

FEASIBILITY OF MOLTEN SALT FAST REACTOR FOR EMERGING NATIONAL TASKS

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A safe, economic and proliferation risk free, robust stand-alone nuclear fission energy supported by solely depleted uranium for indefinite period is the perpetual global task. Meanwhile, producing no other bulky waste but the compact highly radioactive waste of which potential radio-toxicity dissipating to low as natural radioactive ore around 400 years, is our emerging national task. The feasibility of the system directing to these tasks are discussed supported by preliminary work. Nevertheless we specify the technological challenges to be addressed in application of the technology particularly associated with maintaining high solubility of fertile fluoride in specific carrier salts characterized by a low melting point. A more favorable system using a non-lithium fluoride solvent to mitigate technological and economical challenge is proposed.

I. INTRODUCTION

Japan selected the closed fuel cycle at the early stage of nuclear power usage, eventually using the LMFBR due to poor reserves of domestic natural resources. The public believe that the nuclear energy is a sustainable alternative, provided that people and the environment will be kept safe from suspected threats of radioactivity, released accidentally from nuclear installations and/or consistently from nuclear wastes. There are 44.4 GWe of potentially available power generation capacity, but no power station could have obtained the safety permit to re-start by the present. This situation forced Japan to cause tremendous international trade deficit and escalating electricity cost for the imported fossil fuel consumption.

The geological disposal of the High Level Solid Waste (HLSW) derived from the commercial reprocessing has been legitimated without the retrievability clauses in Japan since 2000 under the "Designated Radioactive Waste Final Disposal Act." It was presumed that a reverse bid system would have worked effectively, however nothing has happened since

then. The public is being convinced ethically, namely it is in return for cheap energy that the public have enjoyed.

Even though it was not stipulated by the above act, there is a big project to develop the Partitioning & Transmutation (P&T) technology, to partition Minor Actinides (MA) from the aqueous raffinate of the LWR spent fuel reprocessing, then to eliminate MA by a sophisticated accelerator driven technology. This extraordinary effort is discreetly associated with pure plutonium production process, and ought to be indispensably required unless there is no alternative as the reliable measure to secure national energy resources in future, rather than the originally stipulated LMFBR. It sounds to us nothing but excuses for an inappropriate choice made well prior to full deployment of LWR.

Nevertheless this situation may unforeseeably increase constitutional vulnerability and economical burden to secure the future energy in Japan.

Molten salt fast reactors are under consideration for the following national tasks to make 1; plutonium stockpile¹ eliminated, 2; national energy resources secure, 3; LWR spent fuel manageable, 4; HLSW acceptable to be disposed by reducing radio-toxic source term, 5; all measures be safe, simple, robust, stand-alone, viable and economical.

II. PRELIMINARY WORK

II.A. Evaluation of Tri-fluoride Solubility in the FLINAK

There have been several attempts using the Molten Salt Reactor (MSR) technology taking advantage of its unique feature not only of intrinsic safety but also of ability to accept extremely versatile forms of fuel composition^{2,3}. Thus, we are initially interested in an application of this technology as a burner of Trans Uranium (TRU) elements separated altogether from the LWR spent fuel.

It was inspired by the Bowman's reactor⁴ in order to burn domestic surplus plutonium along with MA, with integrated chemical processing but without an accelerator, and tried to make its critical version⁵ as suggested by Prusakov⁶, however a limited solubility of tri-fluorides in the LiF-BeF₂ series of solvents⁷ was recognized as the critical challenge to be overcome.

The LiF-NaK-KF (46.5-11.5-42 mole%) eutectic mixture (called "FLINAK") has been appraised as mainly heat transfer media owing to its low melting point and excellent hydro-thermal characters. However, it has not attracted interest as a carrier salt of the molten salt fuels particularly for thermal neutron applications, exclusively due to its high neutron absorption cross-section.

During reevaluation of the MSR technology as a candidate of Gen-IV reactors by the Molten Salt reactor Technology (MOST) project⁸, it was revealed that the Molten Salt Breeder Reactor (MSBR) concept could have been somewhat reactively instable⁹, and removal of graphite moderators from the core was identified as an effective way to give a higher negative reactivity coefficient of temperature as well as a higher breeding capability^{10, 11}. This fact certainly stimulates and accelerates our interest to use an un-moderated MSR as burner application to eliminate TRU to cope with the post Fukushima situations in Japan¹²⁻¹⁵.

The knowledge of FLINAK as a fuel carrier salt was once limited in the Institute for Trans-Uranium Elements of the EC. They initially thermo-dynamically estimated that the LiF-NaF-BeF₂-1.3 mole% PuF₃ as the designated fuel for the Molten Salt Actinide Recycler & Transmuter (MOSART) had a 50°C higher eutectic point than that of pure solvent salt¹⁶. Meanwhile they also suggested that FLINAK favorably produced a eutectic mixture with 1.3 mole% of PuF₃ at 450°C¹⁷, which could make 100°C lower than that of the original fuel salt for the MOSART³.

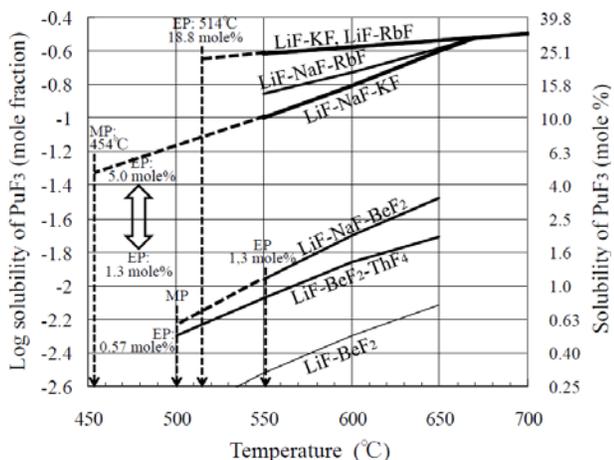


Fig. 1. Temperature Dependence of PuF₃ Solubility in Alkaline Fluoride and Reference Eutectic Mixtures.

Although the relevant phase-diagram has not been presented, it was predicted that 19.3 mole% of PuF₃ would be dissolved in FLINAK at 600°C¹⁶, which was supposed as the core inlet temperature. Their results are summarized in Fig. 1 with relevant solubility data.

We should respect the fact in which the standard design principle assumes that the fissile material never solidify prior to the rest of components of fuel salt at the eutectic point. That is 0.57 mole% of PuF₃ at 500°C in the LiF-BeF₂-ThF₄-PuF₃ fuel for the Pu start-up MSBR¹⁸, and 1.3 mole% at 550°C in the LiF-NaF-BeF₂-PuF₃ for the MOSART start-up as well as the make-up³.

II.B. Evaluation of Density for Alkaline Fluoride Salt Mixtures with UF₄

Density of the fuel salt is indispensable information to perform reactor physics calculations. There were not so many available density data of alkaline metal fluorides containing UF₄ as shown in Fig.2^{19, 20}. However, it seems that all systems are expressed by an identical second order function for very wide concentration range of UF₄, i.e. 0 to 45 mole%. The averaged temperature coefficients were reported as 0.0008/°C in the range of 0~4 mole%, however as 0.0011/°C in the range higher than 22 mole%^{19, 20}.

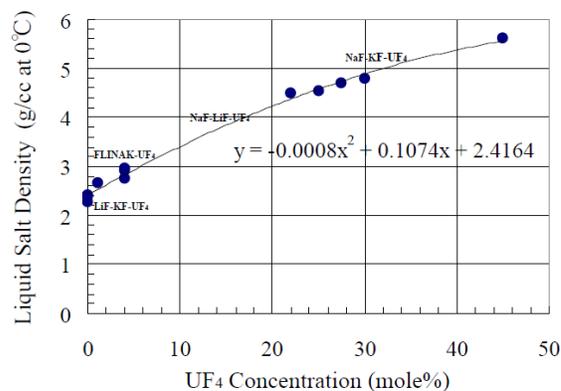


Fig. 2. Density of Alkaline Fluoride Containing UF₄

Based upon the density data for pure UF₄, UF₃, PuF₄ and PuF₃, i.e. 6.72, 8.97, 7.0 and 9.32 g/cm³ respectively²¹, it is observed that the density of tri-fluorides is 4/3 times higher than that of tetra-fluorides, and uranium fluorides are 0.96 times lower than that of plutonium fluorides. Thus it is hypothetically assumed that PuF₃ can be substituted by 1.389 molecules of UF₄ in the sense of density effect and UF₃ by 1.33 molecules of UF₄.

This procedure may give the density of 4.23 g/cm³ for FLINAK-UF₄-PuF₃ (74-23-5 mole%), that of 4.61 g/cm³ for NaF-KF-UF₄-PuF₃ (42-23.5-27.5-7.0 mole%), and that of 4.65 g/cm³ for NaF-KF-UF₄-UF₃-PuF₃ (42-23.5-25-2.5-7.0 mole%) all at 627°C.

II.C. Chemical Engineering Evaluation of the Molten-Salt TRU Burning System¹⁴

This work intended to propose a suite of chemical processes to be applied to all phases covering from the head-end through the back-end of TRU burning MSR, in order to specifically apply to a 3.93 GWt ABWR, but unique with a full MOX core intending to reduce Pu stockpile¹. The spent MOX fuel has been supposed to be the first candidate for geological disposal, due to technological incapability of the reprocessing plant for the MOX fuel.

There have been several attempts to burn TRU nuclides based on fluoride molten-salt reactors, however some chemical issues associated with head-end process as well as with on-line process, such as the electrochemical winning/dissolution and reductive metal extraction, etc., have not been adequately solved, mainly because of sophisticated nature of such chemical processes.

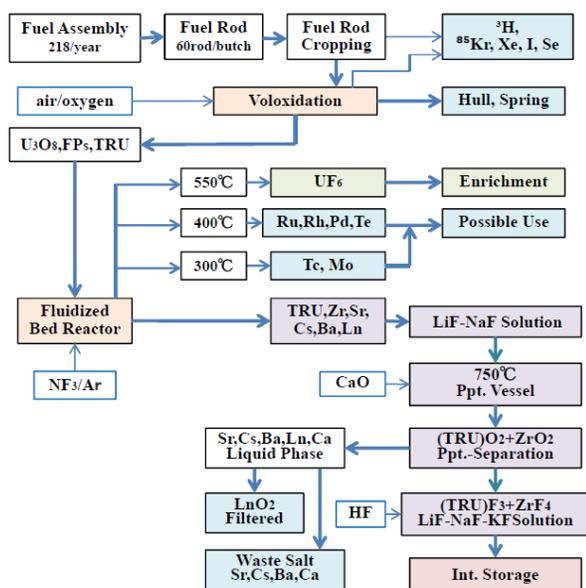


Fig. 3. Flowsheet of the “Fluor-Ox” Process Applicable to the Fuel Processing of the ABWR with a MOX Core.

It was logical to constitute the chemical process using the fluoride volatility process which originally converted all components into volatile fluorides by using a fluorine flame reactor, and then separate them into fractions according to their properties²².

However, we were rather interested in the recently developed process using NF_3 as a thermally-sensitive reagent; it would react with different compounds at different temperatures²³. For example, NF_3 reacts with Tc and Mo oxide near 300°C, Ru and Rh near 400°C, while U oxides required near 500°C to form a volatile fluoride.

This process eventually yields the non-volatile fraction containing all TRU fluorides²³. Then we intended to apply the oxide precipitation process, suggested by Ignatiev²⁴, to separate TRU elements from other residual materials of fluorination process effectively.

The oxide precipitation process for tri-fluorides separation after oxidized to tetra-fluorides had been evaluated in ORNL before the reductive metal extraction process had been developed²⁵, however the suggested process is based upon a successful experimental attempt, which was made to precipitate mixed solutes from the LiF-NaF molten salt solution by fluor-oxide exchange with other oxides (for example CaO , Al_2O_3 , etc.) at 700-800°C. 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 5×10^{-4} mole%, while rare earths still in solution²⁶. This represents that the TRU leak rate is 100 ppm (0.01%) from 5 mole%-TRU solution.

A suite of processes arbitrary named as “Fluor-Ox” process, and is shown as the flowsheet specifically for the ABWR spent fuel processing, however reasonably applicable to the ordinal LWR in Fig.3 and to the on-line processing of TRU burner reactor in the Fig. 4¹⁴.

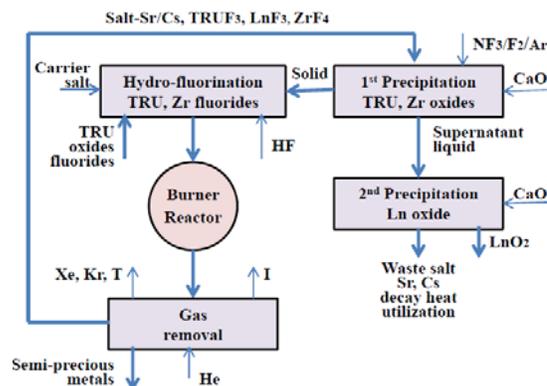


Fig. 4. Flowsheet of the “Fluor-Ox” Process Applicable to the on-line processing of the Molten Salt TRU Burning Reactor

This configuration of process is attractive due to, the ultimate simplicity in chemistry as well as in technology, least process waste stream, innate nature, gentle chemical environment to material, easy to control, specifically applicable to highly concentrated fuel, and may viable provided that the TRU product does not require strict purity and the solvent salt does not contain materials affected by oxidation (such as BeF_2 or ZrF_4). Fission product Zr is assumed to be always accompanied with TRU all through phases, and produces TRU-Zr oxide solid solution as the final residue. CaO (or Al_2O_3) accompanies with the salt seeking fission products as a major component of the HLSW.

We created the new criterion to judge the acceptable radio-toxic source term of the waste by TRU content in the solid fission product to make the radio-toxicity be adequately low after 400 years cooling, as sufficiently less than 0.15% from the current assessment of required capability of the partitioning process.

TRU content of a typical solid fission product from the LWR spent fuel can be decreased to 0.15% by the Fluor-Ox process with 0.02% of the leak rate, comparing with 0.5% by the PUREX reprocessing and with typically 0.1% by the advanced high-temperature-dry processes³.

It was found that in our TRU burning scheme assuming the annual burning rate of 5% to the total inventory, the average interval of chemical processing should be no less than 3 years, however practically about 5 years, using the Fluor-Ox process.

Reducing the potential radio-toxicity should be more critical target than simply reducing the TRU inventory in Japan, though this might not be so serious for some countries, such as France, which has already established the public consensus with the geological disposal of HLSW including 600 tons of MA by the end of century².

II.D. Reviewing the Challenges for Realization of the Molten-Salt TRU Burning System

II.D.1. Characteristic Arrangement for the un-moderated MSR.

The authors will have never dared to realize molten-salt fast reactors for burning TRU, unless we could have seen a tank-within-tank layout proposed by Forsberg²⁷ and reproduced in Fig. 5, to ensure characteristic safety of the un-moderated MSR based on the technology for the Fluoride High-temperature Reactor (FHR).

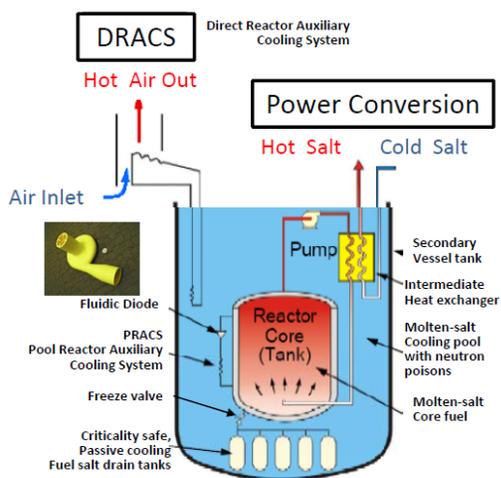


Fig. 5. The Tank-within-Tank Arrangement of Un-Moderated MSR

A unique criticality safety challenge associated with un-moderated MSR is that criticality can occur if the fissile materials leak from the system and come near neutron moderators, such as concrete. This has to exclude the “Catch Pan” arrangement to transfer gravitationally the spilled fuel material into the drain tank, which has been traditionally adapted by graphite moderated MSR.

The combination of the Direct REActor Cooling System (DRACS), the Pool Reactor Auxiliary Cooling System (PRACS) and the buffer-salt pool which includes drain tanks in bottom and is located in the underground silo can accommodate the decay heat removal and criticality issues under the design-basis as well as the beyond-design-basis accident, even including the outer vessel failure.

II.D.2. Material Compatibility in the Fuel Salt

The corrosion control of the Hastelloy-N in fluoride salt including Te effect was established by keeping U(IV)/U(III) ratio at maximum 70 but not excessively less than that to avoid metallic uranium deposition on the graphite, during the post MSRE study²⁸. This practice is understood to be applied to only in the neutral fuel salt, such as LiF-BeF₂. It was also suggested that the strong basic nature of alkaline metal salts such as FLINAK should be applied under more reduced conditions than that of the neutral salts, due to causing different oxidized chemical species of Cr (Cr⁺⁺⁺ instead of Cr⁺⁺)²⁹.

Recently it was informed by the study³⁰ with molten LiF-BeF₂-ThF₄ (75-5-20 mole%) salt mixture fuelled by 2 mole% of UF₄ and containing additives of Cr₃Te₄, included 250 hrs tests with exposure of nickel-based alloys specimens at temperatures from 700 to 750°C and under mechanical loading, that there were no traces of tellurium inter-granular cracking on specimens in the fuel salt with [U(IV)]/[U(III)] ratio from 0.7 to 20 and no nickel-uranium inter-metallic film on the specimens with fuel salts characterized by the ratio larger than 3, as shown by the acceptable Red-Ox voltage range in Fig. 8.

It was traditionally understood by Thoma³¹ that during the UF₄ fissioning, the net oxidation state of fission products was less than four and yet four fluorine atoms were released, leaving a somewhat oxidizing event per fission, meanwhile, in the case of PuF₃, it was expected to produce a slightly reducing event per fission.

Nevertheless it is suggested that the fission of PuF₃ could produce 0.53 moles of excess free fluorine per a mole of fission in the fluoride salt according to the LMFBR fission product yield data³². In a system to burn 0.39 tons-TRU/GWt-y will produce 861 moles of free fluorine a year. In addition to that, transmutation of LiF in neutron fluxes was expected to produce substantially additional free fluorine.

From the chemical engineering point of view, the last but an inevitable challenge to overcome was to specify the proper Red-Ox buffer material in the system containing no uranium at all, such as a TRU burner. It was suggested that Np(IV)/(III) as a major component of the TRU might satisfy the need in this respect, as shown in Fig. 6²⁹, however it has not been demonstrated yet.

Nevertheless when the process transmutes U tetra-fluoride to TRU tri-fluoride, 0.25 moles of excess free fluorine per fission should be taken into account. One should need at least 300 kg of U(III) a year as the Red-Ox buffer in a system to burn 0.39 tons-TRU/GWt-y.

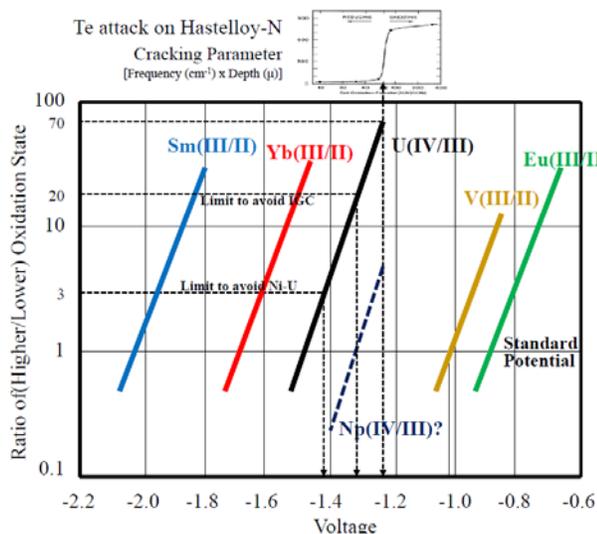


Fig. 6. Dependence of the Red-Ox Potential on the Oxidation State Ratio of Buffer Couple in Fluoride Molten-Salt Fuel.

II.D.3. Developing Technologies

The development of the FHR is under way taking advantage of inherited technologies developed in the ORNL, as a university integrated research project in the US³³. It is noteworthy that this project intends to develop various innovative but originally had been suggested during the MSR project such as: Salt compatible canned mortars; Nickel lined high temperature allowable construction materials; Mechanical check valves as well as throttle valves with salt compatible; Various novel sensors and instrumentations, and so on. In order to realize a molten-salt TRU burner reactor according not to those in 1970's but to the modern criteria, it is strongly perceived that the presently proposed system should also observe the FHR's practice³³ as far as possible.

It has been envisaged by the current ORNL that the coupling steam cycle to MSR was complicated because of the need to avoid freezing of the salt, diffusion of tritium through hot heat exchangers from the reactor core into the

steam, and other constraints. The double walled heat exchanger associated with the closed gaseous Brayton power cycle is currently proposed to the FHR to mitigate characteristic tritium diffusion issue due to the use of Li containing salt coolant by the capability to collect tritium not as a hydride but to trap as an elemental form³³.

III. MOLTEN SALT FAST BREEDER REACTOR (MSFBR)

Many attempts have been made to realize an effective breeder reactor based on the U/Pu cycle using liquid fuels specifically using molten-salt fuels, such endeavors were limited in the chloride salts, because of non-elastic scattering of high energy neutron by ¹⁹F atoms¹⁶. The mixture of alkaline chloride(s) and highly concentrated actinide chlorides can produce the sufficiently hard neutron spectrum and give us efficient breeding capability on the U/Pu cycle; however unavailability of the conventional structural materials to endure high temperature chloride and safety issue of soluble long life activity of ³⁶Cl, were perceived as serious challenges^{34,35}.

Recently confirmed high solubility data of various fluorides of actinides as well as lanthanides specifically in FLINAK³⁶⁻³⁸, reportedly could allow utilizing the high enough energy neutrons for the U/Pu breeding cycle^{39,40}. These findings stimulated our new approach to cope with the contemporary as well as the perpetual national tasks.

III.A. Reactor Physics Evaluation of the MSFBR

We made reactor physics calculations to verify the feasibility of the MSFBR using FLINAK with breeding capability (i.e.; can keep criticality by feeding ²³⁸U only) without a blanket, taking originally proposed configurations of the reactor and the definition of the terms⁴⁰, such as the power out put, the effective density of fuel salt and the temperature coefficient of reactivity, as the reference, but other factors, such as the actinides isotopic composition, neutron flux buckling, the fuel temperature (i.e., 627°C), the fuel salt density (ref. **II B** of this paper) and the salt clean-up and make-up condition (i.e., annual clean-up and every 25 days make-up), etc., were discretely specified to give verified number of heavy element masses and concentrations in FLINAK carrier salt to give designated criticality (i.e., $k_{eff} = 1.007$) from the start up to the equilibrium state (40 years).

It was found that the MSFBR was reactor-physically feasible provided that 2.71 tons of Pu were supplemented as a pseudo-exponential decay pattern after the startup, presumably to compensate the initial isotopical depletion of fissile plutonium as shown in the second column of Table I, although they have not yet be optimized.

The above calculation assumed that pure ^{238}U was used as the initial feed as well as the make-up. In reality, as the depleted uranium from enrichment contains 0.2 to 0.4% of ^{235}U , and that from reprocessing typically contains 0.8%, some saving in fissile material loading will be available actually, for the initial charge. This may contribute some save of plutonium as the initial inventory, as well as for the successive supplement.

The reactor physics calculations were made using the SRAC2006 standard nuclear analytical code with PUBURN routine⁴¹ based upon JENDLE3.3⁴² and the buckling in taking account of neutron leak.

Calculation procedures and results in detail will be published in elsewhere.

III.B. Constraints for Realization of the MSFBR

The present work revealed that the defined breeding could not be realized unless initial concentration of ^{238}U was at least 27.3 mole% in the use of FLINAK.

There have been few confident data concerning the solubility of actinide fluorides in FLINAK other than the references³⁹⁻⁴¹, which had been the basis for the feasibility of the original proposal^{39,40} shown in Fig. 7.

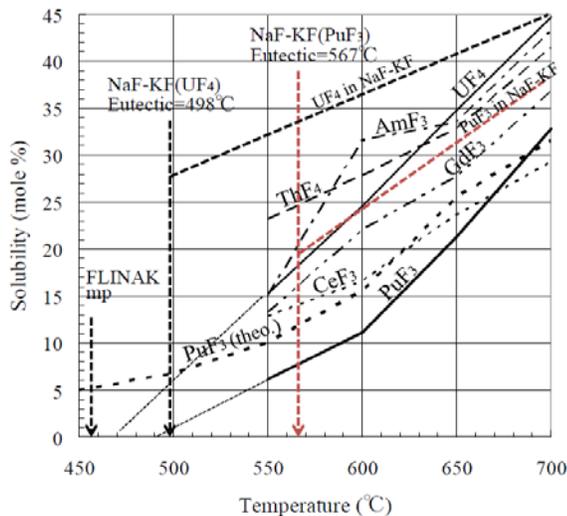


Fig. 7. Temperature Dependence of Actinide and Lanthanide Fluorides Solubility in Alkaline Fluoride Mixtures. (Solvent is FLINAK unless otherwise stated.)

The temperature to dissolve required UF_4 and PuF_3 is 625°C for 27.3 mole% of UF_4 , accompanied with 6.1 mole% of PuF_3 at start up condition and for 8.0 mole% of PuF_3 accompanied with 26.7 mole% of UF_4 at equilibrium condition as shown in Fig. 8 with no interference in solidification is to be assumed between tri-fluorides and tetra-fluorides. Where the line segment of A to C represents the effective solubility of UF_4 and that of B to C does that of PuF_3 .

This system can be feasible unless the temperature is not allowed lower than 625°C , even though it is contradict with the design principle of the molten salt reactor as discussed in the section II.A of this paper. The real question is what will happen, when the fuel temperature drops under any un-stable and un-foreseeable conditions until somewhere between 625°C and 454°C of the eutectic temperature of pure FLINAK as the carrier salt.

Since these solubility measurements have been made using heating (ascending) cycle, it should be in mind that the data only express how much materials could dissolve in the solvent at the designated temperature.

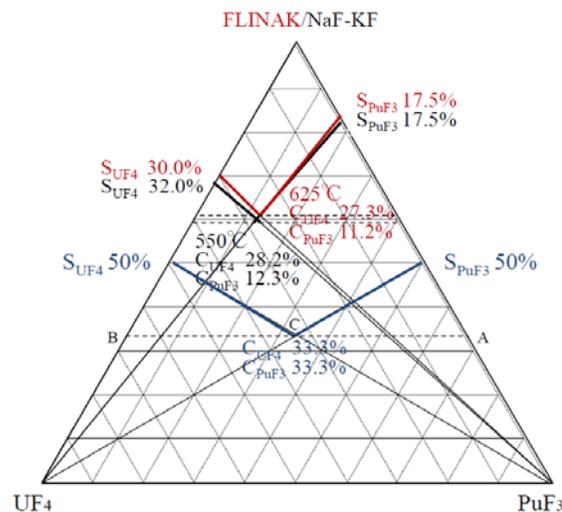


Fig. 8. Effective Solubility Affected by Accompanied Component. (S_{UF_4} , S_{PuF_3} : solubility of UF_4 and PuF_3 , C_{UF_4} , C_{PuF_3} : Acceptable Concentration of UF_4 and PuF_3)

III.C. Resolutions for Realizations of the MSFBR

The authors perceive there might be the other approach, which will pursue a possibility of not the solvent but the established eutectic system of alkaline fluorides containing UF_4 as an analogous to $\text{LiF}-\text{BeF}_2-\text{ThF}_4$ for thorium molten-salt reactors. There would be $\text{LiF}-\text{NaF}-\text{UF}_4$, $\text{LiF}-\text{KF}-\text{UF}_4$, $\text{LiF}-\text{RbF}-\text{UF}_4$, $\text{NaF}-\text{RbF}-\text{UF}_4$ and so on, however, the possible candidate at the date is $\text{NaF}-\text{KF}-\text{UF}_4$ eutectic (46.5-26-27.5mole %, 498°C , arbitrarily named as FUNAK hereafter.) for which the phase diagram⁴³ and some physical properties including density and viscosity have been established¹⁹.

The temperature dependence of UF_4 solubility elucidated from the phase diagram of this system is quite less steep and a substantially higher value is available than that in FLINAK at the eutectic point as shown in Fig. 7.

There was also thermodynamically calculated phase diagram of $\text{NaF}-\text{KF}-\text{PuF}_3$ eutectic (28.4-58.4-19 mole%,

567°C)¹⁷ with also significantly higher solubility value as shown in Fig.7; however no information is available of NaF-KF-UF₄-PuF₃ phase system.

It is anticipated that FUNAK should be much less sensitive to partial solidification issue than FLINAK, and that a NaF-KF-UF₄-PuF₃ mixture might represent a new eutectic temperature somewhere between 498°C and 567°C. The phase stability of the proposed fuel salt until final solidification should be evaluated by extensive experiments precisely according to the 1950's ORNL experience⁴⁴.

Preliminary calculations were performed for the cores fueled hypothetically dissolved PuF₃ or PuF₃+MAF₃ in FUNAK, using the same practice as in FLINAK, and found that two types of MSFBR were reactor-physically feasible provided that the core out-put was no less than 3,200MWt and 2.48 tons or 1.49 tons of fissile materials should be supplemented after startup, similar to that in FLINAK, as shown in the third and fourth columns of TABLE I.

The FUNAK reactor with TRU charge behaved much favorable as a MSFBR, smaller need of initial supplement of fissile materials and larger breeding gain, presumably by better burning capability of MA than that of Pu charge.

The temperature to dissolve required U and TRU shown in the Table I in FUNAK are supposed to be 550°C as shown in Fig. 8 taken into account the presence of UF₃ as the Red-Ox buffer material and some residual fission product tri-fluorides. Thus originally assumed core inlet

temperature of 600°C and the temperature condition for reactor physics calculation (627°C) were satisfactorily validated in this case.

The inventory of TRU is almost kept at constant however that of MA decreases to 54%. Very high transmutation rate of Np to 6.5% comparing with other TRU burners and limited build up of Cm are characteristic features of the MSFBR as shown in Fig. 9 comparing with the MOSART³ of MOX derived TRU charge normalized at 3.2 GWth. The averaged annual core breeding gain of TRU after 10 years from the start to 40 years is 0.0042 comparing with 0.03 of the REBUS-3700 concept using 0.55NaCl-0.45(U+TRU)Cl₃ fuel salt³⁴.

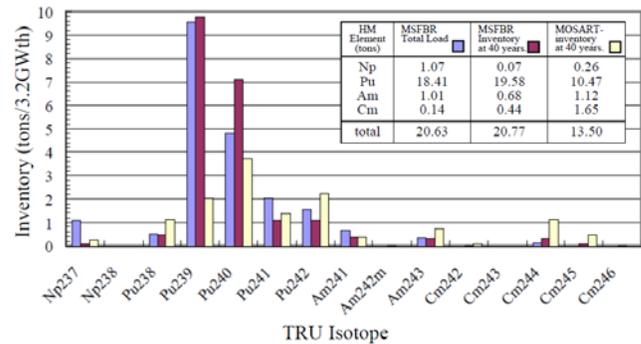


Fig. 9. Evolution of Actinide Inventory in the MSFBR and Comparison with that in the MOSART.

TABLE I. Main Characteristics under which the MSFBR is Physically Feasible

Solvent of the Molten-Salt Fuel	FLINAK	FUNAK	
•Power, MWt	3,200	3,200	3,200
•Height/radius ratio, h/r	1.85	1.85	1.85
•Core volume, m ³	21.2	21.2	21.2
•Specific power, W/cm ³	150	150	150
•Effective fuel density, g/cm ³ ⁴⁰	3.10	3.12	3.14
•Initial fuel loading ²³⁸ U/Pu ^(a) , tons	72.1/16.1	71.3/15.6	
•Initial fuel composition ²³⁸ U/Pu, mole %	27.3/6.1	28.2/6.1	
•Supplementary annual loading of Pu, tones, (feed pattern)	2.71, (0.872e ^{-0.25y})	2.48, (0.655e ^{-0.25y})	
•Equilibrium fuel inventory U/Pu/MA, tons	71.9/20.3 ^(c) /1.2	71.2/19.2 ^(d) /1.3	
•Equilibrium U/Pu/MA composition, mole %	26.7/7.5/0.5	27.6/7.6/0.5	
•Initial fuel loading ²³⁸ U /Pu/ MA ^(b) , tones			71.3/17.08/2.05
•Initial fuel composition ²³⁸ U / Pu/MA, mole			28.1/6.65/0.87
•Supplementary loading of TRU, tones, (feed pattern)			1.49, (0.711e ^{-0.454y})
•Equilibrium fuel inventory U/Pu/MA, tons			71.2/19.58 ^(e) /1.19 ^(f)
•Equilibrium U/Pu/MA composition, mole %			27.9/7.6/0.5
•Annual breeding gain at equilibrium, (years after startup)	0.0031, (19~)	0.0037, (17~)	0.0042, (10~)
•Neutron Leak Rate (initial~equilibrium), %	5.1~4.4	4.7~4.2	4.4~4.2
•k _∞ (initial~equilibrium)	1.063~1.054	1.059~1.051	1.055~1.052
•k _{eff} (initial~equilibrium)	1.009~1.007	1.009~1.007	1.009~1.007
•Void coefficient. (initial~equilibrium), dk _{eff} /(dp/p) ⁴⁰	-0.079 ~ -0.066	-0.082 ~ -0.063	-0.068 ~ -0.063
•Temperature coefficient., [dk _{eff} /(dp/p)] / [(dp/p)/dT], 1/K ⁴⁰	-8.7×10 ⁻⁵ ~ -7.3×10 ⁻⁵	-8.2×10 ⁻⁵ ~ -6.9×10 ⁻⁵	-7.5×10 ⁻⁵ ~ -7.0×10 ⁻⁵
^(a) Separated from BWR 45,000 MWD/T UOX after 5 years cooling, ^{238/239/240/241/242} Pu %: 2.80/51.77/25.98/11.07/8.38.			
^(b) Pu/MA comp. Pu/Np/Am/Cm %: 89.22/5.188/4.901/0.691. ^{(c)(d)(e)} Pu _{fiss} /Pu _{total} %: 55.2, 55.3, 55.6.			
^(f) Pu/MA comp. Pu/Np/Am/Cm %: 93.525/0.358/3.695/2.422, ^{238/239/240/241/242} Pu %: 2.50/50.01/36.21/5.60/5.60.			

Once proposed novel salt systems are qualified, there will be substantial advantages. Firstly, not only separated Pu but also TRU can be used to start up the MSFBR even with higher performances. Secondly, non-Li salts allow us to avoid use of problematic enriched ^7Li , and to mitigate the highly technological challenge of tritium control.

Nevertheless, because the NaF-KF has considerably inferior hydrothermal characters than those of FLINAK, some special cautions should be examined.

III.D. Potential Usefulness of the MSFBR

Once the MSFBR can be verified physically as well as chemically, the following scenario can be envisaged, according to the present knowledge and the experience.

Deployment of Reactor:

- (1) Use 44.2 tons of Pu stockpiles¹ as 18% MOX for the initial load to start up 3.6 GWe (7.82 GWth).
- (2) Use 600 tons of TRU recovered from 50,000 tons of the projected LWR spent fuel by the end of century to start up 42.8 GWe (91.44 GWth).
- (3) Do not discriminate any isotopic compositions of U and Pu and to provide versatile fuel utilizations.

Design of Reactor:

- (4) The unit thermal power of a core is no less than 3,200MWt to provide breeding capability.
- (5) Tank-within-tank arrangement with drain tanks, installed in underground silo.
- (6) Core inlet temperature is allowed to be as 600°C.
- (7) Convert fission energy into electricity by gas turbines with a Brayton cycle or into chemically usable heat.
- (8) Recover tritium as elemental form using a separated cycle and storage as the metal hydride.
- (9) The reactor vessel should be durable for certain period with a high fluence of very hard neutron spectrum, and the shield-reflector design³⁴ should be considered.

Operating of Reactor:

- (10) Total TRU inventories in the operating 46.4 GWe are indefinitely kept virtually constant at 648 tons, comparing with 644 tons as initial charges.
- (11) Domestic stockpile of recovered U shall be utilized 2,200 tons for initial loading and at the rate of only 46 tons a year for make up and maintain the current available capacity for over a millenary.
- (12) U make-up can be done on-line without solvent, by dissolving UF_4 and metallic uranium in the fuel salt. Temporary fissile make-up can be done by TRU oxide in small butch-wise, but its capacity may be realistically optimized
- (13) Assuming 7%/y of burning rate of TRU, the interval of chemical processing will be 3-5 years using the Fluor-Ox process with 0.02% of TRU leak rate.
- (14) The only appreciable drawback of the Fluor-Ox process is building up of fission product Zr. The

electrochemical Zr removal process developed for the MSRE remediation⁴⁵ should be taken into account.

- (15) The Fluor-Ox process originally arranged for the TRU burner should be accommodated to the MSFBR containing U and TRU.
- (16) $\text{Na}_2\text{O}+\text{K}_2\text{O}$ will be a potential candidate of the oxidizer instead of CaO to give the NaF-KF eutectic mixture as the final waste salt form.
- (17) As molten fluorides are transparent, primary system can be optically inspected from the outside in the buffer tank.

When the above mentioned scenario would be implemented, the following merits are available without virtually any drawbacks:

Radioactive Waste Decreasing:

- (1) Yields no other massive fuel cycle wastes but the compact HLSW of which the potential radio-toxicity can be made low as the natural radioactive ores after 400 years without additional P&T practice.
- (2) Virtually negligible process loss of fuel material.
- (3) Some useful fission product elements can be recovered as clean from alpha contaminations.
- (4) Easy to use decay heat of fission products for several decades prior to final solidification of the waste salt.

Sustainably Save of Resources:

- (5) Maintain quantity and composition of the national TRU inventory at constant indefinitely.
- (6) Sustain operation of the current national power level indefinitely, by decreasing recovered U stockpile.
- (7) Fissile material required for deployment is less than that of the solid fuel alternative.

High Grade of Safety:

- (8) Can stop operation safely under plant black-out and no water supply as the walk away mode.
- (9) Keep major activities within low volatile salt forms.
- (10) Large negative temperature coefficient of reactivity may stop reactor automatically with a margin of 100°C.

Realization, Viability and Economy:

- (11) Robust stand-alone central power plant, capable to be installed off-shore sites, less disturbed from outsides, to eliminate all fuel cycle associate waste management.
- (12) Straightforward chemical process without Th, Li or Be.
- (13) Large inventory of U can make support negative Red-Ox potential easily to protect material from corrosion.
- (14) Operation for 50 years of a UOX-LWR, or for 20 years of a MOX-LWR will provide enough amount of TRU to start an MSFBR with the equivalent out-put.
- (15) Encourage operation of LWR by saving the cost of spent fuel management and creating a vast value to it.

Proliferation Resistance:

- (16) Plutonium kept in the reactor core is not a subject of the NPT and the IAEA Safeguard.
- (17) Plutonium should never be isolated from other actinides during the chemical process.

IV. CONCLUSIONS

The MSFBR was evaluated addressing for the emerging national tasks to make 1; plutonium stockpile¹ eliminated, 2; national energy resources secure, 3; LWR spent fuel manageable, 4; HLSW acceptable to be disposed by reducing radio-toxic source term, 5; all measures be safe, simple, robust, stand-alone, viable and economical.

The MSFBR will accept separated Pu as well as TRU from the LWR spent fuel and keep in a core to produce energy indefinitely by consuming depleted U inventory from enrichment and reprocessing. This makes any NPT concern material not exist in Japan, meanwhile national energy resources secure using only leavings of the LWR operation, and resulting easily manageable HLSW. This encourages LWR (and MOX-ABWR) with open-cycle operation so far as the natural uranium will be available.

The MSFBR will produce no spent fuel but the HLSW consisting of virtually only fission products. The radio-toxic source term will be low enough to dissipate within several hundred years by applying the cleaning process with a low TRU leak rate.

There should be significant technical challenges to be overcome to realize the MSFBR, however owing to the characteristic stand-alone feature; the MSFBR can be robust and economically viable by excluding all fuel-cycle-associated fragile infra-structures in addition to all medium-low level but highly massive radioactive wastes.

The calculations which suggested the reactor physics feasibility of the MSFBR are preliminary and conservative nature without optimization however could provide many hints to be theoretically addressed to obtain more beneficial conclusions might be available by utilizing more advanced and elaborating evaluation technologies.

The capability of this concept provides us with revolutionarily enormous potential benefit as a tool to use nuclear fission energy not relying on existing infrastructure with ultimate safety and simplicity. On the other hand, the viability of the concept is fundamentally owing to not any sophisticated theories or mechanisms but to an extraordinarily high solubility of actinide fluorides in very common alkali fluoride eutectic mixtures.

Nevertheless we specified that the experimental and theoretical verifications of phase stability as well as the physical properties of the fuel salts over extended temperature range were the critical technological challenges to be addressed immediately, in order to proceed to the next engineering technology step.

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