamic potentials for actinides and lanthanides.

Regarding technological aspects, the electrochemical method has the important advantage consisting in the possibility of a continuous quantitative control of the process. During the process its rate and also depth of fuel processing are set and controlled by the value of a potential on electrodes of an electrolytic bath.

Note the following drawbacks of this method:

- Space limitation on processes area (bath electrodes), decreased capacity of units as contrasted to chemical processes in volume, especially for the end phase of the process.
- Necessity of dendrites control when use solid electrodes.

## 4. Container material studies

### 4.1 Fuel and coolant circuits

An important part of Russian MSR program dealt with the investigation of the container materials [3,11,12]. The development of domestic structural material for MSR was substantiated by available experience accumulated in ORNL MSR programme on nickel based alloys for  $UF_4$  containing salts [1,2]. In addition, compatibility tests were conducted to re-examine the possibility of using iron based alloys as container materials for MSRs. These alloys are more resistant to tellurium penetration and generate less helium under irradiation than Ni based alloys. However, their use would limit the redox potential of the salt and operating temperature in MSR.

Corrosion resistance of materials was studied in RRC KI by two methods. The first is the method of capsule static isothermal test of reference specimens in various molten salt mixtures. Also, flibe, flinak and sodium fluoroborate eutectic salts have been circulated for thousands of hours in natural and forced convection loops constructed of iron and nickel based alloys to obtain data on corrosion, mass transfer, and material compatibility. Not only normal, but also lowered and high oxidation conditions were present in the loops.

The alloy HN80MT was chosen as base. Its composition (in wt.%) is Ni(base), Cr(6.9), C(0.02), Ti(1.6), Mo(12.2), Nb(2.6). The development and optimisation of HN80MT alloy was envisaged to be performed in two directions:

- Improvement of the alloy resistance to a selective chromium corrosion.
- Increase of the alloy resistance to high temperature helium embrittlement and to tellurium intergranular cracking.

The results of combined investigation of mechanical, corrosion and radiation properties various alloys of HN80MT permitted to suggest the Ti and Al-modified alloy as an optimum container material for the MSR. This alloy named HN80MTY (or EK-50) has the following composition (in wt.%): Ni(base), Fe(1.5), Al(0.8–1.2), Ti(0.5–1), Mo(11–12), Cr(5–7), P(0.015), Mn(0.5), Si(<0.15), C(<0.04). The comparison of our corrosion data with those obtained at ORNL for Hastelloy-N indicates that corrosion resistance of HN80MTY is higher and it's maximum working temperature could be up to 750°C.

The high temperature, salt redox potential, radiation fluence and energy spectrum poses a serious challenge for any structural alloy in a MSB system. Data for  $UF_4$  containing salts provides a roadmap for establishing the corrosion properties for  $PuF_3$  containing salts, but additional corrosion testing under MSB conditions will be required to quantify corrosion rates of candidate container materials.

NaF, LiF,  $BeF_2$ ,  $CaF_2$ ,  $ThF_4$ ,  $PuF_3$ ,  $AmF_3$ ,  $CmF_3$  and  $NpF_3$  can not be oxidised in the system considered and can be reduced only to the metals, and then only by reducing agents very much stronger than the constituents of Hastelloy-N. Mixtures of these materials would not be expected to be corrosive. Recent capsule experiments in Cheljbinsk-70 [13] have demonstrated that  $PuF_3$  addition in LiF- $BeF_2$  solvent system did not make the corrosion situation worse on both nickel- and iron-base alloys. Same time the chemistry of  $PuF_3$  needs further testing in corrosion loops studies for redox control.

Also, included in further evaluation should be an assessment of lower salt redox potentials from the standpoint of allowing the use of stainless steels as structural materials and establishing the

potentials that must be maintained to avoid intergranular cracking for nickel-based alloys. Techniques developed under other reactor programs to improve the resistance of stainless steels to helium embrittlement should be extended to include nickel-base alloys.

#### **4.2 Fission product clean up unit**

The materials required for fission product clean up unit depend of course, upon the nature of the chosen process and upon the design of the equipment to implement the process. For MSB the key operation in the fuel treatment is removal of rare earth, alkali-metal, alkaline-earth fission products from the fuel solvent before its return, along with the TRUs, to the reactor. The crucial process in most of the processing vessels is that liquids be conducted to transfer selected materials from one stream to the another.

Such a fission product cleanup unit based on metal transfer process at least will present a variety of corrosive environments, including:

- Molten salts and molten alloys containing e.g. bismuth, lithium or other metals at 650°C.
- HF-H<sub>2</sub> mixtures and molten fluorides, along with bismuth in some cases, at 550-650°C.
- Interstitial impurities on the outside of the system at temperature to 650°C, particularly if graphite and refractory metals are used.

Certainly, the R&D on the materials for the fission product clean up unit for MSB is at very early stage. A layer of frozen salt will probably serve to protect surfaces that are worked under oxidising conditions, if the layer can be maintained in the complex equipment. RRC-KI preliminary tests at molten salt loops [11,12] showed that the thickness of the frozen film on the wall was predictable and adhered to the wall.

The only materials that are truly resistant to bismuth and molten salts are refractory metals (W, Mo, Ta) [1] and graphite (e.g. graphite with isotropic pyloric coating tested in RRC-KI [12] for both working fluids), neither of which is attractive for fabricating a large and complicated system. Development work to determine if iron base alloys can be protected with refractory metal coatings should probably be considered for higher priority. The approach taken to materials development could be to initially emphasise definition of the basic material capabilities with working fluids and interstitial impurities, and then to develop a knowledge of fabrication capabilities.

### **5. ISTC supported R&D**

To solve some of the mentioned in previous sections essential issues, Russian Institutes (RFNC-All-Russian Institute of Technical Physics, (Chelyabinsk-70), RRC-Kurchatov Institute (Moscow), Institute of Chemical Technology (Moscow) and Institute of High Temperature Electrochemistry (Ekaterinburg)) have submitted to the ISTC the Task #1606 *Experimental Study of Molten Salt Technology for Safe and Low Waste Treatment of Plutonium and Minor Actinides in Accelerator Driven and Critical Systems*.

The general purpose of the project is to perform an integral evaluation of potential of the selected molten salt fuel as applied to safe, low-waste and proliferation resistant treatment of radwaste and Pu



management. The major developments that will be pursued in the framework of the project are the following:

- Experimental study of behaviour and fundamental properties of prospective molten salt compositions.
- Experimental verification of candidate structural materials for fuel circuit.
- Reactor physics and fuel cycle consideration and recommendations on the key points of MSB concept development.

First two objectives are considered as the most crucial to the MSB further development. There are no doubts that the candidate constituents of the solvent system for MSB concept must be LiF, NaF, BeF<sub>2</sub>, maybe with minor additions of some other fluorides in order to provide required fuel properties. In contrast with well-established 2LiF-BeF<sub>2</sub> solvent system, for the other prospective compositions there is essential uncertainty in fundamental data, necessary to estimate their potential for MSB fuelled by plutonium and minor actinides.

Some of the salt components required for the experiments are commercially available and will be purchased, but some other salts have to be prepared basing on the technology developed by the participating institutes. Particularly, Pu, Np and Am trifluorides will be prepared at VNIITF site by pyrochemical method of HF gas treatment. Although starting materials of very high purity will be used in production of the fluorides, a careful analysis and, in some cases, a further purification will be needed before the salts usage in cell and loop systems. Two steps of purification would be required: one for the removal of oxides and sulfides and one for the removal of structural metal fluorides (NiF<sub>2</sub>, FeF<sub>2</sub>, CrF<sub>2</sub>, etc.)

The measurements of basic properties of prospective molten salt fuels will be carried out by means of techniques, which are mastered or will be developed in the participating institutions. First, it is planned to evaluate a phase diagram for the selected solvent and perform its experimental verification on several points. Also, it is planned to measure molten salt properties, such as actinides/lanthanides solubility, viscosity, thermal conductivity, standard potentials of actinides and rare-earth fluorides and equilibrium distribution coefficients of lanthanides in the two-phase system. Heat capacity, vapour pressure and density of the selected mixture can be calculated with a reasonable accuracy. Measurements of the properties will be carried out in a range of temperatures from 350°C to 800°C.

Two main types of candidate structural materials for the MSB primary circuit will be tested. First ones are samples of the Ni-based alloy Hastelloy-N, modified according to the ORNL and RRC-KI recommendations, the second will be prospective stainless steel samples. The corrosion tests will be carried out with selected fuel salt under conditions simulating the design ones with a working temperature up to 700-750°C, fuel salt heat up about 100°C, as well as with additions simulating the main fission products. The tests will be carried out at thermal convection loops with the samples exposure time in a flow till to 1 000 hours. It is planned to develop a technique for redox potential control and monitoring and apply it at the corrosion test loops. After the samples exposition, their detailed examinations will be carried out and corrosion rate will be measured. On the basis of these studies, the conclusion about the candidate structural materials compatibility with the selected salt will be made.

The experimental data will be fed into the conceptual design efforts. The objectives of the conceptual studies are to consider a candidate flow-sheet for MSB concept, that would be feasible.

Based on experimental data received for the project, the recommendation on choice of the fuel composition, the core configuration and the fuel cycle parameters, will be done.

The specific expected results of the Task #1606 will be:

1. Identification of the place of the molten salt technology in future fuel nuclear power system and suggestions on the strategy of its implementation for the fuel cycle harmonisation.
2. Recommendations on the choice of fuel compositions, core configuration and fuel cycle of the MSB fuelled by Pu and MA.
3. Measurements of the key properties of the selected molten salt fuel composition. A whole set of experimental facilities will be created. It allows continuing examinations of other candidate fuel compositions according to requests of the foreign collaborators.
4. Tests of candidate structural materials in corrosion loops and verification the materials compatibility with the selected fuel composition.
5. Fission product clean up feasibility study, including experimental studies of basic data for the pyrochemical processes as applied to MSB needs.

In March 2000 the Governing Board of the ISTC has approved Task #1606 for financial support. Funding source is EU. In the meantime the work plan for the Task #1606 has been developed. Numerous comments and suggestions of foreign collaborators have been taken into account.

## **6. Summary**

It is obvious from the discussion above that the use of molten fluorides as fuel and coolant for a reactor system of energy production and incinerator type faces a large number of formidable problems. Several of these have been solved, and some seem to be well on the way to be solved. But it is also clear that some still remain unsolved. The molten salts have many desirable properties for such applications, and it seems likely that – given sufficient development time and money – a successful TRU free or burner system could be developed. The properties of the MSR basic option salts however, are not all near the optimum for MSB applications. Only performing some additional experimental work give us the possibility to understand the practicability of operating an MSB.

It is still too early to guarantee that a MSR could truly operate as a reliable long term incinerator of TRUs and producer of energy in U-Pu or U-Th fuel cycle. It may even be uncertain whether such a system would serve a useful purpose if its successful development were assured. It is certain that effort to date has thrown light on e. g. much elegant high temperature non-aqueous chemistry and has shown how molten salts can operate under hard and strong conditions. Finally, it opens perspectives significantly different to the present reactor and fuel cycle technology.

### *Acknowledgements*

The authors acknowledge the ISTC for its financial support. They also thank the staff members for their friendly assistance, which was very much appreciated.

## REFERENCES

- [1] H.J. MacPherson, *Development of Materials and Systems for the Molten-salt Reactor Concept*, Reactor Technology, Vol. 15, No. 2, (1972).
- [2] J.R. Engel *et al.*, *Development Status and Potential Program for Development of Proliferation-resistant Molten Salt Reactors*, ORNL/TM-6415, March, (1979).
- [3] V.M. Novikov, V.V. Ignatiev, V.I. Fedulov, V.N. Cherednikov, *Molten Salt Nuclear Energy Systems – Perspectives and Problems*, Energoatomizdat, Moscow, (1990), (in Russian).
- [4] V.I. Oussanov *et al.*, *Long-lived Residual Activity Characteristics of Some Liquid Metal Coolants for Advanced Nuclear Energy Systems*, In Proc. of the Global'99 international conference, Jackson Hole, USA, (1999).
- [5] C.J. Barton, J.D. Redman, R.A. Strelow, *J. Inorg. Chem.*, 1961, Vol. 20, No. 1-2, p. 45.
- [6] V.F. Afonichkin *et al.*, *Interaction of Actinide and Rare-earth Element Fluorides with Molten Fluoride Salts and Possibility of their Separation for ADTT Fuel Reprocessing*, In Proc. of the second international conference on ADTTA, Kalmar, Sweden, 1996, June 3, pp. 1144-1155.
- [7] R.Y. Zakirov, V.N. Prusakov, *Role of Electrochemistry for Fuel Processing in Molten Salt Reactors*, Pre-print IAE-6061/13, RRC-KI, Moscow, (1998).
- [8] F. Lemort, X. Deschanel, R. Boen, *Application of Pyrochemistry to Nuclear Waste Processing*, In Proc. of the Global'99, Jackson Hole, USA, (1999).
- [9] C. Pernel *et al.*, *Partitioning of Americium Metal from Rare Earth Fission Products by Electrowinning*, In Proc. of the Global'99, Jackson Hole, USA, (1999).
- [10] V.F. Gorbunov *et al.*, *Experimental Studies on Interaction of Plutonium, Uranium and Rare Earth Fluorides with Some Metal Oxides in Molten Fluoride Mixtures*, Radiochimija, 1976, Vol. 17, pp. 109-114, (in Russian).
- [11] V.V. Ignatiev, V.M. Novikov, A.I. Surenkov, V.I. Fedulov, *The State of the Problem on Materials as Applied to Molten-salt Reactor: Problems and Ways of Solution*, Pre-print IAE-5678/11, Moscow, (1993).
- [12] V.V. Ignatiev, V.M. Novikov, A.I. Surenkov, *Molten Salt Test Loops (In and Out Reactor Experimental Studies)*, Pre-print IAE-5307/4, Moscow, (1991).
- [13] V.V. Ignatiev, K.F. Grebenkine, R.Y. Zakirov, *Experimental Study of Molten Salt Reactor Technology for Safe, Low-waste and Proliferation Resistant Treatment of Radioactive Waste and Plutonium in Accelerator-driven and Critical Systems*, In Proc. of the Global'99 international conference, Jackson Hole, USA, (1999).

