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MOLTEN SALT FUEL VERSION OF LASER INERTIAL FUSION FISSION ENERGY (LIFE)

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ABSTRACT

Molten salt with dissolved uranium is being considered for the Laser Inertial Confinement Fusion Fission Energy (LIFE) fission blanket as a backup in case a solid-fuel version cannot meet the performance objectives, for example because of radiation damage of the solid materials. Molten salt is not damaged by radiation and therefore could likely achieve the desired high burnup (>99%) of heavy atoms of ^{238}U. A perceived disadvantage is the possibility that the circulating molten salt could lend itself to misuse (proliferation) by making separation of fissile material easier than for the solid-fuel case. The molten salt composition being considered is the eutectic mixture of 73 mol% LiF and 27 mol% UF_4, whose melting point is 490°C. The use of ^{232}Th as a fuel is also being studied. (^{232}Th does not produce Pu under neutron irradiation.) The temperature of the molten salt would be ~550°C at the inlet (60°C above the solidus temperature) and ~650°C at the outlet. Mixtures of U and Th are being considered. To minimize corrosion of structural materials, the molten salt would also contain a small amount (~1 mol%) of UF_3. The same beryllium neutron multiplier could be used as in the solid fuel case; alternatively, a liquid lithium or liquid lead multiplier could be used. Insuring that the solubility of Pu^{3+} in the melt is not exceeded is a design criterion. To mitigate corrosion of the steel, a refractory coating such as tungsten similar to the first wall facing the fusion source is suggested in the high-neutron-flux regions; and in low-neutron-flux regions, including the piping and heat exchangers, a nickel alloy, Hastelloy, would be used. These material choices parallel those made for the Molten Salt Reactor Experiment (MSRE) at ORNL. The nuclear performance is better than the solid fuel case. At the beginning of life, the tritium breeding ratio is unity and the plutonium plus ^{233}U production rate is ~0.6 atoms per 14.1 MeV neutron.

I. INTRODUCTION AND BACKGROUND

As a backup to the solid-fuel version of Laser Inertial Confinement Fusion Fission Energy (LIFE)[1] we are considering the use of a molten salt fuel consisting of UF_4, ThF_4, or a combination of both, dissolved in flibe. This study of fusion-fission systems using molten salt builds on many past studies, for example Ref. 2 and 3. If the solid fuel options are unable to achieve the high burnup (>0.95 fissions per initial heavy metal atom, usually written as 95% FIMA) desired for the LIFE mission due to radiation damage, then molten salt fuels offer an attractive alternative because they do not suffer radiation damage. A disadvantage of molten salt fuels is they circulate outside the reactor and must undergo some amount of processing during operation of the reactor. This could potentially provide opportunities to access the fissile materials in the fuel, and hence raises proliferation concerns that can be addressed by safeguards. If the solid fuels cannot achieve the desired burnup in a single fueling step, then they, too, would need to undergo processing for further burning, so this processing comparative disadvantage would go away.

We assume the fusion systems (laser, target, fusion power, tritium system, chamber clearing) are the same as for the main solid fuel case and refer the reader to other papers[4,5]. The first wall is also the same.

We find that the nuclear performance of the molten salt case is similar enough to the solid fuel case that we only consider the differences in the system. We find the need to coat the steel with either tungsten or nickel to mitigate corrosion of structural materials by the molten salt. With molten salts there has always been the issue of how much plutonium and other actinides can be allowed to build up before some begins to precipitate out, that is, before the solubility limit for a Pu-containing phase is exceeded in the coldest part of the system. Because the trivalent rare-earth elements (REE), which are high-abundance fission products, behave similarly to Pu^{3+}, it is the sum of these that cannot exceed about 2 mol% at 600 °C. This limitation will need to be verified by more study and experiments and leads us to consider processing the molten salt to remove the rare-earth fission products. How this processing impacts the nonproliferation assessment needs study.

Four possible missions for LIFE

With the same neutron source, the blanket can be designed for different missions. The following are four such possible missions for LIFE to consider.

1- **The power mode with its own wastes burned**. ^{238}U (depleted uranium) and or ^{232}Th dissolved in molten salt and burned to completion (FIMA----->100%).

2- **The power mode with spent nuclear fuel burned**. Remove cladding from spent fuel, reduce oxides and convert to fluorides and dissolve in molten salt and burn to completion (FIMA----->100%).

3- **The Pu and minor actinides burner.** Remove cladding from spent fuel, separate the ^{238}U, and burn to completion the Pu and minor actinides (FIMA---->100%). The Pu could be burned in fission reactors leaving only the minor actinides to burn in LIFE. The feedstock could also be surplus weapons-grade Pu.

4-**The fuel factory mode.** ^{232}Th dissolved in molten salt; remove ^{233}U for use in fission client reactors ($^{232}U/^{233}U$>>2.4% for non-proliferation)[a] fission is suppressed, FIMA<<100%.

Preliminary studies indicate that keeping the concentration of (Pu, REE)F_3 below ~2 mol% is needed to prevent precipitation of an actinide+rare-earth element solid-solution phase. This would suggest replacing some of the U in the melt with Th because this would reduce the amount of Pu generated. We could start with a small amount of UF_4, for example 2 mol% and ThF_4 22 mol%, or even have no UF_4. This would be appropriate if we removed the uranium in spent fuel and feed the Pu and minor actinides to LIFE for burning. The Pu production would be greatly reduced and one could add Pu just as it is burned and keep it below the solubility limit. Even with an all Th system we would need to ensure the REE do not exceed their solubility limit.

II. LIFE mechanical design summary

The LIFE engine configuration is shown in Fig. 1 with the main parameters in Table I. Thermal hydraulic designs and calculations for the temperature distribution in the primary loop of LIFE with a molten salt fuel yield initial estimates for the temperatures at key points given in Table II. An estimate of the liquidus temperature from phase relationships in the PuF_3-UF_4-ThF_4-LiF system is ~520°C for a melt containing 1.5 mol% PuF_3, 15 mol% UF_4, 9 mol% ThF_4 as discussed in section VI. The coldest part of the heat exchanger needs to be safely above the liquidus temperature to avoid Pu precipitation so we have assumed a 30°C margin, giving T_{cold}=550°C. Assuming an additional 30°C drop across the boundary layer of the molten salt results in an inlet temperature to the molten salt blanket of T_{in}=580°C. If the molten salt in the blanket heats up by 50°C due to fission, then the blanket outlet temperature is T_{out}=630°C. If the hottest part of the blanket is cooled by the exiting molten salt with same boundary-layer drop as assumed before (30°C), then the structural hot spot temperature is $T_{structural\ hot\ spot}$ =660°C. If the hottest part of the structure is cooled with other than exiting flibe then $T_{structural\ hot\ spot}$ = T_{out} = 630°C. The important temperatures are constrained by (Pu, REE)F_3 solubility considerations on the low end and structural materials temperatures limits on the high end.

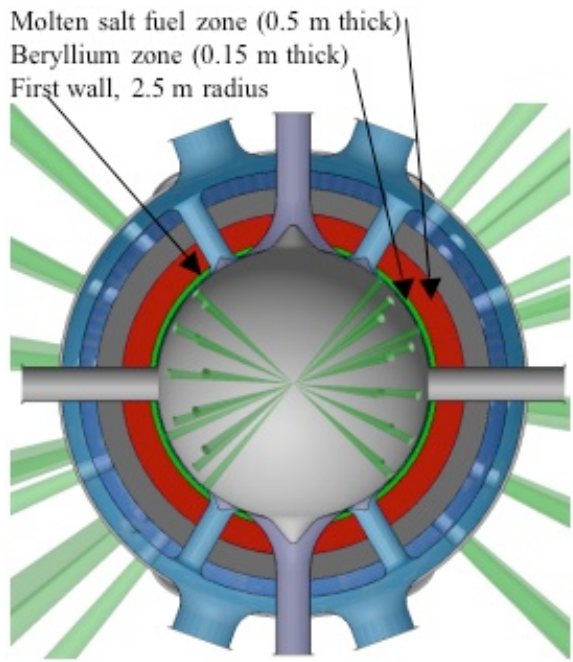

Fig. 1. LIFE engine with a molten salt fuel blanket.

TABLE I.
Typical LIFE engine parameters.

First wall radius, m	2.5
Beryllium zone thickness, m	0.15
Molten salt zone thickness, m	0.5
Laser energy in, MJ	1.3
Target gain	29
Fusion yield, MJ	38
Pulse rate, Hz	13.3
Fusion power, MW	500
Blanket thermal power, MW	2000 to 2600
Neutron wall load, MW/m^2	5.1
Surface heat load, MW/m^2	1.3
Net electric power, MWe	700 to 900

a. The dose rate at 1 m from 5 kg of ^{233}U one year after chemical separation is 100 rem/hr or 1 Sv/hr for $^{232}U/^{233}U$ =2.4%.

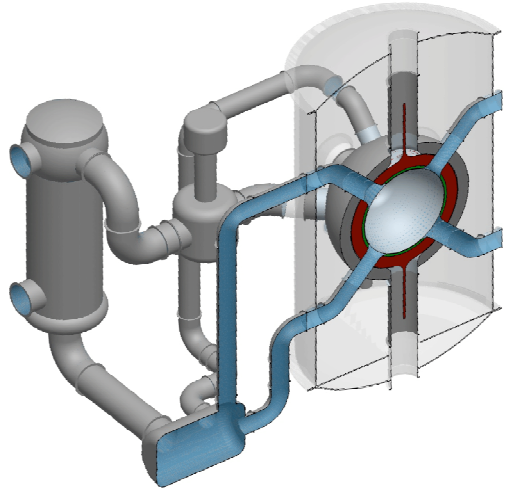

Fig. 2. Illustration of LIFE blanket and flow loopincluding a pump and heat exchanger.

TABLE II.
Temperature at key points in the molten salt loop.

$T_{structure\ hot\ spot}$	630 to 660°C
T_{out}	630°C
T_{in}	580°C
T_{cold}	550°C
T_{melt}	520°C

III. Structural materials

The radiation damage considerations for the steel used for molten salt version of LIFE are very similar to the version using solid fuel. However, corrosion is specific to the molten salt fuel design.

The molten salt containing UF_4 and ThF_4 is aggressive towards corrosion of structural materials. Our choices are:

1-use a frozen salt layer on the structural material to slow corrosion

2-use stainless steel (304) and allow leaching to enrich nickel on the surface to look like Hastelloy

3-coat steel with Mo, W or Ni to inhibit corrosion.

We cannot use the frozen layer approach to corrosion inhibition because of the potential for accumulation of solids containing fissile materials (e.g., Pu) in the layer, with attendant criticality concerns. We cannot use 300-series stainless steel for first wall materials because the thermal conductivity is two to three times lower than for ferritic steel, an important parameter in thermal stress analyses, and the radiation damage is more severe in 300-series austenitic steel than it is in ferritic steel. This leaves us with using coatings for corrosion resistance. Tungsten is assumed for the first wall and could be used to coat the structural components that face the molten salt. A study of the coating of steel with W as armor for the first wall seems to show no problem with regard to fatigue at the interface. Iron and W form stable intermetallics and the bonding is very good.[6] The choice of coating material needs further study.

IV. Molten salt composition

To have a better appreciation of what the issues are, let us consider the phase diagram of the binary molten salt, LiF-UF_4 that is shown in Figure 3. The curve that delineates the high-temperature region of the phase diagram defines the domains of stability of the liquid (at high temperatures) and of the solid or a mixture of solid and liquid (at low temperatures). The lowest melting temperature is about 490°C at the composition 0.73 LiF-0.27 UF_4. The phase diagram for the binary molten salt, LiF-ThF_4, is shown in Figure 4. The lowest melting temperature is about 570 °C with ThF_4 ranging between 21 and 29 mol%. Finally, Figure 5 shows the LiF-PuF_3 phase diagram. The eutectic temperature in this system is about 740°C at ~20 mol% PuF_3 and even more at lower concentration. The high melting point is the source of solubility concerns and the actual system of interest will be combinations of these three binary phase diagrams along with fission products as is discussed in section VI

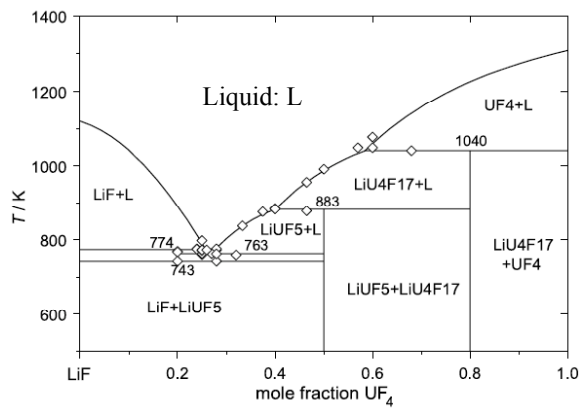

Fig. 3. LiF-UF_4 phase diagram[7].

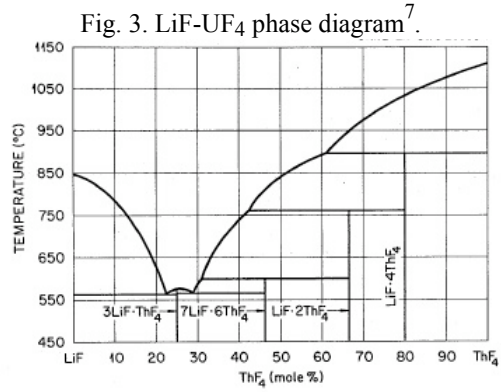

Fig. 4. LiF-ThF_4 phase diagram.[7]

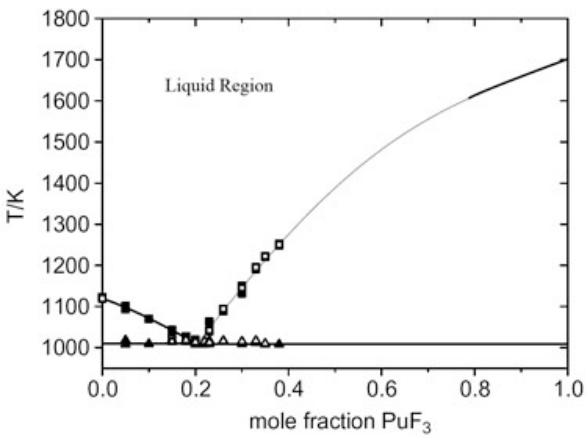

Fig. 5. LiF-PuF_3 phase diagram.[8]

V. Neutronics performance

Preliminary studies of molten salt candidates for LIFE were performed by P. Song[9] and more extensive studies were carried out by J. Powers[10]. The results of that work are summarized here. Neutron and photon transport calculations were performed using the Monte Carlo transport code MCNP5[11]. Burnup calculations were performed using Monteburns 2.0[12], which, in turn, uses ORIGEN2[13] for depletion/decay calculations. Improvements to Monteburns, as well as additional custom code developments, were required to perform the burnup calculations for LIFE. The nuclear data used were from ENDF/B-VII[14].

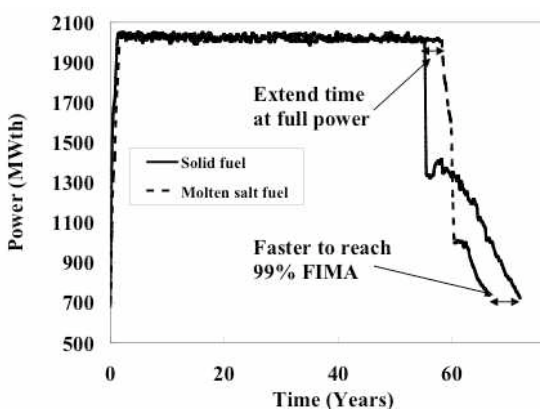

Fig. 6 Power Curves for molten salt fuel compared to solid fuel.

The starting composition of the molten salt was: LiF (76 mol%) + UF_4 (12 mol%) + ThF_4 (12 mol%). The mixture of U and Th was used to lower the peak concentration of Pu in the salt. With a steady 500 MW of fusion power, the total thermal power was kept at 2000 MW by adjusting the ^{6}Li to ^{7}Li ratio. Graphite was added as a moderator. The optimum discussed in the Powers report was 70 vol% graphite and 30 vol% molten salt. The results of the burn calculations are illustrated in Figure 6. As Pu and ^{233}U build up, the fission power rapidly rises to the total desired power of 2000 MW and stays at this level for almost 60 years. Then, as the fertile plus fissile inventory is exhausted, the power declines until FIMA of 99% is reached As expected, the performance is somewhat better than for the solid fuel case owing to the absence of the two fission product classes: gases, Kr and Xe and the noble and semi-noble metals (see discussion of processing below).

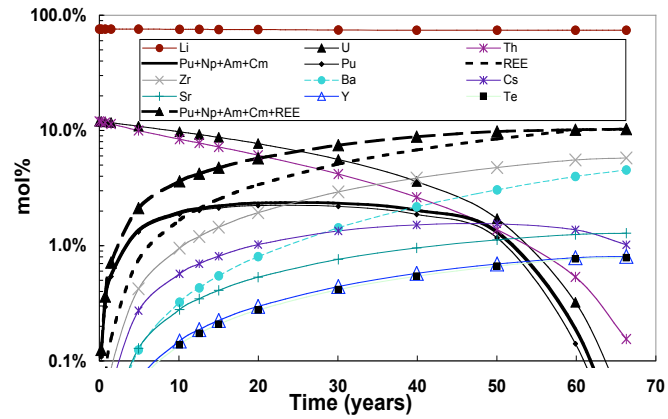

Fig. 7. Evolution of the molten salt composition with time.

The evolution of the composition of the molten salt for this case is shown in Figure 7. The Pu concentration peaked at 2.2 mol%, and the concentration of all the transuranic elements (Pu+Np+Am+Cm) is only slightly larger than Pu alone. This plot also shows the real issue, namely that the sum of the trivalent actinides and rare-earth elements (REE) would approach 10 mol% in the salt. These elements essentially behave as a single element in molten fluoride salts, and form a mixed actinide+REE trifluoride solid solution. The solubility of this phase, as we discuss below, is only on the order of 1-3 mole% in low-BeF_2 compositions (Figure 11). A salt composition containing 10 mol% trivalent actinides and REE would therefore precipitate this solid trifluoride. We therefore need to keep the dashed line in Figure 7 below the appropriate solubility (i.e., the solubility at the minimum temperature of the low-temperature loop). This will require removal of the REE during operation. Going to a Th fuel cycle will not help the REE solubility problem because the high-mass peak of the fission yield curve for ^{233}U still lies in the region of the light REE; however, no Pu is made.

Another set of neutronics calculations was performed assuming removal of REE as well as the noble gases and noble/semi-noble metals and, as expected, the performance was a somewhat better, but similar to that shown in Figures 6 and 7.

The evolution of salt composition during the burn for this case is shown in simplified form in Figure 8. The Pu peaked at 2.3 mol% and the Pu+Np+Am+Cm peaked at 2.5%. In order to reduce the peak PuF_3 concentration, a case was run for 6 mol% UF_4 and 18 mol% ThF_4 and the peak Pu concentration dropped to 1.2 mol%.

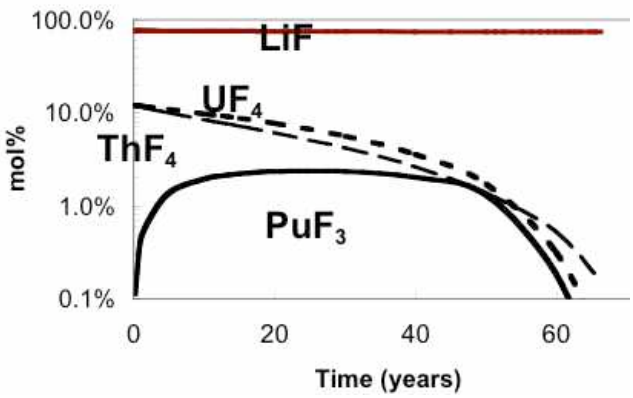

Fig. 8. Simplified evolution of the molten salt composition with time, assuming REE removal

An inventory of ^{233}U and Pu builds up to a peak of about 2 tons peak during the burn. The U is self protected somewhat by the buildup of ^{232}U as is shown by Figures 9 and 10.

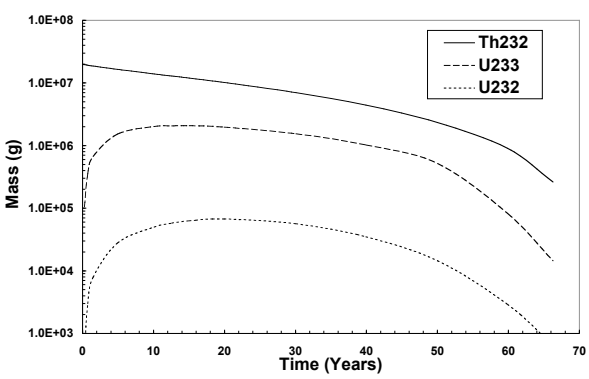

Fig. 9. Time-dependent Isotopic Masses.

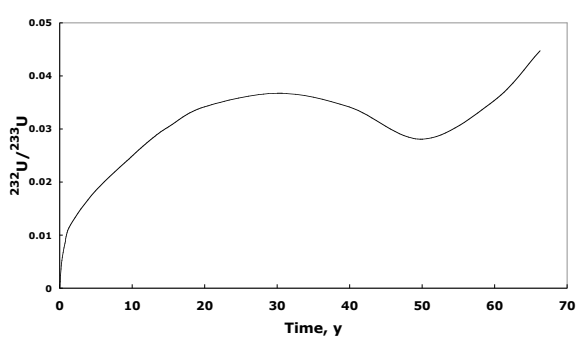

Fig. 10. $^{232}U/^{233}U$ ratio versus burn time.

If we wanted to use LIFE in a breeder mode, we would process fissile material before it is allowed to build up and fission; that is, suppress fissioning. Our preliminary work found a breeding ratio of about 0.6 at the beginning of life of a LIFE engine. That is, 0.6 fissile atoms are produced for each fusion neutron produced. This amounts to about 1000 kg per 500 MW_{fusion}·year. In the case of a fertile, Th-only blanket, the fissile product would be ^{233}U, which would be separated from the molten salt by fluorination. The ^{232}U that is also produced would self protect this material, somewhat reducing proliferation concerns; nevertheless, the facility would need to be safeguarded. In this case the ^{232}U fraction of the product might be higher than that shown in Figure 10 owing to the buildup of ^{231}Pa and lower concentration of ^{233}U, and the ratio of $^{232}U/^{233}U$ would be well over 2.4%. With $^{232}U/^{233}U$=10% the dose rate at 1 m from 5 kg of bare ^{233}U one year after separation is 420 rem/hr or 4.2 Sv/hr which would make this materials rather unattractive to an aspiring proliferant.

VI. Solubility of plutonium and rare-earth elements

We want to know the minimum temperature of the molten salt at which there is only liquid present (*i.e.*, the "liquidus" temperature) as a function of composition. We assume the rare-earth elements are removed. The Th and U concentrations were varied from 24 mol% ThF_4 to 24 mol% UF_4 with ThF_4+UF_4=24 mol%. The equilibrium PuF_3 concentration is assumed to vary from 0 for all Th to 4 mol% for all U case with the remainder of the salt being LiF. The liquidus obtained using these assumptions is plotted in Figure 11. The minimum temperature of 520 °C occurs at 15 mol% Th, 9 mol% U and 1.5 mol% Pu. This means the solubility of Pu limits its concentration in the melt to 1.5 mol% under these conditions.

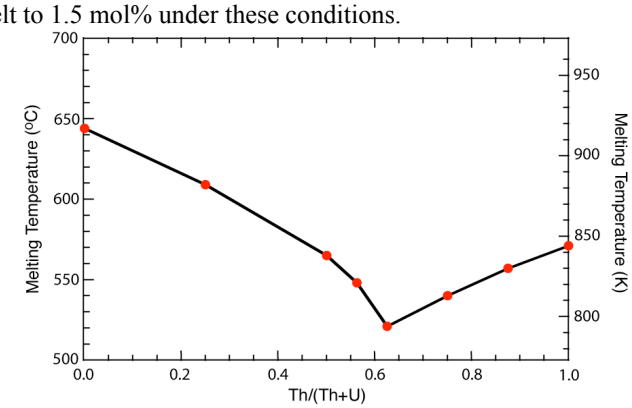

Fig. 11. Liquidus temperature for various Th/U concentrations along the pseudobinary 24%ThF_4 – 24%UF_4+4%PuF_3 in the quaternary LiF-UF_4-ThF_4-PuF_3 system.

VII. Processing and clean up

Some fission products in a molten salt reactor are removed almost automatically and some can be removed by applying processes to side streams. In addition it is necessary in any molten salt system to clean up the molten salt including what is called "doctoring." This process can take place continuously or in batch mode. Impurities and oxygen must be removed. Hydro-fluorination is commonly used to remove moisture. The salt is kept reduced to control corrosion of structural materials by keeping the ratio of $UF_3/UF_4 \approx 0.05$ as a buffering agent.

Gas sparging

Bubbles of inert gas, He, are injected into the molten salt flow. This serves to sparge gaseous fission products, including the volatile fluorides. In this way, ^{85}Kr ($T_{1/2}$ = 11 y), ^{133}Xe ($T_{1/2}$ = 5.2 d), and ^{135}Xe ($T_{1/2}$ = 9.1 h) are removed in an estimated time of 50 s[15]. These gases are held up in pressurized vessels and allowed to decay to stable ^{85}Rb and ^{133}Cs, and long-lived ($T_{1/2}$ =2.3 x 10^6 y) ^{135}Cs. This last nuclide would need to be combined with the alkali waste stream from the final processing of the molten salt at the end of the engine's life and immobilized in a waste form for disposal in a repository.

Precipitation extraction

The noble and semi-noble metal fission products do not form soluble fluoride complexes at the fluoride potential of the melt and are insoluble. They are: Zn, Ga, Ge, As, Nb, Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, and Sb. Experience with the ORNL Molten Salt Reactor Experiment (MSRE)[16] suggests that these elements form multi-atom metallic clusters in the melt. In the MSRE, some of these elements were removed as particulates in the sparge gas, some plated out on the metal walls of blanket structures and piping and heat exchangers, and the remaining fraction circulated in the molten salt. The proportion of these elements in each of these reservoirs varied unpredictably during operation of the MSRE, and additional study of the chemical behavior of these elements will be needed.

If these elements circulate many times around the pumped circuit before plating out it may be possible to preferentially capture them on filters in a slipstream process loop in a time estimated at 2.4 h. One suggestion is to pass the molten salt through a nickel wire filter where plating occurs. Another suggestion is to use a centrifuge to enhance or concentrate these elements. This would be effective if precipitation is occurring homogeneously throughout the molten salt, so that these elements are present in the salt as solid metallic particles. If, however, precipitation is occurring preferentially heterogeneously at metal surfaces then centrifugation would not be very effective. We need to consult those more familiar with the problem to make a prediction of which mechanism is most operative.

Cowell et al. suggested[17] electro-winning as a means to remove the noble and semi-noble metals from the fluoride salt blanket of an accelerator-based conversion (ABC) scheme for transmuting surplus Pu. Electro-winning is commonly used in industry (e.g., it is the sole commercial method for producing aluminum) to yield pure metals from oxide or halide feed materials that have been dissolved in a molten salt. In the case of LIFE, we are interested in the inverse problem, namely the cleanup of a molten salt rather than the production of a pure metal. In the case of a beryllium-containing salt (e.g., flibe), the electro-winning system would use an electrochemical cell consisting of a consumable anode fabricated from metallic Be, and a Ni cathode onto which the noble/seminoble metals and zirconium are plated. The reaction that describes the process is

$$n\ Be(s) + 2\ MF_n(salt) = n\ BeF_2(salt) + 2\ M(s)$$

So long as the free energy difference between BeF_2 and MF_n is negative, this reaction will be spontaneous; thus, the cell could be operated passively. Cowell et al. further suggest that the application of an externally generated potential difference between the electrodes would enhance the rate of mass transfer. The electrochemical potentials of plutonium, other actinides, and lanthanides lie between those of beryllium and lithium, so this process will not remove these elements from the molten salt. In addition to removing fission product metals, the cell could also provide control of the redox potential of the fuel salt. These elements would then be in metallic form, and ready to be converted to an appropriate waste form discussed in the next section.

The remaining fission product classes form stable fluorides and will circulate in the molten salt. They can be removed if we wish to add the necessary process equipment (metal transfer/reductive extraction).

Metal transfer/reductive extraction of fluorides

Considerable work has been conducted on the selective extraction of elements from molten fluoride salts into liquid metals. Partitioning data for many of the elements of interest are available for extraction into molten bismuth with variable amounts of lithium added as a reductant[18]. This process can be described by the reaction:

$$MF_n(salt) + n\ Li(Bi) = M(Bi) + nLiF(salt),$$

where MF_n is a metal fluoride dissolved in the molten salt. This process can remove noble and semi-noble metals, the actinides, Y, the REE, and Zr. In the LIFE application, we are primarily interested in removing fission product REE from the salt, while leaving the actinides and other fission products in the salt so they can be transmuted. This appears to be possible using a sequential extraction of the actinides and REE into molten Bi:Li, followed by extraction of the REE into a molten LiCl salt. The separated actinides would be electrolytically transferred from Bi:Li back into the fuel salt. A simplified conceptual flow sheet for this process is shown below. Considerable research on processing has been done that we hope will be applicable or can be modified for the LIFE application.

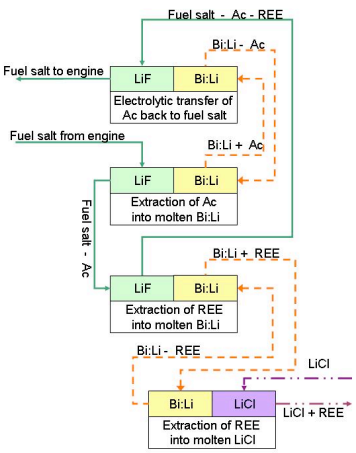

Fig. 12. Conceptual scheme for REE removal.

VIII. Waste streams from processing suitable for disposal

Because the processes that will be used for the on-line and end-of-life processing have not been determined, it is difficult to be specific about the nature of the waste streams resulting from these processes. Nevertheless, one can guess that they will be similar to those of the electrometallurgical process developed by Argonne National Laboratory (ANL) for a variety of spent nuclear fuels[19]. One might therefore expect the waste streams from a LIFE engine to include a metallic waste stream containing Zr and the noble/semi-noble metals (*e.g.*, Mo, Nb, Ru, Rh, Ag, Cd, Tc), and at least one additional waste stream containing the REE, along with some actinides, alkalis, and alkaline earths. The chemical form of this second waste stream is not well defined – most likely it would be a fluoride or chloride stream; however it could be an oxide stream or some other material, depending on the nature of the on-line processing used. In the case of the ANL electrometallurgical process, this stream consisted of a chloride salt.

In addition to these two (or more) streams, there will be the waste stream generated by the offgas treatment process. As discussed above, the radioactive noble gases sparged from the molten salt during operation will either decay to stable isotopes of Rb and Cs, or to long-lived ^{135}Cs, which will need to be separated and combined with the alkali-containing waste stream generated by direct treatment of the salt. The offgas will also contain some quantity of other chemical species that have high vapor pressure in molten fluoride salts (*e.g.*, ZrF_4, I_2), as well as entrained particles of salt.

Waste forms suitable for disposal

Assuming the waste streams outline above, one can further assume that one of the final waste forms from a molten-salt-fueled LIFE engine will be a metallic alloy. For the metal waste form resulting from their electrometallurgical process (which was already Zr-rich due to the presence of Zircaloy cladding hulls in this waste stream), ANL added additional Zr to produce a lower melting point alloy, and to improve the corrosion behavior of the alloy under repository conditions. At this point, one cannot predict what types of further processing will be needed to convert the metallic waste stream into an alloy that has good performance characteristics as a waste form. This will only be possible once process flow sheets and material balance estimates are available for proposed processing systems for a molten-salt LIFE engine. Nevertheless, the development of a suitable metallic waste form should be a relatively straightforward matter.

The waste stream containing the REE, actinides, alkalis, and alkaline earths is more problematic than the metal waste form. If a process similar to that shown in Figure 12 is used, in which the REE are separated into a LiCl salt, then the process used for converting chloride salt waste stream of ANL electrochemical process to a form acceptable for disposal in a repository could be adapted for this wastestream. The ANL process involved occluding the molten chloride salt within the channels and cages of an alumino-silicate zeolite, then mixing this "salt-loaded" zeolite with a borosilicate glass frit, and sintering the mixture in a hot isostatic press. The resulting waste form consists of a multi-phase, glass-bonded ceramic. If the REE stream consists of a fluoride salt, however, this approach is unlikely to work. Molten fluoride salts are not compatible with silicates, due to the formation of stable (and highly volatile) SiF_4. A silicate- or alumino-silicate-dominated waste form is unlikely to be possible for LIFE wastes unless the waste stream is first processed to remove the bulk of the fluoride. Although there are fluoride-containing crystalline silicates, and some level of fluoride can be tolerated, the presence of fluoride in a silicate glass

waste-form will severely degrade the long-term performance of the glass.

Little work has been done to develop waste forms for fluoride–containing waste streams; however, there are numerous options that could be explored for immobilizing either a fluoride salt, or a fluoride-contaminated oxide waste stream. The possibility of reacting the fluoride salts with phosphate-, titanate-, or zirconate-containing precursors to produce suitable phosphate (e.g., monazite, apatite), titanate (e.g., pyrochlore, hollandite), or zirconate (e.g., fluorite-structure REE-zirconate) phases should be explored. If the fluoride salt is first converted to an oxide, then production of either a glass, or a multi-phase ceramic involving the above-mentioned phases would be a straightforward extension of previous work. These, and related, phases are known to accept actinides, rare-earth elements, and alkaline earths into their structures, and have a history and relevant scientific literature to draw upon. More speculative possibilities that might be produced directly from a fluoride salt, but for which there are little literature data, include such exotic phases such as hsianghualite ($Ca_3Li_2Be_3(SiO_4)_3F_2$), semenovite $(Ca,Ce,La,Na)_{10\text{-}12}(Fe^{++},Mn)(Si,Be)_{20}(O,OH,F)_{48}$, or a host of other naturally occurring mineral phases. Clearly, the possible chemistry is rich, and poorly explored to date. Unfortunately, this also means that the *de novo* development and qualification of a completely novel waste-form will take more time than following a previously explored path.

IX. ECONOMICS

An Excel-based systems code is being developed for the solid-fueled version of LIFE. The code contains costing for all the major subsystems including fission fuel fabrication, fusion fuel targets, laser, engine, power conversion and other balance of plant systems, and end-of-life spent fuel containment. Work will be needed to add subsystems and features unique to the molten-salt version of LIFE so that systems can be compared on a consistent basis. The question is how to properly credit LIFE for its ability to destroy weapons Pu or long-lived waste from SNF. For example if the estimated cost of electricity is higher than competing options, the additional operating cost can be assigned to the material destruction mission. In that way, one can calculate the cost per unit mass destroyed. This cost can be expressed in dollars per ton of Pu burned and compared to other proposed uses of this Pu or methods of disposing of this Pu such as burning in LWRs as mixed oxide fuel (MOX).

Strategy for Future Work and gaps in knowledge

We expect to need more study of the phase diagrams relevant to this system, including PuF_3 and fission products, especially the REE. Processing to remove REE fission products will be studied and wasteforms suitable for geological disposal will need to be identified and developed. Economic analyses can be performed to assess the value of burning actinides to a high degree.

The opportunities offered by molten salts as candidate fuels in general, and more specifically in the context of LIFE, Gen-IV and the thorium cycle option, should lead to increased interest. Future studies should be undertaken in five areas: fuel salt chemistry, design and safety, reactor physics, fuel salt clean-up (salt control, online processing, waste form and waste disposal), and materials and corrosion.

An integrated materials design and testing facility will be required to make an optimal choice of salt for the LIFE engine. Salt chemistry, corrosion chemistry, and alloy selection should be jointly studied, and the establishment of a coordinated effort with other activities carried out for the Gen-IV molten-salt fueled reactor should be considered.

We need to determine the radiation damage lifetime of the moderator graphite. We need to develop the design in the direction of steady-state operation allowed by having processing of fission products.

X. Conclusions

We have considered substituting molten salt containing the same actinides that the solid fuel mission was considering in a very preliminary study. Corrosion of the structural steel is mitigated by use of nickel or tungsten plating. As transuranic (primarily Pu) and rare-earth elements are produced during operation of the LIFE engine, they would exceed their solubility limit of about 2 mol%. The separation of these rare-earth elements will therefore be needed. The neutronic performance is better than for the solid fuel case as a result of the removal of neutron-absorbing fission products. Processing also allows the continuous feed of materials and steady-state operation that will improve economics. The impact of processing on nonproliferation need study. Whether LIFE goals can still be sufficiently met with the molten salt design is a key unanswered question. We call for more study of the solubility and processing question that will go hand in hand with the blanket design and optimization. The cost of the molten salt option has not yet been estimated.

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