FUEL CYCLE FLEXIBILITY OF TERRESTRIAL ENERGY'S INTEGRAL MOLTEN SALT REACTOR (IMSR®)

J. Choe[1], M. Ivanova[1], D. LeBlanc[1], S. Mohaptra[1] and R. Robinson[1]
[1] Terrestrial Energy Inc., Oakville, Ontario, Canada
Corresponding Author dleblanc@terrestrialenergy.com

Abstract

The Integral Molten Salt Reactor (IMSR) is being developed at Terrestrial Energy as an SMR (Small Modular Reactor) with an output of 400 MWth of 600 °C industrial heat and/or 190 MWe of electrical production. The use of a graphite moderator and a highly thermalized neutron spectrum gives multiple advantages for standard operations; (i) Superior negative temperature reactivity coefficients, (ii) The ability to start up on SEU fuel (<2% enrichment), (iii) The use of commercially available makeup fuel of standard assay LEU (<5% enrichment), (iv) Low Pu and other transuranic accumulation. The IMSR also offers future fuel flexibility including the ability to utilize spent LWR (Light Water Reactor) fuel without reprocessing but by conversion to fluoride in a dry process. Various other fuel cycle options are also reviewed.

1. IMSR® Technology Overview

The IMSR® power plant design uses a fundamentally different reactor technology – a liquid fueled, low pressure, high temperature, molten salt reactor, within an innovative design.

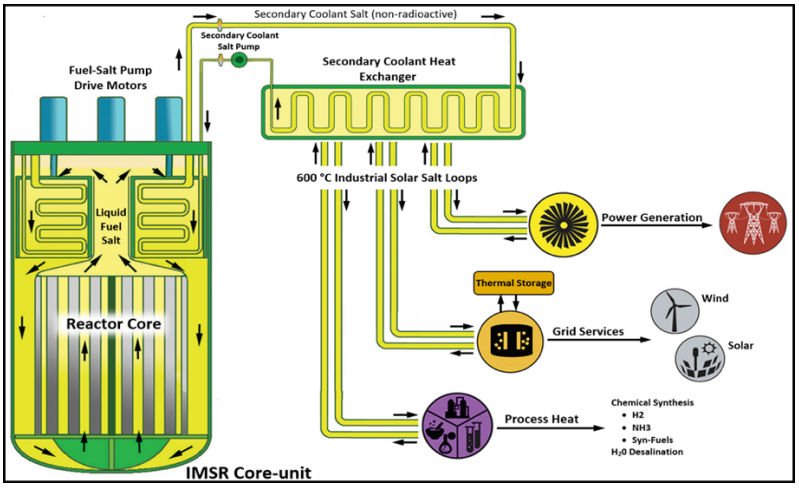

Figure 1 IMSR Power-Plant Basic Schematic.

TEI's IMSR® is fundamentally different from Conventional Reactor systems, and it approaches "Control, Cool and Contain" in fundamentally different ways. This distinction is central to the technical and business case for the IMSR® development.

As illustrated in Figure 1, heat is conveyed from the reactor core by a secondary, non-radioactive Coolant Salt, also a molten fluoride, which flows in a closed loop through Secondary Heat Exchangers external to the Core-Unit. Heat is then removed from the Secondary Heat Exchangers by an inexpensive and common molten "Solar" salt, which is piped from the Nuclear Island to the Balance-of-Plant for electricity production or industrial process-heat uses, or both.

The IMSR® generates a total power of 400 MWth and approximately 190 MW of electrical generation with a peak fuel salt temperature of over 700 °C and a solar salt temperature of up to 600 °C. The Fuel Salt is a highly stable coolant and has intrinsically high radio-nuclide retention capabilities. It can employ standard assay low enriched uranium (LEU) of less than 5% U-235. Such LEU is advantageous due to the availability of this fuel type in existing supply chains.

The IMSR® design greatly eases the high capital cost burden and improves operating cost performance due to the improved overall thermal efficiency, using a high temperature liquid molten-salt fuel in a highly innovative design. This translates to a smaller footprint, and simpler construction.

The IMSR® employs a graphite moderator whose 7-year lifetime at the power densities employed is managed by swapping out of the integral Core-unit in a switch loading fashion. At the same time this eases qualification of metal structures (vessel wall, heat exchanger tubes) contacting the fuel salt as only a 7year lifetime is required. Graphite use opens up numerous advantages for reactor operation and fuel flexibility.

2. Graphite Use and Reactivity Coefficients

The IMSR® utilizes passive and inherent safety features throughout design and benefits from the ability of Molten Salt Reactors (MSR) to attain strong negative temperature coefficients that inherently reduce and/or halt the fission process for any temperature rise. There have been numerous proposed designs, including fast spectrum concepts and the details here can differ significantly.

In a graphite moderated MSR, like the IMSR® , there are three main contributors to the overall temperature coefficient, 1) Doppler 2) Fuel Salt Density and 3) Graphite Temperature. Doppler is always a desirable negative term coming from the broadening of fertile resonant absorption peaks. Its strength can vary greatly, however, being the strongest for epi-thermal spectrum concepts with limited amount of moderation. Fuel salt density will decrease with rising temperature and this has two main effects, reducing the fuel to moderator ratio and increasing neutron leakage from the active core. Graphite temperature effects come from the shifting energy position of the Maxwell Peak of thermalized neutrons. Its behaviour can be complex and reflects the Maxwell peak shifting into or out of the fission cross section peaks in the various fissile elements found in the fuel salt. In the case of Thorium U-233 fuelled MSR concepts, a U-233 cross section peak at ~2 eV leads to a strongly positive graphite temperature term as the leading edge of the Maxwell peak overlaps more with this fission peak as temperature increases. For LEU fuelled concepts, U-235 lacks any such peak and features a very strong negative graphite temperature term. With LEU use however, the build-up of Pu, is a significant fraction of the fissile fuel, and leads to a reduction in the negative graphite term.

Pu effects are subtler in nature as the Maxwell peak tends to almost straddle the peaks of Pu-239 and Pu-241. A fertile capture resonance of Pu-240 at about 1eV also plays a role.

The density coefficient can be counter intuitive and often mischaracterized. An oversimplification is sometimes employed which describes heating up of the reactor pushing fuel salt out of the core and thus driving down reactivity. This is not correct for many MSR concepts including the 1970s MSBR (Molten Salt Breeder Reactor) design. [1] The MSBR featured a positive density term as well as a positive graphite term which were countered by an even stronger negative Doppler term. The reason the MSBR and other MSR designs can have a positive density term is that they are "under-moderated" (like most reactors including LWRs) and by removing fuel from the core, a new lower fuel to graphite ratio can in fact increase reactivity. A positive density term was deemed acceptable in early ORNL development but also means that the sudden creation or injection of bubbles can also lead to a reactivity increase.

The IMSR® has sought to maintain a near zero density coefficient and has done so primarily by going to a quite thermalized spectrum with a low salt fraction. The figure below shows the IMSR's neutron spectrum being somewhat more thermal than that of the MSRE [2] (Molten Salt Reactor Experiment). Thus, the fuel to moderator ratio of the IMSR® results in a core quite close to the maximum reactivity and helps in attaining a negative density reactivity term. In operation the IMSR starts with a very strongly negative temperature coefficient of reactivity with a strong graphite term benefitting from U-235 as the sole fissile element. As Pu builds up the temperature reactivity coefficient reaches an equilibrium of around -3 pcm/°K. For comparison the 1970s MSBR was only calculated to have a net of -0.89 pcm/°K while the operating MSRE [2] had a much larger -15 pcm/°K coefficient. It must be noted that it is easier for a small reactor core, such as the MSRE that feature a large fraction of neutrons leaking from the core, to have a negative density term as a lower density leads to a larger net loss and a beneficial effect upon reactivity.

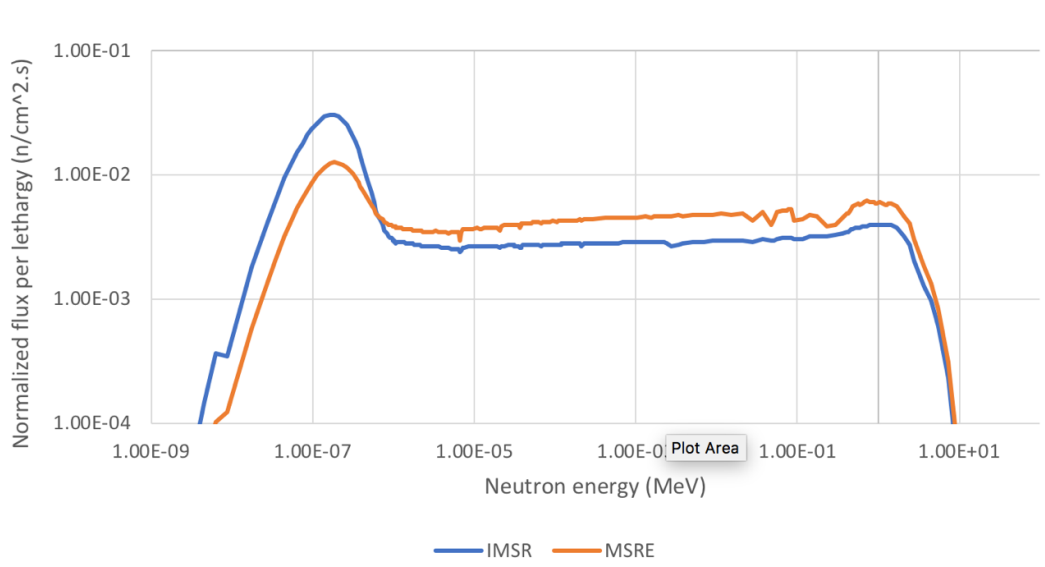

Figure 2 Neutron Spectrum of IMSR and MSRE. Maxwell peak at approximately 0.2 eV but leading edge to just under 1 eV.

The highly thermalized spectrum of the IMSR leads to other important benefits. The starting U-235 enrichment is quite low, below 2%, and allows the makeup fuel additions to remain standard assay (< 5% enrichment) in low enough volume to avoid the need for any fuel salt to be removed over the 7-year design life of each Core-unit. This starting point has also raised very interesting fuel cycle options.

3. Simple Fuel Cycles: Once Through or Feed, Seed and Breed

The most obvious simple fuel cycle for the IMSR is a true once through option. In such operation each Core-unit is started on a substantial volume of fuel salt (carrier fluoride salts plus UF_4 as SEU <2%) and then makeup fuel salt of a similar mix but 4.95% enrichment is added over the 7-year period. The entire final salt volume (about 50% larger volume than at the start) then being moved to the nearby salt storage vault and a new load of SEU startup fuel salt is employed for the next 7-year Core-unit.

Such a mode is relatively efficient on uranium fuel needs, being on par with small LWR designs and because of significant in-situ consumption of Pu produced, the amount of Pu in used fuel is well below that of LWR or CANDU (Canadian Deuterium Uranium) reactor per MWe. However, this mode does result in a substantial volume of fuel salt accumulating over the planed ~60-year facility lifetime within the adjacent salt vault. Per MWe more mass and volume than LWR but less than CANDU.

3.1 Feed, Seed and Breed: Long Term Partial Fuel Salt Reuse

3.1.1 Breed and Burn Cycles Proposed for Fast Reactor Designs

The relatively large volume addition over the 7-year cycle of the IMSR, has allowed consideration of an alternative fuel cycle which in some ways is similar to what is termed a “Breed and Burn” cycle 0, 0 and 0 in fast spectrum reactor designs. The “Breed and Burn” concept, is similar to “feed and bleed” in engineering terms, in that makeup fuel is added, either solid or liquid fueled, as natural, depleted or LEU while the same volume of fuel is removed or “bled”. Such a system reaches an equilibrium where the amount of fission products removed within the spent or “bled” fuel equals production of fission products. Thus, an indefinite fuel cycle lifetime is possible in the “Breed and Burn” concept without reprocessing. With an extremely high breeding ratio, the system can even continue with only natural or depleted uranium in the fresh “feed” makeup fuel. If the breeding ratio is insufficient, LEU must be used. In either case this avoids any reprocessing of fuel but in practice would lead to relatively massive production of Pu and other transuranics in the spent fuel due to the very high concentration of Pu in the “bled” fuel of a fast spectrum system.

3.1.2 Feed, Seed and Breed Cycles Proposed for the IMSR

For the IMSR® with its highly thermalized spectrum, a related fuel cycle option now exists which involves the partial direct reuse of fuel salt and might be termed a "Feed, Seed and Breed" (FSB) fuel cycle. The concept being, after 7 years and adding a makeup fuel salt "Feed", that the now significantly larger salt volume can be split to two locations, one to "Seed" the next Core-Unit with its needed starting fuel and the remaining excess sent to tank storage, representing useful startup material for other IMSR® reactors. Thus "bred" fuel. In practice, if this cycle is repeated, fission product loading will build up during the first few 7-year cycles but reach an equilibrium where the fraction going to storage will be removing the same amount of fission products produced in the last cycle. Thus, reaching an equilibrium that can continue indefinitely. While the makeup needs during each subsequent cycle will be slightly higher than the first cycle, overall fuel economy can improve significantly as each subsequent cycle does not require new LEU in the starting fuel load.

Such a cycle offers many benefits. A significant improvement of Uranium utilization, lower Pu content in the used fuel, greatly reduced buildup of used fuel within the facility and the fact that such used fuel then represents possible starting material for additional IMSR on site or even off site. A main challenge being to assure the somewhat higher loading of fission products and Pu content does not lead to reactor physics concerns or solubility limitations of trifluoride species such as PuF_3 and LnF_3 (Ln being Lanthanide fission products).

3.1.3 Feed, Seed and Breed (FSB) Simulation Results

The EQL0D computer program [3], developed at the Paul Scherrer Institute, is used to calculate fuel salt burnup for the IMSR, and the SERPENT program [3] is used to calculate neutron flux and power using the EQL0D fuel burnup files. As with all MSR burnup routines there are projected removal rates from the fuel salt of both noble gas (by bubbling out) and semi noble and noble metals by plate out on surfaces (HX walls and dedicated metallic filters). Previous ORNL studies [1] assumed a 50 second gas removal time and 2.4 hr noble removal time. In this study 240 seconds for gas and 5 hrs for metals is assumed as removal half lifes.

After a 7-year operation cycle, the addition of makeup fuel has risen the total salt volume by about 50%. Thus 2/3rds of the final salt volume is piped as startup fuel for the next Core-unit and 1/3rd is sent to nearby storage in the adjacent vault. This cycle is repeated for 8 cycles or 56 years.

The figures below show the buildup of total fission products as a function of EFPY (Effective Full Power Years), the fraction of those being Lanthanides and the total transuranics (Pu, Np, Am, Cm etc). Next being the breakdown of Pu vs Minor Actinides showing that Pu dominates. Shown as atomic density percentage in the total fuel salt these figures represent the composition in the reactor. The composition of the fraction sent to the salt vault will have slightly different composition and is also tracked and aged in modeling.

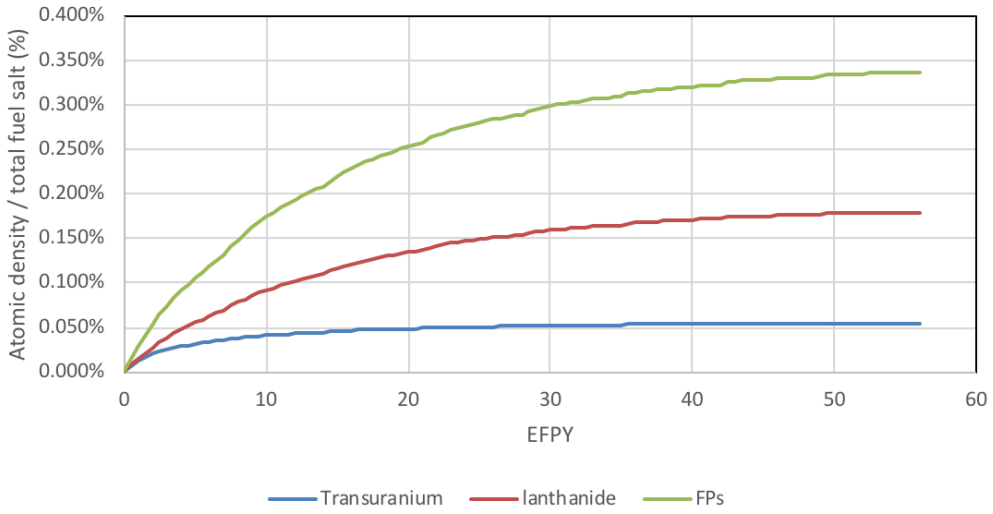

Figure 3 AtomicDensity buildup of Transuranium, Lanthanides and total fission products versus Equivalent Full Power Year

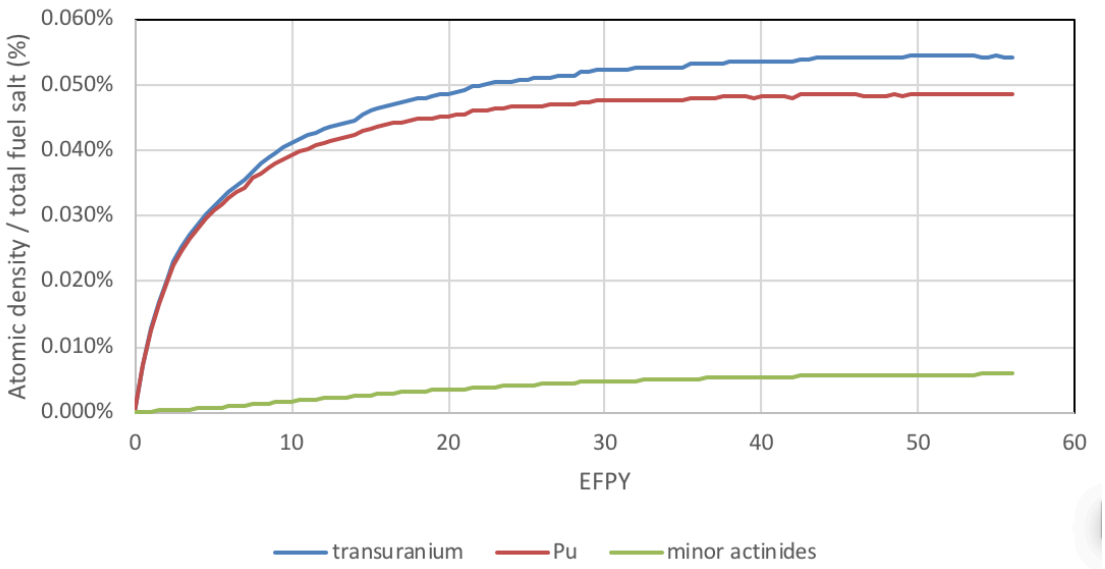

Figure 4 AtomicDensity of total Transuranium and showing large majority being Pu

In figure 5 below the atomic density of the different Pu isotopes in the fuel salt as a function of core burnups is shown. Note that the fraction of Pu-238 and Pu-242 build up significantly (both increasing proliferation resistance). Figure 6 shows the ratio of Pu isotopes at end of plant life (combining last load of fuel salt and that in salt storage). It compares 8 once through cycles to the FSB option. As a point of comparison the isotopic blend of PWR and CANDU spent fuel have far higher fractions of Pu-239 and very little Pu-242.

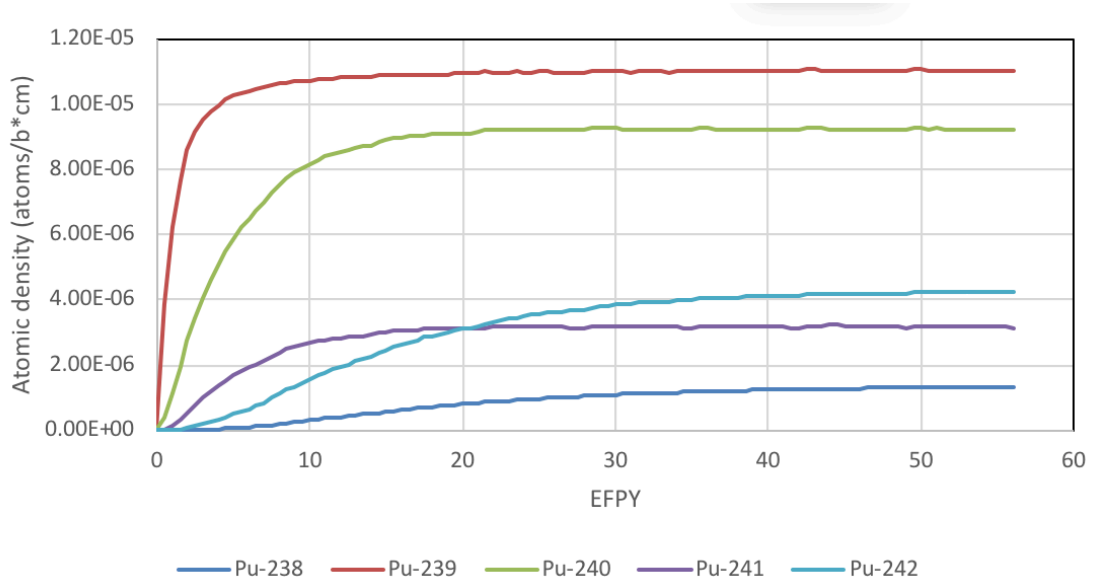

Figure 5 Shows changing isotopic ratio of Pu Isotopes with time

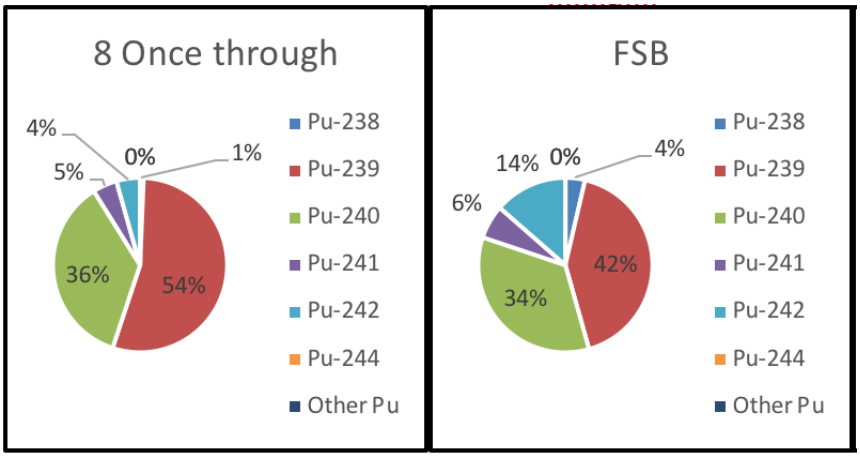

Figure 6 Mass ratio of Pu isotopes to total Pu mass for IMSR once through Cycle and Reuse Cycles (FSB)

As the FSB option also avoids the need for subsequent starting fuel loads there is a substantial savings in uranium needs as well. Table 1 shows uranium needs for the once through option being somewhat more than a large LWR per GWe-Year but dropping by a third with FSB IMSR cycle which matches the LWR usage. It should be noted that the IMSR has logical evolutionary changes to its fuel salt or cycle in the future that can reduce uranium needs to a fraction of LWR (for example the eventual adoption of enriched lithium use once availability and tritium handling issues are addressed).

Table 1 Various parameters for IMSR fuel cycle option vs Standard 1000 MWe LWR.

	8 Once-through	FSB	LWR
Lifetime Fuel Salt Vol	Unity	55% Reduction	
Transuranium [Atom%]	0.036%	0.054%	
Lanthanide [Atom%]	0.069%	0.179%	
FPs [Atom%]	0.132%	0.336%	
Pu [Atom%]	0.034%	0.048%	
Minor actinides [Atom%]	0.001%	0.006%	
Pu mass [kg]			
Pu-238	12.23	40.53	
Pu-239	1012.43	466.76	
Pu-240	669.40	384.00	
Pu-241	84.10	69.72	
Pu-242	81.34	150.96	
Total Pu in 56 years	1859.51	1111.99	
Total Pu per GWe-year	174.8	104.5	~250
Reactivity coefficients [pcm/K]			
Temp Reactivity coefficient	-2.76	-3.38	
Burn-up [MWd/MTU]	14341	31943	33000
Ave U-235 kg/ GWe-year	1510	1033	~1000
Ave Nat U t/GWe-year	277	194	~200
kSWU / GWe-year	219.5	177	~200

Fluoride fuel salts typically have a limit on solubility of trifluoride species which include PuF_3, AmF_3 and CmF_3 along with Lanthanide species LnF_3. Also, while uranium is predominately UF_4, a fraction is maintained as UF_3 to provide a proper redox potential in the salt to reduce corrosion. The IMSR's chosen fuel salt has been measured at a solubility limit of about 5 mole% at minimum expected fuel temperature. Between a once through cycle, and FSB, the total trifluoride molar percentage sees only a very modest increase. With once through having a 0.10 mole % PuF_3 and total of 0.8 mole % trifluoride species while FSB has a maximum of only 0.14 mole% PuF_3 and a total of 1.1 mole% trifluoride species. All cases thus well under the solubility limit.

The Feed Seed and Breed (FSB) concept of partial fuel salt reuse offers a significant benefit to the IMSR concept as it significantly reduces on site needs for spent fuel salt storage while improving uranium utilization and giving a modest improvement in reactivity coefficients. The excess stored used fuel salt then representing viable starting fuel for an expanding IMSR fleet, even further improving resource needs.

4. Closing the IMSR's Fuel Cycle

Fuel cycle options for the IMSR to date have been focused upon relatively simply Once Through

options or proposed partial reuse of fuel salt described as Feed, Seed and Breed. While such options are commercially pragmatic and do represent an improvement over current LWR or CANDU in terms of the amount of Pu and other Transuranics (Np,Am,Cm) going to waste, it is of importance to understand the possible options for the IMSR to potentially eliminate any Transuranic (TRU) waste stream.

There is a significant value obtained by eliminating TRUs from the waste stream of the IMSR. For geological sequestration proposals such as Yucca Mountain it TRUs such as Pu that dominate activity after about 300 years and call for the systems designed for 10,000 years and beyond. Thus beyond commercial considerations of potentially reducing uranium needs, closing of the fuel cycle for the IMSR's own fuel is important to understand the challenges and limitations.

It is often supposed that a fast spectrum is needed to close a fuel cycle. In a thermal spectrum it is true there are more capture vs fission so that the ratio is higher of Pu isotopes and higher actinides to a start point of Pu-239. Countering this fact though is all cross sections, both fissile and fertile are several orders of magnitude higher. Thus, in practice a thermal spectrum is very effective in consuming transuranics. What is the true differentiator is the U-Pu cycle a thermal spectrum is not capable of breeding. The end result though is that a thermal spectrum reactor can be capable of consuming all its own transuranic with the aid of LEU additions and the allowance of some unused uranium leaving the cycle (to join fission products in sequestration or to be saved for future fast spectrum designs). It may not be the definition many are used to but closing the fuel cycle (no TRUs to waste) is possible for non-breeder reactors.

In practice closing the fuel cycle for thermal spectrum reactors can be a challenge for two main reasons. One being that the added difficulty of fuel fabrication for minor actinides, especially americium can be a daunting hurdle due to far higher radiation levels. Second is a complete mix of higher actinides may bring in reactor physics limitations due to effects upon reactivity coefficients. In the case of the IMSR or other MSR-Burner concepts with no solid fuel fabrications needs, dealing with minor actinides is far less a concern. In terms of reactivity effects, this must be fully evaluated but work to date is appearing promising for the IMSR to close its own fuel cycle. The question then remains what cost or complexity is involved to enable TRU reuse to closing the fuel cycle.

4.1 Transuranic Reuse Options for the IMSR

The used fuel salt coming from IMSR operations contains considerable amounts of unused uranium. This however has very little value as the enrichment level of unused uranium is very low and of little economic value. It also represents very limited concern in terms of radioactive waste as the unused uranium has about the same activity as natural uranium. Thus, if it is removed in any process in a pure state, its disposal or conversion to other use or storage for future fast reactor options pose few difficulties. What is of value are methods to remove used fuel salt transuranics (Np, Pu, Am, Cm) as a group and reuse them in subsequent IMSR fuel cycles. If this is performed it offers a way to close the IMSR fuel cycle and improve the waste profile.

4.1.1 Fully Closed IMSR Fuel Cycle by Use of Reductive Extraction

A possibly attractive option to fully close the IMSR fuel cycle comes from the removal of TRUs

by reductive extraction. Most reductive extraction work at ORNL involved liquid bismuth as a carrier metal for a small amount of lithium acting as a reductant. This was very challenging work with the goal of a process for Thorium U-233 breeder concept. Challenging as removing rare earths without pulling out chemically similar Thorium as well as the toxicity and corrosive nature of liquid bismuth. In the case of the IMSR however, the goal is quite different and one would wish to simply remove the TRUs from the fuel salt and leave all else behind.

A fascinating option for this is offered by work out of first Russia and then the CEA (the French Atomic Energy Commission) on the use of aluminum metal as both the bulk carrier metal and the reductant. The first step being uranium being removed from the fuel salt by simple fluorination. Note it is difficult to avoid Np join the U but Np can be later isolated and added back to the other TRUs. Then next step is then the simple act of contacting the fuel salt with molten aluminum will cause all the remaining transuranics Pu, Am and Cm to trade places into the metal in their metal form and AlF_3 joining the waste fuel stream now only consisting of fission products in the carrier salt but virtually no Transuranics.

In studies to date [6] a small fraction of rare earth fission products are likely to also end up in the aluminum but even if not separated later from the TRUs, they would likely represent only a small fuel economy penalty for use of this TRU material as a part of makeup fuel to close the fuel cycle. A study of Gibbs free energy levels of various fission products would be necessary to predict what other fission products might have levels closer to the actinides and possibly also accompany the actinides in transfer to the aluminum melt. Some entrainment of fission products may in fact be desirable to further increase safeguards.

Once removed from the molten aluminum, the TRUs (with Pu never isolated) can be converted back to fluorides and used as a makeup fuel. Likely with a carrier salt and in combination with LEU. Thus, both closing the fuel cycle and improving uranium utilization.

In the IMSR Once Through mode, approximately 215 kg of TRUs would be collected from a 7-year cycle (1.3 GWe-years) and of this about 55% being fissile actinides. If used as a fraction of the makeup in future cycles, the equilibrium amount of TRUs would be modestly higher. For the partial reuse option (FSB) the amount at equilibrium would be moderately higher. In normal makeup fuel additions of LEU there is over a tonne of fissile U-235 in the makeup salt added over 7 years so for each IMSR to close its own fuel cycle, only a small fraction of its makeup fuel would be from TRUs. Maintaining makeup TRU feed as a small fraction may prove important as the effect on reactivity coefficient of increased Pu use may be limiting.

5. The IMSR for the Consumption of LWR/CANDU Spent Fuel Transuranics

The use of spent LWR and/or CANDU fuel as either startup or makeup fuel source for IMSR use could follow numerous parameters. At the simplest level would be the integration of current stockpiles of separated Pu from recycling plants or weapons grade Pu instead of use in LWR. Such an option presumes separation has already been performed as such cost for PUREX are excessive. In addition, processes that are simpler, lower cost and with higher inherent safeguards given the unique ability of MSR-Burners such as the IMSR, could prove an attractive way to help with the issue of spent LWR and CANDU fuel.

Several options are reviewed briefly below. Concepts that focus upon a large fraction of fission power coming from Pu isotopes such as all makeup fuel coming from LWR/CANDU transuranics have preliminarily been shown to be limiting in terms of reactivity coefficients for the IMSR. There exist methods to improve reactivity coefficients but a more secure option may be to limit any externally sourced TRU additions to only a fraction of the makeup fuel as discussed relating to IMSR self TRU consumption.

5.1 Stockpile Pu as IMSR fuel source

Existing Pu stockpiles such as the UK's +100 tonne PUREX derived stockpile which is expected to reach 140 tonnes by 2020 represent an obvious source of IMSR fuel. Any attempt to consume only Pu (i.e no UF_4) in the IMSR might require an extensive redesign of core physics and carrier salt. With the IMSR's reference salt, added Pu might be used as a PuF_3 containing makeup salt. This represents a relatively straightforward option but amount of PuF_3 in the makeup fuel may be limiting based on reactivity concerns.

5.2 Nearly Direct use of LWR Spent Fuel for IMSR Startup with Mild Fluoride Volatility

Similar to the DUPIC concept of direct use of LWR spent fuel as a CANDU fuel, the concept of "nearly direct" would be the conversion of LWR spent fuel to fluoride after decladding and the use of entire mass of stable fluorides as a starting fuel after mixing with appropriate amounts of carrier fluoride salts. Thus, all actinides and most fission products would represent a replacement startup fuel for the IMSR.

Such a process could represent the simplest use of LWR spent fuel from a chemistry standpoint. Potentially the direct contact and mixing of declad LWR pellets with Sodium BiFluoride could be a relatively simple process to convert all oxides in the LWR pellets to fluorides. In the case of CANDU, such "direct use" is found to be more reactive than natural uranium. Thus, in IMSR with its low starting enrichment of ~2% this fuel source may be sufficient on its own or mixed with small amounts of 4.95% LEU. The "age" of spent LWR fuel will play a strong role here for two reasons, past lower burnup levels and degree of aging which converts valuable Pu-241 to Am-241 for instance. A full variety of burnup levels are available. For the most modern spent fuel with burnup of 33 to perhaps 45 GWd/tonne or higher, the spent fuel is far less reactive as there is much less U-235 remaining and far more fission products. However older spent fuel was typically at much lower burnup and, even taking into account the increased loss of Pu-241 to decay, low burnup spent fuel is expected to be more reactive. Reference [7] is a good source here. This document shows that for instance there is very large quantities of lower burnup material and details average burnup and enrichment levels. With these starting points one can use modeling, including the on-line uranium fuel cycle simulator of the IAEA to get details of spent fuel makeup from Pu isotope content to fission product levels.

It is thus of interest to examine if the nearly "Direct Use" of spent LWR fuel could represent a stand alone start up material or in combination with LEU to increase reactivity or decrease unwanted effects on reactivity coefficients. Preliminary work has shown that such direct use will function quite adequately as IMSR startup fuel but that the latest high burnup LWR fuel may start to be insufficient on reactivity without LEU aid.

5.3 CANDU or LWR "Ash" as an IMSR Fuel Source

Another promising source at the next level of difficulty in terms of processing complexity is what has been termed Spent Fuel "Ash". The basic concept being that the use of fluorine gas can selectively remove uranium from LWR spent fuel while also converting all actinides and most fission products to fluorides (termed fluoride volatility). By removing the very low reactivity Uranium (as its enrichment at end of burnup is very low), the reactivity of the remaining "ash" can be substantially higher even though many fission products remain. The use of CANDU spent fuel, in these regards, is far more attractive given that the ratio of Pu to fission products is far lower in CANDU than LWR but of course more tonnage of spent fuel is needed to be processed for the same amount of transuranic "ash".

In a study by Rozon and Lister [8] they assumed that around 5% of the original uranium in CANDU spent fuel would be left behind and be part of the "Ash". This choice of 5% seems to have been due to other contemporary work on fluoride volatility but this fraction can be significantly lower.

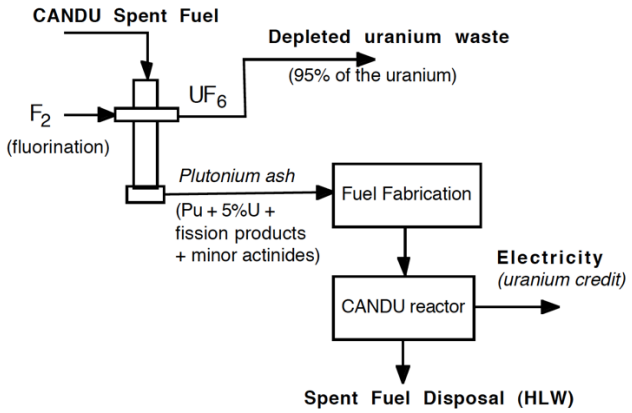

Figure 7 From Rozon et al. Proposal for use of CANDU spent fuel within CANDU reactors.

For use in a CANDU reactor, the steps of conversion back to oxide and new fuel fabrication are far more demanding than if the "Ash" is used directly in IMSR as a makeup fuel form.

The big advantage of such a process is that it can reuse all transuranics from the spent CANDU fuel with a relatively simple process. The behavior of Np during fluoride volatility remaining somewhat of a question as some volatility methods might leave it behind in the ash, others might require Np be removed from the depleted uranium UF_6 stream by NaF beds to rejoin the TRUs.

For a similar proposal of using LWR "ash" the remaining ash will be less reactive as it will have a higher ratio of Pu to fission products and an isotopic blend of Pu containing less fissile. It is of interest to study nonetheless.

5.4 TRU Recycle from by Fluoride Volatility and Aluminium Reductive Extraction

Another option worth investigating would be to extend the concept of LWR/CANDU "ash" in terms of first removing most U by fluoride volatility but then to use aluminum based reductive extraction to separate out the majority of fission products and provide a much cleaner and more

reactive product as IMSR fuel. The extra advantage being far less concern of reaching trifluoride solubility limits due to the absence of Lanthanide trifluorides in the "Ash".

6. Conclusion

Terrestrial Energy is developing the IMSR® in Canada as an advanced Gen IV design with significant economic advantages due to its adherence to a passive and inherent safety profile. It employs a simple standard assay fuel supply and with the aid of partial reuse of fuel salts between 7-year cycles features a significant reduction of Pu and other transuranic with a good fuel economy. Its low fuel cycle cost further aided by the lack of solid fuel fabrication needs. The adoption of a thermal spectrum with graphite moderation offers great leeway in fuel cycle and evolution advances will allow it to close its own fuel cycle to remove Pu and other transuranics going to a waste stream. It also affords pragmatic options with far simpler processing options should nations wish aid in ridding transuranics from existing LWR and/or CANDU stored spent fuel.

7. Reference

[1] R.Robinson ORNL 4541, June 1971

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