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

Charles Forsberg (cforsber@mit.edu)

Molten Salt Workshop 2016
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

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

Reactor Basis and Description

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

C. Forsberg and P. F. Peterson, "Basis for Fluoride Salt-Cooled High-temperature Reactors with Nuclear Air-Brayton Combined Cycles and Firebrick Resistance Heated Energy Storage", *Nuclear Technology*, 196 , Oct. 2016

Market Defines Reactor Strategy

Understand 2030 Market

Higher Revenue with Variable Electricity Output

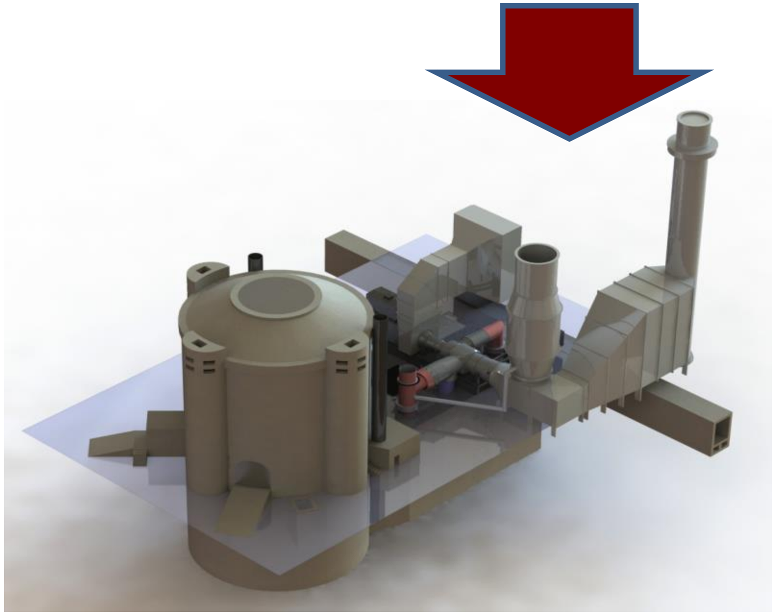
Power Conversion System to Meet Market Requirements

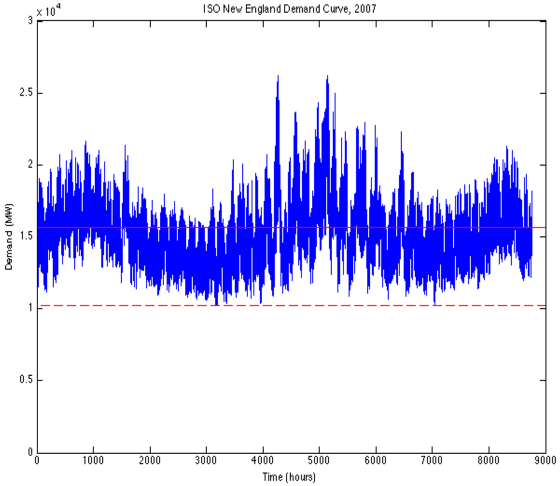
Base-Load Reactor with Variable Electricity to Grid Using Nuclear Air Brayton Combined Cycle (NACC)

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

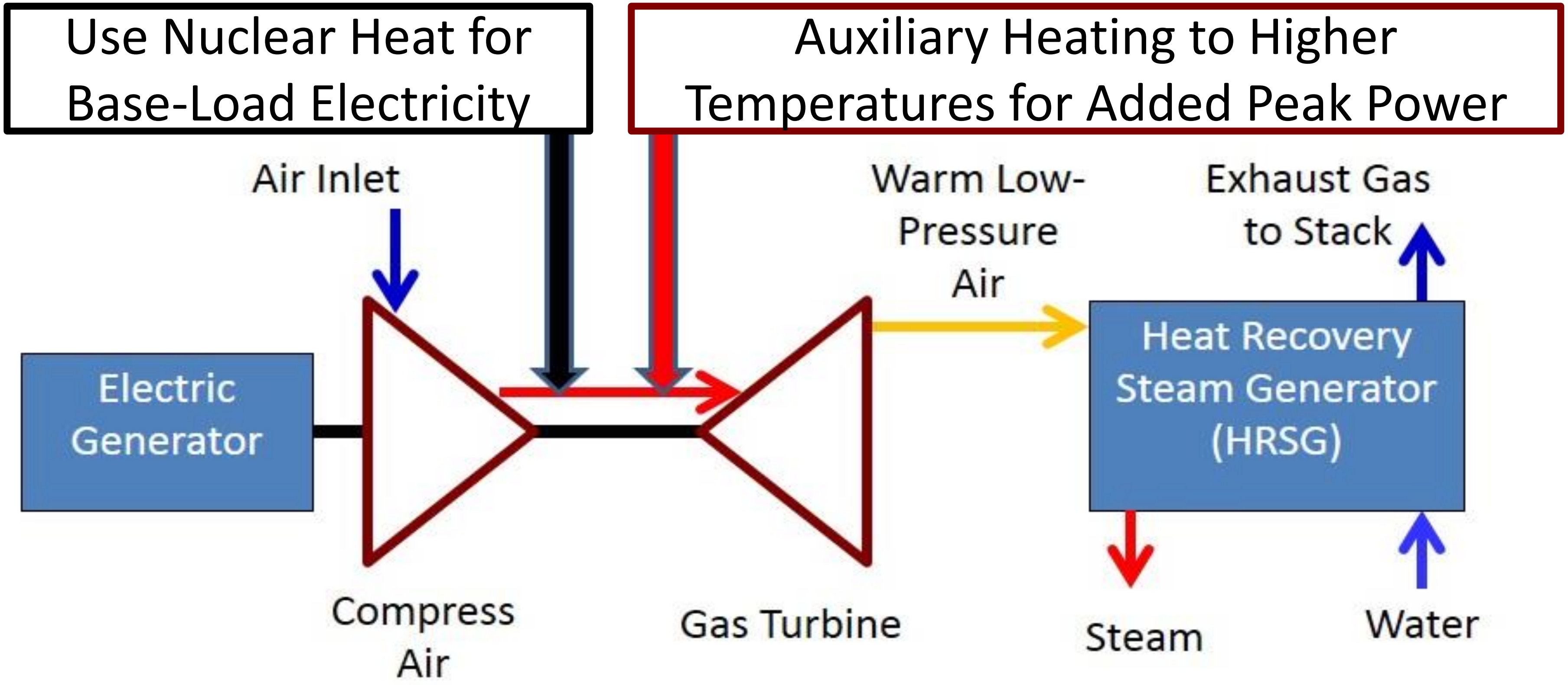
Variable Electricity And Steam

- 50 to100% 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

Air Inlet

Warm Low-Pressure Air

Exhaust Gas to Stack

Electric Generator

Heat Recovery Steam Generator (HRSG)

Compress Air

Gas Turbine

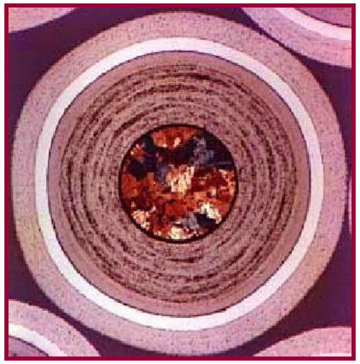
Steam

Water

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

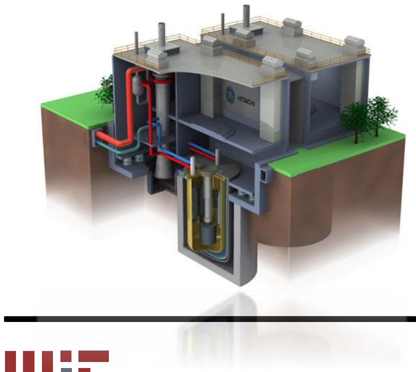
Auxiliary: Heat Natural Gas, Hydrogen or Stored Heat; Simplified Schematic of Power Cycle

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
2. Performing lab-scale experiments to validate computer models
3. Building a new collaboration network to advance FHR technology
4. Developing capabilities to license FHRs and shape future of nuclear power

- Tritium Control and the Role of Carbon (MIT and UW)
- Corrosion Control with Redox Control, Impurity Control, and Materials Selection. (UW and MIT)
- Experiments and Modeling for Thermal Hydraulics, Neutronics and Structural Mechanics (UCB)
- Evaluation Model Benchmarking and Validation Workshops (UCB)

Tritium Control (MIT)

Lithium Salts Generate Tritium: Must Prevent Tritium Release to Environment

$$^{6}\text{LiF} + n \rightarrow {}^{4}_{2}\text{He} + {}^{3}_{1}\text{HF}$$

$$^{7}\text{LiF} + n \rightarrow {}^{4}_{2}\text{He} + {}^{3}_{1}\text{HF} + n'$$

$${}^{19}_{9}\text{F} + n \longrightarrow {}^{17}_{8}\text{O} + {}^{3}_{1}\text{H}$$

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

$${}^{6}_{2}\text{He} \rightarrow {}^{6}_{3}\text{Li} + e^{-} + \overline{v}_e \quad \left(t_{1/2} = 0.8\,\text{sec}\right)$$

And Fission Product Tritium In MSR with Dissolved Fuel

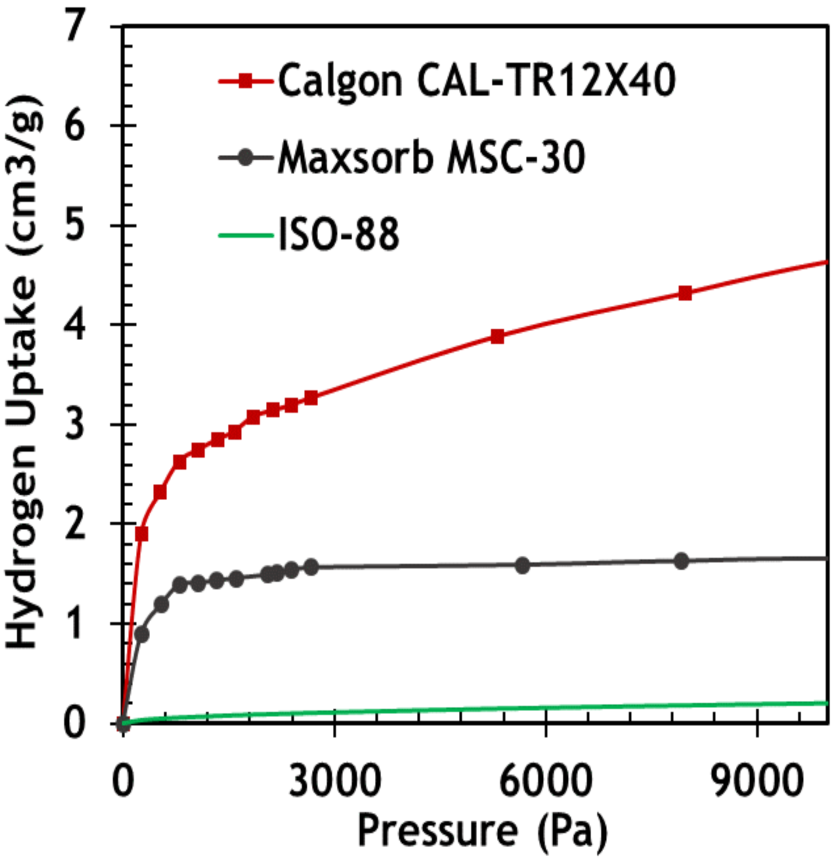
Tritium Removal from Liquid Salt Using Carbon Beds

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**

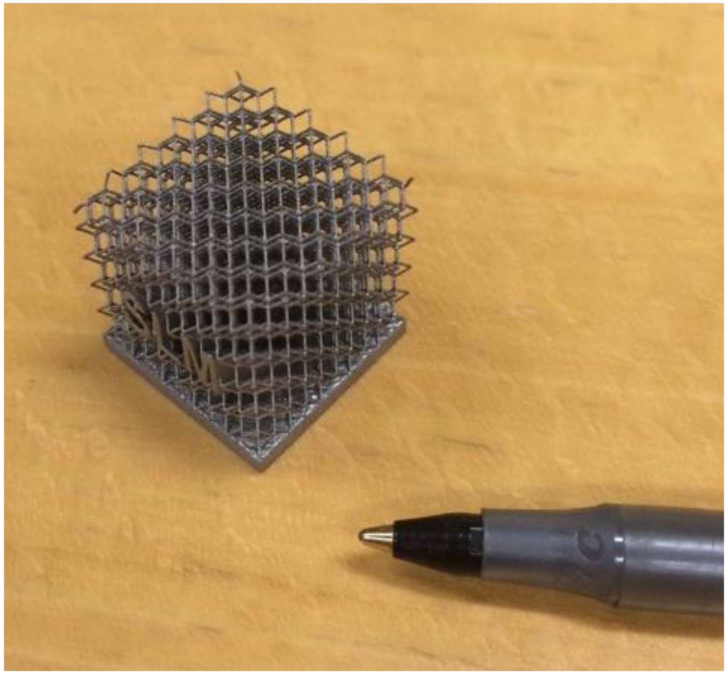
- **Outside the reactor core one can chose a carbon with high tritium sorption for a tritium removal bed (similar to ion exchange system in an LWR)**

- **Initial assessment suggests can control tritium levels in FHR with carbon bed external to the reactor core**



Systems Good for Tritium Removal in Clean Salts May Remove MSR Noble Metals

High Surface Area; Good Mass Transfer

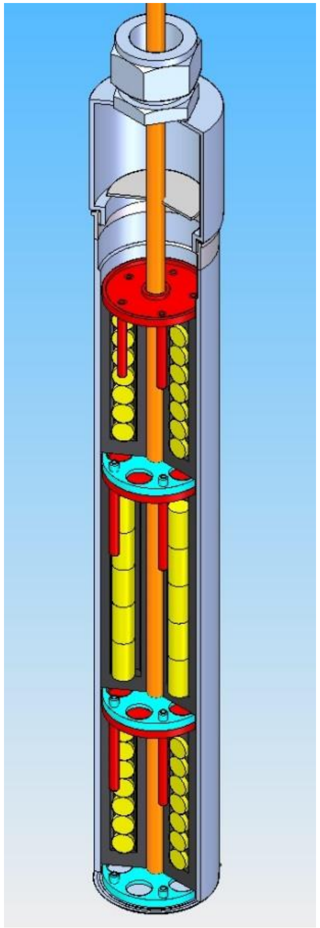


High-Surface-Area Additive Manufacture Adsorber Bed

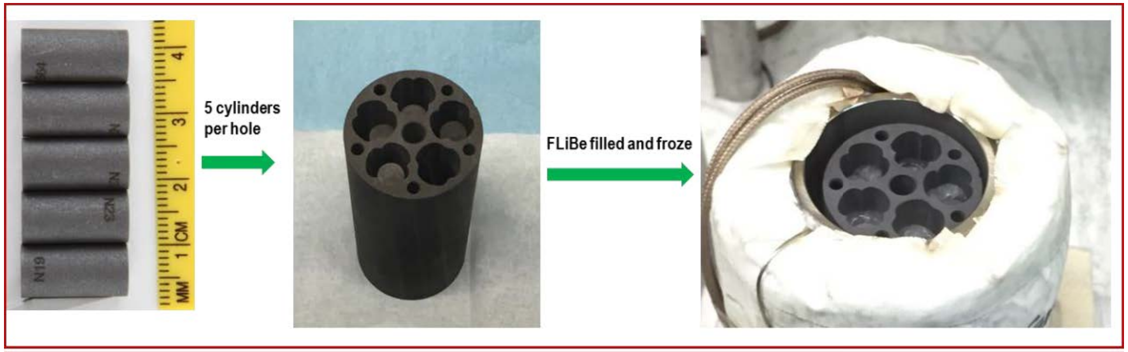
Platinum on High-Surface-Area Carbon (Commercial Catalyst for Hydrogenation Reactions)

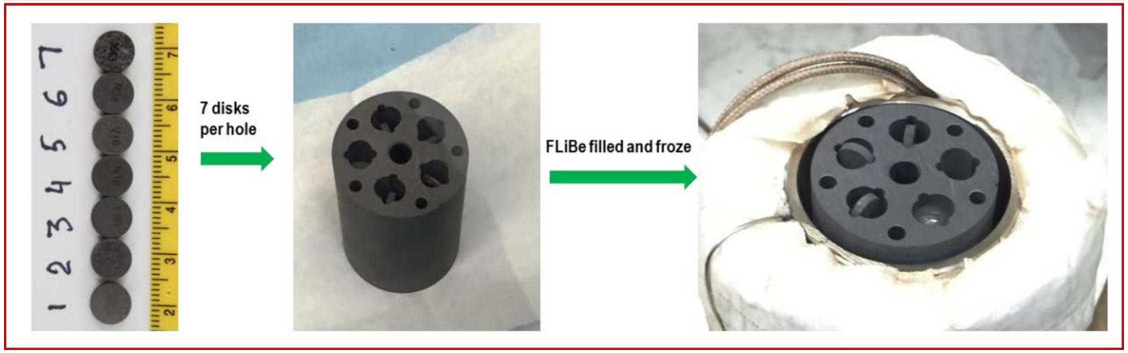
In-Reactor Materials Testing Underway for FHR

3rd FHR Irradiation in MITR (Fall 2016)



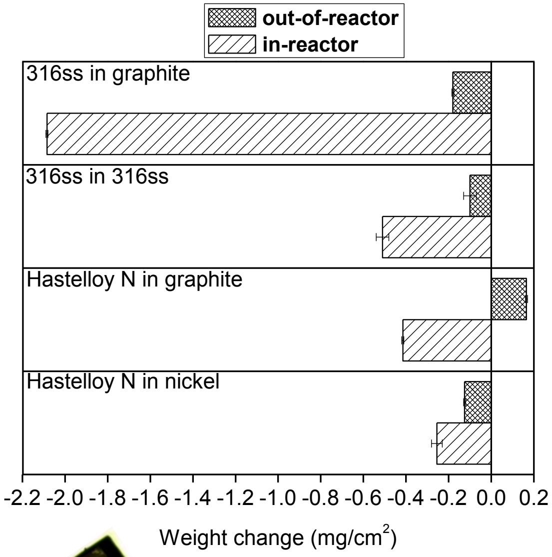
- 1000 hours at 700°C in enriched flibe
- Graphite and C/C specimens
  (previously irradiated SiC, 316SS, Hastelloy-N, TRISO)

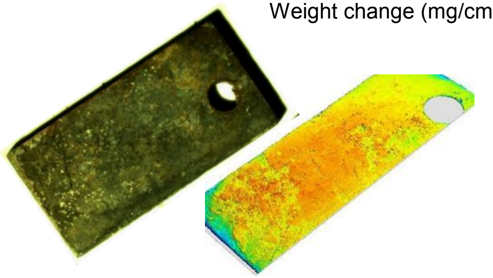




Post-Irradiation Exam of FHR Materials

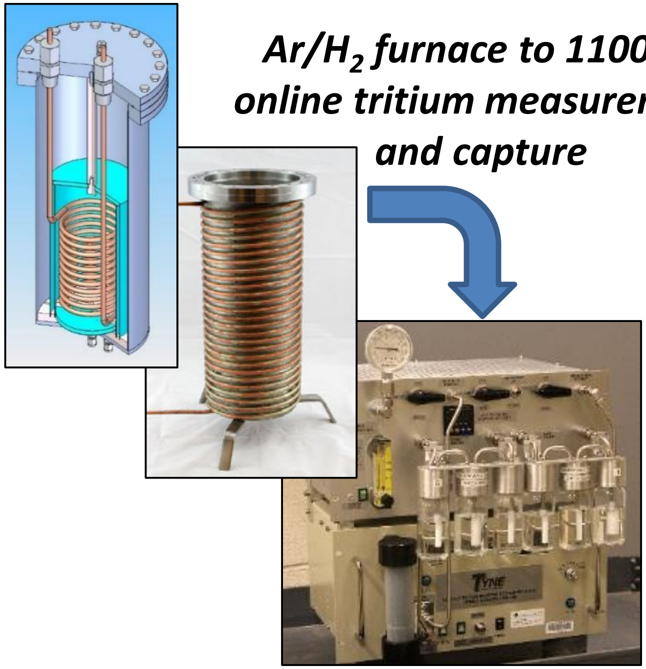
Irradiation-accelerated corrosion in flibe





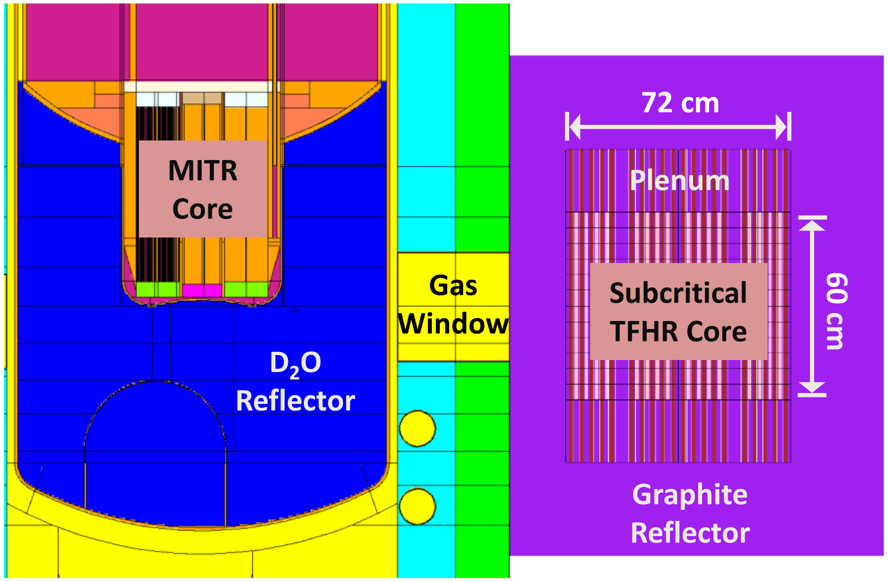
SS316 irradiated in flibe w/ IG-110U graphite crucible

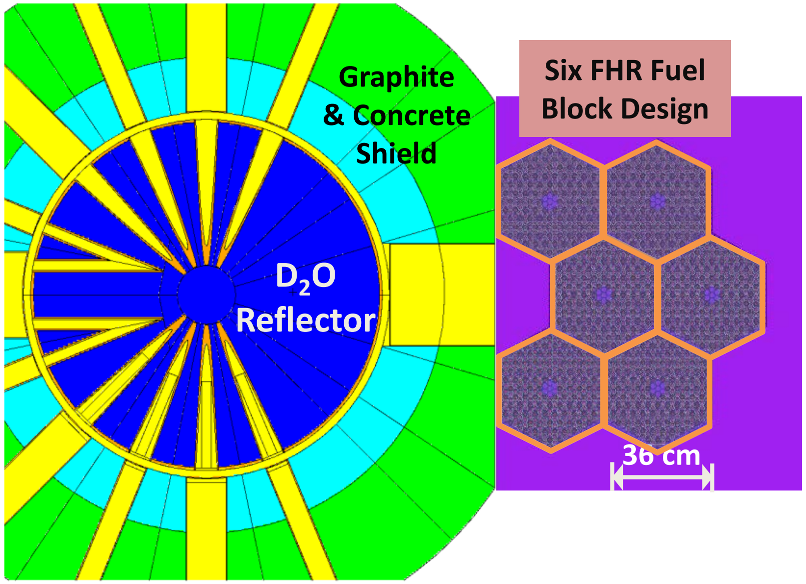
Desorption of tritium from irradiated components: Understanding tritium behavior in an FHR



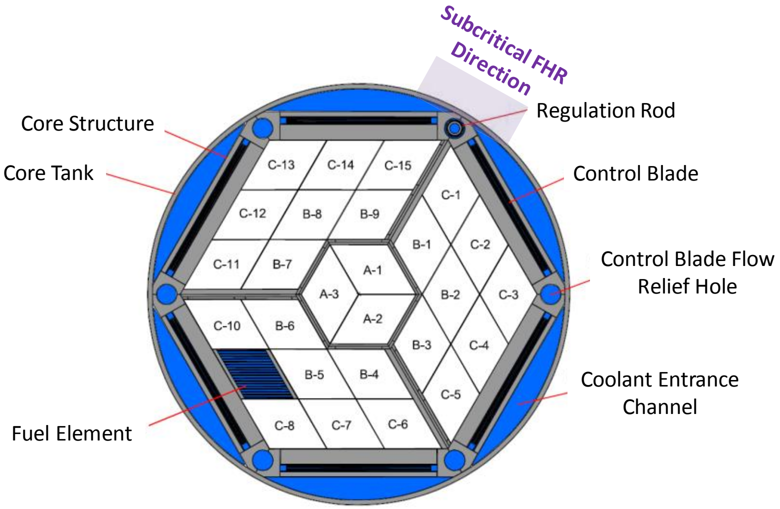
Ar/H$_2$ furnace to 1100°C; online tritium measurement and capture

Developing Reactor-Driven Subcritical FHR Demonstration Option using MITR

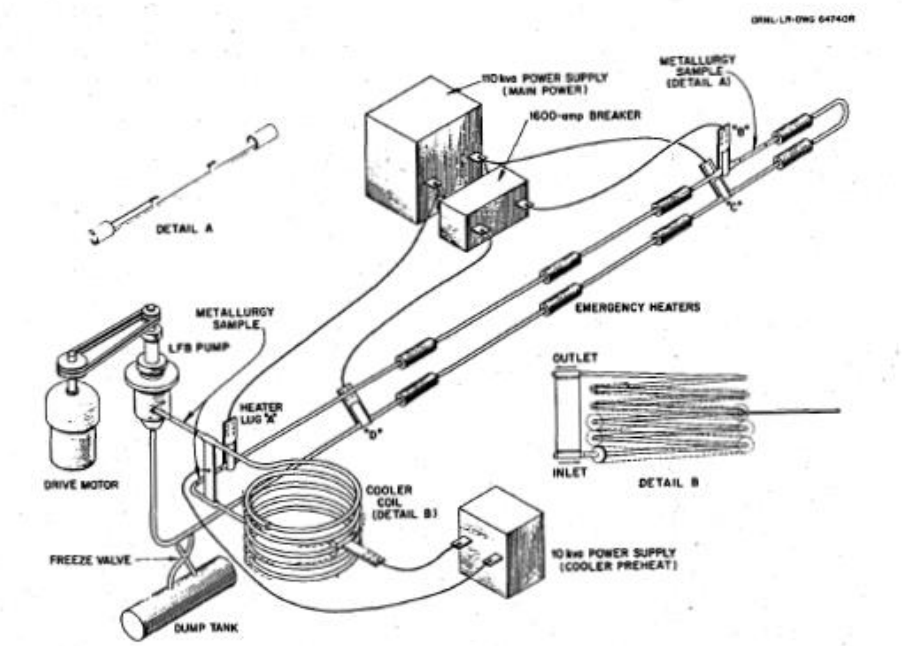


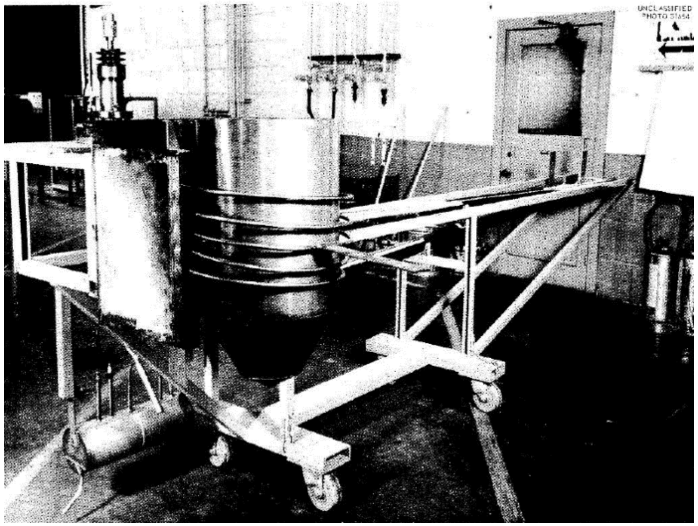


- An MIT Reactor Driven Subcritical System (RDSS) is designed to demonstrate FHR technology.
- A subcritical system with k_{src} of 0.98 is expected to generate 1 MW thermal power ~ 60% average FHR power density.
- Licensed as an MITR *experimental facility*, not as a new reactor.
- A cost-effective integrated experiment facility suitable for code validation, operation, maintenance, instuments, and component testing. The novel concept reduces risks for licensing a full-scale demonstration reactor.



Forced circulation flow loops provide the potential next step

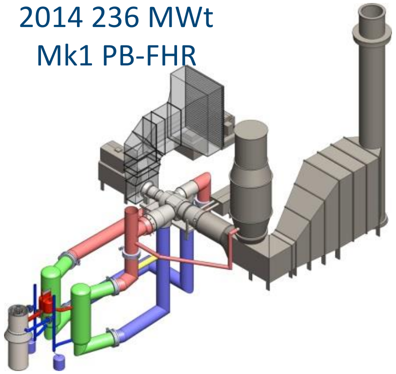




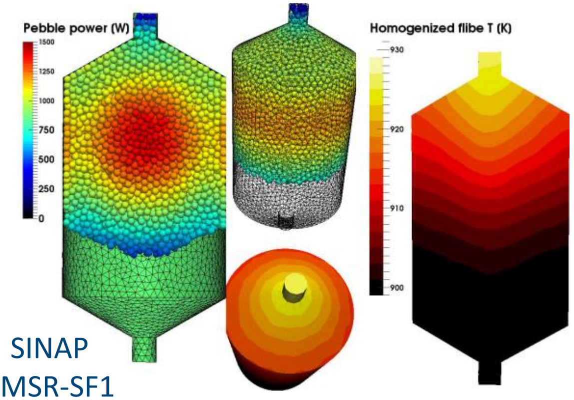
Inserted into reactor beam port

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

UC Berkeley FHR research focuses on thermal hydraulics, neutronics, safety and licensing



Conceptual Design Studies



SINAP
TMSR-SF1

Coupled neutronics and thermal hydraulics

Separate and integral effect tests

X-PREX Pebble Bed Tomography

Preliminary FHR Subsystems Definition, Functional
Requirement Definition, and LBE Identification
White Paper
UCBTH-12-001

Preliminary Fluoride-Salt-Cooled, High-Temperature Reactor (FHR) Subsystems Definition, Functional Requirement Definition, and Licensing Basis Event (LBE) Identification White Paper

March 2012
Revision 3

Integrated Research Project Workshop 1
Department of Nuclear Engineering
University of California, Berkeley

Panel Members
Todd Allen (University of Wisconsin, Madison)
Syd Ball (Oak Ridge National Laboratory)
Edward Blandford (Stanford University)
Tom Downar (University of Michigan)
George Flanagan (Oak Ridge National Laboratory)
Charles Forsberg (Massachusetts Institute of Technology)
Ehud Greenspan (University of California, Berkeley)
David Holcomb (Oak Ridge National Laboratory)
Lin-wen Hu (Massachusetts Institute of Technology)
Regis Matzie (*ret.* Westinghouse Electric Company)
John McGaha (*ret.* Entergy)
Dan Mears (Technology Insights)
Matthew Memmott (Westinghouse Electric Company)
Per Peterson (University of California, Berkeley)
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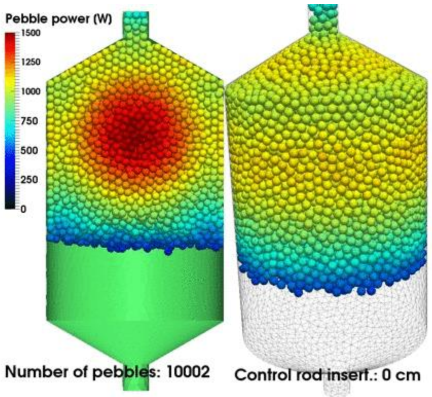
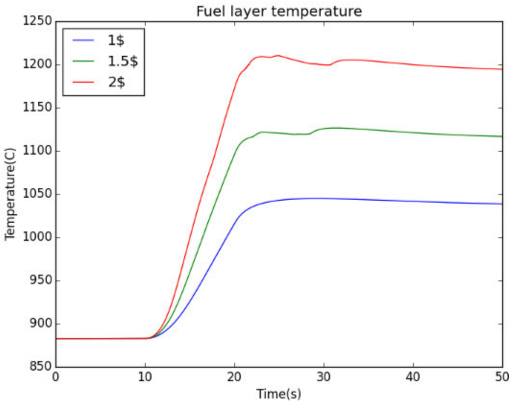
Organize Expert Workshops and White Papers

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

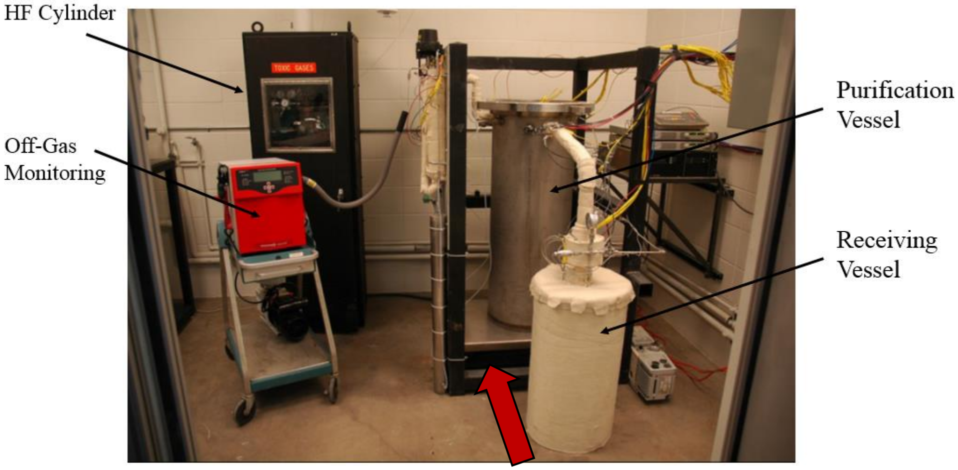


Coupled Serpent-OpenFOAM simulation of control rod insertion in TMSR-SF1

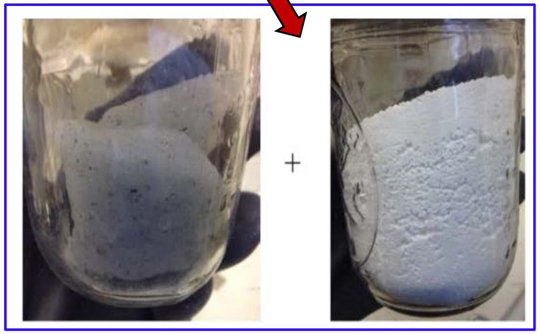


Fuel temperature after a prompt reactivity insertion in TMSR-SF1

University of Wisconsin - Production, Purification, and Reduction of FLiBe



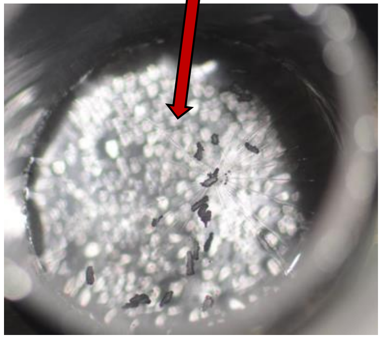
Raw Materials → $HF/H_2 \rightarrow H_2 \rightarrow$ Filtration, 600 – 630°C → UW Flibe → Beryllium and Filtration, 650°C → Beryllium Reduced



As-received BeF_2

As-received LiF

Melted FLiBe Salt



UW Materials Corrosion in FLiBe Salt at 700°C

Selected Tests Duplicated by MIT with In-Reactor Tests

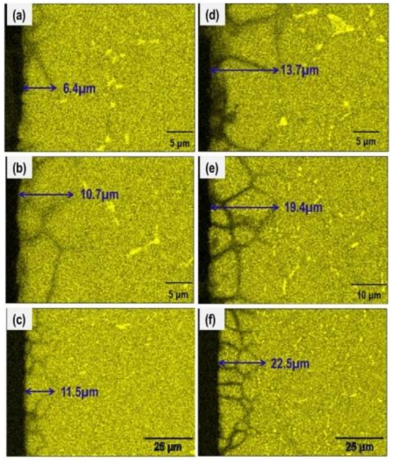
- **Materials Investigated:**
    - **316 stainless steel**
    - **Hastelloy-N**
    - **SiC-SiC composites**
    - **C-C composites**
    - **Graphite**
- **Additional Materials to be investigated:**
    - **SiC coated SiC-SiC**
    - **Diffusion bonded SiC-SiC**
    - **Mo-Hf-C alloy**
    - **W-ZrC cermet**
- **Comparison of corrosion behavior in Be-reduced and unreduced FLiBe**

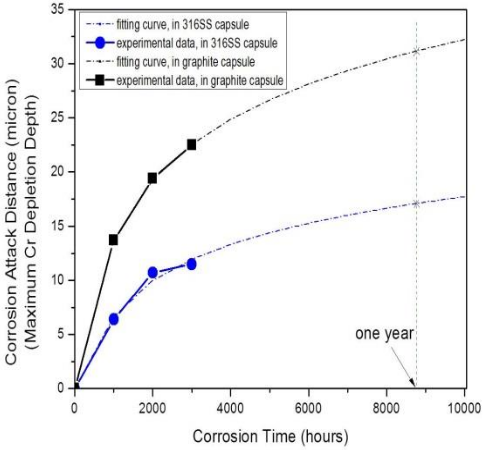
Six Compartment Graphite Crucible for Corrosion Tests

1000h

2000h

3000h



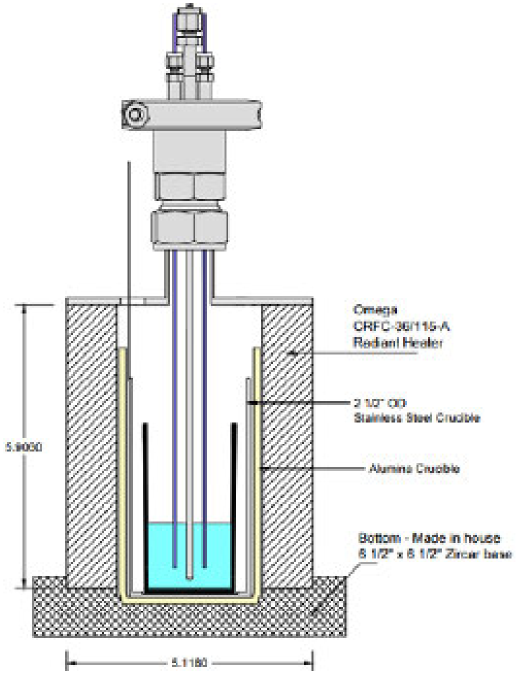


Results for 316 stainless steel tested up to 3000 hours

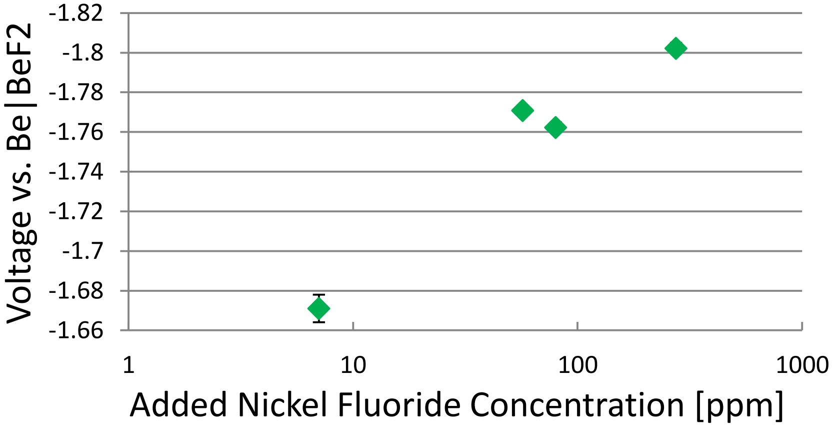
UW Electrochemistry for Redox Potential Measurements

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**



Schematic of redox measurement system (right) and Be/BeF_2 reference probe being developed from Afonichkin's (2009) below

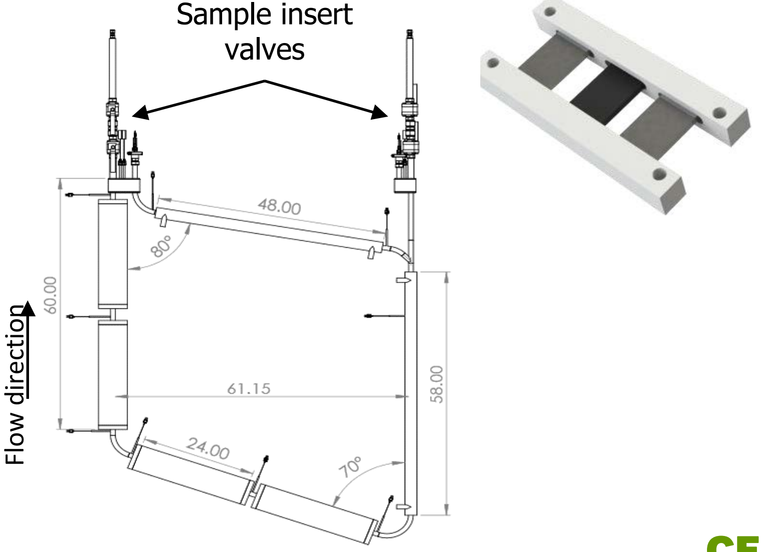


Measurement of the oxidizing effect of metal impurity fluorides on the FLiBe salt redox potential

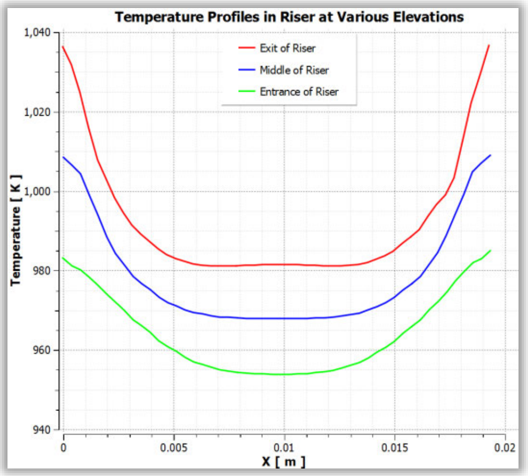
FliBe electrochemistry globe box

UW Natural Circulation Molten FLiBe Salt Flow Loop nearly complete

Enable Measuring Corrosion Under a Wider Set of Conditions



Flow-loop schematic and sample holder



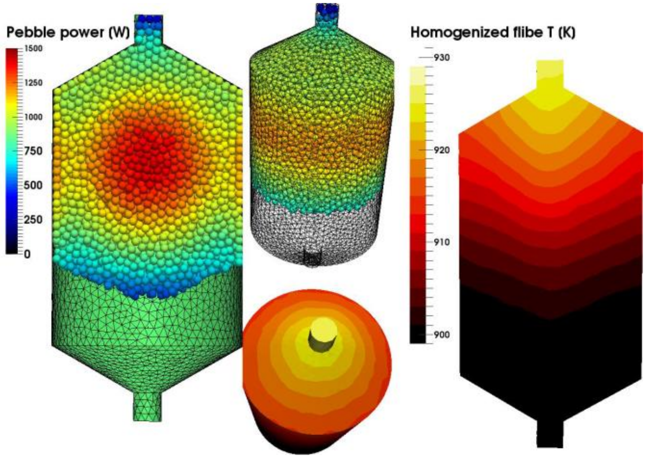
CFD predictions of temperature profiles at the bottom, middle, and top of the heated riser

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

- **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**

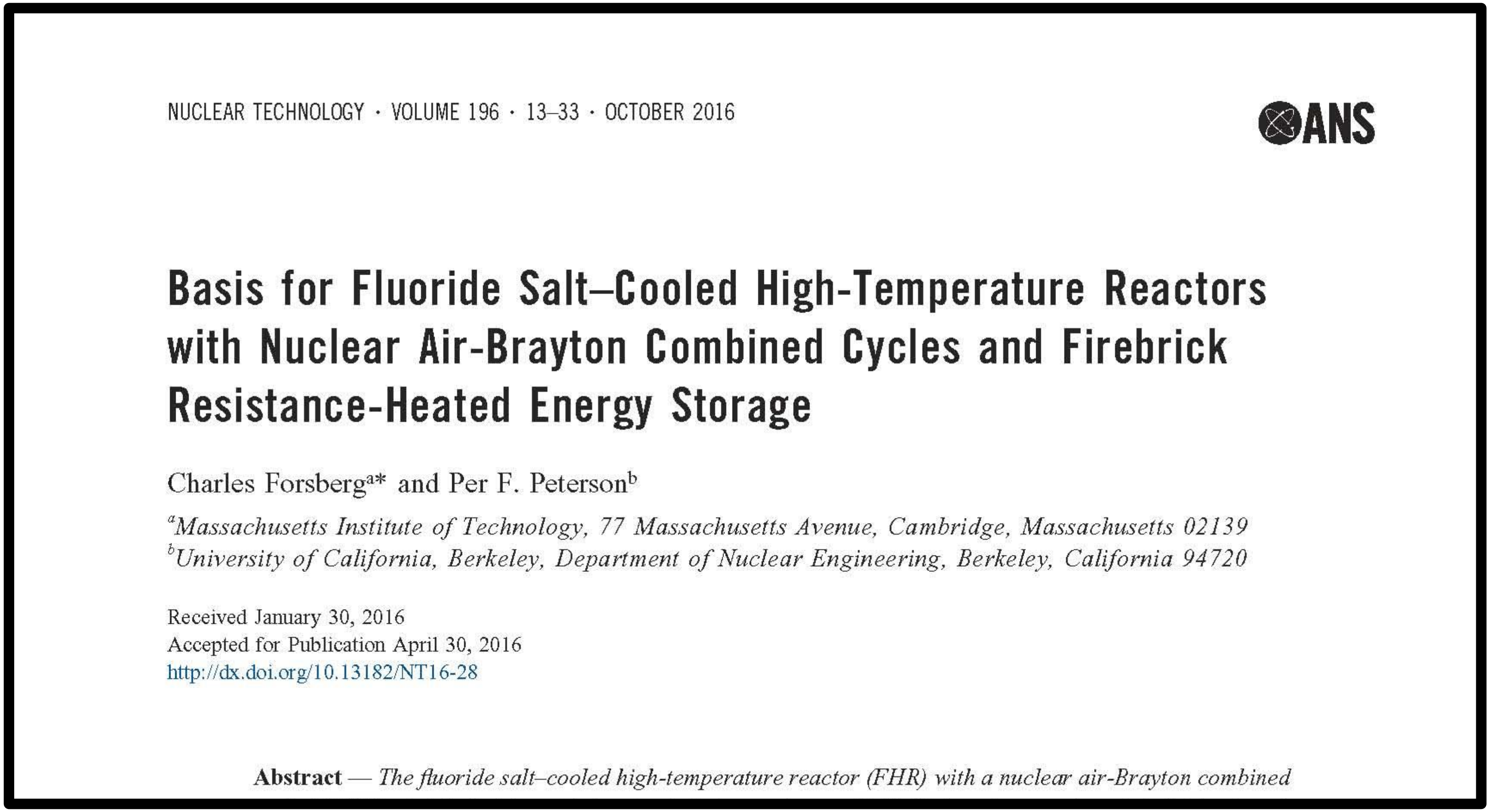
IRP-SINAP Interactions

- SINAP 10 MWt test reactor based on IRP design
- Multiple activities at multiple levels
  - Benchmarking
  - Participation in workshops
  - Joint work on tritium control strategie
    - 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

Recent Paper Summarizes Basis for FHR



NUCLEAR TECHNOLOGY · VOLUME 196 · 13–33 · OCTOBER 2016

Basis for Fluoride Salt–Cooled High-Temperature Reactors with Nuclear Air-Brayton Combined Cycles and Firebrick Resistance-Heated Energy Storage

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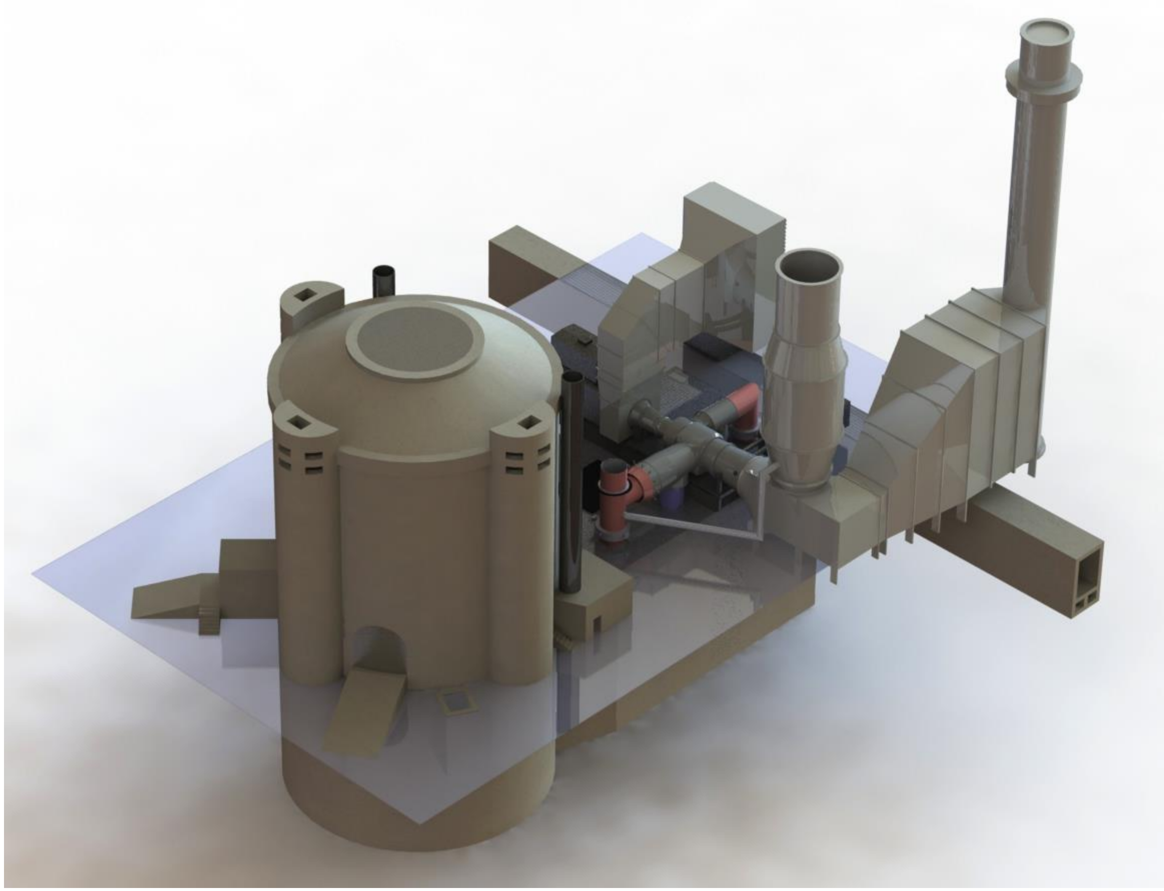
Abstract — *The fluoride salt–cooled high-temperature reactor (FHR) with a nuclear air-Brayton combined*

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.

http://web.mit.edu/nse/people/research/forsberg.html

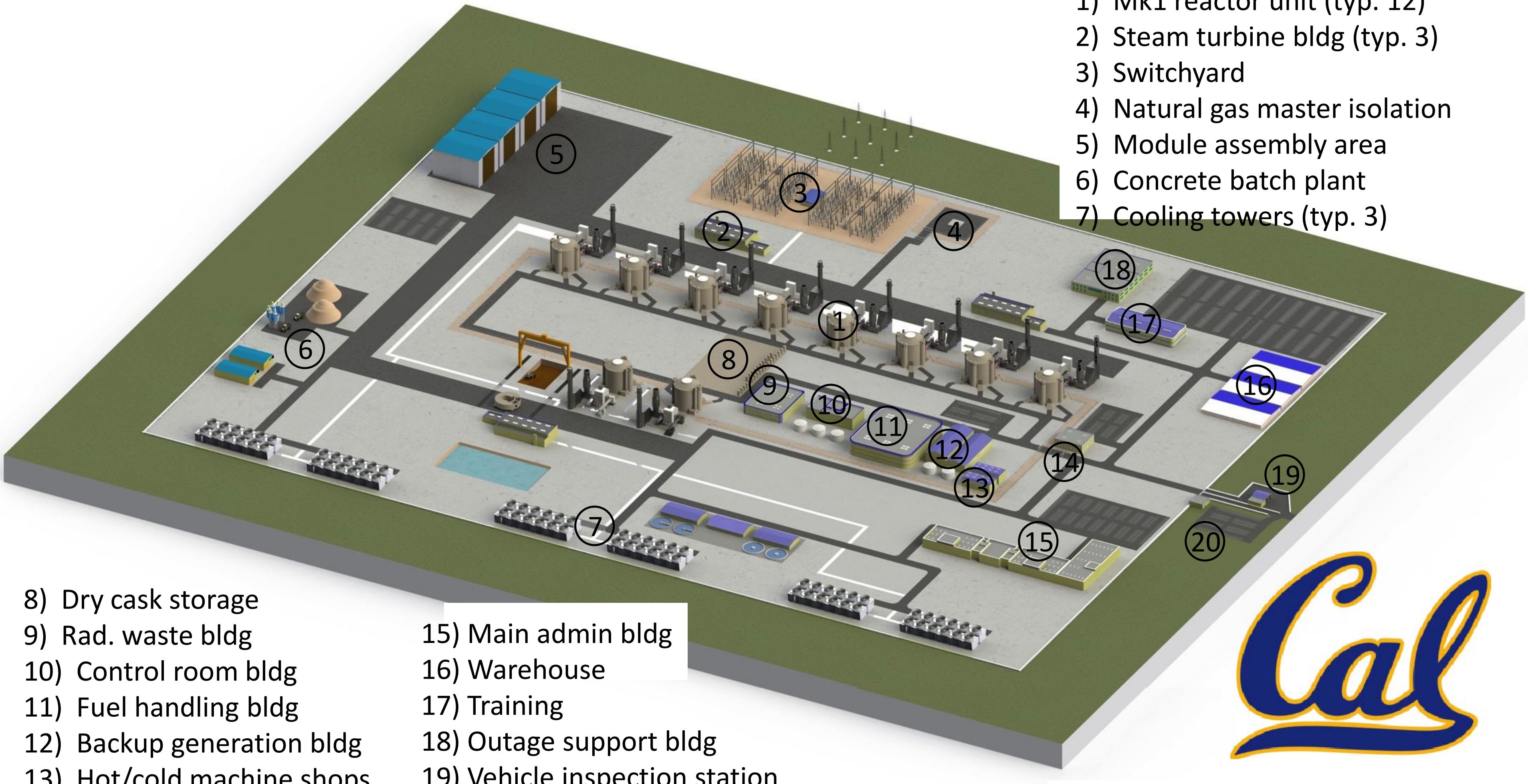
List of IRP-2 Tasks

- 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

Massachusetts Institute of Technology

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

	FHRs		PWR	HTGR	SFR
	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)

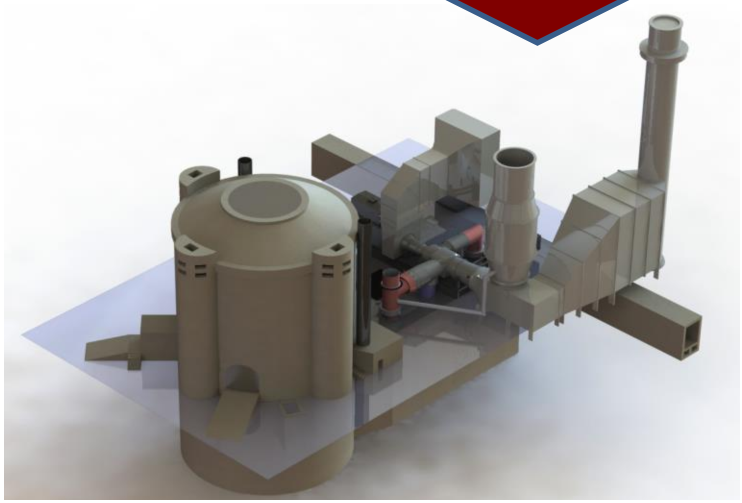
** Assumes a uranium tails assay of 0.003.

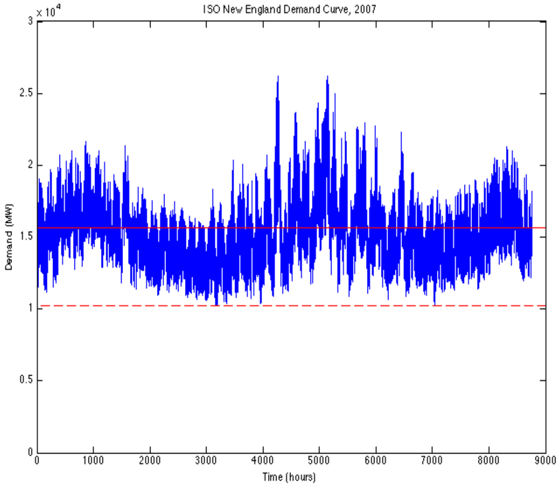
FHRs have unique safety characteristics for accidents resulting in long-term off-site land use restrictions from Cs-137

	FHRs	LWRs
Low Cs-137 inventory	~30 g/MWe	~105 g/MWe
High thermal margin to fuel damage	T_{damage} > 1800°C	T_{damage} ~ 830 – 1250°C
High solubility of cesium in coolant	CsF has high solubility	Cs forms volatile compounds
Intrinsic low pressure	High coolant boiling temperature and chemical stability	High vapor pressure at accident temperatures

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

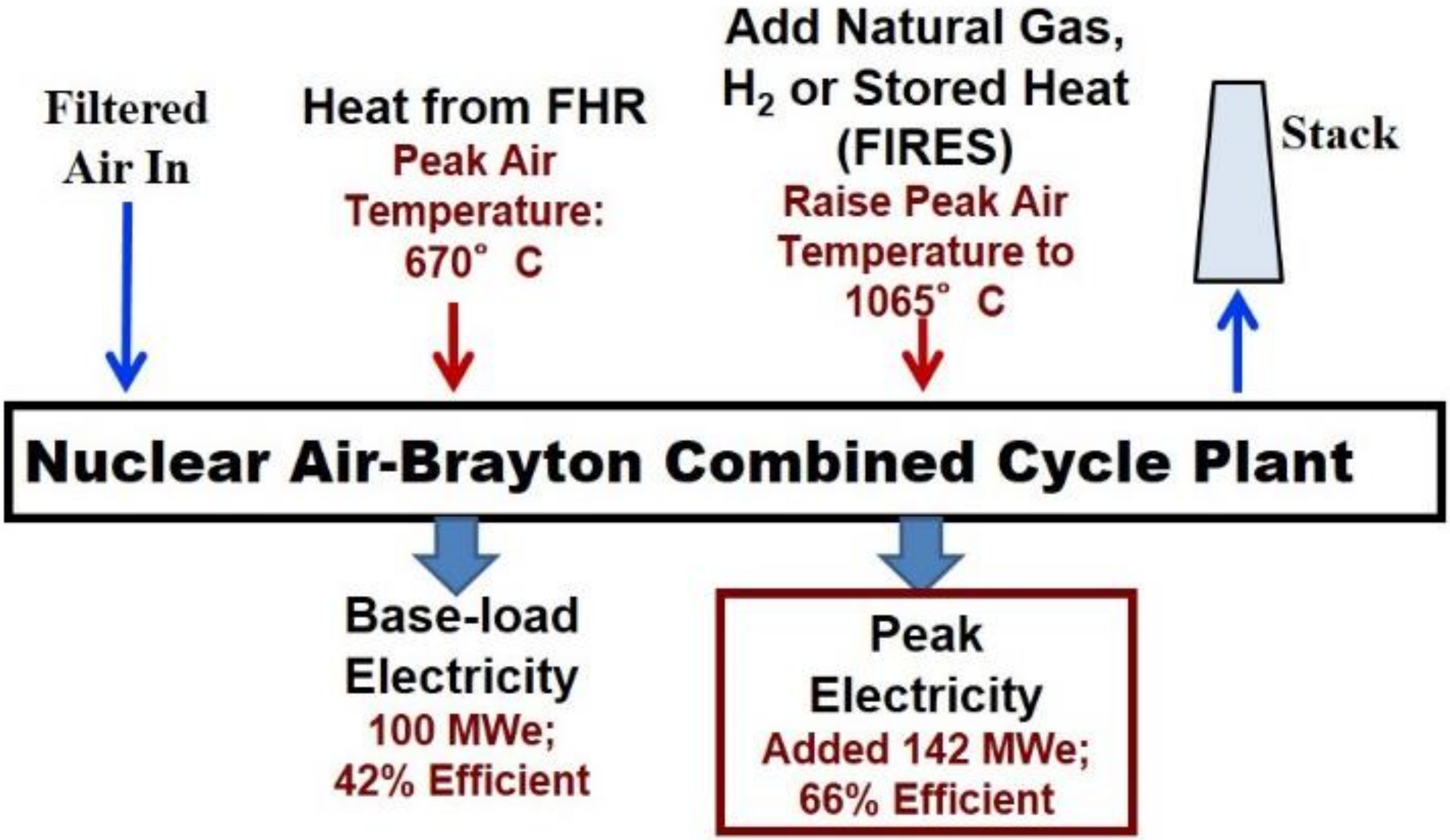
Variable Electricity And Steam

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, H_2 or Stored Heat (FIRES)

Raise Peak Air Temperature to 1065° C

Stack

Nuclear Air-Brayton Combined Cycle Plant

Base-load Electricity

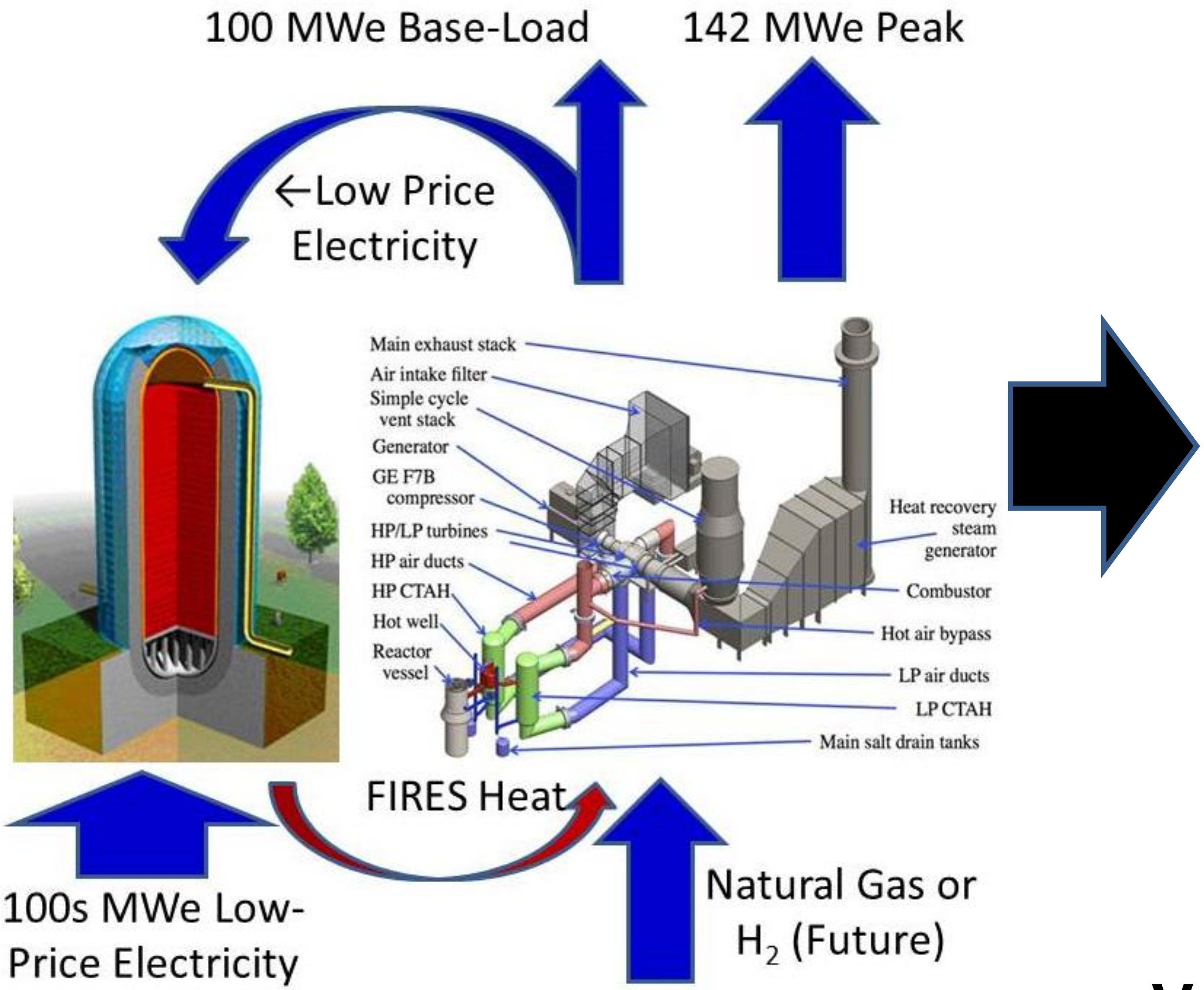
100 MWe; 42% Efficient

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

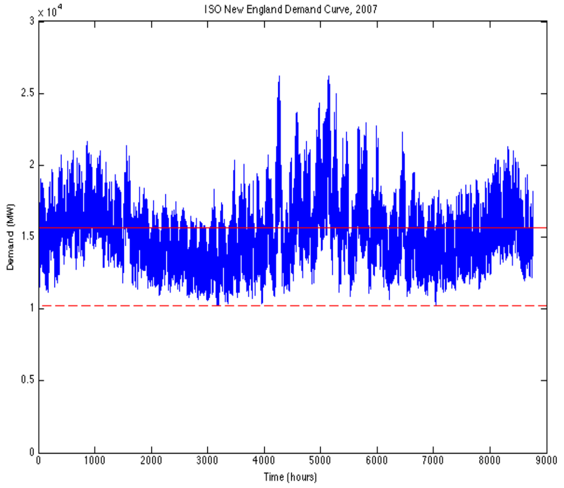
142 MWe Peak

←Low Price Electricity

FIRES Heat

100s MWe Low-Price Electricity

Natural Gas or H_2 (Future)



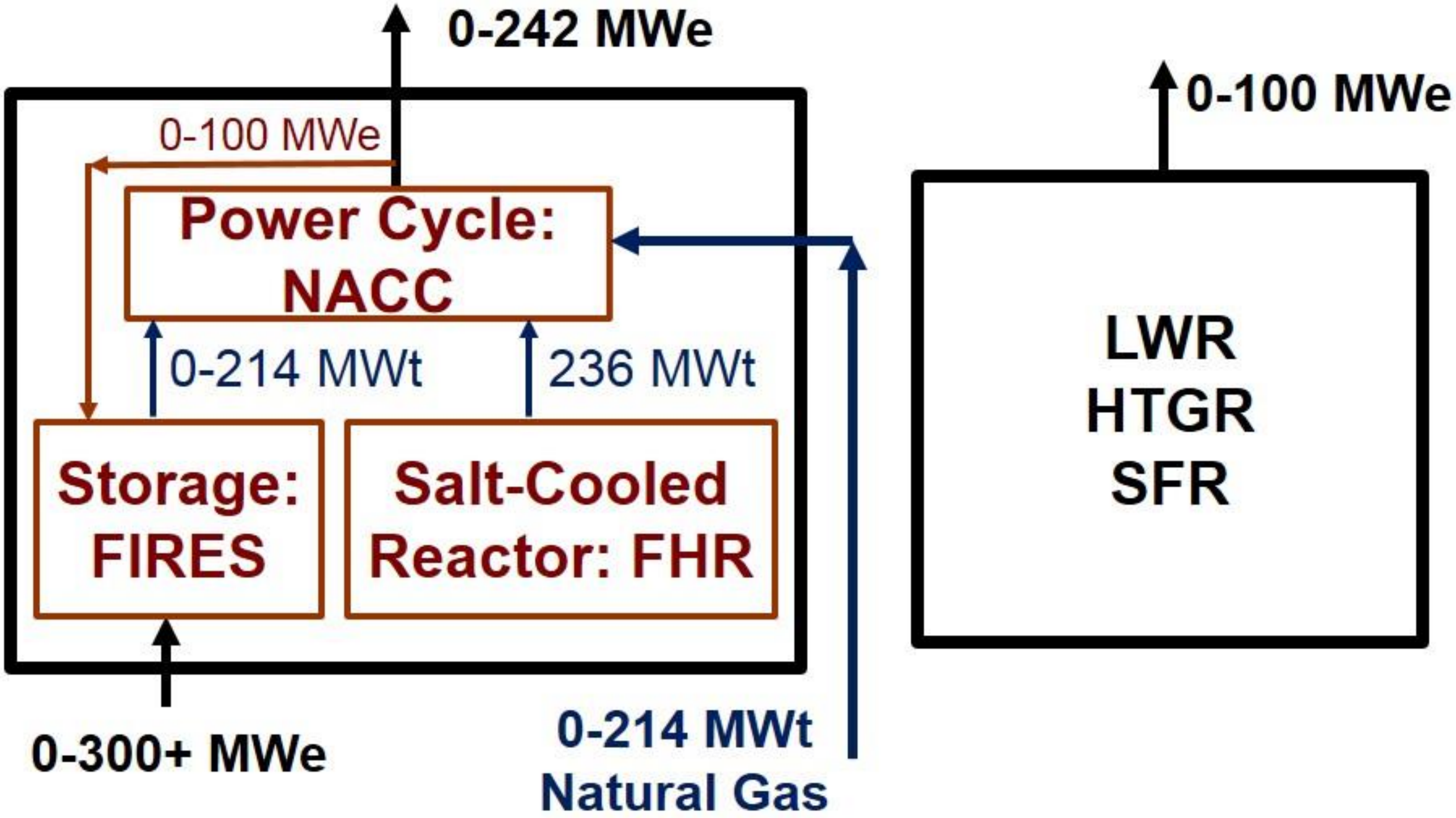
Base-Load Reactor, NACC and FIRES

Variable Electricity And Steam

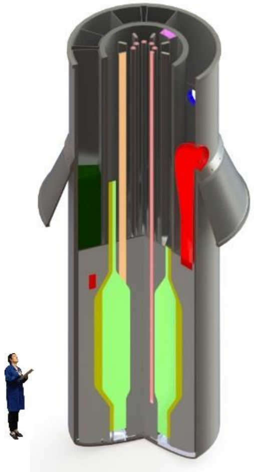
Economic Basis for FHR with NACC

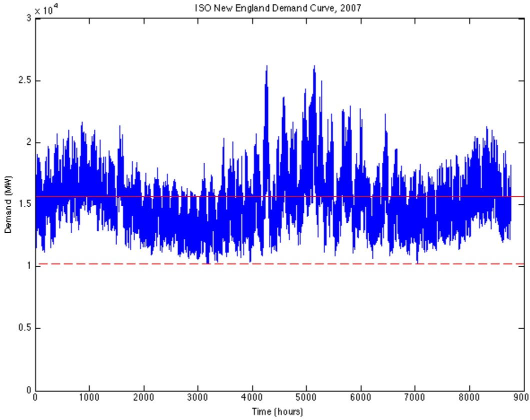
- Competing with Natural Gas (NG)
  - Base-load heat-to-electricity efficiency: 42%
  - Peak electricity with incremental NG gas efficiency: 66%
  - For peak electricity, more efficient than stand-alone NG plants (60%) and thus 50% increase in revenue over base-load-only nuclear plants after pay for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

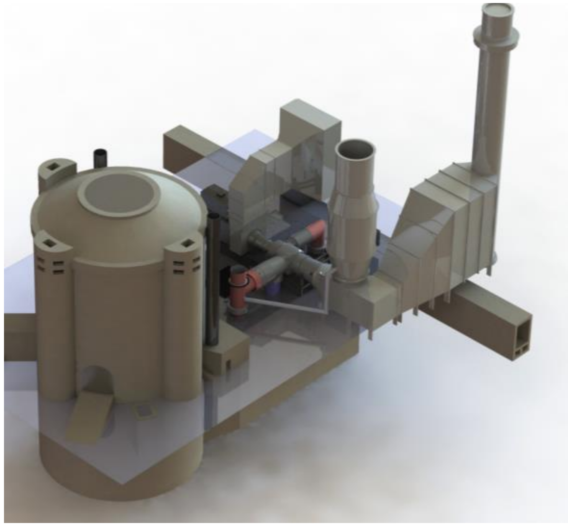
From the Grid Perspective, FHR/NACC/FIRES is a Second Class of Nuclear Power



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

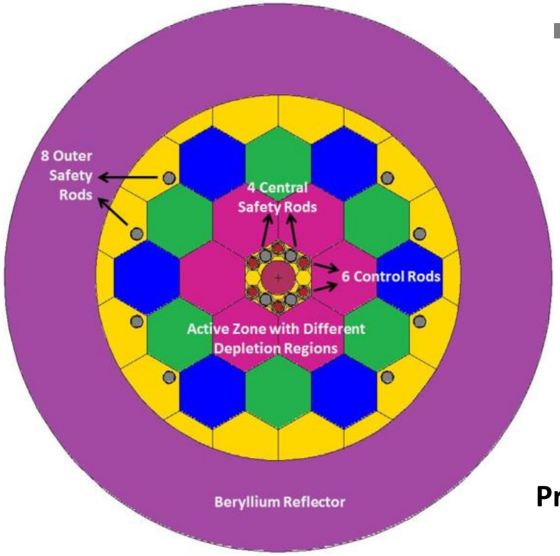
1. Operated a flibe laboratory since 2012: walk-in fume hood, and glove-boxes. Purification, salt-loop, salt transfers, and glove-box experiments.
2. Additional hazards associated with flibe handling: HF, F2, high temperature, voltage, challenges of salt transfer while ensuring purity.
3. Ensuring inert atmosphere goes hand in hand with containing Be

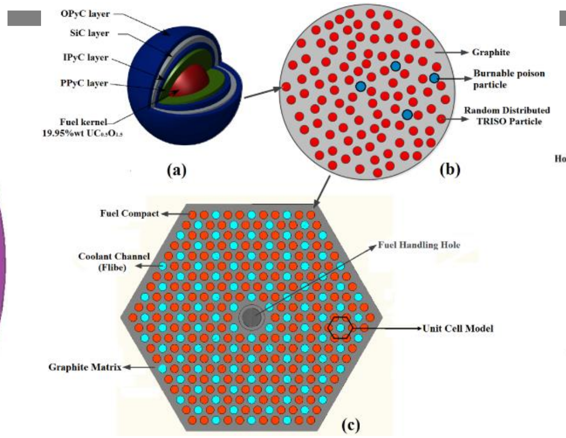
Flibe Researchers at UW. Jan 2016, next to Flibe Glove-box

Flibe Walk-in Fume-Hood With HF purification

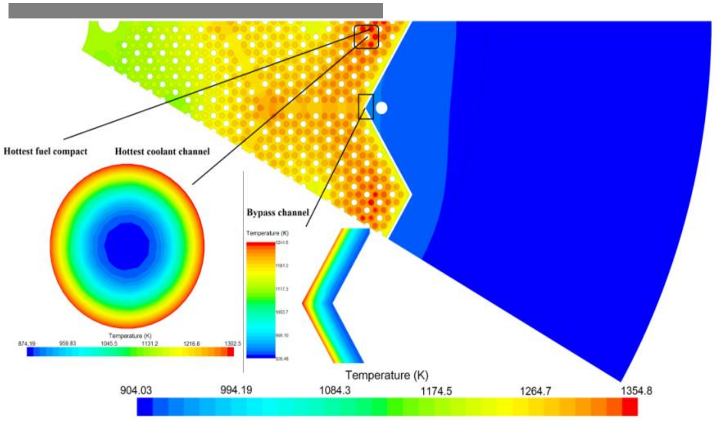
4. Air monitoring: Met OSHA PEL: 2 ug/m3 and Action Level: 0.2 ug/m3
5. Surface swipes housekeeping: 3 ug/100 cm2, general release: 0.2 ug/100 cm2. Occasional swipes above 0.2 ug/100 cm2 were followed by clean-up to ensure good housekeeping.
6. Better understanding of source term and particulate size 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)

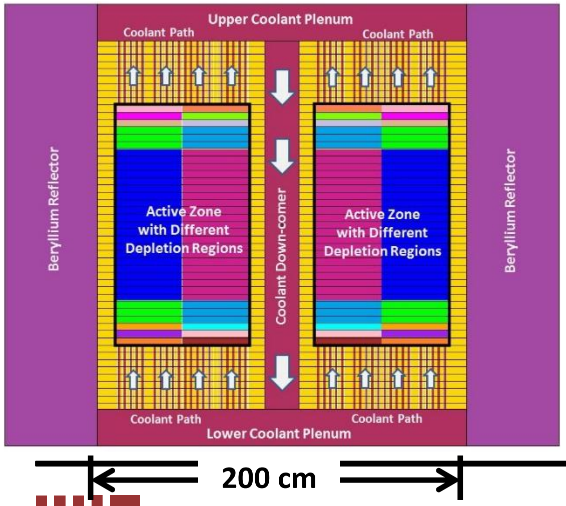




Prismatic assembly with TRISO fuel particles



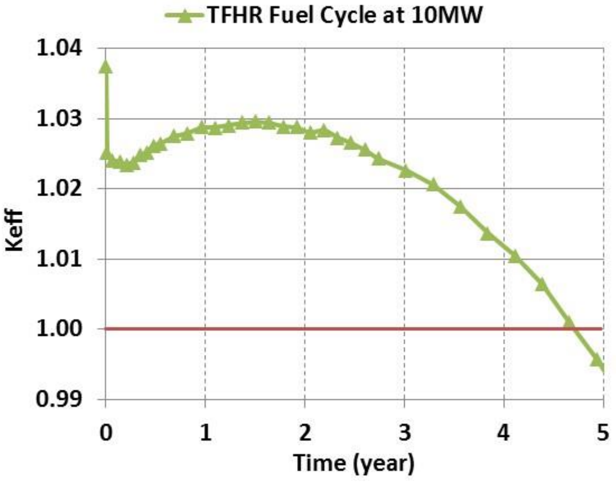
Full 3-dimensional CFD modeling



Design Features

- ➢ 10 MWth with ~ 5-yr fuel cycle
- ➢ Compact core ~ 2-m diameter
- ➢ Tranportable 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



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