

**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**

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*The FHR uses graphite-matrix coated-particle fuel (same as high-temperature gas-cooled reactors (HTGRs)) and a clean liquid salt coolant. It delivers heat to industry or the power cycle at temperatures between 600 and 700°C with higher average heat delivery temperatures than other reactors. The liquid-salt-coolant melting point is above 450°C. The high minimum temperatures present refueling challenges and require special features to control temperatures—avoiding excessively high temperatures and freezing of the coolant that could impact decay heat cooling systems. We describe herein a pre-conceptual FHR design that addresses many of these challenges by adopting features from the British AGR and alternative decay heat cooling systems. The basis for specific design choices are described.*

*The AGRs are carbon-dioxide cooled and graphite-moderated reactors that use cylindrical fuel subassemblies with vertical refueling at 650°C—meeting FHR high-temperature refueling requirements. The 14 AGRs have operated for many decades. The AGR uses 8 cylindrical fuel sub-assemblies each a meter tall coupled axially together by a metal stringer to create a long fuel assembly. The stringer assemblies are in vertical channels in a graphite core that provides neutron moderation. This geometric core design is compatible with an FHR using graphite-matrix coated-particle fuel. The FHR uses a once-through fuel cycle. The design minimizes used nuclear fuel volumes relative to other FHR and HTGR designs. The primary system is inside a secondary liquid-salt-filled tank that (1) provides an added heat sink for decay heat, (2) helps ensure no freezing of primary system salt, and (3) helps ensure no major fuel failures in a beyond-design-basis accident. The refueling standpipes above each stringer fuel assembly in the AGR core with modifications can be used in an FHR for refueling and provide efficient heat transfer between the primary system and the secondary liquid-salt-filled tank. The passive decay heat removal system uses heat-pipes that turn on and off at a preset temperature to avoid overheating the core in a reactor accident and avoid freezing the salt coolant as decay heat decreases after reactor shutdown.*

Keywords: Advanced Gas-cooled Reactor (AGR), Fluoride-salt-cooled High-Temperature Reactor (FHR), Refueling, Decay Heat Removal, Heat Pipe

## I. INTRODUCTION

The Fluoride-salt-cooled High-temperature Reactor (FHR) is a new reactor concept<sup>1-3</sup> that uses graphite-matrix coated-particle fuel developed for high-temperature gas-cooled reactors (HTGRs) and clean liquid fluoride salt coolants originally developed for molten salt reactors (MSRs). In an MSR the fuel is dissolved in the coolant. The core of a thermal-spectrum MSR is graphite that provides neutron moderation with the salt containing dissolved fuel flowing through channels in the graphite. An 8-MWt Molten Salt Reactor Experiment (MSRE) was built and operated successfully between 1965 and 1969 at Oak Ridge National Laboratory (ORNL). The MSRE demonstrated the compatibility of liquid fluoride salts with carbon at high temperatures—the feature that enables the FHR with solid fuel and clean coolant. The use of a salt coolant enables delivery of heat to the power cycle or industrial process at a temperature between 600 and 700°C.

We describe an FHR design in the early stages of development that uses many of the features of the British Advanced Gas-cooled Reactor (AGR). AGRs have successfully operated for decades in the United Kingdom. The salt coolant options are described first because it is their ability to deliver high-temperature heat that creates (1) the commercial basis for the FHR and (2) most of the technological challenges of the design. This is followed by a description of the AGR, the refueling technology that enables high-temperature FHR refueling, the proposed FHR design and the basis for selection of specific design features. This includes the temperature control systems necessary to avoid either freezing of the salt or overheating of the reactor under accident conditions. The system could be built in sizes up to thousands of megawatts thermal. Within this paper we use the term FHR to refer to characteristics of FHRs in general and the term AGR-FHR to refer to characteristics of an FHR that has incorporated features from the AGR.

### I.A. Salt Coolant Choices and Economic Implications

There are several coolant choices<sup>4-5</sup> with the leading candidates shown in Table 1. All coolants are mixtures of multiple salts to reduce melting points. For comparison, water properties under light-water reactor (LWR) conditions are given. The requirement to minimize neutron absorption in a thermal spectrum limits the coolant choices used to cool the reactor core. If the salt contains lithium, it must be isotopically-separated lithium-7, which has a low neutron absorption cross section. *Flibe, a mixture of lithium fluoride and beryllium fluoride, is the base-line coolant because of its superior nuclear and heat transfer characteristics.*<sup>5-6</sup> Flibe<sup>7-8</sup> has a very low nuclear cross section and thus ensures a negative void coefficient in large reactor cores for many different designs. The neutronic core design options with the sodium-zirconium salts are more limited. The better thermal-hydraulic characteristics of Flibe allow higher power densities<sup>6</sup>.

The advantage of the zirconium salts is that they avoid the use of toxic beryllium. The zirconium in other salt coolants is natural zirconium; however, Klydon Corporation<sup>9-11</sup> is operating a zirconium isotopic separation pilot plant in South Africa. Their goal is to isotopically separate zirconium isotopes to reduce neutron adsorption in zirconium-alloy fuel clad used in water-cooled reactors. This may create the option of using isotopically-separated zirconium and reduce its neutron absorption by a factor of four or more. This would significantly reduce the neutronic penalties of zirconium salts and could change preferred coolant options in the future.

TABLE I. Candidate Reactor Coolant Properties

Coolant	$T_{\text{melt}}$ (°C)	$T_{\text{boil}}$ (°C)	$\rho$ (kg/m <sup>3</sup> )	$\rho C_p$ (kJ/m <sup>3</sup> °C)
<sup>7</sup> Li <sub>2</sub> BeF <sub>4</sub> (Flibe)	459	1430	1940	4670
59.5 NaF-40.5 ZrF <sub>4</sub>	500	1290	3140	3670
26 <sup>7</sup> LiF-37 NaF-37 ZrF <sub>4</sub>	436		2790	3500
51 <sup>7</sup> LiF-49 ZrF <sub>4</sub>	509		3090	3750
Water (7.5 MPa)	0	290	732	4040
Water (15.5 MPa)	0	345	709	4049

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 is shown at 290°C (7.5 MPa, boiling point) and PWR conditions with water properties at 309°C.

From a thermo-mechanical design perspective, the defining characteristic of these salts relative to other nuclear reactor coolants is the high melting point. The practical implication is that typical reactor designs have reactor core coolant inlet temperatures near 600°C to minimize the risk of salt freezing and exit temperatures near 700°C based on temperature limits of economic materials for the construction of the heat exchangers.

The recent increase in interest in all salt-cooled reactor types is partly because salt reactors deliver a larger fraction of their heat at higher temperatures than any other class of reactors (Table 2)—a consequence of using a high-temperature liquid salt coolant and the small temperature rise across the reactor core. That implies the ability to deliver more higher-temperature heat to industry per MWt output and a higher heat-to-electricity efficiency with the ability to efficiently couple to combined-cycle gas turbines and other advanced power cycles.

TABLE 2: Typical Reactor Coolant Temperatures

Coolant	Average Core Inlet Temperature (°C)	Average Core Exit Temperature (°C)	Average Temperature of Delivered Heat (°C)
Water	270	290	280
Sodium	450	550	500
Helium	350	750	550
Salt	600	700	650

The higher average FHR temperatures of delivered heat have major economic implications. When the MSR was developed in the late 1960s, the peak steam power-cycle temperatures were at 550°C and the available steam cycles were not capable of taking advantage of the high-temperature heat that could be delivered by the reactor. The development and deployment of steam cycles to 650°C can now take advantage of these higher temperatures.

Equally important, this opens the option for Nuclear Air Brayton Combined Cycles (NACC). These cycles<sup>1,12,13</sup> operate in two modes: (1) base load and (2) peak power, where auxiliary fuel (natural gas, oil, biofuels, hydrogen) or stored heat raises the peak gas-turbine temperatures after nuclear heating to produce added electricity. The incremental heat-to-electricity efficiency for thermodynamic topping cycles for peak

electricity production can be 70% or higher—significantly above stand-alone natural gas plants. Because such plants would produce electricity with less natural gas per kWh, they would be dispatched before natural gas plants in peak power mode and have enhanced revenue from peak power production due to lower fuel costs.

Last, the high temperatures enable efficient coupling to heat storage technologies<sup>14-15</sup> for base-load reactor operation with variable electricity to the grid to maximize electricity revenue. Today the levelized cost of electricity from wind and solar is significantly lower than from nuclear plants. Operating costs are lower. In areas with good wind or solar resources this results in low or negative wholesale prices at times of good wind and solar output and low electricity demand—but higher prices at other times create large economic incentives to maximize sales at times of higher prices. Heat storage options include the use of intermediate loops with nitrate salts with gigawatt-hour heat storage. Hot stored salt is used as the storage media. There is also the secondary advantage that nitrates trap any tritium from the FHR in the form of steam that can be removed from the cover gas. These heat storage and power conversion systems are similar to existing heat storage systems used in solar power towers that have gigawatt-hour heat-storage systems to maximize their electricity output at times of higher prices and thus maximize revenue.

### I.B. British Advanced Gas-Cooled Reactor (AGR)

The 14 British AGRs have operated for decades. The AGR is a graphite-moderated carbon-dioxide-cooled reactor with exit temperatures of 650°C in a prestressed concrete reactor vessel (Fig. 1). The reactor powers a high-temperature steam cycle. The reactor core is graphite (the neutron moderator) with vertical holes for fuel assemblies. A partial AGR core cross section is shown in Fig. 2 including details of how the various graphite pieces are assembled. The larger holes are for fuel stringers and the smaller holes are for control rods and instrumentation.

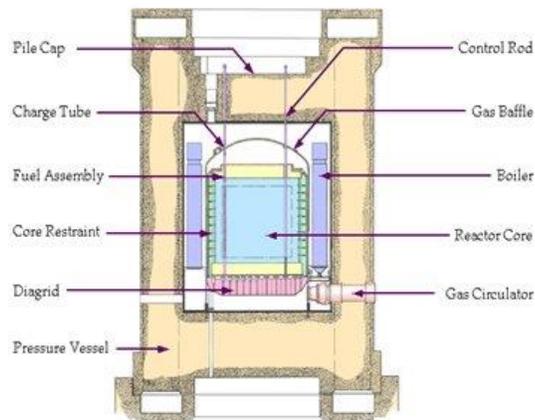


Fig. 1. General schematic of an Advanced Gas-cooled Reactor<sup>16</sup>

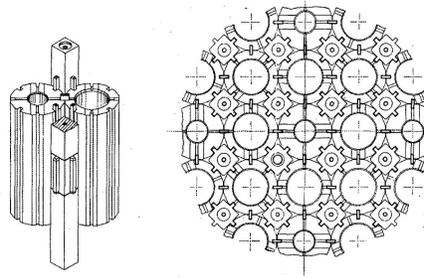


Fig. 2. Partial cross section of AGR core graphite components that are keyed together<sup>17</sup>

The AGR fuel consists of uranium dioxide in stainless-steel-clad pins. The fuel sub-assembly has pins in a circular array and includes an annular graphite shell (Fig. 3). The graphite shell receives most of the irradiation damage from fast neutrons and is part of the fuel assembly. This small amount of graphite is included within the fuel assembly and ensures a longer life for the graphite moderator blocks that remain in the core for the reactor lifetime. The same logic can be applied to an FHR with a similar design of core—the long-life graphite moderator receives a much lower radiation dose than the graphite in the fuel assembly.

Each AGR fuel sub-assembly is a meter in length. Multiple fuel sub-assemblies are tied together by a tie bar so that all of the fuel in a channel can be removed at one time. The AGR refuels at operating pressures and temperatures (650°C)—refueling temperatures much higher than any other existing reactor. Some of the AGRs refuel at low power levels. Refueling at full power is not done because if a fuel assembly is dropped and damaged, there may be local overheating of the fuel assembly—a constraint that may not exist for an FHR with liquid coolant that would limit drop speeds and provide better cooling than a gas. The AGR refueling strategy<sup>18</sup> is to have a standpipe above each fuel assembly with fuel pulled vertically from the core, through thermal insulation, through the concrete reactor vessel and to the refueling floor and into a fuel transfer cask. The fuel is then transferred to spent nuclear fuel (SNF) storage. This enables most mechanical components except the fuel grabber to be in air at room temperature—no complex equipment inside the reactor vessel. It is a simple way to refuel a reactor at high temperatures.

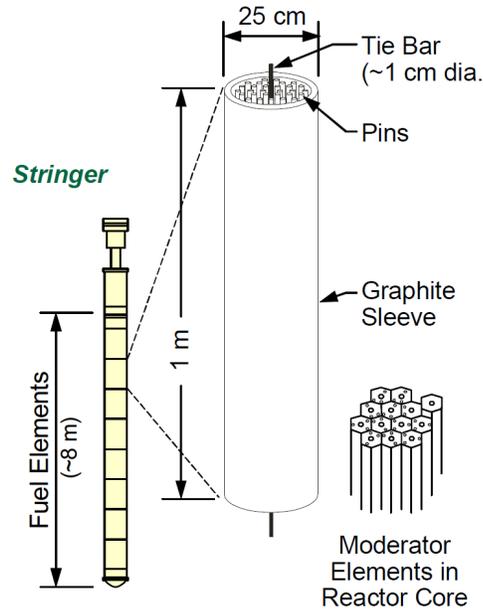


Fig. 3. AGR fuel assembly and graphite reactor core with fuel assembly holes

The refueling is done at reactor pressure; thus, the transfer cask operates at high internal gas pressures and has a high-pressure seal system to prevent leakage of carbon dioxide from the reactor onto the refueling floor and a gas control system within the transfer cask to avoid introducing air and other impurities from the transfer cask into the reactor. For an AGR-FHR a similar sealing system and gas control system is required to avoid introducing oxygen into the reactor.

The AGR refuels at high temperatures. Most other reactors refuel at low temperatures. The only other exception is the pebble-bed HTGR where pebbles flow through the reactor core. The AGR is the only reactor type where there is large-scale multi-decade experience in instrumentation, core inspection and maintenance at high temperatures—the temperatures that an FHR must be maintained at to avoid freezing of the salt. Like the AGR, the FHR has a graphite core—similar challenges. This experience base provides the basis for our work to consider an FHR variant of the AGR.

## II. FHR CORE DESIGN

*The base-line fuel for the proposed AGR-FHR herein is a variant of the HTGR prismatic fuel block where the fuel geometry is compatible with the geometric layout of the AGR core—including the stringer fuel design.* The geometric similarity enables the use of AGR technology such as high-temperature refueling for the proposed AGR-FHR. The HTGR graphite-matrix coated-particle fuel in the hexagonal form has been used in HTGRs. There are multiple fuel options including near-term options that use carbon-matrix coated-particle fuel (existing fuel technology) and more advanced fuel options that require significant efforts in fuel developments.

### II.A. Graphite-Matrix Coated-Particle Fuels

Recent work has developed a Fuel in Radial Moderator (FIRM) fuel assembly (Fig. 4) for application in an FHR test reactor<sup>19-20</sup>. The fuel assembly is a hexagonal block. The design characteristics of this fuel

assembly enable the use of AGR technology for an alternative design of AGR-FHR. The fuel assembly is similar to an HTGR prismatic fuel assembly except the fuel and coolant channels have been moved to the center of the block. This is possible because liquid salts are excellent coolants relative to helium and thus fewer coolant channels are needed if salt cooling replaces helium cooling in an HTGR-type fuel block.

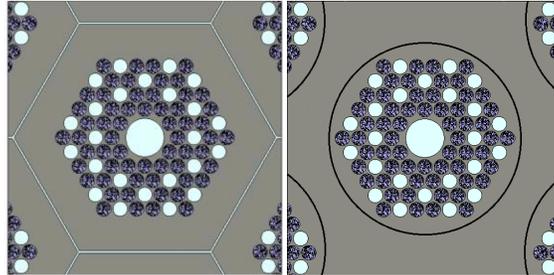


Fig. 4: FHR FIRM test reactor assembly (left) and a notional AGR-FHR FIRM assembly with separate fuel and moderator regions (right). Coolant channels are shown in light blue.

The TRISO coated-particle fuel is embedded in a carbon matrix in the form of cylindrical compacts. The compacts are in holes drilled into the hexagonal graphite block and sealed at both ends. The coolant channels are drilled through the graphite block. The manufacturing limit for graphite block height is 1 to 2 meters; thus the fuel assembly length is limited.

The basic core geometries of the AGR-FHR with FIRM and the AGR are similar—graphite with circular zones containing the fuel and coolant channels. The AGR fuel assemblies are replaced with the fuel-coolant zone of the FIRM assemblies—a graphite cylinder with coolant channels and holes filled with fuel compacts (graphite matrix with coated-particle fuel). The surrounding graphite would be part of the permanent reactor graphite structure. Multiple short fuel sub-assemblies would be held together by a stringer so that all the fuel can be removed from a single fuel channel—same refueling strategy as discussed earlier for the AGR. The similar heights of the AGR-FHR and AGR sub-assemblies are controlled by the ability to fabricate graphite structures that can withstand the reactor environment. The FHR FIRM has temperature limits in excess of 1650°C (the limit of the TRISO fuel) allowing much higher temperature operations and excellent performance under accident conditions.

The FIRM fuel assembly design was a result of an optimization process to increase fuel burnup by reducing parasitic neutron losses. Fast neutrons are generated in the fuel zones, migrate to the graphite, are thermalized with minimal neutron absorption in the graphite away from the fuel, and migrate back toward the fuel to cause thermal fissions. It is at a fundamental level the same neutronic strategy used in CANDU and AGR reactors with some differences to increase fuel burnup. This leads to the following similarities.

- *Moderator.* In the CANDU reactor the moderator is heavy water whereas in the AGR and FHR the moderator is graphite.
- *Fuel.* In the CANDU reactor and AGR the fuel is low-enriched uranium dioxide fuel pellets. In the FHR the fuel is uranium oxy-carbide particles coated with layers of SiC and carbon. The fuel particles are embedded in graphite-matrix compacts that are then embedded in channels in a circular graphite fuel assembly.
- *Coolant.* In the CANDU reactor the coolant is pressurized heavy water, in the AGR the coolant is pressurized carbon dioxide and in the FHR it is a low-pressure liquid fluoride salt.

The FIRM assembly design shown in Fig. 4 involves the lowest risk of designs with cores similar to the AGR because of the experience with HTGR hexagonal fuel blocks; but, there are other fuel variants—each with specific disadvantages:

- *High-Temperature Test Reactor (HTTR) variant.* The Japanese HTTR<sup>21</sup> has the fuel pellets inside graphite cylinders that are in the middle of the coolant channels. There is no fuel in the hexagonal graphite block. For this concept, the hexagonal graphite block can be replaced by a cylindrical graphite structure. This option minimizes fuel temperature and so improves safety margins; but there is a neutronic penalty for an FHR variant. Thermal neutrons are absorbed by the salt while migrating from graphite through the coolant to the fuel.
- *Annular fuel tubes.* If an annular fuel tube can be fabricated with coated-particle fuel, the fuel tubes can be used to line the holes drilled in the cylindrical graphite sub-assemblies. This design also minimizes fuel temperatures. It has a neutronic advantage. Fast neutrons enter the graphite and slow down. The resulting thermal neutrons in the graphite migrate towards the fuel and are absorbed in the fuel before reaching the coolant, resulting in lower parasitic absorption of neutrons by the coolant—much reduced neutron coolant interactions

## II.B. Advanced Fuel Options

More recent work<sup>7-8</sup> has examined a much broader set of AGR-FHR fuel and coolant options while maintaining the general geometric arrangement of the AGR. This has included (1) different fuel types, (2) different salt coolants, and (3) parametric studies of fuel, coolant and carbon ratios to maximize fuel burnup while maintaining negative coolant temperature coefficients. Recent work<sup>7</sup> has examined core designs with Flibe coolant and fuel pins made of silicon carbide containing uranium carbide fuel. The results, based on single pin models, indicate that the best designs in terms of beginning of cycle performance have fuel pins with (1) pitch to diameter ratios between 4.4 to 5.2, (2) 40% to 100% salt mass fractions in the non-fuel (graphite plus coolant) zones, and (3) enrichments between 5 wt% to 10 wt%. This is work in progress.

Other studies<sup>8</sup> have examined the potential to use sodium-zirconium fluoride salts with isotopically separated <sup>90</sup>Zr. Such coolants avoid the complications of using Flibe (tritium production and beryllium toxicity) but have higher neutron absorption with concerns about positive void coefficients in large reactors with small neutron leakage. Potentially viable designs that meet the multiple constraints of a reactor core have been identified. South Africa has a zirconium isotopic separation pilot plant where the initial market is isotopically-separated zirconium for LWR fuel clad and CANDU reactor pressure tubes. If isotopically-separated zirconium<sup>9-11</sup> becomes commercially available, it would make fluoride salts containing zirconium much more attractive for use in FHRs.

## III. FHR DECAY HEAT REMOVAL SYSTEMS

### III.A. Heat Removal Requirements

The FHR heat removal systems have two challenges: (1) remove decay heat upon reactor shutdown to prevent reactor damage or fuel failure from high temperatures and (2) avoid freezing of the coolant salt when the reactor shuts down and the decay heat decreases with time. If salt freezes in a decay-heat removal

system, it could prevent the removal of decay heat. This is not a concern for light-water, sodium or gas-cooled reactors. This is not unique to salt-cooled reactors. Lead-cooled reactors<sup>22</sup> must also address some of these same challenges because lead also has a relatively high melting point of 327°C.

### III.B. System Design

To address the temperature control challenge, the base-line design has the primary reactor vessel be enclosed in a second vessel filled with a second, low-cost salt, as shown in Fig 5. There are several advantages with this approach:

- *Freeze control.* The secondary salt (such as Flinak) bathes the outsides of standpipes, control rods systems and other high-surface-area components in a constant temperature bath thus reducing the risk of salt freezing. One converts a system with a complex outer geometry that requires full insulation into a simple geometry to insulate or use as a cooling surface if desired for decay heat cooling. Pump casings, auxiliary primary salt tanks including spent nuclear fuel storage systems and other equipment, can be kept warm in the secondary salt tank.

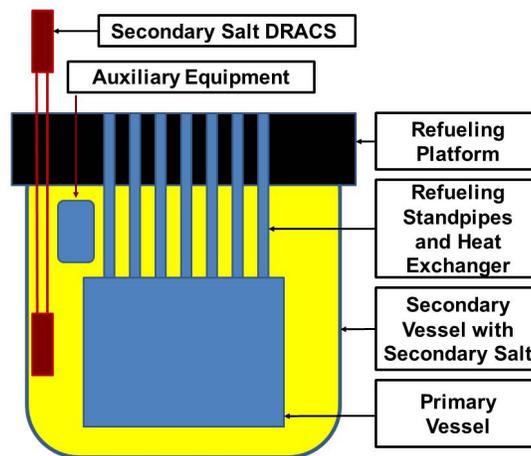


Fig. 5. Temperature management and refueling in AGR-FHR with FIRM fuel assemblies

- *Heat sink.* The secondary-tank salt provides a massive heat sink that can absorb decay heat resulting in (1) slower heat-up of the entire system after shutdown and (2) slower cool-down after reactor shutdown. One could have a single primary tank but the primary coolant salts are expensive. With a low-cost secondary salt, one does not need to minimize volumes because of salt costs. Figure 6 shows the decay heat versus time for one nominal FHR design. While the absolute decay heat depends upon the reactor power level, the shape of the curve is nearly identical for all thermal-spectrum reactors operating on a uranium fuel cycle.

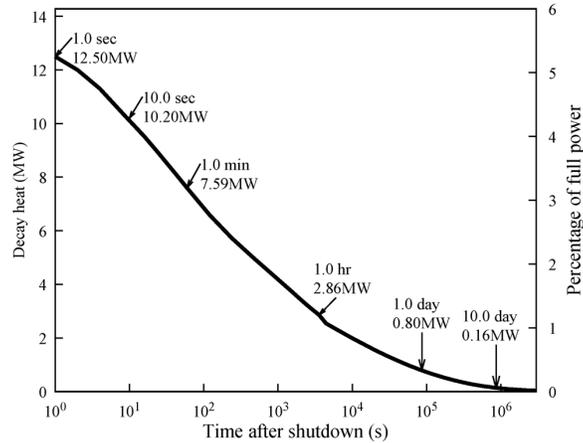


Fig. 6. Decay heat versus time in a thermal spectrum reactor. Initial power 236 MWt

The decay heat initially decreases very rapidly with time but then falls off at a slower rate—the curve becomes concave after an hour. With a salt system, there is the requirement to remove decay heat but not overcool and freeze up the decay heat system with frozen salt. If one has to remove large quantities of decay heat immediately after an accident, the decay heat removal system will have to reduce its decay heat removal capabilities to very low levels within a few days to avoid freezing the salt flowing through the decay heat removal system. If there is a large heat capacity per unit of thermal reactor power level (secondary salt tank), the “turn down” requirements for the decay heat removal system to avoid freezing salt several days after an accident are reduced. As described below, this greatly simplifies design of a decay heat removal system that shuts off below a preset temperature.

There is one standpipe per string of sub-assemblies in the reactor core with a diameter slightly larger than the fuel assembly. This creates, in effect, a massive heat exchanger directly above the core to equilibrate reactor coolant temperatures with the secondary salt.

- *Radiation shielding.* A secondary salt provides a liquid shield that is effective in avoiding radiation streaming toward the refueling floor. Fluoride salts have short-lived fluorine activation products ( $^{16}\text{N}$ :  $t_{1/2} = 7.1$  seconds and  $^{19}\text{O}$ :  $t_{1/2} = 27$  seconds) with hard gamma rays. Because the secondary salt does not go through the reactor core, it lowers the radiation levels near the operating floor. There is primary salt in the standpipes that are close to the refueling floor but that salt moves at a slow rate and thus has relatively low radiation levels. During refueling, the refueling machine provides added radiation shielding when transferring fuel out of the reactor core.
- *Decay heat removal.* The FHR will have an active decay heat removal system for normal operations to carefully control temperatures to reduce thermal fatigue. As discussed earlier, there are the dual requirements to prevent overheating and overcooling (salt freezing) the reactor while minimizing heat losses during normal operations. Two options are described below. A decay heat cooling system may also be installed in the secondary salt system.

The standpipes are used for refueling and as heat exchangers between the primary tank and the secondary tank. During refueling, cooling is required for the fuel assembly. If there is on-line refueling, there will be significant decay heat in a fuel assembly just extracted from the core. Each standpipe as a cooling system has several components (Fig. 7):

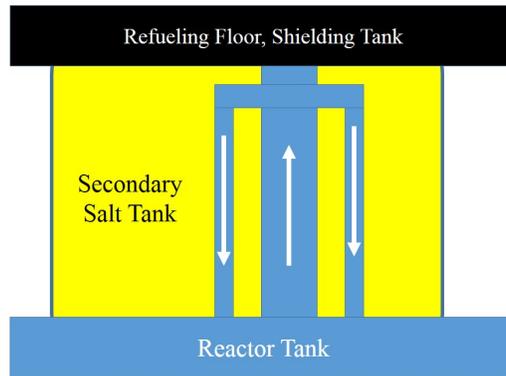


Fig. 7. Schematic of single standpipe with two return channels

- *Standpipe.* The standpipe is sized to enable vertical removal of a stringer fuel assembly.
- *Return tubes.* There are one or more tubes that return the salt from near the top of each standpipe back into the main reactor vessel. If the secondary salt is cooler than the primary salt, primary hot salt will flow up each standpipe and down the secondary tubes. The secondary tubes are designed with better heat transfer (higher surface to flow ratio) so the salt is cooler and thus denser in the tubes and flows by natural circulation downwards. During refueling there will be spent fuel in the standpipe generating decay heat. This will heat the salt in the standpipe resulting in higher rates of primary salt circulation and cooling. The surface area of the tubes determines the rate of heat transfer from primary to secondary salt. The return tube or tubes with each standpipe can be a single piece of hardware.

The size of the secondary tank will be controlled by (1) the equipment that must fit within it (primary reactor vessel, standpipes, auxiliary equipment, etc.), (2) shielding requirements to minimize neutron and gamma dose external to the secondary tank, (3) size of secondary decay heat sink that is desired or (4) the Beyond-Design-Basis-Accident (BDBA) safety systems that partly depend upon secondary salt volumes. The secondary tank may include pump bowls and intermediate heat exchangers depending upon reactor system design goals—as do some designs of integral SFRs. Shielding requirements may dictate shielding in the secondary tank that includes steel for gamma shielding and graphite for neutron moderating and shielding. Graphite may be used for neutron moderation and shielding because of its chemical compatibility with high-temperature salts. All other components add to the heat capacity of the secondary tank. .

### III.C. Decay Heat Removal

Multiple strategies for decay heat removal have been identified<sup>23</sup> (Direct Reactor Auxiliary Cooling System (DRACS) and a heat pipe assembly) that meet the goals of (1) avoiding overheating by decay heat and (2) avoid overcooling with the danger of freezing parts of the decay heat removal system that would then allow the reactor core to overheat due to frozen salt elsewhere in the system. Two options are described herein. *The baseline option uses heat pipes because of their demonstrated temperature control capability to turn on and off at a preset temperature but this option is less developed.*

Central to the choice of decay heat cooling systems are the design goals. Our goal herein is a passive walk-away safety system. A conventional DRACS could be used but after several days there would be the need to degrade its performance to avoid freezing salt—such as reducing air flow through the DRACS with louvers.

### III.C.1. Direct Reactor Auxiliary Cooling System

References describe one freeze-tolerant DRACS<sup>24-25</sup> that has been designed for a single-tank FHR. This option has received the most study. The FHR and DRACS use FLiBe as the coolant. The nominal reactor core inlet temperature is 600°C and the exit temperature is 700°C. The maximum core outlet temperature depends upon materials of construction. A recent assessment<sup>26</sup> concluded that in an accident the peak temperature must not exceed 970°C to ensure against structural failure of the 316 stainless steel. The high-temperature fuel and coolant implies that temperature limits are defined by components outside the reactor core.

A schematic layout of DRACS is shown in Fig. 8. It is a natural circulation heat transfer system that transfers heat from the primary system to the atmosphere. The DRACS heat exchanger (DHX) in the primary system is located between the lower higher-pressure cold-salt plenum and upper hot-salt plenum. 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 during normal operations. If the main salt circulation pumps stop, there is natural circulation of primary salt from the hot-salt plenum above the reactor core through the DHX to the bottom cold-salt plenum and back through the reactor core. In the DHX the heat is transferred to the DRACS salt that by natural circulation transfers the heat to an external heat exchanger and ultimately to the atmosphere.

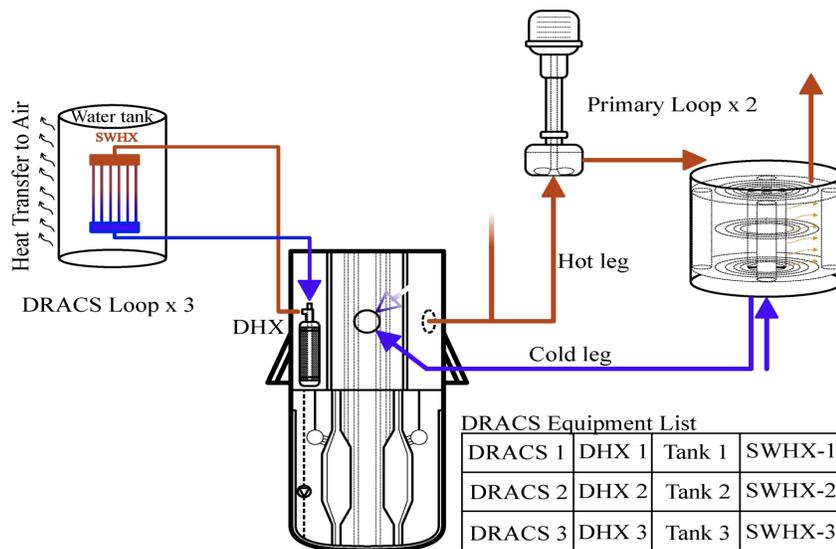


Fig. 8. Direct Reactor Auxiliary Cooling System for FHR

The hot salt in DRACS rejects its heat through the Salt Water Heat Exchanger (SWHX) and the water transfers heat to air through the surface of the water tank. Some water boils off during initial operation of DRACS when decay heat rates are high. To avoid freezing of the DRACS salt when not in operation, the SWHX is a double tube heat exchanger with helium between the two tubes to increase the temperature drop from hot circulating salt to water.

As the reactor decay heat decreases and the salt temperature decreases in DRACS, the salt may freeze in the SWHX tubes. There are different lengths of SWHX tubes in DRACS with the longest tubes, which have the largest area for heat transfer, providing the greatest amount of cooling with the slowest salt flow rates. Thus, the liquid salt going through the longest SWHX tubes freezes first. As some heat exchanger tubes freeze up, decay heat removal decreases and the reactor temperatures increase. The hotter salt entering DRACS stops the freezing process in the shorter tubes. The primary coolant temperature oscillates in multiple cycles with additional tubes being blocked with frozen salt in each cycle. The system can ensure decay heat removal for hundreds of hours before freezing of all of the SWHX's tubes or requiring the implementation of other features to avoid freezing. Decay heat removal can be extended for much longer times by degrading DRACS performance—by draining some of the water out of the SWHX after the first hundred hours so that the top of the tubes is exposed to a gas environment which reduces heat transfer efficiency between the SWHX and the air-cooled tank walls.

The addition of a secondary salt tank increases the system heat capacity. Decay heat rapidly decreases (Fig. 6) during the first few hours and then decreases at a much slower rate. The secondary salt tank (1) lowers the required peak heat removal capacity of DRACS and (2) enables DRACS to operate for a much longer period of time before freeze-up of components. The above system has two important design points. The first is the ratio of peak decay heat removal to nominal heat rejection through DRACS during normal

reactor operation. In this specific design, that ratio is 2.29. The second design point is the ratio between peak decay heat removal rate and heat removal rate when the DRACS totally freezes unless operator action is undertaken to reduce system performance. In this specific design, that ratio is 19.02. For any given peak-to-freeze heat rejection ratio for DRACS, the time between these two events increases rapidly when the time of peak heat rejection occurs later because of the secondary salt tank and concave shape of the decay heat curve. The larger the secondary salt tank, the longer the time before peak heat rejection occurs.

There is also a second effect. As system temperatures increase there are increased heat losses by conduction everywhere in the system. This slows down the rate of temperature increase with time after reactor shutdown. At some point heat losses by conduction become important and ensure decay heat removal independent of DRACS.

There are a variety of other options to boost the effective peak to minimum decay heat removal rates before salt freezing, such as replacing the helium in the double-wall SWHX with vacuum. Table 3 shows radiative heat transfer as a function of temperature and emissivity. The heat transfer goes up by more than a factor of four from 600 to 1000°C. It drops off rapidly as the salt temperature decreases. Such mechanisms slow down decay heat removal as temperatures decrease.

TABLE 3. Heat Flux (kW/m<sup>2</sup>) Between Two Infinite Black Plates with Increased Temperature (°C) Versus Emissivity

Emissivity	Temperature (°C)									
	100	200	300	400	500	600	700	800	900	1000
1.0	0	1.74	5.02	10.5	19.2	31.9	49.8	74.1	106.	148.
0.8	0	1.16	3.35	7.03	12.8	21.3	33.2	49.4	70.9	98.6
0.5	0	0.58	1.67	3.51	6.39	10.6	16.6	24.7	35.4	49.3

### III.C.2. Decay Heat Removal Using Heat-Pipe Assembly

An alternative heat removal system can be built using heat pipes containing metallic lithium, potassium or sodium. Depending upon the plant configuration, the heat pipes would transfer heat (1) from the primary salt coolant to air or (2) from a DRACS loop to the air (replacing the SWHX and water tank as described above in section III.C.2). Heat pipes have the interesting characteristic that their heat transfer capability is a strong function of temperature—they shut down at lower temperatures. There has been large-scale development of sodium, potassium, and lithium heat pipes, primarily for space nuclear reactors.<sup>23,27</sup> These studies and experiments have included startup and shutdown under hot and cold temperatures far beyond those that would be expected any place on earth.

TABLE 4. Heat Pipe Coolant Options

Fluid	Boiling Point (°C) at 1 atm	Typical Operating Range (°C)
Lithium (Li)	1330	825-1125
Sodium (Na)	883	625-825
Potassium (K)	759	325-525

The baseline decay heat removal system for our proposed AGR-FHR is a heat pipe system<sup>23</sup> that uses sodium because of its physical properties (Table 4) and the expected operating conditions. Each heat pipe (Fig. 9) would have its bottom half in the hot primary coolant salt and its top half in the DRACS air flow. 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. 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 to the hot salt zone where it is evaporated. The heat pipe operates independent of orientation with a constant sodium vaporization temperature. If there were a pool of sodium, the boiling point of the sodium would be greater at the bottom than further up the heated zone. This strategy also minimizes the heat-pipe sodium inventory. Assuming there is sufficient heat transfer area, the rate of heat transfer depends primarily upon two coupled parameters that result in high heat transfer at high temperatures and low heat transfer at low temperatures:

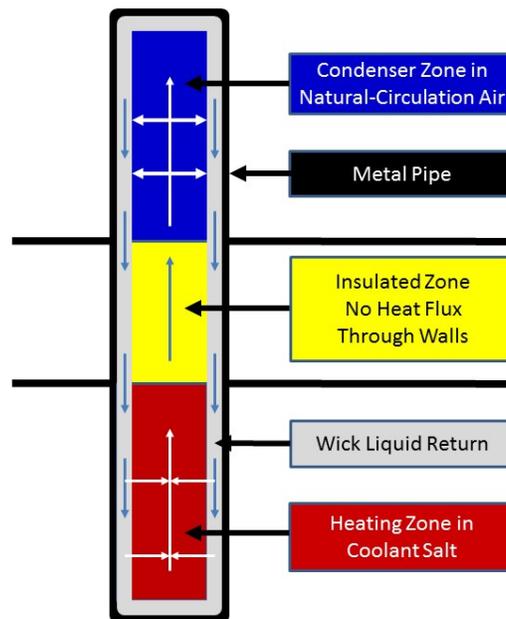


Fig. 9. Heat Transfer Regions within a Heat Pipe

- *Sodium vapor pressure.* As the temperature goes up, the sodium vapor pressure<sup>28</sup> goes up and more heat is transferred per cubic meter of vapor from the hot zone to the cold zone.
- *Sonic velocity.* The limit of sodium vapor flow is the sonic velocity of the gaseous sodium that is controlled by heat pipe geometry.

The startup temperature of the heat pipe can be more precisely controlled by the addition of an inert gas to the heat pipe (Fig. 10). The 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 inert gas and the sodium vapor is pushed into the condenser section. It 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 shuts down as temperatures go down.

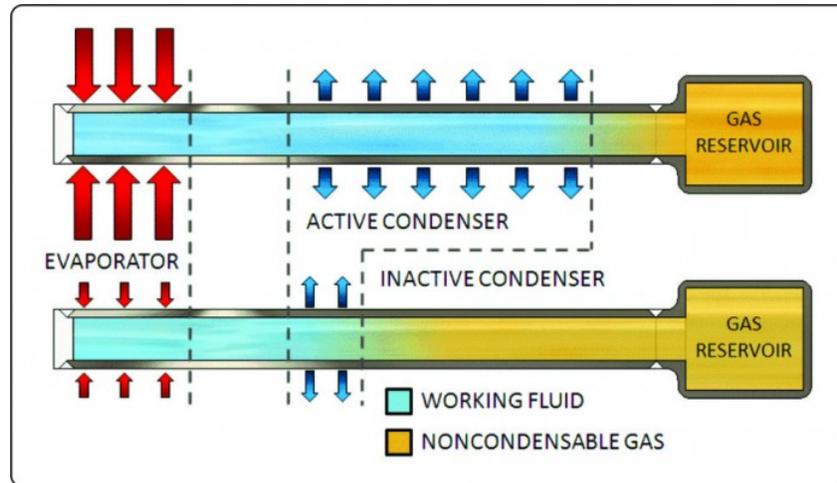


Fig. 10. Schematic of Variable Conductance Heat Pipe

There have been many sodium heat pipes built and tested for space applications including long-duration runs (>100,000 hours) 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<sup>29-30</sup> where heat would be transferred at temperatures ranging from 900 to 1100°C.

Initial analysis indicates viability, but detailed studies have not yet been undertaken for this application. However, because of the massive amount of work that has been done for space and other programs, the range of performance and operational limits are well understood.

Heat pipes can be designed to be large or small. In an FHR 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 the reactor. Sodium is a strong chemical reducing agent that will reduce beryllium fluoride and many other fluorides that may be salt components<sup>31</sup> 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 and potentially degrade fuel. These parameters will place upper limits on individual heat-pipe sodium inventories to limit consequences of leaks. In this context the use of a wick is important for it minimizes the sodium inventory in any individual heat pipe.

Another consideration is tritium transport from the primary coolant to the environment via the decay heat removal system. Primary salts that contain lithium and beryllium generate tritium. FHRs have tritium removal systems but tritium leakage must be limited. Tritium barrier materials such as tungsten are chemically compatible with sodium and have been used in sodium heat pipes—providing one option to minimize tritium releases.

#### III.D. Choice of Secondary-Tank Salt

There are many secondary-tank salt options depending upon requirements. The primary options are fluoride and chloride salts that are chemically compatible with the primary salt in the event of a leak between the two systems. If the AGR-FHR has a heat transfer loop between the reactor coolant and power cycle, there will be incentives to use the same coolant in the secondary tank and intermediate heat transfer loop for ease of operations. One must also consider how one would recover from a leak between the two systems—is there an easy way to purify the primary coolant after a leak? The Molten Salt Reactor Experiment (MSRE), an 8-MWt test reactor built in 1964 and operated until 1969, used the same Flibe salt in the primary and the intermediate-loop heat-transfer system with isotopically-separated lithium-7 to avoid addressing such challenges.

*The base-line secondary-tank and intermediate-loop salt is Flinak<sup>5</sup>, a eutectic alkaline metal fluoride salt mixture LiF-NaF-KF (46.5-11.5-42 mol %) with a melting point of 454°C and a boiling point of 1570°C. This salt is well understood and relatively non-toxic (no beryllium). For many proposed salt-cooled reactors it is the preferred choice for an intermediate heat transfer loop. From the cost perspective, one of the major questions is whether the secondary-tank salt can contain a lithium salt, and if it contains lithium, whether it should be isotopically-separated lithium 7. Lithium is used in these salt mixtures to lower the melting point. If normal lithium is used, the cost is low. However, if there is a leak between the primary and secondary-tank systems there will be mixing of the separated lithium-7 with natural lithium that will require isotopic separation to recover the lithium-7.*

Lithium is a light element where there are large differences in the chemical behavior of the two isotopes—creating many potentially low-cost isotopic separation options. Different processes are being developed in China and at ORNL to replace the old process that used mercury. The question is the size of the market. There are large economics of scale in isotopic separation processes. There are potentially significant markets for isotopically-separated lithium. Lithium is used in lithium batteries where using lithium-6, with its higher diffusion rate in batteries results in a higher performance lithium battery—more power per unit weight. That is important for applications such as aircraft and spacecraft. If lithium isotopic separation is scaled up to supply the high-end battery market, it would imply low costs for lithium isotopic separation. A second market is high-performance space vehicles where lithium is an alloying ingredient in aluminum and other alloys. In the long-term, fusion may require isotopically separated lithium-6.

The low nuclear cross-section non-lithium fluoride salt is the sodium zirconium fluoride salt that is a candidate reactor coolant (Table 1). There are many other options with cations with high neutron cross sections. If one of these fluoride salts is used, the question will be how to purify the primary coolant salt if there is a leak and some secondary salt enters the primary system. Ongoing research on MSRs may provide lower-cost technologies that would shift the choice of secondary salt. Most separations in the industrial world are done by distillation—an option that was examined in the 1970s at ORNL in the context of the MSR program. The difficulty at that time was finding materials of construction that were corrosion resistant and could operate at extremely high temperatures. The recent development of additive manufacturing for refractory alloys now creates the option to build distillation columns from moly and other very high-temperature alloys.

The leading chloride salt candidate is a sodium potassium magnesium chloride salt (24.5 wt % NaCl, 20.5 wt % KCl and 55 wt % MgCl<sub>2</sub>) eutectic<sup>32-33</sup> that has a very low cost and a very low melting point of 387°C. In the last several years there has been a rapidly expanding effort to develop this salt as a coolant and storage medium.

- *Heat storage.* This salt has become the primary candidate for advanced concentrated solar power towers that would operate at peak temperatures above 700°C. Existing concentrated solar power plants operate at temperatures below 600°C because nitrate salts decompose at higher temperatures. In a solar power tower, the mirrors are the largest cost; thus, there are very large economic incentives to go to higher temperatures to boost plant efficiency. The chloride salt would be used as the primary coolant and for large-scale sensible heat storage to enable the solar power tower to produce electricity at times of higher prices after the sun has set. This same chloride salt because of its very low cost is the leading candidate for advanced higher-temperature heat-storage systems<sup>14</sup> that might be coupled to an FHR or a MSR to enable base-load reactor operation with variable electricity to the market to maximize revenue.
- *Intermediate heat transfer loop coolant.* The same salt is a candidate for the intermediate heat transfer loop for salt reactors—particularly the molten chloride fast reactor. The salt is compatible with the primary coolant where a little leakage into the primary coolant would not be a big concern.

### III.E. Radiation Heat Transfer

In the FHR, thermal radiation becomes an important heat transfer mechanism in addition to convection and conduction under normal operation, and particularly under accident conditions. Thermal radiation can have a significant impact on the temperature distribution in the reactor systems, structures and components, and therefore could potentially change reactor safety characteristics. This poses a challenge in modeling and simulation of such coupled convection and radiation heat transfer.

Conduction and convection heat transfer in most systems increase linearly with the temperature drop from hot to cold. Radiative heat transfer increases with the fourth power of the absolute temperature and begins to become significant above 600°C. In most systems insulation involves trapping of gases with their low thermal conductivity—whether in conventional insulation or ceramic insulation. Radiative heat transfer boosts heat transfer across those gas gaps. Furthermore, liquid salts are semitransparent and can be a participating media for radiative heat transfer.<sup>34-35</sup> This is a system that bleeds heat as temperatures increase. In real systems this will become a significant factor in limiting temperature rises under any over-temperature transients.

## IV. FHR BEYOND DESIGN BASIS ACCIDENT (BDBA) SYSTEMS

*A BDBA strategy has been developed for the proposed AGR-FHR; but, there is no single base-line BDBA system because the system design will depend upon the thermal output of the reactor and other design features.* For a small FHR, there is no need for a BDBA system because decay heat will leave the system via conduction and other pathways to maintain coolant temperatures below fuel failure temperatures. For a large FHR, there is the potential to heat fuel above its failure temperature and release radionuclides if all the decay heat removal systems fail. In all cases the BDBA system is based on the combination of a high-temperature fuel and a high-temperature liquid salt coolant. The high volumetric heat capacity of the salts will help slow down core heat-up and thus increase coping time for accident mitigation. Conceptually the BDBA system to ensure no major fuel failure under any conditions is simple. The normal reactor operating temperature is 700°C with salt boiling points above 1400°C and fuel failure temperatures above 1650°C. If there were no insulation, there would be more than 1200°C temperature drop to move decay heat from the reactor core to the earth—an extraordinarily large temperature driving

force to move decay heat out of the system before fuel failure temperatures are exceeded. The primary factor stopping massive heat losses is the insulation—thus a mechanism is required to destroy the insulation in a severe accident.

Limited work on a BDBA system has been done<sup>36</sup> using the secondary-tank salt to ensure no large-scale fuel failure for relatively large reactors. The requirements are (1) sufficient secondary salt to ensure the salt level inside the reactor silo compartment will be significantly above the reactor core level if the secondary salt tank fails and (2) the secondary tank fails before boiling of the primary or secondary salt. Given the very high boiling points of these salts, the second requirement is the likely reality with most designs—whether a design goal or not.

In a BDBA, the secondary-tank salt after vessel failure is used to destroy the insulation and allow efficient heat transfer to the ground heat sink. The nominal design of the reactor silo from inside to outside (Fig. 11) is primary system, secondary-tank, insulation, silo wall cooling system, concrete silo wall and earth structure. The silo wall cooling system removes heat that leaks through the insulation during normal operations. There are several features of this system.

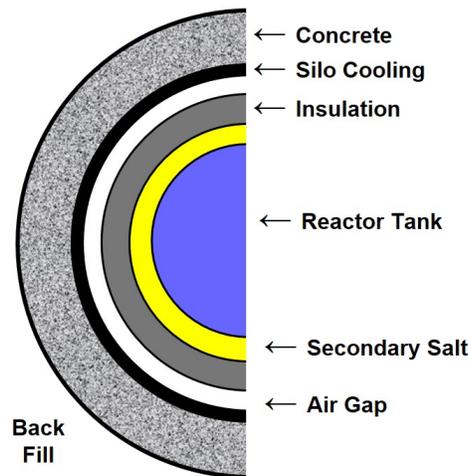


Fig. 11. Silo layout

- *Reactor system.* The primary and secondary systems are efficient in transferring heat to the outer wall of the secondary system by natural circulation of salt and conduction.
- *Insulation failure.* The insulation system is designed to fail if contacted by hot high-temperature liquid salt. One class of high-temperature insulation is mirror insulation made of alternating thin layers of stainless steel and air. The air is the insulator. If this insulation is made with holes in appropriate locations, hot secondary liquid salt upon tank failure will replace the low conductivity air converting the insulation into a conductor. Alternatively, firebrick insulation is traditionally used in glass and steel furnaces. These types of firebrick have small voids that provide the insulation capabilities of these materials. Recent work<sup>37</sup> on using salts for high-temperature heat storage has discovered that in many cases molten salts will quickly enter firebrick, fill the microscopic voids, and dramatically increase the thermal conductivity of the brick.

- *Insulation to silo heat transfer.* Filling the silo between the insulation and silo wall with liquid salt results in convective salt heat transfer with liquid flowing upward near the vessel and downward at the silo wall. This results in more uniform temperatures along the silo wall.
- *Silo cooling system.* The silo cooling system typically consists of air or water channels in the steel liner of the silo. If operational after a severe accident, it will create a freeze wall of frozen salt and stop the accident.
- *Heat conduction to ground.* If the silo-wall cooling system fails, heat is conducted to ground. Special concretes can be used to minimize high-temperature degradation and gas generation such as granite or other high-temperature aggregate and alumina cement. Alternatively, the inner wall can be made of steel rings with the cooling coils on the outside of the rings. Steel rings are used to line mine shafts. In terms of an FHR BDBA system, steel rings have the advantage of providing a high thermal-conductivity structure to spread heat over the total internal area of the silo including above the salt level with more uniform temperatures to assist heat transfer to ground in an accident.

For the first three heat transfer steps, the temperatures are high enough that there is an additional radiative heat transfer through the salt where the salt is a participating medium and heat transfer goes up as the fourth power of the absolute temperature<sup>35</sup>.

These BDBA options only exist because of the combination of a fuel that fails at very high temperatures (above 1650°C) and salt coolants with very high boiling points that approach the melting point of iron. These characteristics enable BDBA decay heat removal by brute-force high-temperature heat conduction—with over a 1200°C temperature drop through many materials with high conductivity from the fuel to the environment before fuel failure.

There are several other characteristics of FHRs that separate them from other reactor types in terms of BDBAs:

- *Fuel failure.* Fuel failure results in actinides and most fission products dissolving in the salt. One can conceptually view fuel failure as turning an FHR into a MSR with fuel dissolved in the same coolant.
- *Vessel insulation.* Almost all the initial temperature drop from reactor core to environment is across the vessel insulation. The operating temperatures are at temperatures where radiative heat transfer becomes important. This may create options for insulation designs where heat losses through the insulation increase rapidly with temperature in an accident scenario. That, in turn, creates the potential for the silo cooling system to become an accident stopping point before vessel failure by limiting vessel temperature increases in BDBAs below vessel failure temperatures.

There are many variants of these systems including having a third frozen salt above the reactor vessel level, such as the sodium potassium magnesium chloride salt described earlier, that melts as temperatures go up and floods the silo with liquid salt to higher levels for BDBA decay heat removal.

## V. SPENT NUCLEAR FUEL (SNF) AND GRAPHITE WASTE MANAGEMENT

*The proposed base-line FHR design uses a once-through fuel cycle.* The AGR and proposed AGR-FHR layout separate graphite into two categories: graphite that is part of the fuel assembly and graphite moderator that remains with the reactor core. This minimizes the volume of SNF and graphite wastes

relative to other FHR designs and HTGRs. The SNF has some graphite with it. Most of the graphite moderator remains in the core until radiation damage requires its replacement.

Graphite-matrix coated-particle SNF has superior performance as a waste form relative to other types of SNF<sup>38</sup> because of the chemical characteristics of graphite and SiC. These fuels are also superior in terms of safeguards relative to LWR SNF.

Depending upon the neutron fluence (power density), the graphite moderator may require replacement once or twice during the reactor lifetime due to radiation damage. Methods to accomplish this were developed for the MSR where the fuel is dissolved in the coolant. The MSR used the same Flibe coolant salts.

## VI. CONCLUSIONS

The development of a new reactor concept is a large undertaking and only a small fraction of the FHR design space has been explored thus far. For the FHR there now is one complete reactor/plant design—for a pebble-bed FHR<sup>2</sup>. There has also been significant work on FHR designs using a carbon-composite plate fuel<sup>39</sup>. This work is the first examination of a third set of design options where this paper defines the major design features and the basis for their selection relative to other options. A three year effort has begun to understand this option. The circumstances are similar to the state of LWR design in the early 1960s—only a partial understanding of design options has been achieved.

The proposed AGR-FHR uses AGR technology and design features including stringer fuel with sub-assemblies to reduce FHR operational challenges (refueling, maintenance, etc.) and associated development costs. The core design minimizes SNF volumes and secondary graphite moderator waste generation relative to other FHR and HTGR designs. The SNF contains the graphite that rapidly sees significant fast-neutron radiation damage and thus must be replaced, increasing lifetimes of graphite moderator that remains in the reactor. The reactor uses a once-through fuel cycle. The double tank system leads to a simplified approach to avoid salt freezing and high-temperatures beyond design limits. The passive decay-heat cooling system using heat pipes removes decay heat upon reactor shutdown to avoid excessive temperatures and shuts down at lower temperatures to avoid freezing of the salt coolant as the decay heat decreases with time. The BDBA system is based on the high-temperature fuel and coolant that provides very large temperature driving forces to move decay heat to the environment and thus limit temperatures below fuel failure temperatures.

The FHR as a reactor concept is less than 20 years old.<sup>40</sup> There has been remarkable progress in the development of FHRs within the last decade. This includes (1) the formation of the startup company Kairos Power<sup>41</sup> with over 70 employees and 5000 m<sup>2</sup> of laboratory space to develop the pebble-bed FHR, (2) expanding programs at ORNL and other national laboratories, (3) many university programs and (4) development work at the Shanghai Institute of Applied Physics of the Chinese Academy of Sciences in China. Like water-cooled reactor technology in the 1950s, there are many design options but not the experience to define the preferred option or options. Design studies, such as this effort, are exploring that option space that will help define the features of future commercial machines.

## ACKNOWLEDGEMENTS

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