SEABORG

Seaborg Wasteburner

Molten Salt Reactor

Whitepaper

Seaborg Technologies
info@seaborg.co

Ref: SEAB-WP-2015-001
Date: March 2015

Executive Summary

The proposal presented before you assess the possibility of building a green and sustainable molten salt reactor, the Seaborg Technologies Wasteburner (SWaB), which burns conventional nuclear waste and produce power at a price comparable to coal. Herein, Seaborg Technologies propose to initiate further investigation into a more detailed design of such a waste-burning molten salt reactor with the aim of constructing a pilot plant.

The high temperature, single salt, thermal-epithermal core, is designed to be highly modular, thus it can be mass produced and decommissioned on assembly line basis. The pilot plant as well as the commercial versions investigated in this report are designed to operate at 50 MW_t. However, the neutronics calculations indicate that it would be favourable to scale up power to 150 MW_t or 250 MW_t.

The SWaB reactor is designed around passive safety mechanisms and is one of the safest reactor designs imaginable. The passive safety features not only minimise the risk of accidents, but also significantly reduce the consequences in case of extreme events (e.g terrorism, one-in-a-million year natural disasters, etc.). One novel safety feature of the Seaborg Technologies reactor is the use of an overflow system in addition to the commonly used salt plug system. This safety system prevents meltdowns, hinders accidents from human operator error, automatically shuts down in case of out of scope operation conditions, and flushes the fuel inventory to a passively cooled and sub-critical dump tank below the core vessel in case of a loss of operation power.

A key feature in the present design is the use of thorium in conjunction with the spent nuclear fuel. The thorium fuel cycle produces significantly less of the long-lived and problematic transuranic waste than both a conventional reactor and a fast plutonium breeder reactor. As a consequence, the core produces much less transuranic waste than it consumes; hence, it is a waste-burner. The 50 MW_t core reduces the amount of transuranic waste in the world by approximately 1 ton over its 60 years power plant lifetime, while building up ^{233}U concentration in the fuel salt to a level where it could potentially sustain a closed thorium fuel cycle.

The reactor relies on a novel on-board chemical fluoridation flame reactor, which can continually extract fission products from the salt during operation. The flame reactor is also used to adjust the fuel levels in the salt such that no absorbing control rods are needed during normal operations; this facilitates a better neutron economy in the reactor. Though no actinide element in this highly proliferation resistant reactor has weapon quality isotope composition at any point during the reactor lifetime, it is worth noting that the chemical reprocessing system is designed such that plutonium cannot be separated from the significant amount of neptunium present in the reactor.

Seaborg Technologies propose building a 50 MW_t liquid fuelled molten salt pilot reactor. The purpose of this is to showcase the viability of the SWaB reactor and to provide valuable insights into corrosion and swelling of the graphite moderator; two of the biggest issues limiting the operational lifetime of this class of reactors.

Contents

1. Process Description

1.1. Reactor core design layout

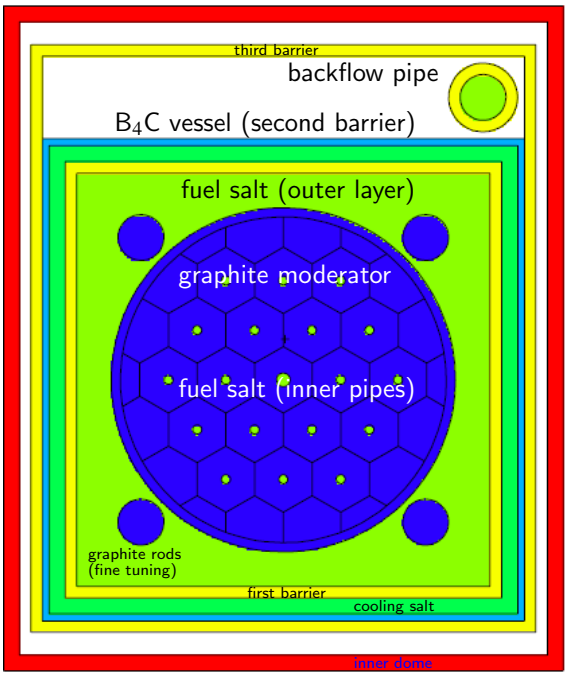

Figure 1: Cross-section of the SWaB reactor core simulated. The dimensions as measured from the outer edge of the inner dome (red box) are 2.4 m by 2.6 m.

The core of the Seaborg Technologies Wasteburner (SWaB) reactor is designed based on a hexagonal graphite assembly; a cross-section of this is shown in figure 1. Central in each hexagon, resides a cylindrical volume containing the fuel salt. Each flow loop includes passing the length of the core three times from outside and inward during which the temperature increases from about 700 °C to 900 °C.

The fuel salt is pumped to the core from an overflow system (OS), from where it flows by natural means through the critical core, into the primary heat exchanger and de-gassing system, and then back into the overflow system.

The OS is the key safety feature in the core design. It is a water-lock-type device connected to a pump which circulates the fuel back into the reactor vessel. In case of overheating, sudden pressure change, or loss of operational power (see section 7), the OS overflows directly to the subcritical and passively cooled dump tank. The OS is the only place where the first and second barriers are breached, as small amounts of fuel are diverted into the chemical reprocessing system (CRS) or to the dump tank.

Core vessel (1st barrier): The core is contained in a 5 cm thick square inner core hastelloy-N vessel. There is one outlet and one inlet for the core fuel. The inner container is cooled to the operation temperature (700 °C) by a spiral pipe of flowing secondary salt.

The inner core container and the main fuel loop are surrounded by a 3 cm thick outer B_4C shielding vessel (2nd barrier) encased in a hastelloy-N container. During normal operation, the B_4C will act as both a neutron shield and a heat shield. This container is designed to withstand high temperatures, thus it can act as a secondary core vessel should the primary container melt during extreme accident scenario (e.g. terrorism).

Reactor container (3rd barrier): The core, the primary heat exchanger, de-gassing system, and the overflow system are contained in a 5 cm thick hermetically sealed Ni-alloy vessel. When the core has served its lifetime or if it fails, the fuel will be drained to the dump-tank and this vessel can be removed and replaced by a new core module. The vessel dimensions are such that it can be extracted to a lead casket in 20 foot container dimensions, which can be transported off-site.

Inner dome (4th barrier): the entire reactor core, chemical reprocessing system, and dump tank are encased in a 5-10 cm thick steal dome buried under large blocks of concrete. At no point during the power plant lifetime will fuel leave the inner dome.

The design outlined above results in a thermal-epithermal spectrum as observed in figure 2.

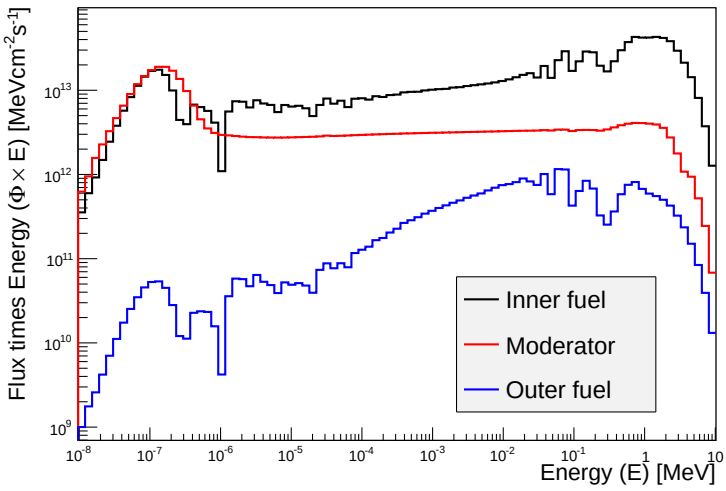

Figure 2: Neutron energy spectrum shown on a logarithmic scale (lethargy).

1.2. Radiochemistry

A quantitative chemical analysis of the elements present in the fuel salt will be included in the chemical reprocessing system. The analysis is based on a real time analysis method which does not require samples to be transferred to the external environment. A patent application has been filed on this approach.

F_2 gas is needed for various purposes in the processing steps for operating the reactor. An on-site F_2 generator is commercially available, which will enable low cost on-site generation of F_2 gas as needed. This removes the need for storage capacity of highly corrosive gasses.

1.3. Schematic

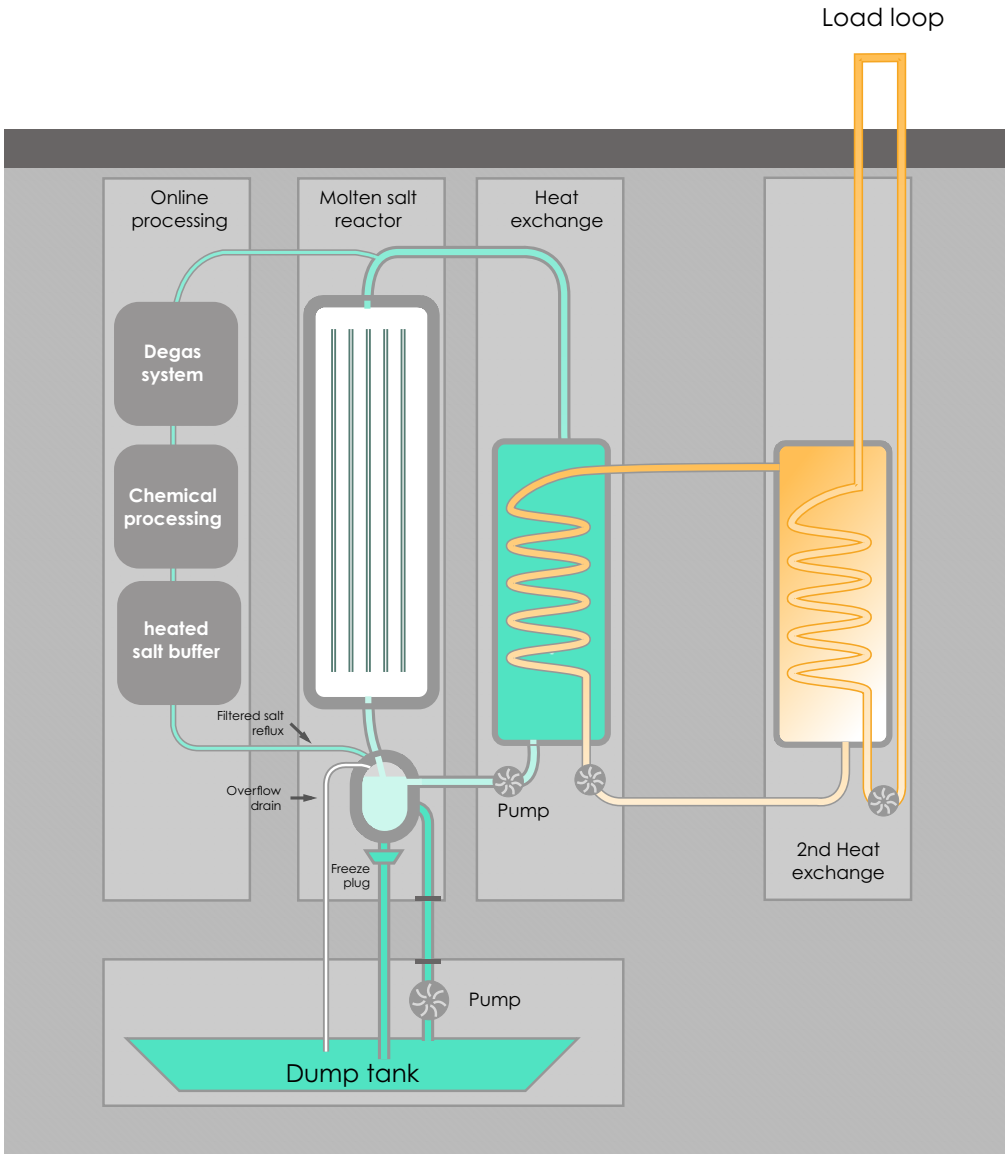

Figure 3: Tentative schematic of the SWaB reactor.

Figure 3 shows the reactor with the chemical reprocessing system, the salt loops, heat exchangers, overflow system, and dump tank. As many aspects of the design is still in heavy development, this schematic is only meant to illustrate the process.

1.4. Energy Output Parameters and Load-following Capabilities

The pilot plant reactor is designed to operate at a nominal power output of 50 MW_t.

In terms of load-following, the reactor is expected to be able to ramp up or down significantly faster and in a broader range than conventional reactors. There are several reasons for this, the main one being that the power output of the reactor is controlled by the cooling salt pump. The large amount of cooling and fuel salts with high heat capacity can store a significant amount of energy so that dramatic load following on a very short time scale can be achieved by simply shutting down the turbines, and storing the energy in the fuel and cooling salts. When heat removal from the secondary cooling salt is reduced after shutting down the turbines, the salt gradually heats up causing the temperature of the inner primary cooling salt to increase. This in turn increases the fuel salt temperature so that reactivity decreases as a result of the negative temperature coefficient (explained in section 3.1). This process is expected to take place on a time scale of minutes and the fine-tuning logs should be adjusted to match the new power level. While decreasing power is rapid, increasing power is somewhat slower. Increasing the power level is accomplished by increasing the cooling pump speed, causing a temperature decrease in the core, hereby increasing the reactivity. However, the increased pump speed results in a decrease in cooling salt outlet temperature which means that the actual power produced by the turbines is slowly increasing while the temperature saturates back to normal operation conditions.

In case of a complete shut-down, the fuel salt will eventually solidify. Even in the absence of external heating, the residual decay heat production is expected to prolong the liquid phase so that the solidification only happens after some days. However, before this occurs, the salt should be drained into the dump tank below the reactor, by simply disabling the fuel salt pump. In order to restart the reactor, the salt must first be melted using external heaters surrounding all salt containing volumes, including piping. The reactivity of the reactor is not expected to suffer significantly from ^{135}Xe poisoning, so once the fuel salt is melted and pumped back to the core, the reactor can be restarted. The reason that xenon poisoning is not expected to constitute a significant obstacle for restarting the reactor is due to the large inventory of about 23 tonnes of fuel salt from where $\sim$13.5 tonnes are actinides; a conventional light water reactor (LWR) has $\sim$40 tonnes of actinides. However, the average power output is only $\sim$1.4%, hence the ratio of ^{135}Xe to fissionable nuclei is very favourable compared to LWRs. In addition, most xenon is extracted using the de-gassing system.

1.5. Operational Lifetime

1.5.1. Expected lifespan

The goal of a pilot plant is to demonstrate that the technology works and allow us to assess in which areas it needs to be optimised before a full scale production power plant is built.

We suggest a 1 year operation life for the pilot plant. The actual commercial power plant will be designed to last 60 years. The first commercial reactor core will be designed for a 4 year lifespan. We plan to increase this to around 10 years in future models. One of the critical issues limiting the lifetime of the reactor core is the swelling of graphite and corrosion from the harsh environment in the reactor core. One of the aims of the pilot plant will be to investigate these two issues.

1.5.2. Pilot plant construction time

The vision of Seaborg Technologies is to serially produce Wasteburners in a central factory. This relies on short construction time per plant, but for the pilot plant of course, this can not be realized.

Rather than providing estimates inflicted with uncertainties beyond what is reasonable, we simply state the construction time will be highly impacted by funding, the degree of political support as well as the turnaround time between nuclear regulative authorities in order to obtain the required licensing.

2. Components

2.1. Component List

The set of tables in the appendix lists some of the components needed for a small power plant with a single Seaborg Technologies Wasteburner reactor, including estimated prices. Note that some components will be superfluous for the pilot plant (e.g. as the turbine and generator - some of the most expensive components), which can reduce the price of the pilot plant compared to the commercial versions.

2.2. Specifications for Key Components

Table 1 lists some of the key specifications and overall design parameters of the current design.

2.3. Corrosion Issues

The molten salt used in the reactor consists of $^{7}LiF-ThF_4$ (78 - 22 mol %) in connection with actinides of the form $LiFAcF_{n(n=3 or n=4)}$. This salt was chosen because it is relatively simple to reprocess and has desirable neutronic and physical properties as detailed in section 3.1. The salt does not contain any toxic Be, greatly simplifying its handling.

The inner fuel loop will be engineered in Hastelloy-N. In order to control the corrosion rate, the redox potential is adjusted chemically by adjusting the ratio of UF_3/UF_4, this way the corrosion can be kept at acceptable levels[1]. The adjustment of UF_3/UF_4 is done in the hydrogenation reactor, where conditions such as temperature, reaction rate, and hydrogen content, are adjusted to convert small amounts of UF_4 to UF_3, that will affect the redox potential and thereby slow down the corrosion rate.

The second cooling salt loop will consist of FLiNaK. The corrosion from FLiNaK is too high in Hastelloy-N, especially due to the presence chromium in the alloy. Therefore an alloy without chromium and with a high level of nickel is needed, e.g. the Ni-201[1] alloy which consists of 99% nickel.

In the chemistry loop, Hastelloy-N is used where the levels of highly oxidised metals like U(VI), Np(V), Pu(VI) etc. are sufficiently low. For other systems, where the oxidisation

Description	Note	Value
MSR power	($>70\%$ from ^{239}Pu)	50 MW_t
Wasteburner container dimensions	WxHxL	2.4 m x 2.6 m x 12.2 m
Reactor core tank dimensions (excluding heat exchanger)	WxHxL	1.8 m x 2.0 m x 3.0 m
Moderator	Graphite core	1.4 m (diameter) x 2.4 m
Fuel salt		78LiF-22ThF_4
Fuel salt		mole fraction
^{6}Li	stable	0.017%
^{7}Li	stable	29.240%
^{19}F	stable	62.260%
^{232}Th	14×10^{9}years	7.193%
^{235}U	0.7×10^{9}years	0.006%
^{236}U	23×10^{6}years	0.003%
^{238}U	4.5×10^{9}years	0.507%
^{237}Np	2×10^{6}years	0.026%
^{238}Pu	88 years	0.009%
^{239}Pu	24×10^{3}years	0.305%
^{240}Pu	6.6×10^{3}years	0.127%
^{241}Pu	14 years	0.045%
^{242}Pu	0.37×10^{6}years	0.026%
^{241}Am	433 years	0.028%
Other		0.207%
Fuel salt outlet temperature		900 °C
Fuel salt inlet temperature		700 °C
Fuel salt heat capacity		6.567 kJ / liter · K
Fuel salt specific heat		1.594 kJ / kg · K
Fuel salt volume		5.7 m^3
Fuel salt density (700 °C)		4140 kg/m^3
Fuel salt flow through primary heat ex.		38.1 liter/s
Circulation time for fuel salt volume		2.5 minutes
Salt in coolant salt loops		LiF-NaF-KF (46.5 - 11.5 - 42)
Efficiency of Air-Brayton turbine		40%
Expected life time of SWaB core	First version	4 years
Expected life time of pilot plant		1 year
Fissions per second		$\sim 10^{18}$
Fission products per year		18.5 kg per year
Size of primary heat exchanger	2 pcs.	1.4 m (diameter) x 2 m
25 MW Air Brayton turbine and generator	WxHxL	4.6 m x 7.8 m x 18.7 m

Table 1: Design specifications.

levels are significant, as well as for the flame reactor, a high nickel content alloy like Ni-201 will be used.

2.4. Component Availability

Components in contact with the salt need to be made from Hastelloy-N or other nickel based alloys. Several metal foundries exist that are able to supply Hastelloy-N pipes and plates - this section discusses components which are not readily available from suppliers but must be custom made.

Given the present level of design it is premature to identify the exact specifications of each individual component.

Many components for the SWaB reactor need further R&D, much of this could be contracted. Other elements, such as the bismuth−thorium extraction unit, as well as several components situated in the vicinity of the MSR core need substantial R&D by Seaborg Technologies.

Equipment to monitor: temperature, pressure, and gamma spectrum (broad wavelength range) are key in the control of the chemical separation systems. However, measurement equipment that can sustain operations in the very radiation harsh and corrosive environment may not be readily available and could require significant development by Seaborg Technologies, hopefully in collaboration with other molten salt reactor startups who face similar challenges.

Currently it is not possible to estimate the costs associated with possible R&D, but it could eventually constitute a substantial fraction of the combined R&D costs. The availability of measurement equipment will be addressed in the near future.

3. Fuel & Salt

3.1. Fuel Type Justifications

In this section, important results from simulations and features of the reactor core, which relate to fuel salt and burnout will summarised. More details and results from the simulations and calculations will be presented in the subsections below.

Though alternatives exist, as shown in table 2 on page 12, the core is planned to reach criticality on nuclear waste from a conventional pressure water reactor (PWR) from where 98.9% of the uranium, 99% fission products, and all noble gasses have been removed using the CRS. Note that plutonium and neptunium are not separated from each other in the CRS. Furthermore, none of the actinides in the reactor will be of weapon quality at any point during the operation of the reactor, as they are highly contaminated with isotopes decaying via spontaneous fission (SF isotopes) and high γ-radiation emitters (this would make them impractical to work with). These facts make the Wasteburner design amongst the most proliferation resistant reactor types imaginable.

In order to burn transuranics faster than they are produced, thorium is added to the carrier salt replacing the removed uranium. This reduces the long lived actinide production significantly and improves the achievable burnup. In the longer term thorium also facilitates

the possibility of converting into a closed Th-cycle (see details below). This is one of the primary goals of the Seaborg Technologies. All figures and calculations presented in this document assume that the fuel used is reprocessed spent nuclear fuel mixed with thorium, as shown in table 2.

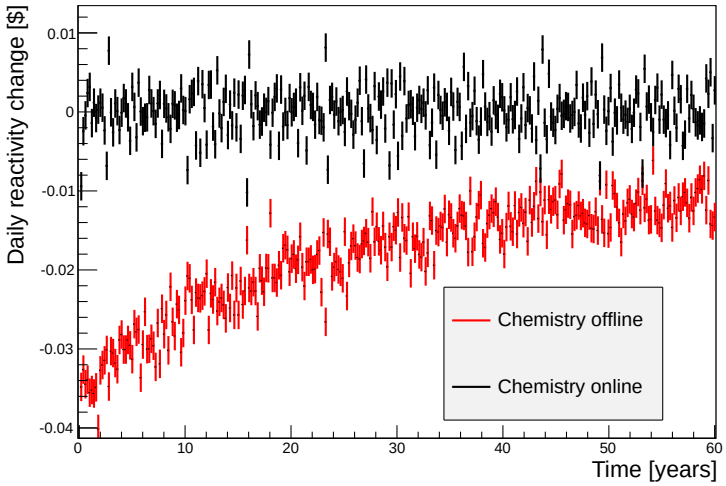

Figure 4: Reactivity change per day with and without the CRS system online. It is observed that the reactivity without the CRS online is approaching 0$ which means that the refuelling rate goes down over time (reactivity of 0$ would mean that no further refuelling would be required (except for replacement of the ∼50 g of fuel burned daily)).

One important feature in the SWaB core is that the use of absorbing control rods is avoided, which results in a better neutron economy. Though a shut-down-rod exists, it is meant to be either fully inserted during shut down or fully extracted during operation.

Power production is controlled on three time scales: on a day to day basis reactivity[1] is controlled by increasing or decreasing the thorium fraction in the carrier salt using the refuelling capability of the CRS (see figure 4); on an hourly basis the power output is controlled using four movable graphite logs capable of changing reactivity with 0.88$, the waste to thorium ratio should be such that the reactor can be shut down using these fine control rods; on a second-minute scale the reactivity is controlled by a strongly negative temperature coefficient, as shown in figure 5.

Figure 6 and figure 7 show how the reactor slowly evolves towards the Th-cycle, while reducing the amount of long lived transuranic waste existing in the world. Note that the neutronic quality of the waste is reduced over time which results in the refuelling rate being faster than the burn rate (see more below). As the initial design is optimised for running

[1] Reactivity is a measure of criticality in units of $: < 0 $ is subcritical, 0 $ is critical (stable power output), and 1$ is prompt criticality.

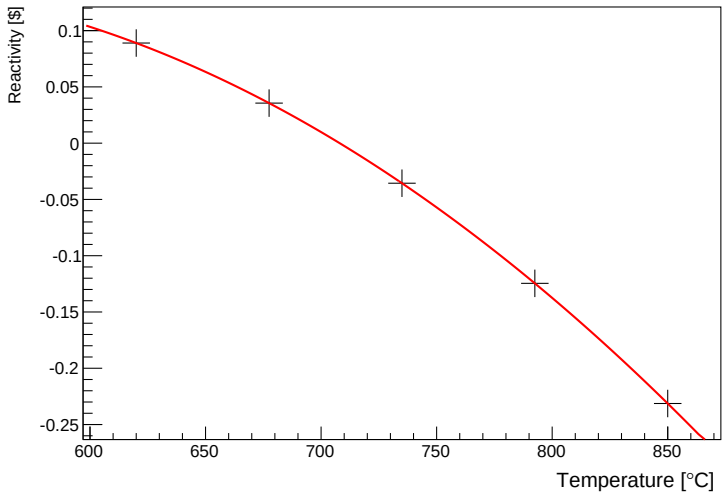

Figure 5: Reactivity as a function inlet temperature. The salt density decreases by 4.8% as the temperature increases from 620 °C to 850 °C, causing the reactivity to decrease.

on transuranic waste with thorium as an absorber, and as the core will be replaced regularly while fuel will stay the same, it is imaginable that as the fuel cycle evolves over time new cores evolve with it. However, the results presented here are based on a design which does not change over time and which is optimal for the initial fuel.

3.1.1. Simulations

Neutronics simulations are carried out using MCNPX [2, 3] using a Cinder90 coupling the neutronic results are used to time evolve the inventory of the reactor core. The neutron flux map at startup is shown in figure 8.

The time evolution is performed stepwise; after 73 days (0.2 year) of static burnup calculations, the results are transferred to ROOT [4] via c++. Dedicated c++/ROOT algorithms were developed allowing recalculation of the 73 day time step with reprocessing online, i.e. CRS and de-gassing system, and to (automatically) refuel based on a user defined reactivity requirement (we used k_{eff}=1.0005). The refuelling algorithm makes use of the neutronics results from MCNPX and can perform three operations: it can remove thorium from the reactor (which is not 100% ^{232}Th), it can add natural thorium, or it can add nuclear waste. This is done with the constraint that the total fuel volume should be constant over time. Note that in these simulations, the refueller adds fuel from the same batch as the initial batch, i.e. the fuel added at a later stage of the burnout calculation is older. After these calculations, the algorithm writes a new MCNPX input file and MCNPX is restarted for the next iteration.

The burnout simulations are computationally heavy. To simulate the entire core including

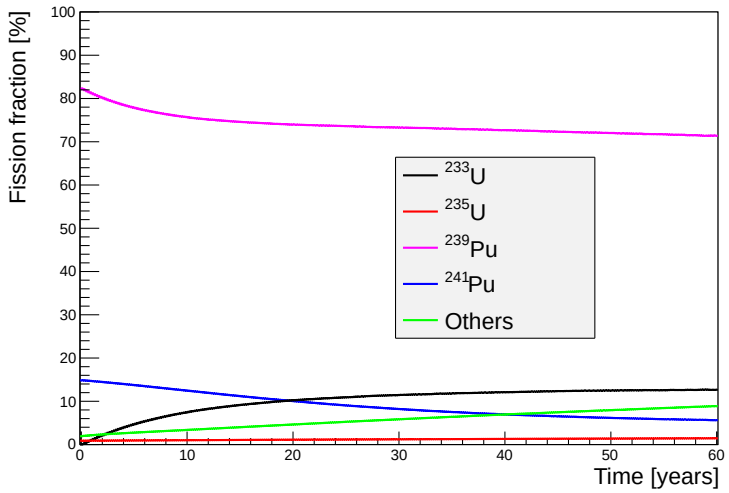

Figure 6: This figure shows which fissile isotopes contribute to the energy production. Note how the reactor slowly evolves towards the Th-cycle. The thorium inventory is decreased over time (see figure 9), which impacts the ^{233}U production rate.

surrounding structure with high precision on burnout, each 73 day simulation cycle requires approximately 75-85 CPU hours on a modern computer. As a consequence of time limitations in combination with the lengthy burnout simulations, it has not been possible to optimise the reactor design, taking into account results from burnout simulations. As the design/simulations are still ongoing at the time of writing, the results presented here are preliminary. The experience and results gained from this first burnout run will be very valuable for the redesign of the reactor, into a mark 1 version.

3.1.2. Initial fuel

Thorium is not fissile by itself and thus requires some other fuel to initiate the cycle. As the pilot plant will run an open fuel cycle, fissile fuel will need to be continually added.

The reactor could be started using either nuclear waste from e.g. PWRs, enriched uranium or decommissioned weapons (weapon grade plutonium), as shown in table 2. From a reactor core perspective, highly enriched uranium is favourable as it is a better initiator for the thorium cycle. However, since the desired enrichment is beyond the allowed limit, it is not realistic as initial fuel.

Though the present focus for the power plant simulations discussed in this section is put on using spent nuclear fuel to start the thorium breeding process, it may be favourable from regulative or handling perspectives having the pilot plant run on a mixture of low enriched uranium and thorium, see table 2, page 12.

In the burnout simulations, the initial fuel vector is derived by first simulating a burnout of 4.5% enriched uranium in a detailed model of a 17×17 assembly of a 2.5 GW_t conventional

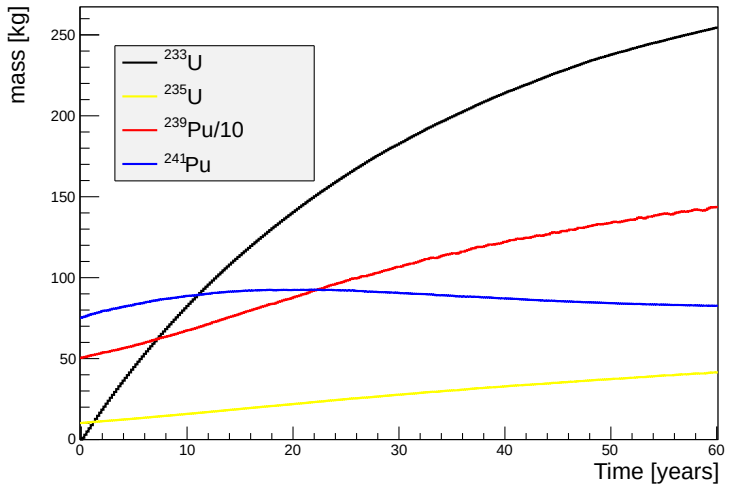

Figure 7: Fissile inventory over time. Note that the thorium inventory is slowly decreased (see figure 9), which has an impact on the ^{233}U production rate.

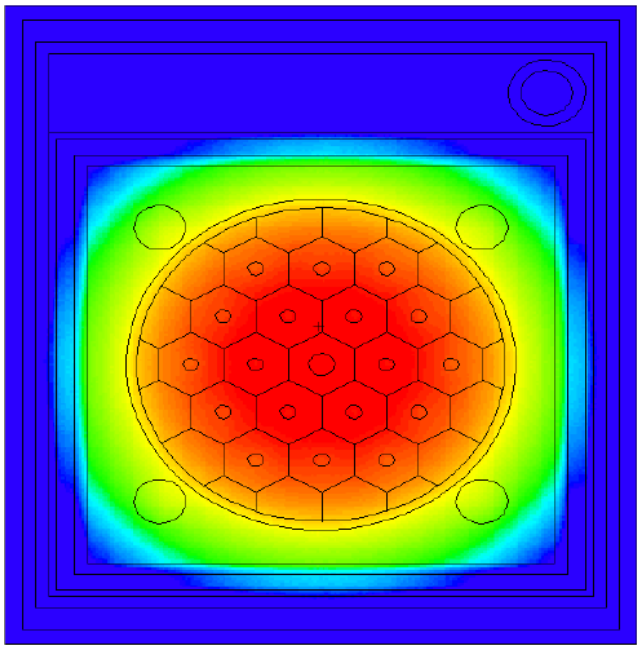

Figure 8: Cross-section of the reactor core (see figure 1) overlaid with a neutron flux map corresponding to the first time step of the burn-out simulation.

PWR reactor. The fuel is burned for 3 years to 45 GWd/tHM. The spent fuel is stored for 10 years, then added into the FLiTh salt until $k_{eff} \approx 1.0005$ (at this point the reactivity span using the fine tuning rods is $-.3\$$ to $.5\$$).

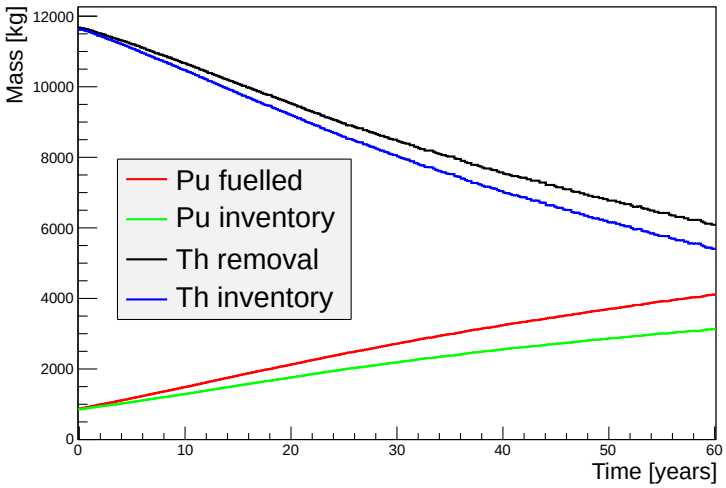

Figure 9: Development of plutonium and thorium inventory over time. The fuelled amount is the sum of the initial fuel and the amount added during operations, hence the difference between the two curves indicates the amount burned. Note that thorium is removed at approximately double the rate of the plutonium is added, this is because the plutonium comes from fuel which contains $\sim 50\%$ and the total fuel volume is kept constant over time.

	processed waste	4.5% ^{235}U	19.99% ^{235}U	93% ^{235}U	100% ^{239}Pu
^{232}Th fraction	86.7%	6.9%	73.3%	93.9%	97.6%

Table 2: The table shows the thorium fraction to be added in order to arrive at an initial $k_{eff} = 1.0005$ for different types of initial fissile material.

An itemised list of the simulation procedure is found in the appendix A.

3.1.3. Fuel cycle

One important design goal of the Seaborg Technologies Reactor is that it eventually should run a closed fuel cycle based on thorium. This is not possible in the present design. The reason is that the neutron per neutron rate is η =1.059 in the case of a fully equilibrated isotope composition of the Th-cycle, see figure 10 (note that η is calculated using all actinides from ^{228}Th to ^{248}Cm including decays, (n,fission), (n,γ), (n,2n), (n,3n), (n,p) and (n,α), and not only the isotopes and reactions illustrated in the figure 10).

The η-value of 1.059 means that, in theory, a reactor with non-actinide absorption and leakage summing to 5.9% would run forever with no transuranic production if it was only supplied with fresh thorium. In comparison the U-cycle shown in figure 11 reveals η=0.878,

which makes it theoretically impossible to achieve a fully closed U-cycle in any reactor with a similar design. Using the MCNPX implementation of the present reactor design, the neutron leakage is calculated to exceed 6.5%. The substantial leakage is mainly a consequence of the core compactness. Moreover some neutrons are lost to absorptions in the moderator, carrier salt (lithium and fluorine), and fission products. This is illustrated in figure 12.

A detailed look into the U- and Th-cycles (figures 10,11) illustrates an important feature concerning the waste produced by burning either uranium or thorium, namely that the Th-cycle only gives rise to a fraction of the problematic transuranic isotopes as compared to the U-cycle. In fact, calculating through the thorium diagram it is found that the Th-cycle yields only 2.7% waste in the form of ^{237}Np, which could be either be removed using chemistry or bred further into fissile plutonium. In the latter case, the waste would be 4.56 ppm americium per thorium atom burned through the cycle. In comparison the U-cycle (given the spectrum used in this reactor design) reveals approximately 16.9% americium per ^{238}U burned and a further 21.9% ^{237}Np per ^{235}U atom burned. Thus, the waste produced as a result (not considering fission products, which are approximately equal in the different cases) of the Th-cycle is far less problematic compared to that resulting from the U-cycle (even in the closed cycle case).

3.2. Salt type and justifications

The choice of salt type in the Seaborg Technologies Reactor is based on an initial study of a range of possible candidates. Two promising candidates, in terms of neutronics, were found for the given design are FLiBe and FLiTh. The Seaborg Technologies Reactor uses a FLiTh salt, assuming a ^{7}Li enrichment of 99.95%, as FLiTh is favoured by several arguments. Neutronically, this salt is similar to the more commonly considered FLiBe, as can be inferred from figure 13.

A major argument against the use of FliBe is the toxicity of beryllium and the strict regulations regarding its use. This is a concern for both the design study phase and the long term maintenance.

In addition the main benefit of using beryllium as opposed to other elements relies on the fact that the neutron absorption is vanishing. However, neutrons absorbed in the thorium that replaces beryllium in the present design are by no means lost. On the contrary, they breed fissile material to be exploited later in the reactor cycle. It should be noted that both beryllium and highly enriched ^{7}Li are expensive materials. Approximately 1400 kg lithium will be needed as the initial fuel for the SWaB reactor, thus the ^{7}Li expense is expected to be a significant cost driver of the pilot plant.

The chemical composition on which the Seaborg Technologies reactor is based is: 78LiF-22ThF_4. This mixture corresponds to the eutotectic ratio which is chemically favourable due to the low melting point: 568 °C [5]. The viscosity of the salt ranges from 4.2 cP to 15.6 cP in the temperature range of interest: 700-900 °C.

3.3. Fuel Reprocessing Required

3.3.1. Fuelling With Spent Fuel

Figures 14 and 15 show the schematics of the chemistry reprocessing. The pilot plant is planned to run on a mixture of low enriched uranium (LEU) and thorium, whereas the full scale production plant is expected to receive spent fuel elements from commercial LWRs, which will be processed and loaded into the core. The main process is based on the FLUOREX[6] reprocessing method. This method is well documented, and is based on a technology that is suitable for removing fission products from molten salt during operations. The process backbone is an improved fluoride volatility[7], where plutonium is never isolated from neptunium, and protactinium is never isolated from uranium by which proliferation issues are avoided.

The system is fed shredded steel cladded spent nuclear fuel. A thermal decladding is carried out with H_2 and O_2 in which the cladding and volatile fission products are removed (I, Xe, Kr)[7]. After this process, the waste is ground (in an on-board grinder) and injected into a flame reactor with dilute F_2 in fluidized alumina. The volatile fraction with UF_6 is distilled under pressure (twice) in order to remove NbF_5, MoF_6, TcF_6, and RuF_6. The UF_6 has a decontamination factor, DF $\sim 10^2$. After conversion to its oxide, it can be vitrified and stored or sent to enrichment.

By running cycles with an increasing F_2 concentration adjusted with helium, the remaining UF_4 and PuF_4 can be converted to UF_6 and PuF_6 and mixed with the molten salt DF $\sim 10^2$. The remaining non-volatile fission products, such as CsF and SrF_2, are left in the flame reactor and can be sent to a hydrogenation reactor together with the other non-volatile fission products. Under high temperature and mixed with H_2, H_2O, they are converted to their oxides and then vitrified for storage.

The PuF_6 is contaminated with NpF_6, UF_6, and other actinides, making it proliferation resistant. The mixture of PuF_6 and contaminants enter into a hydrogenation reactor and are treated with H_2 in order to convert it to PuF4 and pumped to the molten salt core.

The uranium hexafluoride is processed through rectification, followed by two successive sodium fluoride traps similar to those used for molten salt processing in order to remove more volatile fission products.

3.3.2. Continuous Reprocessing of the Fuel Salt From the Core

As shown in figure 14, the salt is processed by three types of chemical cleanup: a continuous helium de-gassing where xenon and krypton are removed; a novel fluoridation method that fluoridates most of the plutonium, which is recycled with uranium and neptunium and separated from volatile fission products through consecutive sodium fluoride traps; and a liquid bismuth extraction[8, 9] that removes and recycles actinides and separates lanthanides.

The helium degassing is done by continuously bubbling helium through the molten salt, either in the reactor or in a loop. The noble gasses are removed and stored in storage tanks for a few days in order to let short lived isotopes decay to solid decay products. After some time of decaying, the container will be cooled to extract noble gasses into a cryo-cooling unit

where helium is recovered and krypton and xenon are liquefied and transferred to storage. Depending on the contamination, storage for approximately one year is expected, after which the xenon fission products have decayed to stable isotopes and can be distilled from krypton through the cryo unit and safely released.

The fluoridation[6] is done in a flame reactor where Pu, U, Np, Nb, Ru, Te, I, Mo, Cr, and Tc are fluorinated so that they become volatile. The reactor will be made of a chromium free nickel alloy such as Ni-201[1]. The flame reactor processes 40 L per day, and will work in conjunction with sodium fluoride traps in absorption-desorption cycles (the NaF is 12-20 mesh salt and run for one hour cycles of absorption at 100°C and desorption at 400°C). There are three successive NaF traps which makes the system able to isolate Nb, Ru, Te, I, Mo, Cr, and Tc with a DF $\sim 10^5$ and a uranium recovery of more than 99.9%[10]. Seaborg Technologies is currently working on improving the fluorination process to gain better yields of PuF_6 and UF_6. This is process is being patented, so the details of this are omitted from this document.

The liquid bismuth extraction[6] proceeds in two stages. First the actinides: Pu, Pa, Am, and Cu are extracted with liquid bismuth containing metallic thorium. The metallic thorium assists as a reducing agent so the actinides are extracted from the molten salt and enter the molten bismuth. After the bismuth extraction an anodic oxidation is carried out in cleaned molten salt $LiFThF_4$, where thorium is reduced at the cathode to metallic thorium and the actinides are oxidised to AcF_3.

The extraction is repeated but using a higher thorium content in the bismuth pool, resulting in lanthanide extraction [6]. The back extraction of lanthanides is performed in a LiCl–KCl pool at 500°C by anodic oxidation where the lanthanides are oxidized to their chlorides and metallic lithium is formed at the cathode. The chlorides are separated by acid-base properties and reacted with H_2O to form lanthanide oxides ready for vitrification and storage. The byproduct HCl is used to react with metallic lithium to give LiCl.

Only fission products will leave the molten salt system, and when they are separated from the salt stream they are treated in a hydrogenation reactor, leaving fission products as oxides.

The chemical processing system is designed to be housed in a 40' container and will be able to sustain itself if electric power is supplied during the lifetime of the power plant.

3.4. Method of Start-up

The SWaB is designed to use spent nuclear fuel from commercial light water reactors for continuous refuelling. As discussed in section 3.1.2 the fission products and the bulk of the uranium are separated during reprocessing, leaving behind approximately equal amounts of plutonium and uranium as fuel which are contaminated by minor actinides. This is supplemented with thorium, which over time are bred into fissile U_{233}.

However, for regulative reasons as well handling considerations, we suggest that the pilot plant be fuelled with LEU. As shown in table 2, this should be relatively straightforward as long as the thorium fraction is adjusted accordingly. The fuel salt will be mixed in the chemical reprocessing system from where it is pumped into the (pre-heated) overflow system and into the core.

3.5. Method of Refuelling

Fresh fuel will be continuously added using the on-board reprocessing unit. The reprocessing unit will have a storage of sufficient fresh fuel for the lifetime of the reactor core (which is replaced regularly throughout the power plant lifetime).

3.6. Fuel Availability

Spent nuclear fuel from commercial light water reactors form the basis of the fuel for the envisioned power plant. However, the SWaB will be able to run on a large variety of other fuel types, as shown in table 2.

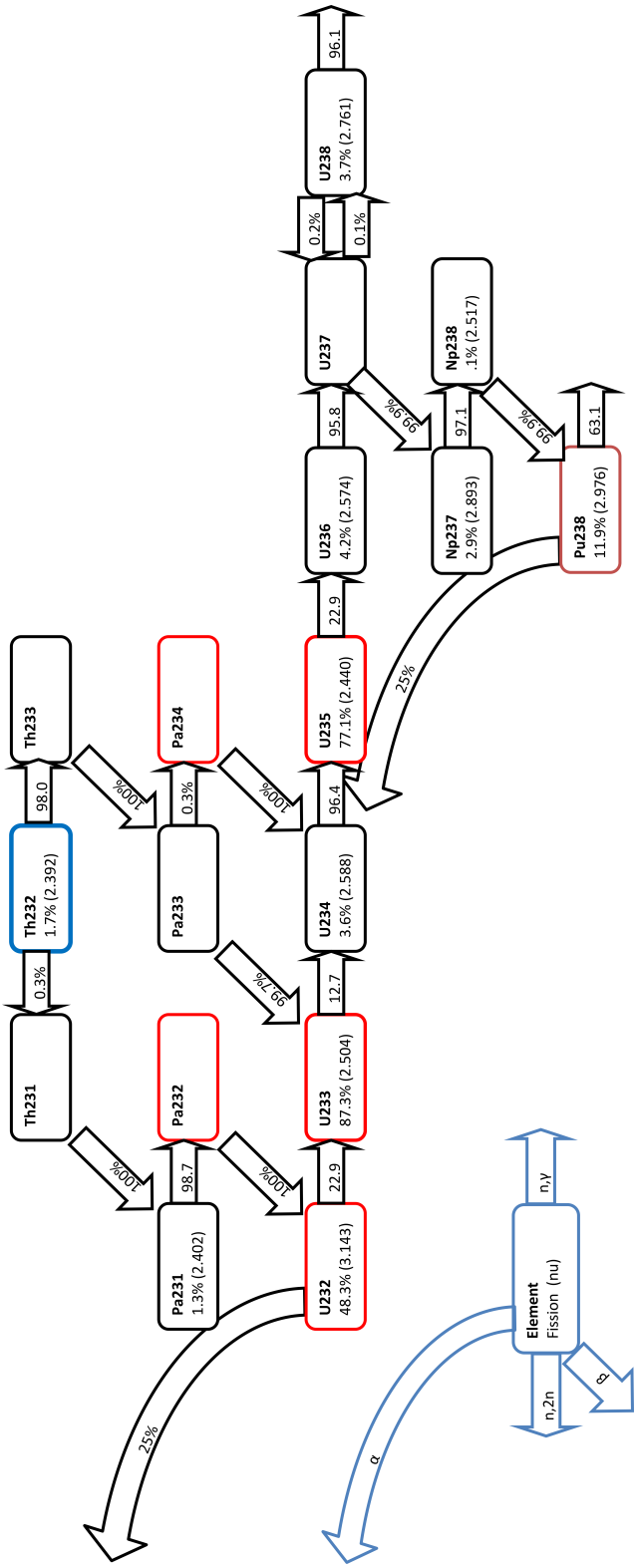

Figure 10: Thorium cycle

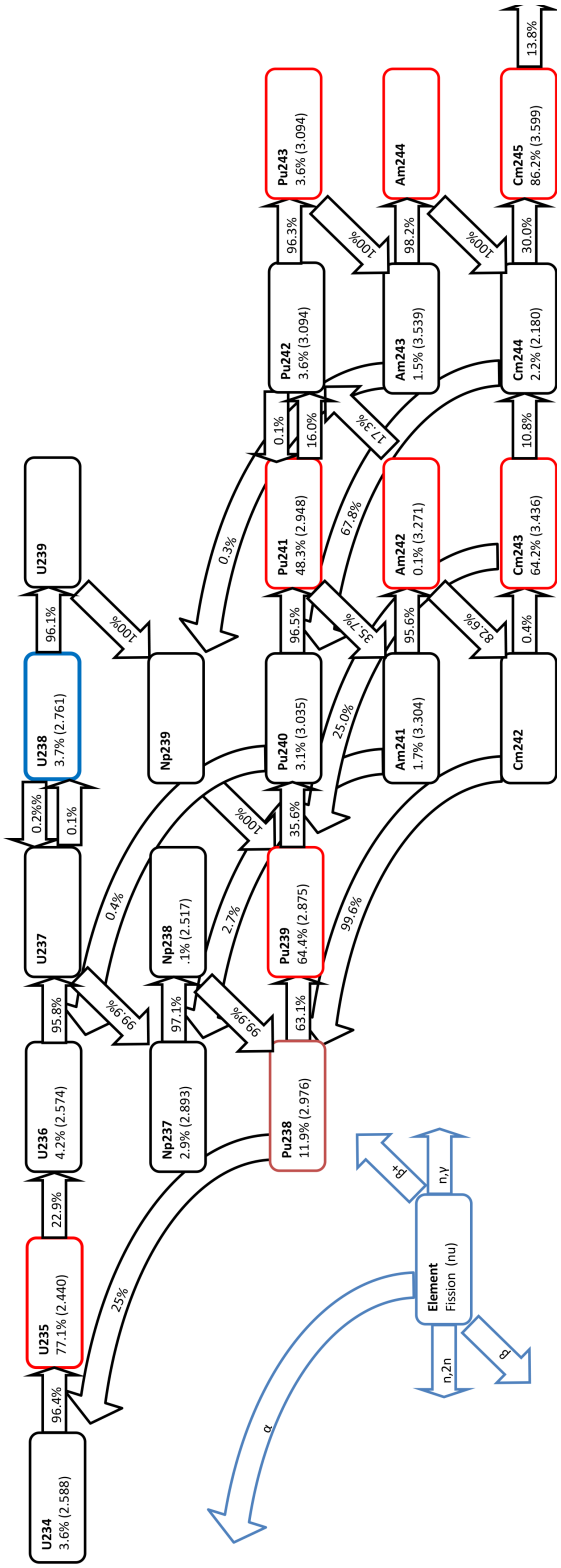

Figure 11: Uranium cycle

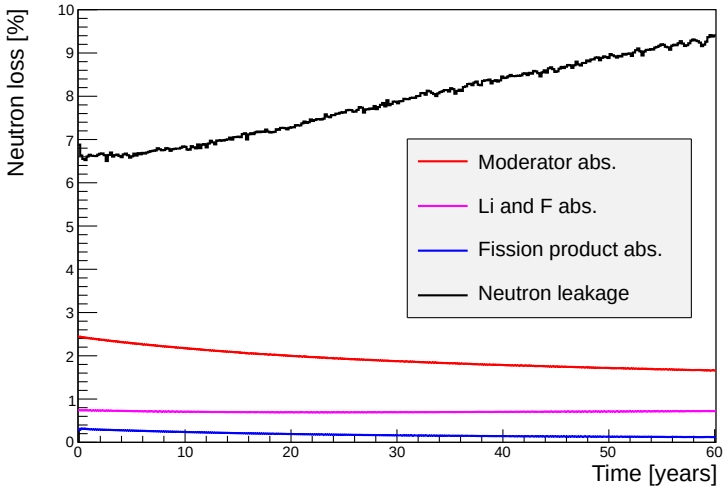

Figure 12: Overview of the neutron loss from the fuel cycle by different means. To run a truly closed fuel-cycle, the sum of neutrons lost from the cycle must be less than the excess neutrons produced in the fuel cycle (i.e. $\eta - 1$=5.9% - the current design does not supply a closed fuel cycle). The raise in neutron leakage over time is explained by changes in the actinide composition - see figure 9.

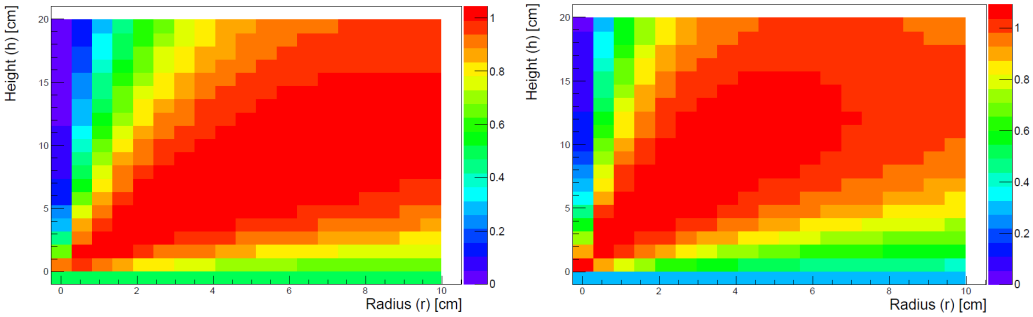

Figure 13: k_{eff} (colour scale) dependence on the hexagonal size (h) and salt containing cylinder radius (r) in a infinite assembly for FLiBe (left) and FLiTh (right). The simulations are based on ^{235}U as fissile and a ^{232}Th as fertile materials and illustrate that the performance in terms of reactivity of the two fuel salts is similar.

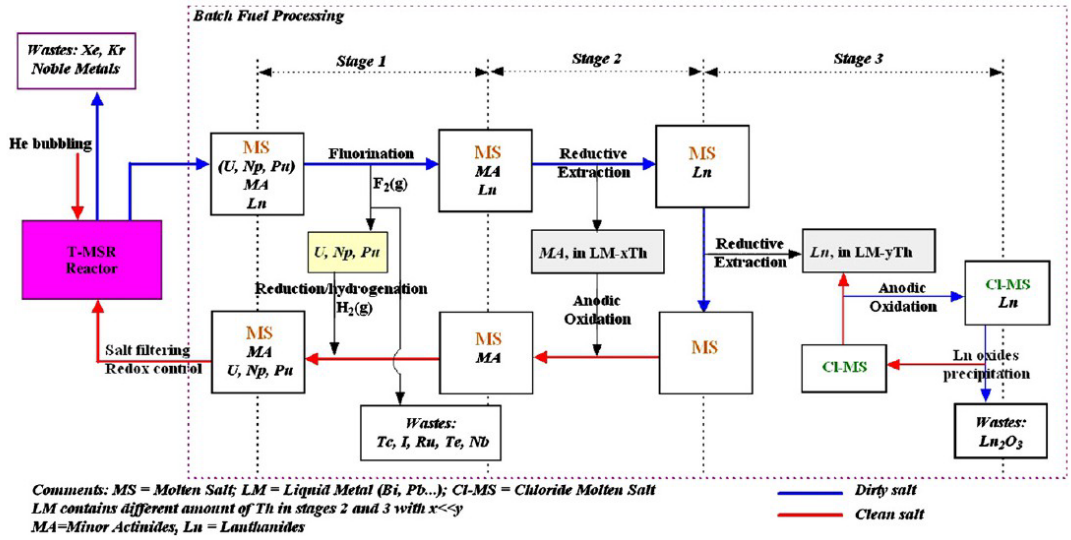

Figure 14: Schematics of the online fission product separation. Figure taken from [8].

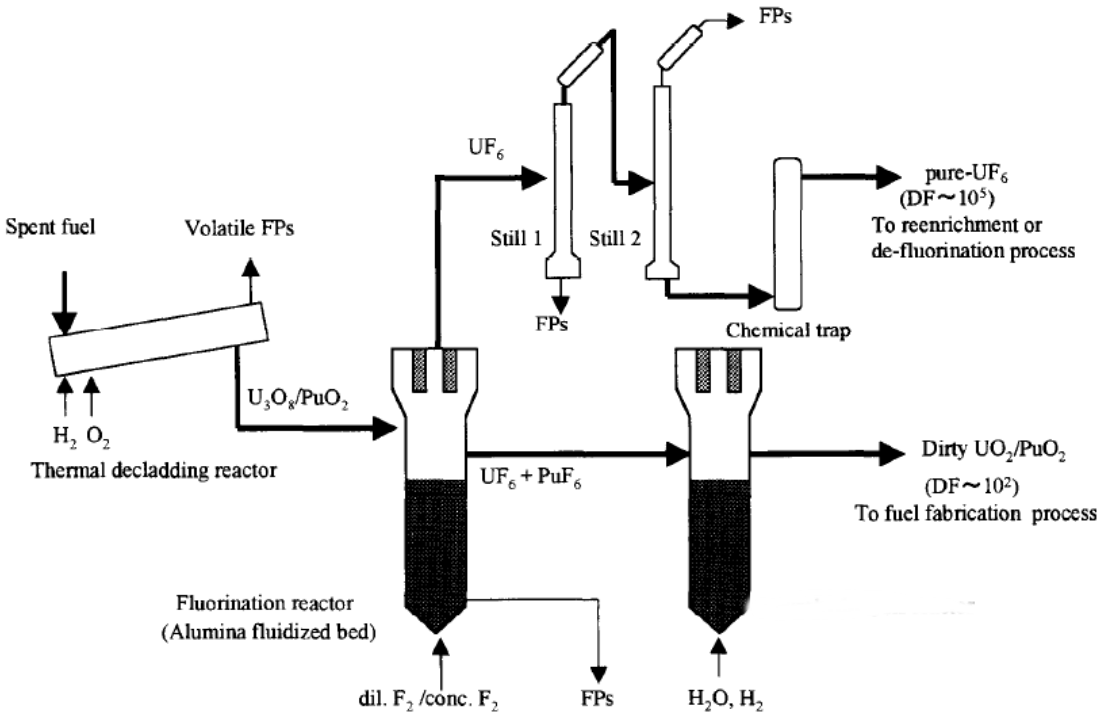

Figure 15: Schematics of the spent nuclear fuel processing. Figure taken from [7].

4. Waste

4.1. Type of Waste

The waste generated by the Seaborg Technologies pilot plant has not been studied in terms of actual burnout as the power plant simulation section handles the commercial case where spent nuclear fuel is used and as the pilot plant is planned to use LEU.

The nuclear waste from the SWaB falls into two categories: fission products and actinides.

The fission products will be continually extracted from the fuel salt through the on-board reprocessing system where it is vitrified and deposited in an underground storage container. Approximately 18.5 kg per year of fission products is produced (at 50 MW), of which 2.7 kg are noble gasses and 10-11 g is tritium. The isotope composition of the fission products produced is similar to that of MOX fuel.

When considering the waste produced by the SWaB power plant, it should take into account that it will be fuelled with waste. Effectively the waste produced by the SWaB power plant is the difference between its initial fuel and the actinides and fission products produced/burned while running. From this point of view, over the 60 year lifespan of the SWaB power plant, approximately 1 tonne of plutonium is converted to 1 tonne of fission products. Long term, the activity of the fission products is dominated by a half-life of 30 years, whereas plutonium and decay daughters have lifetimes on the order of hundreds of thousands of years; i.e. final storage of fission products is a solvable problem, whereas depositing transuranics for hundreds of thousands of years poses significant challenges.

Some thorium is removed over time to allow for addition of 'new' spent fuel (since the fuel salt volume is kept constant), see figure 9.

4.2. Storage & Disposal

The CRS will extract fission products and deliver separated, vitrified waste, simplifying storage and disposal. The fission products should be left in an underground, on-site storage (which is connected to the CRS), until they have decayed to a level where local regulations allow them to be moved to dry cask storage at an existing nuclear waste depository.

During operation, a fair amount of thorium will be removed from the reactor. After a short on-site storage, it is expected that this thorium can either be transferred to a new SWaB reactor module or disposed of.

At the end of the reactor lifetime, the uranium and thorium in the reactor has a high burn value due to huge build up of ^{233}U, and this will be of high value for future reactors. The remaining actinide content in the fuel salt can be extracted by the CRS and must be deposed of as conventional nuclear waste. It should be noted that at the end of the reactor lifetime, the amount the transuranic waste will have been reduced by approximately one tonne compared to what has been fuelled in.

5. Method of Material Movement & Specific Requirements

The Seaborg Technologies Wasteburner reactor is designed to be isolated underground and run without human access throughout its operational lifetime. For this reason, it will contain as few moving parts as possible, relying on gravity and convection where possible. For the pilot plant it may be desirable to allow for easier access in order to study conditions during the trials.

6. Scalability

We propose a pilot plant with a design power of 50 MW_t. It is our opinion that this strikes a good balance between usefulness and practicality for a pilot plant. The production version will be scaled up to 150 MW_t or 250 MW_t, in order to improve the economy of the plant and to reduce neutron leakage, potentially sustaining a closed thorium fuel cycle.

Corrosion and swelling of the graphite moderator are expected to constitute the main limitations on the operational lifetime for the core of this class of MSR. Since these effects scale with operational power, the lessons learned from a pilot plant with a much lower power output would be of significantly less valuable.

7. Safety, Potential Failures, and Remediation

The design of the SWaB is in its early stages and all failure scenarios have been assessed through hypothesis, estimates, and rough simulations. All the scenarios listed below (and more) need much deeper investigation during the next stages of the design phase.

The key safety feature in the Seaborg Technologies Reactor design is the overflow system (OS). This device will passively dump the fuel to the dump tank in most critical failure scenario. The system is designed such that all fuel salt flows by gravity into the OS from where it must be actively pumped back into the reactor core. If the fuel salt is not pumped actively back into the core (by choice or due to loss of operation power (LOOP)), if the salt overheats, or in case of a sudden pressure change, it will be dumped to the dump tank where it is cooled by passive means. The salt can be electrically heated in the dump tank and pumped back into the OS and onwards back into the reactor to restart operation.

- Pump failure: If cooling pumps fail for an extensive time period the reactor will shut-down it self (overheating scenario). If the fuel pump is stopped, the fuel is flushed to the dump tank by gravity. This is an undesired scenario, as the fuel has to be reheated and pumped back into the reactor core to restart operation. However, this is not considered an accident scenario.
  Remediation: The fuel salt pump is one of the most vulnerable and most difficult to replace components of the plant design. Therefore, the fuel loop is best fitted with redundant pumps installed in parallel so that a failure of a single pump will not interrupt normal operations. In the unlikely event of permanent failure of all redundant fuel pumps, the reactor container would have to be replaced. The primary coolant salt

pump could be housed in a intermediate radiation zone, where a replacement will be possible during a maintenance shut down. The secondary coolant salt pump will be housed on the surface and is therefore easily accessible for maintenance.

- Overheating: The core is designed with a negative temperature coefficient (see figure 5). In case of overheating of the core, the reactivity will decrease and consequently power production will decrease. If the situation is not acted on (by adjusting the fine tuning rods), the moderator will heat up and start expanding. This could potentially increase the reactivity, thus increase the power level causing an automatic fuel dump.
  Remediation: Overheating is self-correcting. Should this for some reason fail and the fuel is dumped, the dump tank pump can be activated to restart the reactor.

- Prompt criticality: The reactor is designed to have a maximal reactivity of .5$. It is not controlled by removable control rods and as the fuel salt has only a very small ratio of burnable poisons to fuel, significant reactivity changes are not expected. Nevertheless, the prompt criticality scenario should be considered. Assuming that prompt criticality is suddenly reached, the rapid expansion of the fuel salt in the inner core will cause an immediate reduction in k_{eff} (negative temperature coefficient) and the inner part of the core will be flushed rapidly through the overflow system. Though the reactor is designed to operate below atmospheric pressure, the container vessels are all designed to withstand high pressure and temperature.
  Remediation: Prompt criticality is classified as a severe accident and will most likely result in decommissioning of the core. However, it is not expected to result in any leakage of fuel from the 2nd barrier of the reactor container (see figure 1).

- Failure to dump: As the reactor does not rely on a salt plug (though one exists) but rather on an overflow system, it is difficult to imagine a scenario where the fuel should fail to dump. However, for the sake of argument, let us consider the extreme case of severe sabotage or extreme natural disaster has blocked the draining pipes from the overflow system and disabled mechanical/electronic safety systems, while the reactor needs to be dumped. One could imagine that the inner reactor vessel might melt during such an extreme scenario and gasses from the salt could evaporate. However, both fuel and gasses would remain contained within the second barrier in the reactor pressure vessel. The B_4C 2nd barrier is directly connected (at the bottom) to a secondary dump tank which is used only in case of failure of the primary dump tank.
  Remediation: Core vessel meltdown as a result of failure to dump would result in permanent shut down of the entire power plant and make decommissioning costly. This far-fetched scenario is likely the worst imaginable accident, yet no release of radioactivity outside the second barrier is expected.

- Chemical reprocessing system (CRS) breakdown: Should the CRS fail, the reactivity will decrease over time, due to lack of refuelling capability. The core fine control rods can adjust the reactivity by .88$, which means that the reactor can be kept critical for 1-3 months depending on the fuel cycle state (see figure 4), while replacing the CRS system.

Remediation: The CRS system will need replacement. It is designed to be removed and replaced independently of the rest of the reactor, and is therefore foreseen to be housed in a separate container. After the CRS is brought back on-line, fission products and fuel levels will over time automatically adjust itself to the original equilibrium state, as before the CRS system was taken off-line.

- De-gassing breakdown: If the de-gassing system breaks, it is expected that the reactor can be kept critical for approximately 6 months while replacing the de-gassing system. Some of the gasses from the salt will leak into the dump container, which will have to be flushed once the de-gassing system is back on-line.

- Fuel pipe breaks (inside 2nd barrier): In the case of a core fuel pipe break, almost all fuel salt is contained within the 2nd barrier (the B_4C - Ni alloy container), which is connected to the secondary dump tank should fuel be spilled from the leakage. In this case, reactivity will decrease (negative void coefficient) and shut down the reactor and the fuel will be dumped to the dump tank.
  Remediation: As no fuel salt will leak from the reactor in case of this failure, it is not considered an accident scenario. However, the reactor core will have to be retired early (note that the core is expected to be replaced regularly).

- Fuel pipe breaks (outside 2nd barrier): In the more severe scenario a CRS pipe breaks (outside the 2nd barrier) and fuel might leak into the inner dome. However, the CRS piping system only contains small amounts of fuel at any given time, thus only small amounts of fuel can leak in this way.
  Remediation: The pipes are isolated with granulated vermiculite which again is housed in a secondary casing. If the secondary casing fails, the molten salt will leak onto the floor where there is a drain to the secondary dump tank.

- De-gassing pipe or container breaks: The container is filled with helium, and the xenon and krypton in the helium gas is recaptured from the container by cycling it through the cryo unit. If that is not possible, a secondary system can capture xenon and krypton through a charcoal filter system.
  Remediation: The reactor should be shut down for an extended time period while the dome gas is being purified in the cryo system such that the gas container or pipe can be be replaced.

- Primary or secondary salt pipe breaks: The primary coolant salt will be activated by the neutron irradiation from the fuel salt during normal operation of the reactor. As the primary coolant loop does not penetrate the concrete lid of the reactor, a pipe break would not cause a radiation hazard. The most extreme consequence of a break in either loop would be complete primary cooling failure of the reactor, causing a passive shut down. This would however not damage the reactor or cause an emergency, as explained previously in this section. The secondary coolant loop does penetrate to the surface, but this salt will not be activated during operations, and as such a break here would not cause a radiation hazard either. Since the neutronic characteristics of the

secondary cooling salt are irrelevant, a non-toxic salt can be used here.
Remediation: Parts of the primary coolant loop are inside highly irradiated zones. A severe break would require an extended shut down to repair, or a full replacement of the reactor container. The secondary coolant loop will be accessible for maintenance during a shut down.

8. Operations & Decommissioning

8.1. Controls

By exploiting the CRS, use of absorbing control rods can be avoided. The reactor is controlled by four movable graphite rods (fine tuning rods) which can alter the reactivity and thereby adjust the power output, see section 1.4. The core has an absorbing control rod, but this is only used for full shut-down.

8.2. Operational Requirements

To reduce construction costs, the turbine and electrical generator will likely not be installed at the pilot plant. The heat from the reactor can be discharged into a river or another large body of water as discussed previously. The purpose of the pilot plant is to operate the SWaB under a number of different scenarios. As mentioned previously, the main control mechanism of the SWaB is to change the temperature of the return salt in the secondary salt loop.

8.3. Decommissioning Process

The SWaB reactor is highly modular, which enables Seaborg Technologies to transport the critical modules to a centralised dedicated decommissioning facility where they can be disassembled and either reused or decommissioned. This is expected to be a much more cost effective procedure than conventional decommissioning. Fuel salt disposal is discussed in section 4. For the first version, the core structures have a four year life expectancy, however it is the ambition to extend this to 10 years in later versions. If possible the graphite of the moderator will be re-used - studies are ongoing within the nuclear reactor community to establish procedures to recycle irradiated graphite.

After removal of critical components and fuel from the power plant site, and the plant structures should be decommissioned following normal regulations.

As centralised decommissioning of critical components is unavailable for the pilot plant, the decommissioning of the pilot plant is expected to be significantly more costly than what is foreseen for later commercial versions. However, as the decommissioning cost is a driver for the full plant cost, the decommissioning of the pilot plant constitute a very important exercise which will be essential in the development progress towards a commercial product.

References

[1] Sridharan, K.; Allen, T. R., Corrosion in molten salts, Molten Salts Chemistry pp. 241-267 (2013).

[2] Waters L S, McKinney G W, Durkee, J W, Fensin M L and Hendricks J S et. al. The MCNPX Monte Carlo radiation transport code. *AIP Conf.Proc.* **896**, 81-90 (2007).

[3] X-5 Monte Carlo Team, MCNP – A General Monte Carlo N-Particle Transport Code, Version 5, *LA-UR-03-1987* (1987).

[4] Antcheva I et al, ROOT — A C++ framework for petabyte data storage, statistical analysis and visualization. *Computer Physics Communications* Anniversary Issue; **180**, Issue 12, Pages 2499-2512 (2009).

[5] Gheribia A.E. et al, Prediction of the thermophysical properties of molten salt fast reactor fuel from first-principles, Molecular Physics, Vol. **112**, Nos. 9–10, Pages 1306-1312 (2014).

[6] Hiroaki Kobyashi et al., Fluorex reprocessing system for thermal reactors cycle and future thermal fast reactors (coexcistence) cycle, Progress in nuclear energy, Vol 47, No. 1 – 4, pp. 380-388 (2005)

[7] M. Kamoshida et al., A New comcept for the nuclear fuel recycle system: Application of the fluoride volatility reprocessing, Progress in nuclear energy, Vol 37, No. 1 – 4, pp. 145 – 150 (2000)

[8] S. Delpech et al., Reactor physic and reprocessing scheme for innovative molten salt reactor system, Journal of fluorine chemistry, p 130, 11 – 17 (2009)

[9] Ferries et al., Equilibrium distribution of actinide and lanthanide elements between molten fluoride salts and liquid bismuth solutions, J. inorg. Nucl. Chem., Vol. 32, pp 2019 to 2035 (1970).

[10] CEA-EDF report "Filie' re sels fondus", CEA no. 76-51 (1976).

[11] H. A. Schmutz et. al., Tritium Formation and Mitigation in High Temperature Reactors, technical report, Idaho National Laboratory (2012), http://www.inl.gov/technicalpublications/Documents/5581212.pdf

Appendix

A. Simulation Steps

The initial waste and burnout procedure is as follows:

1. MCNPX is used to simulate 4.5% enriched UO_2 fuel being burned to 45 GWd/tHM over the course of 3 years in a conventional PWR (17x17 assembly type).
2. The fuel (from above) is stored for 10 years.
3. Fuel is put into the flame reactor, where:
    a) 99% of all elements below $Z = 90$ are removed.
    b) 100% of all noble gasses and oxygen is removed.
    c) 98.94% of all uranium is removed.
    d) The fuel is mixed 1:12.8 with ^{232}Th (the critical solution)
    e) Li (99.95% enriched) and F are added until the composition is: 78LiF-22AcF_4
4. (loop) MCNPX simulates and time evolves 73 days of running at 50 MW_{th} power.
5. The MCNPX output file is passed to c++, where:
    a) All noble gasses are removed, along with hydrogen.
    b) The reprocessor simulation extracts 99% of all fission products contained in 40 liters of salt each day of the run.
    c) Li and F levels are re-balanced to 78LiF-22AcF_4 (new Li is 99.95% enriched)
    d) The original PWR fuel is time evolved to the current step time (current run time plus 10 years), and step 3a-3c and 3e are carried out.
    e) The fresh fuel salt is added, while (denatured) thorium is removed such that the fuel volume is constant (some FPs were removed) and such that the prediction yields keff=1.0005.
6. A new MCNPX input file is written.
7. Repeat from step 4