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
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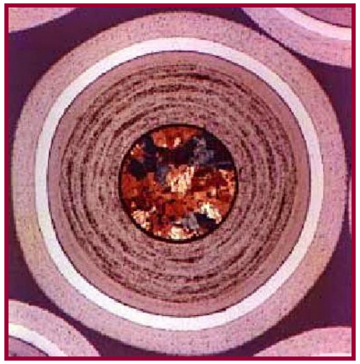
FHR Combines Existing Technologies



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

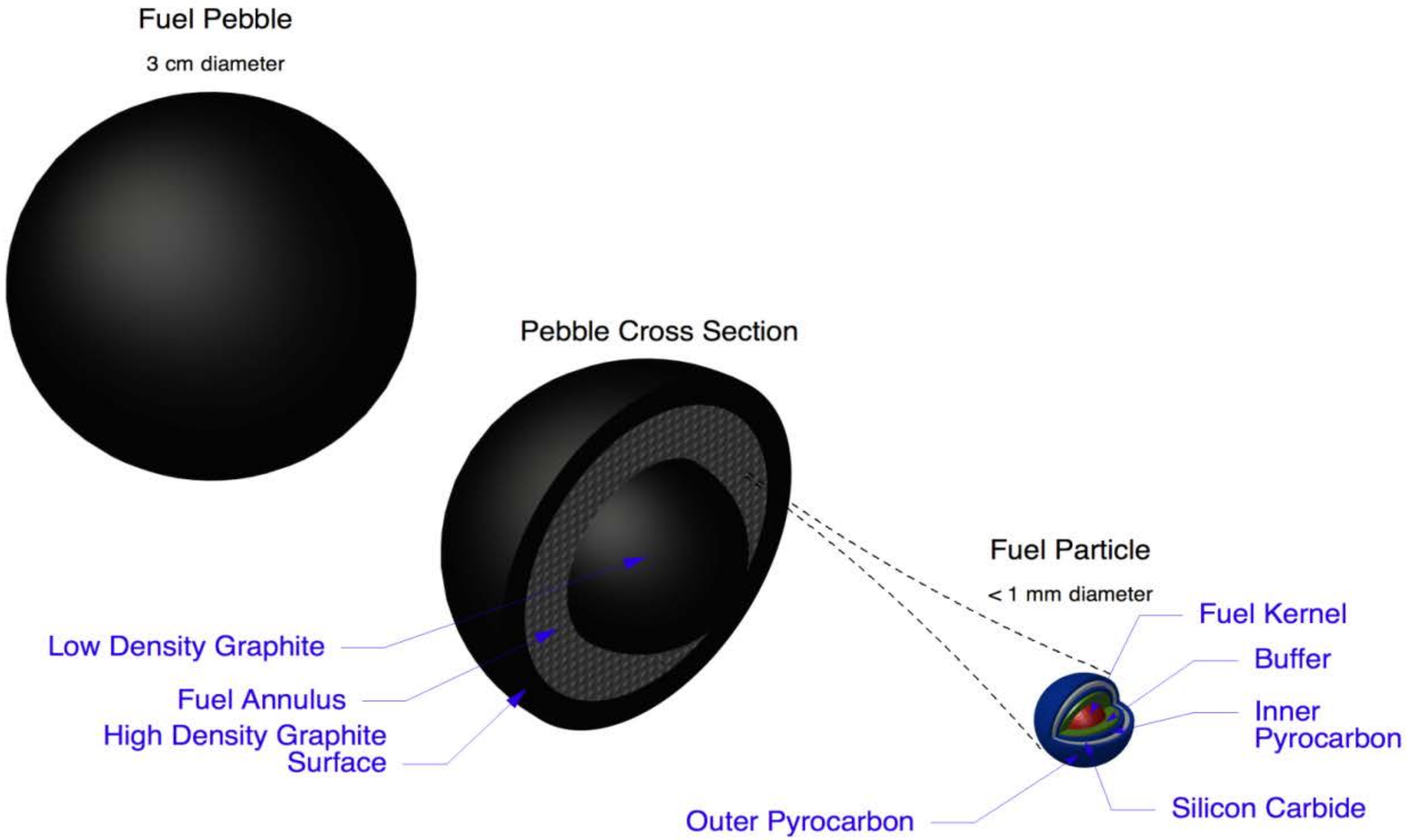
- Designs from 50 to 3000 MWt
- Different fuel geometries
- Different high-temperature salts
- Different power cycles

The Fuel

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

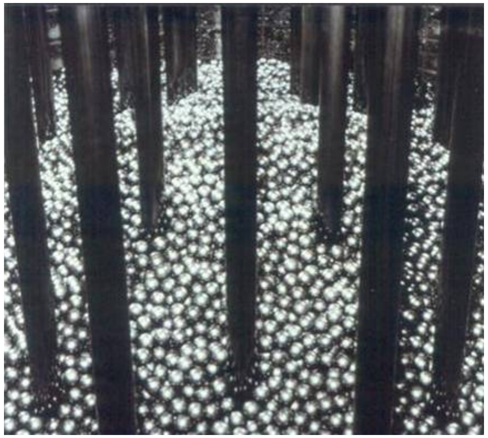
FHR Uses Graphite-Matrix Coated-Particle Fuel

Same Fuel as High-Temperature Gas-Cooled Reactors in Several Different Geometric Forms



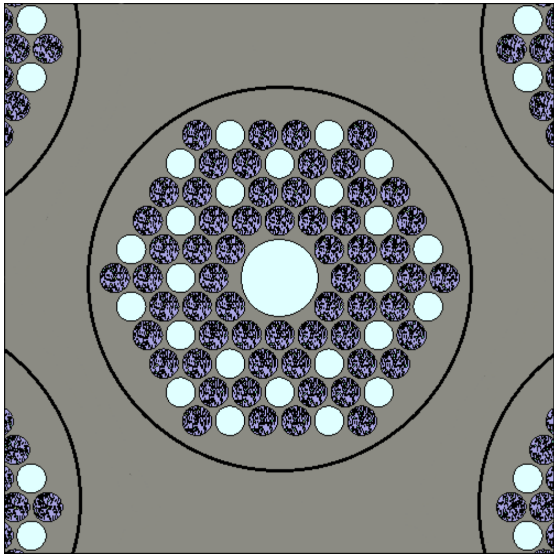
Many Fuel Options

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

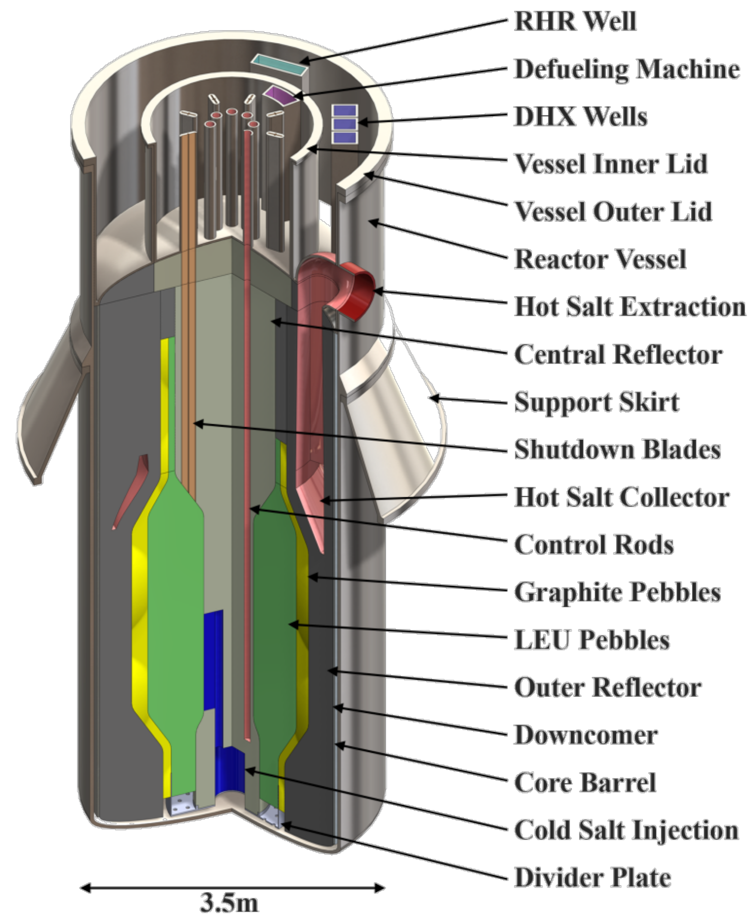
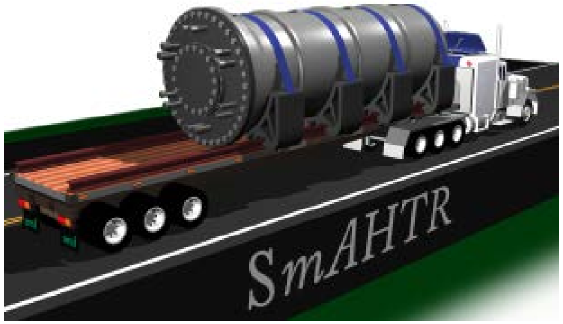


Plate-Fuel FHR Built Upon Sodium Fast Reactor Plant Designs

- Hexagonal fuel assembly
- Plant designs similar to sodium fast reactors (low pressure, high-temperature coolant)
- "New" fuel
  - Coated-particle fuel
  - Carbon composite plates



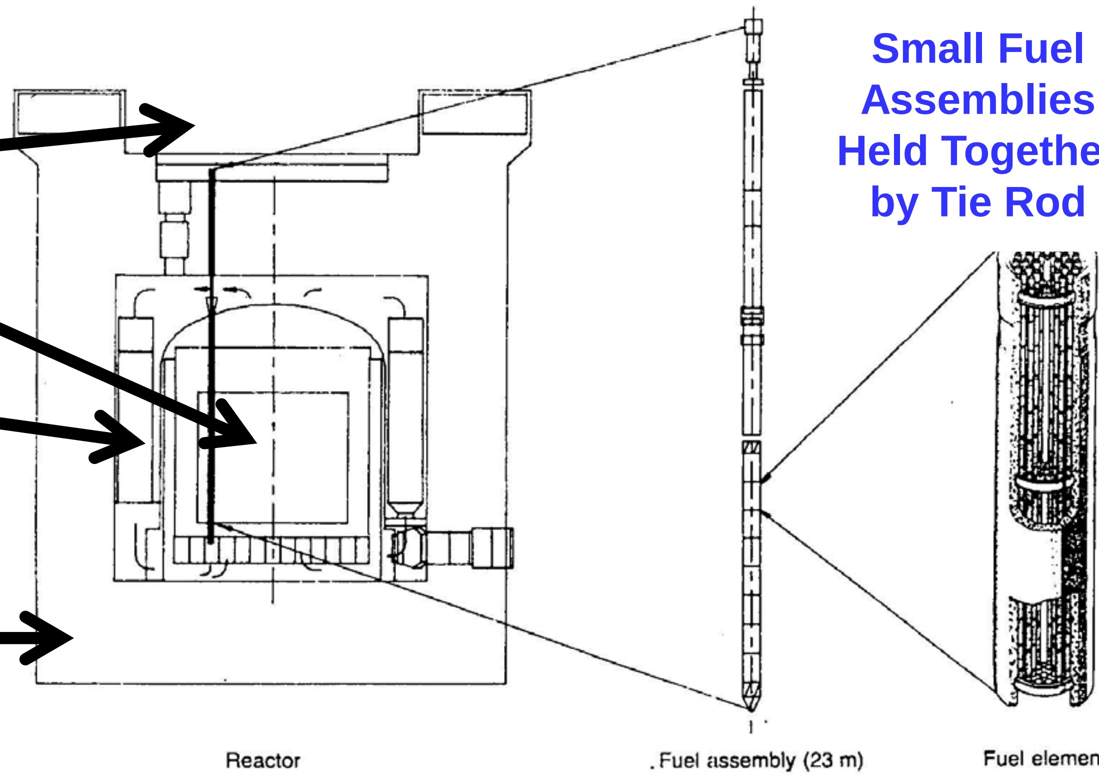
2010 125 MWt

2012 3600 MWt

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

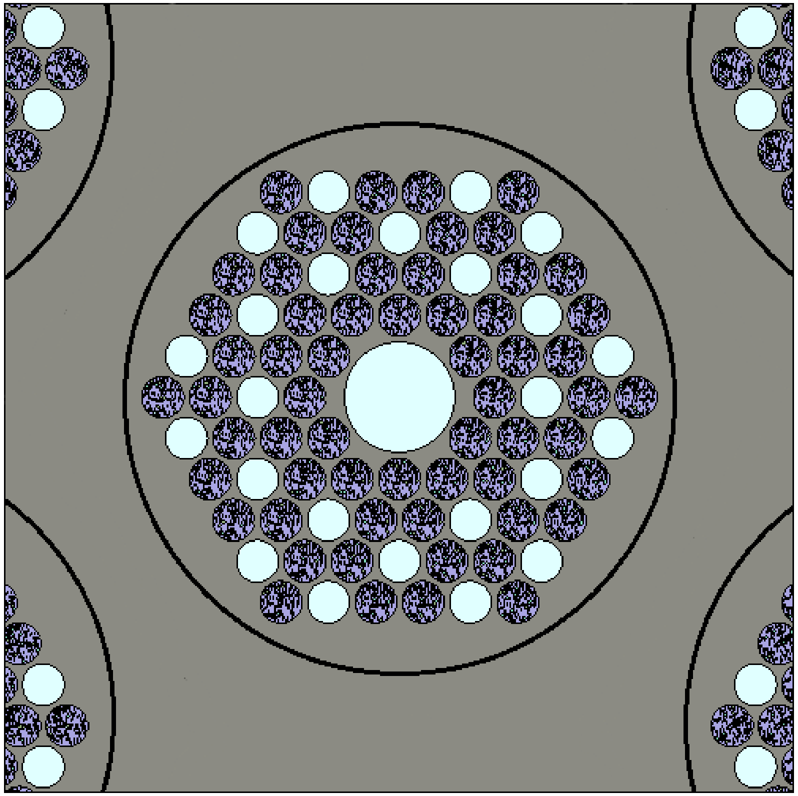


Small Fuel Assemblies Held Together by Tie Rod

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 $^7Li_2BeF_4$ (Flibe)

There Are Alternative Coolant Salts

Coolant	T_{melt} (°C)	T_{boil} (°C)	ρ (kg/m³)	ρC_p (kJ/m³ °C)
$^7Li_2BeF_4$ (Flibe)	**459**	**1430**	**1940**	**4670**
59.5 NaF-40.5 ZrF_4	500	1290	3140	3670
26 ^{7}LiF-37 NaF-37 ZrF_4	436		2790	3500
51 ^{7}LiF-49 ZrF_4	509		3090	3750
Water (7.5 MPa)	0	290	732	4040

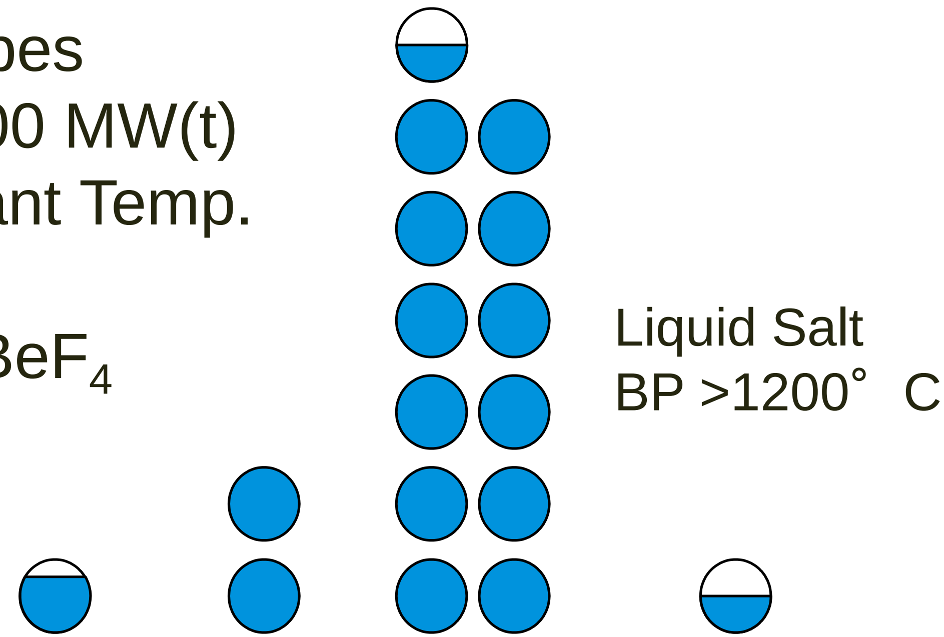
Salt compositions are shown in mole percent. Salt properties at 700ºC and 1 atm. Sodium-zirconium fluoride salt conductivity is estimated—not measured. Pressurized water data are shown at 290°C for comparison.

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 100ºC Rise in Coolant Temp.

Baseline salt: Flibe: $^{7}Li_2BeF_4$



	Water (PWR)	Sodium (LMR)	Helium	**Liquid Salt**
Pressure (MPa)	15.5	0.69	7.07	**0.69**
Outlet Temp (ºC)	320	540	1000	**To 1000**
Coolant Velocity (m/s)	6	6	75	**6**

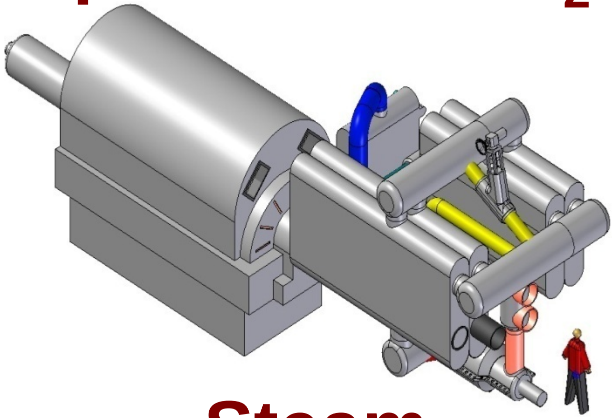
The Power Cycle

Power Cycle Options with 700°C Salt

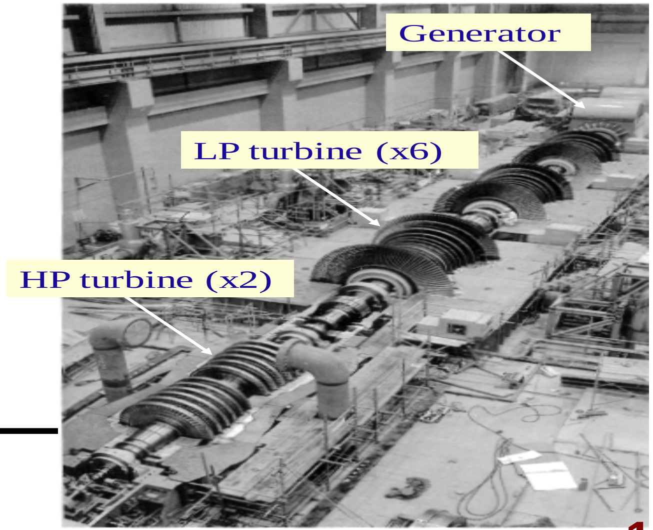
NACC or NHCC

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

Supercritical CO_2



Steam



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 Bryaton 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
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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 (INEST) 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.

Many Teams Working on the FHR

- MIT, UCB, UW, and UNM

Organization	PI	Area
MIT	Charles Forsberg Project Leader	Market case Severe Accidents
	Lin-wen Hu	Irradiation experiments
University of California, Berkeley	Per F. Peterson	Thermal-hydraulics, safety
	Massimiliano Fratoni	Neutronics
University of Wisconsin, Madison	Kumar Sridharan	Materials
University of New Mexico	Edward Blandford	Thermal-hydraulics, safety

- 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

FISSION REACTORS

MOLTEN-SALT–COOLED ADVANCED HIGH-TEMPERATURE REACTOR FOR PRODUCTION OF HYDROGEN AND ELECTRICITY

KEYWORDS: *molten salt, high-temperature reactor, hydrogen production*

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The molten-salt–cooled Advanced High-Temperature Reactor (AHTR) is a new reactor concept designed to provide very high-temperature (750 to 1000°C) heat to enable efficient low-cost thermochemical production of hydrogen (H_2) or production of electricity. This paper provides an initial description and technical analysis of its key features. The proposed AHTR uses coated-particle graphite-matrix fuel similar to that used in high-temperature gas-cooled reactors (HTGRs), such as the General Atomics gas turbine–modular helium reactor. However, unlike the HTGRs, the AHTR uses a molten-salt coolant and a pool configuration, similar to that of the General Electric Super Power Reactor Inherently Safe Module (S-PRISM) liquid-metal reactor. Because the boiling points for molten fluoride salts are near ~1400°C, the reactor can operate at very high temperatures and atmospheric pressure. For thermochemical H_2 production, the heat is delivered at the required near-constant high temperature and low pressure. For electricity production, a multireheat helium Brayton (gas-turbine) cycle, with efficiencies >50%, is used. The low-pressure molten-salt coolant, with its high heat capacity and natural circulation heat transfer capability, creates the potential for robust safety (including fully passive decay-heat removal) and improved economics with passive safety systems that allow higher power densities and scaling to large reactor sizes [>1000 MW(electric)].

I. INTRODUCTION

The Advanced High-Temperature Reactor (AHTR) is a new reactor concept developed by the authors to produce high-temperature heat (750 to 1000+°C) for efficient production of electricity and thermochemical H_2. The AHTR is based on four technological developments:

1. high-temperature, low-pressure molten-fluoride-salt reactor coolants from the aircraft nuclear propulsion program[1] of the 1950s and the molten-salt breeder reactor program of the 1960s and 1970s.[2] Today, development continues on using these salts and associated advanced high-temperature materials for cooling fusion reactors.

2. coated-particle graphite-matrix fuel developed for high-temperature gas-cooled reactors[3] (HTGRs) in the United States and Germany, starting in the 1960s

3. passive safety systems for gas-cooled and liquid-metal reactors introduced in the 1980s

4. advanced gas turbines—including commercialization in the last 5 yr of magnetic bearing systems that can permit these turbines to be used in closed helium cycles.

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