FLUORIDE-SALT-COOLED HIGH-TEMPERATURE REACTORS (FHR)

Implications for Tritium Management

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Workshop on Tritium Control and Capture in Salt-Cooled Fission and Fusion Reactors: Experiments, Models, and Benchmarking
Salt Lake City
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FHR Combines Existing Technologies

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

**Coolant:** High-Temperature, Low-Pressure Liquid-Salt Coolant developed for the 1950s Aircraft Nuclear Propulsion Program

**Power Cycle:** Salt Cooling Creates New Options including Brayton Power Cycles
The FHR Is a Family of Reactors

- Designs from 50 to 3000 MWt
- Different fuel geometries
- Different high-temperature salts
- Different power cycles
The Fuel

Advances in Carbon-Matrix Coated-Particle Fuel are the Enabling Technology for the FHR
FHR Uses Graphite-Matrix Coated-Particle Fuel

Same Fuel as High-Temperature Gas-Cooled Reactors in Several Different Geometric Forms
Many Fuel Options
Graphite-Matrix Coated-Particle Fuels

Pebble Bed
Fuel Plates in Hex Configuration
Fuel Inside Radial Moderator (FIRM)

- Pebble bed: Base-Case: Current technology
- Plate Fuel: Existing materials, New Design
- Fuel in Radial Moderator: Variant of HTGR Prismatic Block Fuel
Pebble-Bed FHR Reactor Built on Helium-Cooled Pebble-Bed Reactor Technology

- Most developed design and the near term option
- Similar to helium-cooled pebble bed reactors
  - FHR power density 4 to 10 times higher because liquids are better coolants than gases
  - On-line refueling (but pebbles float in salt so pebbles out top
Plate-Fuel FHR Built Upon Sodium Fast Reactor Plant Designs

- Hexagonal fuel assembly
- Plant designs similar to sodium fast reactors (low pressure, high-temperature coolant)
- “New” fuel
  - Coated-particle fuel
  - Carbon composite plates
FIRM FHR Built upon British Advanced Gas-Cooled High-Temperature Reactor (AGR)

14 AGRs Operating (2-Reactor Plants)
Graphite Moderated, Carbon-Dioxide-Cooled, Metal-Clad Pin Fuel

- Refueling Floor
- Graphite Core
- Boiler
- Pre-Stress Concrete Reactor Vessel

Small Fuel Assemblies Held Together by Tie Rod

Use AGR Core, External Fuel Geometry and Refueling Designs
Fuel Inside Radial Moderator (FIRM) Assembly Design

- Surround fuel and coolant channels with solid graphite region
  - 54 fuel channels
  - 24 coolant channels
  - Central hole for handling and materials irradiations
- Introduces spatial resonance self-shielding:
  - Enhances resonance escape probability
  - Significantly increases fuel burnup

Fuel Design is Variant of Ft. St. Vrain Gas-Cooled High-Temperature Reactor Fuel
Similar FHR and AGR FIRM Fuel Geometry → Similar Core Designs

- Similar refueling (AGR 650°C versus 700°C peak FHR coolant temperatures)
- Similar in-core graphite inspection / maintenance
- Similar instrumentation
- Similar control rod systems
- 50-year AGR operational experience base to build upon

But FHR is Low-Pressure with Liquid Cooling so Much Smaller Machine
Advanced Fuel Option: Work at General Atomics and Elsewhere May Enable FHR Pin-Type Fuel Assemblies

- Lower fuel fabrication costs
- Lower enrichments with higher fuel loading
- Longer fuel cycle and higher burnup (less waste)
- Work in progress—being developed as part of LWR accident tolerant fuel program
The Salt Coolant
For Most Proposed FHRs The Base Case Salt is $^{7}\text{Li}_2\text{BeF}_4$ (Flibe)
There Are Alternative Coolant Salts

<table>
<thead>
<tr>
<th>Coolant</th>
<th>$T_{\text{melt}}$ (°C)</th>
<th>$T_{\text{boil}}$ (°C)</th>
<th>$\rho$ (kg/m$^3$)</th>
<th>$\rho C_p$ (kJ/m$^3$°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td>$^{7}\text{Li}_2\text{BeF}_4$ (Flibe)</td>
<td>459</td>
<td>1430</td>
<td>1940</td>
<td>4670</td>
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<tr>
<td>59.5 NaF-40.5 ZrF$_4$</td>
<td>500</td>
<td>1290</td>
<td>3140</td>
<td>3670</td>
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<tr>
<td>26 $^{7}\text{LiF}$-37 NaF-37 ZrF$_4$</td>
<td>436</td>
<td>2790</td>
<td>3500</td>
<td></td>
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<tr>
<td>51$^{7}$ LiF-49 ZrF$_4$</td>
<td>509</td>
<td>3090</td>
<td>3750</td>
<td></td>
</tr>
<tr>
<td>Water (7.5 MPa)</td>
<td>0</td>
<td>290</td>
<td>732</td>
<td>4040</td>
</tr>
</tbody>
</table>

Salt compositions are shown in mole percent. Salt properties at 700°C and 1 atm. Sodium-zirconium fluoride salt conductivity is estimated—not measured. Pressurized water data are shown at 290°C for comparison.
Liquid-Salt Coolant Properties Can Reduce Equipment Size and Thus Costs
(Determine Pipe, Valve, and Heat Exchanger Sizes)

Number of 1-m-diam. Pipes Needed to Transport 1000 MW(t) with 100°C Rise in Coolant Temp.

Baseline salt: Flibe: $^7\text{Li}_2\text{BeF}_4$

<table>
<thead>
<tr>
<th></th>
<th>Water (PWR)</th>
<th>Sodium (LMR)</th>
<th>Helium</th>
<th>Liquid Salt</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pressure (MPa)</td>
<td>15.5</td>
<td>0.69</td>
<td>7.07</td>
<td>0.69</td>
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<tr>
<td>Outlet Temp (°C)</td>
<td>320</td>
<td>540</td>
<td>1000</td>
<td>To 1000</td>
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<tr>
<td>Coolant Velocity (m/s)</td>
<td>6</td>
<td>6</td>
<td>75</td>
<td>6</td>
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</table>
The Power Cycle
Power Cycle Options with 700°C Salt

NACC or NHCC

Nuclear Air or Nuclear Helium Brayton Combined Cycle based on natural-gas plants

Supercritical CO$_2$

Steam

Generator

LP turbine (x6)

HP turbine (x2)
Power Cycle Choices May Impact Tritium Control Strategies

- Can trap tritium in some power cycles because cold side of power cycle prevents tritium releases
  - Supercritical carbon dioxide
  - Helium Brayton cycles
- Tritium major challenge if enters some power cycles
  - Steam cycles
  - Air-Brayton power cycles
The FHR Tritium Challenge
FHR Tritium Challenge

- Reactor environment
  - Clean fluoride salt coolant containing Lithium-7
  - Tritium generation varies by order of magnitude depending upon FHR design
  - Carbon-based fuel that absorbs tritium and impacts chemistry
  - Potential for small quantities of fission products from leaking fuels

- Tritium challenge
  - FHRs produce tritium significantly above levels requiring controls
  - Tritium absorbed in graphite fuel in significant quantities
    - Possibility in pebble bed to capture tritium on pebbles and recycle pebbles with tritium removal during recycle for tritium control
    - Must consider off-normal events where tritium inventory in fuel may be released if increased core temperatures

- Tritium is a waste—can recover for use but not a requirement
Questions

Tritium Environment Below-Previous Slide

- Reactor environment
  - Clean fluoride salt coolant containing Lithium-7
  - Inventory varies by order of magnitude depending upon design
  - Carbon-based fuel
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Biography: Charles Forsberg

Dr. Charles Forsberg is the Director and principle investigator of the High-Temperature Salt-Cooled Reactor Project and University Lead for the Idaho National Laboratory Institute for Nuclear Energy and Science (INESI) Nuclear Hybrid Energy Systems program. He is one of several co-principle investigators for the Concentrated Solar Power on Demand (CSPonD) project. He earlier was the Executive Director of the MIT Nuclear Fuel Cycle Study. Before joining MIT, he was a Corporate Fellow at Oak Ridge National Laboratory. 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 for outstanding chemical engineering contributions to nuclear energy, including his work in hydrogen production and nuclear-renewable energy futures. 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 11 patents and has published over 200 papers.

http://web.mit.edu/nse/people/research/forsberg.html
Many Teams Working on the FHR

- MIT, UCB, UW, and UNM

<table>
<thead>
<tr>
<th>Organization</th>
<th>PI</th>
<th>Area</th>
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</thead>
<tbody>
<tr>
<td>MIT</td>
<td>Charles Forsberg</td>
<td>Market case</td>
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<td>Lin-wen Hu</td>
<td>Irradiation experiments</td>
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<tr>
<td>University of California, Berkeley</td>
<td>Per F. Peterson</td>
<td>Thermal-hydraulics, safety</td>
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<td></td>
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<tr>
<td>University of Wisconsin, Madison</td>
<td>Kumar Sridharan</td>
<td>Materials</td>
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<tr>
<td>University of New Mexico</td>
<td>Edward Blandford</td>
<td>Thermal-hydraulics, safety</td>
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- National Laboratories: ORNL, INL, etc.
- Georgia Tech Consortium
- Chinese Academy of Science (2020 FHR test reactor)
- Vendors
The Idea of a Fluoride-salt-cooled High-temperature Reactor (FHR) dates to 2002

- No FHR has been built
- Compelling reasons must exist to develop a new reactor type
  - Commercial: Improve economics
  - Government: Meet national goals
  - Public: Safety against major accidents