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Liquid Fuel Molten Salt Reactors for Thorium Utilization[1]

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ABSTRACT

Molten salt reactors (MSRs) represent a class of reactors that use liquid salt, usually fluoride- or chloride-based, as either a coolant with a solid fuel (such as fluoride salt-cooled high-temperature reactors) or as a combined coolant and fuel with the fuel dissolved in a carrier salt. For liquid-fueled MSRs, the salt can be processed online or in a batch mode to allow for removal of fission products as well as introduction of fissile fuel and fertile materials during reactor operation. The MSR is most commonly associated with the U-233/thorium fuel cycle, as the nuclear properties of U-233 combined with the online removal of parasitic absorbers enable the design of a thermal-spectrum breeder reactor. However, MSR concepts have been developed using all neutron energy spectra (thermal, intermediate, fast, and mixed-spectrum zoned concepts) and with a variety of fuels including uranium, thorium, plutonium, and minor actinides. Early MSR work was supported by a significant research and development (R&D) program that resulted in two experimental systems operating at ORNL in the 1950s and 1960s: the Aircraft Reactor Experiment and the Molten Salt Reactor Experiment. Subsequent design studies in the 1970s focusing on thermal-spectrum thorium-fueled systems established reference concepts for two major design variants: (1) a molten salt breeder reactor (MSBR) with multiple configurations that could breed additional fissile material or maintain self-sustaining operation, and (2) a denatured molten salt reactor (DMSR) with enhanced proliferation resistance. MSRs have been selected as one of the Generation IV systems, and development activity has been seen in fast-spectrum MSRs, waste-burning MSRs, and MSRs fueled with low-enriched uranium as well as more traditional thorium fuel cycle-based MSRs. This paper provides an historical background of MSR R&D efforts, surveys and summarizes many of the recent developments, and provides analysis comparing thorium-based MSRs by way of example.

Keywords: Molten Salt Reactors, Thorium Fuel Cycle, MSBR

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I. INTRODUCTION

Molten salt reactors (MSRs) represent a class of reactors that involve the use of a liquid salt (either a fluoride or chloride salt) either as a coolant with a solid fuel (as in fluoride salt-cooled high-temperature reactors [1]) or with a fuel dissolved in liquid salt that also serves as the coolant material. In dissolved-fuel MSRs, the salt can be processed additionally, either online or in a batch mode, to allow for the removal of fission products (FPs) and the introduction of fissile fuel and fertile materials during reactor operation. MSR concepts have been developed with both thermal and fast neutron spectra and with uranium, thorium, and plutonium fuels. The MSR has been selected as a Generation IV system. [2]

The dissolved-fuel MSR is most commonly associated with the U-233/thorium fuel cycle, as the nuclear properties of U-233 combined with the online removal of parasitic absorbers enable the design of a thermal-spectrum breeder reactor. [3] There has been recent research and development (R&D) activity on fast-spectrum MSRs using fuel cycles based on uranium and/or thorium, [4,5] waste-burning MSRs [6,7], and MSRs fueled with low-enriched uranium (LEU), [8] as well as thermal-spectrum thorium fuel cycle-based MSRs. [3, 9 - 11]

II. CHARACTERISTICS OF MOLTEN SALT REACTORS

The use of liquid rather than solid fuel in MSRs allows many reactor design features that are not possible with solid fuel. These include circulation of the fuel-containing liquid to act as a coolant and heat transfer mechanism, online chemical processing to remove parasitic absorbers and optimize the breeding and burning of materials, and different means of passive safety, such as draining the fuel from the core. The MSR designs of the 1960s and 1970s were focused on optimizing the thorium cycle to achieve a high level of breeding performance by online chemical processing. At the time, it was envisioned that there would be very quick growth in nuclear energy, with fissile material representing a limit to growth. MSR fuel consists of fissile and fertile actinides dissolved in a liquid carrier salt. The most commonly considered carrier salt is a LiF-BeF_2 salt with lithium enriched in Li-7 content to minimize absorption and tritium production. Many other salts have been considered, based on sodium, zirconium, rubidium, and other materials. In addition, chloride-based salts have been considered for fast-spectrum systems. However, most R&D to date has been with fluoride salts, which is the primary focus of the discussion below.

The online chemical processing system is fundamentally based on fluoride chemistry to allow effective removal of the uranium from the salt, followed by vacuum distillation and/or reductive extraction for the removal of FPs. Gaseous FPs are readily removed from the fuel salt by helium sparging. Removing the highly absorbing FPs and allowing Pa-233 to decay to U-233 outside the core results in an optimal breeding system for a thermal-spectrum reactor using thorium. An MSR has a relatively low fissile mass inventory, and advantageous attributes of a liquid-fueled system allow it to retain most of the fissile inventory of the entire fuel cycle in the reactor by minimizing holdup times elsewhere in the fuel cycle (e.g., no long hold time requirements for cooling fuel before separations, fuel fabrication, or transportation between fuel cycle facilities). The breeding performance of a thermal-spectrum molten salt reactor utilizing thorium can be compared to other systems by examining the doubling time, which is the time it takes for the amount of fissile material in a breeder reactor to double [10]. The linear doubling time (t_{Dl}) is given by the following equation [10]:

$$t_{Dl} = \frac{m_o}{(BR-1)wP}$$

where m_o is the initial fissile inventory, BR is the breeding ratio, w is the fissile consumption rate per unit power by fission and capture, and P is the reactor power level. As this equation shows, the doubling time is proportional to the initial inventory divided by the breeding gain (BR-1) for a fixed power level. Therefore, in comparison to fast-spectrum systems, the lower breeding ratio of a thermal-spectrum thorium-fueled MSR is offset by the lower fissile requirements resulting in similar doubling times. In addition, when considering the overall fuel cycle, performing online processing in the MSR minimizes the fissile inventory outside of the reactor thereby reducing the system doubling time.

A typical MSR core region consists of a matrix of graphite blocks that provide moderation to create a critical system with fissile fuel and fertile blanket regions. Two design approaches were considered at Oak Ridge National Laboratory (ORNL) during the Molten Salt Breeder Reactor (MSBR) project: a "two-fluid" system with separate fuel and fertile salts, and a "single fluid" design in which the fertile materials are included in the fuel salt. The single fluid concept requires a more sophisticated chemical processing system to separate the lanthanide FPs from the fertile thorium materials. The graphite moderator in these designs requires occasional replacement as a result of radiation damage, with typical replacement times being 4–8 years, depending upon the core power density. Alternative designs have been considered with lower power densities that would allow the graphite lifetime to be extended to much longer periods (see, for example, Table 5.1 of reference 9), and fast-spectrum systems that eliminate the use of a graphite moderator.

The final ORNL MSBR concept [3] was designed to operate at a high temperature, with a fuel salt core exit temperature of 700°C, and was based on a high-temperature Rankine power conversion system with 44% power conversion efficiency. Current concepts consider other power conversion systems, such as a Brayton cycle. The safety of an MSR relies on negative reactivity coefficients and decay heat removal systems. The MSBR concept used freeze plugs and a drain tank system with passive decay heat removal; if the fuel salt temperature increased above a design value, the freeze plugs would melt and the fuel salt would drain into tanks designed to have a subcritical configuration and sufficient passive decay heat removal. Given that the fuel salt would distribute radioactive materials throughout the primary fuel circuit, the system was designed for remote maintenance, which was at least partially demonstrated during MSRE operation.

III. MSR DEVELOPMENT HISTORY

MSRs were first proposed at ORNL shortly after World War II as a means to power military aircraft as part of the Aircraft Nuclear Propulsion (ANP) program. This concept was chosen primarily because a very high-temperature, high-power-density reactor was needed. As part of the program, a 2.5 MWt proof-of-principle test reactor (Aircraft Reactor Experiment) was developed and operated for 100 hours at high temperature (860°C) in 1954. [13] Based on this successful test, the ANP program went on to develop a higher- power version. However, the program ended as a result of national policy decisions, and the technology was adapted to a civilian nuclear power program.

The civilian MSR program started in the mid-1950s and continued into the early 1970s with the progressive development of advanced reactor concepts and fundamental R&D on high-temperature materials, salt chemistry, and separation sciences. The work was focused on the development of a thermal-spectrum breeder based on the Th-232/U-233 cycle for nuclear power sustainability, in parallel with the ongoing Liquid Metal Fast Breeder Reactor (LMFBR). [13, 14]

As part of the program, the 8 MWt Molten Salt Reactor Experiment (MSRE) was operated from 1965 to 1969 with over 13,000 fuel power operation hours, including an 8000-hour continuous period of operation. The reactor operated on U-235, U-233 (the first reactor to do so), and U-233/Pu-239. The design used a single fluid consisting of fuel salt with no thorium. MSRE operation provided a successful demonstration of a one-fluid MSR concept and several specific MSR technologies.

An MSBR design concept was completed in 1972 as a 1000-MWe thermal breeder with on-line refueling, high thermal efficiency, and costs comparable to those for light-water reactors (LWRs) based on economic analysis at that time. [3] MSR development ended in the mid-1970s, when the Atomic Energy Commission focused its efforts solely on the LMFBR. In the late 1970s, the denatured molten salt reactor (DMSR) concept [15] was developed at ORNL; the goal of was to reduce proliferation risk by avoiding the separation of U-233and maintaining a uranium composition equivalent to LEU from a nonproliferation standpoint, based on the relative combined fraction of U-233 and U-235 in the total uranium mass. [16]

III.A The Molten Salt Breeder Reactor

The MSBR concept that represented the final ORNL design was a thermal-spectrum single fluid system. [3] The design employed ~43 m^3 of fuel salt (71 mol % ^7LiF, 16 mol % BeF_2, 12 mol % ThF_4, and ~0.3 mol % $^{233}UF_4$). The plant was a four-loop design with an average core power density around 22 kW/liter. The number of loops was chosen in part according to what was thought to be reasonable pump design capacity limits, based on an extrapolation of the pumps tested in the MSRE.

A total of 295,000 kg of graphite was called for in the design, approximately 205,000 kg of which was to be replaced approximately every 4 years. The fissile fuel inventory of the reactor system and processing plant was estimated to be 1500 kg, and the thorium inventory was estimated to be 68,000 kg. The breeding ratio was estimated to be 1.06, producing a doubling time of approximately 22 years. A summary of the MSBR key design and operating parameters is presented in Table I.

Notably, this design included a significant amount of thorium that was discarded, resulting in a thorium utilization of about 10%. At the time, this was seen to be reasonable, given the relatively high abundance of thorium and the substantial improvement of this system compared with the uranium utilization in LWRs (<1%). A conceptual layout of the reactor core and vessel is shown in Fig. 1.

III.B The Denatured Molten Salt Reactor

In addition to breeder reactors operating on the thorium fuel cycle, MSRs include designs that use uranium as a fuel. Although it is possible to develop MSRs that operate on the uranium/plutonium fuel cycle as thermal convertor reactors and fast-spectrum breeder reactors, development work was also performed for DMSRs using both uranium and thorium as fuel. [15] This concept was developed in the late 1970s as a result of President Ford's nuclear policy statement of 1976 [17] that imposed restrictions on fuel reprocessing. Later, it was continued under the Nonproliferation Alternative Systems Assessment Program (NASAP). [18]

The DMSR was developed directly from the MSBR design and retains many of the features described in the previous section. The primary difference from the MSBR is that the DMSR does not include an online chemical processing plant. As a result, only gaseous FPs are extracted, and the noble and semi-noble metals are assumed to plate out in the reactor system. LEU was introduced to provide the necessary reactivity to overcome the FP penalty. The DMSR was designed to operate as a once-through system without fuel processing. Another design difference is a reduction in the reactor power density, which extended the lifetime of the graphite moderator to match the designed plant lifetime (30 years). The reactor is fueled as follows:

1. Thorium is added to the initial loading and allowed to decline over the life of the reactor.
2. Enriched uranium is added as needed to maintain criticality (19.75% enrichment). A specific fuel addition schedule is provided in [15] with an average addition of about 790 kg/year.
3. U-238 is added as needed to maintain all uranium in a denatured state per the inequality:

$$\text{U-238 density} \geq (6 \cdot \text{U-233 density}) + (4 \cdot \text{U-235 density}) .$$

4. Gas sparging removes gaseous fission products from the fuel salt, and noble metals plate out.

Key design, inventory, and operating parameters of the DMSR are summarized in Table II. Even with the introduction of U-238, the inventory of transuranics (TRU) at end of life (EOL) is reasonably low; the total Pu inventory is 736 kg. The reactor requires 17.5 tonnes of 19.75%

enriched uranium at startup and a total of 23 tonnes of enriched uranium over the reactor lifetime. The wastes during operation are primarily gaseous FPs. The entire EOL core inventory after 30 years of operation is the final waste produced by the system. Alternatively, the uranium in the discharge salt could be recovered via fluorination and used in a subsequent DMSR, which would reduce uranium requirements by about 50%.

Other design options for uranium-fueled MSRs are possible and were evaluated as part of the DMSR study. They include options with partial FP removal, enhanced designs with break-even breeding, and batch mode salt replacement.

III.C Comparisons of the MSBR and DMSR

Table III summarizes key MSBR and DMSR parameters that can be used for fuel cycle assessments. These parameters can be used to analyze the nuclear fuel cycle performance of these designs in terms of waste generation and resource utilization.

III.C.1 Resource Utilization

For the MSBR, the mined resource is thorium that is loaded in the initial core and used as feed during operation. The total thorium used, shown in Table III, was 68 tons at startup and 6 tons per year of operation. If the MSBR is started with U-233 bred from operation of previous MSBRs, then no enriched uranium is required. However, assuming a startup with LEU (19.75%), an initial core loading of about 8 tons is required, resulting in 320 tons of natural uranium before enrichment. The energy produced over the 30-year operation period of the

MSBR is 22.5 GWe-yrs (assuming a 75% capacity factor). These values provide the resource sustainability results given in Table IV.

For the DMSR, the mined resource includes both uranium and thorium. The initial loading of uranium is 17.5 tons (19.75% enrichment), and the total enriched uranium additions over the life of the reactor are 0.8 tons per year of operation. The thorium in the initial core load is 110 tons. This system also produces 22.5 GWe-yrs over its operation lifetime. The calculated values for resource sustainability are also given in Table IV.

Compared with the current operation of an LWR, these thorium-based MSRs have significantly lower uranium utilization per GWe-yr: an MSBR with a U-233 initial core requires no uranium, and an MSBR with a U-235 start-up core requires about 14 times less uranium. In terms of total heavy metal (HM) resources, the MSBR with U-233 initial core uses about 18 times less HM and an MSBR with a U-235 initial core about 8 times less than the LWR reference system. A DMSR uses significantly more HM resources than an MSBR, but only approximately 1/3 as much as the LWR reference system. It should also be noted that any of these MSR systems offer potential savings in separative work units (SWU) during enrichment by reducing enrichment requirements or even completely eliminating enrichment.

III.C.2 Waste Management

Waste management deals with the long-term environmental burden imposed by the disposal of waste materials from a nuclear fuel cycle. The environmental impact can be characterized by the quantity of actinides, decay heat, and long-term radiotoxicity of the waste. The primary assumption for the MSR systems is that all materials are stored on the reactor site until the end of the 30-year period of operation, followed by a 5-year cooling and waste processing period. As shown in Table V, both the MSBR and the DMSR have significantly lower actinide and TRU mass per unit of energy generation than current LWRs. However, the DMSR has an increased level of TRU production because it has a higher uranium inventory than the MSBR. The decay heat is lower than in LWRs, since the overall thermal efficiency of the molten salt systems is higher. In addition, FPs are separated (only gaseous and noble metal FPs in the case of the DMSR) and can be disposed of independently.

IV. ANALYSIS OF SELF-SUSTAINING THERMAL-SPECTRUM MOLTEN SALT REACTORS

IV.A MSR Inventory Analysis for Evaluation and Screening

In support of an evaluation and screening of potential fuel cycle options, [19] the Office of Fuel Cycle Technologies of the United States Department of Energy Office of Nuclear Energy supported the analysis of a self-sustaining (unity breeding ratio) concept to inform the study. Thermal-spectrum thorium-fueled MSRs were analyzed to ensure that a complete range of potential fuel cycle options was considered. As part of this fuel cycle assessment work, various parametric studies were performed to understand the design space and sensitivity to input parameters exhibited by the MSRs of interest. The referenced fuel cycle option reported here used “full recycling,” in which all primary fuel materials undergo continuous recycling, and active separations processes remove other materials. Calculations of other options based on once-through and limited recycling of materials were also considered. [20]

IV.B Modeling Parameters

MSR analysis requires modeling several important material feed and removal functions: direct discard of fuel salt, salt treatment processes, separations processes that actively extract species such as rare earth element FPs, and fresh feed of thorium. These processes are applied sequentially in this work in the following order: (1) salt discard (if any), (2) uranium and protactinium separations, (3) salt treatment and separations, and (4) thorium addition.

Salt treatment and separations calculations were performed using effective cycle times from the MSBR program [3] and batch removal calculations. [20] The MSBR program defined a *cycle time* as the amount of time required to remove 100% of a given element from a fuel salt. Cycle times were converted to removal fractions for this work, with a removal fraction of 1.0 occurring when the depletion time-step length matched the cycle length for an element. Table VI summarizes the cycle times used for modeling salt treatment and separations, assuming full recycling of the fuel salt.

IV.C Equilibrium MSR Inventories

An equilibrium inventory analysis for a thorium-fueled MSR with full recycling demonstrates how the models and reactor physics insights described above can be combined and applied to support the U.S. fuel cycle assessment. The objective of this fuel cycle analysis was the self-sustained operation of a critical, thermal-spectrum MSR with equilibrium requirements of no enrichment and only a thorium feed.

Calculations indicate near-equilibrium after 20 years of operation based upon tracking the changes in several metrics, including eigenvalues, mass flow rates, and isotopic number densities in the fuel salt. Figure 2 shows the infinite multiplication factor as a function of time during this 20-year operation period, and Table VII summarizes key equilibrium fuel cycle parameters. The equilibrium infinite multiplication factor (1.0255) allows sufficient excess reactivity for neutron leakage and a k_{eff} of 1.0.

Note that unlike the MSBR previously discussed, this self-sustaining system assumes that thorium loss is only the result of processing losses and not salt discard in order to focus on full recycle of the thorium. This results in a thorium feed rate of less than 1 ton per year compared with about 6 tons per year for the MSBR.

V. FAST-SPECTRUM MOLTEN SALT REACTORS

MSRs can also be implemented as fast-spectrum reactors that can offer some advantages over the thermal-spectrum concept and fast-spectrum solid fuel designs. The elimination of the moderator (typically graphite) can simplify maintenance compared with thermal-spectrum MSRs, since the moderator typically has a limited lifetime and may require replacement. In addition, the use of a fast spectrum results in lower parasitic absorption loss to FPs or parasitic capture in intermediate species in breeding cycles (e.g., Pa-233 in the Th-232/U-233 cycle). This may enable fast-spectrum MSRs to use a simplified salt processing system, eliminating or minimize salt-handling facilities and processes, and therefore may also offer a benefit for proliferation resistance due to less material access. This comes at the expense of a higher fissile inventory and presents additional development challenges beyond those previously demonstrated for the thermal-spectrum design. Fast-spectrum MSRs can serve as both breeders and converter systems using either uranium or thorium fuel cycles, and they can serve as waste transmutation systems.

Compared with solid-fueled fast-spectrum reactors, liquid-fueled MSRs offer a fuel that has no limit on fuel life related to cladding fluence limits, and it can provide a design that has large negative void reactivity coefficients. However, fast-spectrum MSRs are at an early stage of development, and no experimental or demonstration systems have been developed.

A fast-spectrum thorium-fueled MSR concept is under development in France, known as the Molten Salt Fast Reactor (MSFR). [2,5] This reactor concept uses a different salt (LiF-ThF4 vs. LiF-BeF-ThF4), and it contains a larger thorium content (22.5 mol % vs 12%) from that considered in the thermal-spectrum concept. The MSFR has a simpler core configuration than the MSBR, with a cylindrical region containing only salt surrounded by a blanket salt region. The liquid fuel salt circulates as a coolant through heat exchangers. The MSFR core configuration is shown in Fig. 3.

Analysis of the MSFR concept [5] shows that it can achieve a higher breeding ratio than the thermal-spectrum configuration (1.12 vs. 1.06). However, the MSFR requires a higher fissile inventory (~5 tons vs. ~1 ton). Analysis of reprocessing times shows that the MSFR can operate as a self-sustaining system with a 6-month processing time and can operate for periods >20 years with no processing at all. [5]

VI. CONCLUSIONS

Because of their historical development as a concept that can operate as a thermal-spectrum breeder with a significant breeding gain, MSRs using liquid fuel are frequently associated with the utilization of thorium as a fuel. The concept was supported by a significant research and development program that resulted in an experimental system operating at ORNL in the 1960s. The resulting MSBR concept represents just one of numerous possible configurations that can effectively use thorium for breeding additional fissile material, or self-sustaining operation, or conversion (with fissile support). Recently, activities in liquid fuel MSR designs using thorium have focused on fast-spectrum concepts that may offer some advantages over the original thermal-spectrum designs. Both thermal- and fast-spectrum systems represent viable designs for future research and development.

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Table I. Key MSBR design and operating parameters [3]

Parameter	Value
Reactor thermal power (MW)	2250
Reactor electrical power (MWe)	1000
Fissile fuel inventory (kg)	1501
Thorium inventory (kg)	68,100
Thorium feed rate (kg/yr)	~6000
Inventory, U/Np/Pu/Am/Cm (kg)	1988/15.3/13.4/2.3/6.2
Waste, Th/Np/Pu/Am/Cm (kg/GWe-yr)	5400/0.72/0.63/0.11/0.29
Waste, total transuranics (kg/GWe-yr)	1.74
Breeding ratio	1.06
Doubling time (years)	22
Fuel salt components	$^{7}LiF-BeF_2-ThF_4-UF_4$
Fuel salt composition (mol %)	71.7-16-12-0.3
Core inlet/outlet temperature (°C)	566/704 °C

Table II. DMSR key design and operating parameters [15]

Parameter	Value
Reactor thermal power (MW)	2250
Reactor electrical power (MW)	1000
Fissile fuel inventory (kg) (includes U-233, U-235, Pu-239, Pu-241)	3,450 BOL (19.75% U235) 3,440 MOL (15 yrs) 3,490 EOL (30 yrs)
Thorium inventory (kg)	110,000 BOL 103,000 MOL (15 yrs) 92,900 EOL (30 yrs)
Uranium inventory (kg)	17,450 BOL 20,620 MOL (15 yrs) 29,850 EOL (30 yrs)
Plutonium inventory (kg)	0 BOL 492 MOL (15 years) 736 EOL (30 years)
Uranium feed rate (kg/yr)	790 kg/year (average)
Uranium feed U-235 enrichment (wt %)	19.75
Thorium feed rate (kg/yr)	0
Conversion ratio	~0.9
Fuel salt components	$^{7}LiF\text{-}BeF_2\text{-}ThF_4\text{-}UF_4$
Fuel salt composition (mol %)	71-20-8-1
Core inlet/outlet temperature (°C)	566/704

BOL = beginning of life; MOL = middle of life

Table III. Summary of key parameters for the illustrative MSR concepts

Parameter	MSBR	DMSR
Power level (MW)	2250 MWt 1000 MWe	2250 MWt 1000 MWe
Fissile enrichment of the fuel (uranium)	70% fissile (U-233+U-235) at equilibrium [a]	19.75% U-235 startup and feed
Uranium content in fuel (fraction of heavy metal)	2% U/(U+Th)	14% U/(U+Th)
Burnup	105 MWt-day/kgTh [b]	460 MWt-day/kgU [c] 120 MWt-day/kg(U+Th) [d]
Fuel residence time (years)	N/A (continuous recycle)	30 (with some fuel "bleeding" to allow room uranium feed)
Refueling method	Continuous online (graphite moderator has a 4 year lifetime)	Continuous online (graphite does not need replacement)
Reactor capacity factor (CF)	75% (assumed value in 1972, modern design would be ≥90%)	75% (assumed value in 1980, modern design would be ≥90%)
Reactor lifetime	30 years (per original design report, modern design would be a minimum of 40 years)	30 years (per original design report, modern design would be a minimum of 40 years)
Fuel physical form	$^7LiF-BeF_2-ThF_4-UF_4$ (liquid)	$^7LiF-BeF_2-ThF_4-UF_4$ (liquid)
Primary fissile material	U-233	U-233 and U-235
Primary fertile material	Th-232	Th-232 and U-238
Separation process	Fluoride volatility, bismuth metal reduction, helium sparging	No separations other than gaseous FP extraction
Separation recycle product	Uranium, Thorium, Salt	N/A
Nature of waste stream	FPs (gaseous, separated from others) Waste salt (containing unextractable Rb/Sr/Cs/Ba FPs, thorium, other actinides, FLIBE) Irradiated graphite with FP	FPs (gaseous) Discharge salt (containing FPs, uranium, thorium, other actinides) EOL fuel salt (containing FPs, uranium, thorium, other actinides)
Actinide waste (kg/GWe-yr)	Th: 5,400, Np: 0.72, Pu: 0.63 Am: 0.11, Cm: 0.29 Total TRU: 1.74	Th: 4,100, Pa: 1.7, U: 1,480, Np: 6.0, Pu: 33 Total TRU: 39

[a]Assumes U-233 is used to start up reactor. Alternatively, the reactor could be started with LEU fuel, which would be replaced over a period of a few years by U-233. This case has not been analyzed.
[b]Value based on energy generated over 30 year operation period (2250 MWt, 75% CF) divided by makeup thorium (6000 kg/yr, 29 years). This assumes that the discharge fuel will be used in a follow-on system. If not, then the denominator should include the initial thorium loading (68000 kg), giving a burnup value of 75 MWt-day/kgTh.
[c]Value based on energy generated over 30 year operation period (2250 MWt, 75% CF) divided by initial and feed uranium (17,450 kg + 790 kg/yr × 29 years).
[d]Same as in footnote *b*, except denominator includes initial thorium loading (110,000 kg)

Table IV: Summary of resource utilization for MSRs and reference LWR system

Parameter	LWR (reference system)	MSBR (U-233 initial core)	MSBR (U-235 initial core)	DMSR
Natural Uranium utilization (tonnes/GWe-yr)	200	0	14	70
Thorium utilization (tonnes/GWe-yr)	0	11	21	6
HM (U+Th) utilization (tonnes/GWe-yr)	200	11	35	76

Table V. Summary of waste management metrics for MSRs and reference LWR system

Parameter	LWR (reference system)	MSBR (U-233 initial core)	MSBR (U-235 initial core)	DMSR
Actinide mass per unit energy (kg/GWe-yr)	22,000	5,400	5,400	5,500
TRU mass per unit of energy (kg/GWe-yr)	260	1.7	1.7	39
Decay heat per unit energy (MW/GWe-yr)	Reference	Lower than reference [a]	Lower than reference	Lower than reference

[a]Qualitative comparison with reference system.

Table VI. Material cycle (removal) times

Processing group		Cycle time
Salt treatment	Volatile gases	20 sec
	Noble metals	20 sec
	Semi-noble metals	200 days
	Volatile fluorides	60 days
Separations	Protactinium	3 days
	Rare earths (non-Europium)	50 days
	Europium	500 days
	Other FPs	9.4 years

Table VII. Self-sustaining MSR parameters

Parameter	Value
Conversion ratio	1.001
Th feed (kg/yr)	926
Fissile mass in salt (kg)	1,304
Th mass in salt (kg)	69,737

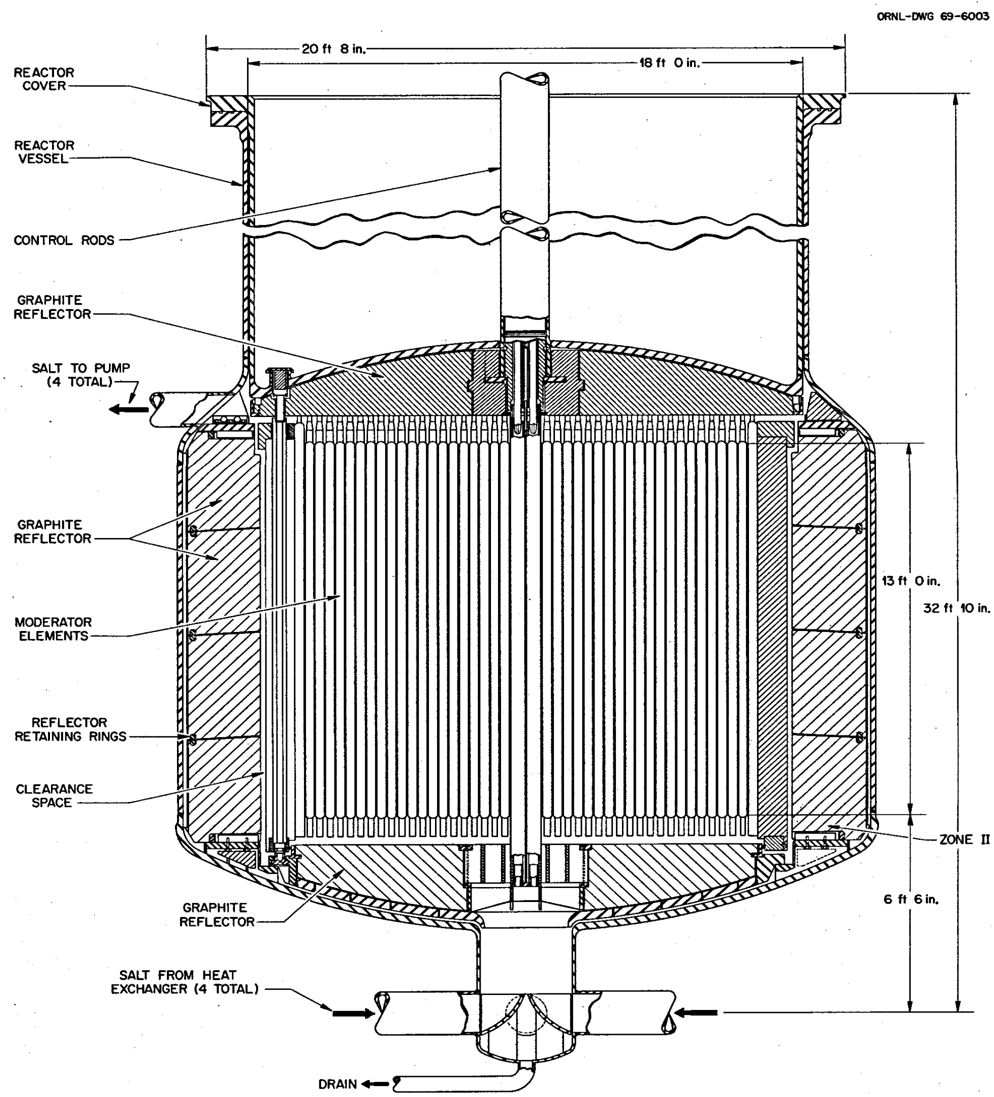

Fig. 1. Single fluid MSBR core diagram. [3]

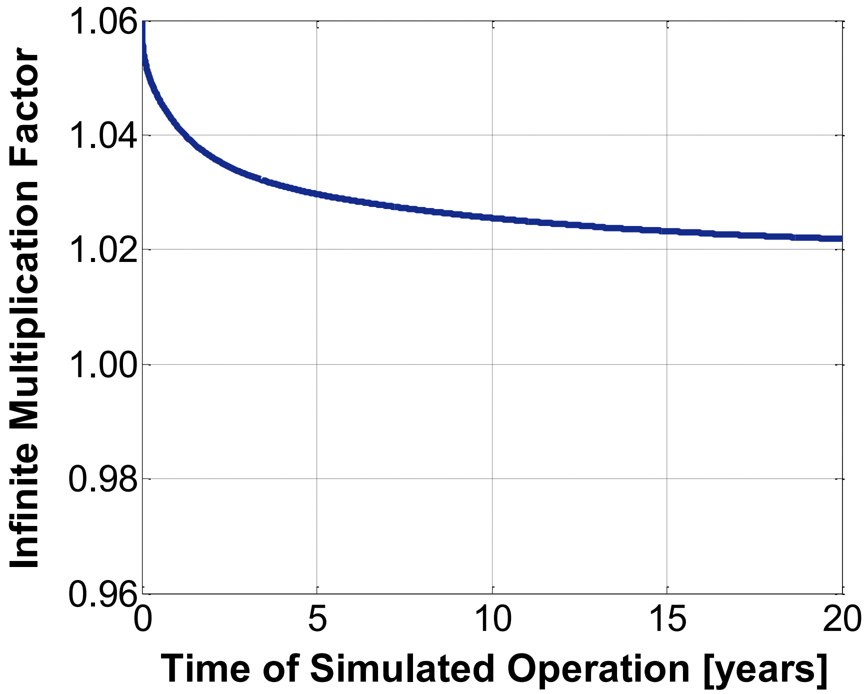

Fig. 2. Infinite multiplication factor as a function of time for a thorium MSR with full recycle.

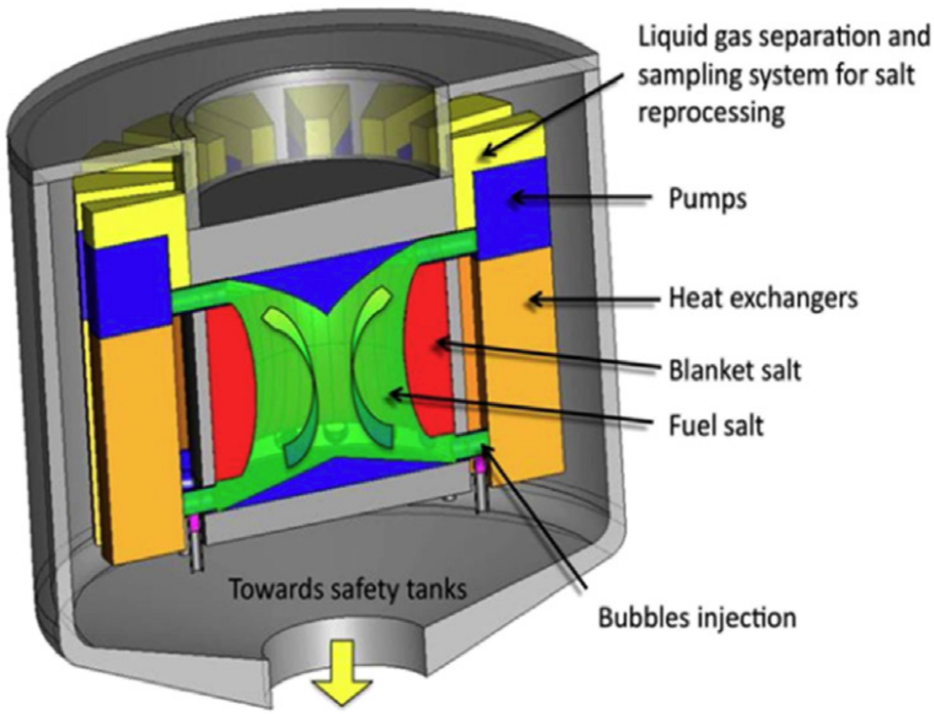

Fig. 3. Molten salt fast reactor concept diagram. [2]