# MOLTEN SALT REACTORS: A NEW VISION FOR A GENERATION IV CONCEPT

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## Abstract

Molten Salt Reactors were developed at Oak Ridge (ORNL) from the late 1940s to the early 1970s, highlighted by two successful test reactors. The Molten Salt Breeder Reactor (MSBR) evolved into a single fluid, graphite moderated design. Until very recently, this 1970s design version has been taken to be the starting point for any resurgence of the Molten Salt concept. This paper will show that a Molten Salt Reactor can in fact take many different forms. Through new solutions and applying new technology, it is hoped that an improved design can be brought to such a level that it can no long be ignored as a practical ally in the resurgence of nuclear power.

## 1. Introduction

A Molten Salt Reactor (MSR) is one in which fluorides of fissile and/or fertile elements such as $UF_4$, $PuF_3$ and/or $ThF_4$ are combined with carrier salts to form a fluid. Single Fluid designs have both fertile and fissile combined in one salt, whereas the lesser known 2 Fluid design has separate salts for fissile ($^{233}UF_4$) and fertile ($ThF_4$). Typical operation sees salt flowing between a critical core and an external intermediate heat exchanger. A secondary coolant salt then transfers heat to steam or closed gas cycle. The vast majority of work has involved fluoride salts as corrosion resistant alloys have been shown to be compatible with these salts. Chloride based salts have also been proposed, especially for fast breeder designs, but have unique problems and no operational experience to draw upon. Designs specifically for the thorium-$^{233}U$ cycle using fluoride salts have also been termed Liquid Fluoride Thorium Reactors (LFTR).

Fluid fuel reactors and MSR in particular have numerous operational and safety advantages over solid fuel designs. A detailed review is beyond the scope of this presentation but briefly:

-Fluid nature of the fuel means meltdown is an irrelevant term and allows the fuel salt to be automatically drained to passively cooled, critically safe dump tanks.

-Most fission products quickly form stable fluorides that will stay within the salt during any leak or accident. The volatile fission products such as the noble gases and noble metals come out of the salt as produced. Noble gases simply bubble out and are stored outside the reactor loop. Noble and semi noble metals will plate out on metal surfaces and can be collected by replaceable high surface area metal sponges within the loop.

-The continuous removal of the noble gas Xenon means that there is no "deadtime" of the reactor after shutdown or a power decrease that solid fueled reactors must deal with due to the production of $^{135}Xe$ from the decay of $^{135}I$. As well, no excess reactivity need be in place to deal with such events.

-Most MSR designs have very strong negative temperature and void coefficients which act instantly, aiding safety and allowing automatic load following operation.

-No pressure vessel is needed as the salts run at atmospheric pressure. No water or sodium means no possible steam explosion or hydrogen production within the containment. In designs without graphite moderator, there is not even combustible material present.

- Fuel concentrations are easily adjusted on a continuous basis meaning no excess reactivity and no need for control rods or burnable poisons. Shutdown rods are often included but even these are not necessary given the ability to drain fuel out of the core to storage tanks.

-Utilization of the thorium to $^{233}U$ cycle produces several orders of magnitude less transuranic wastes than a conventional once through cycle and about one order of magnitude less than a U-Pu fast breeder (based on 0.1% losses during fuel processing). This leads to waste radiotoxicity being less than equivalent uranium ore levels within a few hundred years.

-Fuel processing and utilization of thorium permits break even breeding with ease and ability to reach a breeding ratio of 1.06 or even up to 1.13. Adding $^{238}U$ to denature the uranium content and still break even is also possible.

-Break even operation requires approximately 800 kg of thorium per GW(e) year added simply as $ThF_4$. Startup fissile requirements can be as low as 200 kg/GW(e) or as high as 5.5 tonnes in harder spectrum designs, with 700 to 1500 kg more common. Thorium startup inventory varies from 50 to 200 tonnes.

-Thorium is 3 times as abundant as uranium. Proven reserves are large even with the small current industrial use of thorium and lack of prospecting. As example, a single new deposit in Lemhi Pass Idaho has added 600,000 tonnes to the world's proven reserves of 1.2 million tonnes. The USGS quotes a price of 27$/kg for thorium nitrate and 80$ to 100$ for high purity thorium oxide.

-Without fuel processing, MSRs can run as simple converters with excellent uranium utilization.

-Offer many advantages for the destruction of transuranic wastes from traditional once through reactors. TRUs may also be used as startup fissile inventory for the thorium to $^{233}U$ cycle in many designs.

## 2. Background

Molten salt reactors were developed primarily at Oak Ridge National Laboratories beginning in the late 1940s. Almost 30 years of funded research and development followed a design evolution leading to the adoption in the late 1960s of what is known as the Single Fluid, graphite moderated Molten Salt Breeder Reactor (MSBR). What is important to realize however, is that this evolution was guided by goals and limitations that are far different than would now exist. In particular, the overwhelming priority given to the MSBR program was a minimization of the doubling time, the time to breed the startup inventory of the next reactor. The two routes for this are decreasing the startup fissile inventory and increasing the breeding ratio. This mandated priority was due to the early belief that nuclear power would follow an exponential growth and that uranium supplies were severely limited. Another fact was that MSBRs main competition was the heavily funded liquid sodium cooled fast breeder whose potential doubling time has always been impressive.

In order to properly evaluate potential molten salt reactors designs, it is important to first re-establish priorities. Given the ability of these reactors to start up on wide variety of fissile material, the doubling time is no longer of any real importance. Reaching a break even breeding ratio of 1.0 and not beyond should be a high priority as this allows extremely low fuelling costs and no fissile material need enter or exit a plant after start up. This simple change alone gives great leeway to reconsider options that may be more practical but neutronically inferior, such simplifying fission product removal. An examination of molten salt design from first principles can lead to novel new solutions to unsolved problems which may further improve the prospects of this unique reactor.

Proliferation and long term waste concerns are also a more prevalent concern today. Transuranic waste production (Pu, Am, Cm etc) from the Th-$^{233}$U cycle is several orders of magnitude lower[1] than for a LWR and will remain one its greatest advantages. Proliferation concerns[1] of a Th-$^{233}$U cycle, while beyond the scope of this paper, might be considered roughly on par with other commercial reactors. Almost no plutonium is produced and it is of far lower fissile/fertile ratio than for LWR once through cycles. Weapons useable $^{233}$U is however produced but is always contaminated by significant amounts of $^{232}$U whose decay chain emits an extremely energetic 2.6 MeV gamma ray. This would aid detection and make handling nearly impossible. If deemed necessary though, a combination of depleted uranium and thorium can be used as fertile makeup to keep all uranium denatured. This complicates reactor operation somewhat but would result in designs with very high proliferation resistance.

## 2.1 The "traditional" MSBR

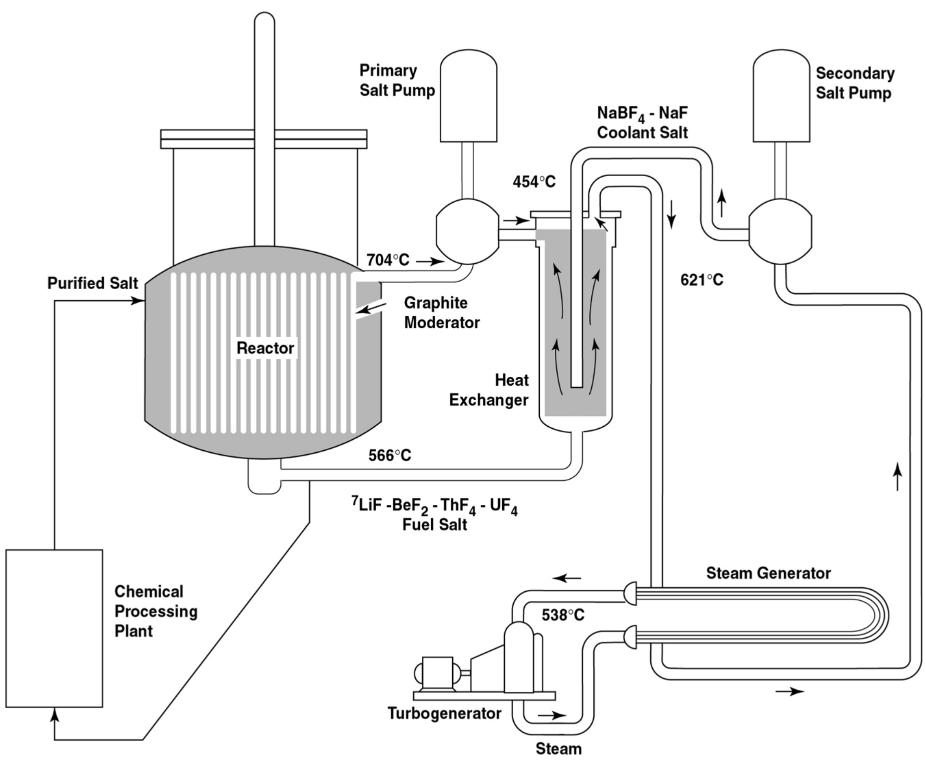


Figure 1 The 1970s Single Fluid, graphite moderated Molten Salt Breeder Reactor. 1000 MWe with a specific fissile inventory of 1500 kg. Reproduced from ORNL 4812.

Before a review of newly discovered and re-discovered molten salt designs it is useful to examine what might be viewed as the "traditional" MSBR which is the 1970s, graphite moderated, Single

Fluid design[2,3]. Both thorium and uranium are combined in a single fluid with a carrier salt ($2^7LiF-BeF_2$). The core consists of graphite blocks with small channels through which the salt flows. The salt is pumped between the core and an intermediate heat exchanger where it transfers heat to a secondary coolant. In the original design, the secondary loop then transfers heat to a steam Rankine cycle but a Brayton closed gas cycle[4] is now considered a better fit to the salts high temperature. Another newly proposed modification of the traditional system is to employ carbon based, compact heat exchangers[4,5] with can dramatically lower the out of core salt volume. The nickel alloy Hastelloy N is used for all piping and is rated for upwards of 750 Celsius with very good corrosion behaviour. The processing of fission products was to be by the liquid bismuth reductive extraction method which is briefly reviewed in the next section.

Potentially the largest drawback of this design is in terms of the significant processing needs for the salt. In a Single Fluid design, the thorium within the salt behaves very much like the rare earth fission products. This rules out the use of many simpler potential processing methods and greatly increases the complexity of the proposed liquid bismuth reductive extraction method.

Another requirement that is particular to this Single Fluid design is protactinium removal. $^{233}Pa$ is the 27 day half-life intermediate between $^{232}Th$ and fissile $^{233}U$. The moderately high neutron flux of this design would result in too high a neutron loss to $^{233}Pa$ if it were not removed from the salt with a fast cycle time, 3 to 10 days being typical. The removed $^{233}Pa$ is stored for several months to allow it to decay to $^{233}U$ which is then reinjected into the salt. This rapid removal of $^{233}Pa$ is both costly and complex. As well, it adds a significant proliferation risk as the $^{233}U$ produced in decay tanks outside the core flux can be relatively free of $^{232}U$. As will be shown however, many other molten salt designs can omit this entire procedure.

In order to evaluate new or even abandoned molten salt designs, a review of basic principles and a historical background is of benefit.

## 2.2 Salt processing methods

There are 2 types of fuel processing commonly used to increase the conversion or breeding ratio in molten salt reactors. Protactinium removal is also sometimes needed but is costly and introduces proliferation concerns and should be avoided if at all possible.

The first process is to remove uranium from the fuel salt. This is typically done before the salt is further processed for fission products. This is known as the fluoride volatility[6] process and has been well known since the 1950s. It is one of the main advantages of working with these salts is that by simply bubbling first HF then $F_2$ gas through the salt, the uranium content in the salt will convert from $UF_4$ to $UF_6$ which comes out of the salt as a gas. This $UF_6$ can be later converted back to $UF_4$ and re-injected into the reactor as needed. Fluorination of higher actinides such as $PuF_3$ to gaseous $PuF_6$ is technically possible but much more problematic due to corrosion issues[7].

Fission product removal is the main need and many methods were investigated at ORNL and elsewhere. Before 1964, there were various methods proposed with perhaps the most simple and attractive being salt replacement. In this, the fissile $^{233}U$ is first removed and transferred to clean new carrier salt. The used salt with fission products can be sent to long term storage or further treatment to concentrate the fission products. The drawbacks are that any contained thorium would be lost and that the best carrier salt ($2^7LiF-BeF_2$) is quite expensive as the lithium requires isotopic enrichment.

In 1964 a breakthrough was made called Vacuum Distillation[6]. In this method $^{233}U$ is first removed from the salt followed by distillation at low pressure to recover the carrier salt and leaving the majority of fission products in the still bottoms. This process would leave behind any thorium contained in the carrier salt.

In 1968 a new method was developed that could allow processing for fission products for salts with both uranium and thorium (i.e. Single Fluids). Known as liquid bismuth reductive extraction, this involved contacting a side stream of molten salt with liquid bismuth and a reducing agent such as lithium. The lithium trades places with various fission products which then entrain with the bismuth. While the process can function in the presence of thorium it is far simpler to employ if it is absent.

### 2.3 The evolution of the MSBR program

The very first molten salt reactor project was the overly ambitious Aircraft Reactor Program for the U.S. Air Force to design a nuclear powered bomber using heat transferred from a molten salt reactor to replace combustion heat in a jet engine. While the practicality of such a concept remained far from proven, the significant funding and manpower assigned to it allowed great progress to be made in terms of molten salt reactors in general. The highlight of this project was the Aircraft Reactor Experiment which was a low power test reactor but which demonstrated operation at salt temperatures up to 860 C. It used highly enriched $^{235}UF_4$ in a $NaF\text{-}ZrF_4$ carrier salt with canned beryllium oxide for added moderation.

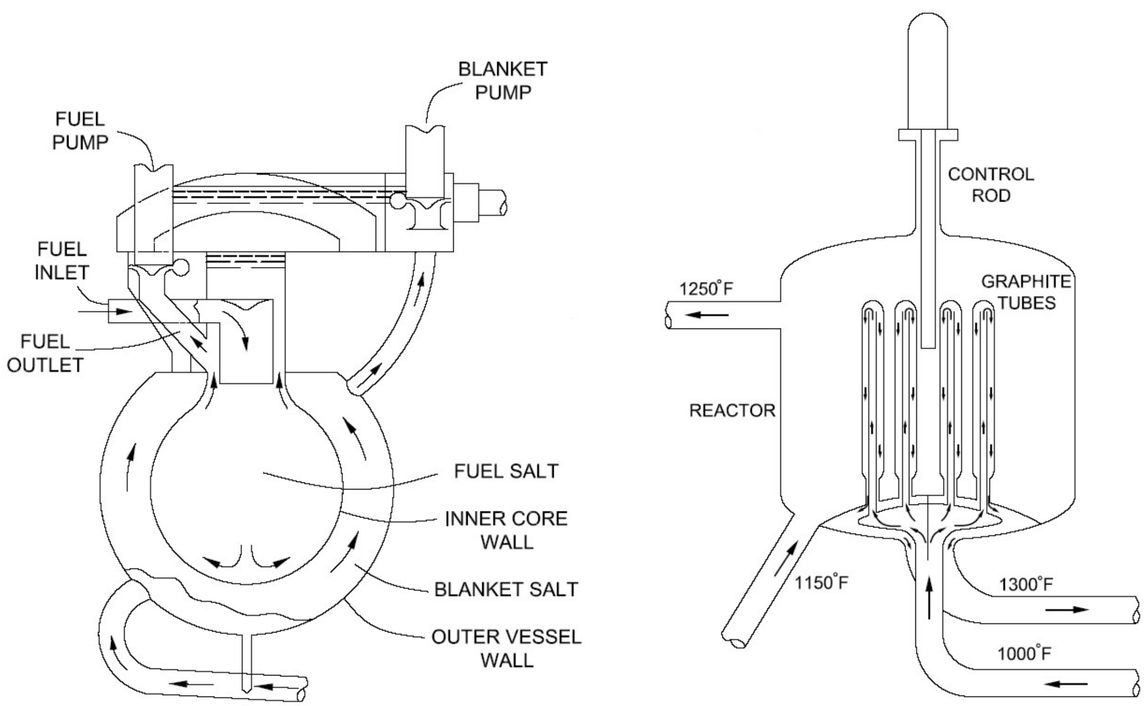


Figure 2. (Left) Depicts the 1950s graphite free, two-region concept. Reproduced from ORNL 2474. (Right) Depicts the 1960s intermixed 2 Fluid MSBR design using internal graphite plumbing. Reproduced from ORNL 4528.

In the mid 1950s a true Molten Salt Breeder Reactor program got underway at ORNL. A very simple homogeneous design was proposed[8] as either a $^{235}U$ burner reactor or a thorium breeder on the Th-$^{233}U$ cycle. It was a two zone system as shown in the left side of Figure 2. A central spherical core contains fuel salt and was separated from an outer thorium blanket salt by a 1/3 inch Hastelloy N barrier. The fuel salt contained a mix of both fissile ($PuF_3$, $^{235}UF_4$ or $^{233}UF_4$) and fertile $ThF_4$ in a carrier salt while the blanket salt contained $ThF_4$ in a carrier. In some studies, if the central core was very small, the fuel salt might lack any $ThF_4$ and would then qualify as a true 2 Fluid design.

While the simplicity of such a design was extremely attractive, in 1959 it was decided to switch focus to employing graphite moderation in order to improve the potential doubling time. It was recognized that a true 2 Fluid system can improve the neutron economy and simplifies fuel processing. The simplest 2 Fluid design would be a central core zone with fissile salt plus graphite surrounded by the fertile blanket salt (with or without graphite). The problem is such an arrangement has little power producing volume as the critical diameter would be small. ORNL workers concluded that blanket salt must also be intermixed within the central core region to allow a larger diameter core. This led to a design[9] with complex graphite plumbing that ran fuel salt up and back down graphite tubes with blanket salt in the space between tubes and around the whole core to catch leakage neutrons. The right side of Figure 2 depicts the core with just a few of the hundreds of graphite tubes depicted.

This design proved highly complex, especially due to the fact that graphite will first shrink and then expand under neutron irradiation. This led to what was termed the "plumbing problem" that was never solved to satisfaction. This basic design remained the focus for nearly a decade however, which gives testament to the advantages seen in separate fissile and fertile salt streams. It should be noted that while graphite swelling is an issue, there is no safety concern of stored Wigner energy since the graphite operates at high temperature.

Also during the 1960s, the highly successful test reactor, the Molten Salt Reactor Experiment was constructed and operated. It was an 8 Mw(th) design chosen to be a single fluid for simplicity. Almost 5 years of operation saw very few operational difficulties. Two unknown issues with Hastelloy N did surface, one was corrosion induced by the fission product tellurium and the other was irradiation damage caused by (n,alpha) reactions in nickel. Both these issues were addressed by modifying the alloy makeup of the Hastelloy but it is now recognized that Hastelloy N may have a limited lifetime if used within the full neutron flux of the core. Outer vessel and piping use should pose no problems.

In 1968 the liquid bismuth technique was brought to light that could potentially process a fuel salt that also contained thorium. Even given the great complexity of this new process, the "plumbing problem" was just too great an issue and ORNL switched focus to the Single Fluid graphite moderated design of Figure 1. In the early 1970s however, for reasons many would argue more political than technical, the MSBR program was terminated by the AEC. The Single Fluid design became the textbook design and little mention of alternatives has appeared until very recent years.

## 2.4 A resurgence of interest

The late 1970s until the late 1990s saw only modest activity worldwide. Several voices attempted to keep the concept relevant including Charles Forsberg at ORNL and Kazuo Furukawa in Japan. In recent years though there has been a resurgence of interest as the many advantages of the general design are recognized and the limited potential for improvement of other reactors has become evident.

The selection of molten salt reactors as one of the six Generation IV reactors in 2002 reactors has certainly contributed to the increase in interest. Much recent activity has also been based on molten salt reactors acting as transuranic waste burners. Most initial TRU burner work looked to modify graphite moderated designs and/or employ subcritical accelerator driven concepts. The latest work[10] points to graphite free systems being the optimal route. A technical issue in TRU burning designs is the fact the $PuF_3$ is much less soluble in most carrier salts compared to $UF_4$ or $ThF_4$. The carrier salt NaF-LiF-$BeF_2$ has recently been shown to be more than adequate and forms the basis of the MOSART[11] design out of Russia.

The most intensive new efforts have been from a group in France, centred in Grenoble which have undertaken a major modelling, design and salt chemistry program. This work has included discovery of a reactivity problem with the traditional Single Fluid MSBR. While the temperature coefficient has the needed fast acting negative term, as graphite heats up the overall temperature coefficient becomes positive. They have proposed remedies to this but they too have reached the conclusion that moving away from graphite moderation will attain the best results. Their latest design offering utilizes a 78%LiF-22%(Th+U)$F_4$ fuel salt as core, surrounded by radial blanket of LiF-$ThF_4$ in a graphite matrix. Termed the Thorium Molten Salt Reactor[12] (TMSR), the combination of high fissile concentration (5.5 tonnes $^{233}$U/GWe) and at least a partial blanket results in a high breeding ratio of 1.13 with a 6 month fission product removal rate and the ability to extend this processing time to 20 years and still break even.

Work involving molten salts has also increased in the U.S. but in a rather different way. Charles Forsberg and others are promoting the use of molten salts as simple coolants for high temperature solid fuel reactors. These designs are termed a Molten Salt Cooled Reactors[13] (MSCR) as opposed to molten salt fuelled designs. Molten salts have high heat capacity and other excellent heat transfer qualities. This lowers pumping requirements, results in smaller heat exchangers and allows large cores to have adequate decay heat removal by natural circulation of the salt. The major design constriction this work faces is assuring a negative coolant void coefficient which has proved challenging but attainable. This work could entail much engineering development that would also be relevant to molten salt fuelled designs but has undoubtedly meant a diversion of expertise and attention away from thorium fuelled MSBR designs.

## 3. Solving the 2 Fluid "plumbing problem"

All original fluid fuel reactor designs involved utilizing two zones, a central core or seed zone surrounded by a fertile blanket (i.e. thorium). For the $^{233}$U-Th cycle, the core might contain a mix of fissile and fertile in a carrier medium or in some cases only fissile. Molten salt work differentiated between these two cases by the terms "2 Fluid" for only fissile in the core and "1 and ½ Fluid" designs if the core contained thorium as well.

Early in development, the advantages of a 2 Fluid design became evident. If the core salt lacked thorium, it would be far easier to process for fission products. However a core without thorium will have a quite small critical diameter if the fissile concentration is kept high enough to limit losses to the carrier salt and/or graphite. The critical diameter is on the order of 1 meter for both pure salt cores or heterogeneous cores with graphite. ORNLs solution was to use plumbing to intermix the 2 fluids within the core zone which as previously reviewed, proved unmanageable.

A solution to this dilemma may in fact be surprisingly simple. Traditionally reactor cores are spherical or short cylinder primarily to minimize neutron leakage. With an encompassing outer fertile blanket in a 2 Fluid design, leakage is not an issue. The simple solution thus proposed is core geometry switch to increase volume while maintaining the relatively small critical diameter.

As a first approximation the critical diameter will be the ratio of the Buckling constants between the given geometries. Thus, for the same graphite and/or fuel salt combination, an infinite cylinder will have a critical diameter approximately 76.6% that of a sphere. If a specific combination of fissile concentration, graphite percentage and carrier salt gives a critical diameter of 1 meter for a sphere, then for comparison, a 5 meter long cylinder would have critical diameter of 0.77 m and a 4m by 4m slab would be 0.51 m thick.

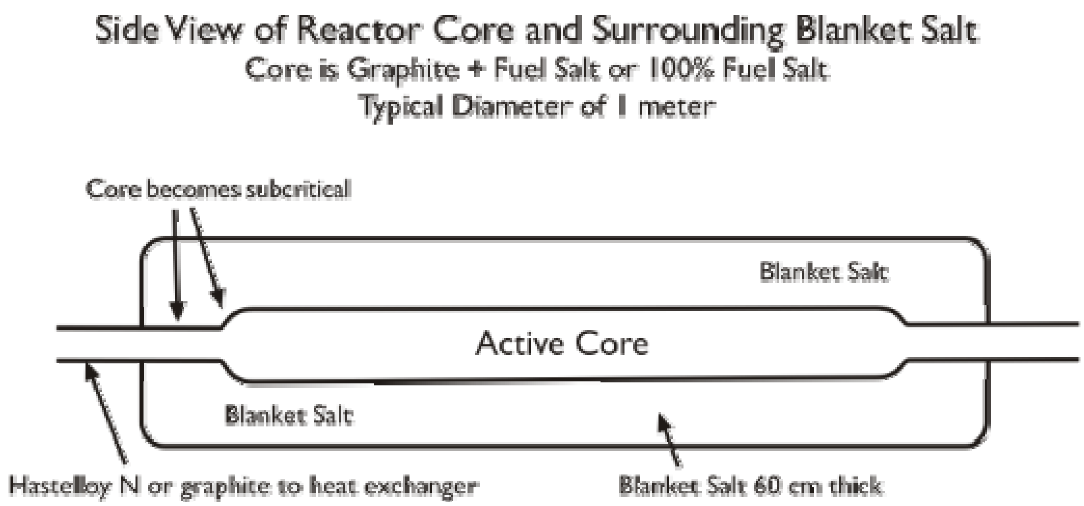


Figure 3 Generalized depiction of an elongated cylindrical 2 Fluid core with encompassing blanket salt. Inlet/outlet for blanket salt cooling are not shown.

The great advantage of going to an elongated cylinder or slab is the fact that a practical total power can now be obtained without intermixing but by simply extending the length of the core. While a barrier needs to be maintained between the core and blanket regions, this will be far less complex than the intimate intermixing of fuel and blanket salts in ORNL 2 Fluid designs. In terms of end plenums on these cylindrical cores, the simplest arrangement would be to taper the ends to a sub critical diameter while still surrounded by the blanket salt (see Figure 3). This should all but eliminate leakage of neutrons. While modelling efforts are ongoing, previous calculations from ORNL work of homogeneous designs of the late 1950s and 2 Fluid graphite work of the 1960s can be used to a significant degree to predict characteristics.

Such a design will have a strongly negative temperature and void coefficient for the fuel salt which is true for any 2 Fluid design. A major improvement over ORNLs intermixed 2 Fluid design is that the blanket should also have negative coefficients. This is due to the fact that the outer blanket acts as a very weak neutron reflector, thus lowering its density decreases this reflective quality and lowers reactivity in the core.

As with any fluid fuelled, two zone design, the leakage of core fluid into the blanket must be guarded against. The simplest method, proposed for all ORNL designs is to run the blanket fluid at a slightly higher pressure. As the blanket salt is far denser than the core salt, hydrostatic pressure accomplishes this automatically. Thus any leak through the barrier will add fertile to the core and lowers reactivity.

### 3.1 Graphite moderated version

There are some advantages to employing graphite moderation including very low fissile specific inventories and providing a built in structure to aid in the barrier between core and blanket. The much lower overall power density of graphite designs results in the need for much greater overall core volumes to attain power plant levels. This may mean multiple units per plant but this fact also brings other operation advantages. The limited lifetime of graphite due to fast neutron damage would also entail periodic replacement as is true of most MSBR designs. The small dimension and multiple units should assist in this aspect.

Using ORNL studies[14] leads to an estimate of a 100 cm diameter for a long cylinder with 0.3%$^{233}UF_4$ in fuel salt and a 20% salt/graphite ratio. Other parameters based on ORNL work are a salt power density of 400 kW/L (80 kW/L core) and an inlet of 565 C and outlet temperature of 705 C. Using the volumetric heat capacity of the salt, $\rho C_p$ = 4.69 J/cm$^3$K and a choice of a 4.5 m/sec in core salt velocity results in a 464 MW(th) power production and a core length of 7.4 meters. Connection to steam cycle at the 44.4% efficiency ORNL predicted yields a 206 MW(e) output. The Brayton gas cycle is projected to produce an even higher efficiency. Graphite lifetime would be on the order of 2 to 5 years depending upon whether flux flattening methods are employed. It is proposed that core arrangement would be horizontal for this design. The 1 meter diameter graphite core would be surrounded by a 60 to 100 cm of a 27%$ThF_4$-73%LiF blanket salt. This blanket will result in extremely low neutron flux reaching the outer vessel wall.

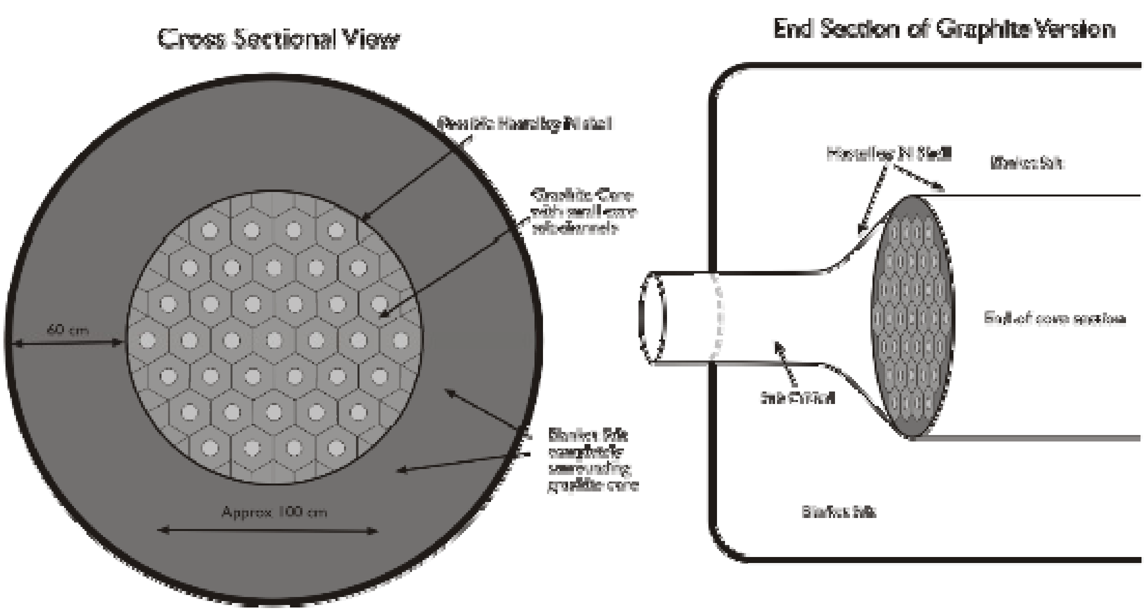


Figure 4 Cross sectional and end view of the 2 Fluid reactor using graphite moderator.

The total volume of salt and the fissile molar concentration dictate the specific inventory. For a graphite moderated design it should be possible to reach 0.15% $^{233}UF_4$ or even 0.1% or lower and still break even. Taking into account salt volume needed out of core leads to a conservative estimate of 20 m$^3$ as adopted in French studies with a lower limit of perhaps 10 m$^3$ given the use of new compact heat exchangers. These estimates give a potential lower limit of start up fissile inventory of a mere 130 kg/GW(e) with 400 kg/GW(e) being a more conservative goal. For comparison ORNL 2 Fluid work was about 700 kg/GW(e), ORNL Single Fluid 1500 kg/GW(e), an LWR is 3 to 5 tonnes/GW(e) and liquid metal cooled fast breeders about 10 to 20 tonnes/GW(e).

### 3.2 Homogeneous, graphite free versions

Perhaps more impressive are the possibilities with homogenous designs lacking graphite moderator. With the entire volume of the core producing power, the needed volume is far less. Single cores for 1000 MW(e) are readily attainable although there are still advantages to smaller unit sizes. Without graphite moderation the assumption is often made that this means a much higher specific inventory and a quite hard spectrum. However, the carrier salt itself is a modest moderator and a wide variety of fissile concentration and neutron spectrum are in fact attainable. Recent French work requires a high specific inventory of 5.5 tonnes/GW(e) partly due to the fact that they choose to remove $BeF_2$ from the carrier salt due to toxicity concerns. In order keep the melting point low enough, the combined $ThF_4$ + $UF_4$ content needs to be 22% (about 2% $^{233}UF_4$). As well, with only a radial blanket in the TMSR design, attempting a much lower concentration would see a significant increase in neutron losses to the top and bottom reflectors.

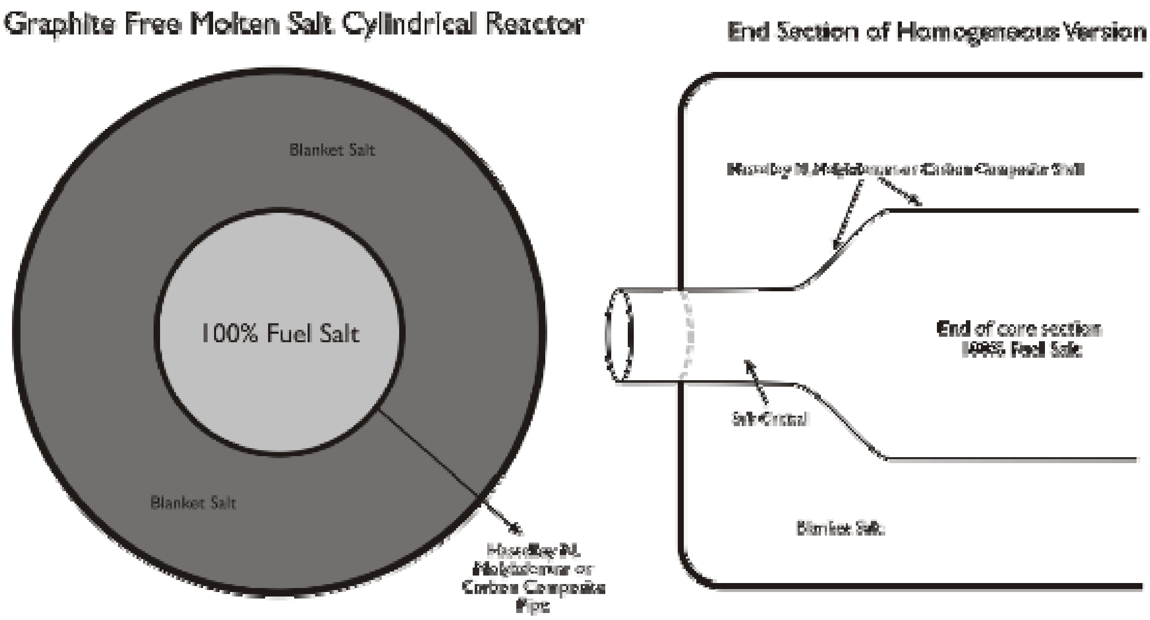


Figure 5 (Left) Cross sectional view of graphite free version (Right) End section showing tapering to a sub-critical while still within the blanket salt

ORNL calculations[15] from the spherical cores of the 1950s design provide an excellent tool for estimation. While the accuracy of such early data must of course remain suspect, it is hoped adequate for at least cursory investigations. This study assumed a 1/3 inch (8.5 mm) thick Hastelloy N barrier for cores up to 12 feet (3.7 m) diameter, thus for much smaller cylinders a thinner wall should suffice. The study also assumed a 2 foot (60 cm) blanket which allowed significant leakage in some cases, expanding this to 100 cm should convert most of those losses to thorium absorptions.

The values of Table 1 give the initial breeding ratios, thus no losses to fission products or protactinium. ORNL also projected[16] long term breeding ratios for the 8 foot core case in detail. Even with a relatively long 1 year processing time for fission product removal and no protactinium separation, the breeding ratio only dropped from 1.078 to 1.044.

TABLE I. Initial State Nuclear Characteristics of Spherical Two Region, Homogeneous, Molten Fluoride Salt Reactor with $^{233}U$ ORNL 2751 (1959). Values in italics are projected by the author

| Inner Core Diameter | 3 feet | 4 feet | 4 feet | 6 feet | 8 feet |
|---|---|---|---|---|---|
| Thorium in Fuel Salt | 0 % | 0 % | 0.25 % | 0 % | 7% |
| $^{233}UF_4$ in Fuel Salt | 0.592% | 0.158% | 0.233% | 0.048% | 0.603% |
| Neutrons per absorption in $^{233}U$ | | | | | |
| Be, Li and F in Fuel Salt | 0.0639 | 0.1051 | 0.0860 | 0.318 | 0.078 |
| Hastelloy N Core Wall | 0.0902 | 0.1401 | 0.1093 | 0.1983 | 0.025 |
| Li and F in Blanket Salt | 0.0233 | 0.0234 | 0.0203 | 0.0215 | 0.009 |
| Leakage | 0.0477 | 0.0310 | 0.0306 | 0.016 | 0.009 |
| Neutron Yield | 2.1973 | 2.1853 | 2.1750 | 2.2124 | 2.200 |
| Median Fission Energy | 174 ev | 14.2 ev | 19.1 ev | 0.33 ev | 243 ev |
| Initial Breeding Ratio | 0.9722 | 0.8856 | 0.9288 | 0.6586 | 1.078 |
| *Projected B.R. Thinner Wall** | *1.060* | *0.9836* | *1.011* | *0.7722* | *1.099* |
| *Projected B.R. Carbon Wall*** | *1.105* | *1.054* | *1.066* | *0.8714* | *1.112* |

* Projected assuming a thinner Hastelloy core wall of 1/6 inch (4.2 mm) and 90% leakage reduction by using a thicker blanket
** Projected assuming a Graphite or Carbon-Carbon core wall and 90% leakage reduction by using a thicker blanket salt

Taking the 3 foot (91 cm) case as example, this would equate to a 70 cm wide cylindrical. Going to a more modest power density of 200 kW/L still gives impressive results. Using the same 140 K temperature change and a much slower salt speed of 2 m/s gives a 505 MW(th) output from a 6.6 meter long core. At 44.4% for steam cycle, this is 224 MW(e). Even including a meter thick blanket and outer vessel wall still results in an extremely simple to manufacture design that can fit within a tractor trailer for transport.

It must be noted that Hastelloy N at the time of these early studies was thought to be good for 10 to 20 years in core. Thermal neutron induced damage discovered in the MSRE means that Hastelloy N might not have a very long lifetime in the full flux of the core. ORNL had success in limiting this damage by modifying the alloy makeup, this trend could perhaps be continued with further study. As well, maintaining a harder spectrum at the barrier might actually improve lifetime as it is predominately thermal neutrons that contribute to the damaging (n,alpha) reactions. Potentially a much superior metal barrier is a high molybdenum alloy. Molybdenum is known have a much greater tolerance to neutron damage. It has been suggested for use not only in molten salt fission designs but also for the barrier between plasma and a $2LiF\text{-}BeF_2$ coolant salt in fusion studies. As well, less expensive iron alloys including the common stainless steels 304 and 316 have also shown promise at somewhat lower operating temperatures. Given the simplicity of the core wall and outer vessel combination it is also not unreasonable to assume that periodic replacement even as short as annually could be still quite economical.

Carbon based material or a simple graphite tube would be ideal if their usability can be assured. The limited lifetime of graphite is well documented and would require periodic replacement. The irradiation tolerance of carbon based materials such as silicon impregnated carbon-carbon composites is an important question. There are thus several choices for a barrier material but it should be highlighted that this issue is of central importance to the proposed design.

### 3.3 Adding fertile, the 1 and ½ fluid or denatured options

While the pure 2 Fluid system has many advantages, adding a limited amount of thorium to the fuel salt does not necessarily detract from the fission product processing advantages. This is true if the thorium present in the fuel salt is allowed to be removed with the fission products. Traditionally this option would not be considered, for example in the Single Fluid 1970s design with 68 tonnes of thorium in the salt and a 20 day cycle time would mean wasting 1241 tonnes of thorium per GW-year. However for homogenous designs, a lower thorium concentration and more importantly much longer processing times afforded by the harder spectrum can result in new options. As an example the 8 foot (244 cm) example with 7% $ThF_4$ and 0.6% $^{233}UF_4$ would contain roughly 14 tonnes of thorium if the fuel salt volume was 15 $m^3$. The processing time could easily be extended to 2 years or more for this version and still break even on breeding. Thus with a thorium discard option, only 7 tonnes per year would be wasted. With the low cost and abundance of thorium the added expense is practically negligible and there would still a roughly 30 fold improvement over LWR once through for resource utilization. Furthermore, thorium is far more abundant than uranium.

Adding thorium to the fuel salt also results in wider critical diameters for similar fissile concentrations. For example 0.6% $^{233}UF$ has a 3 foot (91cm) critical diameter without thorium but an 8 foot (244 cm) diameter with 7% $ThF_4$. Thus a return to near spherical geometry for larger volumes is possible by the addition of fertile into to the core salt.

A similar result of wider, shorter cores is also be attained by adding fertile $^{238}U$ to the fuel salt to run a denatured cycle that keeps $^{233}U$ at less than 12% of uranium content. Thus uranium in all stages of operation will remain unfit for weapons use and allow easier compliance to existing regulations and the Nuclear Non-Proliferation Treaty. It is hoped that a pure Th-$^{233}U$ cycle can be shown to be equally proliferation resistant but it is obviously prudent to plan for both options. In practice running denatured would entail having both thorium and depleted uranium in the blanket salt such that the uranium remains denatured in the blanket as $^{233}U$ is produced. Running a denatured cycle entails significantly increased production of plutonium and other transuranics which would need to be recovered and re-injected to the core during processing for fission products. This adds to complexity but if the processing cycle time is greatly lengthened compared to the traditional 20 day cycle, this results in very small daily processing needs. Running a break even denatured cycle was calculated to be possible for a graphite moderated Single Fluid design[17] in the late 1970s so it should be little problem to break even for a 2 Fluid graphite moderated version. For homogeneous versions, there will undoubtably be a lower limit on the fissile molar concentration and thus spectrum hardness to overcome $^{238}U$ resonant absorptions. This would mean a larger starting fissile load is required, but this is easy to provide as low enriched uranium is ideal for startup in this case.

As a final reactor example, the 8 foot (244 cm) core case of ORNL can be examined. This had 7% $ThF_4$ and 0.6% $^{233}UF_4$ which represents approximately a 1200 kg/GW(e) starting fissile load for a total salt volume of 15 $m^3$. In cylindrical geometry this would be a core close to 2 m in diameter. Again assuming a modest power density of 200 kw/L, and in this case a salt velocity of 1.3 m/s gives an output of about 1000 MW(e) with a core length of only 4.2 meters.

Going to higher fissile plus fertile molar concentrations in the core salt and the resultant harder neutron spectrum has many advantages. Losses to fission products and protactinium are significantly lowered as their cross sections drop off faster than for fissile elements. This results in far less fuel processing requirements. Improving the neutron economy gives the ability to employ other carrier salts that do not contain $^7$Li or Be as these elements are expensive and produce tritium. Disadvantages include a shortening of the prompt neutron lifetime which can complicate reactor control. The very strong negative reactivity coefficients aids in this respect. Also, the issue of accidental criticality if salt spills can reach moderator has been raised. Proper design with boronated leak pans guards against this and any potential energy release of a spill reaching criticality should be small given that the salt would simply flash to vapor.

A potential plant layout for the above example is shown in Figure 6. Vertical core orientation is thought best for a shorter core. The thick blanket salt means almost no neutron flux reaches the outer vessel wall. Thus it need not be very thick or contain reflective material. A drain line activated by a freeze plug, drains the core to critically safe dump tanks should the salt temperature rise for any reason. A spill drain to dump tanks is also shown at the low point of the containment structure.

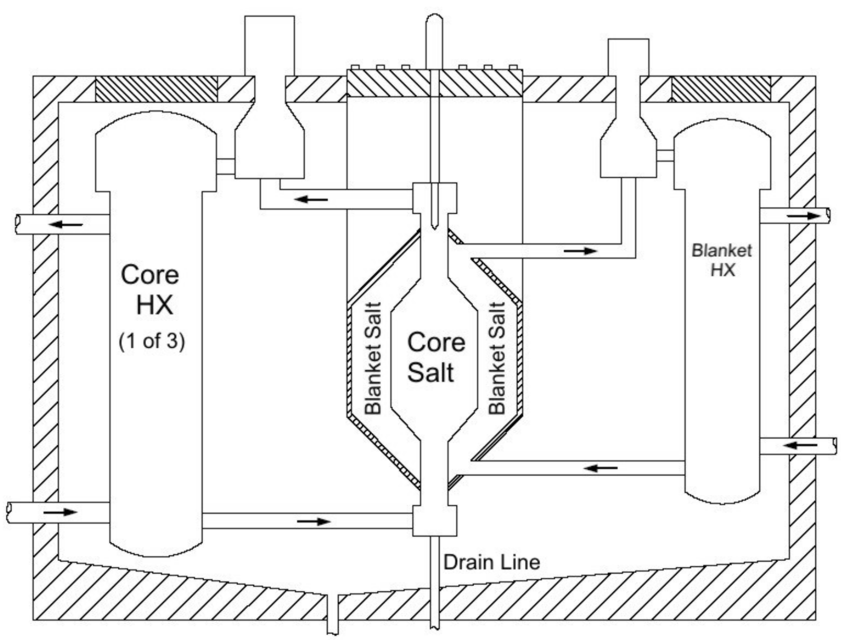


Figure 6. Potential plant layout within containment for an inner core of about 2 meter diameter. Secondary coolant salt transfers heat out of containment to drive a steam or closed gas cycle.

## 4. Conclusions

As the advantages of employing thorium have become widely recognized, it is time to formally reexamine the reactor specifically designed for its use. As is hopefully now evident, molten salt reactor designs offer great flexibility and advantages in almost all operational aspects. Costs of the traditional Single Fluid design have been estimated to be roughly on par with LWR costs, such that the great simplifications in design and fuel processing proposed here and elsewhere offer great saving potential. Overall safety sees a multitude of advantages over other reactor designs and denatured operation can be employed if even greater proliferation resistance is desired. Design and modeling work is ongoing on these presented designs and numerous others giving the versatility to adapt to design obstacles. For example, if a barrier between core and blanket proves unfeasible, barrier free alternative designs are already being modeled.

While at present, government and industry support is sorely lacking worldwide, the research and development needs[18] are far less than many may imagine. Perhaps ORNLs greatest legacy in this respect has been their dedication to fully document all aspects of their work and this wealth of information is now readily accessible. While the lack of after sale profits (enrichment, solid fuel fabrication etc) may require a different business model to attract corporate interest, the potential rewards are indeed great for any government, corporation or agency willing to take a leading role in this vital effort.

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