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OVERVIEW OF THE PHYSICAL PROPERTIES OF MOLTEN SALT REACTOR USING FLIBE

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Abstract: Currently, there are six Generation IV nuclear reactor designs in development. Four are fast neutron reactors, and all designs operate at higher temperatures that permit hydrogen production. Thus, the interest in fluoride salts has grown due to their high-temperature application in fission and fusion reactor designs. The aircraft propulsion project was the initial plan, which used molten salt as a coolant and was started by Bettis and Briant in the 1940s. The molten salt reactor has been designed to operate at temperatures of 700 to 800°C with fissile material dissolved in a molten fluoride salt composition. Molten fluoride salts are stable at high temperatures, show good thermodynamic properties, and can also dissolve actinides and fission products easily. It creates a candidate for a thorium reactor with more than 45% efficiency. The purpose of this work was to investigate the physical characteristics of two systems of fluoride salt combinations, namely LiF-BeF2 (FliBe) and LiF-NaF-KF (FliNaK), including melting temperature, density, and heat capacity. The aim is to characterize the advantages of the various designs proposed for Generation IV by reviewing properties evidenced by safety improvements and limitations.
Keywords: MSR, FLiBE, LFTR, FLiNaK, Generation-IV.

INTRODUCTION

In the last two decades, nuclear systems have tried to improve safety after three generations of nuclear power plants. Early prototypes were the first reactor generation, which started in 1950, and the commercial power reactor represents the second generation, which began in 1960 (Dolan et al., 2017). During the 1990s, a few news designs started. Around the world, in 2018, there were 413 reactors in operation in 31 countries, predominantly light water reactors (LWRs) and heavy water reactors (HWRs). Still, in 2002, most reactors of the second generation were active. Among the various reactor types, the United States Department of Energy (DOE) began efforts for the next generation in 2002. Generation IV, proposed for 2030, selected six designs: supercritical water-cooled reactor (SCWR), very-high-temperature reactor (VHTR), lead fast reactor (LFR), sodium fast reactor (SFR), molten salt reactor (MSR), and gas-cooled fast reactor (GFR) (Grape, Sophie, et al., 2014).

A thermal analysis of nuclear units involves many aspects, mainly thermodynamic behavior that shows coolant dependence. Nowadays, it works as light water H_2O, heavy water D_2O, helium, carbon dioxide CO_2, liquid metals, and molten fluoride salts (Holocomb et al., 2013). Water reactors have predominated in the last six decades, generating electricity in more than 20 countries for over six decades. A representative reactor fleet of around 70% is the pressurized water reactors (PWRs), which use water as a coolant and moderator at a very high pressure of 15 MPa to 17.1 MPa at 343 °C in the primary circuit and steam formed in a secondary loop (Romatoski et al., 2017). The Westinghouse AP1000 is a conventional PWR 2-loop design working in thermal spectrum neutrons with solid fuel UO_2 (Schulz Terry, 2006). Operating with a Rankine cycle, each loop comprises a single hot leg, two cold legs, a steam generator (SG), and two reactor coolant pumps.

Molten salt reactors (MSR) operate with fissile materials, and fluoride salts are mixed uniformly to create the liquid fuel. The fast reactor uses liquid salts that have reached temperatures exceeding 450 °C, such as FLiNaK salt (LiF-NaF-KF) or FLiBe salt (LiF-BeF_2). Reactor operating temperatures range from 550 °C to 750 °C. For thermal and fast-flux breeder reactors, the molten salt (LiF-BeF_2) can be used with dissolved fuels (MacPherson, H. G., 1985). In order to quickly fill the core geometry, maximize electricity output, and control criticality, something immediately transmits fission energy, which is then produced as heat in the fluid. Reprocessing the fuel also takes place without shutting down the reactor. Reprocessing the fuel also takes place without shutting down the reactor.

For instance, LiF-BeF_2 (69–31 mol%) melting at 505 °C at a temperature of 600 °C reveals a density of 1.96 g/cm^3, a heat capacity of 2093 J/Kg-K, and a kinematic viscosity of 7.488 g/m-s (Cantor, S., 1968). Even though all intermediate spectra may exist, the molten salt reactor can use two spectrum categories. The thermal spectrum means that most of the neutrons that cause a decrease in fissions have an energy corresponding to the thermal agitation around them. After emitting around 2 MeV, the neutrons suffer successive shocks. This moderation is standard for the light-water reactor, the heavy-water reactor, or graphite. The MSR moderator is molten salt. A fast spectrum means neutrons are slightly slower; they produce fission or are captured long before reaching thermal energy.

Ideally, the coolant must have a higher melting point and a suitable solubility for fissile U-235 or fertile Th-232 that dissolves, forming the liquid fuel. The physical properties are essential for calculating the thermal behavior of the core: density, heat capacity, viscosity, thermal conductivity, and vapor pressure. Early experiments worked on the thermal spectrum, which needs graphite as a moderator in the primary circuit. Early in the MSR project, it operated with LiF-BeF_2-ThF_4-UF_4 (72,16,12,0.4) mol%. In contrast, we employed NaF-$NaBF_4$ as the secondary coolant, running at 705 °C.

Since 2002, MSR and MSBR have concentrated on a few types of molten fluorides used in fast and breeder reactors. These fluorides comprise lithium, beryllium, sodium, and potassium (LiF, BeF_2, NaF, and KF). Studies for the molten chloride compounds KCl and LiCl have also been discovered (Cadwallader, Lee Charles, and Longhurst, Glen Red, 1999). The MSR design envisioned for generation IV contains the fissile (UF_4) and the fertile (ThF_4) dissolved in ionizing fluoride salts (Jerden, James, 2019). The liquid fuel state offers several advantages as it improves reactor safety. The liquid fuel also serves as a primary coolant, showing low vapor pressures with high boiling points >1400 °C, and the total pressure is 0.1 MPa (Davis Cliff, 2005).

Thus, the fissile material (U-235, U-233, or Pu-239) forms an inorganic liquid pumped at low pressure through the reactor vessel and the primary circuit. The fission generates heat transferred in a heat exchanger to a secondary coolant, a molten salt (Cervi E. et al., 2019). A natural evolution of concepts pretended to breed U-233 from Th-232 and Pu-239 from U-238 and run the thermal spectrum using molten salt as a coolant. In the 1976s, when ORNL started, the denatured molten salt reactor (DMSR) arose using low enrichment, or denatured U-235. Then, the DMSR can run with about 3450 kg of 20% enriched 235U and work for 30 years.

THEORETICAL BACKGROUND

One of the advanced designs incorporated into the most recent evolution of nuclear reactors is fluoride-salt high-temperature reactors (FHRs). Combining the tristructural particle fuel (TRISO) with fluoride salts as a coolant produces a ground-breaking concept operating on high safety. The core of TRISO is a fissile kernel with a sphere-like shape of 3cm diameter, usually UO2, UCO, or UN. A porous carbon buffer encircles the kernel, and the discharge of fission gas follows the central fuel sphere. Kernel and buffer show additional layers of pyrolytic carbon (IPyC), followed by silicon carbide (SiC) and outer pyrolytic carbon (OPyC), which are placed after the buffer surrounds.

However, the FHRs comprise a natural evolution of the Very-High-Temperature Reactor (VHTR), use a thermal neutron spectrum, and are helium-cooled with graphite as a moderator. The primary goal of VHTR is the cogeneration of hydrogen and electricity through thermochemical processes at temperatures higher than 1000 °C. The technological foundation for VHTR is the tristructural isotropic (TRISO)-coated particle fuel. TRISO spheres of 3 cm diameter comprise a core of uranium oxycarbide (UCO) and three carbon layers, one silicon carbide layer, and one silicon carbide layer. It increased the thermal efficiency by 50% at this high temperature. The steam generator (SG) in the VHTR uses heat transfer tubes made of Inconel 617 alloy, Incoloy 800H alloy, and Hastelloy X alloy as alternatives. Table 1 shows six GEN-IV reactors designed and characterized by the neutron spectrum.

The project for a supercritical water-cooled reactor is based on supercritical (25 MPa, 540 °C) and super-supercritical (35–37 MPa, 620–700°C) water steam. SCWRs are basically LWRs operating at higher pressure and temperatures with a direct once-through cycle. The primary system has been simplified for the better thanks to SCWR's adoption of a direct thermal cycle and the removal of the steam generators, pressurizers, and central circulation pumps (Schulenberg et al., 2011).

Additionally, the SCWRs can operate in the fast, thermal, or mixed spectrum, divided into two design concepts: pressure tube and pressure vessel. The primary circuit operates at 25.0 MPa. The feed water temperature is 280 °C, and the average core outlet coolant temperature is about 500 °C. Besides, SCWR designs can use UO_2, $(Pu\text{-}U)O_2$, and thorium fuels with an active fuel length of 4.2 m. SCWR designs have used alloys such as austenitic steel, ferritic-martensitic alloys, and nickel, chrome, aluminum, and yttrium (NiCrAlY) alloys, in part because of their high resistance against creep and radiation.

Around the world, several institutes have used liquid metal technology as a coolant for fast reactors. The clementine reactor first used mercury as a liquid metal but switched to sodium and lead because of some advantages. Then, sodium and potassium (NaK) were used as coolants in the experimental breeder reactor (EBR-I). In the United States, sodium-cooled fast reactor prototypes were relevant between the 1960s and the early 1990s. Recently, European initiatives built the Advanced Lead-cooled Fast Reactor European Demonstration (ALFRED) in Europe, a 300 MW pool system shaped to show the practicality of the LFR architecture for commercial use. Heavy liquid metals such as lead, or Pb-Bi, composed of Pb (44.5%) and Bi (55.5%) have been used as coolants for fast reactors since the 1950s (Gromov et al., 1997). Some liquid-fuel reactors have used LBE as a coolant. LBE eutectic is also considered part of the Generation IV reactor initiative for use in critical and subcritical speed-up-driven systems (ADS).

In the early 1990s, Russia developed the BREST, a pool-type lead-cooled fast reactor (LFR) loaded with uranium-plutonium mononitride (PuN-UN) (Subbotin et al., 2002). The molten lead fast reactor does not significantly moderate neutrons, acts as a reflector for neutrons, and undergoes almost no neutron activation (Weeks, J. R., 1971). The combination of the properties of nitride fuel and the low-moderating lead coolant permits a breeder ratio greater than one. Table 2 describes the thermal and physical properties of liquid metal and fluoride molten salt coolants.

In sodium-cooled fast reactors (SFRs), liquid metals have excellent heat transfer

Fast Neutron Spectrum		
Reactor Design	Coolant/Moderator	Temperature (°C)
Gas-Cooled Fast Reactor (GFR)	Helium	490-850
Sodium-Cooled Fast Reactor (SFR)	Sodium	370-550
Lead-Cooled Fast Reactor (LFR)	Lead, Lead-Bismuth (LBE)	Pb(550-800), LBE (420-540)
Molten Salt Fast Reactor (MSFR)	Fluoride salt +(UF_4/TF_4)	T(inlet) =700-800
Thermal and Epithermal Spectrum		
Very-High Temperature Reactor (VHTR)	Helium / Graphite	490-1000
Supercritical Water-Cooled Reactor (SCWR)	Light water/Light water	300-625
Molten Salt Reactor (MSR)	Fluoride salt+ (UF_4)/Graphite	T(outlet) =700-800

Table 1. Generation IV, six reactor designs, and neutron spectrum

Physical properties	He 7.5 MPa	Na (700 °C)	$^{7}LiF-BeF_2$ (66-34)	$NaF-ZrF_4$ (59.5-40.5)	^{7}LiF-NaF-KF (46.5-11.5-42)	$^{7}LiF-ThF_4$ (72-28)
Weight (g/mol)	4	23	33.0	92.7	41.3	87.99
Melting point (°C)	-272.2°C	98	459	500	454	555
Boiling point(°C)	-268.93	883	1727	1350	1570	1674
Density (Kg/m^3)	3.8	845	1.994	2.8264	2.718	4470
Heat capacity (kJ/kg·K)	5.19	1.27	2.414	1.161	1.151	1.05
Thermal conductivity (W/m·K)	0.281	62	1.0	0.49	0.92	1.2
Viscosity (cp)	0.041	0.182	5.6	5.1	2.9	16.74

Table 2. Comparative of a few physical properties of fluoride salt and liquid metals as coolant

Reactors Design	MK-1	MSFR	AHTR	MSR FUJI	PBMR
Power (MWth)/(MWe)	236/100	3000/1500	3400/1530	450/200	400/175
Fuel Material	TRISO/UCO	$^{233}UPuF_3-ThF_4$	1530	$^{233}UF_4-ThF_4$	UO_2
Coolant	$^{7}LiF-BeF_2$	LiF	$^{7}LiF-BeF_2$	$^{7}LiF-BeF_2$	Helium
Moderator	Graphite	-	Graphite	Graphite	Graphite
Inlet/outlet temperature (°C)	600-700	650-750	650-700	565-704	250-750
Power density (MWth/m^3)	22.7	48	12.9	7.3	4.0
Thermal efficiency (%)	42.37%	50%	45.0%	44.44%	40.0%

Table 3. High-temperature thermal designs FHR-PB, AHTR, MSR, PBMR, and the fast design MSFR

Fluoride salts	$LiF–BeF_2$ (66–34)	$NaF-NaBF_4$ (8–92)	LiF–NaF–KF (46.5–11.5–42)	$LiF–ThF_4$ (78–22)	$LiF–BeF_2–ThF_4$ (71.7–16–12.3)
Specific heat (J/Kg-K)	2.39	1.506	1.88	1.5	1.5

Table 4. Specific heat capacity of FLiBe and fuel compositions

Molten salt reactors	$T_{(Inlet)}$= (°C)	$T_{(Outlet)}$ = (°C)	Coolant	Moderator	Power (MWe)
MSRE	635	663	Fluoride salt	Graphite	2.6
LFTR	500	650	Fluoride salt	Graphite	250
MSFR	630	650	Fluoride salt	--	1500
Thorcon	565	704	Fluoride salt	Graphite	500
MSR-FUJI	565	704	Fluoride salt	Graphite	200

Table 5. The inlet and outlet coolant temperatures of MSRs

characteristics. In addition, it can withstand low pressure despite sodium's high thermal conductivity and lower Prandtl number (Tenchine, D., 2010). Besides, liquid metals have excellent heat transfer properties and are adaptable to low pressure in sodium-cooled fast reactors (SFRs).

Reactor designs like the high-temperature gas reactor (HTGR) were Peach Bottom Unit 1, which used helium as a coolant and was built, tested, and operated commercially with success from 1966 to 1974. During the 1980s, the U.S. government established HTGR research and development plans. In 1991, Japan began operating the High-Temperature Engineering Test Reactor (HTTR), which comprised a gas-cooled reactor with an outlet temperature of 850 °C to 950 °C and employed a prismatic core like the Chinese HTR-10 (Suh et al., 2022). Between 1994 and 2009, South Africa developed the pebble bed modular reactor (PBMR), which comprises an HTGR using a helium-cooled, graphite-moderated pebble bed reactor with a multi-type of fuel (Reitsma, 2006). Table 3 depicts the thermal characteristics of a few high-temperature reactor designs.

In 2006, Oak Ridge built the Advanced High-Temperature Reactor (AHTR) based on the benefits of ceramic-coated particle fuel coupled with high-temperature molten salt coolant at low pressure and planned for hydrogen production. The AHTRs operate at 900 °C, like the VHTR, using helium as a coolant. The AHTR concept supports several coolants, such as $^{7}LiF\text{-}BeF_2$, $NaF\text{-}BeF_2$, LiF-NaF-KF, and $NaF\text{-}ZrF_4$ (Williams et al., 2006).

PHYSICAL PROPERTIES OF FLUORIDE MOLTEN SALTS

The fluoride salt mixtures have both beneficial and undesirable aspects; the two most important are Li-7 fluoride and beryllium fluoride. The lithium isotope Li-6 represents 7.5% of natural lithium and is far too absorbent of neutrons to be a suitable component of a reactor fluid. In molten fluoride salt, the density decreases with increasing temperature (Sohal et al., 2010). Figure 1 shows the density behavior of the fluoride salt $LiF\text{-}BeF_2$ mixed with fertile ThF_4.

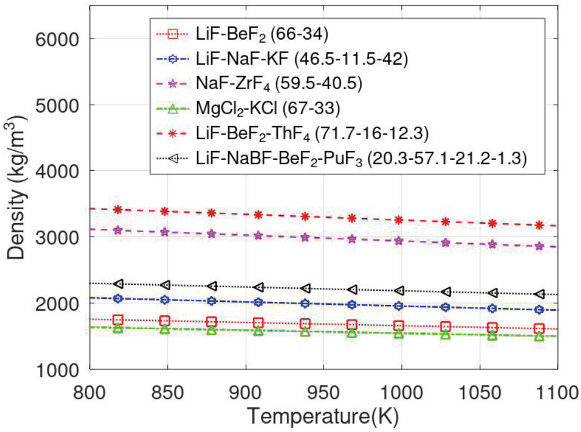

Figure 1. The density of liquid molten fluoride salts

The thermal conductivity of $LiF\text{-}BeF_2$ (66–34 mol%) is practically constant in a liquid state of 1.1 W/mK, independent of the temperature. Equations 3 and 4 describe the correlation given as a function of temperature in K for FLiNaK and $LiF\text{-}BeF_2\text{-}ThF_4$.

$$k(FLiNaK) = 0.36 + \frac{0.00056}{T} \quad (773K < T < 923K) \tag{3}$$

$$k(LiF - BeF2 - ThF4) = 0.42 + \frac{0.005}{T} \quad (773K < T < 923K) \tag{4}$$

The thermal conductivity, on the other hand, declines and somewhat rises with temperature. The thermal conductivity of FliBe is slightly higher than FLiNaK, around 20%, and FLiNaK has a higher specific heat capacity than FliBe. Figure 2 illustrates the

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thermal conductivity of fluoride molten salts.

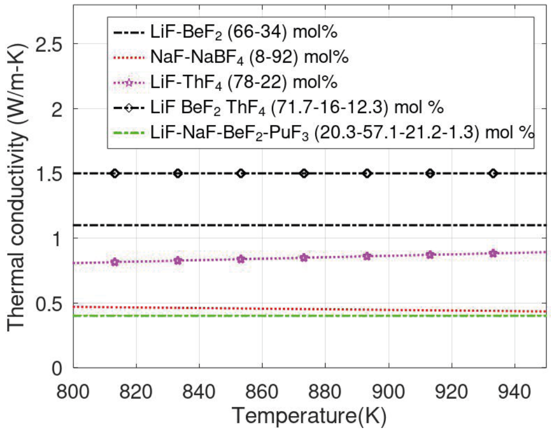

Figure 2. Thermal conductivity of fluoride salts LiF-BeF_2, LiF-ThF_4, and NaF-$NaBF_4$.

Liquid metals, such as sodium and lead, show a little viscosity compared to molten fluoride salts. Figure 3 shows the dynamic viscosity of molten salts and dissolved fuels.

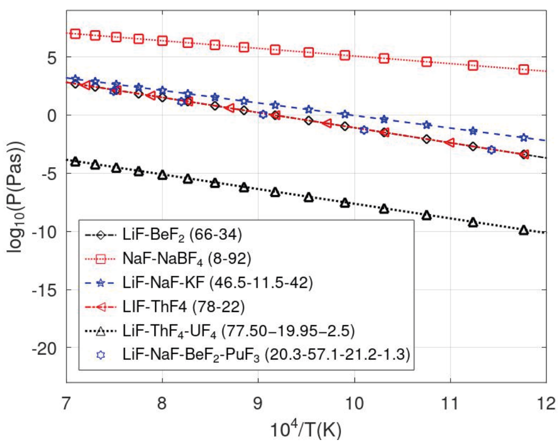

Figure 3. Dynamic viscosity of molten salts and dissolved fuels

The heat capacity of liquid in the temperature range (773–873 K), LiF–BeF_2 (66–34 mol%), is 2.41 J/Kg-K and 2.39 J/Kg-K (Douglas, T. B., Payne, W. H., 1969). Table 4 describes the specific heat capacity of liquid molten fluorides.

The molten salts FLiBe and FLiNaK exhibit several notable thermodynamic properties, including low pressure and high heat capacity, a high boiling point, satisfactory viscosity, and a suitable Prandtl number. There is good agreement with the Sieder-Tate correlation found for LiF–BeF_2–ThF_2–UF_4 and $NaBF_4$–NaF in the turbulent region at Reynolds above 15000. The heat capacities of liquids in the temperature range of 773–873 K for LiF–BeF_2 are 2.41 and 2.39 J/kg·K. Figure 4 depicts the Prandtl number in the temperature function used to calculate the convection heat transfer coefficient.

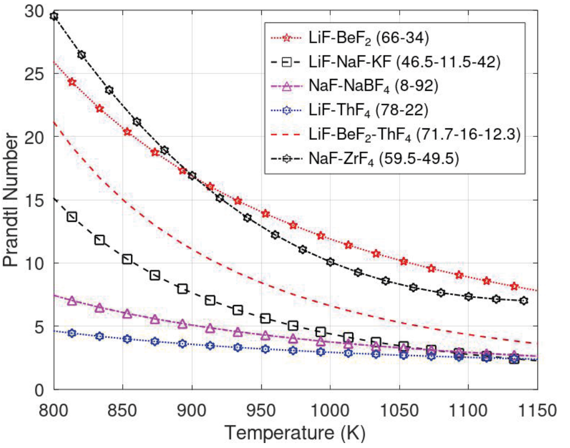

Figure 4. Prandtl number as a function of temperature at near atmospheric pressure

For many other designs, such as LWRs, micro-reactors, nuclear thermal propulsion, and salt-cooled reactors, the option TRISO is possible. TRISO kernels have included thorium carbide (ThC_2), thorium dioxide (ThO_2), plutonium dioxide (PuO_2), uranium dioxide (UO_2), and uranium oxycarbide (UCO) with diameters ranging from 100 µm to 500 µm and an enrichment of 19.74% U-235. TRISO fuels are structurally more resistant to extreme temperatures, corrosion, oxidation, and neutron irradiation than standard fuels, which impact fuel performance most. Figure 5 shows the thermal conductivity of TRISO-coated layers.

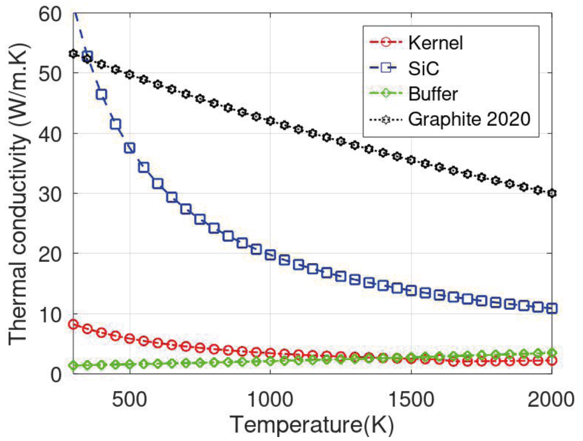

Figure 5. Thermal conductivity as a function of temperature for TRISO coatings

Over the years, the FliBe system has been the subject of experimental studies on the thermodynamics of the phase diagram, mixed enthalpies, heat capacities, and vapor pressure. The CALPHAD (CALculation of PHAse Diagrams) method became a tool for materials analysis for many years. Also, we can use FactSage, a thermochemical software, and databases containing nuclear libraries to produce phase diagrams. These databases contributed to developing phase diagrams with extrapolation methods for describing binary and ternary systems. Figure 6 displays the phase diagram of Pu_4/(LiF+PuF_4).

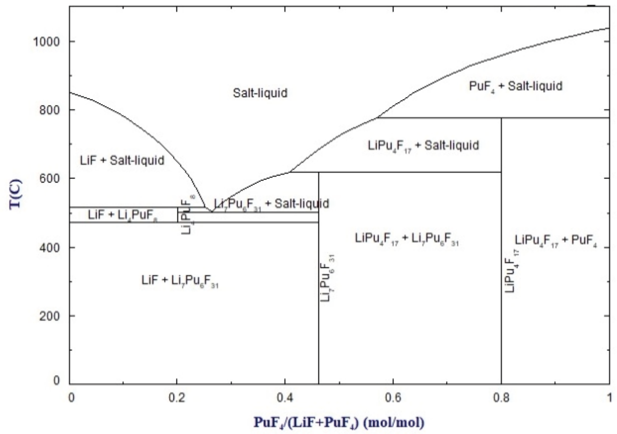

Figure 6. Phase diagram PuF4/(LiF-PuF_4), FactSage

Thermodynamic models for the molten salt system have shown a few methods permitting thermodynamic optimization and building phase diagrams. The molten salt composition can show variations from thermal reactors using (66LiF–34BeF2) mol% to fast reactors with (46.5LiF–11.5NaF–42KF) mol%. A solution comprises the Modified Quasi-chemical Model (MQM) in the Quadruplet Approximation (MQMQA). Thus, MQM has proven an effective model for representing fluoride salt systems. Research implemented the MQM in the open-source equilibrium thermodynamic library. Thus, anyone can execute sophisticated multi-physics simulations of the molten salt in a nuclear reactor system. Equation 6 describes the Gibbs energy equation for relevant compounds as a contribution of the enthalpy of formation and absolute entropy in reference.

$$G(T) = \Delta_f H(298.15K) - S(298.15)T + \int_{298.15}^{T} C_P(T)dT - T\int_{298.15}^{T}\left(\frac{C_P(T)}{T}\right)dT \quad (6)$$

Extensive experimental research on LiF-BeF2 focused on phase diagrams exists. The results from the experimental database developer for LiF–BeF_2–UF_4, LiF–ThF_4–UF_4, and BeF_2–ThF_4–UF_4 had created numeric functions for Gibbs energy. After investigating the thermodynamic tables of fluoride salts, we can find the polynomial correlations that can describe the Gibbs energy functions. The equation describes Gibbs's energy as a function of numeric parameters. Equation 7 defines the polynomial Gibbs energy mathematical model.

$$G(T) = a + bT + cTln(T) + \sum d_i T^i \quad (7)$$

As in ceramic fuels, the liquid molten salt fuel will also be subject to various types of radiation that can cause damage, such as α and β-decay, γ-radiation, and neutron and fission products.

DISCUSSION AND RESULTS

LIQUID FUEL

The fuel used for molten salt reactors works with fissile materials (U-235, U-233, or Pu-239), which are dissolved in the fluorinated molten salt. Then, the heat exchangers receive part of the thermal energy fusion generates. These heat exchangers operate with a secondary refrigerant, which is often molten salt. The primary circuit contains the coolant-fuel mixture in the reactor vessel at low pressure, operating with FLiBe as the primary coolant. Table 5 shows the inlet and outlet temperatures of the molten salt reactors.

The fuel working in a liquid state offers reactors several advantages, such as better safety margins. The secondary circuit forms an intermediate loop introduced for safety reasons: to avoid direct contact between steam and fuel. Inside, the core has an operating temperature of 525 °C to 725 °C. The lower limit determines the liquid phase using the melting temperature of the salt, while the higher limit limits the corrosion rate of the structural material. In its liquid form, the fuel also serves as a primary coolant, presenting low vapor pressure. Then, the total pressure of the primary circuit is shallow, around 0.1 MPa, compared with light water reactors above 15 MPa. A second aspect that improves the safety margins of the liquid fuel is that the reactor has a strong negative temperature coefficient. Hence, the chain reaction automatically slows down when the temperature increases. The thermal expansion of the primary coolant produced a volumetric expansion of the fuel, which decreased the fuel density. The ability to drain liquid fuel to emergency dump tanks in accident scenarios would be the third characteristic that increases the reactor's safety.

SOLID FUEL

A primary method of keeping nuclear reactors safe is using reliable methods to remove the decay heat. Thus, the FHR-safe system inherits three systems from other reactor designs, including decay heat removal. The first is the Direct Reactor Air Cooling System (DRACS) (Liu et al., 2018). The second approach uses the Reactor Vessel Air Cooling System (RVACS), which generates decay heat through natural air circulation through the reactor vessel (Liu et al., 2018). The third option is the silo cooling system (SCS) used for the decay heat removal system in a beyond-design accident.

Performance evaluation of decay heat removal systems for FHR resulted in three concepts: DRACS, RVACS, and SCS. These systems are all built on technology initially created for sodium-cooled reactors. However, sodium shows a lower heat capacity of 1.25 J/kgK at 673 K than FliBe, with 2369 J/kgK. Thus, the volumetric heat capacity of FliBe is significant and needs adaptations to change the size of the heat exchangers, internal piping, valves, and internal components.

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The DRACS has three coupled loops operating through natural circulation and convection. The DRACS Heat Exchanger (DHX) and the Natural Draft Heat Exchanger (NDHX) are the two heat exchangers that connect these loops. The fluidic diode reduces parasitic flow into the DRACS primary loop.

Under accident scenarios, the DRACS is prepared for activation. DRACS transfers heat to a thermosiphon-cooled or natural heat exchanger that rejects heat from ambient air, which serves as the ultimate decay heat sink. Figure 7 depicts the half-PB-FHR Mark-1 core design operating with a DRACS heat exchanger and fluid diode.

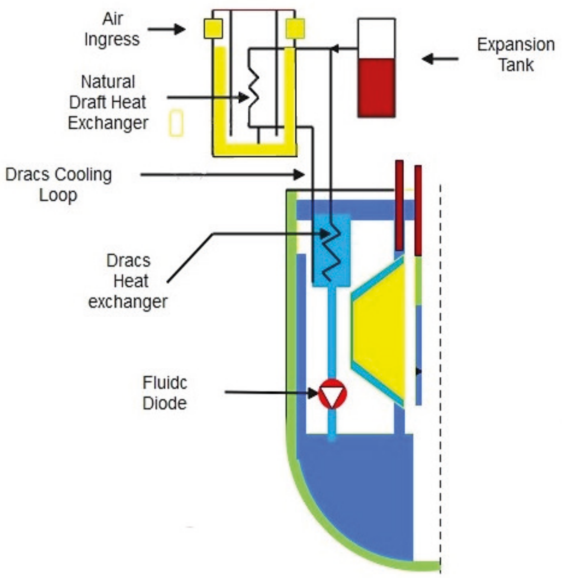

Figure 7. The Direct reactor auxiliary cooling system is a passive heat removal system in FHRs.

The thermal conductivity value of 1.0 W/mK used for FLiBe heat capacity is 2390 J/kgK. While for the FLINaK, it is 0.92 W/mK and 1940 J/kg·K. The packing factor adopted for FHR Mark-1 is 0.60, and the coolant velocity is around two m/s for both options, FLiBe and FLiNaK. In FHR, the core average Reynolds number is around 1250, and the core average Prandtl number is 18.42. The average effective thermal conductivity of a TRISO sphere is around 15 W/mK. They composed spheres of TRISO fuel with a spherical fuel kernel and four coating layers: a porous carbon buffer layer, an IPyC layer, a SiC layer, and an OPyC layer. Mark 1's core contains around 470,000 fueled pebbles and 218,000 unfueled pebbles. One pebble produces 500 watts. Each pebble contains 1.5 g of uranium enriched at 19.9%, showing an external area of 0.00282 m2 with an estimated core heat flux of 0.189 MWth/m^2. Figure 8 shows the convection heat transfer of PB-FHR using FliBe compared with FLINAK.

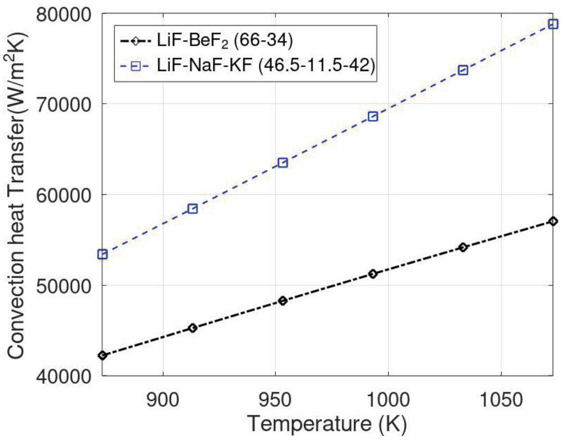

Figure 8. Convection heat transfer for FliBe and FLiNaK

The active volume of fuel is 10.4 m3, with a power density of 22.7 MWth/m^3, resulting in 236 MWth. The nominal coolant flow rate in the Mk1 design is 0.54 m3/s, which uses approximately 47 m3 (91,970 kg) of FliBe as its main salt. Many pebble bed codes are used to calculate the coolant temperature throughout the reactor and the fuel temperature, using the coolant temperature as a boundary condition.

CONCLUSION

The benefits of molten fluoride salt systems such as FLiBe and FLiNaK have several applications in MSR and MSFR and can achieve a high-power efficiency of about 45%, which is higher than the 35% limit of PWRs. Compared to FLiNaK, FLiBe has a better neutron response, but FLiNaK has a more alluring heat capacity. Using molten salts in high-temperature reactors operating near atmospheric pressures is advantageous over helium-gas-cooled reactors. The features of the molten salt FLiBe include its low melting point of 459 °C, high boiling point, and thermal conductivity of around 1.1 W/mK. In addition, it shows a low vapor pressure of 1.2 mmHg at 900 °C and a high specific heat capacity of 2390 J/kgK, supporting the use of this material.

FLiBe and FLiNaK are preferable alternatives to dissolved fertile and fissile materials, allowing for the use of thorium and plutonium, along with decreased cross-section and the ability to deliver clean fission products into the core reactor They were showing a high Prandtl number of 10 to 20. This way, options such as FLiBe FLiNaK and NaF-ZrF_4 meet better specifications. The high density of these salts in the liquid state and their high volumetric capacity are positive factors, enabling more compact nuclei. They use encapsulated, coated particles in TRISO fuel with a power of 500 W per sphere of 3 cm diameter. The FHR Mk1's operating parameters in the thermal spectrum are between 600 °C and 700 °C. The expertise accumulated over the years in Germany for helium-cooled reactors with a diameter of 6 cm and spherical fuels has undergone significant progress.

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RESPONSIBILITY NOTICE

The author is the only answerable for the printed information included in this study.

REFERENCES

Cadwallader, L. C., & Longhurst, G. R., 1999. *Flibe Use in Fusion Reactors-An Initial Safety Assessment*. Idaho National Lab. (INL), Idaho Falls, ID (United States).

Cantor, S., 1973, *Density and Viscosity of Several Molten Fluoride Mixtures*, ORNL-TM-4308.

Cantor, S., J. W. Cooke, A. S. Dworkin, G. D. Robbins, R. E. Thoma, and G. M. Watson, 1968., "Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts", ORNL-TM-2316.

Cervi, E., Lorenzi, S., Cammi, A., & Luzzi, L., 2019. "Development of a multiphysics model for the study of fuel compressibility effects in the Molten Salt Fast Reactor". Chemical Engineering Science, Vol. 193, pp. 379-393.

Cohen, S. I., Jones T. N., 1957., "Viscosity measurements on molten fluoride mixtures". No. ORNL-2278. Oak Ridge National Lab., Tennesse.

Davis, Cliff., 2005. Implementation of molten salt properties into RELAP5-3D/ATHENA. No. INEEL/EXT-05-02658. Idaho National Lab.(INL), Idaho Falls, ID (United States).

Dolan, T. J., 2017. "Molten salt reactors and thorium energy". Woodhead Publishing.

Douglas, T. B., Payne, W. H., 1969. "Measured enthalpy and derived thermodynamic properties of solid and liquid lithium tetrafluoroberyllate, Li2BeF4, from 273 to 900 K". *Journal of Research of the National Bureau of Standards*. Section A, Physics and Chemistry, Vol. 73(5),pp- 479.

Fütterer, M. A., Fu, L., Sink, C., de Groot, S., Pouchon, M., Kim, Y. W., Tachibana, Y., 2014. "Status of the very high temperature reactor system". Progress in Nuclear Energy, Vol. 77, pp. 266-281.

Gehin, J. C., & Powers, J. J., 2016. "Liquid fuel molten salt reactors for thorium utilization". *Nuclear Technology*, Vol.194(2), pp. 152-161.

Grape, S., Svärd, S. J., Hellesen, C., Jansson, P., & Lindell, M. Å., 2014. New perspectives on nuclear power—Generation IV nuclear energy systems to strengthen nuclear non-proliferation and support nuclear disarmament. "*Energy Policy*", Vol. *73*, pp- 815-819.

Gromov, B. F., Belomitcev, Y. S., Yefimov, E. I., Leonchuk, M. P., Martinov, P. N., Orlov, Y. I., Stepanov, V. S., 1997. "Use of lead-bismuth coolant in nuclear reactors and accelerator-driven systems". *Nuclear Engineering and Design*, Vol. 173(1-3), pp. 207-217.

Jerden, J., 2019. *Molten Salt Thermophysical Properties Database Development: 2019 Update*. Argonne National Lab.(ANL), Argonne, IL (United States).

Holcomb, D. E., Flanagan, G. F., Mays, G. T., Pointer, W. D., Robb, K. R., & Yoder Jr, G. L., 2013. *Fluoride salt-cooled high-temperature reactor technology development and demonstration roadmap*. ORNL/TM-2013/401, ORNL, Oak Ridge, Tenesse.

MacPherson, H. G., 1985. "The molten salt reactor adventure". *Nuclear Science and engineering*, Vol. 90(4), pp. 374-380.

Powers, W. D., S. I. Cohen, and N. D. Greene, 1963, "Physical Properties of Molten Reactor. Fuels and Coolants," *Nuclear Science and Engineering*, Vol. 17, pp. 200-211

Romatoski, R. R., & Hu, L. W., 2017. "Fluoride salt coolant properties for nuclear reactor applications: A review". *Annals of Nuclear Energy*, Vol. 109, pp. 635-647.

Schulenberg, T., Starflinger, J., Marsault, P., Bittermann, D., Maraczy, C., Laurien, E., Toivonen, A., 2011. "European supercritical water cooled reactor". *Nuclear Engineering and Design*, Vol. 241(9), pp. 3505-3513.

Schulz, T.L., 2006. "Westinghouse AP1000 advanced passive plant". *Nuclear engineering and design*, Vol. 236(14-16), pp.1547-1557.

Sohal, M. S., Ebner, M. A., Sabharwall, P., & Sharpe, P., 2010. *Engineering database of liquid salt thermophysical and thermochemical properties*. Idaho National Lab.(INL), Idaho Falls, ID (United States).

Subbotin, V. I., Arnol'Dov, M. N., Kozlov, F. A., & Shimkevich, A. L., 2002. "Liquid-metal coolants for nuclear power". *Atomic energy*, Vol. 92(1), pp. 29-40.

Tenchine, D., 2010. "Some thermal hydraulic challenges in sodium cooled fast reactors". *Nuclear Engineering and Design*, Vol. 240(5), pp. 1195-1217.

Torgerson, D.F., Shalaby, B.A. and Pang, S., 2006. "CANDU technology for Generation III+ and IV reactors". *Nuclear Engineering and Design*, Vol. 236(14-16), pp.1565-1572.

Weeks, J. R., 1971. *Lead Bismuth, Tin and Their Alloys as Nuclear Coolants*. Brookhaven National Lab., Upton, NY.