Integrated Research Project: Salt Loop Irradiation

- Charles Forsberg (MIT)
- David M. Carpenter (MIT)
- Ayman I. Hawari (NCSU)
- Raluca O. Scarlat (UCB)
- and Kevin Robb (ORNL)

- 2021 Virtual Molten Salt Reactor Workshop
- October 12-13, 2021
- Salt Irradiation Session: Tuesday: 12:00 to 1:30

Charles Forsberg
cforsber@mit.edu

Massachusetts Institute of Technology

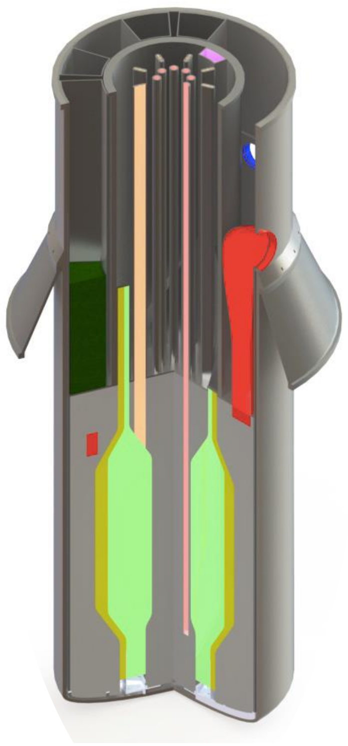
NC STATE UNIVERSITY

Berkeley UNIVERSITY OF CALIFORNIA

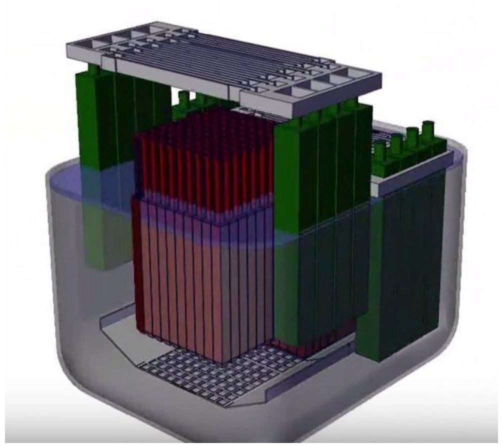
OAK RIDGE National Laboratory

Multiple Technologies Dependent on Salt Technology

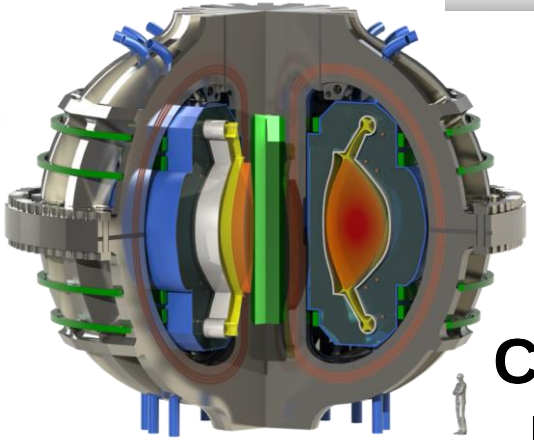
Clean Fluoride Salt Coolant



Kairos Power FHR



Moltex



Commonwealth Fusion ARC

Fuel in Fluoride Salt

MSR: Many variants

Molten Fluoride Salt Fast Reactor (Europe)

Fuel in Chloride Salt

Molten Chloride Fast Reactor

Fuel Salt in Tubes with clean salt coolant (Moltex)

Project Goals

- Design, build, and test a general-purpose instrumented molten-salt test loop at the MIT reactor where flowing salt is irradiated by neutrons with temperature variations around the loop to duplicate conditions in a salt reactor.
  - Provide near-term capability
  - Provide learning experience for future loops at ATR, VTR and university reactors
- Provide experimental data on tritium and fission product retention, diffusion and transport properties.
- Provide an experimental test bed for chemistry control, salt cleanup, tritium control and instrumentation
- Strong interactions with industry and national laboratories

Team Members and Responsibilities

Massachusetts Institute of Technology

- MIT. Design, build, and test a general-purpose instrumented molten-salt test loop at the MIT reactor

NC STATE UNIVERSITY

- NCSU. Develop, design, build and test off-gas sensor system capable of measuring tritium, fission products and actinides

Berkeley UNIVERSITY OF CALIFORNIA

- University of California at Berkeley: Develop, design and build instrumentation for measurement and control of redox salt chemistry

OAK RIDGE National Laboratory

- Oak Ridge National Laboratory. Supporting Role

Massachusetts Institute of Technology

Department of Nuclear Science and Engineering

MIT Reactor Laboratory

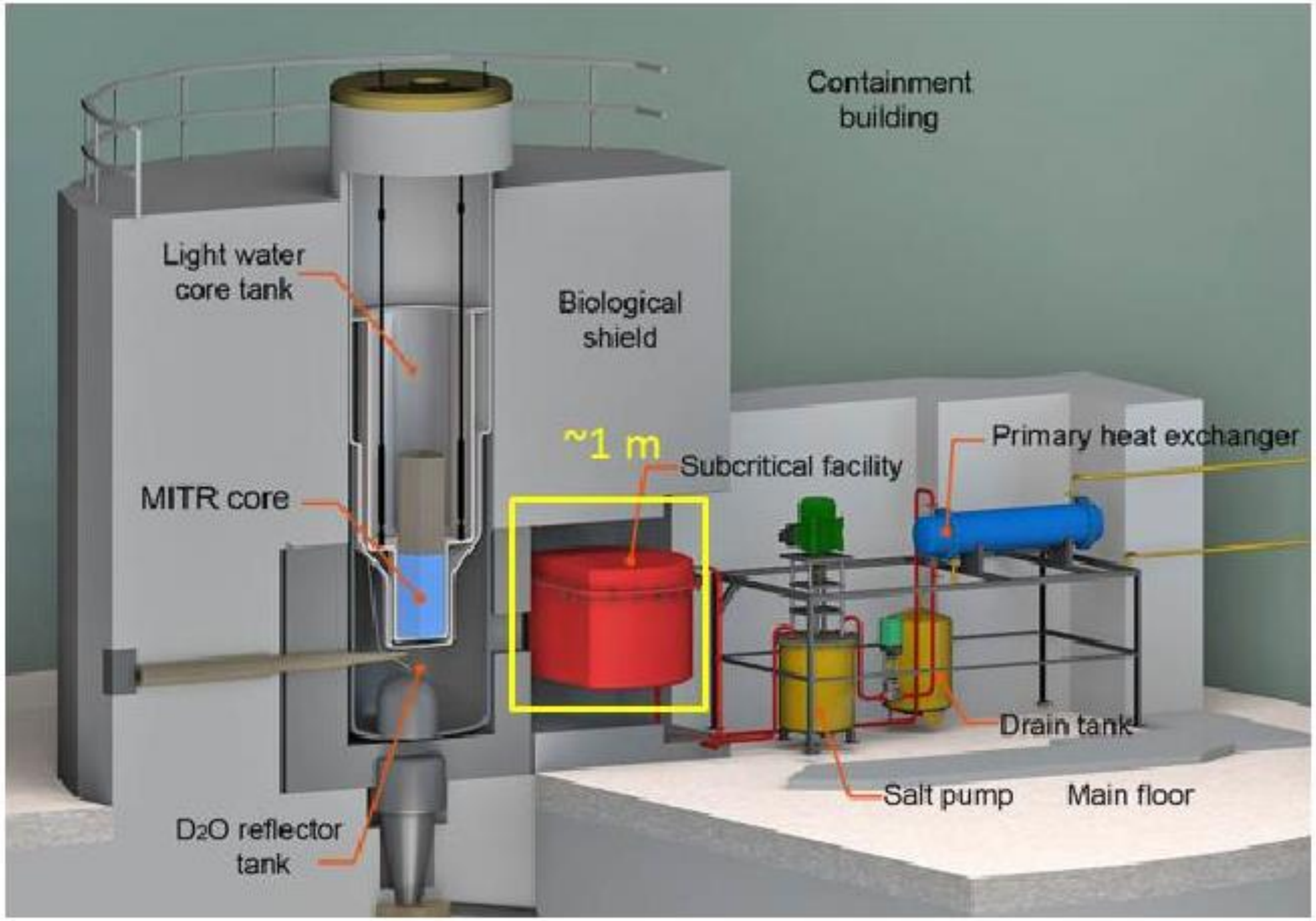
C. W. Forsberg, D. Carpenter, G. Zheng,

G. Su and N. Cetiner

MIT Has Initiated Design and Construction of a Salt Loop at MIT Reactor

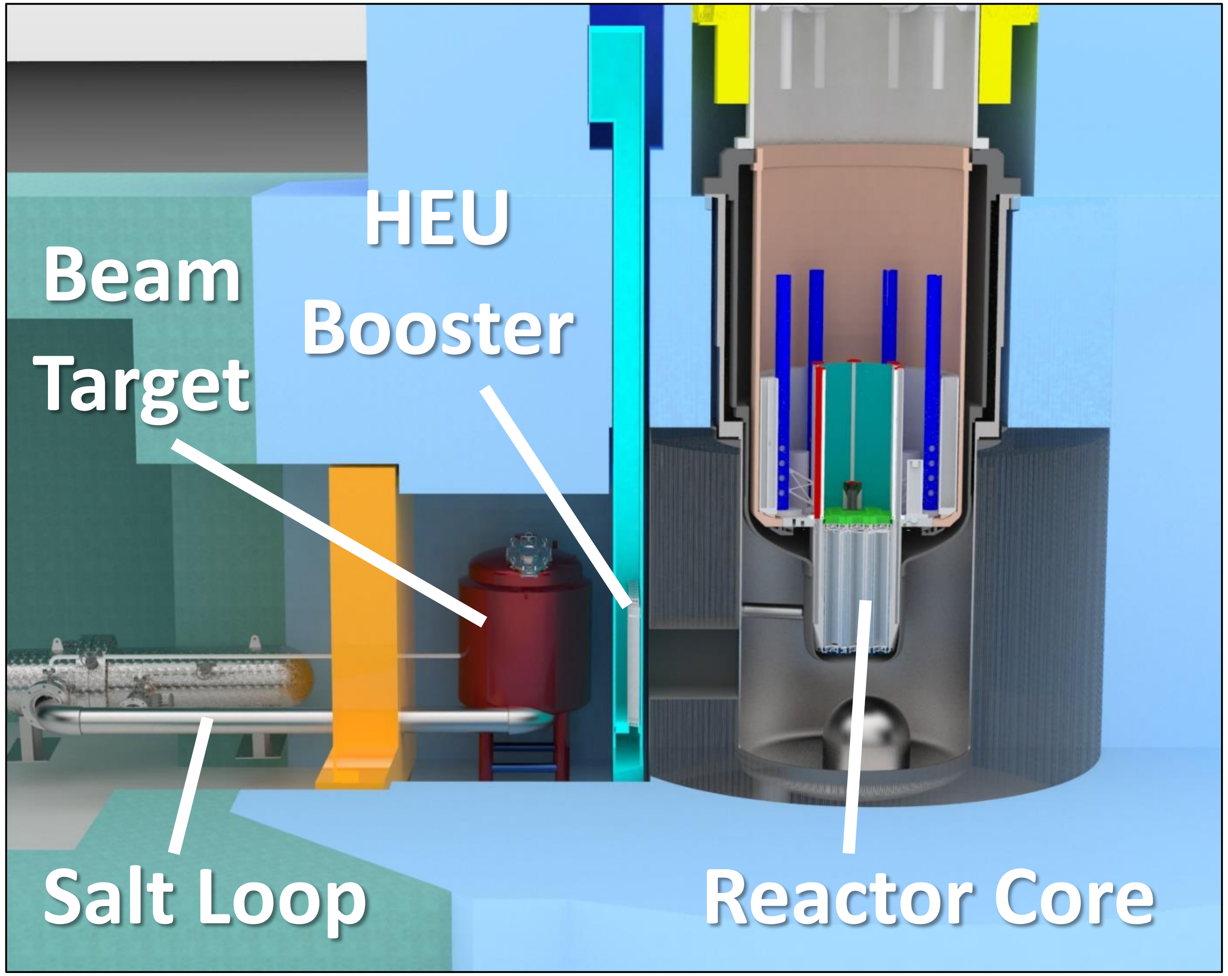
- MIT reactor: 6 Megawatts

- Forced circulation salt loop, heat and cool
  - High-temperature
  - Fully instrumented
  - 3-year project (DOE IRP)



MIT Facility Enables Loop Design with Fissile Material

- Loop outside the reactor tank that partly decouples reactor neutronics from loop
- Avoids large feedback effects and enables use of fissile materials in loop
- Can adjust fissile and lithium-6 content of salt to obtain desired salt behavior



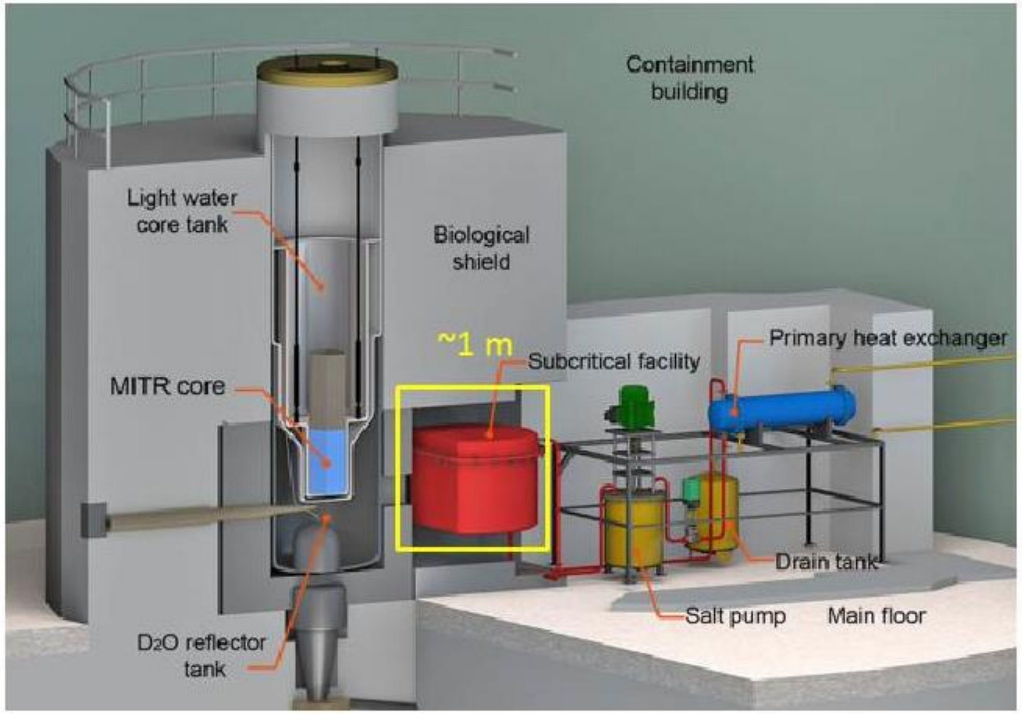
Preliminary Reactor Loop Parameters

Parameters	Value	Description
Main loop dimension H×W×D [m]	2.1 × 1.2 × 0.8	More compact than most of the existing salt test loops in US
Main loop material	316ss	Minor parts can be graphite or Inconel
Main tubing size [mm, inch]	12.7, 0.5	0.5 inch OD with 0.049 inch wall thickness
Salt constituent	flinak and flibe	Use flinak for loop shakedown; use flibe for the neutron irradiation test
Salt tank vol. [L]	~10	Can be further adjusted
Operating temperature [°C]	Up to 700	Steady state; short transients to 750 °C
Salt velocity [m/s]	0.1 - 2	Cover the PB-FHR Mk1 and KP-FHR design range
Temperature gradient [°C]	Up to 100	Depends on the salt flow rate
Continuous operation [hr]	Up to 1000	Progressively upgrade to 1000

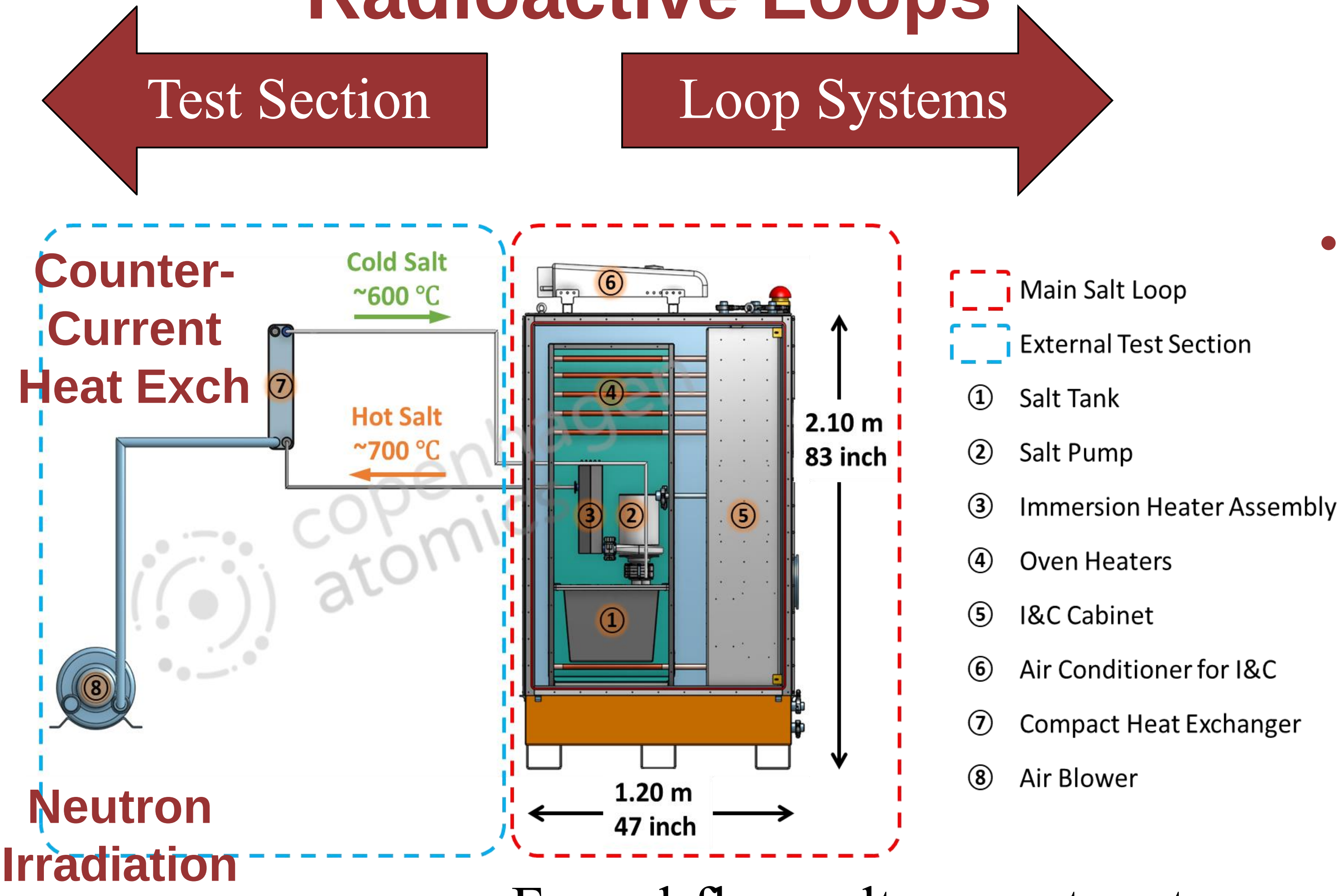
MIT Developed Design Specifications and Technical Requirements for Two Forced-Flow Salt Loops

Air test loop to non-radioactive loop with Flinak salt for learning and testing options

Neutron irradiation loop with Flibe salt and experiments in the M³ Facility



Forced Flow Loop Design for Non-Radioactive and Radioactive Loops



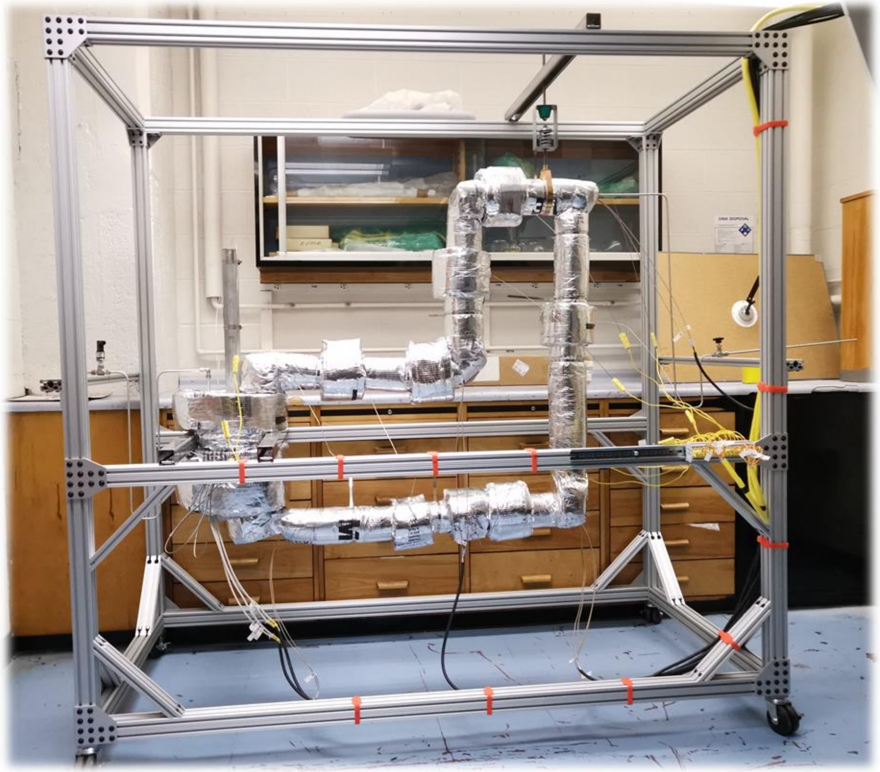
Forced-flow salt support systems from Copenhagen Atomics

- **Design has two sections**
  - Self-contained forced-flow salt system (right)
  - Reconfigurable test section
- **Test sections will be designed to achieve various objectives**
  - Coupon, component, and sensor testing
  - Thermal gradients
  - Irradiation target
  - Tritium removal
  - Chemistry monitoring & control

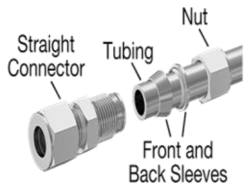
Dry Test Facility

- Filled with low-pressure Argon
- Currently in shakedown testing at low temperature

Initial testing focusing on insulation and candidate fittings

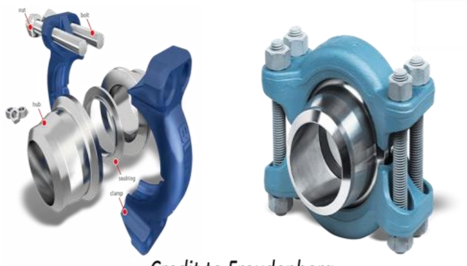


❑Swagelock



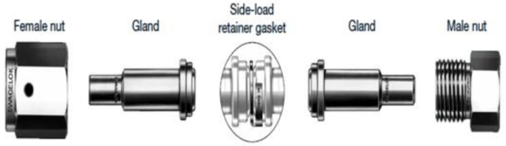
Credit to Swagelock and McMaster Carr

❑Clamp connecter (Grayloc type)



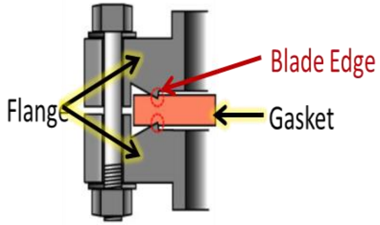
Credit to Freudenberg

❑VCR connector



Credit to Swagelock

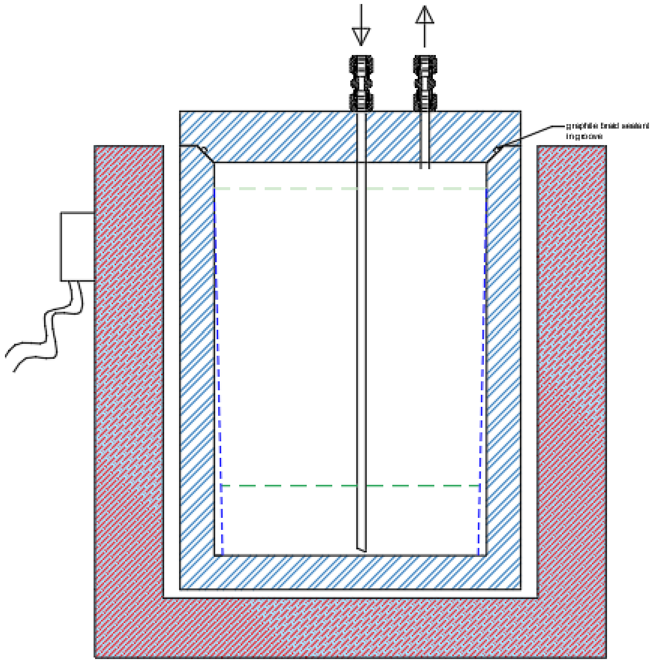
❑CF or KF flange



Credit to sev-vacuum

Building Supporting Salt Handling Systems

Expanding our existing capability (3 salt furnaces in 2 glove boxes, one for clean and the other irradiated salts) with larger furnaces and crucibles



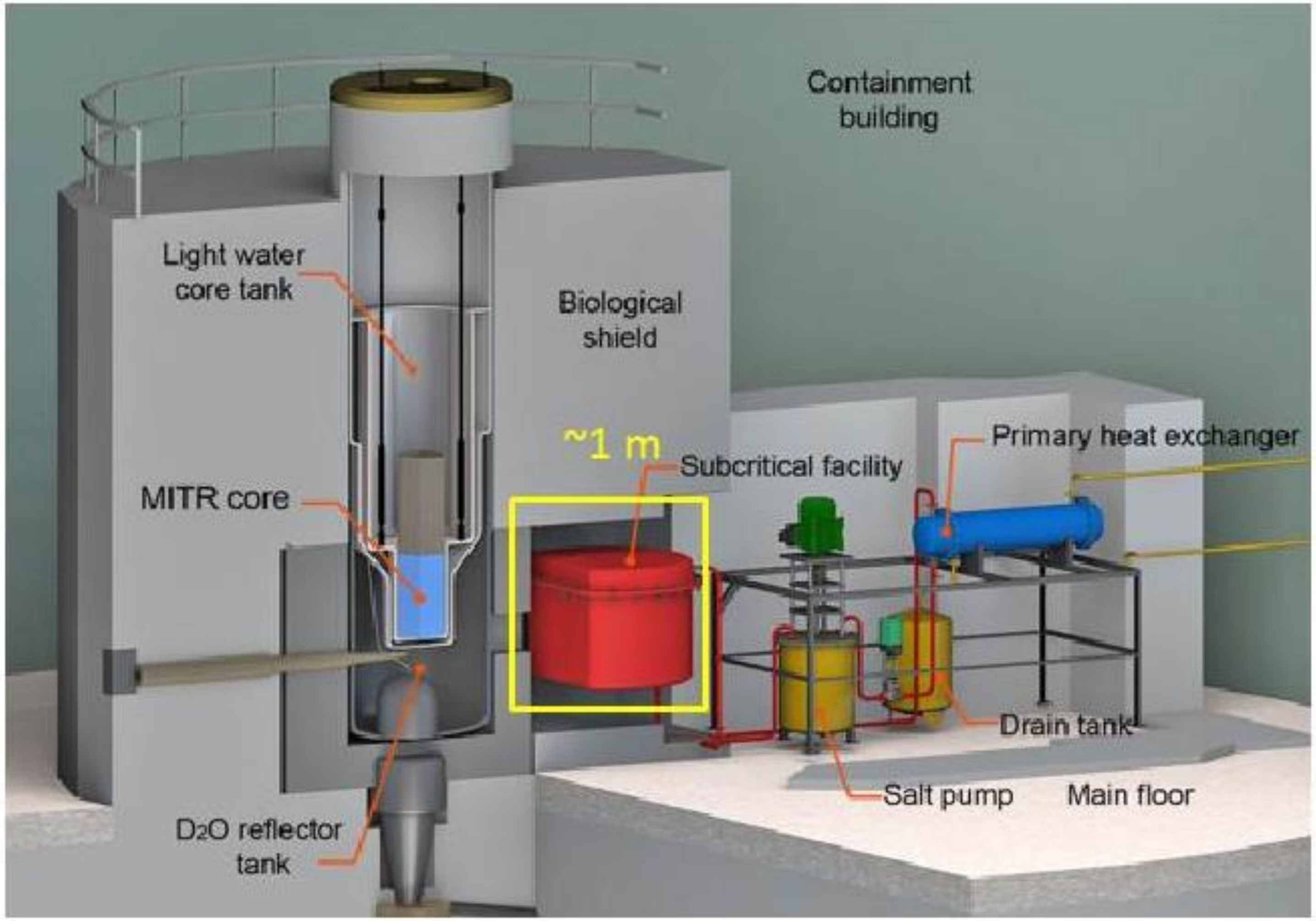
Graphite container with CVD SiC coating on inner and outer surface, ~5kg FLiNaK/batch
Vendor: Ibiden USA

Cylindrical ceramic furnace for heating above container
Vendor: DS Fiber Cooperation

New furnace received

Work Underway to Prepare M^3 Reactor Facility

- Removal and relocation of previous experiments and support equipment
- Repairs to neutron shutter system
- Allocation of additional power, cooling, and ventilation
- Planning for removal of high-activity components (neutron filters) and reloading of HEU booster elements



North Carolina State University

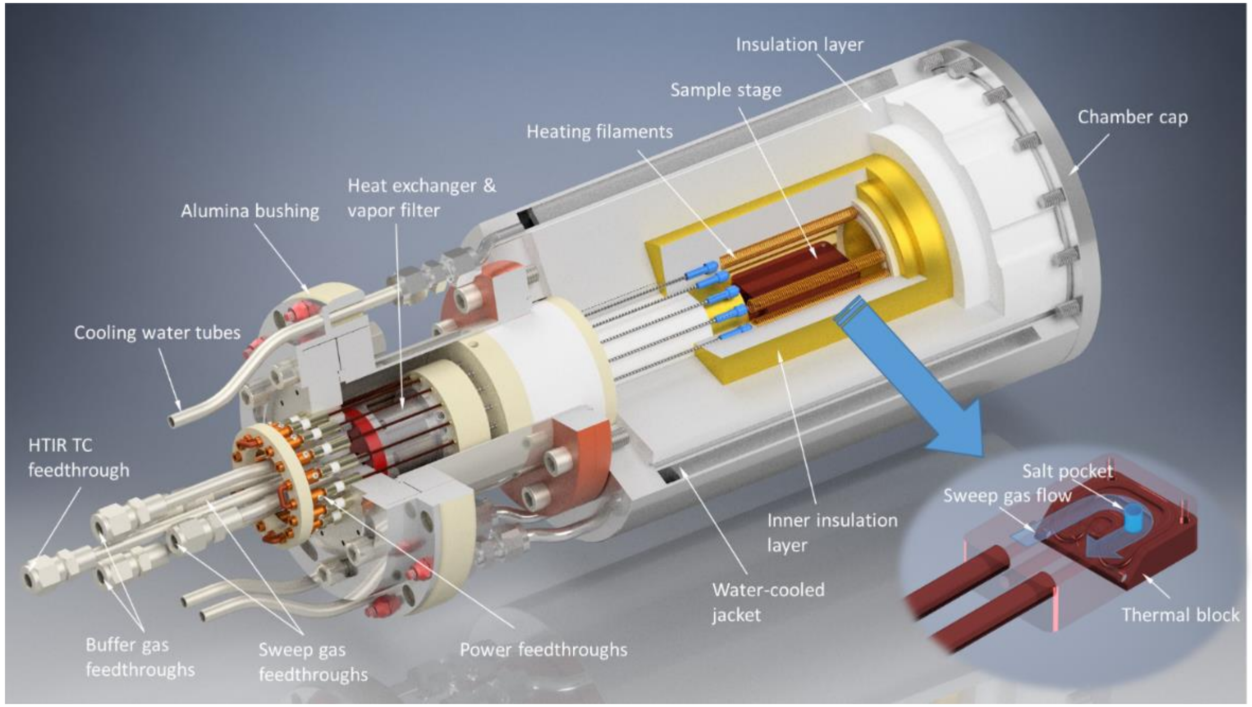
Nuclear Reactor Program, Department of Nuclear Engineering

A. I. Hawari, S. Lassell, M. Liu

Detailed Presentation by NCSU Later in This Session

North Carolina State University is Developing an Off-gas Monitoring System

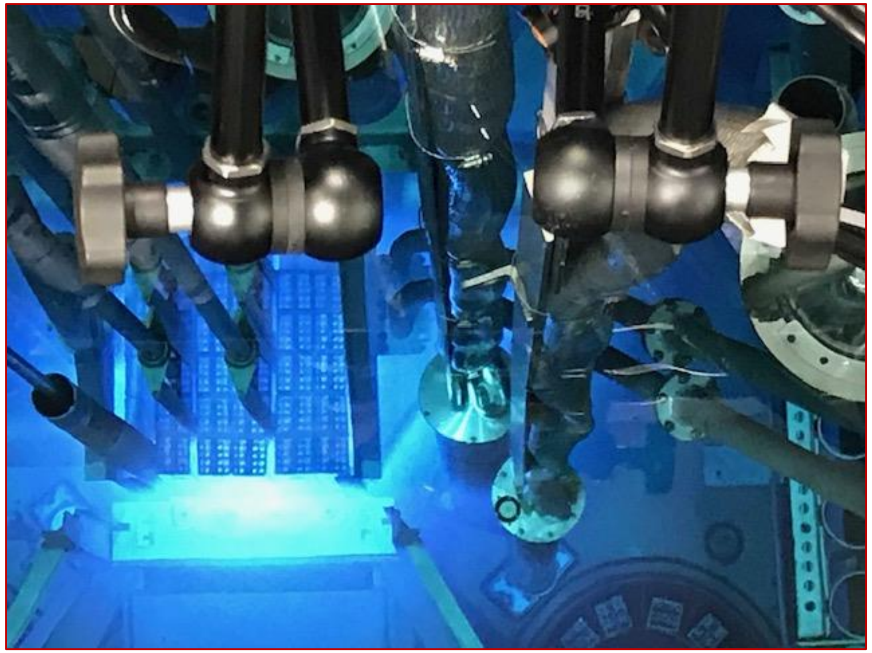
- Measure full MSR fission product spectrum with off-gas between 600 and 700°C
- Initial testing in NCSU PULSTAR reactor



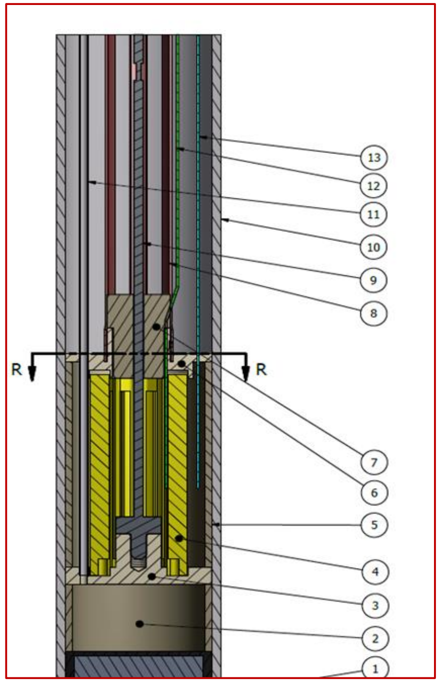
Conceptual design of the fission gas and tritium measurement irradiation chamber

NCSU Building Off-gas Sensor System and Off-gas Source (Molten Salt Materials)

- Sensor system
- Irradiation of molten salt reactor (MSR) materials in intense irradiation and high temperature environments
    - Facility design at advanced stage
    - Equipment acquisition underway



View of NCSU PULSTAR core



Extreme Environment In-Pool Irradiation facility

The University of California at Berkeley

Department of Nuclear Engineering

R. O. Scarlat, L. Vergari, H.E Williams,

A.M. Kennedy, M. Borrello

THE SALT RESEARCH GROUP

Berkeley UNIVERSITY OF CALIFORNIA Nuclear Engineering

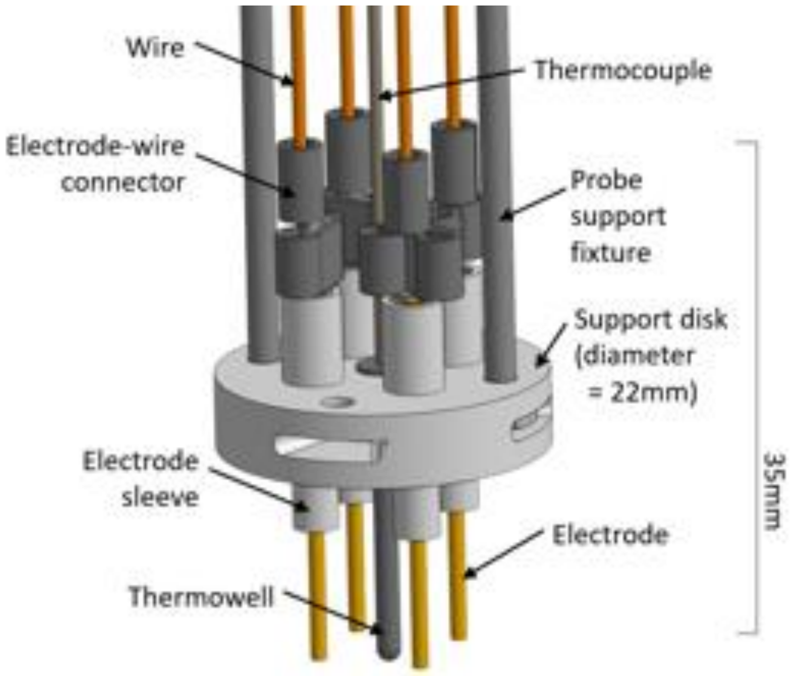
U.C Berkeley Is Developing Chemical Control Strategies

Redox Chemistry Control Determines Corrosion Rates and What Fission Products are Metals versus Fluorides

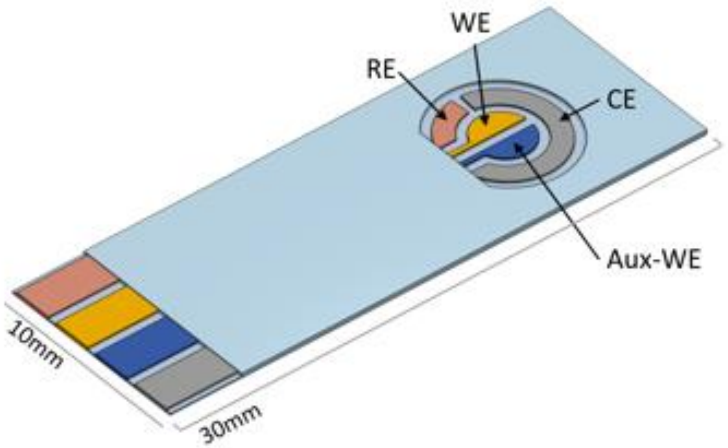
- Tritium and fission product transport experiments
- Development of redox measurement probes for loop
- Development of redox control strategies

Sensor Development at U.C Berkeley

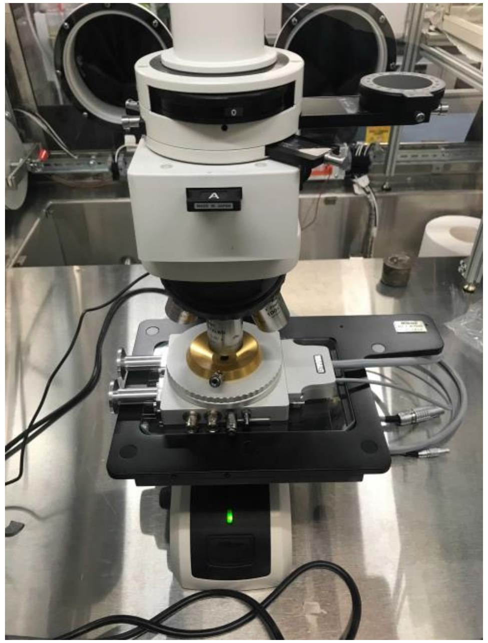
Electrochemical probe for standard molten salt electrochemical cell. The probes will be inserted in the MIT irradiated loop



Thin film sensor for high-throughput electrochemical experimentation

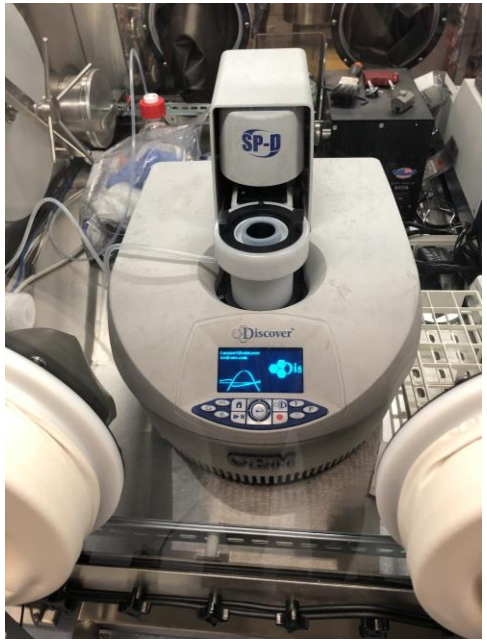


New capabilities for salt and samples analysis



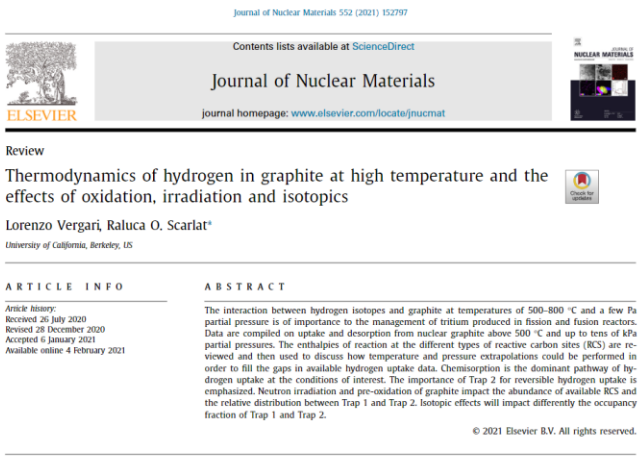
High temperature stage for electrochemistry and spectroscopy, mounted on polarized light microscope

Commissioned glovebox train for experiments with irradiated materials and beryllium containing salts. The train is equipped with a deep well and an oil-free pump, which can be helpful for hydrogen and tritium experiments.



CEM Microwave digester in the glovebox train. The digester will help correlate redox measurements to elemental analysis of salt samples from the loop

Publications


Journal of Nuclear Materials 552 (2021) 152797
Contents lists available at ScienceDirect
Journal of Nuclear Materials
journal homepage: www.elsevier.com/locate/jnucmat

Review

Thermodynamics of hydrogen in graphite at high temperature and the effects of oxidation, irradiation and isotopics

Lorenzo Vergari, Raluca O. Scarlat*
University of California, Berkeley, US

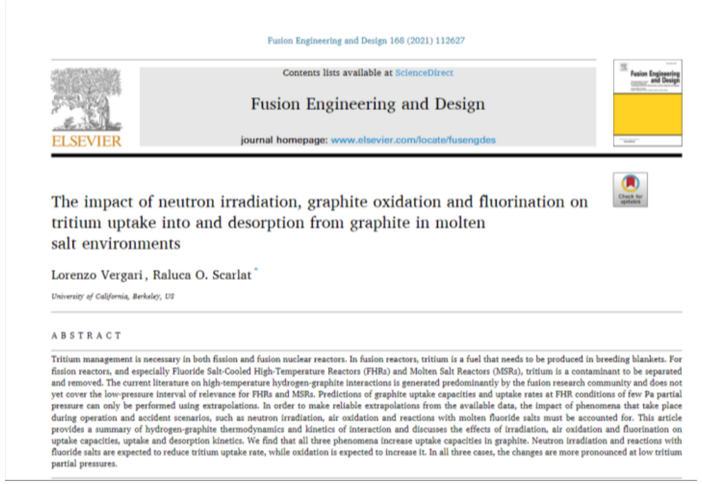
ARTICLE INFO

Article history:
Received 26 July 2020
Revised 28 December 2020
Accepted 6 January 2021
Available online 4 February 2021

ABSTRACT

The interaction between hydrogen isotopes and graphite at temperatures of 500–800 °C and a few Pa partial pressure is of importance to the management of tritium produced in fission and fusion reactors. Data are compiled on uptake and desorption from nuclear graphite above 500 °C and up to tens of kPa partial pressures. The enthalpies of reaction at the different types of reactive carbon sites (RCS) are reviewed and then used to discuss how temperature and pressure extrapolations could be performed in order to fill the gaps in available hydrogen uptake data. Chemisorption is the dominant pathway of hydrogen uptake at the conditions of interest. The importance of Trap 2 for reversible hydrogen uptake is emphasized. Neutron irradiation and pre-oxidation of graphite impact the abundance of available RCS and the relative distribution between Trap 1 and Trap 2. Isotopic effects will impact differently the occupancy fraction of Trap 1 and Trap 2.

© 2021 Elsevier B.V. All rights reserved.

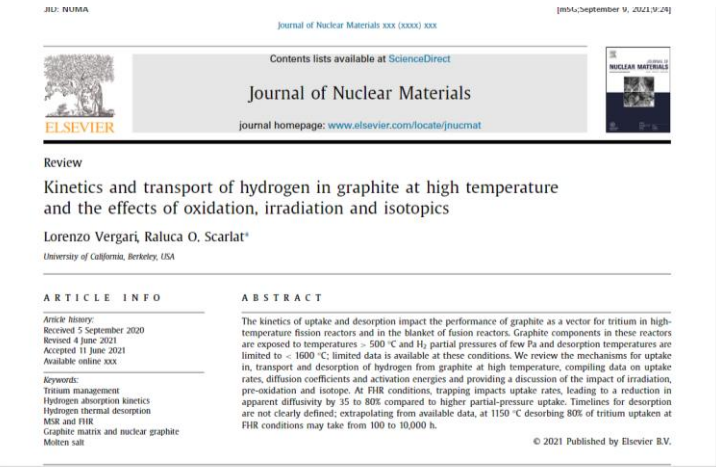

Fusion Engineering and Design 168 (2021) 112627
Contents lists available at ScienceDirect
Fusion Engineering and Design
journal homepage: www.elsevier.com/locate/fusengdes

The impact of neutron irradiation, graphite oxidation and fluorination on tritium uptake into and desorption from graphite in molten salt environments

Lorenzo Vergari, Raluca O. Scarlat*
University of California, Berkeley, US

ABSTRACT

Tritium management is necessary in both fission and fusion nuclear reactors. In fusion reactors, tritium is a fuel that needs to be produced in breeding blankets. For fission reactors, and especially Fluoride Salt-Cooled High-Temperature Reactors (FHRs) and Molten Salt Reactors (MSRs), tritium is a contaminant to be separated and removed. The current literature on high-temperature hydrogen-graphite interactions is generated predominantly by the fusion research community and does not yet cover the low-pressure interval of relevance for FHRs and MSRs. Predictions of graphite uptake capacities and uptake rates at FHR conditions of few Pa partial pressure can only be performed using extrapolations. In order to make reliable extrapolations from the available data, the impact of phenomena that take place during operation and accident scenarios, such as neutron irradiation, air oxidation and reactions with molten fluoride salts must be accounted for. This article provides a summary of hydrogen-graphite thermodynamics and kinetics of interaction and discusses the effects of irradiation, air oxidation and fluorination on uptake capacities, uptake and desorption kinetics. We find that all three phenomena increase uptake capacities in graphite. Neutron irradiation and reactions with fluoride salts are expected to reduce tritium uptake rate, while oxidation is expected to increase it. In all three cases, the changes are more pronounced at low tritium partial pressures.


JID: NUMA
Journal of Nuclear Materials xxx (xxxx) xxx
Contents lists available at ScienceDirect
Journal of Nuclear Materials
journal homepage: www.elsevier.com/locate/jnucmat

Review

Kinetics and transport of hydrogen in graphite at high temperature and the effects of oxidation, irradiation and isotopics

Lorenzo Vergari, Raluca O. Scarlat*
University of California, Berkeley, USA

ARTICLE INFO

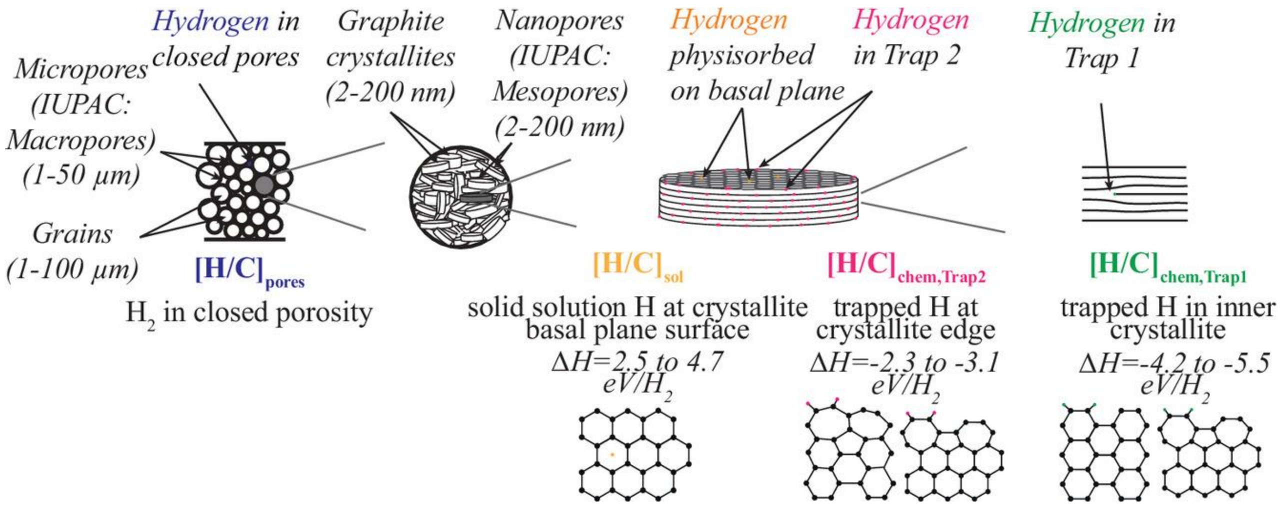
Article history:
Received 5 September 2020
Revised 4 June 2021
Accepted 11 June 2021
Available online xxx

Keywords:
Tritium management
Hydrogen absorption kinetics
Hydrogen thermal desorption
MSR and FHR
Graphite matrix and nuclear graphite
Molten salt

ABSTRACT

The kinetics of uptake and desorption impact the performance of graphite as a vector for tritium in high-temperature fission reactors and in the blanket of fusion reactors. Graphite components in these reactors are exposed to temperatures > 500 °C and H_2 partial pressures of few Pa and desorption temperatures are limited to < 1600 °C; limited data is available at these conditions. We review the mechanisms for uptake in, transport and desorption of hydrogen from graphite at high temperature, compiling data on uptake rates, diffusion coefficients and activation energies and providing a discussion of the impact of irradiation, pre-oxidation and isotope. At FHR conditions, trapping impacts uptake rates, leading to a reduction in apparent diffusivity by 35 to 80% compared to higher partial-pressure uptake. Timelines for desorption are not clearly defined; extrapolating from available data, at 1150 °C desorbing 80% of tritium uptaken at FHR conditions may take from 100 to 10,000 h.

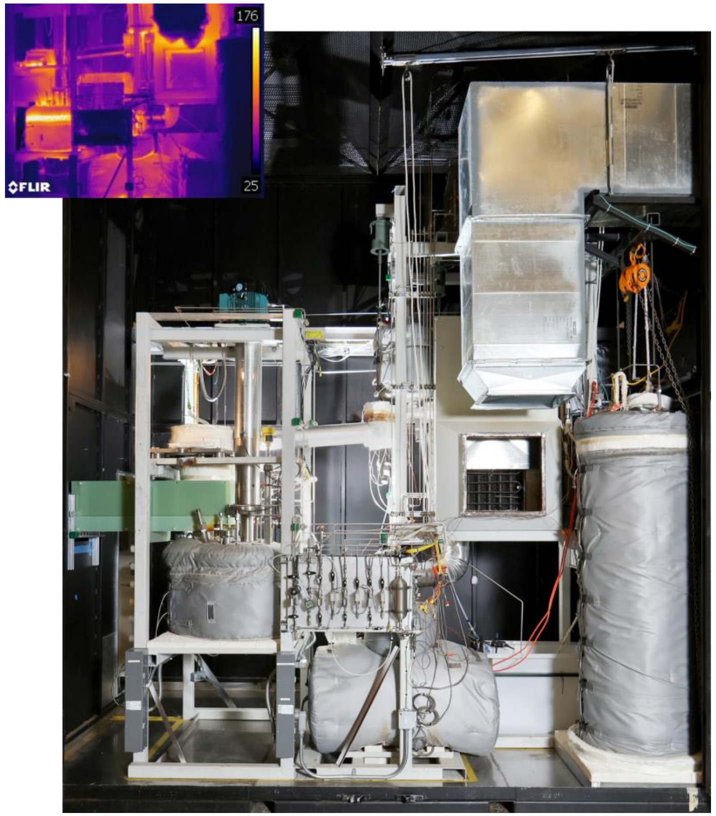
© 2021 Published by Elsevier B.V.

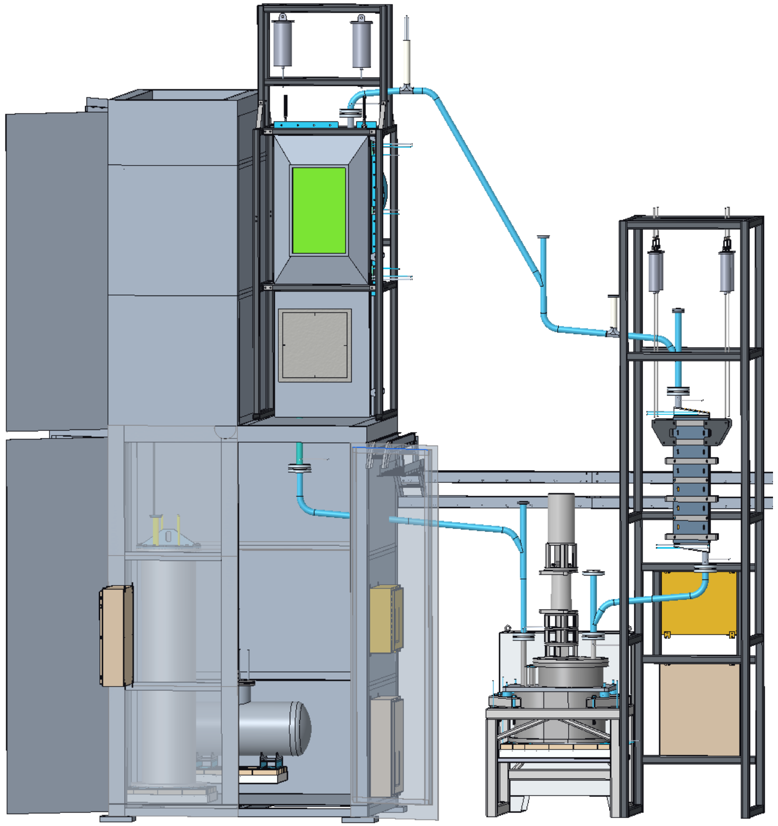


Recently published literature reviews on hydrogen – graphite chemistry
https://doi.org/10.1016/j.jnucmat.2021.152797
https://doi.org/10.1016/j.fusengdes.2021.112627
https://doi.org/10.1016/j.jnucmat.2021.153142

ORNL is Supporting the Project Based on Experience with Out-of-Reactor Loops

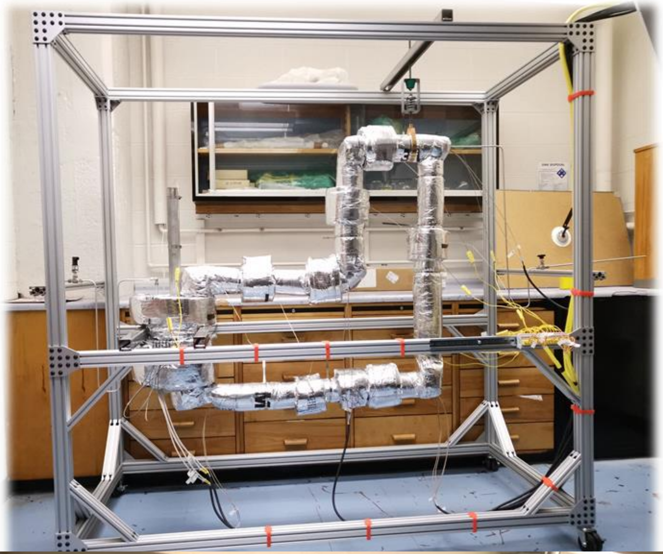
- ORNL has been designing and operating multiple loops, including forced flow systems
- Experience used to support loop project





Conclusions

- Designed and building a instrumented salt flow loop is underway
  - Neutron irradiation—initially clean flibe salt then with fissile materials
  - Variable temperature around the loop
- Designed as long-term facility for change-out of experiments with lessons learned for future DOE/University salt loops
- Major procurements on order, cold testing of subsystems underway



Biography: Charles Forsberg

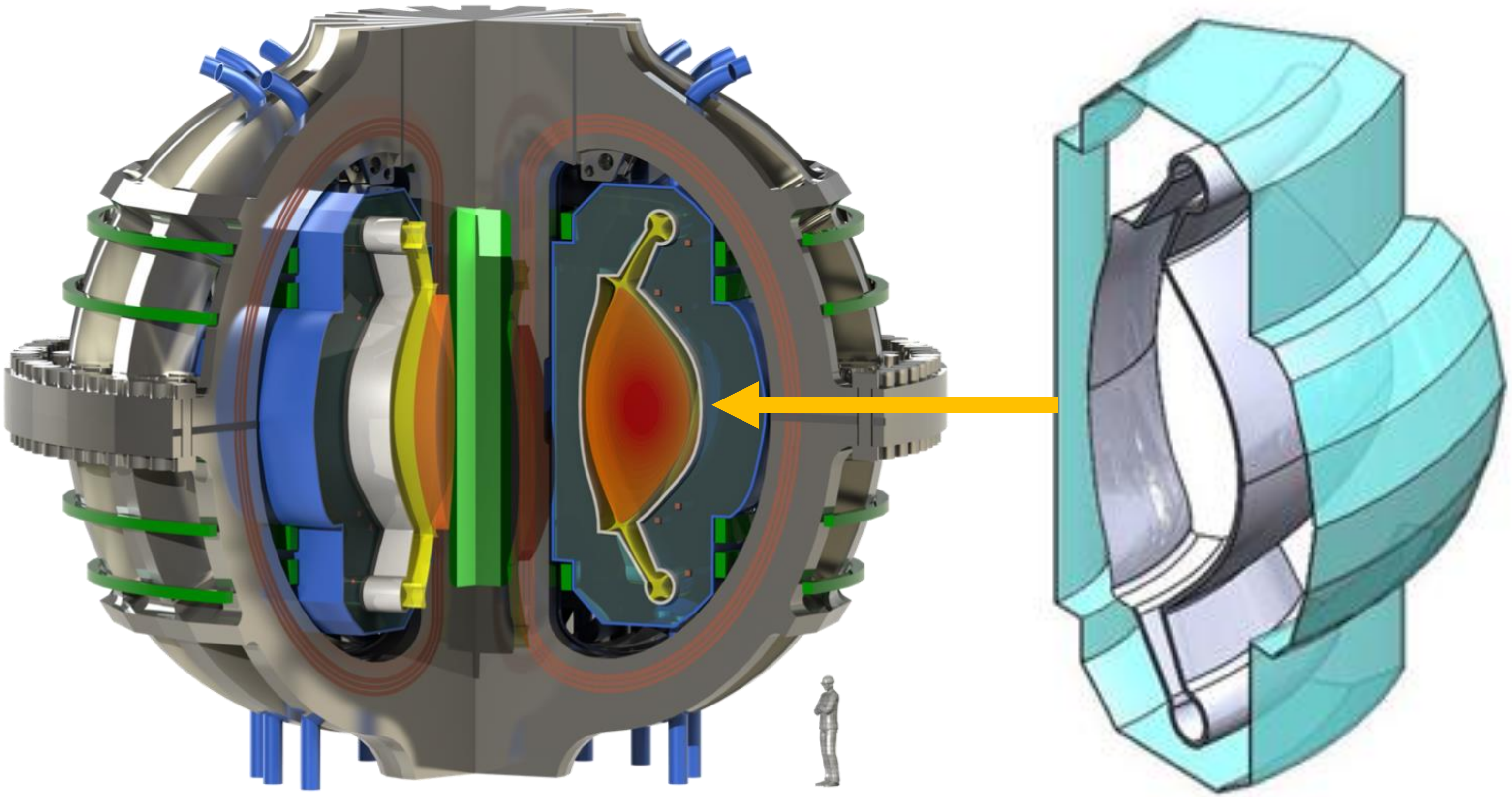
Dr. Charles Forsberg is a principal research scientist at MIT. His current research areas include Fluoride-salt-cooled High-Temperature Reactors (FHRs), hybrid energy systems and utility-scale 100 GWh heat storage systems. He teaches the fuel cycle and energy systems classes. Before joining MIT, he was a Corporate Fellow at Oak Ridge National Laboratory. Earlier he worked for Bechtel Corporation and Exxon.

He is a Fellow of the American Nuclear Society (ANS), 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 waste management, hydrogen production and nuclear-renewable energy futures. He received the American Nuclear Society special award for innovative nuclear reactor design and is a Director of the ANS. 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 published over 300 papers.

Potential Implications of Recently Announced MIT / Commonwealth Fusion Breakthrough

- MIT successfully tested large magnet that enables doubling magnetic fields in fusion machines
- **Size of magnetic fusion system for any given power output varies as *one over the fourth power of the magnetic field***
- Higher magnetic fields can shrink fusion system size by an order of magnitude with massive cost savings
- **Power density in the fusion blanket increases by an order of magnitude creating incentive for liquid flibe salt blanket that is coolable, solid blankets may melt**

ARC Fusion with Liquid Flibe Salt Blanket



ARC　　　Flibe Salt Blanket

- Breed tritium fusion fuel from lithium in salt
- Convert energy in 14-Mev neutrons to heat for power cycle
- Radiation Shielding

Flibe Coolant Becoming a Priority for Fusion Systems