Advanced Nuclear Power Report Series

Fluoride-Salt-Cooled High-Temperature Test Reactor (FHTR): Goals, Options, Ownership, Requirements, Design, Licensing, and Support Facilities

Charles Forsberg, Lin-wen Hu, Joshua Richard, Rebecca Romatoski, Benoit Forget, John Stempien, Ron Ballinger, Kaichao Sun, and Dave Carpenter

MIT-ANP-TR-154

December 2014

Center for Advanced Nuclear Energy Systems

Massachusetts Institute of Technology
77 Massachusetts Avenue, 24-215
Cambridge, MA 02139-4307

(617) 452-2660
canes@mit.edu
mit.edu/canes

CANES

Abstract

This report discusses the path forward for a Fluoride-salt-cooled High-temperature Test Reactor (FHTR) including goals, requirements, ownership, design, and licensing. The goal is to develop and demonstrate the basic technology that would lead to the commercial development of a Fluoride-salt-cooled High-temperature Reactor (FHR). The FHTR is the test. In this context, it is similar to the DRAGON reactor (the first high-temperature gas-cooled reactor) and the Experimental Breeder Reactor I (the first sodium-cooled fast reactor). This is in contrast to test reactors that are used to test fuels and materials such as the High-Flux Isotope Reactor at Oak Ridge National Laboratory and the Advanced Test Reactor at Idaho National Laboratory. In those reactors the reactor is a tool to test materials but the reactor technology is proven technology.

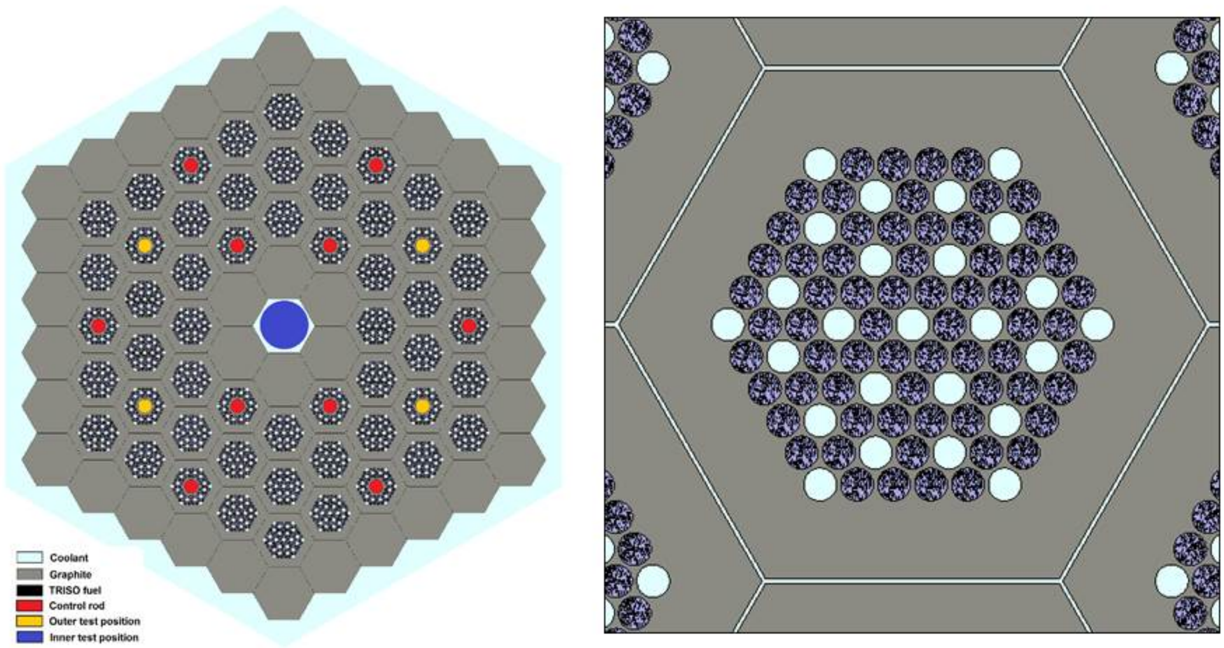
The FHR is a new reactor concept developed in the United States that combines (1) liquid salt coolant, (2) graphite-matrix coated-particle fuel originally developed for High Temperature Gas-cooled Reactors (HTGRs), (3) a nuclear air-Brayton combined cycle power system (NACC) adapted from natural-gas combined cycle plants and (4) Firebrick Resistance-Heated Storage (FIRES). The concept is enabled by advances in HTGR fuel and gas turbine technology.

The unique capabilities of the FHR create the incentives to develop it as a commercial reactor and the justification for building a FHTR as part of the development process. The FHR can produce base-load electricity using nuclear heat from the reactor and additional peak electricity using auxiliary natural gas or stored heat. Because the peak power heat input is added after nuclear heating (a topping cycle), the efficiency in converting natural gas or stored heat to electricity is 66%--far above stand-along natural-gas-fired power plants. Because the FHR can sell more electricity when the price is high, the plant revenues are 50 to 100% larger than a base-load nuclear plant. This is after subtracting the cost of the stored heat or auxiliary natural gas used to produce the peak electricity. The ability to operate the FHR as a base-load reactor with variable dispatchable output (including using stored heat for peak power) improves economics and makes the FHR an enabling technology for a zero-carbon nuclear-renewable electricity grid. The fuel and coolant combination enable avoidance of major fuel failures under accident conditions.

In the United States, the owner of the FHTR would be the federal government—all first-of-a-kind test reactors have been funded by governments. The time between the test reactor and a commercial product is too long for private funding. The FHTR would be designed to enable testing of alternative fuels and coolants to support multiple vendor designs and government missions. There are four funding options: (1) full U.S. government funding—the traditional model, (2) a joint partnership with China [planning to build 10 MWt FHTR by 2020] to jointly develop the FHR with a small FHTR in China followed by a more capable FHTR in the U.S., [3] a U.S. led international consortium to reduce the financial risks for each partner—a model that has worked in the past for test

reactors, and (4) a public-private partnership. The second and third funding options are considered realistic options at this time.

A FHTR is a two level design. At the facility level sufficient space is included to enable replacement of the entire reactor core. Like the Shippingport reactor that investigated alternative light-water reactor core designs in the 1960s, the option for major changes in the test reactor over its lifetime is maintained. At the reactor core level, the core is designed to test of alternative graphite-matrix coated-particle fuels and alternative fluoride-salt coolants. Figure A.1 shows the core and fuel assembly designs.



A.1 Top-down view of FHTR core with inner/outer radial reflectors and fuel assembly design (The gray region around the outside of the block is graphite, the purple-and-black cylinders in the interior of the assembly are the fuel compacts and the light-colored cylinders are the liquid salt coolant channels)

The U.S. regulatory structure for test reactors is different than for commercial reactors. Test reactors are required to have large safety margins but not required to follow commercial rules such as using code-qualified materials. That follows from the test reactor mission—testing new materials and concepts. A U.S. Department of Energy (DOE) test reactor can be licensed by DOE or at the request of DOE be licensed by the U.S. Nuclear Regulatory Commission. A review of FHTR licensing options by licensing experts at the test reactor workshop (Appendixes D and E) resulted in different opinions on the preferred licensing strategy for a FHTR; thus our recommendation that a separate study be conducted to identify the preferred licensing option.

The development and building of an FHR will require several significant supporting test facilities including a salt test loop in an existing materials test reactor, an integral non-nuclear high-temperature component test facility, non-nuclear NACC power system testing, and additional testing of the HTGR graphite matrix coated-particle fuel at higher power levels but lower temperatures.

ACKNOWLEGEMENTS

We would like to thank the U.S. Department of Energy and Idaho National Laboratory for their support of this work through the Nuclear Energy University Program, Oak Ridge National Laboratory, Westinghouse Electric Company and our advisory panel, the participants of the 6th FHR Workshop on FHR test reactors, and the many reviewers of this report.

Canes Publications

Topical and progress reports are published under seven series:

Advances in Nuclear Energy Disciplines (ANED) Series
Advanced Nuclear Power Technology (ANP) Series
Nuclear Fuel Cycle Technology and Policy (NFC) Series
Nuclear Systems Enhanced Performance (NSP) Series
MIT Reactor Redesign (MITRR) Series
Nuclear Energy and Sustainability (NES) Series
Nuclear Space Applications (NSA) Series

Please visit our website (mit.edu/canes/) to view more publication lists.

MIT-ANP-TR-154	C. Forsberg, L-W. Hu, J. Richard, R. Romatoski, B. Forget, J. Stempien, and R. Ballinger, **Fluoride-Salt-Cooled High-Temperature Test Reactor (FHTR): Goals, Options, Ownership, Requirements, Design, Licensing, and Support Facilities** (2014).
MIT-ANP-TR-153	C. Forsberg, D. Curtis, J. Stempien, R. MacDonald and P. Peterson, **Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy**. **A Fluoride-Salt-Cooled High-Temperature Reactor (FHR) with a Nuclear Air-Brayton Combine Cycle (NACC) and Firebrick Resistance-Heated Energy Storage (FIRES)** (2014).
MIT-ANP-TR-152	A. Briccetti, J. Buongiorno, M. Golay, N. Todreas, **Siting of an Offshore Floating Nuclear Power Plant** (2014).
MIT-ANP-TR-151	M.J. Minck, and C. Forsberg, **Preventing Fuel Failure for a Beyond Design Basis Accident in a Fluoride Salt Cooled High Temperature Reactor** (2014).
MIT-ANP-TR-150	YH. Lee, T. McKrell, and M.S. Kazimi, **Safety of Light Water Reactor Fuel with Silicon Carbide Cladding** (2014).
MIT-ANP-TR-149	Y. Sukjai, E. Pilat, K. Shirvan, and M.S. Kazimi, **Silicon Carbide Performance as Cladding for Advanced Uranium and Thorium Fuels for Light Water Reactors** (2014).
MIT-ANP-TR-148	D.A. Bloore, E. Pilat, and M.S. Kazimi, **Reactor Physics Assessment of Thick Silicon Carbide Clad PWR Fuels** (2013).

MIT-ANP-TR-147 C. Forsberg, L-wen. Hu, P.F. Peterson, and K. Sridharan, **Fluoride-Salt-Cooled High-Temperature Reactors (FHRs) for Base-Load and Peak Electricity, Grid Stabilization, and Process Heat,** (January 2013).

MIT-ANP-TR-146 Tingzhou Fei, E. Shwageraus, and M. J. Driscoll, **Innovative Design of Uranium Startup Fast Reactors** (November 2012).

MIT-ANP-TR-145 K. Shirvan and M.S. Kazimi, **Development of Optimized Core Design and Analysis Methods for High Power Density BWRs** (November 2012).

MIT-ANP-TR-144 G.L. DeWitt, T. McKrell, L-W Hu, and J. Buongiorno, **Investigation of Downward Facing Critical Heat Flux with Water-Based Nanofluids for In-Vessel Retention Applications** (July 2012).

MIT-ANP-TR-143 C. Forsberg, L-Wen Hu, P.F. Peterson, and T. Allen, **Fluoride-Salt-Cooled High-Temperature Reactors (FHRs) For Power and Process Heat. Advanced Nuclear Power Program** (January 2012).

MIT-ANP-TR-142 J. DeWitte and N.E. Todreas, **Reactor Protection System Design Alternatives for Sodium Fast Reactors** (September 2011).

MIT-ANP-TR-141 G. Lenci, N.E. Todreas, M.J. Driscoll and M. Cumo, **Alternatives for Sodium Fast Reactor Cost-Effective Design Improvements** (September 2011).

MIT-ANP-TR-140 M.R. Denman, N.E. Todreas and M.J. Driscoll, **Probabilistic Transient Analysis of Fuel Choices for Sodium Fast Reactors** (September 2011).

MIT-ANP-TR-139 R.P. Arnold, T. McKrell, and M.S. Kazimi, **Vented Silicon Carbide Oxidation in High Temperature Steam** (September 2011).

MIT-ANP-TR-138 F. Vitillo, N.E. Todreas, M.J. Driscoll, **Vented Inverted Fuel Assembly Design for an SFR** (June 2011).

MIT-ANP-TR-137 B. Truong, L-W Hu, J. Buongiorno, T. McKrell, **Effects of Surface Parameters on Boiling Heat Transfer Phenomena** (June 2011).

MIT-ANP-TR-136 J. Dobisesky, E.E. Pilat, and M. S. Kazimi, **Reactor Physics Considerations for Implementing Silicon Carbide Cladding into a PWR Environment** (June 2011).

MIT-ANP-TR-135 J.D. Stempien, D. Carpenter, G. Kohse, and M. S. Kazimi, **Behavior of Triplex Silicon Carbide Fuel Cladding Designs Tested Under Simulated PWR Conditions** (June 2011)

MIT-ANP-PR-134 M.S. Kazimi, J. Dobisesky, D. Carpenter, J. Richards, E. E. Pilat, and E. Shwageraus, **Feasibility and Economic Benefits of PWR Cores with Silicon Carbide Cladding** (April 2011).

MIT-ANP-TR-133 R.C. Petroski and B Forget, **General Analysis of Breed-and-Burn Reactors and Limited-Separations Fuel Cycles** (February 2011).

MIT-ANP-TR-132 D. M. Carpenter and M. S. Kazimi, **An Assessment of Silicon Carbide as a Cladding Material for Light Water Reactors** (November 2010)

MIT-ANP-TR-131 Michael P. Short and Ronald G. Ballinger, **Design of a Functionally Graded Composite for Service in High Temperature Lead and Lead-Bismuth Cooled Nuclear Reactors** (October 2010)

MIT-ANP-TR-130 Yu-Chih Ko and Mujid S. Kazimi, **Conceptual Design of an Annular-Fueled Superheat Boiling Water Reactor** (October 2010)

MIT-ANP-TR-129 Koroush Shirvan and Mujid S. Kazimi, **The Design of a Compact, Integral, Medium-Sized PWR: The CIRIS** (May 2010)

MIT-ANP-TR-128 Tingzhou Fei and Michael Golay, **Use of Response Surface for Evaluation of Functional Failure of Passive Safety System** (March 2010)

MIT-ANP-TR-127 Rui Hu and Mujid S. Kazimi , **Stability Analysis of the Boiling Water Reactor: Methods and Advanced Designs** (March 2010).

MIT-ANP-TR-126 Paolo Ferroni and Neil E. Todreas, **An Inverted Hydride-Fueled Pressurized Water Reactor Concept** (October 2009).

MIT-ANP-TR-125 M.S. Kazimi, P. Hejzlar, Y. Shatilla, Bo Feng, Yu-Chih Ko, E. Pilat, K. Shirvan, J. Whitman, and A. Hamed, **A High Efficiency and Environmentally Friendly Nuclear Reactor (HEER) for Electricity and Hydrogen** (October 2009).

MIT-ANP-TR-124 Joshua J. Whitman and Mujid S. Kazimi, **Thermal Hydraulic Design Of A Salt-Cooled Highly Efficient Environmentally Friendly Reactor** (August 2009).

MIT-ANP-TR-123 Matthew J. Memmott, Pavel Hejzlar, and Jacopo Buongiorno, **Thermal-Hydraulic Analysis of Innovative Fuel Configurations for the Sodium Fast Reactor** (August 2009).

MIT-ANP-TR-122 Sung Joong Kim, T. McKrell, J. Buongiorno, Lin-wen Hu, **Subcooled Flow Boiling Heat Transfer and Critical Heat Flux in Water-Based Nanofluids at Low Pressure** (April 2008).

MIT-ANP-TR-121 M. Memmott, J. Buongiorno, and P. Hejzlar, **Development and Validation of a Flexible RELAP5-3D-Based Subchannel Analysis Model for Fast Reactor Fuel Assemblies**(December 2008).

MIT-ANP-TR-120 Jiyong Oh and M.W. Golay, **Methods for Comparative Assessment of Active and Passive Safety Systems** (February 2008).

MIT-ANP-TR-119 M. J. Driscoll, **Comparative Economic Prospects of the Supercritical CO_2 Brayton Cycle GFR** (February 2008).

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

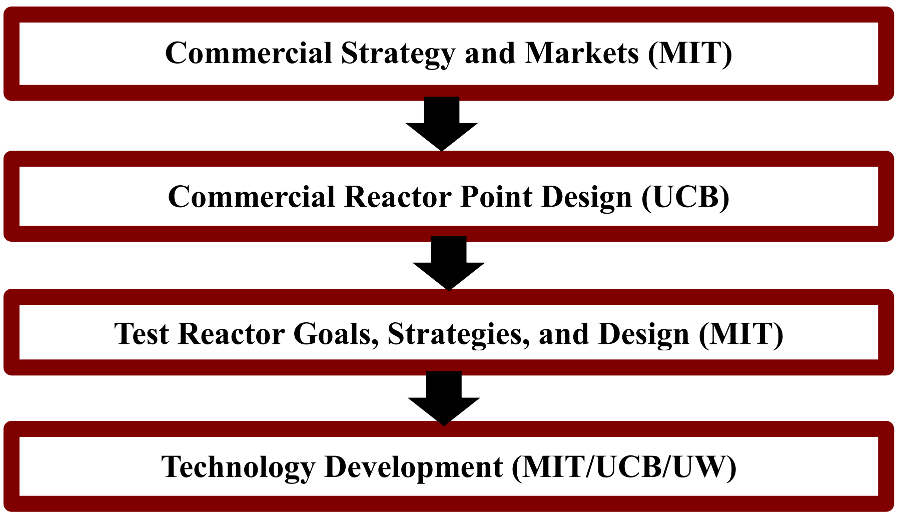


Fig. PP.1 Structure of FHR Project

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. This report is the Test Reactor Goals, Strategies, and Design report. The three reports are:

- *Commercial Strategy and Electricity Markets*: Charles Forsberg, Daniel Curtis, John Stempien, Ruaridh MacDonald, and Per. F. Peterson, *Fluoride-salt-cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy*, MIT-ANP-TR-153, Massachusetts Institute of Technology, Cambridge, MA, December 2014.
- *Commercial Reactor Point Design*: Charalampos "Harry" Andreades, Anselmo T. Cisneros, Jae Keun Choi, Alexandre Y.K. Chong, Massimiliano Fratoni, Sea Hong, Lakshana R. Huddar, Kathryn D. Huff, David L. Krumwiede, Michael R. Laufer, Madicken Munk, Raluca O. Scarlat, Nicolas Zweibaum, Ehud Greenspan, Per F. Peterson, *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 Goals, Strategy, and Design (This report)*: Charles Forsberg, Lin-wen Hu, John Richard, Rebecca Romatoski, Benoit Forget, John Stempien, Ron Ballinger, Kaichao Sun, and Dave Carpenter, *Fluoride-salt-cooled High-temperature Test Reactor (FHTR): Goals, Options, Ownership, Requirements, Design, Licensing, and Support Facilities*, MIT-ANP-TR-154, Massachusetts Institute of Technology, Cambridge, MA, December 2014.

Executive Summary

Fluoride-Salt-Cooled High-Temperature Test Reactor (FHTR): Goals, Options, Ownership, Requirements, Design, Licensing, and Support Facilities

This report summarizes goals, options, ownership, requirements, design, licensing, and support facilities required for an FHTR. After the draft report was prepared, it was followed by a workshop of international experts in test reactors and reactor development. The workshop participants received copies of the draft report before the workshop. The results of the workshop were used in preparation of the final report. The workshop presentations and conclusions are included in several appendixes.

Fluoride-salt-cooled High-Temperature Reactor (FHR)

No FHR has ever been built. If the FHR is to be commercialized, a FHTR is required. The prerequisite for building an FHTR is a strong set of incentives to commercialize the FHR that provide the driving force for such investments. The other requirement is to understand the key characteristics of the FHR that need verification to help define the test goals for the FHTR.

The FHR is a new reactor concept that combines (1) liquid salt coolant, (2) graphite-matrix coated-particle fuel originally developed for High Temperature Gas-cooled Reactors (HTGRs), (3) a nuclear air-Brayton combined cycle power system (NACC) adapted from natural gas combined cycle plants and (4) Firebrick Resistance-Heated Energy Storage (FIRES). The concept is a little over a decade old and enabled by advances in high-temperature gas-cooled reactor (HTGR) fuel and 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 designed to provide high-temperature heat 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 utility combined-cycle gas turbine for commercial power generation. Advances in fuel from the Next Generation Nuclear Plant (an HTGR) have developed the fuel to make an FHR viable.

The coupling of NACC to the FHR enables an FHR to operate as a base-load nuclear plant or in a peaking mode with auxiliary natural gas, jet fuel, or stored heat. In base-load mode, the plant efficiency is 42%. The plant can produce peak electricity by the using auxiliary natural gas or hydrogen or stored heat to raise gas turbine temperatures after nuclear heating of the compressed air. The auxiliary fuel to electricity efficiency is 66%--greater than the best stand-alone natural-gas-fired combined cycle plant. The high efficiency is because the auxiliary heat is added after nuclear preheating of the compressed air to 670°C—a topping cycle. With these capabilities the FHR can produce added electricity at times of high electricity prices more efficiently than any other technology. Using California and Texas hourly wholesale electricity rates and natural gas as the fuel for peak electricity production, the revenue is 50% higher than a base-load nuclear plant in these markets.[1] This revenue is after subtracting the cost of natural gas used for peak electricity production. At higher natural gas prices the revenue is double that of a base-load nuclear plant.

In a future zero-carbon world, it is also the most efficient method to convert hydrogen or biofuels to electricity for peak electricity production. Alternatively, FIRES can be used for peak power production. The firebrick is heated using electricity when electricity prices are less than natural gas to provide high-temperature heat for peak electricity production. In electricity systems with large capacities of wind or solar, electricity prices are depressed and may approach zero when there are good wind or solar conditions. The round trip electricity-to-heat-to-electricity efficiency is 66%. For a zero-carbon nuclear renewable electricity grid, the need is for economic dispatchable power to match electricity production with demand. The FHR with NACC and FIRES can meet that need.

Need for FHTR

The FHR is a new reactor concept. A test reactor is required to (1) demonstrate technical viability, (2) provide the data required for design and licensing of a pre-commercial demonstration plant, (3) provide operational and maintenance experience and (4) provide a test bed for alternative fuels, fluoride salt coolants, and other systems. Test reactors can be divided into two classes. The goal of a Class I test reactor is to develop the reactor technology. Dragon (the first high-temperature gas-cooled reactor) and Experimental Breeder Reactor I (the first sodium fast reactor) are examples of Class I test reactors. The FHTR would be a Class I test reactor. Class I test reactors can be further categorized by goals. Class I-A test reactors are designed for versatility to allow testing of a wide set of fuels or other system features whereas a Class I-B test reactor is designed to lead to a specific design of commercial or special purpose reactor. No Class I test reactor has been built in several decades anywhere in the world. Class II test reactors are neutron irradiation sources to test materials and fuels. HFIR at ORNL and ATR at INL are Class II test reactors. A goal of a Class II test reactor designer is to minimize risk in the reactor design.

This report is a first effort to define what is required for a FHTR including: goals, options, ownership, requirements, design, licensing, and support facilities. Definition of goals, design options, ownership, and requirements are required input to design a test reactor—an essential part of reactor design. There are several challenges.

---

[1] C. Forsberg et al., *Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy*, MIT-ANP-TR-153, Massachusetts Institute of Technology, December 2014.

- *Experience base*. No new reactor concept has been tested anywhere in the world for over 40 years. In that time conditions have changed.
- *Licensing.* A new test reactor if built by the U.S. Department of Energy (DOE) can be licensed by DOE or the U.S. Nuclear Regulatory Commission. All test reactors build in the U.S. were built before the current regulatory structure was created. A safety licensing strategy is required.
- *Chinese Academy of Sciences (CAS).* Three years ago, the CAS announced that it plans to develop an FHR and have a 10 MWt FHTR startup by 2020. This raises the important policy question of what is the relationship of any U.S. program to the CAS and whether a strategy can be developed that keeps open options for (1) an integrated program with the CAS or (2) and independent program. Central to that strategy is the strategy for an FHTR.

FHTR Ownership and Top-Level Goals

Class I FHTRs can be broken into two sub-classes as defined by goals: (1) a Class I-A general purpose FHTR allowing testing of different fuel, coolant salt, and system options and (2) a Class I-B FHTR to provide the information for design and licensing a specific design of pre-commercial FHR. The choice of goals depends upon ownership. The Chinese FHTR will be a Class I-B FHTR focused on addressing questions to build a larger pebble-bed FHR. In the United States, the owner of the FHTR would be the federal government based on the following considerations.

- *Vendor response*. Discussions with vendors indicate that the time to develop a new reactor concept starting at the test reactor stage is beyond the time line for any commercial organization. Commercial funding becomes possible after technical viability is demonstrated and when one transitions from a test reactor to a pre-commercial demonstration reactor
- *Government roles*. All first-of-a-kind reactors have been funded by governments. The Energy Policy Act that authorized the Next Generation Nuclear Plant (NGNP) explicitly recognizes that a test reactor is a government responsibility.

If the U.S. government funds a test reactor, it would likely be a Class I-A general-purpose FHTR that can test different fuel, coolant, and systems options. This is based on the following considerations.

- *Competitive vendors*. The U.S. government as a national policy has always supported at least two vendors for each advanced reactor concept. This was true for light water reactors (LWRs), sodium fast reactors (SFRs), and HTGRs. It is true for the current small modular reactor program funded by DOE. Competitive vendors imply alternative designs. That requires a test reactor that is capable of testing key components of alternative commercial FHR designs.

- *Government missions*. The FHR with a nuclear air Brayton combined cycle is an option for some government missions such as providing electricity and heat to remote sites and surface naval propulsion. The unique advantage it brings to such missions is that the reactor can be sized for average power demand but with the addition of jet fuel for meeting peak power demand. This combination (1) can substantially reduce total reactor cost and (2) enable very high peak power capabilities. If the base power level is 100 MWe, auxiliary fuel can increase output by an additional 142 MWe. Potential government missions require the capability to test a wide variety of fuels and systems because some of the requirements may be very different than a commercial power plant.
- *Institutional challenges*. There are several strategies for developing a new reactor. The first is a series of test reactors such as was done in the development of fast reactors. The Experimental Breeder Reactor I (EBR-I) was built and quickly followed by the Experimental Breeder Reactor II (EBR-II) and later by the Fast-Flux Test Reactor (FFTF). Alternatively, more general purpose facilities can be built. An example of this is the Shippingport pressurized water reactor (PWR) where three totally different reactor cores were tested in a single facility. The U.S. government today has difficulties in funding and licensing major projects. Such constraints favor the second strategy with fewer steps.

Four FHTR ownership strategies were evaluated

- *U.S. government*. This is the traditional model.
- *U.S. foreign partnership such as with China*. China will build a small 10 MWt FHTR by 2020. The U.S. could be a partner in this reactor or in an expanded joint program with a U.S. FHTR.
- *U.S. led international consortium*. There is a successful history of international consortiums to fund test reactors. DRAGON, the first high-temperature gas-cooled reactor, was an international consortium led by the United Kingdom with the U.S. as a partner. LOFT, the Loss of Flow Test reactor, was an international consortium led by the United States to do safety tests on LWR emergency core cooling systems. Several existing reactors such as Halden[2] in Norway are operated as international consortiums. Today there is one new reactor that is being built in France as an international consortium—the Jules Horowitz Reactor led by the French in cooperation with research institutes of Spain, Belgium, Czech Republic, Finland, Israel, India, Japan, and the U.K. and utilities and industrial partners[3]. This is a 100 MWt water-cooled general purpose test reactor. The success of several such international consortiums is based on several factors.

[2] http://www.oedc-nea.org/jointproj/halden.html
[3] http://www.cad.cea.fr/fjh/index.html

    - *Budget*. These are one to two billion dollar projects where the cost is spread over five to ten partners. The financial burden is limited for each partner, although the host country may have financial liabilities if the project is significantly over budget.
    - *Project management*. The historical track record for this size project has been good. In contrast, the track record for 10 billion dollar international projects has been poor (ITER, Space Station). This may reflect the difficulties of managing such large projects that span a long duration.
    - *Commercial considerations*. Test reactor projects are early in the development of a new reactor concept. The strong commercial drivers and the challenges of managing intellectual property rights are limited. However, once it reaches the stage of a pre-commercial demonstration project, this becomes a major challenge.
- *Public-private partnership*. This would be a joint effort by the U.S. government and industry. There are no examples of such partnerships for first-of-a-kind Class-I test reactors.

Our analysis leads to the recommendation that the U.S. should build a Class I general-purpose FHTR as an international project based on the following considerations:

- *Financial constraints*. An international consortium reduces the financial risk to each party and brings other partners with associated technologies and capabilities into the project.
- *Coupling with the Chinese Academy of Science (CAS).* The CAS plans to have a 10 MWt FHTR operational by 2020. This Chinese FHTR is designed to lead to a specific design of pre-commercial FHR. There are large incentives for a FHR program that (1) can be a cooperative program with the CAS to accelerate the technology or (2) independent of the CAS because of the unpredictability of international relations. A general-purpose international FHTR enables both options to remain open. Working with the CAS FHTR would provide valuable data for design of a more capable general-purpose FHTR. For the CAS there would be large incentives to be a major partner in a general-purpose FHTR because such a machine could test coolants, materials, and fuels under much more extreme conditions (including neutron flux) to strengthen the design, safety, and licensing case for a commercial FHR. It would also provide backup options if major problems were identified in the development pathway they had chosen. In effect, the goal is a win-win strategy for all partners for alternative futures.
- *Broader international community support.* The development of a new reactor with transformational capabilities is a large undertaking. Wider participation can increase the potential for long-term success in providing greater access to technological and financial resources.

Test Reactor Pre-Conceptual Design

A preliminary FHTR design has been developed that meets the requirements for a Class I-A test reactor. It is a two level design to allow flexibility to test many alternative FHR fuels, coolants, and designs. At the facility level sufficient space is included to enable replacement of the entire reactor core if it is desired to test different reactor core designs. Like the Shippingport reactor, the option for major changes in the test reactor is maintained. This feature may or may not be used.

At the reactor core level, the core is designed to enable testing of alternative graphite-matrix coated-particle fuels and alternative fluoride-salt coolants. Figure S.1 shows the core design, Figure S.2 shows the fuel assembly design, and Figure S.3 shows the thermal neutron flux. The design includes the following features.

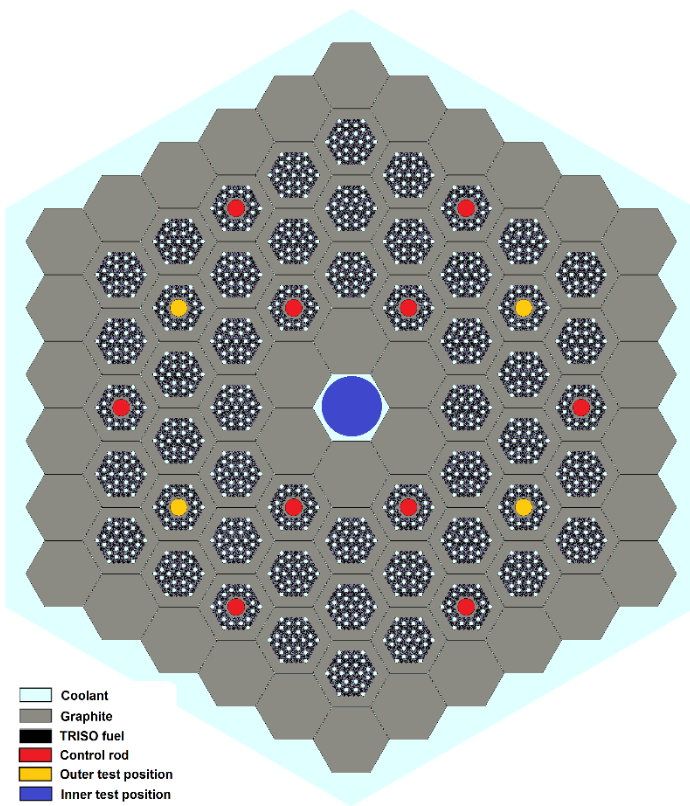


Fig. S.1 Top-down view of FHTR core and inner/outer radial reflectors

- *Core design*. The reactor core has driver fuel and a flux trap in the center experimental hole to enable high thermal fluxes to fuels for fuel tests. The fuel blocks can be assembled in a variety of configurations to meet experimental needs.
- *Fuel assembly*. The fuel assembly is a prismatic block similar to that used in the Ft. St. Vrain HTGR. This is a proven HTGR fuel that allows three dimensional variations in

enrichment and particle packing fractions to meet neutron flux design goals. The assembly design, termed Fuel Inside Radial Moderator (FIRM), places fuel compacts and cooling channels inside a region of solid graphite moderator. The solid graphite regions introduce spatial self-shielding of the fuel resonances, increasing the resonance escape probability of the neutrons born in fission which in turn increases core reactivity. It also implies that the smaller test positions inside a single assembly have a relatively hard neutron flux for accelerated materials testing.

- *Coolant salt*. The FHTR is designed to operate with different coolant salts from $^{7}Li_2BeF_4$ (flibe) to NaF-ZrF_4 coolant salt. There are four somewhat similar candidate coolant salts for an FHR. The NaF-ZrF_4 has the lowest cost but poorest neutronic and thermal hydraulic behavior. Flibe has the best neutronic and thermal hydraulic behavior but is more costly because of the need to enrich ^{7}Li. The expectation is that the FHTR over its operational lifetime would operate on several different salts as part of its test program. There are complex operational and cost tradeoffs on salt coolants.

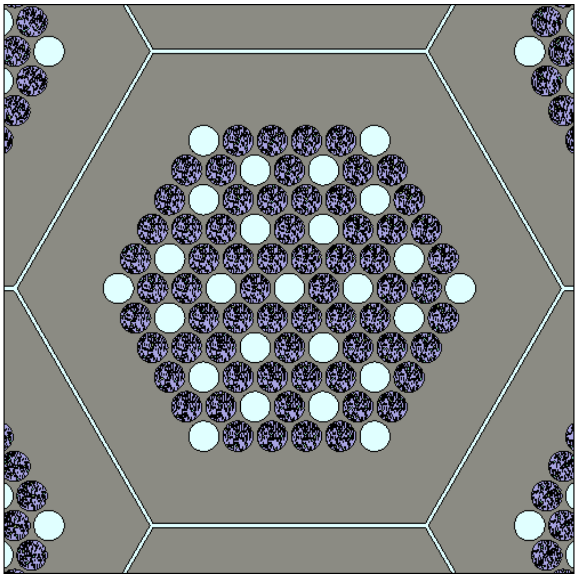


Fig. S.2 Top-down view of the FHTR standard fuel assembly. The gray region around the outside of the block is solid graphite, while the purple-and-black cylinders in the interior of the assembly are the fuel compacts. The light cyan colored cylinders are the liquid salt coolant channels.

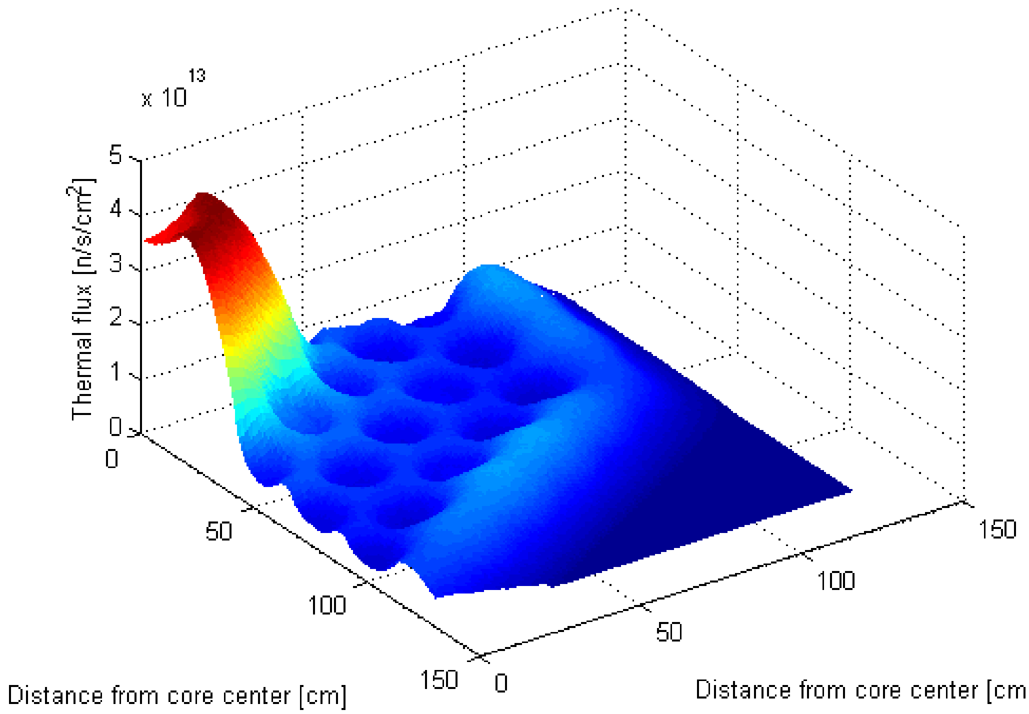


Fig. S.3 Test Reactor Thermal Flux Profile

Regulatory Strategy

In the United States the regulatory structure for test reactors is different than for commercial reactors. Test reactors are required to have large safety margins but not required to follow commercial rules such as using code-qualified materials. That follows from the test reactor mission—testing new materials and concepts. 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. The current DOE policy is for the NRC to license new DOE reactors. A review of FHTR licensing options by licensing experts at the test reactor workshop (Appendixes D and E) resulted in different opinions on the preferred licensing strategy for a Class I test reactor.

The different licensing perspectives for a Class I test reactor follow from the fundamental differences in Class I reactors versus other reactor types. Most reactors are designed to produce a product: electricity, heat, neutrons for medical isotope production and neutrons for research (Class II test reactors). Those missions imply that there will be few changes in such reactors over time—the reactor is licensed with the expectation it will operate with few changes for decades. For commercial reactors, the NRC licenses a design with the expectation that multiple reactors of that type will be built. In contrast the goal of a Class I test reactor is to test the reactor technology. As a consequence there is the expectation that the reactor will be modified over the course of time as the need arises to examine alternative fuels, coolants, control strategies, etc. That implies a different safety and licensing strategy.

In the United States different licensing strategies are used for reactors with different missions. The responsibility for licensing is different for commercial reactors (NRC), navy reactors (U.S. Navy), and nuclear space power systems (NASA) that reflect different goals and environments. There appears to be no difference in the levels of safety. In this context the DOE licenses many one-of-a-kind nuclear facilities. This leads to our recommendation to evaluate the licensing options for the FHTR to determine the preferred option for an FHTR license based on assuring safety and meeting reasonable schedule requirements. Because no Class I test reactor has been built in over 40 years, there has been no incentive to examine licensing strategies for such reactors.

Other Major Test Facilities

The development and building of an FHR will require several significant test facilities. These facilities would also likely be used for development of a pre-commercial FHR demonstration reactor. A partial list of these facilities is provided below.

- *Salt test loop in existing test reactor*. A full-scale salt test loop in an existing Class-II test reactor would enable testing of driver fuel, chemistry control strategies including those for tritium and transient fuel behavior.
- *Integrated test facility*. There are multiple mechanical systems that must operate in salt at 700°C. This implies the need for a non-nuclear test facility using the same salt at temperature to develop and test systems before startup of the FHTR, to test new systems before being tested in the FHTR, and to provide training. This could be a non-nuclear version of the FHTR. Integral or separate with such a facility is the need for an integrated thermal hydraulics test facility to benchmark the computer codes for safety.
- *Nuclear-Air Brayton Combined Cycle (NACC) power system*. Components of this system, particularly the salt-to-air heat exchanger, need to be demonstrated in a non-nuclear test facility
- *Higher-power-density coated-particle fuel tests*. The optimum FHR has a higher power density than an HTGR. Confirmatory testing of coated particle fuels at higher power densities is needed. This could be done in existing test facilities developed for high-temperature gas-cooled reactors.

Conclusions

The FHR is a new reactor concept; thus, a test reactor is required. The starting point is the definition of ownership, technical requirements, and licensing strategy as inputs to the design of a test reactor. These inputs drive the FHTR design. Changing any of these inputs will result in changes in test reactor design. This report lays out a proposed basis for selection of these starting conditions as well as a preliminary FHTR design.

Table of Contents

List of Figures

List of Tables

1 Introduction

The goal of this report is to address the major aspects of an FHTR. This is required because factors such as reactor ownership drive top level design criteria. One can't design a Class-I test reactor without addressing these issues. We start by providing a short description of what an FHR is, the development pathway to a test reactor, and the contents of this report.

1.1 Concept Description

The Fluoride-salt-cooled High-temperature reactor (FHR) is a graphite-moderated, fluoride-salt-cooled low-enriched-uranium thermal-neutron-spectrum reactor[4]. The concept is about a decade old[5]. No FHR has been built and thus a Fluoride-salt-cooled High-temperature Test Reactor (FHTR) will be required before a pre-commercial power reactor is built.

The Massachusetts Institute of Technology (MIT), the University of California at Berkeley (UCB) and the University of Wisconsin (UW) have a DOE-sponsored joint R&D project to develop a conceptual design of an FHR, a conceptual design of an FHTR, and a roadmap to a commercial machine. The FHR design[6] has three goals: (1) 50% to 100% increase in revenue relative to base-load nuclear power plants with capital costs similar to existing LWRs, (2) the enabling technology for a zero-carbon nuclear-renewable grid, and (3) no major fuel failures and thus no major radionuclide releases in a beyond design basis accident (BDBA). The development of a new reactor is a major undertaking. It will not happen unless there is a compelling economic (utility) and societal (government) case as defined by goals.

The baseline FHR fuel consists of tri-structural isotropic (TRISO) coated-particle fuel embedded in 3-centimeter diameter graphite spheres—the same basic fuel used in pebble-bed HTGRs. There are alternative fuel geometries but all consist of coated-particle TRISO fuel in a graphite matrix. The primary coolant is a lithium-beryllium-fluoride salt known as flibe ($^{7}Li_2BeF_4$). There are alternative candidate fluoride salt coolants. 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. The characteristics of the fuel and coolant provide a unique safety case. The basis to prevent major fuel failures in a BDBA is described in several papers[7, 8].

---

[4] C. FORSBERG, D. Curtis, J. Stempien, R. MacDonald, P. Peterson, *Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy*, MIT-ANP-TR-153, Massachusetts Institute of Technology, December 2014.

[5] C. W. FORSBERG, P. S. Pickard and P. F. Peterson, "Molten-Salt-Cooled Advanced High-Temperature Reactor for Production of Hydrogen and Electricity", *Nuclear Technology,* 144**,** pp. 289-302 (December 2003).

[6] C. ANDREADES et al., *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. (2014).

[7] M. J. MINCK, M. P. Short and C. W. Forsberg "Fluoride-salt-cooled High-Temperature Reactor Severe Accident Strategy with Vessel Insulation Designed to Fail," Paper No. FA244, *Proc. of ICAPP 2013,* Jeju Island, Korea, April 14-18, 2013.

The FHR is coupled to a nuclear air Brayton combined cycle (NACC) that operates on nuclear heat with the capability to produce peak power using auxiliary natural gas, jet fuel, stored heat or ultimately hydrogen. This power cycle is similar to a natural-gas combined cycle plant where air is compressed, heated, flows through a turbine to generate electricity and exhausted to a heat recovery steam generator (HRSG), In the HRSG, steam is produced that generates added electricity. The base-load thermal efficiency is 42% with a cooling water demand 40% of a LWR per unit of electricity. The low cooling water demand is because of (1) higher efficiency and (2) an air Brayton combined cycle where much of the heat is rejected as hot air from the HRSG. Peak power is produced by using auxiliary natural gas, another fuel, or stored heat to raise the compressed air temperature after nuclear heating (~670°C) to over 1100°C before entering the turbine. When producing peak electricity, the auxiliary heat to electricity conversion efficiency is 66%--above that of a stand-alone natural gas plant because the auxiliary heat source acts as a topping cycle above the nuclear-heated air.

In deregulated markets such as California and Texas this power cycle increases plant revenue[9] by more than 50% relative to base-load nuclear plants by producing added peak electricity when electricity prices are high. Revenue is defined as revenue from sales of electricity minus the cost of the natural gas or other source of heat used to produce the added peak electricity. The analysis used current low natural gas prices. The revenue advantage relative to base-load plants increases with natural gas prices.

The power cycle imposes two requirements on the FHR: (1) delivery of salt coolant at or above 700°C and (2) control of tritium and other radionuclides to avoid transfer to compressed air in NACC.

1.2 Development Pathway

To develop a path forward for the FHR we adopted a top-down strategy (Fig. 1) that defined goals that led to the commercial reactor concept as described previously. The commercial reactor design specifics the minimum goals for the test reactor. However, the test reactor may have multiple owners and customers that impose requirements beyond those needed for a single design of a commercial FHR. The test reactor has three goals that can’t be achieved by simulations, laboratory test loops, and irradiations of fuel or materials in existing test reactors:

- Demonstrate the technical viability of an FHR.

[8] M. J. MINCK and C. W. Forsberg, “Preventing Fuel Failure for a Beyond Design Basis Accident in a Fluoride Salt Cooled High Temperature Reactor,” Paper 14119, *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, North Carolina, April 6-9, 2014.

[9] C. W. FORSBERG and D. Curtis, “Meeting the Needs of a Nuclear-Renewable Electrical Grid with a Fluoride-Salt-Cooled High-Temperature Reactor Coupled to a Nuclear Air-Brayton Combined Cycle Power System, *Nuclear Technology* (March 2014).

- Provide the required information for design and licensing of a commercial demonstration FHR.
- Provide the test bed for different fuels, salt coolants, and materials that may be part of the commercial plant design.

The test reactor does not determine the economic viability of the concept. That is determined by a larger pre-commercial demonstration reactor.

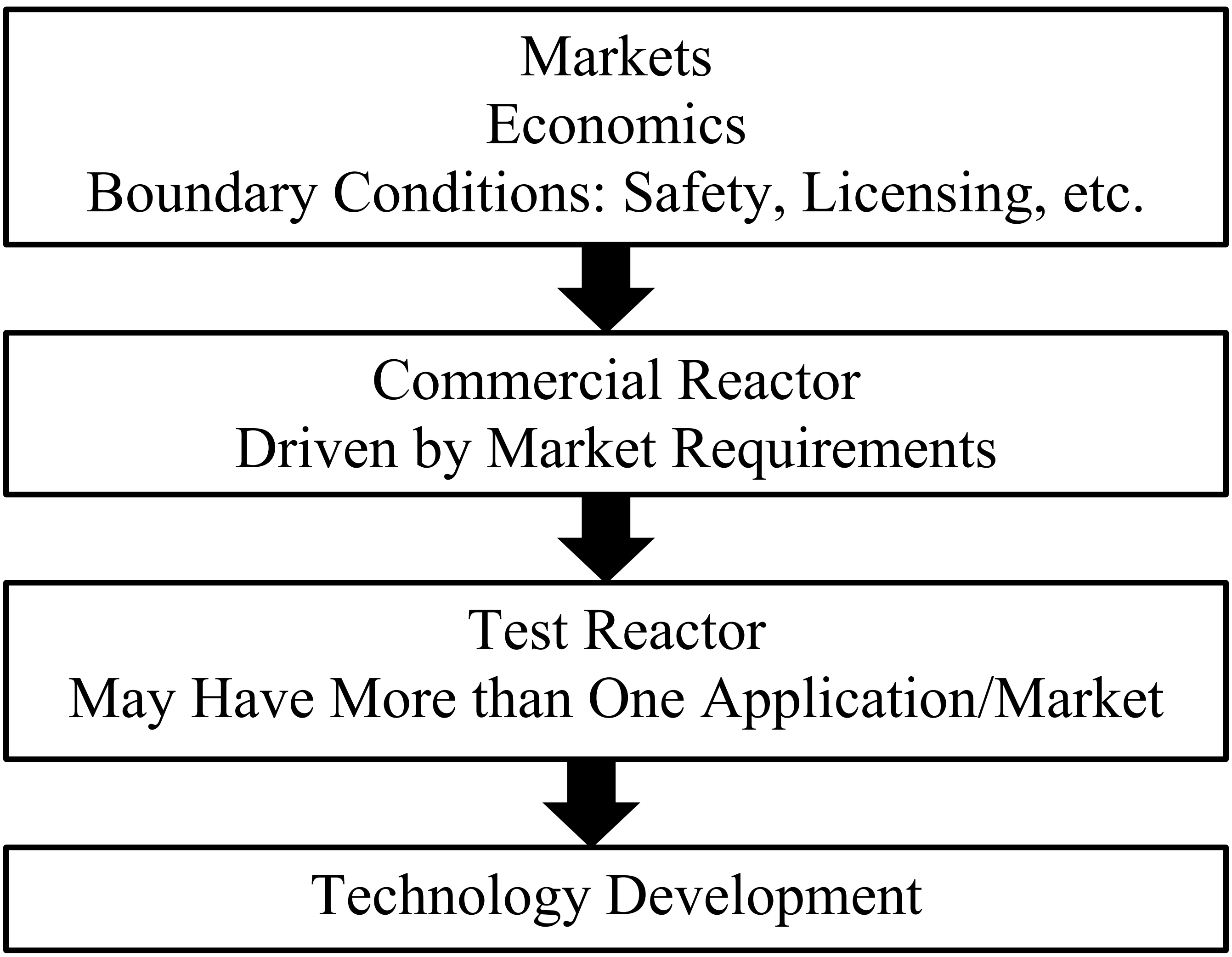


Fig. 1.1 Strategy to Develop a Path Forward for the FHR

1.3 Test Reactor Report

To prepare this report, a three step process was used. A draft report was prepared with recommendations and conclusions. This was followed by a test reactor workshop with international experts to discuss key questions associated with development and design of a test reactor. The draft report was sent to participants in advance of the workshop. The final report was prepared partly based on the results of the workshop. Major workshop conclusions are in Appendix D and the workshop agenda, attendance, and presentations are in Appendix E. In some areas there was consensus but in other areas there were different perspectives—both are reported. This report provides a first roadmap to a test reactor. The report has eight chapters.

- *Chapter 2: Commercial Reactor Design*. This chapter describes the family of FHR concepts and the commercialization bases. The viable design option space for the FHR ultimately drive the requirements for a test reactor
- *Chapter 3: Test Reactor Markets and Owners*. Who owns the test reactor will define many of the top-level design requirements. The different types of ownership and the implications of that ownership on test reactor requirements are described.
- *Chapter 4: Test Reactor Requirements*. The requirements that led to the design of a test reactor are defined.
- *Chapter 5: Test Reactor Design*. The pre-conceptual design of the FHTR is presented.
- *Chapter 6: Licensing*. The licensing basis for the FHTR and safety basis is described.
- *Chapter 7: Other Major Facilities.* The development of an advanced reactor requires a variety of major facilities. The test reactor is the largest among those but is not the only facility. Other major facilities are defined.
- *Chapter 8: Recommendations and Conclusions*

2 Commercial FHR Design

The starting point for a test reactor design is the definition of long-term goals—a commercial power plant with specific capabilities. Those goals define the characteristics of the power reactor that, in turn, define the requirements for the test reactor. The FHR concept is described herein.

There are multiple proposed FHR concepts (Fig. 2.1). All are graphite-moderated, fluoride-salt-cooled low-enriched-uranium thermal-neutron-spectrum reactors[10]. Fluoride salt coolants have boiling points in excess of 1200°C thus all of these reactors operate at low pressure. All current designs use graphite-matrix coated-particle fuel developed for high-temperature gas-cooled reactors (HTGRs) but with different geometric shapes for the fuel assemblies. One specific reactor design (the Mark I Pebble-bed FHR [MK-1 PB-FHR]) is described herein to provide an understanding of FHRs and the incentives to develop this class of reactors. This is the design developed by University of California at Berkeley (UCB) that is part of the U.S. Department of Energy Integrated Research Project led by MIT with partners at the UCB and the University of Wisconsin. Three components of this particular reactor are described: (1) goals, (2) baseline design, and (3) economic basis.

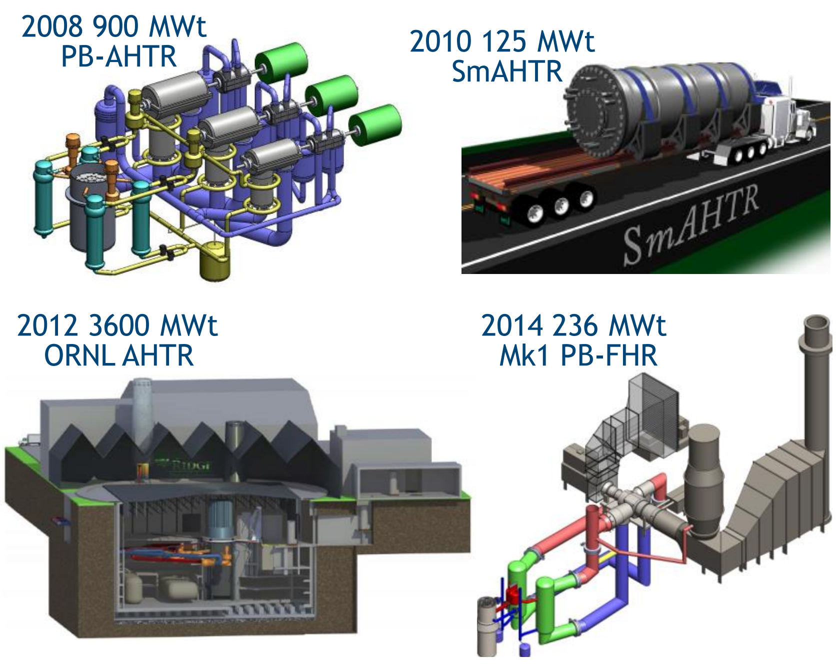


Fig. 2.1 Alternative FHR Designs

[10] C. W. FORSBERG, L. Hu, P. F. Peterson and K. Sridharan, *Fluoride-Salt-Cooled High-Temperature Reactors (FHRs) for Base-Load and Peak Electricity, Grid Stabilization, and Heat*, MIT-ANP-TR-147, Department of Nuclear Science and Engineering, Massachussetts Institute of Technology, Boston, MA (2013).

2.1 FHR Goals

Our proposed FHR design[11] has three goals[12]:

- Economics: A 50% to 100% increase in revenue relative to base-load nuclear power plants with capital costs similar to light-water reactors (LWRs).
- Environment: The enabling technology for a zero-carbon nuclear-renewable grid by providing an economic dispatchable source of zero-carbon electricity.
- Safety: No major fuel failures and thus no major radionuclide releases in a beyond design basis accident (BDBA).

2.2 Baseline Design

The choice of fuel and coolant define most reactor characteristics. The baseline FHR fuel consists of tri-structural isotropic (TRISO) coated-particle fuel embedded in 3-centimeter diameter graphite spheres—the same basic fuel used in pebble-bed HTGRs. The primary coolant is a lithium-beryllium-fluoride salt known as flibe ($^{7}Li_2BeF_4$). 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. Figure 2.2 shows the reactor vessel while Table 2.1 defines the major design parameters.

The characteristics of the fuel and coolant provide a unique safety case with large margins before fuel failure. The peak fuel temperature is ~800° with a fuel failure temperature above 1600°C. The nominal peak coolant temperature is 700°C with the coolant boiling above 1400°C. The high temperature capabilities in a BDBA may allow the decay heat to conduct to the environment at temperatures below major fuel failure[13, 14]. If there are no major fuel failures, there can't be large-scale radionuclide releases.

The FHR is coupled to a nuclear air Brayton combined cycle (NACC) that operates on nuclear heat with the capability to produce peak power using auxiliary natural gas, jet fuel,

---

11 C. ANDREADES et. al., *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. (2014)

12 C. W. FORSBERG et al., *Fluoride-salt-cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy*, MIT-ANP-TR-153, Massachusetts Institute of Technology, Cambridge, MA, December 2014.

13 M. J. MINCK, M. P. Short and C. W. Forsberg, "Fluoride-salt-cooled High-Temperature Reactor Severe Accident Strategy with Vessel Insulation Designed to Fail," Paper No. FA244, *Proc. of ICAPP 2013*, Jeju Island, Korea, April 14-18, 2013.

14 M. J. MINCK and C. W. Forsberg, "Preventing Fuel Failure for a Beyond Design Basis Accident in a Fluoride Salt Cooled High Temperature Reactor," Paper 14119, *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014),* Charlotte, North Carolina, April 6-9, 2014.

stored heat or ultimately hydrogen. This is similar to a natural-gas combined cycle plant where air is compressed, heated, flows through a turbine to generate electricity and exhausted to a heat recovery steam generator (HRSG), In the HRSG, steam is produced that generates added electricity. A plant flow schematic is shown in Figure 2.3 while Figure 2.4 is a drawing of the plant.

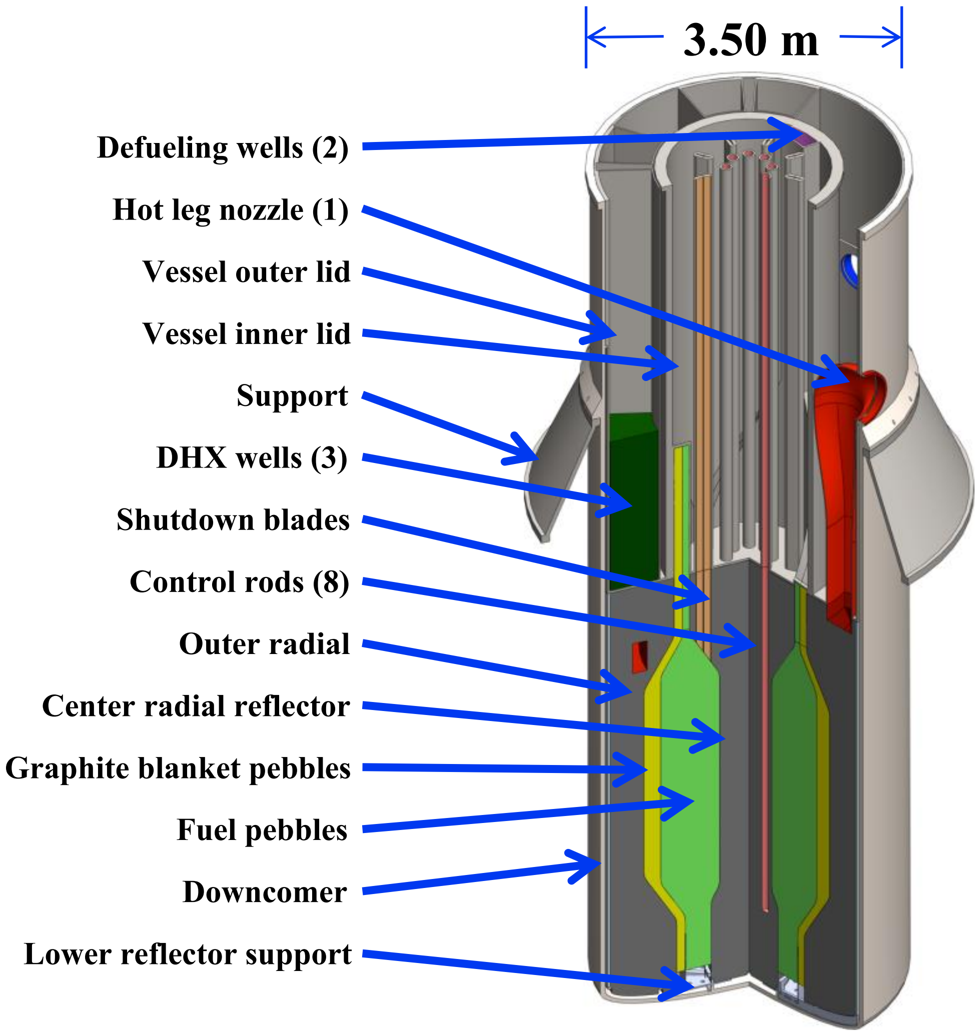


Fig. 2.2 Mark I Pebble-bed FHR (MK-1 PB-FHR)

The base-load thermal efficiency is 42% with a cooling water demand 40% of a LWR per unit of electricity. The low cooling water demand is because of (1) higher efficiency and (2) an air Brayton combined cycle where much of the heat is rejected as hot air from the HRSG. Peak power is produced by using natural gas, other fuels or stored heat to raise the compressed air temperature after nuclear heating (~670°C) to over 1100°C before entering the turbine. When producing peak electricity, the auxiliary fuel to electricity conversion efficiency is 66%--above that of a stand-alone natural gas plant because the natural gas heat acts as a topping cycle above the nuclear-heated air. The air Brayton cycle requires peak reactor coolant temperatures near 700°C and thus the power cycle defines the FHR peak coolant temperatures.

Table 2.1 Mark I Pebble-bed FHR Design Parameters

Reactor Design	
Thermal power	236 MWt
Core inlet temperature	600°C
Core bulk-average outlet temperature	700°C
Primary coolant mass flow rate (100%power)	976 kg/sec
Primary coolant volumetric flow rate (100% power)	0.54 m^3/sec
Power Conversion	
Gas turbine model number	GE 7FB
Nominal ambient temperature	15°C
Elevation	Sea level
Compression ratio	18.52
Compressor outlet pressure	18.58 bar
Compressor outlet temperature	418.7°C
Compressor outlet mass flow (total flow is 440.4 kg/s; conventional GE-7FB design uses excess for turbine blade cooling)	418.5 kg/sec
Coiled tube air heater outlet temperature	670°C
Base load net electrical power output	100 MWe
Base load thermal efficiency	42.5 %
Co-firing turbine inlet temperature	1065°C
Co-firing net electrical power output	241.8 MWe
Co-firing efficiency (gas-to-peak-power)†	66.4 %

This is a modular reactor design based on the following considerations.

- *Factory production*. All components can be built in a factory and be shipped by rail to the reactor site. This enables the advantages of mass production in a factory environment.

- *Scaleup*. While large FHRs can ultimately be built, the development of an FHR would start from the test reactor with several intermediate steps to a large reactor. This design could be the first step beyond a FHTR and the first size of commercial interest.
- *Power cycle*. The air-Brayton power system is the GE-7FB, the largest rail portable gas turbine made by GE. The reactor is sized to match that existing commercial turbine. If a larger reactor was to be built, multiple gas turbines would be used.

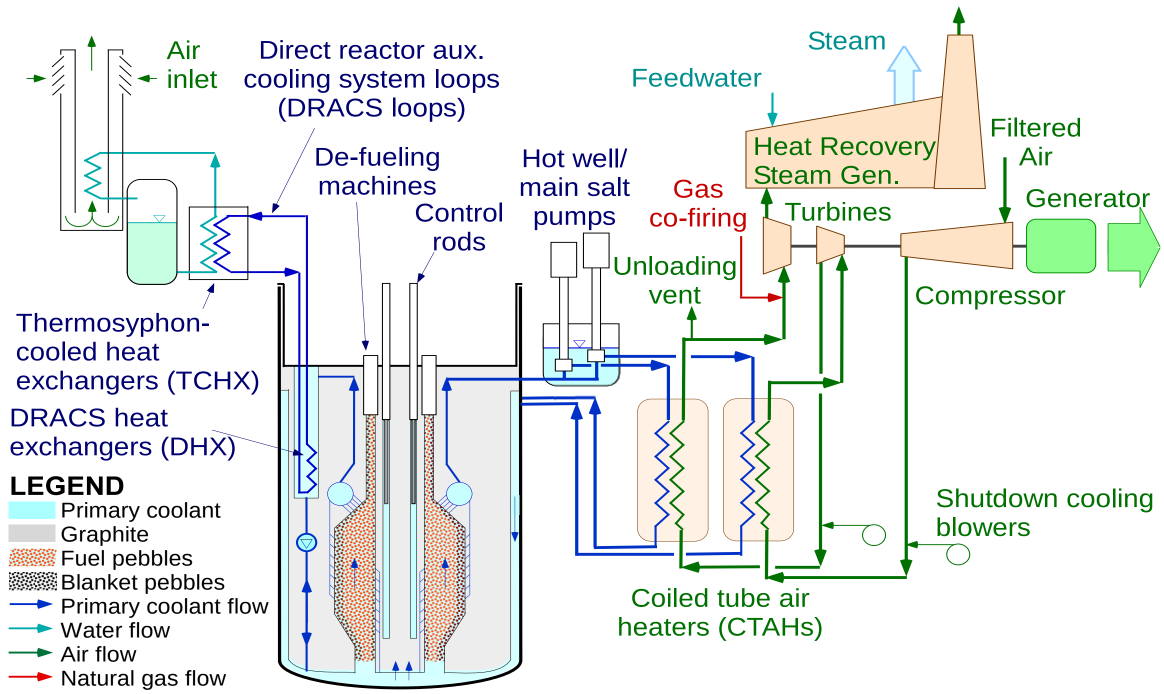


Fig. 2.3 FHR Process Schematic

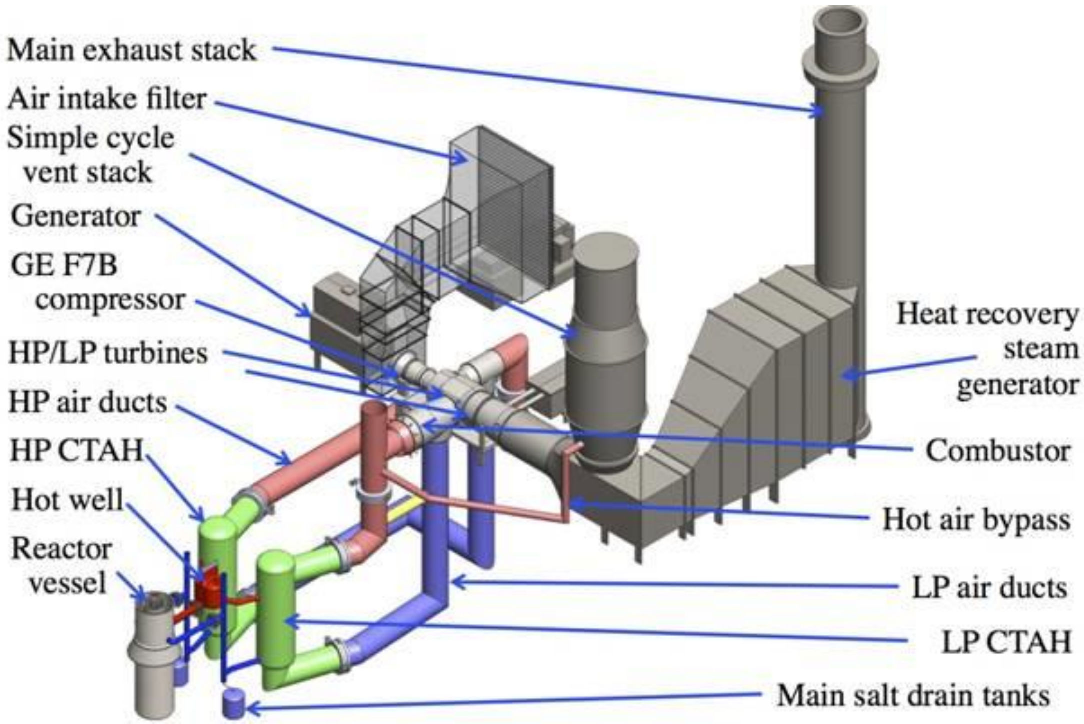


Fig. 2.4 FHR Plant Schematic

2.3 Economic Basis

The FHR economic case[15, 16] is based on maximizing revenue by the production of base-load and peak electricity. This capability is a consequence of choosing a Nuclear Air-Brayton Combined Cycle NACC) power conversion system. The original development of salt-cooled reactors in the 1950s was for the Aircraft Nuclear Propulsion Program where the goal was building a nuclear-powered bomber. The reactor was to power an aircraft jet engine. The reactor that was developed was the molten salt reactor (MSR) where the fuel was dissolved in the coolant—versus the FHR that uses solid fuel. The temperature requirements of the jet engine drove the development of the coolant salt. This was followed by the Molten Salt Breeder Reactor program with the goal to build a commercial power plant. The program did build a successful 8 MWt test reactor. However, the proposed design of the commercial MSR power plant was to use

---

[15] C. FORSBERG and D. Curtis, "Meeting the Needs of a Nuclear-Renewable Electrical Grid with a Fluoride-Salt-Cooled High-Temperature Reactor Coupled to a Nuclear Air-Brayton Combined Cycle Power System," *Nuclear Technology*, 186 (2014).

[16] C. FORSBERG et al., *Fluoride-salt-cooled High-Temperature Reactor (FHR) Commercial Basis and Commercialization Strategy*, MIT-ANP-TR-153, Massachusetts Institute of Technology, Cambridge, MA, December 2014.

a steam cycle because the Brayton cycle in the 1950s and 1960s was very inefficient for electricity generation.

Fifty years of advances in gas turbine technology now make it possible to couple an FHR to NACC. The proposed design uses the GE 7FB, a currently-available commercial air-Brayton gas turbine system, with modifications to deliver heat to the pressurized air stream from the reactor salt coolant via coiled tube air heaters (CTAHs). Figure 2.5 shows the power cycle. In the power cycle, air is filtered, compressed, heated by hot salt, flows through the first gas turbine, heated by hot salt, flows through the second gas turbine and sent to the heat recovery steam generator (HRSG). The hot exhaust air generates steam that can produce added electricity or sold to industrial customers. The base-load efficiency of 42% with peak compressed air temperatures of only 670°C is possible because this is a reheat gas-turbine power cycle—with some similarities to reheat steam cycles. Conventional natural-gas turbines do not have reheat cycles because there is insufficient oxygen in the compressed gas stream to inject added natural gas to reheat the air.

With NACC, peak power can be produced by injecting natural gas after using the hot salt to reheat the compressed air. The natural gas-to-electricity efficiency is above 66% because this heat is added above the "low-temperature" nuclear heat at 670°C. Table 2.2 gives the power cycle parameters.

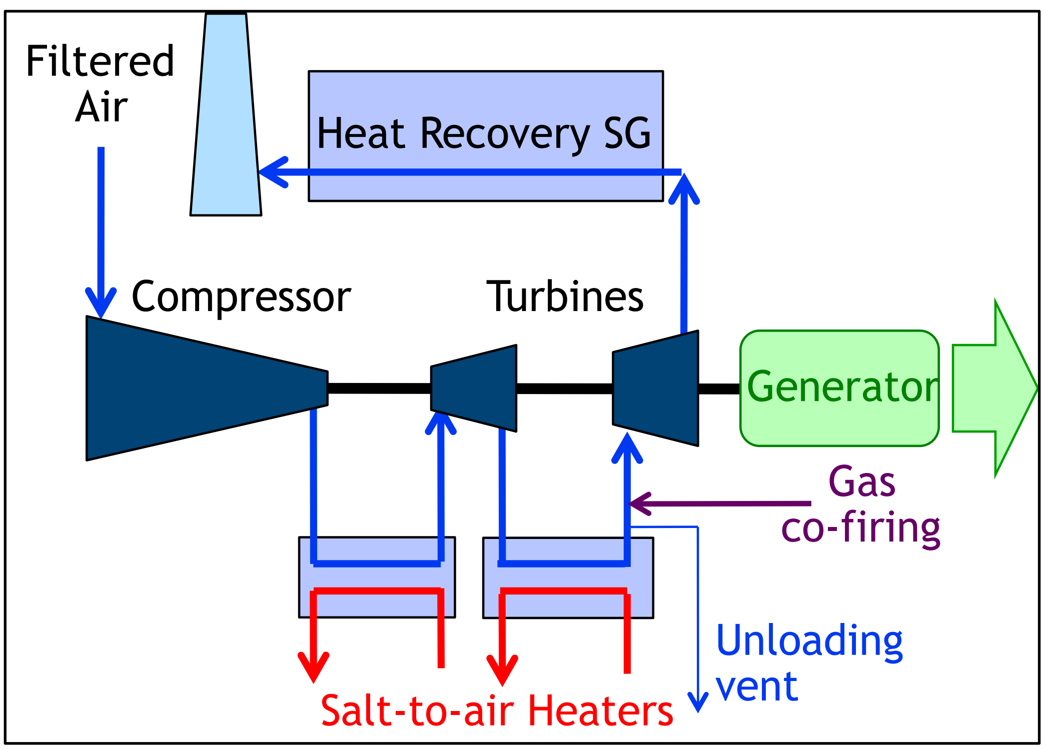


Fig. 2.5 NACC Power Cycle

Table 2.2 System Parameters for the GE 7FB Gas Turbine

Parameter	GE 7FB
Peak FHR Coolant Temperature (°C)	700
Compressor Exit Temperature (°C)	418.7
Air Temperature After Nuclear Heat (°C)	670
Base-load Heat (MWt)	235.3
Base-load Electricity (MWe)	100
Base-load Efficiency (%)	42.5
Natural Gas Heat Input (MWt)	213.5
Natural Gas Electricity (MWe)	141.8
"Co-firing efficiency" (%)	66.4
Peak Electricity (MWe)	241.8

In deregulated electricity markets, the price of electricity varies with time. The ability to produce peak electricity or sell steam enables increased revenue relative to a base-load nuclear plant. There are five possible operating modes.

- *Base load*: The reactor runs at full rated power. No supplemental fuel is injected. The Brayton and Rankine cycles both produce electricity for sale.
- *Peak load*: The reactor runs at full rated power. Supplemental natural gas is injected at full capacity. The Brayton and Rankine cycles both produce electricity for sale.
- *Steam sales*: The reactor runs at full rated power. No supplemental fuel in injected. The Brayton cycle produces electricity for sale, and the HRSG steam is directed to the industrial steam distribution system for process heat sales when the price of electricity is low. This process heat is assumed to have a value of 90% of the price of natural gas per unit energy. (Here, natural gas is considered the most convenient alternative source of industrial process heat.)
- *Mixed mode*: The reactor runs at full rated power. Supplemental natural gas is injected at full capacity. The Brayton cycle produces electricity for sale, and the HRSG steam is directed to the industrial steam distribution system for process heat sales.

- *No output*: There is the option of dumping hot air from vents for no power to the grid while the reactor continues to operate. This provides a grid "black start" capability but is not considered in this economic analysis.

The FHR with NACC allows the operator to choose for each hour of the year the operating mode that maximizes revenue—a capability that exists with gas turbine power cycles but not with other power cycles. The full set of advanced operating modes provides an increase in revenue over baseload-only operation by 90% in the Texas market and by 114% in the California market. For the case where only base load and peak power production modes are used (assuming that steam sales and mixed mode operation are unavailable), the revenue increase is 42% in Texas and 67% in California. In that case, the plant operates in peak mode for 6,776 hours (77% of a year) in Texas and 7,041 hours (80% of a year) in California. The large number of hours of peak electricity operations is because NACC is more efficient at converting natural gas to electricity than a stand-alone natural gas plant. As a consequence, it is dispatched before any natural gas plant in Texas and California. This increases plant revenue after subtracting the cost of natural gas. Current natural gas prices are very low. If those prices rise or there is a carbon tax, the competitive advantage of an FHR with NACC versus stand-alone natural gas plants will increase. Table 2.3 shows revenue relative to base load-only revenue for all calculated cases.

Table 2.3 Relative Revenues for Different Operating Modes

Allowable Operating Modes	**Utility Markets (2012 Wholesale Prices used in Analysis)**	
	Texas	**California**
Base load	100%	100%
Base load + Peak	142%	167%
Base load + Steam	142%	132%
Peak + Steam	167%	182%
Peak + Mixed + Steam	190%	214%

The potentially large economic benefits are a result of combining a high-temperature nuclear heat source with a combined cycle gas turbine. This is a new option because of advances in gas turbine technology in the last 20 years. An FHR with NACC could not have existed 20 years ago—the gas turbine technology was not sufficiently advanced. This has implications for the future. Almost all R&D being done on power systems is being done on gas turbines because of their use in aircraft jet engines and natural gas plants. As a consequence, the performance of these systems for base-load and peak power is expected to substantially improve by the time an FHR is commercialized.

2.4 Zero-Carbon Dispatchable Electricity

The world will transition to a low carbon grid in this century. The FHR with NACC can be used to produce variable zero-carbon electricity using hydrogen, biofuels or stored high-temperature heat (Fig. 2.6). Hydrogen made from electrolysis or biofuels can substitute for natural gas. Alternatively, high-temperature stored heat may be used for peak electricity production.[17]

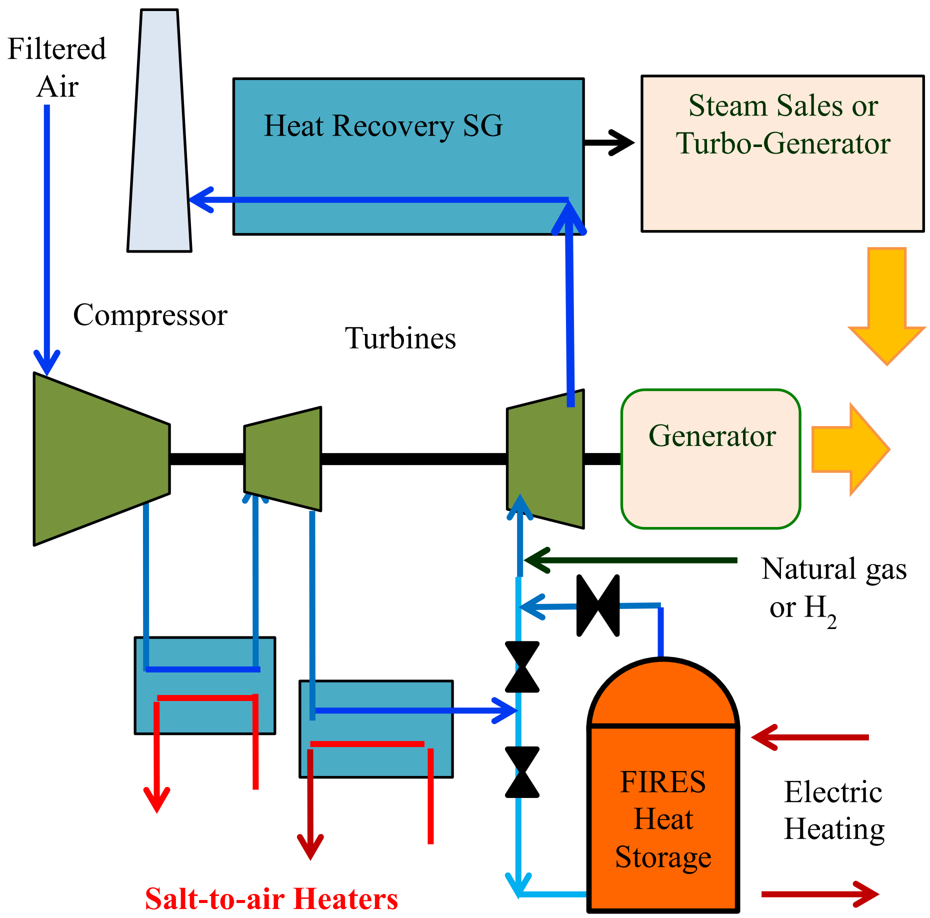


Fig. 2.6 FHR with NACC and FIRES

Firebrick Resistance-Heated Energy Storage (FIRES), the stored heat option, involves heating firebrick inside a prestress concrete pressure vessel with electricity to very high temperatures at times of low prices—less than the price of natural gas. When peak power is

[17] C. W. FORSBERG, "Dispatchable Electricity with Salt-Cooled Reactors, Heat Storage, and Base-load Reactor Operation," *Trans. American Nuclear Society*, Anaheim, California, November 9-13, 2014.

needed, compressed air after nuclear heating and before entering the second turbine would be routed through the firebrick, heated to higher temperatures and sent to the turbine. The efficiency of converting electricity to heat is 100%. The efficiency of converting auxiliary heat (like auxiliary natural gas) to electricity in our current design is 66%. This implies a round trip efficiency of electricity to heat to electricity of ~66%. Improvements in gas turbines in the next decade are expected to raise that efficiency to 70%. This storage efficiency is similar to many other electricity storage technologies. Like an FHR with NACC, reasonable electricity storage efficiencies via electricity to heat to electricity are a consequence of advances in gas turbine technology.

Much of the heat-storage technology is now being developed by General Electric and partners for a new adiabatic compressed air energy storage (CAES) system called Adele (German abbreviation) where the first prototype plant is expected to be operational in several years (Fig. 2.7). The baseline design involves compressing air to 70 bars and 600°C, cooling the compressed air by heating firebrick in a prestress concrete pressure vessel, and storing the cool compressed air in underground caverns when the price of electricity is low. At times of high electricity prices the compressed air goes through the firebrick, is reheated, and sent through a turbine to produce electricity. The expected round-trip storage efficiency is 70%. For NACC using high-temperature stored heat FIRES peak pressure would be about a third of Adele but with the complication of the need to electrically heat the firebrick at times of low electricity prices.

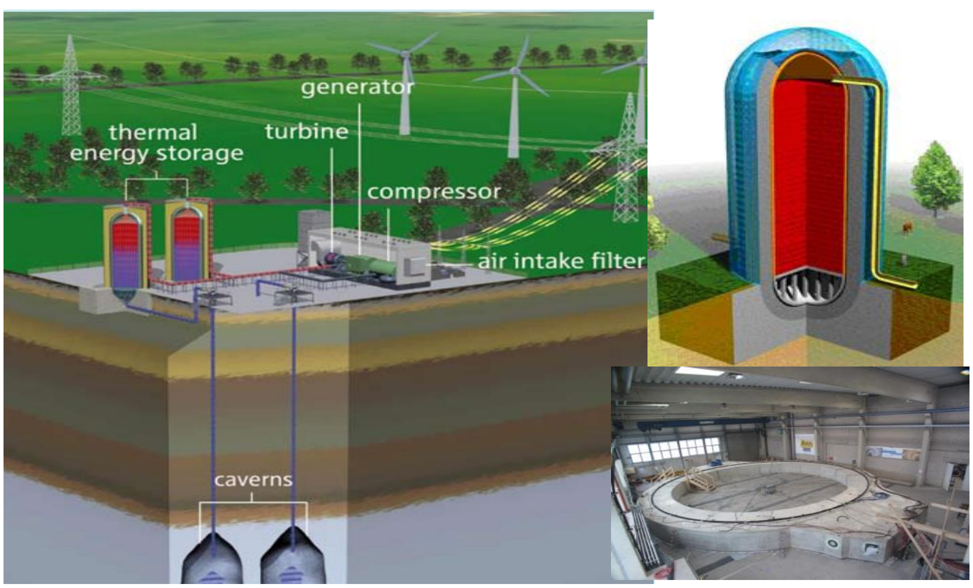


Fig. 2.7 Adiabatic Compressed Air Energy Storage System: Adele System, Prestress Concrete Pressure Vessel, and Test Section of Vessel

In some markets such as California there are times of very low electricity prices—including negative-priced electricity at times of high wind or solar input. When prices are low, the reactor operating at base-load would recharge the heat storage system. If prices per unit of heat were below the price of natural gas, the plant would buy electricity from the electricity grid for heat storage to avoid the cost of natural gas for peak power production. Such operating modes imply very large revenue streams relative to a base-load nuclear plant.

In the context of a zero-carbon nuclear-renewable electricity grid, this storage system is fundamentally different than batteries or pumped storage. With traditional storage systems the electricity charging rate is close to the discharge rate. In this system low-capital-cost resistance heating enables fast charging rates to buy large quantities of low-priced electricity when available—such as for two or three hours in the middle of the day in a grid with large PV output. Second, storage (MWh) by itself does not enable use of renewables. One also needs electricity generating capacity (MW) because conventional storage systems will become fully discharged if there are multiple days of low solar or wind conditions. Heat storage embedded in NACC provides both storage and generating capacity.

The FHR with NACC and FIRES is an enabling technology for the large-scale use of renewables. Large-scale solar or wind result in excess electricity at times of high solar or wind output. That results in collapse of electricity prices at those times. For higher latitudes (United States, Europe, etc.) revenue collapse[18] in deregulated electricity markets occurs between 5 and 10% of total electricity provided by solar and between 20 and 30% of total electricity provided by wind—independent of the solar or wind technology. What is required for large-scale use of renewables is a zero-carbon electricity generation technology that incorporates storage to absorb excess electricity when available and provide electricity at times of low wind or solar output. A similar conclusion was reached by the Google® renewables energy team[19] that new technologies are required to stop climate change and that renewables in their current forms can't stop climate change. An FHR with NACC and FIRES matches those requirements.

---

[18] L. Hirth, "The Optimal Share of Variable Renewables: How the Variability of Wind and Solar Affects Their Welfare-Optimal Deployment," *The Energy Journal*, **36** (*1*), 2015.

[19] R. Koningstein and D. Fork, "What It Would Really Take to Reverse Climate Change," *IEEE Spectrum*, November 18, 2014, http://spectrum.ieee.org/energy/renewables/what-it-would-really-take-to-reverse-climate-change.

3 Test Reactor Markets, Owners, and Structures

3.1 Goals

Test reactors can be divided into two classes. The goal of a Class I test reactor is to develop the reactor technology. Dragon (the first high-temperature gas-cooled reactor) and Experimental Breeder Reactor I (the first sodium fast reactor) are examples of Class I test reactors. The FHTR would be a Class I test reactor. Class II test reactors are irradiation sources to test materials and fuels. HFIR at ORNL and ATR at INL are Class II test reactors. No Class I test reactor has been built in several decades anywhere in the world. A Class-I test reactor is required to:

- Demonstrate technical viability
- Provide the data required for design and licensing of a pre-commercial demonstration plant
- Provide a test bed for alternative fuels, fluoride salt coolants, and other systems.

The U.S. government is considering building a new test reactor. There is proposed funding in the House of Representatives appropriations legislation for FY 2015 that directs the U.S. Department of Energy to initiate a study of what type of test reactor should be built.[20] This includes the options of a Class I test reactor or a Class II test reactor. There are several candidate Class I and Class II test reactor options. The largest Class II test reactor in the United States is the Advanced Test Reactor (ATR) at Idaho National Laboratory. Studies have been conducted on a long-term replacement for the ATR and what would be the requirements for such a reactor.[21, 22]

3.2 Test Reactor Options

There are two FHTR options as defined by goals:

- *General Purpose Class I-A Test Reactor*. The FHTR would be a Class I-A test reactor designed with broader test capabilities to provide required information for a variety of FHR concepts. This lowers the risk of premature selection of technical features but requires more careful planning for the transition between a FHTR and a pre-commercial

[20] U.S. House of Representatives Appropriations Bill 113 for FY 2015 Energy and Water, Washington D.C.

[21] M. A. Pope, H. D. Gougar and J. M. Ryskamp, Evaluation of Concepts for Multiple Application Thermal Reactor for Irradiation eXperiments (MATRIX), INL/EXT-13-30045 (September 2013).

[22] M. A. Pope, H. D. Gougar, and J. M. Ryskamp, "Design Studies for a Multiple Application Thermal Reactor for Irradiation Experiments (MATRIX), *PHYSOR 2014 – The Role of Reactor Physics Toward a Sustainable Future*, Kyoto, Japan, September 28 – October 3, 2014.

FHR. Such an FHTR would be functionally similar to DRAGON[23]—the first high-temperature gas-cooled reactor. This was a 20-MWt reactor built by the OECD in the United Kingdom. Such general purpose machines require greater resources to build, are more versatile and keep design options open for a longer period of time. While not designed as a Class 1A test reactor, the U.S. pre-commercial Shippingport PWR over its lifetime operated as such a reactor in terms of core design with three radically different cores with different fuels, geometries, and control systems (plate fuel, a HEU thorium core, and a pin fuel).

- *Class I-B Test Reactor*. A class I-B FHTR is designed with a more restrictive set of goals. It may be designed as a proof-of-concept first-of-a-kind test reactor or to provide the necessary information for a specific pre-conceptual design of a commercial reactor. It is the option being pursued by the Chinese Academy of Science with their goal to build an FHTR by 2020.

The two types of Class I test reactors reflect different test reactor strategies that can lead to a pre-commercial FHR. One can build a very simple test reactor with a short lifetime followed by a more capable test reactor. This strategy reduces risk by taking smaller steps. It was used to develop the early fast reactors in the U.S.: EBR-I was built and followed by EBR-II. A second strategy is to build a test reactor where major changes can be made—including replacement of the entire reactor core. This was part of the development strategy that included the Shippingport PWR that had three radically different core designs that were sequentially tested. It implies a larger investment in the first FHTR.

There is not a sharp line of division between a Class I-A and Class I-B test reactor but rather a spectrum of options. The test reactor choice depends upon the government and commercial structure[24] of the country building the reactor. Discussions with vendors indicate that they would not be willing to finance a test reactor because the time is too long and the risks are too great to start with a new concept and commercialize that reactor. Globally, all first-of-a-kind reactors have been built by governments. The implication is that an FHTR if built in the United States will be built by the U.S. Government. In that case one must consider government goals that imply a general purpose Class I-A FHTR would likely be chosen.

- *U.S. National goals*. The government interest in a commercial FHR is driven by multiple national policy goals such as environment (low-carbon electrical grid), nonproliferation

[23] DRAGON PROJECT, *O. E. C. D. High-Temperature Reactor Project (DRAGON): 1959-1976: A Summary & Evaluation of the Achievements of the Dragon Project and Its Contribution to the Development of the High-Temperature Reactor*, Report 1000, A.E.E. Winfrith, November 1978.

[24] D. CARPENTER et al., *Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Development Roadmap and Test Reactor Performance Requirements White Paper*, Department of Nuclear Engineering, University of California, Berkeley, UCBTH-12-004, 2013, http://FHR.nuc.berkeley.edu/.

and safety—as well as competitive economic goals. Multiple goals imply a more capable design of test reactor.

- *Competitive vendors*. Historically the U.S. government has been unwilling to choose a vendor and support that vendor to develop a national product. That was true for the light-water reactor, sodium-cooled fast reactor, the high-temperature gas-cooled reactor, and the current DOE small modular reactor (SMR) program. In each case the U.S. government provided support to two or more vendors to help the early development of their concepts. This is in contrast to Russia, France, and several other countries with national vendors. This implies that a test reactor for the United States will need the capability to support multiple FHR design concepts that may be developed by different vendors.
- *U.S. Government markets*. The government interest will be driven by commercial and government needs such as the general purpose capability to undertake high-temperature irradiations for government missions. There are potential government markets[25] for an FHR with NACC where the reactor design and fuel would be significantly different than a commercial design. The federal government has a potential need and is examining the use of nuclear reactors to provide power to isolated facilities where it is very expensive to bring in fossil fuels. The FHR with NACC is potentially attractive because the reactor could be sized to meet the average needs of the facility with NACC using auxiliary fossil fuel to provide peak power. The reactor would not need to be sized for peak demand but fuel logistics would be dramatically decreased.
- *Institutional constraints*. The option of building in series of increasingly more capable test reactors as a route to a commercial machine is only viable if there is a massive national commitment. That must include fast licensing of test reactors to have a credible development schedule. At the current time only a few defense programs have such a commitment.

3.3 Ownership, Financing, and Project Structure

A test reactor requires major funding and appropriate project structures to succeed because it combines major R&D with a significant nuclear construction project. Currently three research reactors are in the planning or construction phase (Table 3.1) that may provide relevant ownership, cost, schedule, and organizational information for an FHTR. These will be discussed below.

---

[25] R. MACDONALD and C. W. Forsberg, "Strategies for a Small Transportable FHR with Reduced Security, Safeguards, and Safety Systems," *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, North Carolina, April 6-9, 2014.

Table 3.1 Selected Research Reactors Being Planned or under Construction

Reactor	Operating Organization	Country	Power (MWt)	Planned Construction	Applications
Jules Horowitz Reactor (JHR)	CEA	France	100	2007-2016	Materials testing; Radioisotope production
MYRRHA	SCK-CEN	Belgium	50-100	2015-2019	Multi-purpose materials testing reactor
TMSR-SF	Shanghai Institute of Applied Physics	China	2	2015-2020	FHR physics and operational tests

Four ownership and financing options have been identified for management and funding of an FHTR.

3.3.1 U.S. Centric Program

This would be similar to the U.S. strategy in the 1950s and 1960s where the U.S. government fully funds the test reactor and development program. Light water reactors were developed using this strategy. The U.S. Navy wanted nuclear powered submarines and developed the pressurized water reactor (PWR) for this application because the reactor could fit within the submarine hull. The Shippingport commercial demonstration project managed by the navy and based on navy reactor design was the demonstration reactor that led to commercial PWRs.

The history of the boiling water reactor (BWR) was different. Much of the general light-water reactor technology was developed by the navy program. However, there was a separate Argonne National Laboratory (ANL) program for the BWR. ANL designed and built a series of test reactors (Borax) that developed the technology leading to commercial deployment.

In both cases the federal government was the funding source through the test reactor stage and a major funder for the reactor demonstration projects. The long lead times imply that no commercial venture will fund a first-of-a-kind test reactor because the time for return on investment is measured in decades. Corporate investments are typically limited to a decade or less.

The demonstration phase was aided by the existence of regulated utilities in the 1950s and 1960s that had a public service requirement in their charters and a steady revenue streams. The changing institutional structure would require changes in the demonstration phase. The recent experience of the Next Generation Nuclear Plant (NGNP) to commercialize high-temperature reactors provides lessons learned in this context.

Because of the close technological connection between the FHR and MSR, the history of the molten salt reactor (MSR) is relevant. In a MSR the fuel is dissolve in the salt coolant. Salts as reactor coolants were developed by this program. Solid fuels that could operate with liquid coolant salts did not exist when the MSR was being developed; thus, the FHR was developed later after development of HTGR fuels.

The MSR was originally developed for the Aircraft Nuclear Propulsion Program that had the goal of developing a nuclear powered bomber—a program parallel to the U.S. Navy submarine program. The reactor was to be coupled to a jet engine—an air Brayton power cycle. The military program was cancelled because of (1) development of intercontinental ballistic missiles and (2) the weight of the reactor shielding.

After cancellation of the military program, a civilian version of the MSR was partly developed as a breeder reactor. It adopted a steam power cycle because air-Brayton power cycle technology was not sufficiently developed to be used in a utility application. That program was ultimately cancelled when the U.S. decided to concentrate resources on a single type of breeder, the sodium-cooled fast reactor.

The current environment makes a U.S.-only option for building an FHTR difficult in the United States at this time. This reflects current low natural gas prices and the historical short-term planning horizons in the U.S. This may change because of concerns about climate change and the need to reduce greenhouse gas emissions or potential government missions.

3.3.2 Joint Program with the Chinese Academy of Science (CAS)

Chinese Academy of Sciences (CAS) started a strategic science and technology program "Future advanced nuclear fission energy" in 2011, aimed to support innovations in line with China's national policy for large-scale development of nuclear energy. Within the scope of this program, the Shanghai Institute of Applied Physics (SINAP) is developing a Thorium-based Molten Salt Reactor (TMSR) nuclear energy system. A solid fuel 10 MWt test reactor will be constructed as the initial step with startup by 2020 with the expectation of a larger 100 MWt FHR shortly thereafter. This reactor, designated as Solid Fuel Thorium-based Molten Salt Reactor (TMSR-SF), will be a fluoride-salt-cooled pebble-bed reactor. TMSR-SF design is evolved from a Pebble Bed FHTR (PB-FHTR) concept developed by UCB[26]. This reactor will be the first FHR ever built.

The TMSR-SF1 will operate at a power density of 5.13 MWt/m^3–below a power reactor power density. FHR fuels and materials can be tested at prototypical power densities and temperatures in existing test reactors, as with the testing of U.S. NGNP fuel in the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). This and other capabilities creates the option of a US-China strategic partnership to reduce costs, schedule, and risks for both partners with a cooperative program. There are several paths forward.

[26] J. E. BICKEL, M. R. Laufer, L. Li, A. T. Cisneros and P. F. Peterson, "Conceptual design, experiments, and analysis for the core of an FHR-16 test reactor," *Proc. International Congress on Advances in Nuclear Power Plants 2010*, ICAPP 2010, June 13 - 17, 2010, American Nuclear Society, pp. 1281-1291.

- *U.S. test reactor*. Data from the TMSR-SF1 would enable the U.S. to build a sophisticated FHTR on a shorter schedule at lower costs with lower risks. A U.S. FHTR would, in turn, accelerate the next step in FHR commercialization.
- *Precommercial power reactor*. Working with the CAS could also be one route to a first demonstration FHR that is built in the United States. The CAS program is driving to a pebble-bed commercial reactor thus would favor that specific design.

The U.S. has growing connections with the Chinese nuclear program. Westinghouse has commercial agreements with the Chinese and a joint program that is building commercial reactors in China. More recent agreements include Chinese suppliers in the supply chain for construction of AP-1000 reactors worldwide. More recently Terrapower® has developed agreements with U.S. government approval[27] for joint development of their traveling wave reactor with China. Beyond these nuclear-specific agreements, there is a now broader agreement between the U.S. government and the Chinese government to reduce greenhouse gas emissions. The FHR development could become a component of this much larger program.

3.3.3 U.S. Led International FHTR

There is a long and successful history of cooperative research projects with the size and scope of an FHTR. In each case there is a lead country with a clear responsibility for the project. Two historic and two ongoing nuclear projects are discussed herein.

The first high-temperature gas-cooled reactor was DRAGON[28, 29], which was built in the United Kingdom with U.K leadership. It was an OECD project[30] that paved the way for all later HTGRs. A FHTR could be organized in a similar fashion. The project was led by the United Kingdom with a strong project structure with a board of directors above the project manager that represented the various nations. There was a common understanding that DRAGON was a science project—not a commercial project. This allowed maximum sharing of information and simplified forming of partnerships.

After the Three Mile Island Accident, the United States led an international consortium[31] that built the Loss of Fluid Test (LOFT) reactor facility at the Idaho National Laboratory. This was a 50 MWt pressurized water reactor that started operations in 1976. The objective was to

[27] DEPARTMENT OF ENERGY, "Proposed Subsequent Arrangement", *Federal Register*, 78 (243) December 18, 2013.
[28] L. R. SHEPARD, "Dragon Project: European Cooperation on High-Temperature Reactors," *Nature* 237, 215-216, May 26, 1972.
[29] A.A.E. WINFRITH, O.E.C.D. High Temperature Reactor Project (DRAGON): 1959-1976: A Summary & Evaluation of the Achievements of the Dragon Project & Its Contribution to the Development of the High-Temperature Reactor, Dragon Project Report 1000, England, November 1978.
[30] E. N. SHAW, *History of the OECD Dragon Project: Europe's Nuclear Power Experiment*, Pergamon Press, 1983.
[31] S. M. MODRO et al., *Review of LOFT Large Break Experiments: OECD LOFT Project*, NUREG/IA-0028, U. S. Nuclear Regulatory Commission, October 1989.

experimentally duplicate large and small pipe break accidents in commercial reactors in a small specially built reactor where a core melt accident would be acceptable.

These partnerships were successful because (1) there were potentially large benefits and (2) strong leadership by at least one partner. In this context, the top level goals for the FHR and the credibility that those goals may be achievable is central to the success of a FHTR international consortium. There are several recent examples of international test reactor consortiums.

- *Jules Horowitz Reactor (JHR).* This Class II test reactor is nearing completion at the Cadarache site in France. It is a light-water test reactor that will serve as a European research reactor for an international consortium of research institutes. As of March 2013, the JHR is supported by 9 research institutes in Spain, Belgium, Czech Republic, Finland, France, Israel, India, Japan, and United Kingdom; 3 utilities and industrial partners including EDF, AREVA, VATTENFALL; and the European Commission[32]. JHR is a 100 MWt Class-II research reactor with more than 20 high flux in-core irradiation positions for testing of fuels and materials. This facility is anticipated to fill the gap in materials test reactors availability when the several of the existing reactors in Europe are expected to be shutdown between 2015 and 2020. JHR's primary missions are in materials and fuel testing, similar to the ATR, but also for medical radioisotopes production. The JHR consortium finances construction and guarantees members access to the facility for proprietary experiments. The construction budget is estimated close to $1 billion dollars.
- Multi-purpose hYbrid Research Reactor for High-tech Applications (MYRRHA). Belgium has initiated this reactor project with international partners. This is a flexible fast spectrum research reactor (50-100 MWt). It is based on an accelerator driven system (ADS), and is designed to operate in sub-critical and critical modes. It consists of a proton accelerator of 600 MeV, a spallation target and a core with MOX fuel, cooled by liquid lead-bismuth. This facility will support research and development in the following areas: lead-bismuth technology; MOX fast reactor driver fuel qualification; material qualification; component qualification; reactor physics code validation. Construction of the facility is planned for 2015-2019, followed by three years of startup tests. MYRRHA is a European project[33] with many similarities to DRAGON. MYRRHA is a 100 MWt reactor that can operate as a critical facility or as an accelerator-driven critical system. It is a materials test facility using lead coolant. However, like Dragon, is a first-of-a-kind facility and thus is in many ways similar in its challenge to a first of a kind FHTR. Its estimated cost[34] is $1.3 billion dollars.

[32] http://www.cad.cea.fr/rjh/index.html
[33] A. ABDERRAHIM et al., "MYRRHA – A Multi-Purpose Fast Spectrum Research Reactor", *Energy Conversion and Management*, **63**, 4-10, 2012.
[34] "SCK-CEN Awards Major MYRRHA Contract," *Nuclear News*, 56 (12) 62-63, November 2013.

All of the above international projects are similar in size and scope to an FHTR providing a first estimate that an FHTR, including the supporting R&D programs, will cost between one and two billion dollars. The FHTR is a smaller reactor but because the goal is ultimately a power reactor, there will be a larger associated R&D program. The actual physical cost of the reactor will be a fraction of the total costs.

In many of these projects, the partners provide financing and support in kind (fuel, special equipment, etc.). For example, in the MYRRAH project AREVA is providing the MOX fuel from its commercial MOX plant. By using existing industrial facilities of partners with a vested interest in project success, costs, risks, and schedule can be reduced. The same would apply to an FHTR.

The history of such projects is good provided there are well-defined goals and a strong centralized project management organization that has full responsibility for all aspects of the project. The success may also be due to the project size. A one to two billion dollar project is large enough to draw the best talent but small enough that it is manageable. The total funding when divided among partners over several years is sufficiently limited that it is manageable within most national budgets. In contrast, the record of very large high-tech projects by the private and public sector is not good.

3.3.4 Public-Private Partnership with Domestic and Foreign Partners

This strategy would have a significant early commercial input. Based on expertise, likely partners could include Japan, China, and Westinghouse because of their high-temperature gas-cooled reactor programs. Candidate commercial partners include the vendors for natural-gas combined-cycle plants: General Electric, Toshiba, Alstom, Siemens, etc. However, the long lead times between a test reactor and commercial product make it unlikely that there would be serious private funding for an FHTR.

3.4 Path Forward

The ultimate path forward will be determined by who makes the major decisions. It may be a combination of the above strategies. At the current time the most viable options appear to be (1) a joint program with the CAS, (2) a U.S. led FHTR with multiple partners, and (3) some combination of options 1 and 2.

An international partnership is more complicated to set up but it reduces financial and political risks for each partner. There are no massive financial costs for any partner. Because this is a test reactor the challenges associated with addressing commercial issues are reduced.

In this context, there is one very attractive option—a joint program between the CAS and the U.S. The CAS plans to have a 10 MWt FHTR operational by 2020. This FHTR is designed to lead to a specific design of pre-commercial FHR. There are large incentives for a FHR program that (1) can be a cooperative program with the CAS to accelerate the technology or (2) independent of the CAS because of the unpredictability of international relations. A general-purpose international FHTR enables both options to remain open. Working with the CAS would accelerate their program to build an FHTR and provide valuable data for design of a more capable general-purpose FHTR. For the CAS there would be large incentives to be a major partner in a general-purpose FHTR because such a machine could test coolants, materials, and fuels under much more extreme conditions to strengthen the design and safety case for a commercial FHR. It would also provide backup options if major problems were identified in the development pathway they had chosen. In effect, it could be a win-win strategy for all partners for many alternative futures.

4 Test Reactor Requirements

The starting point of any reactor design is to define goals. A recent example[35] of defining goals is that for a new Class II materials test reactor in the U.S. The same is required for an FHTR. This chapter defines those goals for a general purpose Class I-A FHTR.

4.1 Test Reactor Capabilities

The above considerations lead to the following test reactor capabilities. Chapter 5 defines these goals at the next level of detail. There are a set of broad requirements for any FHTR. These include:

- *Peak outlet temperature will be at least 700°C.* FHR coolant temperatures of ~ 700°C are required to couple efficiently to air-Brayton power cycles and their capability to produce base-load electricity, peak electricity with auxiliary heat (natural gas, other combustible fuels, and stored heat), and low cooling water requirements.
- *Inlet temperature will be at least 600°C.*The fluoride salts considered for use as coolants in the FHTR have freezing temperatures between 350 and 500°C. The inlet temperature should be at least 100°C above the freezing point of the coolant to preserve sufficient safety margins and improve salt properties by decreasing viscosity[36].
- *Coolant salt(s) will (1) be compatible with graphite-matric fuel, graphite and structural materials, (2) have freezing points below 500°C, (3) be stable in high radiation environments, and (4) have low neutron cross sections.* The requirements limit the coolant to mixtures of fluoride salts. The 100°C safety margin and minimum temperatures of 600°C limit coolant options to those that remain liquid at 500°C or lower. The most likely salt choice is flibe but as discussed earlier there are other options.
- *Uranium enrichment will be less than 20 percent.* Enrichment of the fuel microspheres is limited to less than 20% ^{235}U to avoid proliferation concerns[37].

There are specific requirements for a Class I-A general purpose FHTR

- Capability to test multiple fuel forms in a large central irradiation position
- Capability to operate with two or more different coolants recognizing that changing coolants may require changing out the driver fuel. The coolant choice is so central to the

---

[35] H. D. Gougar and M. A. Pope, *Requirements, Supporting Technologies, and Recommended Investigations for a New Test Reactor Fueled with Low-Enriched Uranium*, INL/EXT-12-27111 (Rev. 1), August 2014.
[36] C. W. FORSBERG, "Goals, Requirements, and Design Implications for the Advanced High- Temperature Reactor," *Proc. 14th Int. Conference on Nuclear Energy*, Miami, Florida, July 17–20, 2006, Paper 89305, American Society of Mechanical Engineers, 2006.
[37] *Office of Global Threat Reduction*. Web link*:* http://nnsa.energy.gov/gtri, last accessed on Jan. 14, 2012.

concept that the test reactor may be required to test different fluoride salt coolants to fully understand the operational implications of each coolant.

- Peak flux in the central core position 3X the FHTR driver fuel average flux to accelerate fuel testing. This limit may be adjusted as design studies define the peak to average flux in alternative commercial designs. In this context it is noteworthy that in a commercial FHR the fuel lifetime in the core is between 1 and 1.5 years—unlike LWRs where fuel lifetime in the reactor core is typically 4 years. As a consequence, the need for accelerated fuel testing is less than for other reactor types.
- Experimental volume in the center core position corresponding to a full-size hexagonal lattice assembly position. For our specific design we have chosen a volume of 95L with a flat-to-flat diameter of 25 cm and stretching the entire core height (175 cm).
- Multiple additional small hard-spectrum irradiation positions for non-fuel materials testing.
- Minimum fuel cycle length of 183 EFPD (0.5 EFPY).
- Maximum peak fuel temperatures below 1250°C during normal operations and 1600°C during accident testing. There will also be a requirement on the average fuel temperature as this affects the reactor's response during reactivity transients.
- Negative reactivity coefficients (fuel temperature, coolant temperature, and void) throughout burnup. For safety this is a property of the core, not of specific reactivity coefficients.
- Shutdown margin of -$0.5 in the most reactive configuration.
- Minimizing the power level (20 to 40 MWt) consistent with meeting the above goals.

4.2 Reactor Viability Testing

The first goal of the FHTR is to demonstrate the technical viability of the FHR concept—can one reliably operate an FHR? This is separate from testing any specific fuel or coolant. This includes a variety of operations.

- *The test reactor will demonstrate requisite machinery, systems, and procedures for refueling salt-cooled high-temperature reactors.* No FHRs have ever been built so there is no prior experience with refueling methods for such reactors. The fuel density may be less than the density of the coolant (like lead-cooled reactors), meaning the fuel will float in the coolant. The need to hold down the core while refueling, coupled with the need for high-temperature refueling, may pose significant engineering challenges[38].
- *Reliable, scalable, and economic systems for preventing salt freezing and for monitoring and regulating salt chemistry and salt volume will be demonstrated.* These systems

[38] D. T. INGERSOLL, C. W. Forsberg, and P. E. MacDonald, *Trade Studies for the Liquid-Salt-Cooled Very High-Temperature Reactor: Fiscal Year 2006 Progress Report*, ORNL/TM-2006/140, Oak Ridge National Laboratory, 2006.

include filtering particulates as well as controlling activation products and fission products, including tritium, in the coolant. When flibe is used, tritium is produced that has a high permeability in most high-temperature metal alloys. Tritium control methods were partially developed but not demonstrated in earlier molten salt reactors and may be adapted for use in the FHR[39]. Much applicable work has since been done to control tritium in fusion systems[40] but the FHR contains graphite that absorbs much of the tritium. A robust system for controlling coolant salt chemistry will be necessary to prevent excessive corrosion in salt-facing components, especially heat exchangers. If salts containing beryllium are used in the FHTR, appropriate and practical methods to assure worker safety must be demonstrated[41].

- *The fabrication and operation of commercially-viable salt-facing components and structures must be demonstrated in high-temperature fluoride-salt environments under realistic reactor conditions.* The structural materials must withstand thermal stress fatigue, low and high cycle fatigue, creep fatigue, static tensile and compressive load types, and high temperatures, while remaining compatible with the liquid fluoride salt. Alloy-N, also called Hastelloy-N® and INOR-8, is the leading candidate for the structural components in the FHTR. Although tests have demonstrated good corrosion resistance of Alloy-N that was developed specifically for nuclear applications, Alloy-N has not been codified for use in Class 1 nuclear components, meaning it currently cannot be used in the design and construction of any commercial nuclear reactor. However, advanced materials can be used in a test reactor with appropriate test and inspection programs to assure the reactor stays within its safety limits[42]. The alternative is a low-nickel stainless steel assuming that the redox chemistry can be controlled to limit corrosion—a different option with a different set of challenges. Laboratory experiments will define the preferred option for an FHTR
- *Structures, systems, and components must be reliable, enable in-service inspection, be easily maintained, and be readily accessible for repair and replacement.* The FHTR will be used to demonstrate methods for in-service inspection, maintenance, and repair. One unique aspect of liquid salts is that they are transparent and thus the option of using various optical methods for inspection—with equipment modified for the high temperatures. By designing and arranging FHTR systems for in-service monitoring, the

---

[39] P. BARDET et al., "Design, Analysis and Development of the Modular PB-AHTR," *Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP '08)*, Anaheim, CA, June 8–12, 2008, Paper 8211, American Nuclear Society (2008).

[40] D. PETTI et al., *Jupiter-II Molten Salt Flibe Research: An Update on Tritium, Mobilization and Redox Chemistry Experiments*, International Symposium on Fusion Nuclear Technology, INEEL/CON-04-02465, May 2005.

[41] P. BARDET et al., "Design, Analysis and Development of the Modular PB-AHTR," *Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP '08)*, Anaheim, CA, June 8–12, 2008, Paper 8211, American Nuclear Society (2008).

[42] W. REN, et al., "Considerations of Alloy N for Fluoride Salt-Cooled High-Temperature Reactor Applications," *Proc. ASME 2011 Pressure Vessels & Piping Division Conference (PVP2011)*, Baltimore, Maryland, July 17-21, 2011, American Society of Mechanical Engineers (2011).

FHTR will provide extensive data on the corrosion resistance and mechanical performance of materials and components under prototypic conditions. By incorporating accessibility in the design and layout requirements, individual components can be quickly replaced. This would allow several variations of a single component to be tested without dramatically altering the plant[43]. In this context there is a major difference between a test reactor and a commercial reactor. The goals of a test reactor and its small size allow replacement of major components—components that would not normally be considered as replacement items in a commercial reactor.

- *The design will enable comprehensive quantification of nuclear, thermal, hydraulic, chemical, and material phenomena.* The FHTR may incorporate significantly more instrumentation than a commercial FHR will require. The primary mission of the FHTR is reactor system performance testing to enable the design and licensing of a FHR demonstration power reactor. Hence, this will require detailed spatial and temporal mapping of temperature and neutron flux. Additional requirements will be developed for mechanical, fuel, neutronic, and safety testing[44].
- *Safety tests will demonstrate inherent safety features, including feedback mechanisms and passive decay heat removal systems.* Transient tests will be designed to demonstrate the viability of passive systems, most importantly the Direct Reactor Auxiliary Cooling System (DRACS), for removing decay heat under normal and accident conditions. Operational tests must also be performed to demonstrate fuel loading, zero-power critical testing, startup and shutdown procedures, and preventative maintenance and in-service inspection methods[45].

4.3 What to Test

Before a test reactor is built, it is required to make a best estimate of what will be tested. This is required to assure the FHTR has the appropriate capabilities. Potential testing needs for a general-purpose FHTR are defined herein.

4.3.1 Fuels for Testing

Five classes of fuel are being considered for various FTRs that a FHTR may be required to test during its operating lifetime. In the early development of the LWR, there was rapid evolution of fuel types; a parallel history occurred with high-temperature reactor fuel. It would not be

[43] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 2013).
[44] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 2013).
[45] P. BARDET et al., "Design, Analysis and Development of the Modular PB-AHTR," *Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP '08),* Anaheim, CA, June 8–12, 2008, Paper 8211, American Nuclear Society (2008).

surprising if FHR fuels have a similar evolution and thus the need to consider future fuel options when considering design of a test reactor.

Graphite-Matrix Coated-Particle Pebble-Bed Fuel. The base-line fuel option is the graphite-matrix coated-particle fuel because of its (1) demonstrated high-temperature capability and (2) compatibility with high-temperature salts. It can be fabricated into many different forms. Our base-line FHR uses pebble-bed fuel (Fig. 4.1) partly because this geometry has been demonstrated in HTGRs in Germany and is currently being used in China. Relative to other concepts, refueling at high-temperatures with this fuel may be easier to accomplish.

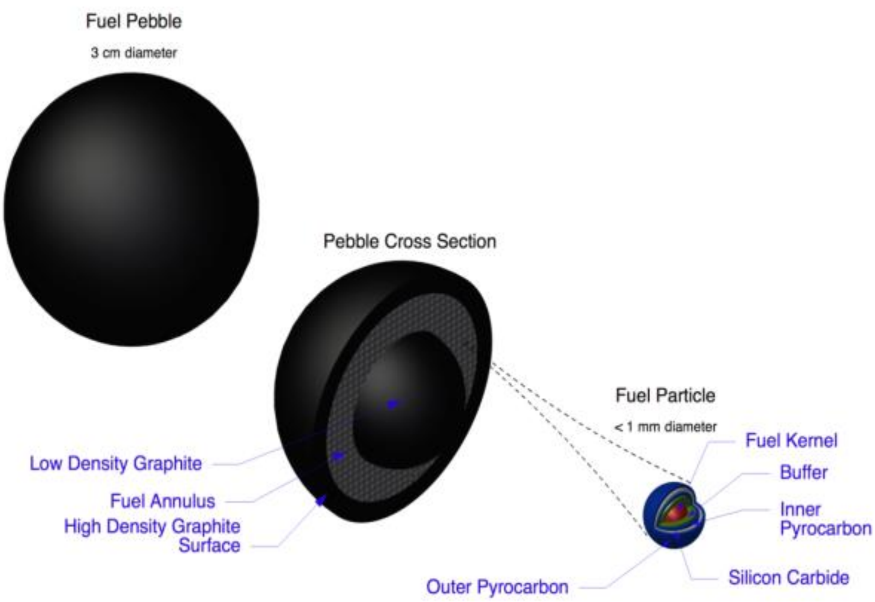


Fig. 4.1 Pebble-bed Coated-Particle Fuel

Graphite-Matrix Coated-Particle Fuel in Radial Moderator (FIRM). The other demonstrated fuel form in HTGRs is the prismatic block fuel. The HTGR prismatic fuel for a commercial FHR was first examined and then rejected. It did not have any advantages over the pebble bed fuel and there were other complications. The fuel is lighter than the coolant; i.e., the fuel floats. We early judged that refueling a core with stacks of floating prismatic blocks would be a significant engineering challenge[46, 47]. The prismatic fuel was adopted for our test reactor because this fuel design is proven and allows 3-dimensional loading of uranium and variations in enrichment and packing fraction that enables control of the flux distribution—something highly desirable in a

[46] C. W. FORSBERG, "Fuel Geometry Options for Salt-Cooled Advanced High-Temperature Reactors," CD-ROM, Paper 7405, *Proc. 2007 Int. Congress on Advances in Nuclear Power Plants (ICAPP '07),* Nice, France (May 13–18, 2007).

[47] C. W. FORSBERG, *Refueling Options and Considerations for Liquid Salt Cooled Very High-Temperature Reactors*, ORNL/TM-2006/92, June 2006.

test reactor. Furthermore, the test reactor is only one fuel block high and thus there are no issues associated with multi-block high refueling. Two developments resulted in our reconsideration of a hexagonal fuel assembly for the FHTR.

- *FIRM*. The FHTR design effort[48] resulted in a hexagonal fuel assembly design (Fig. 4.2) with significantly better neutronics and fuel cycle characteristics in terms of lowering enrichments and extending burnup than the more conventional designs of hexagonal blocks used in HTGRs. This assembly design is discussed in detail later in this report. Because salts are better coolants than helium, a prismatic fuel block for the FHR requires fewer or smaller coolant channels. This allows reconfiguration of the fuel assembly with the coolant channels and fuel in the center of the fuel assembly surrounded by graphite. Neutronically this allows fission neutrons to escape the fuel area into the surrounding graphite, slow down, and return to the fuel. Neutron moderation in fuel-free graphite significantly reduces parasitic neutron losses, reduces fuel enrichments, and extends fuel lifetimes.
- *Refueling*. A stack of prismatic fuel blocks can be held together by a tie rod to enable handling the entire stack as a “single” fuel assembly. This addresses the refueling challenge of prismatic blocks. The strategy also enables adding weight at either end of the stack so the fuel column will not float in the coolant salt. Tie rods are used to hold together smaller fuel bundles as a single assembly in the British Advanced Gas-Cooled Reactor to simplify refueling.

[48] J. RICHARD, B. Forget, C. Forsberg, and K. Smith, “Neutronic Comparisons of Liquid Salt Primary Coolants and Novel Assembly Design for a Fluoride Salt Cooled High-Temperature Test Reactor,” *Transactions American Nuclear Society*, Anaheim, California, November 9-13, 2014.

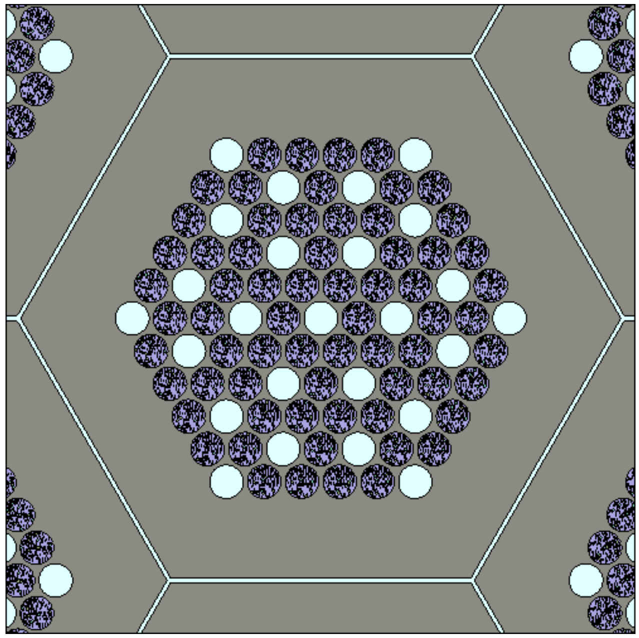


Fig. 4.2 Fuel in Radial Moderator Assembly

For the FHTR the fuel compact is in drilled holes in the graphite—the proven fuel assembly design used at the Fort St.Vrain high-temperature gas-cooled reactor. There are other more-advanced variants that potentially could lower fuel temperatures that has a positive impact on FHR reactivity control and transients. The Japanese High-temperature Test Reactor (HTTR), a helium cooled reactor, has the fuel compacts in graphite cylinders with annular coolant flow around each compact. There is also the option of lining each cooling channel with an annular compact with the coolant through the center of each fuel compact.

No full analysis of a commercial reactor core of this design has been completed. The uncertainties are significantly greater than with a pebble bed fuel design. However, it emphasizes that we are early in the development of an FHR—similar to the development of light-water reactors in the early 1950s. The design space is not well understood.

Graphite-Matrix Coated-Particle Plate Fuel. ORNL is proposing a plate fuel (Figure 4.3) that has a carbon-carbon backbone with a graphite-matrix coated-particle fuel layer on the flat surfaces. It is similar in geometry to many plate fuels used in research reactors. It enables a traditional core design using demonstrated materials; however, no such fuel assembly has been built or tested. Plate fuel as been proposed to address the complications of prismatic fuel blocks in a commercial FHR including refueling[49].

[49] V. K. VARMA et al., *AHTR Mechanical, Structural, and Neutronic Preconceptual Design*, ORNL/TM-2012/320, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September, 2012).

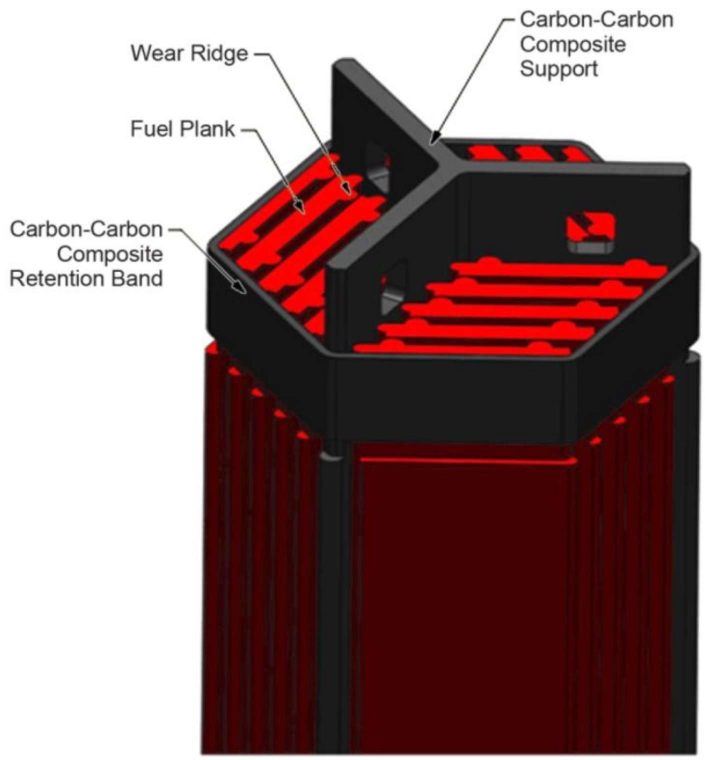


Fig. 4.3 Plate Assembly Coated-Particle Fuel

Silicon-Carbide-Matrix Coated-Particle Fuel. Silicon-carbide-matrix (SiC_m) coated-particle fuel is a variation of graphite-matrix coated-particle fuel that replaces the graphite matrix with SiC (Figure 4.4). The coated-particle fuel is unchanged. ORNL has done limited work on this advanced fuel. This fuel was originally proposed as a new matrix fuel for accident-tolerant LWR fuels but more recently has been proposed for the FHR[50]. The objectives of this substitution are to exploit SiC's resistance to radiation damage and to create a fuel form that is more robust under accident conditions. In a reactor, graphite first shrinks and then swells as a function of fast neutron fluence. SiC_m fuel provides much more dimensional stability under irradiation.

[50] C. W. FORSBERG et al., "Fluoride-Salt-Cooled High-Temperature Reactor (FHR) with Silicon-Carbide-Matrix Coated-Particle Fuel," Paper 6514, *Transactions of the American Nuclear Society, San Diego, CA* (Nov. 11–15, 2012).

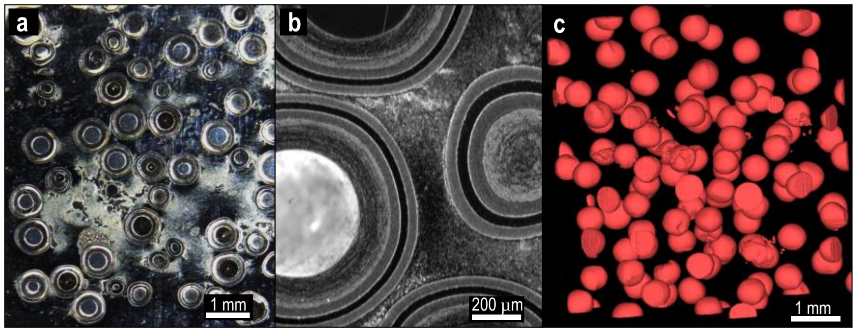


Fig. 4.4 Silicon-Carbide Matrix Coated-Particle Fuel

For small FHRs this fuel could be highly attractive because of its extreme resistance to external assault. SiC is a common component of armor. There is the potential to design a reactor with greatly reduced requirements for security and safeguards based on the ability of the fuel to withstand extreme events. A recent paper provides details on this option[51]. This is a longer term option than the other fuels described above.

Pin Fuel. The long-term option may exist to create an FHR with a pin-type fuel assembly. The graphite (the moderator) would be separated from the fuel. Conceptually this would be similar to the British Advanced Gas-Cooled Reactors (AGRs) with a graphite core and pin-type fuel assemblies (Figure 4.5). The fuel would be in pellet form such as UO_2 rather than a coated-particle fuel. This could potentially be a lower-cost fuel to fabricate and the pin design opens up a wider set of reactor design options. This is a longer-term option because of the need to develop a fuel clad. Two advanced clad options have been identified:

- *SiC clad*. While SiC has been developed as the cladding for coated particle fuel, the joining technology has not been fully perfected for sealing tubes. SiC[52] cladding is being developed to create an accident-tolerant fuel for LWRs. If it is successfully developed for LWRs, it becomes a potential option for FHRs.
- *Nickel alloy cladding*. Hastelloy®-N, which is compatible with fluoride salts, could be used for cladding except it has low radiation resistance. Recent work[53] indicates that advances in centrifuge technology may enable creating nickel alloys depleted in ^{58}Ni that could withstand high radiation fields. A major long-term R&D program would be required to determine the viability of such a cladding material.

[51] R. MACDONALD and C. W. Forsberg, "Strategies for a Small Transportable FHR with Reduced Security, Safeguards, and Safety Systems," *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, North Carolina, April 6-9, 2014.
[52] Y. KATOH et al. "Continuous SiC Fiber, CVI SiC Matrix Composites for Nuclear Applications: Properties and Irradiation Effects," *J. of Nuclear Materials* (2013).
[53] D. A. BLOORE and C. W. Forsberg, "Nuclear Applications of ^{58}Ni-Depleted Nickel Alloys and Superalloys", Paper 9343, *Transactions 2013 American Nuclear Society Winter Meeting*, Washington D.C., Nov. 10-14, 2013.

Areva examined the use of pin assemblies in FHRs[54, 55] and concluded major advantages for using pin assemblies in FHRs—assuming that a viable clad can be found. The fuel should be cheaper to manufacture. Such designs decouple fuel and moderator in the core. Graphite lifetime in a reactor is limited by fast neutron fluence, but in typical FHR configurations, the fuel lifetime in the reactor core is much less than the graphite lifetime. This results in unnecessary graphite waste production in configurations in which the moderator and fuel are inseparable (as is the case with pebble bed, plate, and prismatic-block fuel). The separation of fuel and moderator in the core simplifies refueling operations. The experience in refueling AGRs would be directly applicable except it would be easier because the FHR is a low-pressure system. The pin-type fuel assembly dramatically increases design options. One of the surprising results of early studies is that an FHR with this design could be the most efficient method to transmute actinides of any solid fuel reactor[56]. However, fuel assemblies using pins in FHRs are a longer-term option several decades into the future.

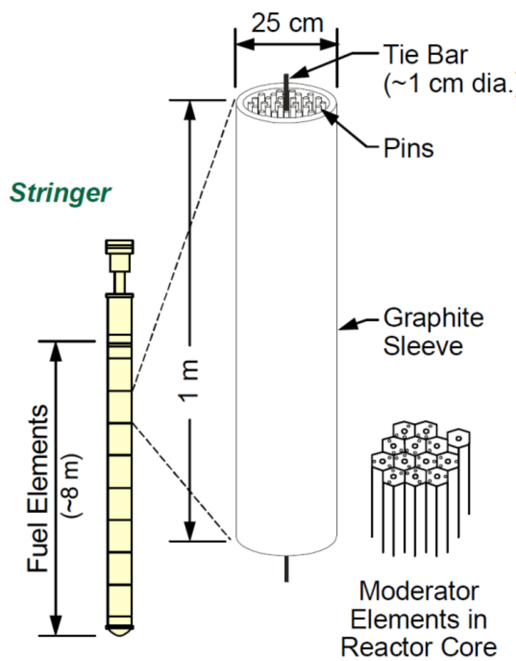


Fig. 4.5 British Advanced Gas-Cooled Reactor Fuel (Fuel Pins in Graphite Pile)

---

[54] W. A. CASINO, Jr., "Investigation of an Alternative Fuel Form for the Liquid Salt Cooled Very High Temperature Reactor (LS-VHTR)," CD-ROM, Paper 6244, *Proc. 2006 International Congress on Advances in Nuclear Power Plants (ICAPP '06), Reno, NV* (June 4–8, 2006).

[55] K. O. STEIN, R. A. Kochendarfer, and J.W. Maddox, "Use of a Liquid-Salt Nuclear Reactor (LS-VHTR) to Transmute Minor Actinides," *International Congress on Advanced Nuclear Power Plants, Anaheim, California, June 8–15, 2008,* American Nuclear Society, La Grange Park, Illinois, 2008.

[56] K. O. STEIN, R. A. Kochendarfer and J. W. Maddox, "Use of a Liquid-Salt Nuclear Reactor (LS-VHTR) to Transmute Minor Actinides," *International Congress on Advanced Nuclear Power Plants, Anaheim, California, June 8–15, 2008,* American Nuclear Society, La Grange Park, Illinois, 2008.

Independent from the fuel geometry or cladding is the choice of fuel. The coated-particle fuels use a uranium oxy-carbide fuel. There is significant work on uranium nitride fuel[57, 58] which offers the advantage of higher fuel densities that would lower the required fuel enrichment. These fuels may be future options for an FHR.

4.3.2 Coolants for Testing

There have been multiple studies of the advantages and disadvantages of different fluoride salt coolants for the FHR. The choice of a coolant involves complex tradeoffs among the salt's nuclear properties (for efficient and safe use of fuel), melting point, heat transfer properties, corrosion potential, level of induced radioactivity, handling properties, and cost. The coolant choices are coupled to the choice of the reactor's fuel, structural materials, size, and application. Table I summarizes the properties of the primary candidate coolants for the FHR. Appendix A provides added detail on choices.

Table 4.1 Candidate FHR Coolant Salts

			Heat Transfer Properties at 700°C					
Salt[a]	Melt point (°C)	900°C vapor press, (mm Hg)	Density (g/cm^3)	Vol. heat capacity (cal/cm^3-°C)	Vis. (cP)	Thermal conductivity (W/m-K)	Neutron capture relative to graphite	Moderator ratio
$LiF-BeF_2$	460	1.2	1.94	1.12	5.6	1.0	8	60
$NaF-BeF_2$	340	1.4	2.01	1.05	7	0.87	28	15
$LiF-NaF-BeF_2$	315	1.7	2.00	0.98	5	0.97	20	22
$LiF-ZrF_4$	509	77	3.09	0.90	> 5.1	0.48	9	29
$NaF-ZrF_4$	500	5	3.14	0.88	5.1	0.49	24	10
$KF-ZrF_4$	390	---	2.80	0.70	< 5.1	0.45	67	3
$Rb-ZrF_4$	410	1.3	3.22	0.64	5.1	0.39	14	13
$LiF-NaF-ZrF_4$	436	~5	2.79	0.98	6.9	0.53	20	13

[a]Molar compositions and boiling points: $LiF-BeF_2$(67-33) bp=1400°C; $NaF-BeF_2$ (57-43) bp=1400°C; $LiF-NaF-BeF_2$ (31-31-38); $LiF-ZrF_4$ (51-49); $NaF-ZrF_4$ (59.5-40.5) bp=1350°C; $KF-ZrF_4$ (58-42); $Rb-ZrF_4$ (58-42) bp=1450°C; $LiF-NaF-ZrF_4$ (26-37-37). Nuclear calculations used 99.995% Li-7. Advances in isotopic separation could change the relative ranking—particularly for zirconium where there has been significant work to reduce cross sections for use in LWR clad.

[57] T. M. BESMANN et al., "Uranium Nitride as LWR TRISO Fuel: Thermodynamic Modeling of U-C-N", *J. of Nuclear Materials*, **427**, 162-168, 2012.
[58] T. M. BESMANN et al., "Fission Product Release and Survivability of UN-Kernel LWR Triso Fuel", *J. of Nuclear Materials*, 2013.

Flibe was chosen as the baseline FHR coolant in previous studies for several reasons: lowest neutron absorption (efficient uranium usage), good heat transfer coefficients, negative void coefficient (reactor safety), reasonable physical properties, very low residual radioactivity, low corrosion potential, low vapor pressure with no "snow" generation in cover gas spaces, solubility in water that simplifies cleanup of equipment, and successful experience as the coolant in the Molten Salt Reactor Experiment (MSRE) – an 8 MWt molten salt reactor (with UF_4 and UF_3 dissolved in the flibe coolant) that successfully operated in the late 1960s. There is a recent thermodynamic evaluation of its properties[59]. However, flibe has several disadvantages:

- *Cost*. It is potentially expensive because of the need for isotopically separated ^7Li with an enrichment of >99.99% (typically 99.995%). If the cost of ^7Li is very high, there would be incentives to seriously consider alternative salts such as salts composed of sodium fluoride and beryllium fluoride.
- *Toxicity*. Beryllium is toxic. If beryllium toxicity becomes a major challenge, there would be incentives to consider alternative salts such as mixtures of lithium fluoride and zirconium fluoride.
- *Cost and Toxicity*. If both cost and toxicity become major challenges, there would be incentives to consider alternative salts such as mixtures of sodium fluoride and zirconium fluoride.
- *Tritium*. Neutron reactions with lithium and beryllium result in tritium production. The conversion of LiF to ^3HF causes primary system corrosion with the generation of $^3H_2$—a gas that can diffuse through hot heat exchangers. There are strategies to control corrosion and tritium but with added complications. Sodium zirconium salts avoid tritium generation.

Activities in other fields may impact salt choices. The current demand for ^7Li is ~1 ton per year that is used to control pH in pressurized water reactors. The primary use of lithium is in lithium batteries where isotopically-separated ^6Li may[60] significantly improve performance. If ^6Li is used in batteries, it would likely drive down the price of isotopic separation by several orders of magnitude eliminating ^7Li cost as a major consideration in the choice of coolants. Similarly there is ongoing work to separate zirconium isotopes to reduce parasitic absorption by zirconium clad in LWRs. If that technology is commercialized, it would significantly improve the relative performance of zirconium salts in FHRs. These factors indicate that the credible range of coolant salts when looking out several decades vary from lithium beryllium to sodium zirconium fluoride salts.

---

[59] O. BENES and R. J. M. Konings, "Thermodynamic Evaluation of the LiF-NaF-BeF_2-PuF_3 system", *The Journal of Chemical Thermodynamics*, **41,** 1086-1095, 2009.

[60] C. W. FORSBERG, "Future Cost of Isotopically Separated Lithium for PWRs, Fluoride-salt-cooled High-Temperature Reactors (FHRs), and Lithium Batteries", Paper 8712, *Transactions 2013 American Nuclear Society Winter Meeting*, Washington D.C., Nov. 10-14, 2013.

4.3.3 Subsystem Tests

The FHTR has many balance-of-plant components where there are alternative options that may be tested. Four systems deserve special attention because they require the radiation field of the reactor core for full scale tests.

- *Instrument and control*. Most instrumentation and control systems are similar to those in other reactors—particularly HTGRs. However there are exceptions. The coolant is a high-temperature transparent coolant that opens up new instrumentation options with unique capabilities that do require a reactor test bed[61, 62]. A periscope with laser range finding capability can do 3-dimensional mapping of surfaces. It may be possible to measure neutron flux (Cherenkov radiation), temperature, fluid flow, chemical redox, and impurity concentrations by remote optical methods. In some cases the gamma and neutron irradiation-generated light in the visible to blue spectrum (non-black body) is the signal. In other cases a laser light is sent in and the question is whether the light generated in that reactor will mask that signal. Only in-core tests can determine viability. If the instruments are to be used as part of safety and control in commercial reactors, a test reactor is required.
- *Chemistry and tritium control*. There are alternative methods to limit corrosion, control tritium, and cleanup the salt. A FHTR is the test bed to validate those options. Corrosion control and tritium behavior are tightly coupled to the neutron radiation field[63]. Neutron irradiation of the coolant generates tritium. Depending upon the coolant redox potential, the tritium is in the chemical form of H_2 or HF. The hydrogen form (H_2) tends to permeate through heat exchangers whereas the HF tends to corrode the heat exchangers. There are complex tradeoffs between corrosion, tritium permeation through heat exchangers, salt cleanup systems, and tritium control systems. Appendix A discusses in further detail some of the tritium control challenges.
- *Air heat exchangers*. Most of the surface area in the primary system is in the heat exchangers thus the heat exchanger surface drives system corrosion. The temperature changes across the heat exchanger drives temperature-driven corrosion mechanisms. Tritium losses out of the plant are via the heat exchanger. These considerations make testing of heat exchangers a FHTR mission. The FHTR is not required to test the ability of the heat exchangers to transfer heat—that can be done in non-nuclear facilities. It's the

[61] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 2013).

[62] C. W. FORSBERG, V. K. VARMA and T. W. BURGESS, "Three-Dimensional Imaging and Precision Metrology for Liquid-Salt-Cooled Reactors', Paper 163631, 5th International Topical Meeting on Nuclear Plant Instrumentation, Controls, and Human Machine Interface Technology, Embedded Topical: American Nuclear Society 2006 Winter Meeting, Albuquerque, New Mexico, November 12-16, 2006.

[63] J. D. STEMPIEN, R. G. Ballinger and C. W. Forsberg, "The Coupled Corrosion and Tritium Challenges of Fluoride-Salt-Cooled High-Temperature Reactors," Paper 14026, Proc. of ICAPP 2014, Charlotte, USA, April 6-9, 2014.

integrated reactor system chemical and tritium behavior as a function of the composition and design of the heat exchangers—tube metallurgy and any tritium control barriers.

- *SNF treatment/storage systems*. The behavior of these systems is dependent upon irradiation, decay heat, and salt-coolant carryover with the SNF. As such, the FHTR becomes the test bed for alternative SNF systems[64].

[64] C. W. FORSBERG and P. F. Peterson, "Spent Nuclear Fuel Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal, *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014),* Charlotte, North Carolina, April 6-9, 2014.

5 Test Reactor Design

5.1 Technical Requirements

Once the top level goals for the FHTR have been chosen, technical performance targets and constraints can be identified. Since the top-level requirements of an FHTR differ significantly from those of a commercial design, the technical requirements will also differ. These technical requirements can be classified into three broad groups: [1] performance targets; [2] feasibility/operability constraints; and [3] safety considerations. The specifics of each of these groups are discussed in the following sections.

The top level goal is a Class 1-A test reactor that is capable of supporting a wide set of FHR commercial design options. There are two strategies to achieve that goal.

- *Enable full test-reactor core replacement*. Test reactor facilities can be designed to enable replacement of the entire reactor core with a new core of a different design. The classic example is the Shippingport reactor that operated over its lifetime with three reactor core designs (Appendix E). Switching from one core design to another is a major undertaking.
- *Design reactor core to test variable fuels and coolants*.

We propose that the test reactor facility be designed with both capabilities. The first option primarily involves allowing sufficient space near the reactor core to enable core replacement. It imposes an added space requirement on the building and may require some additional facility requirements such as large crane lift capability. The second requires special core design. This chapter describes that test reactor design. Maintaining both capabilities is a method to reduce long-term programmatic risk. Because the FHR is a new concept, we do not know the full range of potential design options.

5.1.1 Performance Targets

5.1.1.1 Materials Irradiation Capability

The FHTR is envisioned as a Class 1-A general-purpose test reactor, one that will help form the licensing basis for commercial FHRs and also provide a high-temperature irradiation capability for multiple liquid salt coolants and fuel types. Key among the irradiation positions is the need for a large, high-flux irradiation position for near-full-size fuel testing. Two key performance targets have been identified to meet these irradiation needs: the size of the fuel irradiation position, and the irradiation flux available in that position. While smaller irradiation positions for materials irradiations are also important, they are more easily included into the design due to their smaller size.

- *Fuel Testing Position Size.* The design target size for a large fuel irradiation position is full core height with a diameter of at least 24 cm, as an empty bundle position in the core assembly lattice. Thus, the core assembly size must also be at least 24 cm in flat-to-flat hexagon distance. This position would be fully exposed to the high-temperature (650-700 °C) coolant conditions that the rest of the core is operating at, allowing for representative irradiation and burnup testing to occur. The size of the position was chosen to be larger than any single in-core position currently available in the DOE test reactor complex, and would allow for testing of pebble bed fuel in a fixed-pebble configuration[65], [66]. It would also allow for testing of ORNL plate-type fuel, though at a slightly reduced size[67].

- *Fuel Testing Position Irradiation Flux.* The large fuel testing position thermal flux is desired to be 3X greater than the driver fuel. This target was chosen to maximize the testing acceleration within the fuel performance constraints posed by thermally-driven failure mechanisms at higher power densities[68]. Setting a maximum on the thermal flux irradiation acceleration reduces the thermal gradients in the fuel matrix and across each TRISO micro particle, which helps to prevent fuel failures.

- *General Materials Irradiation Positions* Multiple small irradiation positions should be included in the initial design to facilitate smaller-scale irradiations, as is common with test and research reactors. Multiple positions should be included to enable concurrent irradiation campaigns to occur. While not all additional positions need to be specified in the preliminary design, the concept should include sufficient design flexibility to add or reposition irradiation volumes as the irradiation mission of the reactor is refined and clarified, or as new needs and capabilities are identified.

5.1.1.2 Primary System Coolant Flexibility

The FHTR should be able to support operation with either of two coolants, $NaF\text{-}ZrF_4$ or flibe, with potential for additional coolants to be used in the future. The choice of coolant impacts and is influenced by many factors, including cost, availability, and performance (both neutronic and thermal-hydraulic). Since $NaF\text{-}ZrF_4$ is more restrictive both from a neutronic (more parasitic capture) and a thermal-hydraulic (poorer thermal conductivity and heat capacity) perspective, it is far easier to design the FHTR to operate with $NaF\text{-}ZrF_4$ initially and later convert to flibe than

---

[65] Idaho National Engineering And Environmental Laboratory**,** *Users Handbook for the Advanced Test Reactor.* 2002. INEEL/EXT-02-01064.

[66] D. L. KRUMWIEDE et al., *Design of a Pre-Conceptual Pebble-Bed, Fluoride-Salt-Cooled, High-Temperature Reactor Commercial Power Plant.* Charlotte, NC: Proceedings of ICAPP 2014, April 6-9, 2014. 14231.

[67] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September. 2013)

[68] J. MAKI et al., *The challenges associated with high burnup, high temperature and accelerated irradiation for TRISO-coated particle fuel.* 270-280, Journal of Nuclear Materials, 2007, Vol. 371.

vice versa. Thus, the preliminary design of the FHTR will use NaF-ZrF_4 as the design coolant, and will identify the necessary design modifications (if any) for switching to flibe.

Designing for operation with either NaF-ZrF_4 or flibe also provides flexibility for potential future operation with other salt coolants. This is because flibe and NaF-ZrF_4 effectively bracket the range of potential binary salt melts, in that flibe is a neutronically-transparent lightweight salt with excellent heat transfer properties, while NaF-ZrF_4 is a neutron-absorbing (and thus activating) heavy salt with relatively poor thermal-hydraulic properties (heat capacity and thermal conductivity roughly ½ that of flibe)[69]. If alternate salts such as LiF-ZrF_4 or NaF-BeF_2 are desired for testing, they can be incorporated relatively easily since their neutronic and thermal-hydraulic properties tend to fall in between those of flibe and NaF-ZrF_4.

5.1.2 Feasibility/Operability Constraints

As may be obvious for any reactor, the design specified must be able to be manufactured and operated for it to be considered acceptable. However, determining the manufacturability of an advanced design is difficult, since no established supply chain is likely to exist and only limited research and development is likely to have been performed. This difficulty is exacerbated for a test reactor, since this reactor itself is to serve as a research and development tool for future designs. Thus, identifying practical design limitations is a necessary yet indeterminate challenge, with many requirements existing only as general estimates subject to significant uncertainty. Nonetheless, an effort has been made to determine reasonable appraisals of these limitations and to account for them in the preconceptual design.

5.1.2.1 Fuel Configuration

The initial design of the FHTR should use fuel and materials that are as well-understood and highly developed as possible. Since the FHTR is intended as a first step towards a viable commercial FHR design, it will necessarily be constructed well before commercialization commences. Thus, using fuels and materials that have had prior characterization and testing will facilitate this shorter time horizon for deployment.

The FHTR will thus use *coated-particle* fuel in a *prismatic-block* graphite matrix. Coated-particle TRISO fuel has been under development for high temperature gas cooled reactors for the last 30+ years, and is one of the most mature advanced fuel forms capable of high-temperature (above 700 °C) operation[70]. The prismatic-block graphite matrix structure was selected for the FHTR driver fuel because it allows for very fine and consistent control over fuel, moderator, and coolant volume fractions in the core (in contrast to pebble fuel) and can presently be manufactured (in contrast to plate-type fuel). One key limitation on the use of prismatic block fuel is the maximum fuel height; the blocks cannot be manufactured to a height greater than 2 m,

[69] D. F. WILLLIAMS, L. M. Toth, and K. T. Clarno, *Assessment of Candidate Molten Salt Coolants for the Advanced High-Temperature Reactor (AHTR).* ORNL/TM-2006/12 : Oak Ridge National Laboratory, 2006.
[70] D. A. PETTI, P. A. DEMKOWICZ and J. T. MAKI, TRISO-Coated Particle Fuel Performance. [book auth.] R. R. Hobbins. *Comprehensive Nuclear Materials.* Waltham, MA: Elsevier, 2012.

so core heights larger than 2 m must be obtained via block stacking[71]. Since block stacking adds considerable complexity, it is desired that the total core height not exceed 2 m, and be preferably closer to 1 m. The structural materials in the FHTR will generally be composed of graphite (for the in-core structures) or metal alloys such as Alloy N or, potentially, stainless steel 316 (for use as primary and secondary system piping, vessels, and heat exchangers).

5.1.2.2 Cycle Length

The minimum tolerable cycle length is specified as *6 months*. This cycle length was chosen to balance the desire for high fluxes and power using a relatively small fissile mass with the operational difficulties of refueling frequently. The total fissile mass of the FHTR is limited by the upper limit on TRISO particle packing fraction of 0.35, which is the fraction of the fuel compact volume occupied by TRISO microspheres. This stands in contrast to ceramic pellet UO_2 fuel found in light water reactors, which typically have an as-fabricated density of approximately 90%[72]. Furthermore, the actual fissile fuel volume is even less than 35% of the fuel channel volume, since the TRISO micro particle is comprised of both the UCO fuel kernel and several surrounding layers of protective graphite and silicon-carbide coatings. Thus, a short cycle length of six months (as compared to the typical commercial cycle length of 18-24 months[73]) was determined to be acceptable, given that the facility's capacity factor and uranium utilization are relatively unimportant for meeting the design goals provided the peak flux requirements are satisfied.

5.1.2.3 Fuel Enrichment

The maximum allowable fuel enrichment was specified as *20 weight %* uranium-235. This limit comes as a result of seeking to limit the reactor's proliferation threat, such that the fuel used remains below the IAEA definition of low-enriched uranium (LEU)[74]. LEU is considered less a proliferation risk than highly-enriched uranium (HEU).

5.1.2.4 Power Level

The power level of the FHTR should be between *20-40 MWt*. The power should be as low as possible to meet the minimum irradiation flux target in the fuel testing position. Selecting a lower core power alleviates the difficulties associated with engineering and technology development of various plant systems, such as heat exchangers, pumps, electrical heaters, and the like, by reducing the size and cost and improving the reliability of these systems. However, everything else being equal, a higher power will result in greater flux in the irradiation positions, so a minimum power has also been specified to help meet the irradiation flux design goals.

[71] T. BURCHELL, Personal communication. Oak Ridge National Laboratory, 2013.
[72] N. E. TODREAS and M. S. Kazimi, *Nuclear Systems I: Thermal Hydraulic Fundamentals.* New York: Taylor and Francis, 1990.
[73] N. E. TODREAS and M. S. Kazimi, *Nuclear Systems I: Thermal Hydraulic Fundamentals.* New York: Taylor and Francis, 1990.
[74] R. G. MURANAKA, *Conversion of research reactors to low-enrichment uranium fuels.* s.l.: IAEA Bulletin, vol. 25, No.1, 1980.

5.1.3 Safety Considerations

Every nuclear reactor must be able to operate safely during both normal and accident scenarios. However, the approaches to safety can be different in a test reactor than in a commercial power reactor. Test reactors are low power, low source term systems where the total risk is greatly reduced relative to large commercial designs. This is reflected in the alternate licensing path available to test reactors as part of class 104(c) per 10 CFR 50.21 as discussed in the next chapter.

5.1.3.1 Fuel Temperature

The peak TRISO fuel temperature must be kept *below 1250 ºC* during normal operation and *below 1600 ºC* during short-duration beyond design basis accident situations[75]. The operating temperature limit is set to maintain the integrity of the TRISO layers such that fission product gases are retained. The accident limit is based on successful retention of the fission products within the TRISO coatings. The 1600 °C limit was selected based on historical German experience with HTRs and is conservative: recent investigations by Idaho National Laboratory as part of the Next Generation Nuclear Plant program suggest that the actual limit may well be higher[76].

Separate from peak temperature is the average fuel temperature. There are large incentives to minimize this temperature to improve accident response capability of the FHTR and to enhance the ability to conduct transient testing.

5.1.3.2 Reactivity Coefficients

Net reactivity coefficients should be sufficiently negative such that passively safe behavior is achieved during both nominal and off-nominal conditions. The net power coefficient and isothermal temperature coefficient must be negative throughout all operating conditions of the FHTR. The coolant temperature coefficient and void worth will be made negative or as close to 0 as possible. A net negative power coefficient ensures safe neutronic behavior of the reactor in anticipated transient without scram (ATWS) or reactivity insertion (RIA) beyond-design-basis accidents, ensuring uncontrolled reactivity increases are precluded even in the absence of operator intervention.

---

[75] C. FORSBERG et al., *Fluoride-Salt-Cooled High-Temperature Reactors (FHRs) for Base-Load and Peak Electricity, Grid Stabilization, and Process Heat.* Cambridge : MIT CANES, 2013. MIT-ANP-TR-147.

[76] IDAHO NATIONAL LABORATORY, Next-generation nuclear fuel withstands high-temperature accident conditions: https://inlportal.inl.gov/portal/server.pt?open=514&objID=1555&mode=2&featurestory=DA_612467, 2013.

5.1.3.3 Shutdown Margin

The reactor's control systems must be able to reduce the reactivity of the reactor upon shutdown to *\$0 with a minimum margin of \$0.5*, for total core reactivity with all rods in (ARI) of at least -\$0.5 in the most reactive core configuration. The reactor's chain reaction must be able to be stopped using the control devices (in this case, control rods) intended for the purpose, with some margin included to account for modeling and data uncertainty. This must be the case for all materials irradiation campaigns and throughout the driver fuel's burnup lifetime. For research reactors, there is no "stuck-rod criterion" per the U.S. NRC as there is for commercial power reactors, where the shutdown margin must be satisfied with the most reactive rod stuck out.

5.2 Preliminary FHTR Design

5.2.1 Modeling and Simulation Methods

The Monte Carlo neutron transport code Serpent[77] was used to perform reactor physics simulations to analyze the core reactivity and burnup of the FHTR. ENDF/B-VII.0 continuous-energy cross sections were employed, with thermal scattering libraries used for graphite. The full double-heterogeneity of the TRISO-based fuel was modeled explicitly without the burden of excessive computational runtime. This was made possible due primarily to Serpent's use of the Woodcock delta-tracking transport method and a unionized energy grid for storing the continuous-energy reaction cross-sections[78, 79, 80]. Monte Carlo uncertainties associated with the presented results were 70 pcm or below in all cases.

1.1.2 Baseline Design Overview

This section presents an overview of the FHTR core, including important dimensions and performance characteristics. The core was developed as the result of exploring different assembly designs, core assembly layouts, reflector materials, and radial/axial reflector geometries. References to these design iterations are made only where exceptionally relevant in this section: interested readers are referred to an ICAPP conference paper[81] for a more complete discussion. Likewise, this current core is not a final preconceptual design: work remains to apply a rigorous optimization framework to result in a fully converged solution. However, while the specific dimensions of the coolant channels, fuel kernels, U-235 fuel enrichment, and core height

[77] J. LEPPÄNEN, *Serpent Progress Report 2011.* VTT Technical Research Center of Finland, 2012. VTT-R-05444-12.
[78] C. GENTRY et al., *Core Physics Parametric Studies for Liquid Salt Cooled Reactors.* Atlanta, GA: Transactions of the American Nuclear Society, June 16-20, 2013.
[79] J. LEPPÄNEN, *Performance of Woodcock Delta-Tracking in Lattice Physics Applications Using the Serpent Monte Carlo Reactor Physics Burnup Calculation Code.* 715-722, Annals of Nuclear Energy, 2010, Vol. 37.
[80] J. LEPPÄNEN, *Two Practical Methods for Unionized Energy Grid Construction in Continuous-Energy Monte Carlo Neutron Transport Calculation*. 878-885, Annals of Nuclear Energy, 2009, Vol. 36.
[81] J. RICHARD, B. Forget and C. W. Forsberg, "Core Design Evolution of a Fluoride-Salt-Cooled High-Temperature Test Reactor," *Proc. of ICAPP 2014,* Paper 14212, Charlotte, USA, April 6-9, 2014.

(among other features) are subject to change, the final design will closely resemble this design, since the present design has provided an acceptable basis for fine-tuning core design parameters.

5.2.1.1 Core Configuration and Dimensions

The preliminary FHTR is a Class I-A 20 MWt, graphite-moderated, liquid salt cooled reactor. The design coolant for the FHTR was chosen to be the most limiting acceptable coolant, which was 59.5% NaF-40.5% ZrF_4. Since NaF-ZrF_4 has increased neutron absorption, reduced heat capacity, and reduced thermal conductivity relative to flibe, a core designed to operate with NaF-ZrF_4 can be converted to operate with flibe, whereas the reverse would not be possible. See Fig. 5.1 for a top-down view of the core, Fig. 5.2 for a side view, and Table 5.1 for some of the principal design parameters.

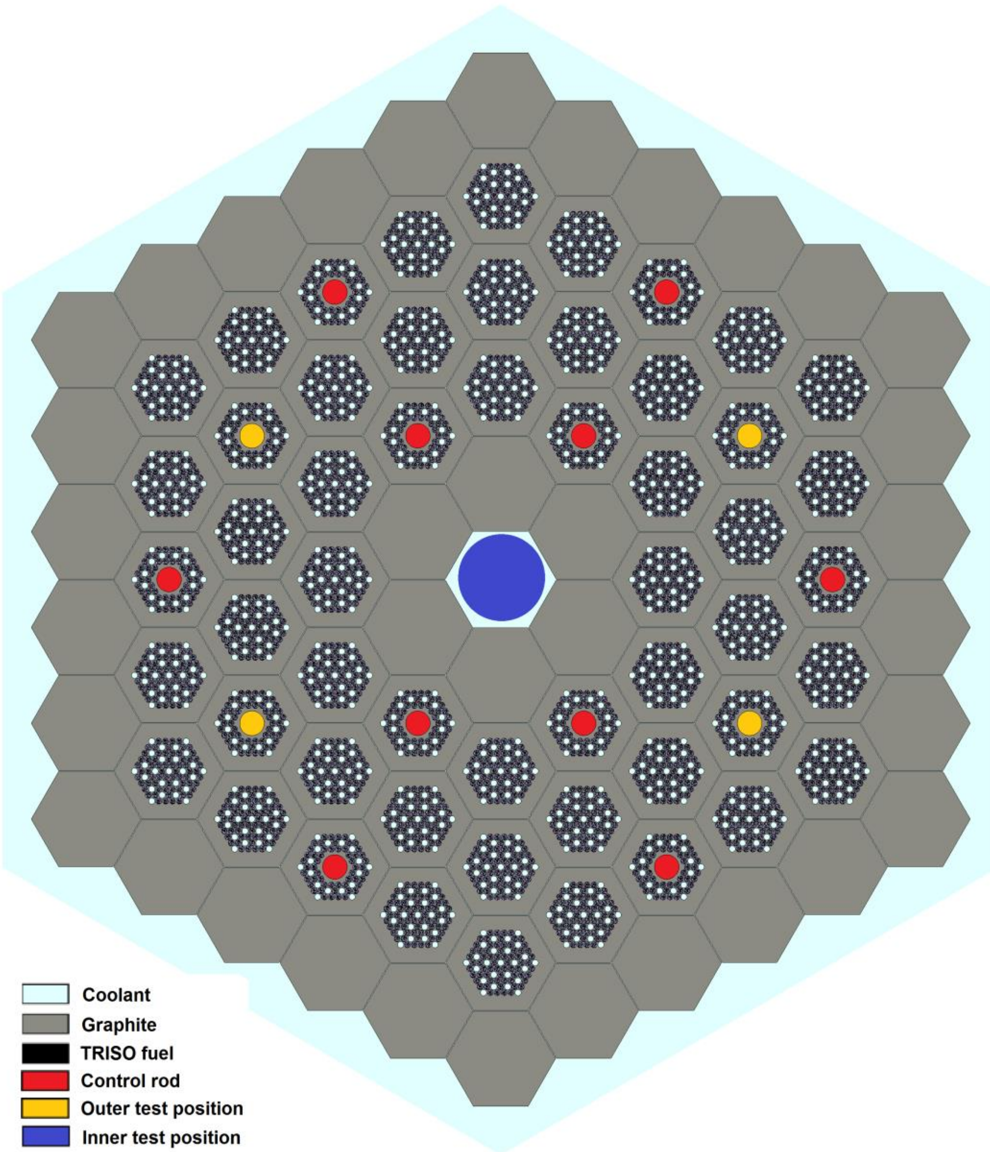


Fig. 5.1 Top-down view of FHTR core, as modeled in Serpent

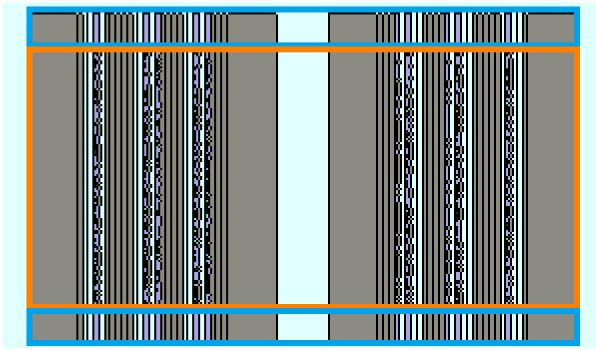


Fig. 5.2 Axial view of FHTR core, as modeled in Serpent

Table 5.1 Selected Design Parameters of the Preliminary FHTR Design

Power	20 MWt
Fuel form	UCO-kernel TRISO
UCO kernel radius	350 μm
Assembly type	Prismatic graphite block
Fuel enrichment	19.5 a% U-235
Primary coolant salt	NaF-ZrF_4 or LiF-BeF_2
Core outlet temperature	700 °C
Core inlet temperature	650 °C
Number of Fuel Assemblies	54
Total Fissile Mass	146 kg
Assembly flat-to-flat width	24.8 cm
Assembly pitch	25 cm
Fuel/coolant channel radius	8 mm
Fuel/coolant channel pitch	1.8 cm
Inner fuel irradiation position radius	12 cm
Outer materials irradiation position radius	3.2 cm
Number of outer irradiation positions	6
Active core outer diameter	2.25 m
Active core inner diameter	0.75 m
Active core height	1.35 m
Outer reflector thickness	0.25 m
Inner reflector thickness	0.25 m
Axial reflector thickness (each)	0.15 m
Core diameter	2.75 m
Core height	1.75 m

The core (including the internal and external reflector assemblies) has a diameter of 2.75 m and a total core height (including axial reflectors) of 1.75 m. The FHTR uses a pool-type primary system configuration, where the core is submerged in a large vessel with an integral heat exchanger, thus limiting the primary flow loop to a single component. Thus, while smaller than proposed commercial designs[82], the FHTR is larger than traditional water-cooled test reactors in operation today, such as the Advanced Test Reactor (ATR) at Idaho National Laboratory[83]. ATR has a square-cylinder core with an active height and width of 1.2 m and employs a loop-type primary circuit, where the primary coolant (light water) is pumped into the core vessel, out through external piping to a heat exchanger, and then back into the core vessel.

The FHTR core is necessarily larger than the ATR core for two primary reasons. First, the pool-type arrangement necessitates that the reflectors sit inside the vessel, which is why the FHTR total core size includes the reflector dimensions. But second, and more importantly, the mean free path of fission neutrons (fast neutrons) traveling through light water is ~2 cm, whereas for graphite the mean free path of fission neutrons is ~4 cm[84]. Since the fission neutrons can be expected to travel 2-3 mean free paths before being absorbed, this means that for a graphite-moderated reactor, the neutrons will tend to travel twice as far before causing an additional fission than in a water-moderated reactor. If the reactor core is small relative to the diffusion length, neutrons are likely to leak before causing fission, which decreases the core's reactivity substantially. Thus, for a graphite-moderated core to have similar reactivity to a water-moderated core, the graphite core will be larger, all else being equal.

The larger mean free path of neutrons in graphite also reduces the FHTR's power density relative to water-cooled reactors. The ATR typically operates at a power level of 110 MWt, which gives an average power density of 81 kW/L. However, the FHTR's nominal power is 20 MWt, which gives an average power density of 2 kW/L. Since power density is tightly coupled to peak neutron flux in a given volume, it is apparent that the peak irradiation flux of the FHTR cannot approach that of a small water-cooled reactor like the ATR because of the basic design selections of a pool-type primary system and a graphite-moderated core. These two design selections were made in support of the top-level goals of the FHTR: The pool-type primary system was selected to maximize passive safety (such as passive decay heat removal), and the graphite moderator was selected because: [1] a water moderator is not capable of staying liquid at the design temperatures without excessive pressurization, which is a cost, feasibility, and safety concern, and [2] the commercial FHR intends to use graphite as the primary moderating material, so using it in the test reactor will be an important part of enabling the FHTR to successfully form a licensing basis for the commercial design. Furthermore, alternate reflector

---

[82] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 2013).

[83] Idaho National Engineering and Environmental Laboratory**,** *Users Handbook for the Advanced Test Reactor.* 2002. INEEL/EXT-02-01064.

[84] P. RINARD, *Neutron Interactions with Matter.* In *Passive Nondestructive Assay of Nuclear Materials*, U.S. Nuclear Regulatory Commission, 1991. NUREG/CR-5550.

materials such as beryllium metal and beryllium oxide are not compatible with being in direct contact with the fluoride salt coolant, and so would necessitate some sort of untested cladding for operating in the pool-type primary system, which adds increased technology risk and should be avoided for the test reactor wherever possible.

The FHTR is designed as a Class I-A test reactor—reactor technology development. It can operate as a Class II test reactor as an irradiation source for fuels testing for salt-cooled reactors. One could install isolated loops to allow testing of HTGR fuel in a helium loop with higher power densities than found in helium-cooled reactors. It would be possible to install water or liquid metal loops, however, the basic neutronics as described above would limit its capabilities as a Class II test reactor for these fuels. The reactor physics enables it to be a very capable Class II test reactor for reactors with graphite moderator but more limited capabilities to test fuels from other reactor types.

The FHTR uses tri-structural isotropic (TRISO) coated-particle fuel in a prismatic-block configuration. TRISO fuel is the most highly developed of any high-temperature fuel form available today[85], making it the most credible choice for a first-of-a-kind high temperature test reactor. The prismatic block configuration was chosen to allow for tight three-dimensional control over important parameters such as fuel packing fraction, moderator-to-fuel ratio, and fuel-to-coolant ratio. As mentioned earlier, these design considerations cannot be satisfied by pebble-bed configurations, and other proposed fuel forms (such as plate-type fuel) have no manufacturing basis. Prismatic block fuel, however, has a reasonable experience and manufacturing base, having been used in the currently-operating Japanese HTTR high-temperature gas cooled reactor (HTGR)[86], as well as General Atomics' Fort St. Vrain HTGR[87], among others.

[85] D. PETTI, P. Demkowicz and J. Maki, *TRISO-Coated Particle Fuel Performance*. In *Comprehensive Nuclear Materials*, Elsevier, 2012.
[86] J. BESS et al., *Evaluation of the Start-Up Core Physics Tests at Japan's High Temperature Engineering Test Reactor (Fully-Loaded Core).* Idaho National Laboratory, 2009. INL/EXT-08-14767.
[87] A. L. HABUSH and A. M. Harris, *330 MW(e) Fort St. Vrain High-Temperature Gas-Cooled Reactor.* 4, 1968, Nuclear Engineering and Design, Vol. 7, pp. 312-321.

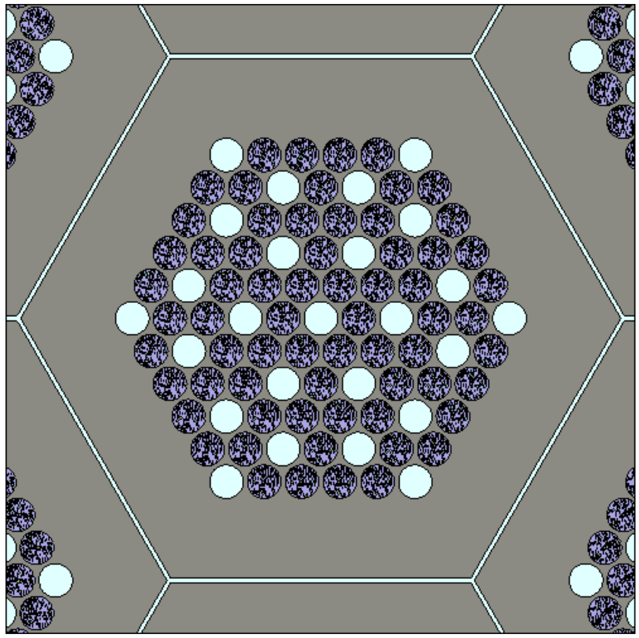


Fig. 5.3 Top-down view of the FHTR fuel assembly. The gray region around the outside of the block is solid graphite, while the purple-and-black cylinders in the interior of the assembly are the fuel compacts. The light cyan colored cylinders are the liquid salt coolant channels.

The FHTR assembly design, in particular the arrangement of fuel pins, differs substantially from that of previous prismatic-block assemblies (see Fig. 5.3). The outer region of each block contains no fuel or coolant channels, which are all located in the interior of the assembly. Thus, the fuel is surrounded by a solid, fuel-free moderator region. Fuel Inside Radial Moderator (FIRM). This stands in contrast to previous HTGR prismatic block assemblies, which evenly distribute fuel and coolant channels in a 2:1 ratio throughout the assembly[88]. The FHTR's distinctive FIRM assembly design was selected to increase the moderator to fuel (M/F) ratio to improve core reactivity, as infinite-assembly simulations showed that the core tended to be significantly under moderated at high fuel particle packing fractions. Note that because liquid salts are much better coolants than helium, the volume fraction of the core dedicated to cooling channels is much smaller enabling the outer zone of graphite in the fuel assembly.

In large cores where leakage effects are less important, increasing the M/F ratio can be accomplished by either reducing the fuel particle packing fraction (as was investigated), or by reducing the radius of the fuel compacts. However, in a small core such as the FHTR, this is less effective because as the packing fraction is reduced, the average neutron mean free path increases, thus increasing the leakage and negating the improved reactivity associated with a higher M/F. Additionally, in a small core with few fuel assemblies, exceptionally low packing

[88] D. T. INGERSOLL et al., *Status of Physics and Safety Analyses for the Liquid-Salt-Cooled Very High-Temperature Reactor (LS-VHTR).* Oak Ridge National Laboratory, 2005. ORNL/TM-2005/218.

fractions can mean a very low fissile mass loading, making it difficult to meet even the modest goal of a six-month cycle length.

However, spatially separating the fuel and moderator regions enables the core to achieve a higher average M/F while still retaining zones with significant neutron absorption (aka a small mfp) in the fuel pins when using a high particle packing fraction. These fuel absorption zones reduce the neutron leakage such that the core's reactivity is increased relative to having an assembly with all fuel pins. Perhaps even more importantly, this geometric heterogeneity provides significant spatial self-shielding of the absorption resonances in U-238, increasing the resonance escape probability substantially, resulting in a much improved neutron economy.

5.2.1.2 FHTR Burnup Performance

As noted in Section 5.1, the FHTR core was not designed to maximize cycle length, but to meet a minimum cycle length target of 6 months. This limit was chosen not to minimize fuel cycle costs (as in a commercial reactor) but to reduce the operational burden of having to perform refueling operations frequently. Using Serpent's internal depletion capability, the cycle length achievable with the FHTR's preliminary design was estimated, as shown in Fig. 5.4.

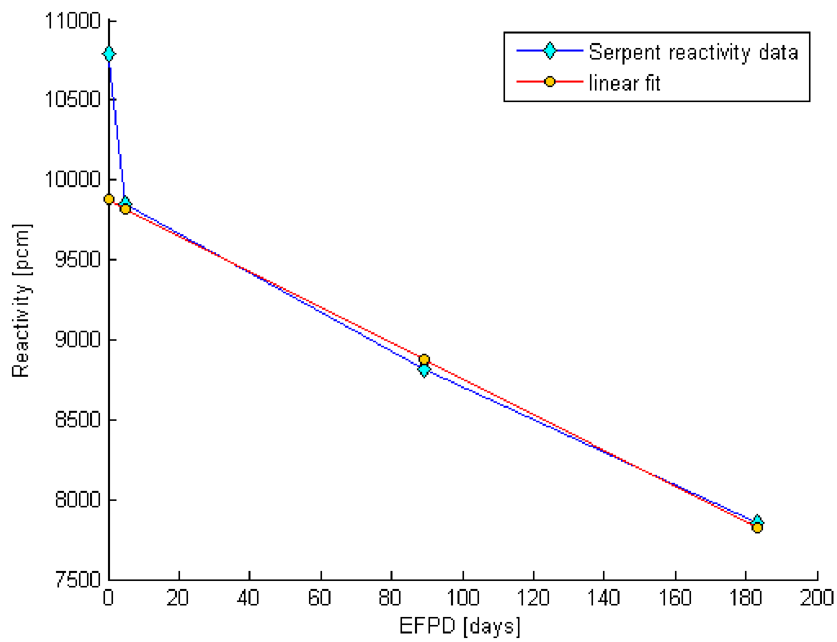


Fig. 5.4 Core reactivity (in pcm) as a function of cycle length (in effective full power days, or EFPD) for both Serpent-generated data and an associated linear regression

As shown in Fig. 5.4, the preliminary design of the FHTR has significant margin to the minimum cycle length of 6 months (~184 EFPD). The initial reactivity drop is due to xenon-135 fission product poisoning, and is worth approximately 940 pcm of reactivity (9% of initial reactivity). Fitting the post-xenon reactivity vs. burnup data with a linear regression, the predicted cycle length is 880 days, or close to 30 months. In reality, however, the actual achievable burnup would be limited not only by reactivity but also total fission gas release in the TRISO kernels, which would result in a much lower actual burnup than the reactivity-limited burnup.

This burnup analysis does suggest that the U-235 fuel enrichment could be reduced while still satisfying the cycle length constraint. Reduced enrichment is desirable from cost, criticality safety, and nonproliferation perspectives. Additionally, the flux in the irradiation positions is inversely proportional to the fuel enrichment (all else being equal), so lower enrichments also benefit the FHTR's optimization objective function of maximizing irradiation fluxes. Further quantification of the minimum enrichment required to satisfy the irradiation flux goals and performance constraints will be performed as part of the forthcoming optimization analysis.

5.2.1.3 FHTR Irradiation Fluxes

The FHTR has two broad categories of irradiation positions: a single large irradiation position at the center of the core surrounded by moderating graphite blocks intended for commercial FHR and high-temperature reactor fuel testing, and four smaller irradiation positions located inside hybrid fuel/irradiation position assemblies in the active fuel region of the reactor. The size, location, and number of irradiation positions was selected based on a very early estimate of the needs of a general-purpose FHR test reactor; a large fuel irradiation position for accelerated testing of FHR commercial fuel at near-scale, and several smaller irradiation positions with a harder flux spectrum for structural materials irradiations. The FHTR core was designed to be as reconfigurable and flexible as possible in this preconceptual stage to meet evolving irradiation needs as the design progresses, with fully removable radial reflector assemblies and outer irradiation positions integrated into fuel assemblies to allow for increasing or decreasing the number of irradiation positions as required for any given irradiation campaign. Soft irradiation spectra positions can be obtained by replacing solid graphite reflector assemblies with graphite reflector assemblies that contain the desired number and size of irradiation positions, and the size of the integrated-fuel irradiation positions can also be altered depending on the desired number of positions or spectrum required.

For the initial configuration of FHTR irradiation positions, irradiation fluxes were estimated using Serpent. All flux estimates were obtained with the coolant salt $NaF\text{-}ZrF_4$ flowing through the irradiation volumes. The resulting neutron fluxes of the inner fuel irradiation position and a single outer irradiation position are displayed in Table 5. Note that given the symmetry of the present core design, the outer irradiation positions are all expected to experience identical neutron flux.

Table 5.2 Irradiation Fluxes and Volumes for the Preliminary FHTR Core

	BOC	EOC	% change
Central fuel irradiation position thermal flux [n/cm^2-s]	4.45E+13	4.53E+13	1.97%
Central fuel irradiation position fast/thermal flux ratio	0.943	0.949	0.643%
Central fuel irradiation position volume [cm^3]	71,906		-
Outer small irradiation position fast flux [n/cm^2-s]	4.07E+13	4.25E+13	4.35%
Outer small irradiation position fast/thermal flux ratio	12.0	12.7	5.71%
Outer small irradiation position volume [cm^3]	4,343		-

The central fuel irradiation position has an average thermal flux of 4.45E13 n/cm^2-s at the beginning of cycle (BOC) over the entire 24.8-cm flat-to-flat, 135 cm tall hexagonal prism that forms an empty assembly-sized lattice position in the FHTR core. This flux value is less than the average thermal flux of a similar-sized research reactor, the 10 MWt Missouri University Research Reactor (MURR), which has an average thermal flux of 5E14 n/cm^2-s[89] over a smaller volume (a 8.75-cm diameter, 100 cm tall cylindrical flux trap at the center of the core)[90]. However, given the limitations on graphite-moderated pool-type reactors as discussed previously, this irradiation flux is acceptable given the FHTR's strategic goals of design flexibility and feasibility. The fast flux, which, in this case, uses the reactor engineering definition of fast flux to mean all flux above the thermal cutoff of 0.625 eV, is 4.07E13 $n/s/cm^2$ at BOC in an irradiation volume that consists of one of six 6.4 cm diameter, 135 cm tall cylindrical positions located inside specially designated fuel assemblies.

As expected, the irradiation fluxes of the FHTR increase slightly throughout the six month cycle length. The thermal flux in the central irradiation position increases 2% from beginning-of-cycle (BOC) to end-of-cycle (EOC), while the fast flux in the outer irradiation positions increases 5.7%. This is primarily due to the burnup of fissile material which requires an increase in fluxes to maintain a constant power level, given that power level is directly proportional to the fission reaction rate in the core. A two dimensional flux map of the thermal spectrum is shown in Fig. 5.5.

[89] N. J. PETERS, J. D. Brockman and J. D. Robertson, .*A new approach to single-comparator instrumental neutron activation analysis.* 2, 2012, Journal of Radioanalytical and Nuclear Chemisty, Vol. 291, pp. 467-472.

[90] C. CATHY, *Current and future use of radionuclides produced from nuclear reactors for medical applications.* Boston, MA: s.n., 2010. NUCL Symposium: Radiochemistry at the Facility for Rare Isotope Beams (FRIB).

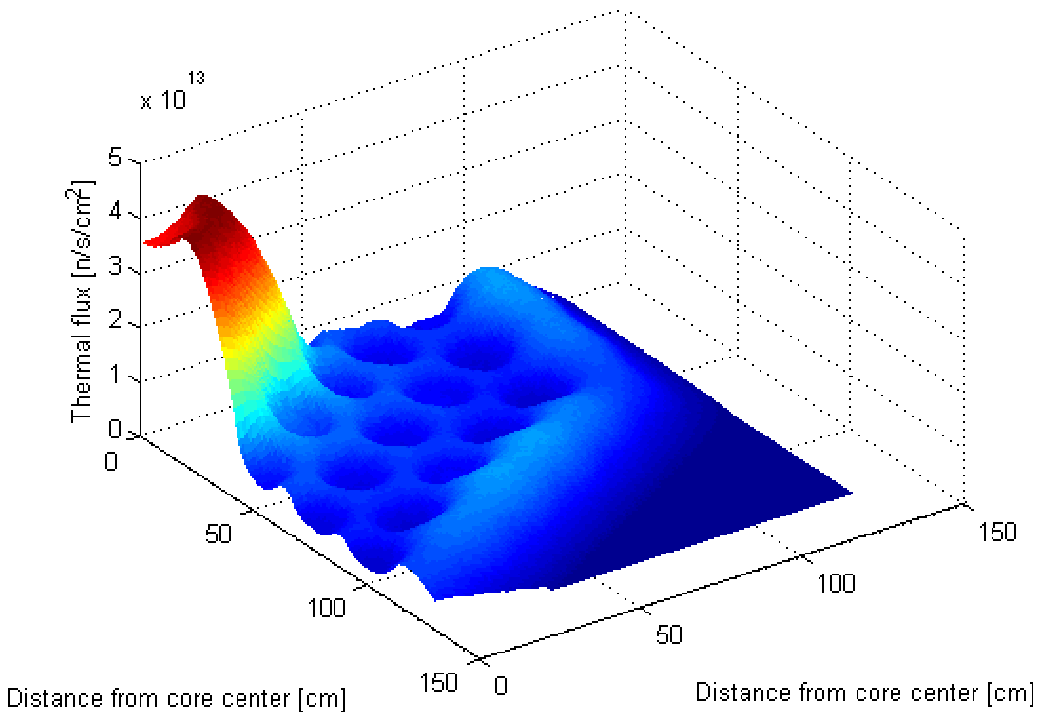


Fig. 5.5 Test Reactor Thermal Flux Profile

6 Licensing

6.1 Licensing Strategies

In the United States are two test reactor licensing strategies if the reactor is owned by the U.S. Department of Energy

- *U.S. Department of Energy (DOE).* If the DOE is the owner of the FHTR, it can chose to license (regulate) the reactor
- *U.S. Nuclear Regulatory Commission (NRC).* The NRC licenses all non-DOE non-U.S. Navy non-NASA reactors. At the request of DOE, the NRC can license a DOE reactor; thus, DOE has the option to license (regulate) its own reactors or use the licensing (regulation) process of the NRC.

The current policy is to license new DOE reactors through the NRC; however, a separate decision would have to be made about an FHTR. The current policy is based on considerations of licensing an existing type of advanced reactor such as a high-temperature gas-cooled reactor or a sodium fast reactor where there is previous experience. It was not based on licensing a first of a kind Class-I test reactor—something that has not occurred in over 40 years. Furthermore, one would expect that a FHTR would be built on one of the large DOE reservations.

DOE Licensing

If DOE is the authorization authority, the owner of the site would be responsible for the certification of the reactor. Nominally this would be a particular office within the DOE. If the reactor is greater than 20 MWt it will follow REG.Guide 1.70 for the form and content of the SAR. This is the same as used for a commercial reactor under the NRC.

Department of Energy (DOE) licenses and regulates several test reactors located at national laboratory sites. This includes licensing the two largest DOE reactors: the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL) and the High Flux Isotope Reactor (HFIR) at ORNL. In addition, the DOE licenses a large number of one-of-a-kind DOE nuclear facilities such as the Spallation Neutron Source that contain significant radioactivity.

NRC Licensing

If the decision was made to license an FHTR using the NRC, the basic approach is defined. According to the Code of Federal Regulations (CFR), there are two categories of NRC nuclear reactor licenses. 10 CFR 50.22 states that Class 103 licenses are for commercial and industrial facilities and are issued to authorize[91] "a production or utilization facility for industrial or commercial purposes." 10 CFR 50.21 states that class 104 licenses are for medical therapy and

[91] Class 103 licenses; for commercial and industrial facilities, http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0022.html. Last updated March 1, 2013.

research and development facilities. Class 104(c) licenses are issued to authorize[92] "production or utilization facility, which is useful in the conduct of research and development activities."

A test reactor, as defined by the U.S. Nuclear Regulatory Commission (NRC) in 10 CFR 50.2, is authorized to operate at thermal power higher than 10 MW or contains (1) a circulating loop through the core for fuel experiments; or (2) a liquid fuel loading; or (3) an in-core experimental facility with cross section in excess of 16 square inches. Design, operation, and safety considerations apply to both test and research reactors; however, test reactors may be subject to additional requirements such as conduct of license hearings and review by the Advisory Committee of Reactor Safety (ACRS)[93]. The only test reactor currently licensed by the NRC is the 20-MW Neutron Beam Split-core Reactor (NBSR) operated by National Institute of Standards and Technology (NIST), a federal technology agency.

There are cases when a research reactor facility incurs significant revenue from production activities, such as isotope and radiopharmaceutical productions. To determine whether a Class 103 license is required for such facility, 10 CFR 50.21 states that

> *"such facility is deemed to be for industrial or commercial purposes if the facility is to be used so that more than 50 percent of the annual cost of owning and operating the facility is devoted to the production of materials, products, or energy for sale or commercial distribution, or to the sale of services, other than research and development or education or training."*

Hence, the criterion for Class 104(c) license is that more than 50% of the cost of owning and operating the facility is devoted to research, development, education, or training. Rarely would a research or test reactor be Class 103 since they typically operate with production revenue much lower than facility operating cost. This clause, however, may potentially enable the FHTR to operate with an experimental NACC power cycle and generate revenue from power production and still maintain a 104(c) license.

If licensed by the NRC the FHTR will follow NUREG 1537 which is not LWR specific. In this context is noted that the NRC has recently undertaken the start of licensing activities for the proposed B&W Mo-99 Aqueous Homogenous Reactor—a type of reactor never before licensed (the AEC built several aqueous homogenous reactors in the 1950s). That reactor will be licensed as Class 103 reactor because it is a net revenue producer. However because of the unusual characteristics of this reactor the application will be reviewed by the research reactors branch of NRR, and will use an Interim Staff Guidance Document (ISG) that has been prepared[94] for use with NUREG 1537 especially for this one-of-a-kind reactor. This suggests the likely pathway for a FHTR if licensed by the NRC—a rewrite of NUREG 1537 explicitly for the FHTR. This, with

---

[92] Class 104 licenses; for medical therapy and research and development facilities, http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0021.html. Last updated March 1, 2013.

[93] NUREG-1537: *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, 1996.

[94] S. Bajorek et al., *Aqueous Homogenous Reactor Technical Panel Report*, BNL-94462-2010, 2010.

other licensing actions[95] for new reactors, is defining the pathway for an FHTR license if licensed by the NRC.

The licensing strategy for the FHTR should minimize uncertainty and the time required to complete the licensing process. If the FHTR is licensed by the NRC, the FHTR will primarily be used for research and development; therefore, Class 104(c) license would be pursued. A phased licensing approach is recommended—an initial low-power license and a power uprate to its full power level. The content requirements for the Class 104(c) test reactor Safety Analysis Report (SAR) for the license application and the standard review plan for the NRC staff are detailed in NUREG-1537[96]. The primary objective of the SAR is to demonstrate that there is a reasonable assurance that the general public around facility site will be protected from radiological risks resulting from the steady-state operation and various accident scenarios of the reactor facility. NUREG-1537 provides guidance from the NRC on the analysis required to demonstrate sufficient confidence in reactor safety. Since NUREG-1537 was developed specifically for light water reactors (LWRs), there are some areas of NUREG-1537 that would have to be revised for an FHTR. However, the licensing process of the FHTR would likely be similar to existing research reactor facilities. Based on the definition of Class 103 and 104(c) licenses, a regulatory evaluation indicates an FHTR would be licensed as a 104(c) test reactor by the U.S. Nuclear Regulatory Commission (NRC).

Evaluation of Licensing Options

An FHTR is a Class I test reactor—a reactor type that has not been built in the U.S. in over 40 years. As a consequence, there is no experience base for Class I reactors for making decisions on the appropriate licensing strategy. Neither the NRC or the DOE or the existing licensing regulations existed when the last Class I reactor was built in the U.S. The preparation of this report included the preparation of a draft report followed by a workshop of experts on test reactor design, operations and licensing (Appendix D and Appendix E). One of the major challenges addressed by the workshop was FHTR licensing where the licensing options were discussed with agreements on some aspects of licensing and disagreements on other aspects of licensing.

The development of a reactor involves three phases: test reactor, pre-commercial demonstration reactors, and commercial reactors. There was agreement that pre-commercial demonstration reactors and commercial reactors must be licensed by the NRC. One of the goals of a pre-commercial reactor is to provide information to determine reactor economics. That requires an understanding of licensing requirements for commercial reactors and thus the pre-commercial reactor must be licensed by the NRC. There was agreement that both FHTR licensing approaches provide equivalent safety and public input—but by different mechanisms

[95] NUREG-0800: *Standard Review Plan*, U.S. Nuclear Regulatory Commission.

[96] Class 104 licenses; for medical therapy and research and development facilities, http://www.nrc.gov/reading-rm/doc-collections/cfr/part050/part050-0021.html, Last updated March 1, 2013.

There were strong disagreements about the appropriate licensing strategy for the FHTR. Part of this discussion was about the differences between a Class I versus Class II test reactors and demonstration reactors. The FHTR would be a Class 1 test reactor—a first of a kind machine where many changes in its design and operation would be expected because the reactor is what is being tested. This is different from a Class II materials test reactor such as the Advanced Test Reactor at INL where the reactor is not the test—it's the test machine that would be expected to use proven technology with few changes after it is licensed.

The rational for NRC regulation of an FHTR is that:

- It is current DOE policy
- The NRC needs to understand the technology for future licensing of a pre-commercial FHR demonstration plant.
- The NRC has acquired recent experience from the early licensing steps for a homogenous aqueous reactor for medical isotope production.

The rational for DOE regulation of an FHTR is that:

- The NRC regulatory structure is not designed to license Class I test reactors and thus it would be difficult to license and operate such a reactor on a reasonable schedule. The NRC is organized to license a reactor on the assumption that it is built and does not undergo large changes; that is, the design remains fixed. That is not true for a Class I test reactor. Any license of a Class I test reactor would be a one-of-a-kind license for the NRC with one-of-a-kind procedures.
- The DOE has a successful history of regulating one-of-a-kind nuclear facilities in addition to reactors—such as the Spallation Neutron Source at ORNL. The DOE has isolated sites where part of the mission is one-of-a-kind testing of nuclear systems.
- Licensing is of a test reactor at a reactor test site where the safety case is coupled to the site—reverse of the NRC where licensing the reactor design is site independent.
- In law and regulation it is recognized that different licensing systems are required for different types of reactors. Government licensing for government-owned reactors is not limited to the DOE. The U.S. Navy has its own licensing strategy for navy reactors as does NASA for space launch of reactors and radioisotope power systems. These different licensing and safety strategies are driven by the unique requirements and missions.
- The NRC is an FHTR customer. The rational of a separate licensing agency is independence—but that does not apply when one of the goals is to provide safety information for the NRC.
- There is sufficient time between the building of a FHTR and any demonstration reactor for the NRC to develop a licensing strategy for FHRs. However, it is essential that the NRC begin to build that licensing capability as the FHTR progresses so they acquire the specialized knowledge for licensing a pre-commercial FHR.

These discussions lead the report authors to conclude that one of the first steps in going forward for an FHTR is the need for a study to examine the options for licensing an FHTR and recommend a preferred licensing strategy. The characteristics of a Class I test reactor are sufficiently different from other reactors that licensing options must be evaluated in total.

6.2 Safety Analysis

A preliminary safety analysis of an FHTR was conducted to understand FHR safety limits. Safety limits depend upon the reactor-specific combination of fuel and coolant

6.2.1 Limiting Safety System Settings (LSSS) Analysis

A major component of licensing a new test reactor is defining the safety envelop for operations and safety system activation—something that is codified for light water reactors. Central to this is the limiting safety system settings analysis (LSSS). The Nuclear Regulatory Commission (NRC) glossary defines LSSS as:

> *"Settings for automatic protective devices related to those variables having significant safety functions. Where a limiting safety system setting is specified for a variable on which a safety limit has been placed, the setting will ensure that automatic protective action will correct the abnormal situation before a safety limit is exceeded."*

As described in Section 6.1, NUREG-1537 is the NRC documentation providing guidelines for the necessary content needed in a Safety Analysis Report (SAR) for a non-power reactor license. Part 1 of NUREG-1537 describes the content and Part 2 describes the criteria to assist NRC staff review. For licensing, "The core configuration with the highest power density possible for the planned fuel should be analyzed as a basis for safety limits and limiting safety system settings in the thermal-hydraulic analyses" where the settings are "chosen to maintain fuel integrity when safety system protective actions are conservatively initiated at the LSSSs"[97] [Part 2 p4-27]. Each applicant must develop technical specifications to protect the environment and ensure the health and safety of facility staff and the public which encompasses an envelope of safe operation as required by 10 CFR 50.36. These limits should also protect "the integrity of the primary barrier against the uncontrolled release of radioactivity. For non-power reactors, the radioactivity of concern is generally the fission products in the fuel" [Part 1, Appendix 14.1, p3].

The limits should address normal operating conditions, off-normal operations, and all pertinent postulated accident scenarios where for each parameter with a safety limit, a protective measure (such as automatic reactor scram function) should prevent exceeding of the safety limit. The calculated set point for this protective action including uncertainties is defined as the limiting safety system setting (LSSS).

[97] NUREG-1537: *Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors*, 1996.

6.2.2 Coolant, Fuels and Materials Limits

For a new test reactor, the starting point of the safety analysis is defining the limits of various components in the reactor: coolant, fuels, and materials. The unique features of the FHTR, compared to conventional research reactors, include the use of high temperature salt coolant, metallic alloy structural material and TRISO particle fuel. Special considerations should be made in developing the test reactor licensing methodology to account for the characteristics of these materials. For example, limited measurement data of flibe's thermophysical properties are available in the literature and some are associated with significant uncertainties. Hence, these uncertainties should be evaluated explicitly, in addition to the conventional engineering hot channel factors in the licensing analysis. Other constraints, such as the structural material's temperature limit and salt freezing temperature limit are also unique features associated with the FHTR that need to be evaluated.

6.2.2.1 Primary Coolant Options

LiF-BeF₂

The baseline coolant for the FHTR is $^7Li_2BeF_4$ (flibe) with a 66.7-33.3 mol% composition of LiF and BeF_2, respectively. Flibe is a transparent fluoride salt that melts at 460°C and boils at over 1430°C[98]. A literature review of the thermophysical properties was conducted. Table I lists the recommended properties for flibe and corresponding uncertainties.

The Molten Salt Reactor (MSR) project resulted in documenting most of the data on molten or liquid salt thermophysical properties of flibe during the 1960s and 1970s, but the data on flibe is not extensive. The data available is very limited, both in amount of experimental work, number of data points, and range of temperatures considered. Experimental measurement of liquid salt properties proves difficult due to high melting temperatures and toxicity of flibe. There have been technology improvements for making property measurements. The Chinese Academy of Science (CAS) plans to measure flibe properties with improved instrumentation and provide more detailed and accurate property data as part of their efforts in developing Thorium Molten Salt Reactor (TMSR) technology. The density of flibe is fairly well documented; however, there is very little data on thermal conductivity and heat capacity, especially for wide temperature ranges. A review of thermophysical properties of flibe is summarized in[99] and the recommended correlations are reproduced in Table 6.1.

[98] C. FORSBERG et al., "Fluoride-Salt-Cooled High-Temperature Reactors (FHRs) for Base-Load Electricity and Process Heat," Cambridge: MIT Center for Advanced Nuclear Energy Systems, MIT-ANP-TR-147 (2013).
[99] R. R. ROMATOSKI and L. W. Hu, "Review of Fluoride Salt-Cooled High Temperature Reactor LiF-BeF2 (Flibe) Coolant Properties," Transactions of the American Nuclear Society, American Nuclear Society, 2014.

Table 6.1 Thermophysical Properties of Flibe Coolant

Property	Value/Correlation	Uncertainty	Reference
Density (kg/m^3)	2413 – 0.4884*T[K]	2%	[100]
Viscosity (Pa-s)	0.000116*exp(3755/T[K])	20%	[101, 102]
Heat Capacity (J/kg-K)	2386	3%	[103]
Thermal Conductivity (W/m-K)	1.1	10%	[104]

Flibe has excellent heat transfer properties. Its heat capacity is similar to water and its thermal conductivity is almost twice that of water. Flibe also excels neutronically. It has a large thermal moderating ratio and negative void and coolant temperature coefficients. Flibe melts around 459°C and boils around 1400°C allowing for a large liquid temperature region for operation[105]. However, flibe does produce significant amounts of tritium. Flibe is toxic because it contains beryllium; however, since it will be used in a radiation environment this is less of a concern. Neutronically, flibe must be 99.99% Li-7 enriched to maintain enough reactivity[106]. Thus, flibe is quite expensive.

NaF-ZrF$_4$

The mole composition of NaF-ZrF_4 is 59.5% NaF and 40.5% ZrF4. NaFZrF_4 has heat capacity and thermal conductivity roughly half that of flibe. Also, its neutronic properties are not as good as flibe. Its thermal moderating ratio is only 10, and it has positive void and coolant density coefficients. NaFZrF_4 melts around 500°C and boils around 1350°C which more slightly

[100] G. J. JANZ et al., "Molten Salts: Volume 4, Part 1, Fluorides and Mixtures Electrical Conductance, Density, Viscosity, and Surface Tension Data," *Journal of Physical and Chemical Reference Data*, **3**, *1*, 32-34 (1974).
[101] D. F. WILLIAMS, L. M. Toth and K. T. Clarno, "Assessment of Candidate Molten Salt Coolants for the Advanced High-Temperature Reactor (AHTR)," Oak Ridge National Laboratory, (March 2006).
[102] O. BENES and R. Konings, "Thermodynamic Properties and Phase Diagrams of Fluoride Salts for Nuclear Applications," *Journal of Fluorine Chemistry*, 130, *1*, 22-29 (Jan. 2009).
[103] O. BENES and R. Konings, "Thermodynamic Properties and Phase Diagrams of Fluoride Salts for Nuclear Applications," *Journal of Fluorine Chemistry*, 130, *1*, 22-29 (Jan. 2009).
[104] S. CANTOR et al., "Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts," Technical Report ORNL-TM-2316, Oak Ridge National Laboratory (1968).
[105] S. CANTOR et al., "Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts," Technical Report ORNL-TM-2316, Oak Ridge National Laboratory (1968).
[106] J. RICHARD et al., "Preliminary Design of a Prismatic Core Fluoride Salt-Cooled High Temperature Test Reactor (FHTR)," Proceedings of ICAPP, (2013).

limits its liquid operating range compared to flibe[107]. $NaFZrF_4$ does not produce tritium production, does not require lithium isotopic separation capability and costs less than flibe.

6.2.2.2 *Fuel and Moderator*

The reactor moderator is the graphite prismatic block. Currently, the graphite is assumed to be H451, and has been conservatively assumed to have been irradiated for 1000 hours with a corresponding thermal conductivity of 0.30 W/m-K[108].

The TRISO particle fuel thermal conductivity (W/cm-K) is based on German fuel in the temperature range 450 to 1300°C as

$$k_f = \left(\frac{-0.3906 \cdot 10^{-4} T + 0.06829}{DOSIS + 1.931 \cdot 10^{-4} T + 0.105} + 1.2881 \cdot 10^{-4} T + 0.042\right) * 1.2768 \tag{1}$$

where DOSIS is the fast neutron radiation dose (10^{21})[109] which is assumed to be that of the SINAP design. TRISO particle fuel is manufactured into graphite matrix fuel compacts, which are assumed to have the same thermal conductivity as the TRISO particles.

6.2.2.3 *Material Constraints on Thermal Limits*

Four thermal limits for the FHTR arise from the limits of the materials used. Two arise from the coolant properties, one from the choice of structural material, and the last based on the characteristics of the TRISO fuel.

The coolant should remain above its melting temperature, therefore the bulk temperature of coolant must start above the melting temperature. To allow for margin between operating conditions and safety limits, a 10°C safety margin is assumed to limit the minimum bulk coolant temperature. Further experiments are required to define this safety margin because the viscosity varies significantly near the melting point. Additionally, to avoid boiling of the coolant at the wall, the maximum coolant temperature must remain below the boiling point during steady state operation. A 200°C margin is chosen between the boiling point and the maximum coolant temperature allowed. If the FHTR uses flibe coolant with a liquid temperature range of 460 to 1400°C, the minimum bulk inlet temperature of 470°C and maximum coolant temperature of 1200°C are the LSSS criteria. For $NaF\text{-}ZrF_4$ coolant, the coolant temperature range will be smaller—510°C minimum bulk inlet temperature and 1150°C maximum coolant temperature.

[107] S. CANTOR et al., "Physical Properties of Molten-Salt Reactor Fuel, Coolant, and Flush Salts," Technical Report ORNL-TM-2316, Oak Ridge National Laboratory (1968).
[108] L. L. SNEAD, "Accumulation of Thermal Resistance in Neutron Irradiated Graphite Materials," *Journal of Nuclear Materials*, 381, *No. 1*, 76-82 (2008).
[109] Z. GAO and L. Shi, "Thermal Hydraulic Calculation of the HTR-10 for the Initial and Equilibrium Core," Nuclear Engineering and Design 218, *No. 1–3,* 51-64 (October 2002).

Table 6.2 LSSS Criteria

	Flibe	$NaFZrF_4$
Minimum bulk inlet temperature, T_{in}	470°C	510°C
Maximum bulk outlet temperature, T_{out}	720°C	720°C
Maximum coolant temperature, $T_{c,M}$	1200°C	1150°C
Maximum fuel temperature $T_{f,M}$	1300°C	1300°C

The primary loop structural material Hastelloy N has been demonstrated to perform with a maximum temperature of 730°C[110]. To ensure the structural material integrity, the bulk outlet temperature of the coolant limit was chosen as 720°C. It is assumed sufficient mixing occurs at the core outlet so that the outlet Hastelloy N pipe is in contact with coolant close to the bulk coolant temperature. Lastly, the TRISO particle fuel's maximum temperature to maintain fuel integrity and contain fission products for steady state operation is 1300°C and for transient conditions is 1600°C. At high temperatures the SiC layer starts decomposing and then diffusion of fission products through the PyC layer follows[111]. Therefore, the fuel temperature limit of 1300°C is chosen as a maximum.

Overall, the FHTR material specifications for the fuel, coolant, and structural materials establish the criteria for LSSS for the bulk inlet, bulk outlet, maximum coolant, and maximum fuel temperatures as summarized in Table 6.2. The safety margins for the temperature limits will require analysis and may need to be updated and incorporated into the LSSS criteria in Table 6.2.

6.2.3 LSSS Analysis Methodology

A one-dimensional, steady-state, thermal hydraulic analysis code was developed to model a fixed pebble bed FHTR, based on the SINAP design[112]. This code was modified to analyze a prismatic core reactor to provide LSSS. The model is based on conservation of mass, momentum, and energy as originally detailed by Xiao et al. in[113]. For one-dimensional flow, a single channel model was adopted with both average and hot channels to simulate fuel and coolant temperatures. Single channel models are simplified but conservative. Thus, a single

[110] D. T. INGERSOLL et al., "Status of Preconceptual Design of the Advanced High-Temperature Reactor (AHTR)," ORNL/TM-2004/104, United States Department of Energy, (2004).
[111] NUREG/CR-6844, "TRISO Coated Particle Fuel Phenomenon Identification and ranking Tables (PIRTS) for fission Product Transport due to Manufacturing, Operations, and Accidents," Office of Nuclear Reactor Regulations, Nuclear Regulatory Commission (2004).
[112] Y. XIAO et al., "Licensing Considerations of a Fluoride Salt Cooled High Temperature Test Reactor," International Conference on Nuclear Engineering, ICONE21-16383 (2013).
[113] Y. XIAO et al., "Licensing Considerations of a Fluoride Salt Cooled High Temperature Test Reactor," International Conference on Nuclear Engineering, ICONE21-16383 (2013).

channel model is well suited to quantify thermophysical uncertainty propagation and will be studied in future work.

A vertical finite volume approach was used in the axial direction such that the core was divided into control volumes or nodes. Each node is analyzed using an equivalent cylinder for the one-dimensional unit cell model as shown in Fig. 6.1. The unit cell consists of one coolant channel and effectively two fuel channels. The radius of the coolant channel is preserved while the radius of the graphite ring conserves the volume of graphite in the unit cell. The fuel ring similarly conserves fuel volume.

Unit cell models tend to under predict the maximum fuel temperature[114]. Thus, either the thermal conductivity of the fuel needs to be adjusted to account for the smaller radius of the fuel ring compared to the fuel channel or the fuel and coolant needs to be decoupled. Davis and Hawkes[115] compared a one-dimensional annular model with a 0.324 thermal conductivity factor to a finite element calculation. The thermal conductivity factor is needed to account for the different between the exact solution of the annular and cylindrical case. The temperature rise between the coolant and the wall, through the graphite and through the fuel were nearly the same. The centerline fuel temperature for the annular model was 0.6 to 2.2°C higher than the finite element calculation depending on the volumetric heat generation, concluding that a one-dimensional annular model can accurately predict fuel centerline temperatures with the correct thermal properties[116]. These simple models are helpful to understand basic aspects of heat transfer in a prismatic fuel block and are economical for their low computational intensity.

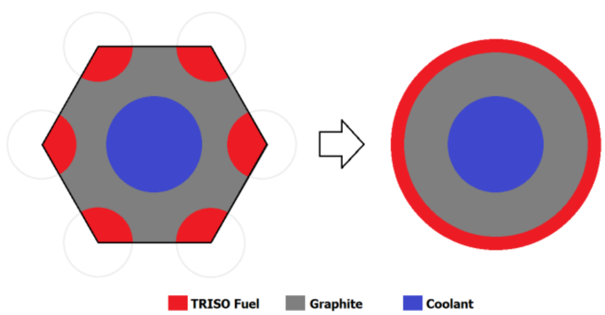


Fig. 6.1 Unit Cell and Equivalent Cylinder Annular Models.

The heat transfer in the single channel analysis is calculated using the energy balance equation. For each node the coolant temperature is

[114] N. TAK, M. H. Kim and W. J. Lee, “Numerical Investigation of a Heat Transfer within the Prismatic Fuel Assembly of a Very High Temperature Reactor,” *Annals of Nuclear Energy*, 35, 1892–1899 (2008).
[115] C. B. DAVIS and G. L. Hawkes, “Thermal-Hydraulic Analyses of the LS-VHTR,” Idaho National Laboratory (INL), June 1, 2006.
[116] C. B. DAVIS and G. L. Hawkes, “Thermal-Hydraulic Analyses of the LS-VHTR,” Idaho National Laboratory (INL), June 1, 2006.

$$hA(T_w - T_c) = \dot{m}\frac{\partial h_c}{\partial z} \tag{2}$$

where h is the heat transfer coefficient between the coolant and the wall, A is the heat transfer area, T_w is the wall temperature, T_c is the bulk coolant temperature, $\dot{m}$ is the mass flow rate of the coolant, h_c is the enthalpy of the coolant, and z is the axial height.

Heat transfer of molten salt experimental studies show molten salts behave like ordinary working fluids[117, 118, 119]. The Reynolds number is around 300 situating the reactor flow in the laminar region. Shaw's heat transfer correlation for laminar flow is adopted.

$$\begin{aligned} Nu &= 1.302\left(\frac{x^+}{2}\right)^{-1/3} - 0.5, \quad x^+ \le 0.003 \\ Nu &= 4.364 + 0.263\left(\frac{x^+}{2}\right)^{-0.506} e^{-41(x^+/2)}, x^+ > 0.003 \\ x^+ &= \frac{2(x/D)}{Re\ Pr} \end{aligned} \tag{3}$$

where Nu is the Nusselt number defined as Nu=hD/k, x is the distance from the entrance (bottom), D is the channel diameter, Re is the Reynolds number and Pr is the Prandtl number[120]. This correlation has been shown to match computational fluid dynamic (CFD) results obtained for flibe[121]. Experimental work has not been conducted specifically for flibe coolant; however since molten salts behave like ordinary fluids and CFD results show a good correlation, Eq. (3) is adopted in this code.

Given the wall temperature, conduction occurs across the graphite in the fuel block, and the temperature at the outer edge of the graphite ring, T_G, is calculated by solving the heat equation as

$$T_G(z) = T_w(z) + \frac{q'(z)}{\pi k_G}\ln\left(\frac{R_G}{R_c}\right) \tag{4}$$

where R_G is the outer graphite ring radius, R_c is the coolant radius, T_W is the temperature at the coolant wall, q' is the linear heat generation rate per fuel channel, k_G is the thermal conductivity

---

[117] J. W. COOKE and B. Cox. "Forced-convection heat-transfer measurements with a molten fluoride salt mixture flowing in a smooth tube," Technical Report ORNL-TM-4079, Oak Ridge National Laboratory (1973).

[118] H. W. HOMAN and S. I. Cohen, "Fused salt heat transfer part III: forced-convection heat transfer in circular tubes containing the salt mixture NaNO2-NANO3-KNO3," Technical Report ORNL-2433, Oak Ridge National Laboratory (1960).

[119] M. D. SILVERMAN, W. R. Huntley and H. E. Robertson, "Heat transfer measurements in a forced convection loop with two molten-fluoride salts: LiF-BeF2-ThF4-UF4 and EUTECTIC NaBF4-NaF," Technical report, Report No. ORNL/TM-5335, Oak Ridge National Laboratory (1976).

[120] U. REA et al., "Laminar convective heat transfer and viscous pressure loss of alumina–water and zirconia-water nanofluids," *International Journal of Heat and Mass Transfer*, 52, 2042–2048 (2009).

[121] W. C. CHENG et al., "CFD Analysis for Asymmetric Power Generation in a Prismatic Fuel Block of Fluoride-salt-cooled High-temperature Test Reactor," Proceedings of ICAPP, (2014).

of graphite, L is the height of the core, and N is the number of nodes. The maximum fuel temperature at the centerline, T_{CL}, of the cylindrical fuel compact is then

$$T_{CL}(z) = T_G(z) + \frac{q'(z)}{4\pi k_{f'}}\left[2\ln\left(\frac{R_f}{R_G}\right) - \left(1 - \left(\frac{R_G}{R_f}\right)^2\right)\right] \tag{5}$$

where $k_{f'}$ is the thermal conductivity of the TRISO particle fuel compact times 0.34 to account for the smaller radius of the fuel ring as discussed above and R_f is the outer radius of the fuel ring.

Hot Channel

The hot channel is a hypothetical coolant channel that is the most limiting coolant channel in the reactor. Hence, the hot channel impacts the thermal safety limits of a reactor. The hot channel model uses hot channel factors to account for spatial variation of power and flow and other uncertainties like fuel fabrication tolerances. It is conservatively assumed that the hot channel has maximum radial peaking and receives the minimum amount of flow. The hot channel thus establishes the upper bound of thermal-hydraulic limits. The hot channel factors accounting for deviations from the nominal or average channel are used to find the maximum coolant, wall, and fuel temperatures

$$T_{c,M} = T_{in} + F_H\Delta T \tag{6}$$
$$T_{w,M} = T_{in} + F_H\Delta T + F_{\Delta T,w}\Delta T_w \tag{7}$$
$$T_{f,M} = T_{in} + F_H\Delta T + F_{\Delta T,w}\Delta T_w + F_{\Delta T,f}\Delta T_f \tag{8}$$

where F_H is the enthalpy rise engineering hot channel factor (EHCF), $F_{\Delta T,w}$ is the film temperature rise EHCF, and $F_{\Delta T,f}$ is the fuel temperature rise EHCF. The EHCFs are determined from a statistical combination of sub factors by

$$F = 1 + \left[\sum_j \left(f_j - 1\right)^2\right]^{1/2} \tag{9}$$

where f_i are the sub factors that account for the uncertainties. The sub factors for the enthalpy rise, film temperature rise, and fuel temperature rise EHCFs are obtained from

$$f = 1.0 + \frac{n\,\sigma}{\mu} \tag{10}$$

where n is the number of standard deviations, σ is the standard deviation and μ is the mean or nominal value of the sub factor. The EHCFs used in the present analysis are listed in Table 6.3. Furthermore, a coolant flow factor of 0.92 and channel flow disparity factor of 0.86 are assumed.

These hot channel factors can each be treated as normal distribution with a nominal value and associated standard deviation.

The limiting safety systems settings (LSSS) are derived from thermal hydraulic limits considering deviation from design specifications as described in Subsection 0. The thermal limits provide safe margin over the designed operating region to ensure the integrity of the fuel.

Table 6.3 Engineering Hot Channel Sub factors for MITR-II [122]

Enthalpy Rise	
Reactor power measurement	1.050
Power density measurement/calculation	1.100
Plenum chamber flow	1.080
Flow measurement	1.050
Fuel density tolerances	1.026
Flow channel tolerances	1.089
Eccentricity	1.001
Statistical $\boldsymbol{F_H}$	***1.173***
Film Temperature Rise	
Reactor power measurement	1.050
Power density measurement/calculation	1.100
Plenum chamber flow	1.060
Flow measurement	1.040
Fuel density tolerances	1.050
Flow channel tolerances	1.124
Eccentricity	1.003
Heat transfer coefficient	1.200
Statistical $\boldsymbol{F_{\Delta T,w}}$	***1.275***
Fuel Temperature Rise / Heat flux	
Reactor power measurement	1.050
Power density measurement/calculation	1.100
Fuel density tolerances	1.050
Eccentricity	1.003
Statistical $\boldsymbol{F_{\Delta T,f}}$	***1.123***

[122] MIT Nuclear Reactor Laboratory, "Safety Analysis Report for the MIT Research Reactor," MIT-NRL-11-02, August (2011).

LSSS Operating Region

Using the thermal limit criteria leads to definition of an allowable operating region for the FHTR. In Fig. 6.2, an example LSSS calculation result is depicted with the operating region shaded. The LSSS analysis considers these thermal limits to ensure the material and fuel integrity of the reactor during steady state operation. The thermal limits define the maximum reactor power, maximum primary coolant bulk outlet temperature, and minimum primary coolant inlet bulk temperature. To find LSSS, the single channel analysis for an average and hot channel model is used to establish the operating region pictured in Fig. 6.2. The four thermal limits (for minimum bulk inlet temperature, maximum bulk outlet temperature, maximum coolant temperature, and maximum fuel temperature) provide the bounding design range.

The ‘Inlet’ line in Fig. 6.2 is determined by the minimum inlet temperature. Iteration of reactor power, given the mass flow rate and inlet temperature of 470°C, leads to the 'Inlet' line and the three points on that line that correspond to the maximum power level where the thermal limits of maximum coolant, fuel, and bulk outlet temperature are reached. The 'Outlet' line is determined by iteration of the inlet temperature starting at 470°C and reactor power for which the maximum bulk outlet temperature of 720°C is reached. The points along this line are determined when the thermal limits of maximum coolant and fuel temperature are reached. Given the thermal limits, the bounding LSSS thus defines the operation region, which is shaded in Fig. 6.2. This shaded region ensures the thermal limits are met and thus defines the safe operating range of the FHTR without automatic protection needed. Note that the operating region is smaller than traditional light water reactors since the flibe coolant has a bounding melting temperature greater than standard room temperature. Thus, the FHTR has unique operating requirements that warrant further study of required temperature margins at the low operating temperatures as the coolant temperature approaches its melting point. The liquid viscosity increases rapidly as temperatures approach the melting point, and thus a larger temperature above the freezing point may be required. In the low temperature range, further study will determine the need for and operational use of electric heating.

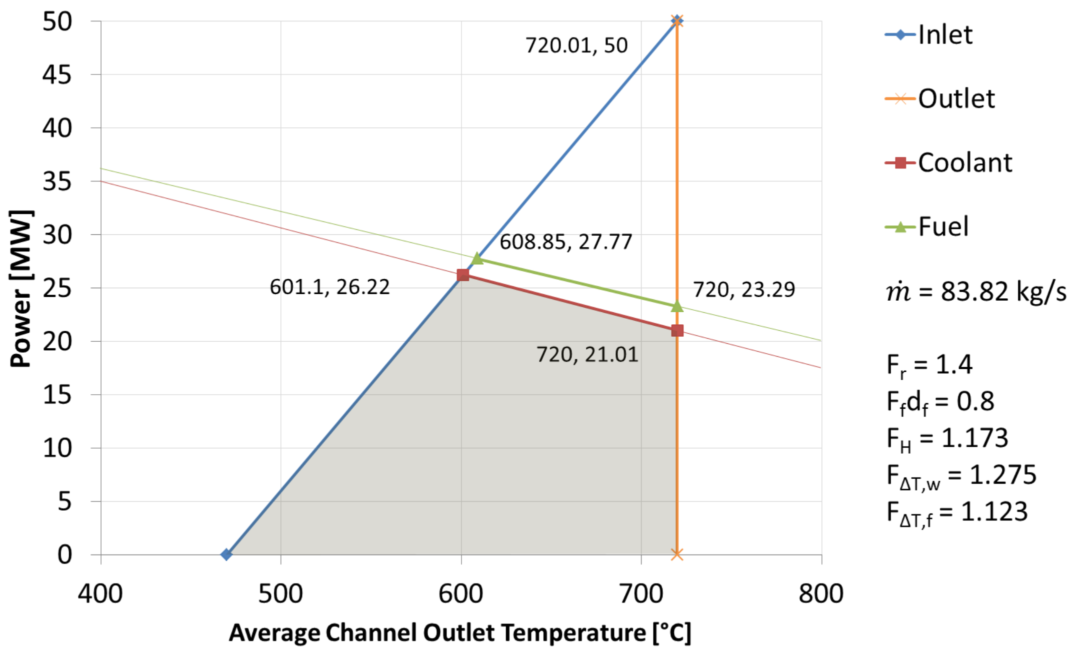


Fig. 6.2 LSSS Forced Convection

This preliminary thermal hydraulic licensing analysis outlines the operational limits based on criteria evaluated with respect to LSSS ensuring sufficient margin to fuel and material limits during steady-state operation. The maximum reactor power level is constrained by the material limits of the fuel, coolant, and structural materials of the reactor design. The most limiting is the structural material Hastelloy N. The next most limiting is the maximum coolant temperature. These two limits dictate the maximum reactor power for a given flow rate. It is important to note that the FHTR operating range is smaller than a water-cooled reactor and is discussed in further detail in Section 6.4. The results depend on the underlying assumptions of the design, for example EHCFs, radial power peaking, axial power distribution, and core geometry. Several iterations on neutronics and thermal hydraulics analyses are necessary for before the test reactor design can be finalized. Thus, future work will refine the hot channel factors and assumptions. Additionally, the low operating temperature limit requires further analysis of low coolant temperature conditions on reactivity feedback and operational control to determine the need and use of electric heaters especially during start up conditions. Finally, given the limited data on the thermophysical properties of the reactor coolant, a sensitivity and uncertainty propagation analysis of physical properties on thermal hydraulic behavior is required. This may also result in modifications of the LSSS Criteria.

Sensitivity and Uncertainty Propagation Analysis

Reactor parameters, engineering hot channel factors, and coolant thermophysical properties have associated uncertainties. Each of these parameters will be treated as normal distributions. The uncertainties inherent in these parameters will propagation throughout the thermal hydraulic analysis. To understand the implications of the uncertainties, several sensitivity and uncertainty propagation analyses will be conducted.

The simplest analysis is to apply a sensitivity factor for each parameter to see how that parameter influences the results. These sensitivity factors can be applied to each parameter separately or to multiple parameters concurrently. For the concurrent sensitivity case, each parameter will be evaluated based on a random sampling of each parameter from its normal distribution that holds constant throughout the calculation or history. Running multiple histories using the Monte Carlo method will propagate the uncertainty of each parameter.

6.3 Comparison of Operating Limits of FHTR to Conventional Research Reactors

In conventional research reactors using water, like the MITR-II, the limiting safety systems settings (LSSS) involve coolant phenomena that do not apply to liquid salt coolants. Additionally, the fuel type, TRISO fuel, differs from conventional light water reactor (LWR) fuel and other research reactor fuel. These differences lend to alternate LSSS.

For water coolant reactors, the goal is to protect the reactor from reaching critical heat flux (CHF) or departure from nucleate boiling (DNB). Other phenomena to protect against may include onset of flow instability and onset of nucleate boiling. In the case of liquid salt coolant, the boiling point is of concern however if temperatures in the reactor are too low then the liquid salt will solidify. This phenomenon adds an additional constraint at low coolant temperature. In conventional research reactors with water, this is not a concern because the melting point is very low.

The TRISO particle fuel has very different operational requirements than conventional LWR fuel. The fuel consists of small microspheres coated with multiple layers that provide cladding to encapsulate fission products. The TRISO particles are incorporated into a graphite matrix structure. The materials and structure of the fuel allows for operation under normal conditions up to 1250°C and under accident conditions 1600°C for limited periods of time without significant fuel failure. No other fuel has demonstrated such high temperature capabilities. Hence, the FHTR appears to have a larger safety margin to fuel failure than conventional research reactors.

7 Supporting FHTR Research and Development

The FHTR is the largest facility in a program to develop a commercial FHR before a pre-commercial demonstration plant. However, there are other significant facilities and significant R&D. Most of the R&D is required to support the design and construction of the FHTR is required to develop a pre-commercial FHR. We discuss herein the total development program with an emphasis on major test facilities for support of the FHTR. Most of these facilities are also required for development of a pre-commercial demonstration plant.

A series of four workshops were held at UCB, UW, and MIT with outside experts to address key design and licensing strategies for the FHR[123] that included (1) licensing[124], (2) methods and experiments[125], (3) materials[126], and (4) developmental roadmap and test reactor performance requirements[127]. Detailed technical reports were produced. Recently, ORNL published a reactor technology development roadmap. The ORNL technology roadmap and parallel efforts with HTGRs are discussed followed by a description of the major supporting facilities for the FHTR and the broader development program.

7.1 ORNL Technology Roadmap

The ORNL technology roadmap[128] (Fig. 7.1) identifies the major technical tasks for the development of an FHR. There are differences between the ORNL strategy and the strategy herein because of different assumptions.

- *Goals*. We chose an FHR with NACC based on our assessment of the needs for the 2030 electricity grid to minimize market risk and enable a zero-carbon electricity grid. The NACC power system has driven many detailed design decisions. The ORNL roadmap goals are to minimize technology development and licensing risk. This included selection of a steam cycle and other design decisions.

[123] R. O. SCARLAT et al., "Design and Licensing Strategies for the Fluoride-salt-cooled High-temperature Reactor (FHR) Technology," *Progress in Nuclear Energy* (in Press).

[124] A. T. CISNEROS et al., "Fluoride Salt-Cooled High-Temperature Reactor (FHR) Subsystems Definition, Functional Requirements Definition, and Licensing Basis Events Identification White Paper," Department of Nuclear Engineering, University of California, Berkeley, UCBTH-12-001, 2013, http://FHR.nuc.berkeley.edu/.

[125] A. T. CISNEROS et al., "Fluoride Salt-Cooled High-Temperature Reactor (FHR) Methods and Experimental Program White Paper," Department of Nuclear Engineering, University of California, Berkeley, UCBTH-12-002, 2013, http://FHR.nuc.berkeley.edu/.

[126] G. CAO et al., "Fluoride Salt-Cooled High-Temperature Reactor (FHR) Materials, Fuels, and Components White Paper," Department of Nuclear Engineering, University of California, Berkeley, UCBTH-12-003, 2013, http://FHR.nuc.berkeley.edu/

[127] D. CARPENTER et al., *Fluoride-Salt-Cooled High-Temperature Reactor (FHR) Development Roadmap and Test Reactor Performance Requirements White Paper*, Department of Nuclear Engineering, University of California, Berkeley, UCBTH-12-004, 2013, http://FHR.nuc.berkeley.edu/.

[128] D. E. HOLCOMB et al., *Fluoride Salt-Cooled High-Temperature Reactor Technology Development and Demonstration Roadmap*, ONRL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee, September 2013.

- *Test reactor*. We have chosen a general purpose test reactor because of the decision that keeping open a set of technology options is important at this stage of development. This decision increases the total R&D but reduces technology and commercialization risks.
- *CAS/US Path forward*. Our strategy is based on maintaining two paths forward: a cooperative program with the CAS and an independent program.

Fig. 7.1 ORNL Roadmap

7.2 Modular High-Temperature Reactor Development

The United States has had a large effort to develop modular high-temperature gas cooled reactors—the Next Generation Nuclear Plant (NGNP) program. That resulted in a systematic effort[129] to identify and quantify technology readiness levels to focus the research, development, and demonstration program. This included addressing licensing challenges. This program provides (1) detailed information on the development status of FHR and HTGR common components and systems and (2) a recent example of how to systematically develop a detailed technical roadmap that identifies and prioritizes required research and development.

7.3 Test Facilities to support FHTR

Significant test facilities are required to support the FHTR development. These activities are designed to provide high confidence that the FHTR will operate as expected and provide key data for commercialization activities. These are summarized with the highest-capital cost long-lead-time facilities described first. There is no requirement that facilities be located in the United States. If the FHTR is an international project, some of these facilities may be located elsewhere in the world. Because of the overlap of technologies, some existing high-temperature reactor and sodium fast reactor test facilities may be able conduct some of this research.

7.3.1 High-Flux Salt Test Loop in Existing Test Reactor

The installation of a 700°C test loop in an existing test reactor such as the Advanced Test Reactor (ATR) at INL is recommended with the capability to pump salt at velocities that significantly exceed expected commercial conditions and power densities. Large Class II test reactors can provide fully realistic fuel and coolant conditions for the test position while the test reactor provides the neutrons and reactor safety features. Test loops provide early learning for an FHTR including full control of coolant chemistry, tritium management, and fuel testing. Test loops can't prove FHR viability that includes all the complexity of a reactor operation at significant power. Test loops can be used to validate performance of driver fuel for a FHTR. This is the single most expensive support facility. Based on experience, the total cost including operations will be in the range of $100 million.

7.3.2 Integrated Test Facility (ITF)

The ITF is physically similar to the FHTR but has no nuclear operations and thus no radiation or nuclear safety requirements. It is heated to the expected operating temperatures, has pumps, and uses the same salt as the FHTR. Its purpose is to realistically test mechanical, chemical, and instrument systems for the FHTR including control rods, fuel handling, and

[129] H. D. Gougar, *Baseline Concept Description of a Small Modular High Temperature Reactor*, INL/EXT-14-31541, Idaho National Laboratory, Idaho Falls, Idaho, May 2014.

instrumentation. After the FHTR is operational, it remains as a test facility for (1) proposed new salts or equipment to be tested in the FHTR under nuclear conditions and (2) support development of a pre-commercial demonstration plant. It also acts as a training facility. The facility would be expected to be operational for the life of the FHTR with a lifetime costs of several tens of millions of dollars.

There is a large experience in the United States in designing, building, and operating such facilities from the sodium-cooled fast reactor program. That experience provides a pathway and checklist of what is required and not required. Both the FHR and SFR are low-pressure high-temperature reactors with many common features from refueling to DRACS decay heat removal systems. There are differences, (1) the FHR operates at significantly higher temperatures, (2) the coolant is transparent that simplifies inspection and refueling, and (3) radiation heat transfer becomes important at the required operating temperatures.

There are a set of other facilities thermal and mechanical test facilities that could be part of an ITF or separate from such a facility. These include DRACS and the silo cooling system. There are a variety of HTGR and SFR test facilities that potentially could be used as FHTR test facilities at different national laboratories and universities.

7.3.3 Reactor-Driven FHR Subcritical Test Facility

A subcritical test facility uses a test reactor with a large-diameter neutron beam that can be used to drive part of an FHTR or FHR core at operating conditions. It is similar to a test loop except (1) the experimental volume is much larger to enable use of prototypical fuel assemblies, instrumentation, and other components and (2) the power densities are lower—typically 10 to 30% of reactor power densities. It is a second approach to understand the behavior of a FHTR or FHR at partial scale. It is a complementary capability to a test loop. It is an option to obtain required information to (1) support design of an FHTR by providing a much larger volume than a loop to enable integrated testing of reactor instrumentation, chemistry control and auxiliary systems, and (2) investigate at lower costs a wider set of core design options than are possible in a test reactor.

Limited work has been done to define the capabilities of such a facility at the MIT reactor (MITR).[130] Appendix C provides an overview of the preliminary design and neutronics analysis of a reactor-driven subcritical FHR facility utilizing neutrons from the MITR. There is a long history of using reactor driven subcritical test facilities in the development of fast reactors.

The MITR has designed and built sub-critical facilities for various applications. In early 1970's, a feasibility study was performed for a convertor assembly for fusion blanket experiment.[131] A fission converter-based epithermal neutron irradiation facility was designed and built in late 1990's at the MITR. The fission coverter was licensed as an experimental facility by

[130] K. Sun, L. Hu and C. Forsberg, "Preliminary Neutronic Study of an MIT Reactor Driven Fluoride-salt-cooled Sub-critical Facility," *Transactions of the American Nuclear Society* Anaheim, California, November 9-13, 2014.
[131] A. PANT, Feasibility Study of a Convertor Assembly for Fusion Blanket Experiments, SM thesis, Department of Nuclear Engineering, MIT (1971).

the U.S. Nuclear Regulatory Commission (NRC) for neutron capture therapy (NCT) research.[132]. A sub-critical facility can be used for multiple research projects such as addressing tritium and chemistry control, materials corrosion performance, instrumentation development for FHR as well as tritium breeding using Flibe for fusion reactor research.

7.3.4 High-Power Triso-Coated-Particle Fuel Testing

The FHR uses the same fuel as HTGRs and thus takes advantage of the Next Generation Nuclear Plant (NGNP) program that has developed methods to produce very reliable fuel and strategies to reduce fuel manufacturing costs. The proposed activities herein assume that this highly-successful program continues. There are differences between HTGRs and FHRs. FHRs use salt coolants that are better coolants than helium. The most economic designs of FHRs operate at higher power densities but lower temperatures than HTGRs.

Figure 7.2 shows the operating window[133] for an optimized FHR fuel relative to the NGNP fuel and several other HTGR plant designs. The economically optimum FHR triso fuel particles have higher power densities and lower temperatures. The fuel performance models indicate that this combination of performance characteristics should be achievable with existing NGNP fuels—but this has not been fully demonstrated—only limited testing has been done. The existing NGNP experimental facilities and program are fully capable of testing triso particles to higher particle power densities. An additional irradiation campaign of coated particles at higher power densities is required to provide confidence and qualify fuel can operate at more economic higher power densities.

[132] O. K. HARLING et al., The Fission Converter–Based Epithermal Neutron Irradiation Facility at the Massachusetts Institute of Technology Reactor, Nuclear Science and Engineering, vol. 140: 223-240.

[133] A. T. CISNEROS, JR., Pebble Bed Reactors Design Optimization Methods and their Application to the Pebble Bed Fluoride Salt Cooled High Temperature Reactor (PB-FHR), Ph.D. Thesis, Department of Nuclear Engineering, University of California, Berkeley (2013).

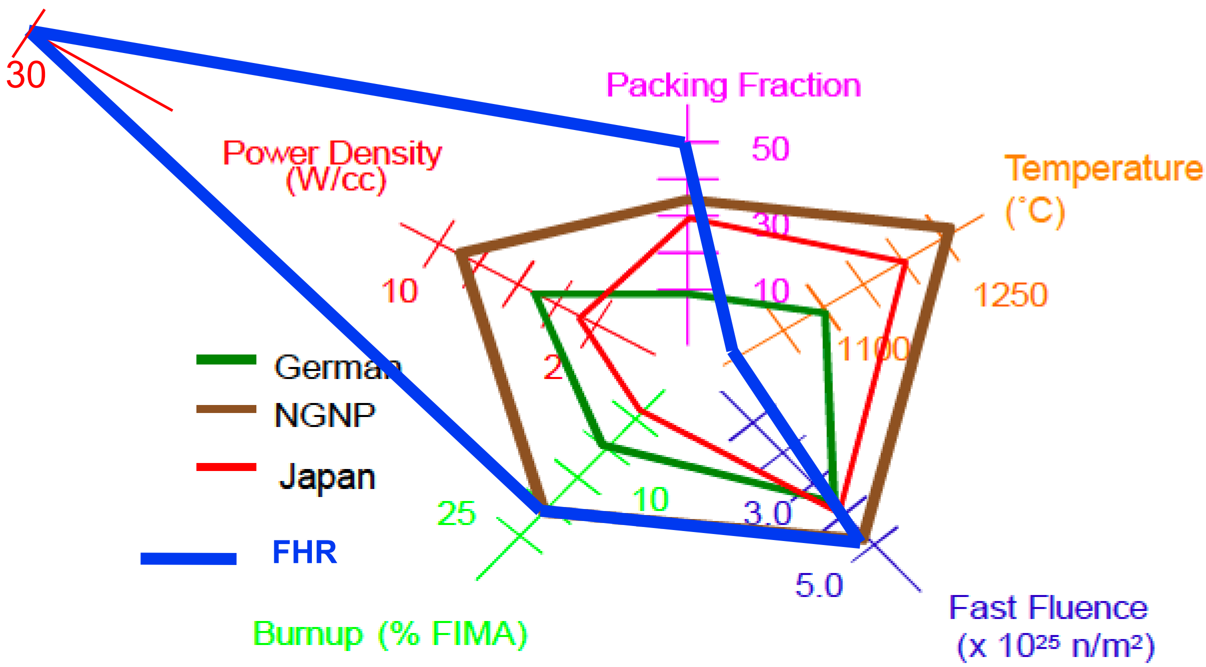


Fig. 7.2 Performance Envelope for FHR and HTGR coated-particle fuel

7.3.5 Air Heat Exchanger Test Facility

The commercial basis for the FHR is its coupling to the nuclear air Brayton combined cycle (NACC)[134, 135] that enables production of (1) base-load electricity and (2) peak electricity with auxiliary natural gas. That capability (after subtracting natural gas costs) increases plant revenue by 50 to 100% relative to base-load nuclear plants in deregulated electricity markets such as California and Texas. The coiled tube air heater (CTAH) must be tested to demonstrate: (1) mechanical integrity including transient behavior, (2) low air pressure drops, (3) ability to conduct maintenance, (4) the ability to control tritium releases, and (5) the ability to assure no damage from salt freezing.

---

[134] C. FORSBERG and D. Curtis, "Meeting the Needs of a Nuclear-Renewable Electrical Grid with a Fluoride-Salt-Cooled High-Temperature Reactor Coupled to a Nuclear Air-Brayton Combined Cycle Power System," *Nuclear Technology*, **186** (2014)

[135] D. CURTIS and C. W. Forsberg, "Market Performance of the Mark 1 Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor", *Transactions of the American Nuclear Society*, Paper 9751; Reno, Nevada, June 15-19, 2014.

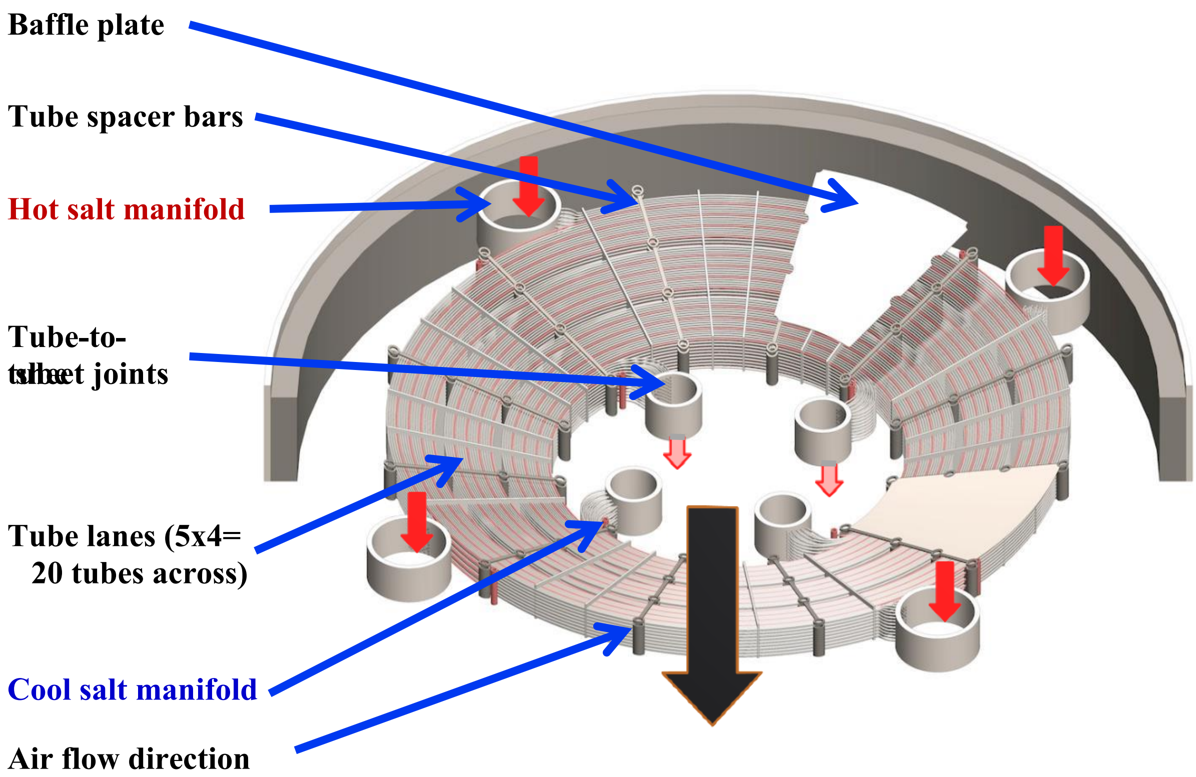


Fig. 7.3 Coiled Tube Air Heater Bundle

The facility will be a non-nuclear facility that uses electricity or natural gas to heat the salt and has a compressor to provide realistic temperatures and pressures for the CTAH. Some of the characteristics of the CTAH system, such as oxide barriers to tritium on the outer heat exchanger surface partly depend upon the pressure and oxygen content of the gas and thus the need for realistic testing conditions. As discussed below, this will also include testing of tritium permeation barriers associate with CTAH under realistic conditions. Such tests may be conducted with trace quantities of tritium or may be conducted with deuterium—a non-radioactive isotope of hydrogen. Added work will be required to determine the required size for credible tests. These tests may be conducted at a gas-turbine test facility.

There is one caveat associated with this work. The most promising power cycle option is NACC. The backup option is a closed helium Brayton power cycle. This cycle, like NACC, can include FIRES heat storage. However, it can’t include burning an auxiliary fuel to produce peak power. If the power cycle is helium, the heat transfer characteristics will be different and there will be different concerns. There are two factors that could drive a decision to a helium power cycle.

- *Tritium*. If control of tritium proves more difficult than expected, there is the option of trapping tritium in the power cycle—a very straight forward low-risk option.
- *Intermediate loop*. One can design FHRs with and without intermediate loops. If there are safety or other concerns, an intermediate loop might be required for an open-air cycle but not for a closed helium cycle. In such a case, the economics might drive one to choose a helium cycle versus NACC with an intermediate loop.

7.3.6 Thermal Hydraulics Test Facilities

The development of LWR, HTGR, and SFR technologies required very-large expensive thermal hydraulic test facilities that are major contributors to the development cost and schedule. This is a consequence of (1) the high operating pressures of LWRs and HTGRs and (2) the high temperatures and chemical reactivity associated with sodium. By good fortune it was discovered that the organic Dowtherm® A at temperatures less than 100°C can match high-temperature salt at 700°C in terms of Reynolds, Prandlt, and other relevant thermohydraulic parameters. This implies that small thermal-hydraulic test loops at atmospheric pressure and low temperatures can simulate the full scale system at operating conditions[136]. However, there will be a need to benchmark the results with high-temperature thermal hydraulics test using real salts to confirm scaling laws before presentation to the safety authority (DOE or NRC).

7.3.7 Tritium Test Facilities

A significant program in tritium control in salt is required to demonstrate containment and recovery of tritium. While this is a significant R&D activity, there are existing tritium research, processing and handling facilities at Idaho National Laboratory, Pacific Northwest National Laboratory, Savannah River National Laboratory, and Los Alamos National Laboratory. These facilities are associated with national defense activities and fusion research. There are also major programs in Japan and Europe associated with ITER—the large fusion experiment.

The tritium R&D challenge is somewhat different than many of the above activities. There has been a massive amount of research to support defense needs and fusion—as well as early research associated with the molten salt reactor program in the United States. The need is to sort through this massive experience base and couple it with the specific requirements of an FHR to define what added R&D is required and what needs to be demonstrated. Appendix A summarizes the tritium recovery options and how the requirements differ from the MSR and fusion requirements.

7.3.8 Materials Test Facility

Salt-cooled reactor materials testing has received much attention and by itself does not require a major facility although testing materials is the center of many of the test facilities

[136] N. ZWEIBAUM et al., "Phenomenology, methods and experimental program for fluoride-salt-cooled, high-temperature reactors (FHRs)". *Progress in Nuclear Energy*. Gen IV Reactors Special Issue (Submitted December 2013).

described above. However, directly or indirectly materials testing will drive much of the test program and the requirements for test facilities. In the context of a test reactor program, there are several key considerations.

- *Code-qualified materials are not required for a test reactor*. Safety must be demonstrated but this can be achieved by short-term testing of materials as long as the testing proceeds in advance of use in test reactors. Test reactors can operate at higher temperatures than most nominal temperature limits. The temperature limits for most materials is that at higher temperatures there is high-temperature creep that limits component lifetime. This is a major constraint in a power reactor; however, in a test reactor it is acceptable to have faster high-temperature creep with the recognition that it implies regular replacement of selected components. In this context, the experience of the High-temperature Test Reactor is relevant. The HTTR is the very high-temperature test reactor in Japan. The testing of some components at high temperatures implies accelerated mechanical creep in other components that, in turn, implies regular replacement of selected components.
- *Advanced materials*. As discussed earlier, the current preferred material of construction is Hastelloy-N with design temperature limit of ~700°C. It would be the preferred choice of materials for construction of a test reactor if built today. Alloys with higher temperature capabilities and compatible with salts are being developed at Oak Ridge National Laboratory. These new alloys are expected to allow peak FHR operating temperatures at significantly higher temperatures. Because of the very large economic payoff for higher-temperature operations, a major programmatic question is whether to accelerate that materials program for early use of those materials. An accelerated materials program would have a large impact on relative near-term priorities as discussed herein.

7.3.9 Zero-Power Criticality Facility

A zero-power criticality facility may be required to confirm reactor physics parameters under appropriate conditions. Limited tests are currently planned in the Czech criticality facility. These facilities operate at near-zero power conditions and thus do not involve significant amounts of radioactivity beyond the natural radioactivity of the uranium fuel.

7.3.10 Lithium-7 Production

The candidate salts require production of significant quantities of separated lithium-7 (^{7}Li). The Chinese Academy of Science is building production facilities; thus, if they are partners in the test reactor they are the likely suppliers. If they are not partners, it may be feasible to buy the required quantities of ^{7}Li from China. There is also the separate option of building lithium isotopic separations facility. There have been various studies of this option and a GAO study[137] on future lithium isotopic production needs and options for the United States.

[137] United States Government Accounting Office, Managing Critical Isotopes: Stewardship of Lithium-7 Is Needed to Ensure Stable Supply, GAO-13-716 (September 2013).

There is a wildcard in lithium isotopic production[138]. The use of lithium-6 in batteries can significantly improve their peak power capability relative to the use of natural lithium. As a consequence, there is also the possibility that one or more companies may develop lithium isotopic separation capabilities to supply lithium-6 for high-value battery applications such as space and military systems. If this occurs, it would also imply the capability to produce lithium-7. If lithium-6 is commercialized, it implies a production capability orders or magnitude greater than would be required for a test reactor and low cost Li-7.

7.3.11 Other R&D

The FHR has multiple unique systems including instrumentation in hot salt, salt redox chemistry to control corrosion, and SNF management[139]. The characteristic of these systems is that they would not be expected to require major facilities and the associated lead times associated with those facilities.

There are other potential facilities not required for the FHTR but could be required by other programmatic considerations. The facilities may be required as one moves toward a pre-commercial demonstration plant.

- *Beyond design basis accident test facility*. The combination of a high-temperature fuel and a high temperature coolant may enable system designs that even in major accidents there would be no fuel failure. These designs are dependent upon the design of the specific reactor and silo containment structure. A test facility would test a specific design and be a non-nuclear test facility. A BDBA is not a consideration for the test reactor. Given the small size of this reactor, the concern will be cooling accidents that freeze salt, not accidents that overheat the reactor core.
- *Gas turbine heat storage with NACC*. The FIRES heat storage option associated with NACC has not been demonstrated. If there were large commercial incentives (high natural gas prices, restrictions on greenhouse gases or very low-cost electricity for limited hours per day—such as an electricity grid with a large installed PV capacity), this technology would require demonstration. The major component to be demonstrated would be the electricity heating systems since the other system components will likely be demonstrated by the development of Adiabatic Compressed Air Storage Systems. The use of heat storage with gas turbines can be applied to gas turbines burning natural gas and thus in the time-frame of FHR commercialization, this technology may be developed for other markets.[140]

[138] C. Forsberg, "Future Cost of Isotopically Separated Lithium for PWRs, Fluoride-salt-cooled High-temperature Reactors (FHRs) and Lithium Batteries," Paper 8712, *Transactions of the American Nuclear Society*, Washington D.C., November 10-14, 2013.

[139] C. W. FORSBERG and P. F. Peterson, "Spent Nuclear Fuel Management for Salt-Cooled Reactors: Storage, Safeguards, and Repository Disposal", *2014 International Congress on the Advances in Nuclear Power Plants (ICAPP 2014)*, Charlotte, North Carolina, April 6-9, 2014.

[140] C. Forsberg, "Electricity Storage Using High-Temperature Resistance-Heated Storage and Gas Turbines", *Energy Conversion and Management* (In Press).

- *Alternative power cycles*. NACC was chosen because (1) it provides the greatest revenue stream and thus maximizes economic benefits and (2) the massive public (military) and private (utility vendor) investments in open-air gas turbines implies major improvements in performance in the next several decades. There are two other power cycles that should be considered as backup options. Both are being developed for other applications and thus progress in those programs will partly determine their viability for use with an FHR.

    - *Helium power cycles.* Helium Brayton power cycles are being developed for HTGRs. They have two technical advantages: (1) inert coolant that avoids all concerns about salt-power cycle interactions and (2) easy recovery of tritium from the power cycle. Limited analysis suggests that heat storage could be incorporated into a helium power for peak power production but one could not produce peak power using natural gas or hydrogen. There have been no studies on incorporating heat storage into helium Brayton power cycles.
    - *Supercritical carbon dioxide cycles.* Supercritical carbon dioxide cycles are potentially the most efficient power cycle in the range of 600 to 700°C and thus potentially the preferred power cycle for production of base-load electricity. Any tritium leaking from the primary system could be captured in this power cycle. Cycle. This power cycle is early in its development.

7.4 Other Considerations

One of the unusual aspects of such a program is the strong synergism with other reactor programs: fuel and materials from HTGRs; refueling, decay heat removal, control rod drives, containments, and high-temperature materials from sodium and lead fast reactors; and tritium handling and high temperature materials from fusion programs. The existing facilities associated with those programs will provide much of the technology and many of the required test facilities.

8 Recommendations and Conclusions

The path forward to an FHTR will depend upon technical and institutional developments over the next few years. Because of the inherent uncertainty looking forward in time, this report defines the various options and pathways as a guide for future decisions.

Based on current technical understandings and constraints, the recommended FHTR path forward has the following features: (1) general purpose FHTR capable of testing multiple fuels and coolants, (2) government ownership with support through an international consortium, (3) licensing as a test reactor, and (4) smaller test facilities relative to other reactor types due to the availability of an excellent organic coolant simulant and fuel development work from the ongoing DOE NGNP program. The cost of such an FHTR program based on similar programs is expected to be ~2 billion dollars—primarily for development activities.

A significant fraction of the R&D is not directly associated with the FHTR but rather collecting the experimental data and conducting the required design tests that when coupled to the FHTR will enable a decision to be made on whether to build a pre-commercial facility.

At the same time, a serious effort should be undertaken for a large cooperative effort with the CAS as either a second pathway to general-purpose FHTR or as a broader program for joint development of the FHR. The U.S. and China are the largest economies in the world and the largest emitters of greenhouse gases; thus, they have the most to benefit from an advanced reactor that can address the needs of a zero-carbon nuclear-renewable energy system by providing dispatchable low-carbon electricity.

Appendix A: Liquid Salt Coolant Options for the FHR

A.1 Introduction

There have been multiple studies of the advantages and disadvantages of different fluoride salt coolants for the FHR. The choice of a coolant involves complex tradeoffs among the salt's nuclear properties (for efficient and safe use of fuel), melting point, heat transfer properties, corrosion potential, level of induced radioactivity, handling properties, and cost. The coolant choices are coupled to the choice of the reactor's fuel, structural materials, size, and application. Table A.1 summarizes the properties of some of the primary candidate coolants for the FHR. All of the candidate salts are binary salts. This is because the melting points of binary salts are far below that of pure compounds.

Table A.1 Potential Candidate Coolant Salts for the FHR

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m^3)	ρC_p (kJ/m^3 °C)
66.7 ^7LiF-33.3 BeF_2 (flibe)	**459**	**1430**	**1940**	**4670**
59.5 NaF-40.5 ZrF_4	500	1290	3140	3670
26 ^7LiF-37 NaF-37 ZrF_4	436		2790	3500
51 ^7LiF-49 ZrF_4	509		3090	3750
Water (7.5 MPa)	0	290	732	4040

Salt compositions are shown in mole percent. Salt properties at 700ºC and 1 atm. Sodium-zirconium fluoride salt conductivity is estimated—not measured. Pressurized water data are shown at 290°C for comparison

Flibe (66.7 ^7LiF-33.3BeF_2) was chosen as the baseline FHR coolant for several reasons: lowest neutron absorption (efficient uranium usage), negative void coefficient (reactor safety), reasonable physical properties, very low residual radioactivity, low corrosion potential, low vapor pressure with no "snow" generation in cover gas spaces, solubility in water that simplifies cleanup of equipment, and successful experience as the coolant in the Molten Salt Reactor Experiment (MSRE) – an 8 MWt molten salt reactor (with UF_4 and UF_3 dissolved in the flibe coolant) that successfully operated in the late 1960s. It is also the best understood fluoride salt coolant with a recent thermodynamic evaluation of its properties[141]. From the perspective of the

[141] O. BENES and R. J. M. Konings, "Thermodynamic Evaluation of the LiF-NaF-BeF_2-PuF_3 system", *The Journal of Chemical Thermodynamics*, 41, 1086-1095, 2009.

reactor core designer it is the preferred salt because it has the best nuclear properties (Chapter 5) and the best heat transfer properties (Chapter 6).

However, flibe has several disadvantages: cost, toxicity and tritium production. In each case there are engineering solutions to address these challenges. The question is whether the optimum strategy is to use flibe and address these challenges or consider alternative salts with their own challenges. Because of the complexity of the tradeoffs that include operational issues, there may be strong incentives for a test reactor to use different salts to determine the preferred salt. The other complication is that advancing technology and developments in other fields may change the preferred salt option over time.

A.1.1 Cost

Flibe is potentially expensive because of the need for isotopically separated ^{7}Li with an enrichment of >99.99% (typically 99.995%). If the cost of ^{7}Li is high, there would be incentives to seriously consider alternative salts where sodium replaces the isotopically separated ^{7}Li. The cost of the fluoride salt has major impacts on many design features of the reactor.

- *Safety*. Large liquid coolant volumes add heat capacity to the reactor core and assure the core remains covered with coolant if there is a vessel leak. There are engineering solutions to minimize coolant volumes but these also have costs. For example, one can have the primary system in a secondary vessel filled with a low-cost fluoride salt that provides added heat capacity and assures excess liquid if pipe failures occur.
- *Operations*. If the coolant salt is relatively inexpensive, the primary vessel can be made larger to provide more internal space for inspection and refueling operations.

There is the potential that the cost of isotopically separated lithium may drop dramatically thanks to advancing technology and massive increases in demand for isotopically separated lithium for other markets (Section A.2). If the isotopically separated lithium is considered too expensive, the most likely replacement would be a salt composed of sodium fluoride and beryllium fluoride where the sodium replaces the lithium.

A.1.2 Toxicity

Beryllium is toxic. If beryllium toxicity becomes a major challenge, there would be incentives to consider alternative salts such as mixtures of lithium fluoride and zirconium fluoride. Zirconium becomes the replacement for beryllium (A.4). There is ongoing work to separate zirconium isotopes to reduce parasitic absorption by zirconium clad in LWRs. If that technology is commercialized, it would significantly improve the relative performance of zirconium versus beryllium salts in FHRs.

A.1.3 Tritium

Neutron reactions with lithium and beryllium result in tritium production. The conversion of LiF to ^{3}HF causes primary system corrosion with the generation of $^3H_2$—a radioactive gas that can diffuse through hot heat exchangers. There are strategies to control corrosion and tritium but with added complications. Sodium zirconium salts avoid tritium generation—as well as beryllium toxicity, and the need for enriched lithium (See section A.3).

A.2 Cost of Lithium Isotopic Separation

Lithium isotopic separation technologies were originally developed in the early 1950s for weapons. Since then, a small commercial market has developed for isotopically-separated lithium that consumes perhaps a ton of ^{7}Li per year. Current demand is met from stockpiles and from defense facilities in Russia and China. The U.S. has decommissioned its lithium isotopic separation facilities. New technologies for lithium isotopic separation have been developed and there is the potential for the market for isotopically separated lithium to grow by several orders of magnitude independent of any salt-cooled reactor program. As a consequence, there is a very large uncertainty in the future costs of ^{7}Li. This appendix[142] summarizes what has happened and may happen.

A.2.1 Lithium Isotopic Separation Economics

In the last 20 years there have been major advances in lithium isotopic separation processes[143, 144, 145] that use classical chemical engineering separations processes such as solvent extraction and ion exchange instead of the diffusion processes (gaseous diffusion and centrifuge) used for uranium enrichment. The Chinese Academy of Science recently started up a pilot plant using one of these new processes that is based on using crown ethers as the separating agent in a solvent extraction system. Traditional separation processes can be used because with very light elements (hydrogen and lithium) the relative differences in atomic mass have significant impacts on physical and chemical behavior. As a consequence, these processes can be scaled up quickly and the scaling laws are well known[146] . As the scale of the production operation increases, the capital costs per unit of production decrease by inverse of the production rate to the 0.6 or 0.7 power. This implies that a major factor in determining lithium isotopic separation cost is the size

[142] C. Forsberg, "Future Cost of Isotopically Separated Lithium for PWRs, Fluoride-salt-cooled High-temperature Reactors (FHRs) and Lithium Batteries." *Transacs American Nuclear Society*, Paper: 8712; Washington D.C., November 10-14, 2013.

[143] D. A. LEE, "The Enrichment of Lithium Isotopes by Extraction Chromatography," in *Isotope Effects in Chemical Processes*, W. Spindel Editor, Advances in Chemistry, American Chemical Society, 1969.

[144] G. ZHIGUO, L. ZAIJUN, and Y. JIE, "Lithium Isotope Separation", *Progress in Chemistry* (China), **23** (*9*), September 2011.

[145] K. G. HEUMAN, "Isotopic Separation in Systems with Crown Ethers and Cryptands" pp. 77-132, *Topics in Current Chemistry*, Springer, **127**, 1985.

[146] R. H. PERRY et al., *Perry's Chemical Engineering Handbook. 6th Edition*, McGraw Hill Book Company, New York. 1984.

of the commercial market [A.2.2]. The typical scaling factors are between 0.6 and 0.7. The implications on the cost per kilogram versus production are shown in Table A.2.

Table A.2 Relative Capital Cost of Lithium Production Facilities with Different Production Rates Using Classical Scaling Factors for Any Given Enrichment

Increase in Production Rate	Exponential Factor	
	0.6	0.7
	Relative Capital Cost Li Production	
10	0.25	0.20
100	0.063	0.040
1000	0.016	0.0080

Based only on capital costs, separation costs per kilogram would be expected to decrease by a factor of a hundred if the market size grows by a factor of 1000. There are several caveats. The most important caveat is that there is a market for the depleted lithium isotope—it's not a waste. Economics-of-scale ultimately level off. The leveling off of the economics of scale is often because of some minimum energy requirement per unit of product, the cost of the feed materials becomes the biggest factor, maintenance costs become dominating or minimum labor cost per unit of production is reached.

For traditional processes like solvent extraction and assuming a market for the tails stream, the next major costs are likely to be labor and maintenance since power usage is relatively small. For these costs, the cost of the final product is likely to be inversely proportional to the volume produced. It rarely takes more labor to operate and maintain a large chemical unit operation (such as a solvent extraction column and its associated feed and product tankage) than for a small one. The same is also true for the maintenance labor though parts replacement (if significant for the process due to corrosion or other aggressive conditions) may not scale at all with product volume.

With current demand for isotopically separated lithium (< 1 ton/year), the facilities are small and thus a factor of a hundred reduction in separation costs may be achievable by increasing the scale of operations. The question is: "what are the lithium isotopic separation markets?"

A.2.2 Markets for Isotopically Separated Lithium

PWR Water Chemistry

The traditional nuclear market for ^{7}Li is control of the pH (water chemistry) in pressurized water reactors (PWRs). PWRs use boric acid to control nuclear reactivity with the boric acid concentration changing with burnup. Lithium hydroxide is added to the water to control the pH and hence corrosion rates in the primary system. Isotopically separated ^{7}Li is used because of the high neutron absorption cross section of ^{6}Li and because neutron absorption in ^{6}Li results in the production of radioactive tritium. Only a few kilograms of ^{7}Li are required per reactor per year. This implies a commercial market of ~ a ton of ^{7}Li per year.

FHRs and molten salt reactors (MSRs)

The second market for salt-cooled reactors such as the FHR is discussed in this report. If these reactors are commercialized, the annual market for isotopically separated lithium would increase by two to three orders of magnitude compared to the current market for PWR chemistry control. The required enrichment would be higher—99.995% ^{7}Li.

Fusion

The proposed fuel for fusion reactors is tritium primarily produced by neutron absorption in ^{6}Li. Depending upon the specific design of fusion machine, a fusion reactor may use isotopically separated lithium[147, 148, 149]. The demand would be similar to that for FHRs and MSRs if fusion was commercialized. The timeframe of development and potential deployment is highly uncertain. A small market for isotopically-separated lithium-6 may be generated by the ITER—the large fusion experiment.

Lithium-ion batteries

Lithium batteries are the dominant battery technology in personal electronics (cell phones, laptops, etc.), are becoming the battery of choice for aircraft, and may become the battery of choice for automobiles. Some sports cars have a lithium battery option to replace the traditional lead-acid battery to save weight. This is because of their light weight and high power density relative to other battery technologies.

Battery technology is changing rapidly but independent of that technology, the ultimate rate of battery discharge is controlled by lithium ion diffusion through the liquids, pastes, or solids of the battery from one electrode to the other. Classical diffusion rates are proportional to one over the square root of the mass of the lithium isotope. This effect is insignificant in traditional batteries using zinc, nickel, or lead where the relative differences in masses of different isotopes of these elements are small. Only in a battery or fuel cell using hydrogen or lithium is the effect significant.

There are second order effects. With the low atomic masses of lithium, the chemical kinetics are significantly different for the two isotopes; but, kinetic effects depend upon the specific battery chemistry. The kinetics are faster for ^{6}Li than for ^{7}Li.

[147] L. A. EL GUEBALY and S MALANG, *Toward the Ultimate Goal of Tritium Self-Sufficiency: Technical Issues and Requirements Imposed on ARIES Fusion Power Plants*, University of Wisconsin, UWFDM-1330, April 2008 (Revised August 2008).

[148] L. A. El-GUEBALY and the ARIES Team, "Nuclear Performance Assessment of ARIES-AT," *Fusion Engineering and Design*, 80, 99-110, 2006.

[149] E. MOSES, T. DIAZ DE LA RUBIA, J. F. LATKOWSKI, et al., "A Sustainable Nuclear Fuel Cycle Based On Laser Inertial Fusion Energy (LIFE)," *Fusion Science and Technology*, 56, 2, 566-572 (2009).

Measured lithium isotopic effects[150, 151, 152] on diffusivity depend upon the system and have been measured from near zero to 25%. The lowest measured isotopic effects occur in systems where the lithium carries a molecular solvation shell and has to drag along four water molecules. However, any lithium battery designer will want the highest performance that occurs with the highest lithium ion diffusivity. The net effect is that a battery using ^{6}Li would ultimately be expected to have a power density at least 8% higher and perhaps 15% higher than one using ^{7}Li. That implies that the power output is 8 to 15% larger or alternatively the battery can be 8 to 15% lighter and smaller if battery size is based on the need for instantaneous power output by using ^{6}Li. As batteries improve, the diffusion and chemical kinetic limits become more important. The incentives to use isotopically separated lithium increase. The first market would be spacecraft where there are large incentives to reduce mass.

In applications such as aircraft where the electricity is needed to start a jet engine, batteries are sized based on output (kW)—not the total kilowatt hours of stored energy. Formula-1 race cars currently use hybrid drivetrains where kinetic energy is harvested in batteries during braking in order to supply an additional 160 horsepower from an electric motor during acceleration—a few seconds later. High end sports cars for public use are now beginning to adopt the same technology. In these applications the weight advantage is important. The potential commercial and military aircraft market is hundreds of thousands of kilograms of isotopically-separated lithium per year. The ultimate market would be the automobile market where high-performance hybrid or electric vehicles could use hundreds of thousands of tons of isotopically-separated lithium per year. For other applications such as laptops, batteries are chosen on total energy delivered (kWh). This is not significantly altered by the choice of lithium isotope.

If ^{6}Li batteries become commercial, there will be other powerful drivers to reduce separation costs. No major country will want lithium isotopic separation and the lithium battery market controlled by a single foreign country so governments will take actions to assure domestic supplies. The economic incentives would be so large that lithium isotopic separation would become a major area of research with costs being driven down by technological advances as well as economics of scale.

Lithium-6 is a controlled material because it is a component in some weapons. The large commercial and defense applications (space, military aircraft and missiles) for lithium batteries will require a reassessment of current material restrictions. The global investments in lithium batteries and associated lithium chemistry are rapidly eliminating any remaining barriers to lithium isotopic separation. It's the same chemistry from mining/milling to batteries to isotopic separations. The large difference in the performance of lithium batteries with different isotopes

---

[150] E. M. PELL, "Diffusion of Li in Si at High T and the Isotopic Effect," *Physical Review*, 119 (*3*), 1014-1021, August 1, 1960.
[151] A. FEINAUER, G. MAJER, and A. SEEGER, "Self-Diffusion in Lithium and Sodium," *Defect and Diffusion Forum*, Vol 143-147, pp 881-886, 1997.
[152] M. OMINI, "Diffusion Processes in a Liquid Mixture of Lithium Isotopes," Philosophical Magazine, 89 (1), pp. 1-25, January 2009.

of lithium implies advances in lithium ion battery technology open up new lithium isotopic separation techniques based on battery chemistry. In terms of plant design, the separation factors for lithium isotopic separations are larger than the separation factors for many commercial chemical separations implying the chemical engineering and scale-up will be easier than for some existing chemical plants.

A.2.3 Implications for the FHR

The existing isotopic lithium market is measured in hundreds of kilograms per year. Large-scale deployment of FHRs or MSRs could increase that by two to four orders of magnitude. Lithium batteries could increase demand for lithium isotopic separation by four to six orders of magnitude within 10 to 30 years. Whether lithium isotopic separation costs are a few dollars or hundreds of dollars per kilogram will be determined by market size for isotopically separated lithium.

A.3 Tritium Production, Corrosion, and Control Strategies

If a salt containing lithium or beryllium is used in the FHR, tritium will be generated. Tritium generation can cause corrosion and tritium releases must be controlled. There are three considerations: production rates, corrosion[153], and control[154].

A.3.1 Production of Tritium

In an FHR it is proposed that the flibe be enriched to 99.995 % Li-7 with the remainder being Li-6, a nuclide with a very high neutron absorption cross section. Under neutron irradiation, the Li-6 is converted to tritium—converting LiF into tritium fluoride (^{3}HF). The Li-7 itself has a small cross section for tritium generation. Additionally, there is a small *n,*α cross section in Be which produces additional Li-6. This newly generated Li-6 will produce additional tritium. Tritium production is both a corrosion concern and a radiological concern.

Table A.3 Tritium Control Environment by Reactor Type

Reactor	Relative ^{3}H Production	Salt Coolant Characteristics	Materials of Construction	Tritium Recovery for Reuse
FHR	1	Clean	Metals, Carbon	Optional
MSR	1	Fission Products	Metals, Carbon	Optional
Fusion	10^3	Clean	Metals	Required

[153] John D. Stempien, Ronald G. Ballinger, and Charles W. Forsberg, "Tritium Transport and Corrosion Modeling in the Fluoride Salt-Cooled High-Temperature Reactor", *Transactions 2014 American Nuclear Society Winter Meeting*, Anaheim, California, November 9-13, 2014.

[154] C. Forsberg, J. Stempien, and R. Ballinger, "Tritium Capture in Salt-Cooled Fission and Fusion Reactors," *Transactions 2014 American Nuclear Society Winter Meeting*, Paper 11032, Anaheim, California, November 9-13, 2014.

FHRs, Molten Salt Reactors (MSRs) and fusion reactors generate significant tritium (^{3}H). In an MSR the fuel is dissolved in the flibe coolant. Tritium is produced by the same routes plus ^{3}H from ternary fission whereas fission product tritium in an FHR is retained in the fuel. The tritium production per MWt in FHR and MSR fission reactors is 2 to 3 orders of magnitude larger than in a PWR, similar to a heavy water reactor and several orders of magnitude less than proposed fusion machines. The ^{3}H challenge was first recognized in the Molten Salt Reactor Experiment (MSRE) at Oak Ridge National Laboratory in the early 1970s[155]. There have been fusion experiments that operated with ^{3}H and provide some engineering experience. Table A.3 summarizes the different ^{3}H environments for the three reactor types.

If flibe is used as a coolant in an FHR or MSR, the ^{3}H generation rate will be higher at beginning of life (BOL). As the small fraction of ^{6}Li is burnt out, the rate will decrease to a constant value from tritium generated by neutron absorption in ^{7}Li and ^{6}Li generated by neutron absorption in beryllium. A typical BOL production rate is 10 Ci/MWd ($3\cdot10^{-4}$ g ^{3}H/MWd). The time to reach the equilibrium ^{3}H production rate could range from 5 to 20 years depending upon the plant specifics (e.g. core coolant volume to total primary coolant volume ratio).

A.3.2 Corrosion

Neutron absorption by LiF produces ^{3}HF—a chemically corrosive compound dissolved in the salt as $^{3}H^{+}$ and F^{-}. The ^{3}HF then interacts with impurities in the coolant, redox control agents in the salt and structural materials, resulting in the creation of $^{3}H_2$ or some mixture of ^{3}H and ^{1}H if normal hydrogen is in the system. This implies that corrosion control strategies[156] for HF determine the chemical form of ^{3}H and the relative amount of the ^{3}H existing as $^{3}H^{+}$ and $^{3}H_2$ in the system. While ^{3}HF remains dissolved in the salt, $^{3}H_2$ can diffuse through hot metals to the environment. The rate of escape of $^{3}H_2$ from the system (Section A.3.4) depends upon the permeability of different reactor surfaces, the surface area of each metal and the tritium removal system.

There are three possible outcomes for the HF that is generated from conversion of lithium into tritium.

- *Metal Corrosion*. Like hydrogen fluoride (HF), tritium fluoride is a strong oxidant, and the accumulation of dissolved TF in the salt creates a corrosive chemical potential capable of preferentially dissolving Cr in metal heat exchangers as CrF_2 according to Eq. (1).

$$Cr_{(s)} + 2TF_{(d)} \rightarrow CrF_{2(d)} + T_{2(g)} \quad (1)$$

[155] R. B. Briggs, "Tritium in Molten-Salt Reactors," *Reactor Technology*, 14 (*4*), 335-352, Winter 1971-1972.

[156] J. D. STEMPIEN, R. G. BALLINGER, and C. W. FORSBERG, "The Coupled Corrosion and Tritium Challenges of Fluoride-Salt-Cooled High-temperature Reactors", Paper 14026, *Proc. of the International Congress on Advanced Nuclear Power Plants*, Charlotte, U.S.A., (2014).

Almost all the metal surface is associated with the heat exchangers; thus, corrosion will primarily occur there.

- *Redox Control*. The FHR will have a chemical redox control system to minimize corrosion. Redox control implies adding a redox agent that preferentially reacts with the TF, generating a metal fluoride and tritium. In effect, a sacrificial material is put in contact with the salt to preferentially react with the TF to prevent corrosion of heat exchangers.

- *TF Removal*. TF will absorb on carbon and can be partly stripped from the salt.

A model called TRItium Diffusion EvolutioN and Transport (TRIDENT) has been developed to analyze tritium generation and its subsequent distribution in the Fluoride salt-cooled High-temperature Reactor (FHR). Because tritium production is intimately connected to corrosion in an FHR, a corrosion model capable of simulating corrosion product mass transport throughout the reactor primary coolant system has been developed in TRIDENT. Figure A.1 shows the model.

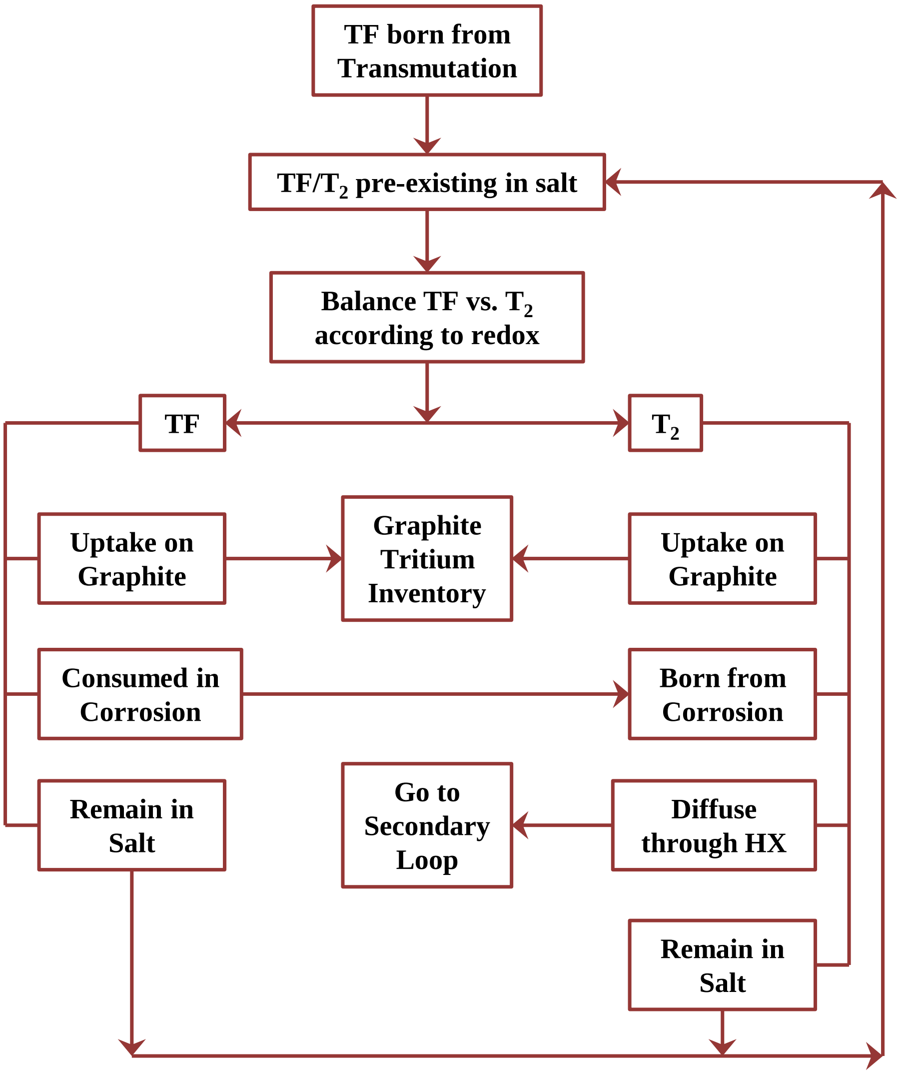


Fig. A.1 Flow chart for primary loop model in TRIDENT

Salt selection, tritium production, and corrosion are tightly coupled in FHRs with the tritium production dependent upon neutron irradiation. Consequently investigations of salt selection ultimately require test loops in a test reactor and/or a FHTR.

A.3.3 Control of Tritium

The health hazard of ^3H necessitates methods to capture and prevent releases of ^3H. Many papers[157, 158, 159, 160, 161] discuss barriers to the release of ^3H by putting coatings with low

[157] G. W. HOLLENBER et. al., "Tritium/hydrogen Barrier Development," *Fusion Engineering and Design*, 28, 190-208 (1995).

[158] S. R. SHERMAN and T. M. ADAMS, *Tritium Barrier Materials and Separation Systems for the NGNP*, WSRC-STI-2008-00358, Rev 1, Savannah River National Laboratory (Nov. 2008).

permeability to hydrogen on all surfaces of such reactors. The other requirement is to capture the ^{3}H. Both requirements must be met and are equally important. This section reviews methods to capture ^{3}H.

For power reactors, ^{3}H capture must be economic and fast. The longer ^{3}H remains in the coolant, the greater the probability of escape and the higher concentration and driving force for $^{3}H_2$ diffusion through hot heat exchanger surfaces. Seven capture options have been identified.

Tritium Capture on Carbon

Carbon strongly absorbs hydrogen[162]—including ^{3}H. At one time, carbon was being considered for hydrogen storage for cars. Furthermore, neutron irradiation can dramatically increase hydrogen absorption by graphite[163]. In the Molten Salt Reactor Experiment[164] where there was graphite moderator in the core, somewhere between 15 and 20% of the ^{3}H was ultimately absorbed by the carbon—an indicator of the capability of carbon to trap ^{3}H. If one does not want tritium in the graphite, major modifications such as coating graphite with SiC would be required[165]. There are two complementary options for tritium removal by absorption on graphite in a FHR.

- *Fuel as tritium absorber*. Some reactor design features enhance ^{3}H absorption. The heat must be transferred from the graphite-matrix coated-particle fuel to the clean salt coolant; thus, the fuel and carbon surface area are maximized to improve heat transfer—and resulting in high surface areas for ^{3}H absorption on the graphite. The graphite processing temperatures are limited by the need to avoid damage to the coated-particle fuel—resulting in carbon forms with relatively high internal surface areas favoring ^{3}H absorption. Preliminary results indicate that for some FHR designs, the carbon-matrix fuel is a major sink for ^{3}H recovery system[166]. In a pebble-bed FHR, the fuel consists of

[159] R. A. CAUSEY et al. "Tritium Barriers and Tritium Diffusion in Fusion Reactors", *Comprehensive Nuclear Materials, Vol. 4: Radiation Effects in Structural and Functional Materials for Fission and Fusion Reactors*, 511-549, (2012).

[160] T. TERAI et. al., "Tritium Permeation Through Austenitic Stainless Steel with Chemically Densified Coating as Tritium Permeation Barrier," *J. of Nuclear Materials*, 212-215, 976-908 (1994).

[161] N. ANDREWS and C. FORSBERG, "Tritium Management in Fluoride-salt-cooled High-temperature Reactors (FHRs), Paper 6407, *Trans. American Nuclear Society*, San Diego, California (Nov. 2012).

[162] R. A. CAUSEY et al. "Tritium Barriers and Tritium Diffusion in Fusion Reactors", *Comprehensive Nuclear Materials, Vol. 4: Radiation Effects in Structural and Functional Materials for Fission and Fusion Reactors*, 511-549, (2012).

[163] R. A. CAUSEY et al., "The Effects of Neutron Irradiation on the Trapping of Tritium in Graphite," *Fusion Technology*, 19, 585-588 (May 1991).

[164] R. B. Briggs, "Tritium in Molten-Salt Reactors," *Reactor Technology,* 14 (*4*), 335-352, Winter 1971-1972.

[165] X. HE et al, "SiC Coating: An Alternative for the Protection of Nuclear Graphite from Liquid Fluoride Salt", *J. of Nuclear Materials,* 448 (*1-3*) pp. 1-3 (May 2014).

[166] J. D. STEMPIEN, R. G. BALLINGER, and C. W. FORSBERG, "The Coupled Corrosion and Tritium Challenges of Fluoride-Salt-Cooled High-temperature Reactors", Paper 14026, *Proc. of the International Congress on Advanced Nuclear Power Plants,* Charlotte, U.S.A., (2014).

pebbles that flow through the core, are sorted for recycle or disposal as SNF and reinjection of recycle pebbles. In such a system it may be possible to use the pebbles as a ^{3}H capture system and extract the ^{3}H by heating the pebbles while they are being recycled to extract most of the ^{3}H.

- *Carbon particle bed*. An alternative strategy is to flow the salt coolant through a bed of small carbon particles that absorb the ^{3}H and use it as a ^{3}H absorber bed. There are large differences in ^{3}H absorption by different forms of carbon[167]. With an absorber bed, the carbon form can be optimized for tritium removal. Preliminary calculations indicate that tritium could be virtually completely captured in a system that recycled 1/60th of the absorber bed per day if the bed has a surface area 4 times that of the core fuel surface area. The carbon would be heated to drive off the tritium and returned to the reactor.

A secondary consideration is the fate of tritium fluoride in an FHR. Carbon also absorbs TF and thus the potential option of using a carbon bed to remove tritium in the form of hydrogen and hydrogen fluoride.

Tritium Capture on Hydrides

Hydrogen (^{3}H) forms highly-stable hydrides with materials such as yttrium[168]. However, these hydrides react with fluoride salts. Chemical reactions can be avoided by enclosing the hydride-forming compound between metal plates that have good corrosion resistance to fluoride salts and high permeability to ^{3}H—such as nickel [169]. The encapsulated hydride containing plates can be rolled into thin sheets—similar to the fabrication of some types of aluminum-clad research-reactor plate fuel elements. The thickness of the sheet is limited by structural requirements. The sheets can be used to fabricate a honeycomb structure with salt flows through the openings. The ^{3}H from the salt diffuses through the metal layer, is converted to a hydride, and captured. Alternatively wire can be manufactured with the hydride on the inside and fabricated into a high-surface area mesh. The tritium absorber is a solid that can be replaced when saturated and placed into long-term storage for decay of tritium, used as a disposal waste form or sent to a facility for tritium recovery for useful purposes. A variant of this concept[170] has been developed as a ^{3}H gas pump where at lower temperatures the tritium is absorbed as a hydride and at higher temperatures the ^{3}H is released.

---

[167] L. WANG et. al., "Unique Hydrogen Adsorption Properties of Graphene," *AIChE Journal*, **57**, 2902-2908, Oct 2011.

[168] S. R. SHERMAN and T. M. ADAMS, *Tritium Barrier Materials and Separation Systems for the NGNP*, WSRC-STI-2008-00358, Rev 1, Savannah River National Laboratory (Nov. 2008).

[169] R. A. CAUSEY et al. "Tritium Barriers and Tritium Diffusion in Fusion Reactors", *Comprehensive Nuclear Materials, Vol. 4: Radiation Effects in Structural and Functional Materials for Fission and Fusion Reactors*, 511-549, (2012).

[170] R. SCOTT WILLIAMS et. al. "Initial Testing of a Low Pressure Permeator for Tritium Processing," *Fusion Engineering and Design*, 49-50, 963-970 (2000).

A variety of such systems have been developed for defense and research facilities but these have been one-of-a-kind systems. The technical questions for this option are the preferred choice of metal (likely a high nickel content alloy), the preferred hydride, manufacturability, ^3H decay heat, and performance. The economic questions are the cost of materials and the very high surface area required to maintain low tritium inventories.

Tritium Capture by Vacuum or Gas Stripping.

Tritium from liquid salt can be removed by gas or vacuum stripping. With gas stripping, small bubbles of helium, another inert gas, or hydrogen are injected into the salt creating small bubbles, the $^3H_2$ or HT (if hydrogen is injected) diffuses through the salt to the liquid-gas interface and into the gas bubbles, the gas is separated from the liquid, and the $^3H_2$ is removed from the gas in an exterior loop. Experience from the Molten Salt Reactor Experiment[171] indicates that some HF can also be removed in this manner. Vacuum stripping is similar. The salt is sprayed into a vacuum creating small drops, the H_2 diffuses through the salt droplet to the liquid-vacuum interface, and the gas is pumped out of the system.

Early calculations[172] predicted DFs of 10^5 for vacuum stripping but later work indicated lower efficiencies because of several factors: (1) measured diffusion rates of $^3H_2$ through salt 20 times smaller than earlier estimates[173] and (2) gas molecular flow resistance from the salt-vacuum surface to the exit to the vacuum pump. Practical devices with small volumes imply large concentrations of salt drops that provide a resistance to $^3H_2$ molecules going the vacuum exit line—the other salt drops get in the way. Similar considerations apply to gas stripping. Newer techniques to improve gas stripping are under development—such as ultrasonic methods[174] to create very small gas bubbles in the salt with very high surface areas. This decreases the diffusion distance for the $^3H_2$ from the salt to a gas surface.

Tritium Capture by Double-Wall Heat Exchangers

For laboratory and defense applications, permeators have been developed for ^3H recovery[175, 176]. A permeator is a metal tube where the fluid with ^3H flows by the tube, the tritium diffuses through the tube, and the ^3H is captured on the other side of the tube wall. For fast ^3H removal, the surface area of the permeator has to be equal to or greater than the surface area of the heat

[171] R. B. Briggs, "Tritium in Molten-Salt Reactors," *Reactor Technology*, 14 (*4*), 335-352, Winter 1971-1972.

[172] T. J. DOLAN, G. R. LONGHURST, and E. GARCIA-OTERO, *A Vacuum Disengager for Tritium Removal from HYLIFE Reactor Flibe*, Idaho National Laboratory, EGG-M-91508, Idaho Falls, Idaho (1992).

[173] P. CALDERONI et al. "Measurement of Tritium Permeation in Flibe (2Li-BeF_2)", *Fusion Engineering and Design*, 83, 1331-1334 (2008).

[174] F. RUBIO, L. BOND, and E. D. BLANDFORD, "Sonoprocessing Applications for Advanced Nuclear Technologies," *Trans. American Nuclear Society*, Anaheim, California (November 2014).

[175] S. R. SHERMAN and T. M. ADAMS, *Tritium Barrier Materials and Separation Systems for the NGNP*, WSRC-STI-2008-00358, Rev 1, Savannah River National Laboratory (Nov. 2008).

[176] R. SCOTT WILLMS et al. "Initial Testing of a Low Pressure Permeator for Tritium Processing," *Fusion Engineering and Design*, 49-50, 963-970 (2000).

exchanger. The practical permeator for a salt-cooled reactor is to incorporate the permeator into the heat exchangers in the form of a double-walled heat exchanger[177]. Double wall heat exchangers are widely used in industry where heat must be transferred with high assurance of no leaks between the two fluids.

In the nuclear field, most research on double-wall heat exchanges has been associated with liquid metal fast breeder reactors (LMFBR) where the goal is to assure no interactions between the sodium and water. There is a small space between the tubes that allows some gas flow. The Japan Atomic Energy Agency (JAEA) has had a major development program in this area[178]. Double-walled tubes have also been examined as ^{3}H permeation barrier systems for use in Li-Pb fusion energy system heat exchangers, eliminating the necessity for an intermediate loop. In such a system, a DF of $>10^{5}$ was calculated with a ~25% increase in heat exchanger surface area.

There are four options for the ^{3}H capture between the double tubes. The zone between the two tubes can be a vacuum; but, this substantially reduces heat transfer. Alternatively this zone can contain a helium oxygen mixture where the helium is chosen to maximize heat transfer and the oxygen converts the ^{3}H to water that does not diffuse through tubes. Third, lithium metal can flow between the tubes. Lithium metal has an extraordinary solubility for hydrogen and has very good heat transfer properties. Last, a hydrogen getter can be put between the tubes[179] to capture and hold the ^{3}H. The quantities of ^{3}H are sufficiently small that all the ^{3}H could be held over the lifetime of the reactor assuming that there is not another source of hydrogen that consumes the hydride—such as from steam-metal oxidation or from the power cycle fluid diffusing into the heat exchanger. The disadvantages of all double-tube heat exchangers are their higher costs and added temperature drops across the heat exchanger.

Tritium Capture in Intermediate Heat Transfer Loops

At high temperatures if nothing is done, ^{3}H will diffuse through the heat exchangers. If the reactor has an intermediate loop, the tritium can be removed in that loop. The MSR program in the 1970s investigated the use of a special salt in the intermediate loop[180] to trap the ^{3}H. This salt was a mixture of sodium fluoroborate and sodium fluoride; captured ^{3}H was removed by a gas purge system.

If the intermediate loop is to be used for ^{3}H removal, it requires that the intermediate loop salt and materials decisions be driven by the requirements to remove ^{3}H. This can substantially

[177] L. GILMAN and C. FORSBERG, "Optimum Double-Wall Heat Exchanger for Containment and Trapping of Tritium in a Salt-Cooled Reactor." Paper 8637, *Trans. American Nuclear Society*, Washington D.C., (November 2013).

[178] H. HAYAFUNE, "Double-walled-straight-tube Steam Generator for future SFRs in Japan," in *Technical Meeting on Innovative Heat Exchange and Steam Generator Designs for Fast Reactors*, Vienna, 2011.

[179] D. E. HOLCOMB et al., *Fluoride salt-cooled High-temperature Reactor Technology Development and Demonstration Roadmap*, ORNL/TM-2013/401, Oak Ridge National Laboratory, Oak Ridge, Tennessee (September 2013).

[180] H. MacPHERSON, "The Molten Salt Reactor Adventure," *Nuclear Science and Engineering*, 90, 374-380, 1985.

increase costs. In the specific case of the MSR, the primary loop has the fuel dissolved in the salt and thus requires an intermediate loop to isolate this radioactivity from the environment.

Tritium Capture in Power Cycle

In some power cycles, the ^{3}H is trapped in the power cycles and can be easily extracted from the working fluid. Tritium diffusion is strongly temperature dependent; thus, ^{3}H diffusion through cold heat-rejection heat exchangers is low. Tritium can be trapped and removed if the power cycle uses helium or supercritical carbon dioxide. Removal of ^{3}H from helium is a well-developed technology because such systems are used in high-temperature gas-cooled reactors.

This is not a viable option for a FHR coupled to a Nuclear air-Brayton Combined Cycle where the heat is rejected to an open-air combined cycle at high temperatures. It is not viable for steam cycles because there are no efficient ways to remove ^{3}H from steam.

Avoid tritium production

The last option is to use a fluoride salt that does not generate ^{3}H; that is, no lithium or beryllium in the salt. This includes salt options such as a sodium-zirconium fluoride salt. There are neutronic and thermal hydraulic penalties but it is a viable option. Some of the implications of alternative salts for an FHR are discussed elsewhere in this appendix and in other papers[181].

A.3.4 Status of Tritium Control

There has been a large amount of work done on tritium control for defense programs and fusion programs. This includes fusion experiments that use megacuries of tritium where tritium control is a high priority. Tritium can be controlled; but, significant work will be required for practical cost-effective systems. The results of our initial modeling indicate that carbon absorbers and gas stripping appear to be the most practical and economic tritium control technologies.

A.4 Zirconium and Beryllium

The alternative to beryllium in the salt is zirconium. Zirconium salts tend to have poorer heat transfer characteristics, higher melting points, and higher nuclear cross sections. The physical properties are intrinsic characteristics. However, there has been significant work to develop zirconium isotopic separation technologies to lower nuclear cross sections of zirconium clad in light-water fuel. Lower absorption cross section zirconium would reduce uranium consumption and uranium enrichment requirements for LWRs. Thus far, the economics have not favored zirconium isotopic separation. If such an industry is created, there would be incentives to consider isotopically separated zirconium for the FHR.

[181] J. RICHARD, B. FORGET, C. FORSBERG, and K. SMITH, "Neutronic Comparison of Liquid Salt Primary Coolants and Novel Assembly Design for a Fluoride Salt Cooled High-Temperature Test Reactor," Paper 2014, *Trans. American Nuclear Society*, Anaheim, California, November 9-13. 2014.

A.5 Observations

In terms of reactor core design, flibe is the preferred coolant salt by a large margin. However, it brings its own challenges in terms of uncertainties about the cost of isotopically separated lithium, tritium, corrosion, and beryllium toxicity. The question remains is what is the optimum salt and will that salt remain the preferred choice over time. There is no right or wrong answer, which provides the basis for considering a test reactor with the capability to test closely related alternative salts.

Appendix B: Existing Test Reactor Capabilities for Salt Loops

B.1 Goals of Salt Loops in Test Reactors

Several different stages of testing will be required to develop and qualify the fuel, materials, and equipment needed for the primary circuit of an FHTR. In many cases non-nuclear component and separate-effects tests will be sufficient. However, before operating an FHTR a larger-scale reactor-based test loop will be required to reduce development risks. A reactor test loop, with either forced or natural convection salt flow at prototypical temperatures through a neutron- and gamma- irradiated test section, allows components to be tested under a more realistic corrosion, nuclear heating, radiation damage, and radionuclide transport environment.

A major component of this testing is confirmatory testing—testing that the fuel and other components will behave as expected. While such tests are expensive, the costs are much less than if it is discovered in the FHTR that there is a major problem with the driver fuel or other in-core component that necessitates a reactor core replacement. A second component of the testing is to validate the performance of models used to predict in-core behavior. These models are used to predict FHTR behavior. After the FHTR is operational, its operation will be used to validate models for larger FHRs. There is a step-wise progression from simple non-nuclear experiments and models to partly-integrated nuclear experiments (test loop) and more complex models to the FHTR and full system models that will ultimately be used to design a power reactor.

The production of activation products, in particular tritium, in the in-core section and subsequent transport to colder out-of-core areas such as the heat exchanger and chemistry control systems is difficult to simulate in a laboratory environment. A reactor not only provides an ample supply of tritium, but also modifies the salt chemistry due to activation and radiolysis in the same manner expected in an FHR. Tritium is transported through the primary circuit by the salt and will adsorb onto surfaces and diffuse through most materials at FHR temperatures. Therefore a reactor salt loop facility will be crucial for testing chemistry and tritium handling systems; demonstration of control of the reactor tritium inventory (e.g. preventing unintended releases through the power cycle or to the reactor containment) will be crucial for ultimate FHR design approval. Such a facility also provides an integrated effects environment that can be used to verify the performance of the materials in the FHR primary loop – fuel compacts, moderator/reflector structures, piping and welds, in-core instrumentation and their guide thimbles, pumps, and interfaces with outside systems (e.g. flanges, valves, diffusion barriers).

B.2 Test Reactor Capabilities for Loops

Currently no facility exists in a test reactor to irradiate flibe in a flow loop configuration. In support of molten salt reactor development several loops were installed in reactors such as the Oak Ridge Research Reactor in the 1960s. These reactors are no longer operating but reports on these activities, as well as experience from the modern out-of-core salt loops like those being constructed at ORNL, the University of Wisconsin-Madison, and The Center for Thorium Molten Salt Reactor Systems in China will be helpful in developing an in-reactor facility.

There are only a few test and research reactors in the U.S. capable of hosting a salt loop and providing a prototypical (or higher) neutron and gamma flux. While a few of these reactors also have experience with pressurized water loops used for LWR materials and fuels testing, a salt loop will likely require a larger footprint and support system due to the need for trace heating, cover gas control, and double encapsulation of the entire system to prevent interaction with water and contain tritium releases.

MIT Research Reactor (MITR), MIT

MITR has current experience in the handling, irradiation, and post-irradiation examination of flibe and flibe-bearing materials. In-reactor tests were done at 700°C in flibe salt. The MITR normally operates a single LWR-type pressurized water loop in core for materials testing. It is limited, however, by its research reactor license that forbids fuel testing in a thermally-isolated loop. This limits testing to materials (graphite, metals, SiC, etc.) and surrogate fuel (ZrO_2 rather than UO_2) in 700°C salt. In-core positions are available up to 2-inches in diameter, and the reactor has facilities for handling irradiated loop structure installation and maintenance. At 6 MW the neutron flux and spectrum of the MITR is similar to that of the proposed FHR, making it a good simulation of the in-core environment. Maximum in-core irradiation position neutron fluxes are 3.6×10^{13} n/cm^2-s thermal and 1.2×10^{14} n/cm^2-s fast.

Advanced Test Reactor (ATR), INL

The ATR is the largest test reactor in the United States. It currently hosts five pressurized water loops, although only one loop is available for outside users. The flux traps for test loops available in the ATR are up to 5 inches in diameter, and when operating around 110 MW the flux is the highest available in the U.S., which would allow the most accelerated materials testing. The traps provide up to 1×10^{15} n/cm^2-s thermal and 5 $\times 10^{14}$ n/cm^2-s fast neutron flux. The ATR is licensed to test fuel in its loops, and has a full suite of post-irradiation facilities available within INL.

High Flux Isotope Reactor (HFIR), ORNL

HFIR offers similar flux levels to the ATR, and has a variety of irradiation positions available within the reflectors up to 2.75 inches in diameter. The larger reflector positions' neutron fluxes are up to 7.5×10^{14} n/cm^2-s thermal and 5×10^{13} n/cm^2-s fast. However, HFIR does not operate any loop experiments, and therefore does not currently have the systems for such a test; a salt loop facility would require additional administrative planning and investment. ORNL has substantial facilities for handling irradiated material on site up to LWR-sized assemblies.

Annular Core Research Reactor (ACRR), SNL

The ACRR at Sandia is a pool-type reactor that normally operates at 2 MW but is used for transient testing up to 1000 times that for short pulses. It has a large dry irradiation facility (9.25 inches diameter) located in the center of the core, and a second dry facility (20 inches diameter) located adjacent to the core. The ACRR is used for fuel testing up to and beyond failure, and has facilities for installing and monitoring these large instrumented tests. While it does not currently have a loop facility, there has been previous interest in installation of a sodium loop. The lower power level and epithermal spectrum ($4x10^{13}$ n/cm^2-s total flux with 56% >10keV) does not match the FHR as closely as other reactors, but the experimental volume is large and it may be useful for transient testing not available elsewhere.

Belgian Engineering Test Reactor-2 (BR2), Belgium

The BR2 is a 120 MW reactor with three current water loop facilities used for fuel testing, including full-sized fuel elements (neutron fluxes available up to 9 $x10^{14}$ n/cm^2-s thermal and $7x10^{14}$ n/cm^2-s fast). Within the vessel there are irradiation locations with up to 20 cm diameter, however the current loop positions accept only up to 9.8 cm. There are on-site PIE facilities designed to couple to the reactor pool for the transfer of capsules in water to large hot cells.

Open Pool Australian Light-Water Research Reactor (OPAL), Australia

The OPAL reactor is recently-constructed 20 MW pool-type core with irradiation positions available in an external heavy-water reflector. These positions are up to 30 cm in diameter with a neutron flux up to $3x10^{14}$ n/cm^2-s thermal and $8x10^{12}$ n/cm^2-s fast. It does not currently have any irradiation loops or the corresponding facilities, however there is interest in exploring such capability.

Jules Horowitz Research Reactor (JHR), France

The JHR is currently under construction in France, and is designed to be the new premier light water reactor materials and fuels testing reactor in Europe. It will have up to $1x10^{15}$ n/cm^2-s fast neutron flux available in up to 8 cm diameter in-core positions or 20 cm reflector positions. One of the experiments currently being planned is an in-core double-encapsulated NaK loop for fuel testing (CALIPSO). This lays potentially useful groundwork for the design and approval of a salt loop at a later time.

Transient Reactor Test Facility (TREAT), INL

TREAT is a transient testing reactor used to test fuel up to and beyond failure. It is air-cooled and operates up to 100 kW steady-state. While TREAT was shut down in 1994, there is a DOE program underway to support its restart. While not sufficiently powerful for nominal operation studies, it may be an option for transient testing.

Appendix C: Reactor Driven FHR Subcritical Facility

C.1 Introduction

No FHR has ever been built, thus the question of how to perform physics tests and integrated materials and instrumentation tests in radiation environment and to obtain operation and maintenance experience for development of an FHR needs to be addressed. There are three types of facilities to obtain experimental data in neutron fields at a scale and cost significantly less than a test reactor: 1) critical assemblies that can be full scale but operate at zero power for neutronics code validation, 2) loops with coolants, materials and fuels in test reactors that operate at high power levels but where the test volume is very small and 3) reactor-driven sub-critical systems with intermediate test volumes and power levels.

The first two options are commonly adopted in the nuclear industry to demonstrate newer technologies. Sub-critical systems are a less conventional approach for nuclear system demonstration and have not received much attention in the last several decades because the United States has not developed new reactor systems for decades. The subcritical system enables integrated systems tests not possible with an in-pile salt loop because of its much larger volume. A preliminary design, based on the MIT Research Reactor (MITR), is given in this appendix as an example for implementing the reactor-driven sub-critical system.

C.2 Key Features

The key features of the sub-critical system are briefly discussed in this sub-section, in particular, the pros and cons relative to the other two options, i.e., zero power critical assemblies and loops (with fuel sample implemented) in test reactors that operate at high power levels.

The zero power critical assemblies are able to provide full-scale demonstration of the neutronic features of the examined nuclear system. Such an option is typically used to explore neutronic concepts. An in-pile test loop is able to couple the effects of thermal-hydraulics and irradiation effects. Due to limited test volume, however, neutronic features and integrated system behavior cannot be fully addressed. This option is commonly adopted for investigating nuclear components, such as new fuel, cladding, instrumentation etc.

The reactor-driven sub-critical system enables the integrated demonstration of part of a FHR reactor core. First, it can represent the core design and coolant system features. Second, it is capable of providing intermediate neutron flux level, up to 30% of that in the reference system. This feature for the FHR allows one to investigating the coupled effects of thermal-hydraulics, tritium generation and control, and radiolysis. Nuclear heating can maintain high temperature environment for material irradiation tests with some additional electrical heating. Last but not least, reactor-driven sub-critical systems can incorporate many components that are required in an FHR such as instrumentation and control rods. The integrated effect at representative operating condition can be studied—including many of the operational and maintenance challenges. Such systems have also been considered for fusion engineering studies[182] and may receive more attention with new proposed fusion systems[183] that use salt coolants.

C.3 Preliminary Design

The MITR is a 6 MW research reactor located on MIT campus in Cambridge, MA. It is a tank-type reactor with a compact core design, and is moderated and cooled by light-water and has a heavy-water reflector[184]. The reactor is designed primarily for experiments using neutron beams and in-core irradiation facilities. Adjacent to the MITR core there is a fission converter-based epithermal neutron irradiation facility licensed as an experimental facility by the U.S. Nuclear Regulatory Commission (NRC) for neutron capture therapy (NCT) research.[185] Thirteen years after the initial operation commenced in 2000, the fission converter was de-fueled in 2013. An isometric view of the entire facility at the MITR is shown in Fig. C.1.

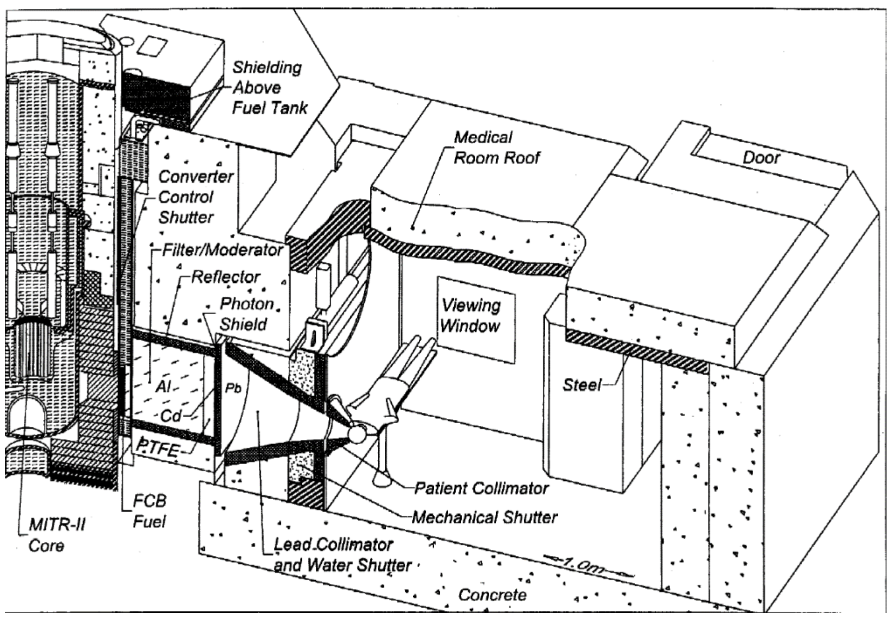


Fig. C.1 Isometric view of the fission converter-based epithermal neutron irradiation facility constructed at the MITR.

This facility can be replaced by a reactor-driven fluoride-salt-cooled subcritical system that can demonstrate many of the FHR technical features at prototypic operating temperatures and somewhat lower power densities. It is an option to obtain required information to 1) support construction of a test reactor and 2) investigate a wider set of core design options than are

[182] A. Pant, *Feasibility Study of a Converter Assembly for Fusion Blanket Assemblies*, MS Thesis, MIT, 1971.
[183] B. N. Sorbom et al. "ARC: A Compact, High-Field, Fusion Nuclear Science Facility and Demonstration Power Plant with Demountable Magnets" *Fusion Engineering and Design*, September 15, 2014.
[184] MITR-STAFF, "Safety Analysis Report for the MIT Research Reactor", 2011, MIT Nuclear Reactor Laboratory.
[185] O. K. HARLING et al., The Fission Converter–Based Epithermal Neutron Irradiation Facility at the Massachusetts Institute of Technology Reactor, Nuclear Science and Engineering, vol. 140: 223-240.

possible in a test reactor. The fission converter facility and associated medical room provide the space needed for the proposed MITR driven fluoride-salt-cooled sub-critical system.

The MITR driven fluoride-salt-cooled sub-critical system presented herein, as an example, adopts the design features of a 20 MWth transportable FHR (TFHR) proposed in Ref [186]. The subcritical system would contain two concentric fuel element rings rather than the three rings in the design of the TFHR. Also, the central location will contain a fuel element instead of instrumentation and control rods. The core height is reduced from 130 cm to 80 cm; whereas the width across flats of each hexagonal element remains at 36 cm. The fuel block dimensions are the same as the original General Atomics fabricated graphite block for Gas-Turbine-Modular Helium Reactor (GT-MHR)[187]. The above-mentioned re-sizing makes the active zone of the sub-critical system about eight times smaller than that of the 20 MWth TFHR. The compact design enables a higher power density and an increased neutron flux. Similar to the fission converter for NCT, the sub-critical system would be located next to the 14-inch window of the MITR, where the thermalized neutrons come through. The center of the sub-critical system will be located against the center of the 14-inch window. The vertical and horizontal cross-sections of the MITR core and subcritical system are shown in Fig. C.2.

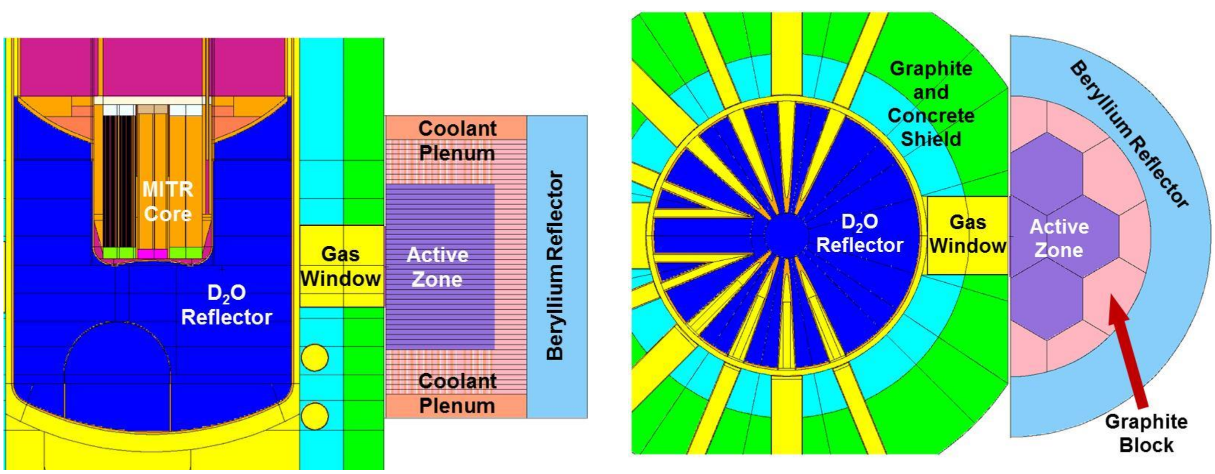


Fig. C.2 Vertical and horizontal cross-sections of the MITR driven sub-critical system

The adopted TRISO fuel particle dimensions are the same as those used in the transportable FHR (i.e. 425/100/50/35/40 μm for kernel/buffer/IPyC/SiC/OPyC). The fuel material ($UC_{0.5}O_{1.5}$) and the graphite matrix also remain the same. The packing fraction of the fuel compact is 0.35, which is the identical as the current FHR pebble fuel option and is also recommended in Ref [188]

[186] K. SUN and L. HU, "Neutronic Design Features of a Transportable Fluoride-salt-cooled High Temperature Reactor", Proceedings of ICAPP 2014, paper 14166.
[187] POTTER and A. SHENOY, "Gas Turbine-Modular Helium Reactor (GTMHR) Conceptual Design Description Report", GA Report 910720, Revision 1, General Atomics, 1996.
[188] K. SUN and L. HU, "Neutronic Design Features of a Transportable Fluoride-salt-cooled High Temperature Reactor", Proceedings of ICAPP 2014, paper 14166.

as an optimal value from the fuel cycle viewpoint. As mentioned earlier, the selected FHR coolant is FLiBe, which consists of highly enriched ^{7}Li (99.995% wt%). The ^{235}U enrichment is considered as a variable in this study, since it can easily adjust the neutron multiplicity of the sub-critical system. The upper limit of the ^{235}U enrichment of low enrichment uranium (LEU) is 19.95 wt% due to the non-proliferation constrains. Burnable poison will not be implemented, since the main focus of the sub-critical system is not the fuel cycle. It should be noted that the reactivity control device and experimental channel have not been included in this scoping study.

In the context of a sub-critical driven system, k_{eff} (effective multiplication factor without neutron source) is no longer the most essential parameter to ensure the system can be maintained throughout a reasonable duration as in a reactor system. The real multiplicity of the system due to external source, k_{src} (multiplication factor with neutron source taken into account), has to be evaluated for licensing and criticality purposes[189]. Accordingly, the present study evaluates three sets of multiplication factor: k_{src_cold} (300 K), k_{src_hot} (900 K), and k_{eff_cold} (300 K). k_{src_cold} represents the highest neutron multiplicity of the sub-critical system and it is the one should be assessed for licensing limit. k_{src_hot} provides a rough estimate of the neutron multiplicity during actual operation. It will also largely determine the system fission power. k_{eff_cold} represents the reference state, when the sub-critical system is isolated by closing the shutters of the neutron beam. The multiplication factors will be investigated as function of ^{235}U enrichment (wt %). In addition, the system fission power will be quantified according to the neutron multiplicity. The calculations were done using the general purpose Monte Carlo code MCNP5_v1.60 with nuclear data library of ENDF/B-VII.0.

The multiplication factor of the sub-critical system is largely dependent on the ^{235}U enrichment (wt %) in fuel. Typically, the higher ^{235}U enrichment will enhance the neutron multiplicity in the system as shown in Fig. C.3. It is also seen that the maximum neutron multiplicity is achieved with 19.95 wt% ^{235}U enrichment. In addition, one can see that k_{src_cold} is generally higher than k_{src_hot} at the same ^{235}U enrichment level. Accordingly, the neutron multiplicity of the system due to external source at the room temperature is considered for licensing purposes.

[189] A. GANDINI and M. SALVATORES, "The Physics of Subcritical Multiplying Systems", Journal of Nuclear Science and Technology, vol. 39 (6): 673-686.

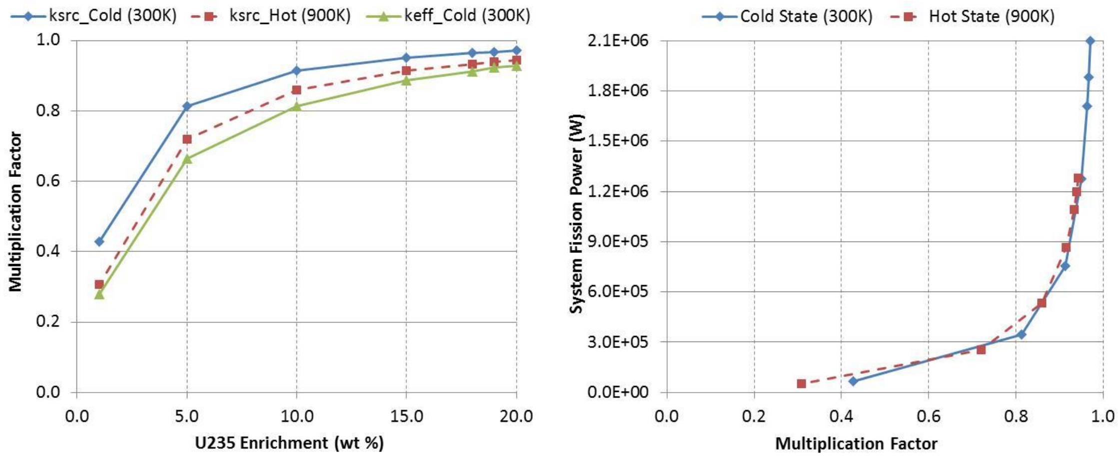


Fig. C.3 Multiplication factor as a function of the ^{235}U enrichment (Left). Fission power as a function of the multiplication factor (Right)

The fission power of the sub-critical driven system is mostly determined by the neutron multiplicity that depends upon design features, operating temperature, and other factors. It is seen in Fig. C.3 that the fission power will increase exponentially, when the neutron multiplicity becomes more intensive. As the multiplication factor approaches 0.9, the system fission power will reach ~800 kW. If the licensing sets 0.9 as the limitation of k_{src_cold} for the sub-critical driven system, k_{src_hot} will be slightly reduced due to a higher temperature at the actual operating condition. However, the fission power output is expected to be higher than 500 kW at the hot state. This power level implies the studied sub-critical system would have 15 - 20% average power density comparing to that of the reference FHR.

Table C.1 Neutron Flux Comparison of Different Systems

Unit: n/cm^2/s	MITR (6 MW)	Ref_FHR (20 MW)	Sub-critical (500 kW)
Thermal flux (< 1 eV)	3.33E+13	2.38E+13	3.97E+12
Fast flux (> 0.1 MeV)	1.23E+14	3.43E+13	4.68E+12
Total (Full range)	2.57E+14	1.33E+14	1.73E+13

The core average neutron spectra of the sub-critical system and the reference FHR are compared in Fig. C.4. It shows that two spectra have generally similar characteristics. The neutrons in the sub-critical system are slightly more thermalized, since the surrounding moderator (i.e. the graphite block) is more effective in terms of slowing down neutrons for a compact core. The energy-group-wise neutron fluxes are compared between different systems in Table C.1.

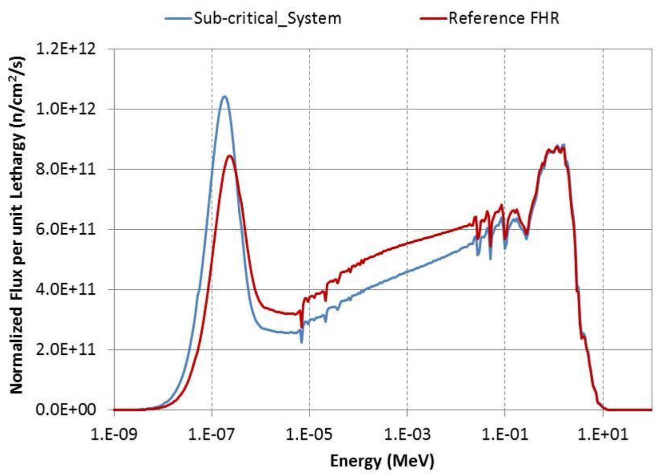


Fig. C.4 Comparison of neutron spectrum between sub-critical system and reference FHR

C.4 Summary

The main design features of the reactor-driven sub-critical systems are discussed in this appendix. In general, such a facility is considered as the best candidate for demonstrating the integrated behavior of nuclear reactor system with substantial lower capital costs than a test reactor because of 1) representative neutronic characteristics, 2) coupled thermal-hydraulics and radiolysis, and 3) large and flexible test volume compared to a loop.

A preliminary design based on the MIT Reactor is given as an example. The neutronic performance of the proposed sub-critical system is summarized as follows:

1. k_{src_cold} is generally larger than k_{src_hot} and k_{eff_cold} at the same ^{235}U enrichment level. It should be adopted as a conservative estimate for the licensing limit.
2. If licensing limit of the sub-critical system for k_{src_cold} is 0.9, the system fission power at the hot state (900 K) is expected to be above 500 kW.
3. Taking into account the compact active zone of the sub-critical driven system, 500 kW implies 15 - 20% power density of that of the reference FHR.
4. The neutron spectrum in the sub-critical driven system shows very similar characteristics as the reference FHR, with somewhat higher thermal-to-fast ratio.

Appendix D: Test Reactor Workshop Conclusions

A *Workshop on Fluoride-salt-cooled High-temperature Test Reactor (FHTR) Goals, Designs and Strategies* was held at Massachusetts Institute of Technology on October 2-3, 2014. The purpose of the workshop was to obtain the perspectives of experts on a proposed FHTR—goals, strategies, finance, and design. The United States has not built a new test reactor in over 40 years. In that time there have been major changes in technology, licensing, and other institutions.

The workshop participants received draft copy of this report in advance and were requested to read the abstract and summary—and any other sections that they had an interest in. This appendix summarizes major conclusions of that workshop—both areas of consensus among participants and areas where there were differing perspectives. The results of the workshop were used to revise this report. Appendix E includes workshop materials (Agenda, List of Participants, Presentations). Table D.1 is the agenda. Six major questions were identified (Table D.2) and sent to the workshop participants as the starting point of workshop discussions.

The workshop was the 6th workshop held as part of the Integrated Research Project, a joint project led by the Massachusetts Institute of Technology in partnership with the University of California at Berkeley and the University of Wisconsin (see project perspective). As with previous workshops, the many of the workshop materials and presentations were by graduate students working on the FHR.

The major conclusions from the workshop are summarized in Section D.1. In some cases there was general agreement among participants—that is noteworthy for it provides confidence in the conclusions given the expert knowledge of the participants. In other cases there were major differences in opinion among the participants. We believe that where there are major differences of opinion among experts, the area requires further analysis before conclusions can be reached. The reasons for the different perspectives are included herein. Section D.2 includes results of discussions on specific topics.

Table D.1 Workshop Agenda on Fluoride-salt-cooled High-temperature Test Reactor (FHTR) Goals, Designs, and Strategies

FHR Integrated Research Project Overview: Markets → Commercial Design → Test Reactor

Review of Workshop Objectives

Session 1: FHR Options: Commercial Design Space: Implications for Test Reactor

Session 2: Test Reactor Goals

Lunch Talk: Christian Gonnier (CEA) "The Jules Horowitz Reactor (JHR), a new high performance Materials Testing Reactor in Europe"

Session 3: FHR Test Reactor Designs

Design Considerations of SINAP's Test Reactor: Z. Dai

MIT General Purpose Test Reactor Design: J. Richard

Session 4: Test Reactor Licensing Strategy: DOE or NRC

Session 5: What is the financial and organizational strategy for an international test reactor project

Regis Matzie: What is the role of the commercial vendor for a Class I test reactor?

C. Gonnier: What required for an international Class II test reactor

P. Leysen: What required for an international test reactor

Dinner Speaker: Richard K. Lester: Innovation in Nuclear Power

Session 6: What are the capabilities of existing test reactors to install and operate 700°C salt loops or capsules? What is the role of reactor-driven subcritical systems? Are criticality facilities required?

Session 7: What major support facilities are required for a test reactor?

Lunch talk: Paul Leysen (SCK-CEN) "Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA): Goals, Design, and Organization"

Session 8: Roadmap: Path Forward

Session 9: Expert Feedback on the Path Forward

Table D.2 The Challenge and the Questions to the Workshop

No FHR has been built. As a consequence large-scale integral experiments will be required to commercialize the FHR. The Fluoride-salt-cooled High-temperature Test Reactor (FHTR) in terms of schedule, budget, and mission is the most important of these experiments. This workshop is to address the challenges associated with an FHTR and other supporting large-scale integrated experiments. This includes addressing the institutional (organization/funding), licensing, and technical challenges. The subject of a test reactor is relevant today. The Chinese Academy of Sciences plans to build a 10 MWt FHTR by 2020. In the United States the House of Representatives budget for this year includes appropriations for the U.S. Department of Energy to address the question of whether the U.S. should build a test reactor. Neither the mission nor goals for such a U.S. test reactor have been defined. This workshop will be one of the inputs to that DOE study. Some of the key questions to be addressed include:

1. There is recent experience in design, organization, and financing of new test reactors (JHR, MYRRHA). While no test reactors have been built in the United States in decades, new test reactors are being built in Europe, Asia, and South America. What has been learned (definition of goals, technical development effort, costs, organization)?

2. FHR test loops with flowing salt at 700°C and fuel can be built in existing and future test reactors to better understand fuel and coolant behavior before an FHTR is built. What are the testing capabilities (flux, power levels, fluid flow, geometry, etc.) of existing and soon-to-start test reactors (ATR, JHR, HFIR, etc.)? What are the needs for criticality facilities? What role for reactor-driven subcritical facilities?

3. What are the requirements for FHTR? What are the advantages and disadvantages of a general purpose test reactor versus a specialized test reactor going forward with time? The Chinese Academy of Science is developing a specialized FHTR to become operational by 2020. The MIT/UCB/UW project has proposed a more general purpose test reactor. Both will be described.

4. In the U.S., the licensing requirements for a test reactor are different than those for a power reactor. What is the difference and what is the same? What are the implications for a U.S. FHTR? What is the NRC capability to licensing a non-light-water test reactor? Should the alternative of a DOE licensed reactor be considered? What are the advantages and disadvantages of each?

5. By what means can a test reactor cost to the U.S. Department of Energy be minimized? If it is an international project, what does it imply in terms of organizing the R&D program to bring partners on board? How should an FHTR be financed and organized? What is the

viability of an international test reactor similar to the DRAGON—the first high-temperature gas-cooled reactor that was built in the United Kingdom?

6. What other large-scale integrated tests are required (thermal hydraulics, etc.). What can be left to the test reactor?

D.1 Major Workshop Conclusions

D.1.1 An Internationally-Funded FHTR is a Realistic Option

How to fund an FHTR? There was general agreement that international funding of a test reactor is a realistic option. The two lunch speakers (Christian Gonnier and Paul Leysen) described two test reactor projects that have international funding.

- *Jules Horowitz Reactor (JHR).* This is a high-performance water-cooled Materials Test Reactor in France that is under construction and nearing completion.
- *Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA).* This is a lead-cooled test reactor in the design stage that will have two operating modes: (1) an accelerator-driven subcritical system or (2) a fast reactor.

There is an important difference between the two reactors. The JHR is a Class II research tool to test materials—not the item to be tested. MYRRHA is a first-of-a-kind accelerator-driven system and the first lead-cooled fast reactor to be built outside Russia. The Russian lead-cooled reactors are submarine reactors. MYRRHA is partly a Class I test reactor and partly a Class II test reactor where one of the major test reactor goals is to develop the reactor technology—similar to an FHTR. The reactor is the test. The workshop presentations that describe these reactors are in Appendix E and include details on the funding mechanisms. There was a general consensus on what is required for a successful international project.

- *Project Leadership.* Successful international projects must have a lead country that is fully committed with a strong management structure where the project manager has complete control. There may be a corporate board based on ownership shares that the project manager reports to but that board does not manage the project.
- *Funding.* Funding by partners may be in the form of cash or services where the services may be required R&D for the test reactor, reactor components, or fuel—subject to the above constraint.
- *Alternative support options.* Several countries are now building test reactors that have very different test capabilities and goals. This may create for the United States the option of paying for the FHTR but including foreign nations as partners in exchange for access time at on their test reactors or partners in their nuclear R&D programs.

D.1.2 There is No Consensus on How to License an FHTR

Assuming that the U.S. Department of Energy owns the FHTR, there are two licensing options: (1) the DOE licenses the reactor or (2) the NRC licenses the reactor. Workshop session 4 addressed the question of how should an FHTR be licensed—with additional discussions at other times in the workshop. There was no agreement among the workshop participants on the preferred licensing strategy.

The development of a reactor involves three phases: test reactor, pre-commercial demonstration reactors, and commercial reactors. There was agreement that pre-commercial demonstration reactors and commercial reactors must be licensed by the NRC. One of the goals of a pre-commercial reactor is to provide information to determine reactor economics. That requires an understanding of licensing requirements for commercial reactors and thus the pre-commercial reactor must be licensed by the NRC.

The disagreement was on the licensing strategy for the FHTR. There was agreement that both FHTR licensing approaches provide equivalent safety and public input—but by different mechanisms. The discussion was about the differences between a Class I versus Class II test reactors and demonstration reactors. The FHTR would be a Class 1 test reactor—a first of a kind machine where many changes in its design would be expected during construction and operation because the reactor is what is being tested. This is different from a Class II materials test reactor such as the Advanced Test Reactor at INL where the reactor is not the test—it's the test machine that would be expected to use proven technology.

The rational for NRC regulation is that:

- It is current DOE policy
- The NRC needs to understand the technology for future licensing of a pre-commercial FHR
- The NRC has acquired recent experience from the early licensing steps for a homogenous aqueous reactor for medical isotope production

The rational for DOE regulation is that:

- The NRC regulatory structure is not designed to license Class I test reactors. It would be difficult to license and operate such a reactor on a reasonable schedule. The NRC is organized to license reactor designs with the expectation that the reactor is not the experiment and that the reactor design is fixed. Any license of a Class I test reactor would be a one-of-a-kind license where the characteristics of a Class I test reactor implies changes in the reactor during construction and operation.
- The DOE has a successful history of regulating one-of-a-kind nuclear facilities with its licensing structure—such as the Spallation Neutron Source at ORNL. It does license nuclear experiments. The DOE has isolated sites where part of the mission is one-of-a-kind testing of nuclear systems. Licensing is of a test reactor at a reactor test site where the safety case is coupled to the site—reverse of the NRC where licensing the reactor design is site independent.
- Government licensing for government-owned reactors is not limited to DOE. The U.S. Navy has its own licensing strategy for navy reactors as does NASA for space launch of reactors and radioisotope power systems. These separate and different licensing and safety strategies are driven by the unique requirements and missions.
- The NRC is an FHTR customer. The rational of a separate licensing agency is independence—but that does not apply when one of the primary goals is to provide safety information for the NRC.
- There is sufficient time between the building of a FHTR and any demonstration reactor for the NRC to develop a licensing strategy for FHRs. However, it is essential that the

NRC begin to build that licensing capability as the FHTR progresses so they acquire the specialized knowledge for licensing a pre-commercial FHR.

These discussions lead the report authors to conclude that one of the first steps in going forward for an FHTR is the need for a study to examine the options for licensing an FHTR and recommend a preferred licensing strategy.

D.1.3 How Many Test Reactors before a Pre-Commercial FHR?

There are several test reactor strategies that can lead to a pre-commercial FHR. One can build a very simple test reactor with a short lifetime followed by a more capable test reactor. This strategy reduces risk per step by taking smaller steps. It was used to develop the early fast reactors in the U.S.: EBR-I was built and followed by EBR-II. The Chinese Academy is planning to build the first FHTR by 2020. If a long-term agreement is made with the CAS, this route could be quickly implemented with the CAS test reactor being the first step followed by a U.S. FHTR shortly thereafter. Alternatively the U.S. could build a series of reactors as was done in the 1950s.

A second strategy is to build a test reactor where major changes can be made—including replacement of the entire reactor core. This was part of the development strategy that included the Shippingport PWR that had three radically different core designs that were sequentially tested. It implies a larger investment in the first FHTR. Many of the participants preferred this strategy because of the difficulty of funding and licensing multiple experimental reactors except in the context of a military mission.

D.2 Other Workshop Conclusions

Major recommendations and conclusions from the discussions at the workshop are summarized below by session. The conclusions are in *italic*.

FHR Options: Commercial Design Space and Implications for Test Reactor (Session I)

Need to define test reactor mission and lifetime—and with it the FHTR strategy. In the 1950s many test reactors had short lifetimes. A test reactor was built, experiments were done in several years, and one moved onto the next test reactor or larger demo depending upon what the results were. Some test reactors were designed to address a single question such as the SEFOR to determine the safety of sodium fast reactors by measuring the Doppler effect in oxide fuels. This reactor had a short operational lifetime (1969 to 1972). This strategy limits the financial commitments for a long-term reactor operational budget.

The strategy of multiple large-scale system tests is used today with the development of some weapons systems. The question is whether the U.S. institutional structure (licensing, funding, etc.) allows such an option for an FHR development program with test reactors? It is not a viable option if one has a licensing activity that extends for many years. A major factor in making such decisions will be the coupling between the CAS test reactor and any U.S. reactor program. If there is a large-scale cooperative program, then one has a pathway with a simple first test reactor in China followed by a more capable larger test reactor in the U.S.

What should a test reactor not do? The scope of the test reactor must be limited if goals are to be accomplished on a reasonable schedule. Examples of what not to test in an FHTR and why by workshop participants included:

- Power conversion systems. This can be done elsewhere
- Fuel cycle technology including actinide burning. This is not suitable for the first test reactors where operational issues must first be addressed.
- Determine commercial viability. This can't be credibly done in a test reactor, it requires a pre-commercial demonstration reactor.
- Severe accident testing. Extreme safety tests are not appropriate for the first test reactor;

Who is the customer? The customers for the test reactor must be defined because they will define goals that define design requirements.

- Vendors. The big questions for the vendors will likely be what types of fuel and coolants should be tested. For the first test reactor, the question will be whether more than one combination should be tested.
- U.S. Government. Many of the test reactor goals for the government will be associated with safety. The Nuclear Regulatory Commission will ultimately determine licensing rules for commercial reactors and thus may have specific goals in terms of tests to determine safety boundaries. This is separate from safety tests to assure safety of the test reactor. If the reactor has a limited lifetime, one should consider tests that may end the functional use of the reactor but would yield valuable information for safety. There is a successful history of using Class I test reactors for this purpose immediately before decommissioning—which then allows cutting apart parts of the reactor to determine test results. Many high-temperature systems can withstand one high-temperature transient—but then can't be used a second time safely at high temperature.

Test Reactor Goals (Session 2)

Must define limited goals at the beginning and not allow mission growth. Expansive goals increase technical risks. Mission growth destroys schedules and will dramatically increase costs. A Class I test reactor can't do all things.

The replaceable reactor core option deserves serious consideration. In the early development of nuclear energy there were a number of test reactors where the reactor core was changed out to allow alternative nuclear reactor core designs to be tested. The best known example was the Shippingport reactor that tested three very different light-water reactor core designs. This strategy is a half-way strategy between building a single test reactor with major capabilities and associated risks versus separate test reactors to test different features of the FHR.

Test reactor is only for tests that require neutrons. If the test does not require neutrons, it should be done in a non-nuclear facility. Cost and schedule demand this strategy.

Must have a strong central management team that defines who is responsible for each phase. Getting the management structure right with clear lines of responsibility will determine success or failure. If there is international participation, there must be a very clear division of specific responsibilities between partners.

FHR Test Reactor Designs (Session 3)

The Chinese Academy of Sciences and MIT FHTR designs were described (Appendix E).

Must use low-risk driver fuel. Fuel development is the long-lead time item for any new reactor. The CAS plans to use proven HTGR pebble-bed fuel. The MIT proposed design uses a modified Ft. St. Vrain hexagonal block HTGR fuel.

There is significant other relevant experience from the HTGR community. Because the FHR uses HTGR fuel, there is relevant experience in fuels, tritium control (United Kingdom) and other areas that can be relied upon to reduce development risks.

Test Reactor Licensing Strategy: DOE or NRC (Session 4)

Licensing of a Class I test reactor was viewed as a major challenge in the U.S. and Europe. This was discussed in multiple sessions and thus is summarized earlier in section D.1.2. The presentations in Appendix D describe the licensing approach by the DOE and the NRC. There was no consensus about the right strategy for licensing a Class I test reactor in the United States.

What is the financial and organizational strategy for an international test reactor project (Session 5)

This session evaluated the viability of organizing the FHTR as an international project lead by the U.S. This was discussed in this session and in the two lunch-time talks that discussed the Jules Horowitz Reactor being built in France and the planned MYRRHA project planned for Belgium. Both projects have international partners. The presentations discuss the requirements and strategies for the two projects. The workshop conclusions are in section D.1.1.

What are the capabilities of existing test reactors to install and operate 700°C salt loops or capsules? What is the role of reactor-driven subcritical systems? Are criticality facilities required? (Session 6)

There are four classes of nuclear test facilities that can provide experimental information for the design of the FHTR and future commercial FHRs to reduce development risks and times. The first three enable testing of materials in radiation environments to understand irradiation damage and corrosion effects in radiation fields. The costs of these facilities are an order of magnitude less than a test reactor.

- *Test loop*. This is a salt test loop in a materials test reactor that can duplicate the irradiation and thermal hydraulic conditions in an FHR. This is the most expensive test capability short of a test reactor. Appendix B summarizes existing facilities with this capability.

- *Reactor-driven subcritical facilities*. These irradiation facilities have more volume to represent part of a reactor but the power levels are a fraction of that of a test loop (Appendix C). They enable integrated tests that include fuel, coolant, control rods, and other components.
- *Salt capsule irradiations*. This is a capsule in a test reactor for testing materials and fuel. It can match the neutronics of a FHR but not the thermal hydraulic conditions in a test or power reactors.
- *Critical facilities*. These are near-zero power facilities that mock-up a reactor core to allow reactor physics tests.

There is a need of a detailed PIRT and gap analysis to define what tests are needed. The FHR is sufficiently different that experience with other reactors must be used with caution. For example, the FHR fuel lifetime is less than a year and a half. As a consequence there is no need for long-term accelerated fuel testing—a major use of test reactors with other fuel types. Considering development risks must be a major component of such a PIRT study. While there are strong arguments to use modeling and simulation in the place of expensive experiments to reduce costs and schedules, there is no recent experience in building Class I test reactors to define what can be calculated and what must be tested.

There are large incentives for a test loop in an existing test reactor to reduce test reactor development risks. One of the early missions is to qualify the test reactor fuel. While Class I test reactors have been used to qualify their own fuel, it implies a much slower startup of a test reactor and higher risks of replacing the entire reactor core if a fuel problem is identified.

The need for reactor-driven subcritical systems must be better defined. At one time such facilities were common—but today few people know of this option because no new Class I test reactors have been built in decades.

What major support facilities are required for a test reactor? (Session 7)

Whenever possible use non-nuclear test facilities that include long-term testing of components and systems rather than the test reactor. The cost and time for non-nuclear testing is substantially less than in a reactor.

Many non-nuclear test facilities will be required for the test reactor and any follow-on pre-commercial FHR. The design of test facilities should consider the long-term development pathways.

There are large incentives for a full-scale non-nuclear test facility of the FHTR (currently planned for the Chinese test reactor). This would be similar to the reactor except no nuclear materials. It would have three functions: testing for the FHTR, testing new systems for the FHTR after it is operational, and training.

Roadmap: Path Forward (Session 8)

Must get the stakeholders involved early in the process to assure well defined and agreed upon goals. Stakeholders include utilities and the major organizations that impact utility requirements (utilities, EPRI, NEI, state regulators, etc.), the vendors and their suppliers, regulators (NRC whether or not it licenses the test reactor, EPA considerations of long-term water use, greenhouse gas regulations, hazardous materials, other), the Federal government beyond the DOE (other interests from non-proliferation to military applications), and concerned groups. Consultation through the appropriate organizations is often the preferred route.

Require early vendor involvement. Given that the goal is a commercial reactor, there must be a strong interface with vendors. It may be desirable to consider the NASA model of industrial development—large trade studies that result in early vendor involvement that provide an industrial perspective on what a commercial FHR might look like and thus provide input into what the FHTR must test.

Economics is central. The reactor will not be developed unless compelling economic case for the electricity grid in 2030.

Invited Expert Comments (Session 9)

- Long-term goals determine whether it is worth going forward. The clear identification and justification of those goals is central.
- Define early what are the things <u>only</u> a test reactor can do? That defines test reactor goals. One needs to drive down options and justify each goal.
- Must define the U.S./CAS relationship because it can drive schedule and costs. There is the potential for a joint program that could leapfrog into a commercial product for a zero-carbon electricity grid.
- Need to define the licensing strategy for the FHTR. This can control schedule independent of all other factors.
- R&D early in the program on critical issues that can strongly influence design decisions such as lithium isotopic separation costs and tritium control costs that could impact salt coolant selection.
- Need for a highly detailed gap analysis to fully understand what is required and the sequence in which technologies must be developed and tested.
- Keep research pragmatic – focus on licensing and engineering.

Appendix E: Workshop Agenda, Participation List and Presentations

Fluoride Salt-Cooled High-Temperature Reactor (FHR) Workshop 6

Workshop on Fluoride-salt-cooled High-temperature Test Reactor (FHTR) Goals, Designs, and Strategies

Workshop Agenda

Taylor meeting room (3rd Floor)
Le Meridien Hotel
20 Sidney St
Cambridge, Massachusetts
October 2-3, 2014

Day 0: Wednesday, October 1, 2014

2:00-5:00 **Tour of salt properties laboratory and MIT reactor**
Reception Area of the Reactor Building
NW12, 138 Albany Street

5:30-7:00 **Reception**
Executive Foyer, Le Meridien Hotel (3rd floor)

Day 1: Thursday, October 2, 2014

8:00 – 8:30 **Continental Breakfast**
Executive Foyer, Le Meridien Hotel (3rd floor)

8:30 – 8:45 **Welcome and Participant Introduction**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

8:45 – 9:35 **FHR Integrated Research Project Overview: Markets → Commercial Design → Test Reactor**
New Test Reactors: Status U.S. and China: Charles Forsberg (MIT)
Market Basis for Commercialization: Charles Forsberg (MIT)
Design of Commercial Reactor: Per Peterson (UCB)

9:35 – 9:45 **Review of Workshop Objectives**

9:45 – 10:35 **Session 1: FHR Options: Commercial Design Space: Implications for Test Reactor**

Facilitator: D. Curtis and R. MacDonald
Discussants: Per Peterson, Z. Dai

FHRs have been proposed for commercial production of electricity and heat, transportable reactors for remote locations and ships, actinide burning, and radionuclide production. The test reactor owner will define what missions are important and thus the goals of the test reactor. Some applications are commercial whereas others are government. The different missions lead to different fuels and salts.

- *Does the potential for multi-missions define test reactor requirements and ownership? This leads to the question of a test reactor that is somewhat prototypical of the expected commercial machine versus a more general purpose test reactor (Driver fuel and test zone).*
- *For the U.S, is it credible to narrow the FHR design space at this time or should a test reactor be designed for the maximum versatility?*

10:35 – 10:50 **Break**

10:50 – 12:00 **Session 2: Test Reactor Goals**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

Facilitator: J. Stempien and J. Richard
Discussants: Lin-wen Hu, H. Gougar, J. Rushton, X. Yu

Historically there have been two categories of test reactors. Class I test reactors develop and demonstrate a new reactor concept—examples include EBR-1 (first U.S. fast reactor) and Dragon (first high-temperature gas-cooled test reactor). Some Class I test reactors over their lifetimes tested radically different reactor cores that included radically different control systems-such as Shippingport that could be viewed as a Class IA test reactor. Other Class I test reactors had narrow goals such as SEFOR and LOFT. Class II test reactors provide an irradiation machine to test materials and fuels—examples include ATR, HFIR, and the MIT reactors.

- *Is it credible to design a Class I test reactor to have significant Class II test reactor capabilities beyond testing FHR materials and salts?*
- *Should a Class I reactor be designed for shorter lifetimes to allow reduction in costs or testing over a much wider range of conditions*
- *What are the tradeoffs?*

12:00 – 1:20 **Lunch:** *Executive Foyer, Le Meridien Hotel (3rd floor)*

Lunch Talk: Christian Gonnier (CEA) "The Jules Horowitz Reactor (JHR), a new high performance Materials Testing Reactor in Europe"

1:20 – 3:15 **Session 3: FHR Test Reactor Designs**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

Design Considerations of SINAP's Test Reactor: Z. Dai
MIT General Purpose Test Reactor Design: J. Richard

The MIT FHTR is a Class IA test reactor with the goal to explore many design options. The SINAP FHTR is a Class IB test reactor with the goal to lead to an early commercial design. What is the range of capabilities for an FHTR? What are the challenges and issues that have not been addressed? What are the high-temperature irradiation tests (e.g., fuel, materials, instrumentation) that can be performed in the FHTR?

3:15 – 3:30 **Break**

3:40 – 4:15 **Session 4: Test Reactor Licensing Strategy: DOE or NRC**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

Facilitator: George Flanagan (ORNL)
Discussants: Steve Lynch (NRC), R. Budnitz, T. O'Connor, Rick Wright

The licensing options for the U.S. are by the DOE on a DOE site or the NRC. For reactors where there is previous experience (SFR, HTGR, etc.), the NRC is the likely to be the preferred strategy. Is that the right strategy for a first-of-a-kind reactor? What would be the NRC path forward? Are there incentives to consider the DOE licensing model?

4:15 – 5:00 **Session 5: What is the financial and organizational strategy for an international test reactor project**

Session Chair: Regis Matzie (Each expert will provide 5 to 7 minute perspectives followed by discussion)

Discussants: T. O'Connor, P. Ferroni, D. Moncton

Regis Matzie (Westinghouse CTO, Ret.): What is the role of the commercial vendor for a Class I test reactor?
C. Gonnier (CEA): What required for an international Class II test reactor
P. Leysen (SCK-CEN): What required for an international test reactor

There are two strategies for new reactors: single government ownership or an international partnership. Both have been successful in the past. The first HTGR, the Dragon Project, was an international project. The single government ownership model is country specific. International projects can reduce financial commitments by a single country but impose organizational constraints.

- *How could an FHTR be organized as an international project?*
- *What is the role of the traditional vendor at this very early stage of technology development*

5:00 **Adjourn**

6:00-9:00 **Dinner**
Loft Room, Le Meridien Hotel (1st floor)

Speaker: Richard K. Lester
Topic: Innovation in Nuclear Power

Day 2: Friday, October 3, 2014

8:00 – 8:30 **Continental Breakfast**
Executive Foyer, Le Meridien Hotel (3rd floor)

8:30 – 8:45 **Summary of Workshop Progress and Overview of Day 2**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)
Charles Forsberg (MIT)

8:45 – 10:00 **Session 6: What are the capabilities of existing test reactors to install and operate 700°C salt loops or capsules? What is the role of reactor-driven subcritical systems? Are criticality facilities required?**
Facilitators: Dave Carpenter and Kaichao Sun
Discussants: G. Kohse; L. Hu; P. Leysen (SCK-CEN); C. Gonnier (CEA), M. Ho (ANSTO)

Test reactors (ATR, Joule Horowitz, etc.) can enable testing of fuel and coolant before test reactor is built and complement the capabilities of an FHTR.

- *What are the capabilities of test reactors that will be available in the next several years?*
- *What level of resources (time and budget) are required for such test loops?*
- *Should reactor-driven subcritical systems be used to develop designs? Such facilities may have ~20% of the power density of a test reactor with larger experimental volumes.*
- *What is needed in terms of criticality facilities?*

10:00 – 10:15 **Break**

10:15 – 11:30 **Session 7: What major support facilities are required for a test reactor?**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

Facilitators: Edward Blandford
Discussants: M. Laufer; D. Carpenter, R. Matzie, K. Sridharan, D. Holcomb, M. Anderson

In addition to the radioactive test facilities (test loops, reactor driven subcritcal facilities, low-power critical facilities), there are other major support facilities for an FHTR: high-temperature component test facility, thermal hydraulic facilities, etc.

- *Is the identified list complete? What else is required?*
- *Do the unique capabilities of some salt simulants reduce the scale and size of these facilities?*
- *What is required to support the test reactor?*
- *What is required to support decisions for a pre-commercial demonstration project?*
- *What is required to support both activities?*

11:30 – 1:00 **Lunch**
Executive Foyer, Le Meridien Hotel (3rd floor)

Lunch talk: Paul Leysen (SCK-CEN) "Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA): Goals, Design, and Organization"

1:00 – 2:25 **Session 8: Roadmap: Path Forward**
Taylor Meeting Room, Le Meridien Hotel (3rd floor)

Facilitators: Charles Forsberg
Discussants: H. Khalil, M. Kazimi,

- *How does one go from concept through the test reactor to a commercial reactor?*
- *What activities for a test reactor and a demonstration commercial reactor are done in series versus parallel?*
- *How does one create a robust long-term program where the test reactor is only one component? What is the right model for the U.S.?*

2:25 – 2:40 **Break**

2:40 – 3:45 **Session 9: Expert Feedback on the Path Forward**
Facilitator: Charles Forsberg, Lin-wen Hu, Per Peterson

This discussion concludes the series of FHR Workshops. This is the final opportunity to raise important technical and economic bounding aspects for success of the FHTR program. Each of the expert participants will be asked to provide their perspectives on key issues.

3:45 – 4:00 **Concluding Remarks and Next Steps**
Charles Forsberg (MIT)

4:00 **Adjourn**

The Challenge and the Questions

No FHR has been built. As a consequence large-scale integral experiments will be required to commercialize the FHR. The Fluoride-salt-cooled High-temperature Test Reactor (FHTR) in terms of schedule, budget, and mission is the most important of these experiments. This workshop is to address the challenges associated with an FHTR and other supporting large-scale integrated experiments. This includes addressing the institutional (organization/funding), licensing, and technical challenges. The subject of a test reactor is relevant today. The Chinese Academy of Science plans to build a 10 MWt FHTR by 2017. In the United States the House of Representatives budget for this year includes appropriations for the U.S. Department of Energy to address the question of whether the U.S. should build a test reactor. Neither the mission nor goals for such a U.S. test reactor have been defined. This workshop will be one of the inputs to that DOE study. Some of the key questions to be addressed include:

1. There is recent experience in design, organization, and financing of new test reactors (JHR, MYRRHA). While no test reactors have been built in the United States in decades, new test reactors are being built in Europe, Asia, and South America. What has been learned (definition of goals, technical development effort, costs, organization)?

2. FHR test loops with flowing salt at 700°C and fuel can be built in existing and future test reactors to better understand fuel and coolant behavior before an FHTR is built. What are the testing capabilities (flux, power levels, fluid flow, geometry, etc.) of existing and soon-to-start test reactors (ATR, JHR, HFIR, etc.)? What are the needs for criticality facilities? What role for reactor-driven subcritical facilities?

3. What are the requirements for FHTR? What are the advantages and disadvantages of a general purpose test reactor versus a specialized test reactor going forward with time? The Chinese Academy of Science is developing a specialized FHTR to become operational by 2017. The MIT/UCB/UW project has proposed a more general purpose test reactor. Both will be described.

4. In the U.S., the licensing requirements for a test reactor are different than those for a power reactor. What is the difference and what is the same? What are the implications for a U.S. FHTR? What is the NRC capability to licensing a non-light-water test reactor? Should the alternative of a DOE licensed reactor be considered? What are the advantages and disadvantages of each?

5. By what means can a test reactor cost to the U.S. Department of Energy be minimized? If it is an international project, what does it imply in terms of organizing the R&D program to bring partners on board? How should an FHTR be financed and organized? What is the viability of an international test reactor similar to the DRAGON—the first high-temperature gas-cooled reactor that was built in the United Kingdom?

6. What other large-scale integrated tests are required (thermal hydraulics, etc.). What can be left to the test reactor?

FHR Test Reactor Workshop
Participants List

Name	Affiliation
Universities	
Charles Forsberg	MIT
Lin-wen Hu	MIT
Gordon Kohse	MIT
David Carpenter	MIT
Michael Ames	MIT
Kaichao Sun	MIT
Ron Ballinger	MIT
John Dennis Stempien	MIT
Rebecca R. Romatoski	MIT
Joshua Richard	MIT
Nestor Sepulveda	MIT
Daniel Stack	MIT
Melanie Tetreault-Friend	MIT
Chenglong Wang	MIT
David Moncton	MIT
Richard Lester	MIT
Mujid Kazimi	MIT
Kumar Sridharan	Univ. of Wisconsin
Mark Anderson	Univ. of Wisconsin
Raluca Scarlat	Univ. of Wisconsin
Guiqiu Zheng	Univ. of Wisconsin
Brian Kelleher	Univ. of Wisconsin
Per Peterson	UC Berkeley
Michael Laufer	UC Berkeley
Nicolas Zweibaum	UC Berkeley
James Kendrick	UC Berkeley
Ed Blandford	Univ. of New Mexico
Preet Singh	GA Tech
Dean Wang	UMass Lowell

National Laboratories & Government	
Hussein Khalil	Argonne
George Flannagan	Oak Ridge Nat. Lab.
David Holcomb	Oak Ridge Nat. Lab.
Tom O'Connor	Dept. of Energy
Hans Gougar	Idaho Nat. Lab.
Steven Lynch	NRC
Advisory Panel	
Regis Matzie	Westinghouse
Jim Rushton	Oak Ridge Nat. Lab.
Robert Budnitz	Lawrence Berkeley Lab.
Invited Speakers	
Christian Gonnier	CEA
Paul Leysen	SCK-CEN
Industry	
Paolo Ferroni	Westinghouse
Richard Wright	Westinghouse
Andrew Sowder	EPRI
Test Reactors	
Mark Ho	OPAL:ANSTO
Lance Maul	OPAL:ANSTO
Zhimin Dai	CAS
Guimin Liu	CAS
Zhaozhong He	CAS

Denotes student

NEUP Integrated Research Project Workshop 6:
Fluoride Salt-Cooled High Temperature Reactor (FHR)
Test Reactor Goals, Designs, and Strategies

Review of Workshop Objectives

October 2-3, 2014
Charles Forsberg

U.S. Department of Energy

The Workshop Objective is to Help Define a Roadmap to an FHTR

- There is no single pathway and the roadmap will change with time
- The roadmap is in the context of the United States; Roadmaps are country specific
- We have developed a pathway; what are the weaknesses and strengths? Other Options?
- Major components of roadmap
  - Defining test reactor goals (Coupled to ownership)
  - Who owns the FHR and how is it financed?
  - How is it licensed?
  - What are the design options?
  - What support facilities are required?

NEUP Integrated Research Project Workshop 6:
Fluoride Salt-Cooled High Temperature Reactor (FHR)
Test Reactor Goals, Designs, and Strategies

FHR Integrated Research Project (IRP) Overview

Markets → Commercial Reactor → Test Reactor

Charles Forsberg

8:45-9:35 am; October 2, 2014

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Workshop Objective: Define Pathway to Fluoride-salt-cooled High-temperature Test Reactor (FHTR)

- FHR is a new reactor concept and thus requires a test reactor to prove the concept
- Question is relevant today
  - The CAS plans to have a 10 MWt test reactor by 2020
  - The U.S. House of Representatives has included funding for a DOE study to define what should be the next test reactor built by the United States
- ***Workshop goals defined in the context of a possible future U.S. FHTR***
  - Workshop initiated before legislation but legislation is changing emphasis

The FHR Is a New Reactor Concept

- Concept is about a decade old
- Enabled by two advancing technologies
  - Natural-gas-fired combined cycles
  - Graphite-matrix coated-particle fuel
- Rapidly growing interest
  - Expanding R&D
  - Chinese Academy of Science decision two years ago to build first FHR test reactor: 10 MWt
- Why the interest?

The United States Has Successfully Commercialized only One Reactor Type

Light Water Reactor

- Basis for LWR commercialization
  - Developed LWR because it would revolutionize submarine warfare
  - Requirements for submarine propulsion close to utility power-plant requirements
- **Need compelling case for any new reactor**

FHR Integrated Research Project Strategy

This Presentation will Discuss the Top Two Activities

Commercial Strategy and Electricity Markets (MIT)
Definition of Near-term and Long-term Goals

↓

Commercial Reactor Point Design (UCB)

↓

Test Reactor (Workshop Topic)

↓

Technology Development (MIT/UCB/UW)

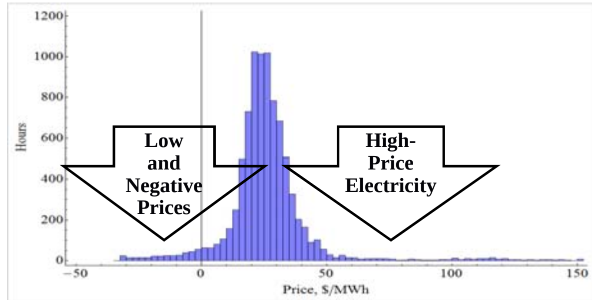
Goals for the Compelling FHR Market Case

- **Economic: Increase revenue 50% relative to base-load nuclear power plants**
- **Environment: Enable a zero-carbon nuclear-renewable (wind / solar) electricity grid by providing economic dispatchable (variable) electricity**
- **Safety. No major offsite radionuclide releases if beyond-design-basis accident**

Common Solution

The Electricity Market

In a Free Market Electricity Prices Vary

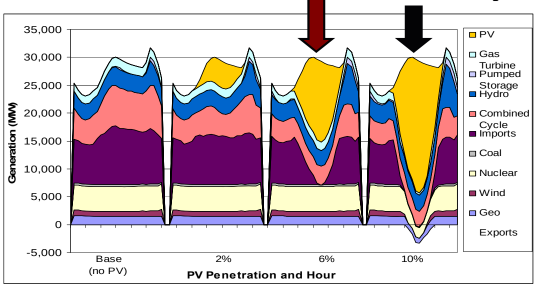


2012 California Electricity Prices

Adding Solar and Wind Changes Electricity Prices & Price Structure

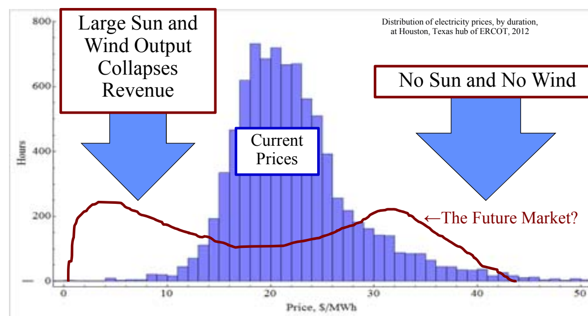
Unstable Electrical Grid

Excess Electricity with Price Collapse



California Daily Spring Electricity Demand and Production with Different Levels of Annual Photovoltaic Electricity Generation

Low-Carbon Electricity Free Market Implies More Hours of Low / High Price Electricity



Future Reactor Economics: Make and Buy Low-Price Electricity and Sell High-Price Electricity

FHR Economic Strategy

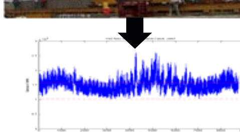
Reactor Core Operates Base-Load
Power Cycle Has Variable Output to Grid
Increase Revenue Relative to Base-load Plants

Combining FHR with a Nuclear Air-Brayton Combined Cycle (NACC)

Constant High-Temperature Heat Supply (600 to 700 C)
Reactor (FHR)

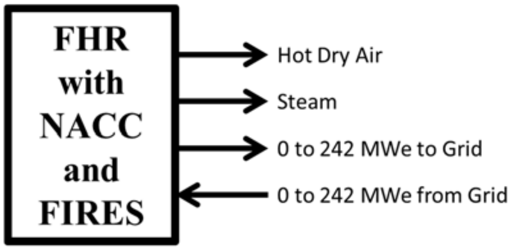
+

Combustible Fuels or Stored Heat
+
Gas-Turbine (NACC)



Variable Electricity

Modular FHR as a Black-Box
Can be Built in Different Sizes



NACC: Nuclear Air-Brayton Combined Cycle
FIRES: Firebrick Resistance-Heated Energy Storage

Not Your Traditional Nuclear Reactor

14

FHR Combines Existing Technologies

Fuel: High-Temperature Coated-Particle Fuel Developed for High-Temperature Gas-Cooled Reactors (HTGRs) with Failure Temperatures >1650° C

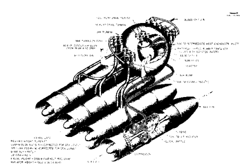
Coolant: High-Temperature, Low-Pressure Liquid-Salt Coolant ($^7Li_2BeF_4$) with freezing point of 460° C and Boiling Point >1400° C (Transparent)

Power Cycle: Modified Air Brayton Power Cycle with General Electric 7FB Compressor

MIT

15

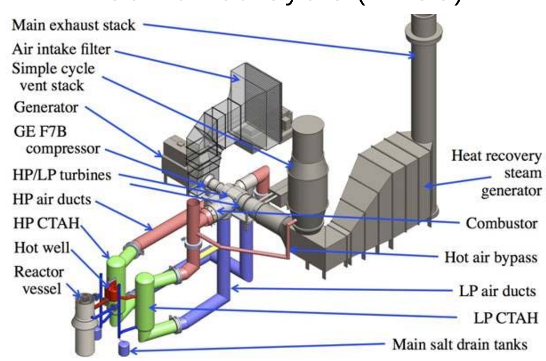
Fluoride Salt Coolants Were Developed for the Aircraft Nuclear Propulsion Program
Salt-Cooled Reactors Designed to Couple to Jet Engines



It Has Taken 50 Years for Utility Gas Turbine Technology to Mature Sufficiently to Enable Coupling with an FHR

16

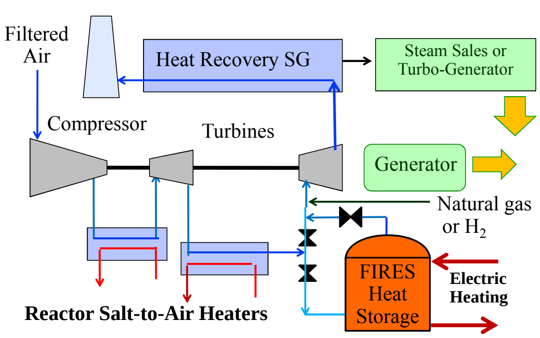
FHR with Nuclear Air-Brayton Combined Cycle (NACC)



Reactor ⟵ Power Cycle ⟶

17

NACC Power System
Base-load and Peak Electricity (Auxiliary Natural Gas or Stored Heat)

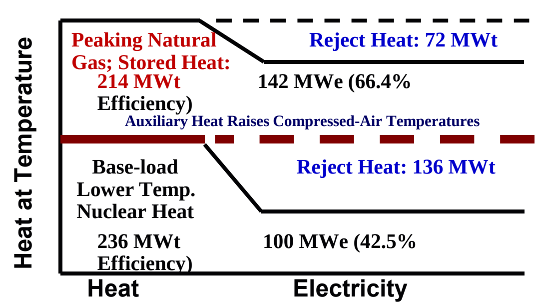


MIT

18

Base-Load Nuclear With Peak Power
High Natural Gas/ Stored Heat-to-Electricity Efficiency

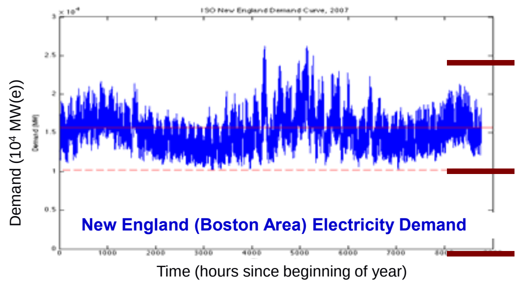
Base load: 100 MWe; Peak: 241.8 MWe



C. Andreades et. al, "Reheat-Air Brayton Combined Cycle Power Conversion Design and Performance under Normal Ambient Conditions," *J. of Engineering for Gas Turbines and Power*, **136**, June 2014

FHR with NACC Can Meet Variable Electricity Demand

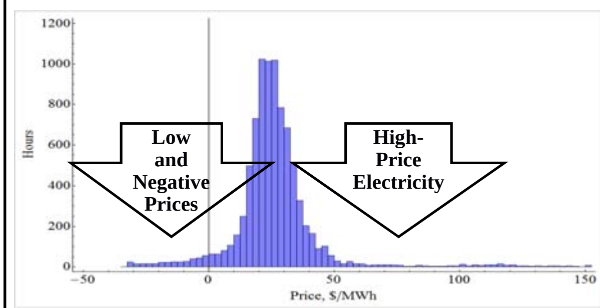
For Every GW Base load, 1.42 GW of Peaking Capability



Dispatchable Nuclear Electricity Option for Zero-Carbon Electricity Grid with Base-Load Reactor Operations

Natural Gas Peaking Option

Base-load When Low Electricity Prices; Natural Gas Peaking When High Electricity Prices



2012 California Electricity Prices

FHR Revenue Using 2012 Texas and California Hourly Electricity Prices

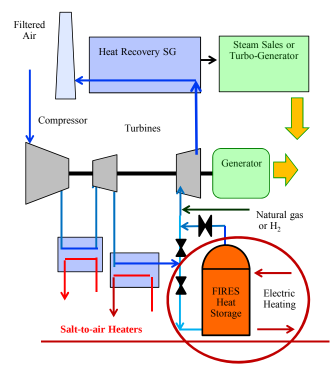
After Subtracting Cost of Natural Gas

Grid→ Operating Modes	Texas	California
	Percent (%)	Percent (%)
Base-Load Electricity	100	100
Base With Peak (NG)	142	167

1. Base on 2012 Henry Hub natural gas at $3.52.
2. Methodology in C. W. Forsberg and D. Curtis, "Meeting the Needs of a Nuclear-Renewable Electrical Grid with a Fluoride-salt-cooled High-Temperature Reactor Coupled to a Nuclear Air-Brayton Combined Cycle Power System," *Nuclear Technology*, March 2014
3. Updated analysis in D. Curtis and C. Forsberg, "Market Performance of the Mark I Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor, *American Nuclear Society Annual Meeting*, Paper 9751, Reno, Nevada, June 15-19, 2014

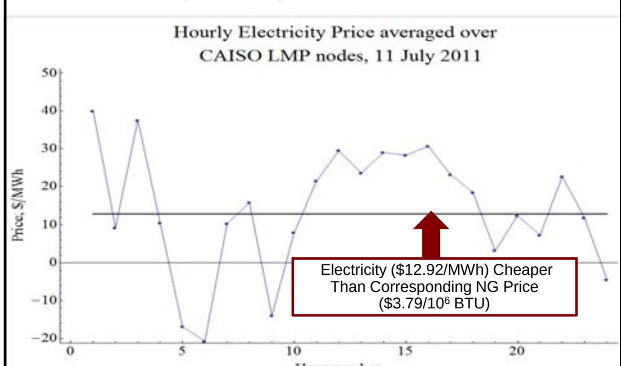
Stored Heat Option for Peak Power

Firebrick Resistance-Heated Energy Storage (FIRES)



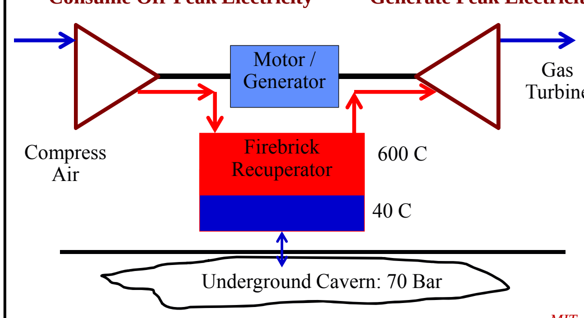
- Electrically heat firebrick in pressure vessel
- **Firebrick heated when low electricity prices; less than natural gas**
- **Use hot firebrick as substitute for natural gas peak electricity**
- Reasonable round-trip efficiency
  - 100% electricity to heat
  - 66+% heat-to-electricity efficiency (peak power)

California Price Curve Shows Times When Electricity Cheaper then Natural Gas



Gas-Turbine Firebrick Heat Storage Is Being Developed by GE/RWE for Adiabatic Compressed Air Storage Systems

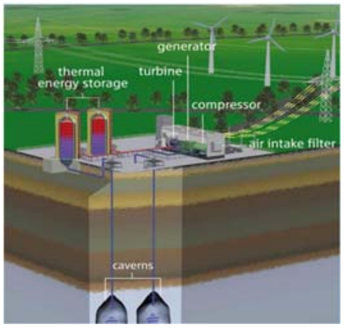
Consume Off-Peak Electricity **Generate Peak Electricity**



25

General Electric - RWE Adiabatic Compressed Air Storage (Adele) Project

Developing Most of the Technology Required for FHR Heat Storage



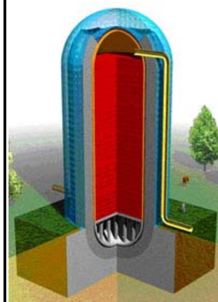
- Grid Electricity into Storage
  - Compress air to 70 bar and 600° C
  - Cool air to 40° C by heating firebrick
  - Compressed air to underground storage
- Electricity from Storage to Grid
  - Heat compressed air with firebrick
  - Turbine produces electricity

MIT

26

Adele Storage Vessel Testing Underway

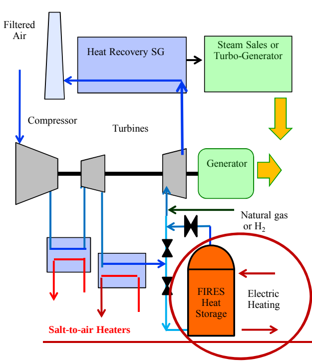
Integrating Heat Storage and Gas Turbine Technology



FHR NACC with Stored Heat Differences: Lower Pressure, Higher Temperature and Electric Heating

27

FHR "Electricity Storage" Does Not Require Backup Generating Capacity



- Batteries and other storage technologies require backup generating capacity for when storage capacity is depleted
- **FHR backup is natural gas or hydrogen if heat storage depleted**
- **FHR/NACC/storage has economic advantage over traditional storage technologies**

MIT

28

FHR Commercial Case Defines FHR Technical Requirements

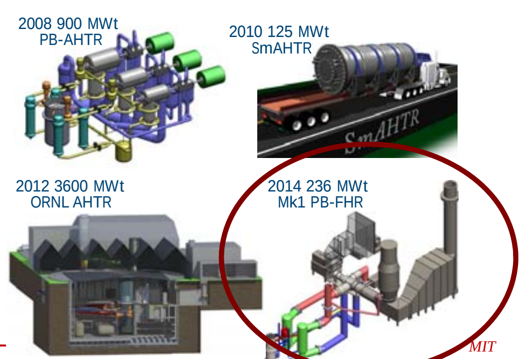
- Front-end air compressor exit temperature between 350 and 500° C—Nuclear heat must be at higher temperatures
- Nuclear heat delivery temperatures: 600 to 700 ° C
- FHR matches NACC requirements

MIT

29

Alternative FHR Designs Can Be Coupled to NACC

Base-line UCB/MIT/UW in Oval



MIT

Comparison of LWR and FHR

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

MIT

Commercial Fluoride-salt-cooled High-Temperature Reactor Design

U. of California—Berkeley
Per Peterson

UC BERKELEY
NUCLEAR
ENGINEERING
Thermal Hydraulics Laboratory

Overview of the UCB Mark-1 PB-FHR Commercial Prototype Preconceptual Design

FHR Test Reactor Workshop
Per F. Peterson
October 2, 2014

FHR

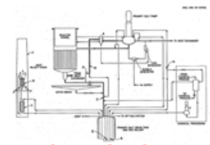
NEUP Nuclear Energy University Programs
U.S. Department of Energy

The UC Berkeley Mk1 PB-FHR design effort had 4 goals

- Demonstrate a plausible, self-consistent Nuclear Air Combined Cycle (NACC) system design
  - Believable predictions for base-load and peaking power levels using an industry-standard design code (Thermoflex)
    - » 2 archival articles now published in the *ASME Journal of Engineering for Gas Turbines and Power*
  - Self-consistent approach to heat air directly with primary coolant
- Provide detailed design for decay heat management systems
  - Provide basis for establishing CIET experiment test matrix
  - Enable TH code validation and benchmarking exercises

These goals expand the design space covered by earlier FHR studies

Aircraft Reactor Experiment, 1954



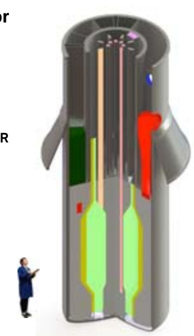
MSBR drain tank cooling system

The Mk1 PB-FHR design had 4 goals (con't)

- Develop a credible, detailed annular FHR pebble core design
  - Inner and outer graphite reflector including assembly method
  - Pebble injection and defueling
  - Coolant flow distribution and pressure loss calculations
  - Provide basis for future FHR code benchmarking
  - Neutronics/depletion/control-rod worth calculations are now documented in A.T. Cisneros doctoral dissertation
- Identify additional systems and develop notional reactor building arrangement
  - "Black-box" level of design for many of these systems
  - Include beryllium and tritium management strategies

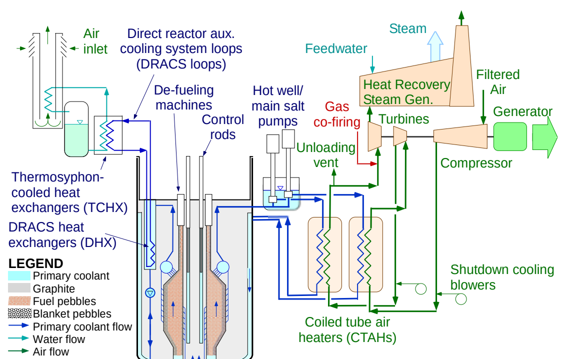
Nominal Mk1 PB-FHR Design Parameters

- Annular pebble bed core with center reflector
  - Core inlet/outlet temperatures 600°C/700°C
  - Control elements in channels in center reflector
  - Shutdown elements cruciform blades insert into pebble bed
- Reactor vessel 3.5-m OD, 12.0-m high
  - Vessel power density 3 x higher than S-PRISM & PBMR
- Power level: 236 MWth, 100 MWe (base load), 242 MWe (peak w/ gas co-fire)
- Power conversion: GE 7FB gas turbine w/ 3-pressure HRSG
- Air heaters: Two 3.5-m OD, 10.0-m high CTAHs, direct heating
- Tritium control and recovery
  - Recovery: Absorption in fuel and blanket pebbles
  - Control: Kanthal coating on air side of CTAHs



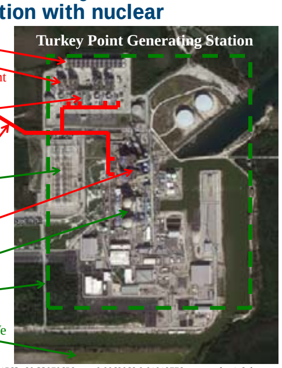
PB-FHR cross section

Mk1 PB-FHR flow schematic



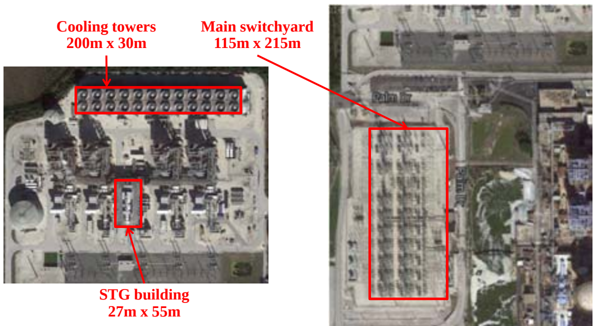
The Turkey Point Generating Station was used as a reference for the Mk1 site design: combines natural gas generation with nuclear

Cooling towers
Unit 5: 4-unit, 1150-MWe GE-7FA combined cycle plant
Steam-turbine generator building
0.60-m diameter, 5.2 MPa natural gas supply
Main switchyard
Units 1-2, 400 MWe gas/oil steam plants
Units 3-4, 886 MWe PWRs
Outline of the baseline 950m x 750 m 12-unit, Mk1 PB-FHR site
Proposed Units 6-7, 1150 MWe AP1000s, to the south



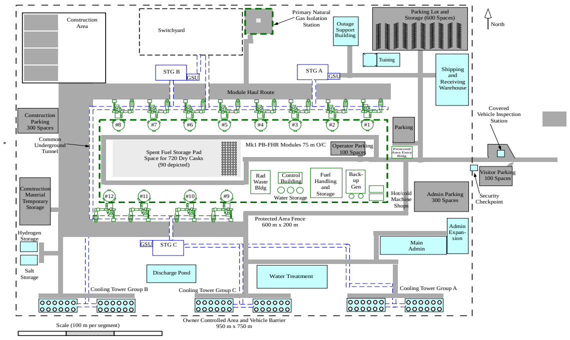
https://maps.google.com/maps?t=h&ll=25.4371563,-80.3307065&spn=0.0069068,0.0101977&output=classic&dg=opt

Key dimensions for Mk1 site were based upon the Turkey Point Generating Station



https://maps.google.com/maps?t=h&ll=25.4371563,-80.3307065&spn=0.006...

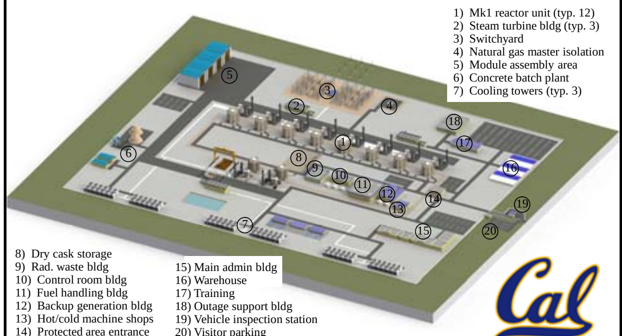
Notional 12-unit Mk1 station



- 1200 MWe base load, 2900 MWe peak station output

Notional 12-unit Mk1 station

1200 MWe base load; 2900 MWe peak



Mk1 Fuel and Materials

Mk1 PB-FHR average stresses are low

- **The Mk1 design reactor vessel and CTAH tubes have low stresses compared to ASME allowable values**
  - **The air pressures in the Brayton cycle (18.7 and 5 bar) are much lower than in a steam Rankine cycle (180 bar)**
  - **Large margins for BDBE creep rupture**

	Reactor vessel	HP CTAH tubes	LP CTAH tubes	S-PRISM reactor vessel†
Outside diameter (cm)	350.0	0.635	0.635	919.5
Wall thickness (cm)	6.0	0.0889	0.0889	5.0
Maximum pressure differential (bar)	2.50	16.72	2.95	1.41
Circumferential stress (MPa)	7.30	5.97	1.05	13.01
Axial stress (MPa)	3.71	3.47	0.61	6.54
Von Mises stress (MPa)	6.32	5.19	0.91	11.26
Von Mises stress (ksi)	0.92	0.75	0.13	1.63
Nominal operating temperature (°C)	600	700	700	355
ASME allowable stress for 316 SS for 100,000 hr (MPa)	60.0	23.00	23.00	110.00
Ratio of allowable to actual stress	9.49	4.43	25.19	9.77

† Pressure differential for S-PRISM based upon sodium hydrostatic head of 16.7 m

- **Key caution: localized thermal stresses may be large and must be evaluated carefully**

Insufficient information exists to select a final structural alloy

- **Mk1 selected 316 SS as reference alloy**
  - **Used 316 SS properties (thermal expansion coefficient, thermal conductivity, density) in design calculations**
  - **304 SS, Alloy N, and advanced ORNL alloy remain options; final selection not currently justified or possible**
- **The large experience base with 304 and 316 SS make them interesting candidates but issues for corrosion/sigma-phase must be resolved**

Application experience at FHR temperatures	304 > 316 >>> Alloy N
Cost	304 < 316 << Alloy N
Corrosion in flibe with redox control	Alloy N >> 304 = 316
Allowable stress	Alloy N > 316 > 304
Sigma phase formation	Alloy N << 304 < 316
Neutron irradiation damage	304 > 316 >> Alloy N
ASME Section III code status	316 = 304 >> Alloy N
Thermal conductivity	Alloy N > 316 = 304
Thermal expansion coefficient	316 < 304 < Alloy N

FHRs have remarkably small Cs-137 inventories (key isotope for long-term land use restrictions)

Implies unique safety characteristics

	Mk1 PB-FHR	ORNL 2012 AHTR	Westing-house 4-loop PWR	PBMR	S-PRISM
Reactor thermal power (MWt)	236	3400	3411	400	1000
Reactor electrical power (MWe)	100	1530	1092	175	380
Fuel enrichment †	19.90%	9.00%	4.50%	9.60%	8.93%
Fuel discharge burn up (MWt-d/kg)	180	71	48	92	106
Fuel full-power residence time in core (yr)	1.38	1.00	3.15	2.50	7.59
Power conversion efficiency	42.4%	45.0%	32.0%	43.8%	38.0%
Core power density (MWt/m3)	22.7	12.9	105.2	4.8	321.1
Fuel average surface heat flux (MWt/m2)	0.189	0.285	0.637	0.080	1.13
Reactor vessel diameter (m)	3.5	10.5	6.0	6.2	9.0
Reactor vessel height (m)	12.0	19.1	13.6	24.0	20.0
Reactor vessel specific power (MWe/m3)	0.866	0.925	2.839	0.242	0.299
Start-up fissile inventory (kg-U235/MWe) ††	0.79	0.62	2.02	1.30	6.15
EOC Cs-137 inventory in core (g/MWe) *	30.8	26.1	104.8	53.8	269.5
EOC Cs-137 inventory in core (Ci/MWe) *	2672	2260	9083	4667	23359
Spent fuel dry storage density (MWe-d/m3)	4855	2120	15413	1922	-
Natural uranium (MWe-d/kg-NU) **	1.56	1.47	1.46	1.73	-
Separative work (MWe-d/kg-SWU) **	1.98	2.08	2.43	2.42	-

† For S-PRISM, effective enrichment is the Beginning of Cycle weight fraction of fissile Pu in fuel

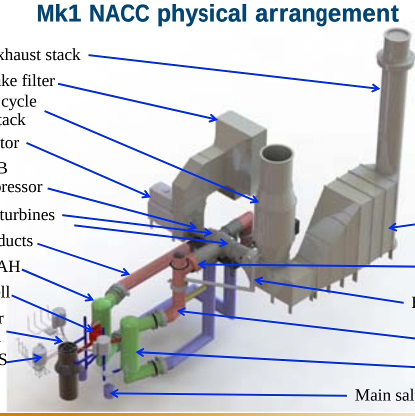
†† Assume start-up U-235 enrichment is 60% of equilibrium enrichment; for S-PRISM startup uses fissile Pu

* End of Cycle (EOC) life value (fixed fuel) or equilibrium value (pebble fuel)

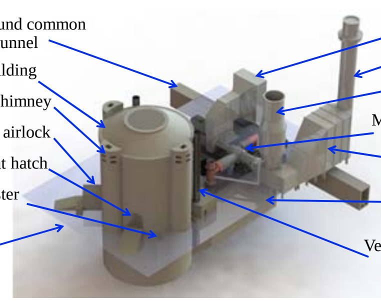
** Assumes a uranium tails assay of 0.003.

Mk1 Systems

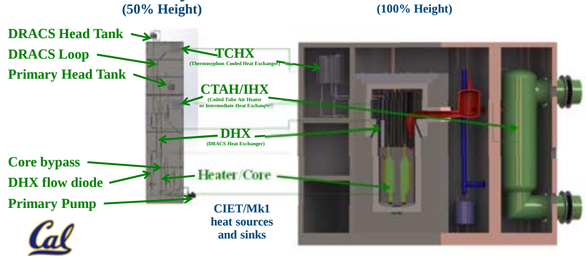
Mk1 NACC physical arrangement



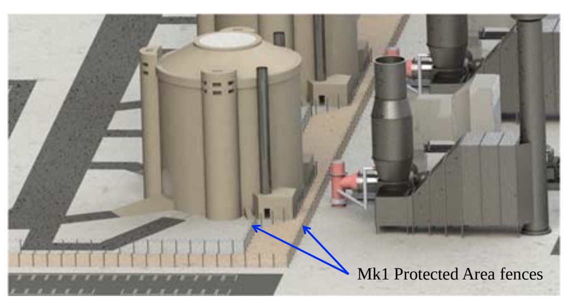
The Mk1 structures are designed for modular construction



The UCB Compact Integral Effects Test (CIET) facility scaling matches the Mk1 reactor design



The Mk1 design places the shield building inside the protected area



The Mk1 uses the same steel-plate composite modular construction as AP1000


Vogtle Unit 3 shield building wall panels, May 2014

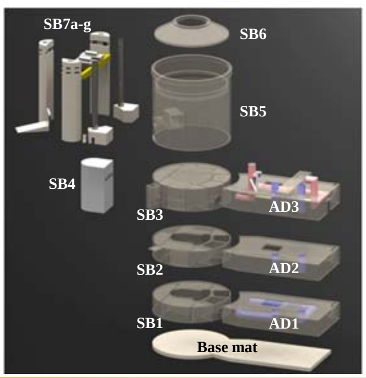

CA20 being set in place by heavy crane


Summer Unit 2 CA20 Transported from MAB

The Mk1 SolidWorks CAD model provides estimates for commodity inputs

- Structural steel
  - Total Mk1 structural steel estimated to be 39 t/MWe
  - 1970's era PWRs used a total of 36 t/MWe
- Concrete
  - Total Mk1 concrete 160.0 m^3/MWe
  - 1970's era PWRs used a total of 81 m^3/MWe

The Mk1 design uses 10 primary structural modules



The Mk1 uses a lift tower for modular construction

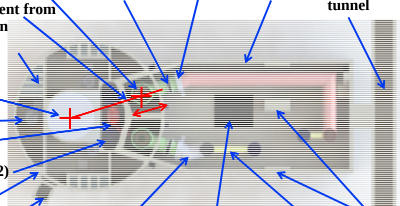

(A) Excavation occurs adjacent to an existing Mk1 unit, with protected area fence rerouted


(B) Basemat is poured after common tunnel and lift-tower have been installed

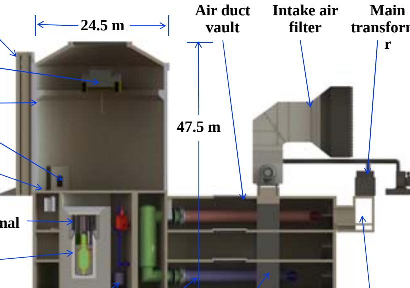

(C) Below-grade structures installed as six structural modules and reactor cavity module


(D) After above-grade modules are installed, fence is rerouted before loading fuel

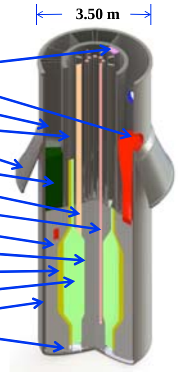
Plan view of reactor, main salt piping, and CTAH systems in cylindrical shield building



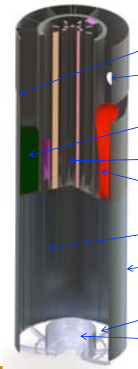
Mk1 Reactor Building Elevation View



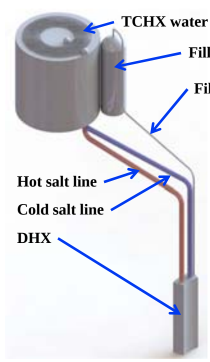
Mk1 Reactor Cross Section



The Mk1 core metallic internals are installed as a single module

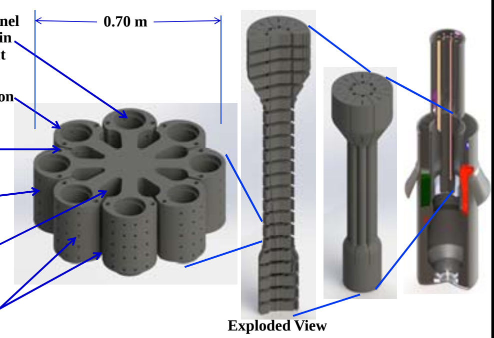


The Mk1 DRACS is modular and uses a TCHX



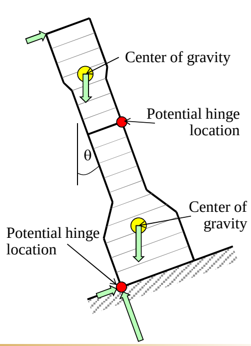
Each TCHX transfers heat from sloping, coiled salt tubes to vertical water cooled thermosyphon pipes

The Mark-1 center reflector block geometry minimizes stresses induced by neutron irradiation

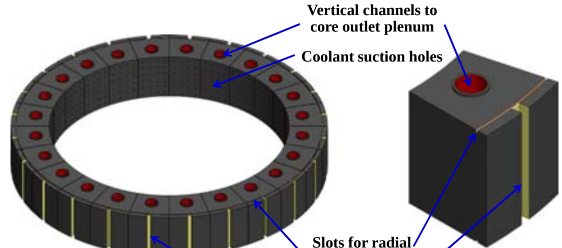


A simplified push-over model for seismic loading treats the center column joints like hinged joints

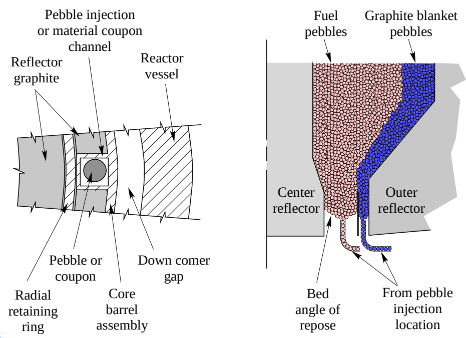
- Effect of horizontal seismic acceleration can be approximated using a static push-over force
  - Net buoyancy force on blocks is upward
- Because ends of column are restrained from horizontal motion, failure occurs by "hinging" at joints between blocks
- Important to design key structures between blocks to carry hinging loads



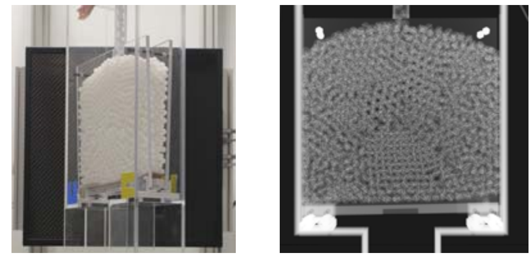
The Mk1 outer reflector uses radial and axial retaining rings similar to the MSBR



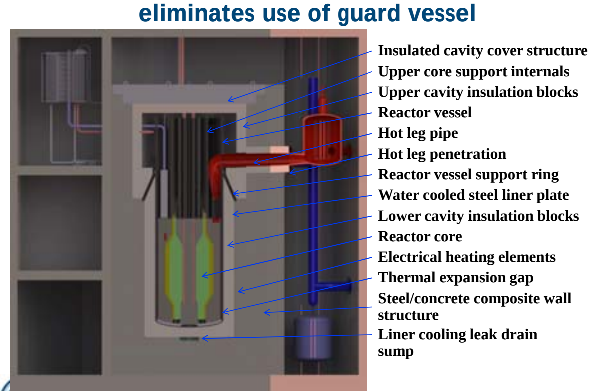
Mk1 pebble injection feeds pebbles to the bottom of the core at a controlled rate



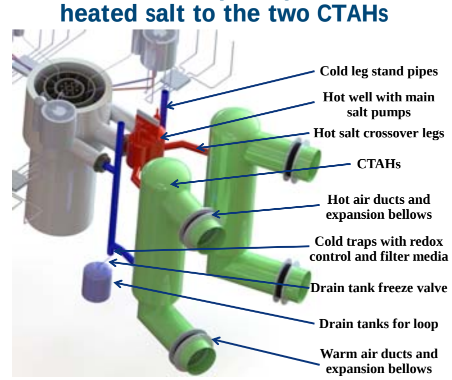
Preliminary X-PREX experiments show pebble heap will have a shallow angle



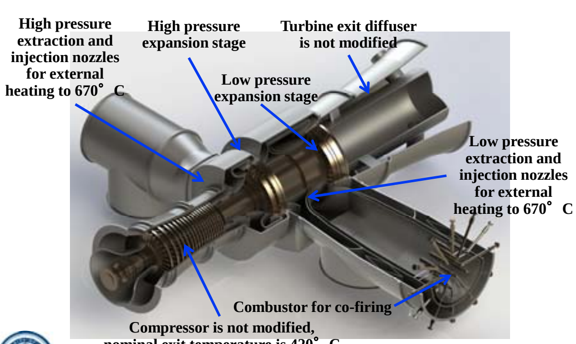
Mk1 Refractory Reactor Cavity Liner System eliminates use of guard vessel



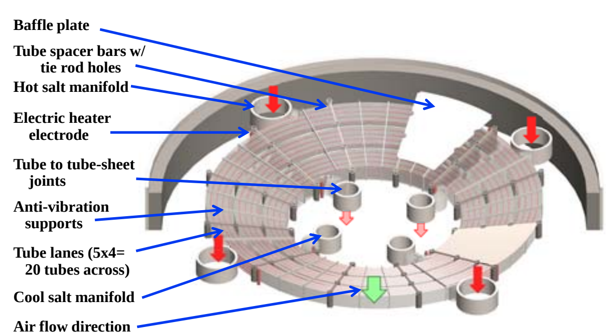
The Mk1 heat transport system delivers heated salt to the two CTAHs



The GE 7FB turbine design has been modified to implement nuclear heating



Mk1 CTAHs have 36 annular sub-bundles



Mk1 CTAH Tube Sub-bundle Model

Questions?

NEUP Integrated Research Project Workshop 6: Fluoride Salt-Cooled High Temperature Reactor (FHR) Test Reactor Goals, Designs, and Strategies

Session One:

FHR Commercial Space & Implications for Test Reactor Design

October 2nd, 2014

Facilitators: Ruaridh Macdonald (MIT)
Daniel Curtis (MIT)

U.S. Department of Energy

Session Goal & Motivation

Derive test reactor goals and requirements from consideration of potential applications and owners

Test reactor (FHTR) intended as a long term resource

- Could be a platform for a focused or broad range of technologies
- Currently undetermined reactor owner will decide the test reactor mission

How does the potential for multi-missions define test reactor requirements and ownership?

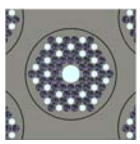
- Owner will prioritize studying technologies for applications they value
- What applications are most interesting and relevant today and in the future?
- Which technologies are common to multiple applications?
- How does this assessment change under different owners or organization and partner structures?

Is it possible to narrow the FHR & FHTR design space at this time?

Application-Neutral Capabilities for FHTR

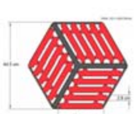
Some testing capabilities are fundamentally required in the FHTR

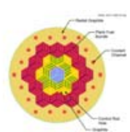
- System performance—can it be operated and maintained
- High temperature fuel
- Fuel cladding materials
- Salt chemistry control
- Tritium control
- Salt freezing in accident / shut down scenarios
- Core reactivity control: startup, steady state, shut down
- Safety / emergency systems
- Material performance under salt and radiation



What else can be done and how are these effected by owner preference?

- Materials and design affect some of the above but are decided by preferred end use





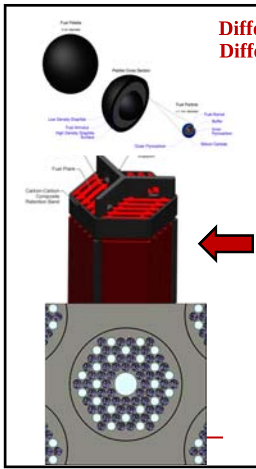
Application-Based Design Methodology

1) What social, economic, or technical **demand** does the system meet?
   How competitive and large is the market?
2) What **technical capabilities and features** are required to serve that application?
3) What **technologies** must we develop to provide those capabilities?
4) **What must be proven** in the FHTR?

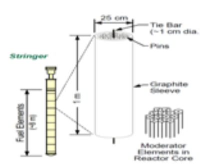
Example applications	
Grid power generation	Actinide Transmutation
Industrial process heat generation	Radionuclide production
Remote site power generation	R&D for other molten salt concepts, e.g. MSR—or HTGRs

---

Different Applications May Lead to Different Fuel and Coolant Choices



Graphite-Matrix Coated-Particle Fuel Forms



Pin SiC Clad UO_2

SiC Matrix Coated-Particle

Test Reactor Could Be Testing Different Fuels/Core Designs/Salt Coolants for Different Purposes

---

Application: Power and Industrial Heat Generation

1) Electricity

- Well established market
- Standardized
- Many suppliers

Heat

- Custom systems
- Large potential markets
- No single technology dominates
- No market presence by reactors (yet!)

2) Technical capabilities:

- Flexible electricity
- High temperature output

3) Technologies:

- Nuclear Air-Brayton Combined Cycle

4) Testing:

- High burnup testing
- Interfacing with power cycle

Application: Remote Power and Propulsion

1) Remote sites currently pay up to x6 grid prices for electricity and heat
- Remote sites include: Antarctic bases, military bases, Alaskan communities, large vessels
- The isolated site cannot support traditional security guns, gates & guards

2) Technical capabilities:
- Flexible electricity
- High temperature output
- Proliferation resistance
- Low manpower security

3) Technologies:
- Nuclear Air-Brayton Combined Cycle
- Integral security
- Proliferation resistant, high temperature fuel (SiC matrix?)

4) Testing:
- SiC matrix fuel testing – larger P/D
- Supplementary chemical and destructive testing of fuel

Application: Actinide Transmutation & Radionuclide Production

1) Nuclear waste concerns has renewed interest in closed fuel cycle
- AREVA and others have investigated using salt-cooled reactors in place of SFRs—more grams Pu burnt per MWt
- Epithermal / thermal FHR performance affected less by actinides in fuel

Radionuclide production is decreasing in many countries
- Minimal neutron cross section of fuel allows flux to be tailored

2) Technical capabilities:
- Flux zones tailored to transmuting different actinides

3) Technologies:
- Flux traps in core
- Driver fuels

4) Testing:
- Cladding and materials testing (pin type fuels?)
- Flexible testing positions

Application: General R&D Support

1) Other reactor tech. may be able to test technologies in FHTR
- HTGR
- MSR technology

2) & 3) Capabilities:
- Dissolved fuel loop
- Flexible fuel / test positions: plate, pin, pebble, prism

Session 2: Test Reactor Goals

Joshua Richard and John Stempien

Massachusetts Institute of Technology FHR

Classification of Test Reactors

Massachusetts Institute of Technology FHR

General Classes of Test Reactors

- Two classes, distinguished by intended mission:

Class I
Develop and demonstrate a new reactor concept

Class II
Serve as an irradiation machine

Massachusetts Institute of Technology FHR

Class I Sub-classes

- Two sub-classes, distinguished by scope of mission:

Class I (a)
Flexible test reactor to allow evaluation of different fuel, coolant salt, and system options

Class I (b)
Limited-capability test reactor to provide data for licensing a specific pre-commercial reactor type

Massachusetts Institute of Technology FHR

Owner of Test Reactor Defines Goals Missions and Constraints

Massachusetts Institute of Technology FHR

Ownership options—1

US Government Owned

- US government fully funds the test reactor development program and owns the facility
- Similar to US strategy in 1950s and 1960s
- Current budget environment makes this difficult
- Budget environment may change in the future

Massachusetts Institute of Technology FHR

Ownership options—2

US-CAS Joint Program

- Collaborate with Chinese TMSR-SF program
- Several paths forward:
  - US Test Reactor: data from TMSR-SF1 enables U.S. build
    - Potentially reduced costs and lower risks with increased experience
  - US Precommercial Power Reactor: TMSR-SF1 becomes stepping stone to demonstration FHR in US
    - Would favor a particular FHR design

Ownership options—3

US-led International FHTR

- Multinational cooperative research project with US as lead country
- Similar to historical projects: DRAGON HTGR project in UK, LOFT in US
  - More recently JHR, MYRRHA
- Successful when:
  1. Potentially large benefits to all participants
  2. Strong leadership by at least one partner
  3. Early technology development that limits issues associated with intellectual property

Ownership options—4

Public-private partnership with domestic and foreign partners

- Include both national programs and private companies
- Potential national partners include Japan and China due to their HTGR program experience
- Potential commercial partners could include vendors for natural-gas combined cycle plants
- Private investment unlikely due to long lead times from test reactor to commercial product

Ownership drives class selection

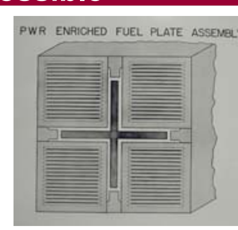
- **The ownership structure will determine the project goals, which will drive the selection of the test reactor class**
- **Propose that the most credible option at this time for the U.S. is some kind of international partnership**
- **International partnership suggests may want flexibility in design characterization and testing (as with DRAGON)**
- **A Class I (a) test reactor can satisfy the goals of many involved partners**

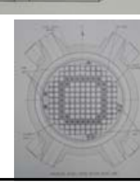
Some Test Reactors Have Been Used to Explore Reactor Design Space

Example: Shippingport-I

Test Reactors with Multiple Capabilities are Possible

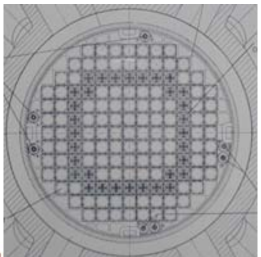
- **Shippingport**
- **First PWR designed to generate electricity**
- **Tested multiple core designs, power levels**
  - Plate core
  - Pin core
  - Thorium core

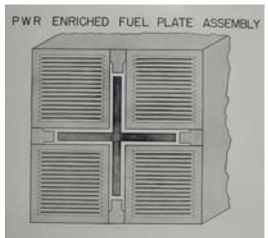




Test Reactors with Multiple Capabilities are Possible-I

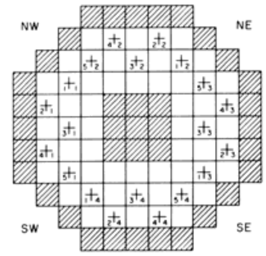
- **Shippingport Core 1, 60 MWe**
  - Plate-type fuel
  - Seed-and blanket core





Test Reactors with Multiple Capabilities are Possible-2

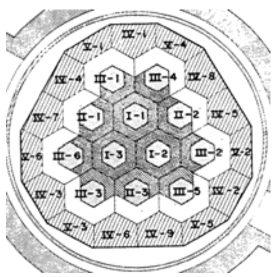
- **Shippingport Core 2, 100 MWe**
  - Pin-type fuel
  - Seed-and-blanket core
  - **Fewer assemblies than in Core I**
  - **Different core layout**



MIT Massachusetts Institute of Technology FHR

Test Reactors with Multiple Capabilities are Possible-3

- **Shippingport Core 3,**
  - Approx 60 MWe
  - ThO_2-$U^{233}O_2$ pin-type fuel
  - Seed-and-blanket core for breeding
  - Radically different core geometry
  - Movable seed assemblies!



MIT Massachusetts Institute of Technology FHR

MIT Test Reactor Goals

MIT Massachusetts Institute of Technology FHR

Goals for FHTR

- **MIT proposes the FHTR will be a Class I (a) (general-purpose) high temperature salt-cooled test reactor capable of testing:**
  - Fuel types – pebble, compact, plate, pin
  - Alternative Core configurations
  - Salt coolants – LiF-BeF_2 (flibe), NaF-ZrF_4, NaF-BeF_2, etc.
  - Materials – metals, graphite, composites (C-C and SiC-SiC)
  - Tritium handling
  - Spent fuel handling
  - Coolant chemistry control
  - Instrumentation and control
- **Focus on feasibility and flexibility not on matching geometry or precise conditions of commercial FHR**
- **<u>Not</u> trying to be prototypical of commercial design**

MIT Massachusetts Institute of Technology FHR

Required Technical Characteristics

- **For materials testing**
  - Thermal flux traps: for fuel
  - Fast flux traps: for materials irradiations
  - Reasonable axial and radial flux shapes
  - One or two large irradiation positions for experiments (ATR's largest is 4' long, 5" diameter
  - Rabbit for short irradiations
- **For coolant testing**
  - Piping and structures able to support variable coolant weight
  - Ability for variable flow rates based on coolant physical properties
  - Piping compatibility with range of potential coolants

MIT Massachusetts Institute of Technology FHR

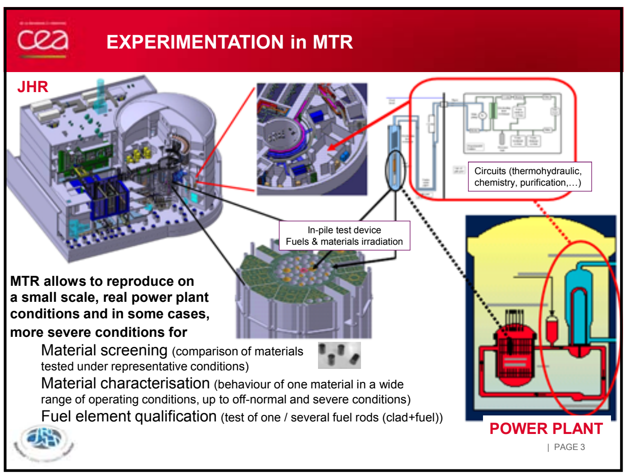
Questions

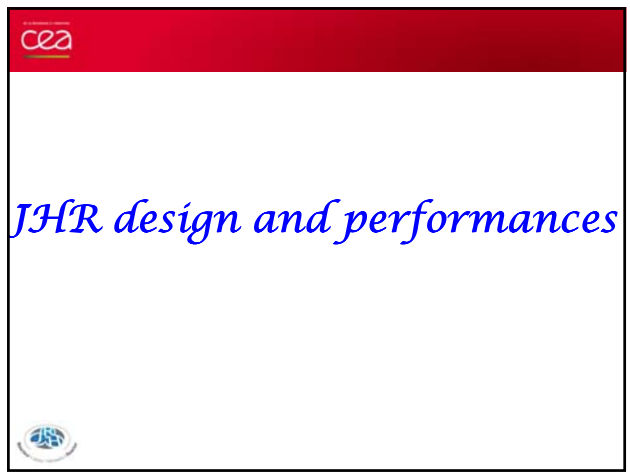
Many Questions

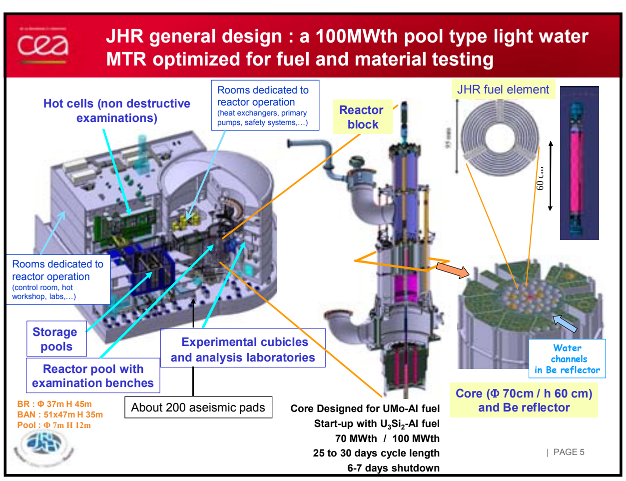
- What ownership options should be considered for a U.S. FHTR?
- Should the U.S. build a Class I (a) general purpose or a Class 1 (b) pre-commercial FHTR
- Is it credible for a Class I test reactor (testing concept) to have credible Class II (general purpose materials testing) capabilities beyond testing other high-temperature reactor components and fuels?
- Should one design with expected short reactor lifetimes?

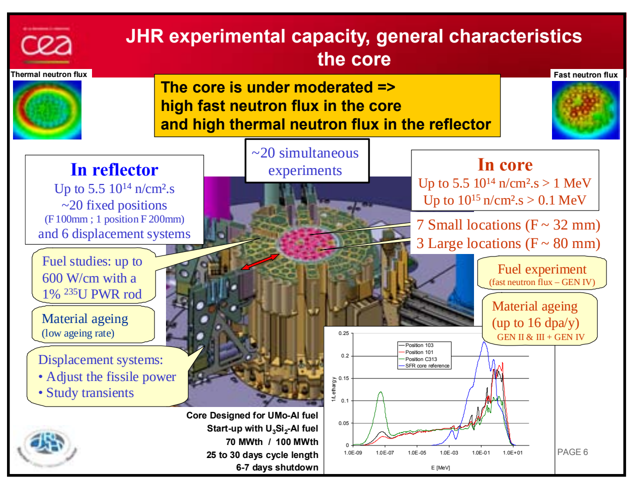

DE LA RECHERCHE À L'INDUSTRIE
cea
The Jules Horowitz research reactor
A new high performance MTR* in Europe
*MTR: Material Testing Reactor
Short Version of Original Talk
CEA/Nuclear Energy Directorate ; Ch Gonnier
October 1-3, 2014
FHR workshop - MIT
www.cea.fr


JHR 3 MAIN OBJECTIVES
R&D in support to nuclear Industry
Safety and Plant life time management (ageing & new plants)
Fuel behavior validation in incidental and accidental situation
Assess innovations and related safety for future NPPs (either LWR or GENIV concepts)
Radio-isotopes supply for medical application
MOLI production
JHR will supply 25% of the European demand (today about 8 millions protocols/year) and up to 50% upon specific request
JHR will be a key tool to support expertise
Training of new generations
Maintaining a national expertise staff and credibility for public acceptance
Assessing safety requirements evolution and international regulation harmonisation

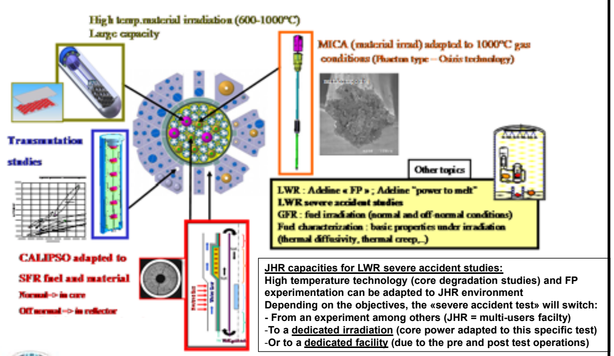

EXPERIMENTATION in MTR
JHR
Circuits (thermohydraulic, chemistry, purification,…)
In-pile test device
Fuels & materials irradiation
MTR allows to reproduce on a small scale, real power plant conditions and in some cases, more severe conditions for
Material screening (comparison of materials tested under representative conditions)
Material characterisation (behaviour of one material in a wide range of operating conditions, up to off-normal and severe conditions)
Fuel element qualification (test of one / several fuel rods (clad+fuel))
POWER PLANT
| PAGE 3


JHR design and performances


JHR general design : a 100MWth pool type light water MTR optimized for fuel and material testing
Hot cells (non destructive examinations)
Rooms dedicated to reactor operation (heat exchangers, primary pumps, safety systems,…)
Reactor block
JHR fuel element
60 cm
Rooms dedicated to reactor operation (control room, hot workshop, labs,…)
Storage pools
Experimental cubicles and analysis laboratories
Reactor pool with examination benches
Water channels in Be reflector
BR : Φ 37m H 45m
BAN : 51x47m H 35m
Pool : Φ 7m H 12m
About 200 aseismic pads
Core Designed for UMo-Al fuel
Start-up with U3Si2-Al fuel
70 MWth / 100 MWth
25 to 30 days cycle length
6-7 days shutdown
Core (Φ 70cm / h 60 cm) and Be reflector
| PAGE 5


JHR experimental capacity, general characteristics the core
Thermal neutron flux
Fast neutron flux
The core is under moderated => high fast neutron flux in the core and high thermal neutron flux in the reflector
~20 simultaneous experiments
In reflector
Up to 5.5 10^14 n/cm².s
~20 fixed positions
(F 100mm ; 1 position F 200mm)
and 6 displacement systems
In core
Up to 5.5 10^14 n/cm².s > 1 MeV
Up to 10^15 n/cm².s > 0.1 MeV
7 Small locations (F ~ 32 mm)
3 Large locations (F ~ 80 mm)
Fuel studies: up to 600 W/cm with a 1% 235U PWR rod
Fuel experiment (fast neutron flux – GEN IV)
Material ageing (low ageing rate)
Material ageing (up to 16 dpa/y) GEN II & III + GEN IV
Displacement systems:
• Adjust the fissile power
• Study transients
Core Designed for UMo-Al fuel
Start-up with U3Si2-Al fuel
70 MWth / 100 MWth
25 to 30 days cycle length
6-7 days shutdown
PAGE 6

Hosting experimental systems under conceptual design



JHR capacities for LWR severe accident studies:
High temperature technology (core degradation studies) and FP experimentation can be adapted to JHR environment
Depending on the objectives, the «severe accident test» will switch:
- From an experiment among others (JHR = multi-users facilty)
-To a dedicated irradiation (core power adapted to this specific test)
-Or to a dedicated facility (due to the pre and post test operations)

Status of JHR project

April 2014

JHR OPERATING RULES

JHR CONSORTIUM & GOVERNING BOARD

19/03/2007 Signature of the JHR consortium

JHR consortium gathers organizations which take part financially in the construction of JHR (1 representative / organization)

JHR Consortium current partnership: Research centers & Industrial companies

VATTENFALL, AREVA, EDF, Ciemat, SCK•CEN, cea, VTT, UJV, IAEC

Associated Partnership IAEA

NNL is the UK representative to JHR
UK/CEA agreement – March 2013

In some cases, the organization (member of the JHR consortium) is itself the representative of a domestic consortium which gathers organizations among industry, R&D organizations, TSO, or safety authority

JHR CONSORTIUM & GOVERNING BOARD

JHR Consortium, economical model for investment & operation

- JHR Members have financial commitment to the JHR construction (minimum contribution 2% of JHR initial cost)
- JHR members - are owner of Guaranteed Access Rights for JHR lifetime
  - have a voting right in the Governing Board

  in proportion of their financial commitment to the construction

  Example, a Member with 2% contribution of the construction cost

  2% Participation (/500M€$_{2005}$) = 10M€$_{2005}$ Contribution (15 M€ $_{EC2014}$)
  = 2% guaranteed access to JHR experimental capability and
  = 2% voting rights in the JHR Consortium
- CEA is the Reactor Owner & the Nuclear Operator with all liabilities

Access Rights of JHR Experiments

Access right definition and its corresponding cost is under discussion (at the Governing Board)

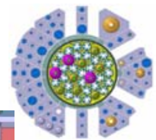
They will depend on:

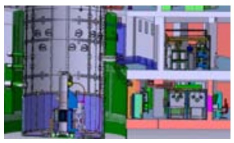
- the « neutron flux » (duration of irradiation + location in the core)
- the « equipment complexity » (need of a displacement system, of a cubicle,…)
- the utilities (water, electricity, gases,…)

But also on:

- operation complexity
- the services (NDE, FP lab, hot cells,
- waste managements, …)

+ ……





JHR CONSORTIUM & GOVERNING BOARD

JHR Consortium, economical model for investment & operation

- A Member can **use totally or partly** his access rights
  - For implementing **proprietary programs** with full property of results for his own business
  - And/or for participating to **Joint International Programs** (results are open to the participants)
- Access rights can be cumulated to some extend from one year to the following (2% → 5%max)
- Access rights can be cumulated with other Members to implement proprietary shared programs
- Reactor Operating Cost are paid only if the access rights are used.

JHR : an International Users Facility working as an International Center of Excellence

Project leader appointment
Validation of operation plan, business strategy, economy of the project

Governing Board
(JHR Consortium Members)

Nuclear safety ; Technical and Economical performance (operation cost)

CEA
(Nuclear Operator)

Operation plan fulfilment
programs definition (preparation of next Operation Plan with users)

Project leader

JHR Reference Operation Plan (4 years plan)
For Members of the Consortium and Non-Members *

Proprietary Programs & Joint international Programs

* Non-Member access under decision & commercial policy of the Governing Board

JHR, a Users-Facility
Rights & Duties for JHR Consortium Members

Reference Operation Plan (ROP)

- 4 year period plan built by the project leader and validated by the GB (yearly update)
- ROP: Loading plan (Proprietary programs and International programs) and provisionnal budget
  - Guidelines are defined by the GB (strategy for the ratio « proprietary programs / joint programs » and for the access to JHR capacity for non-members)
  - Each member declares his needs of Access Rights (for proprietary programs and for international joint programs)
  - The project leader plans a share of the access rights first between members, second between non-members

Present organization

1] **a yearly Governing Board meeting**

2] **a yearly experimental seminar alternatively** :
- one year restricted to Consortium Members
- one year partly open to dedicated non-Members

3] recently: **setting-up of 3 Working Groups**
fuel issue, material issue, and technology issue with participants of all Consortium members for
- the preparation of future R&D programs in JHR
- the development of new technologies / test devices

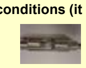
CONCLUSION…

Experiment in MTRs will remain essential in the next decades in support for simulation and in support for the development and the qualification of nuclear fuel and structural materials

JHR, will be a major facility for the European and International community, considering its hosting capacities and its high characteristics about neutron fluxes and spectra

Its experimental capacity is innovating and with a large range of operating conditions (test loops, FP laboratory, NDE equipment,..)

The on-line measurements are essential for the control of the experimental conditions (it guaranties the quality of the experiments).



The complementarity of JHR with the hot labs dedicated to fuel and material examinations (LECA – LECI) will make possible to propose, to the scientific international community, a high level of expertise and competences in the fields of simulation, in-pile and out-of-pile experiments, and sample characterizations.

Full operation is expected in 2020

Thank you for your attention

FHTR Design Overview

Joshua Richard, Ben Forget, Charles Forsberg, Kord Smith

Massachusetts Institute of Technology

FHR

MIT Test Reactor Goals

Massachusetts Institute of Technology

FHR

FHR and FHTR Strategic Goals

- **Commercial FHR Goals:**
  1. Enhanced revenue via multi-mission design
  2. Limited severe accident consequences
  3. Improved nonproliferation and waste characteristics
- **Three overarching goals to be accomplished with a FHTR:**
  1. Develop the safety and licensing basis for a commercial FHR
  2. Demonstrate that a FHR can be operated reliably
  3. Test structures, fuels, and coolants in an actual reactor configuration

Massachusetts Institute of Technology

FHR

MIT FHTR Project Goals

- **Design a Class I (a) <u>general-purpose</u> high temperature salt cooled test reactor capable of testing multiple fuel and coolant types**
  - Focus on design feasibility and flexibility, not on matching geometry or precise conditions of commercial FHR
  - <u>Not</u> trying to be prototypical of commercial design

Massachusetts Institute of Technology

FHR

FHTR Design Constraints

- **Neutronic constraints:**
  - 6 month cycle length, 20% U-235 enrichment, TRISO packing fractions ≤ 0.35, negative reactivity coefficients
- **Thermal-hydraulic constraints:**
  - peak fuel temperature of 1250 C (normal ops) and 1600+ C (BDBA), capability to operate with multiple salt coolants (flibe or NaF-ZrF_4), reasonable limit on pumping power
- **Manufacturability constraints:**
  - Total core height <2 m, graphite reflectors

Massachusetts Institute of Technology

FHR

FHTR Major Design Decisions

- **Use NaF-ZrF_4 as the design salt, since has more limiting:**
  - Neutronics: Na and Zr parasitic absorption xs 1-2 orders of magnitude greater than Li and Be
  - Thermal hydraulics: heat capacity and thermal conductivity ½ that of flibe, density 50% greater
  - A reactor designed to operate with NaF-ZrF_4 can be easily modified to operate with flibe, but the reverse is not possible
- **Use prismatic block driver fuel in the first core:**
  - Currently be manufactured (Japan), prior use in HTGR (Fort St. Vrain, US)
  - Allows for fine three-dimensional control over fuel packing fraction, moderator/fuel ratio, and coolant volume fraction
  - Block stacking not an issue since FHTR core will have short height (<2 m)

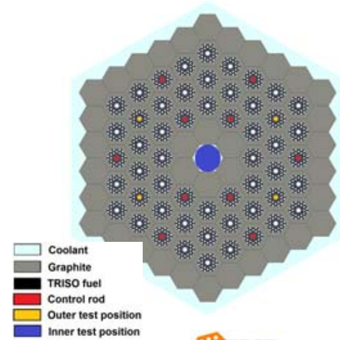
Massachusetts Institute of Technology

FHR

MIT Test Reactor Design Overview

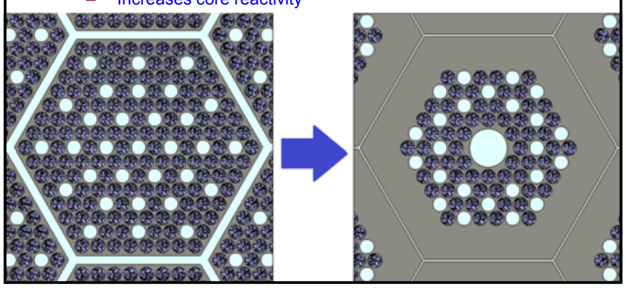
FHTR Preliminary Design

Power	20 MWth
Fuel form	UCO-kernel TRISO
Assembly type	Prismatic graphite block
Fuel enrichment	19.5 a% U-235
Primary coolant salt	NaF-ZrF_4 or LiF-BeF_2
Core outlet temperature	700 °C
Core inlet temperature	650 °C
Number of Fuel Assemblies	54
Assembly pitch	25 cm
Core diameter	2.75 m
Core height	1.75 m

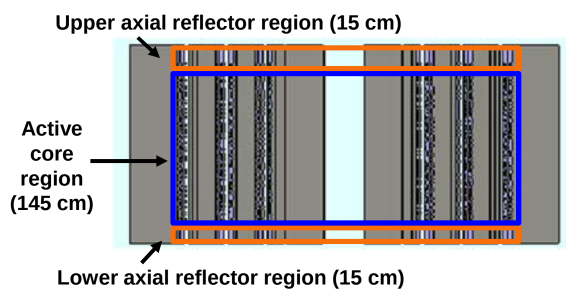


Fuel Inside Radial Moderator (FIRM) assembly design

- **Intra-assembly geometric heterogeneity:**
  - Spatial resonance self shielding
  - Enhances resonance escape probability
  - Increases core reactivity

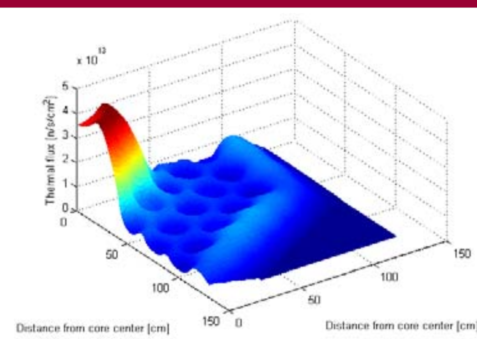


FHTR Preliminary Design



Irradiation fluxes and related considerations

FHTR Radial Thermal Flux Profile



Other Considerations

- **Facility design to allow alternative core designs and fuel types (like Shippingport) if replace entire core**
  - First core designed to minimize driver-fuel risks while providing testing of alternative FHR fuel types
  - Later cores could have alternative driver fuel or totally different core designs
- **High temperatures will constrain design**
- **Difficult to push up peak fluxes relative to other fuel / coolant systems—characteristic of graphite as moderator and reflector**
  - FHTR peak thermal flux ~10^{13}
  - ATR peak thermal flux ~10^{14}

Questions

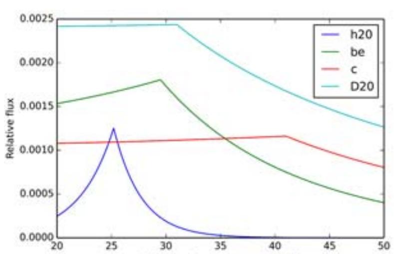
Questions

- **What is the required range of capabilities for an FHTR?**
- **What are the challenges and issues not addressed?**
- **What are other interesting core designs?**
- **How should core configuration evolve throughout FHTR lifetime?**

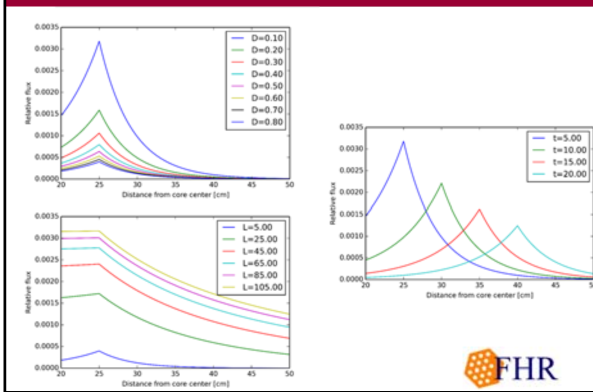
Additional Technical Information

Limits on FHTR Irradiation Flux

- **Maximum irradiation flux limited by use of graphite as primary moderator and reflector material**



Limits on FHTR Irradiation Flux



FHR and FHTR Strategic Goals

- **Commercial FHR Goals:**
  1. Enhanced revenue via multi-mission design
  2. Limited severe accident consequences
  3. Improved nonproliferation and waste characteristics
- **Three overarching goals to be accomplished with a FHTR:**
  1. Develop the safety and licensing basis for a commercial FHR
  2. Demonstrate that a FHR can be operated reliably
  3. Test structures, fuels, and coolants in an actual reactor configuration

MIT FHTR Project Goals

- **Design a Class I (a) <u>general-purpose</u> high temperature salt cooled test reactor capable of testing multiple fuel and coolant types**
  - Maximize thermal irradiation flux in a designated fuel testing position
  - Test multiple different fluoride salt coolants
  - Satisfy safety, feasibility, and operational constraints
  - Focus on design feasibility and flexibility, not on matching geometry or precise conditions of commercial FHR
    - <u>Not</u> trying to be prototypical of commercial design

Massachusetts Institute of Technology FHR

FHTR Design Constraints

- **Neutronic constraints:**
  - 6 month cycle length, 20% U-235 enrichment, TRISO packing fractions ≤ 0.35, fuel kernel size based on fission gas release, negative net power coefficient of reactivity, negative coolant temperature coefficient of reactivity, negative void worth
- **Thermal-hydraulic constraints:**
  - peak fuel temperature of 1250 C (normal ops) and 1600+ C (BDBA), capability to operate with multiple salt coolants (flibe or $NaF\text{-}ZrF_4$), reasonable limit on pumping power
- **Manufacturability constraints:**
  - Total core height <2 m, prismatic block driver fuel, graphite block reflectors

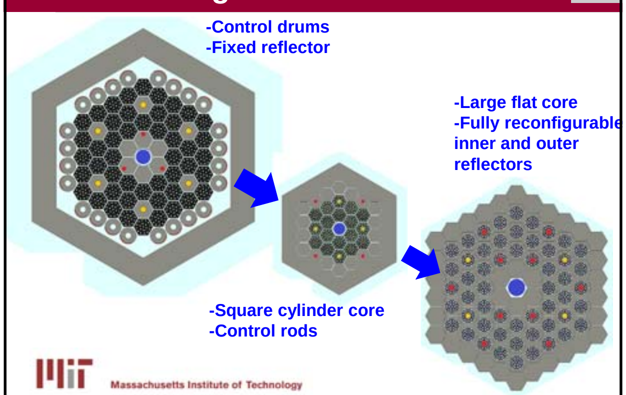
Massachusetts Institute of Technology FHR

FHTR Analysis methodology

- **Externally-coupled neutronic, thermal-hydraulic, and fuel temperature simulations**
  - <u>Serpent</u>: Continuous-energy Monte Carlo transport code for neutronic analysis
  - Simple purpose-built single channel analysis code for thermal-hydraulic analysis
  - <u>ABAQUS</u>: Finite-element analysis code for fuel temperature distribution calculation
- **Currently investigating the optimization approach**
  - Will likely be surrogate-based to reduce number of expensive neutronics calculations
  - Evaluation of existing algorithms will include both local (pattern/direct search) and global (SA/GA)

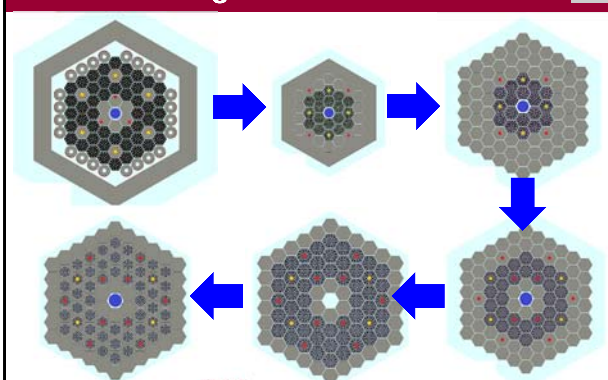
Massachusetts Institute of Technology FHR

FHTR Design Evolution



Massachusetts Institute of Technology

FHTR Design Evolution

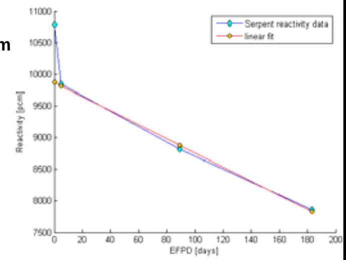


FHTR Irradiation Position Spectra

	BOC	EOC	% change
Central fuel irradiation position			
Fast flux [n/s/cm2]	2.70E+13	2.77E+13	2.56%
Thermal flux [n/s/cm2]	4.45E+13	4.53E+13	1.97%
Total flux [n/s/cm2]	7.15E+13	7.31E+13	2.19%
Fast/thermal flux ratio	0.608	0.611	0.58%
Outer materials irradiation position			
Fast flux [n/s/cm2]	2.44E+14	2.55E+14	4.35%
Thermal flux [n/s/cm2]	2.03E+13	2.01E+13	-1.29%
Total flux [n/s/cm2]	2.65E+14	2.75E+14	3.92%
Fast/thermal flux ratio	12.0	12.7	5.71%

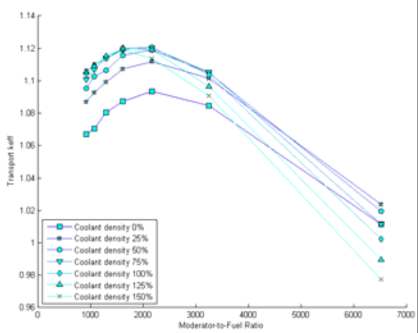
FHTR Depletion Analysis

- Initial reactivity = 10,750 pcm
- Xenon worth = 940 pcm (9% of initial reactivity)
- Linear reactivity fit predicts cycle length of 880 days (~30 months)
  - Can reduce enrichment



Reactivity Coefficient/Void Worth

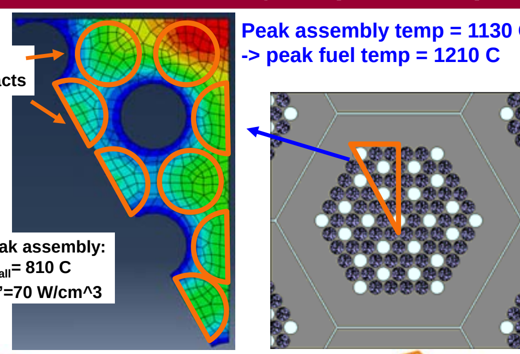
- Negative void worth for realistic pf (35%)
  - -3,100 pcm (full core)
- Negative coolant temp reactivity coefficient
  - -12 pcm/%void (-0.36 pcm/K)



FHTR Thermal Analysis (ABAQUS)

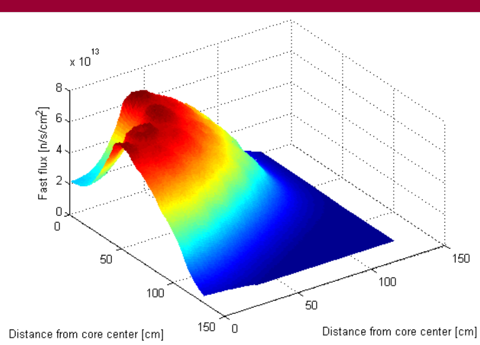
Fuel compacts

Peak assembly temp = 1130 C -> peak fuel temp = 1210 C



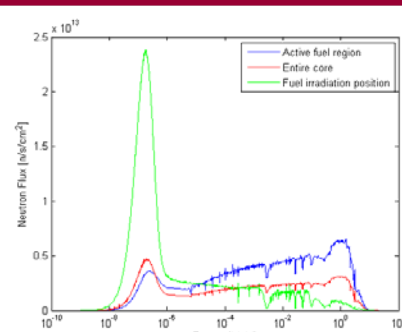
peak assembly:
T_{wall}= 810 C
q'''=70 W/cm^3

FHTR Radial Fast Flux Profile



- 2-group thermal cutoff = 0.625 eV

FHTR Flux Spectra



Conclusions and Future Work

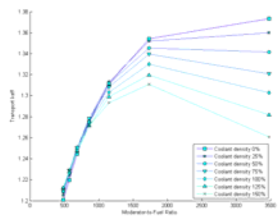
- **The nontraditional design goals of a test reactor drive design in a different direction that conventional power reactors**
  - Fuel utilization/burnup less important than irradiation flux, design flexibility, first-of-a-kind feasibility
- **A preliminary FHTR design has been developed to meet these irradiation flux goals while satisfying constraints**
  - FIRM assembly separates fuel and moderator regions to increase core reactivity and achieve the desired cycle length

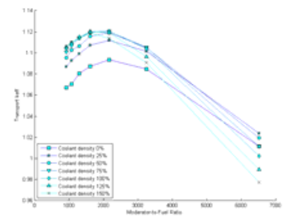
---

- **Develop a formalized optimization framework and use it to obtain a more finalized design**
- **Analyze the proliferation attractiveness of the spent fuel throughout burnup**
- **Perform limited system-level safety analysis**

K_{eff} Sensitivity to Mod/Fuel Ratio

Assembly model **Fullcore model**

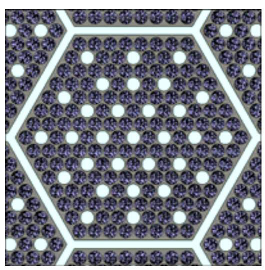




- **16,000 pcm more reactivity for fullcore with PF=35%**

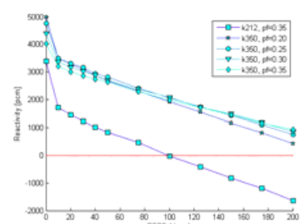
FHTR Original Assembly Design

- **Original design based off that of prismatic HTGR**
  - Fuel pins and coolant channels distributed throughout assembly
  - Coolant flow in interassembly gaps
- **However, core was too undermoderated with this configuration**



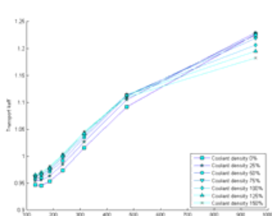
K_{eff} /Burnup Sensitivity to Fuel Kernel Size

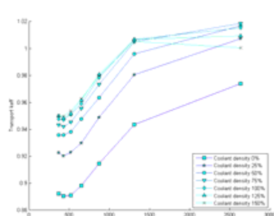
- **Increasing kernel size increases fissile loading of core (keeping enrichment constant)**
  - Cycle length increase of 168% when increasing from kernel radius of 212.5 μm to 350 μm
- **Kernel sizing constrained by need to keep SiC in compression**
  - Tensile forces on SiC generated by fission gas production



K_{eff} Sensitivity to Mod/Fuel Ratio

Assembly model **Fullcore model**

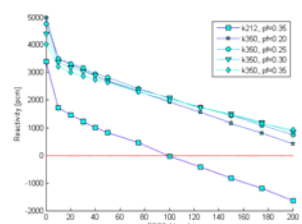




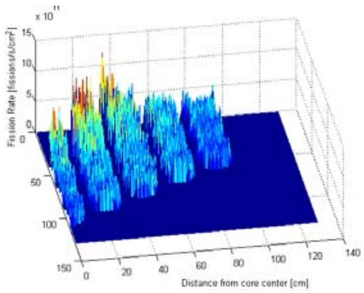
- **Core needs additional moderation, but cannot decrease PF below 5%**
- **Increasing leakage in full core at low PF begins to offset reactivity gains from increased moderation**

K_{eff} /Burnup Sensitivity to Fuel Kernel Size

- **Increasing kernel size increases fissile loading of core (keeping enrichment constant)**
  - Cycle length increase of 168% when increasing from kernel radius of 212.5 μm to 350 μm
- **Kernel sizing constrained by need to keep SiC in compression**
  - Tensile forces on SiC generated by fission gas production

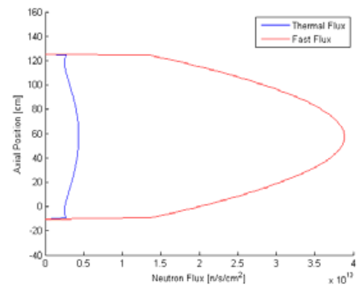


FHTR Radial Fission Rate Profile

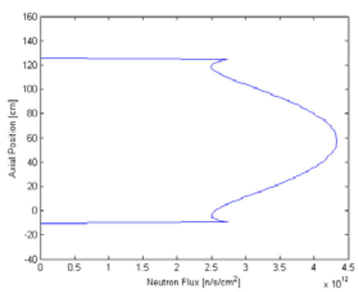


- Max pointwise peaking factor = 5.0

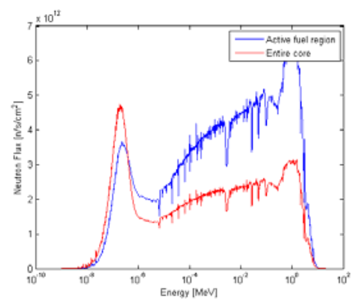
FHTR Axial 2-Group Flux Profiles



FHTR Axial Thermal Flux Profile



FHTR Flux Spectra



Design Considerations of Chinese TMSR Test Reactor

Zhimin DAI

CAS Center For Excellence in TMSR

Shanghai Institute of Applied Physics, CAS

Jialuo Road 2019, Jiading, Shanghai 201800, China

Outline

- **TMSR Project Overview**
- **Considerations of TMSR Test Reactor Design**
- **Progress of TMSR-SF1 Design**

Outline

- **TMSR Project Overview**
- Consideration of TMSR Test Reactor Design
- Progress of TMSR-SF1 Design

Backgroud

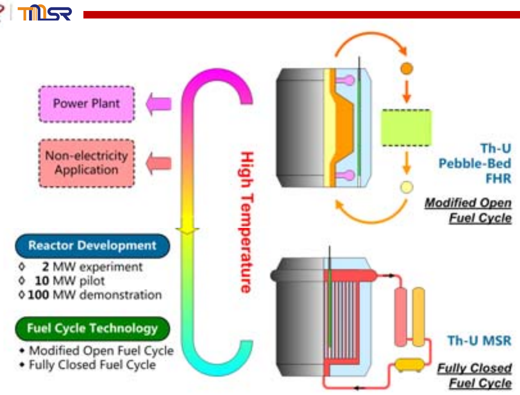
- January 2011 , Chinese Academy of Sciences (CAS) initiated the "Thorium Molten Salt Reactor Nuclear Energy System" **(TMSR).**
- August 2013, TMSR Project has been chosed as one of the National-Energy Major R&D projects of China National Energy Administration(CNEA).
- September 2014, Shanghai Government plans to start a major new-Energy project to support the TMSR project, including developing the capabilities for manufacture of the special materials & devices.

Aims of TMSR Project

- The Aims of TMSR Project is to develop Th-Energy, Non-electric application of Nuclear Energy based on Liquid-Fuel TMSR and Solid-Fuel TMSR during coming 20-30 years.
  - Liquid-Fuel TMSR (TMSR-LF)--- MSRs
  - Solid-Fuel TMSR (TMSR-SF1)--- FHRs

TMSR-SF: Optimized for high-temperature based hybrid nuclear energy application (Non-electric application).

TMSR-LF: Optimized for utilization of Thorium.



Scheme of TMSR Developing Strategy

TMSR Tasks and Funding

	2015 Research	2025 Demonstration	2035 Commercialization
TMSR-SF	2MW test reactor	10MW test/100MW demo	1GW commercial reactor
TMSR-LF	2MW test reactor	2MW test/10MW online repro.	100MW demo reactor

CAS	National&Local	Market
project funding	project funding	project funding
request : 2.588 B appro. : 2.172 B (without budget for infrastructure and Utility)	request : R&D : 1 B Infrastructure &Utility:>1B Demo : 9.5 B	Funding mainly from industry
(Jiading Cite)	(Dafeng Cite)	

Partly from Industry

TMSR Milestones

2015年

- Start construction of
  - **TMSR-SF Simulator**
  - **10MW TMSR-SF test Reactor**
  - **2MW TMSR-LF test Reactor** & Pyro-Process Facility
- Full capability of non-radioactive laboratories in Jiading.

2017年

- Commissioning of
  - TMSR-SF Simulator
  - 10 MW TMSR-SF (Reach full power)
  - 2MW TMSR-LF (Reach criticality)
- TMSR Nuclear Park (Phase I) in DaFeng.

2022年

- Commissioning of 100 MW-class TMSR-SF and 10 MW-class TMSR-LF
- Demonstration of pyroprocessing with TMSR-LF.
- TMSR Nuclear Park (Phase II)

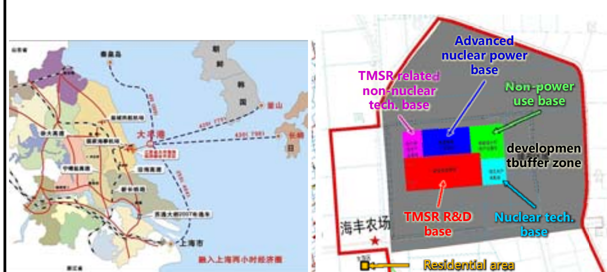
TMSR Research Bases

- **Fundamental research base at Jiading Campus of SINAP**
  - Material, Simulation, Thermal hydraulic, Safety facilities, Education and training et al., (without high radioactivity)
- **TMSR nuclear park (Reactor Site)**
  - Experimental reactors, High radioactivity Lab., Chemical reprocessing Lab. et al.,

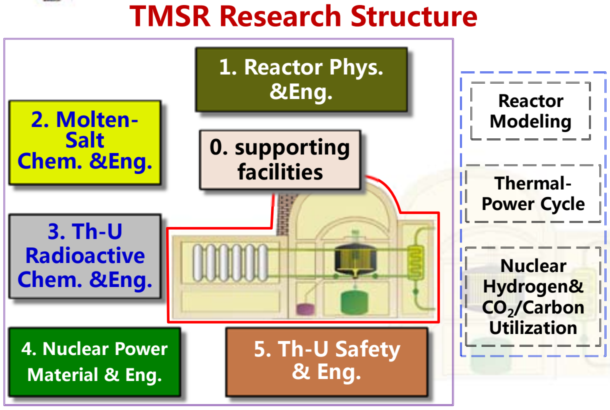
9

TMSR Reactor Site

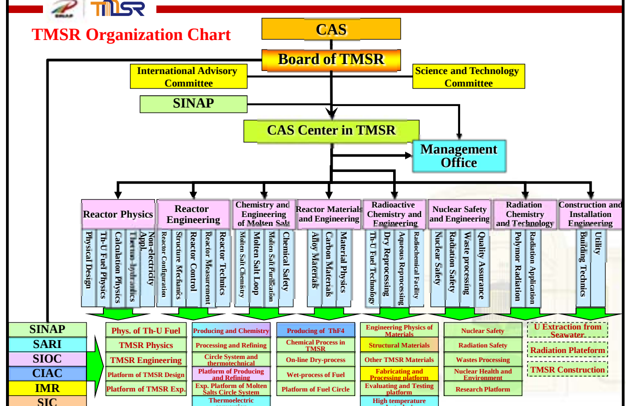
Shanghai and Jiangsu support, MEP NNSA agree in principle



TMSR Research Structure

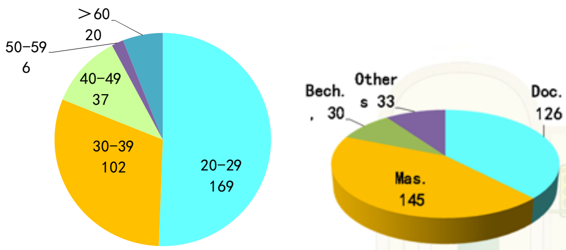


11



TMSR Team

- ~600 staffs and more than 200 graduate students , ~410 staffs and 120 students from SINAP



◆TMSR center staff Average Age ~31
◆Key personnel Average Age ~38

Cooperation of TMSR Design

- Nuclear Island of Test Reactor
  - Nuclear Power Institute of China (NPIC) , which has built many test reactor & related test bench.
  - Qinghua University, which has built HTR-10 test reactor, and being HTR-PM Demonstration.
- Site selection, Building & Utilities of Test Reactor
  - Shanghai Nuclear Engineering Research and Design Institute (SNERDI), etc., which have experience on design of building on soft soil.
- Demonstration Reactor
  - SNERDI etc.

Outline

- TMSR Project Overview
- **Considerations of TMSR Test Reactor Design**
- Progress of TMSR-SF1 Design

Goal of TMSR test reactor

- Gain Experience on reactor design, construction and operation.
- Development and integration of key technologies and components.
- Build experimental platform for future TMSR development.

Design Ability	Synthetically Ability	Behavior Verification	Materials Behavior
Reactor physics Thermal-hydraulic Safety systems Engineering design	Integration Construction Operation Maintenance	Reactor physics Thermal-hydraulic Safety concept	Fuel Materials Molten salt Device

Design & Construction of Test Reactor

Design	Conceptual Design	Preliminary Engineering Design	Engineering Design	
Safety Analysis& Licensing	Design Criteria	Principle of Examination		
	Preliminary Safety Analysis Report (PSAR)	Final Safety Analysis Report (FSAR)		
Verification & Engineering	Experimental Benches	Simulator (Mock up)	Test reactor	

FHRs can be a Precursor to MSRs

- MSRs development requires all of the technologies which is needed for an FHR (such as: materials, pumps, heat exchangers, and salt chemistry and purficiation, and power conversion) , except fuel.
- FHRs deployment does not require some of the MSR longer-term development activities (such as reprocessing of highly radioactive fuel salts).
- FHRs are much easier to control radioactive release than MSRs.

Here I just present the design consideration of TMSR-SF1 (test reactor)

Design considerations of TMSR-SF1

- Safety (is one of the most important issue)
- Conservative design:
  - Immature concept, technology, materials and analysing tool.
  - Design should cover various uncertainties.
  - Large safty margin (e.g. temperature limit) shall be considered.
- Using existing materials and technologies as possible:
  - Fuel elements of Chinese HTR project.
  - Alloy and molten salt loop technology of MSRE.

Choice of Power Level of TMSR-SF1

- Before 2013, the maximum power of the first test reactor of CAS TMSR Project (TMSR-SF1) is 2MW.
  - ➢ Test reactors are classificated as three classes by China National Nuclear Safety Administration (CNSA): Class-I ≤ 2MW, Class-II ≤ 20MW, , Class-III >20MW
- In 2013, the maximum power of TMSR-SF1) was changed to 10MW.
  - ➢ Test reactors are re-classificated as three types by CNSA: Class-I ≤ 500kW, Class-II ≤ 10MW, & Class-III >10MW
  - ➢ TMSR has been chosed as one of CNEA Projects

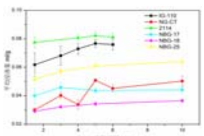
Site Selection

- The Reactor site Dafeng has a soft soil base, but lack of experience for build reactor on soft-soil ground in China.
- We proposed new siting criteria of TMSR-SF1, which is different from that of power plants,
- ➢ The seismic design criteria of TMSR-SF1 (Class-II research reactor) is set to civil building+1 degree
- Siting criteria passed the CNSA review.
- Review meeting for TMSR-SF1 being class-II research reactor has been successfully held in July 2014.

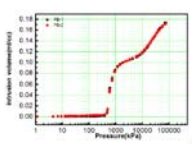
Basic materials

- Fuel: Triso particle fuel, 6cm sepherical elements of HTR-PM. Each element contains 7g LEU of 17% U5 enrichment.
- Main coolant salt: FLiBe for first loop, 99.99% ^{7}Li, FLiNaK for second loop
- Structure materials:
  - Hastelloy-N alloy: Reactor vessel, in core structure, 1st loop.
  - GH-3535 alloy (home made) : 2nd loop
  - Graphite (NG-T-10) (home made) : Reflector.

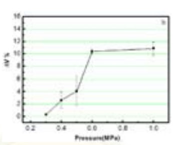
Graphite in Molten Salt



Weight Gain of Candidate reflector Graphite， Infiltrated by FLiNaK in Different Pressure



Volume ratio of FLiBe in graphite shell of fuel elemrnt estimated from mercury penetration experiment. 0.3%@0.4MPa.



Volume ratio of FLiNaK penitrated into graphite shell vs. pressure.

ratio	reflector		fuel element	
	ΔKeff	ΔPrice	ΔKeff	ΔPrice
0.5%	-113	25	-94	30
1%	-120	-87	51	-66
2%	-202	-310	-11	-252
5%	-500	-606	-141	-559
10%	-851	-1271	-276	-1138

NG-T-10 is similar to IG110 & NBG18

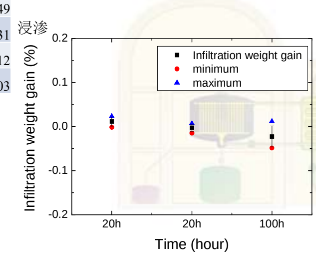
- ➢ Molten salt permeation of graphite shall be considered in neutronics.
- ➢ The graphite shell shows good resistance to FLiBe molten salt. need further and more detailed experiments. Triso particle in FLiBe salt?
- ➢ Better thermal conductance and mechanical strength after salt permeation are observed.
- ➢ Behaviour after irradiation and in moving molten salt need further experiments.
- ➢ Special graphite shall be developed for TMSR-LF, which pore size is less than 1μm

Graphite Pebbles in Molten Salt

Graphite Pebbles Infiltration weight gain (%)			
No.	20h	20h	100h
1	-0.015	0.0004	-0.019
2	0.0055	0.0189	-0.045
3	-0.007	0.0186	-0.049
4	-0.005	0.0233	-0.031
5	0.0069	-0.001	0.012
6	-4E-04	0.0102	-0.003

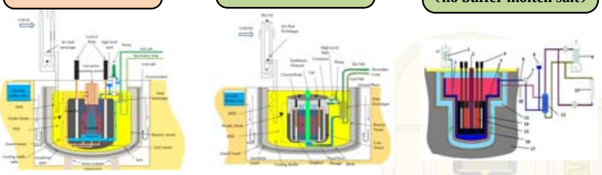
Infiltration conditions
Temp. : 700 ℃
Pressure: 5atm;
Atmosphere: Argon cover gas
Molten Salt: Flibe

Weight gain=100*（W1-W0)/W0 %
W1: weight after infiltration;
W0: Weight before infiltration;

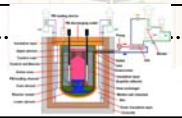


Development of TMSR-SF1

2012.4. Flowing pebble bed → 2012.9. Ordered pebble bed → 2013.1. Ordered pebble bed （no buffer molten salt）



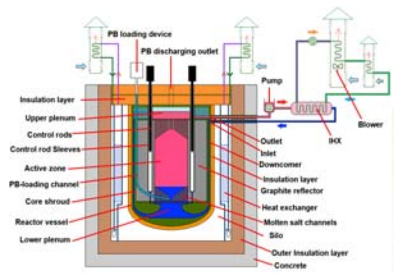
→ 2013.3，2013.7，2013.9 ………..



Engineering design consideration

--Minimum molten salt, easy for fuel handling

Schematic layout of TMSR-SF1



Main systems

- Reactor core
- Coolants: first loop and second loop
- Reactor building: confinement, auxiliary system, etc.

Main design features of TMSR-SF1

- Reactor power: $10MW_{th}$
- Coolant temperature: Inlet 600°C, outlet 650°C.
- Fuel element: Triso fuel, sepherical elements.
- Core: Graphite core, random pebble bed.
- Temperature limitations: Fuel, <1400°C; coolant outlet, <700°C.
- Reactor vessel pressure limitations: <5atm.
- With passive residual heat removal.

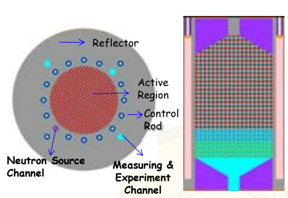
Outline

- TMSR Project Overview
- Considerations of TMSR Test Reactor Design
- **Progress of TMSR-SF1 Design**

--completed the conceptual design

Core layout of TMSR-SF1

Parameters	Value
Active area components	Fuel cone and cylinder
Active area- cylinder height	180 cm
Active area-cylinder diameter	135cm
Cone angle	30°
Reflector height	300.0 cm
Reflector OD	260.0 cm
Side-reflector thickness	75 cm
Channel number	20 ×13.0 cm

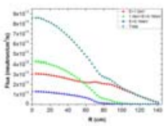


- Have considered uncertainty : boron equivalent, abundance of Li7, packing factor, fuel parameter, reflector density etc.
- Reactivity Control system: 8 shim rods, 2 regulating rods, 6 safety rods.

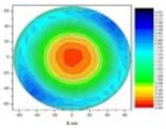
Core Physics Parameter-Neutronics

Neutron Physics Parameter	Value
Fuel pebble number	Triso filled fuel pebble Initial:10800，Full:14650
U inventory	13（Initial）/17.5（Full）kg
235U enrichment	17%
Average power density	4.8（Initial）/3.6(Full) MW/m^3
Power peak factor	1.51
Average discharge burnup	25 GWd/MTHM
Reactivity temperature coefficent	-9.0 pcm/K
Excess reactivity in thermal state	5900 pcm
Total Max neutron flux	1.79×10^{14} n/cm2/s
Max fast neutron flux (>0.1Mev)	3.08×10^{13} n/cm2/s

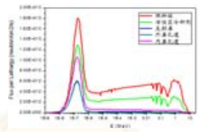
Neutron flux and power distribution



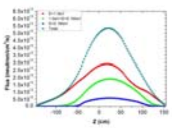
Radial neutron flux distribution



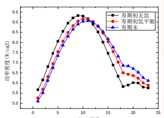
Horizontal power distribution



Neutron Spectra



Axial neutron flux distribution

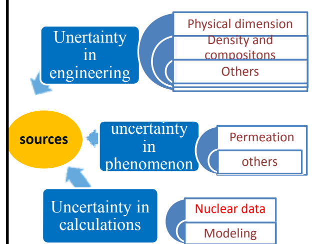


Axial power distribution

The average and maximum heat deposition power of fuel pebble is 926W and 1380W respectively.

Uncertainty consideration

- Significance: Ensure safety analysis and engineering design margins more reasonable
- Sources:

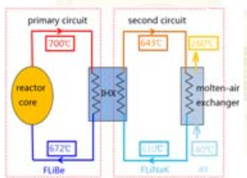


Main sources of Keff	Design value	deviation	(pcm)
Weight of U in a single pebble (g)	7	0. 033	100
U235 enrichment	17. 08%	0. 01%	110
B equivalent			
In coolant（ppm）	2	+0. 5	−70
In reflector graphite （ppm）	1	+0. 5	−86
In matrix graphite（ppm）	4	+1	−100
In UO2（ppm）	4	+0. 5	63
Density of matrix（g/cm3）	1. 73	0. 006	100
Density of reflector graphite（g/cm3）	1. 76	0. 01	100
infiltration volume percent of coolant in matrix graphite and reflector graphite	0	+3%	193
Channels in reflector			
diameter（cm）	13	−0. 1	100
Distance to the internal surface of the side reflector（cm）	6	0. 2	100
Thickness of the side reflector（cm）	80	2	115
Thickness of the top and bottom reflectors（cm）	65	2	60

Thermal hydraulic design

- Limiting value of coolant、fuel and material temperature
- Nomal heat transfer route: core→1st loop →2nd loop →air heat exchanger
- External heating and thermal Insulation.
- Passive residual heat removal.

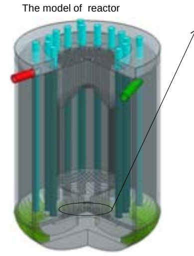
Power (MW)	10	Flow rate (kg/s)	150
Inlet T (°C)	600	Outlet T (°C)	650
Avg. flow speed in core (m/s)	0.071	Cover gas pressure (MPa)	0.15
Pressure loss in core (kPa)	6	Pressure loss from in to out (kPa)	40
Pump head (MPa)	0.4	Passive heat removal (kW)	120



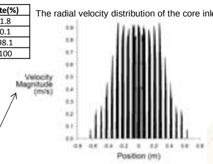
Flow distribution of Reactor Core

Flow distribution	flow(kg/s)	rate(%)
Gap of control rod	2.627	1.8
Gap of Prevent siphon hole	0.117	0.1
The core activity area	147.256	98.1
Total flow	150	100

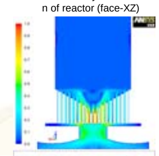
The model of reactor



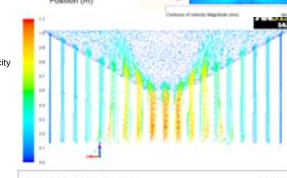
The radial velocity distribution of the core inlet



The velocity distribution of reactor (face-XZ)



The velocity vector of core



Analysis of LOFA (Complete Loss of Forced Flow Accident)

- Some analysis results, Maximum Temperature <1000 °C

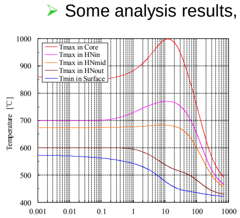
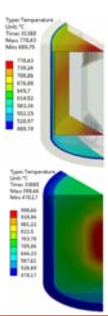


Fig. 1 The temperature curve of internals in reactor core . The Effective Thermal Conductivity is 15 $W/m{\cdot}K$



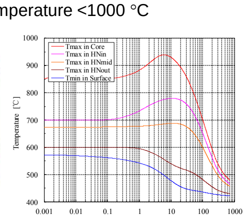
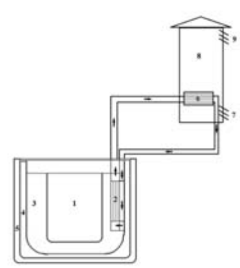


Fig. 2 The temperature curve of internals in reactor core . The Effective Thermal Conductivity is 30 $W/m{\cdot}K$

◆The Tmax in Core, HNin, HNmid, HNout or surface means the Max Temperature in the Reactor Core, the Core Barrel, the Pressure vessel, the Protection Container or the Outer Surface of the Reactor Body.

Passive decay heat removal system



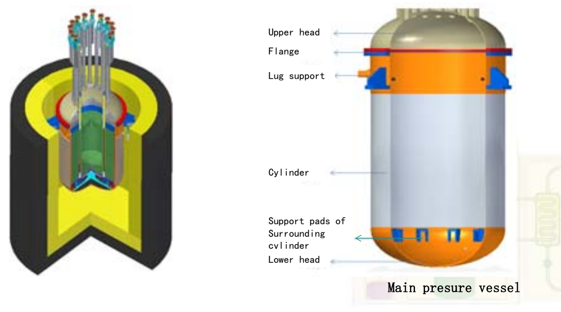
Closed air natural circulation

1.Reactor core 2.Passive air heat exchanger(PAHX) 3.silo 4. Thermal-protective coating 5. Concrete wall 6. Natural draft heat exchanger(NDHX) 7.Inlet of air door 8. Air cooling tower 9.Outlet of air door

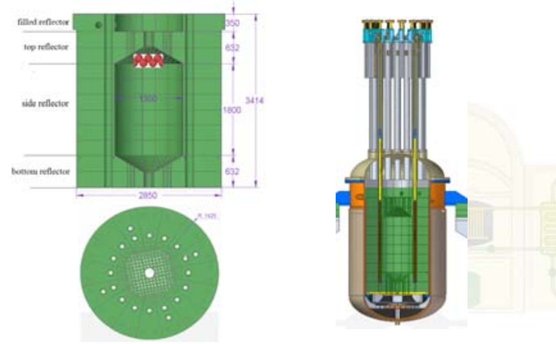
The designed parameters and temperature values of every layer

The thickness of Ar layer（m）	0.024	
The thickness of stainless steel（m）	0.3	
The thickness of thermal-protective coating（m）	0.8	
The thickness of concrete wall（m）	0.7	
Operational states	Preheat and normal operation	Accident shutdown
The inner wall temperature of Ar layer（℃）	600	600
The outer wall temperature of Ar layer（℃）	577	519
The outer wall temperature of stainless steel（℃）	560	480
The inner wall temperature of thermal-protective coating（℃）	552	420
The inner wall temperature of concrete wall（℃）	67	58
The outer wall temperature of concrete wall（℃）	48	46
The environment temperature（℃）	40	40
The heat dissipating capacity of concrete wall（kW）	8.4	5.6
The heat dissipating capacity of 4 PAHX（kW）	21.6（The inlet of air door opens 18%）	120
Total heat dissipating capacity（kW）	30	125.6

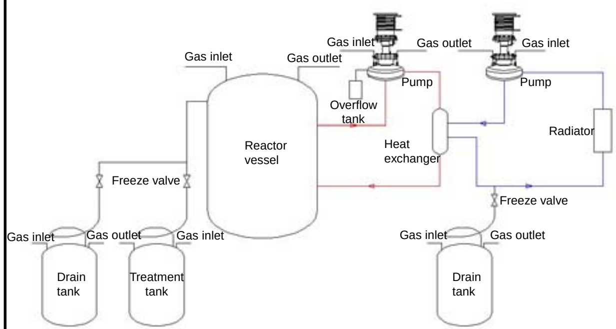
Reactor structure Layout



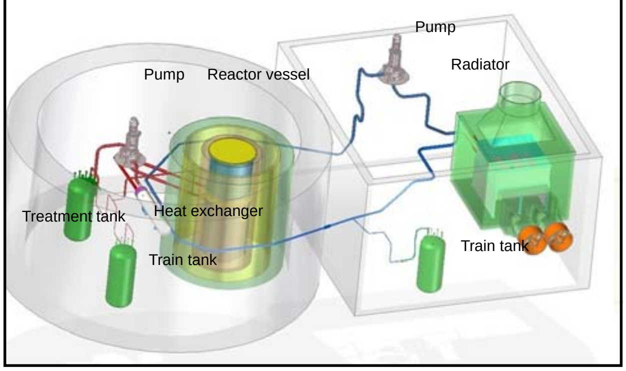
Graphite Reflector & Internal Structure



Layout of coolant system

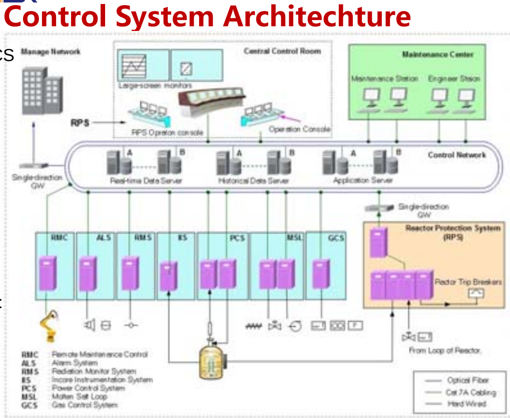


Preliminary design of coolant system



Control System Architechture

- Digital DCS
- 1E / N1E system isolated
- Hrdware: Qualified commercial products
- Software: EPICS /dedicated software

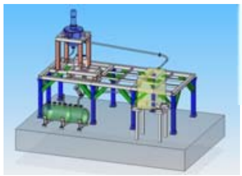


Completed a draft of the PSAR

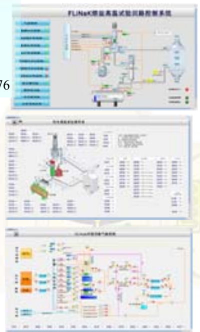
目录	PSAR	FSAR	备注
第一章 前言和反应堆概述	✓	✓	
第二章 厂址特征	✓	✓	
第三章 结构、部件、设备及系统的设计	✓	✓	
第四章 反应堆	✓	✓	
第五章 反应堆冷却剂系统和与之连接的系统	✓	✓	
第六章 专设安全设施	✓	✓	
第七章 仪器仪表控制装置	✓	✓	
第八章 电力系统	✓	✓	
第九章 辅助系统	✓	✓	
第十章 反应堆的应用	✓	✓	
第十一章 放射性废物管理	✓	✓	
第十二章 辐射防护	✓	✓	
第十三章 运行管理	X	✓	
第十四章 初始试验大纲	X	✓或单独提交	单独提交调试大纲作为此章节内容
第十五章 事故分析	✓	✓	
第十六章 技术规格书	X	✓	
第十七章 质量保证	✓或单独提交	✓或单独提交	单独提交不同阶段的质量保证大纲

事故类型	始发事件
反应性事故	1. 一根控制棒在次临界或低功率运行下失控提出； 2. 一根控制棒在功率运行下失控提出； 3. 控制棒误动作； 4. 装料过程中意外临界
堆芯排热减少事故	5. 熔盐泵卡轴； 6. 热交换器故障； 7. 丧失厂外交流电； 8. 热阱丧失； 9. 堆芯熔盐流道少量阻塞；
堆芯排热增加事故	10. 二回路流量增加； 11. 二回路温度降低；
管道破口和设备泄漏事故	12. 一回路管道小破口； 13. 二回路管道小破口； 14. 主换热器传热管破裂； 15. 燃料颗粒破损；
未能紧急停堆的预期瞬态（ATWS）	16. 失去厂外电未能紧急停堆； 17. 控制棒误抽出未能紧急停堆；
灾害（外部和内部的）	17. 地震； 18. 水淹； 19. 强风； 20. 爆炸； 21. 火灾。

FLiNaK High Temperature Test Loop



•Temp.： 550～700℃
•ΔT＜30 ℃
•Flow： 5～15 m³/h
•Heater Power：200kW
•Material： Hastelloy C276

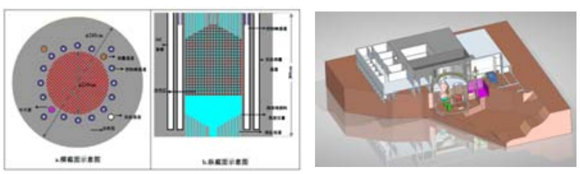


Natural circulation test loop

- Used RELAP5 and CFD to predict the loop performance.

Design of TMSR-SF Simulator

- It is 1:1 simulating the TMSR-SF1, including reactor core, two loops, fuel handing system, control system, et.al.
  - which is planned to be built in 2016.
- Electric heating power 2MW.



NEUP Integrated Research Project Workshop 6:
Fluoride Salt-Cooled High Temperature Reactor (FHR)
Test Reactor Goals, Designs, and Strategies

Session Four:
Test Reactor Licensing Strategy: DOE or NRC

October 2-3, 2014

Facilitator: George Flanagan (ORNL)

U.S. Department of Energy

Session 4 Goals

- **Goals**
  - Identify and summarize the possible licensing options
    - » DOE non-power reactor certification as described in 10CFR 830
    - » NRC non-power reactor licensing as described in 10CFR50
- **Motivation**
  - FHR Test Reactor is a hybrid first of a kind high temperature reactor
    - » Uses fuel from the HTGR
    - » Low pressure similar to the LMR
    - » Molten salt similar to MSR
  - Neither NRC nor DOE have recent licensing experience with this unique reactor
  - NRC and DOE have more experience with non-power LWRs and their reactor regulatory processes are based on LWR technology
  - Approaches to licensing are different between the two agencies
  - Important to the designer that they understand the licensing path forward - reduces uncertainty and possibly cost

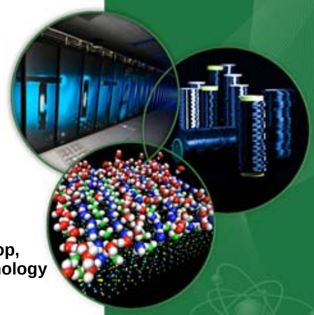
Summary of Session Discussion Topics

- **Description of the NRC non-power licensing process**
- **Description of the DOE non-power certification process**
- **Recent Adaptation of NRC approach for the proposed Aqueous Homogeneous Reactor (AHR)**
- **Discussion of positive and negative aspects of each approach**

Wrap-up Discussion

- **Summary of Discussion**
  - NRC non-power reactor licensing process was recently adapted to allow accommodation of totally unique reactor AHR.
  - Managing safe operation of the reactor after licensing needs to be addressed including implementation of PAAA
  - NRC and DOE will need to reach an agreement on the process
  - Costs and Schedule will be part of the determining factors
- **Closing questions**
  - What appears to be the smoothest path forward?
  - What would be the next step?
  - What are the obstacles that still need to be over come?
  - Are there legal issues that override the technical issues?

DOE or NRC Licensing of a Test Reactor



Dr. George Flanagan
Oak Ridge National Laboratory
Oak Ridge, TN
flanagangf@ornl.gov

For the

Sixth Fluoride-salt-cooled High-temperature Test Reactor Goals, Designs, and Strategies Workshop, Massachusetts Institute of Technology Cambridge, MA

October 2–3, 2014

Background

- Historically the test reactors that have been built and operated at DOE sites were first certified by the Atomic Energy Commission (AEC) and later recertified by DOE under 10CFR820 (Procedural Rules for DOE Activities) and 10CFR830 (Nuclear Safety Management) with the exception of:
  - Naval Nuclear Propulsion
  - Transportation (under DOT)
  - Nuclear Waste Policy Act
  - Launch Approval of Space Nuclear Energy Systems
- 10CFR835 addresses occupational radiation protection

Background (continued)

- All reactor facilities on DOE sites were originally certified by the AEC/ERDA/DOE
- Examples of some of the larger reactors are:
  - Experimental Breeder Reactor II (EBR II) (50MWt)—produced power
  - High Flux Isotope Reactor (HFIR) (100MWt)—DOE-SC
  - Advanced Test Reactor (ATR) (250MWt)—DOE-NE
  - Fast Flux Test Facility (FFTF) (400MWt)

Background (continued)

- FFTF is the most recent DOE site reactor certified by DOE
  - Constructed in the early 1970s under AEC
  - After the 1974 AEC reorganization, ERDA requested NRC review
    - ACRS and NRC staff provided review and comments
  - ERDA accepted some comments and rejected others
  - Certified by DOE in 1979

DOE's Licensing Requirements (10CFR830 Subpart B "Safety Basis Requirements" Parts 200–207)

- New reactors will require a PSAR
  - Category A facilities (reactors > 20 MWt) use NRC RG 1.70 (LWR power reactor SAR format -17 chapters) for form and content
  - No construction, materials acquisition, or component procurement allowed prior to DOE approval.
  - Licensing does not involve public participation
  - Requires an EIS before start of construction (does require public participation)
  - Approval authority is DOE management official (Asst. Sec., Director, or Asst. Administrator) who is responsible for the management of the facility
- DOE (except for FFTF) has not certified a new reactor facility

NRC's Non-Power Reactors

- NRC issues two types of reactor licenses
  - Research and development reactors 104 (AEA requires minimal regulation)
  - Commercial power or heat generation reactors 103 (>50% operational costs from sales)
- NRC has defined two types of non-power reactors (104 license)
  - *Research reactor* means a nuclear reactor licensed by the Commission under the authority of subsection 104c of the Act and pursuant to the provisions of § 50.21(c) of this chapter for operation at a thermal power level of 10 megawatts or less, and which is not a *testing facility* as defined in this section (10CFR170.3).

NRC's Definition of a Test Reactor 10CFR170.3

- *Testing facility* means a nuclear reactor licensed by the Commission under the authority of subsection 104c of the Act and pursuant to the provisions of § 50.21(c) of this chapter for operation at:
  - (1) A thermal power level in excess of 10 megawatts; or
  - (2) A thermal power level in excess of 1 megawatt, if the reactor is to contain:
    - (i) A circulating loop through the core in which the applicant proposes to conduct fuel experiments; or
    - (ii) A liquid fuel loading; or
    - (iii) An experimental facility in the core in excess of 16 square inches in cross-section.

NRC's Non-Power Reactor Licensing Requirements

- Requirements meet most but not all the regulations contained in 10CFR20 and 10CFR50
- Safety Analysis Report (18 chapters)
  - NUREG 1537 Part 1 Form and Content
  - NUREG 1537 Part 2 Standard Review Plan
- Does not require 10CFR50 Appendix A or B conformance
- Uses ANSI/ANS Research Reactor Standards ANS 15 series (2, 7,8,11,15, 17,20) Note 7,15, and 17 have been withdrawn
- Research Reactors use 10CFR20.1201 dose requirements as acceptance criteria for occupational dose (TEDE 5.0rem) and 10CFR20.1301 for public dose (TEDE 0.5 rem)

NRC's Licensing Requirements

- Test reactors
  - Use dose requirements from 10CFR100.11 for public dose acceptance criteria (TEDE 25 rem)
  - Use 10CFR20.1201 for occupational dose acceptance criteria
  - Require an EIS, public hearing, and ACRS review
- Research reactors
  - May use one step licensing at the time of construction permit request

NRC's Experience with Non-Power Reactors

- Non-power reactors
  - Licensed >150
  - Largest with operating license: 20MWt NIST reactor in Gaithersburg, MD.
    - AEC did license larger reactors such as the NASA Plum Brook at 60MWt
- NRC has shown some flexibility in licensing a non-LWR
  - Has provided an Interim Staff Guidance Document to license an Aqueous Homogenous Reactor for Medical Isotope Production
  - Final Interim Staff Guidance Augmenting NUREG–1537,"Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors," Parts 1– 2, for Licensing Radioisotope Production Facilities and Aqueous Homogeneous Reactors FR 2011.
    - ML12156A069 Part 1
    - ML12156A075 Part 2

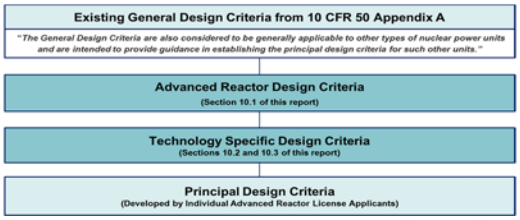
Power and/or Heat Producing Reactors Will be NRC-Licensed (Regardless of Site Location)

- Energy Policy Act 2005 sets a precedence for NRC license of a power or heat production reactor even if built on a DOE site
- Test reactors can produce power as part of its mission (EBR II)
  - Automatic NRC license (EPA 2005)
  - If less than 50% revenue—could be a candidate for a non-power license
  - Otherwise likely a prototype reactor (NGNP) 10CFR50.43(e)(2)
    - Use of prototype reactor to comply with testing requirement due to
      - Lack of data on non-LWR systems, which use simplified, inherent, or passive systems:
        - to demonstrate performance of each safety feature,
        - to understand interdependent effects between safety features, and
        - to assess analytical tools
    - Additional requirements on siting, safety features, or operational conditions may be imposed by NRC on the prototype reactor

DOE/NRC's Joint Initiative for Advanced Reactor Licensing

- Applies to power reactor licensing
  - DOE reports to NRC, November 2014
  - NRC regulatory guidance expected in FY 2015/early 2016
- Modifies the Appendix A 10CFR50 General Design Criteria (GDC) to allow applicant to use as a guide to formulate their Principal Design Criteria (Advanced Reactor Design Criteria)

DOE/NRC's Joint Initiative for Advanced Reactor Licensing (continued)



ANS Standard 20.1 will refine the ARDC to a FHR specific set of Design Criteria

Conclusion

- If DOE licenses an FHR test reactor
  - Will use power reactor requirements RG 1.70
- If NRC licenses a reactor as non-power
  - Will be a test reactor >10MWt
  - May develop reactor specific licensing form and content/review plan for FHR design (AHR)
- If NRC licenses a reactor as a power reactor (connected to grid or produces heat >50% revenue)
  - Prototype Power Reactor License Process similar to NGNP

What is the Role of a Commercial Vendor for a Class I Test Reactor?

Dr. Regis A. Matzie
Cambridge, MA

Definition of Class I Test Reactor

- **Purpose – To develop and demonstrate a specific new reactor concept. Examples:**
  - Shippingport (PWR)
  - EBR-I (SFR)
  - Dragon (HTGR)
- **Flexibility – A Good Value Proposition!**
  Some Class I test reactors made a radical design change during their lifetimes to demonstrate a significantly different concept variant. Examples:
  - Shippingport (light water breeder reactor)
  - BR3 (spectral shift reactor)

Role of Commercial Vendor

- Time Phased
- Experience Based
- Financially Limited
- Low Risk Approach

Timed Phased Roles

- Concept Definition (jointly with other stakeholders)
  - Mission development
  - Objectives articulation
  - Requirements input
- Design Development and Licensing
  - Bring experience based on commercial projects
  - Traditional supplier role in progressing the design and managing the licensing process
  - Project management capabilities, including planning and scheduling, cost estimating, quality assurance, etc.

Time Phased Roles (cont'd)

- Component Supply and Construction
  - Manufacture of specific nuclear components
  - Procure components from sub-suppliers
  - Manage project as part of a consortium/integrated team
- Operations
  - Provide operations and maintenance services (most likely with one or more partners, e.g., power company with operator experience)
  - Utilize reactor for appropriate irradiation and testing services (on a cost competitive basis)

Financial Role

- Commercial Vendor would take a low-risk, limited (if any) financial position
- Reactor would be viewed as a national asset with government owned IP or broadly shared IP
- Lack of technical and regulatory maturity, as well as highly uncertain future market, are key drivers in this approach

Perspective on Test Reactors

- **Class I test reactors are a way of accelerating the development and deployment of new technologies**
  - **They should be sufficiently flexible so that some key options can be assessed and improvements can be tested for eventual deployment**
  - **Reactor vendors are interested in seeing such test reactors built for Generation IV type technologies**
- **Class II test reactors should provide capabilities that are not available in existing test reactors**
  - **They need to be able to accelerate the deployment of new fuels and materials**
  - **They need to cost competitive with already available test reactors**

Questions?

The Jules Horowitz research reactor

CEA/Nuclear Energy Directorate ; Ch Gonnier

October 1-3, 2014
FHR workshop – MIT – session 5

www.cea.fr

JHR CONSORTIUM & GOVERNING BOARD

19/03/2007 Signature of the JHR consortium

JHR consortium gathers organizations which take part financially in the construction of JHR (1 representative / organization)

JHR Consortium current partnership: Research centers & Industrial companies

Associated Partnership (JAEA)

NNL is the UK representative to JHR UK/CEA agreement – March 2013

In some cases, the organization (member of the JHR consortium) is itself the representative of a national consortium which gathers organizations among industry, R&D organizations, TSO, or safety authority

Setting up JHR CONSORTIUM : why a "success story" ?

- Consortium "strategy" has been launched with 80% of the financial needs secured by French organizations (CEA, EdF, AREVA)
- The missing part (20%) was relatively "low"
- "Massive work" to advertize the interest of being a member of the consortium (many contacts, explanations, discussions,…)
- The "entrance ticket" (minimum 2%) is
  - "relatively cheap",
  - but nevertheless significant
    - ✓ JHR is designed for 20 simultaneous experiments (average "value" = 5%)=> 2% ~ access to a permanent irradiation cavity in the reflector with a "simple test device"
    - ✓ Especially through joint international programs (possibility to perform "complex experiments" with shared costs – open to non members)

Setting up JHR CONSORTIUM : why a "success story" ?

- The financial contribution of GB members is "secured" (any extra-cost has no impact on GB members: extra-costs are managed by CEA, partially by selling its own access rights to new comers)
- The JHR performances meet the needs of nuclear industry (involved in Gen 2 & 3 reactors). Industry representatives (and regulators) are members of the GB sometimes through domestic consortiums having a GB representative belonging to another "type" of organization (R&D). Example: Spain, Finland, UK
- Reactor Operating Costs are paid only if the access rights are used.
- Facility is designed to operate simultaneously 20 experiments (cost reduction)
- CEA developed modern instrumentation to improve the "on-line measurements" (cost reduction)
- An already active "scientific life" (yearly experimental seminar, working groups, ..)

NEUP Integrated Research Project Workshop 6: Fluoride Salt-Cooled High Temperature Reactor (FHR) Test Reactor Goals, Designs, and Strategies

Session Six: Salt Loops in Existing Test Reactor and Role of Reactor-driven Sub-critical System

October 2-3, 2014

Facilitators: David Carpenter (MIT) and Kaichao Sun (MIT)

U.S. Department of Energy

Session 6 Goals

- **Motivation**
  - The FHR is a new reactor concept and no FHR has ever been built.
  - For licensing purposes, the FHR fuel and coolant, in particular as a combined form, are required to be tested under irradiation; this is typically accomplished by loops in a test reactor.
  - Prior to construction of a commercial prototype FHR, intermediate steps, such as fluoride-salt based critical facilities, Class I or II FHTRs, or reactor driven sub-critical systems may be required for the sake of technology demonstration and/or reducing uncertainties.
- **Goals**
  - Identify the capabilities of existing test reactors. In particular, identify challenges of implementing loops for testing FHR fuel and coolant.
  - Identify the resources (time and budget) that are required for such fuel and coolant test loops.
  - Identify the needs for fluoride-salt based critical facilities.
  - Identify the role of a reactor driven sub-critical system on the path of FHR development.

SFR Development Path in France

"Rapsodie"
Experimental
1967 – 1984
40 MWth

"Phénix"
Prototype
1973 – 2009
255 MWe

"Super-Phénix"
Commercial-size
1986 –1997
1242 MWe

"ASTRID"
2017 ?

Projected FHR Path

"MSRE"
Experimental
1965 – 1969
7.4 MWth

"Session 6"

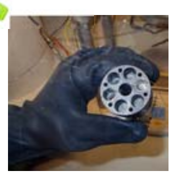
Commercial prototype

Current Status

Next step: In-core salt loop experiment

Salt loops for thermal-hydraulic research (UCB, OSU, UJV, SINAP, XJTU)

Salt chemistry and material research (ORNL, UW)



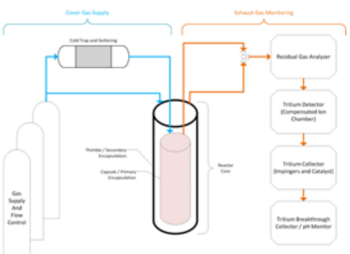
In-core capsule irradiation experiment (MIT)

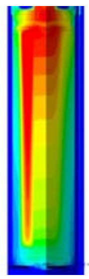
MIT Research Reactor (MITR)

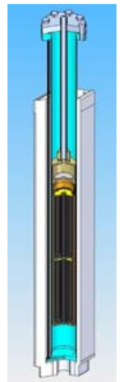
- **Located on MIT campus since 1958, upgraded in 1976**
- **6 MW, second largest university reactor in U.S.**
- **Light water-cooled, heavy water-reflected**
- **Operates 24/7, up to 10-week cycles**

Flibe Irradiation Experience at MIT

- Completed FS-1 (1000 hr) and FS-2 (300 hr) capsule irradiations in the MITR at 700°C
  - Materials segregated in graphite
  - Continuous inert sweep gas through double encapsulation



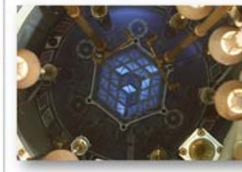




Capsule Irradiations at MIT

- Capsule conditions
  - Passive heating with some active control
  - Compact design
  - Monitor temperature at graphite mid-plane
  - Measure off-gas with mass spectrometry and tritium collection

MITR Core



Irradiation Results

- Post-Irradiation Exams (ongoing)
  - Corrosion, swelling, tritium partitioning
  - Capsule design is robust
  - Small fraction of tritium released
  - Substantial release of activation products at low temperature
- Lessons learned
  - Low-temperature operation hazards (radiolysis and release of F_2, W, Mo, Br?)
  - Tritium release behavior
  - Damage due to volatiles (BeF_2?)
  - Thermal cycling (TRISO)

FS-1 holder

FS-2 holders

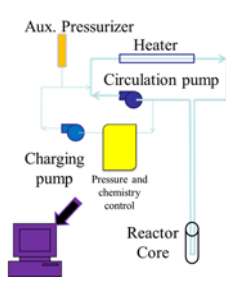
FS-2 capsule

Why a Loop?

- Drawbacks of capsule irradiations
  - Stagnant
  - Small volume of flibe
  - Surface-to-volume ratios
  - Fixed thermal gradients
- Advantages of a loop
  - More typical of FHR in-core conditions
  - More uniform environment (temperature and chemistry)
  - Online salt monitoring and control
  - Transport
- Drawbacks of a loop
  - More expensive
  - Electrical heating, insulation required
  - Cooling may be required
  - Pumps, valves may be required
  - Large double-encapsulation
  - Extraction of specimens more difficult

The MITR Water Loop

- Forced-flow pressurized water loop intended for testing under PWR/BWR conditions
  - 300°C (external heating)
  - 1500 psi
  - H_2 water chemistry
  - B/Li additions
  - On-line chemistry monitoring and re-conditioning

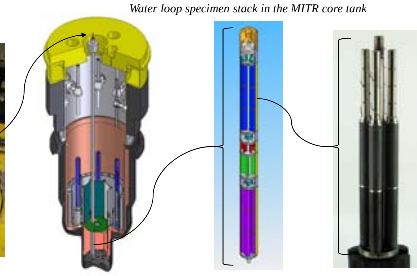


The MITR Water Loop

- Flow is annular – down inside wall of autoclave, up through specimen stack
- Autoclave is encapsulated in aluminum thimble with CO_2 insulating gap
- Capsules can be exchanged during shutdowns
- All conditions except irradiation independent of reactor power

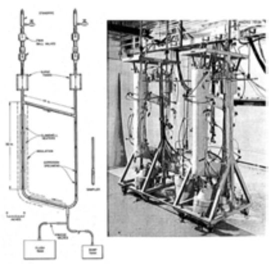
Water loop specimen stack in the MITR core tank

Water loop autoclave head



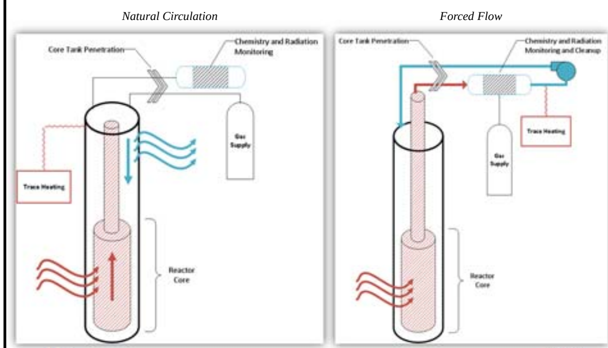
MITR Salt Loop

- Design Considerations
  - Forced vs. natural convection flow
  - Tritium control
  - Loop, specimen extraction
  - Flow out of core tank?
- Loop conditions
  - 700°C in the in-core specimen region (out-of-core testing?)
  - Cover gas sampling
  - Maintain liquid state



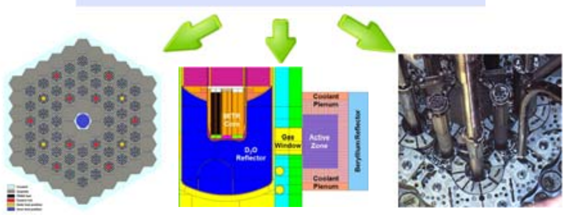
ORNL "harp" natural convection loops

Conceptual Salt Loop Designs



What's Next?

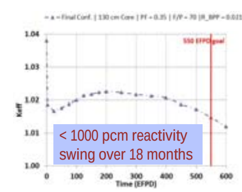
In-core salt loop experiment completed

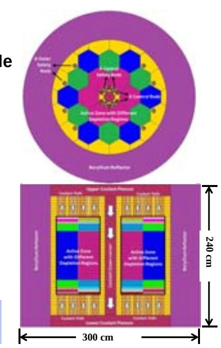


Class-I or -II FHTR	Reactor-driven sub-critical facility	Zero-power critical facility
10 – 100 MWth	0.1 – 1 MWth	e.g. ATRC and PROTEUS
Full capability	Up to 30% power density of FHR	Mainly for neutronics

Transportable FHR Core

- 20 MWth prismatic core
- 18-month once-through fuel cycle
- Burnable poison particles

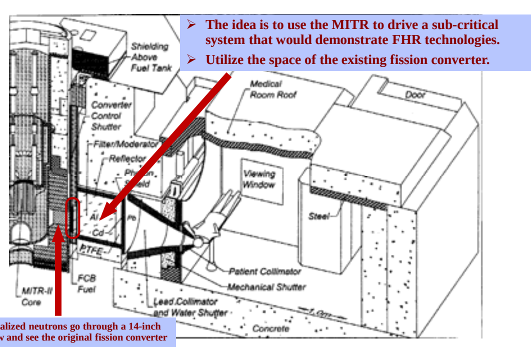




K. Sun and L. Hu. Parametric optimization of 18-month fuel cycle for a transportable fluoride-salt cooled high temperature reactor. Proceeding of ICAPP 2014, Paper 14166

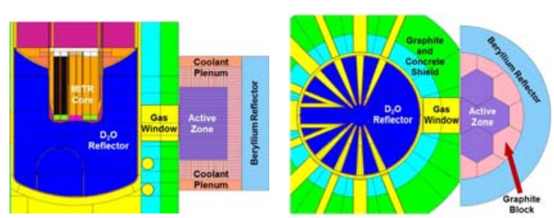
MITR Fission Converter

- The idea is to use the MITR to drive a sub-critical system that would demonstrate FHR technologies.
- Utilize the space of the existing fission converter.



Thermalized neutrons go through a 14-inch window and see the original fission converter

MITR-driven FHR System

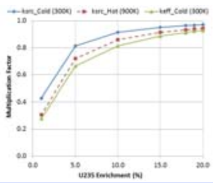


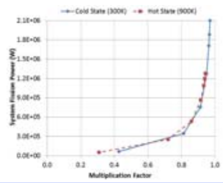
- A two-ring core with an active zone of 50 cm in radius and 80 cm in height
- Using same fuel, graphite, flibe, and reflector to demonstrate FHR technology
- Proposed sub-critical multiplication factor during operation is 0.90 – 0.93
- System fission power is 300 kW – 1 MW (i.e. up to 30% power density of FHR)

System Power Level

Three sets of multiplication factors should be studied for a sub-critical system:

1) k_{src} (with MITR neutron source) at cold state (300K)
2) k_{src} (with MITR neutron source) at hot state (900K)
3) k_{eff} (reference state, no neutron source) at cold state (300K)

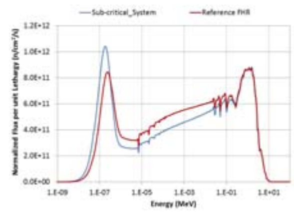




➢ **The multiplication factor of the system effectively determines the fission power level.**

System Flux Level

The average power density of the sub-critical system is about 10% - 30% of the reference transportable FHR core. Accordingly, the neutron flux level in the active zone of the sub-critical system is expected to follow the same trend.



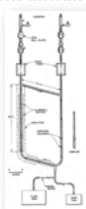
The presented neutron spectra are normalized to the fission source for comparison.

Unit: n/cm²/s	MITR-ICSA (6 MW)	Ref_FHR (20 MW)	Sub-critical (1MW)
Thermal flux (< 1 eV)	3.33E+13	2.38E+13	7.93E+12
Fast flux 1 (> 0.1 MeV)	1.23E+14	3.43E+13	9.35E+12
Fast flux 2 (>1.0 MeV)	5.90E+13	1.29E+13	3.62E+12
Total (Full range)	2.57E+14	1.33E+14	3.46E+13

➢ **The neutron spectrum of the sub-critical system is more thermalized than that of the reference FHR, due to higher moderator-to-fuel ratio.**
➢ **If the sub-critical system is designated to have 1 MW fission power, the flux level is ~ 30% of the reference FHR.**

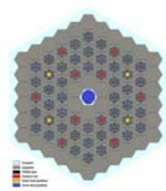
Discussions

- **Which irradiation facilities are practical choices for installation of a salt loop?**
  - **Size, flux, fissile loading, and expense constraints?**
- **What data is needed from a loop irradiation?**
- **How does an irradiation loop fit into FHTR/FHR development timeline?**

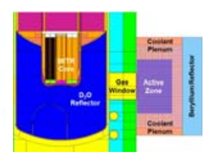


Discussions – 2

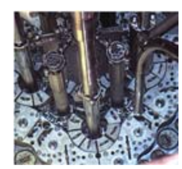
- **What are the roles of reactor-driven sub-critical systems?**
- **Are criticality facilities, such as ATRC, required?**



Class-I or -II FHTR
10 – 100 MWth
Full capability



Reactor-driven sub-critical facility
0.1 – 1 MWth
Up to 30% power density of FHTR



Zero-power critical facility
e.g. ATRC and PROTEUS
Mainly for neutronics

Summary of Session Discussion Topics

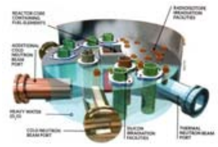
- **Reviewed the loop experience in the existing test reactors**
- **Discovered the capability of salt loops (for FHR fuel and coolant testing) in the next several years**
- **Estimated the required time and budget for such test loops**
- **Discussed the needs of fluoride-salt based critical facilities**
- **Clarified the role of reactor driven sub-critical system**

Back-up Slides…

Salt Loop in Czech

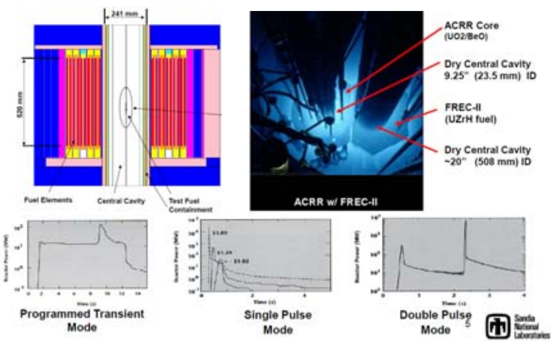
OPAL Reactor in Australia

Type of reactor	Open pool
Nominal power	20 MW thermal
Maximum Neutron flux: – Thermal, in core – Thermal, irradiation position in reflector – Fast, in core – Fast, irradiation position in reflector	 2×10^{14} n cm^{-2}s^{-1} 3×10^{14} n cm^{-2}s^{-1} 2×10^{14} n cm^{-2}s^{-1} 8×10^{12} n cm^{-2}s^{-1}
Fuel type	flat plate U$_3$Si$_2$-Al dispersion, Al clad parallel plates
Enrichment	19.8 wt%
Reference pressure of the facility	Pool open to atmosphere
Coolant (type and flow direction)	Light water, upward flow
Moderator	Light water
Reflector	Heavy water
Maximum heat flux (for 20 MW operation)	Max. (nominal) 173 W cm^{-2} Average 72 W cm^{-2}
Nominal flow rate	Total 2000 m^{3}hr^{-1}; flow rate through core 1972 m^{3}hr^{-1}
Flow rate with natural circulation	Depends on power
Experimental Facilities	Five tangential beam tubes
Irradiation Positions	Out-of-core only



Number of fuel assemblies (FA)		16	
Arrangement of fuel assemblies		4 × 4	
Layout		0 and 90° rotated	
Fuel assembly X × Y lattice dimensions		81.5 mm × 81.5 mm, parallel plate FAs	
Number of control and safety absorbers		5	
Core Coolant & Moderator material		Light Water	
Core Coolant & Moderator density [g cm^{-3}]	Operation (T=42 °C)	Pumps On	0.99180
	Shut down (T=25 °C)	Pumps Off	0.99837
Pool Coolant material		Light Water	
Pool Coolant density [g cm^{-3}]	Operation (T=41.4 °C)	Pumps On	0.99180
	Shut down (T=25 °C)	Pumps Off	0.99837
Composition of pool and core coolant Elements and weight %		H 11.191 wt% O 88.809 wt%	

ACRR at SNL

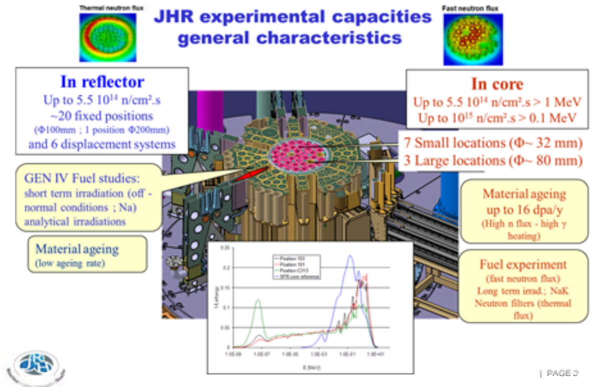


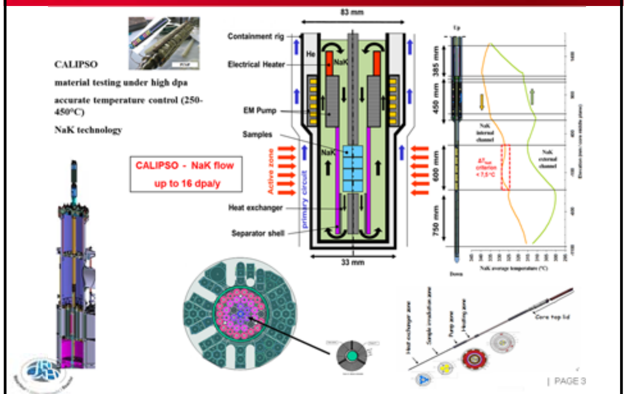
The Jules Horowitz research reactor

CEA/Nuclear Energy Directorate ; Ch Gonnier

October 1-3, 2014
FHR workshop – MIT – session 6

www.cea.fr

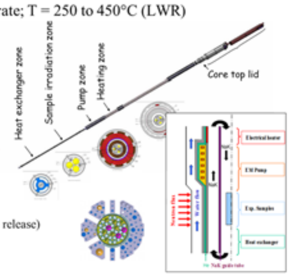




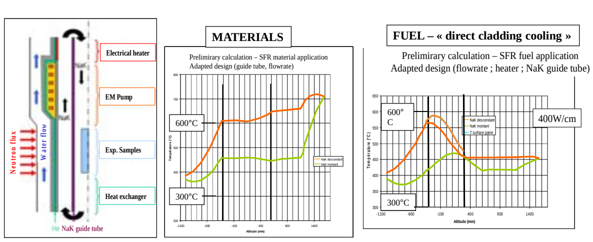
CALIPSO
evolutions toward broader applications

CALIPSO: Irradiation at controlled temperature, at low pressure, under high flux (under forced NaK convection)

- ✓ Present design : Material irradiation
  - ☞In core material irradiation, high ageing rate; T = 250 to 450°C (LWR)
- ✓ Next step (1) : Material irradiation
  - ☞In core material irradiation,
  - ☞high ageing rate;
  - ☞Temp.up to 650°C (liquid metal FR)
- ✓ Next step (2) : Fuel irradiation
  - ☞ Fuel samples irradiation SFR conditions
    - In-core : long term irradiations (NaK-filters)
    - In-reflector : off-normal situations, power and flowrate transients (Na)
    - Clad failure detection (gas sampling line ; FG release)



CALIPSO : EVOLUTION FOR SFR MATERIAL AND FUEL IRRADIATION



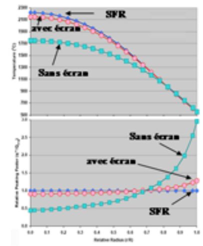
Technical points to be addressed:

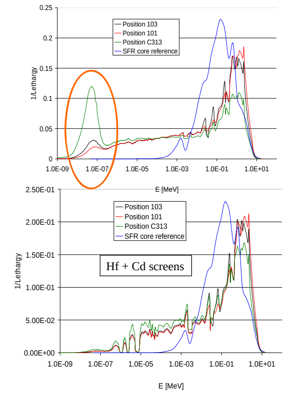
- For JHR, two safety barriers are required (to prevent any interaction between « hot molten materials and water »)
- The lifetime of the experiment will mainly depends on the lifetime of the « hot barrier » (RCCMRx manufacturing rules)
  - Above 450°C, creep must be taken into account in the design. (strong limitation of the life time, few days, even few hours, depending on the temperature)
  - In any case, lifetime will be limited due to the loss of ductility (limitation about 5dpa for 400 – 450°C)
  - Safety studies must take into account the failure of the pumping system (overheating of the structures ?)

Technical points to be addressed:

- Technology: « autonomous capsule » (only electrical connectors – no piping through the pool, expect a gas sampling line for fission gas detection or measurement)
- Location of the experiment:
  - In the core if the experiment requires a « hard neutron spectrum ». (need of « neutron filter » to make the spectrum « harder » without « thermal neutron flux » ? - Acceptable but impact on the core reactivity, => impact on refueling)
  - In the reflector for safety studies (power transient)
  - Technology : behavior of the electrical heaters (to melt the materials) under high neutron flux in the core
- Amount of high temperature molten materials is limited in the JHR core (8kg of Na – NaK)
- Safety instrumentation (temperature, flowrate,…).

CALIPSO
EVOLUTION FOR SFR FUEL IRRADIATION NEUTRON SCREENS

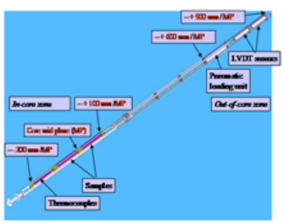


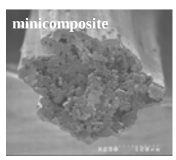


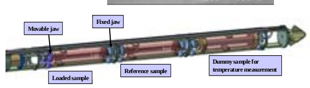
HIGH TEMPERATURE, INSTRUMENTED, MATERIAL IRRADIATION

Test device : Phaeton type - Osiris technology (Chouca-He)

Example : Tests on SiC fibers performed in Osiris reactor at ~1000°C (Cedric – Crocus experiments)







NEUP Integrated Research Project Workshop 6: Fluoride Salt-Cooled High Temperature Reactor (FHR) Test Reactor Goals, Designs, and Strategies

Session Seven: What Major Support Facilities are Required for a Test Reactor?

October 2-3, 2014

Facilitators: Edward Blandford (UNM) and David Carpenter (MIT)

U.S. Department of Energy

Session 7 Goals

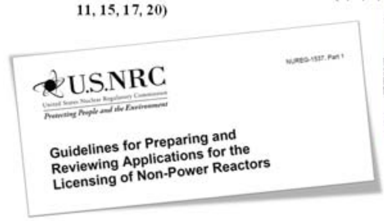
- Goals
  - Identify what Support Facilities will be required for an FHTR
    - » Is FHTR design "general-purpose" or "design-specific"
    - » Regulatory component testing requirements
    - » Identify key component qualification needs
    - » Nuclear vs. non-nuclear testing facilities
    - » Synergism with other R&D projects
  - Extension to a pre-commercial FHR design
- Motivation
  - No FHR has been operated before
  - Need to build confidence in reactor components
  - Gain operational experience
  - Reduce component testing costs
  - Fulfill qualification needs for FHTR / capability to provide qualification data for pre-commercial FHR

Summary of Discussion Topics

- Regulatory component qualification requirements for test reactors
- What historical experience of component testing programs exists for test reactors?
- Outline of FHTR component testing needs
  - How design-specific is the FHTR?
  - Key components/systems
  - Data needs
- Identification of testing facilities
  - Centralized vs. distributed facilities
  - Opportunities for international partnerships
  - Nuclear vs. non-nuclear testing requirements
  - Collaboration with other R&D programs
- Pre-commercial reactor testing requirements
  - Regulatory requirements
  - FHTR vs. CTF

Regulatory Guidelines for Test Reactor Components

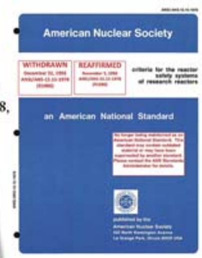
- Guiding language
- Material and component qualification requirements for licensing?
  - Most data appears to come from ANSI/ANS 15 Research Reactor Standards Series (2, 7, 8, 11, 15, 17, 20)


NUREG-1537, Part 1

U.S.NRC

Protecting People and the Environment

Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors



What Historical Component Testing Experience for Test Reactors Exist? What Applies Today?

- HTR-10
- ATR
- EBR-I/II
- MSRE
- ACRR
- Others?


MSRE


Tsinghua Univ. Graphite Seismic Test

Identifying Component Testing Needs

- What is the FHTR design philosophy?
  - General-purpose vs. design-specific
- What are the key components needed for the FHTR?
  - Nuclear instrumentation
  - Temperature monitors
  - Heat exchangers
  - Vessel and piping
  - Fuel elements and fuel handling system
  - Pumps
  - Valves
  - Fluidic diode
  - Membranes
- Specific data needs?

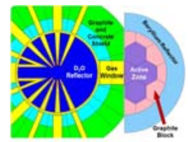
Identifying Component Testing Needs (Nuclear/non-Nuclear)

- Key sub-systems?
  - DRACS
  - Trace heating
  - Salt manufacturing / chemistry control / cleanup
  - Waste salt removal
  - Tritium sequestration / management
  - Be, T, HF, and F_2 personnel safety

Key Facilities

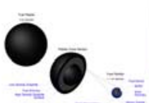
- High-Flux Salt Loop
- Subcritical Facility
- Zero-Power Reactor
- Tritium Testing
- Materials Testing
- Integrated Component Test Facility
- Heat Exchangers / Thermal-Hydraulics Testing
- Li-7 Separations Plant

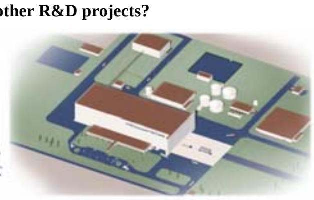
ATR



Component Testing Facilities

- What must be tested in a nuclear facility?
  - Reduce cost, complexity, and time
- Integrated/centralized facility vs. distributed?
  - National center, national labs and universities, public-private consortium?
- International collaboration opportunities?
- Synergies with other R&D projects?
  - HTGR
  - SFR

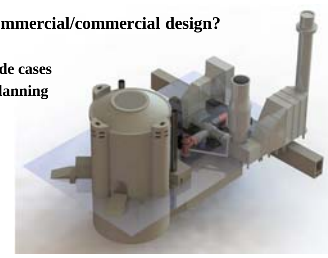




Proposed HTGR CTF

Pre-Commercial Facility Considerations

- Additional regulatory requirements for component testing and qualification?
- How can the FHTR supplement testing facilities for materials and component testing?
- CTFs specific to a pre-commercial/commercial design?
  - Fuel qualification
  - Structural materials code cases
  - Training / operations planning
  - Power conversion
  - Spent fuel storage



Wrap-up Discussion

- Summary of Discussion
  - Regulatory guidance on component testing
  - Historical experience
  - Key component testing needs
  - Required facilities
  - Pre-commercial plant considerations
- Closing questions
  - Framework for funding component testing facilities?
  - Coordination with standards bodies for code qualification data?
  - Where do CTFs fit in FHTR timeline?

The MYRRHA ADS project in Belgium

Paul Leysen
(SCK•CEN, Mol Belgium)

pleysen@sckcen.be

This is an abreviated version of the entire talk

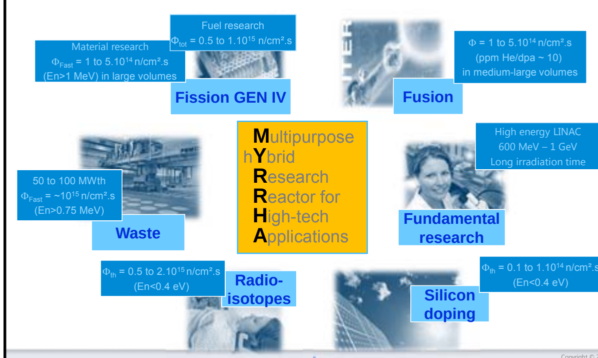
MIT workshop
Boston, USA, October 2 & 3, 2014

Contents

SCK•CEN: background

Studiecentrum voor Kernenergie - Centre d'Étude de l'énergie Nucléaire

MYRRHA objectives : a multipurpose facility



Design input

Objectives	Requirements	Choices

Flexible Fast Spectrum Irradiation Facility	• Flux 1.10^{15} n/cm².s (>0.75 MeV) in large volume (3 l)	• small target
		  • Windowless design with an off-center Spallation Loop (SL)
		  • Loopless window design
		• Liquid Metal (LM) cooling

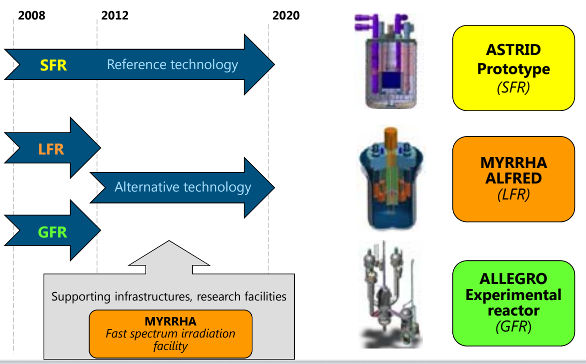
	• Availability (65%)
	• Flexibility
		• Heavy Liquid Metal (HLM) cooling
		• Pool-type
		  • In-vessel storage
		  • FA manipulation beneath core
		  • In-vessel inspection & repair
		• IPS manipulation above core
		• Replaceability

	• No high temperatures required	• Lead Bismuth Euthectic (LBE) cooling

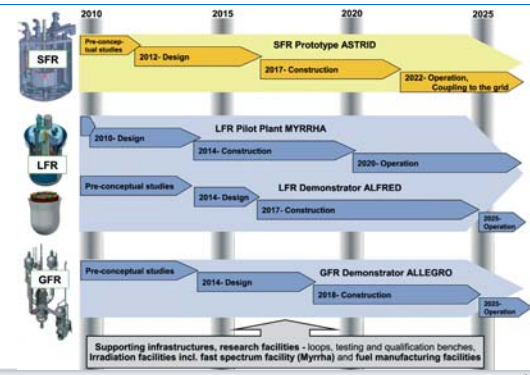
Design input

Objectives	Requirements	Choices
ADS demonstration & transmutation	• High power accelerator • Reliability • Target	• LINAC (600 MeV, 4 mA)
LFR demo	• HLM technology & components • *Critical mode operation*	• *MYRRHA/FASTEF*
Operational in 2026	• Use of mature technology where possible • Innovation where needed	• Fast Reactor MOX 30-35% fuel • Known Materials: 15-15-Ti, 316L • LBE

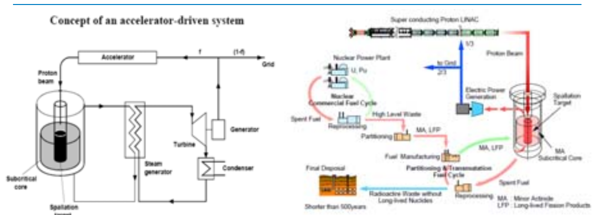
The MYRRHA position within ESNII
European Sustainable Nuclear Industrial Initiative



The ESNII schedule



What is an ADS?



An **Accelerator-Driven-System** is:

- a subcritical neutron multiplication assembly (nuclear reactor, $k_{eff}<1$),
- driven by an external neutron source,
- obtained through the spallation mechanism with high energy (~ 1GeV) protons,
- impinging on massive (high Z) target nuclei (Pb, Pb-Bi, W, Ta, U).

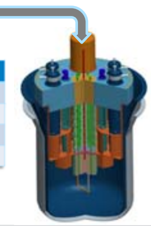
MYRRHA is an Accelerator Driven System

Accelerator	
particles	protons
beam energy	600 MeV
beam current	2.4 to 4 mA
mode	CW
MTBF	> 250 h

Reactor	
power	~85 MW_{th}
k_{eff}	0.95
spectrum	fast (flexible)
fuel	30 to 35% Pu MOX
coolant	LBE

- Demonstrate the **ADS concept** (coupling accelerator + spallation source + power reactor) at pre-indust. scale
- Demonstrate **Transmutation** (experimental fuel assemblies)
- Fast neutron source: **multipurpose and flexible irradiation facility**

Target	
main reaction	spallation
output	$2{\cdot}10^{17}$ n/s
material	LBE (coolant)
power	2.4 MW



Genesis of MYRRHA project

- 1995 – 1997: ADONIS
  - Coupling accelerator – target – subcritical core
  - Dedicated to production of ^{99}Mo
  - Coolant = water, target = ^{235}U, thermal spectrum
  - 150 MeV, 1.5 MW_{th}
- 1998 – 2005: MYRRHA “2005”
  - MTR for fuel & material research, feasibility of transmutation & demonstration of ADS principle
  - Coolant & target = Pb-Bi, fast spectrum
  - 350 MeV, 30 MW_{th} (2002) ⇨ 52 MW_{th} (2005)
- 2005 – 2009: version XT-ADS (FP6 IP-EUROTRANS)
- 2009 – 2012: version FASTEF (FP7 CDT)
- 2013 – 2014: evolution of primary system & building + start of external FEED contract

The MYRRHA LINAC accelerator : Challenges

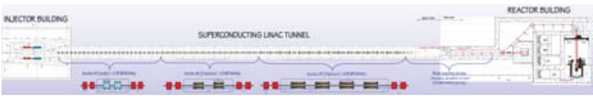
fundamental parameters (ADS)	
particle	p
beam energy	600 MeV
beam current	4 mA
mode	CW
MTBF	> 250 h

challenge !

failure = beam trip > 3 s

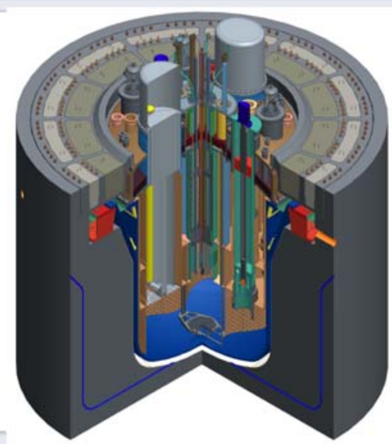
implementation	
superconducting linac	
frequency	176.1 / 352.2 / 704.4 MHz
reliability = redundancy	double injector
	“fault tolerant” scheme

The MYRRHA LINAC accelerator: Scheme

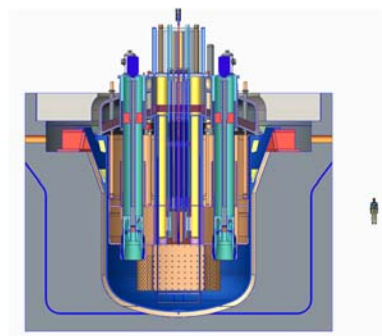


Basis for the current primary design

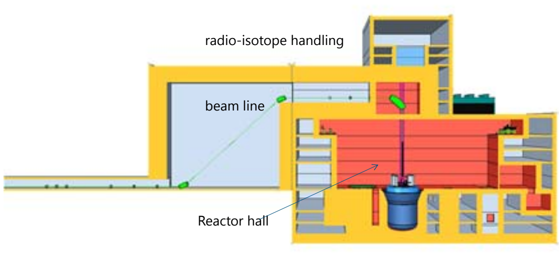
- No electricity, so "low" temperatures preferred
- Fully passive in DBC 2 & 3 conditions: 72 hours grace time
- Maximum 2 dpa for safety critical structural components;
- Reduction of Pu-enrichment from 35% to 30%, so more fuel assemblies are necessary (from 69 to 105);
- Host experiments for material research
- Severe accidents considered (Fukushima consequence), so additional emergency systems needed.
- Ability to produce in thermal islands:
  - Medical isotopes (Mo, …)
  - NTD Silicon
  - Other
- Create conditions of fusion for materials research



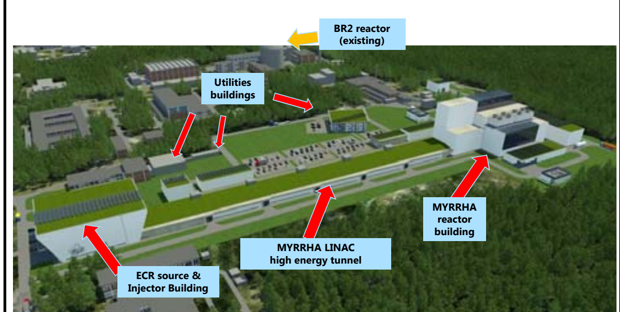
view of reactor vessel & internals



Vertical section in the Reactor building



An artistic view, when constructed



MYRRHA project planning

- 2014 • End yellow phase FEED
- 2015 • Intermediate evaluation by Belgian Government
- 2017 • Start of procurement of buildings & components
- 2021 • Completion of civil engineering work at Mol prior to delivery of components
- 2022 • Assembly of components of MYRRHA at SCK•CEN
- 2024 • Start up of the facility and commissioning
- 2026 • Full power operation

Licensing

- Complete prelicensibility file shall contain:
  - Preliminary safety assessment report
  - Environmental impact assessment report
  - Preliminary dismantling plan
- Staged approach
  - Focus points
  - Prelicensibility statement
  - Licensibility statement
- Focus points
  - Focus on all innovative aspects of LBE and coupling reactor - accelerator
  - 49 in total

Conclusions

- MYRRHA is conceived as a flexible multi-purpose fast spectrum irradiation facility;
- Able to work in sub-critical and critical mode; foreseen to be in full operation by 2026;
- Operated in the first years as an Accelerator Driven System
  - to demonstrate the ADS technology and
  - the efficient demonstration of Minor Actinides burning in subcritical mode.
- In function of needs, MYRRHA can also work as a critical flexible fast spectrum irradiation facility.
- MYRRHA will be able to significantly contribute to the development of LFR Technology and will be the European Technology Pilot Plant in the roadmap for LFR.

MYRRHA is an international project

Outside EC	Industry	Research centres	Universities

SCK•CEN
Studiecentrum voor Kernenergie
Centre d'Etude de l'Energie Nucléaire
Belgian Nuclear Research Centre

Stichting van Openbaar Nut
Fondation d'Utilité Publique
Foundation of Public Utility

Registered Office: Avenue Herrmann-Debrouxlaan 40 – BE-1160 BRUSSELS
Operational Office: Boeretang 200 – BE-2400 MOL

NEUP Integrated Research Project Workshop 6:
Fluoride Salt-Cooled High Temperature Reactor (FHR)
Test Reactor Goals, Designs, and Strategies

Session Eight
Roadmap: Path Forward

October 2-3, 2014
Charles Forsberg and Finis Southworth

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Session 8 Goals

- Goals: Examine major assumptions for commercialization strategy
- Motivation
  - Unlike HTGRs and SFRs, no FHR has been built
  - Timeline includes test reactor
  - Commercialization strategy may impact test reactor strategy

Roadmap Forward: Key Questions

- Basis for compelling case
- Need for government support
- Timing of test reactor and pre-commercial demonstration plant
- Technology development strategy
- Need to determine technology readiness levels of all systems
- Need to engage utilities on goal definition

Project Conclusion 1
Need for Compelling Case

- How should the case be strengthened?

Developing the Case for the FHR

- Base-case has three components
  - Increased revenue
  - Enable nuclear renewable grid
  - Limited radionuclide release BDBA
- Grid undergoing radical changes that support FHR case—How much work should be done to understand those grid changes to better define goals?
- How much emphasis on other goals?
  - Heat for industry
  - Government case: remote sites, navy, etc.

IRP Conclusion 2
Need for Government Support

- Time line through test reactor is too long for commercial funding
- Require government funding for test reactor
  - Atomic Energy Act and NGNP legislation support this perspective

Generation IV Reactor Development and Deployment

Costs and Risks

Dr. Finis H. Southworth
Chief Technology Officer
AREVA Inc.
September 29, 2014
Shared with Prof. Charles Forsberg
MIT

AREVA

MIT

Development Venture

One-time costs to develop SC-HTGR

Development Venture	$ millions
Technology Development (INL 2011 estimate)	316
Conceptual/ Preliminary Design	280
Final Design	200
Licensing through preparation of application[13]	165
Equipment and infrastructure development *	648
Inspections, Testing and Modifications (Demonstration initial operations)	75
Subtotal	1,609

*Note: Most infrastructure costs will be spread over several years and based upon the backlog of orders. Initial investment will be just over $1,000 Million.

MIT

 September 29, 2014

Deployment of the First of a Kind Module

Deployment Project	$ millions
Complete site-specific design	200
Construction permit/license application/review	32
Equipment procurement	432
Construction	625
Startup & testing[14]	55
Initial operations (3 years)	348
Revenue (initial 3 years)	-265
Subtotal	1,427

Infrastructure Framework	$ millions
Nuclear fuel production facility	440
Graphite production facility	150

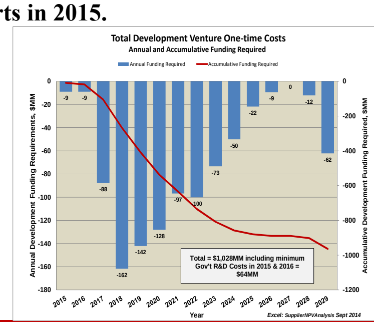
Program Direction	$ millions
Program Support	90

MIT

AREVA Tour, Operational Center of Excellence – Technical Training Center 9/9/2014

Cash flow

- **Cash flow for one-time development costs assuming preapplication discussions with NRC starts in 2015.**

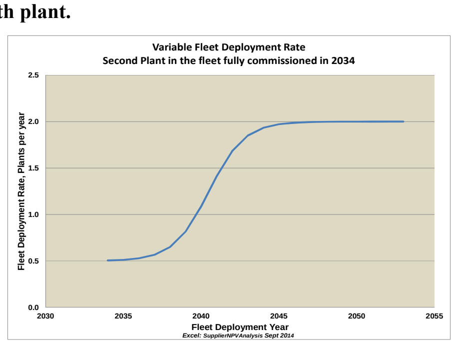


MIT

 September 29, 2014

Examples of Intellectual Property Practices

Assumed deployment rate. Each plant has four modules. Nth of a kind costs are achieved at about the sixth plant.

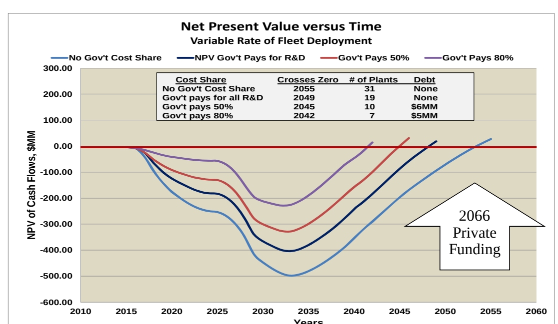


MIT

 September 29, 2014

Time to achieve breakeven for developers

- **Net Present Value versus time for the developers with variable government input. Crossover is after 27 to 40 years.**



MIT

 September 29, 2014

IRP-2 Conclusions
Implications of Development Costs
Reasonable Assumptions?

1. Private support of test reactor is not credible due to length of time before return on investment
2. Test reactor will be government funded
3. Need for public funding recognized in the U.S. Atomic Energy Act and NGNP legislation
4. It's the time to development, not the cost that implies a large initial government role
5. Large incentives to reduce deployment times

IRP Question 3
What is Impact of Pre-Commercial FHR Demo Project on FHTR?

- Technology development for pre-commercial demonstration reactor will partly parallel to test reactor development
- Does this have major implications on test reactor capabilities?

IRP Conclusion 4
Major Questions on Vendor Commercialization Strategy

- Historical commercialization strategy involved national vendors
- International vendor partnerships have become the norm
- Does this change or alter pathway to test reactor?

IRP Conclusion 5
Complex Technology Development Program Relative to other Reactor Systems

- Many component technologies being developed or may be developed by other programs
- How can this be addressed going forward?

Major Systems May Be Developed by Other Programs

- FHR strongly coupled to NGNP fuels and materials programs, Other coupling to SFR
- Developments in other fields could have large impacts on test reactor program
  - Gas turbines with heat storage → Boost commercial incentives for early deployment
  - $^{6}Li/^{7}Li$ Isotopic separation for high-volume Li-ion batteries → Cheap Li-7
  - SiC clad fuel for accident tolerant fuels → Cheaper FHR fuels and wider design choices
  - Concentrated Solar Power on Demand (High-temperature salt cooled system) → Improved salt technologies

Conclusion 6
Need for a Mapping Exercise to Define Full Readiness of Technologies

- Test reactor program leads to commercial reactor system
- Need for detail readiness level evaluation as early part of test reactor program, similar to NGNP effort
- What is the priority? Are we sufficiently advanced that a credible readiness review can be done?

Other Questions?