Fluoride-Salt-Cooled High-Temperature Reactor (FHR) for Power and Process Heat

Final Project Report

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EXECUTIVE SUMMARY

Fluoride-salt-Cooled High-Temperature Reactor (FHR) for Power and Process Heat

In 2011 the U.S. Department of Energy through its Nuclear Energy University Program (NEUP) awarded a 3-year integrated research project (IRP) to the Massachusetts Institute of Technology (MIT) and its partners at the University of California at Berkeley (UCB) and the University of Wisconsin at Madison (UW). The IRP included Westinghouse Electric Company and an advisory panel chaired by Regis Matzie that provided advice as the project progressed. The first sentence of the proposal stated the goals: The objective of this Integrated Research Project (IRP) is to develop a path forward to a commercially viable salt-cooled solid-fuel high-temperature reactor with superior economic, safety, waste, nonproliferation, and physical security characteristics compared to light-water reactors. This report summarizes major results of this research.

Successful development of an advanced reactor in the United States requires three strategic elements: (1) a compelling commercial and national case to begin and sustain the development effort, (2) a reasonable case that the technology can be developed and (3) a strategy for commercialization. We defined three FHR goals that form the basis of the compelling commercial (vendor / utility) and national (government) case for deployment.

- **Economics.** Increase plant revenue by 50 to 100% relative to base-load nuclear power plants with plant costs similar to light-water reactors (LWRs)
- **Environment.** Provide the enabling technology for a zero-carbon nuclear renewable electricity grid
- **Safety.** Assure that fuel failure with large-scale radionuclide releases will not occur under extreme conditions—including beyond design basis accidents (BDBAs).

To achieve the first two goals we propose the FHR use a Nuclear Air-Brayton Combined Cycle (NACC) with Firebrick Resistance-Heated Energy Storage (FIRES). The FHR as a concept is about a decade old. It combines graphite-matrix coated-particle fuel developed for high-temperature gas-cooled reactors (HTGRs) with liquid salt coolants developed for molten salt reactors (MSRs) where the fuel is dissolved in the coolant. MSRs were initially developed for the Aircraft Nuclear Propulsion Program in the 1950s and 1960s with the goal to power a jet bomber with unlimited range. The project was cancelled with the development of Intercontinental Ballistic Missiles (ICBMs). Research was then started on development of the MSR as a commercial power plant using a steam (Rankine) cycle but the reactor was never commercialized. While the salt coolant was developed to transfer heat from a reactor to an air-Brayton power cycle (jet engine), the air-Brayton technology was not sufficiently developed at this time to be useable in a commercial utility environment.

The last 50 years has seen extraordinary developments in gas turbines including development of efficient natural gas combined-cycle plants for utilities. These developments now enable coupling an FHR to a gas turbine. Developments in HTGR fuel now provide a reliable fuel for the FHR. The FHR with NACC and FIRES could not have existed 15 years ago—it’s a new concept made possible by advances in gas turbines and HTGR fuel that in turn make it possible to meet the first two goals defined above.

The power cycle is similar to a natural-gas-fired plant with a base-load nuclear power output of 100 MWe and a plant efficiency of 42%. NACC enables the use of auxiliary natural gas or stored heat to further raise compressed air temperatures after nuclear heat for the production of 142 MWe of additional peak electricity beyond base-load. FIRES enables buying of electricity when the price of electricity is below that of natural gas to
store as heat and use that heat as a replacement for natural gas for peak electricity production. The capabilities of a modular FHR plant treated as a black box are shown in Fig. ES.1. The baseline design has 12 units per station. FHRs with larger thermal power could be developed with multiple NACC power systems per reactor.

![Diagram showing FHR with NACC and FIRES](image)

**Fig. ES.1. Capability of Modular FHR with NACC and FIRES with Base-load FRR Operation**

Using 2012 hourly wholesale electricity prices in the California and Texas markets, the ability to produce peak power increases net revenue by ~50% relative to a base-load nuclear power plant after subtracting the cost of the auxiliary heat. This assumes low-cost natural gas ($3.52/mBTU). If the price of natural gas triples, the net revenue is about double a base-load nuclear power plant after subtracting the cost of auxiliary natural gas. The incremental natural gas or stored heat to electricity efficiency is 66.4% versus 60% for the best stand-alone combined-cycle natural gas plants. If the price of electricity is below that of natural gas, FIRES enables the FHR to buy up to 242 MWe of electricity that goes into heat storage for production of peak power at a later date. The round-trip electricity-to-heat-to-electricity efficiency is 66%. In all cases the reactor operates at full load. The FIRES storage capability or the use of a zero-carbon fuel (hydrogen or biofuels) enables a zero-carbon electricity grid.

The water consumption per MWe is about 40% of a light-water reactor—using conventional evaporative cooling. The efficiency penalties for using hybrid or dry cooling are much lower as well. This is because combined cycle power plants, nuclear or natural gas, reject much of their heat as hot air. Similar to a natural-gas plant, NACC combines a gas turbine with a heat recovery steam generator (HRSG) that allows selling steam from the HRSG to industrial customers rather than production of added electricity.

The high coolant boiling and fuel failure temperatures of the FHR contribute to its robustness and safety in BDBAs, contributing to the third goal above. The fuel failure temperature is above 1650°C. The coolant boiling point is above 1200°C.

The FHR component technologies are based on existing technologies. However, no FHR has been built and thus significant challenges remain to develop the technology into a reliable commercial power plant. To meet the goals of the IRP project, there were three integrating activities with supporting experimental work.
• **Commercial basis.** The commercial basis\(^1\) for an FHR with NACC and FIRES was developed by MIT and led to the three goals defined earlier. The earliest commercialization date for an FHR is about 2030; thus, the emphasis was on a commercial design that could meet the needs for the 2030 electricity grid. That includes likely limitations on greenhouse gas releases and the need for variable low-carbon electricity. The FHR provides variable electricity to the grid while the reactor operates at base-load.

• **FHR point design.** A detailed point design\(^2\) of a modular FHR was developed by UCB to provide a credible design and define the technical challenges. A modular FHR was chosen for two reasons. First, the output couples to a modified GE\(^®\) F7B gas turbine—the largest utility rail-transportable gas turbine made by GE\(^®\). The design is based on existing gas turbine technology. Second, the development of the FHR will start with a test reactor followed by a pre-commercial machine. This is the first size that could be commercialized. Depending upon the economics, larger FHRs could ultimately be built. A schematic of the plant is shown in Fig. ES.2.

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\(^{2}\) C. Andreades et al. *Technical Description of the “Mark 1” Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant*, UCBTH-14-002, Department of Nuclear Engineering, University of California, Berkeley, September 30, 2014
• **Test Reactor.** No FHR has been built and thus a test reactor is required to develop and test the technology. A preconceptual design of a Fluoride-salt-cooled High-Temperature Test Reactor (FHTR) and a test reactor strategy was developed\(^3\). The FHTR would be a Class I test reactor where the goal is to develop and demonstrate the concept—the reactor is the test. It would be similar to the Experimental Breeder Reactor I (the first sodium-cooled fast reactor) and DRAGON (the first HTGR). The last first-of-a-kind Class I test reactor that was built anywhere in the world was the first lead-cooled fast reactor in Russia in the 1980s. This is in contrast to a Class II test reactor where the reactor is a neutron source to test materials and fuels. Class II test reactors use proven technology since the reactor is a tool. Worldwide, several Class II test reactors are under construction. It is recommended that the U.S. build an FHTR as an international project to reduce the financial risks for each partner, demonstrate the technology, and enable exploration of alternative design options for a commercial machine. There are large incentives to develop a cooperative program with the Chinese Academy of Sciences that has a FHR development program and plans to build a small FHTR by 2020. There is a history of successful international funding of test reactors starting with DRAGON (the first high-temperature gas-cooled reactor) and continuing today to the building of the Jules Horowitz Reactor in France. The building of first-of-a-kind test reactors has been a government responsibility.

The IRP included three major experimental programs to begin to lay the technical foundation for development of an FHR. These experimental programs resulted in the creation of major experimental facilities to support the further development of the FHR. This experimental work helps provide confidence that an FHR can be developed as a practical machine and helps define future required development activities.

• **Thermal hydraulic tests (UCB).** The University of California at Berkeley has built large-scale thermal-hydraulic test loops that use an organic simulants and conducted experiments to provide required experimental data for reactor design.

• **Materials Testing (UW).** The University of Wisconsin has built and now operates systems to purify the fluoride salts required for the FHR. These salts have been used in 700°C corrosion tests to test different materials of construction for the FHR.

• **Materials Irradiations (MIT).** MIT has developed, built, and operated test capsules with prototypical materials in 700°C salt in the MIT reactor under prototypical temperature and irradiation conditions expected in the FHR. Identical tests were conducted outside the reactor at UW to understand and separate out the effects of salt corrosion and irradiation on materials.

Historically the variable demand for electricity has been met by the burning of fossil fuels. Restrictions on releases of greenhouse gases require new methods to generate electricity on demand. The introduction of renewables and deregulation are changing the electricity market. The FHR with NACC and FIRES is designed to meet the needs for the 2030 electricity grid. The option could not have existed 15 years ago. The advances in gas turbines and HTGR fuels have created a potentially transformational nuclear power technology. Significant technical challenges remain but a pathway to the FHR has been developed.

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FHR PROJECT PERSPECTIVE

The Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Integrated Research Project (IRP) is a U.S. Department of Energy funded Nuclear Energy University Program led by the Massachusetts Institute of Technology (MIT) with the University of California at Berkeley (UCB) and the University of Wisconsin at Madison (UW). The objective is development of a path forward for a commercially viable FHR. To meet the objective, the project has used a top-down structure where goals drive the reactor design and the reactor design drives the test reactor goals, strategies and design. These, in turn, drive the technology development activities.

The products of the IRP (in addition to supporting students and over a hundred technical reports, papers, and theses), are three project reports that summarize the results of the first three activities in Fig. PP.1. The three major project reports are:


This report is a summary of all the activities.
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1. INTRODUCTION

In 2011 the U.S. Department of Energy through its Nuclear Energy University Program (NEUP) awarded a 3-year integrated research project (IRP) to the Massachusetts Institute of Technology (MIT) and its partners at the University of California at Berkeley (UCB) and the University of Wisconsin at Madison (UW). The IRP included Westinghouse Electric Company and an advisory panel chaired by Regis Matzie that provided advice as the project progressed. The first sentence of the proposal stated the goals: The objective of this Integrated Research Project (IRP) is to develop a path forward to a commercially viable salt-cooled solid-fuel high-temperature reactor with superior economic, safety, waste, nonproliferation, and physical security characteristics compared to light water reactors. This report summarizes the work to develop a Fluoride-salt-cooled High-Temperature Reactor (FHR).

Figure 1.1 shows the structure of the project. We used a top-down approach starting with the market requirements that lead to a point design. That, in turn, helped drive decisions on the commercial reactor, test reactor, and the experimental program to support the IRP.

![Figure 1.1 Structure of FHR Project](image)

We defined three FHR goals that form the basis of the compelling commercial (vendor / utility market) and national (government) case for deployment (Top Box in Fig. 1.1).

- **Economics.** Increase plant revenue by 50 to 100% relative to base-load nuclear power plants with plant costs similar to light-water reactors (LWRs)
- **Environment.** Provide the enabling technology for a zero-carbon nuclear renewable electricity grid
• **Safety.** Assure that fuel failure with large-scale radionuclide releases will not occur under extreme conditions—including beyond design basis accidents (BDBAs).

To achieve the first two goals we propose the FHR be coupled to a Nuclear Air-Brayton Combined Cycle (NACC) with Firebrick Resistance-Heated Energy Storage (FIRES). The FHR as a concept is about a decade old. It combines graphite-matrix coated-particle fuel developed for high-temperature gas-cooled reactors (HTGRs) with liquid salt coolants developed for molten salt reactors (MSRs) where the fuel is dissolved in the coolant. MSRs were initially developed for the Aircraft Nuclear Propulsion Program in the 1950s and 1960s with the goal to power a jet bomber with unlimited range. The project was cancelled with the development of Intercontinental Ballistic Missiles (ICBMs). Research was then started on development of MSRs for commercial power plants using a steam (Rankine) cycle but the reactor was never commercialized. While the salt coolant was developed to transfer heat from a reactor to an air-Brayton power cycle (jet engine), the air-Brayton technology was not sufficiently developed at that time to be useable in a commercial utility environment.

The last 50 years has seen extraordinary developments in gas turbines including development of efficient reliable natural gas combined cycle plants for utilities. These developments now enable coupling an FHR to a gas turbine. Developments in HTGR fuel now provide a reliable fuel for the FHR. We therefor proposed and chose to develop a FHR coupled to a Nuclear air Brayton combined cycle (NACC) with a Firebrick Resistance Heated Energy Storage System (FIRES). Those decisions, in turn, make it possible to meet the first two goals defined above. The choice of fuel and coolant is the basis for meeting the third goal.

Chapter 2 provides a brief description of the entire system with an emphasis on the NACC power cycle with FIRES that enables meeting the first two goals. Chapter 3 describes how the reactor system meets its stated goals. Chapter 4 describes the reactor. Chapter 5 describes the commercialization strategy with the test reactor while Chapter 6 summarizes the supporting experimental work.

The major products of the IRP (in addition to supporting students) are the three major project reports\(^4\),\(^5\),\(^6\) and over a hundred technical reports and papers that summarize the results of these three activities and the experimental work in Fig. 1.1.

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2. REACTOR SYSTEM DESCRIPTION

The FHR is a new reactor concept (Fig. 2.1) that combines (1) a liquid salt coolant, (2) graphite-matrix coated-particle fuel originally developed for High Temperature Gas-cooled Reactors (HTGRs), (3) a NACC power cycle adapted from natural gas combined cycle plants and (4) FIRES. The FHR concept is a little over a decade old and has been enabled by advances in gas turbine technology. The liquid salt coolant was originally developed for use in molten salt reactors (MSRs) where the fuel is dissolved in the salt. The original MSR program was part of the Aircraft Nuclear Propulsion Program of the 1950s to develop a jet-powered nuclear bomber. Consequently, the fluoride salt coolant was developed to transfer high-temperature heat from a nuclear reactor to a gas turbine. Advances in utility gas turbines over 50 years have now reached the point where it is practical to couple a salt-cooled reactor to a commercial stationary combined-cycle gas turbine. It is that combination that enables the FHR to potentially have the transformational capabilities as described below.

![FHR Features](image)

**Fig. 2.1.** FHR Features. From top to bottom: fuel, coolant, gas turbine, reactor vessel and plant layout (bottom right) where the reactor vessel is black and the salt-to-air heat exchangers are green
To provide a basis to understand what is required to commercialize an FHR, a point design for a commercial FHR was developed with a base-load output of 100 MWe. The power output was chosen to match the capabilities of the GE® F7B gas turbine—the largest rail transportable gas turbine made by General Electric. FHRs with higher output could be built by coupling multiple gas turbines to a single reactor or using larger gas turbines. The development of an FHR will require construction of a test reactor—this size commercial machine would be a logical next step after a test reactor. This point design describes the smallest practical FHR for stationary utility power generation. The market would ultimately determine the preferred reactor size or sizes. There are many FHR design variants under study including alternative geometries for the coated particle fuel, fluoride salt coolants, and plant designs.

The FHR power cycle is shown in Fig. 2.2. In the power cycle external air is filtered, compressed, heated by hot salt from the FHR while going through a coiled-tube air heat exchanger (CTAH), sent through a turbine producing electricity, reheated in a second CTAH to the same gas temperature, and sent through a second turbine producing added electricity. Warm low-pressure air flow from the gas turbine system exhaust drives a Heat Recovery Steam Generator (HRSG), which provides steam to either an industrial steam distribution system for process heat sales or a Rankine cycle for additional electricity production. The air from the HRSG is exhausted up the stack to the atmosphere. Added electricity can be produced by injecting fuel (natural gas, hydrogen, etc.) or adding stored heat after nuclear heating by the second CTAH. This boosts temperatures in the compressed gas stream going to the second turbine and to the HRSG.

Fig. 2.2. Nuclear Air-Brayton Combined Cycle (NACC) with FIRES

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7 C. Andreades et al., Technical Description of the “Mark 1” Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant, UCBTH-14-002, University of California at Berkeley, September 20, 2014
The incremental natural gas, hydrogen, or stored heat-to-electricity efficiency is 66.4%--far above the best stand-alone natural gas plants because the added heat is a topping cycle. For comparison, the same GE® F7B combined cycle plant running on natural gas has a rated efficiency of 56.9%. The reason for these high incremental natural gas or stored heat-to-electricity efficiencies is that this high temperature heat is added on top of “low-temperature” 670°C nuclear-heated compressed air (Fig. 2.3). For a modular 100 MWe FHR coupled to a GE F7B modified gas turbine that added natural gas or stored heat produces an additional 142 MWe of peak electricity.

The heat storage system consists of high-temperature firebrick heated to high temperatures with electricity at times of low or negative electric prices. The hot firebrick is an alternative to heating with natural gas. The firebrick, insulation systems, and most other storage system components are similar to high-temperature industrial recuperators. The round-trip storage efficiency from electricity to heat to electricity is ~66%--based on ~100% efficiency in resistance electric conversion of electricity to heat and 66% efficiency in conversion of heat to electricity. That efficiency will be near 70% by 2030 with improving gas turbines.
3. MEETING FHR GOALS

The commercialization of a new reactor requires transformational goals. Otherwise the incentives to develop such a reactor will not be sufficient to obtain the required resources over a multi-decade time frame. Because the commercialization date is ~2030, the goals must be defined in terms of the expected future conditions, not the current environment. The basis for those goals is described herein.

3.1. Superior Economics

The traditional nuclear-reactor economic figure of merit has been levelized cost of base-load electricity (LCOE)—an appropriate metric if comparing two base-load electricity generating technologies. However changes in the market (deregulation, renewables, etc.) have resulted in large variations in the price of electricity with time. This creates large economic incentives to produce variable electricity with higher production at times of higher prices. The FHR with NACC and FIRES produces variable power while the reactor operates steadily at full power. Figure 3.1 shows the plant as a black box and indicates its capabilities, assuming that the base-load electricity production is 100 MWe. The reactor can be built in different sizes. This capability implies that economic analysis must be based on return on investment that accounts for both the production costs and added revenue made possible by variable electricity production.

![Diagram](image)

Fig. 3.1. Inputs and Outputs of a Modular FHR with a Base-load Power Output of 100 MWe

The base-load FHR electricity output is 100 MWe with a thermal-to-electricity efficiency of 42%. An additional 142 MWe of peaking power can be generated by using auxiliary natural gas or stored heat to increase total power to the grid to 242 MWe. If the price of electricity is less than the price of natural gas per unit of heat, up to 242 MWe of electricity can be bought from the grid and used to electrically heat firebrick in the thermal energy storage system (FIRES). If the price of electricity is low, the 100 MWe base-load output will also go into the FIRES energy storage system to produce peak power later at times of high prices. The decision to include FIRES in an FHR facility depends upon whether the
specific electricity market has a significant number of hours with electricity prices below natural gas prices (or other suitable future peaking fuels, possibly including hydrogen) with the incentive to use stored heat to replace the burning of fuels for peak power.

The price of electricity varies with the time of day. We examined deployment of the FHR in the California and Texas electricity markets using the NACC power cycle with natural gas peaking but without FIRES. Based on using 2012 hourly wholesale rates in those states and the corresponding average natural gas price ($3.52/MBTU), the net revenue for base-load and peak electricity was ~50% higher than a base-load-only nuclear plant of equivalent performance. Net revenue is total revenue minus the cost of natural gas used to produce peak power. The incremental natural gas to electricity efficiency is 66% versus 60% for a stand-alone natural gas plant. Because an FHR with NACC is more efficient in converting natural gas to peak power, it is dispatched before any natural gas plant. This also implies that as stand-alone natural gas plants come on-line, they set the market prices for electricity. Because the FHR with NACC is more efficient than stand-alone natural gas plants, the production of peak electricity increases FHR net revenue after subtracting the cost of the natural gas.

Increasing natural gas prices increase electricity prices with two effects: (1) increased revenue for all nuclear plants and (2) relative increases in revenue for the FHR with NACC versus base-load nuclear plants. When the FHR is producing peak power and electricity prices are set by stand-alone natural gas plants, the net revenue from FHR peak electricity production increases with natural gas prices. This is because of the higher efficiency in turning natural gas into electricity than stand-alone natural gas plants. If U.S. natural gas prices were to triple from their historical lows, the FHR revenue from base and peak electricity production would be double a base-load nuclear plant. Natural gas prices in Europe and Asia are about three times those in the United States and thus one would expect a larger economic advantage for the FHR with NACC versus a base-load nuclear plant in those markets.

If industrial markets are available for steam sales from the HRSG, the net plant revenue is about double that of a base-load nuclear plant. This assumes sales of steam at 90% of the cost of natural gas heat to industrial customers at times of low electricity prices with varying electricity and steam sales to maximize revenue. It also assumes that the industrial customer has his own boilers that burn natural gas and turns those boilers down and buys steam when available at prices less than his steam production costs. The revenue gains are larger if there are increases in natural gas prices or any limits on carbon dioxide emissions. In this mode (assuming the FHR does not have FIRES) the gas turbine is sending electricity to the grid but the HRSG is sending steam to industrial customers.

Limited analysis indicates FHR capital costs are similar to LWRs per kWe—implying significantly better economics because of the higher revenue from peak power sales. The economics are helped by intrinsic characteristics of the reactor: low-pressure operation, high-temperature operation with high thermal-to-electricity efficiency, high reactor-vessel power density (slightly less than a boiling water reactor), coupling to a gas turbine power conversion system, and modularization.
3.2. Provide the Enabling Technology for a Zero-Carbon Nuclear-Renewable Grid.

The FHR with NACC and FIRES potentially enables a zero-carbon nuclear-renewable grid. Fig. 3.2 shows the power demand in New England by hour over a year and the capability of a fleet of FHR plants to meet variable electricity demand with the reactors operating at continuous full power. In a zero-carbon world, one would not use natural gas to produce peak electricity. Peaking power would use stored heat or hydrogen as the fuel. The characteristics of this system have major implications in terms of a zero-carbon electricity grid if natural gas is not available.

- **Enabling zero-carbon nuclear renewable grid.** Large scale use of wind or solar imply low prices and excess electricity capacity at times of high wind or solar output. The FHR with NACC and FIRES can store excess electricity as heat when available from renewables that do not have their own built-in storage capacity (like nearly all present renewable facilities other than hydro). Existing storage technologies (hydro pumped storage, batteries, etc.) have a major weakness when coupled to renewables. If there is a multi-day period of no wind or solar, these storage systems are depleted. As a consequence, renewables require backup generating capacity such as gas turbines for reliable electricity. The FHR with NACC and FIRES has that capacity built-in to a single facility that also earns revenue 24/7 from base-load electricity production and possible heat sales, and has the highest efficiency in converting gaseous or liquid fuels to electricity—better than stand-alone natural-gas turbines. This results in major capital cost and operating cost savings relative to other electricity storage systems.

- **Minimizing electricity storage costs.** The two methods to cheaply store energy are heat and hydrogen because the energy storage media are cheap, firebrick for heat storage and underground caverns for hydrogen storage similar to those used for natural gas. There is a difference. In the FHR the round trip electricity-to-heat-to-electricity efficiency is ~66%. The efficiency of electricity-to-hydrogen-to-electricity efficiency in all technologies identified to date is below 50%. Hydrogen is a more expensive method for electricity storage because of the low round-trip efficiency of electricity-to-hydrogen-to-electricity. Hydrogen can be stored seasonally underground like natural gas at low costs whereas FIRES would be expensive for long-term heat storage because the firebrick is inside a pre-stressed concrete pressure vessel that makes it cost prohibitive to have long-term storage of heat. This implies that the optimum FHR system for a zero-carbon grid would store energy in FIRES for daily swings in electricity demand but use hydrogen for longer-term seasonal variations in electricity demand. In practice, maintenance and refueling outages for FHRs would be at times of year with low electricity demand that would reduce the need for seasonal storage using more expensive hydrogen.
Alternative to Hydro Pumped Storage and Batteries. The storage system is built on firebrick with the potential that the total storage system cost will be less than other energy storage systems. The integration of firebrick heat storage with gas turbines is being developed by General Electric for another storage technology—adiabatic compressed air storage (Adele project). Much of this technology development program is directly applicable for NACC with FIRES.

These capabilities may result in the FHR with NACC and FIRES becoming the enabling technology for a zero-carbon electricity grid and for the larger scale use of renewables by addressing the central challenges of renewables—their non-dispatchability and the resultant electricity price and revenue collapse at times of high solar or wind output.

The free market price distribution for electricity has major implications for a low-carbon grid with nuclear and renewables. The addition of a small amount of solar is beneficial because the electricity is added at times of peak demand. However, as additional solar is added, it drives down the price of electricity during the mid-day at times of high solar input. Each owner of a solar power system will sell electricity at whatever price exists above zero. This implies that when somewhere between 10 to 15% of the total yearly electricity demand is met by solar in California, the output from solar systems during midday for parts of the year will exceed demand, the price of electricity will collapse to near or below zero, and the revenue to power plants at those times will collapse to near zero. Each incremental addition of solar will lower the revenue for existing solar and other electricity producers. The same effect occurs as one adds wind capacity but wind input is more random. As wind penetrates the market it drives the price of electricity down on days with high wind conditions and low electricity demand.
Recent studies have estimated this effect in the European market\textsuperscript{8, 9}. If wind grows from providing 0\% to 30\% of all electricity, the average yearly price for wind energy in the market would drop from 73 €/MWe (first wind farm) to 18€/MWe (30\% of all electricity generated). There would be 1000 hours per year when wind could provide the total electricity demand, the price of electricity would be near zero, and 28\% of all wind energy would be sold in the market for prices near zero.

Figure 3.3 shows the price curve for electricity in California; that is, how many hours electricity can be bought each year at different prices. Also shown in red is a notational price curve if one adds significant non-dispatchable renewables. It is the collapse of electricity prices at times of high wind or solar input that limits the use of renewables. The addition of renewables has one other implication. It implies substantially higher electricity prices at times of low solar or wind input. Backup electricity generating plants will not be built that operate a limited number of hours per year unless there is a significant increase in electricity prices during those times. In a free market, the addition of renewables may result in locking in added fossil fuel plants because they are the most economic choice to provide backup power because of their low capital costs. This is happening in Germany. Recent State of California\textsuperscript{10} and Google\textsuperscript{11} studies came to similar conclusions.

![Distribution of Electrical Prices](image)

Fig. 3.3 Distribution of Electrical Prices (bar chart), by Duration, Averaged Over CAISO (California) Hubs (July 2011-June 2012) and Notational Price Curve (Red Line) for Future Low-Carbon Grid.


\textsuperscript{10} California Council on Science and Technology, \textit{California Energy Futures – The View to 2050: Summary Report}, April 2011

The FHR with NACC and FIRES directly addresses this challenge by buying electricity when the price is low and providing electricity at times of high prices. In a nuclear renewable system, the FHR raises the electricity price when prices are depressed by large solar or wind generation by creating added demand and lowers the electricity price when there is little solar or wind generation. The characteristics of the FHR with NACC and FIRES become the enabling technology for the larger-scale use of nuclear.

The combined cycle also enables integration to meet industrial energy demands—a requirement for a zero-carbon world. There is a long history of using LWRs to provide steam to industrial customers in Europe and Asia. However, there have been two historical constraints that add to the cost of steam from nuclear reactors:

- **Radionuclides in steam.** If steam is used from nuclear plants, there is the concern that radionuclides (including tritium) will enter the industrial steam supply. To prevent this, isolation heat exchangers are used between the nuclear steam supply system and the industrial customer—with added capital costs and inefficiencies due to required temperature drops.
- **Reactor shutdown.** Industrial customers require reliable steam supplies. Reactor outages and refueling add major complications in assuring steam supplies.

The FHR with NACC bypasses these challenges. Within NACC hot air transfers heat from the salt-air heat exchangers to the HRSG. The air acts as an isolation system—there is no need for a separate isolation heat exchanger. It implies that the added cost to supply steam to an offsite industrial customer is very low—valves and pipes.

Fig. 3.4. NACC Modified to Assure Steam Supply if Reactor Shutdown (Option to Use Biofuels or Hydrogen if Zero-Carbon World)
The NACC can be designed to provide steam if the reactor is shut down for any reason (Fig. 3.4). Combined-cycle natural gas plants are widely used in the chemical and refinery industry to produce variable quantities of steam and electricity. Many plants use combustible waste gases as fuel with variable feed to the combined cycle plant that produces the required steam with excess electricity sold to the grid. Many chemical plants require ultra-reliable steam supplies for process and safety systems. To assure a reliable steam supply in the event of a gas turbine failure, there are auxiliary natural gas burners that can provide hot air to the HRSG. This technology is directly applicable to an FHR with NACC. In both the power and chemical industry multiple turbines are often connected to a single HRSG. There are large economics of scale associated with HRSGs and such units have higher reliability than gas turbines.

The FHR coupled to NACC has several unique characteristics that may open new markets. NACC has the unique characteristic of generating a stack gas of 97°C hot dry air—unlike a conventional gas turbine where the combustion of natural gas results in moist exhaust stack gas. Low-cost hot dry air (no-cost stack air) enables removal of water from wet biomass that may enable economic recovery of two new large sources of biomass for liquid fuels production: high-water content unconventional sources such as kelp and more efficient use of harvested biomass from the forest products and paper industries. Only limited studies\(^{12}\) have been conducted on markets for very-low-relative-humidity warm air.

3.3. Assure No Major Fuel Failures in Beyond Design Basis Accidents (BDBAs)

The FHR has the traditional safety systems to prevent accidents and thus protect the public and plant investment: (1) active decay-heat cooling systems and (2) Direct Reactor Auxiliary Cooling Systems (DRACS)—a passive decay heat cooling system developed for sodium fast reactors. Several intrinsic characteristics of the FHR improve safety and economics: (1) low pressure coolant, (2) excellent coolant heat transfer properties, (3) a high-temperature fuel, (4) high retention of fission products in the coolant, and (5) high heat capacity in the reactor core.

In addition, the FHR combination of fuel and coolant characteristics has the potential to prevent major fuel failures with large FHRs (thermal outputs significantly greater than 1000 MWt) in beyond design basis accidents (BDBA). The BDBA events could include reactor vessel, containment, and other such failures. The larger the thermal output of the reactor the more difficult it is to prevent fuel failure in a severe accident. As a consequence, LWRs use reactor containments to contain radioactivity if there is an accident with large-scale fuel failures. The largest reactor today that can be built without large-scale fuel failure in a severe accident is a high-temperature gas-cooled reactor (HTGR) with an output of ~600 MWt. Because the FHR uses the same fuel as the HTGR, an FHR of similar output could be built with

this characteristic. A series of studies\textsuperscript{13}, including modeling of severe accidents, was undertaken to develop a pre-conceptual design of an FHR BDBA system for larger FHRs with these capabilities.

When a reactor shuts down, it continues to generate decay heat at a decreasing rate. If this decay heat is not removed, fuel temperatures ultimately increase until the fuel fails and radionuclides are released. It follows that fuel failure can be prevented by finding a way to remove decay heat to keep fuel temperatures below failure temperatures in an accident. The potential to avoid major fuel failures under extreme accident conditions in large FHRs is a consequence of the unique combination of the high-temperature properties of the fuel and coolant. The FHR uses HTGR graphite-matrix coated-particle fuel with failure temperatures of $>1650^\circ$C. The coolants are clean fluoride salts that have melting points above 350°C and boiling points above 1400°C. These high temperature limits relative to other nuclear fuels and coolants may enable systems to be designed to prevent major fuel failures in large FHRs in severe accidents. There are four features of this system.

- **Core heat capacity.** The reactor core has a large heat capacity and there is a 700°C margin between the nominal peak coolant operating temperature and its boiling point. The combination provides the ability to absorb large quantities of decay heat and thus provide time for the decay heat rate to decrease and reducing BDBA decay heat removal rate system requirements.

- **Temperature driving force for decay heat removal.** The rate of decay heat transfer from the fuel to the environment (atmosphere) in an accident is proportional to the temperature difference. There is a 1400°C temperature drop between the coolant boiling point and the environment and 1700°C difference between fuel failure and the environment. The temperature driving forces for decay heat removal before fuel failure or coolant boiling are larger in an FHR than in any other reactor.

- **Removal of heat transfer barriers.** Normally the FHR is highly insulated to prevent heat loses. If decay heat in a BDBA is to be removed, these barriers to heat transfer must be eliminated. There is a 700°C temperature difference between normal FHR operating temperatures and the boiling point of the salt that would remove the coolant salt from the reactor core and allow higher reactor accident temperatures. In an accident the salt coolant temperature will rise. The large temperature rise in an accident before fuel failure and coolant boiling can be used to degrade the insulation system—reducing the resistance for heat transfer from the fuel to the environment. High-temperatures initiate insulation failure that in turn rapidly increase heat loses from the fuel to the environment. Unlike other reactor coolants, there is sufficient temperature margin that this can be done before coolant boiling.

- **Ultimate silo cooling system.** The reactor vessel is in a silo that is designed to efficiently transfer heat in an accident to the environment after insulation failure. The silo contains a low-cost BDBA salt. In an accident the silo temperatures increase, causing this salt to melt and partly

\textsuperscript{13}C. W. Forsberg, J. D. Stempien, M. J. Minck, and R. G. Ballinger, “Eliminating Major Radionuclide Releases in Fluoride-salt-cooled High-Temperature Reactor Severe Accidents,” Submitted to Nuclear Technology
flood the silo. The melting of the BDBA salt absorbs decay heat reducing vessel and fuel temperatures. It thermally couples the reactor vessel to the silo wall to reduce the temperature drop between the fuel and silo wall. This provides over 1000°C in temperature drop to drive decay heat from the silo to the environment in a BDBA with no major insulation barriers.

The combination of mechanisms enables decay heat to move sufficiently fast from fuel to the environment in an accident to prevent exceeding temperatures at which major fuel failures occur. *The BDBA safety system is not dependent upon mechanical system design feature except the physical properties of the fuel, coolant, and materials near the reactor core.* Only limited work has been done on this radically different BDBA control system. Significant research will be required to develop and confirm this unique capability to assure that severe accidents will not result in large-scale fuel failures.

Separate from the above mechanisms, if fuel damage were to occur, in fluoride salts most significant radionuclides are soluble as fluorides. This includes cesium and strontium. It has been shown the iodine largely remains in the salt as I⁻ ion or as an iodide compound such as CsI. Noble gases such as Xe and Kr are not soluble in the salt.

3.4. Superior SNF Waste Forms and Proliferation Resistance

Fuel cycle characteristics such as spent nuclear fuel (SNF) performance in a geological repository and proliferation resistance against diversion are determined by fuel choices. A study was undertaken to understand the fuel cycle implications of the FHR.¹⁴ The FHR uses graphite-matrix coated-particle fuel, the same basic fuel used in high-temperature gas-cooled reactors (HTGRs). Consequently, the FHR and HTGRs have similar fuel cycle characteristics. HTGR SNF is generally recognized to have superior waste form performance in a repository and superior non-proliferation characteristics relative to other types of power reactors. These characteristics also apply to FHR SNF.

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4. REACTOR DESIGN

There are many proposed pre-conceptual FHR designs that could couple to a NACC with FIRES. The project developed a specific design that included the entire facility including materials requirements to provide an understanding of what an FHR is. The fuel and coolant define most FHR characteristics. The FHR fuel consists of tri-structural isotropic (TRISO) coated-particle fuel embedded in 3-cm diameter graphite pebbles—the same basic fuel used in previous and planned HTGRs. The primary coolant is a lithium-beryllium-fluoride salt known as flibe (Li$_2$BeF$_4$). There are alternative fuel and coolant options. The primary coolant system is a closed loop that operates at atmospheric pressure with nominal core coolant inlet and outlet temperatures of 600 °C and 700 °C respectively. Figure 4.1 shows a schematic of the reactor system, Table 4.1 defines the major design parameters, Figure 4.2 shows a schematic of the plant and Figure 4.3 shows a plant site designed for 12 units. The peak fuel temperature is ~800° with a fuel failure temperature above 1650°C. The nominal peak coolant temperature is 700°C, and the coolant boils above 1400°C.

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15 C. Andreades et. al., 2014, Technical Description of the “Mark I” Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant, UCBTH-14-002, Department of Nuclear Engineering, University of California at Berkeley, Berkeley, CA.
17 D. E. HOLCOMB et. al., Core and Refueling Design Studies for the Advanced High-Temperature Reactor, ORNL/TM-2011/365, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 2011.
The reactor core designs are similar to many high-temperature gas-cooled reactors (HTGRs) with two major differences—both consequences of the liquid salt coolant. First, the power densities are much higher because liquids are better coolants than gases. Second, it is a low pressure system. The fluoride salt coolants have boiling points above 1200°C.

Table 4.1. Mark I Pebble-bed FHR Design Parameters

<table>
<thead>
<tr>
<th>Reactor Design</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Thermal power</td>
<td>236 MWt</td>
</tr>
<tr>
<td>Core inlet temperature</td>
<td>600°C</td>
</tr>
<tr>
<td>Core bulk-average outlet temperature</td>
<td>700°C</td>
</tr>
<tr>
<td>Primary coolant mass flow rate (100%power)</td>
<td>976 kg/sec</td>
</tr>
<tr>
<td>Primary coolant volumetric flow rate (100% power)</td>
<td>0.54 m³/sec</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Power Conversion</th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Gas turbine model number</td>
<td>GE 7FB</td>
</tr>
<tr>
<td>Nominal ambient temperature</td>
<td>15°C</td>
</tr>
<tr>
<td>Elevation</td>
<td>Sea level</td>
</tr>
<tr>
<td>Compression ratio</td>
<td>18.52</td>
</tr>
<tr>
<td>Compressor outlet pressure</td>
<td>18.58 bar</td>
</tr>
<tr>
<td>Compressor outlet temperature</td>
<td>418.7°C</td>
</tr>
<tr>
<td>Compressor outlet mass flow</td>
<td>418.5 kg/sec</td>
</tr>
<tr>
<td>(total flow is 440.4 kg/s; conventional GE-7FB design uses excess for turbine blade cooling)</td>
<td></td>
</tr>
<tr>
<td>Coiled tube air heater outlet temperature</td>
<td>670°C</td>
</tr>
<tr>
<td>Base load net electrical power output</td>
<td>100 MWe</td>
</tr>
<tr>
<td>Base load thermal efficiency</td>
<td>42.5 %</td>
</tr>
<tr>
<td>Co-firing turbine inlet temperature</td>
<td>1065°C</td>
</tr>
<tr>
<td>Co-firing net electrical power output</td>
<td>241.8 MWe</td>
</tr>
<tr>
<td>Co-firing efficiency (gas-to-peak-power)†</td>
<td>66.4 %</td>
</tr>
</tbody>
</table>
Fig. 4.2. FHR with NACC
To develop an initial estimate of relative costs compared to other reactors, the material inputs for the Mk1 PB-FHR were compared to other types of power plants. Table 4.2 presents comparisons with a wide range of values found in the literature for other reactors and power systems. It is important to emphasize that the Mk1 values are approximate. This said, it is noteworthy that the quantities of carbon steel and concrete needed to construct a 1200-MWe base load, 12-unit Mk1 station are comparable to the quantities needed to construct large LWRs with similar base-load power output. This comparison is based on base-load power output—not on peaking capabilities but the FHR with NACC has peaking capabilities—for every 1000 MWe there is the capability of an added 1420 MWe of peak power. In particular, the somewhat larger quantities of steel are a consequence of NACC and its peaking capability.
Table 4.2. Comparison of estimated Mk1 material inputs to other power plant types per MWe using baseload FHR output (100 MWe/unit) with no credit for peaking capability (142 MWe/unit).

<table>
<thead>
<tr>
<th></th>
<th>Carbon Steel (1000 kg/MWe)</th>
<th>High Alloy and Stainless Steel (1000 kg/MWe)</th>
<th>Concrete (1000 kg/MWe)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Mk1 PB-FHR (100 MWe)</td>
<td>69.9</td>
<td>11.2</td>
<td>425.1</td>
</tr>
<tr>
<td>ORNL 1970’s PWR (1000 MWe) *</td>
<td>36.1</td>
<td>2.1</td>
<td>179.5</td>
</tr>
<tr>
<td>CRS nuclear plant range **</td>
<td>26 to 72</td>
<td>§</td>
<td>198 to 685</td>
</tr>
<tr>
<td>ABWR (1380 MWe) †</td>
<td>46.0</td>
<td>§</td>
<td>332.7</td>
</tr>
<tr>
<td>GT-MHR †</td>
<td>26.9</td>
<td>§</td>
<td>183.1</td>
</tr>
<tr>
<td>NGCC plant (620 MWe) ††</td>
<td>0.20</td>
<td>2.2</td>
<td>47.8</td>
</tr>
<tr>
<td>CRS NGCC plant range **</td>
<td>34 to 56</td>
<td>§</td>
<td>53 to 108</td>
</tr>
<tr>
<td>Pulverized coal steam plant (1000 MWe) †††</td>
<td>62.2</td>
<td>§</td>
<td>178.3</td>
</tr>
<tr>
<td>CRS coal plant range **</td>
<td>24 to 56</td>
<td>§</td>
<td>175 to 354</td>
</tr>
</tbody>
</table>

** S. Kaplan, Memorandum to Senator John McCain, “Concrete and Steel Requirements for Power Plants,” Congressional Research Service, Table 1, Nov. 27, 2007. Carbon steel values cited in this study may be for a conventional natural gas steam plant, rather than a combined cycle plant.
§ Information not available
5.0. COMMERCIALIZATION STRATEGY AND TEST REACTOR

The development of a new reactor is a multi-decade effort. Private industry will fund large projects with timeframes of approximately a decade—such as large mines and offshore oil fields. Private industry does not fund multi-decade development programs because (1) the financial risks are too great because of the time value of money and (2) the greater uncertainties in predicting future events. As a consequence, government funding is required. This divides the development into two phases—one primarily with government funding and one primarily with private funding. There is a large overlap between these two phases. Because both public and private funding is required, the compelling case for any advanced reactor must be compelling in meeting national goals as well as commercial goals.

Phase I activities are primarily funded by the government with the government taking the major financial and other risks. Many of the activities will be conducted by private industry. For example, the development of the LWR was led by the U.S. Navy, but much of the work was done by private contractors. Phase I must include the involvement of potential vendors and customers (utilities and other users) to provide the essential feedback to define commercial goals and help focus the program so the technology development activities lead ultimately to a competitive commercial product.

Phase I activities have another characteristic—the issues and challenges of intellectual property rights are much less significant because of the longer time horizon. This characteristic enables the use of international consortiums involving multiple countries to spread the risk and costs of developing a new technology. A new reactor technology will require one or more test reactors to develop and demonstrate the technology. Phase I activities go through the building and operation of those test reactors and other support facilities.

Phase II activities are primarily funded by private industry and begin with pre-commercial demonstration plants. At this phase there will be a mixture of government and private sharing of risks and funding. For example, the federal government would likely fund the costs of licensing to reduce those risks and may build specialized test facilities for industrial use. As the technology moves closer to commercial deployment, commercial considerations such as patents, trade secrets, and other types of intellectual property become more important.

This division of roles is not new. The U.S. government has had an historic role to reduce long-term financial risks with new technologies—starting with the development of canals and railroads in the 1800s. The government is the regulator with the understanding that regulatory activities also create barriers for new technologies that require government assistance to lower licensing risks. In the area of nuclear power successful examples in the last several decades include (1) the extended fuel burnup program and (2) the NP2010 program that led to commercialization of the Westinghouse AP-1000 reactors that are now being built in China and the U.S. Last, there are candidate government missions for the FHR such as reactors for remote sites and ships. If these are sufficiently compelling, the government will develop the technology for government missions.
Phase I

Phase I includes developing and demonstrating the technology—including developing the licensing basis for the FHR. The single largest activity is the design, construction, and operation of one or more Fluoride-salt-cooled High-temperature Test Reactors (FHTRs). An FHTR is used to develop and demonstrate the technology. A separate report\(^{18}\) describes the proposed FHTR, including design, licensing, financing, organization, and auxiliary test facilities.

The FHTR may test alternative fuel, coolant, and supporting system options to determine which design options have the best performance. In the early development of LWRs the United States, the government built the Shippingport Pressurized Light-Water Reactor. Over its lifetime this single reactor tested three very different types of reactor core designs that helped provide the basis to choose optimum designs for commercial LWRs. The FHTR is proposed to have similar capabilities.

There are incentives to create an international cooperative program to develop the FHR base technology including the design, construction, and operation of an FHTR to reduce financial risks for each partner, reduce technical risks and accelerate development. There is a history of successful international test reactor projects. The first high-temperature gas-cooled test reactor was DRAGON—an international project led by the United Kingdom that included the United States. There are several test reactors today that are funded by international consortiums (e.g. Halden, Jules Horowitz, etc.). There are two requirements for success: (1) the lead country where the reactor will be built must have a strong commitment and (2) a strong project management team with complete control of the project. If the U.S. led such an international test reactor project, there are several financing options: (1) other countries provide funding, equipment, or services for their contribution or (2) the U.S. pays for the FHTR in return for access to different foreign test reactors with different capabilities to meet other U.S. needs.

In the United States the regulatory structure for test reactors is different than for commercial reactors. The owner of the FHTR would likely be the U.S. Department of Energy (DOE) that can license the reactor or request that the reactor be licensed by the U.S. Nuclear Regulatory Commission. DOE has recently built and licensed other one-of-a-kind nuclear facilities that it owns. The advantages and disadvantages of the alternative licensing strategies for the FHTR are summarized in the test reactor report.

The FHR concept was originated in the United States. Since then the Chinese Academy of Sciences (CAS) has launched a major program with the goal to build a first small FHTR by 2020. That project has grown to hundreds of research personnel, a site has been selected and industrial partners have joined. Research has also started in Europe, Japan, Australia and other locations. Simultaneously, there is

growing interest in the U.S. by universities, vendors, and the Electric Power Research Institute. The foundation for developing a U.S. lead high-performance FHTR is forming.

**Phase II**

Phase II is the transition beyond the test reactor (which tests and demonstrates the technology) to a pre-commercial demonstration plant and then commercial deployment. Because the goal is development of a commercial reactor, it is essential that the vendors, suppliers, and technical experts that will lead Phase II participate in Phase I activities: partly, to focus the goals and drive the system to a viable commercial product, and partly to carry forward essential insights and technical understandings during Phase II commercialization efforts. Commercialization risks can be further reduced by creating opportunities during Phase I for participants to begin developing the infrastructure they will need during Phase II. The design, construction, and operation of the FHTR may be particularly valuable in creating such opportunities.

The United States has not attempted development and commercialization of a new reactor type in over 40 years. As a consequence, the institutional structures are not structured for efficient commercialization of new commercial power reactor technologies. The system has evolved to adopt incremental improvements. A major component of the Phase II activities that must be initiated during Phase I activities is development and implementation of a commercialization strategy that considers financing, business practices and regulation in the context of a new reactor type with innovative features.

One example of the challenges is the licensing the FHR. An understanding of the regulatory-financial challenges can be obtained by comparing how new drugs are licensed versus nuclear reactors. Like the nuclear industry, the development of new drugs is a long and expensive process that involves the investment of billions of dollars. Because new drugs are brought to market each year, the pharmaceutical industry and regulatory structures have evolved to reduce financial risks in development of new drugs while assuring public safety. In the licensing of new drugs, the Federal Drug Administration uses a staged licensing approach. As the drug is developed and tested, the FDA makes decisions. The incremental process allows changes to be made early in the development cycle rather than late in the development cycle where changes are expensive in time and money. The staged regulatory process reduces financial and schedule risks as a drug is developed. The regulatory and schedule risk decreases as progressively larger investments are required and thus reducing the difficulty in financing new drug development.

The Nuclear Regulatory Commission (NRC) has a single-step process where large investments must be made before there are any licensing decisions. The NRC licensing approach evolved over time for light-water reactors (LWRs) where there is a large experience base with a particular reactor type and that once a reactor design is licensed, multiple identical reactors will be built based on that license with no significant changes in design. That is a reasonable model for an existing technology. However, that was
not originally the case with nuclear reactors. The early experience with light-water reactors (LWRs), high-temperature reactors (HTGRs), and sodium fast reactors (SFRs) was rapid evolution of reactor design in the first few years as feedback was obtained from construction and operations. This is what is seen in almost all new technologies as they are first developed. The same fast evolution may be expected occur with FHRs. Because of these different circumstances, we recommend that a comprehensive assessment of the licensing options to be carried out in the near future for the FHR that includes consideration of the implications for financing the FHR. This would include consideration of a staged licensing process similar to that used for the pharmaceutical industry to lower the financial risks associated with developing new reactors by reducing regulatory risks with staged licensing. The licensing risks would decrease as the expenditures increase.

_U.S. Competitive Advantage_

Existing industrial companies have many of the technologies required for the FHR. B&W produces graphite-matrix coated-particle fuel. Westinghouse, Areva, and General Atomics have long experience in designing HTGRs and in their fuels. General Electric is a leader in combined cycle gas turbines and a partner in the IDELE project to integrate firebrick heat storage into a gas turbine for an adiabatic compressed air storage system.

The United States has a competitive advantage in terms of developing and commercializing the FHR with NACC and FIRES. The FHR was invented in the United States. The U.S. leads in development of graphite-matrix coated-particle fuel. There is a domestic market—particularly in the southwest United States with significant solar resources and the Great Plains with significant wind resources where there are large variations in electricity prices that favor the capabilities of an FHR with NACC and FIRES. Recently proposed Environmental Protection Agency rules to limit greenhouse gas emissions will create large incentives for such a technology across the entire country. Last, and most important, the U.S. is the commercial leader in combined-cycle gas turbines, high-temperature materials, and other required technologies for an FHR, including reactor vendors with experience in high temperature reactors.

From a national perspective, the U.S. historical competitive advantage has been in innovative technologies. At one time the U.S. led in worldwide LWR sales. Today it faces many competitors and is losing market share as international competition increases. With the loss of market share there is decreasing influence in nuclear developments worldwide. To maintain economic competitiveness and continued influence in nuclear markets, there are strong incentives for development of advanced reactors such as the FHR that are focused on meeting future utility grid requirements.
6.0. EXPERIMENTAL ACTIVITIES

The IRP conducted in parallel a large experimental program to support the initial development of the FHR and help define the challenges. This chapter briefly describes the major activities and lists some of the major reports. This includes descriptions of the three major experimental capabilities that have been developed to support FHR development and will help FHR development going forward with time. The last major salt-coolant development program was in the 1960s and ended in the mid-1970s. Advances in knowledge imply major changes in the types of work and how the challenges were approached compared to the 1970s. Major advances were made in several areas.

- **Thermal hydraulics (UCB).** The development of LWRs, HTGRs, and SFRs required multiple large very-expensive thermal hydraulic test loops. UCB discovered that Dowtherm® A oil can be used as a simulant fluid for fluoride salts because at relatively low temperatures (50-120°C), Dowtherm® A can match the Prandtl, Reynolds and Grashof numbers of the major liquid salts simultaneously, at approximately 50% geometric scale and heater power under 2% of prototypical conditions. Major experimental facilities using Dowtherm® A were built and are now operating at UCB to provide some of the experimental data required. It implies major savings in cost and time for development of an FHR and should greatly reduce the number of expensive high-temperature salt test loops that are required for thermal hydraulic testing.

- **Measurements (UW and MIT).** Advancing technology allows detailed surface examination of materials corrosion samples and analysis of salts (neutron activation) that now provide an understanding of corrosion mechanisms that was not possible in the 1970s (UW and MIT). This understanding enables better selection of materials of construction and estimating long-term corrosion rates.

- **Modeling (UCB and MIT).** Advances in simulation and modeling now enable experimental measurements to validate models that can then be used for design. The IRP has developed two major sets of models. UCB has developed thermal hydraulic models for the FHR. MIT has developed a corrosion/tritium model that simulates the primary system corrosion and tritium transport.

- **Understanding differences between MSR and FHR (UW and MIT).** Most previous work on the FHR was based on the information developed for MSRs. The experimental work has begun to extend this work to the FHR to take advantage of the characteristics of a clean coolant salt versus the MSR with high concentrations of fission products in the coolant. Initial work at UW suggests that 316 SS may be a viable material of construction for the FHR with its clean coolant rather than more expensive higher-nickel alloys required for the MSR. Second, an understanding of neutron-induced radioactivity in the coolant and implications for design has been developed during MIT irradiations of materials in 700°C salt.

- **What to do, what not to do.** The experimental programs, like all experimental programs, have developed better experimental strategies based on what worked and what did not work. These have been documented to aid future experimental work.
To develop the scientific and technical basis to design and license fluoride-salt-cooled, high-temperature reactors (FHRs), the University of California, Berkeley (UCB) has conducted an experimental test program to validate simulation codes for application to FHRs. These experiments are novel, because they use organic simulants to provide the ability to reliably predict salt cooling behavior in FHRs. The capability to validate separate effect and integral steady-state and transient response models is a key issue for licensing new reactor designs. UCB constructed and operated the pebble scaled high-temperature heat transfer (PS-HT2) experiment to study convective heat transfer in heated tubes and in packed pebble beds. UCB also designed and constructed the first integral effects test (IET) facility to study FHRs, called the Compact Integral Effects Test (CIET 1.0) facility to reproduce the integral transient thermal hydraulic response of FHRs under forced and natural circulation operation. CIET 1.0 provides data to validate simulation codes for direct reactor auxiliary cooling systems (DRACS), used for natural-circulation-driven decay heat removal in FHRs. CIET 1.0, shown in Fig. 6.1, is designed to replicate FHR transient response for a wide range of licensing basis events (LBEs).

The CIET 1.0 facility uses Dowtherm® A oil as a simulant fluid for fluoride salts, because at relatively low temperatures (50-120°C), Dowtherm® can match the Prandtl, Reynolds and Grashof numbers of the major liquid salts simultaneously, at approximately 50% geometric scale and heater power under 2% of prototypical conditions. CIET 1.0 was scaled based on the earlier design of a 900-MWth channel-type pebble bed advanced high-temperature reactor. Subsequently, UCB developed a new, pre-conceptual design of a 236-MWth Mark 1 pebble bed FHR (Mk1 PB-FHR), after the scaling and design of CIET 1.0 were finalized. CIET is scaled so that major components (heater/core, IHX/CTAH, DHX, NDHX/TCHX) are at 50% of the elevation of a prototypical FHR plant, such as the Mk1 PB-FHR, as shown in Fig. 6.2.
Figure 6.1 The UC Berkeley Compact Integral Effects Test Facility.
Figure 6.2 The CIET facility is scaled to approximately match heights of heat sources and sinks in the Mk1 PB-FHR preconceptual design.

To develop experience prior to designing and constructing CIET 1.0, a simplified natural circulation loop was operated on the PS-HT² facility at UCB. Key components (e.g., resistive heating element and instrumentation) were tested in the PS-HT² facility before being adapted to the CIET 1.0 facility. Test loops for CIET were fabricated from stainless steel tubing and welded fittings, allowing rapid construction and design modifications. The simplicity of the construction, compared to the complexity and safety requirements for tests with the prototypical salt and other prototypical reactor coolants, was a key element in enabling the proposed experiments to be performed at much lower cost than previous integral effects tests for other types of reactors. As noted in CIET final design report, the total cost for procurement and fabrication of the CIET facility was $253,000, along with 2480 days of labor for design, assembly, and start up testing of the facility.

Over the course of this three-year, U.S. Department of Energy funded Integrated Research Project (IRP), the CIET 1.0 facility was designed, fabricated, filled with Dowtherm A oil and operated. Isothermal pressure drop tests were completed, with extensive pressure data collection to develop validated models for flow losses in the system. CIET-specific friction loss correlations were compared with handbook values used in the initial design of the facility, and empirically measured values were implemented in the system codes that are to be validated by CIET data. Significant insights were gained about the accuracy of different friction and form loss correlations provided by handbooks and by vendor literature for specific CIET components such as flow meters and static mixers. The project then entered a phase of heated tests, from parasitic heat loss tests to more complex feedback control tests and natural circulation experiments, which will continue on the same facility with no major modifications in the
near future. In parallel, UCB has been developing thermal hydraulic simulation models to predict FHR steady-state characteristics and transient response for a set of reference LBEs. The general FHR strategy, recommended by an expert workshop, is to rely on existing general-purpose thermal hydraulic codes with a significant verification and validation basis for design and licensing by the U.S. Nuclear Regulatory Commission, such as RELAP5. However, UCB has also been developing a one-dimensional FHR advanced natural circulation analysis (FANCY) code for CIET and FHR natural circulation modeling, whose results will be compared with RELAP5 and validated by data from CIET 1.0. Table 6.1 provides a list of selected UCB publications on their experimental work.

Table 6.1. Selected UCB FHR Publications

<table>
<thead>
<tr>
<th>No.</th>
<th>Authors</th>
<th>Title</th>
<th>Details</th>
</tr>
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6.2. Corrosion Experiments at the University of Wisconsin

6.2.1: Overview

The University of Wisconsin (UW) has successfully completed out-of-core corrosion tests of 316 stainless steel and Hastelloy N in molten 7Li-enriched FLiBe salt (procured from ORNL from MSRE program) at 700°C for exposure durations of up to 3000 hours. Surrogate TRISO particles (up to 300 particles), bulk CVD SiC, SiC-SiC composites (Tyranno and Hi-Nicalon), carbon-carbon composites, and IG110 and ARB grade graphite were also tested for corrosion in molten FLiBe salt. In addition to conducting corrosion tests in standard metallic crucibles (e.g., stainless steel) as per procedures developed at the UW, tests were also performed in specially designed graphite crucibles that contained three or six cylindrical compartments. The tests using the compartmentalized graphite crucible design were based on the requirements of the MITR where in-core corrosion tests were performed of the same
materials. Thus by performing out-of-core and in-core corrosion tests under the same conditions, the effect of radiation on corrosion could be established. Figure 6.3 shows a photograph of the compartmentalized graphite crucibles used for some of our corrosion tests.

It is well-recognized that impurities in the salt strongly drive corrosion by increasing the redox potential of the salt. Extensive procedures were developed to purify the salt by optimizing hydrofluorination treatments and beryllium additions. To measure the efficacy of the purification treatments, a probe was developed for measuring redox potential of the molten FLiBe salt purified by different routes which enabled us to perform corrosion tests under two different redox potentials (-151 and -170 kcal/mol $F_2$). Finally, proton irradiation of 316 stainless steel at 450°C to a dose of 0.5dpa was also completed for analyzing the effects of radiation-induced segregation in stainless steel on corrosion.

Figure 6.3 Photographs of compartmentalized graphite crucibles used for out-of-core (UW) and in-core (MIT) corrosion tests.

6.2.2. Corrosion Studies of 316 Stainless Steel

Corrosion of 316 stainless steel was dominated by the dissolution of the Cr into the salt with corrosion attack occurring predominantly along the grain boundaries. The corrosion in graphite capsules exhibited about twice the attack depth compared to those tested in 316 stainless steel capsules, indicating that graphite has a significant accelerating effect on the corrosion of stainless steel. Furthermore, for samples tested in graphite crucible, carburization of 316 stainless steel was observed throughout the depth of the corrosion layer, due to the formation of chromium carbide by the reaction of outward diffusion of Cr from the alloy and inward diffusion of C into the samples.

Figure 6.4 shows the SEM images of the surface and EDS-mapping of Cr element in cross section of the samples tested in 316 stainless steel and graphite liner for 1000 to 3000 hrs. The images and elemental mapping results clearly show the corrosion-accelerating effect of graphite on 316 stainless steel. Extrapolation of this data indicates a corrosion rate of about 17 µm/year in stainless steel crucible tests and about 31 µm/year in graphite crucible tests.
Figure 6.4. SEM images of surface and EDS mapping of Cr element in cross section of the 316 stainless steel after corrosion testing in (a) 316 stainless steel crucible and (b) graphite crucible for 1000, 2000, and 3000 hours in molten FLiBe salt at 700°C.

Figure 6.5. Depth of corrosion attack in 316 stainless steel tested in 316 stainless steel crucible (lower curve) and graphite crucible (upper curve) after testing in molten FLiBe salt at 700°C.

6.2.3. Corrosion of Hastelloy N (Alloy N) in Molten FLiBe

A slight weight loss of 0.12mg/cm² was observed in Hastelloy N after 700°C/1000hr corrosion in nickel crucible, but a weight gain of 0.16 mg/cm² was observed in the same alloy tested in graphite.
crucible. In general corrosion of Alloy N was lower than 316 stainless steel. Analysis of the surfaces of samples tested in graphite crucibles by SEM-EDS and XRD analysis showed the formation of surface deposits as chromium carbides and molybdenum carbides on the surface which accounts for the slight weight gain. A carburized layer was also observed to form in samples tested in graphite crucibles and this layer extended to a depth of about 7 µm underneath a near-surface Ni$_3$Fe layer that also formed as a consequence of the corrosion process. In contrast, no deposits were observed on samples tested in Ni crucible where the near-surface region of the alloy exhibited a 200nm deep porous structure layer. An overall 3.5 µm chromium depleted region or attack depth was observed in Hastelloy N tested in nickel crucible. The EDS mapping of the cross section of Hastelloy N tested in nickel liner and graphite crucible was shown in Figure 6.6.

Figure 6.6 SEM-EDS mapping analysis of Hastelloy N samples tested in (a) nickel crucible and (b) graphite crucible. Cu coating was electroplated on sample surface prior to polishing for edge retention.

6.2.4. Corrosion of SiC/SiC Composites, C/C Composites and Graphite

Figure 6.7a shows the weight loss of monolithic CVD SiC and two types of CVI SiC-SiC composites (Tyranno and Nicalon) after corrosion test in molten FLiBe salt in graphite crucible at 700°C for 1000 hours. The SEM images of SiC and SiC/SiC composites before and after corrosion are shown in Figure 6.8. The weight loss for monolithic SiC is extremely low indicating that SiC as a material is resistant to corrosion attack in FLiBe salt. However, in the composites some loss of material and higher porosity were observed after corrosion tests due to attack at the fiber-matrix interfaces. The higher weight loss for the composites and the differences in the performance of the two composites suggests that the manufacturing process used for these composites may play a role in their performance in molten FLiBe salt. Figure 6.8b shows the weight loss of C/C composite and graphite after corrosion tests in molten FLiBe salt at 700°C for 1000 hours. Slight weight loss was observed in C/C composite after corrosion. For graphite there was no observable corrosion under the SEM.
Figure 6.7. Weight loss of (a) monolithic SiC and SiC-SiC composites and (b) C/C composite, IG110 and ARB graphite after corrosion tests in molten FLiBe salt at 700°C for 1000 hours.

Figure 6.8. SEM images of SiC/SiC composite and CVD SiC (a) before and (b) after corrosion test in molten FLiBe salt.
6.2.5. Corrosion of Surrogate TRISO Fuel Particles

No damage was observed in surrogate TRISO fuel particles after corrosion testing at 700°C for 1000 hours (Figure 6.10). Inspection of a number of fuel particles after corrosion tests showed the graphite and SiC coatings on the TRISO particles to be highly intact after corrosion tests.

![Figure 6.9. Photograph of the TRISO particles tested in molten FLiBe salt and SEM cross-sectional image of a particle after corrosion test in molten FLiBe salt at 700°C (right).](image)

6.2.6. Proton Irradiation of 316 Stainless Steel

316 stainless steel samples were irradiated with 2.0MeV protons to a dose of 0.5 dpa at about 450°C. The goal was to examine the effects of radiation induced segregation (RIS) on corrosion in molten FLiBe salt at 700°C. Figure 6.11 shows the photographs of the irradiation chamber with samples loaded and two proton irradiated 316SS samples. However, after proton irradiation, the stainless steel samples exhibited measureable radioactivity levels due to some Co impurity in the stainless steel. Thus under the advice of the radiation safety personnel at UW, the corrosion tests have been delayed and strategies for the disposal of mixed waste (chemical and radioactive) from the FLiBe corrosion laboratory are currently being evaluated.

![Figure 6.10. Photograph of the chamber for proton irradiation (left) and stainless steel samples after 2.0MeV proton irradiation to 0.5dpa at 450°C.](image)

6.2.7 Purification of $^7$Li-enriched FLiBe Salt from ORNL

Because salt containing impurities such as moisture and oxides can significantly corrode metal systems, fluoride salt free of impurities is required to run a molten salt reactor. UW has created a batch purifier capable of sparging hydrogen fluoride and hydrogen in up to 52 kg of FLiBe at temperatures up to 650°C. To mitigate the risk associated with beryllium work, a negative pressure walk-in fume hood
was shown in Figure 6.12 was developed to contain beryllium dust. The room measures roughly 9' x 12' with a 10' tall ceiling. The room is large enough to contain the purification system as well as a full line of corrosion tests.

![Figure 6.11. Photographs of the walk in fume hood and the FLiBe salt batch purifier.](image)

The purification of the as-received MSRE FLiBe salt has been successfully completed. To facilitate purification, the temperature of the salt was increased from 550°C to 600°C. The total flow rate for purification was 2.5 L/m, with 250 mL/m HF and 2.25 L/m hydrogen. Effluent hydrogen fluoride neutralization was dissolved into a series of water carboys and neutralized with a sodium hydroxide solution in a fume hood when the max HF concentration reached 10%. The hydrofluorination was done for a satisfactory 76 hours. Hydrogen sparging at 631°C with a flow rate of 2.5 L/m for three days was used to reduce the nickel fluoride dissolved in the salt from fluorinating and iron fluoride from the ORNL shipment container. After three days’ sparging, the salt was cooled to the transfer temperature of 600°C and the hydrogen bubble rate was reduced to 0.5 L/m.

6.2.8. Measurement of Redox potential of FLiBe Salt

A compact electrochemical probe, as shown in Figure 6.12, has been developed to measure the redox potential in FLiBe via an in-situ dynamic beryllium electrode in the likeness as that described by Afonichkin et al. This probe is capable of operating on a loop, or static system at salt temperatures exceeding 500°C. The probe has been well-characterized in Li₂BeF₄ salt over wide variety operating conditions, yielding redox measurements with an accuracy of ±4 mV. A 760-1780 μV per measurement reduction was observed during repetitive measurements.

The `dynamic reference electrode' removes the need for an ion permeable container, creating a simpler, more robust probe, which has the potential to operate in varying depths. A measurement begins by applying a voltage to a submerged cathode, creating a current between the cathode and submerged anode, forcing beryllium fluoride to reduce as beryllium at the cathode. After sufficient plating, the circuit is opened, allowing beryllium to donate electrons to the salt, reforming beryllium fluoride. As the beryllium recombines, it creates a distinct voltage on the cathode which can be referenced and compared to the potential on a third, submerged, indicator electrode, yielding the redox potential.

This probe has many advantages such as (1) it can operate under a wide array of salt chemical conditions without corroding or malfunctioning, (2) it can be disassembled, repaired, and assembled.
without difficulty, (3) it is attachable to a wide array of salt experiments, both static and flowing, while holding a gas seal at temperature and (4) it is compact and therefore can be easily loaded into an out of glove boxes and minimize the length of the delicate electrodes.

Figure 6.12. Photograph of the compact probe for measuring the redox potential

![Figure 6.12. Photograph of the compact probe for measuring the redox potential](image)

Six back-to-back measurements of purified, and beryllium reduced, flibe were taken at 500°C using a constant current of 13 mA for 2 s. This current and time combination was found through trial and error and was used for all measurements with desirable results. The runs, shown in Figure 6.13, were found to be reproducible. Out of six runs, all experienced nearly identical plateaus, and all exhibited the same characteristic decline to 0 V.

This probe is ideal for use in both flowing and static loops, where redox potential measurements are necessary for corrosion studies. Future work should include a calibration of the dynamic beryllium reference electrode to a standard reference electrode which is unaffected by impurities.

Figure 6.13. Complete measurement showing the plateau and relaxation to 0 V (left) and a zoom in of the plateau region showing the slope of beryllium dissolution and reduction of the salt (left).

![Figure 6.13. Complete measurement showing the plateau and relaxation to 0 V (left) and a zoom in of the plateau region showing the slope of beryllium dissolution and reduction of the salt (left).](image)
Table 6.2 List of UW publications/presentations

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<th>Publication Details</th>
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<tr>
<td>5</td>
<td>G. Cao, et al., Fluoride-Salt-Cooled, High-Temperature Reactor (FHR) Materials, Fuels and Components White Paper, UCBTH-12-003 Department of Nuclear Engineering, University of California, Berkeley (2013)</td>
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<tr>
<td>6</td>
<td>G. Cao, Corrosion of SiC composite in molten FLiBe Salt, Oral presentation in NuMat 2014, October 27-30, Clearwater Beach, FL</td>
</tr>
<tr>
<td>9</td>
<td>G. Zheng, Corrosion in High Temperature Molten Li2BeF4 (FLiBe), Oral presentation in NuMat 2014, October 27-30, Clearwater Beach, FL</td>
</tr>
<tr>
<td>11</td>
<td>G. Zheng, Corrosion Behavior of Structural Alloys in Molten Fluoride Salts for Next Generation Nuclear Reactors, Poster presentation in 2012 ANS annual meeting, June 24-28, Chicago, IL</td>
</tr>
</tbody>
</table>
6.3. Irradiation Experiments at the Massachusetts Institute of Technology

Completed experimental work at the MIT Nuclear Reactor Laboratory (MITR) consisted of the design, construction, and operation of two in-core, high-temperature (700°C) flibe corrosion tests (referred to as FS-1 and FS-2), Post-Irradiation Examination (PIE) of the samples from the first in-core test, and trace elemental analyses by Neutron Activation Analysis (NAA) of flibe and sample materials used in the UW and MIT purification and corrosion experiments. The MITR in-core tests FS-1 and FS-2 were designed to examine the corrosion and behavior of likely FHR structural and fuel-related materials in high-temperature flibe while subject to reactor-level radiation fields, and to examine the transport and fate of tritium produced in the flibe. The flibe, sample materials, and test temperatures and durations used in FS-1 and FS-2 were designed to be identical to those used in experiments at UW so that the effects of irradiation on the behavior of the flibe and test materials could be assessed.

In-Core Experimental Design and Constricture

Many aspects of both in-core experiments were similar. Both experiments had six independent graphite test compartments (Figure 6.14), inside sealed nickel capsules (Figure 6.15) that were instrumented with thermocouples. The nickel capsules were sealed into titanium thimbles that were inserted into dummy fuel elements in the MITR core (Figure 6.16). The FS-1 was run in the MITR multipurpose In-Core Sample Assembly (ICSA) thimble which limited the capsule height to approximately 17 cm. The FS-2 was run in a new, dedicated in-core thimble which had a height of approximately 34 cm, which covers most of the height of the MITR core. The flibe used in these tests and those at UW was originally used in the secondary (non-fueled) coolant loop of the MSRE. The lithium in this flibe is isotopically enriched to 99.99% $^7$Li (natural enrichment is 92.4% $^7$Li, 7.6% $^6$Li) to reduce the amount of tritium produced through the $^6$Li (n,α) $^3$H reaction.

The tests were heated by absorption of the gamma and neutrons in the inner components and this heat rejected across a He/Ne gas gap between the nickel and titanium and into the MITR primary coolant. Neutronics and nuclear heating analyses of the two in-core experiments were performed using the MCNP-5.1.4 code and a 3D representation of the MITR core. The heating data were fed into the ANSYS Fluent v13 computational fluid dynamics code, which provided feedback for fine-tuning the dimensions of the capsule to achieve the target temperature in the salt chambers. The gases inside and outside the capsule were independent and could be sampled for tritium and trace impurities. Mass flow controllers were used to meter gases into the capsule and thimble; the neon flow rate was adjusted automatically to control the capsule temperature.
Figure 6.14. Graphite capsule components of the FS-1 (a) and FS-2 (b), both filled with samples and flibe prior to their in-core tests.

Figure 6.15. Nickel and graphite capsule components of the FS-1 (a), the assembled nickel capsule of the FS-2 (b), and outer titanium thimble of the FS-2 (c).
Figure 6.16. 3-D models of in-core test facilities at the MITR. Figure 6.3a shows a cutaway of the full MITR core tank. Figure 6.3b shows of the core region only, with three in-core experiments inserted. The FS-1 experiment was run in the light-blue thimble. The FS-2 experiment was inside a dedicated thimble shown in yellow in Figure 6.3b. Figure 6.3c shows a cutaway of the in-core section of the FS-2.

The FS-1 and FS-2 specimen test matrices are shown in Table 6.3. The FS-1 matrix included 316 SS and Hastelloy N samples with and without exposure to graphite and CVD and composite SiC. The FS-2 matrix included 316 SS and SiC/SiC fiber composite materials in flibe at two different fluorine potentials, and C/C fiber composites. Surrogate TRISO particles (with ZrO$_2$ rather than UO$_2$ kernels) were tested in both experiments. The TRISO particles in FS-2 were part of the most recently produced batch from Babcock and Wilcox. The IG-110U graphite used as liner in both experiments and samples in FS-2 is a ultra-high-purity nuclear grade structural graphite, while the ARB-B1-Z009 coupons in FS-2 are used for the outer layer of graphite fuel compacts. Loading of the flibe and specimens into the graphite was performed inside an argon-filled glove box at UW where flibe was added into each compartment directly from a heated storage container. Final assembly of the capsules was performed under helium at MIT.
Table 6.3. Sample test matrices of the FS-1 and FS-2.

<table>
<thead>
<tr>
<th>FS-1</th>
<th>FS-2</th>
</tr>
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<tbody>
<tr>
<td>2 pieces 316 SS, no liner</td>
<td>2 pieces 316 SS, F-pot. = -151 kcal/mol F₂</td>
</tr>
<tr>
<td>2 pieces 316 SS, 316 SS liner</td>
<td>2 pieces 316 SS, F-pot. = -170 kcal/mol F₂</td>
</tr>
<tr>
<td>2 pieces Hastelloy N, no liner</td>
<td>2 pieces SiC composite, F-pot. = -151 kcal/mol F₂</td>
</tr>
<tr>
<td>2 pieces Hastelloy N, nickel liner</td>
<td>2 pieces SiC composite, F-pot. = -170 kcal/mol F₂</td>
</tr>
<tr>
<td>3 pieces SiC/SiC composite, 1 piece SiC</td>
<td>2 pieces IG-110U, 2 pieces ARB-B1-Z009, ~100 surrogate TRISO particles, F-pot. = -151 kcal/mol F₂</td>
</tr>
<tr>
<td>~300 surrogate TRISO particles</td>
<td>2 pieces C/C composite, ~100 surrogate TRISO particles, F-pot. = -151 kcal/mol F₂</td>
</tr>
</tbody>
</table>

**In-Core Experiment Operational Results**

The FS-1 and FS-2 were both designed to operate at 700°C for 1000 hours in order to parallel the tests being done at UW. The FS-1 was inserted into the MITR on September 8, 2013 and began operation at 700°C on September 17; the experiment ran without interruption and at constant reactor power for the next 1000 hours, shutting down on October 29. The FS-2 was inserted into the MITR on July 8 2014 and began operation at 700°C on July 18. On July 31 after 300 hours of operation; the MITR scrammed due to a condition unrelated to the FS-2. When the reactor restarted, it was not possible to maintain stable pressure in the FS-2 capsule, and it was discovered that a leak had developed in the tritium monitoring system. Because these conditions were not readily explained or remedied, and the safe operation of the experiment could not be ensured, the reactor was shut down. The reactor was restarted briefly on August 6 to bring the FS-2 up to a temperature just below 400°C to test several hypotheses (discussed below) about the conditions that occurred on August 31. Figure 6.17 shows the FS-1 and FS-2 temperature and MITR power histories.

![Figure 6.17](image-url)
There were several findings of interest among the data collected during the operation of FS-1 and FS-2. Tritium is created by neutron absorption by lithium. Tritium measurements of the exhaust gases from the nickel capsule that contained the flibe and test materials, and from the titanium thimble outside the capsule were not substantially different. Because there was no evidence of leakage from the capsule into the thimble and because the thimble operated at a higher pressure than the capsule, the equivalence of the tritium levels between these systems indicates that, at 700°C, the nickel capsule is essentially transparent to tritium diffusion. Additionally, the amount of tritium measured in these gases was only small fraction of the tritium produced in the flibe. Near the beginning of the test the gas-phase tritium accounted for approximately 10% of the total produced; as the experiment progressed this fraction dropped to around 0.1%. It is likely that much of the tritium was absorbed into the graphite. A custom furnace has been built to heat the test components to temperatures well above the irradiation temperature to attempt to evolve the tritium and establish a mass balance for the experiment.

In parallel with the experimental work, a model called TRItium Diffusion EvolutioN and Transport (TRIDENT)\(^\text{19}\) had been developed to analyze tritium generation and is subsequent distribution/transport in the FHR and to the environment. Because tritium diffuses through hot metals, the FHR will require methods to block release of tritium to the environment and remove the tritium from the salt. Alternative control strategies were investigated. Understanding tritium behavior and its control is a major experimental and modeling activity for the next three years. Figure 6.18 shows TRIDENT flow chart.

During the operation of both FS-1 and FS-2 two energetic but short-lived activation products, $^{16}\text{N}$ ($t_{1/2} = 7.1$ seconds) and $^{19}\text{O}$ ($t_{1/2} = 27$ seconds), were identified in the exhaust gas from the capsule. These are both produced from fast neutron reactions on fluorine ($^{19}\text{F}(n,\alpha)^{16}\text{N}$ and $^{19}\text{F}(n,p)^{19}\text{O}$). The release of these gases was highest when the flibe was still solid, and decreased to almost undetectable levels when the flibe was at 700°C. No evidence has been found that the presence of $^{16}\text{N}$ and $^{19}\text{O}$ was observed during the operation of MSRE where the continuous high-temperature operation may have delayed the gases release and the presence of volatile fission products would have masked detection. The flibe was at 700°C only when the reactor was at full power and production would be the greatest. Thus at high temperatures the gases must be being delayed from escaping the flibe long enough for them to decay. Nevertheless, future experiments, test reactors, or FHRs there is the possibility for very high radiation fields to exist in head spaces and off gas systems if these gases escape the flibe. The mechanisms for $^{16}\text{N}$ and $^{19}\text{O}$ retention and release needs to be studied further, and designs of nuclear fluorine salt gas systems must consider phenomenon. These short-lived radionuclides in the salt imply a significant radiation field associated with the coolant during operations that rapidly disappears when the reactor is shut down.

The conditions that led to the early termination of the FS-2 were examined and it is believed that they occurred due to production of free fluorine by radiolysis of the flibe. Measurements taken on the capsule exhaust gas during the FS-2 experiment and the brief test following the reactor scram found significantly elevated levels of filterable $^{187}\text{W}$ and $^{99}\text{Mo}$ (with equilibrium $^{99m}\text{Tc}$), and $^{82}\text{Br}$ in the first stage of the tritium impinger (water bubbler). The highest release rates of these activation products occurred when the flibe was at lower temperatures (around 200°C), and were very low when the flibe was at its full operating temperature of 700°C. It is likely the W and Mo were released as fluorides following the evolution of $\text{F}_2$ from the flibe (e.g. $\text{W} + 3\text{F}_2(g) \rightarrow \text{WF}_6$) with the Br behaving similarly to the F. It is believed that these conditions were not as severe in the FS-1 test because the flibe in this earlier test was not at low temperature while in the MISTR core for as long as the FS-2 (due to there being no reactor scrams or prolonged shut-downs during the FS-1 test). The radiolytic formation and evolution of $\text{F}_2$ from flibe at temperatures below 100°C was noted in a 1964 ORNL report and observed during the long-term storage of the fuel salt that was stored at low temperatures and contained high concentrations of fission products. At higher temperatures the back reaction limits the evolution rate. A temperature of 200°C was deemed sufficient to prevent the generation of free fluorine gas. Future experiments, test reactors, and FHRs will need to maintain the flibe at sufficient temperatures in radiation fields to avoid the production of fluorine gas and the potential adverse effects on components in gas systems.

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In-Core Experiment Post-Irradiation Examinations

Following a decay period for the major activation products of the FS-1 components, the test materials were extracted from the test flibe inside a helium filled glove box. The samples of proposed structural materials have been weighed, measured for size, and photographed. The surrogate TRISO particles have been examined by optical microscopy. The nickel test capsule from the FS-2 is currently being stored under helium in the MITR hot cell until the activation products in several of the components have decayed sufficiently to allow MITR personnel to handle them without excessive radiation exposures.

Table 6.4. Volume and weight changes of Hastelloy N, 316SS, and SiC samples during the FS-1 test.

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<th>Compartment material</th>
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<td>—</td>
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<td>—</td>
</tr>
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<td>—</td>
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</tr>
<tr>
<td>SA3 SiC/SiC composite</td>
<td>graphite</td>
<td>-0.18*</td>
<td>-2.2</td>
</tr>
<tr>
<td>HNLS SiC/SiC composite</td>
<td>graphite</td>
<td>-1.23*</td>
<td>-1.4</td>
</tr>
</tbody>
</table>

*Surface areas of SiC/SiC composites not corrected for surface roughness/open porosity.

Weights and dimensions of the 316SS, Hastelloy N, and SiC samples were all measured before and after the FS-1 test. Changes in the masses and volumes of these materials are shown in Table 6.4. The samples all lost measurable mass. Among the metal samples, the 316 SS lost more mass than the Hastelloy N, and the samples in the graphite compartments lost more mass than those in the metal lined compartments. If the weight loss in the metals was due to oxidation of components of the materials into soluble fluorides, then the lower weight losses in the nickel and 316 SS-lined compartments may be due to the losses from the liners raising the metallic fluoride levels in the flibe, thus decreasing the rate of oxidation from the samples. The SiC samples all shrank during the test. The volume changes of the metal samples were smaller than the uncertainty in their measurement.

Following their extraction from the flibe, the surrogate TRISO particles irradiated in FS-1 were observed to have significant cracking of their Outer Pyrolytic Carbon (OPyC) layer (Figure 6.19). This type of cracking was not observed on as-fabricated particles, particles irradiated in helium but not exposed to flibe, nor on particles exposed to flibe at UW but not irradiated. Particles previously irradiated at high temperature under helium in the MITR were tested by immersing them in molten flibe which was then frozen and re-melted three times. These particles exhibited OPyC cracks similar to those seen in the FS-1 TRISO, indicating that the cracks are due to the confluence of an irradiation effect (which degrades the OPyC, perhaps by weakening the bonding between the OPyC and SiC layers and/or
embrittling the OPyC) and a mechanical effect from the freezing-thawing action of the flibe around the particles. In an FHR, the TRISO particles are not expected to be exposed directly to the flibe because they will be embedded in a carbon matrix, and the fuel is not expected to go through multiple freeze-thaw cycles. However, the cracking of irradiated OPyC during flibe freeze-thaw cycles needs to be considered in future experiments and fuel design studies—including the options of (1) maintaining samples above the melting point of the salt after irradiation until the liquid salt is separated from the TRISO particles or (2) embedding TRISO particles in a graphite matrix as they will be embedded in the fuel. Historically bare TRISO particles have been tested in helium with multiple heating and cooling cycles—this experimental approach may need to be changed when testing in salt. The FS-2 test used a more recent batch of surrogate particles (conforming to AGR1 – 1&2 specifications) which have yet to be examined.

Figure 6.19. Surrogate TRISO particles from the FS-1 test showing cracks (a) and loss (b) of OPyC layer.

Trace Element Component Measurements of Flibe by Neutron Activation Analysis

Because corrosion of materials in flibe can be significantly affected by the presence of minor and trace impurities, the MSRE flibe used in the MIT and UW corrosion tests was purified at UW. The effectiveness of the purification and the levels of these impurities after the corrosion tests were measured by Neutron Activation Analysis (NAA) at the MITR. To provide as complete an analysis of the flibe as possible by NAA, samples of the flibe were irradiated in two different facilities at the MITR for three different irradiation periods; gamma spectra of the activation products were collected for five different decay and counting periods. The overall NAA program produced data for a total of 43 different elemental impurities. Analytical results for several elements of interest are shown in Table 6.5. NAA is a standard technique used today but did not have the high resolution capability for impurities when the MSR program was underway. It is one of many areas where a much better understanding of corrosion is possible because of better analytical techniques.

After its use in the MSRE and storage for several decades, the as-received flibe contained fairly high levels of Cr and Fe (above the maximum specified level for the MSRE primary). The first purification preceded the MIT (FS-1) and UW 1000-hour corrosion tests. The purification processes were then improved prior to the second 1000-hour tests. The second purification included the addition of Be to the
flibe. Fe levels were decreased significantly in the first purification and further in the second purification. Cr levels were lessened only slightly by the early purifications, but decreased significantly with the addition of Be to the salt which lowered the fluorine potential and caused CrF$_2$ to precipitate as Cr(s). The Ni levels increased from the concentration in the as-received flibe after the initial purification work (which was carried out in nickel vessels), but was successfully reduced by the second purification. The element Co was present at very low levels in all of the samples and may not be of concern with respect to corrosion, but Co is often a problem in nuclear systems where it activates easily to the long-lived and energetic radionuclide $^{60}$Co. In the purification tests Co behaved similarly to Fe.

Table 6.5. Selected elemental concentrations (µg/g) of flibe in purification and corrosion tests, and the MSRE primary loop maximum specified concentrations.

<table>
<thead>
<tr>
<th>Element</th>
<th>MSR max. spec.</th>
<th>First purification</th>
<th>Second purification</th>
<th>1000-hour UW Corrosion tests</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
<td>As received</td>
<td>Purified</td>
<td>Second step</td>
</tr>
<tr>
<td>Cr</td>
<td>25</td>
<td>46</td>
<td>36</td>
<td>31.5</td>
</tr>
<tr>
<td>Fe</td>
<td>100</td>
<td>144</td>
<td>86</td>
<td>&lt;9.7</td>
</tr>
<tr>
<td>Co</td>
<td>—</td>
<td>0.038</td>
<td>&lt;0.07</td>
<td>&lt;0.0097</td>
</tr>
<tr>
<td>Ni</td>
<td>25</td>
<td>0.38</td>
<td>6.6</td>
<td>0.73</td>
</tr>
</tbody>
</table>

Table 6.6. Selected MIT FHR Publications

<table>
<thead>
<tr>
<th></th>
<th>Publication</th>
</tr>
</thead>
</table>
7.0. CONCLUSIONS

As stated in our proposal: The objective of this Integrated Research Project (IRP) is to develop a path forward to a commercially viable salt-cooled solid-fuel high-temperature reactor with superior economic, safety, waste, nonproliferation, and physical security characteristics compared to light water reactors. There are three requirements for the commercialization of the FHR: (1) a compelling commercial (utility and vendor) and national (government) case for development of the technology, (2) a reasonable case that the technology can be developed and (3) a roadmap for commercialization. The starting point is the definition of goals. Three goals were defined.

- **Economics.** Increase plant revenue by 50 to 100% relative to base-load nuclear power plants with plant costs similar to light-water reactors (LWRs)
- **Environment.** Provide the enabling technology for a zero-carbon nuclear renewable electricity grid
- **Safety.** Assure that fuel failure with large-scale radionuclide releases will not occur under extreme conditions—including beyond design basis accidents (BDBAs).

Fuel, coolant, and power cycle choices enable meeting the goals as summarized in Table 7.1. Meeting the economic and zero-carbon electricity grid goals require NACC and FIRES. NACC defines the top-level reactor requirements and thus drives the choice of fuel and coolant. In modern gas turbines the exit temperature from the air compressor is between 350 and 500°C. That implies any reactor coupled to a utility gas turbine must deliver heat above those temperatures. Neither LWRs nor SFRs have that capability. The capability to be the enabling technology for a zero-carbon grid requires the addition of FIRES heat storage to NACC—a storage technology partly being developed elsewhere for gas turbines. The accident resistance capability of the FHR is a consequence of a high-temperature fuel and a high-temperature coolant. The fuel cycle characteristics are consequences of fuel choices.

**Table 7.1. Mapping of Technologies and Goals**

<table>
<thead>
<tr>
<th>Project Goals → Required technologies ↓</th>
<th>Improved Economic Performance</th>
<th>Zero-Carbon Electricity Grid</th>
<th>Accident Resistance</th>
<th>Fuel Cycle Performance</th>
</tr>
</thead>
<tbody>
<tr>
<td>High-temperature fuel</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td>X</td>
</tr>
<tr>
<td>Liquid salt coolant</td>
<td>X</td>
<td>X</td>
<td>X</td>
<td></td>
</tr>
<tr>
<td>NACC</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
</tr>
<tr>
<td>FIRES</td>
<td>X</td>
<td>X</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

There are similarities between the drivers that favored the development of the LWR in the 1960s and the FHR today. In the 1960s there were concerns about fossil fuel prices and thus the need for new methods to generate base-load electricity. Today the utility concern is how to provide economic variable electricity to the grid because of (1) the increase need for variable electricity output in a grid that contains significant quantities of non-dispatchable renewables and (2) potential limitations on burning fossil fuels—the primary method we use to produce variable electricity. The combined cycle gas turbine is replacing pure steam cycles in utility applications because of higher efficiency and its other unique...
capabilities. Steam technology allowed efficient coupling between the LWR and the electricity grid. For the FHR, gas turbine technology allows efficient coupling between the FHR and the electricity grid. Last, other government programs have developed many of the base technologies required for the FHR.

![Diagram of Market and Power Cycle]

**Market**
- (Constant Reactor Power)
- **LWR**: Base-Load Electricity
- **FHR**: Variable Electricity

**Power Cycle**
- **LWR**: Steam
- **FHR**: Air Brayton
- **Combined Cycle With Storage**

**Reactor**
- **LWR**: Water Coolant
- **FHR**: HTGR Fuel and Salt Coolant

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Fig. 7.1. Comparison of LWR and FHR

The three universities built major test capabilities that have advanced FHR technology and provide the starting point for the further development of the FHR. This includes (1) thermal hydraulic test loops using Dowtherm®-A at UCB, (2) salt purification systems and corrosion test systems at UW, and (3) the materials irradiation capabilities in 700°C salt at MIT. These facilities have begun to provide the experimental data required to validate design tools (thermal hydraulic codes, tritium modeling, etc.) and for making FHR design choices (materials of construction, tritium control strategies, etc.).

The technical, financial and institutional challenges have been identified but should not be underestimated. The required component technologies exist but it requires a large development program that will have significant challenges to develop and integrate those technologies into a practical power plant. We have developed a first detailed point design of an FHR—but there are a wide set of FHR design options that can meet the stated goals. The design space has been only partly explored and needs to be better understood to develop the most economic machine and to assure the R&D program has the appropriate focus. This is an area where early vendor involvement to develop their own pre-conceptual designs can provide essential input. The financial challenge is to provide the required funding over the multi-decade development time before a commercial product is developed, built, installed and begins to operate to generate revenue. There are multiple institutional challenges but the most significant in the United States are likely to be associated with reactor licensing. The experience base is with LWRs—there is little experience in licensing a totally new technology.
Molten salt coolants in reactors were developed for the Aircraft Nuclear Propulsion Program and then for the molten salt reactor (MSR) program with the goal to develop a commercial power reactor. That work stopped by the mid-1970s. About a decade ago there was a renewed interest in using salt coolants in nuclear systems. This salt technology, coupled with fifty years of advancing gas turbine technology, has created new technical options that, in turn, create the possibility of an alternative nuclear strategic vision.

The strategic perspective starts with the goal of an economic zero-carbon electricity grid (Fig. A.1). That drives the choice of power cycle for nuclear reactors that in turn drives the choice of reactor coolant and ultimately the nuclear options. This appendix outlines one strategic vision while acknowledging the large technical uncertainties.

A.1. Strategic Perspective Using Salt Coolants for Zero-Carbon Electricity Grid

The electricity grid must provide variable electricity on demand. Historically this has been accomplished by low-capital-cost high-operating-cost fossil plants. Because of the low capital costs, fossil plants can be operated at part load without large impacts to electricity costs. Low-carbon electricity generating technologies (nuclear, wind, and solar) have high capital costs and low operating
costs. If they are not operated at full capacity, the cost of delivered electricity will be expensive. Furthermore, wind and solar are non-dispatchable sources of electricity. Their use requires another generating technology to provide electricity at times of low wind or solar conditions. That technology today is burning natural gas—not an option for a zero-carbon electricity grid. Traditional storage technologies can’t fill the gap because (1) longer periods of low wind or solar will deplete the storage systems and (2) there are hourly to seasonal storage requirements.

A Fluoride-salt-cooled High-temperature Reactor (FHR) with a Nuclear Air-Brayton Combined Cycle (NACC) and Firebrick Resistance-Heated Energy Storage (FIRES) can potentially meet the variable electricity requirements for a zero-carbon nuclear renewable grid. It is the near-term nuclear option; however, it is also the pathway to more advanced systems including Molten Salt Reactors (MSRs) and fusion machines. There is a potential pathway using common technologies that goes beyond the FHR depending upon future technical developments.

**Fluoride-salt-cooled High-temperature Reactors (FHRs)**

The FHR with Nuclear Air-Brayton Combined Cycle (NACC) and Firebrick Resistance Heated Energy Storage enables base-load reactor output with variable electricity to the grid that (1) increases plant revenue by 50 to 100% relative to a traditional base-load nuclear plant and (2) enables a zero-carbon electricity grid. It may also be the enabling technology for use of significant wind or solar by reducing electricity price collapse that occurs if significant wind or solar on the grid—as well as a potential replacement to utility-scale electricity storage technologies (batteries, pumped storage, etc.). The NACC cooling water requirements are only 40% of a traditional light-water reactor. The FHR is a new reactor concept that is about a decade old, no FHR has ever built. These characteristics follow from three developments.

- **Fluoride Salts.** Fluoride salts were developed to transfer heat from reactor cores to jet engines in the Aircraft Nuclear Propulsion Program where the goal was a nuclear powered jet bomber. The fluoride salts had two requirements to meet: (1) the nuclear requirements of the reactor (low neutron absorption cross sections, etc.) and (2) the heat transfer requirements of the reactor and heat exchangers to a gas turbine.

- **High-Temperature Solid Fuel.** The High-temperature Gas-cooled Reactor programs have developed graphite-matrix coated-particle fuels that are compatible with high-temperature fluoride salts. The U.S. Next Generation Nuclear Plant program (NGNP) has substantially improved these fuels.

- **Air-Brayton Combined Cycles.** Public and private investments of billions of dollars per year since the 1950s have made possible NACC with FIRES. No one in the 1960s would have predicted the extraordinary performance of modern utility combined cycle gas turbines. The FHR with these capabilities could not have exited 15 years ago—the turbine technology was not sufficiently advanced.
NACC as the enabling technology for a zero-carbon electricity grid imposes constraints on nuclear systems. The exit temperatures of the front-end gas compressors require delivery of high-temperature heat that efficiently couples with a salt coolant—the purpose for which these salt coolants were originally developed. If the goal is a zero-carbon electricity grid that requires NACC, salt-cooled systems become a preferred option because they can efficiently couple to NACC with FIRES. The FHR is the simplest nuclear energy technology using currently-existing technologies to couple with NACC—it is the near-term option. However, there are two longer-term options.

**Molten Salt Reactors (MSRs)**

The MSR was originally developed for the aircraft nuclear propulsion program and then further developed as a commercial machine using a Rankine steam cycle. That work was stopped in the early 1970s. Because the MSR uses fluoride salt coolants, it can couple to NACC and FIRES to produce variable electricity with base-load reactor operation for a zero-carbon electricity grid. MSRs can have thermal or fast neutron spectrums and can be designed to have breeding ratios greater than 1; that is, convert fertile $^{238}$U or $^{232}$Th into fertile $^{239}$Pu and $^{233}$U faster than these fertile fuels are consumed. They also have the potential to burn actinides. There have been major developments in MSRs in the last decade.

- **Fast spectrum MSRs.** The European Union$^{21}$ launched a program to reexamine the MSR. With modern computational techniques backed with laboratory experiments they were able to explore a very wide design space including potentially attractive fast spectrum versions of the MSR.
- **Materials.** New materials have the potential to simplify MSR design—particularly the processing of salt and the off-gas streams. For example, it was understood in the 1960s that salt processing could be dramatically simplified if fission products could be removed by distillation from the salt. The constraint was that the available materials of construction could not operate at the high required temperatures. New materials (carbon fiber, SiC, printer moly) may enable building distillation columns that operate at 1200°C. There are many other examples.

The FHR followed by the MSR is an alternative to the traditional nuclear power strategy of developing light-water reactors (LWRs) followed by sodium-cooled reactors for better fuel utilization. MSRs have the fuel dissolved in the fluoride salt coolant. The salt coolant has the additional requirement that the fuel be soluble in the salt. Because the fuel is dissolved in the coolant, the MSR relative to the FHR has the additional complications of fission products and actinides in the coolant. That implies an extremely radioactive coolant and the operational complications associated with such a coolant. It also implies added separations and radioactive waste management facilities associated with each reactor site to process the salt coolant and radioactive off-gases that are generated. A MSR would build upon FHR technology but require significantly more R&D, along with additional licensing and policy challenges. The Chinese Academy of Science plans to build a 10 MWt FHR test reactor by 2020 and a 2 MWt

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MSR. The strategy is to first commercialize the FHR with the longer-term goal to develop the MSR and its potential fuel cycle benefits in terms of fuel utilization.

In this context, there is one other reactor that could potentially couple to NACC—the lead-cooled fast reactor. However, currently there are no materials of construction compatible with lead at the required operating NACC operating temperatures. Like the MSR, there are a wide variety of fuel cycle options. In effect, they are functionally similar in terms of fuel cycle options. As with all reactors designed for better fuel utilization, the economics partly depend upon the price of natural uranium—the competition.

Fusion

There are classes of fusion reactors that use fluoride salts as coolants and to produce tritium. Fusion machines burn a mixture of tritium and deuterium. Deuterium, $^2\text{H}$, is available from water. Tritium must be produced—the classical method is neutron absorption in $^6\text{Li}$, a component of fluoride salts. The adoption of any of these salt-cooled fusion machines enables use of NACC and FIRES, plus all of the salt handling technology being developed for the FHR. Recent developments are of special note.

- **Ultra-high magnetic field fusion confinement systems.** Recent work on magnetically confined fusion systems using new superconductor materials (rare earth barium copper oxide [REBCO]) and superconductor joint technology indicate a potential pathway to a much smaller more-economical fusion machine\(^\text{22}\). It would reduce ITER in size by more than a factor of 10. The high magnetic fields and small machine imply very high power densities. Proposed designs would use fluoride salts as the coolant, primary shield material, and tritium generating blanket. The high power densities drive to liquid salt shielding because of the difficulty in controlling temperatures in solid shielding walls.

- **Fission fusion options.** Fusion generates massive quantities of neutrons creating the option to produce fissile fuels for fission reactors—an alternative to traditional sodium-cooled breeder reactors. The 14-Mev neutrons create from thorium $^{233}\text{U}$ with very high concentrations of $^{232}\text{U}$ ($^{232}\text{U}/^{233}\text{U} > 2.4\%$)—a form of denaturing $^{233}\text{U}$ to make it non-weapons useable\(^\text{23}\). Fission reactor neutrons are not sufficiently energetic to accomplish this goal. A large number of fission reactors could be supported by the fissile fuel from one fusion machine. It creates a new class of fuel cycle options.

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Conclusions

The FHR with its salt coolant, NACC and FIRES potentially opens up a long-term strategic option that includes MSRs and fusion while meeting the variable electricity requirements for a zero-carbon electricity grid. There are (1) challenges in developing an FHR, (2) larger technical, economic and institutional challenges for the MSR and (3) still larger technical and economic challenges for any of the fusion energy options. Our knowledge of fusion at this time is not sufficient to know the technical viability of fusion.