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April 23, 2026 – Albuquerque, NM

Understanding graphite–salt interactions: a multidisciplinary approach

Nidia C Gallego & Team

Oak Ridge National Laboratory

U.S. DEPARTMENT of ENERGY

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Team Effort – Contributors


Jisue Braatz


Tomas Grejtak


David Arregui-Mena


Lianshan Lin


GS Jung

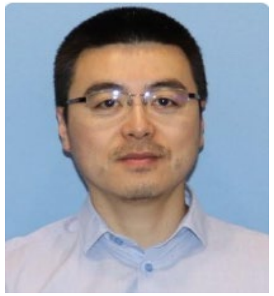

Nidia C. Gallego


J. Wilna Geringer

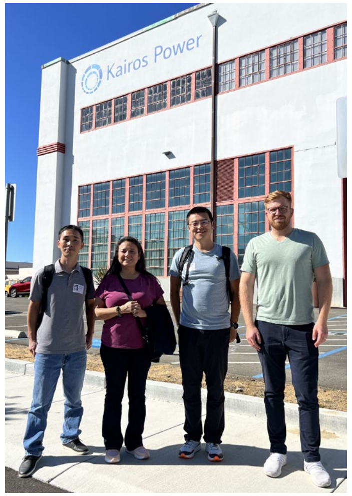

Jun Qu


Yuxuan Zhang


Cristian Contescu


Wenqi Li

INGSM 2024
UC Berkeley, CA



Kairos Power
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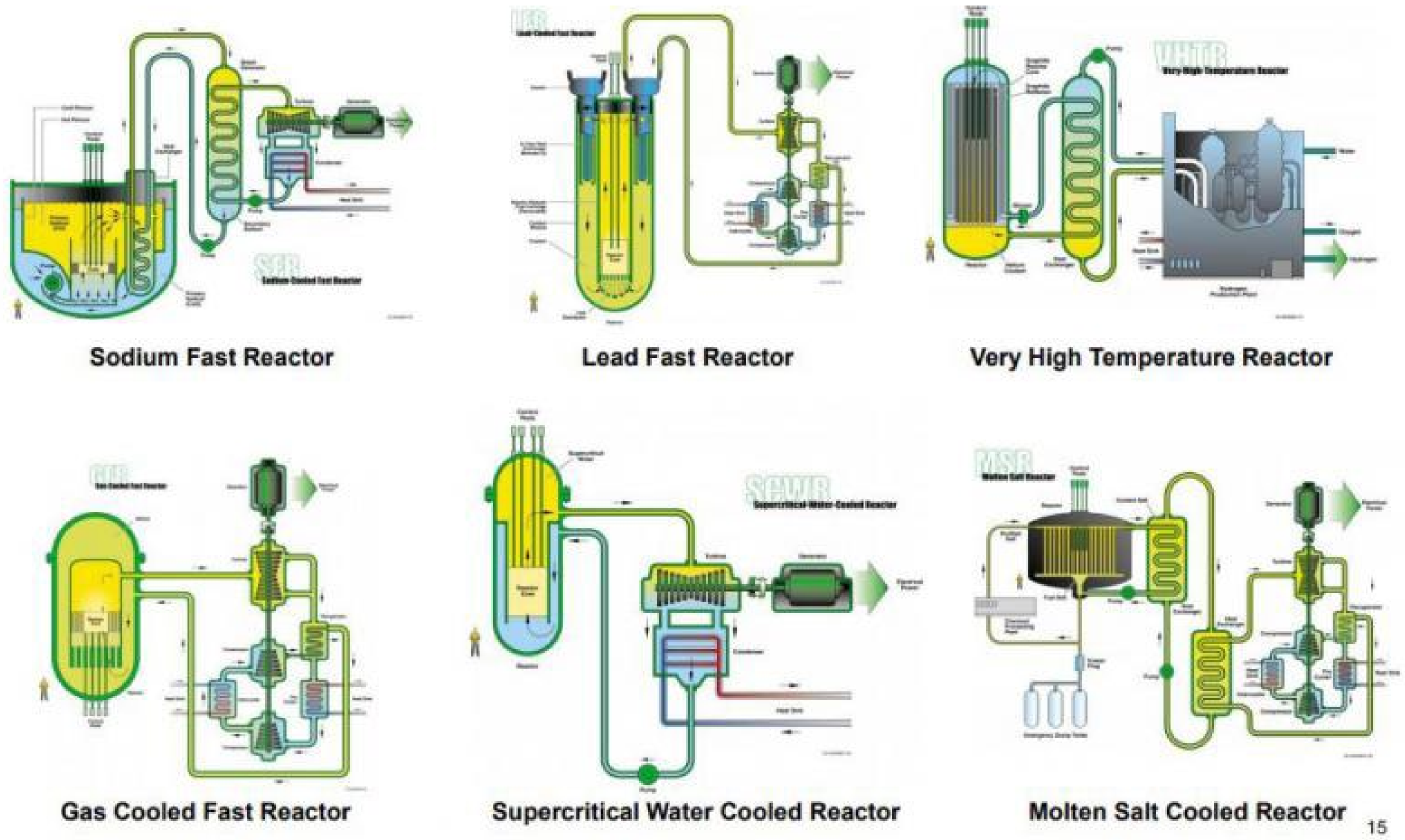
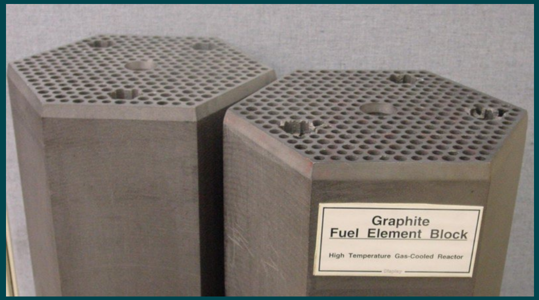
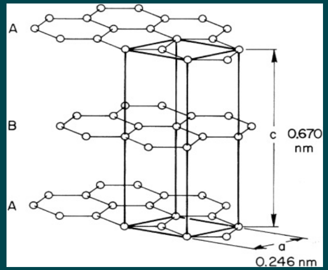
visit 2024

Graphite plays a critical role in several GenIV-type Advanced Reactor concepts, including gas-cooled and molten salt reactors.







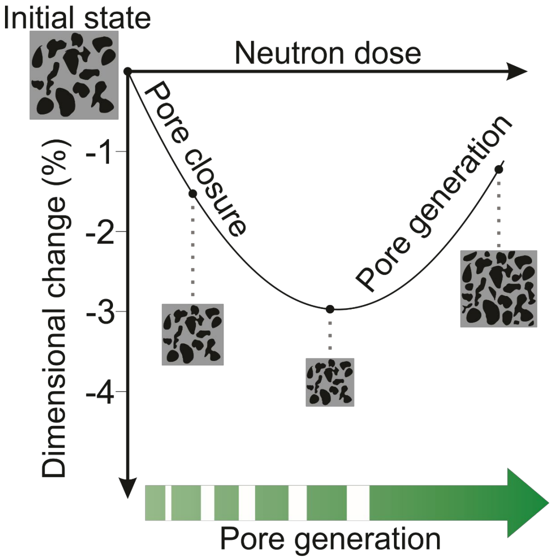
Why graphite?

- Good neutron moderator for thermal reactors
- Low neutron absorption – enhances chain reaction
- High thermal conductivity

Graphite plays a critical role in several GenIV-type Advanced Reactor concepts, including gas-cooled and molten salt reactors.

Effect of fast neutron irradiation and its relationship with microstructure

- Dimensional changes, structural damage
- Change in mechanical and thermal properties



Environment Effects

Gas-cooled reactors

- Chronic oxidation
- Acute oxidation

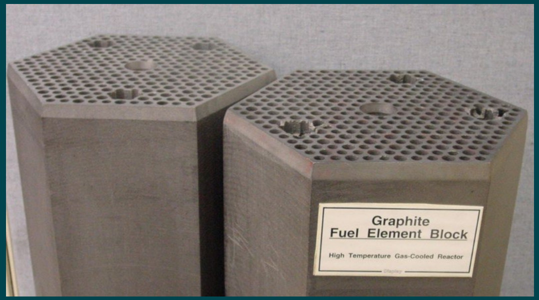
Fluoride salt-cooled reactors

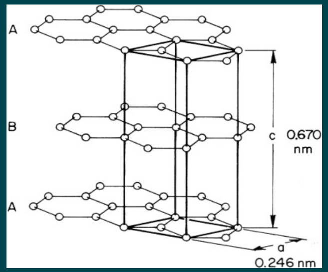
Physical

- Salt intrusion into graphite pores and effect on mechanical/ thermal properties; effect of/on irradiation damage; what if salt is fueled
- Wear and abrasion
- Erosion
- Effect of potential heating/cooling cycles

Chemical

- Chemical Interactions: Fluorination, intercalation, effect on properties of graphite
- Effect of salt impurities
- Effect of graphite dust in salt
- Absorption of fission products
- Galvanic corrosion





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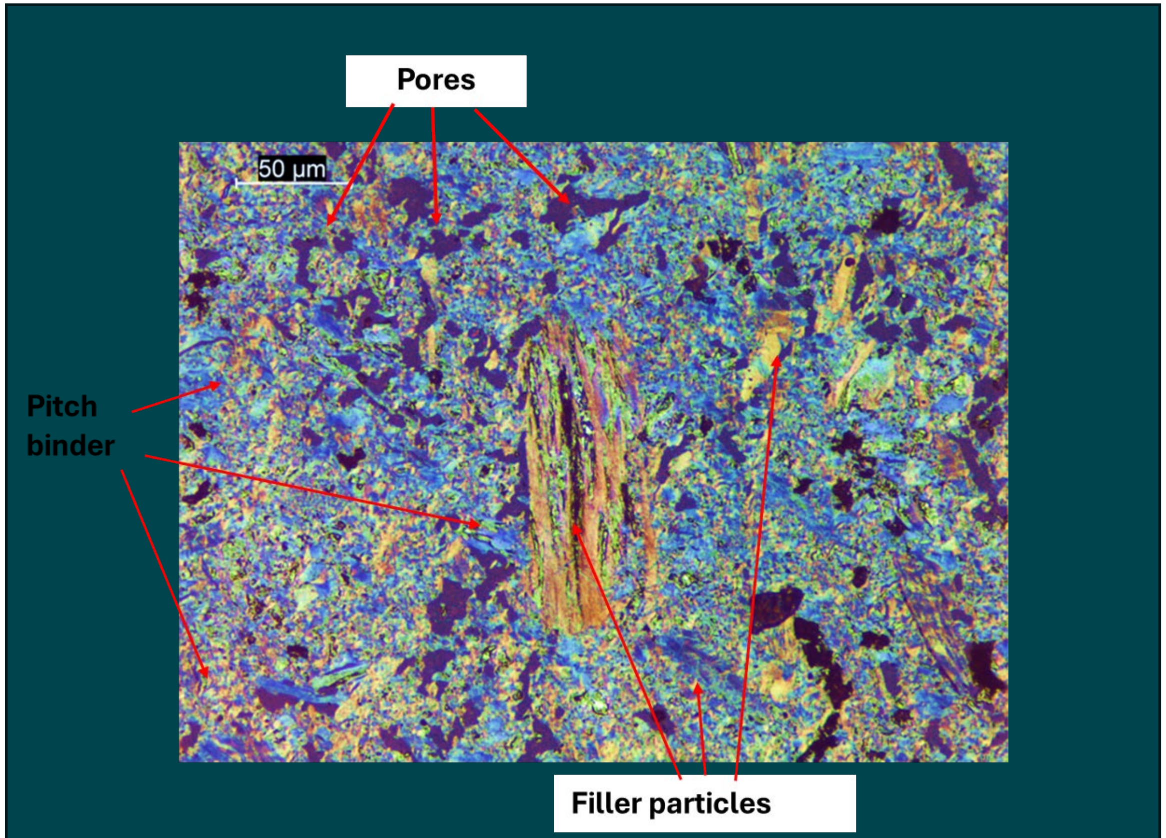
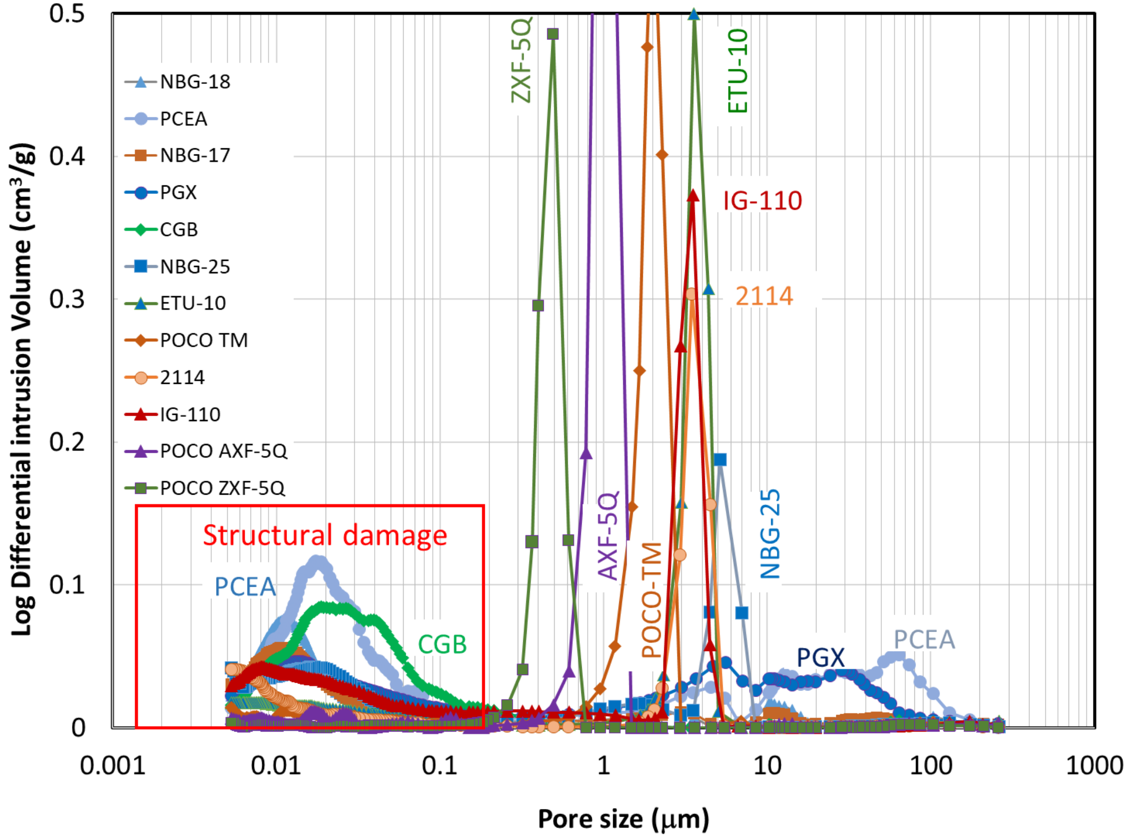
Graphite pore network (**size, shape and connectivity**) varies significantly between graphite grades



Manufactured Graphite has about **20 % porosity**



Graphite pore size distribution obtained from mercury intrusion measurements

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Objective:

Assist in the near-term deployment of MSRs by help defining the safe working envelope for nuclear graphite

We acknowledge Financial Support from Advanced Reactor Technology Program – DOE-NE - Graphite Program under the GCR Campaign

This research used resources at the High Flux Isotope Reactor, a US DOE Office of Science User Facility operated by the ORNL

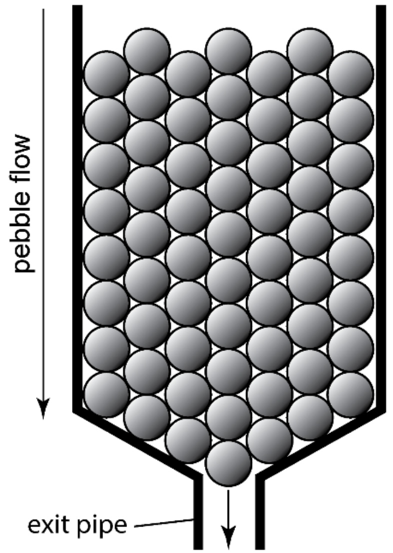
Current Research Focus of Graphite-Salt Studies at ORNL:

- Understanding physical interactions: salt intrusion, penetration depth and salt distribution, as a F(T, P, t) and graphite microstructures.
- Studying wetting behavior of salt on graphite surfaces to develop predictive models for salt intrusion
- Studying wear and erosion behavior of graphite in molten salt
- Working with the ASTM community to develop standards to measure the effect of salt intrusion on graphite properties
- Working with the ASME Community to develop the needed knowledge to address the gaps in the ASME code.

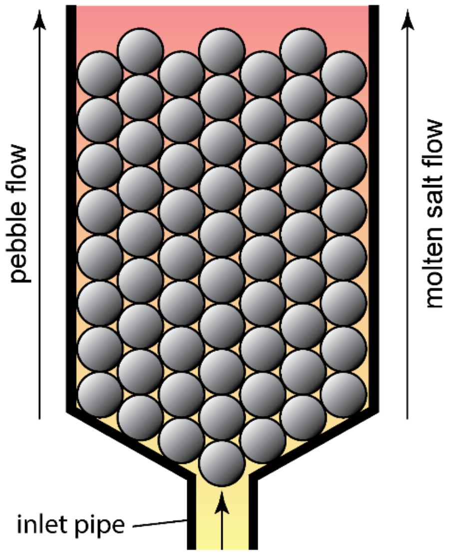
Tribological Studies - Understanding wear properties of graphite

Tribological characterization of graphite in dry argon and molten salt environment

High temperature gas-cooled reactor (HTGR)



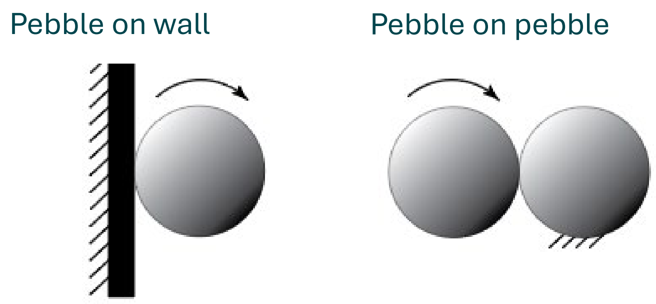
Fluoride salt-cooled high temperature reactor - (FHR)



- Rector contains thousands of pebbles that circulated during operation.
- Each pebble passes through multiple times before it is finally discharged.
- In **HTGR**, the pebbles flow downwards, while in **FHR** the pebbles and molten salt flow upward.
- ***Pebble flow results in their abrasion and potential dust contamination.***
- <u>**Environment:**</u> argon or helium gas, temperatures up to 650 °C.

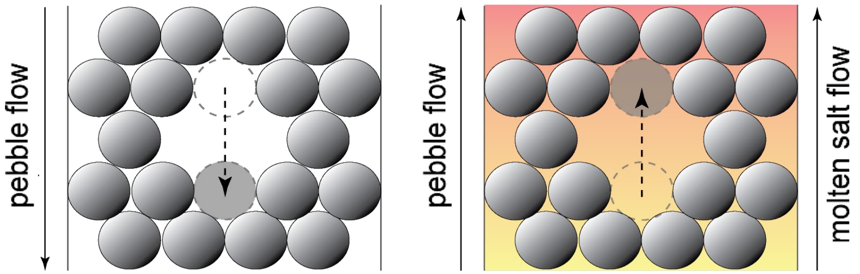
Contact modes

Sliding & Rolling



Pebbles slide and roll against other pebbles and the graphite wall during circulation in the reactor which may results in ***their abrasion and dust contamination***.

Impact



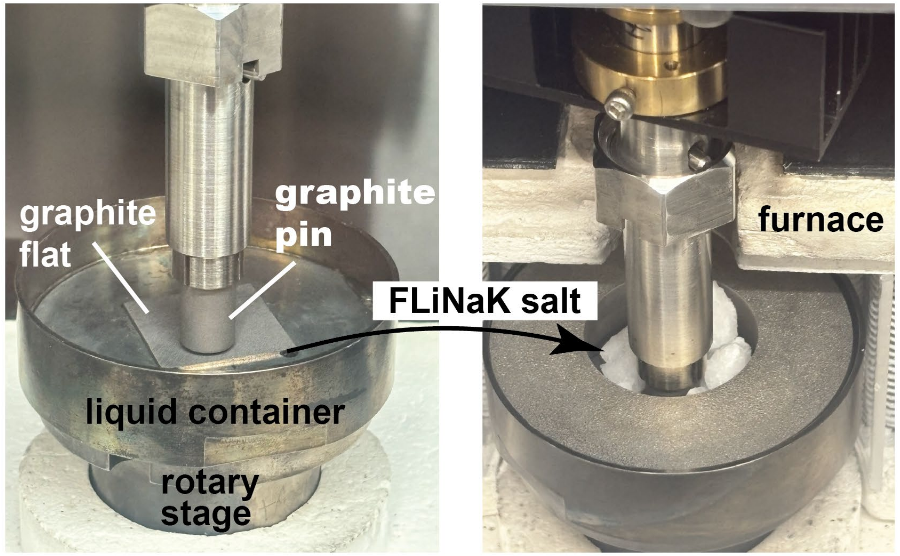
A pebble could move through the void and collide with an opposing pebble at higher contact loads causing an **impact**.

Friction and wear characterization – *sliding* mode

Tribometer placed in a glovebox with a controlled Ar atmosphere

Dry argon

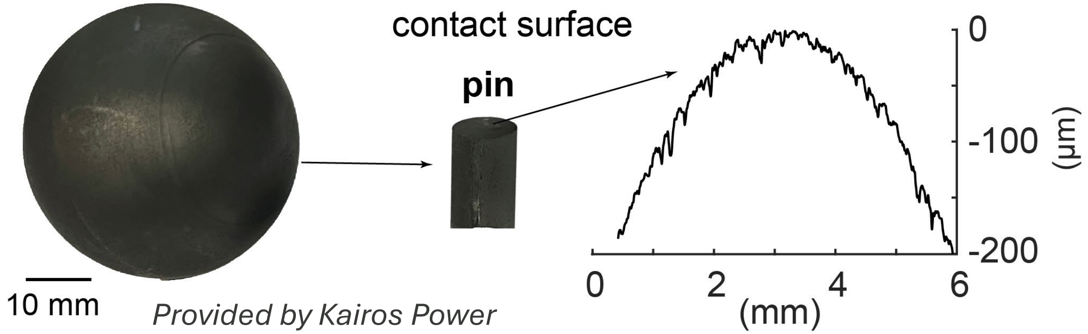
FLiNaK



After the test

ET-10

Testing on as-received surface



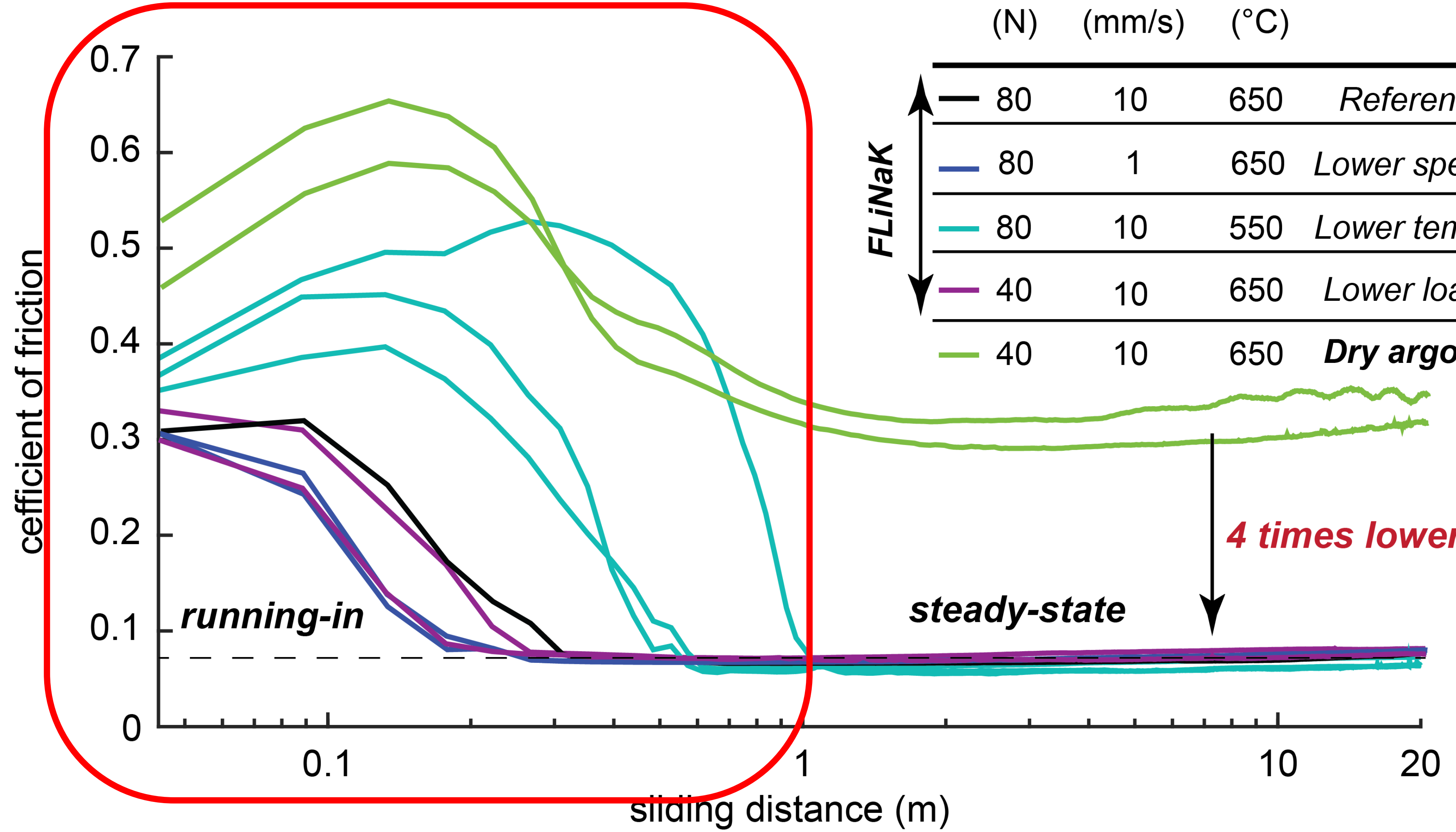
Provided by Kairos Power

Prepared with an end mill and polished

Gas: argon (O_2 < 1ppm ; H_2O < 1 ppm)

FLiNaK: (mol %: 46.5% LiF, 11.5% NaF, 42% KF)

Preliminary Results - Molten Salt Significantly Reduces Friction



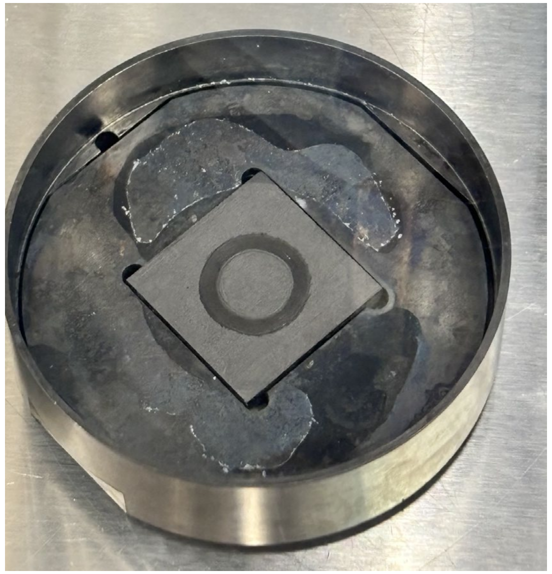
	Test conditions				Friction behavior	
	load (N)	speed (mm/s)	temp. (°C)		max CoF	steady-state CoF
FLiNaK	80	10	650	*Reference*	0.32	0.07
FLiNaK	80	1	650	*Lower speed*	0.31	0.08
FLiNaK	80	10	550	*Lower temp.*	0.40, 0.53	0.07
FLiNaK	40	10	650	*Lower load*	0.30, 0.33	0.08
	40	10	650	***Dry argon***	0.59, 0.65	0.31, 0.35

Future work: understanding friction and wear behavior in the FLiNaK

- Effect of temperature, sliding speed and contact load is not fully understood
- Technical challenges:
  - Materials compatibility with salt
  - Salt coverage of the graphite surface
- Testing under more realistic conditions:
  - Graphite infiltrated with salt
  - Testing in FLiBe
  - Matrix material

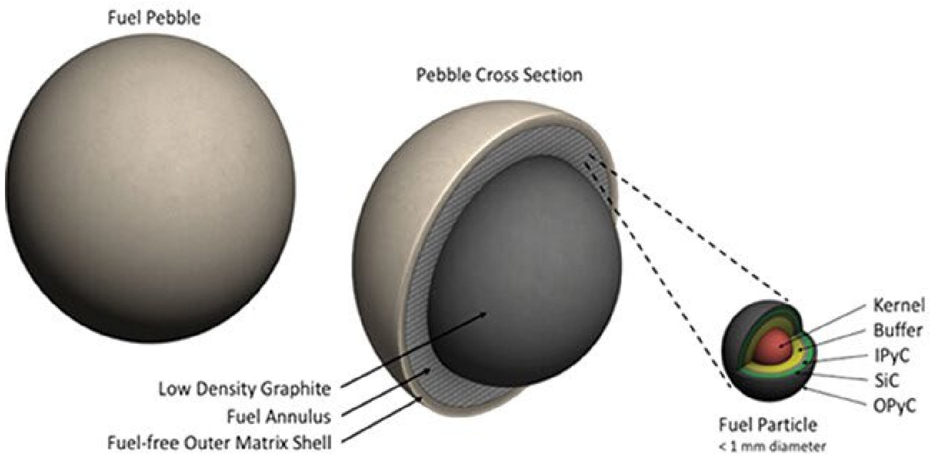
Fully submerged

Starved

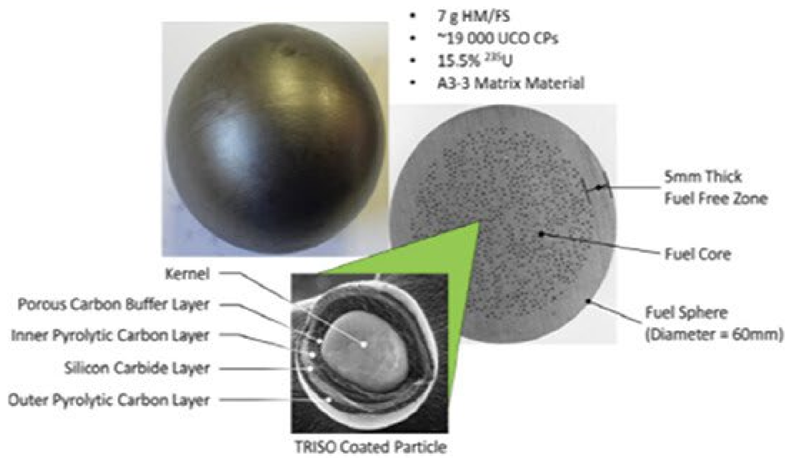


Friction and wear characterization of graphite matrix material

- Outer shell of a pebble is made of graphite matrix
- Preliminary tribological testing in dry argon using graphite matrix provided by **TRISO-X**
- Pin and flat samples machined from a pebble (50 mm diameter)



https://www.kairospower.com/technology



https://x-energy.com/xe-100/

Test conditions

- Temperature: 750°C
- Contact load: 20 N
- Sliding speed: 10 mm/s
- Gas: argon

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Wear 522 (2023) 204706

Contents lists available at ScienceDirect

Wear

journal homepage: www.elsevier.com/locate/wear

Tribocorrosion of stainless steel sliding against graphite in FLiNaK molten salt☆

Xin He [a], Chanaka Kumara [a], Dino Sulejmanovic [a], James R. Keiser [a], Nidia Gallego [b], Jun Qu [a,*]

[a] Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN, 37831, USA
[b] Chemical Sciences Division, Oak Ridge National Laboratory, Oak Ridge, TN, 37831, USA

Sliding friction and wear behavior of nuclear graphite in high temperature inert environment: influence of contact load, speed and temperature

Tomas Grejtak[1*], Nidia C. Gallego[2], J. David Arregui-Mena[1], Jisue Braatz[3], Seokhoon Jang[1], Alexandre C. Foucher[4], Jun Qu[1*]

[1]Materials Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831
[2]Chemical Science Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831
[3]Radioisotope Science and Technology Division, Oak Ridge National Laboratory, Oak Ridge, TN 37831
[4]Center for Nanophase Materials Sciences, Oak Ridge National Laboratory, Oak Ridge, TN 37831

*Corresponding Authors: grejtakt@ornl.gov, qujn@ornl.gov

ORNL/TM-2024/3253

Report on Initial Tribological Studies of Graphite in Dry Argon and Molten Salt Environment

Tomas Grejtak
Jun Qu
Nidia C. Gallego
James R. Keiser

January 2024

ORNL/TM-2025/4003

Friction and wear of nuclear graphite exposed to molten FLiNaK salt environment

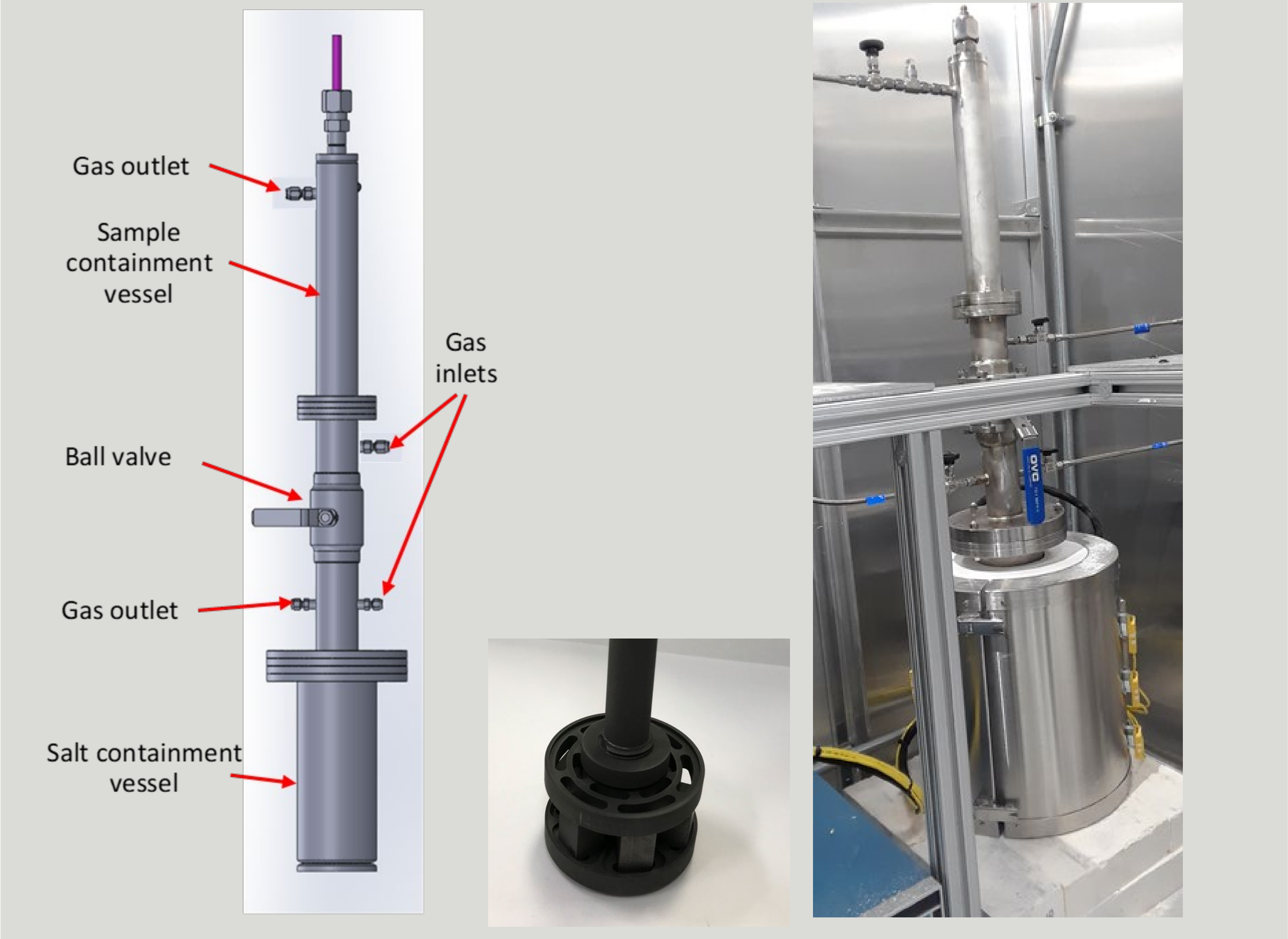
Tomas Grejtak
Nidia C. Gallego
Jun Qu

July 2025

Understanding salt intrusion and salt wetting behavior on graphite surfaces

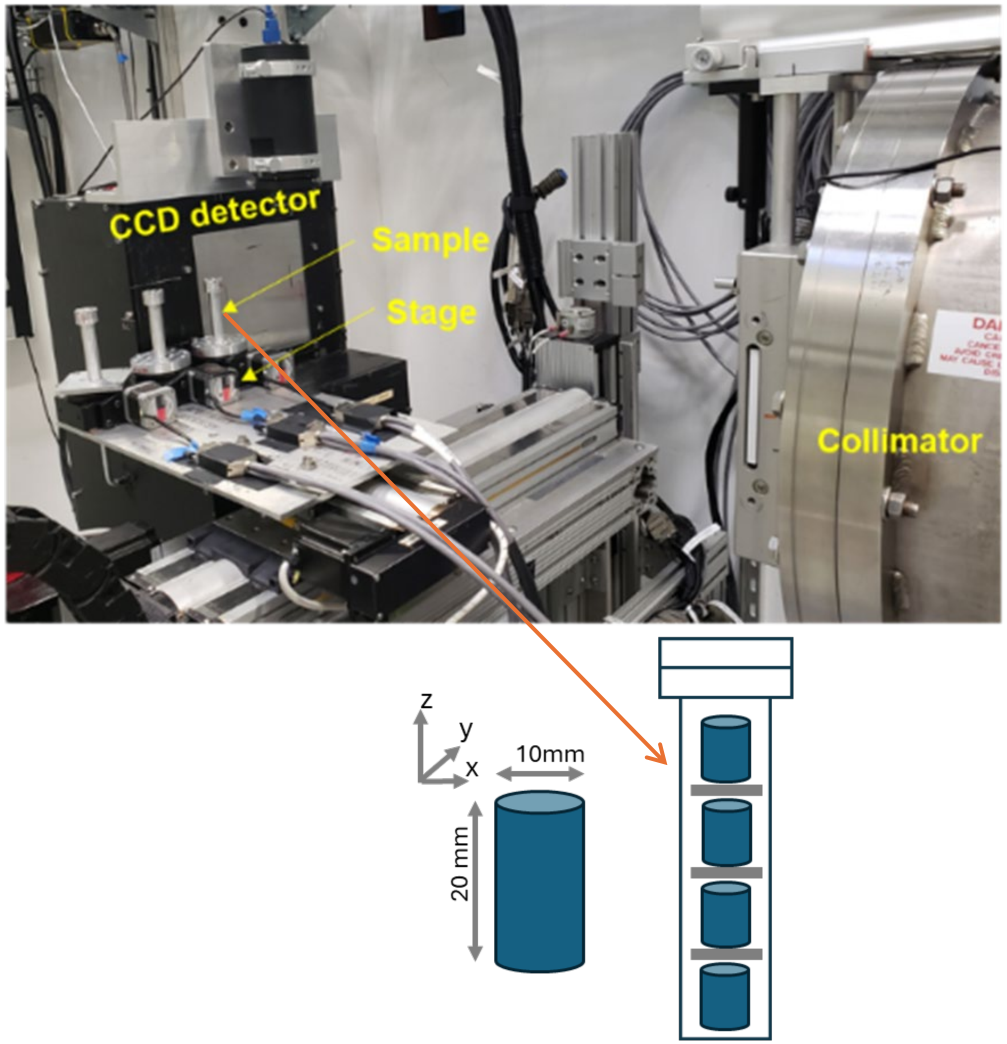
Capabilities for FLiNaK Salt Intrusion and Use of nCT for Analysis

Setup for high-temperature, high-pressure salt intrusion experiments



Neutron tomography

(MARS) beamline at the HFIR, ORNL



Understanding the Effect of Exposure Time on Salt Intrusion

Graphite grade	Grain size (μm)	Grain category
IG-110	<50	superfine
ETU-10	<50	superfine
2114	<50	superfine
NBG-25	<100	fine
PCEA	<1000	medium-fine
NGB-18	<4000	medium-coarse

Workflow:

1) nCT scans of the pristine samples
2) Salt intrusion experiments
3) nCT scans of the salt-exposed samples

Intrusion Conditions
T: 750ºC
P: 3 bar
Salt: FLiNaK

Time: 12 and 336 hours

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Gravimetric results indicated significant differences on weigh uptake as a function of time

Intrusion Conditions
T: 750ºC
P: 3 bar
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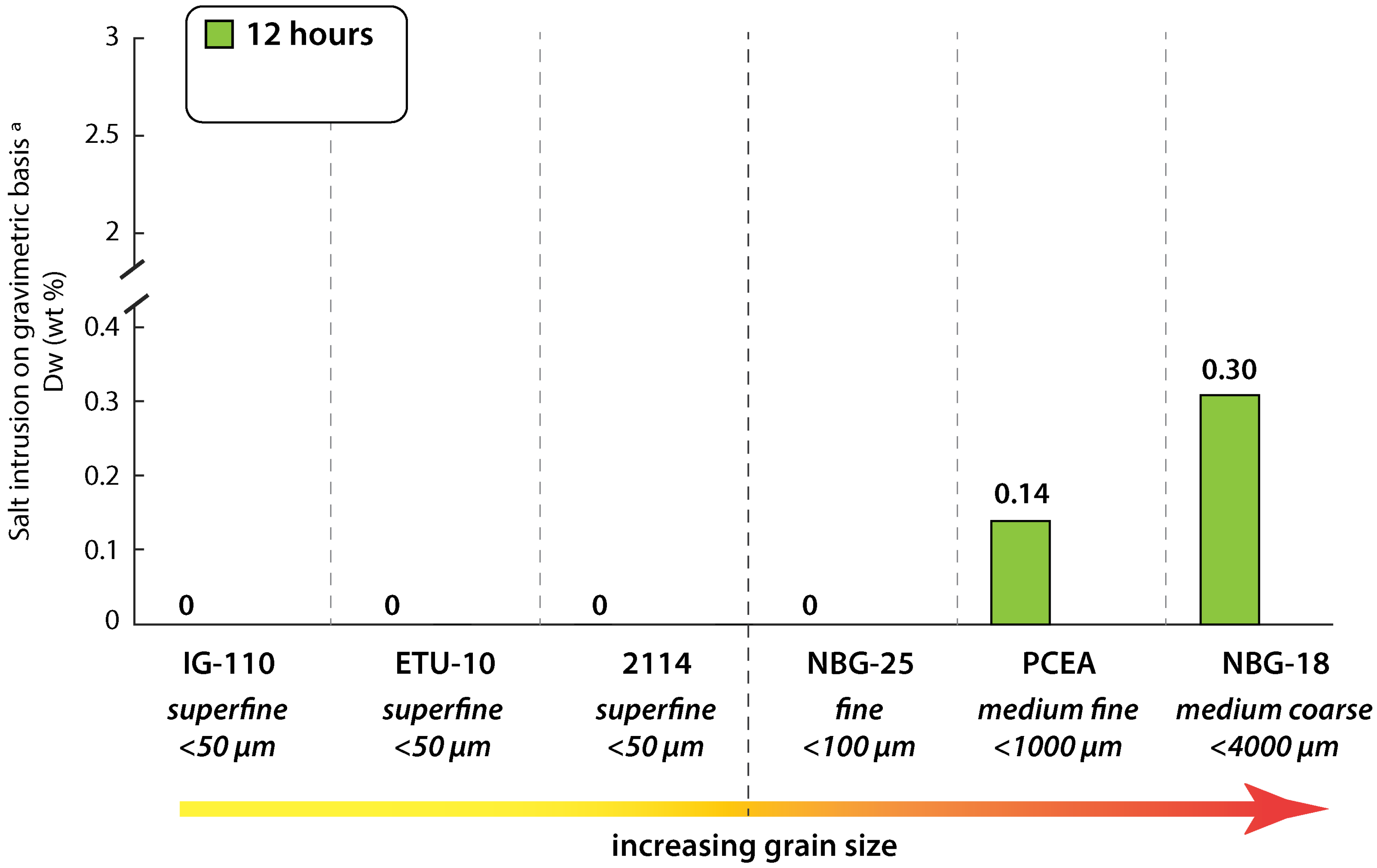

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Intrusion Conditions
T: 750ºC
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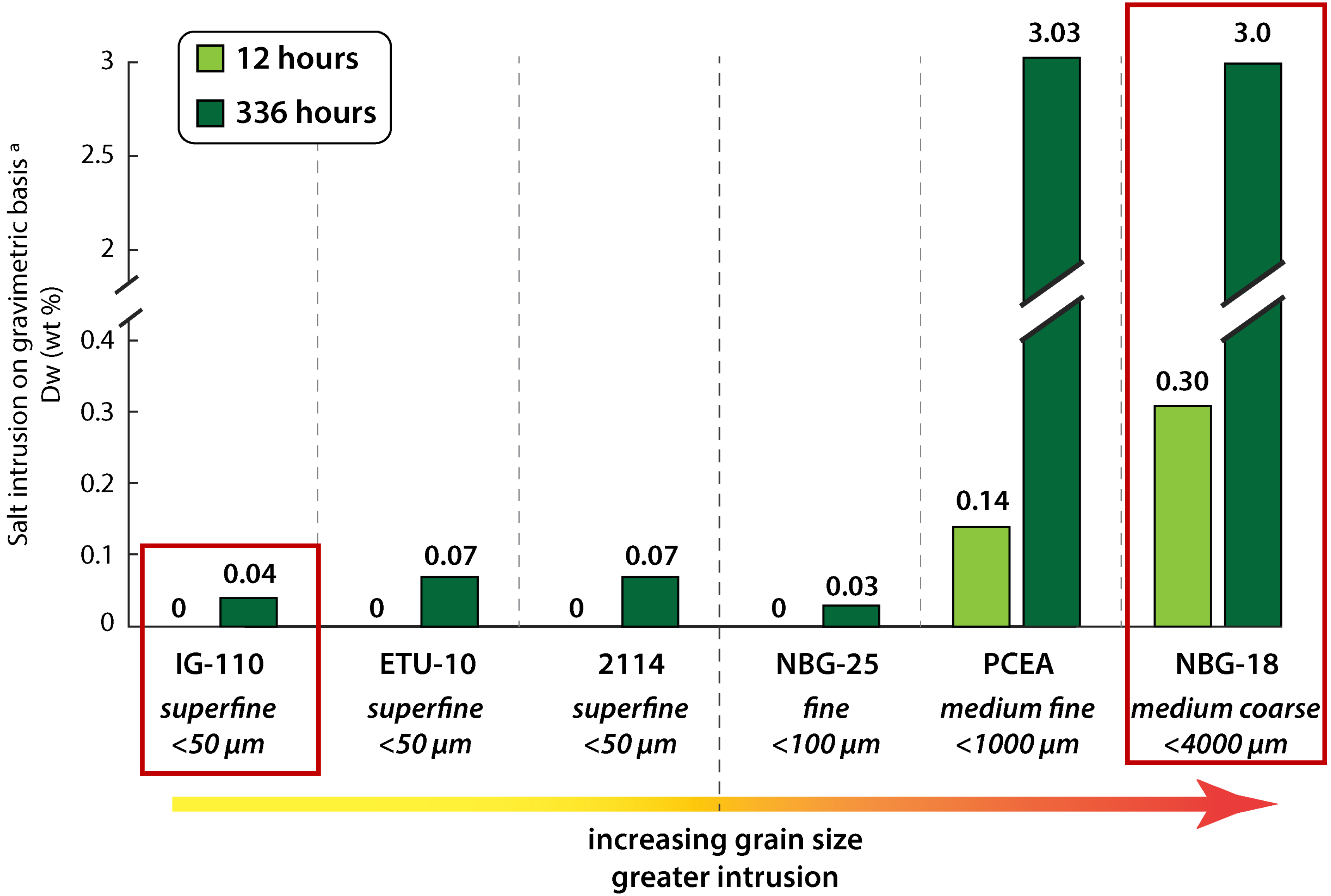
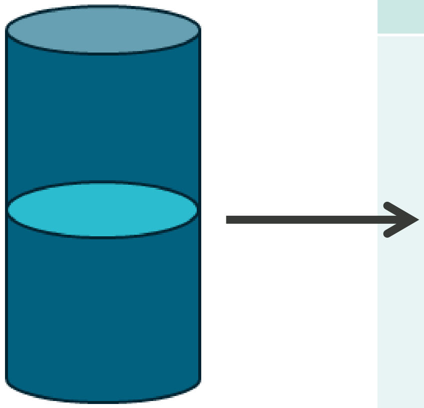
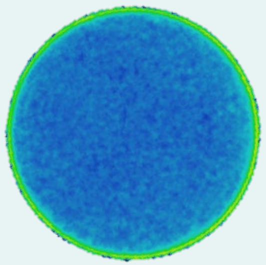
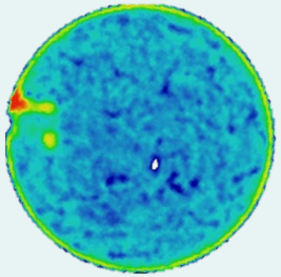
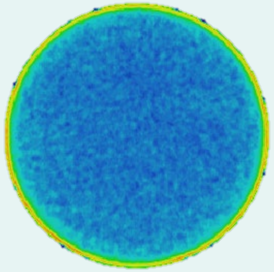
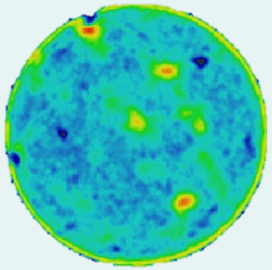


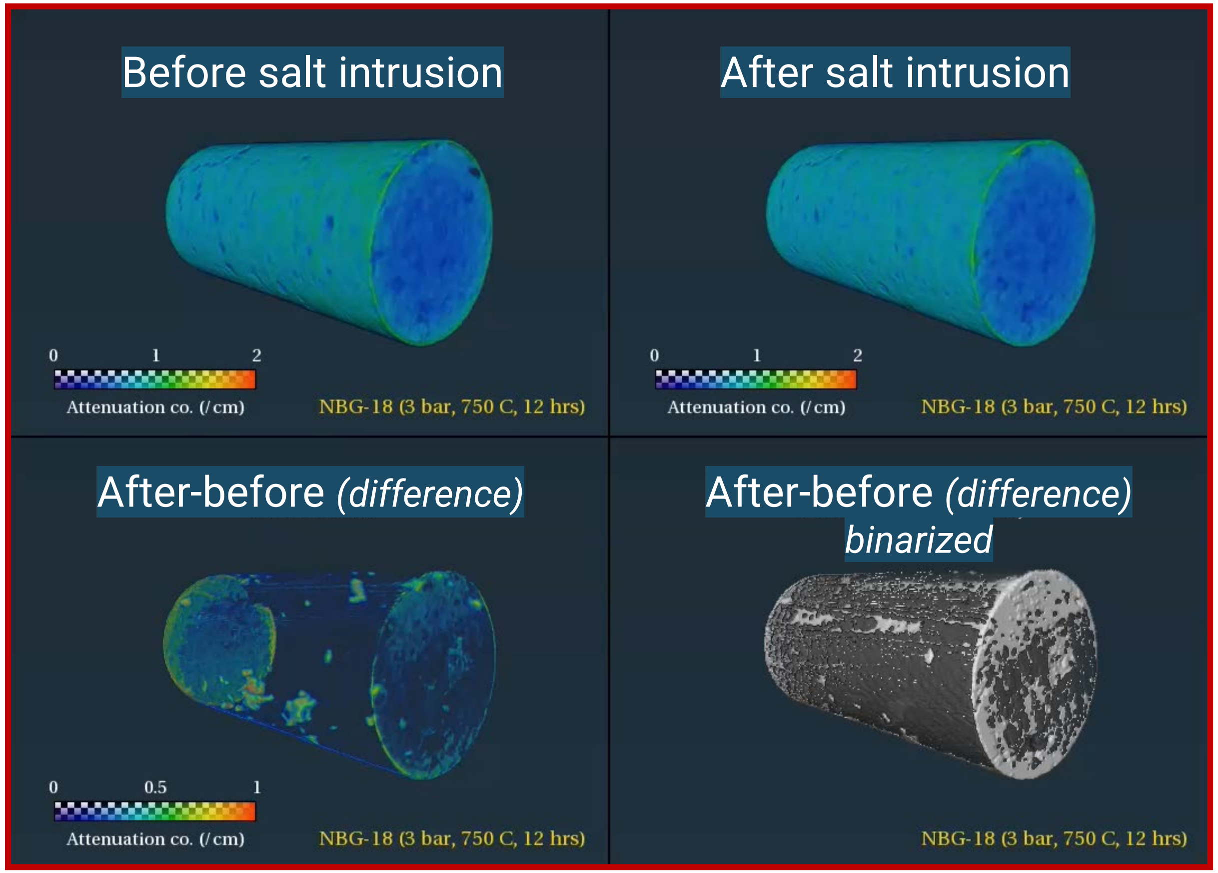
Preliminary Results had Indicated Different Behavior Between Fine and Medium Grain Graphites



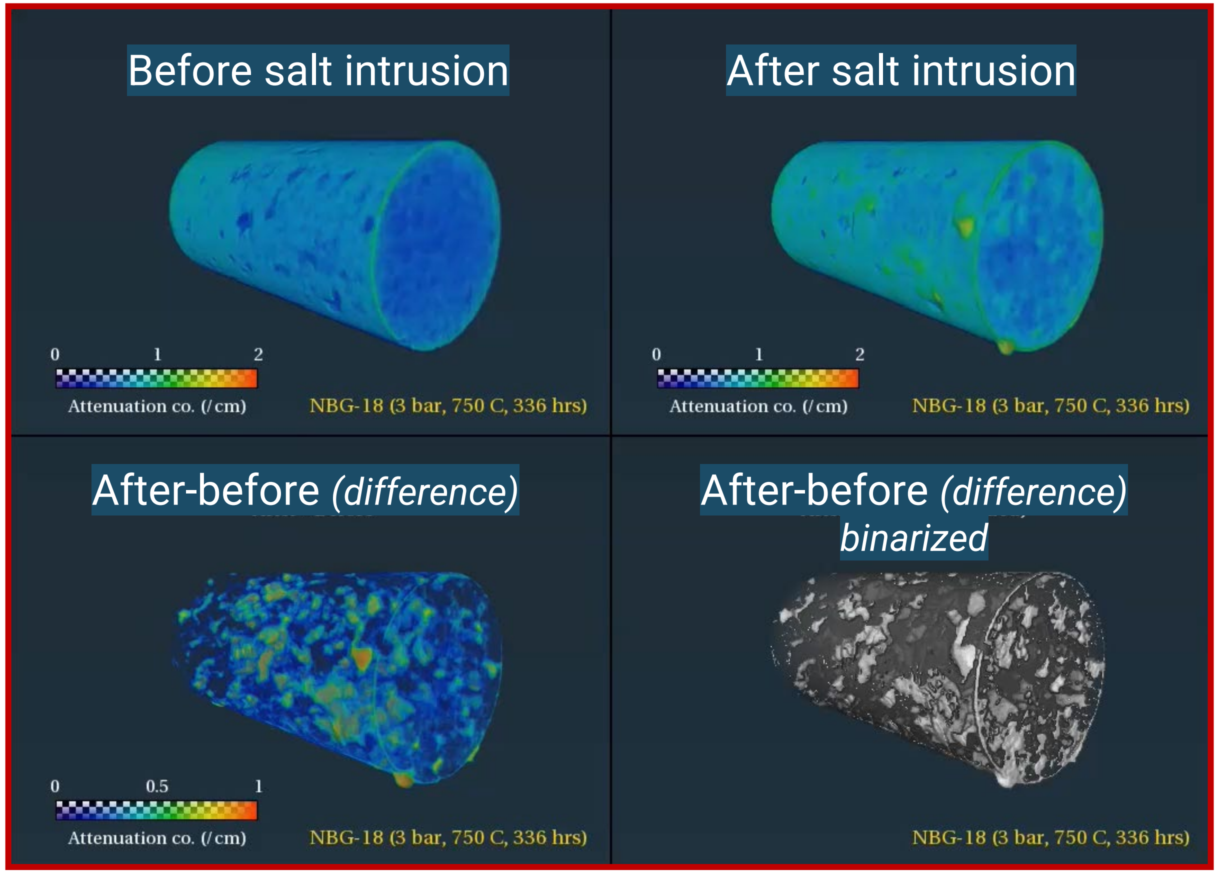
12 hrs		336 hrs	
IG-110 *(superfine)*	NBG-18 *(medium-coarse)*	IG-110 *(superfine)*	NBG-18 *(medium-coarse)*
			

Significant differences for MEDIUM grain graphite grade NBG-18: 12 v 336 hrs

12 hrs

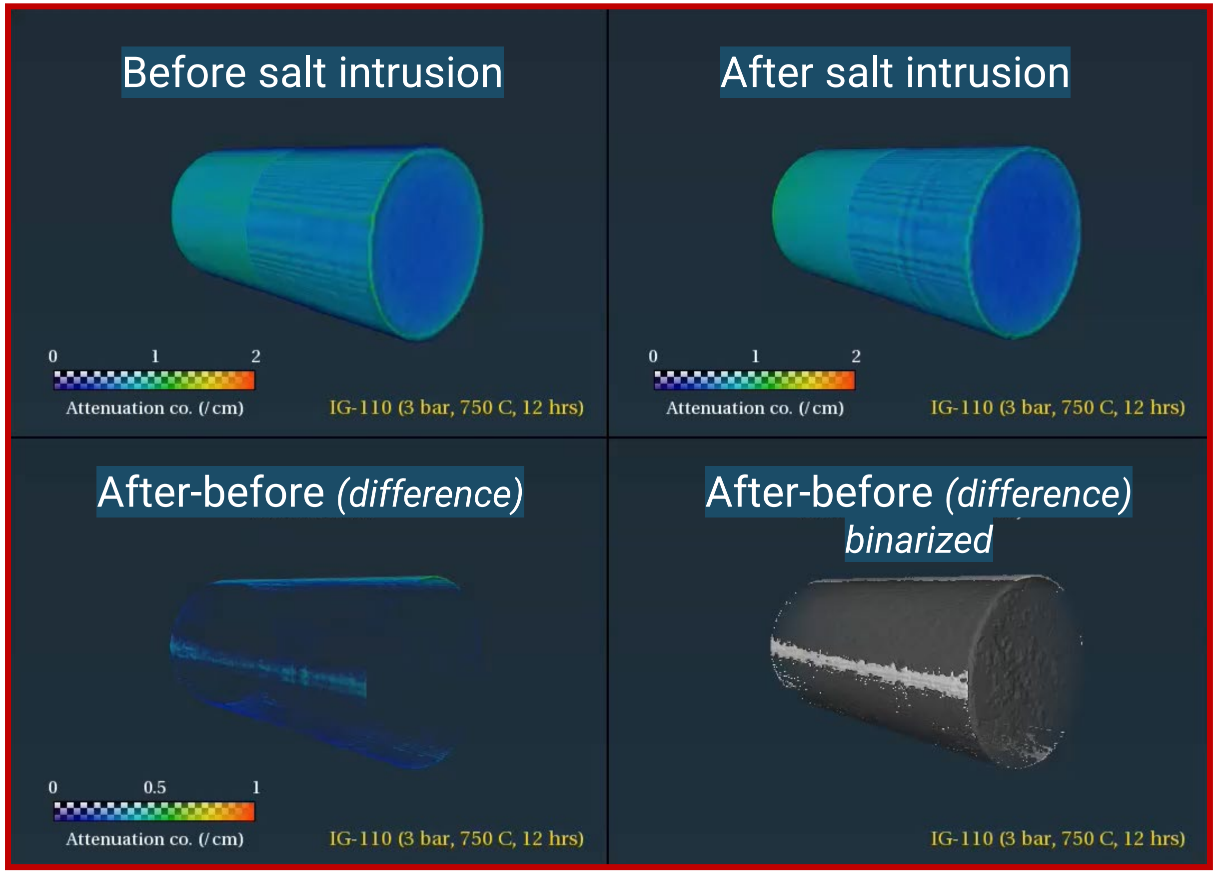


336 hrs

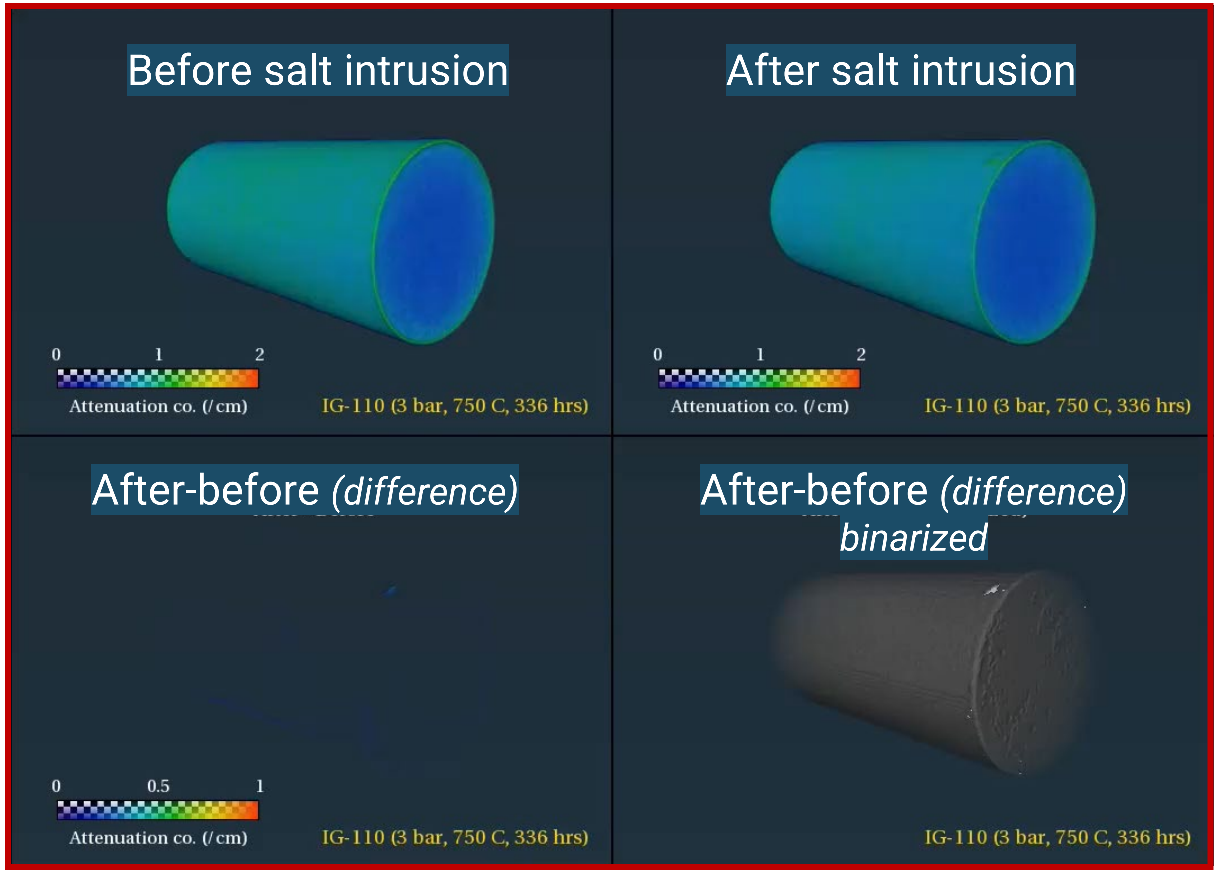


No major differences for FINE-grain graphite grade
IG-110 : 12 v 336 hrs

12 hrs



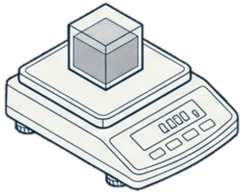
336 hrs



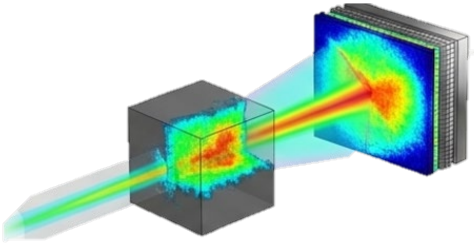
Coupling salt intrusion, nCT and wetting behavior

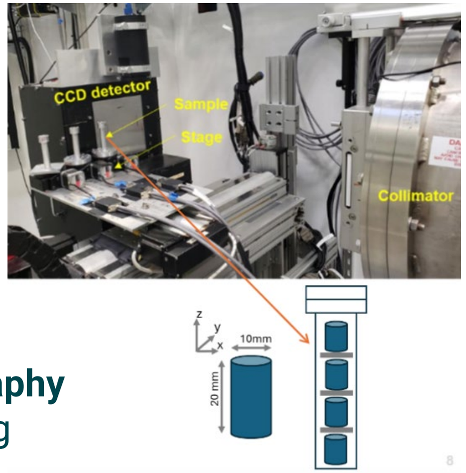
Infiltration study

Set up for high temperature,
pressure salt intrusion experiment



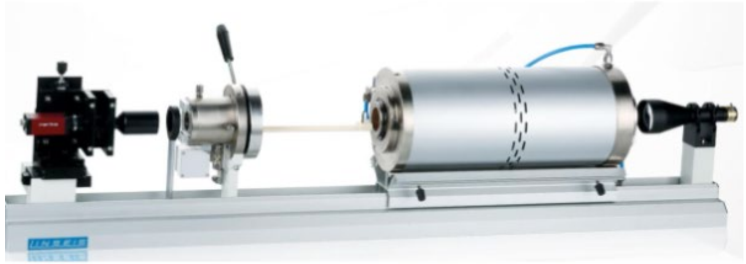
Gravimetric Uptake





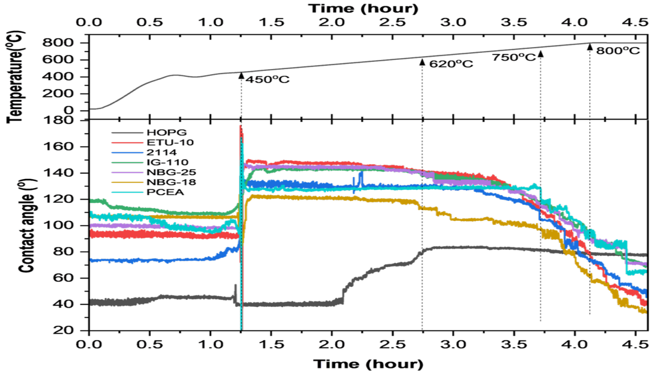
HFIR-MARS neutron tomography
Spatially Resolved 3D imaging

Wettability study



Measuring contact angle of FLiNaK using high temperature optical dilatometer

Temperature (°C)	500 °C	600 °C	700 °C	800 °C
IG-110	141° 142°	143° 142°	135° 134°	96° 98°
HOPG	41° 42°	72° 73°	81° 82°	79° 80°



Observed a lot of variability and inconsistencies in measurements

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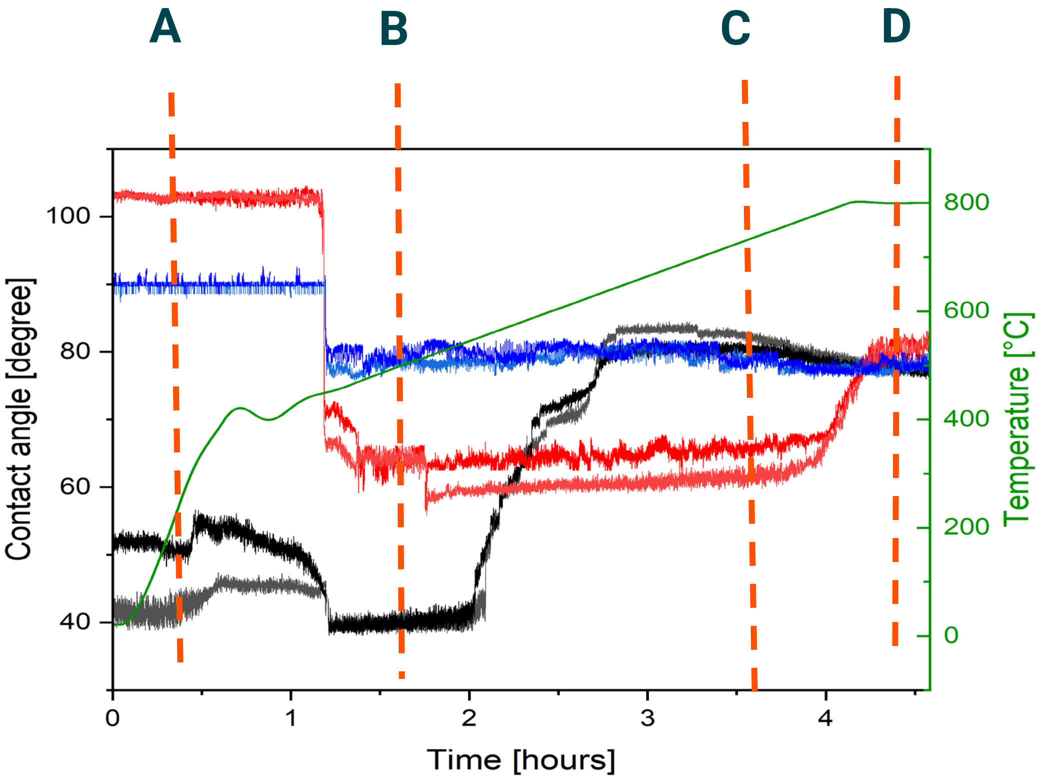
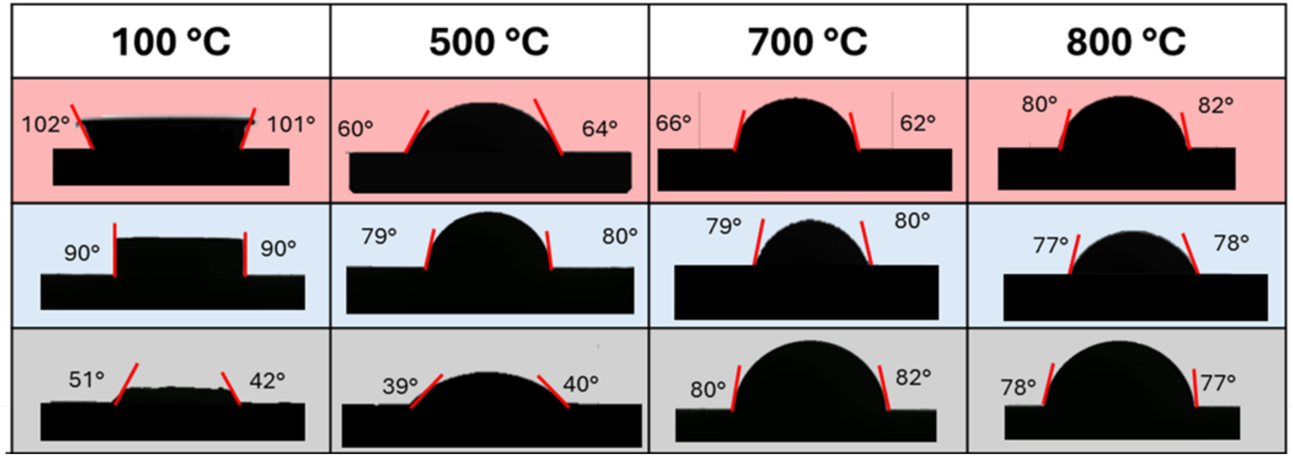
Use HOPG as a reference surface to understand the evolution of FLiNaK contact angle





Δ ~ 39°

Δ ~ 14°

Δ ~ 3°

Initial salt pellet geometry, significantly affects measured contact angle

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High-temperature surface energy and interfacial energy of molten FLiNaK–highly oriented pyrolytic graphite (HOPG) system from selected contact angle measurement

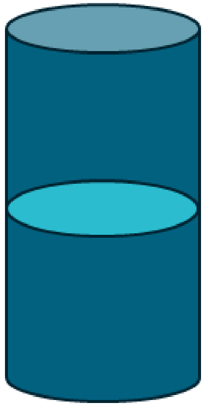
Jisue M. Braatz[1], Gang Seob Jung[2], Nidia C. Gallego[3]*

Submitted to Journal for review

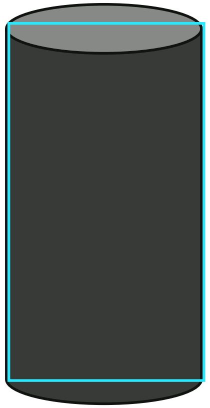
More recent activities using FLiBe

First FLiBe infiltration study: revealing spatial heterogeneity via n-CT

First demonstration of FLiBe infiltration in nuclear graphite under controlled temperature and pressure conditions. Neutron CT enables direct visualization of spatial infiltration, revealing pronounced heterogeneity in ET-10 despite similar bulk uptake.



Graphite: ET-10

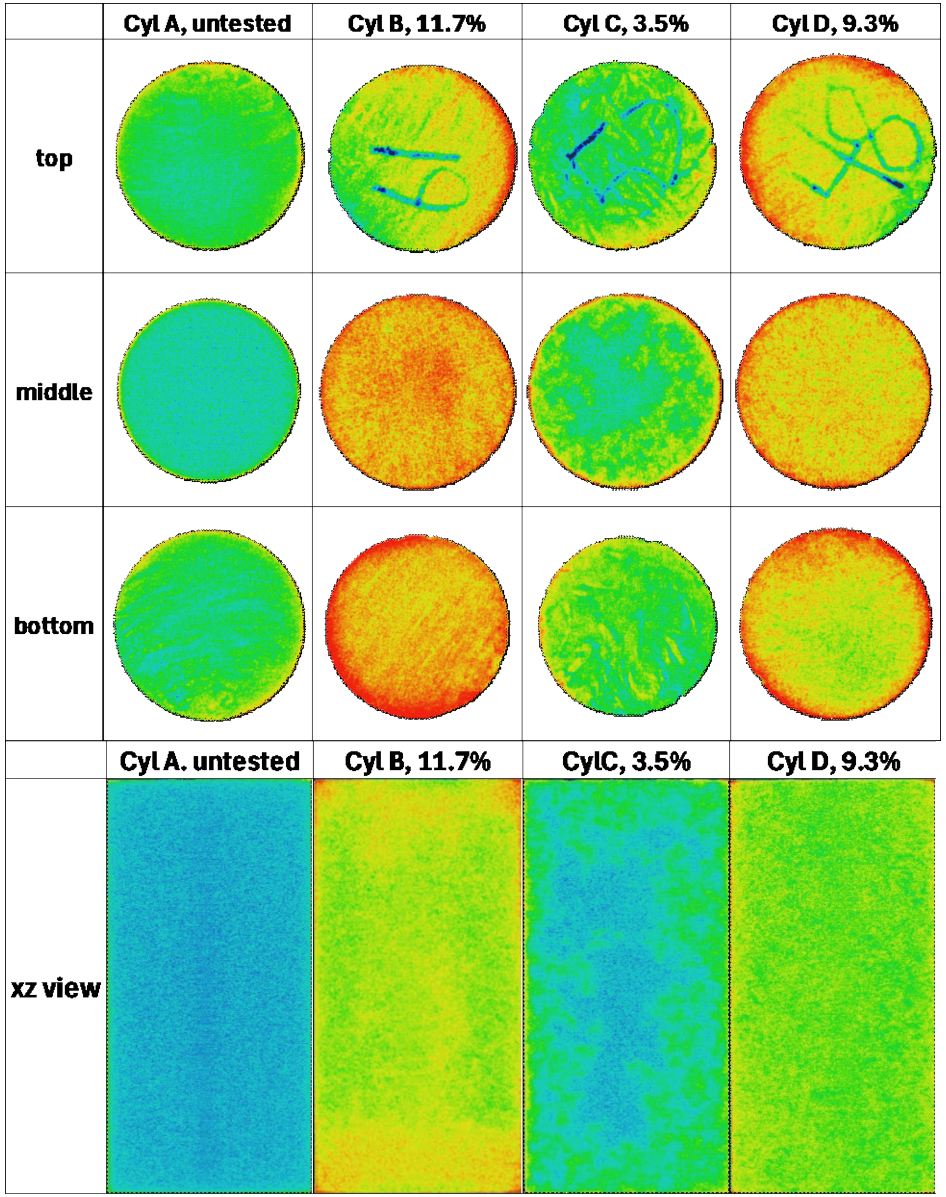


Infiltration result

Cyl A: untested

Cyl B: 11.7%

Cyl C: 3.5%



Ongoing work: Validation and reactor-relevant testing

Reproducibility and data validation

- Replicate key conditions(P,T, duration)
- Quantify variability across graphite grades
- Focus on anomalous/high infiltration cases
- Establish confidence bounds for infiltration measurement

Ensuring observed trends are robust and not condition-specific

Capsule testing (FLiBe & FLiNaK)

- Static capsule experiment at elevated temperature
- Thermal cycling conditions(realistic operation scenarios)

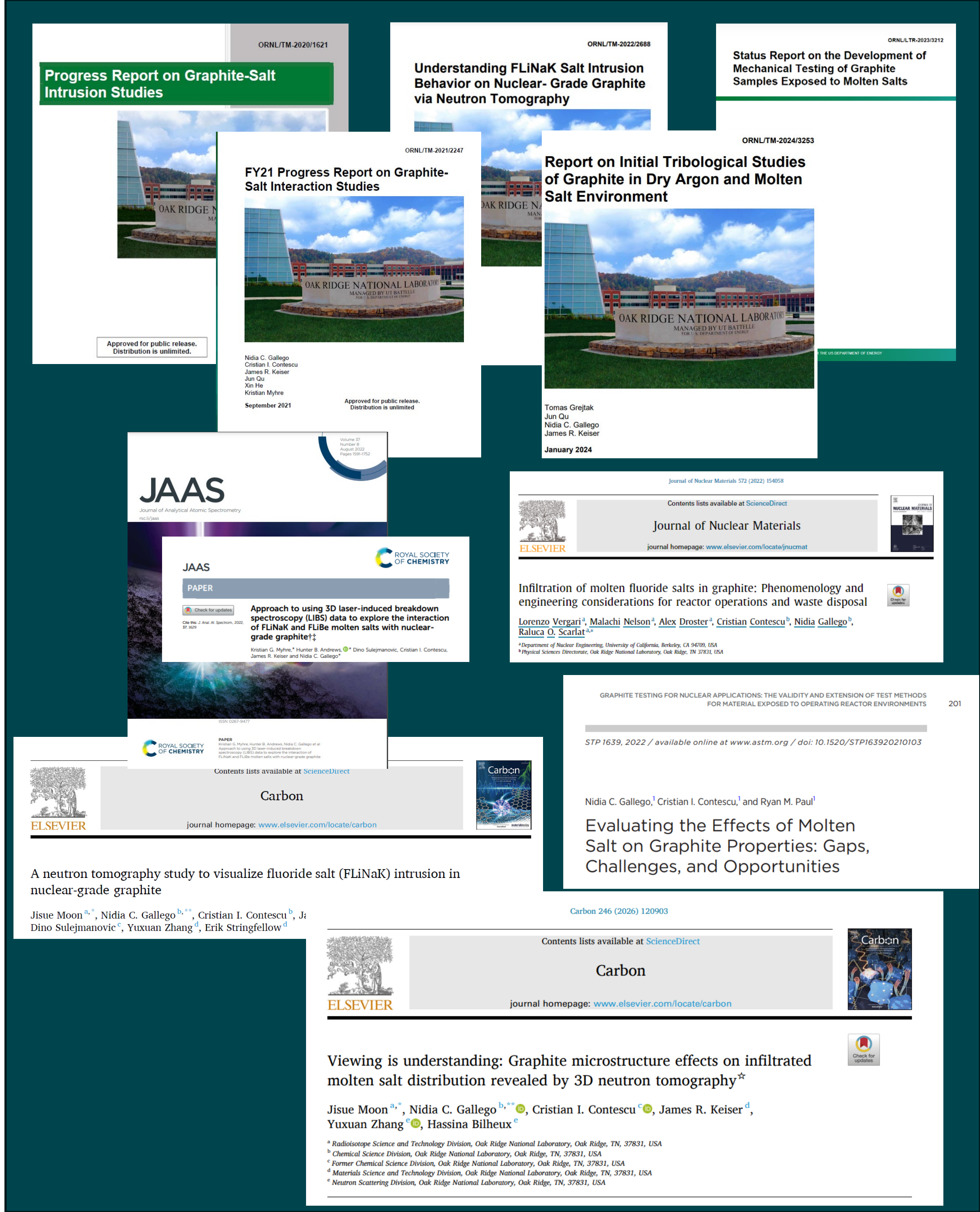
- Compare:
    - FLiNaK(model salt)
    - FLiBe(reactor salt)

- Evaluate:
    - Infiltration evolution over time
    - Stability under cycling

Bridging lab-scale measurement to reactor-relevant environments

Summary - Intrusion Activities

- Salt intrusion happens but it is highly dependent on temperature, pressure, time and graphite grade
- Different graphite grades evolve toward equilibrium at significantly different rates.
- Data analysis of samples exposed to other conditions of Temperature and Pressure continues.
- Accurate contact angle measurements are critical for the development of predictive models
- Static capsules for cyclic exposures

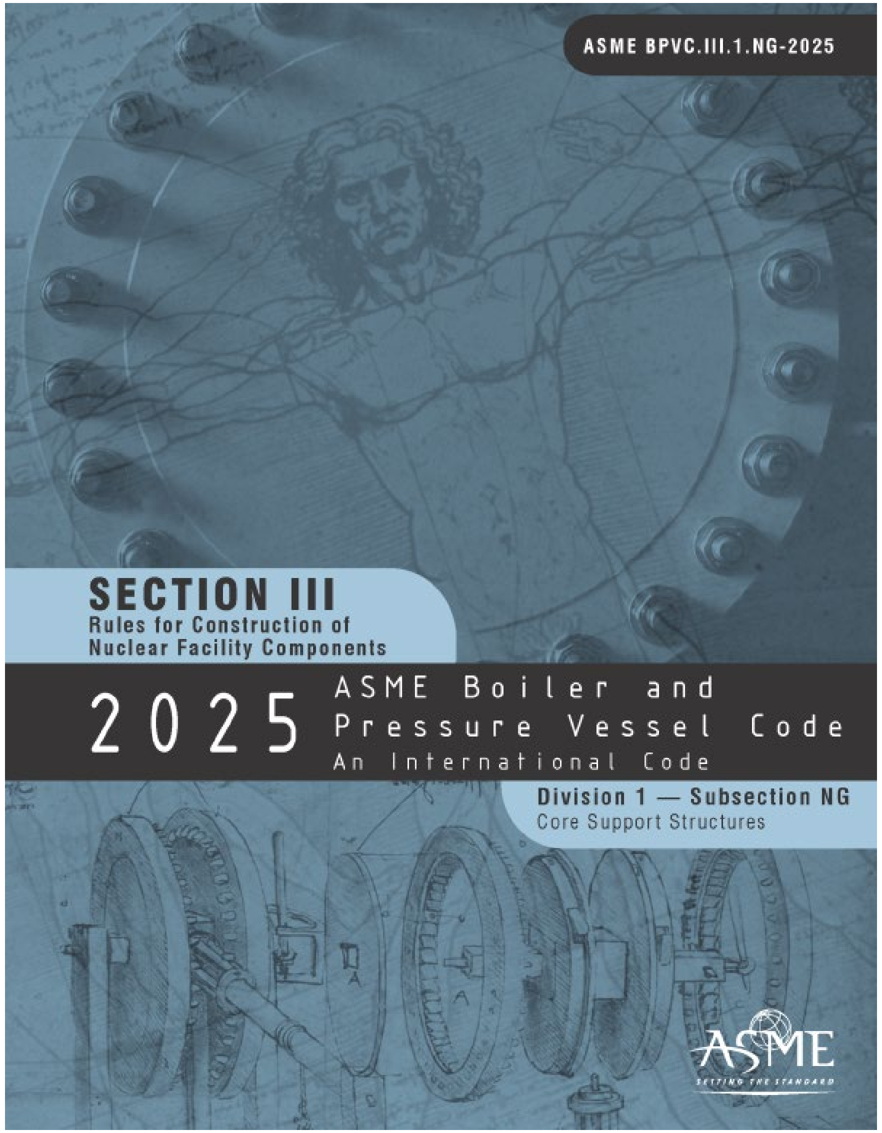


Participation in Codes and Standards Committees

ASME SEC III Division 5
High Temperature Reactors

The current HHA does not address any coolant salt interactions with graphite.

Chemical attack, salt infiltration and retention as well as wear and erosion aspects need to be incorporated in the design rules.



ASTM Standards:
D02.F0 on Manufactured Carbon and Graphite Products

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Upcoming FLiBe capabilities

GOAL

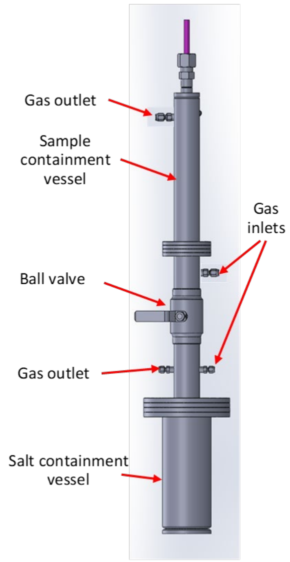
Establish capabilities for handling Be-containing fluoride salts [and fueled salts] and conduct research activities in support of the graphite program and the development of MSRs/FHRs

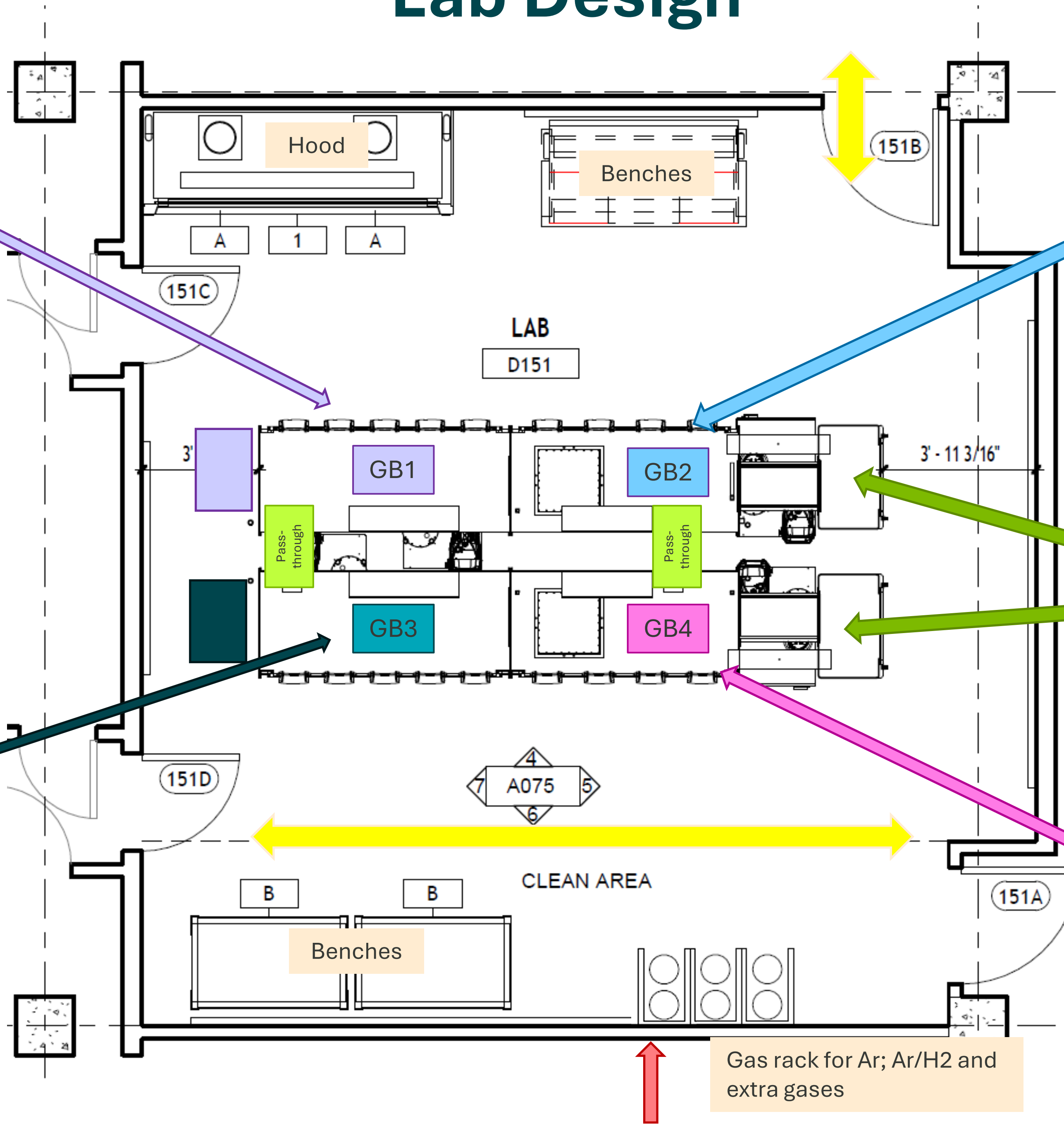
Lab space 4500S – D151 has been assigned for this work

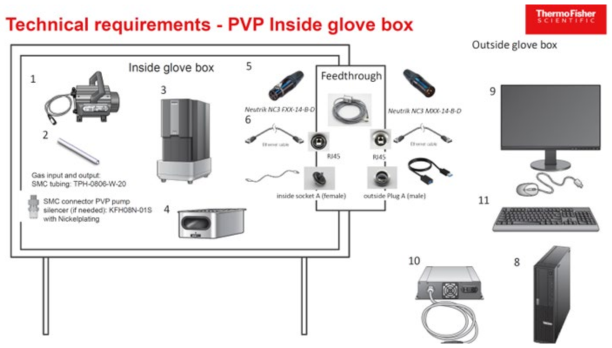
Lab Space Renovation

Lab Design

GB1 will have the salt intrusion system (furnace with high pressure, high temperature vessel)



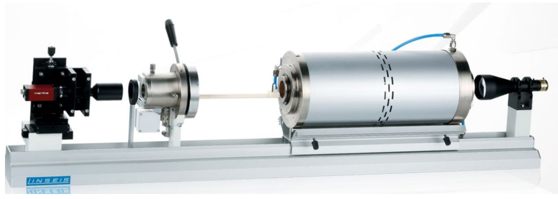




GB2 will have the tabletop SEM.

Two small 'hoods'' for staging, providing additional operator safety.

GB4 will have the contact angle system

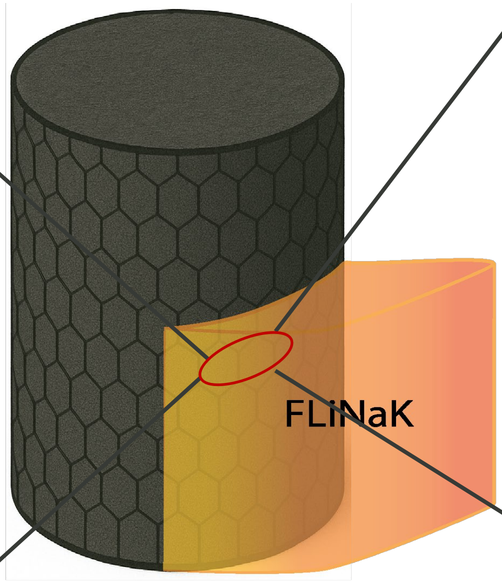


GB2 general purpose: sample preparation, storage, outgassing etc; balance, press, vacuum oven, tools, containers, etc

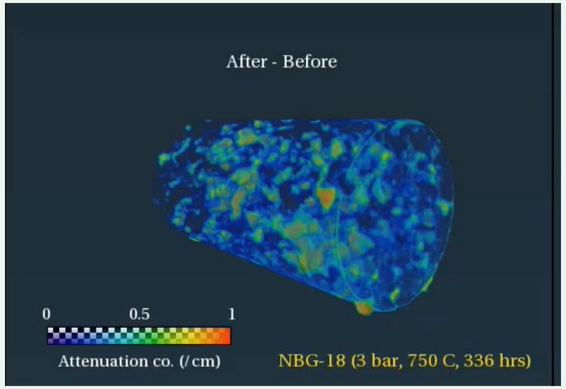
Gas rack for Ar; Ar/H2 and extra gases

OAK RIDGE National Laboratory

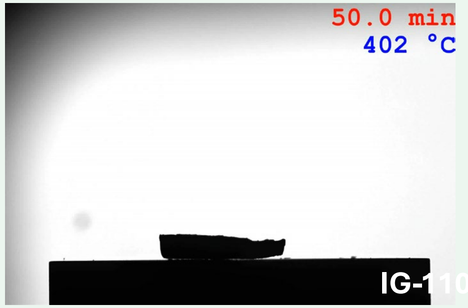
Understanding graphite–salt interactions via a multidisciplinary approach



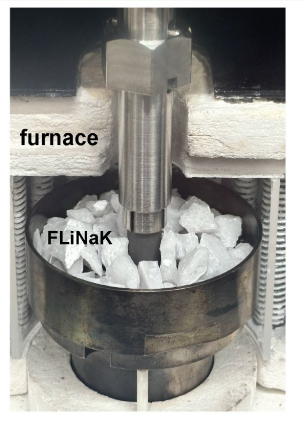
Salt intrusion visualization:
3D neutron tomography

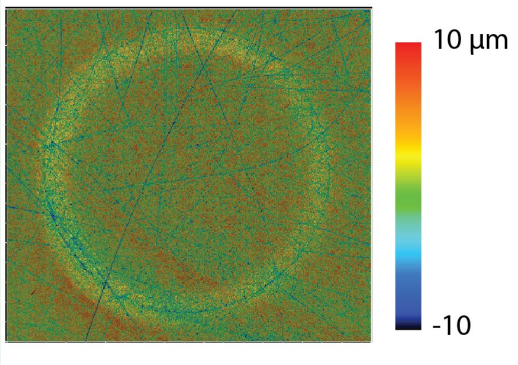


Salt intrusion prediction:
contact angle measurements

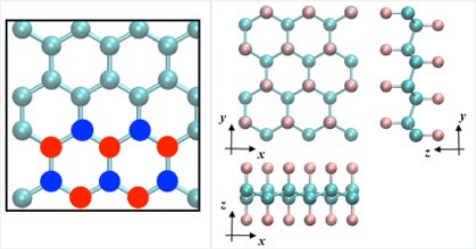


Wear and friction behavior of graphite in molten salts:
tribological characterization



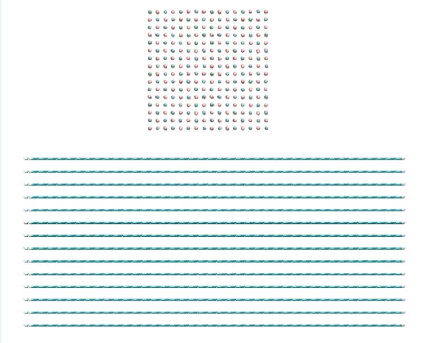


Modeling chemical interactions - Fluorination



NRC-funded

Modeling the wetting behavior:
molecular dynamics simulations



Want additional information on Graphite?

DOE-ART | NEUP Website | NEUP Prj Files

NDMAS

DOE-ART Graphite

The DOE-ART Graphite R&D program is the primary nuclear graphite research program for the USA. This program focuses on research and development activities necessary to qualify and license graphite components for use within nuclear applications, specifically within advanced reactor designs such as High Temperature Reactor designs. The data generated within the ART Graphite program is intended to be used in conjunction with other publicly available nuclear graphite data such as is contained within the IAEA Nuclear Graphite Knowledge Base. The ART Graphite program is divided into 5 primary research areas providing a combination of data, analysis reports, and pertinent references to describe and explain the trends within the data.

- **Unirradiated (Baseline):** Establish as-manufactured (Baseline) values for unirradiated material properties that can be used to determine the quantitative changes during irradiation and degradation during nuclear applications.
- **Irradiation (AGC Experiment):** Establish evolution of material property changes due to irradiation dose and temperature. The AGC Experiment is an irradiation creep experiment which provides creep data for selected graphite grades.
- **Degradation:** Establish effects of irradiation, oxidation, and molten salt interaction on graphite behavior.
- **Behavior models:** Predictive and degradation models for graphite behavior.
- **Licensing and code:** Papers and data supporting ASME code development and NRC license assessment.

Baseline (Unirradiated)

AGC (Irradiated)

IAEA

ART Advanced Reactor Technologies GAS-COOLED REACTORS

NDMAS Nuclear Data Management and Analysis System

IAEA.org NUCLEUS

IAEA NGKB Nuclear Graphite Knowledge Base

Search this site

Open Knowledge Wiki | Restricted Resources | NGKB Meetings

Home

Welcome to the IAEA Nuclear Graphite Knowledge Base

Mission Statement

The aim of this knowledge base is to support the preservation and sharing of expert knowledge and experience, across the international Graphite Community.

The knowledge base contains two levels of knowledge:

- General information on the subject of Nuclear Graphite
- Specialist knowledge, secured for members of the international project

Please click on the below images to link to the desired section of the knowledge base

Databases

INGSM | Graphite Database | Restricted Knowledge Reports | Open Knowledge Wiki

SAVE-THE-DATE

Joint EPRI Nuclear Graphite Workshop & DOE-NE ART Gas-Cooled Reactor Program Review

JULY 07-09, 2026

Join us at Idaho National Laboratory in Idaho Falls, ID.

This joint event brings together industry, national laboratories, reactor developers, and technical experts to assess nuclear graphite supply challenges and opportunities, as well as qualification and performance for advanced gas-cooled reactors. Discussions will highlight graphite manufacturing capabilities, qualification progress, and key challenges impacting future deployment.

The program features an EPRI-led, industry-focused workshop and technical sessions addressing graphite supply chain readiness, high-temperature materials qualification, and HTGR core and system modeling capabilities, along with ongoing efforts supporting gas-cooled reactor deployment. The EPRI workshop (July 7) is in-person only, while the ART-GCR Program Review (July 8–9) will also offer a virtual option. A technical tour of in-town graphite and metals testing and qualification laboratories will be offered on July 8.

Register Here

JULY 07 — EPRI Nuclear Graphite Workshop

This workshop convenes graphite stakeholders to address challenges in nuclear-grade graphite.

- Graphite supply chain readiness, manufacturing, & qualification
- Engineering design considerations
- Operational performance and maintenance
- Regulatory engagement and standards development

JULY 08 — ART-GCR: Graphite

Technical updates on graphite qualification activities supporting the DOE-NE ART program:

- Irradiation testing and materials characterization
- Graphite degradation (oxidation, molten salt, wear, fracture)
- Qualification, modeling, & standards (ASME/ASTM)
- Integration with fuel & core design

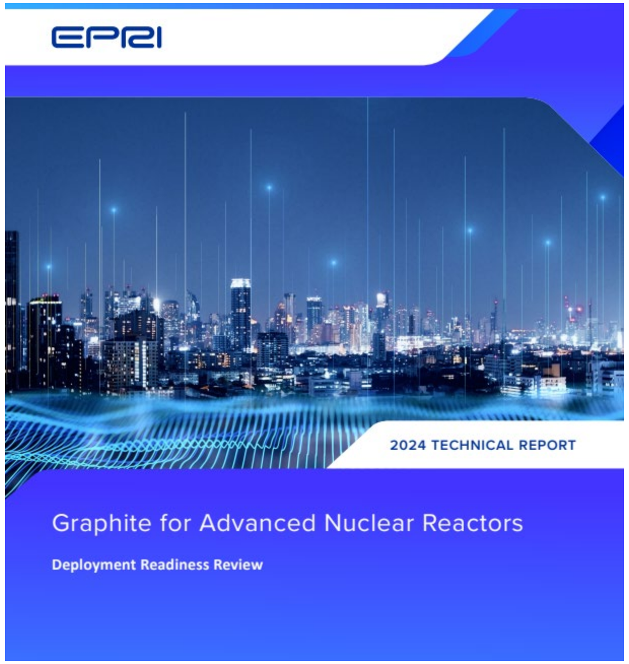
JULY 09 — ART-GCR: Methods and Metals

Program updates on materials and modeling supporting gas-cooled reactor deployment:

- Structural alloy qualification progress
- Environmental effects and materials code development
- Advanced core simulation tools and validation
- International benchmarking activities

HOSTED BY

EPRI ELECTRIC POWER RESEARCH INSTITUTE | GAS-COOLED REACTOR ADVANCED REACTOR TECHNOLOGIES PROGRAM | INL Idaho National Laboratory



INGSM 2026
TOYOTANSO / JAEA

METI | 経済産業省 資源エネルギー庁

Shaping the Future of Nuclear Energy

Together with Graphite

Schedule: October 18 (Sun) – 23 (Fri), 2026

Venue: Grand Front Osaka

https://ingsm26.toyotanso.co.jp/

OAK RIDGE National Laboratory

Questions?

gallegonc@ornl.gov