Molten Salt Reactors

By Thomas J. Dolan

The accidents at Three Mile Island (TMI), Chernobyl, and Fukushima Daiichi were caused by evaporation of coolant, steam explosion, and hydrogen generation. They would not have occurred in a molten salt reactor (MSR), which can operate at low pressure without generating steam or hydrogen.

MSRs with *solid* fuels, such as ceramic pebbles developed for HTGRs, could operate safely at high temperature and low pressure. *Liquid* fueled MSRs could process the fuel online to adjust reactivity and remove some fission products, reducing core radioactivity, avoiding solid fuel clad damage, and extending core life. Either the U-^{239}Pu fuel cycle or the Th-^{233}U fuel cycle could breed additional fissile fuel. MSRs could burn actinides from used light water reactor (LWR) fuel, reducing the need for geological high-level waste disposal. We will discuss MSR history, solid and liquid fuel reactors, tritium, salt processing systems, nonproliferation, reactor design projects, international cooperation, and development issues. [1]

History

Alvin M. Weinberg, Director of Oak Ridge National Laboratory (ORNL), led the development of liquid fuel MSRs. ORNL built the Aircraft Reactor Experiment (ARE) that used liquid fuel of $NaF\text{-}ZrF_4\text{-}UF_4$ (53-41-6 in mol%), flowing through fuel tubes, with BeO moderator, Inconel structure, and liquid Na coolant. It operated successfully at 860 °C, 2.5 MWth, for 100 hours in 1954.

ORNL operated the 8 MWth Molten Salt Reactor Experiment (MSRE) burning liquid $^{7}LiF\text{-}BeF_2\text{-}ZrF_4$- (65-29-5-1 mol%) fuel at 650 °C from 1965-1969. The molten salt flowed vertically upwards through graphite moderator tubes, **Figure 1**.

Figure 1. The MSRE core. Fuel flowed upwards through channels in the graphite. [ORNL]

Heat was removed by a salt-to-air heat exchanger. Hastelloy-N alloy structure was developed to minimize corrosion. The MSRE demonstrated that issues of control, pumping, heat removal, radioactivity containment, and corrosion could be managed well.

The ORNL 1000 MWe Molten Salt Breeder Reactor (MSBR) design had a liquid fuel of LiF-BeF_2 ("FliBe") with dissolved ThF_4 and $^{233}UF_4$. The core inlet/outlet temperatures were 566/704 °C. About 205 tonnes of graphite moderator were to be replaced every four years. The MSBR projections were for a fissile inventory of 1501 kg, a breeding ratio of 1.06, 68100 kg of thorium, and a doubling time of 22 years, but funding was terminated in the 1970s. [2]

Solid fuel MSRs

Solid fuel MSRs would probably be graphite-moderated, cooled by fluoride salts, such as FliBe, with passive decay heat removal. Fluoride salt cooled High-temperature Reactors (FHRs) would be *burners* (conversion ratio CR < 1) using fuel rods, plates, or pebbles, burning ^{235}U, ^{239}Pu, and other actinides.

Pebble Bed

For example, a pebble bed design (PB-FHR) is like a pebble bed high temperature gas cooled reactor (HTGR) with the gas coolant replaced by molten salt, such as FliBe. The 3-cm diameter

graphite pebbles contain thousands of 0.4 mm tri-structural-isotropic (TRISO) fuel kernels that can survive at T ~1800 °C. Many countries have developed the fuel manufacturing capability, and the coolant temperature is limited by interaction with structural alloys, rather than by fuel pebble limits.

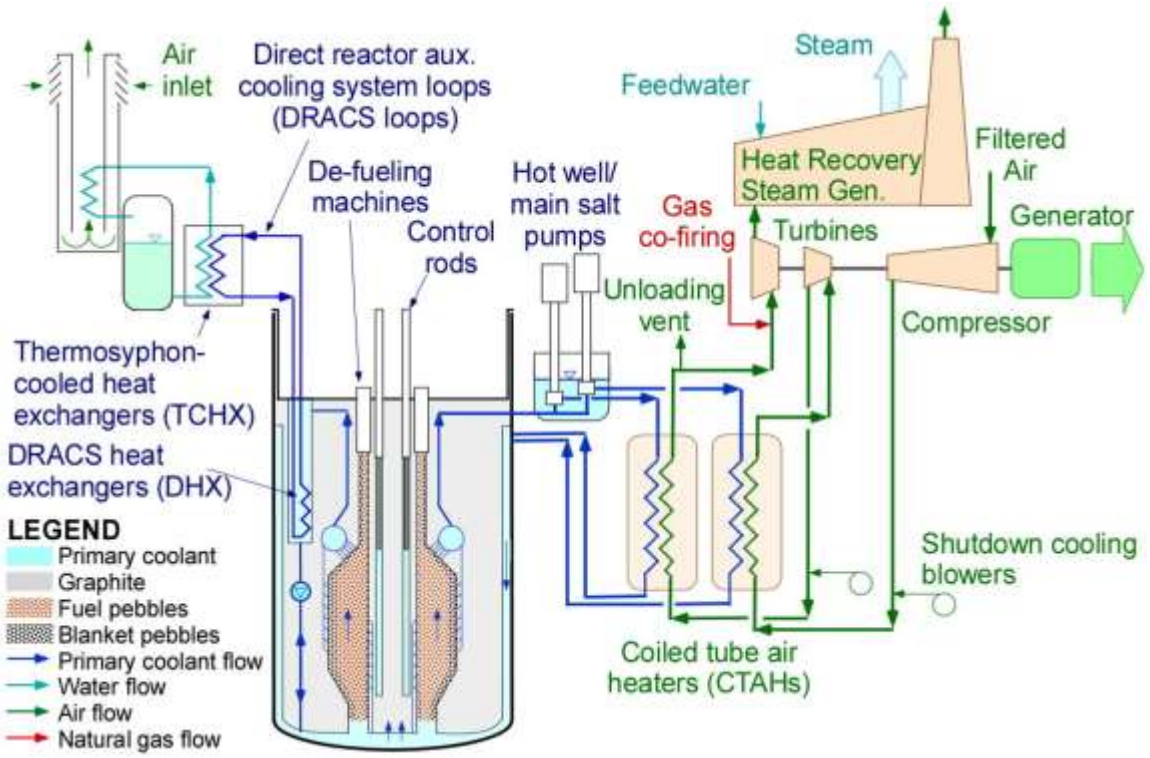

Figure 2. Major components of The Mk1 PB-FHR design of the University of California – Berkeley (UCB) with Nuclear-Air-Brayton Combined Cycle (NACC).

The Mk1 PB-FHR of **Figure 2** has a 3.6 m diameter, 12 m high vessel, which limits core power, but enables rail transport. Core salt with inlet/outlet temperatures of 600/700 °C flows at 2 m/s to coiled tube air heaters to drive a modified GE 7FB gas turbine that produces 236 MWth and 100 MWe per module, giving a core power density of 0.87 MWe/m^3.[3] The pebbles flow very slowly (2.1 months per pass) upward through the core in FliBe molten salt, which facilitates online inspection and refueling. A discharge burnup of 180 GWd/tonne U is estimated, with a pebble residence time of 1.4 effective full-power years. The volumetric heat capacity of FLiBe (4.18 MJ/m^3-K) exceeds that of water, so the salt can have lower flow rates at a given power than

LWRs, HTGRs, and sodium-cooled fast reactors. FLiBe also has excellent neutronics performance, but it is very expensive, and it produces large amounts of tritium. The Mk1 PB-FHR does not have an intermediate cooling loop, which improves efficiency, but increases the tritium leakage potential.

Nuclear power needs to have strong **load-following** capability to compensate for fluctuations of wind and solar power, which can cause severe fluctuations in electric power prices. In a **Nuclear Air-Brayton Combined Cycle** (NACC) system the molten salt coolant heats filtered air to 670 °C in a coiled tube air heat exchanger (CTAH), Figure 2. The air drives two gas turbines with reheat in between, followed by a steam generator, to generate 100 MWe from 236 MWth in the basic design (42% efficiency). Additional heat may be added between gas turbine stages to boost the air temperature to 1065 °C, which raises the co-fired power to 242 MWe. The additional heat can come from combustion of natural gas or hydrogen or from heat stored in firebrick resistance-heated energy storage (FIRES) at T > 1100°C (not shown in Figure 2). The bricks can be heated electrically when power is cheap and can supply heat when power is expensive. The molten salt delivers heat at very high average temperature (~650 °C, compared to helium at 550 °C, sodium at 500 °C, and water at 280 °C), and the incremental heat conversion back into electricity can be done at high efficiency (>65%). The NACC plant power can be varied rapidly (> 24 MW/min.) over a wide power range at prices competitive with natural gas fired plants. In near-term plants sodium-potassium-magnesium chloride salts with heat storage in the 500 to 700°C temperature range can have heat storage costs for the salt below $5/kWh, significantly below the costs of other heat storage options. [4]

In addition to improved safety, solid fuel MSRs could have additional advantages over current LWRs:

- Thinner, less expensive, more reliable vessels and tubes.
- Higher temperature and efficiency, facilitating more compact Rankine and Brayton cycle systems (including air cooling), and chemical processes, such as the sulfur-iodine process for hydrogen generation.
- Useful in arid climates, like HTGRs.
- Lighter components facilitate modular construction, transport to reactor sites, and rapid assembly

Solid fuel MSRs face several challenges:

- Tritium containment
- Graphite lifetime, interaction with coolant, and disposal
- Corrosion of structural alloys by molten salts
- High cost of some salts, such as FliBe
- Possible salt freezing in plumbing and heat exchangers
- Complexity of licensing new technology.

Solid fuel life is limited by fissile burnup and by radiation damage, which may require periodic fuel replacement, flux limitation, or high initial enrichment. These problems could be alleviated by using liquid fuels.

Liquid fuel MSRs

With *liquid* fuels there is no clad damage or refueling shutdown. On-line processing could adjust fissile composition, remove some fission products, and recycle transuranics (TRUs).

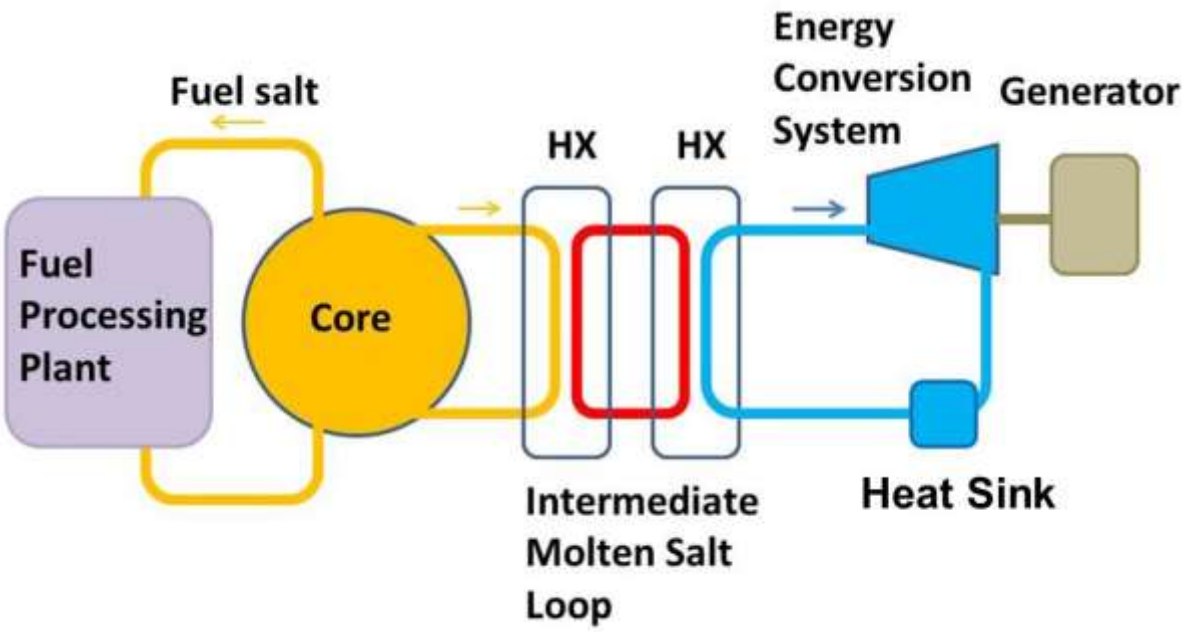

Figure 3. Simplified view of a liquid fuel MSR power plant. HX=heat exchangers. Drain tanks and pumps are not shown

Figure 3 shows a simplified liquid fuel MSR. The fission reactions occur in the central core (yellow) containing fissile materials in a carrier salt, such as $^{7}LiF\text{-}BeF_2\text{-}^{233}UF_4$. Hot fuel salt flows to a heat exchanger HX, where intermediate coolant salt (red) carries the heat to the energy conversion system (blue) that generates electricity. The intermediate coolant loop isolates the energy conversion system from the core radioactivity (fission products, actinides, and tritium).

If the reactor is a *breeder* (CR>1.0) the fissile materials in the fuel salt would sustain the chain reaction, and the extra neutrons emitted could breed fissile ^{233}U from fertile ^{232}Th and ^{239}Pu from ^{238}U, using the following reaction sequences:

$$n + {}^{238}U \rightarrow {}^{239}U \xrightarrow[24\ min]{} {}^{239}Np \xrightarrow[2.4\ day]{} {}^{239}Pu \qquad \text{U-Pu cycle}$$

$$n + {}^{232}Th \rightarrow {}^{233}Th \xrightarrow[22\ min]{} {}^{233}Pa \xrightarrow[27\ day]{} {}^{233}U \qquad \text{Th-U cycle}$$

where the beta decay half-lives are shown below the arrows. The intermediate isotopes ^{239}Np and ^{233}Pa could be destroyed by (n,γ) or (n,2n) reactions, so the breeding yield would be improved by removing these intermediates from the neutron flux. Both cycles can be started with Pu and low-enriched uranium (LEU) from used LWR fuel, which will alleviate the high-level waste (HLW) disposal problem.

The transition to a breeder fuel cycle could extract more of the nuclear energy potentially available from U and Th. About 90% of the thorium energy could be utilized in liquid fuel MSR breeders with online or fast batch processing. Thorium is an inexpensive byproduct of rare earth, iron, titanium, and phosphate mining. About 80,000 tonnes of thorium could be recovered from mines producing 7 million tons of titanium per year, which would greatly exceed the need of a worldwide thorium breeder reactor fleet.

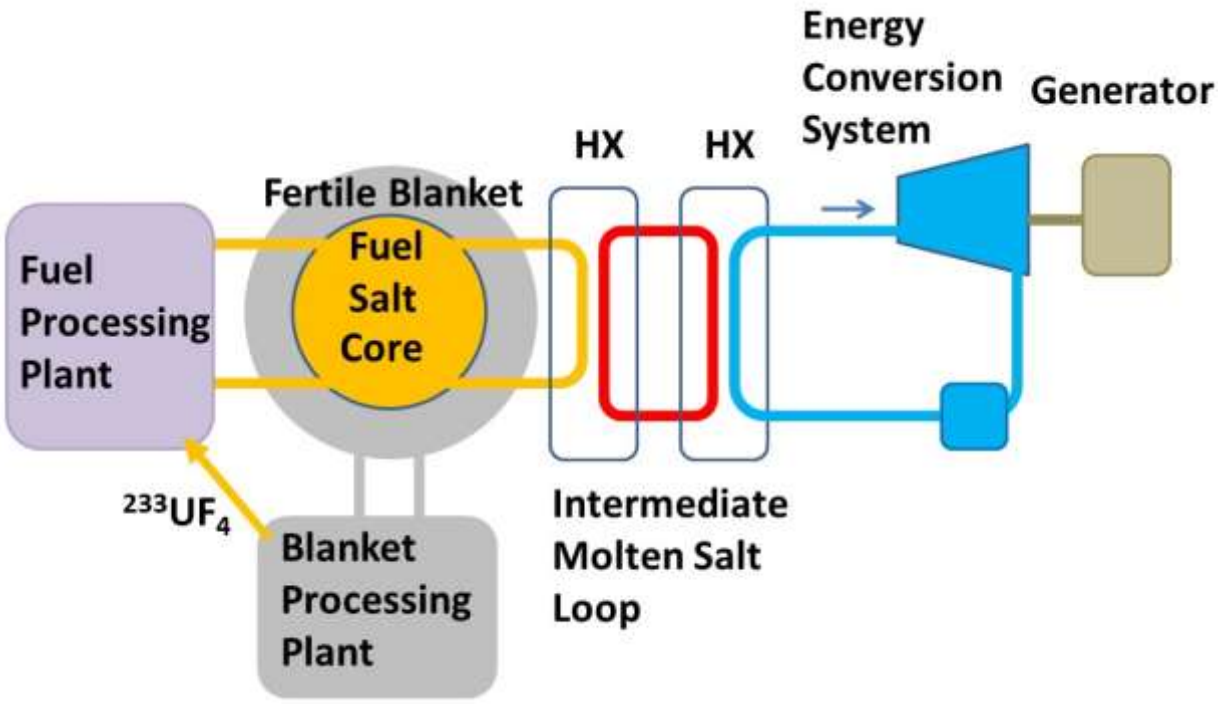

Figure 4. A two-fluid reactor with the Th-U fuel cycle. Drain tanks and pumps are not shown

To improve the breeding ratio a two-fluid breeder reactor could have a separate blanket region around the core, **Figure 4**. The blanket region (gray) contains ^{7}LiF-BeF_2-ThF_4 fertile salt, where neutrons leaking from the core breed ^{233}U. Some of the protactinium may be separated in the blanket salt processing. It decays into $^{233}UF_4$, which is inserted into the fuel salt at a rate that helps control core reactivity. An additional system (not shown) extracts heat from the blanket

salt. Liquid fuel reactors may use a thermal neutron energy spectrum or an epithermal/fast spectrum.

Thermal spectrum

The U-Pu fuel cycle could not breed well at thermal energies, but the Th-U cycle can breed some. One example is the FUJI-U3 design, **Figure** 5, which would produce 200 MWe in a large core (H = 4.66 m, R = 2.36 m) with a low neutron flux ($4x10^{-13}$ cm^{-2} s^{-1}) and power density 5.5 $MWth/m^3$ to achieve 30 year graphite lifetime.

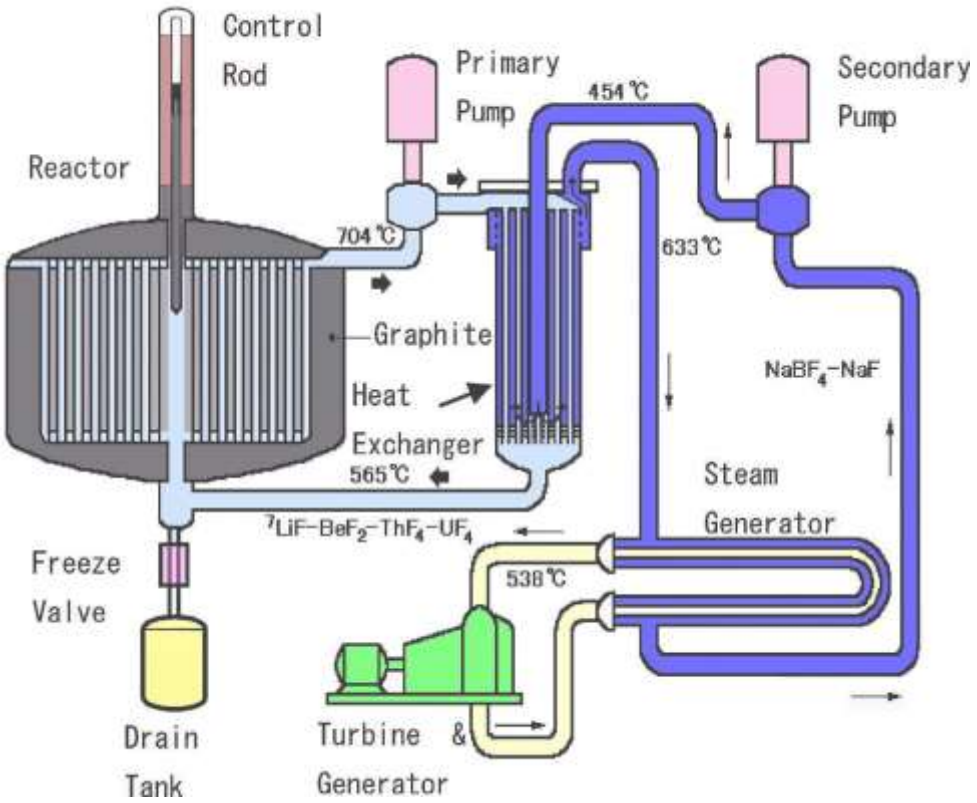

Figure 5. One loop of the FUJI-U3 liquid fuel thermal MSR.

The Th-U fuel cycle salt comprises ThF_4(12%) + UF_4(0.24%) + LiF (71.8%) + BeF_2(16%). It has three core regions to flatten the flux profile, and its CR = 1.01 without reprocessing. The initial primary loop contains 1.13 tons ^{233}U, 56.4 tons Th, and 163 tons graphite. The fissile fuel consumption in 30 years is 7.8 tons, and the core at end of life contains 7.9 tons, which may be transferred to the next reactor. The net production of minor actinides (MA) is 23 kg, compared to 543 kg in a boiling water reactor (BWR). The FliBe coolant will probably require ^7Li enrichment

and multiple barriers to contain tritium. Thermal liquid fuel reactors can have excellent safety features and long core life; but they would probably have larger heat exchangers, higher reprocessing rates, much lower power densities than epithermal/fast reactors (~ 6 MWth/m^3 compared to 300 MWth/m^3), and higher capital cost per MW.

Epithermal/fast spectrum

The molten salt fast reactor (MSFR) under design in Europe is a 3000 MWth reactor with a compact cylinder vessel 2.25 m high and 2.25 m diameter, and total fuel salt volume of 18 m^3, fissile inventory 3.5 tons/GWe. The fuel salt is LiF-ThF_4-UF_4, with 22.5 mol% heavy metal fluorides, peak temperature of 750 °C. The fuel salt flows freely upward through the central part of the core without any solid moderator. The 16 return loops around the core contain pumps and heat exchangers (**Figure 6**).

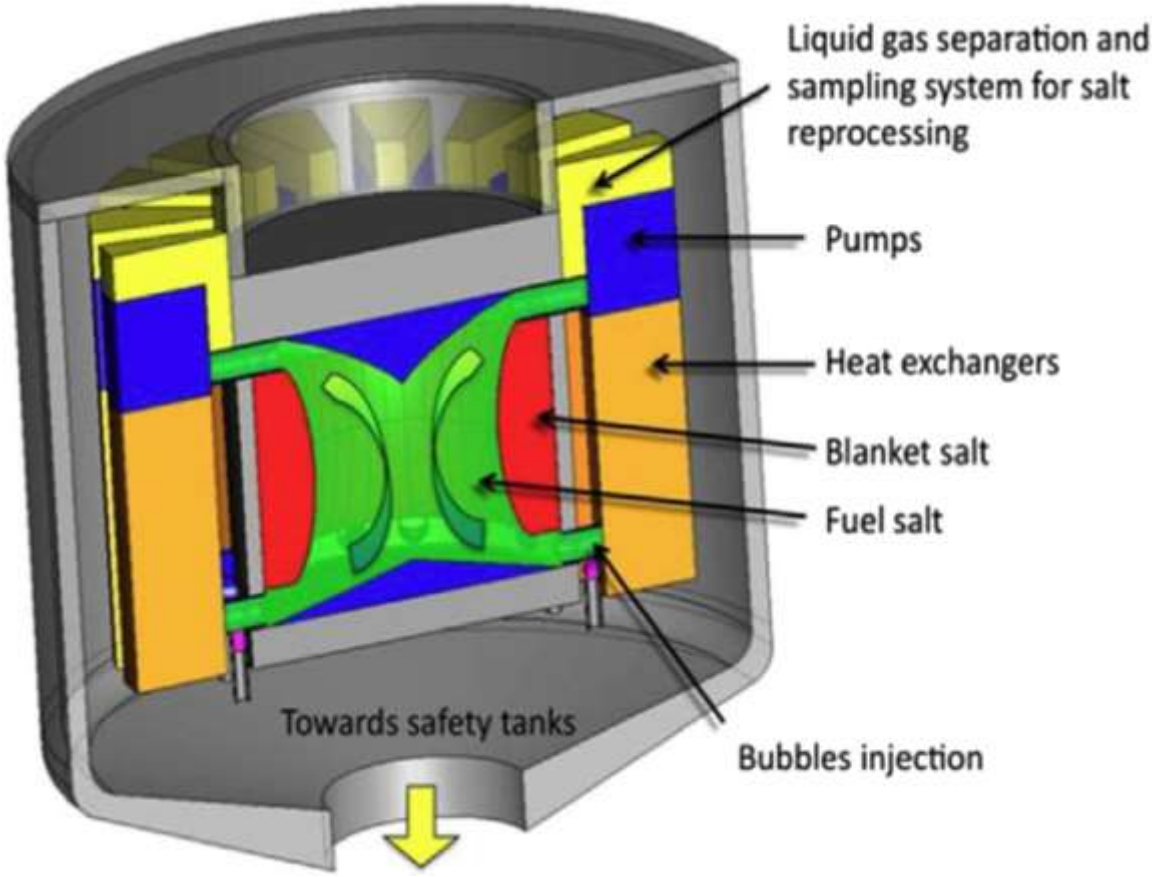

Figure 6. MSFR vessel with 16 heat exchanger loops.

He bubbles are injected at the fuel inlets and separated from the liquid at the core outlet carrying volatile fission products. Half of the fuel salt volume is in the core and half is in the external circuit (salt collectors, salt-bubble separators, fuel heat exchangers, pumps, salt injectors and pipes). The fuel salt runs through the total cycle in 3.9 s.

Liquid fuel MSRs could eventually have several advantages over **solid-fuel** MSRs:

- No **manufacture** or qualification of fuel pebbles, clad, rod assemblies
- Fuel **lifetime** not limited by radiation damage; no clad leaking or fuel melting issues
- Power loss → **drain plugs melt** → core drains into passively cooled tanks
- Removal of some **fission products** → better neutron economy, lower core decay heat
- Easy removal of **noble gases** → no xenon poisoning
- Continuous reactivity adjustment and low core **excess reactivity**
- Large negative **temperature coefficient** of reactivity, due to expansion and Doppler effect
- With ^{233}U or ^{239}Pu breeders less need for **^{235}U enrichment**.
- Fissile materials remain **onsite**. Bred ^{233}U could be "**denatured**" with ^{238}U for transport off site.
- Some **actinides** from LWRs could be burned, reducing the waste disposal volume.
- Capital and fuel costs could be lower than those of LWRs and coal power plants. [5]

Between solid fuel reactors and liquid fuel reactors is a combination called "Stable Salt Reactor".

Stable Salt Reactor (SSR)

Instead of solid fuel rods the SSR design uses static liquid fuel inside metal tubes with a vented gas plenum at the top of each, surrounded by a large pool of coolant salt., **Figure 7.** This separates fuel and primary coolant salts without a separate heat exchanger, and natural convection keeps the liquid fuel peak temperature low. Coolant salt adjustment facilitates good control of corrosion.

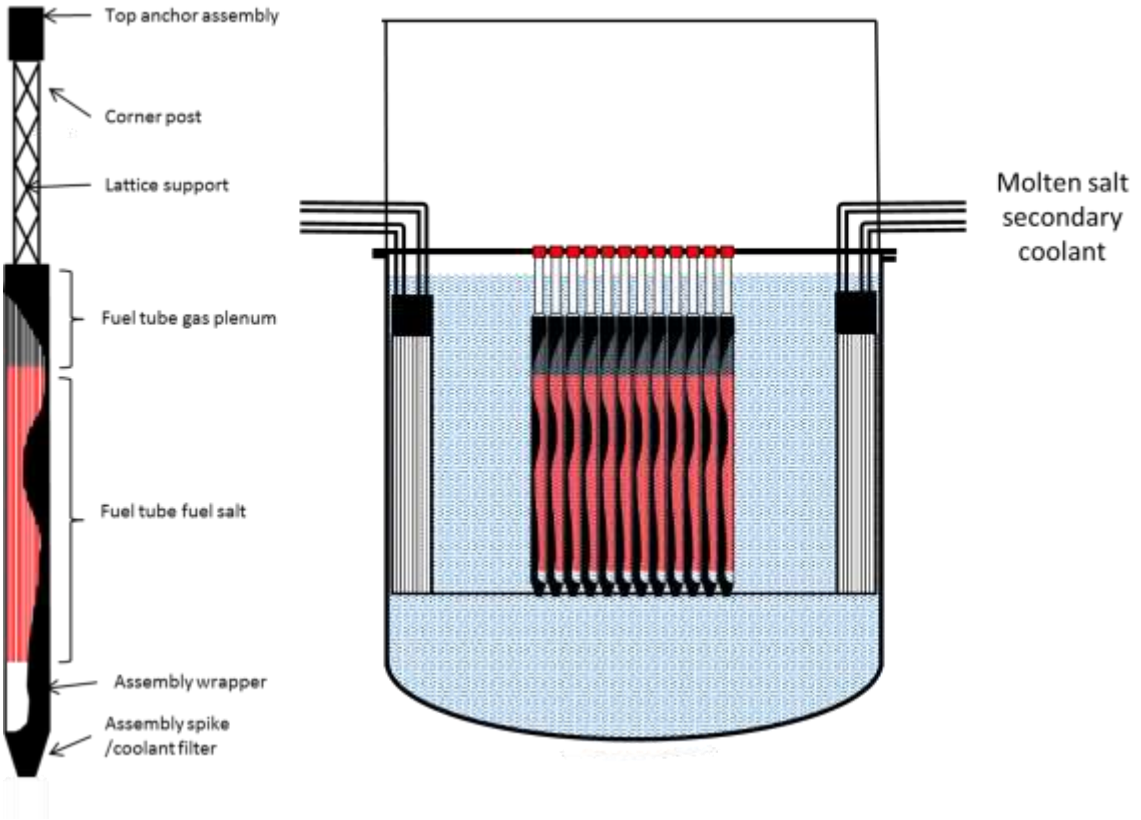

Figure 7. Stable Salt Reactor design

All MSR design aspects -- neutronics, heat transfer, corrosion, induced radioactivity, and cost -- are strongly affected by the choice of salts.

Fluorides and chlorides

A few possible salts are ^{7}LiF-BeF_2 ("FliBe"), LiF, LiF-NaF-KF ("FLINAK"), and $Na^{37}Cl$. The melting temperature of pure NaCl (801 °C) would be too high for MSR use, but addition of heavy metal chlorides reduces it to useable values. Fluorides are usually chosen for thermal spectrum MSRs, both solid fuel and liquid fuel, because chlorides absorb too many neutrons while they are slowing down to thermal energies. For epithermal/fast neutron energy spectra either chlorides or fluorides may be chosen.

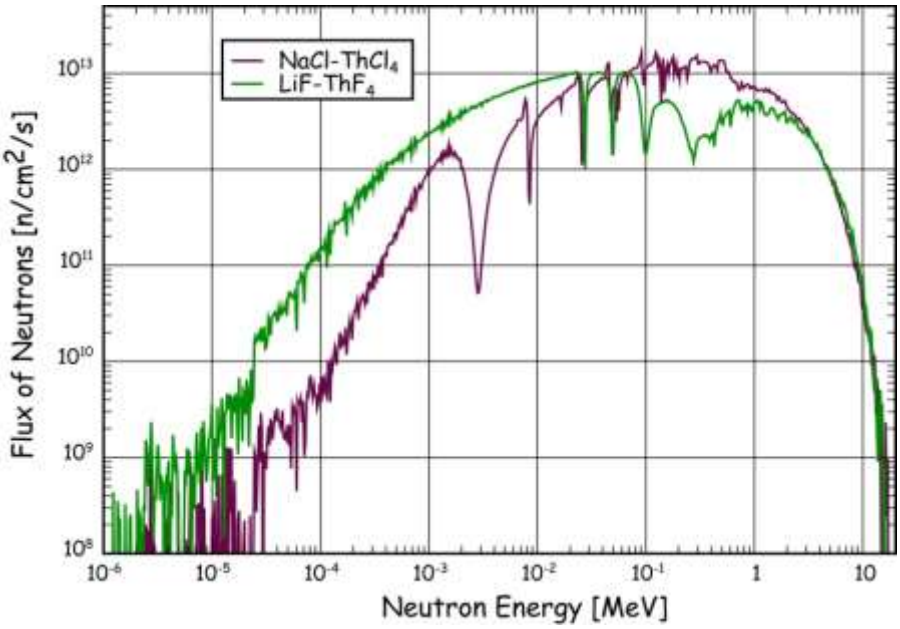

Figure 8. Fast and epithermal energy spectra associated with chloride and fluoride salts

Figure 8 compares the fast spectrum of a chloride salt core (peak at 0.1-2 MeV and strong absorption at 1-20 keV) with the epithermal spectrum of a fluoride salt core (relatively high flux below 100 keV).[6] The hard spectrum of chlorides allows fast neutrons to cause significant radiation damage to structural alloys.

The isotopes of chlorine are ^{35}Cl (76%) and ^{37}Cl (24%). ^{35}Cl has a high neutron cross section, reducing the conversion ratio and yielding ^{36}Cl, which emits a high-energy beta with 0.3 million year half life, creating a radioactive waste concern. Therefore, some designs would employ isotopic separation to use enriched ^{37}Cl, which produces much less ^{36}Cl. The hard spectrum of NaCl also produces more radiation damage in structural materials. In one optimization study of a molten salt fast reactor (MSFR), the fluoride core (LiF-ThF_4) achieved CR = 1.126 and the chloride core (NaCl-ThF_4) had CR = 1.040. **Table 1** compares chlorides and fluorides.

Table 1. Comparison of fluorides and chlorides in an MSFR .

Fluorides	Chlorides
more moderation then chlorides	little moderation
longer materials lifetimes	more radiation damage
lower parasitic capture of F	higher parasitic capture of Cl
good with thermal or fast spectrum	poor with thermal spectrum
tritium production by LiF, BeF_2, …	production of ^{36}Cl

need to remove ^{6}Li	need to remove ^{35}Cl
MSFR CR = 1.126	MSFR CR = 1.040

Fluorine has lower parasitic capture of neutrons than chlorine, but fluorides of Li and Be produce large amounts of tritium.

Tritium

The tritium ***source*** can be reduced by avoiding Li and Be, by isotopic enrichment of ^{7}Li (removing most of the ^{6}Li), or by using NaCl in a fast chloride MSR. China has enriched Li to over 99.95 % ^{7}Li using a cascade extraction process with 160 stages of centrifuges.

Tritium ***transport*** can be reduced by using an intermediate salt loop, such as $NaNO_3$ and KNO_3 (which help trap tritium), and by coating tube surfaces with low-permeability materials.

Tritium can be ***removed*** by permeation through high-permeability windows like palladium and by trapping in materials like titanium and some forms of graphite.

Good tritium technology has been developed for the Canada Deuterium Uranium (CANDU) reactors, for nuclear fusion experiments, such as the International Thermonuclear Experimental Reactor (ITER), and at the Pacific Northwest National Laboratory (PNNL), which is in use at the Watts Barr nuclear power plant. Processing the molten salt fuel is more complex.

Fluoride Salt Processing System

Prior to operation the fuel tetrafluorides are prepared by exothermic reactions of their oxides with hydrogen fluoride at T ~ 500 °C:

$$UO_2 + 4HF = UF_4 + 2H_2O$$

$$ThO_2 + 4HF = ThF_4 + 2H_2O$$

During reactor operation the fuel processing system will remove some fission products, add fertile salt, adjust the fissile fuel content to control core reactivity, and possibly insert some actinides for incineration.

The fission products (FPs) Xe and Kr can be removed by bubbling He through the salt and then stored in activated charcoal for decay. This reduces the excess reactivity that would be required to overcome Xe and Kr poisoning of the neutron flux. Some metals (like Nb, Mo, Ru, Cs, Te, Tc) are also removed by the He gas, because they do not form stable fluorides in the salt.

Uranium can be **volatilized** by a countercurrent of F_2 gas at T ~ 500-600 °C:

$UF_4 + F_2 \rightarrow UF_6$ gas

$ThF_4 + F_2 \rightarrow ThF_6$ gas

Formation of PuF_6 is more difficult. The gaseous products UF_6 and ThF_6 bubble out of the salt. Then they can be trapped and separated from accompanying FP fluorides by successive beds of NaF pellets and MgF_2 pellets at various temperatures.

After U removal by volatilization, liquid Bi containing Li can mix with the salt flow through multistage extraction systems to separate Th, U, and Pa from fluorides and chlorides at 500-700 °C. For example

$4Li(Bi) + PaF_4 \rightarrow Pa(Bi) + 4LiF$

This can remove ^{233}Pa from the core. Then it can be fluorinated and decay to ^{233}U without its destruction by neutrons. Other FPs, such as lanthanides, can be separated by Cyclic Voltammetry in FliBe, in FLINAK, or in $LiF\text{-}CaF_2$.

Development of this system could begin with easy separations (Xe, volatiles, U…) and gradually advance to more difficult separations (such as lanthanides). Sophisticated systems could take many years to develop, but simple ones could be deployed promptly. Another concern that could delay deployment is nonproliferation.

Nonproliferation of Th-U Fuel Cycle

The nonproliferation aspects of the U-Pu fuel cycle are well known, so only the Th-U cycle will be discussed here. ^{233}U has low reflected critical mass (8.4 kg), low specific spontaneous fission yield (0.5 $s\text{-}kg^{-1}$), and low heat output (0.3 W/kg), which make it attractive for weapons. Liquid thorium fuel monitoring needs to measure liquid mass density and flow rate continuously, which will be more difficult than counting individual fuel rods, and some fuel is outside the core. Online fuel processing may remove some fission products, making the fissile material easier to handle. The pumps, valves, tanks, and pipes of the processing facility must be monitored to prevent diversion of small quantities of fissile materials. Removal of ^{233}Pa before its decay to ^{233}U can yield pure ^{233}U, so this process must be strictly controlled. Radioactive fuel rods could be removed and transported in a cask, while theft of liquid fuel would require the thieves to break into the pipes of the primary loop and to pump hot, highly radioactive liquid fuel into a container for removal. The whole primary loop must be inside a safeguarded containment. The 2.6 MeV gammas from ^{232}U daughter product ^{208}Tl makes weapons manufacture hazardous, and they could make stolen fuel detection easier.

Some thorium is already being deployed in LWRs and HWRs. The current readiness level of safeguards technology for ^{233}U is low. We need better gamma and neutron analysis techniques for thorium and ^{233}U. Multigroup analysis codes are needed for ^{233}U gammas. Additional technologies may be applied, such as laser induced breakdown spectrometry. LIBS can identify salt components, but cannot measure the redox level, which can be done by optical spectroscopy and by voltammetric analysis of Th and U salts. [7] An advantage of liquid fuel onsite processing is that no fissile material would need to leave the secure site for fabrication. If desired, the bred ^{233}U could be immediately blended with ^{238}U, so that no weapons-usable uranium would be present (a “denatured” thorium fuel cycle), but this would result in a lower breeding ratio and a higher inventory of TRUs. The appropriate IAEA safeguards should be designed simultaneously with the fuel processing facility (“safeguards by design”) in the many ongoing projects.

Reactor Design Projects

Many MSR concepts are at various stages of pre-conceptual and conceptual design. In North America MSR concepts are studied by Terrestrial Energy,[8] Thorcon, Flibe Energy, Elysium Industries, TerraPower, Oak Ridge National Laboratory, Kairos Power, universities, and others. MSR designs are also proposed by groups in France, UK, Denmark, Germany, Russia, India, Japan, Korea, China, and others.

Europe has a large multinational group developing the Molten Salt Fast Reactor.[9] TerraPower is developing dozens of Microloops (small natural circulation flow loops) to study salt properties and an Integral Effects Test (2019), to be followed by Test Reactor operations in 2025. The Shanghai Institute of Applied Physics has over 700 people working on solid fuel and liquid fuel MSRs, and several other Chinese laboratories are collaborating on component and salt development. [10]

International Cooperation

Several organizations promote MSRs or thorium energy:

Generation IV International Forum (GIF) is a group of countries that collaborate on six advanced reactor designs, one of which is an MSR. https://www.gen-4.org/gif/jcms/c_9260/public and www.samofar.eu

Thorium Energy World is a non-profit international organization, which holds Thorium Energy Conferences (recently Brussels 2018). Furthermore, the organization develops a web-based, open course on Thorium MSR, produces a documentary, and its website provides information on technical news and events. http://www.thoriumenergyworld.com/ http://www.itheo.org/

The **International Thorium Molten Salt Forum** provides information and advice about thorium research. http://msr21.fc2web.com/english.html

The **Thorium Energy Alliance** in the USA promotes MSR research and holds conferences (recently St. Louis 2017) www.thoriumenergyalliance.com/

The **International Atomic Energy Agency (IAEA)** facilitates cooperation, organizes technical meetings (Vienna 2016), and publishes technical documents. (www.iaea.org) The IAEA has activities in nuclear energy, nuclear safety, safeguards, nuclear sciences and applications, and technical cooperation (aid to developing countries).

MSR developers have a unique opportunity to create common international regulatory standards, because the regulations will be new. Such regulatory uniformity could facilitate faster and cheaper deployment of MSR reactors. An IAEA Coordinated Research Project to develop such a standard could be planned, but only if Member States requested the IAEA to do so.

Although conceptual design studies emphasize the potential advantages of MSRs, many issues could delay progress.

Development Issues

Structural materials may be degraded by radiation damage, chemical corrosion, and creep at high temperatures. Liquid fuel flowing out of the reactor continues to emit **delayed neutrons** in the surrounding pipes and valves, which make the plumbing radioactive, so remote maintenance is required. Solid fuel reactors and static liquid fuel reactors have less widespread activation, but radioactive fuel and structure still require remote operations and exposure controls.

The MSRE developed control systems, drain tanks, salt heaters, structural materials, and instrumentation. For MSR power plants the following developments are needed:

- For liquid fuel reactors, **chemical systems** to separate uranium, plutonium, thorium, molten salts, and fission products in a highly radioactive environment with remote handling are the most difficult challenge requiring development.
- Reliable **heat exchangers** for high temperature molten salt, such as molten salt to air heaters, steam generators, and salt-to-salt heat exchangers that prevent freezing, reduce fuel salt volume, and accommodate potential noble metal plateout or clogging.
- **Pumps** and valves
- Better corrosion-resistant, high temperature **materials** for reactor vessel and piping are desirable, especially non-nickel materials to minimize helium generation damage. Designers could develop improved materials or limit the operational parameters to fit existing materials. ASME standards and irradiation qualification standards are needed.
- If the salt contains **lithium,** inexpensive and proliferation resistant isotope separation methods would be desirable to minimize tritium generation by ^{6}Li.
- **Licensing** may be difficult, especially where the regulator has been focused on LWR technology for many decades. Regulators have little experience with liquid fuels.
- Safeguards for **nonproliferation** of nuclear weapons must be thorough. The IAEA has many publications and experts who can help achieve safeguards by design.
- Switching from LWR technology to MSR technology would be a massive paradigm shift. [11] Some companies and people may oppose the new technologies.
- Lack of regulatory experience → slow licensing
- Public ignorance of MSR safety and waste management advantages.

Conclusions

Nuclear power is safe and reliable, but many people are afraid of accidents and radioactive waste, so MSRs with stronger safety, waste incineration, and lower cost might gain better public acceptance. MSRs could

- ensure safety without auxiliary safety systems
- achieve high-temperature process heat and high-efficiency electricity generation
- cost less than water-cooled reactors
- utilize thorium and uranium fuel efficiently

- incinerate actinides from LWR spent fuels, reducing the need for long-term high-level waste storage
- compensate for swings in solar and wind power.

Governments and industry should support a broad spectrum of concepts, until one or two prototype reactors turn out to have superior reliability, cost, and environmental benefits. Construction of test reactors is needed to demonstrate the technology, in addition to more paper studies. Based on manpower and resources devoted to MSR development, China is the world leader.

We envision a world where most countries have affordable molten salt reactors burning thorium, uranium, and spent fuel actinides, producing electricity, hydrogen, desalinated water, and other process heat applications, with no serious accidents.

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