Integrated Research Project

Fluoride-salt-cooled High-Temperature Reactor (FHR) with Nuclear Air-Brayton Combined Cycle (NACC)

Integrated FHR Technology Development: Tritium Management, Materials Testing, Salt Chemistry Control, Thermal-Hydraulics and Neutronics with Associated Benchmarking

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Oak Ridge National Laboratory: October 4, 2016

Massachusetts Institute of Technology
University of California at Berkeley
University of Wisconsin at Madison
University of New Mexico
Reactor Basis and Description

Much of the Reactor Basis and Technology That is Being Developed is Applicable to all Molten-Salt Reactors

Market Defines Reactor Strategy

Understand 2030 Market
Higher Revenue with Variable Electricity Output

Power Conversion System to Meet Market Requirements
Base-Load Reactor with Variable Electricity to Grid Using Nuclear Air Brayton Combined Cycle (NACC)

Fluoride-salt-cooled High-Temperature Reactor
Fluoride-Salt-Cooled High Temperature Reactor (FHR) with Nuclear Air-Brayton Combined Cycle (NACC)

Stored Heat and/or Natural Gas

Base-Load Reactor

Gas Turbine

Variable Electricity And Steam

- 50 to 100% Greater Revenue than Base-Load Plant
- Enable Zero-Carbon Energy System when Coupled to Heat Storage
- Safety Strategy to Assure Fuel Integrity in All Accidents
Nuclear Air-Brayton Combined Cycle (NACC) is a Modified Natural-Gas Combined Cycle

Base-load FHR Heat-to-Electricity Efficiency: 42%
Incremental Natural Gas-to-Electricity Efficiency: 67%

Use Nuclear Heat for Base-Load Electricity

Auxiliary Heating to Higher Temperatures for Added Peak Power

Boost Revenue >50% After Pay for Natural Gas Relative to a Base-Load Nuclear Plant
FHR Combines Existing Technologies

**Fuel:** High-Temperature Coated-Particle Pebble-Bed Fuel Developed for High-Temperature Gas-Cooled Reactors (HTGRs): **Proven Technology**

**Coolant:** High-Temperature, Low-Pressure Liquid-Salt Coolant developed for the 1950s Aircraft Nuclear Propulsion Program: **Enables Coupling to Gas Turbine; Clean Salt to Minimize Licensing, Corrosion and Maintenance Challenges**

**Plant Design:** Inherited from SFR: **Low pressure system, DRACS decay heat removal, Passive shutdown**
IRP Goals Are To Address Major Challenges from Idea to Reactor

1. Combining well-known technologies into an innovative concept
   - Tritium Control and the Role of Carbon (MIT and UW)

2. Performing lab-scale experiments to validate computer models
   - Corrosion Control with Redox Control, Impurity Control, and Materials Selection. (UW and MIT)

3. Building a new collaboration network to advance FHR technology
   - Experiments and Modeling for Thermal Hydraulics, Neutronics and Structural Mechanics (UCB)

4. Developing capabilities to license FHRs and shape future of nuclear power
   - Evaluation Model Benchmarking and Validation Workshops (UCB)
Lithium Salts Generate Tritium: Must Prevent Tritium Release to Environment

\[ ^6\text{LiF} + n \rightarrow ^4\text{He} + ^3\text{HF} \]

\[ ^7\text{LiF} + n \rightarrow ^4\text{He} + ^3\text{HF} + n' \]

\[ ^{19}\text{F} + n \rightarrow ^{17}\text{O} + ^3\text{H} \]

\[ ^9\text{BeF}_2 + n \rightarrow ^4\text{He} + ^6\text{He} + 2\text{F} \]

\[ ^6\text{He} \rightarrow ^6\text{Li} + e^- + \bar{\nu}_e \quad \left( t_{1/2} = 0.8\text{sec} \right) \]
Tritium Removal from Liquid Salt Using Carbon Beds

Low-Pressure Measurements Show Large Differences In Hydrogen Sorption For Different Carbons (MIT)

- Carbon (ISO-88) designed for high fluences has low hydrogen sorption at 700° C.
- Outside the reactor core one can choose a carbon with high tritium sorption for a tritium removal bed (similar to an ion exchange system in an LWR).
- Initial assessment suggests one can control tritium levels in FHR with carbon bed external to the reactor core.

![Graph showing hydrogen uptake vs. pressure for different carbon types](image-url)
Systems Good for Tritium Removal in Clean Salts May Remove MSR Noble Metals

High Surface Area; Good Mass Transfer

High-Surface-Area Additive
Manufacture Adsorber Bed

Platinum on High-Surface-Area Carbon (Commercial Catalyst for Hydrogenation Reactions)
In-Reactor Materials Testing Underway for FHR

3rd FHR Irradiation in MITR (Fall 2016)

- 1000 hours at 700°C in enriched flibe
- Graphite and C/C specimens
  (previously irradiated SiC, 316SS, Hastelloy-N, TRISO)
Post-Irradiation Exam of FHR Materials

Irradiation-accelerated corrosion in flibe

<table>
<thead>
<tr>
<th>Material Combination</th>
<th>Weight Change (mg/cm²)</th>
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<tbody>
<tr>
<td>316ss in graphite</td>
<td>-2.2, -2.0, -1.8, -1.6, -1.4, -1.2, -1.0, -0.8, -0.6, -0.4, -0.2, 0.0, 0.2</td>
</tr>
<tr>
<td>316ss in 316ss</td>
<td></td>
</tr>
<tr>
<td>Hastelloy N in graphite</td>
<td></td>
</tr>
<tr>
<td>Hastelloy N in nickel</td>
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</table>

Desorption of tritium from irradiated components: Understanding tritium behavior in an FHR

Ar/H₂ furnace to 1100°C; online tritium measurement and capture

SS316 irradiated in flibe w/ IG-110U graphite crucible
An MIT Reactor Driven Subcritical System (RDSS) is designed to demonstrate FHR technology.

A subcritical system with $k_{\text{src}}$ of 0.98 is expected to generate 1 MW thermal power ~ 60% average FHR power density.

Licensed as an MITR experimental facility, not as a new reactor.

A cost-effective integrated experiment facility suitable for code validation, operation, maintenance, instruments, and component testing. The novel concept reduces risks for licensing a full-scale demonstration reactor.
Forced circulation flow loops provide the potential next step

- ORNL forced circulation loop MSR-FCL-1 (ORNL-TM-3866)
- Designed to be inserted into reactor beam port to study irradiation effects.

Inserted into reactor beam port
UC Berkeley FHR research focuses on thermal hydraulics, neutronics, safety and licensing.

Conceptual Design Studies

Coupled neutronics and thermal hydraulics

Separate and integral effect tests

2014 236 MWt Mk1 PB-FHR

SINAP TMSR-SF1

2014 236 MWt Mk1 PB-FHR

X-PREX Pebble Bed Tomography

Organize Expert Workshops and White Papers

Recent UCB FHR Neutronics Advances

- Code-to-code verification is in progress
- Nuclear data uncertainty quantification was performed
- Coupled Monte Carlo/CFD tool was developed for high fidelity (benchmark) calculations
- Parallel development of lower fidelity models for production calculations
- Preliminary results for TMSR-SF1 show that in case of a prompt reactivity insertion, reactivity feedbacks limit the fuel temperature and prevent fuel damage
University of Wisconsin - Production, Purification, and Reduction of FLiBe

Raw Materials → HF/H₂ → H₂ → Filtration
600 – 630°C

UW FLibe

650°C

Beryllium and Filtration

Beryllium Reduced

As-received BeF₂
As-received LiF
Melted FLiBe Salt
UW Materials Corrosion in FLiBe Salt at 700°C
Selected Tests Duplicated by MIT with In-Reactor Tests

- Materials Investigated:
  - 316 stainless steel
  - Hastelloy-N
  - SiC-SiC composites
  - C-C composites
  - Graphite

- Additional Materials to be investigated:
  - SiC coated SiC-SiC
  - Diffusion bonded SiC-SiC
  - Mo-Hf-C alloy
  - W-ZrC cermet

- Comparison of corrosion behavior in Be-reduced and unreduced FLiBe

Results for 316 stainless steel tested up to 3000 hours
**UW Electrochemistry for Redox Potential Measurements**

**Salt Redox Potential: Basis for Understanding and Controlling Corrosion**
- A voltage related to the inherent chemical potential energy of the salt
- A measure of a salt’s corrosivity
- Useful for understanding results of corrosion experiments
- **Determines when chemical reduction of the salt is necessary in order to slow corrosion**

**Measurement of the oxidizing effect of metal impurity fluorides on the FLiBe salt redox potential**

**FliBe electrochemistry globe box**
UW Natural Circulation Molten FLiBe Salt Flow Loop nearly complete
Enable Measuring Corrosion Under a Wider Set of Conditions

- Thermal hydraulics
  - Flow velocities
  - Temperature profiles
  - Beryllium transport rates
  - Characteristics of the natural circulation
  - Heat transfer characteristics

- Mass Transport
  - Beryllium redox agent transport throughout system
  - Corrosion products transport

- Corrosion
  - Stainless Steel, SiC/SiC, Alloy 800H etc.
  - Flow-assisted corrosion
  - Dissolution in hot leg and plating on cold leg

Flow-loop schematic and sample holder

IR image during heater testing - inside of the loop is at 700°C

CFD predictions of temperature profiles at the bottom, middle, and top of the heated riser
IRP-SINAP Interactions

- SINAP 10 MWt test reactor based on IRP design
- Multiple activities at multiple levels
  - Benchmarking
  - Participation in workshops
  - Joint work on tritium control strategies
    - Joint papers
    - Irradiations at MIT of Chinese materials to understand Tritium
  - Exchange of students
- Major university consortium supporting CAS-DOE agreements

UC Berkeley IRP coupled full-core neutronics/TH simulations of TMSR-SF1
Basis for Fluoride Salt–Cooled High-Temperature Reactors with Nuclear Air-Brayton Combined Cycles and Firebrick Resistance-Heated Energy Storage

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Abstract — The fluoride salt–cooled high-temperature reactor (FHR) with a nuclear air-Brayton combined...
Questions
IRP Experimental and Analytical Results
Support the FHR and other Salt Concepts
Added Information
Biography: Charles Forsberg

Charles Forsberg is the Director and principle investigator of the High-Temperature Salt-Cooled Reactor Project at the Massachusetts Institute of Technology (MIT). He teaches the nuclear fuel cycle systems and nuclear chemical engineering classes. Before joining MIT, he was a Corporate Fellow at Oak Ridge National Laboratory where he led molten salt reactor studies. He is a Fellow of the American Nuclear Society, a Fellow of the American Association for the Advancement of Science, and recipient of the 2005 Robert E. Wilson Award from the American Institute of Chemical Engineers. He received the American Nuclear Society special award for innovative nuclear reactor design on salt-cooled reactors and the 2014 Seaborg Award. Dr. Forsberg earned his bachelor's degree in chemical engineering from the University of Minnesota and his doctorate in Nuclear Engineering from MIT. He has been awarded 12 patents and has published over 200 papers.
List of IRP-2 Tasks

- FHR Tritium Control and the Role of Carbon (Largest Task)
- FHR Corrosion Control with Redox Control, Impurity Control, and Materials Selection
- FHR Experiments and Modeling for Thermal Hydraulics, Neutronics and Structural Mechanics
- FHR Evaluation Model Benchmarking and Validation Workshops
- Using Lessons Learned From FHR R&D to Advance All Generation IV Technologies
Notional 12-unit Mk1 PB-FHR nuclear station

1200 MWe base load; 2900 MWe peak

1) Mk1 reactor unit (typ. 12)
2) Steam turbine bldg (typ. 3)
3) Switchyard
4) Natural gas master isolation
5) Module assembly area
6) Concrete batch plant
7) Cooling towers (typ. 3)
8) Dry cask storage
9) Rad. waste bldg
10) Control room bldg
11) Fuel handling bldg
12) Backup generation bldg
13) Hot/cold machine shops
14) Protected area entrance
15) Main admin bldg
16) Warehouse
17) Training
18) Outage support bldg
19) Vehicle inspection station
20) Visitor parking

For more info: http://fhr.nuc.berkeley.edu
FHRs differ from other reactor classes in several key ways

- FHR fuel reaches full depletion in a short period of time
- Primary system is compact compared to HTGRs and SFRs
- Core fissile inventory is remarkably small
- Core Cs-137 inventory is remarkably small
- Uranium and enrichment requirements similar to LWRs and HTGRs

<table>
<thead>
<tr>
<th></th>
<th>FHRs</th>
<th>PWR</th>
<th>HTGR</th>
<th>SFR</th>
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<tbody>
<tr>
<td></td>
<td>Mk1</td>
<td>ORNL</td>
<td>Westinghouse</td>
<td>S-PRISM</td>
</tr>
<tr>
<td></td>
<td>PB-FHR</td>
<td>2012</td>
<td>4-loop</td>
<td></td>
</tr>
<tr>
<td>Reactor thermal power (MWt)</td>
<td>236</td>
<td>3400</td>
<td>3411</td>
<td>1000</td>
</tr>
<tr>
<td>Reactor electrical power (MWe)</td>
<td>100</td>
<td>1530</td>
<td>1092</td>
<td>400</td>
</tr>
<tr>
<td>Fuel enrichment †</td>
<td>19.90%</td>
<td>9.00%</td>
<td>4.50%</td>
<td>9.60%</td>
</tr>
<tr>
<td>Fuel full-power residence time in core (yr)</td>
<td>1.38</td>
<td>1.00</td>
<td>3.15</td>
<td>2.50</td>
</tr>
<tr>
<td>Power conversion efficiency</td>
<td>42.4%</td>
<td>45.0%</td>
<td>32.0%</td>
<td>43.8%</td>
</tr>
<tr>
<td>Core power density (MWt/m3)</td>
<td>22.7</td>
<td>12.9</td>
<td>105.2</td>
<td>4.8</td>
</tr>
<tr>
<td>Fuel average surface heat flux (MWt/m2)</td>
<td>0.189</td>
<td>0.285</td>
<td>0.637</td>
<td>0.080</td>
</tr>
<tr>
<td>Reactor vessel diameter (m)</td>
<td>3.5</td>
<td>10.5</td>
<td>6.0</td>
<td>6.2</td>
</tr>
<tr>
<td>Reactor vessel height (m)</td>
<td>12.0</td>
<td>19.1</td>
<td>13.6</td>
<td>24.0</td>
</tr>
<tr>
<td>Reactor vessel specific power (MWe/m3)</td>
<td>0.866</td>
<td>0.925</td>
<td>2.839</td>
<td>0.242</td>
</tr>
<tr>
<td>Start-up fissile inventory (kg-U235/MWe) ††</td>
<td>0.79</td>
<td>0.62</td>
<td>2.02</td>
<td>1.30</td>
</tr>
<tr>
<td>EOC Cs-137 inventory in core (g/MWe) *</td>
<td>30.8</td>
<td>26.1</td>
<td>104.8</td>
<td>53.8</td>
</tr>
<tr>
<td>EOC Cs-137 inventory in core (Ci/MWe) *</td>
<td>2672</td>
<td>2260</td>
<td>9083</td>
<td>4667</td>
</tr>
<tr>
<td>Spent fuel dry storage density (MWe-d/m3)</td>
<td>4855</td>
<td>2120</td>
<td>15413</td>
<td>1922</td>
</tr>
<tr>
<td>Natural uranium (MWe-d/kg-NU) **</td>
<td>1.56</td>
<td>1.47</td>
<td>1.46</td>
<td>1.73</td>
</tr>
<tr>
<td>Separative work (MWe-d/kg-SWU) **</td>
<td>1.98</td>
<td>2.08</td>
<td>2.43</td>
<td>2.42</td>
</tr>
</tbody>
</table>

† 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.
FHRs have unique safety characteristics for accidents resulting in long-term off-site land use restrictions from Cs-137

<table>
<thead>
<tr>
<th></th>
<th>FHRs</th>
<th>LWRs</th>
</tr>
</thead>
<tbody>
<tr>
<td>Low Cs-137 inventory</td>
<td>~30 g/MWe</td>
<td>~105 g/MWe</td>
</tr>
<tr>
<td>High thermal margin to fuel damage</td>
<td>$T_{\text{damage}} &gt; 1800^\circ\text{C}$</td>
<td>$T_{\text{damage}} \sim 830 – 1250^\circ\text{C}$</td>
</tr>
<tr>
<td>High solubility of cesium in coolant</td>
<td>CsF has high solubility</td>
<td>Cs forms volatile compounds</td>
</tr>
<tr>
<td>Intrinsic low pressure</td>
<td>High coolant boiling temperature and chemical stability</td>
<td>High vapor pressure at accident temperatures</td>
</tr>
</tbody>
</table>
FHR Couples with Nuclear Air-Brayton Combined Cycle (NACC) and Firebrick Resistance-Heated Energy Storage (FIRES)

Stored Heat and/or Natural Gas

Base-Load Reactor  Gas Turbine  Variable Electricity And Steam

Base-load Reactor with Power Station that Buys or Sells Electricity As Needed

Designed for Cheap Natural Gas or Zero-Carbon Grid
Gas Turbine Operates in Two Modes
NACC for Variable Electricity Output

Filtered Air In

Heat from FHR
Peak Air Temperature: 670°C

Add Natural Gas,
H₂ or Stored Heat
(FIRES)
Raise Peak Air Temperature to
1065°C

Nuclear Air-Brayton Combined Cycle Plant

Base-load
Electricity
100 MWe; 42% Efficient

Peak
Electricity
Added 142 MWe; 66% Efficient

Topping Cycle: 66% Efficient for added Heat-to-Electricity:
Stand-Alone Natural Gas Plants 60% Efficient
FHR With NACC Can Incorporate Firebrick Resistance-Heated Energy Storage (FIRES)

100 MWe Base-Load 142 MWe Peak

Low Price Electricity

FIRES Heat

100s MWe Low-Price Electricity

Natural Gas or H₂ (Future)

Base-Load Reactor, NACC and FIRES

Variable Electricity And Steam
Economic Basis for FHR with NACC

• Competing with Natural Gas (NG)
  – Base-load heat-to-electricity efficiency: 42%
  – Peak electricity with incremental NG gas efficiency: 66%
  – For peak electricity, more efficient than stand-alone NG plants (60%) and thus 50% increase in revenue over base-load-only nuclear plants after pay for NG

• Competing with renewables and enabling a low-carbon nuclear renewable grid
  – At times of low prices (excess electricity) convert electricity to high-temperature stored heat using Firebrick resistance-Heated Energy Storage (FIRES)
  – FIRES heat replaces burning of natural gas
  – Converting low-price electricity to high-price electricity
From the Grid Perspective, FHR/NACC/FIRES is a Second Class of Nuclear Power
FHR With Nuclear Air Brayton Cycle and FIRES Creates a Second Class of Nuclear Power Systems

Separate from LWR/SFR/HTGR That Compete for Same Energy Market
Beryllium Safety for Flibe Work

1. Operated a flibe laboratory since 2012: walk-in fume hood, and glove-boxes. Purification, salt-loop, salt transfers, and glove-box experiments.
2. Additional hazards associated with flibe handling: HF, F2, high temperature, voltage, challenges of salt transfer while ensuring purity.
3. Ensuring inert atmosphere goes hand in hand with containing Be
4. Air monitoring: Met OSHA PEL: 2 ug/m3 and Action Level: 0.2 ug/m3
5. Surface swipes housekeeping: 3 ug/100 cm2, general release: 0.2 ug/100 cm2. Occasional swipes above 0.2 ug/100 cm2 were followed by clean-up to ensure good housekeeping.
6. Better understanding of source term and particulate size distribution for flibe activity would be valuable
7. Additional options for real-time beryllium monitoring, and health monitoring should continue to be explored
10 MWth Transportable FHR (TFHR)

Design Features

- 10 MWth with ~ 5-yr fuel cycle
- Compact core ~ 2-m diameter
- Transportable by air, rail or truck
- Flibe salt coolant 600-700 °C
- High efficiency air Brayton cycle
- 18 prismatic fuel assemblies
- 6 control rods and 12 safety rods
- Center coolant down-comer

> 10-year fuel cycle optimization in progress