



Heat-Pipe Heat Exchangers for Salt-Cooled Fission and Fusion Reactors to Avoid Salt Freezing and Control Tritium: A Review

Bahman Zohuri, Stephen Lam & Charles Forsberg

To cite this article: Bahman Zohuri, Stephen Lam & Charles Forsberg (2019): Heat-Pipe Heat Exchangers for Salt-Cooled Fission and Fusion Reactors to Avoid Salt Freezing and Control Tritium: A Review, Nuclear Technology, DOI: [10.1080/00295450.2019.1681222](https://doi.org/10.1080/00295450.2019.1681222)

To link to this article: <https://doi.org/10.1080/00295450.2019.1681222>



© 2019 The Author(s). Published with license by Taylor & Francis Group, LLC.



Published online: 11 Dec 2019.



Submit your article to this journal [↗](#)



Article views: 816



View related articles [↗](#)



View Crossmark data [↗](#)



Heat-Pipe Heat Exchangers for Salt-Cooled Fission and Fusion Reactors to Avoid Salt Freezing and Control Tritium: A Review

Bahman Zohuri,^a Stephen Lam,^b and Charles Forsberg^{b*}

^aUniversity of New Mexico, Albuquerque, New Mexico

^bMassachusetts Institute of Technology, 77 Massachusetts Avenue, Cambridge, Massachusetts 02139

Received July 14, 2019

Accepted for Publication October 14, 2019

Abstract — *The fluoride-salt-cooled high-temperature reactor and some proposed fusion reactors use clean fluoride salts as reactor coolants that have melting points above 450°C and generate tritium. Tritium diffuses through most hot metals, thus methods to capture tritium and prevent its release to the environment are required. Molten salt reactors (MSRs) use fluoride or chloride salts with high melting points where the fuel is dissolved in the coolant. MSR systems produce volatile fission products (Xe, Kr, etc.) and some produce significant tritium. We examine the use of heat exchangers with multiple heat pipes for salt-cooled fission and fusion systems that serve four functions: (1) transfer heat from primary coolant to power cycle, secondary loop, or environment; (2) provide the safety function of a secondary loop by isolating the reactor salt coolant from the high-pressure power cycle; (3) stop heat transfer if the reactor coolant approaches its freezing point to prevent blockage of the primary loop; and (4) block tritium escape to the environment with recovery of the tritium. Each of these capabilities in some form has been demonstrated in a heat pipe system, but not all the functions have been demonstrated in a single system because there has been no need for all of these capabilities in a single system. We review the status of heat pipe technology and the limits of heat pipe technology as the starting points for decisions on the development of such heat pipe systems.*

Keywords — *Heat pipes, fluoride-salt-cooled high-temperature reactor, molten salt reactor, fusion, tritium.*

Note — *Some figures may be in color only in the electronic version.*

I. INTRODUCTION

Heat transfer from the primary salt coolant in a fluoride-salt-cooled high-temperature reactor (FHR), molten salt reactor (MSR), or salt-cooled fusion reactor presents two challenges not seen in water-, sodium-, and helium-cooled reactors: (1) liquid salts have melting

points above 400°C and must not freeze to avoid blocking of primary coolant flow and (2) the salt coolant generates tritium that must not be allowed to escape to the environment. Lithium fluoride is used in coolant salts to lower the melting point of the coolant salt; however, neutron irradiation of lithium generates tritium. Isotopically separated ⁷Li is used in salt fission systems to minimize neutron adsorption and tritium production. Isotopically separated ⁶Li is used in fusion systems to maximize tritium production because tritium is the fuel for a fusion reactor. Tritium production rates in salt-cooled fusion machines are about three orders of magnitude larger than in fission machines.

This paper examines the use of heat exchangers that incorporate multiple heat pipes designed to (1) transfer

*E-mail: cforsber@mit.edu

This is an Open Access article distributed under the terms of the Creative Commons Attribution-NonCommercial-NoDerivatives License (<http://creativecommons.org/licenses/by-nc-nd/4.0/>), which permits non-commercial re-use, distribution, and reproduction in any medium, provided the original work is properly cited, and is not altered, transformed, or built upon in any way.

heat from primary coolant to power cycle, secondary loop, or environment; (2) provide the safety function of a secondary loop by isolating the reactor salt coolant from the high-pressure power cycle; (3) stop heat transfer if the reactor coolant approaches its freezing point to prevent flow blockage; and (4) block tritium escape to the environment with recovery of the tritium. No such heat pipe with all of these capabilities exists but each of these four capabilities has been demonstrated in heat pipes. If heat pipes can be developed with these four capabilities, there is the potential to greatly simplify salt reactor design and reduce costs by replacing multiple independent systems with a single system that meets the four requirements. We define the requirements for such heat pipes, review the status of the technology to meet the four functional requirements, and define the path forward.

Figure 1 shows a schematic of a sodium heat pipe system with these capabilities. As discussed later, sodium is the preferred heat pipe fluid for salt systems based on multiple criteria.

I.A. Heat Transfer—The Traditional Application for Heat Pipes

The bottom half of the heat pipe is in the hot salt. The heat vaporizes the sodium that then flows to the colder condenser section of the heat pipe where it condenses and flows back to the hot salt zone. Heat pipes operate at near isothermal conditions where the rate of heat transfer is determined by the rate of sodium vapor flow from evaporator to condenser zone. As the temperature goes up, the density of sodium vapor increases and more heat is transferred

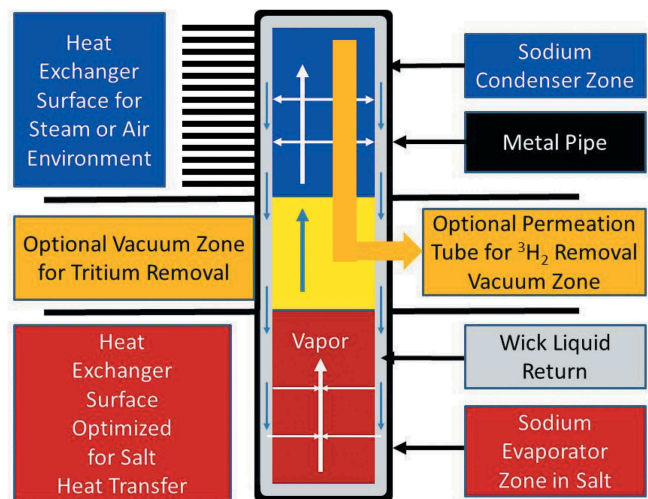


Fig. 1. Heat transfer regions within a heat pipe.

by the evaporation-condensation cycle. The heat pipe is lined with a wick where capillary forces wet the entire heat pipe surface and move liquid sodium from the cold condenser section back to the hot salt section where it is evaporated. The heat pipe operates independent of orientation of the heat pipe. If there were a pool of sodium, the boiling point of the sodium would be greater at the bottom than farther up the heated zone due to the hydrostatic pressure and would require higher temperature drops across the heat pipe for the same heat transfer rate. The use of wicks minimizes the heat pipe sodium inventory and thus the consequences of a failed heat pipe.

I.B. Prevention of Coolant Salt Freezing

Heat pipes can be designed to start up at a preset temperature. Sodium vapor pressure and thus sodium heat transfer rapidly increase with temperature. The sodium atmospheric boiling point is 883°C . In a sodium heat pipe the temperature needs to be above 500°C for significant heat transfer—above the freezing points of salt coolants. Second, one can put a small amount of inert gas into the heat pipe. At low sodium pressure, this gas blocks sodium vapor flow to the condenser section. As the temperature increases, the sodium pressure increases, the inert gas is compressed into smaller gas volume at the top of the heat pipe and most of the condenser section is available for sodium vapor condensation.

I.C. Tritium Capture

Heat pipes can be efficient methods for tritium capture. Most of the surface area in any nuclear system is associated with the heat exchangers. Unless one has methods for rapid removal of tritium ($^3\text{H}_2$) from the salt and diffusion barriers on the surface of the heat exchangers, the tritium will rapidly diffuse through the heat exchangers. With a heat pipe, the sodium condenser section on the inside is coated with tungsten, aluminum oxide, or other material to limit tritium escape via diffusion through the heat pipe. Liquid salts are fluxing agents that dissolve oxides and most high-performance tritium diffusion barriers. In contrast, sodium is chemically compatible with many high-performance tritium diffusion barriers including aluminum-oxide diffusion barriers. The different chemical environment enables a much wider choice of tritium diffusion barriers. The sodium vapor sweeps the tritium gas to the condenser section to be removed by a nickel permeation tube where the tritium diffuses from the sodium vapor to the nickel surface, diffuses through the nickel, and is removed from

a vacuum zone on the other side. This tritium removal system can be small relative to other tritium removal systems because of four system characteristics: (1) the large heat exchanger area enables tritium to diffuse through the salt to the heat pipe surface, (2) noncondensable tritium in the heat pipe is concentrated in the condenser section by sodium vapor movement, (3) tritium diffusion through sodium vapor and sodium liquid to the permeation membrane is rapid relative to the slow diffusion of tritium through liquid salts, and (4) tritium diffusion through the nickel is rapid. Mechanisms number (2) and (3) create the potential of a tritium removal system that is an order of magnitude smaller than alternative methods for tritium removal from salt systems.

Heat pipes are used today for a wide variety of applications from electronics to building heating to space nuclear reactors. Recent work with sodium heat pipes includes a 2018 ground demonstration of a nuclear space reactor with sodium heat pipes¹ to move heat from the reactor to the power cycle. Today sodium heat pipes are being considered for microreactors² with power ratings up to 15 MW(electric). The interest in microreactors implies development of heat exchangers at the multi-megawatt scale using sodium heat pipes—the same base technology that would be used for salt-cooled fission and fusion reactors with the added considerations of (1) heat transfer shutdown as liquid salts approach their melting points to avoid freezing salt and (2) control of tritium.

Section II describes alternative fission and fusion salt systems and the requirements for these heat exchangers. Section III provides a general description of heat pipe operation and how to use heat pipes to prevent salt freezing. Section IV discusses control of tritium and other chemical aspects of the design.

II. REACTOR SYSTEMS AND HEAT PIPE DESIGN REQUIREMENTS

II.A. Salt Fission and Fusion Reactor Systems

There are multiple salt fission and fusion systems that will impose somewhat different requirements on heat pipe systems.

II.A.1. Fluoride-Salt-Cooled High-Temperature Reactor

The FHR (Refs. 3, 4, and 5) uses graphite matrix coated-particle fuel developed for high-temperature gas-cooled reactors (HTGRs) and clean liquid fluoride salt coolants originally developed for MSR. The concept is less than 20 years old and no FHR has been built. Fluoride salt coolants are used because of their chemical compatibility with the graphite fuel and the low neutron adsorption cross section of fluorine. Relative to HTGRs, the use of a liquid salt coolant enables (1) increasing the power density by up to a factor of 10 because liquids are better coolants than gases, (2) near atmospheric pressure operation, and (3) a more robust safety case based on the fuel and a coolant that dissolves fission products that escape the fuel.

The pebble-bed FHR (Fig. 2b) is being developed by Kairos Power that uses flibe (${}^7\text{Li}_2\text{BeF}_4$) salt because of its excellent neutronic and thermal-hydraulic properties^{6–8} relative to other salts (Table I). Isotopically separated ${}^7\text{Li}$ is required because of the very high neutron adsorption cross section of ${}^6\text{Li}$. However, even with the use of isotopically separated ${}^7\text{Li}$, the tritium generation rate is about two orders of magnitude larger than in a pressurized water reactor (PWR). Methods to capture the tritium and prevent its release to the environment are required.

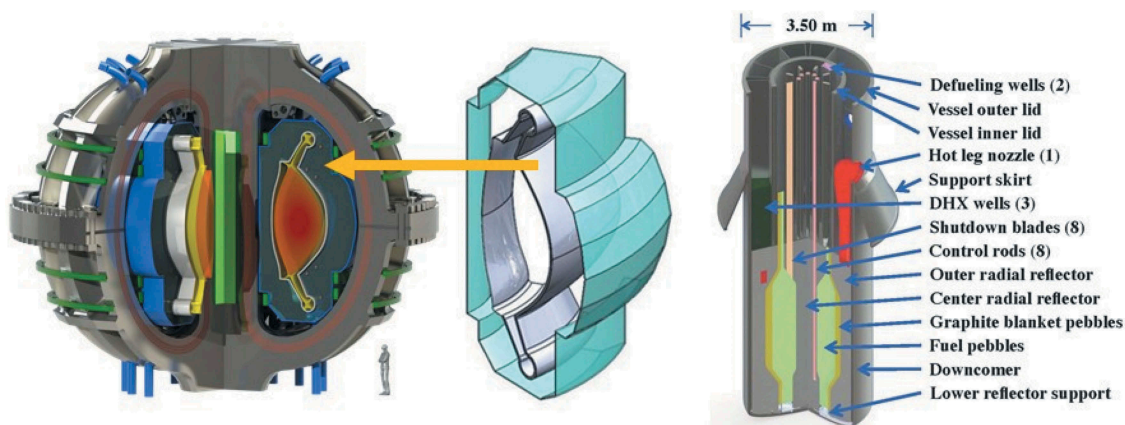


Fig. 2. (a) Affordable, robust, compact (ARC) fusion system and flibe fusion blanket for ARC and (b) FHR fission system.

TABLE I
Candidate Fluoride-Salt Reactor Coolants*

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m ³)	ρC_p (kJ/m ³ °C)
Li ₂ BeF ₄ (flibe)	459	1430	1940	4670
59.5 NaF-40.5 ZrF ₄	500	1290	3140	3670
26 ⁷ LiF-37 NaF-37 ZrF ₄	436	–	2790	3500
51 ⁷ LiF-49 ZrF ₄	509	–	3090	3750
Water (7.5 MPa)	0	290	732	4040
Water (15.5 MPa)	0	345	709	4049

*References 6, 7, and 8. Salt compositions in mole percent. Salt properties at 700°C and 1 atm. The boiling points of the zirconium fluoride salts are not well known. For comparison, water data are shown at 290°C (7.5 MPa, boiling point) and PWR conditions with water properties at 309°C.

These salts will typically deliver heat to the power cycle at between 600°C and 700°C. The minimum salt temperature is significantly above the melting point of flibe (459°C) for two reasons: (1) flibe near its melting point is viscous and (2) a reasonable temperature margin is required to avoid the risk of freezing salt in the heat exchangers. The maximum salt temperature is controlled by the availability of economic materials of construction. With existing materials that limit is near 700°C. For the FHR, the allowable fuel and coolant temperature limits could allow peak coolant temperatures to near 900°C assuming suitable materials of construction for the heat exchangers.

II.A.2. Salt-Cooled Fusion Systems

Clean (no-fuel) flibe salt blankets are being developed for high-magnetic-field fusion machines. In a fusion machine radioactive tritium (³H₂) combines with deuterium (²H₂) to generate helium and 14-MeV neutrons. Figure 2a shows the fusion system and flibe blanket. The flibe salt has four functions: (1) it slows down neutrons and converts that energy into heat, (2) the neutrons are used to breed tritium, (3) the salt acts as the primary radiation shielding to protect the magnets, and (4) flibe salt cools the first wall that separates the fusion plasma from the salt. Flibe salt is chosen to maximize tritium production. The beryllium in the salt acts as a neutron multiplier via a ($n, 2n$) reaction. Neutron adsorption by lithium generates tritium. Current designs propose 90% ⁶Li in the salt to maximize tritium production resulting in tritium production rates about three orders of magnitude greater than in an FHR per unit of thermal power.

While the concept of a flibe fusion blanket is old, recent engineering developments have created large incentives to develop flibe salt blankets relative to other fusion blanket

concepts. In the last 5 years, advances in superconductors have enabled doubling the magnetic field in fusion reactors.^{9,10} In fusion systems, the plasma power density increases as one over the fourth power of the magnetic field. Higher magnetic fields can reduce the machine volume by an order of magnitude for the same total power output, which dramatically improves the economics but creates major challenges. In a fusion blanket the 14-MeV neutrons slow down, delivering most of the heat from the fusion reactor to the blanket. With these higher power densities, it is difficult to cool traditional solid fusion blankets. As a consequence, flibe liquid-salt immersion blankets are proposed to adsorb the heat from the fast neutrons and breed tritium fuel. The first of these high-magnetic-field fusion systems is being developed by Commonwealth Fusion Systems in cooperation with the Massachusetts Institute of Technology. One consequence of these developments is that these advanced fission and fusion systems are tightly coupled¹¹ in terms of salt coolant technology, tritium control, and power cycle development.

II.A.3. Molten Salt Reactors

Molten salt reactors dissolve the fuel in the salt. Fluoride salts are used in thermal spectrum MSRs, whereas fluoride or chloride salts are used in fast spectrum MSRs. Fission products are generated in the liquid salt. Most of the fission products form fluoride or chloride salts. Gaseous volatile fission products (xenon, krypton, etc.) must be removed, and much of the tritium will be removed from the salt at the same time.

The 8-MW(thermal) Molten Salt Reactor Experiment at Oak Ridge National Laboratory demonstrated the concept in the late 1960s. The Shanghai Institute of Applied Physics (SINAP) of the Chinese Academy of Sciences is building a 2-MW(thermal) MSR. Both of these reactors

are thermal spectrum MSR designs with a graphite moderator that uses flibe salt with isotopically separated ^7Li . Some thermal spectrum MSR designs use other fluoride salts. Most but not all MSR designs propose that in an emergency the liquid fuel drain to a critically safe, passively cooled dump tank, typically using a freeze valve of frozen salt where if the reactor heats up, the salt melts, and the salt drains to the dump tanks.¹² In China,^{13–15} work is underway to develop sodium-cooled heat pipes to transfer decay heat from this dump tank to the air. This includes ongoing laboratory experiments. For this application, the heat pipe requirements are relaxed relative to the FHR and fusion applications—tritium generation stops once the reactor becomes noncritical and the decay heat is decreasing with time. There is also ongoing work to develop potassium-cooled heat pipes for decay heat removal in FHRs (Ref. 16). Neither of these systems are designed for tritium recovery or temperature control.

Europeans^{17–19} are developing a fast spectrum MSR using fluoride salts with a composition of 77.5% LiF and 22.5% heavy metals that includes thorium and the fissile fluorides. The salt mixture requires lithium fluoride to reduce the melting point of the liquid salt. Isotopically separated ^7Li is used to minimize tritium production, but as with the FHR there will be significant tritium production.

Last, there is the recent development of molten chloride fast reactors. Only fission product tritium is generated in these systems, which do not include lithium in the chloride salts. The chloride salts are typically mixtures of sodium, potassium, and sometimes magnesium salts. The optimum salt compositions are not fully defined for these systems. The molten chloride fast reactor²⁰ is being developed by TerraPower as a breed-and-burn once-through MSR. Because there is no lithium or beryllium, tritium production will be less than in most other MSR systems, with much of the tritium stripped from the salt with the xenon and krypton gases. In this specific design, the liquid salt remains in the main system when shut-down; dump tanks are not used.

The other molten chloride fast-reactor concept, being developed by Moltex,^{21,22} has the traditional layout of a solid-fuel fast reactor except the fuel pins contain liquid chloride salts. A secondary clean salt transfers heat from the fuel assemblies to the heat exchanger. In this context, it resembles the FHR with fixed fuel, a clean liquid salt coolant, and a requirement to avoid salt freezing.

II.B. Salt Reactor Heat Exchanger Requirements

Heat exchangers will be used to transfer heat (1) from the primary coolant to the power cycle and (2)

remove decay heat after shutdown for the FHR and MSR systems. A simplified schematic of an FHR with these two types of heat exchangers is shown in Fig. 3 using heat pipe heat exchangers. The primary difference between salt reactors and other reactors (water, helium, or sodium cooling) in the context of heat exchangers is that the salt melting point is above 400°C ; thus, a system is required to prevent freezing of the salt within the heat exchanger. The choices are (1) a power cycle where the cold incoming fluid is above the salt melting point, (2) a control system that isolates the heat exchanger or dumps the salt if salt temperatures approach the freezing point of the salt, or (3) a heat exchanger that shuts down before the salt can freeze—a heat pipe heat exchanger.

The primary coolant heat exchangers will send heat to a Rankine (steam) or nuclear air-Brayton power cycle. Within that heat exchanger multiple heat pipes (green in Fig. 3) move heat from the salt to the steam cycle through a tube sheet. There is the option to include a tritium removal system as a component of the heat pipe system. Heat pipes are nearly isothermal heat transfer devices. For transferring heat from the salt to the power cycle, countercurrent heat exchanges are preferred to maximize the temperature of the steam or compressed air to the power cycle. Figure 4 shows schematically how one may use multiple heat pipes to create a countercurrent heat exchanger. Countercurrent heat pipe heat exchangers are used in heat recovery systems.^{23–28}

A more advanced power cycle option is the nuclear air-Brayton combined cycle that has peaking capability.^{3–5,29–32} MSRs were originally developed as part of the U.S. Aircraft Nuclear Propulsion Program to power a jet bomber of unlimited range; that is, salt reactors were originally

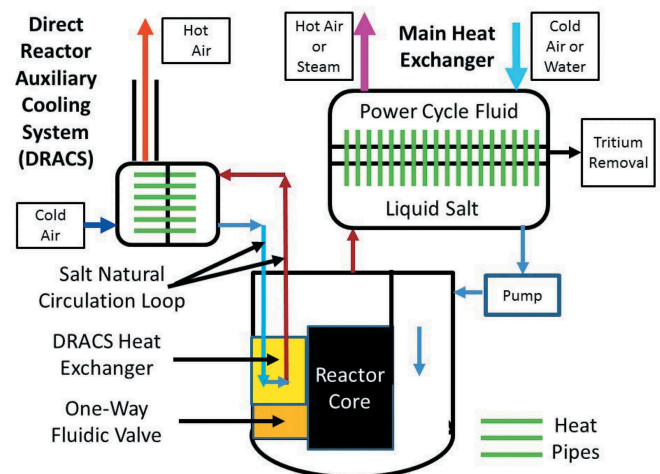


Fig. 3. FHR with primary and decay heat removal heat exchangers incorporating heat pipes.

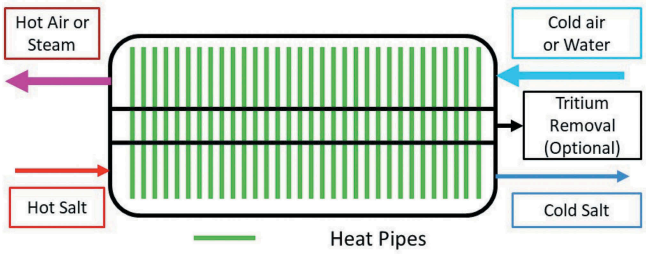


Fig. 4. Countercurrent heat exchanger with heat pipes.

developed to efficiently couple to a Brayton power cycle with a sodium intermediate loop between the reactor and jet engine. The Brayton power cycle requirements drove the reactor design. Because of changes in the electricity market with the addition of wind and solar and advances in gas turbine technology, there are now large incentives to couple salt reactors to gas turbine cycles because of their capabilities to produce peak power using thermodynamic topping cycles as a replacement for the traditional gas turbine in a low-carbon electricity grid. One candidate gas turbine cycle is shown in Fig. 5. Air is compressed, heated using nuclear heat, goes through a turbine, is reheated using nuclear heat, goes through a second turbine, and then goes to a heat recovery steam generator. It is a nuclear variant of a natural gas combined cycle.

Added peak power can be produced by adding high-temperature heat or a combustible fuel after the nuclear reheat step to increase the gas temperature going into the second turbine up to temperatures as high as 1500°C. The combustible fuel could be natural gas, biofuels, or ultimately, hydrogen. If there is low-price electricity, the electricity can be used to heat firebrick (firebrick

resistance-heated energy storage) that then can provide very high-temperature heat for the peaking cycle. The peak power capability is a thermodynamic topping cycle with incremental heat-to-electricity efficiency that can approach 75%, and the cycles can be designed for different base load-to-peak power ratios to match market requirements, including large peak power capabilities to provide large assured generating capacities.

The temperature exiting the front-end air compressor of a modern gas turbine is near 400°C. That requires the reactor to provide heat to the power cycle between 500°C and 700°C and matches salt reactors. The high compressed air temperatures reduce the risk of freezing the salt. However, because it is an open-air gas turbine, it implies that any tritium that goes through the heat exchanger will be released to the environment in seconds. The use of heat pipes avoids concerns about freezing of the salt when the reactor shuts down and can potentially block tritium releases to the environment. Because of the air pressure in the gas turbine, any heat pipe failure will have the air entering the heat pipe, the oxygen reacting with sodium to form sodium oxide, and a resulting shut-down heat pipe filled primarily with nitrogen. There are other implications. The air-side heat transfer coefficient is less than in steam systems; thus, these heat exchangers are considerably larger than salt-to-steam-system heat exchangers.

The FHR has a passive decay heat cooling system where the goals are to minimize heat losses during normal operations, remove heat if the salt coolant temperature increases such as after reactor shutdown, and avoid freezing the salt. Several alternative systems^{33,34} have been proposed to meet these goals. A direct reactor auxiliary cooling system (DRACS) with heat pipes can be used as shown in Fig. 3. DRACS is a natural circulation heat transfer system using flibe or another salt that transfers heat from the primary system. In this specific case, a sodium heat pipe would include an inert gas that would minimize heat to the environment when the salt temperature was below its nominal peak operating temperature.

The DRACS heat exchanger (DHX) in the primary system is located between the lower higher-pressure cold-salt plenum below the reactor core and upper hot-salt plenum above the reactor core. When the pumps are running, the flow through the DHX is restricted by a one-way fluidic valve that allows some bypass flow through the DHX from the cold-salt zone at the bottom of the reactor to the hot zone above the reactor core. If the pump stops, hot salt exiting the reactor flows through the DHX to the bottom of the reactor core by natural circulation. This heats the salt in the separate DRACS loop that flows by natural circulation to the salt-air

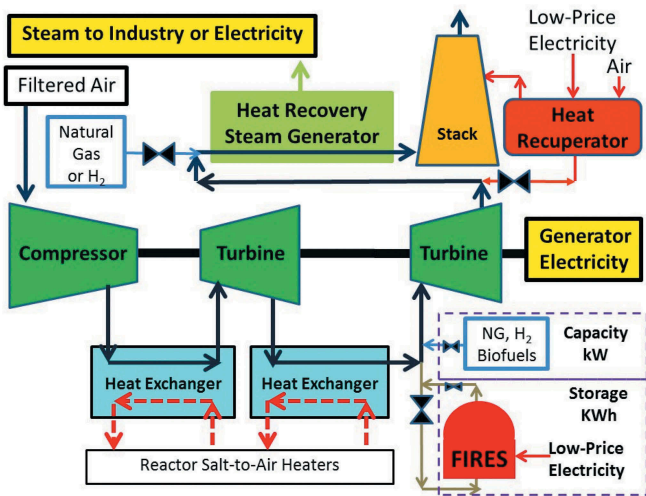


Fig. 5. Nuclear air-Brayton combined cycle with peaking capability.

heat exchanger where the heat is dumped to air. The air flows by this heat exchanger by natural circulation.

In the salt-to-air heat exchanger, heat pipes move heat from the natural circulation salt to the air stream. The heat pipes turn on and off when salt temperatures are above or below 700°C, respectively. A recent assessment³⁵ concluded that in an accident the peak temperature must not exceed 970°C to ensure against structural failure of the 316 stainless steel. This defines the performance requirements for DRACS. The high-temperature fuel and coolant capabilities imply that temperature limits are defined by components outside the reactor core, such as heat exchangers, piping, and the reactor vessel.

Last, heat pipes can be used to cool hot-box systems. Rather than insulate individual pipes, valves, pump heads, sample lines, salt inventory tanks, primary reactor vessel, and other components to prevent salt freezing, there is the option to insulate a box that contains multiple such components. There is less risk of small tubes freezing, inspection is simplified, and total insulation costs may be reduced. For such hot boxes, heat pipes can be used to dump heat from the hot box to the atmosphere if the salt temperatures and thus hot-box room temperatures exceed some temperature limit. Unlike the above cases, in this case the heat pipe evaporator section would be in an inert gas environment rather than in a liquid salt environment. Heat transfer to the heat pipe would be some combination of convective gas flow and radiative heat transfer. The heat pipe could be part of the wall structure.

III. HEAT PIPE DESIGN AND STARTUP TEMPERATURES

III.A. Choice of Fluid

Different salt reactors have somewhat different salt inlet and outlet temperatures, but within the temperature range of 500°C to 750°C. Advanced machines may operate at higher temperatures. Decay heat cooling systems may also operate at higher temperatures. The candidate heat pipe fluids are potassium, sodium, and lithium as shown in Table II. There has been large-scale development of sodium, potassium, and lithium heat pipes,³⁶⁻⁴⁰ primarily for space nuclear reactors. Many of these studies and experiments have included start up and shut down under hot and cold temperatures. Sodium, as discussed below, is the preferred heat transfer fluid for most salt systems.

The operating range for each heat pipe fluid is determined by the liquid metal vapor pressure versus pressure. At very low pressures, there is insufficient flow of vapor from the evaporator to condenser section to move

TABLE II
Heat Pipe Coolant Options

Fluid	Boiling Point (°C) at 1 atm	Typical Operating Range (°C)
Lithium (Li)	1330	825 to 1125
Sodium (Na)	883	600 to 825
Potassium (K)	759	325 to 525

significant heat. As discussed below, in most heat pipe systems the maximum heat transfer is a function of the density of the vapor and the sonic velocity of the vapor. Figure 6 shows the vapor pressure of sodium⁴¹ and potassium as a function of temperature. A potassium heat pipe becomes fully operational substantially below the freezing point of liquid salts⁴⁰ with significant potassium pressures at higher temperatures. Sodium heat pipes begin to transfer heat at higher temperatures¹ as the sodium vapor pressure increases. There is substantial experimental data on heat pipe performance with temperature, primarily with space reactors that go through a startup transient from frozen liquid metal to full operating temperatures. The minimum heat transfer in a heat pipe as the temperature goes down is determined by heat conduction through the heat pipe walls and wick structure between the evaporator zone and condenser zone that contains liquid sodium.

The startup temperature of the heat pipe can be more precisely controlled by the addition of an inert gas to the heat pipe (Fig. 7). For sodium heat pipes the inert gas is neon, which has about the same molecular weight as sodium. The

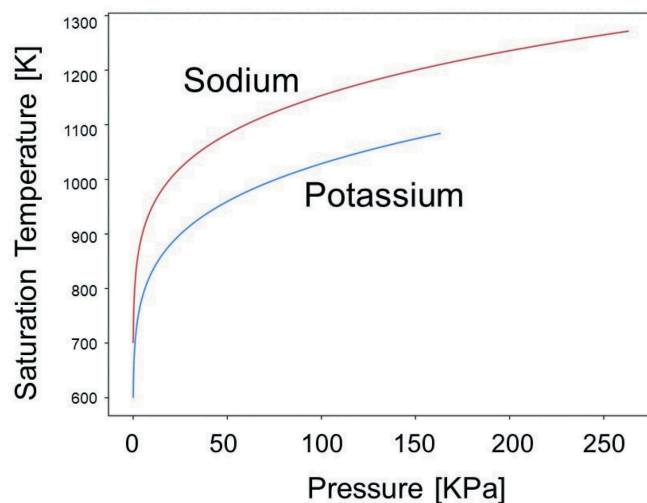


Fig. 6. Vapor pressure (in kilopascals) of sodium (top) and potassium (bottom) versus temperature (in kelvins).

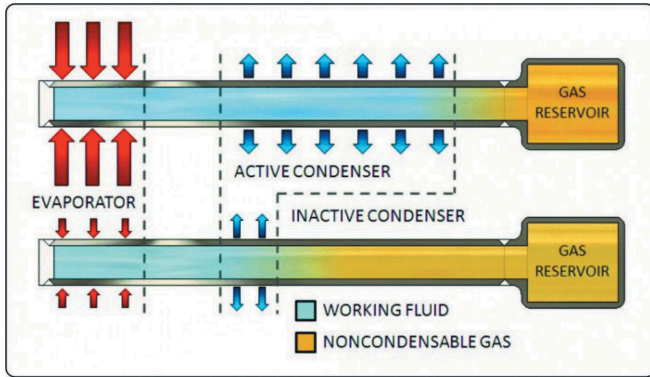


Fig. 7. Schematic of variable conductance heat pipe.

flow of sodium vapor from hot to cold pushes the inert gas to the top of the condenser section. At lower temperatures and thus lower sodium pressures, the inert gas fills the condenser section with very little heat flow. As the temperature increases, the sodium vapor pressure increases, and the flow of sodium vapor pushes the inert gas to the top of the heat pipe. When this happens more of the condenser section is exposed to sodium vapor and is condensed. The amount of inert gas for any given geometry determines at what temperature the boundary between the inert gas and the sodium vapor is pushed into the condenser section. This creates the option of a decay heat removal system with low heat losses during normal reactor operations, high heat removal if the temperature goes up, and shut down as temperatures go down. Experiments⁴² have been done with sodium heat pipes with neon gas for temperature control.

This heat pipe startup and shutdown capability is being considered for concentrated solar power (CSP) towers. Concentrated light from the sun provides the heat to the evaporator section of the heat pipe. The heat is transferred to flowing liquid salt. Liquid salts are used for heat storage. Sodium heat pipes have two potential advantages. First, sodium is very good at transferring heat with very high heat fluxes to the salt. The peak temperature of currently used nitrate salts is limited by the potential for hot spots in the solar receiver. The heat pipe eliminates the potential for excess nitrate temperatures and allows somewhat higher average peak temperatures. Second, the addition of an inert gas can be used to shut down the heat pipe to avoid salt freezing. When a cloud comes over the CSP system, the heat pipe will operate in reverse to cool down and then freeze the salt. The current generation of CSP systems uses nitrate salt with a relatively low melting point, but the next generation of CSP systems is expected to use a sodium-potassium-magnesium-chloride salt that can operate at very high

temperatures and is very cheap. However, that salt has a melting point of approximately 420°C. A heat pipe with the appropriate design can become a one-way heat transfer system to prevent salt freezing—the same challenge as with salt-cooled reactors.

The operating temperature of a sodium heat pipe can be raised—adding lithium to the sodium heat pipes will lower the vapor pressure of the sodium at any given temperature. We are not aware of any experimental work that has investigated this option for sodium heat pipes. Dual-component heat pipes⁴³ have been investigated with other fluids, but have more complicated behavior. In some ways their operation is similar to a distillation column on full reflux. The lower boiling point component preferentially vaporizes resulting in the gas-phase composition changing with temperatures. In addition, molten lithium complexes with tritium—a way to capture tritium. However, in such a heat pipe there may be no simple way to recover the tritium. This is discussed in Sec. V.

The performance of heat pipes is measured by the Merit number as defined in Eq. (1) and plotted in Fig. 8 as a function of temperature for different fluids:

$$N_l = \frac{\rho_l \sigma \lambda}{\mu_l}, \quad (1)$$

where

ρ_l = liquid density

σ = surface tension

λ = latent heat

μ_l = liquid viscosity.

High liquid density and high latent heat reduce the fluid flow required to transport a given power, while high surface tension increases the pumping capability. A low liquid viscosity reduces the liquid pressure drop for a given power. Sodium has the highest Merit number for any heat pipe system; that is, the best performance. The combination of operating temperature range and performance indicates that a sodium-based heat pipe system is the preferred option for salt systems.

IV. HEAT TRANSFER ANALYSIS

We examine herein the limits of heat pipes and then the implications for sodium heat pipes.

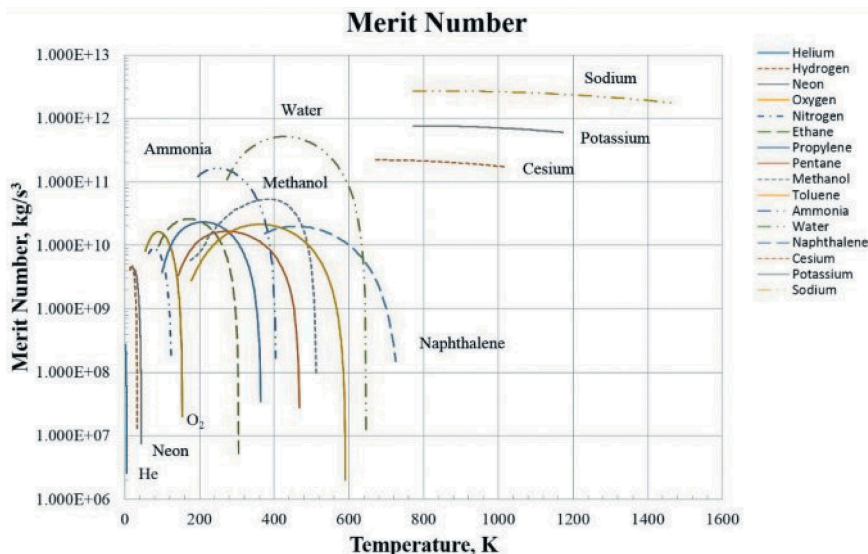


Fig. 8. Merit number for commonly used heat pipe working fluids.

IV.A. Heat Pipe Operational Limits

Heat pipes have multiple operational limits but only a few of these limits apply to a sodium heat pipe in a liquid salt system. Figure 9 shows the operational limits for a specific sodium heat pipe versus temperature and the constraints that are likely to be relevant for fission or fusion salt system—sonic (gas phase) and capillary (liquid phase) limits. We discuss all the limits and why specific limits are important or unimportant for sodium heat pipes in salt systems. This specific heat pipe has an inside diameter of 1.5 cm and a length of 4 m. Under normal conditions (salt reactor

conditions), such a heat pipe would transfer kilowatts to tens of kilowatts of heat. For near-term salt reactor systems, normal peak operating temperatures would be near 700°C.

IV.A.1. Viscous Limit

The viscous limit dominates at low temperature, near the melting point of the working fluid. The high liquid pressure losses in the wick, due to high viscosity and low permeability, limit liquid flow from the condenser to the evaporator section. Salt melting points in salt reactors are far above the melting point of sodium so this is not a constraint for sodium heat pipes in salt systems.

IV.A.2. Entrainment Limit

The entrainment limit is a result of the liquid and vapor moving in opposite directions. It influences wick design. The vapor exerts a shearing force on the liquid at the liquid-vapor interface. If this shear force exceeds the surface tension of the liquid, liquid droplets are entrained into the vapor flow and are carried toward the condenser section. The entrainment limit is typically encountered during a heat pipe startup when the vapor flow at the evaporator section exit is choked (velocity is near sonic). The entrainment limit can be raised by employing a small pore size wick and/or increasing the cross-sectional flow area for the vapor in the heat pipe to lower its velocity at the exit of the evaporator section. This would not be expected in a properly designed sodium heat pipe.

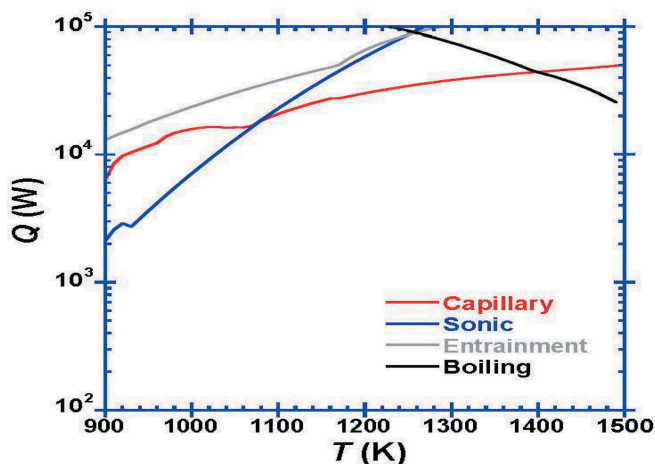


Fig. 9. Typical pipe axial performance limits versus evaporator exit temperature for 1.5 cm inside diameter by 4-m-long sodium heat pipe.⁴⁴

IV.A.3. Boiling Limit

Boiling at the inside surface of the heat pipe wall in the evaporator section is likely when the local liquid superheat exceeds that for incipient nucleate boiling. The ensuing nucleation and growth of vapor bubbles blocks the flow of returning liquid to the evaporator section. In alkali-metal heat pipes, the boiling limit is typically encountered at high wall temperatures, beyond those selected for nominal operation. Boiling limits can occur in heat pipe reactors where the evaporator section of the heat pipe is in the reactor core. For sodium heat pipes in salt systems where liquid salt brings the heat to the heat pipe, this is not a design constraint. Liquid salts have a low Prandtl number; that is, low thermal conductivity that limits heat transfer rates. The heat transfer limits on the salt side of the evaporator are far more limiting than the limits inside the heat pipe.

IV.A.4. Sonic Limit

The sonic limit is dominant at lower temperatures for many designs of sodium heat pipes (Fig. 9). The vapor pressure of the working fluid is a good indicator of reaching this limit.⁴⁵ The vapor pressure and physical state of the heat pipe liquid at ambient temperature, as well as the thermal resistance between the condenser and the adjacent heat sink, have significant influence of the startup behavior of a heat pipe. The sonic limitation is defined in Eq. (2):

$$Q_{s,\max} = A_v \rho_v h_{lv} \left[\frac{\gamma_v T_v r_v}{2(\gamma_v + 1)} \right]^{1/2}, \quad (2)$$

where

- A_v = cross-sectional area of vapor space
- ρ_v = vapor density phase (kg/m³)
- h_{lv} = heat transfer coefficient liquid, vapor (latent heat of vaporization)
- γ_v = specific heat ratio for vapor side
- r_v = radius of vapor space section
- h_{lv} = heat transfer coefficient liquid, vapor.

IV.A.5. Wicking or Capillary Limit

The wicking limit or capillary limit can be a constraint at higher temperatures for sodium heat pipes. This condition occurs when an applied heat flux causes the liquid in the wick structure to evaporate faster

than it can be supplied by the capillary pumping power of the wick. Once this event takes place the meniscus at the liquid-vapor interface continues to withdraw and move back into the wick until all of the liquid has been depleted. This action will cause the wick to become dry and the heat pipe container temperature may continue to rise at the evaporator until a “burnout” condition is reached.⁴⁵ The difference in the capillary pressure across the liquid-vapor interfaces governs the operation of the heat pipes. This is one of the most important parameters that affect the performance and operation of a heat pipe.

The capillary limit is encountered when the capillary pressure is not sufficient to pump the liquid back to the evaporator and thus causes the dry out of the wick of the evaporator end. The physical structure of the wick is one of the most important reasons for this limit, and the type of working fluid affects it. It is less of a concern for vertical heat pipes where gravity supports the return of the liquid to the evaporator section.

Thus, the capillary (or wicking) limit is encountered when the net capillary pressure head is less than the combined pressure losses of the liquid flow in the wick and of the countercurrent vapor flow in the heat pipe. The capillary pressure head for circulating the heat pipe’s working fluid increases with increasing the liquid surface tension and decreasing the radius of curvature of the liquid-vapor meniscus in the surface pores in the wick, as illustrated in terms of R_c in Fig. 10 and Eq. (3):

Capillary Limitation:

$$Q_{c,\max} = \frac{2 \left(\frac{\sigma_p h_{lv}}{\mu_l} \right) \left(\frac{K}{r_c} \right) (2\pi r_v T_w)}{0.5L_e + L_a + 0.5L_c}, \quad (3)$$

where

- σ = surface tension, condensing coefficient (N/m)
- ρ_l = liquid density phase (kg/m³)
- h_{lv} = heat transfer coefficient liquid, vapor (latent heat of vaporization)
- K = permeability (m²)
- r_v = radius of vapor space section
- t_w = temperate of heat pipe wall
- μ_l = liquid viscosity (N·s/m²)
- r_c = effective capillary radius (m)
- L_e = length of evaporator section (m)
- L_a = length of adiabatic section (m)
- L_c = length of condenser section (m)

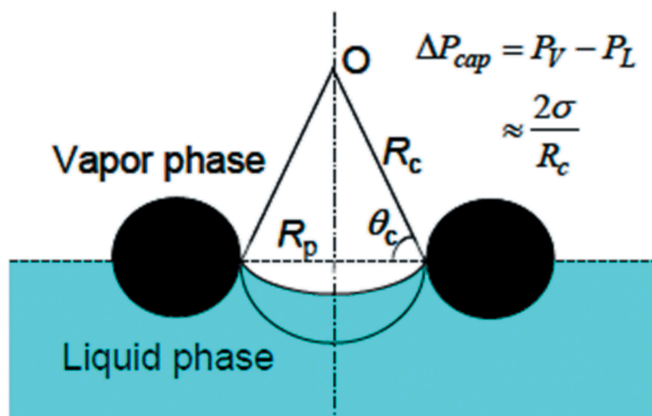


Fig. 10. Capillary pumping of the working fluid in heat pipe.

IV.B. Sodium Heat Pipe Experience

There is significant experience with sodium heat pipes in the temperature range applicable to salt reactor systems. Thermacore, Inc., has carried out several sodium heat pipe life tests to establish long-term operating reliability. A 316L stainless steel heat pipe with a sintered porous nickel wick structure and an integral brazed cartridge heater has successfully operated at 650°C to 700°C for over 115 000 h without signs of failure. A second 316 L stainless steel heat pipe with a specially designed Inconel 601 rupture disk and a sintered nickel powder wick has demonstrated over 83 000 h at 600°C to 650°C with similar success. A representative one-tenth-segment Stirling Space Power Converter heat pipe with an Inconel 718 envelope and a stainless steel screen wick has operated for over 41 000 h at nearly 700°C. A hybrid (ie., gas-fired and solar) heat pipe with a Haynes 230 envelope and a sintered porous nickel wick structure was operated for about 20 000 h at nearly 700°C without signs of degradation. These life test results collectively have demonstrated the potential for high-temperature sodium heat pipes at the temperatures expected in salt fission and fusion systems to serve as reliable energy conversion system components for power applications that require a long operating lifetime with high reliability.³⁹

There have been many sodium heat pipes built and tested for space applications including long-duration runs (>100 000 h) and different operating temperatures. In a space reactor, heat pipes move heat from the reactor around the radiation shield to the power conversion system. Multiple heat pipes are used rather than a typical sodium heat transfer loop so any single failure (tube leak) does not cause mission failure. This experience includes testing heat pipes in the Experimental Breeder Reactor II

for space applications^{38,45} where heat would be transferred at temperatures ranging from 900°C to 1100°C.

Heat pipes can be designed to be large or small. The maximum heat pipe size will likely be determined by the allowable leakage of sodium into the primary system if there is a leak in the heat pipe. That, in turn, determines the total number of heat pipes required for each application. Sodium is a strong chemical reducing agent that will reduce beryllium fluoride and many other fluorides that may be salt components⁴⁶ of the coolant salt to metals. Most of these metals have some solubility in the coolant salt. Small additions of sodium to the primary coolant will not have major impacts; but large additions would cause major changes in salt chemistry with different impacts depending upon the type of salt-cooled system. These parameters will place upper limits on individual heat pipe sodium inventories to limit the consequences of leaks. In this context the use of a wick is important because it minimizes the sodium inventory in any individual heat pipe.

V. TRITIUM CONTROL

For many fission salt-cooled reactors and all salt-cooled fusion reactors, neutron interaction with the coolant generates tritium. Fission reactors use isotopically separated ⁷Li to minimize tritium production whereas fusion machines use isotopically separated ⁶Li to maximize tritium production. Tritium is the fuel for fusion reactors. The tritium production rates in FHRs per megawatt hour are about hundred times those of a PWR. Tritium production rates in a fusion machine will be a thousand times larger per unit of power output than an FHR. Further, fusion machines will have the added requirement for efficient recovery of tritium to recycle as fuel while having the same allowable release limits to the environment.

Most of the tritium is generated by neutron adsorption by lithium in lithium fluoride; thus, the chemical form of the tritium is ³HF. All proposed salt reactors have chemical redox control systems⁴⁶ to convert corrosive ³HF into ³H that then becomes ³H₂ or ³HH if hydrogen gas sparging is used for tritium removal. Tritium becomes a dissolved gas (³H₂) in the liquid salt. To capture tritium and prevent its release to the environment, the fission or fusion reactor will contain barriers to tritium release and methods to capture the tritium. Recent reviews⁴⁷ have examined methods to capture tritium and limit its release from salt-cooled fission and fusion reactors. Three strategies are available. Real systems are

likely to use more than one system, particularly fusion reactors with their very high tritium production rates:

1. *Tritium removal from salt:* Tritium can be removed from liquid salts by beds of carbon and other materials, permeation filters where the tritium diffuses through nickel or other tubing to a tritium removal system, and gas sparging using inert gases or gases that contain some hydrogen for redox control.

2. *Tritium removal within heat exchangers:* The options herein are heat pipes and double-wall heat exchangers. Double-wall heat exchangers require a vacuum or flowing fluid between the double tubes for tritium removal with some loss in heat exchanger efficiency.

3. *Tritium removal in a secondary loop:* If a nitrate salt intermediate heat transfer loop is used, tritium that diffuses through the heat exchangers from the hot salt will be converted to steam and can be removed from the off-gas of the nitrate salt system.

Heat pipes can incorporate (1) barriers to prevent the release of tritium from the primary system and/or (2) a tritium capture system. Materials with low permeability to tritium, such as tungsten, can be used to coat the salt side (evaporator section) of the heat exchanger to minimize tritium diffusion into the heat pipe. Alternatively, tungsten, aluminum oxide, or other coatings can be used to coat the inside sodium condenser side of the heat pipe to not allow tritium to escape from the heat pipe to the environment. Tritium diffusion barriers on the inside of the sodium heat pipe have the potential to be orders of magnitude better than tritium diffusion barriers in salt systems. Liquid salts act as fluxing agents that dissolve almost all proposed tritium diffusion barriers except options such as tungsten. However, sodium is chemically compatible with high-performance tritium diffusion barriers, such as aluminum oxide, that can be applied to metal heat pipes.

Tritium can be removed from the heat pipe by a permeation filter where tritium diffuses into a vacuum or inert sweep gas, as shown in Fig. 1. This can be part of the wall of the heat pipe or a separate permeation filter can be installed in the heat pipe. In each case there is a vacuum to recover the tritium. The sodium vapor will sweep the tritium that diffuses through the evaporator section toward the condenser section of the heat pipe and thus most of the surface area of the permeation filter should be in the condenser section. Tritium will diffuse through the metallic permeation filter into the vacuum zone. Nickel is the likely material of construction for the

permeation tube due to its compatibility with sodium and high permeability to hydrogen.

There have been many studies on removing tritium from liquid salts with permeation membranes.⁴⁸ Permeation rates are a function of the hydrogen diffusivity and hydrogen solubility in the material. Figure 11 shows hydrogen diffusivity for different materials that may be found in an FHR or fusion reactor including salts, metals of construction, and liquid sodium. Figure 12 shows the corresponding solubility coefficients for different materials that could be found in an FHR or fusion machine with heat pipes. There is a caveat on hydrogen solubility. For diatomic gas molecules, the solubility in metals, including liquid sodium, follows Sievert's law where the solubility is proportional to the square root of pressure. For liquid salts, the solubility of gas follows Henry's law where the solubility is proportional to the pressure.

In terms of permeation rates, diffusivity can be considered a measure of the difficulty of a molecule moving through a material whereas the solubility is a measure of the number of pathways for a molecule to move through a material. Liquid salts have low diffusivity relative to most materials and low hydrogen solubility. This implies that any system to remove tritium from the salt must have a high surface area to reduce the diffusion time from the liquid to the removal system, be it a gas sparging system, an adsorber bed, or a permeation filter. If a permeation filter is used for removal of dissolved tritium in the salt, it will most likely be like a cross-flow heat exchanger to create turbulent flow of salt past the tubes to increase the mass transfer of tritium from salt to the permeation filter surface. Once the tritium reaches the surface, it can diffuse through the metal for collection of the tritium. This mass transfer limitation implies large surface areas

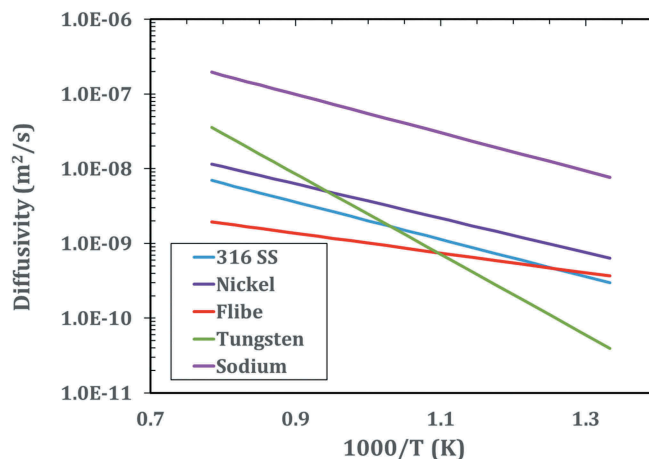


Fig. 11. Diffusivity of flibe compared to different metals⁴⁹⁻⁵¹ and liquid sodium.⁵²

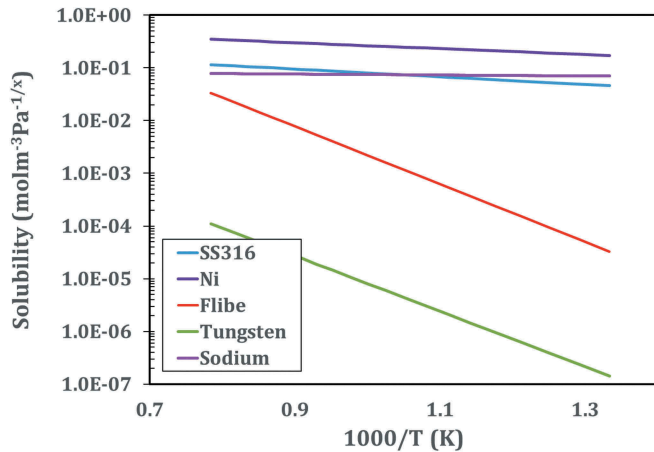


Fig. 12. Hydrogen solubility coefficients versus temperature for different materials including liquid sodium.^{49–52}

for the permeation filter. There is, however, one structure in all salt systems with a large surface area and thus the potential to easily remove tritium from the salt—the primary heat exchanger. If it does not have a highly effective permeation barrier, the tritium will go through the heat exchanger.

A permeation filter to remove tritium in a heat pipe will have a much smaller surface area per unit of tritium removed and thus is attractive as a tritium control strategy. The large heat pipe surface area for salt-to-sodium heat transfer implies tritium will easily enter the heat pipe unless the heat pipes have permeation barriers to prevent tritium diffusion from salt to sodium. Inside the heat pipe the vaporization and condensation of sodium sweeps and concentrates the tritium gas in the condensate section of the heat pipe. The tritium will rapidly diffuse to a permeation membrane tube because of the low density of sodium vapor relative to liquid flibe salt. At 673°C, the pressure will be a tenth of an atmosphere. This is diffusion through a low density gas about three orders of magnitude less dense than the liquid salt. The combination of using sodium vapor transport to concentrate hydrogen and the low density of sodium vapor relative to salt implies that the surface area of the permeation membrane in the heat pipe is much smaller than a permeation membrane to remove an equivalent amount of tritium from the liquid salt.

The alternative option is having a section of the heat pipe exposed to a vacuum and operating as a permeation membrane (Fig. 1). There are two options here. The first option is to have parts of the wall that are used as permeation membranes with no wick material to minimize liquid sodium between the sodium vapor and the

wall. The second option is to have sodium in the wick structure flow by the wall to remove tritium from the sodium. Liquid sodium has a high diffusion coefficient (Fig. 11) and high solubility (Fig. 12) for hydrogen and thus a high permeation rate for hydrogen⁵² where the hydrogen dissociates and is atomic hydrogen in the sodium like in solid metals. It is this combination of characteristics that creates the option to use heat pipes as tritium removal systems in a salt fission or fusion reactor. It also follows from Figs. 11 and 12 that electroplating tungsten on the inside of the condenser section of the heat pipe will limit tritium escape because of the very low solubility of hydrogen in tungsten. However, as discussed earlier, the sodium environment creates the option to use aluminum oxide or several other high-performance tritium permeation barriers—options that can't be considered for surfaces exposed to liquid salts that will dissolve such surface barriers.

Hydrogen diffusion into heat pipes and the use of permeation membranes to remove hydrogen has been demonstrated, but not for this application. Heat pipes are being developed for a variety of chemical processes (gasification, solid oxide fuel cell stacks, etc.) where the sodium evaporation (heating) zone is removing heat from hydrogen-rich gas streams many orders of magnitude greater than the tritium in a salt-cooled system. In these systems⁵³ sufficient hydrogen can diffuse into the heat pipe to create a hydrogen (inert-gas)-flooded condenser zone that shuts down heat transfer in the heat pipe in days. The condenser zone fills with inert hydrogen gas. As a consequence, permeation tubes for hydrogen removal have been developed to remove this hydrogen. Similar systems⁵⁴ with permeation filters have been designed and tested for specialized space applications to remove hydrogen where hydrogen diffusion into the heat pipe can shut it down. Some space nuclear reactors use hydrides in the reactor core to moderate neutrons and reduce the reactor size. These hydrides under some circumstances can lose hydrogen that diffuses into the heat pipes. In all of these systems (1) the quantities of hydrogen are orders of magnitude larger than the quantities of tritium that might enter a heat pipe in a salt reactor system and (2) the permeation of very small amounts of hydrogen through the condenser section of the heat pipe is not a concern. If such systems are used in salt reactor applications, there are tight constraints on allowable tritium escape from the sodium condenser side of the heat pipe system into the environment. We are not aware of any efforts to design and test heat pipes for this combination of constraints and operating conditions.

Tritium will chemically react with sodium to form hydrides⁵⁵ at lower temperatures that decompose at higher temperatures. At low concentrations the hydride is soluble in sodium. At very low temperatures there will be an equilibrium between (1) tritium diffusion into the heat pipe via the evaporator section, (2) tritium forming hydrides, and (3) tritium escaping from the heat pipe via a permeation surface that acts as a tritium removal system or via diffusion from the condenser section of the heat pipe. Lithium forms a much stronger bond with tritium than sodium. If some lithium is mixed with the sodium, one would expect more efficient capture of the tritium. However, any significant lithium will also raise the operating temperatures of the heat pipe. We are not aware of any studies that have undertaken an integrated study of all of the effects and options.

Last, permeation membranes may create other tritium control options. There is the option to use the permeation membrane with a very small surface area to add hydrogen or deuterium at low pressures to the heat pipe to create a hydrogen counter flux through the heat pipe back to the flibe salt to minimize tritium entry into the heat pipe. It is a countercurrent sweep gas. Systems removing the tritium from the salt (gas sparging, absorbers, etc.) would then recover tritium with the added hydrogen and deuterium from the heat pipes. The technical viability of this option depends upon material selection and the flibe surface of the heat pipe. Flibe and some materials such as tungsten have very low solubilities for hydrogen so the addition of a hydrogen/deuterium counter flow can fill the interatomic spaces in the metal or flibe to block tritium diffusion. This is only viable if the required concentration of hydrogen in the heat pipe can be sufficiently low that the hydrogen would not shut down heat pipe operation. To date, there have been insufficient analysis and experimental work to determine if such a system is a technically viable engineering option.

VI. STATUS OF TECHNOLOGY AND PATH FORWARD

Fission and fusion salt systems have two relatively unique challenges relative to other reactor coolants: (1) avoid freezing of the salt and (2) capture the tritium. Heat pipes have the potential to provide a simpler and lower-cost solution to these challenges. Some aspects of such heat pipes are well understood but significant work is required in other areas.

Sodium heat pipes for heat transfer are well understood. There is a significant experience base, they have been used in multiple applications, and work is underway

to develop larger heat pipe systems for applications including microreactors that couple to different types of power cycles at power levels of tens of megawatts. The larger-scale use of heat pipes is resulting in the development of better models for heat pipe design,^{56,57} but none of these models are capable of modeling heat pipes with all of the required capabilities as described herein. We are not aware of any work to develop heat pipe heat exchangers for transferring hundreds of megawatts of heat as may be required in a power reactor. The questions are how to design, manufacture, monitor, and repair large heat pipe systems. This is not a constraint if the heat pipe system is used for decay heat removal systems that are much smaller.

Sodium heat pipes that turn off and on at preset temperatures have been built and tested but there is limited operating experience. Most of the work has been associated with understanding the startup behavior of heat pipes, not systems where the temperatures decrease and the system slows down, such as in the long-term behavior of a decay heat removal systems as decay heat decreases with time.

There has been limited work on the performance of heat pipes that capture and remove hydrogen with permeation membranes built into the heat pipe. This work has been focused on systems where hydrogen diffuses into the heat pipe and shuts down the heat pipe; that is, systems where the heat pipe is removing high-temperature heat from a system with high concentrations of hydrogen. In these systems the goal is to keep the heat pipe operating, the hydrogen can be dumped to the environment, and the concentrations of hydrogen are relatively high. We have not identified work beyond very limited studies to develop heat pipes that include tritium recovery where the hydrogen (tritium) concentrations are low and the goals are to (1) capture the tritium and (2) very low tritium losses to the power cycle or environment. At the same time, there have been in the last five years rapidly growing efforts to remove and capture tritium from liquid salts, both in experimental work and development of models of tritium transport. Tritium transport to heat pipe surfaces and tritium diffusion through heat exchangers is now reasonably well understood. The science and technology base for a heat pipe to remove and capture tritium is being developed.

Acknowledgments

This work was supported by the U.S. Department of Energy and the SINAP of the Chinese Academy of Sciences.

References

- M. GIBSON et al., “The Kilopower Reactor Using Stirling Technology (KRUSTY) Nuclear Ground Test Results and Lessons Learned” (2019); <https://ntrs.nasa.gov/search.jsp?R=201800054352019-04-03T11:44:18+00:00Z> (current as of July 14, 2019).
- A. LEVINSKY et al., “Westinghouse eVinci™ Reactor for Off-Grid Markets,” *Trans. American Nuclear Society*, Vol. 119, Orlando, Florida, November 11–15, 2018 (2018).
- C. W. FORSBERG and P. F. PETERSON, “Basis for Fluoride-Salt-Cooled High-Temperature Reactors with Nuclear Air-Brayton Combined Cycles and Firebrick Resistance-Heated Energy Storage,” *Nucl. Technol.*, **196**, 1, 13 (Oct. 2016); <https://doi.org/10.13182/NT16-28>.
- C. ANDREADES et al., “Design Summary of the Mark-I Pebble-Bed, Fluoride Salt-Cooled, High-Temperature Reactor Commercial Power Plant,” *Nucl. Technol.*, **195**, 3, 223 (Sep. 2016); <https://doi.org/10.13182/NT16-2>.
- C. W. FORSBERG et al., “Integrated FHR Technology Development: Tritium Management, Materials Testing, Salt Chemistry Control, Thermal Hydraulics and Neutronics, Associated Benchmarking and Commercial Basis,” MIT-ANP-TR-180, Center for Advanced Nuclear Energy, Massachusetts Institute of Technology (Oct. 2018); <https://www.osti.gov/search/semantic:1485415> (current as of July 14, 2019).
- D. F. WILLIAMAS, L. M. TOTH, and K. T. CLARNO, “Assessment of Candidate Molten Salt Coolants for the Advanced High-Temperature Reactor (AHTR),” ORNL/TM-2006/12, Oak Ridge National Laboratory (2006).
- R. R. ROMATOSKI and L. W. HU, “Fluoride Salt Coolant Properties for Nuclear Reactor Applications: A Review,” *Ann. Nucl. Energy*, **109**, 635 (2017); <https://doi.org/10.1016/j.anucene.2017.05.036>.
- R. R. ROMATOSKI and L. W. HU, “Fluoride-Salt-Cooled High-Temperature Test Reactor Thermal-Hydraulic Licensing and Uncertainty Propagation Analysis,” *Nucl. Technol.*, **205**, 1495 (2019); <https://doi.org/10.1080/00295450.2019.1610686>.
- B. N. SORBOM, “ARC: A Compact, High-Field, Fusion Nuclear Science Facility and Demonstration Power Plant with Demountable Magnets,” *Fusion Eng. Des.*, **100**, 378 (Nov. 2015); <https://www.sciencedirect.com/science/article/pii/S0920379615302337> (current as of July 14, 2019).
- A. Q. KUANG et al., “Conceptual Design Study for Heat Exhaust Management in the ARC Fusion Pilot Plant,” Cornell University (2018); <https://arxiv.org/abs/1809.10555> (current as of July 14, 2019).
- C. FORSBERG et al., “Fusion Blankets and Fluoride-Salt-Cooled High-Temperature Reactors with Flibe Salt Coolant: Common Challenges, Tritium Control, and Opportunities for Synergistic Development Strategies Between Fission, Fusion, and Solar Salt Technologies,” *Nucl. Technol.*; <https://doi.org/10.1080/00295450.2019.1691400>.
- C. FORSBERG, “Leaky-Vessel Decay-Heat-Removal System for Fluid-Fuel Molten Salt Reactors,” *Trans. Am. Nucl. Soc. Winter Mtg.*, Washington, D.C., November 17–21, 2019 2019.
- C. WANG et al., “Transient Behavior of the Sodium-Potassium Alloy Heat Pipe in Passive Residual Heat Removal System of Molten Salt Reactor,” *Prog. Nucl. Energy*, **68**, 142 (2013); <https://doi.org/10.1016/j.pnucene.2013.07.001>.
- C. WANG et al., “Study on the Characteristics of the Sodium Heat Pipe in Passive Residual Heat Removal System of Molten Salt Reactor,” *Nucl. Eng. Des.*, **265**, 691 (2013); <https://doi.org/10.1016/j.nucengdes.2013.09.023>.
- M. LIU et al., “Experimental Study on the Heat Transfer Characteristics of Fluoride Salt in the New Conceptual Passive Heat Removal System of Molten Salt Reactor,” *Int. J. Energy Res.*, **42**, 1635 (2018); <https://doi.org/10.1002/er.3959>.
- C. WANG et al., “Conceptual Design and Analysis of Heat Pipe Cooled Silo Cooling System for the Transportable Fluoride-Salt-Cooled High-Temperature Reactor,” *Ann. Nucl. Energy*, **109**, 458 (2017); <https://doi.org/10.1016/j.anucene.2017.05.035>.
- M. BROVCHENKO et al., “Neutronic Benchmark of the Molten Salt Fast Reactor in the Frame of the EVOL and MARS Collaborative Projects,” *EPJ Nucl. Sci. Technol.*, **5**, 2 (2019); <https://doi.org/10.1051/epjn/2018052>.
- E. MERLE, “Concept of European Molten Salt Fast Reactor (MSFR),” CNRS-IN2P3-LPSC/Grenoble Institute of Technology/UGA—France (May 23, 2017); https://www.gen-4.org/gif/upload/docs/application/pdf/2017-05/emerle_gif-final23may2017.pdf (current as of July 14, 2019).
- “SAMOFAR: A Paradigm Shift in Nuclear Reactor Safety with the Molten Salt Fast Reactor” (June 2019); http://samofar.eu/wp-content/uploads/2019/06/D6.7_SAMOFAR-Dissemination-summary-report_v1.0_20190630.pdf (current as of July 14, 2019).
- B. FENG et al., “Core and Fuel Cycle Performance of a Molten Salt Fast Reactor,” presented at ICAPP 2019—Int. Congress on Advances in Nuclear Power Plants, Juan-les-pins, France, May 12–15, 2019.
- Advances in Small Modular Reactor Technology Developments*, International Atomic Energy Agency (2018); https://aris.iaea.org/Publications/SMR-Book_2018.pdf (current as of July 14, 2019).
- Moltex Energy; <https://www.moltexenergy.com/>; Supplement to: IAEA Advanced Reactors Information System (ARIS) (2018); https://aris.iaea.org/Publications/SMR-Book_2018.pdf.
- B. ZOHURI, *Heat Pipe Design and Technology: Modern Applications for Practical Thermal Management*, 1st ed., Springer Publishing Company (2016).

24. F. PERETZ and B. HORBANIUC, "Optimal Heat Pipe Heat Exchanger Design," *J. Heat Recov. Syst.*, **4**, 9 (1984); [https://doi.org/10.1016/0198-7593\(84\)90088-2](https://doi.org/10.1016/0198-7593(84)90088-2).
25. X. ZHA, S. B. RIFFAT, and G. GAN, "Heat Recovery with Low Pressure Loss for Natural Ventilation," *Energy Build.*, **28**, 179 (1998); [https://doi.org/10.1016/S0378-7788\(98\)00016-4](https://doi.org/10.1016/S0378-7788(98)00016-4).
26. M. S. YLEMEZ, "Optimum Length of Finned Pipe for Waste Heat Recovery," *J. Energy Convers. Manage.*, **49**, 96 (2008); <https://doi.org/10.1016/j.enconman.2007.05.013>.
27. L. L. VASILIEV, "Review Heat Pipes in Modern Heat Exchangers," *J. Therm. Sci. Eng. Appl.*, **25**, 1 (2005); <https://doi.org/10.1016/j.applthermaleng.2003.12.004>.
28. E. AZAD, "Review: Heat Pipe Heat Exchangers at IROST," *Int. J. Low-Carbon Technol.*, **8**, 3, 173 (Sep. 1, 2013); <https://doi.org/10.1093/ijlct/cts012>.
29. B. ZOHURI and P. McDANIEL, *Combined Cycle Driven Efficiency for Next Generation Nuclear Power Plants: An Innovative Design Approach*, 2nd ed., Springer (2018); <https://www.springer.com/gp/book/9783319705507> (current as of July 14, 2019).
30. B. ZOHURI, P. McDANIEL, and C. R. de OLIVERIA, "Advanced Nuclear Open Air-Brayton Cycles for Highly Efficient Power Conversion," *Nucl. Technol.*, **192**, 1, 48 (2015); <https://doi.org/http://dx.doi.10.13182/NT14-42>.
31. N. J. LAMFON, Y. NAIJAR, and M. AKYURT, "Modelling and Simulation of Combined Gas Turbine Engine and Heat Pipe System for Waste Heat Recovery and Utilization," *J. Energy Convers. Manage.*, **39**, 81 (1998); [https://doi.org/10.1016/S0196-8904\(96\)00175-6](https://doi.org/10.1016/S0196-8904(96)00175-6).
32. A. GREENOP, "Coiled Tube Gas Heater Effectiveness Modeling, Simulation and Experiments for Nuclear Power Conversion Cycles," PhD Thesis, University of California Berkeley (Spring 2018).
33. C. FORSBERG, "Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Temperature Control Options: Removing Decay Heat and Avoiding Salt Freezing," MIT-ANP-TR-183, Center for Advanced Nuclear Energy, Massachusetts Institute of Technology (Jan. 2019).
34. C. FORSBERG et al., "Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Using British Advanced Gas-Cooled Reactor (AGR) Refueling Technology and Decay-Heat Removal Systems that Prevent Salt Freezing," *Nucl. Technol.*, **205**, 1127 (2019); <https://doi.org/10.1080/00295450.2019.1586372>.
35. J. NAKATA et al., "Performance Evaluation of DRACS System for FHTR and Time Assessment of Operation Procedure," presented at the 21st Int. Conf. on Nuclear Engineering (ICONE21), Chengdu, China, July 29–August 2, 2013.
36. M. S. EL-GENK and J. P. TOURNIER, "Uses of Liquid-Metal and Water Heat Pipes in Space Reactor Power Systems," *Front. Heat Pipes (FHP)*, **2**, 013002 (2011); <https://doi.org/10.5098/fhp.v2.1.3002>.
37. B. ZOHURI, *Heat Pipe Applications in Fission Driven Nuclear Power Plants*, Springer (2019); <https://www.springer.com/gp/book/9783030058814> (current as of July 14, 2019).
38. M. G. HOUTS, D. I. POSTON, and W. J. EMRICH Jr., "Heat Pipe Power System and Heat Pipe Bimodal System Development Status," *Proc. Space Technology and Applications Int. Forum (STAIF-1998)*, Vol. 3, pp. 1189–1195, M. S. EL-GENK, Ed., American Institute of Physics, AIP-CP-420 (1998).
39. J. H. ROSENFELD et al., "An Overview of Long Duration Sodium Heat Pipe Tests," Thermacore International, Inc., NASA/TM-2004-212959 (2004); <https://ntrs.nasa.gov/archive/nasa/casi.ntrs.nasa.gov/20130013063.pdf>.
40. P. M. DUSSINGER, W. G. ANDERSON, and E. T. SUNADA, "Design and Testing of Titanium/Cesium and Titanium/Potassium Heat Pipes, Advanced Cooling Technologies," *Advanced Colling Technologies*(Aug. 2019); <https://www.1-act.com/design-and-testing-of-titaniumcesium-and-titaniumpotassium-heat-pipes> (current as of July 14, 2019).
41. W. H. RODEBUSH and E. G. WALTERS, "The Vapor Pressure and Vapor Density of Sodium," *J. Am. Chem. Soc.*, **52**, 7, 2654 (1930);
42. R. S. REID, J. T. SENA, and A. L. MARTINEZ, "Heat-Pipe Development for Advanced Energy Transport Concepts: Final Report Covering the Period January 1999 Through September 2001," LA-13949-PR, Los Alamos National Laboratory (Oct. 2002).
43. C. L. TIEN and A. R. ROHANI, "Theory of Two-Component Heat Pipes," *J. Heat Transfer*, **94**, 4, 479 (Nov. 1, 1972); <https://doi.org/10.1115/1.344997>.
44. P. McCLUE et al., "Design of Megawatt Power Level Heat Pipe Reactors," LA-UR-15-28840, Las Alamos National Laboratory (Nov. 2015).
45. W. R. DETERMAN and G. HAGELSTON, "Thermionic In-Core Heat Pipe Design and Performance," *Proc. 9th Symp. Space Nuclear Power Systems*, p. 1046, M. S. EL-GENK and M. D. HOOVER, Eds., American Institute of Physics, CONF-920104, AIP-CP-246, 3 (1992); <https://doi.org/10.1063/1.41913>.
46. J. ZHANG et al., "Redox Potential Control in Molten Salt Systems for Corrosion Mitigation," *Corros. Sci.*, **144**, 44 (2018); <https://doi.org/10.1016/j.corsci.2018.08.035>.
47. C. W. FORSBERG et al., "Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors: Status, Challenges, and Path Forward (Critical Review)," *Nucl. Technol.*, **197**, February, 119 (2017); <https://doi.org/10.13182/NT16-101>.
48. X. WU et al., "Conceptual Design of Tritium Removal Facilities for FHRs," presented at the 16th Int. Topl. Mtg. Nuclear Reactor Thermal Hydraulics (NURETH-16),

- American Nuclear Society, Chicago, Illinois, August 30–September 4, 2015.
49. T. TANABE et al., “Hydrogen Transport in Stainless Steels,” *J. Nucl. Mater.*, **123**, 1568 (1984); [https://doi.org/10.1016/0022-3115\(84\)90304-0](https://doi.org/10.1016/0022-3115(84)90304-0).
 50. A. NAKAMURA, S. FUKADA, and R. NISHIUMI, “Hydrogen Isotopes Permeation in a Fluoride Molten Salt for Nuclear Fusion Blanket,” *J. Plasma Fusion Res.*, **11**, 25 (2015); <https://pdfs.semanticscholar.org/4d8c/d2273bd3e3a99129105ea7e1bc5d6edcaecb.pdf> (current as of July 14, 2019).
 51. R. A. CAUSEY, R. A. KARNESKY, and C. S. A. N. MARCHI, “4.16-Tritium Barriers and Tritium Diffusion in Fusion Reactors,” *Comprehensive Nuclear Materials*, pp. 511–549, R. J. M. KONINGS, Ed., Elsevier, Oxford (2012).
 52. J. TROUVE and G. LAFLANCHE, “Hydrogen Diffusion and Permeation in Sodium,” presented at the Int. Conf. on Liquid Metal Engineering and Technology in Energy Production, Oxford, United Kingdom, CEA-CONF-7645, April 9–13, 1984.
 53. J. M. LEIMERT, M. DILLIG, and J. KARL, “Hydrogen Inactivation of Liquid Metal Heat Pipes,” *Int. J. Heat Mass Transfer*, **92**, 920 (Jan. 2016); <https://doi.org/10.1016/j.ijheatmasstransfer.2015.09.058>.
 54. M. T. NORTH and W. G. ANDERSON, “Hydrogen Permeation Resistant Heat Pipe for Bi-Modal Reactors: Final Report Period Covered,” Thermacore, Oct 1–September 30, 1994 (1995); https://inis.iaea.org/collection/NCLCollectionStore/_Public/27/063/27063177.pdf (current as of July 14, 2019).
 55. G. PARRY and R. J. PULHAM, “Rate of Reaction of Hydrogen with Liquid Lithium: Comparison with Sodium and Potassium,” *Dalton Trans.*, **19**, 1915 (1975); <https://doi.org/10.1039/dt9750001915>.
 56. J. E. HANSEL et al., “Sockeye: A 1-D Heat Pipe Modeling Tool,” INL/EXT-19-55742, Idaho National Laboratory (Sep. 2019).
 57. J. E. HANSEL et al., “Sockeye Theory Manual,” INL/EXT-19-54395, Idaho National Laboratory (June 2019).