Extended Storage Collaboration Program (ESCP): Collaborative Forum for Addressing Global Technical Challenges Around Used Fuel

Welcome to the ESCP 2025 Meeting

Hatice Akkurt
Used Fuel and High-Level Waste Management, EPRI

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 27-30, 2025

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Extended Storage Collaboration Program (ESCP)

Mission

- **Enhance the technical bases to ensure continued safe, long term used fuel storage and future transportability**

Goals

- Bring together US and International organizations engaged with active or planned R&D in used fuel area
- Share information
- Identify common goals and needs
- Identify potential areas of “formal” collaborations

Phases

- **Phase 1:** Review current technical bases and conduct gap analysis for storage systems
- **Phase 2:** Conduct experiments, field studies, and additional analyses to address gaps
- **Phase 3:** Long-term performance confirmation

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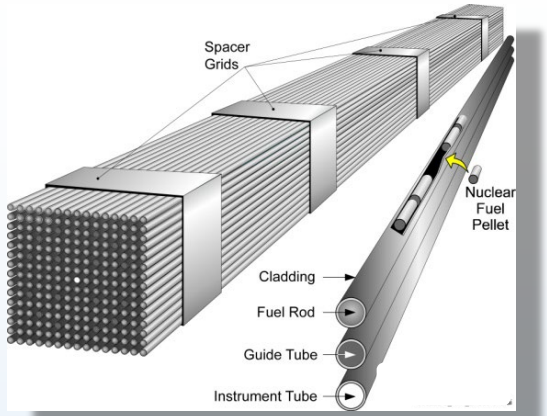
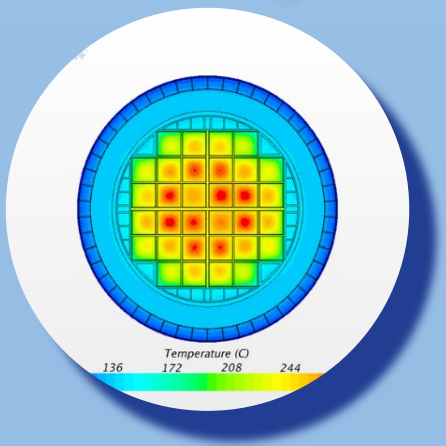
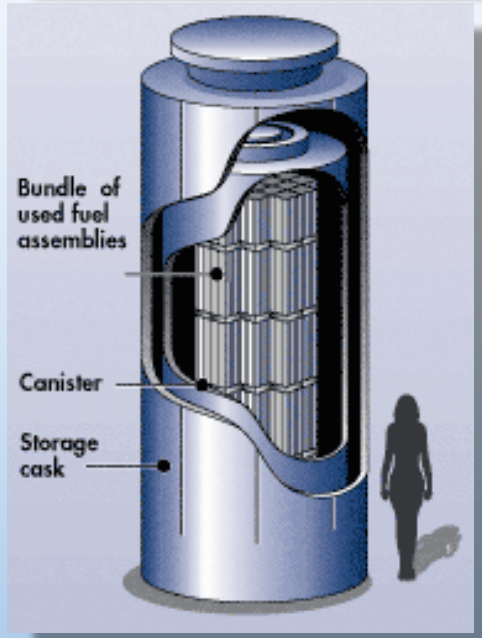
Managing Extended Storage of Used Fuel: Technical Challenges







Fuel Cladding Integrity

Aging Management of Dry Storage Systems

Accuracy of models: Thermal, decay heat, and dose models

Collaborative R&D to Inform and Transform

ESCP is a collaborative forum for addressing global challenges

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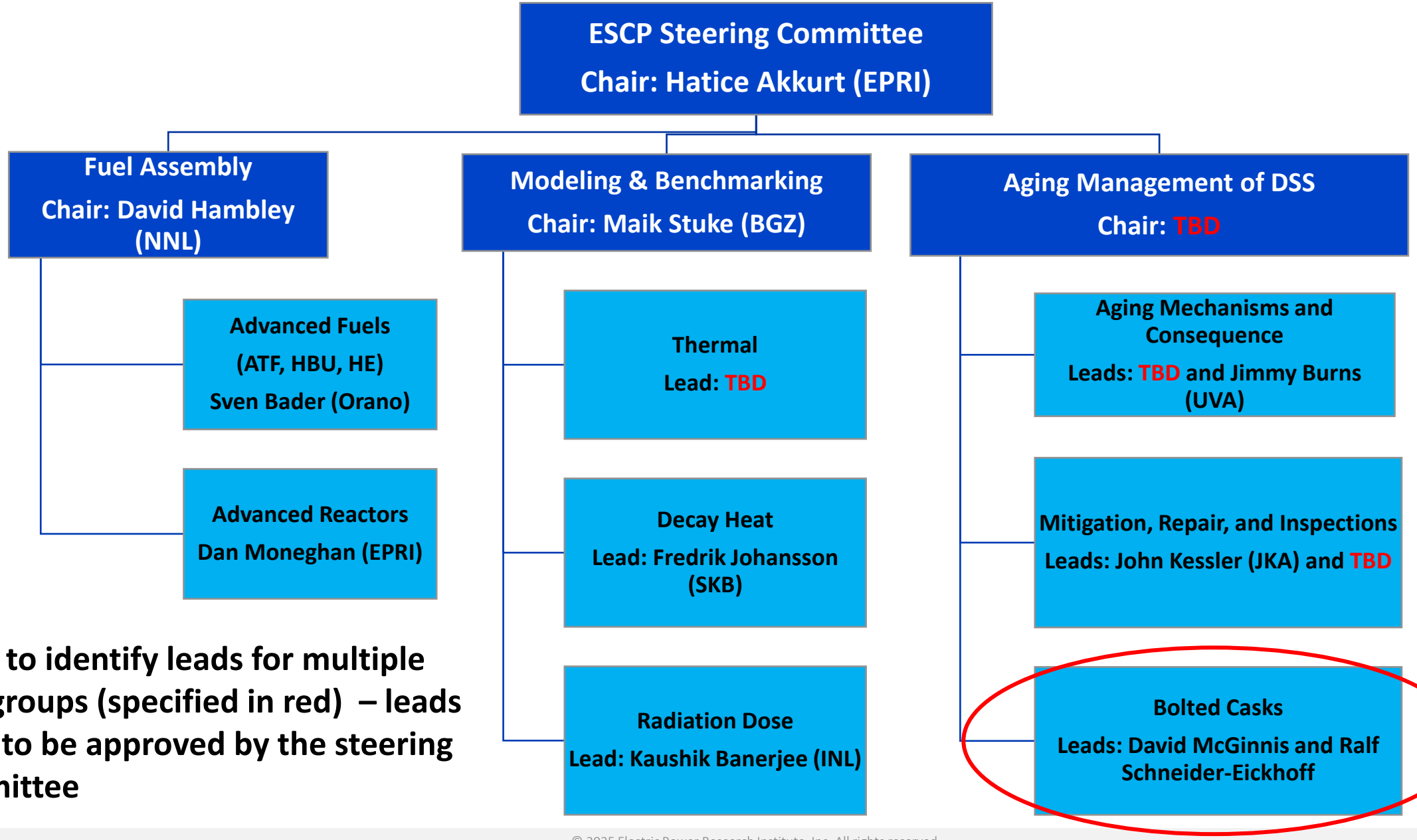
Updates since ESCP 2024 meeting

New ESCP Structure - Subcommittees (SCs) and Task Groups (TG)



Need to identify leads for multiple task groups (specified in red) – leads need to be approved by the steering committee

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ESCP International Thermal Modeling

- International Thermal Modeling Benchmark Project – Results: An Extended Storage Collaboration Program (ESCP) Activity report published on October 23, 2025, and the report is publicly available
- Includes participation from 10 organizations representing 8 countries
- Discussion with the participants on the next phase
- Will continue to be active based on participants' feedback
- **More details on the results and next phase will be presented on Thursday**

EPRI ESCP report, 3002032045, published on October 23, 2025 and the report is publicly available

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ESCP Aging Management Working Group

- Key recommendations from ESCP Aging Management Working Group
- Report is not published yet but there was a PATRAM paper
- While there is some progress on some of the key recommendations, some of them have not even started yet
- **Updates on the status of the key recommendations on Wednesday**

EPRI ESCP report on Aging Management Working Group is expected to be published before the year end

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ESCP Forward Looking

Forward Looking ESCP Focus Areas for Next 2-3 Years

Fuel

- Resolution of creep and annealing issues
- Transport of HBU Demo cask and opening
- Increased focus on ATF/HE/HBU and back-end effects
- Increased focus on Advanced Reactors and back-end issues

Aging Management and Canister Integrity

- Evaluation of inspection results and trending
- Mitigation techniques development
- Demonstration via field tests
- Acceleration of consequence studies

Modeling & Benchmarking

Thermal:

- Completion of international thermal modeling project
- Gathering more benchmark data during inspections

Dose:

- Blind benchmarking activity for dose modeling

Decay Heat:

- Completion of decay heat reports
- New measurements and potential for blind benchmark

Collaborative R&D to Inform and Transform

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Save the Dates: ESCP 2026 Meeting

October 26-29, 2026

EPRI Charlotte, NC office

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Questions/Comments?

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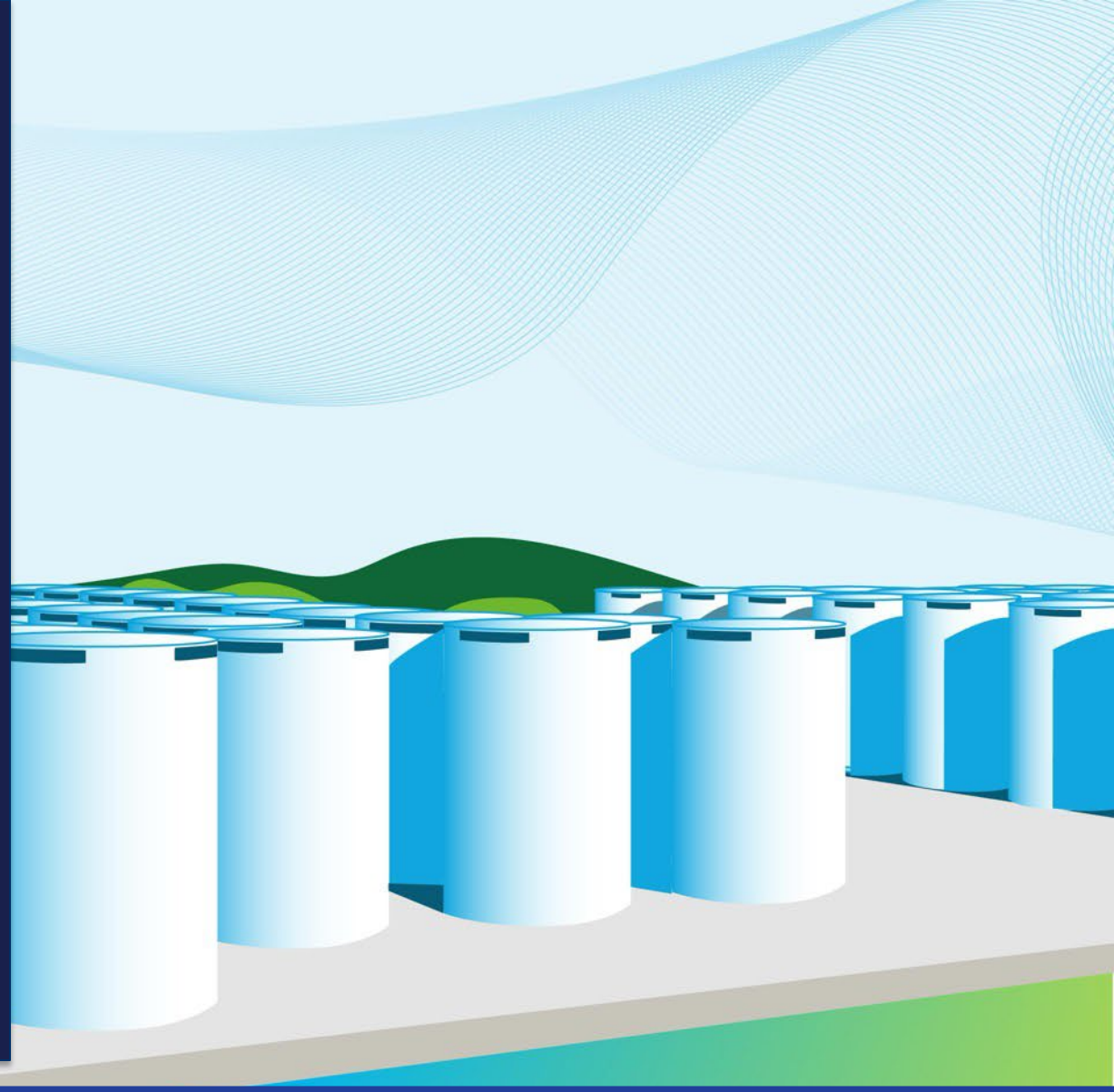
Spent Nuclear Fuel Center for Applied Research

Paul Cantonwine[1] and Steven Vitto[2]

[1] Oak Ridge National Laboratory

[2] Department of Energy

ESCP 2025



U.S. DEPARTMENT of ENERGY | Office of Nuclear Energy

Spent Fuel and High-Level Waste Disposition

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961)

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and constructing of a spent nuclear fuel repository.

Ensuring Transportation and Long-Term Storage of Spent Nuclear Fuel (SNF) and High-Level Waste (HLW)

- The U.S. Department of Energy (DOE) is making plans to deploy the Spent Nuclear Fuel Center for Applied Research (SNF Center).

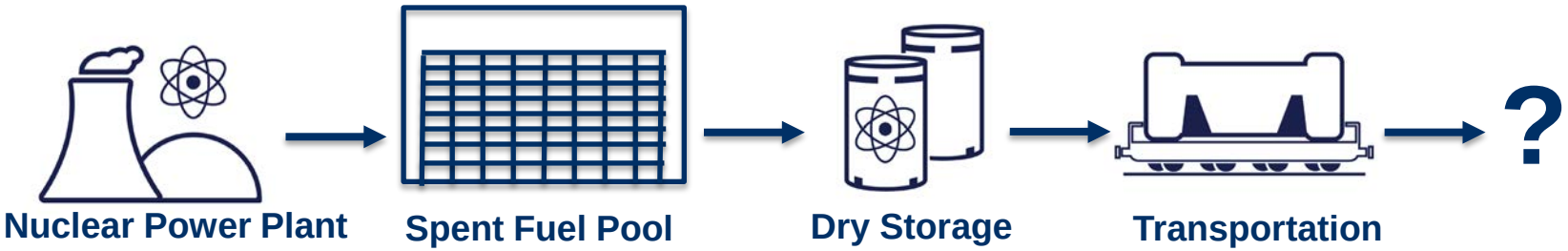
Mission: Coordinate efforts in developing innovative technologies and solutions, conduct research that helps support and maintain regulatory compliance, and ensure public confidence in the storage and transportation of commercial and DOE-managed SNF and HLW

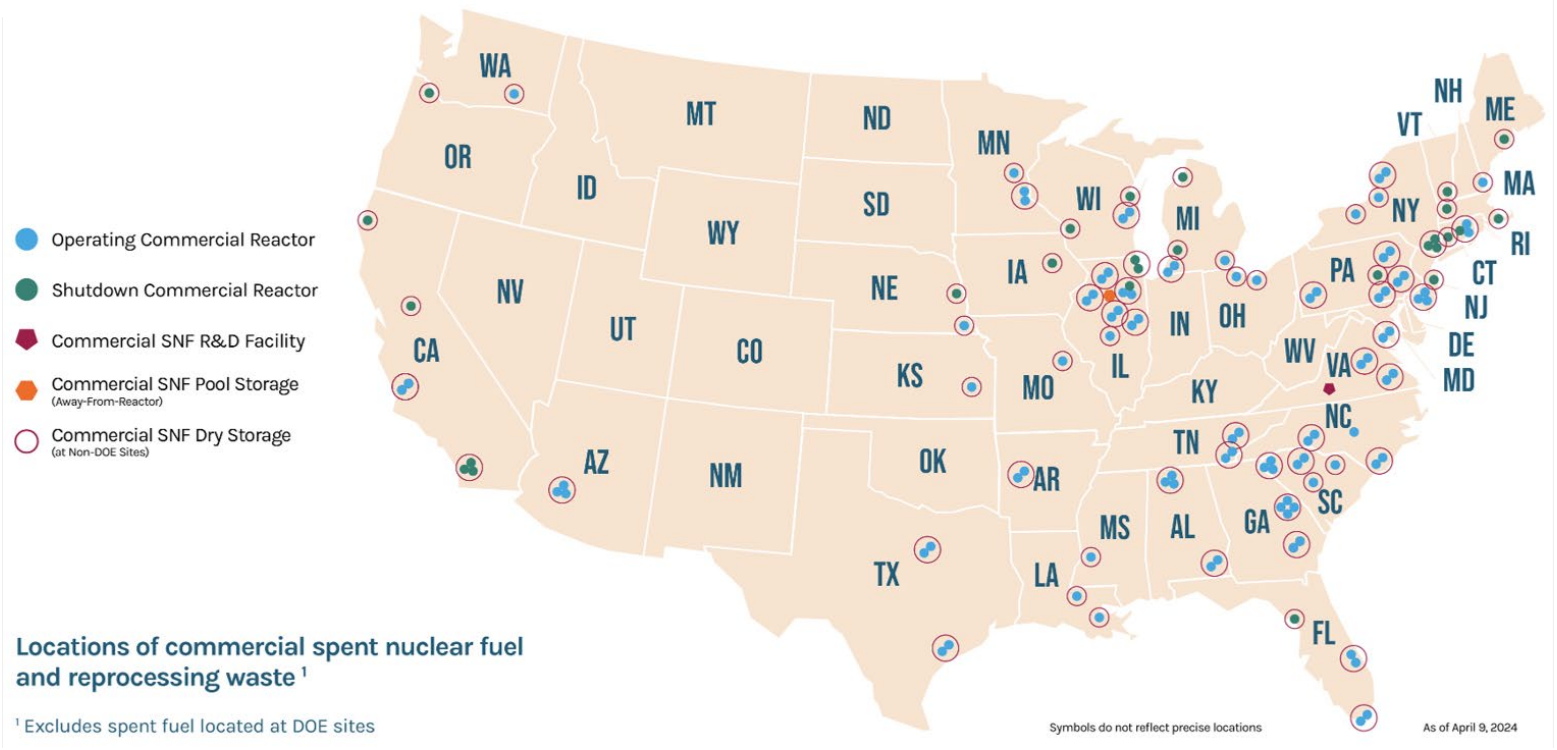
Reduce DOE's contractual and technical liabilities.

Leverage U.S. and international research and operational experience.

Furnish data and methods to support extended storage and transportation.

SNF Storage in the United States [1]





As of 2024

94 operating commercial reactors at 54 nuclear power plants in 28 states

- 20 nuclear power plants have been shut down
- ~95,000 metric tons of SNF

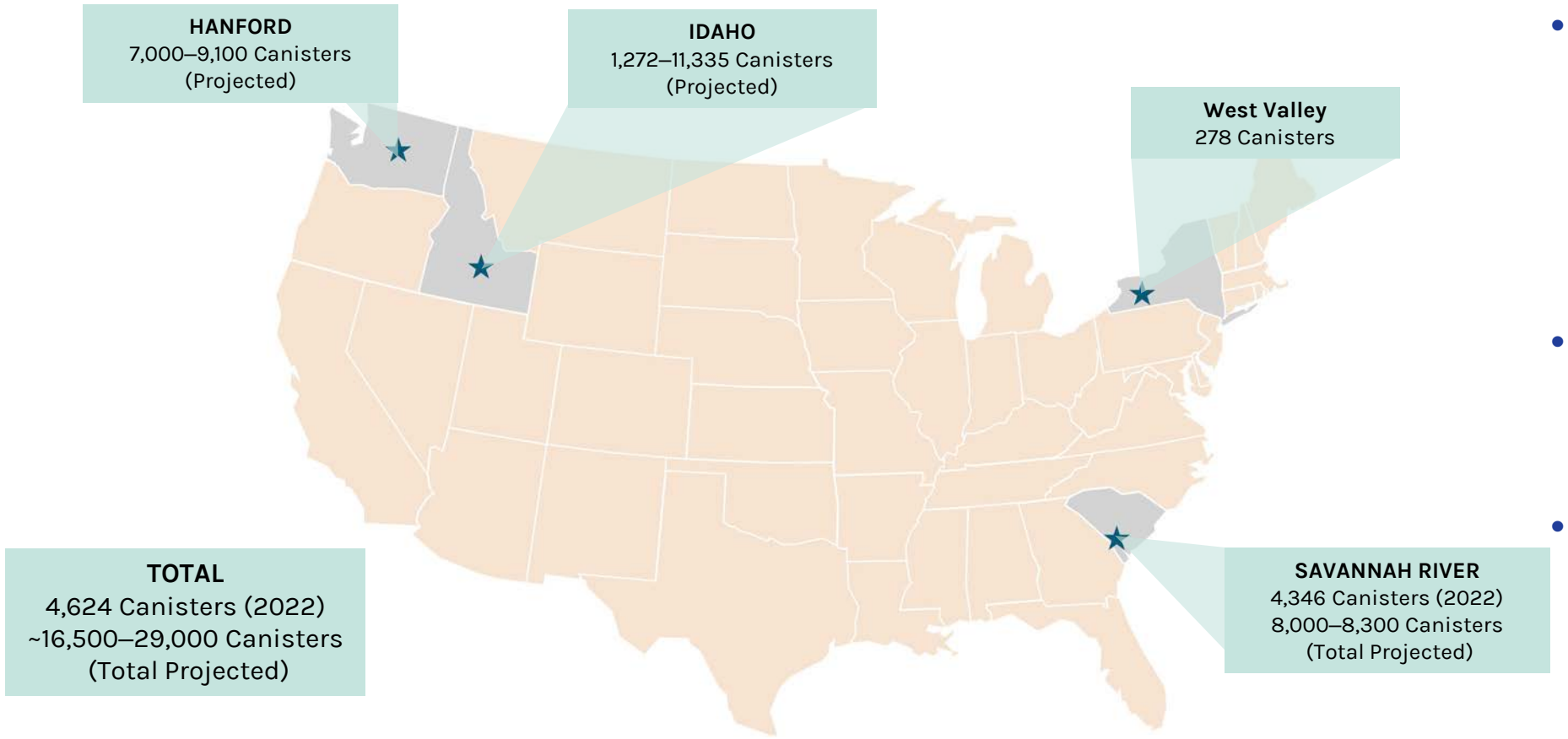
Current Fleet Projections

The United States is estimated to have up to approximately 180,000 metric tons of commercial spent nuclear fuel.

DOE-managed SNF

4 primary sites, containing ~2,450 metric tons of spent nuclear fuel

SNF Reprocessing Waste / HLW [1]



- Savannah River and West Valley have vitrified HLW and packaged it into canisters.
- Hanford is planning to vitrify their tank waste.
- Idaho contains calcine waste and waste stemming from electrochemical metallurgical treatment.

Source: Adapted from information in *Spent Nuclear Fuel and Reprocessing Waste Inventory*, PNNL-33938, Rev. 1, August 2024.

Partners to the SNF Center

- The SNF Center will operate out of Idaho National Laboratory based on a hub-and-spoke model
  - Hub = physical and intellectual location
  - Spokes = principal partners, associates, and other participants
- Accountable to the DOE-NE Deputy Assistant Secretary for Spent Fuel and HLW Disposition
- US Industry/Electric Power Research Institute (EPRI)
- DOE Nuclear Engineering University Program awards
- Small Companies through the small business innovation awards
- International programs

SNF Center Work Scope and Products

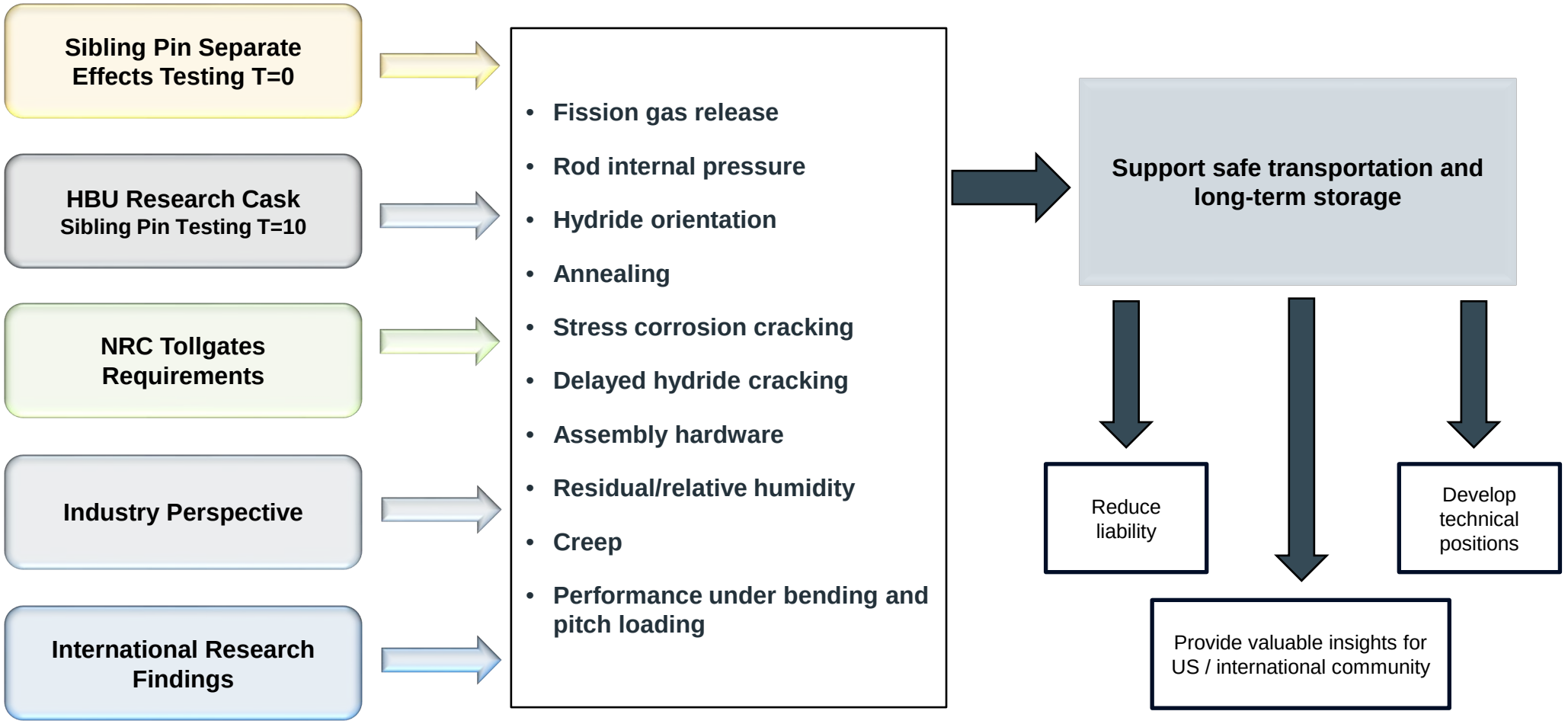
- Organization
  - Principal: representatives from U.S. national laboratories conducting SNF/HLW research
  - Associate: additional laboratory staff, storage and transport industry leads, academia
  - Advisory: Nuclear Energy Institute, Nuclear Regulatory Commission, and EPRI*
- Proposed activities
  - Emerging Issues (1 year)
  - Technical Gap Closures (3 years)
  - First-of-a-Kind Research (>3 years)
- All proposed activities funded by DOE will have clear objectives with defined outcomes

*International collaboration is currently envisioned through EPRI

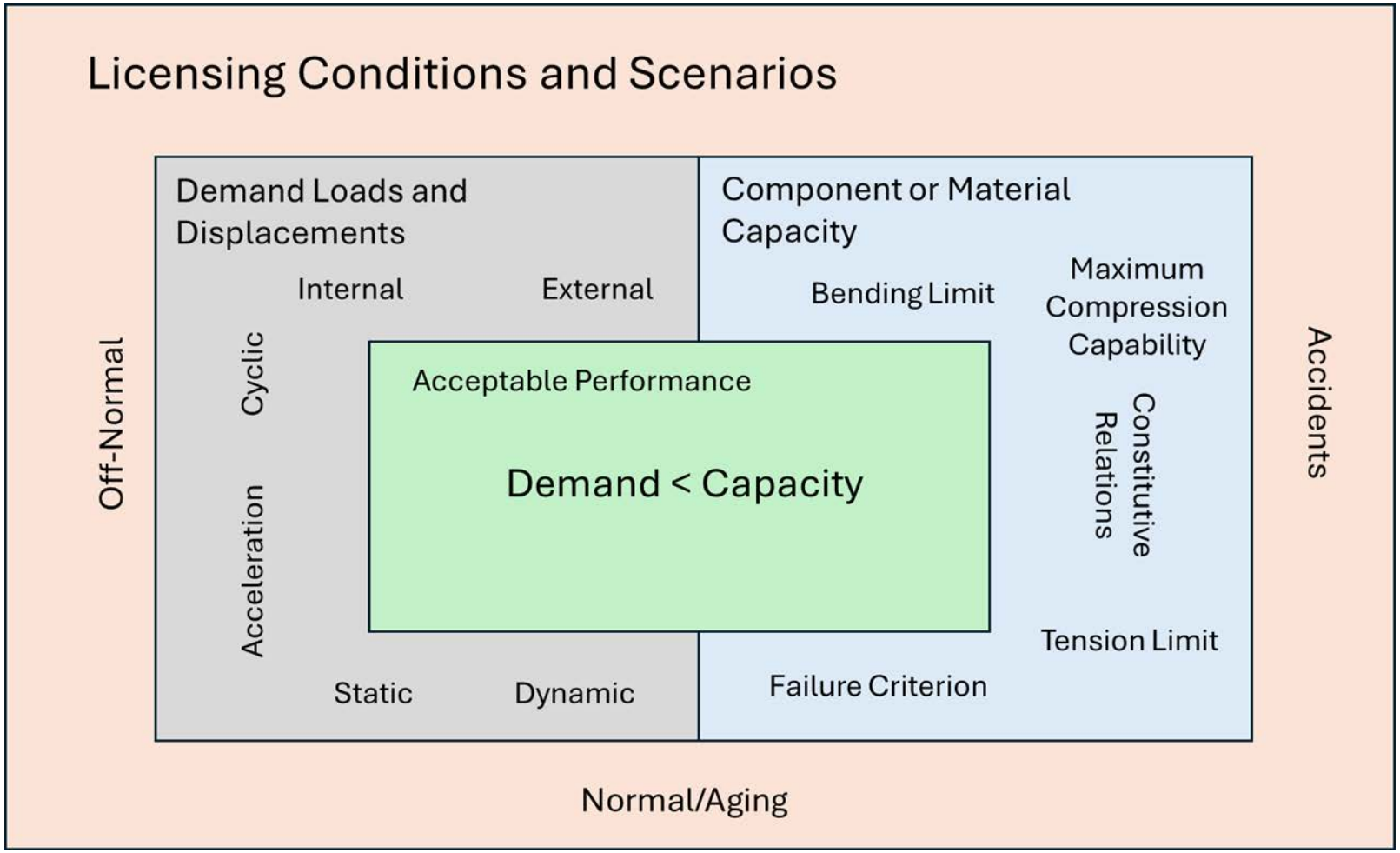
DOE's High Burnup Research Cask is a high priority for the SNF Center

- In 2013, a research project concerning high-burnup SNF started as a collaboration between DOE and the EPRI.
- Thirty-two pressurized-water reactor high-burnup SNF assemblies are stored in the HBURC.
- For the SNF assemblies in the cask, there are four different cladding types:
  - Zircaloy-4, low-tin Zircaloy-4, Zirlo®, M5®
- At the time of shipment, the SNF will have been in dry storage, representing about 10 years' worth of data.
- The project will provide data to support safe storage of SNF for DOE's future federal CSF(s) and for most currently operating U.S. nuclear power plants.
- Information From [3]

SNF Center will plan what pins to pull and what exams to perform after HBURC is opened



Technical gaps will be identified using an evaluation-based gap analysis that considers both predictions of the demand loads and understanding of component capacity



There may be so much margin between demand and capacity that more uncertainty in demand or capacity may be acceptable.

The Spent Nuclear Fuel Center for Applied Research: Summary

- DOE is planning to deploy the Spent Nuclear Fuel Center for Applied Research.
  - Mission: Coordinate efforts in developing innovative technologies and solutions, conduct research that helps support and maintain compliance, and ensure public confidence in the storage and transportation of commercial and DOE-managed SNF and HLW.
  - Goal 1: Reduce DOE's contractual and technical liabilities associated with the management of SNF and HLW prior to their undergoing permanent disposal.
  - Goal 2: Leverage lessons-learned from countries with established SNF storage and transportation programs
  - Goal 3: Furnish data and methods to support extended storage and transportation of SNF and HLW prior to their eventual disposal.
- The Center will operate using a hub-and-spoke model to enhance collaborations with other laboratories, industry, and academia.
- The Center's activities will have clear objectives with defined outcomes.

References

1. Banerjee, K., Rigato, A., & Wilson, V. (2024, December 1). *Spent Nuclear Fuel and Reprocessing Waste Inventory*. PNNL. https://curie.pnnl.gov/system/files/SNF%20and%20Rep%20Waste%20Inventory%20PNNL%2033938%20Rev.%201.1_0.pdf
2. U.S. Department of Energy, *Agency Financial Report, Fiscal Year 2024*, DOE/CF-0211 https://www.energy.gov/sites/default/files/2024-12/fy-2024-doe-agency-financial-report.pdf
3. Waldrop. 2019. Waldrop, K., Mcgee, D., Tomlinson, D. *High Burnup Spent Fuel Dry Storage Research Project*. Institute of Nuclear Materials Management 2019. https://resources.inmm.org/patram-proceedings/high-burnup-spent-fuel-dry-storage-research-project

BGZ Gesellschaft für Zwischen-lagerung mbH

BGZ's Research on Inventory

Dr. Maik Stuke

EPRI-ESCP Winter Meeting

October 28th 2025

BGZ's task: Safe interim storage of LLW and HLW

Sole shareholder of the BGZ:
Federal Republic of Germany, represented by the BMUKN

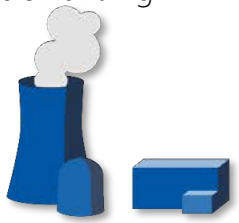
Federal Ministry
for the Environment, Climate Action,
Nature Conservation and Nuclear Safety

100 %

BGZ Gesellschaft für Zwischenlagerung mbH

Our mandate: **Safe interim storage**

Decommissioning & dismantling



NPP

Packaging of different types of waste

Nuclear fuel and waste from reprocessing (high-level waste)

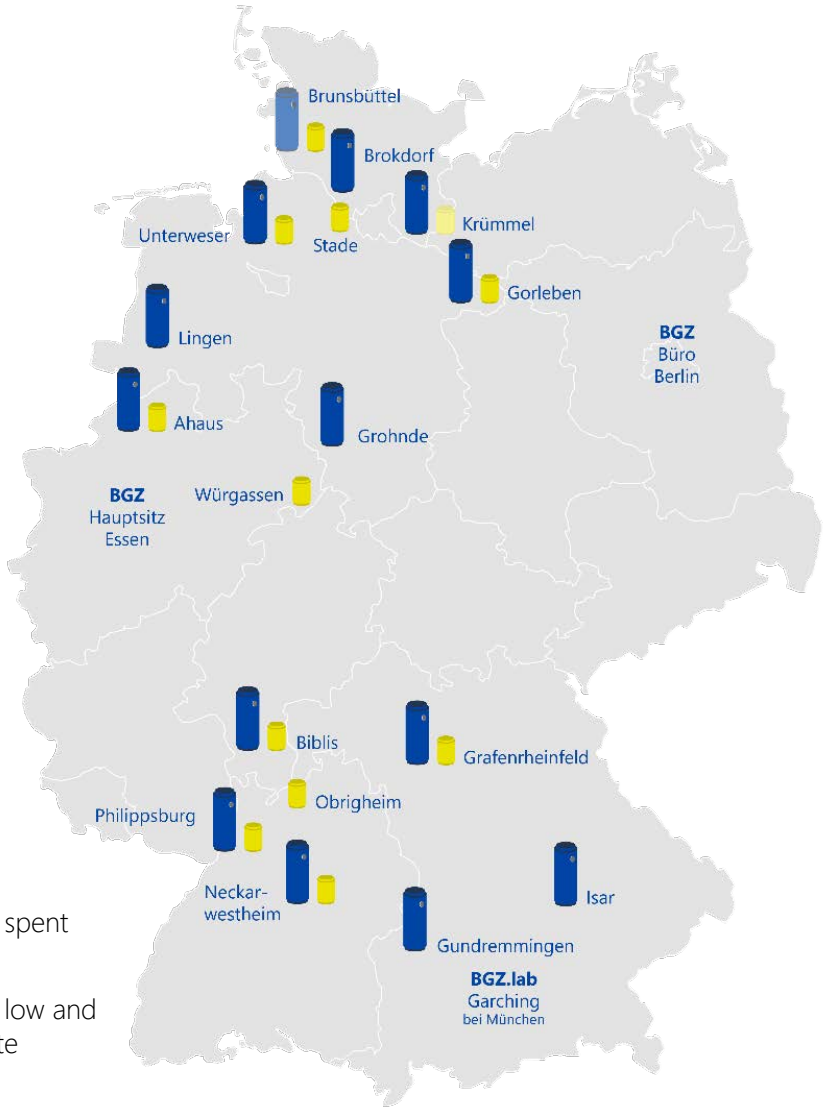
BGZ

Waste from operation and decommissioning (low and medium radioactive)

SNF/HLW geological repository (site selection act)

BGE

Konrad repository



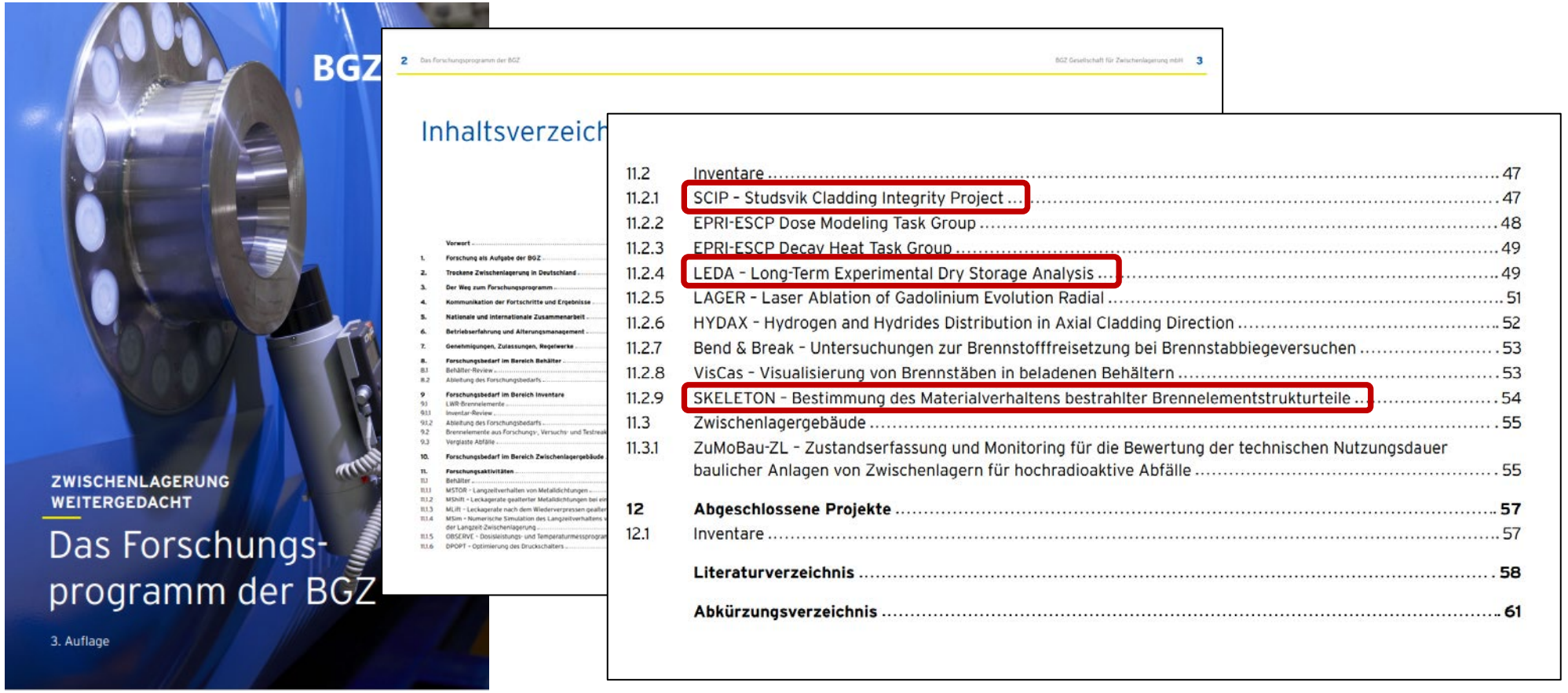
Interim storage of spent nuclear fuel

Interim storage of low and medium level waste

M.Stuke; BGZ's research on inventories; ESCP 2025

BGZ's Research Programme, 3rd Edition

German edition to be published soon, english version in 2026



Link to current 2nd edition, english

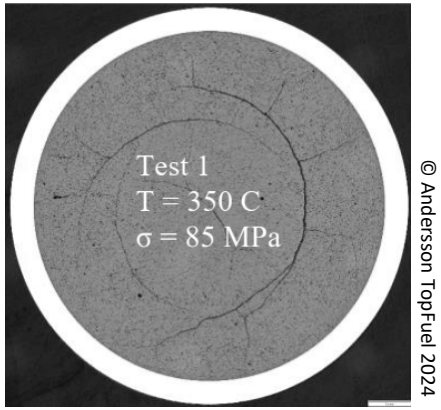
- 9 Research projects focusing in Inventory
- 2 New Projects – Bend & Break and SKELETON

International Project SCIP-V

- The Studsvik Cladding Integrity Project SCIP, is an international OECD/NEA project that started in 2004 operated by Studsvik.Divided in Tasks:
  - Back End
  - LOCA/FFRD
  - ATF

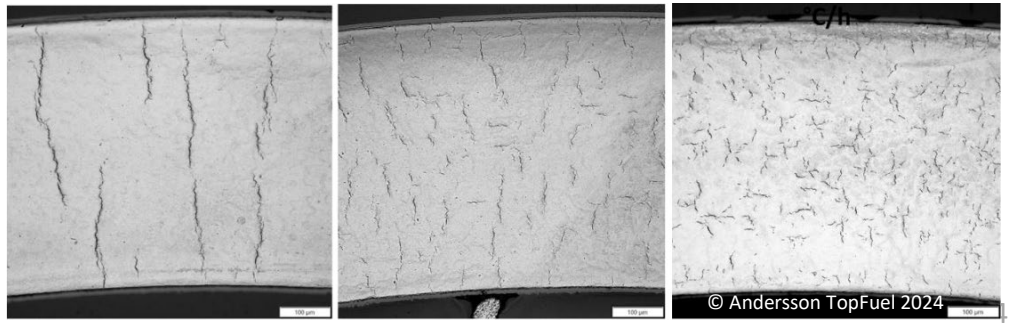
- BGZ interested in "Back End" Task
  - Hydride precipitation
  - Hydride re-orientation
  - Cladding creep
  - Cladding annealing

- BGZ actively contributing in experiment design
  - Parameters of re-orientation
  - Hydrogen behavior



© Andersson TopFuel 2024

Courtesy of Studsvik



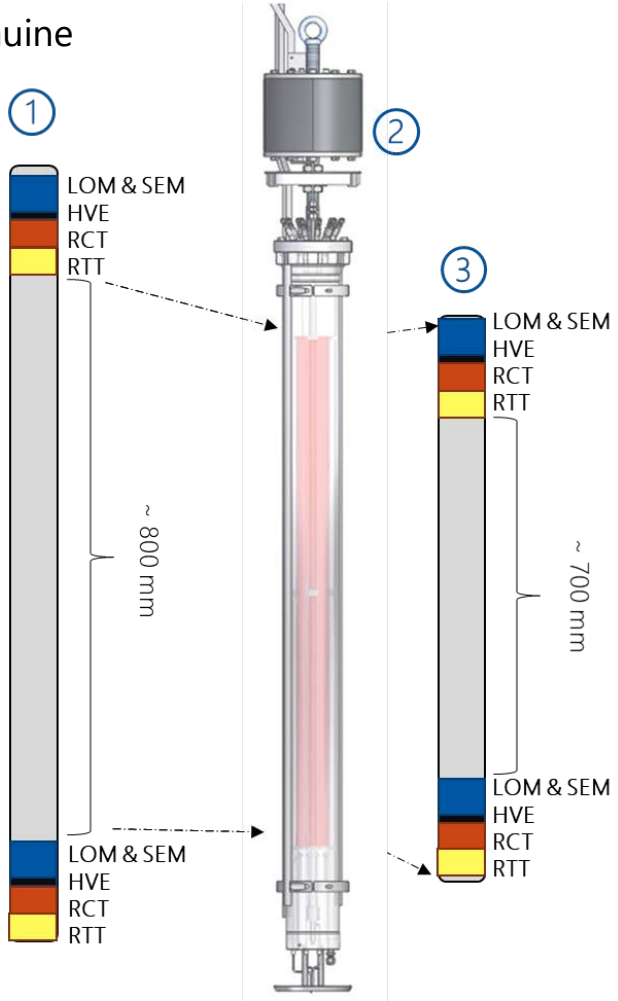
Research Project: LEDA
Long-term Experimental Dry Storage Assesment

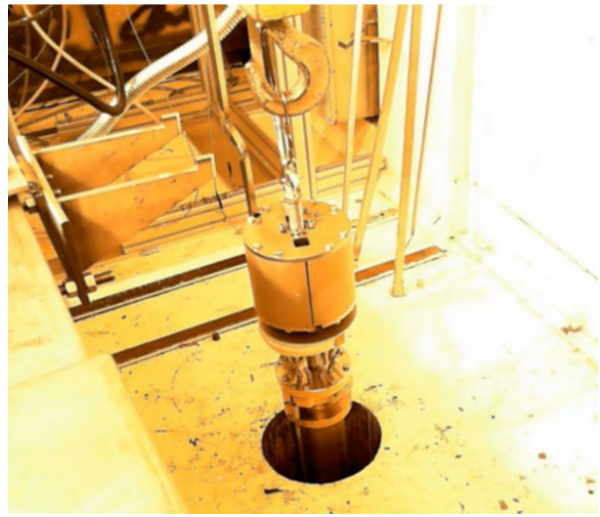
Led by BGZ, multiple stakeholder

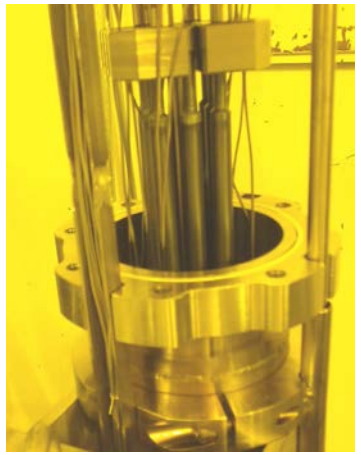
Simulation of long-term storage conditions on genuine irradiated fuels

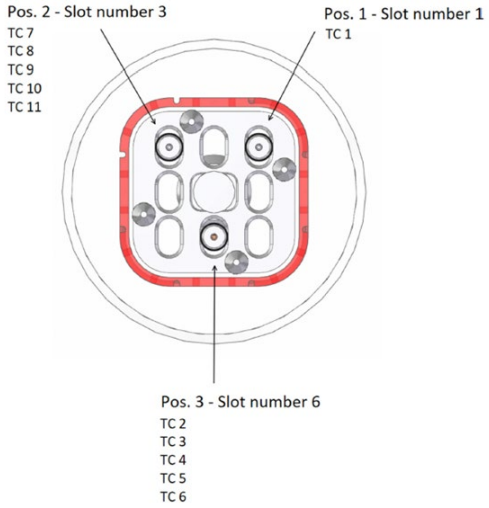
Philosophy

- Pre-characterization ①
  Determining post-irradiation characteristics
- Heat treatment ②
  Simulation of cask drying and dry storage
  Generating proto-typical material
- Post-characterization ③
  Determining post-test characteristics
- Additional material for further testing



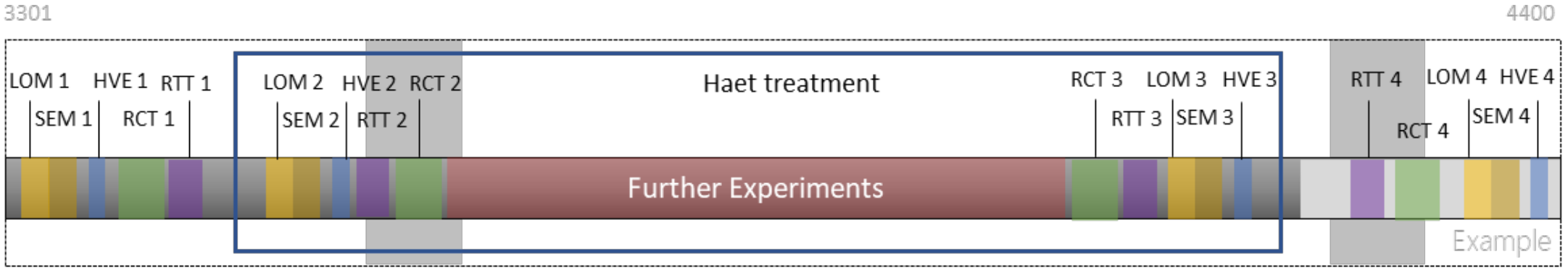




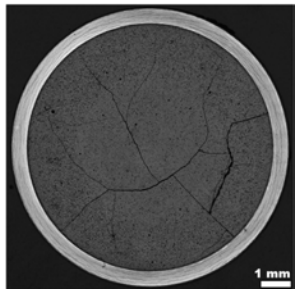


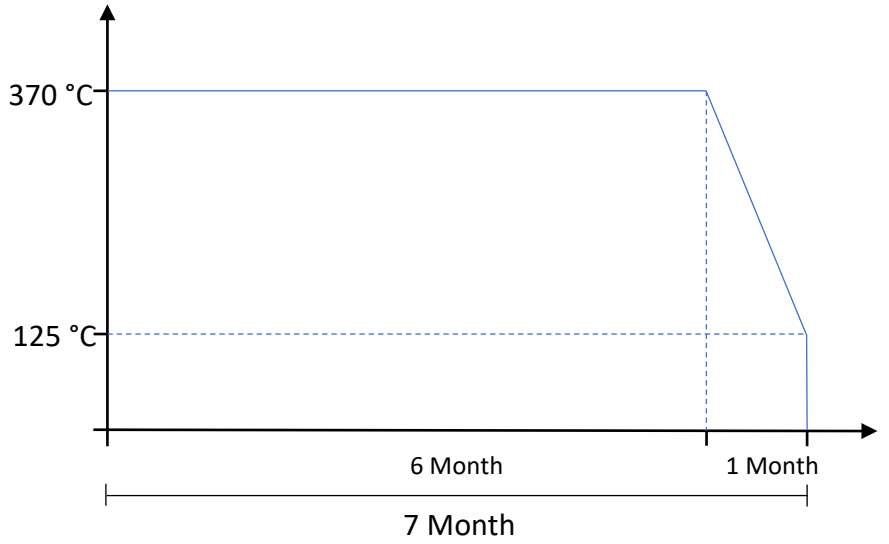
F.Boldt et al., TopFuel 2025

LEDA – Current Status



- 10 full legth rods and 12 segments (~1m) available in LEDA
- Relevant cladding material (Zry4, Zry2(-wL), OptZirlo, M5) and fuel (UO_2, ERU, up to 5,05 wt.% ^{235}U) for Germany
- Pre-characterization of first set of rod segments finished
- Heat treatment of first campaign finished
- Post-characterizations on-going





SKELETON – Evaluation of the material characteristics of key structural components in irradiated fuel assemblies

Aim

- Investigate structural components of Light Water Reactor (LWR) fuel assemblies (FAs) for safe transport and handling.

Motivation

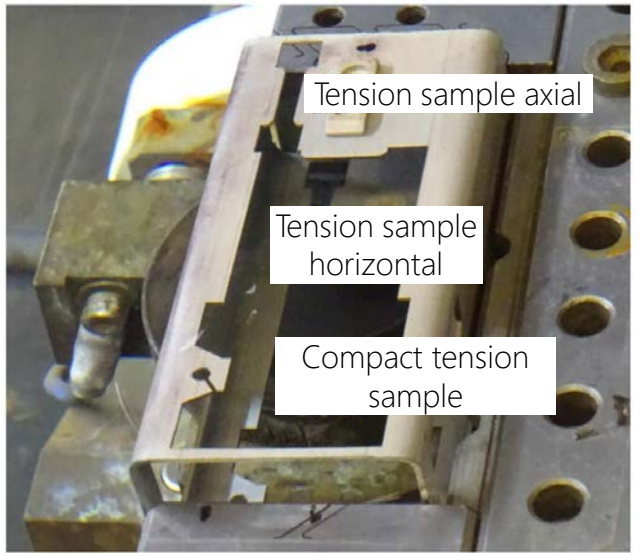
- Limited data on irradiated structural hardware.
- Handling limits are driven by these components, not only by fuel rods.
- Structural parts see the highest neutron fluence and elevated temperatures after the rods.

Approach

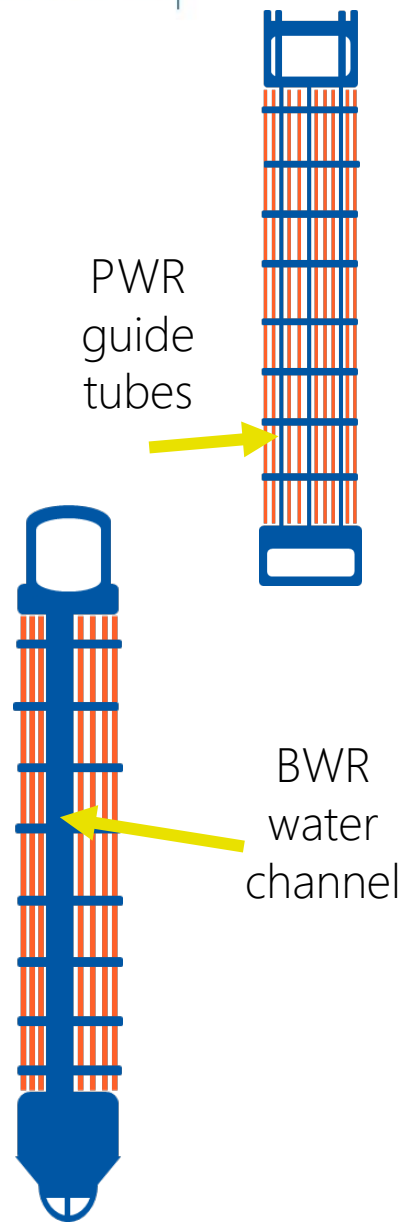
- Perform mechanical experiments on irradiated components (water channels, control-rod guide tubes).
- Derive model parameters from test data to describe component behavior.
- Develop mechanical models for handling and transport scenarios.

Status (Oct 2025)

- First experimental results expected by end of 2025.



Water channel test sample cutting @Framatome, Germany



Fuel Rod Simulation using Transuranus

What isTransuranus?	Usage of Transuranus at BGZ	**Fuel Rod Simulation** (before Cask loading)
• **Fuel performance code** modeling thermal and mechanical behavior of nuclear fuel rods during irradiation in a reactor • **Well established** used for reactor core design calculations over decades • **Developed by JRC Karlsruhe** continuously validated and improved	• **Fuel rod behavior during dry storage** modeling thermal and mechanical behavior of nuclear fuel rods • **Supports experimental analysis** helps to interpret and analyze experimental data • **Predict fuel rod conditions** inside storage cask	

BGZ Gesellschaft für Zwischen-lagerung mbH

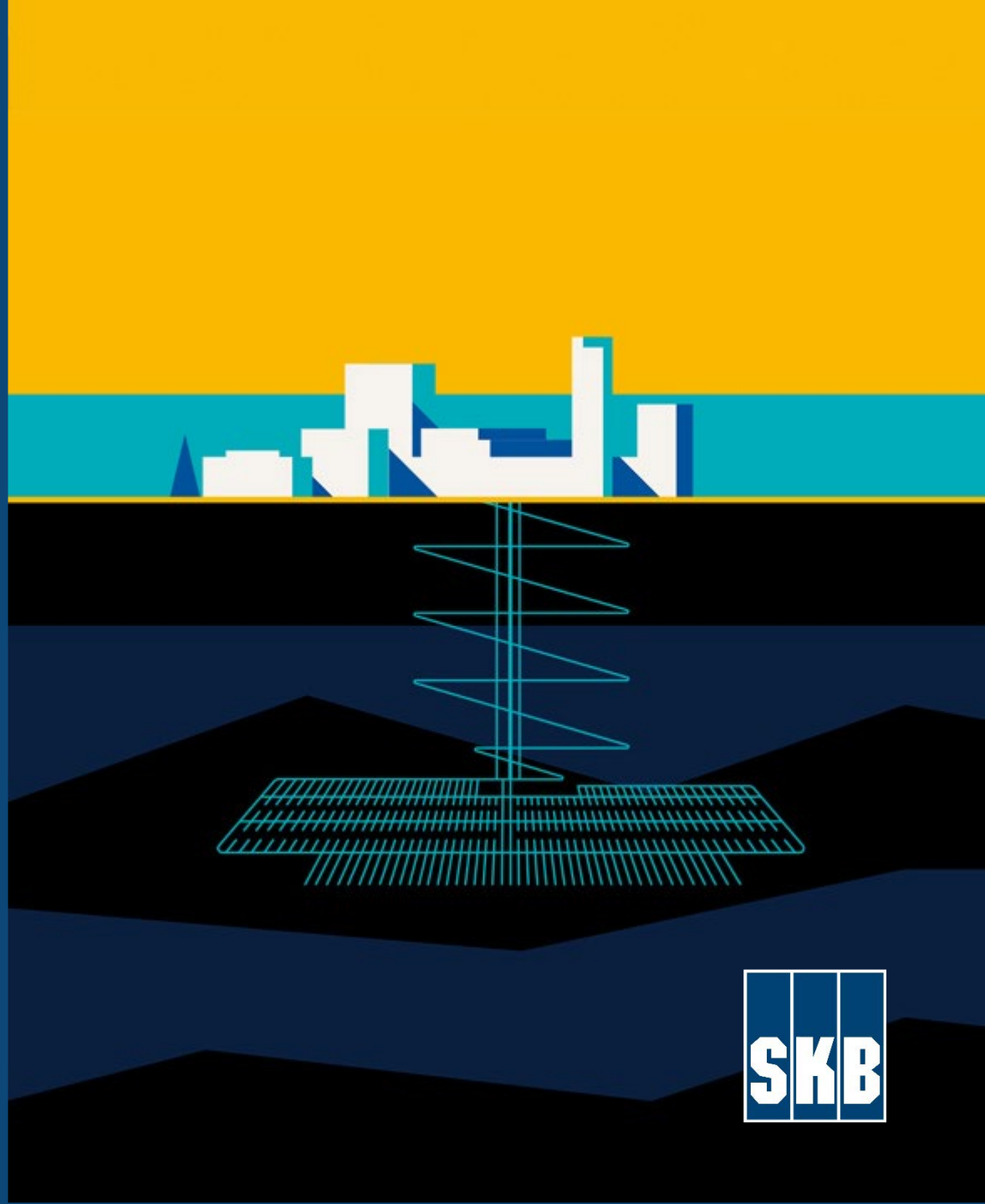
Thank you for your interest !

BGZ's research program

Fuel wet - dry - wet transition in the SKB system

Johan Ljungberg SKB

October 13th 2025



Outline

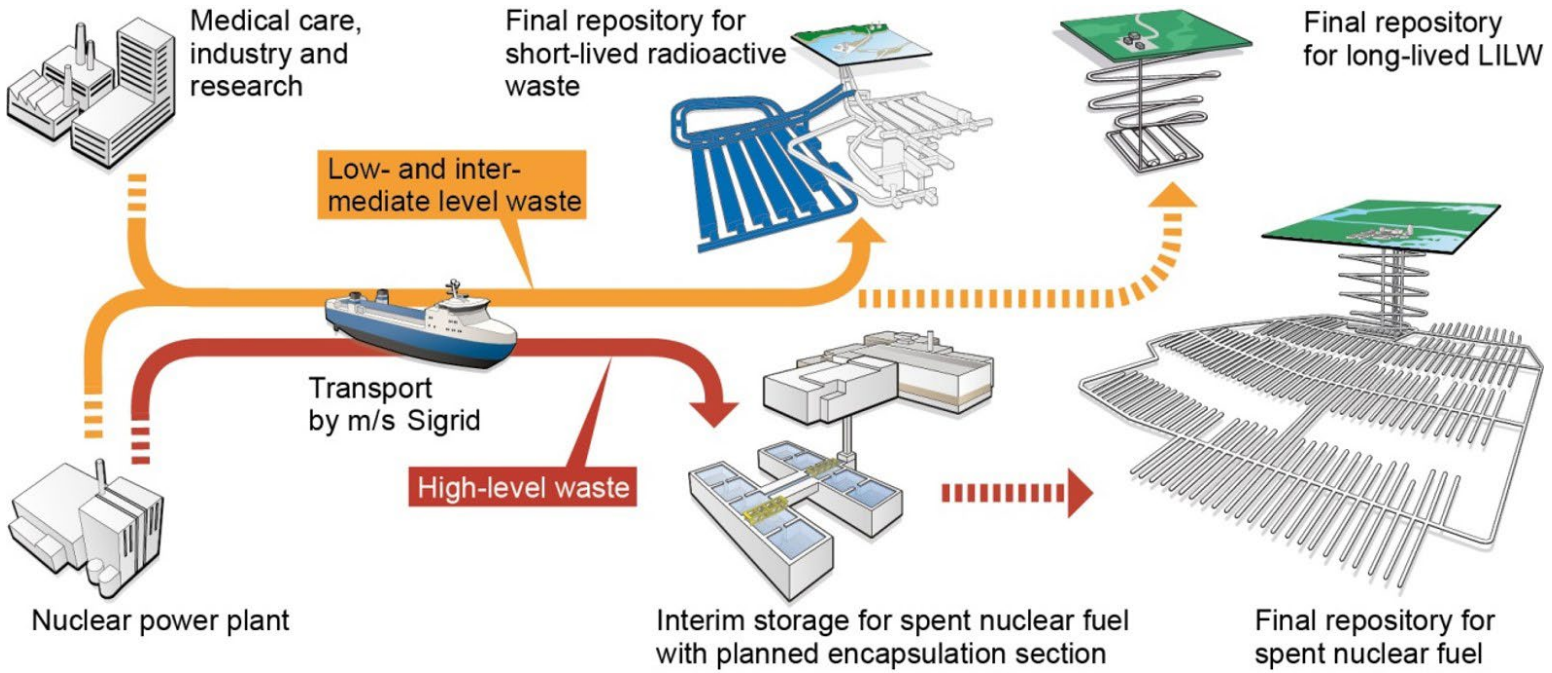
The SKB system

The fuel at our intermediate storage – Gap-analysis

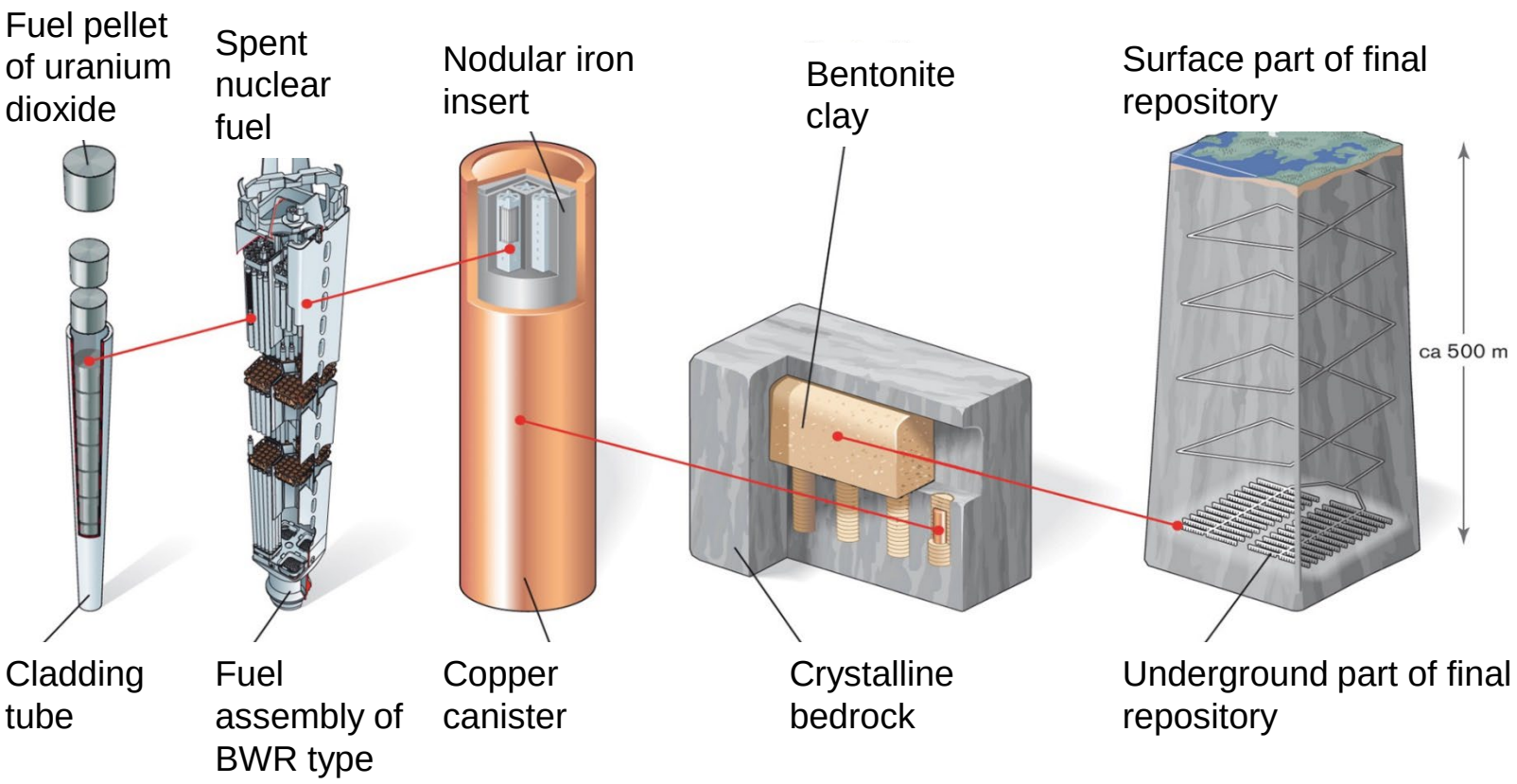
SKB transport system

Findings when transporting

The SKB system



SKB method


Fuel pellet of uranium dioxide
Spent nuclear fuel
Nodular iron insert
Bentonite clay
Surface part of final repository
ca 500 m
Cladding tube
Fuel assembly of BWR type
Copper canister
Crystalline bedrock
Underground part of final repository

GAP analysis

- *Our stored FAs*
- *Which FA are easy to handle*
- *FA with problems*

Fuel to handle

PHWR 19, 37, 5x5

BWR 6x6, 8x8, 9x9, 10x10, 11x11

PWR 14x14, 15x15, 17x17

Burnup

1 - 62 MWd/kgU

Manufacturers

AEG, Telefunken, KWU, ANF, Exxon, Siemens, GE, GNF, Areva, Framatome, Westinghouse, Atomenergi, ASEA-Atom, ABB Atom, TVEL

Countries of origin

France, Germany, Russia, Sweden, USA

Fuel development

Development history in Heavy water fuel (PHWR): (1962-1971) Ågesta

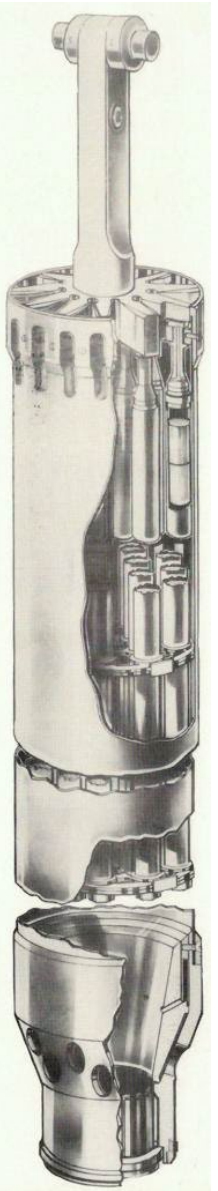
- Zr-spacer to Inconel to mitigate fuel vibrations (The spacers totally reengineered)
- Chamfered UO2 pellets and smaller radius of fuel rods to enhance TMOL (Other design changes also)
- Pre-oxidized cladding for decreasing the debris induced fuel leakers

Development history BWR: (1975 – 2000)

8x8 => 10x10

More controlled density in pellet design

Liner and debris filter



Development history PWR:

- New structural Zr-based alloys mitigating hydriding while being stiff enough to counteract Fuel Assembly bow while adapting to the core
- New Cladding materials enabling higer burnup, Zr-4 => ZIRLO and M5, less corrosion and hydriding
- Debris filter
- Better bonding of topplate to structure - Less welding in top plate

Fuel with challenges in the backend process

PWR FA with IGSCC problem in top plate structure

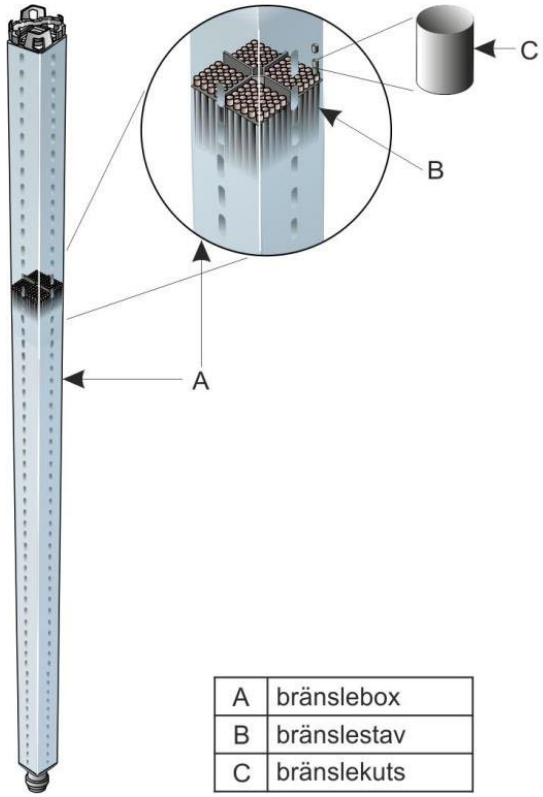
Unchanneled BWR FA w load bearing fuel rods

FA w deviating dimensions

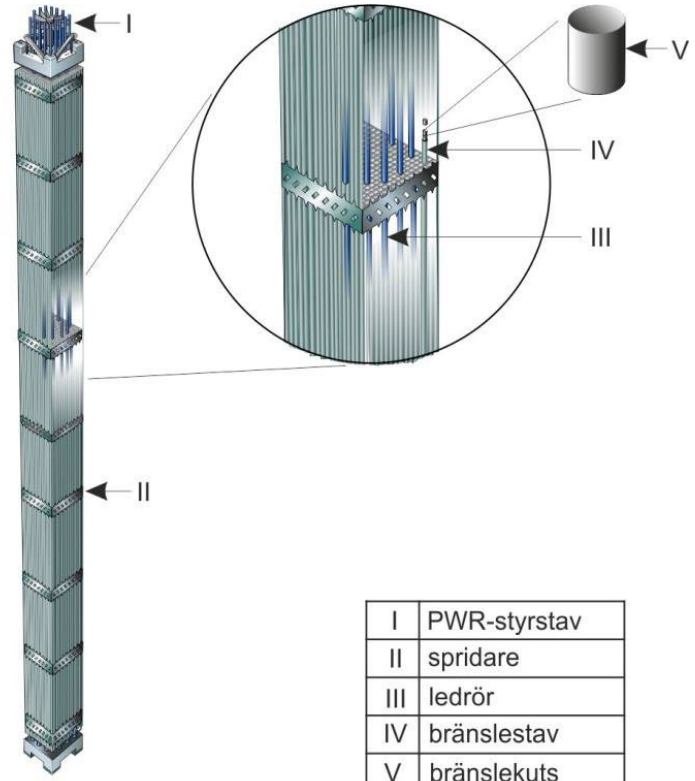
Fuel easy to handle

BWR FA with fuel channels

PWR FA w sound structural parts



A	bränslebox
B	bränslestav
C	bränslekuts



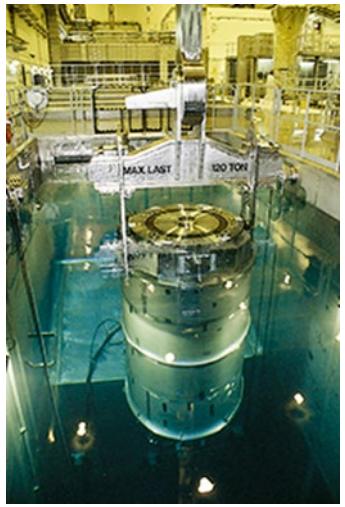
I	PWR-styrstav
II	spridare
III	ledrör
IV	bränslestav
V	bränslekuts

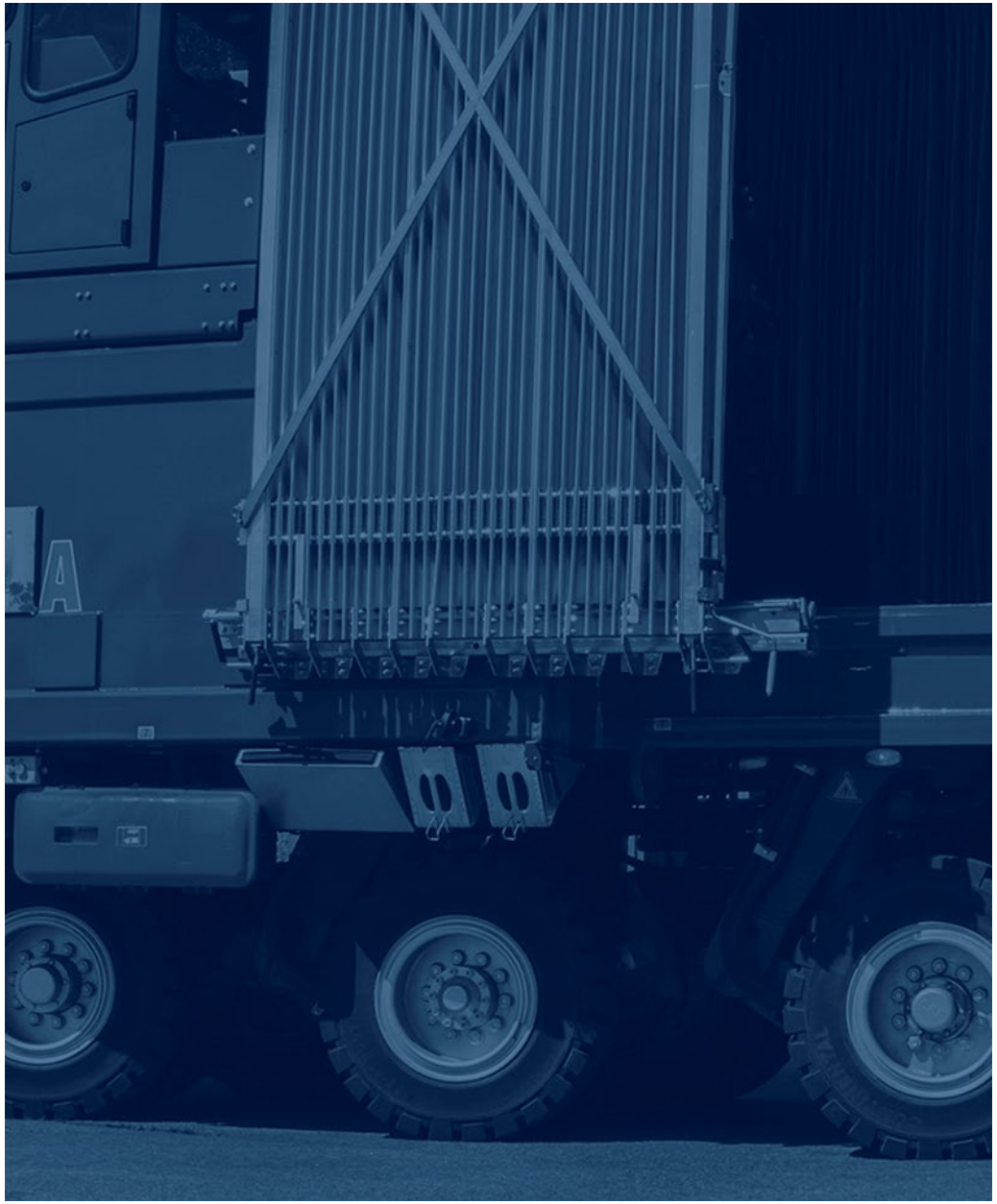
Experiences of fuel transport

Leaking fuel

Fuel for reuse

Thermal history



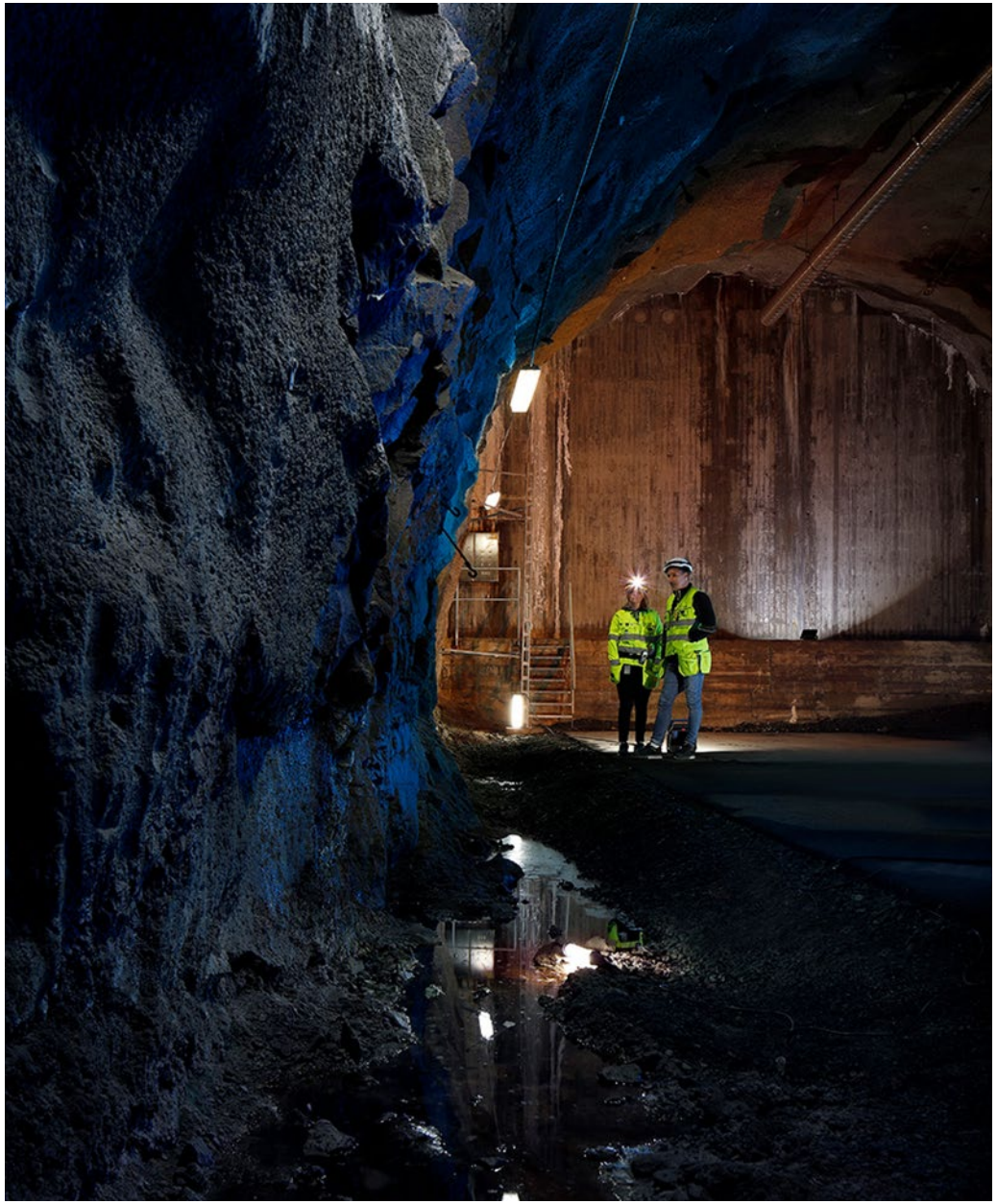


Fuel leakers during transport

Kr-85 gas release – indication on leaker(s)

Contains Cs 134/137 – bigger than pin-hole

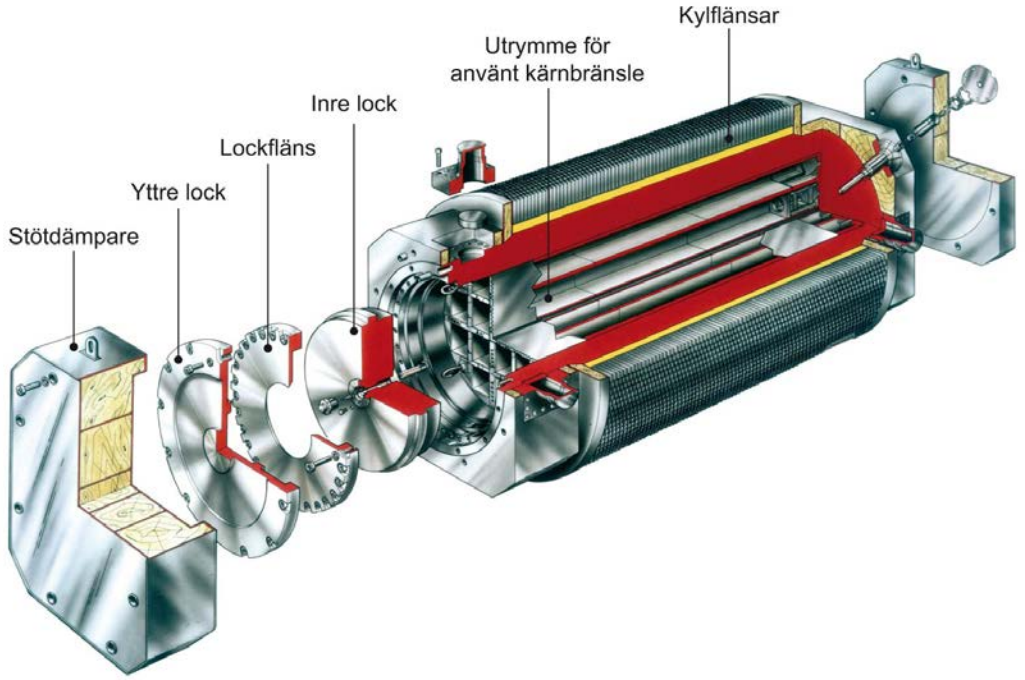
Leak detection of fuel assemblies to verify damaged fuel



Fuel for reuse

When Ringhals 1 NPP closed down, FA (10-30 MWd/kgU) used for 1 or 2 cycles, where sent from Ringhals NPP to Forsmark NPP and have been put to use in the Forsmark 2 reactor. These FA has been operating in F2 for 2 cycles and have no deviations reported.

The FAs where sent using the SKB transport cask. Accelerometers where used to make sure no unnecessary load was applied on the FA.



Thermal calculations

A calculation has been made on all stages of the transport cask transport and handling

Max <400 C

Vacuum drying results in a T exceeding 300 C, close to 400C if we reach equilibrium, when the cask is filled with He, then the temperature falls below 200 C in about 12 hours

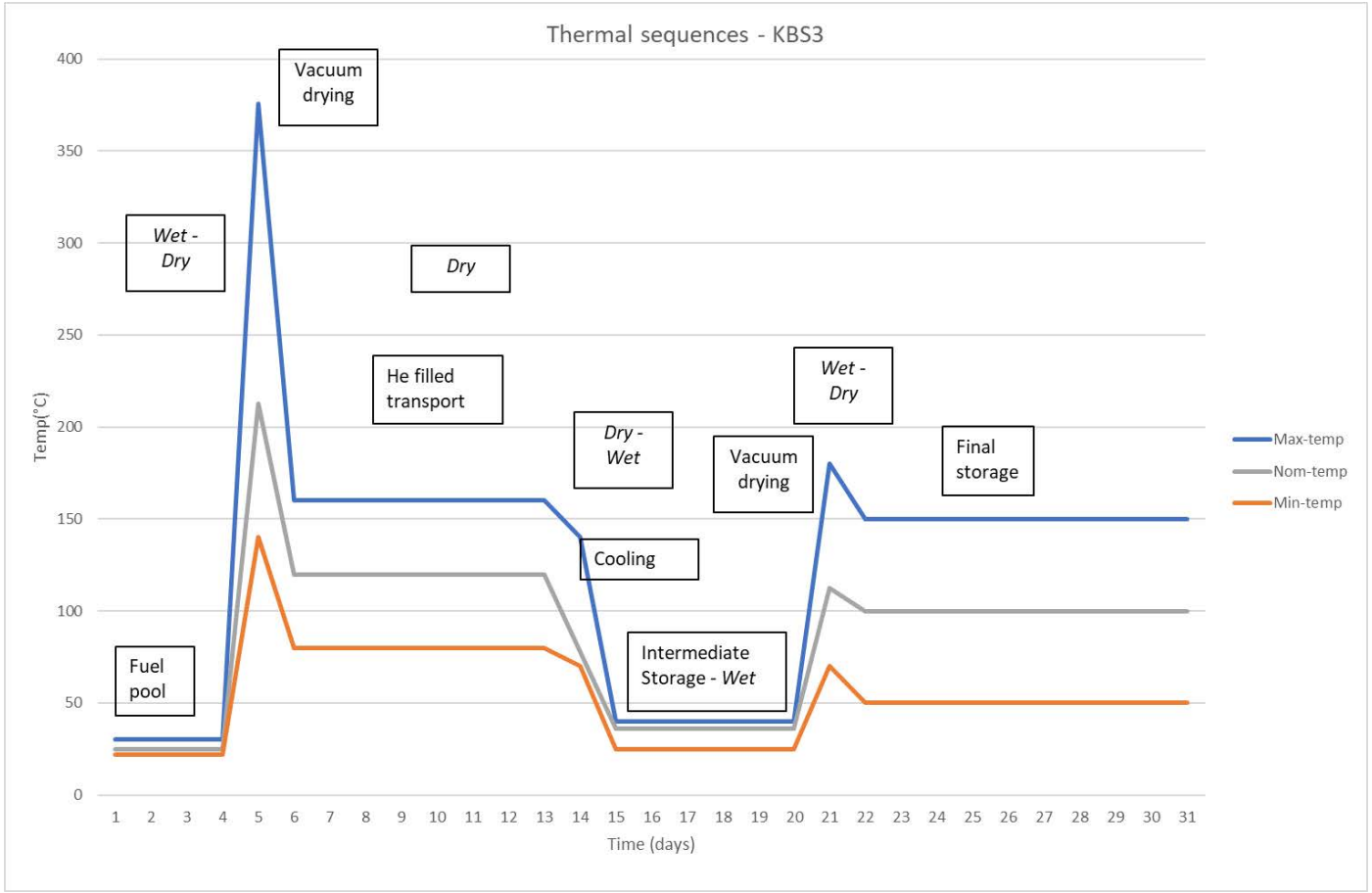
Transport

On the warmest day in Sweden with maximum sun on a parking lot with no wind. 38C (100F) midsummers day, warmest day of 200 years.

→ 160C

Handling at Clab

When standing upright before cooling at 25 C maximum T 140C



Summary

SKB have

- transported and recieved 39 153 FA since 1985
- detected Kr-85 or Cs 134/137 for a very limited number of transports
- made a lot of inspections on nuclear fuel assemblies, but never seen any degradation effects related to the storage in fuel pools or the dry transportation

The effects we have seen are all related to operation in reactors

skb.se

Cask Opening Experience

Laura McManniman
Sr. Technical Leader

EPRI ESCP
October 30, 2025

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Capturing Operational Experiences

- Lot of research undertaken on spent fuel and storage system degradation mechanisms
- From a public perspective, often asked how we know if research results reflect reality
- Operational data, experiences and observations can provide evidence to support research findings

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Cask Opening Experiences

What effects have dry storage had on the condition of spent fuel?

- Data available from four different sites relating to:
  - Five different cask types
  - BWR and PWR fuel
  - Burnups ranging from 9-35 GWd/MTU
  - Pre-inspection dry storage periods ranging from 5-14 years
  - Fuel having undergone a wet-dry-wet storage regime
  - Fuel examined before and following a beyond design basis event
- In all instances fuel condition comparable at inspection to its pre-loading condition

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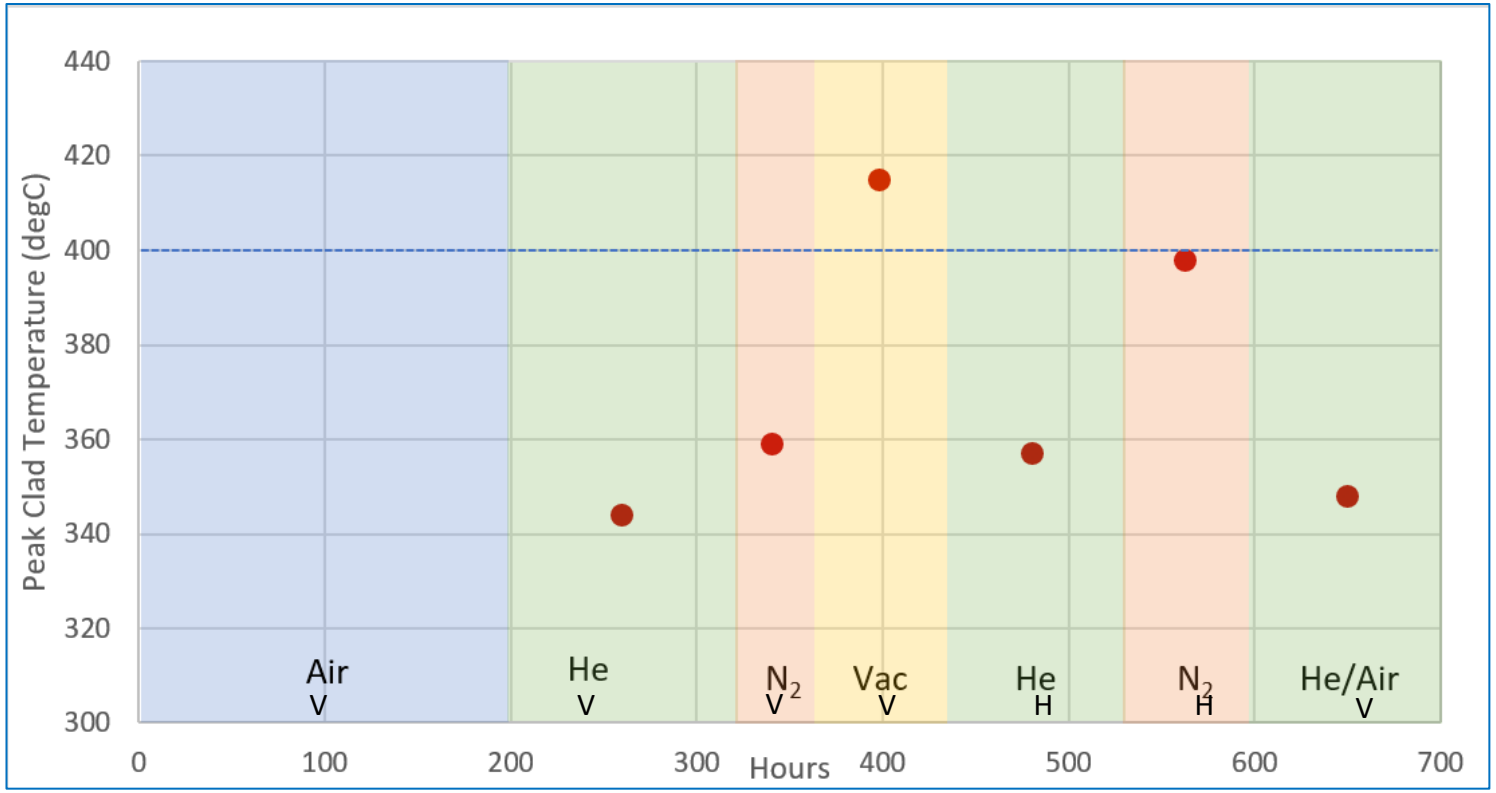
Idaho National Laboratory

Castor V/21 containing 15x15 Surry PWR fuel opened after ~14 years

- Cask and fuel initially used for backfill environment performance testing



- Subjected to different fill gases, cask orientations, and temperature cycling

Atypical start of storage period

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Idaho National Laboratory

Castor V/21 containing 15x15 Surry PWR fuel opened after ~14 years

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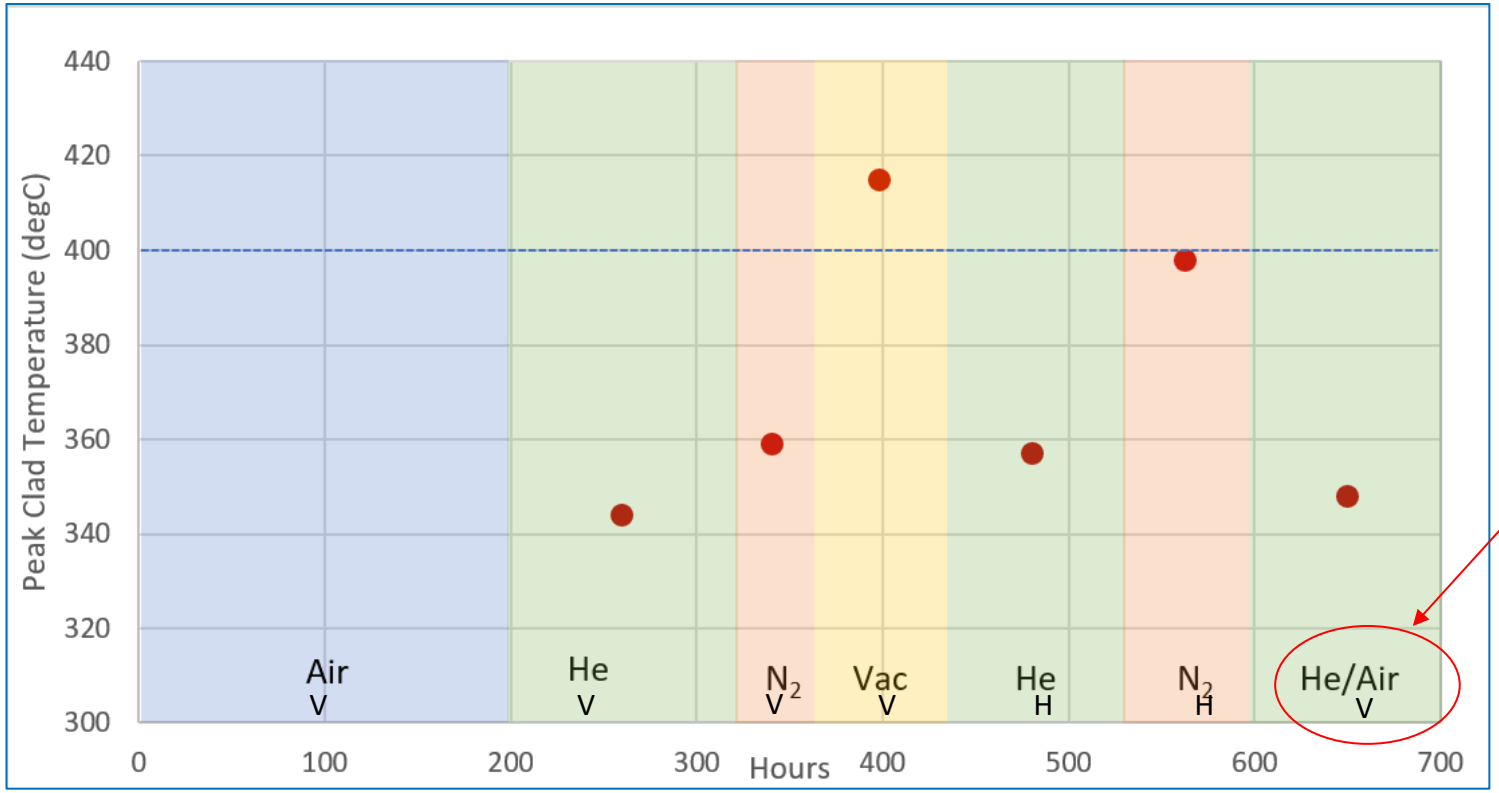

70% He / 30% Air due to sealing failure of quick-connect fitting on lid
Estimated 4-month duration before full He backfill restored for ongoing storage

- Subjected to different fill gases, cask orientations, and temperature cycling

Atypical start of storage period…

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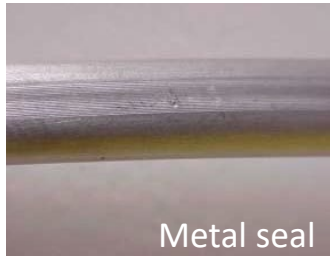
Idaho National Laboratory

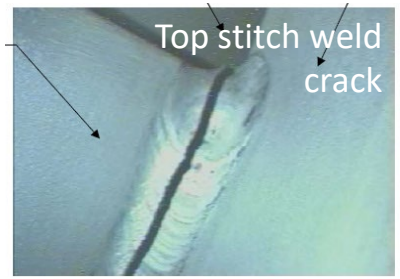
Cask Inspections

- Seals found to be in excellent condition



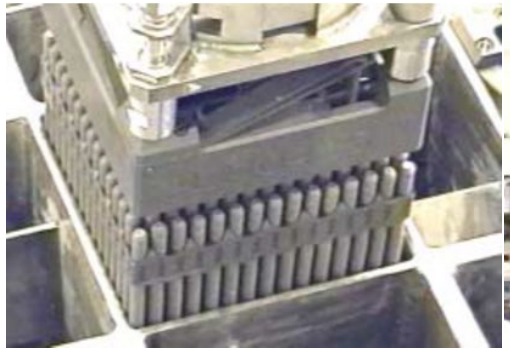
- Cask internals in good condition

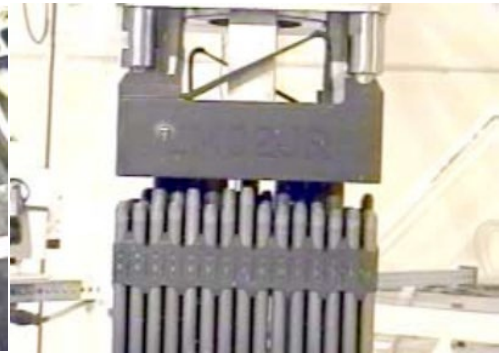
- No change in original weld cracks noted in 1985 inspection



Fuel Inspections

- Cover gas monitoring found no indication of failures during storage
- Lifting force measurements indicative of no significant bowing or adherence due to corrosion product development





- All FAs visually inspected with only slight localized bowing noted in rod sections
- Detailed inspection of 12 pins found creep <0.1%, no fission gas release, no H_2 pickup or reorientation, little-to-no cladding annealing.

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Idaho National Laboratory

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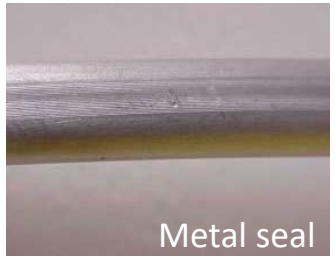

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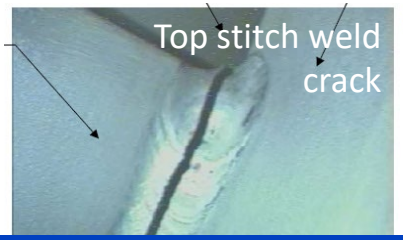

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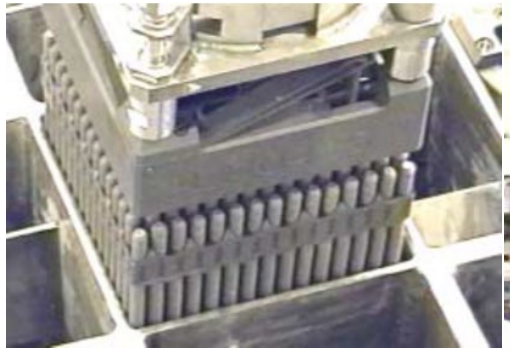

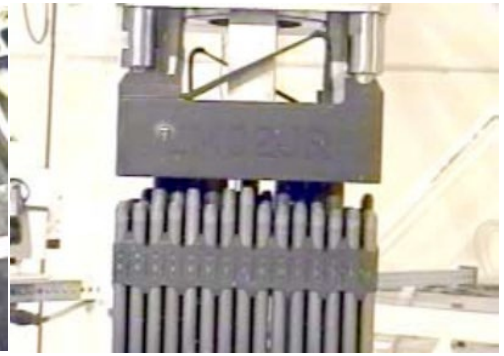

- All FAs visually inspected with only slight localized bowing noted in rod sections
- Detailed inspection of 12 pins found creep <0.1%, no fission gas release, no H_2 pickup or reorientation, little-to-no cladding annealing.

… No long-term effects noted

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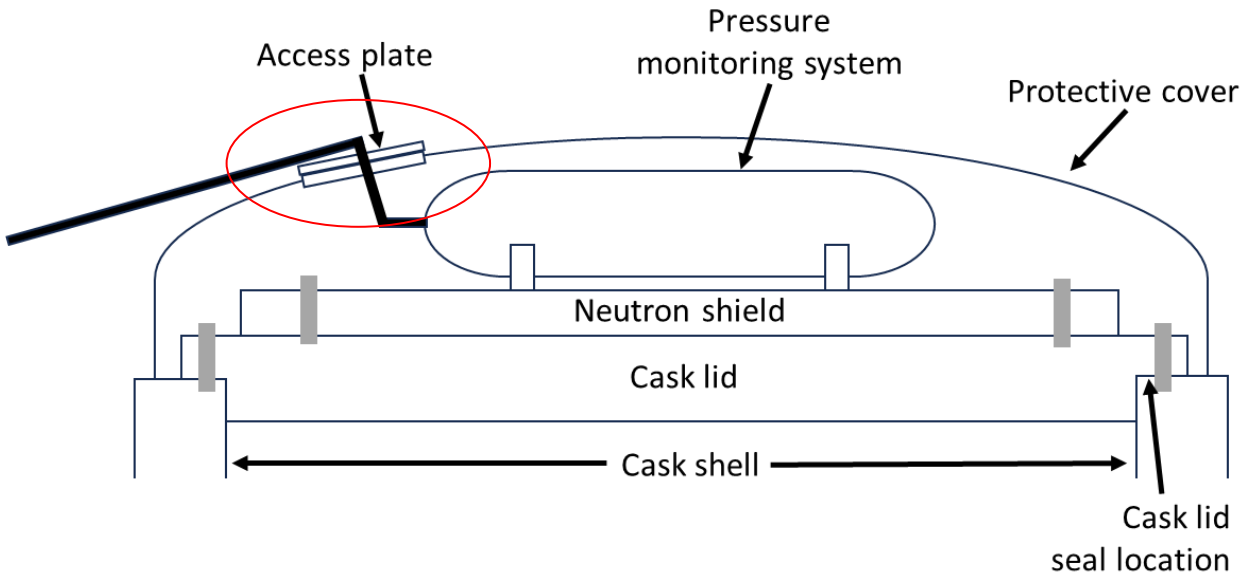
Peach Bottom Atomic Power Station

- TN-68 containing 68 BWR assemblies opened after ~10 years
  - Conservative decision to return to pool following pressure monitoring system alarm



- Inspection found fuel to be in good condition
  - Gas sampling found no evidence of failures, confirmed through vacuum sipping of all assemblies
  - Visual inspection found no change from loading condition for all fuel assemblies

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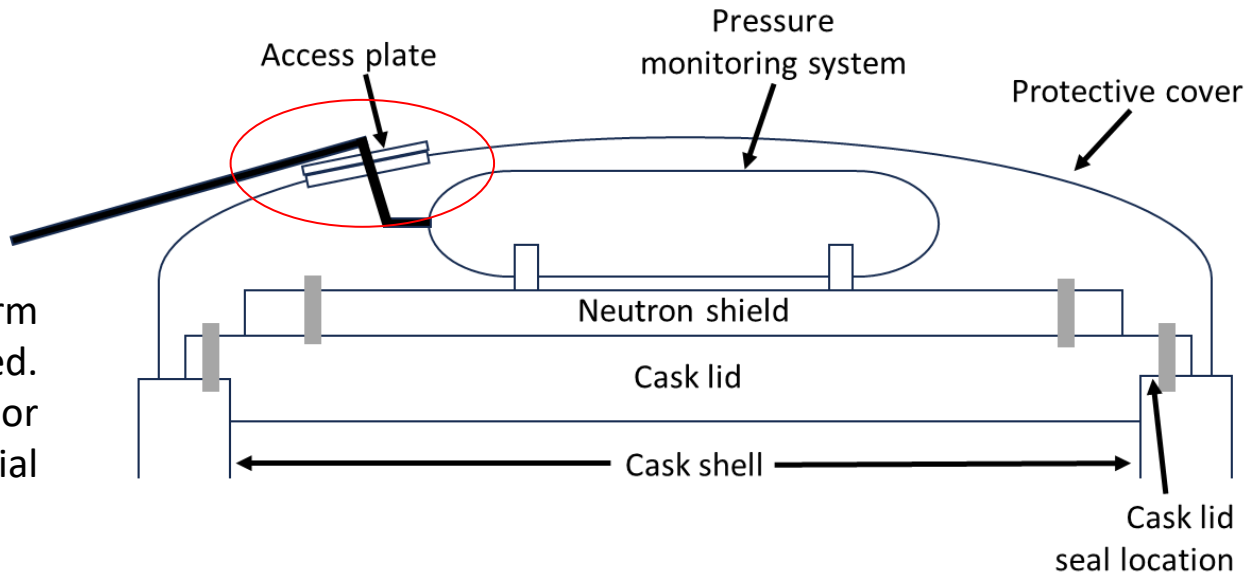
Peach Bottom Atomic Power Station

- TN-68 containing 68 BWR assemblies opened after ~10 years
  - Conservative decision to return to pool following pressure monitoring system alarm

Monitoring system & alarm functioned as intended. No loss of confinement or release of radioactive material



Design & verification improvements introduced for the access plate seal and protective weather cover

- Inspection found fuel to be in good condition
  - Gas sampling found no evidence of failures, confirmed through vacuum sipping of all assemblies
  - Visual inspection found no change from loading condition for all fuel assemblies
  - Cask reloaded and placed back into service

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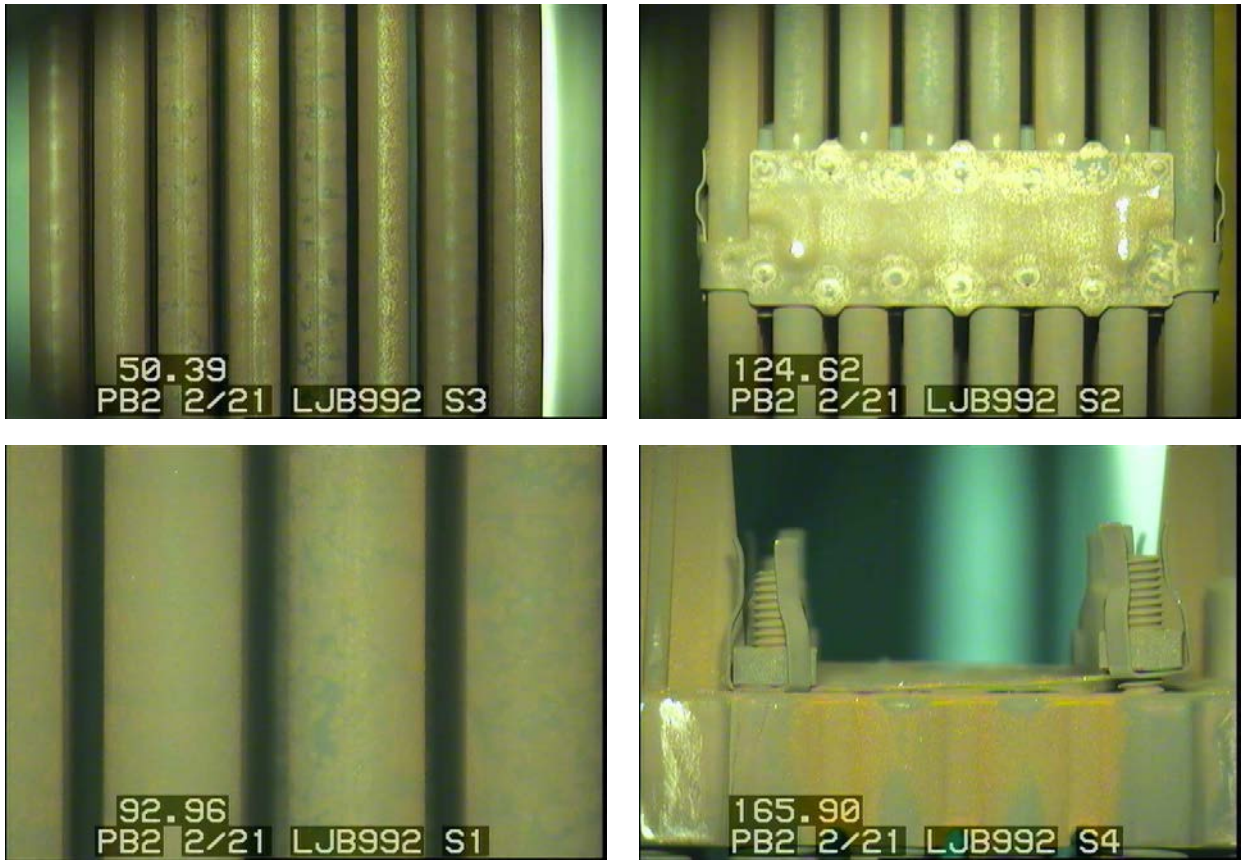
Peach Bottom Atomic Power Station

- Fuel returned to pool for storage following unloading
- One assembly was selected for inspection after ~10 years wet storage
  - No abnormalities identified
  - Fuel and hardware deemed structurally sound



Transition through dry storage – rewetting – wet storage with no identifiable adverse effect

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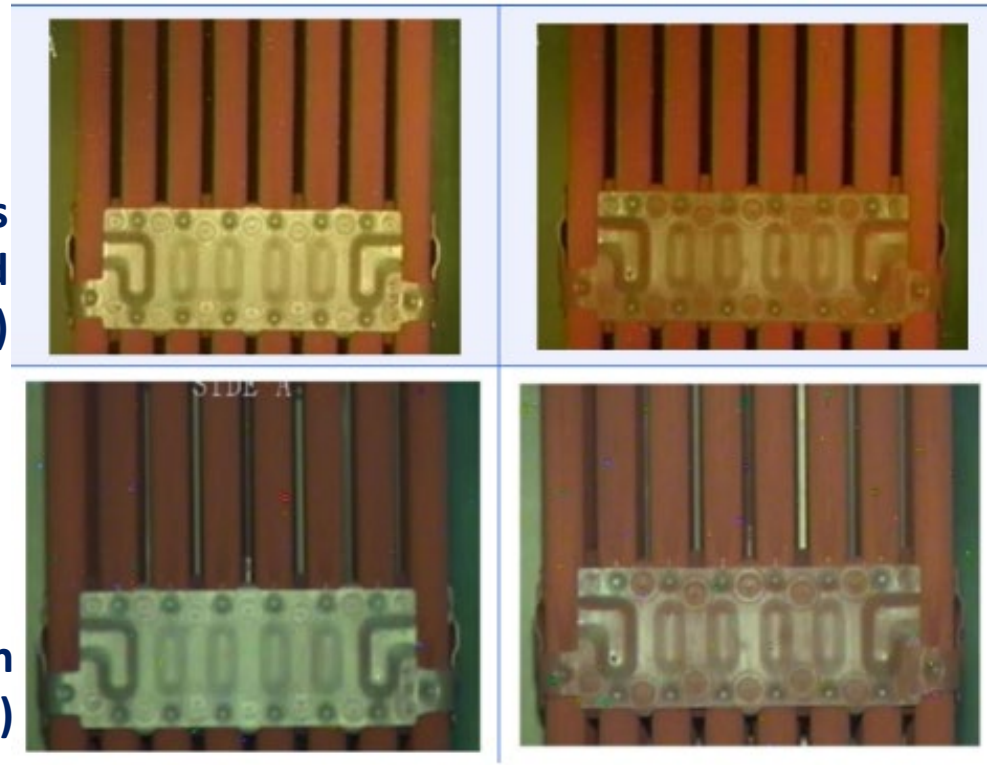
Tokai No2 NPS

- Hitachi Zozen containing 61 BWR fuel assemblies opened after 7 years
  - Cask opening planned as a confirmatory measure
- Cask found to be in good condition
  - Cover gas monitoring found no evidence of failures
  - Seal leak testing confirmed maintenance of containment
  - Examination of seals and surfaces found no evidence of degradation
- Two fuel assemblies selected for inspection & no significant changes observed



No indication of either fuel or cask degradation during storage

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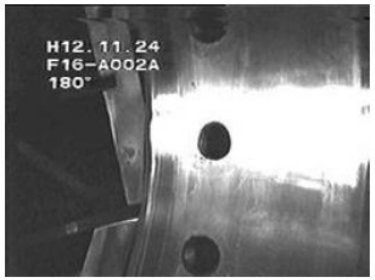
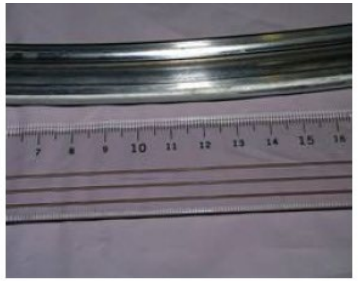
Fukushima Daiichi NPS

- Planned openings as confirmatory measure
  - KOBELCO medium type, 37 BWR after 5 years
  - KOBELCO large type, 52 BWR after 10 years
- Procedure involved:
  - Leak testing of primary lid
  - Cavity gas sampling
  - Examination of cask seal & sealing surface
  - Visual inspection of two highest heat generating fuel assemblies



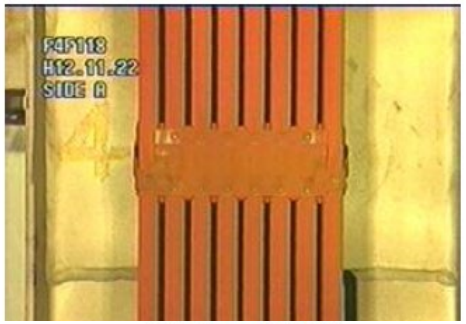
←Flange surface of the cask

Metal gasket of the primary lid →



17/Nov/1995

22/Nov/2000



8/Sep/1995

23/Aug/2005

Images with permission from TEPCO

No indication of either fuel or cask degradation during storage

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Fukushima Daiichi NPS

- Epicenter of 9.0 magnitude earthquake 112 miles from site
  - Max PGA values recorded at Unit 2
    - Horizontal 0.561g
    - Vertical 0.308g
    - Design basis 0.447g
- Site hit by 7 tsunamis, with max height 46-49 feet
  - Cask facility flooded with seawater & casks submerged for a period of time

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Accident conditions experienced beyond design basis

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Fukushima Daiichi NPS

Methodology developed for safety function inspection prior to moving casks to Temporary Cask Storage Facility (TCSF)

Test Subject	Safety Function	Methodology	Cask Applied To	Criteria
Fuel	Fuel Integrity	Visual inspection	F16-A002C	Confirmation of no abnormality (deformation, damage etc.)
Cask	Subcriticality	Basket visual inspection	F16-A002C	Confirmation of no abnormality (deformation, damage, etc.)
		Equivalent neutron dose rate measurement	All	No significant change from pre-tsunami readings
	Confinement	Primary lid leak test	All	Leakage rate $\leq 1\times10^{-6}$ Pa·m^3/s
		Lid interspace pressure test	All	Pressure > 0.29 MPa (abs)
		Metal seal visual inspection	All	No surface abnormalities (crack, etc.)
		Penetration test	All	No indication
	Heat removal	Cask surface temperature measurement	All	37 FA: <69°C 52 FA: <77°C
	Shielding	Survey of equivalent dose rate (total γ and n)	All	Surface: <2000 µSv/hr @1 m: <100 µSv/hr

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Fukushima Daiichi NPS

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Impact	Differentiator	Observations
Earthquake	None	All casks in same location M/L cask almost identical design, no significant seismic differentiator
Seawater	None	All temporarily submersed
Rubble	None	All casks covered by resin cover All casks have thick metal shell No significant impact damage noted
Fuel type	None	Two types of BWR fuel stored (8×8) but no geometric or material differences
Hydrogen embrittlement	Burnup	Link between operational history and hydrogen pickup
Basket storage location	Central	Amplitude of seismic activity larger in central region (see Figure 4-12)

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Fukushima Daiichi NPS

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		Lid interspace pressure test	All	Pressure > 0.29 MPa (abs)
		Metal seal visual inspection	All	No surface abnormalities (crack, etc.)
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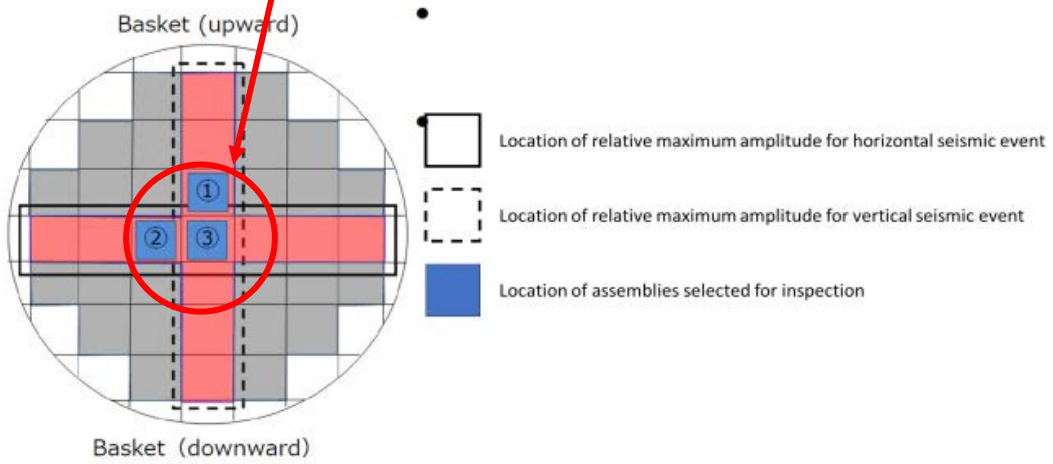

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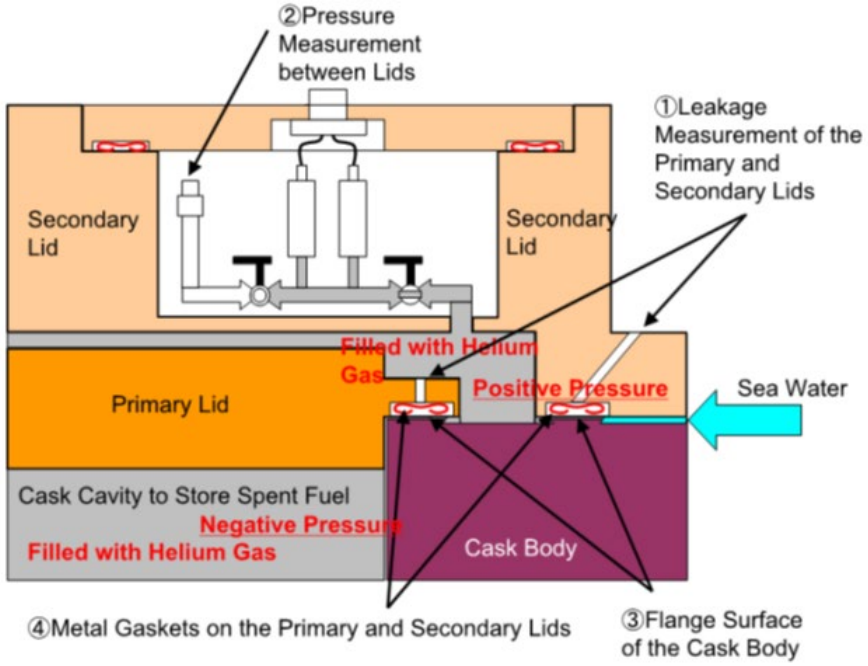
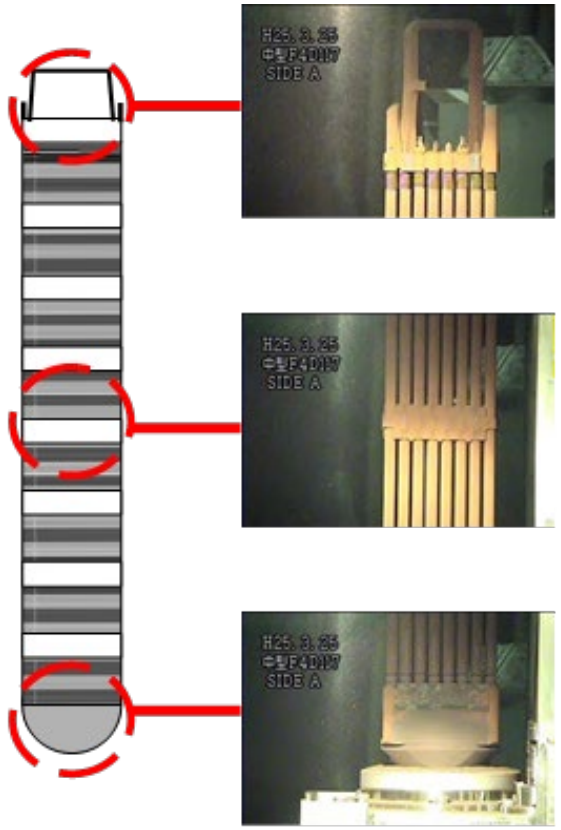
Fukushima Daiichi NPS

Inspection undertaken using ‘representative cask’



Cross-section of cask lid showing inspection points & entry point where seawater contacted secondary lid seal.

With permission from TEPCO



Images from one of three fuel assemblies inspected.
No abnormalities observed such as deformation or damage.
Rods & hardware confirmed to be in good condition.

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No loss of cask confinement or loss of fuel integrity identified

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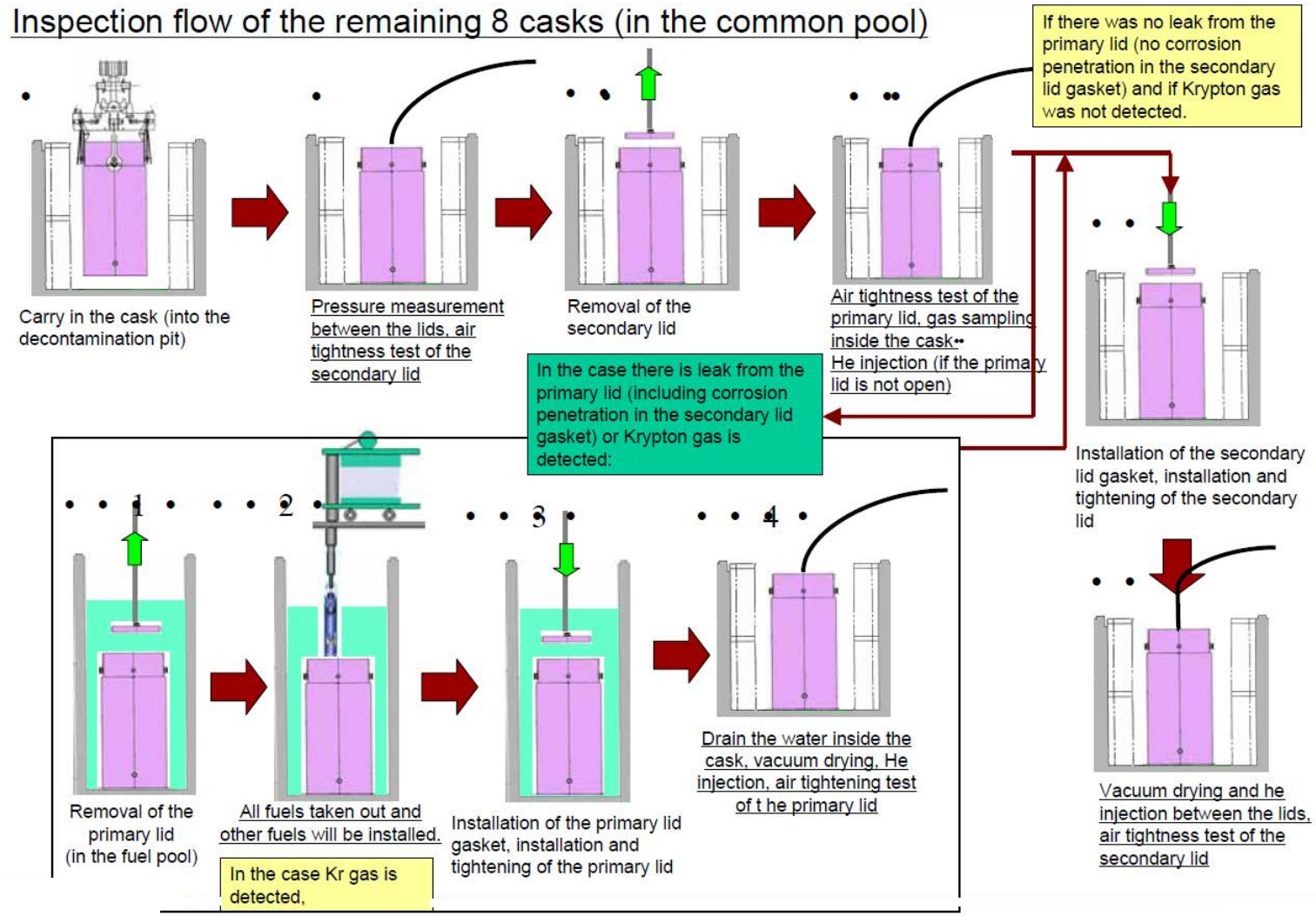
Fukushima Daiichi NPS



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No primary lid leaks or Kr detected of any of the 8 casks

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Summary

	Idaho	Peach Bottom		Tokai No 2	Fukushima Daiichi		
Fuel type	PWR	BWR	BWR	BWR	BWR	BWR	BWR
Storage time (Years)	14	10	10 (wet)	7	5	10	13
Burnup (GWd/MTHM)	24-35	9-30.3	32.2	31.8-33.5	≤33	≤33	≤33
Reason	Atypical storage conditions	Response to pressure alarm	Wet stored following dry storage	Confirmatory inspection	Confirmatory inspection	Confirmatory inspection	Post-accident
Findings	Fuel & cask in good condition, no evidence of loss of integrity or confinement						

- What next?
  - Will add additional case studies as become available
  - Will add examples of shorter term wet-dry-wet transitions

Report available from EPRI.com

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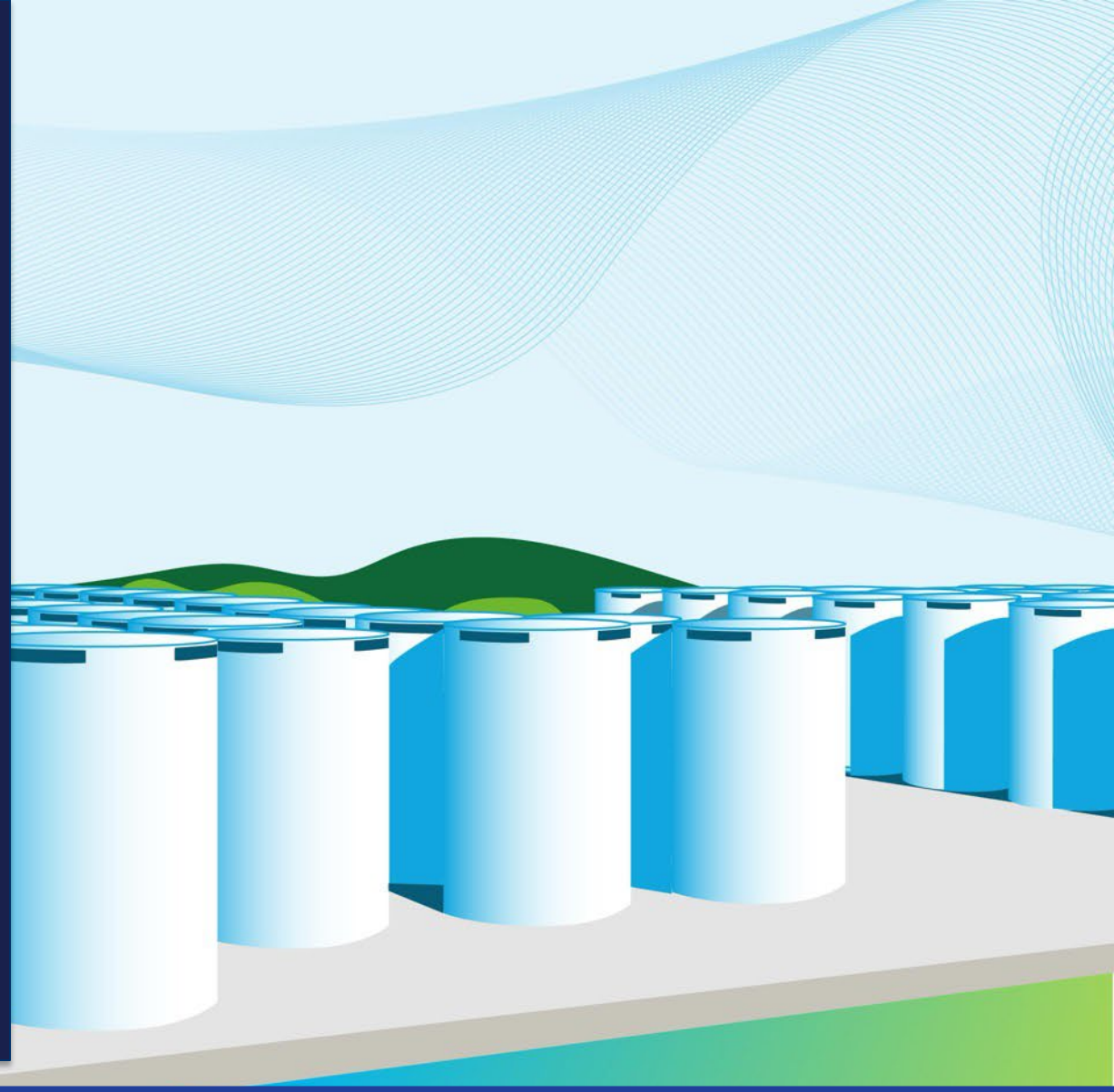
Considerations for Thermal Annealing of Zr-Alloy Cladding During Dry Storage

P. Cantonwine[1], R. Karkkainen[1], R. Montgomery[1], Y. Sasikumar[1], M. Nedim Cinbiz[1], R. Torres[2], R. Shimskey[2], G. Petersen[3], J. Reed[4], M. Aumand[4], I. Porter[5], D. Mitchel[6], G. Pan[6], P. Nair[7], S. Vitto[7], J. Faldowski[8]

[1]Oak Ridge National Laboratory, Oak Ridge, TN;
[2]Pacific Northwest National Laboratory, Richland, WA;
[3]Idaho National Laboratory, Idaho Falls, ID;
[4]Framatome, Lynchburg, VA,
[5]Global Nuclear Fuel – Americas, Wilmington, NC,
[6]Westinghouse, Columbia, SC,
[7]Departement of Energy, Washington D.C.,
[8]Electric Power Research Institute, Charlotte, NC



U.S. DEPARTMENT of ENERGY | Office of Nuclear Energy
Spent Fuel and High-Level Waste Disposition

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

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DOE hosted a workshop at ORNL on thermal annealing with fuel vendors in January 2025

- Goal of the workshop was to discuss potential alternatives to yield strength as a failure criterion for spent nuclear fuel (SNF) rods in dry storage and transportation accidents

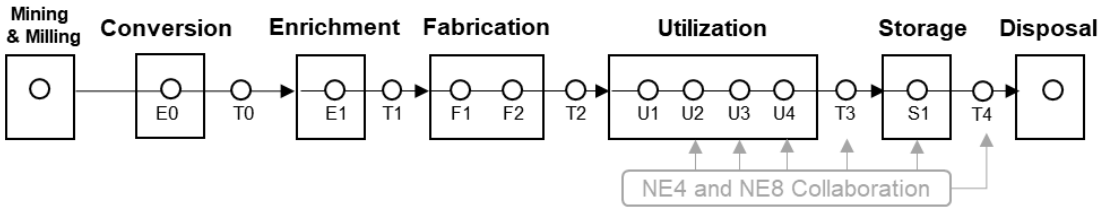
A Cradle-to-Grave Perspective is Valuable to Fully Understand Fuel Performance

For example,

- Manufacturing defects that affect in-reactor performance are minimized through quality inspections
- In-reactor performance history affects performance both in-reactor and for the backend

But,

- Fuel rod design does not explicitly consider backend SNF performance requirements
- New fuel and cladding technology is developed considering only in-reactor performance requirements



E0 – Conversion U3O8 to UF6
T0 – transportation of UF6 to enrichment facility
E1 – UF6 enrichment (<5w/o) and deconversion
T1 – transportation of UF6 to fuel fabrication facility
F1 – fabrication of UO2 fuel pellets
F2 – fabrication of LWR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and loading
U2 – power production
U3 – spent fuel pool/shuffle operations
U4 – on-site dry cask storage
T3 – transportation of spent fuel to off-site storage
S1 – off-site storage
T4 – transportation to disposition

Source: https://www.nrc.gov/docs/ML2108/ML21088A047.pdf

Fuel vendor perspectives on the backend are valuable

Regulatory requirements are written so that responsibility of backend fuel performance belongs to cask vendors

NRC Storage Requirements (10 CFR Part 72)

- The SNF cladding must be protected during storage against degradation that leads to gross ruptures …
- Configuration of SNF rods are inputs for demonstrating heat load, dose rates, and criticality control limits are met
    - Nuclear criticality accident must not be possible when at least two unlikely, independent, and concurrent or sequential changes have occurred

NRC Guidance to Meet Requirements (ISG-11 Rev. 3, NUREG-2215)

- Maintain the temperature in dry storage below 400^{o}C
- Combined stresses in the cladding during a tip-over accident should remain below the as-discharged yield strength of the material or justified otherwise
- Thermal annealing is not explicitly considered in the NRC guidance but was recognized in NUREG-2214 as something that increased ductility

Sibling Pin (or "Sister Rod") Test Campaign

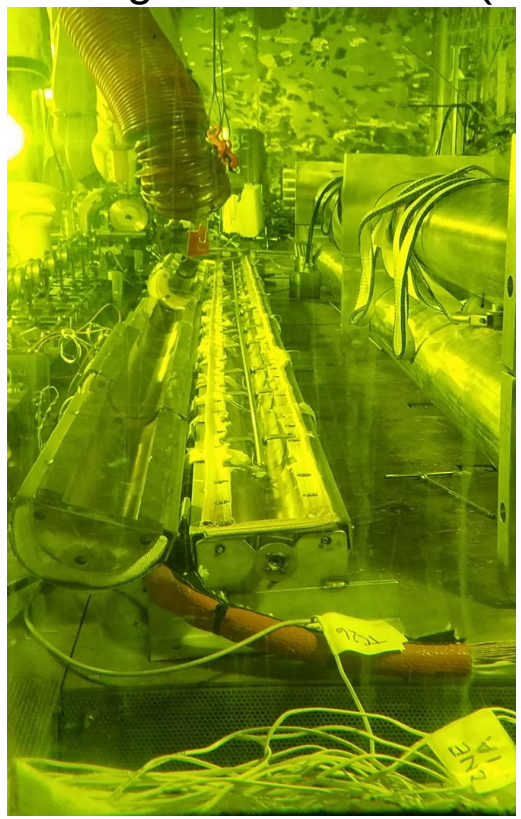
- The Sibling Pin test campaign is a DOE-funded research activity that is part of the High Burnup Spent Fuel Data Project in collaboration with EPRI.
- The campaign is focused on generating characterization, material property, and performance data for high burnup (HBU) fuel rods.
- 25 HBU fuel rods selected with characteristics and histories that closely match those used in the HBU Research Cask.
- Destructive and non-destructive examinations (DEs and NDEs, respectively) were performed at ORNL, PNNL, and ANL of 25 HBU fuel rods.

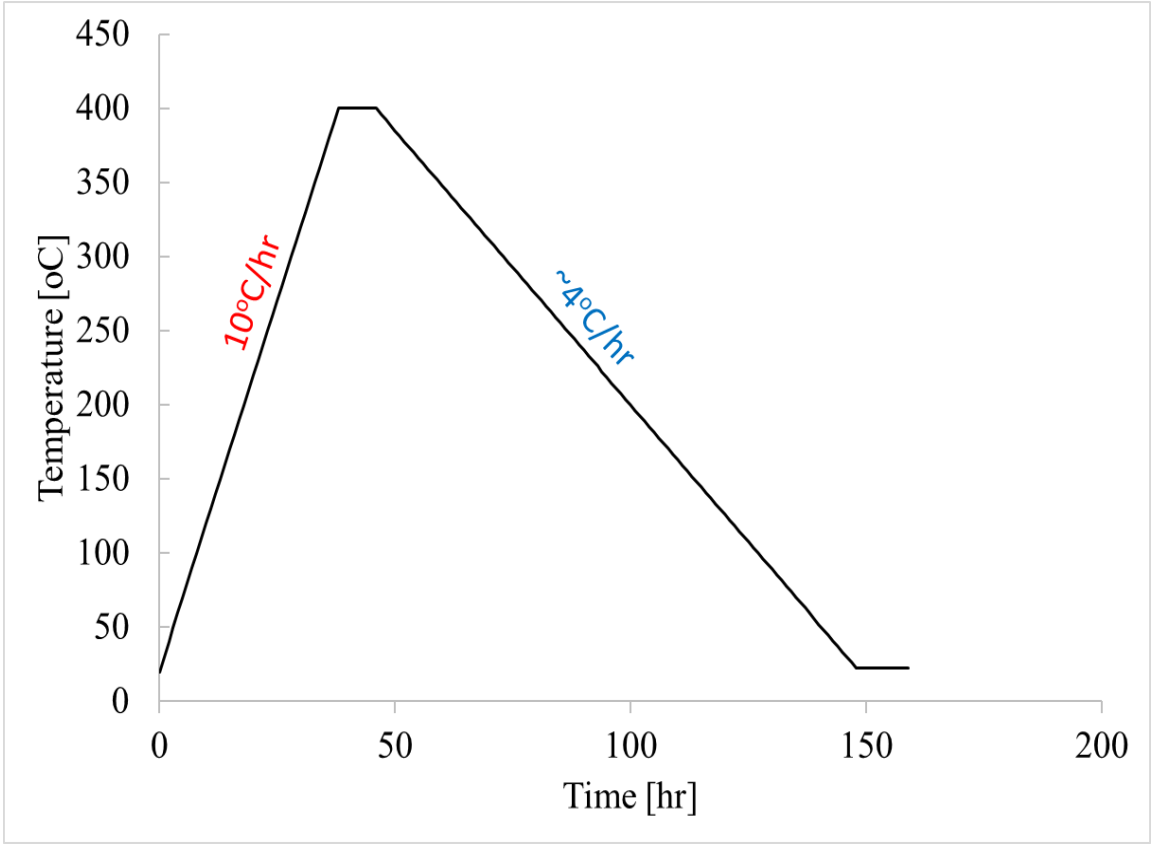
Loaded TN-32B for the High Burnup Research Project at North Anna NPP. Photo Credit: North Anna NPP

Cladding Type	# Rods	Burnup (GWd/MTU)
$M5_{Framatome}$	9	49-57
ZIRLO®	12	49-59
Zr-4	2	58
Low-SN Zr-4	2	48-51

In the Sibling Pin Program, SNF Rods or Pressurized Segments were Heat Treated at PNNL and ORNL at 400ºc Using the Same Temperature Profile

At ORNL, 3 full-length unpunctured SNF rods were full-length heat treated (FHT)



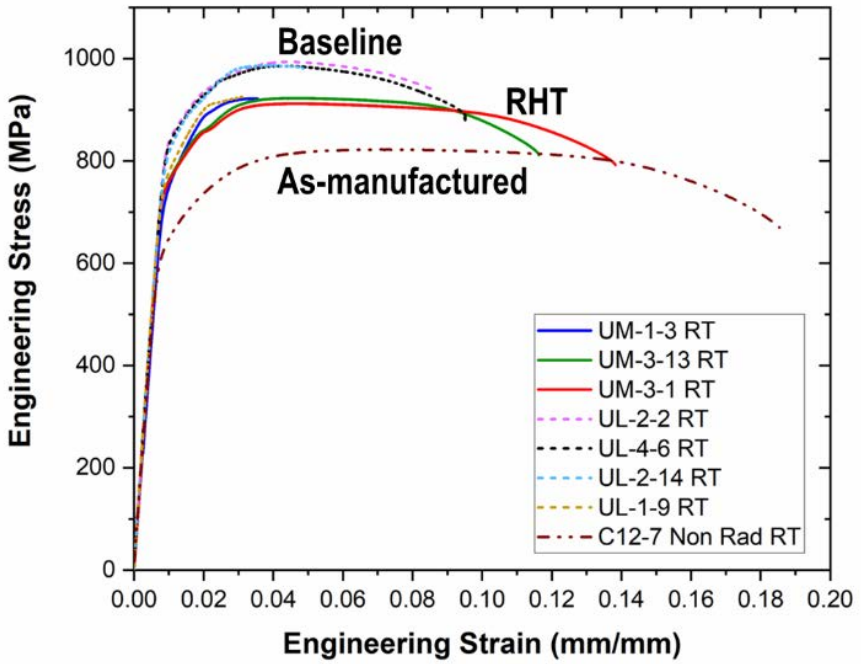


At PNNL, 6” pressurized segments were heat treated; referred to as a Radial Hydride Treatment (RHT)

PNNL observed a decrease in cladding yield stress in simulated vacuum drying at 400°C

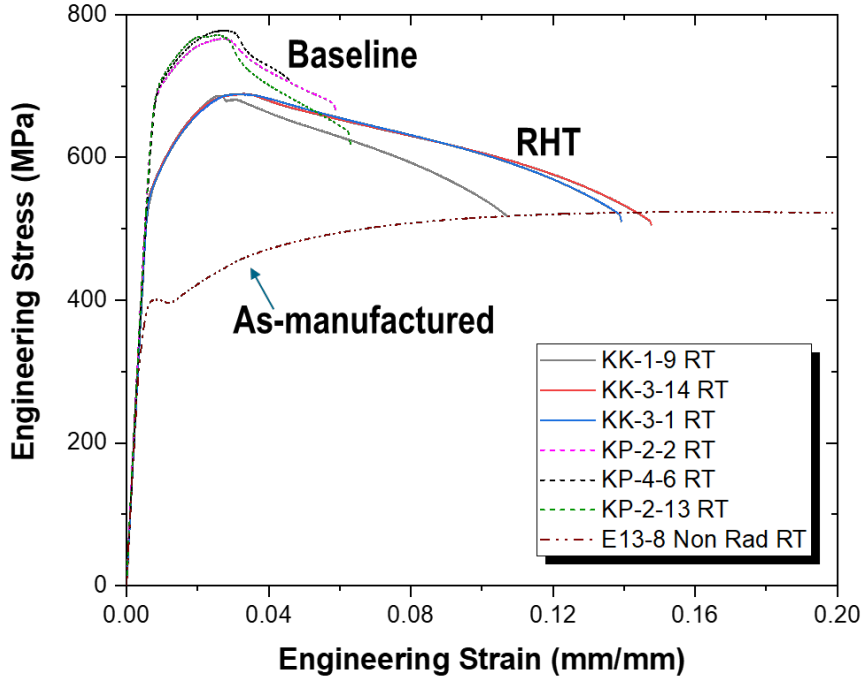
- ASTM E8M
- 6-inch cladding tube samples with 4-inch testing gauge length
- Used Extensometer and Digital Image Correlation (DIC) for axial displacement

Irradiated ZIRLO™ at Room Temperature



Yield Strength ↓ ~10%

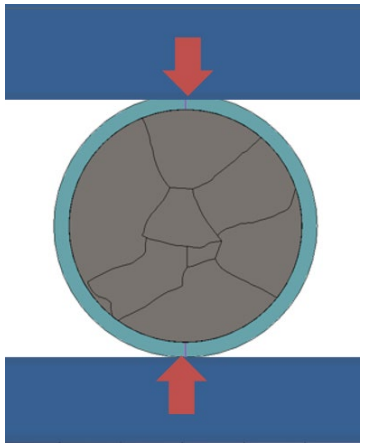
Irradiated $M5_{framatome}$ at Room Temperature



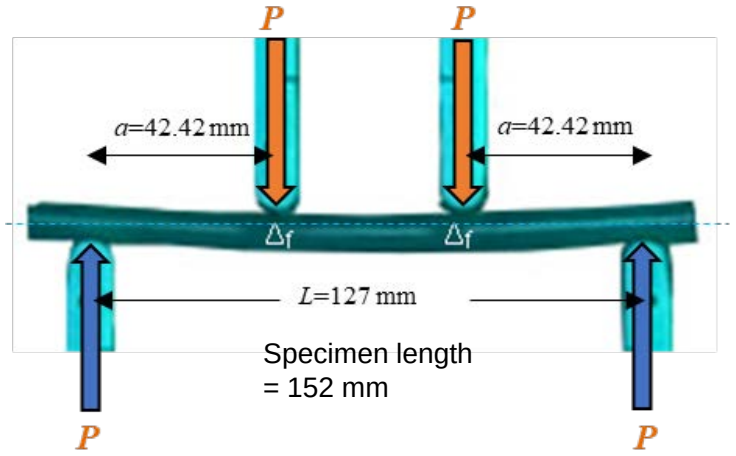
Yield Strength ↓ ~20%

Mechanical tests performed at ORNL used fueled rods (cladding + pellets)

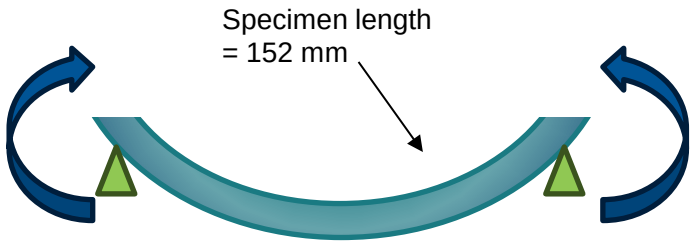
Specimen depth = 25 mm



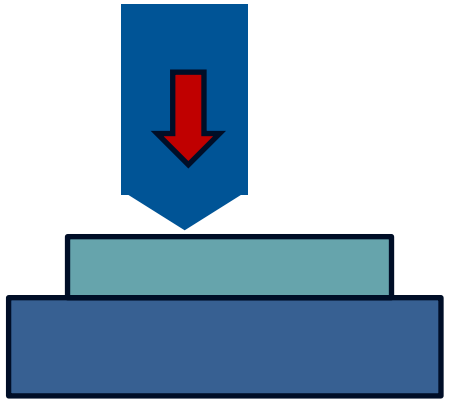
- Fueled Ring Compression Test (RCT)



- Four-point bending (4PB)



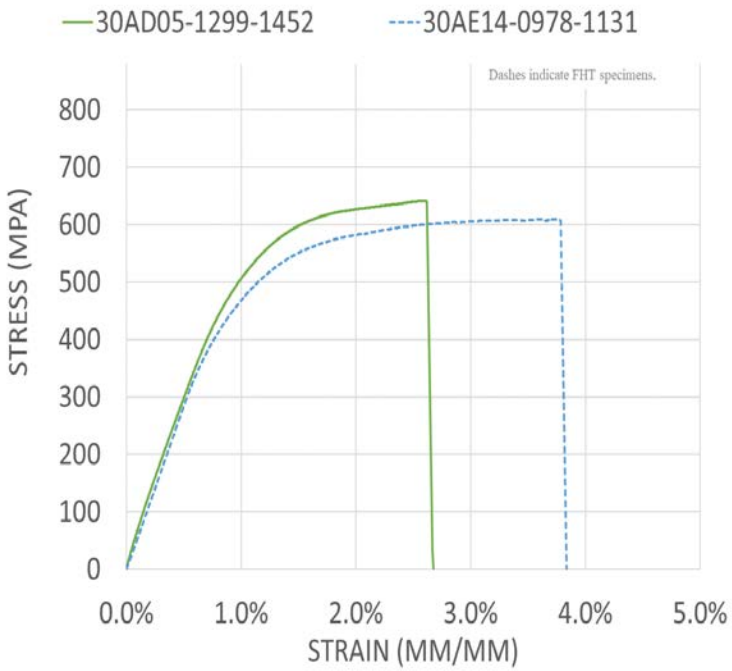
- Cyclic Integral Reversible Fatigue Testing (CIRFT)



- Vickers Micro-hardness

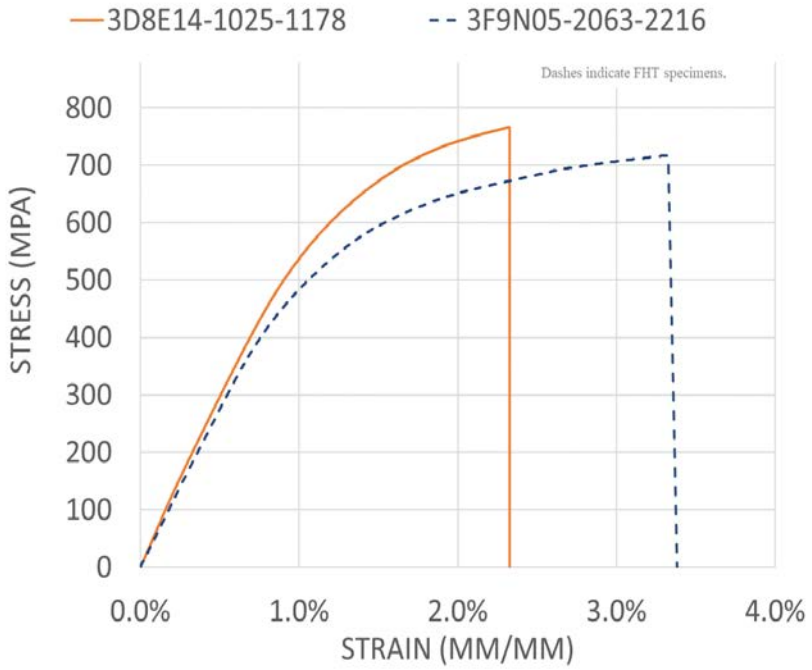
4 point bending test results of SNF rods also show indications of thermal annealing at 400°C (~10% decrease in Yield Strength)

$M5_{Framatome}$ SNF rods at Room Temperature



BU = 58-60 GWd/MTU, rod average

ZIRLO SNF rods at Room Temperature



BU = 59-64 GWd/MTU, rod average

Failure strains are lower than total strains observed in cladding tensile tests

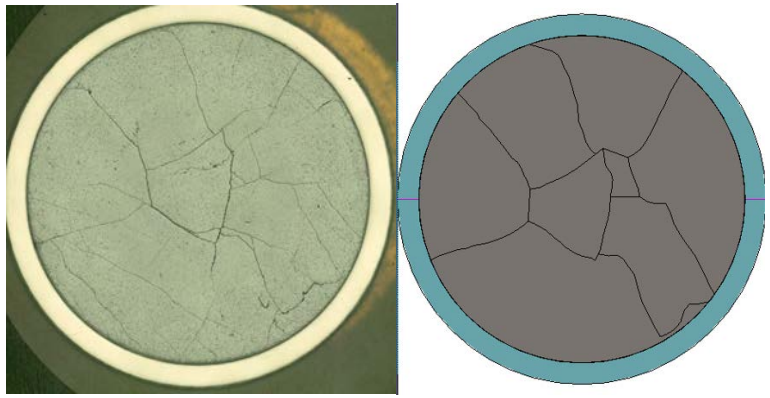
Finite element (FE) modeling performed to understand effect of fuel pellet on performance

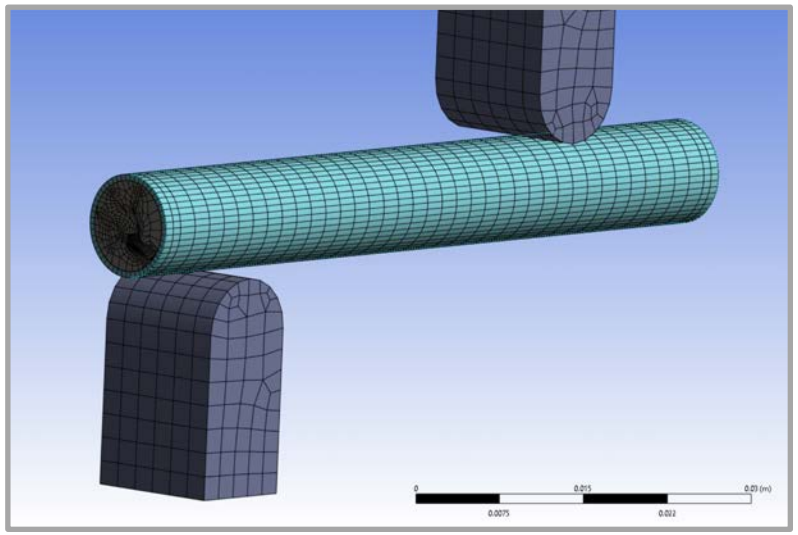
The pellets are segmented into 43 sections (21.5 in. half symmetry)

With each pellet comprised of 8 fragments, there are 176 independent pellet parts in the FE model (with frictional contact)

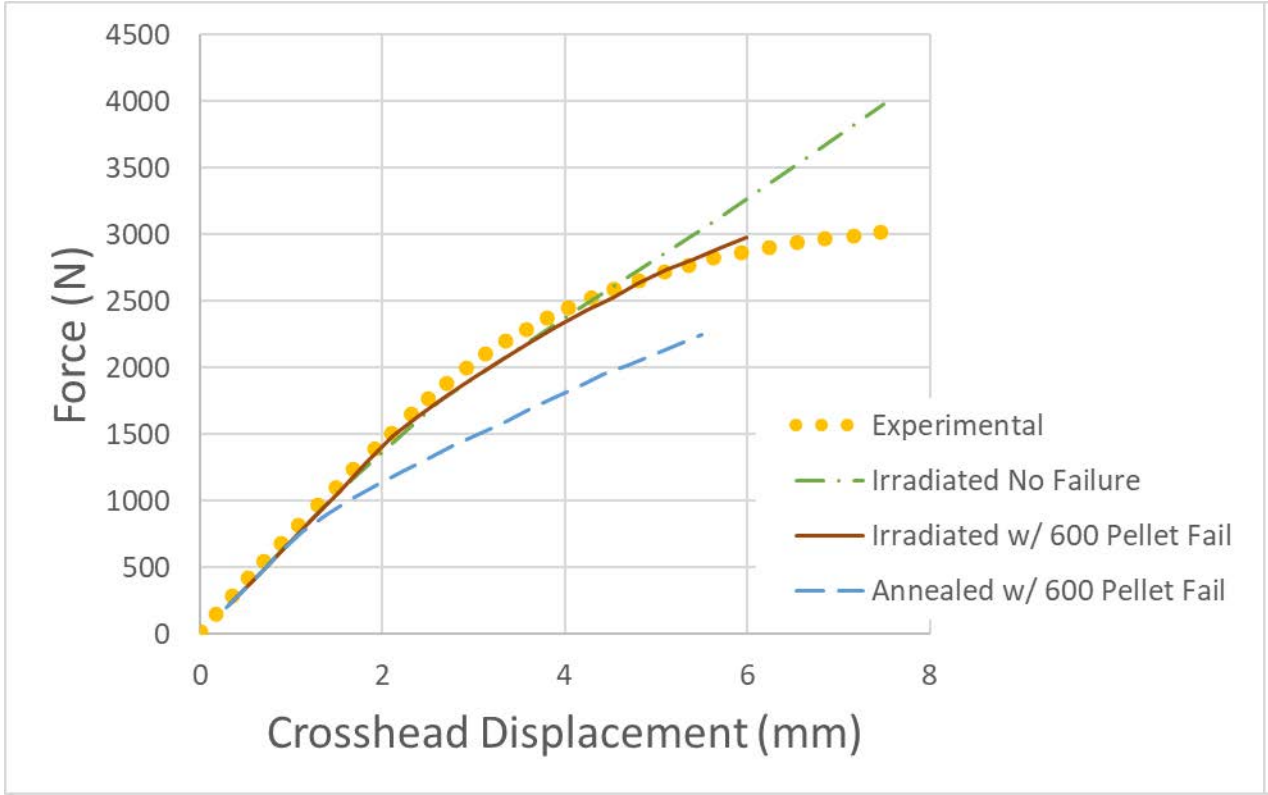
The ANSYS® based quasi-static non-linear finite element model comprised over 60K 3D solid linear elements

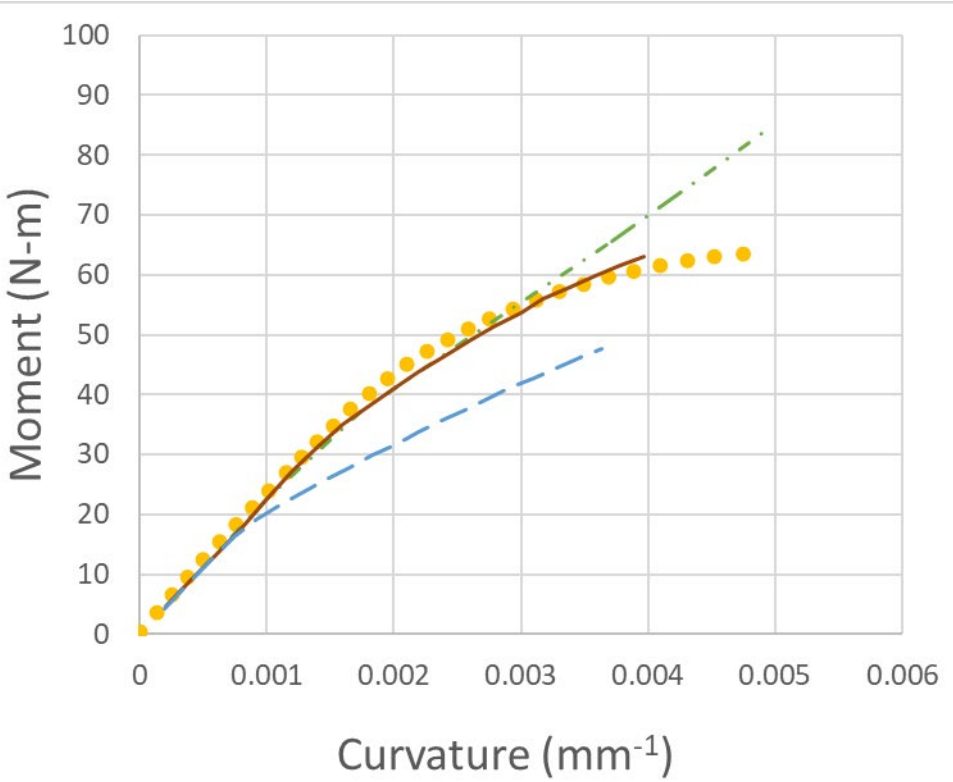
Specimen cross-sectional image is directly used to generate 3D 4PB half-symmetry model





FE predicts that fully annealing the cladding ($\sigma_y \downarrow$ 50%) decreases SNF rod load capacity by only ~20%





Pellet segmenting and fragmenting needed to match Linear response and pellet failure needed to match non-Linear Response

4 point bend data converted to a moment-curvature response, which is likely better for defining a failure criterion that is more applicable to cask vendor drop analysis

Westinghouse has experience providing data for SNF Performance evaluations

- Approved fuel rod models can provide input data for SNF rod evaluations
  - Rod internal pressure (RIP), internal gas inventory in moles, cladding wall thinning due to corrosion, and cladding hydrogen inventory.
- In 1999, Westinghouse submitted a topical for dry cask storage, WCAP-15168, Dry Storage of High Burnup Spent Nuclear Fuel.
- Westinghouse participates in multiparty efforts to obtain data under dry storage conditions for Westinghouse fuel rod cladding alloys: (1) Thermal creep, (2) Hydride reorientation, (3) Maximum allowable creep strain

Framatome has Evaluated the effect of thermal annealing on creep response

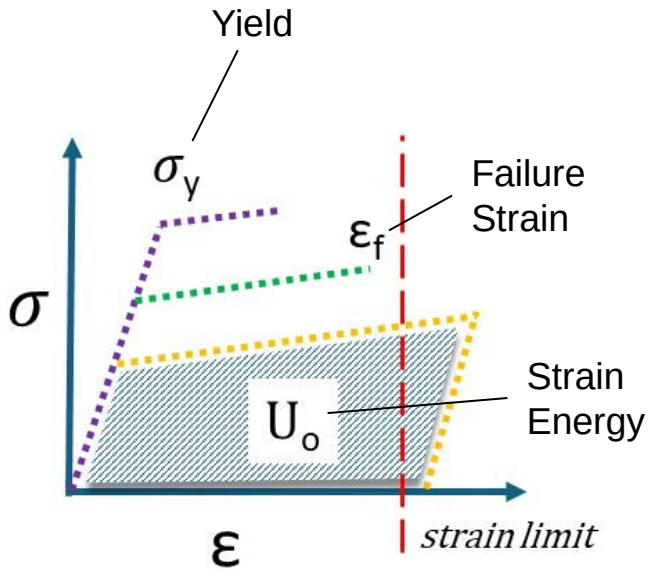
- PWR fuel assemblies with $M5_{Framatome}$ cladding have operated to burnup levels of up to 75 GWd/MTU in Europe and were sent to dry storage facilities successfully
- Thermal creep of irradiated $M5_{Framatome}$ cladding less sensitive than other alloys to thermal annealing[1]
  - Thermal annealing at 450^{o}C had no effect on thermal creep at 400^{o}C, 130 MPa (cladding stress)
  - Niobium-rich nanoprecipitates act as dispersion strengtheners when irradiation damage anneals out
- Thermal creep of irradiated Zircaloy-2 cladding is sensitive to thermal annealing

[1] F. Onimus, S. Doriot, A. Ambard, F. Bourlier, B. Verhaeghe, Th. Le Jolu, C. Cappelaere, "Understanding post-irradiation creep behavior of M5Framatome zirconium alloy," Journal of Nuclear Materials, Volume 597, 2024,

Global Nuclear Fuel (GNF) has considered Thermal Annealing in licensing BWR fuel for over 40 years

- GNF's flagship thermal-mechanical code, PRIME, explicitly considers thermal annealing of the cladding when evaluating overpower conditions resulting from anticipated operational occurrences (AOOs)
    - The maximum allowable cladding temperatures during normal operations and AOOs remain sufficiently low such that full annealing is not expected to occur during the fuel rod's lifetime
- Licensing limits established for normal operation and AOOs may be applicable to back-end applications
- Fuel vendors should be engaged and made aware of decisions that have the potential to push their fuel products outside of the original design basis
- Fuel vendors provide 95/95 confidence for in-reactor fuel performance; what confidence level is acceptable for SNF?
- The limited engagement between fuel vendors and parties engaged in dry storage should change to ensure all parties are aligned on expectations for SNF performance

Failure Criterion that could replace cladding yield strength as a load Limit



Schematic illustration of various metrics associated with failure analyses shown on a stress (σ) vs. strain (ε) curve.

- Failure criterion protecting against fracture (not just yield) should be based on the fuel rod and not just the cladding
- Cladding that has yielded is still capable of carrying mechanical load up to a point of ultimate fracture.
- Failure criteria may be based upon limit stress, total strain allowable, or total strain energy
- Total strain and strain-energy based approaches have the advantage of reflecting considerable post-yield structural capacity
- Demand loads in drop analyses are represented by predicted moments and bending curvatures, which make them good parameters for a failure criterion

Highlights from the Workshop

- Fuel performance requires a cradle-to-grave approach – fuel vendor perspectives are important to consider on backend evaluations
- Thermal annealing found to occur at the current NRC temperature limit (400^{o}C)
  - Decreases yield strength of cladding and SNF rods under bending
  - Has little effect on SNF performance under pinch loading conditions
- Westinghouse has submitted a topical report to the NRC on fuel performance in dry storage conditions
  - Fuel vendor topical reports could include the effect of thermal annealing
- Framatome has evaluated thermal annealing effects on creep
  - Found thermal creep of $M5_{Framatome}$ is less sensitive to thermal annealing than other cladding alloys
- GNF considers thermal annealing for evaluations of AOOs and suggested that failure criteria used in these evaluation may be applicable to the backend
- Yield strength is unnecessarily conservative and alternative failure criterion should be developed based on SNF rod fracture
- All workshop participants thought it appropriate for fuel vendors to be engaged and aware of potential decisions that may push their products outside the original design space

Questions

M5$_{Framatome}$ is a trademark or registered trademark of Framatome or its affiliates, in the USA or other countries

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Spent Fuel and High-Level Waste Disposition

ESCP meeting, October 27-30 2025, Charlotte

Acquired Data and Planned New Testing Campaign at JRC

V.V. Rondinella, T. Wiss, R. Nasyrow, C. Nourry, D. Papaioannou
European Commission, JRC-Karlsruhe

S. Caruso, Gösgen-Däniken Kernkraftwerk, CH

E. Vlassopoulos, Nagra, CH

nagra

Kernkraftwerk Gösgen

European Commission

Outline

- Introduction - context
- Mechanical testing of spent fuel rod segments
- Spent fuel ageing
- Perspectives – new campaign

Acknowledgements

Kernkraftwerk Gösgen

framatome

nagra.

eurad
European Joint Programme on Radioactive Waste Management

ÉCOLE POLYTECHNIQUE FÉDÉRALE DE LAUSANNE

BAM
Bundesanstalt für Materialforschung und -prüfung

SNL, PNNL

Universidad Politécnica de Madrid

European Commission

Spent fuel (SF) safety studies at JRC Karlsruhe

assess SF/wasteform ability to fulfil its expected function over long-term

(Extended) Storage

radionuclides containment, rod retrievability *(≥100 y?)*

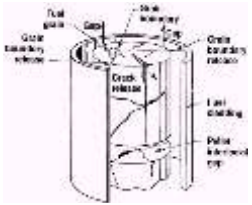
Spent fuel rod mechanical integrity degradation processes during storage

- cladding (hydrides);
- fuel: decay damage, He accumulation;

Accident conditions

pools, handling, transport, storage, retrieval: corrosion, loss of cooling; damaged SF, debris properties.

- Mechanical load resistance: impact, bending;



Geologic Disposal

reduce uncertainties on release of long-lived radionuclides over an *open-ended disposal timescale*

radionuclides “Source Term”, “Instant Release”; matrix corrosion: effects of environment and spent fuel properties

Convey experimental data into models and codes (predictions)

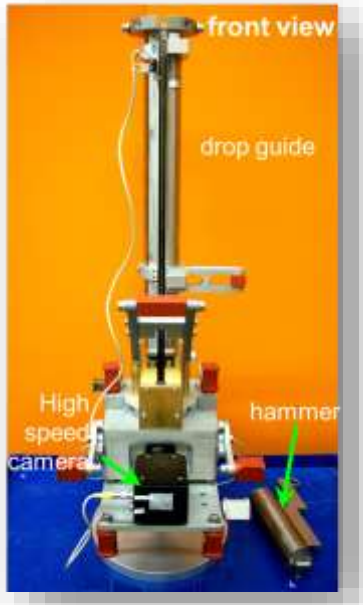
European Commission

Mechanical tests at JRC-Karlsruhe

Reason:

Assessing the mechanical integrity and level of functionality of spent fuel rods retained during handling, storage and transportation, and behaviour under accident conditions.

Macroscopic investigation of spent fuel rod mechanical properties (flexural) under static and dynamic external solicitations.



Scope:

Establish a basis of reference data and evaluate the response of spent fuel rods to external solicitations that might occur under various accident conditions. Feed modelling developments.

Approach:

Bending and impact tests on spent fuel rod segments;

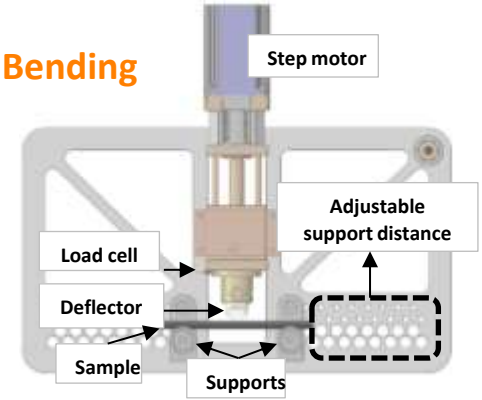
Ring compression tests*;

Burst or long term creep tests on cladding tubes*.

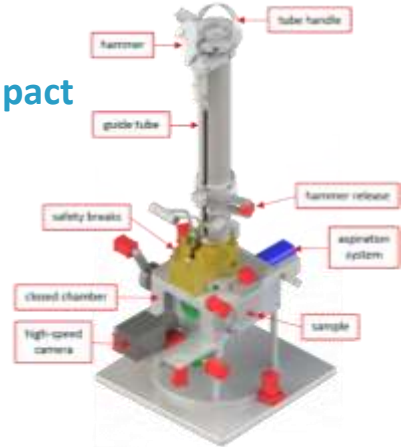
*fueled and defueled

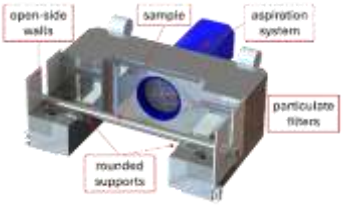
Setup for bending and impact tests

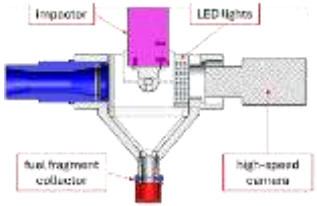
Three-Point Bending

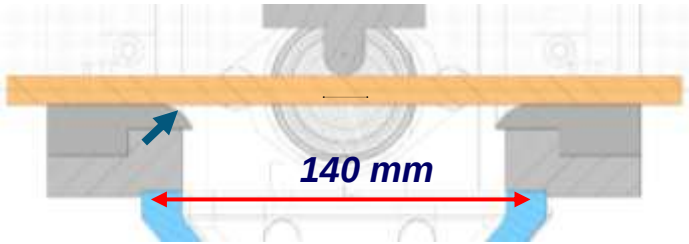


Impact









Similar geometrical configuration enables correlation between quasi-static and dynamic loads

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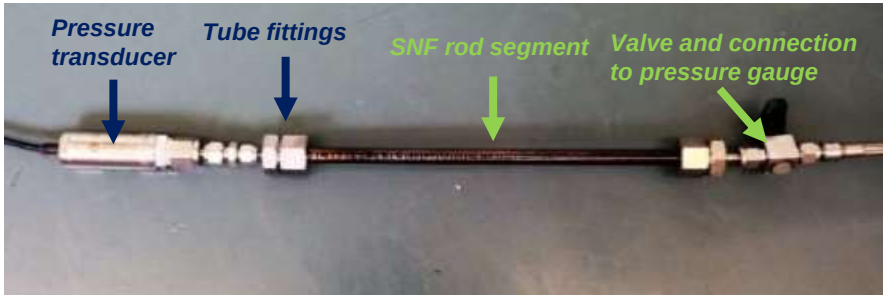
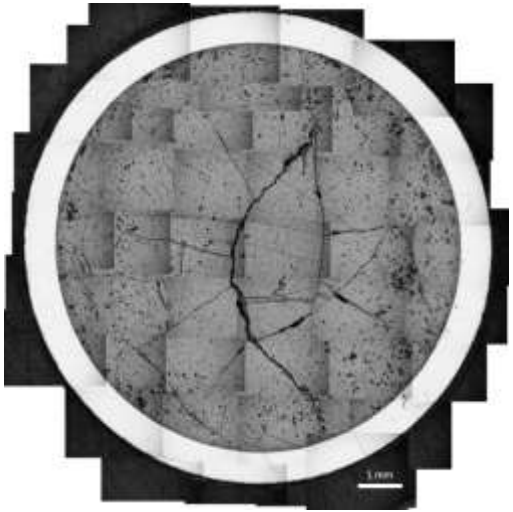
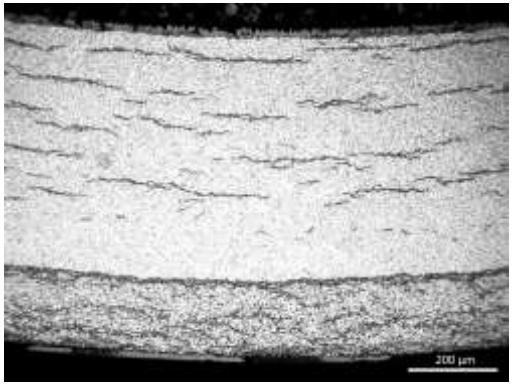
Fuel segments for mechanical tests







Specimen:

- Fuel: UO_2, $(Pu,U)O_2$
- Segment: central axial position, no spacer contact;
- Cladding: DX D4, DX Zr2.5Nb, duplex Zry-4;
- NPP: PWR Gösgen / Grohnde;
- Aver. burn-up: 18-100 GWd/tHM;
- Re-pressurized up to operational level.

Fuel samples for mechanical tests

Owner	Sample ID	Fuel type	Cladding	Avg. burnup [GWd/tHM]	Irradiation cycles	RIP [bar]
KKG	A11	UO2-ERU	DX D4	18.3	1	29.4
KKG	D08	UO2-ERU	DX D4	46.9	3	39.3
KKG	A07	UO2-ERU	DX D4	58.6	4	68.4[1]
KKG	DO2	UO2-AUC	DX Zr2.5Nb	100.5	10	105.9
KKG	6001-03	MOX-MIMAS	DX D4	60.6	4	53.0[2]
KWG	A16-5	UO2	Duplex Zry-4	67.0	5	70.0

[1] For the three-point bending tests, samples from this rod were pressurized to 32 bar.
[2] In this study, samples from this rod were pressurized to 40 bar.

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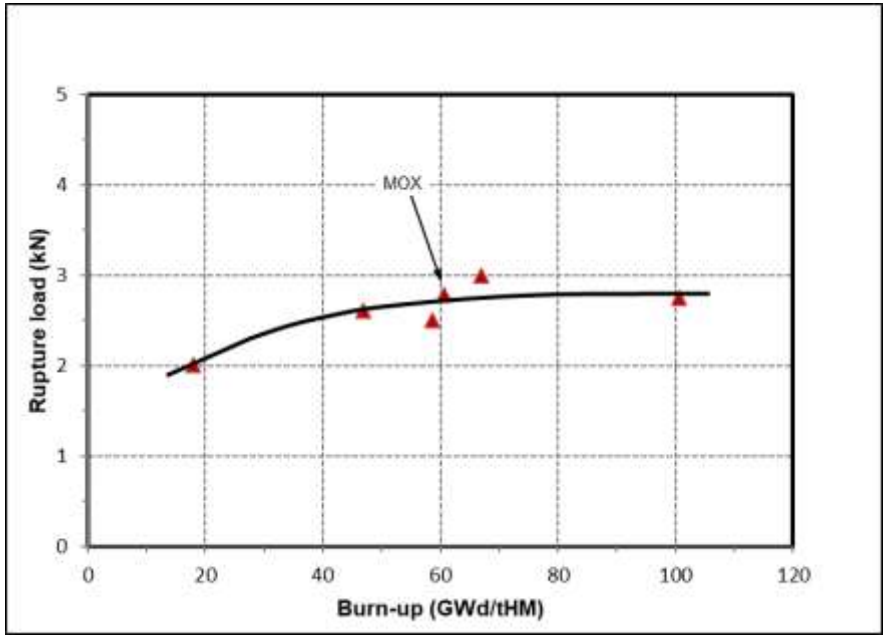
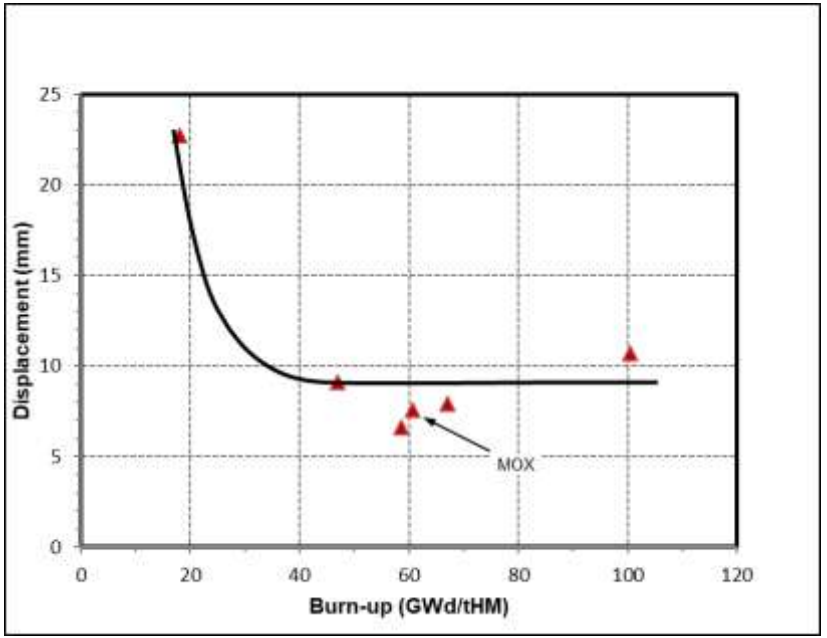
Summary of 3-point bending tests



Rupture load as a function of burnup



Displacement to rupture as a function of burnup

- typical deflector translation: 16.7 µm/s
- load, displacement and pressure monitored online
- fuel mass release determination

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European Joint Programme on Radioactive Waste Management
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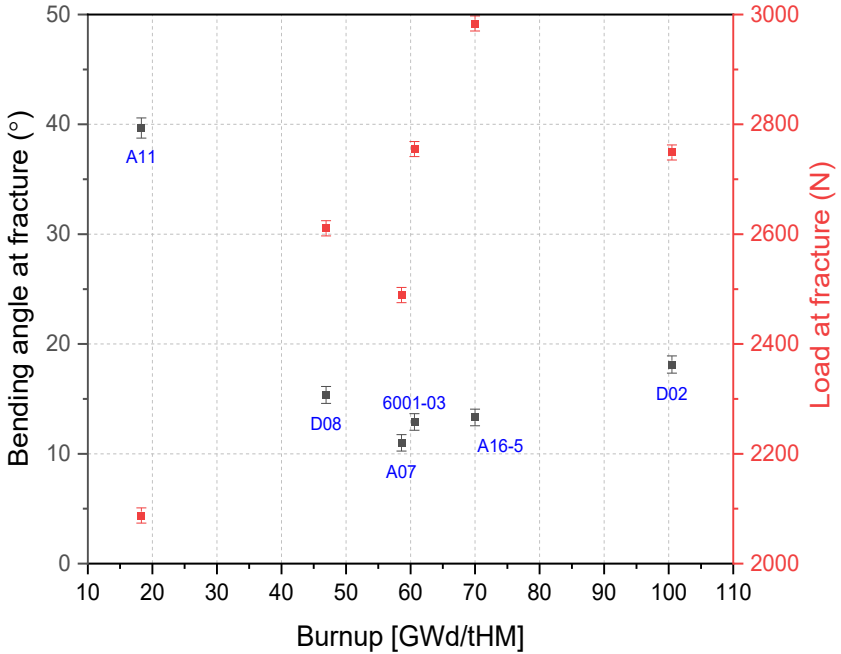
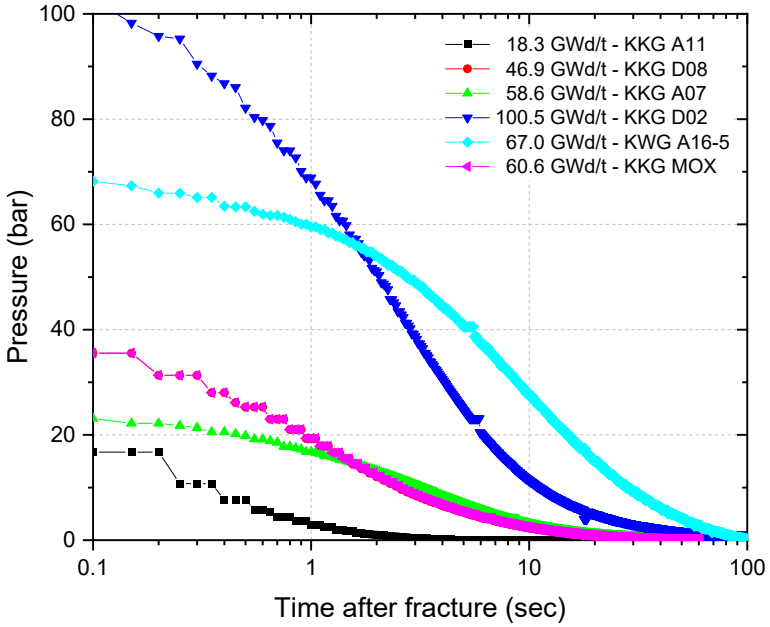
3-point bending results



Bending angle and fracture load vs. burnup

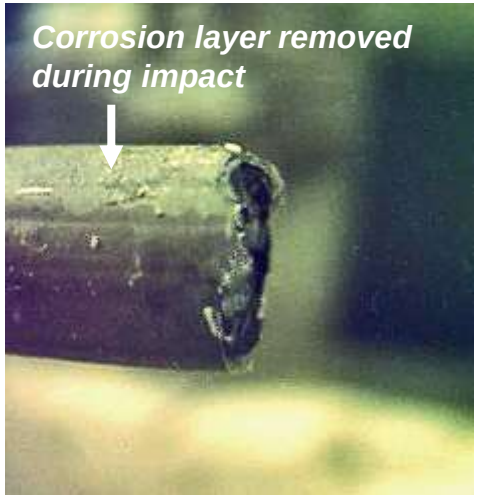


Pressure drop after failure

E. Vlassopoulos et al., 2025

European Commission

Broken segment after bending



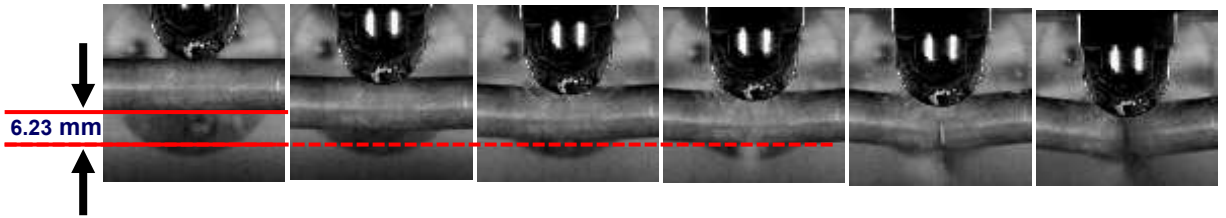
Fracture surfaces after impacting:

- ***Only limited fuel amount released, strong bonding with the cladding;***
- ***Near the impact location, the outer corrosion layer was removed from the cladding.***

European Commission

Impact test

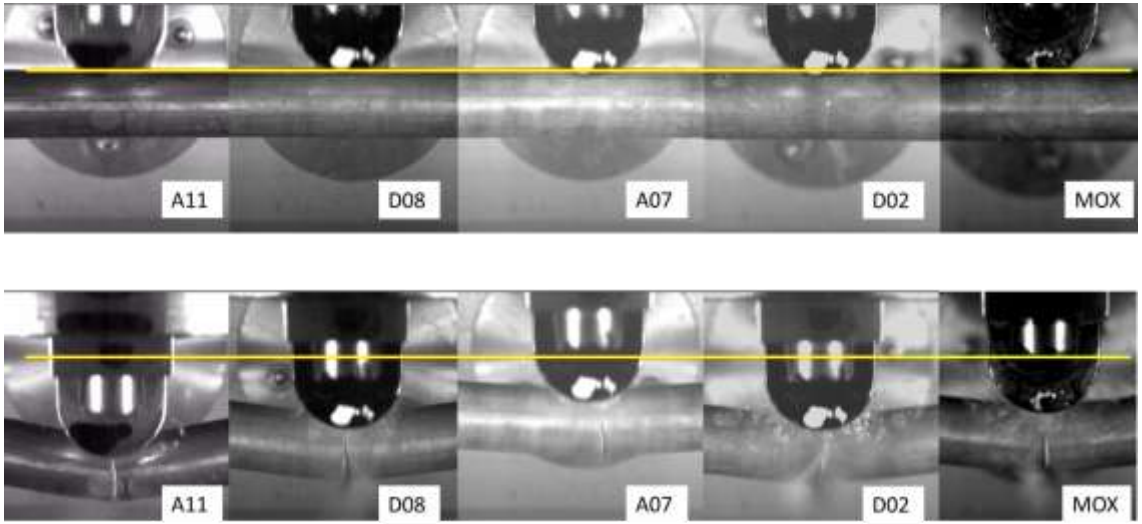
61 GWd/tHM MOX rod segment



- Test recorded by a high speed camera;
- Splits from the outer corrosion layer are first detached
- Due to internal pressure fine particles are released explosively
- Displacement till crack initiation
- Absorbed energy upon fracture
- Fuel mass released
- Released fuel characterization

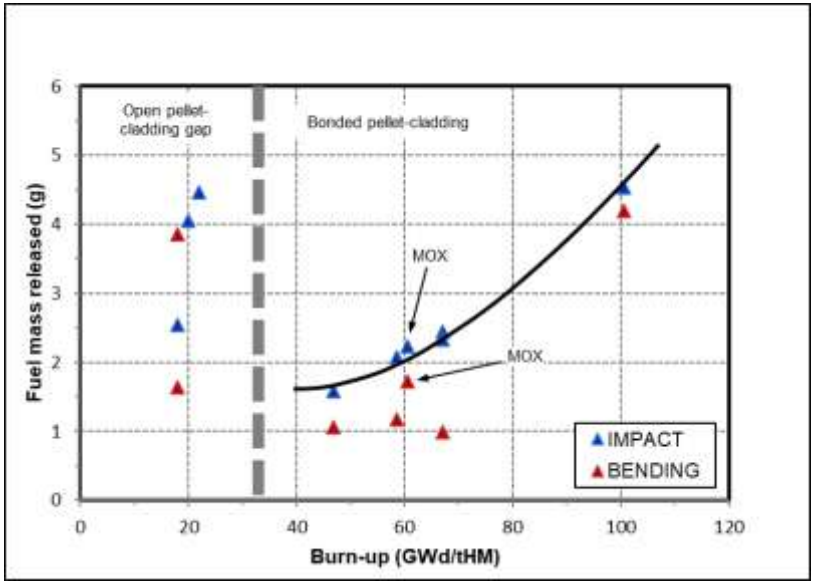
European Commission

Impact tests



European Commission

Mass release during the SF tests



Fuel mass released during impact and bending tests as a function of burn-up.

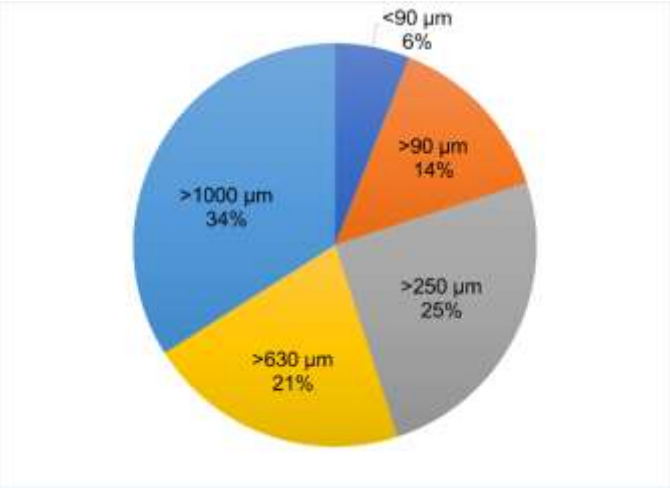
- No significant difference between MOX and UO_2
- The fuel masses released during bending or impact tests are of the same order of magnitude; the higher release during impact is attributed to the higher energy transmitted to the specimen
- The released fuel < the mass of a fuel pellet
- The fuel appears to control the overall SNF behaviour; effects of specific cladding properties (irradiation history, composition) on fuel release do not show clear trends
- Increase of released mass at very high (and very low) burn up was observed.

eurad European Joint Programme on Radioactive Waste Management
European Commission

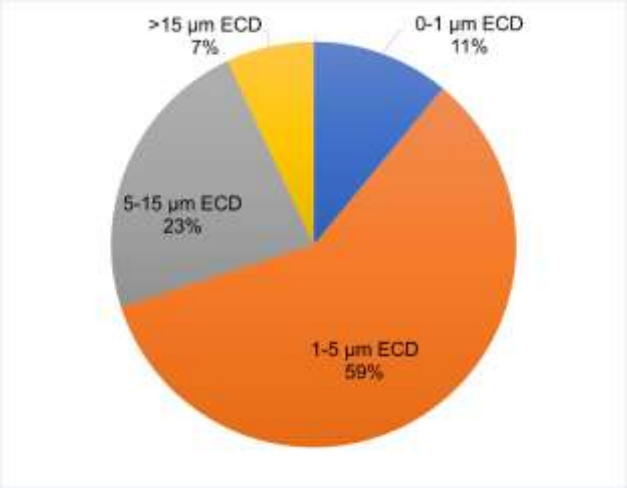
Characterization of released fragments

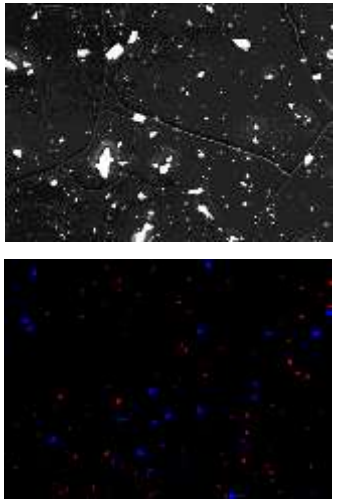
99.1% consists of fragments collected at the bottom of the testing chamber

0.9% is the aerosol and fine particulates fraction.



Size distribution of coarse fragments





Size distribution and SEM image of the aerosol particles; analysis in terms of equivalent circle diameter (ECD).

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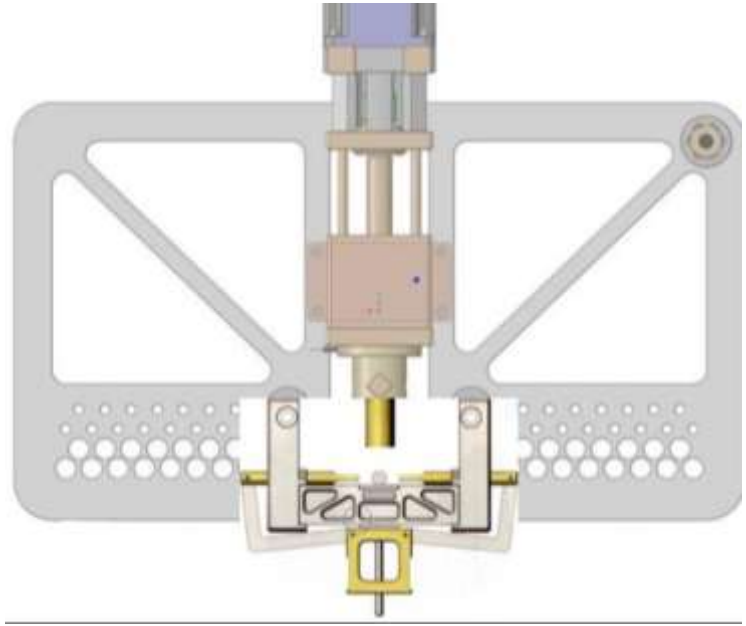
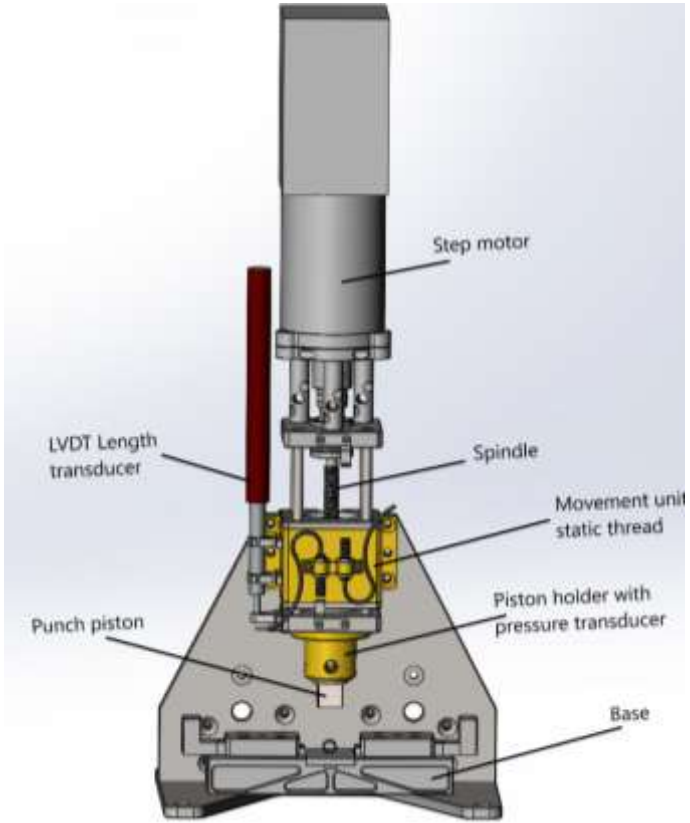
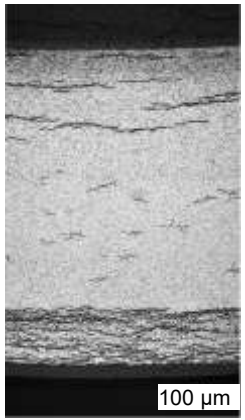
Setups for ring compression tests





Specimens for ring compression tests



67 GWd/tHM

18.3 GWd/tHM

100.5 GWd/tHM

Rings from low burnup regions → no need for mechanical or chemical defueling

67 GWd/tHM

18.3 GWd/tHM

100.5 GWd/tHM

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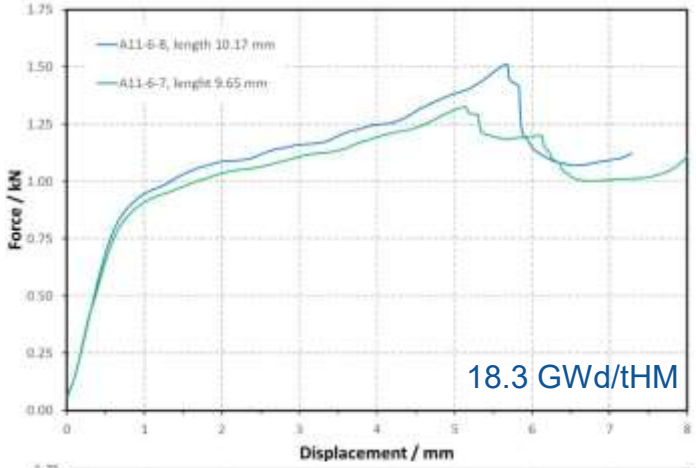
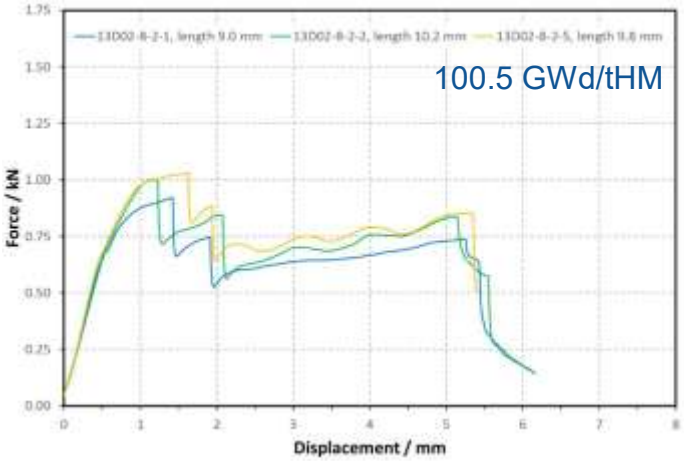
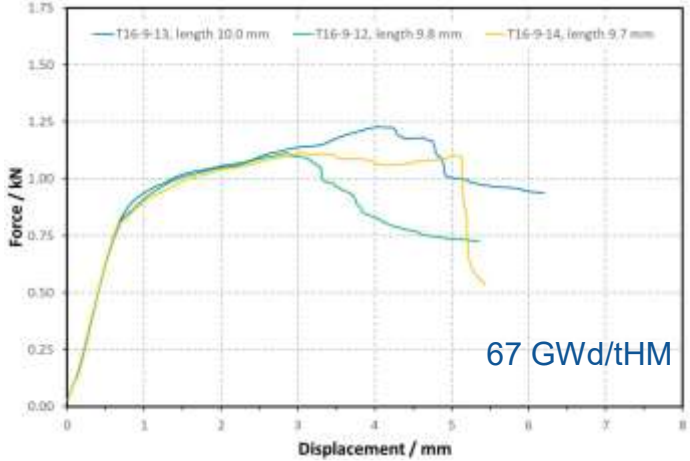
European Joint Programme on Radioactive Waste Management

European Commission

Force-displacement curves



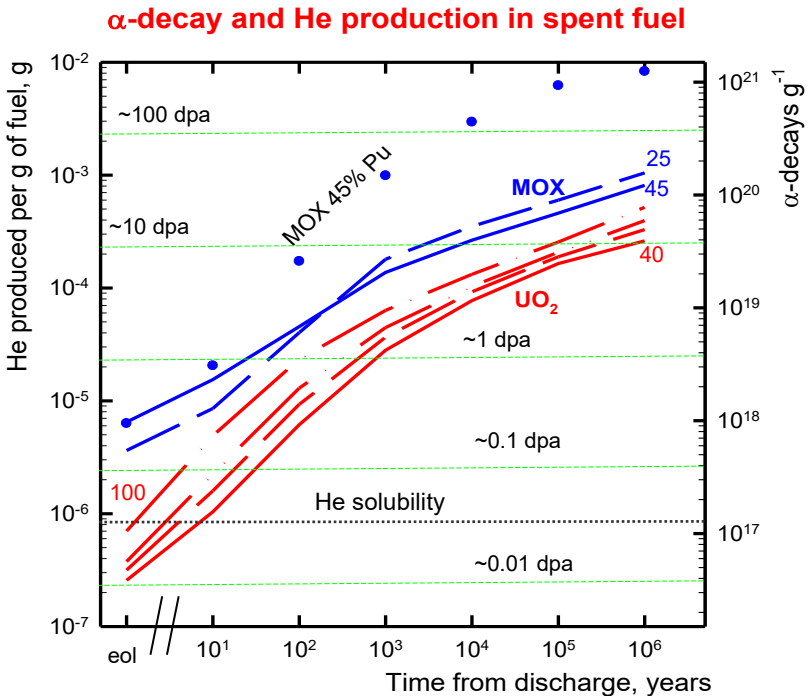




Effect of specimen height
Embrittlement at very high burn-up.

Fuel properties evolution during storage

how to predict SF properties over a timeframe of a few centuries?



Effects at micro- and macrostructure level;

→ tests on high decay rate α-doped UO_2

Analysis of basic mechanisms linked to **accumulation of decay damage/He** and thermal **recovery**:

- microstructure (defects)
- lattice parameter (swelling)
- hardness
- heat effects (apparent C_p)
- thermal diffusivity
- He solubility & release behaviour

→ **assess applicability to spent fuel**

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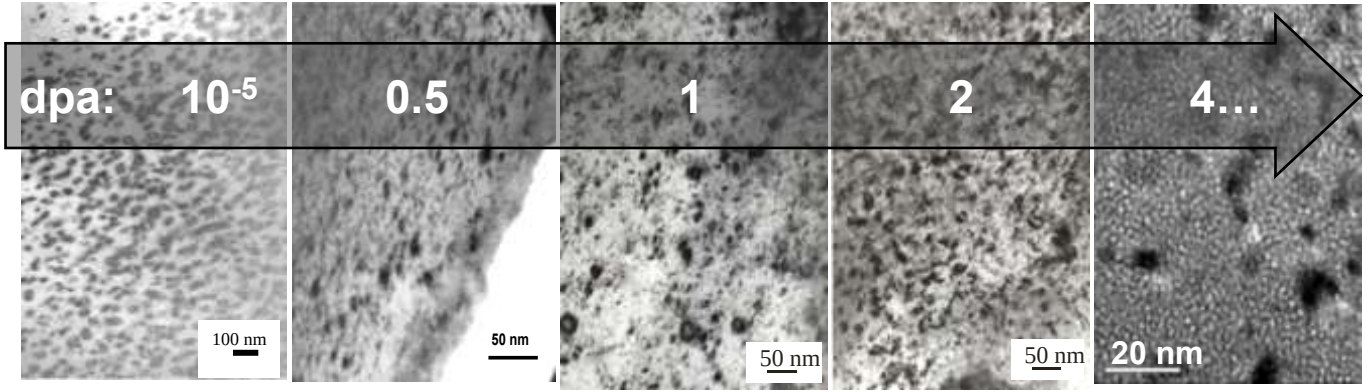
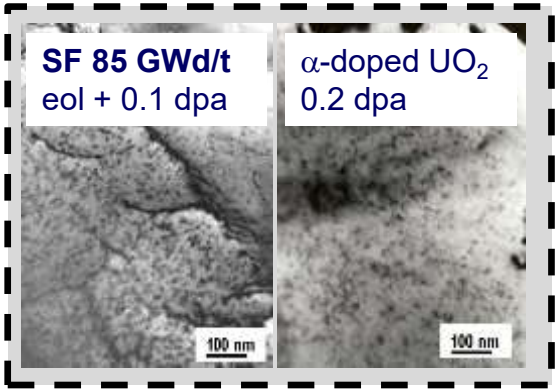
Alpha-decay effects on microstructure

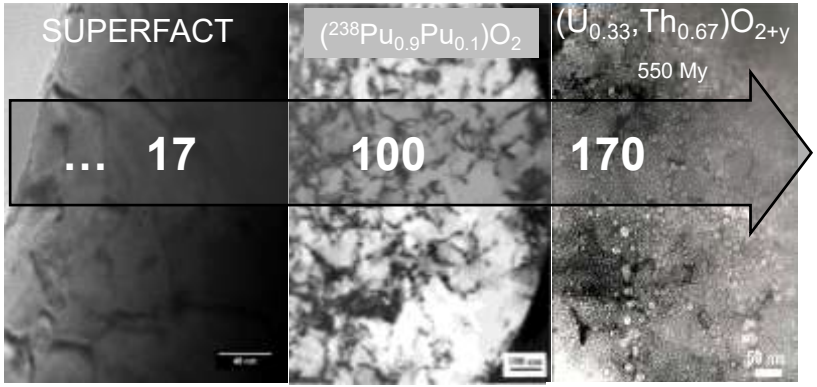
α-doped UO_2



spent fuel vs. α-doped



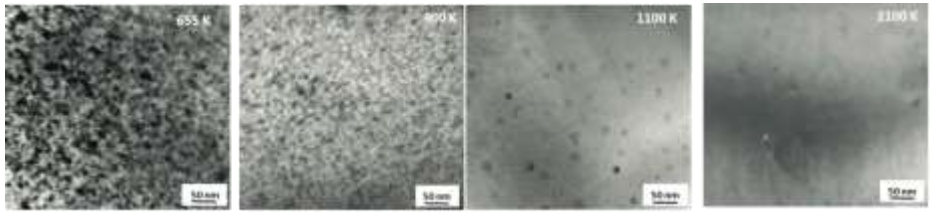
extreme consequences – high dpa

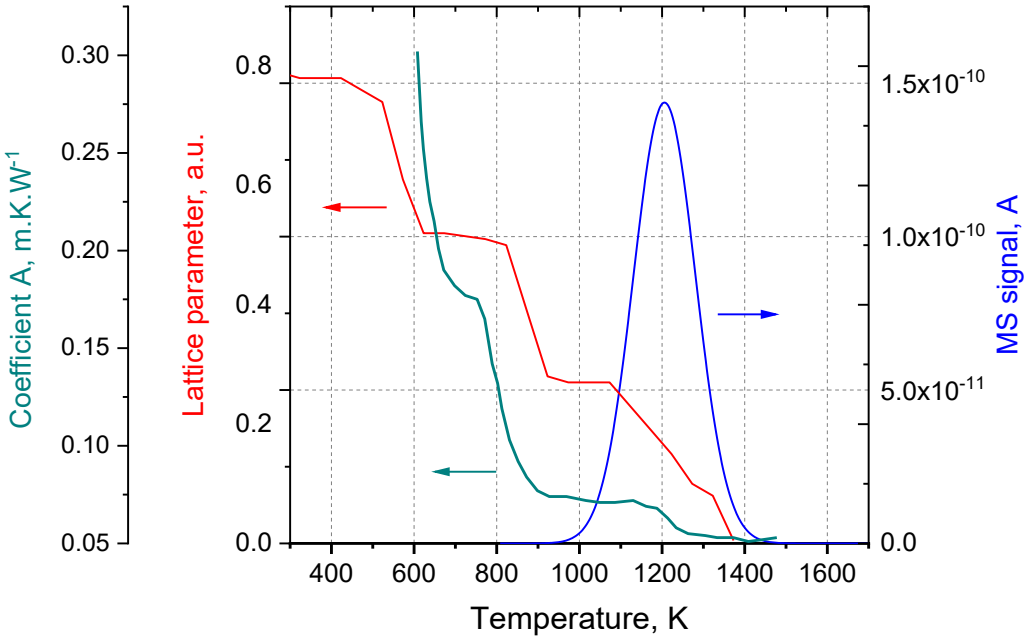


advanced fuel | α-doped | natural mineral

Annealing stages

XRD, thermal conductivity, He release measurements





Transmission Electron Microscope (TEM)

Laserflash for thermal conductivity determination

Knudsen cell for He release measurement

XRD: X-Ray Diffractometry

Fuel properties evolution during storage

SF properties affected at micro- and macrostructural level during storage;

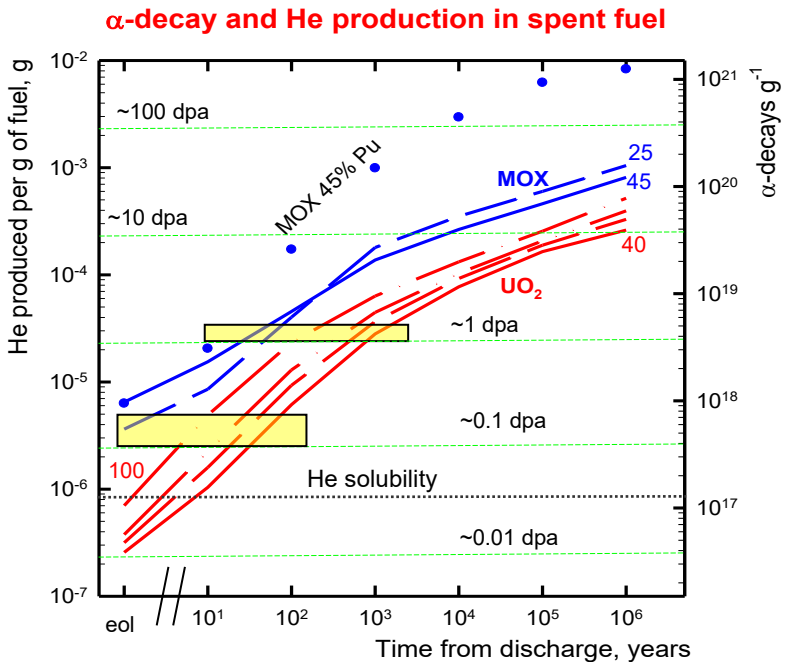
→ tests on alpha-doped UO_2 (accelerated conditions), spent fuel

Most properties show saturation around 0.1-0.2 dpa

Lattice swelling: 0.2-0.3% at 0.1–0.2 dpa, $\leq 10^2$ y storage

max ~0.4% @ ~1.2 dpa, $\geq 10^2$ y range for high burnup UO_2;
if confirmed, such levels of swelling may induce relevant stress on the cladding

→ **validate applicability to spent fuel**



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Perspectives

Extend matrix of irradiated fuel testing:

- M5 cladding (→ new campaign in collaboration / with support of BGZ, Framatome, Gösgen)

- Innovative fuel types (depending on availability)

Extend spectrum of experimental conditions:

- Introduce thermal treatments

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Thank you

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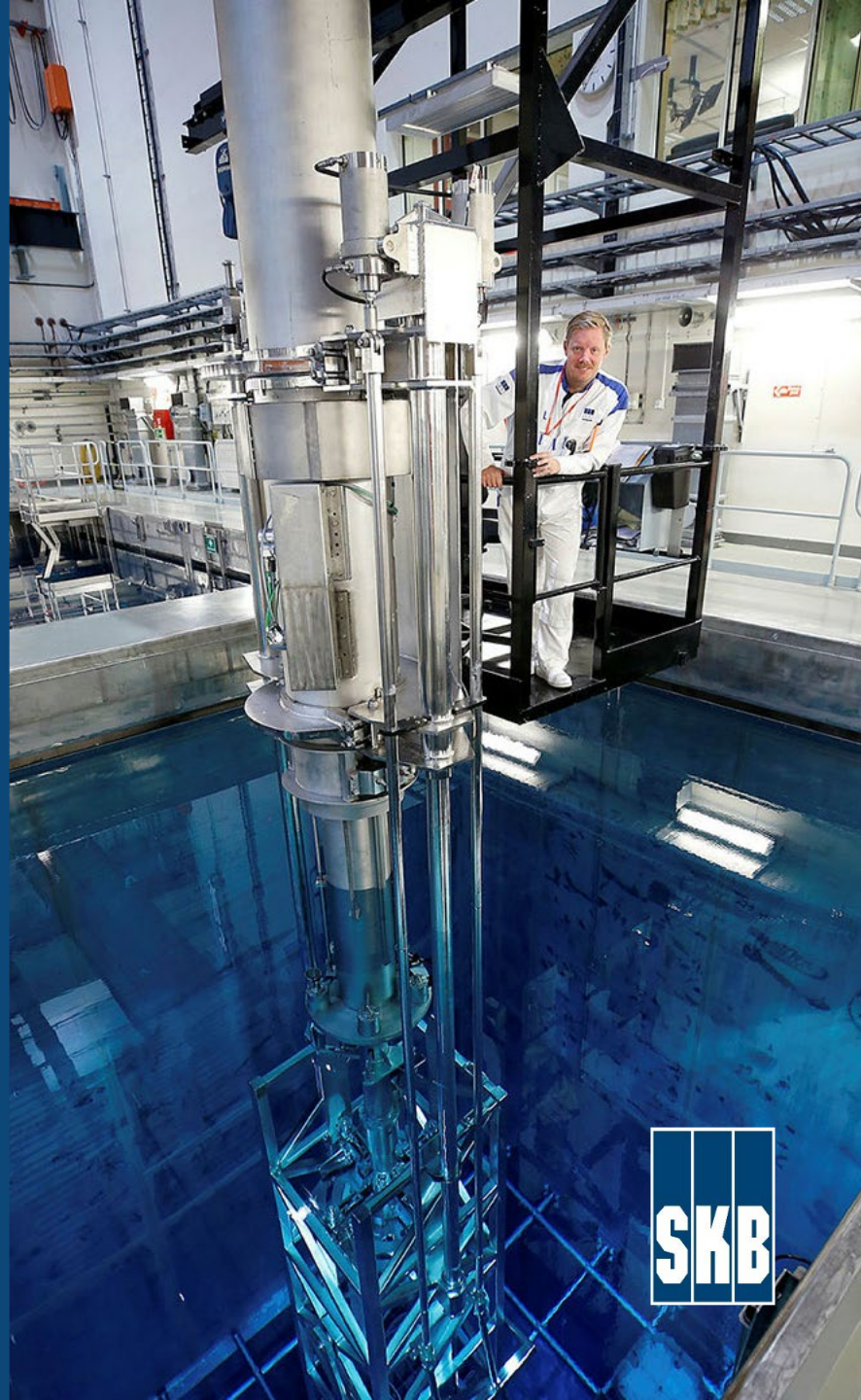
European Commission

Fuel Database and How to Preserve Enough Fuel Information to Optimize the Final Storage

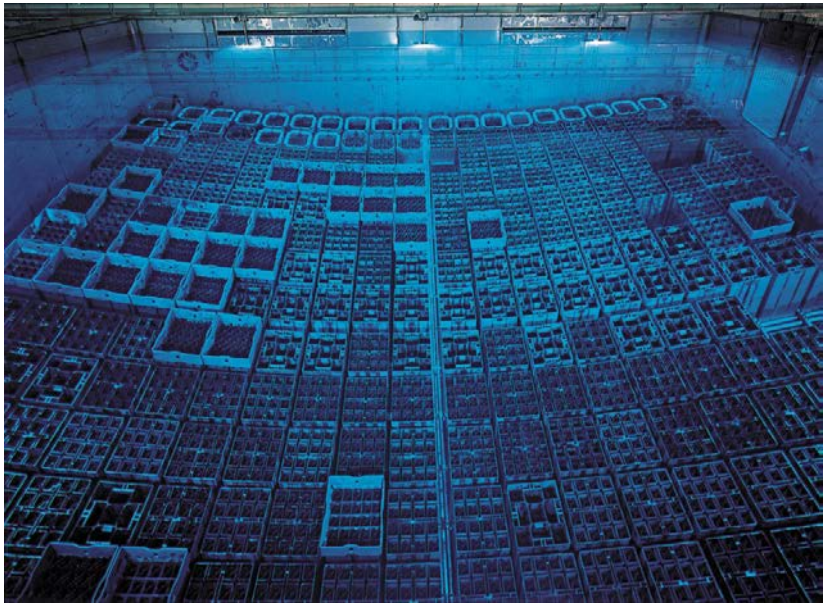
Fredrik Johansson
Swedish Nuclear Fuel and Waste and Management Co

2025-10-27 ESCP Charlotte NC USA

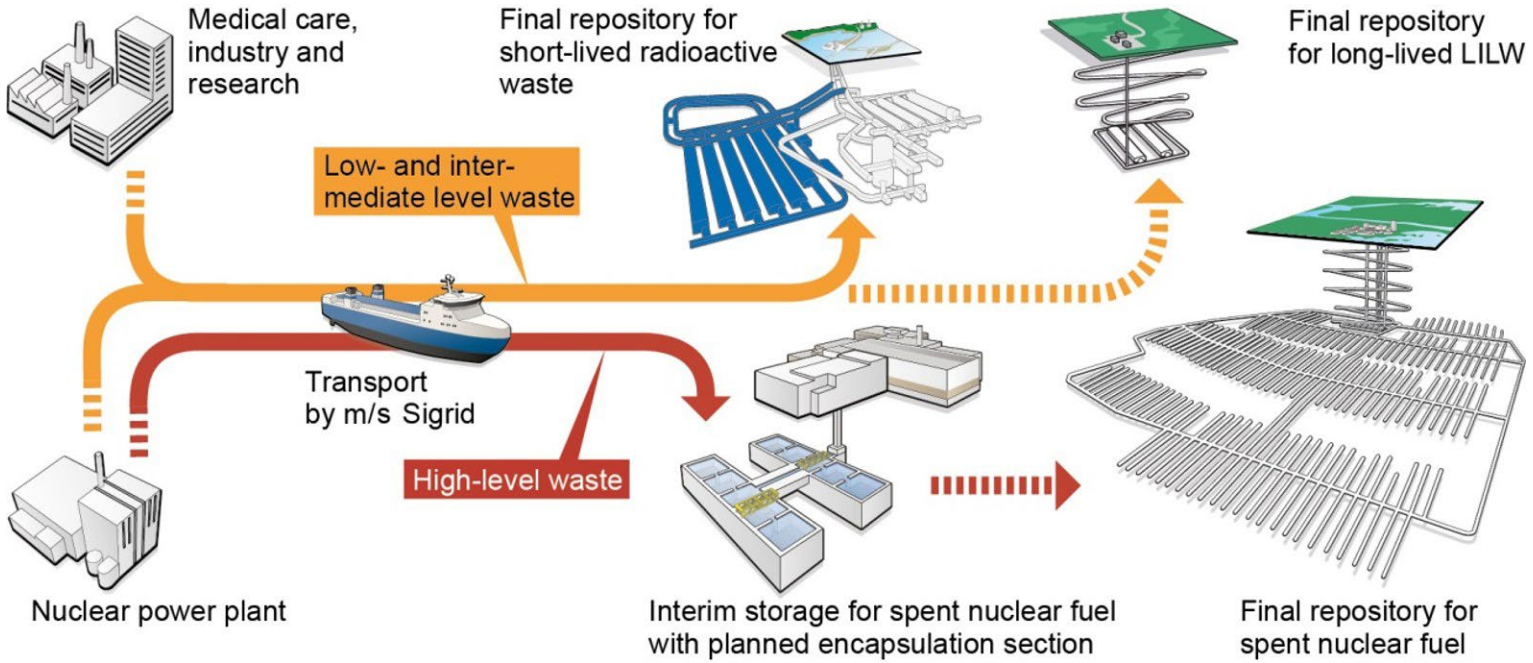


Outline

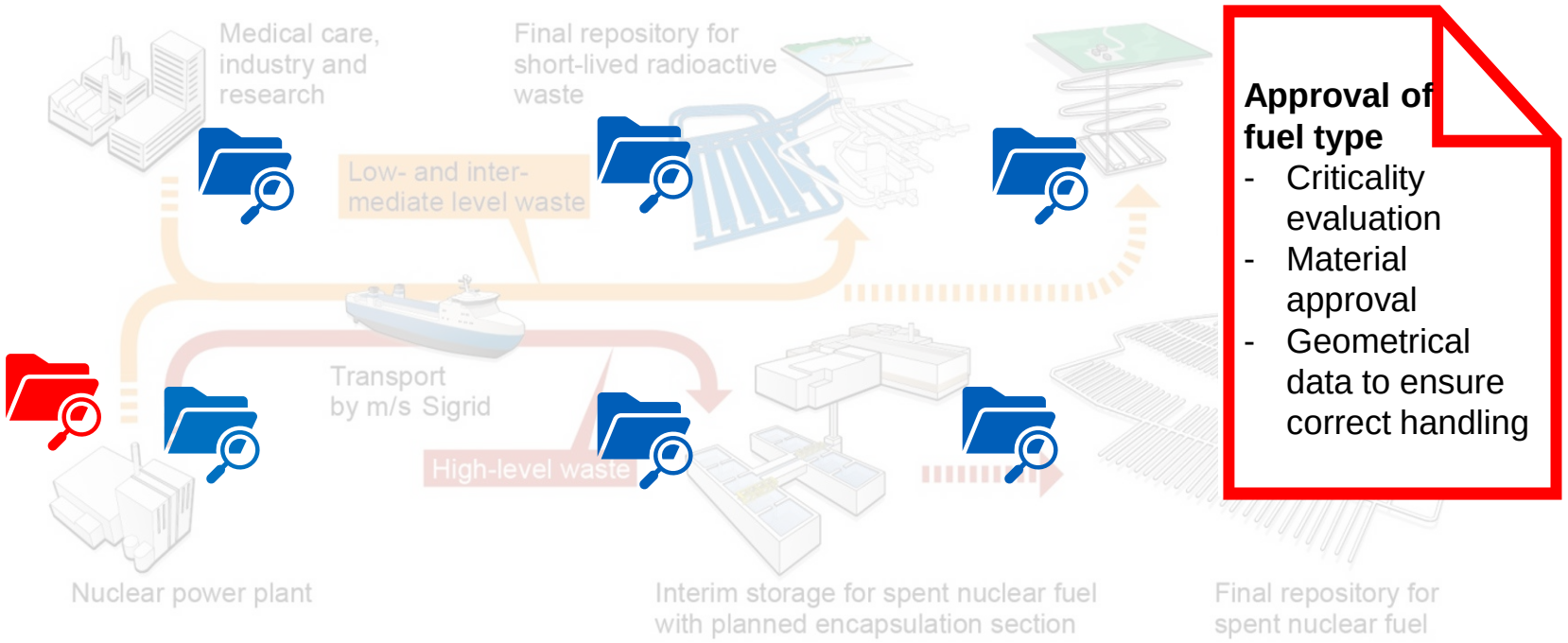
- SKB System and the flow of fuel information
- What information did we miss
- Project SABIS
- The challenges
- Conclusion



SKB's system

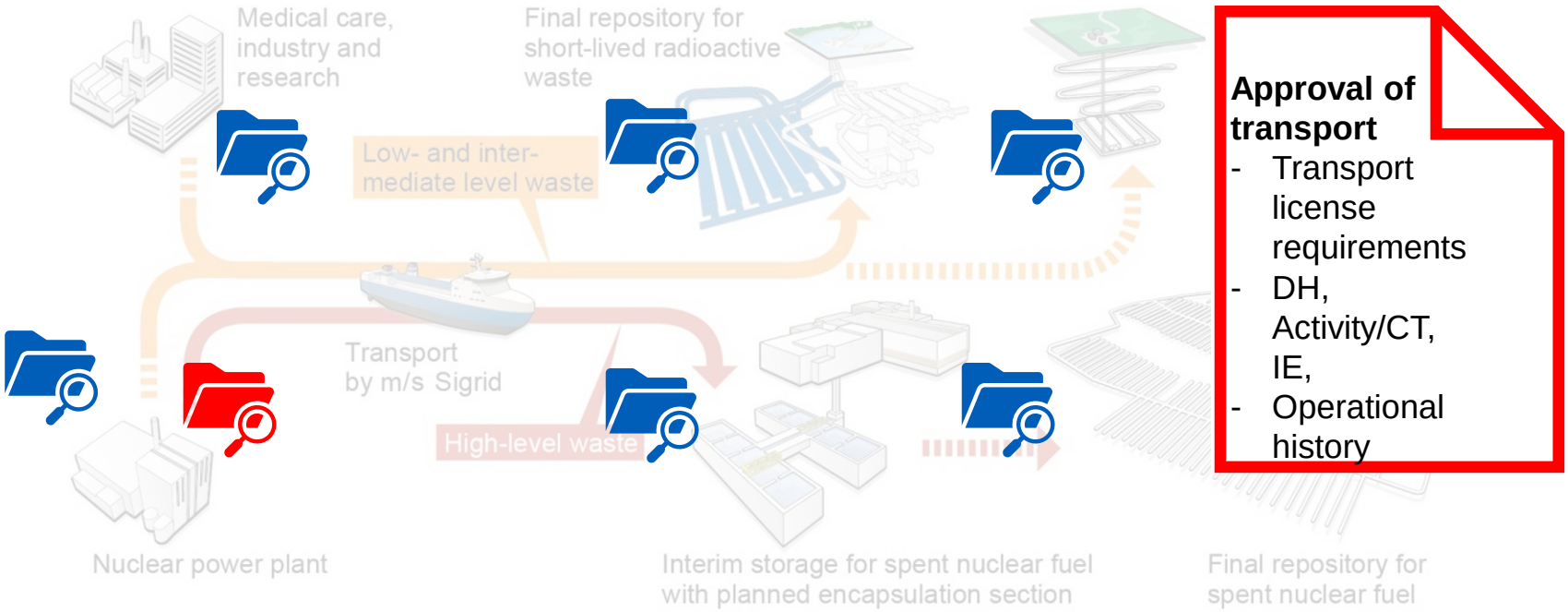

Medical care, industry and research
Final repository for short-lived radioactive waste
Final repository for long-lived LILW
Low- and inter-mediate level waste
Transport by m/s Sigrid
High-level waste
Nuclear power plant
Interim storage for spent nuclear fuel with planned encapsulation section
Final repository for spent nuclear fuel

The flow of fuel information (1)



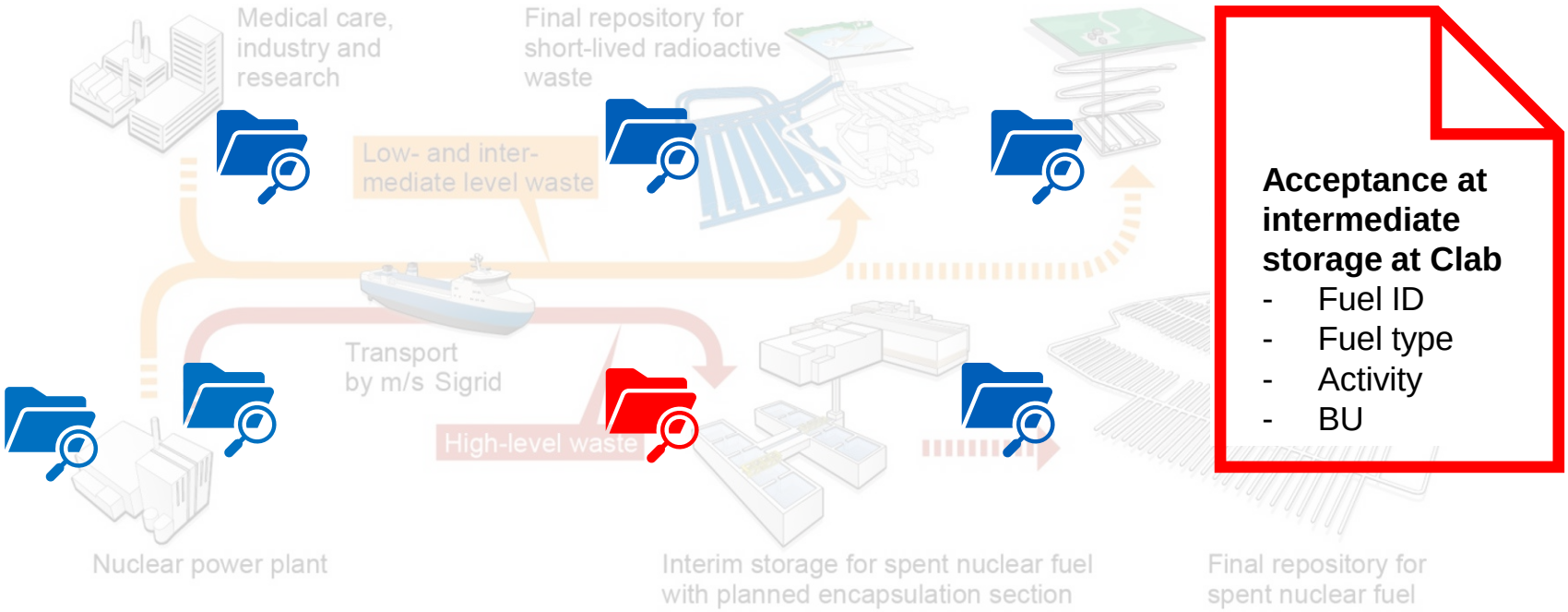
- Historically the requirements of the final storage was unknown
- The requirements on documentation not at todays standard
- The NPPs sometimes throw away information once the fuel was transported

The flow of fuel information (2)



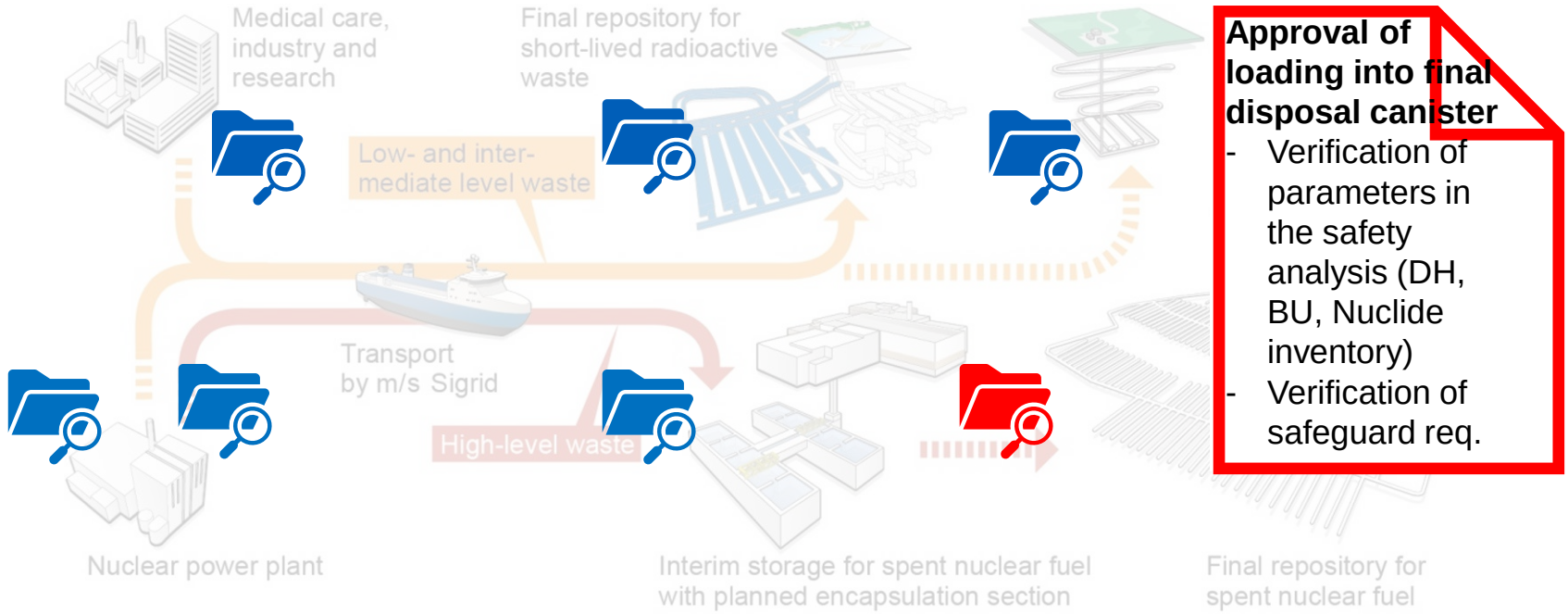
- Historically requirements on DH/Activity checked by requirements on CT, no operational data needed

The flow of information (3)



- Historically no BUC, no BA-credit

The flow of information (4)



Approval of loading into final disposal canister

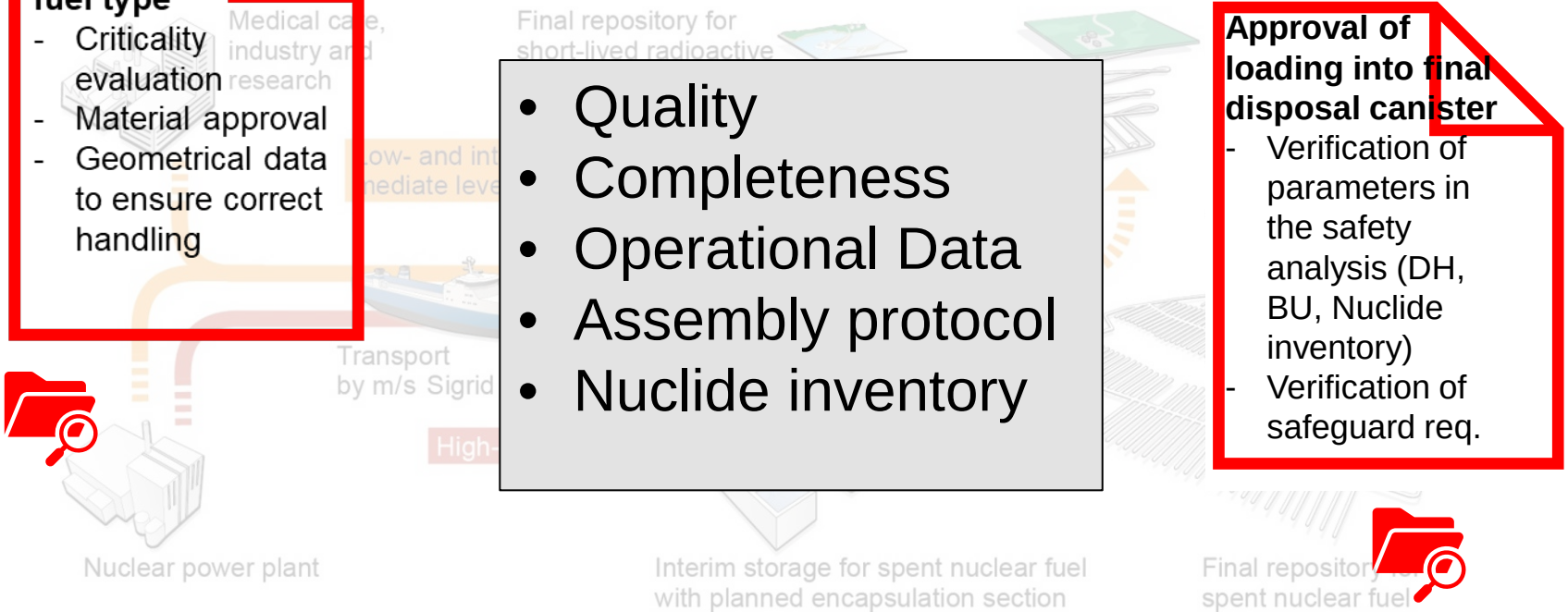
- Verification of parameters in the safety analysis (DH, BU, Nuclide inventory)
- Verification of safeguard req.

The missing parts

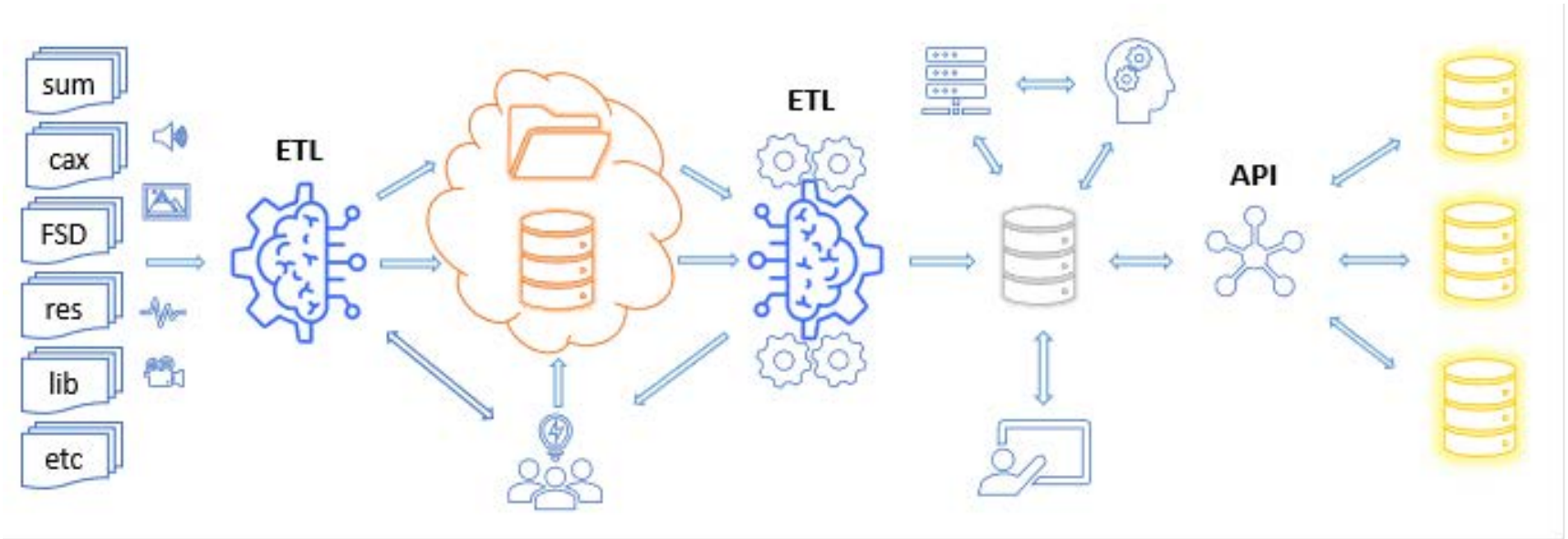
Approval of fuel type
- Criticality evaluation
- Material approval
- Geometrical data to ensure correct handling

- Quality
- Completeness
- Operational Data
- Assembly protocol
- Nuclide inventory

Approval of loading into final disposal canister
- Verification of parameters in the safety analysis (DH, BU, Nuclide inventory)
- Verification of safeguard req.



Project SABIS - Vision

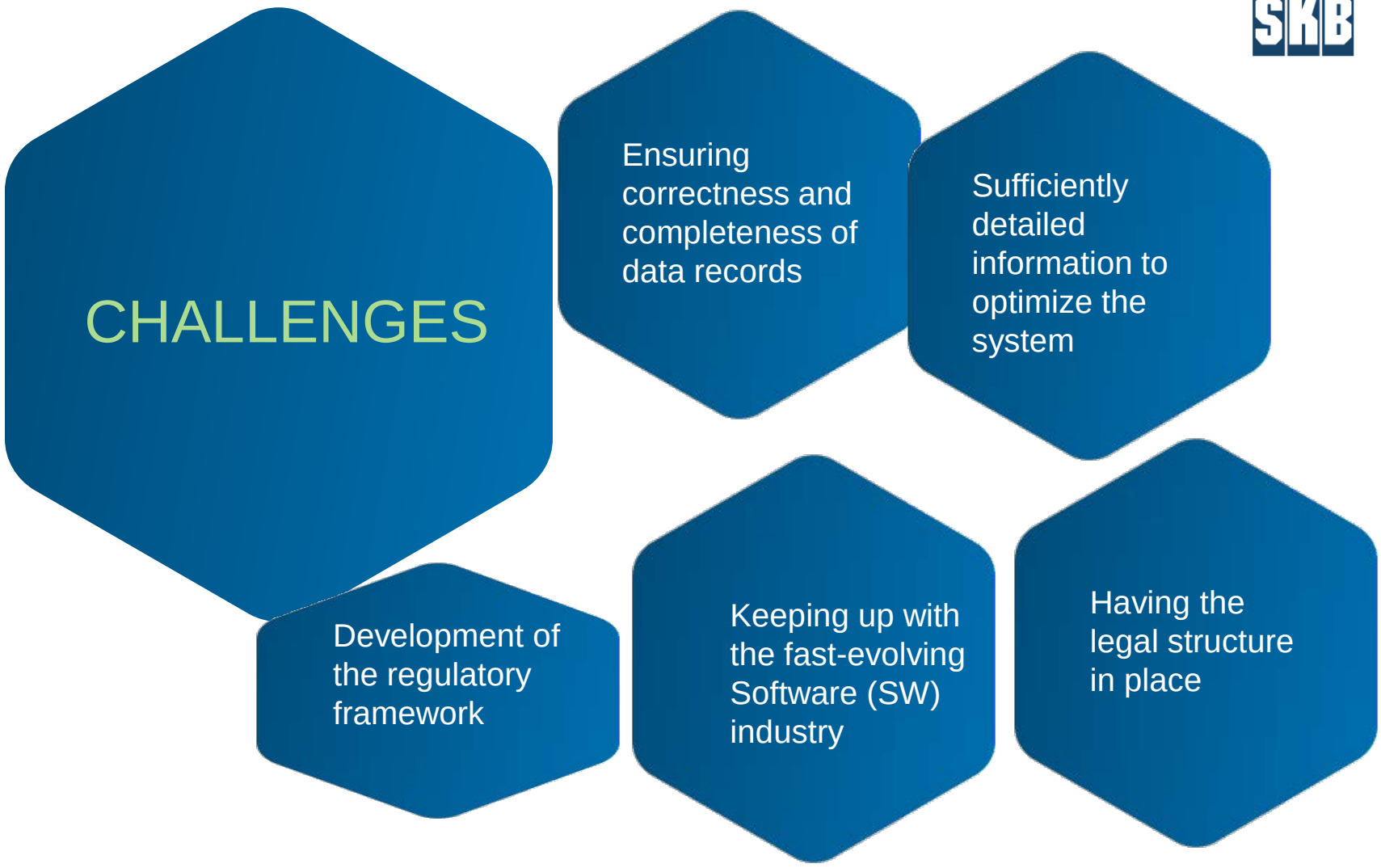


SABIS Demo

Dashboard
User: konhlil
DB: FuelDBdev

Anläggningar Clab

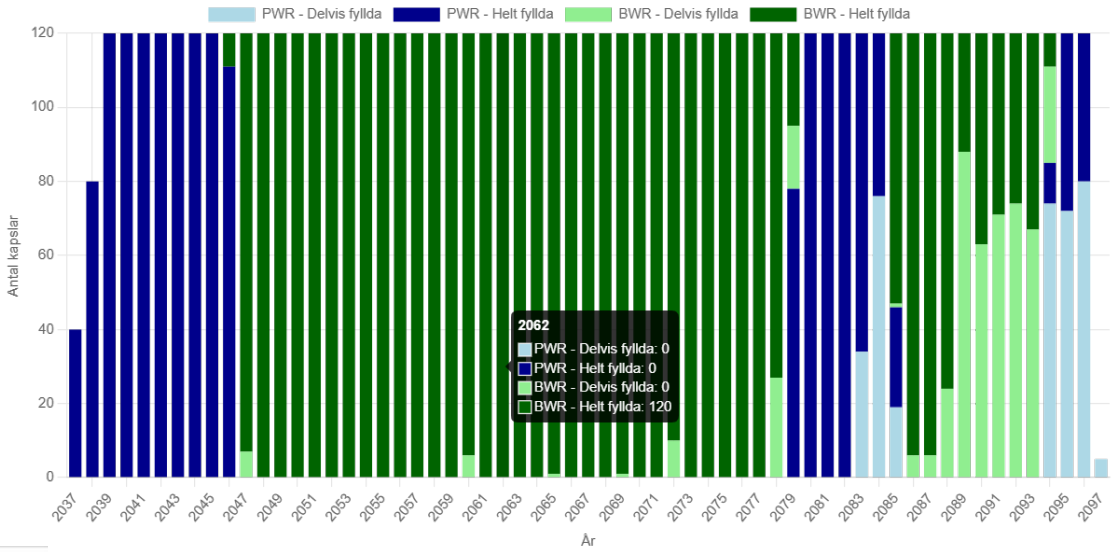
Reset

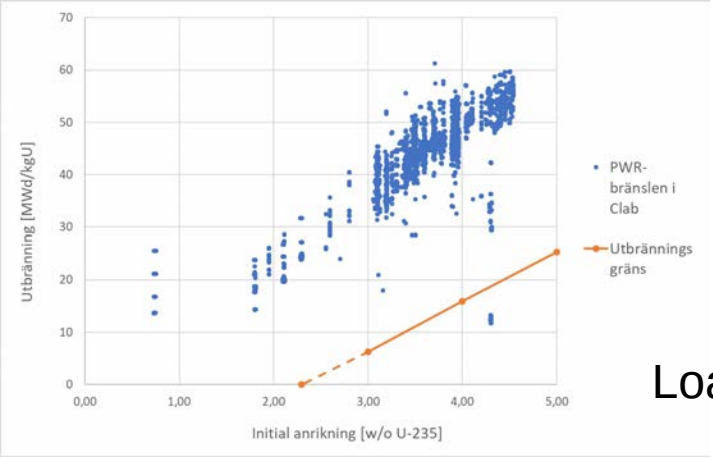

CHALLENGES
Ensuring correctness and completeness of data records
Sufficiently detailed information to optimize the system
Development of the regulatory framework
Keeping up with the fast-evolving Software (SW) industry
Having the legal structure in place

Challenges

Sufficiently detailed information to optimize the system

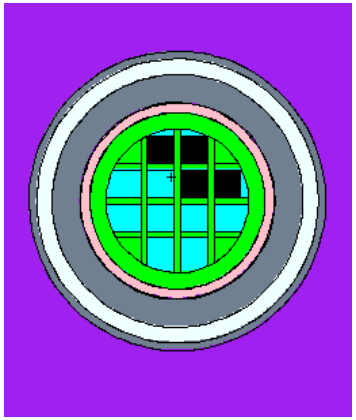
Canister optimization





Loading curve for criticality

Radiation protection



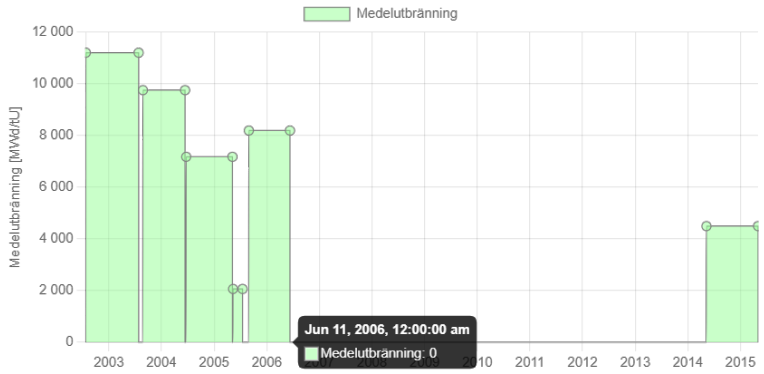
Challenges

Ensuring correctness and completeness of data records

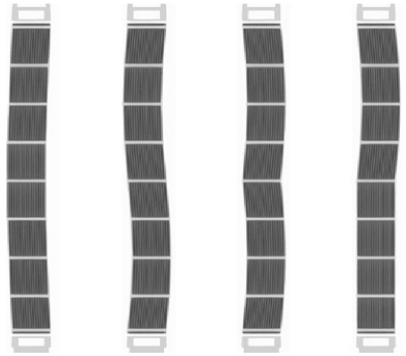
Missing rods

	A	B	C	D	E	F	G	H
8	16185	27886	27858	BLOCKERAD	15948	27869	27846	9560
7	9665	23268	16157	9534	16210	8621	8743	16286
6	16025	16124	16872	22522	15149	15189	18183	17026
5	BLOCKERAD		6229	6183	6220	15540	17091	16946
4	16787	12111	8666	22774	8628	9550	7603	BLOCKERAD
3	27882	9009	9019	8997	22602		27847	16661
2		7308	12921	22676	12242	12238	27877	27843
1	27876	7399	9011	10533	BLOCKERAD	17077	8977	9022

Operational history



Rod bow, damaged fuels, leakers



Recommendations from Eurad project

“It is recommended that the maximum amount of information from online core monitoring is used for SNF characterisation.”

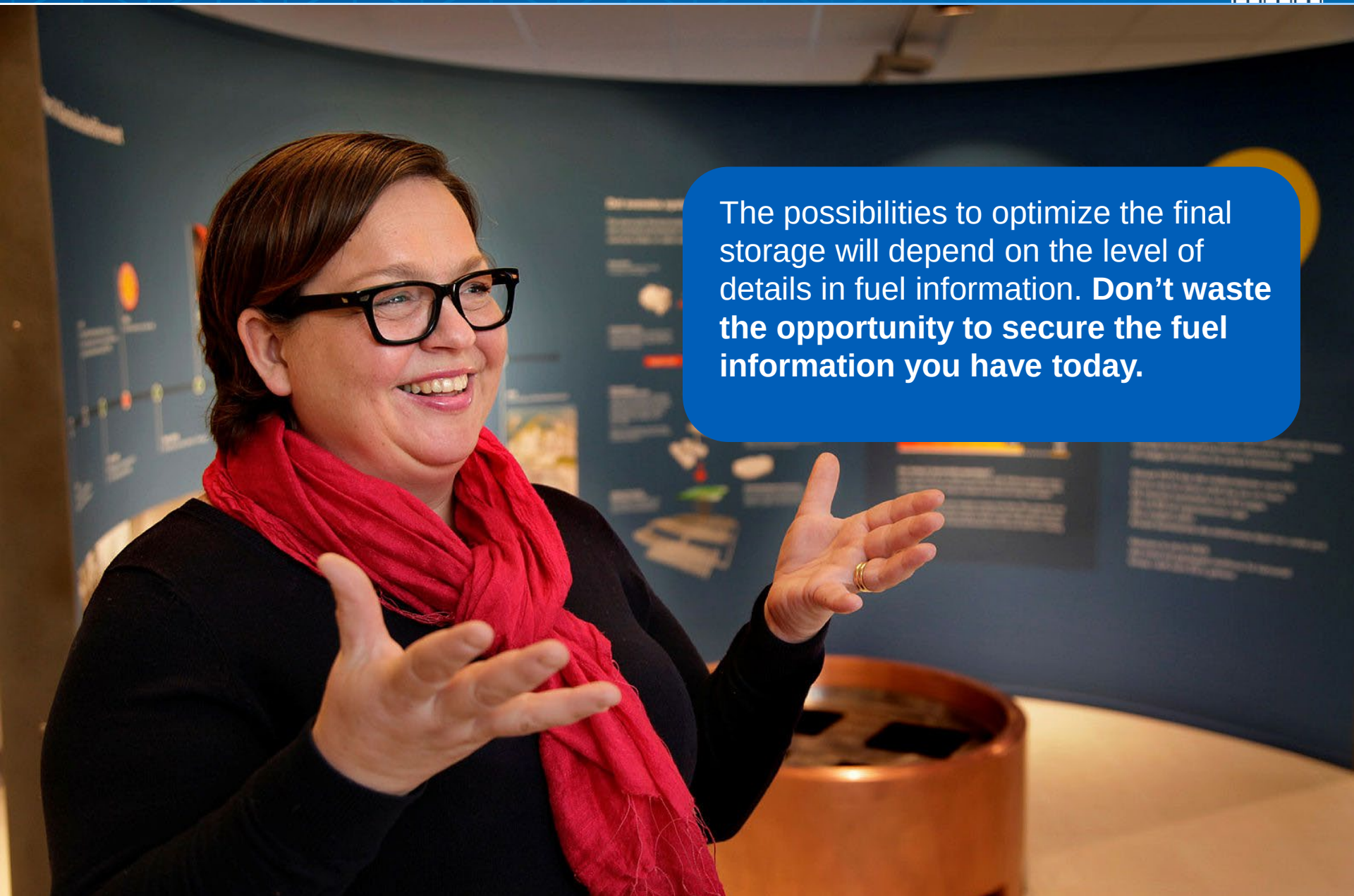
” Nonetheless, best estimate calculations are not always used in the industrial context due to data availability, time, effort constraints, and regulatory requirements. In this case, conservative approaches based on standard methods (e.g. ANSI or ISO standards) are used. It inherently leads to less efficient and under-optimized systems (e.g., cask loading), possibly artificially increasing various costs related to SNF storage and disposal. The advantage of best estimate calculations needs to be emphasised from an economical point of view, but also from a scientific aspect.”

Eurad Deliverable 8.7 Task 2

Summary

- Fuel Information is a valuable asset and should be treated with the same care as other valuable assets.
- Our industry is unique in in the sense that information needs to be kept intact and available for decades. This puts unique demands on the way we handle fuel information.
- The need for information originates from many different steps in the back-end chain, and the more detailed information you possess, the better you can optimise your spent fuel management system.
- In the end, the only thing the next generation will know about the final repository, is the information we have preserved for them.

Final words



The possibilities to optimize the final storage will depend on the level of details in fuel information. **Don’t waste the opportunity to secure the fuel information you have today.**

PSI Center for Nuclear Engineering and Sciences

DRYstars

Updates and Outlook

P. Konarski, B. Torrente

Extended Storage Collaboration Program Winter Meeting,
27.10.2025

Agenda

1 Introduction

2 Journal paper on Wet-To-Dry-To-Wet (W2D2W)

3 Journal paper on Falcon developments

4 DRYstars3

Used nuclear fuel in Switzerland

Swiss Used Fuel Policy

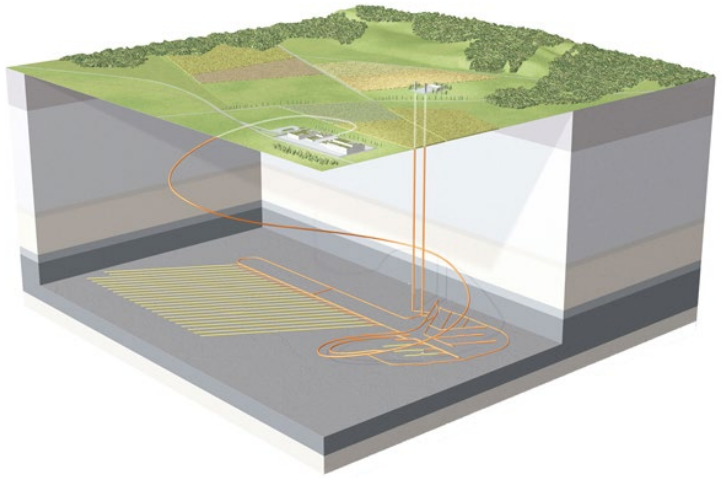
Interim storage:

- Wet storage at reactor sites
- Centralized dry storage facility

Final disposal:

- To be disposed in a geological repository

Credit: ZWILAG



Credit: NAGRA

Issues and Challenges:

Final disposal:

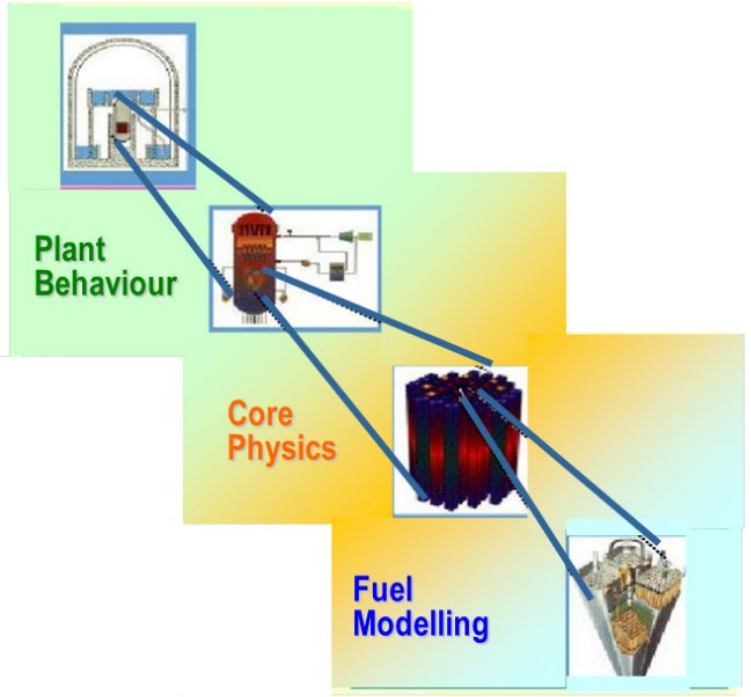
- Operating dates unknown (around 2050)

Interim storage:

- Limited capacities of used fuel pools
- Used fuel behavior in extended dry storage: Safety assessment supported with numerical computations

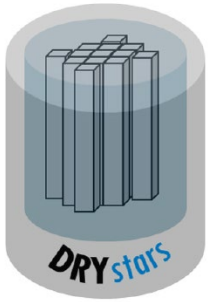
Modelling capabilities at PSI

- **STARS** (Steady-state and Transient Analysis Research for the Swiss reactors), Swiss multiphysics platform:
  - Plant Behavior
  - Core Physics
  - Fuel Modelling
- From “Turbine to Pellet” Integral LWR Safety Analyses
- Capabilities limited to in-reactor behavior
- **DRYstars** (DRY Storage Analyses for the Reactors in Switzerland) launched in 2019



Topical review

- Phenomenology and Safety criteria
- Experiments and Codes
- Swiss Specific Issues



Model development

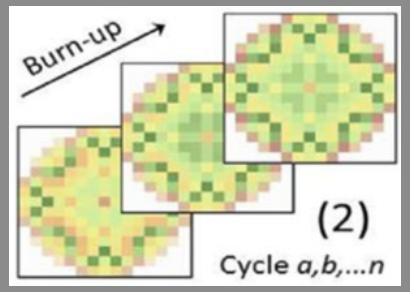
- Pellet Behavior
- Cladding Behavior
- Hydrogen Behavior

Multiphysics Modelling Capabilities

- Multiphysics couplings
- Swiss Fuel Analyses
- Sensitivity Studies

Modelling capabilities at PSI

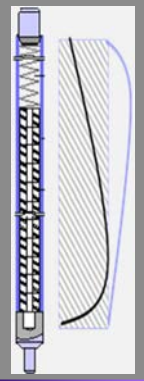
Reactor Core Simulations



CASMO | SIMULATE

- Validated Core Models
- Real Irradiation History
- Core Design and Loading Patterns
- Assembly Design

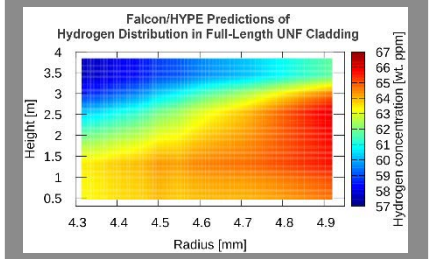
Fuel Behavior Simulations



Falcon

- Burnup
- Rod Internal Pressure
- Fission Gas Release
- Oxide Thickness
- Stresses

Hydrogen Behavior Simulations



HYPE

- Hydrogen Uptake
- Radial Diffusion
- Thermochemistry
- Embrittlement
- Delayed Hydride Cracking

Journal paper on Wet-To-Dry-To-Wet (W2D2W)

PSI

Content of the paper

- Defining W2D2W
- Choosing limiting rods for analyses
- Sensitivity study on the duration of W2D2W
- Studying the impact of W2D2W on thermo-mechanical and hydrogen behavior in dry storage

W2D2W at Gösgen Nuclear Power Plant

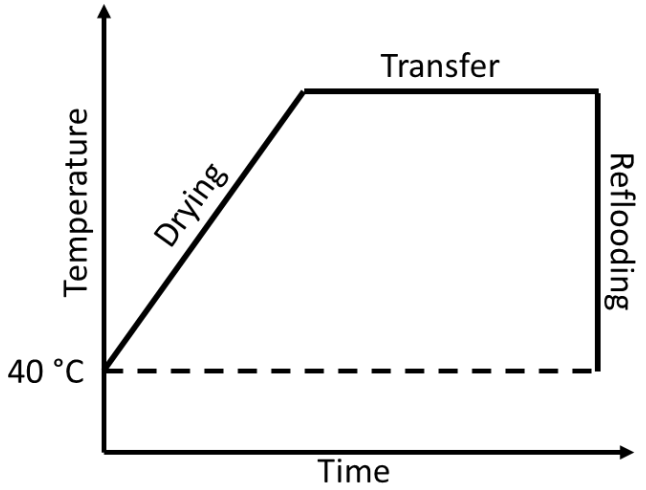
- Very high burnup / MOX
- Two wet storage pools at Gösgen NPP:
  - One inside the reactor building
  - One outside the reactor building
- Transfer of used fuel between the pools using a dry cask known as Wet-To-Dry-To-Wet (**W2D2W**)

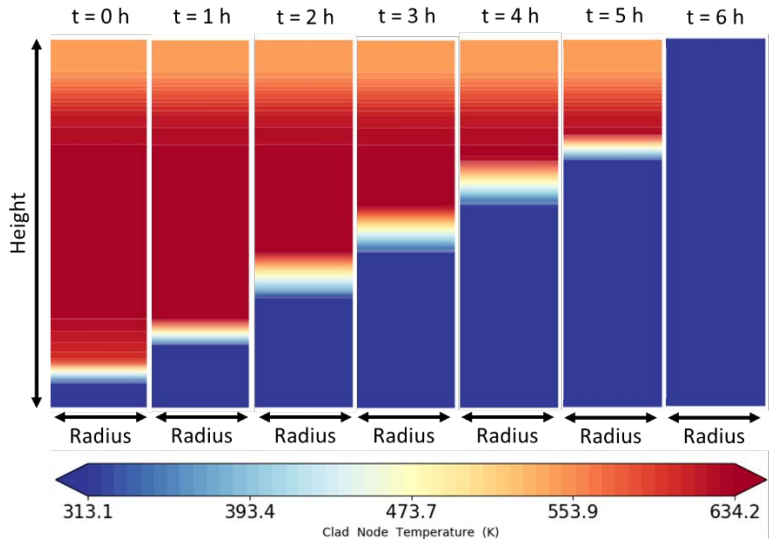
Credit: KKG

Journal paper on Wet-To-Dry-To-Wet (W2D2W)

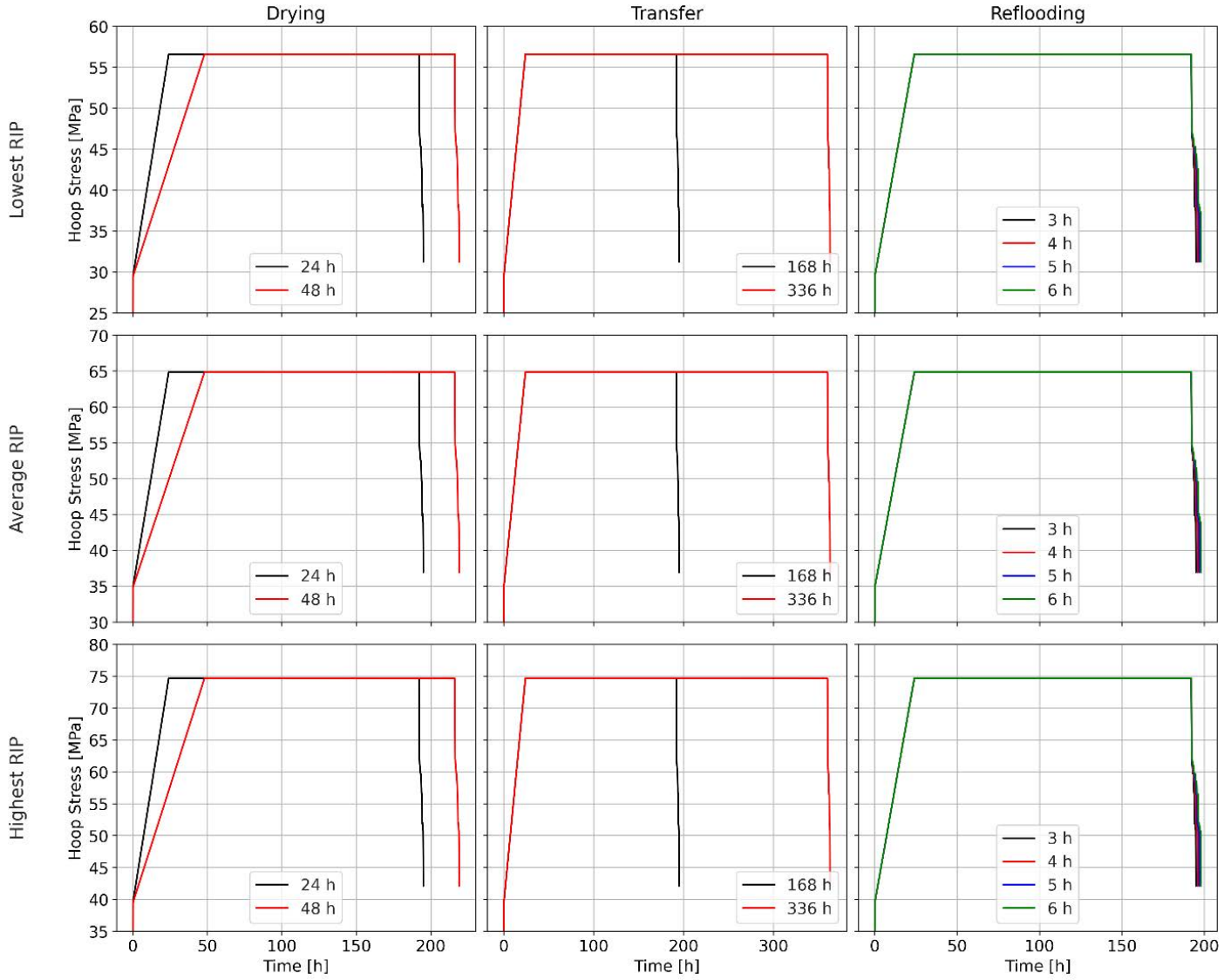
- Impact of W2D2W on dry storage studied with:
  - Falcon: Thermo-mechanics
  - HYPE: Hydrogen behavior
- W2D2W consists of three phases: drying, transfer, reflooding
- Three rods with different Rod Internal Pressure (RIP) at the end of life chosen from a group of the 80 most limiting rods tested
- Liner cladding DXD4 used in the studied rods
- Different durations of drying, transfer and reflooding tested

- Tested rods (3 cases):
  - Lowest RIP (57 GWd/tU)
  - Average RIP (63 GWd/tU)
  - Highest RIP (73 GWd/tU)
- Tested combinations (16 cases):
  - Drying: 12, 24 h
  - Transfer: 168, 336 h
  - Reflooding: 3, 4, 5, 6 h



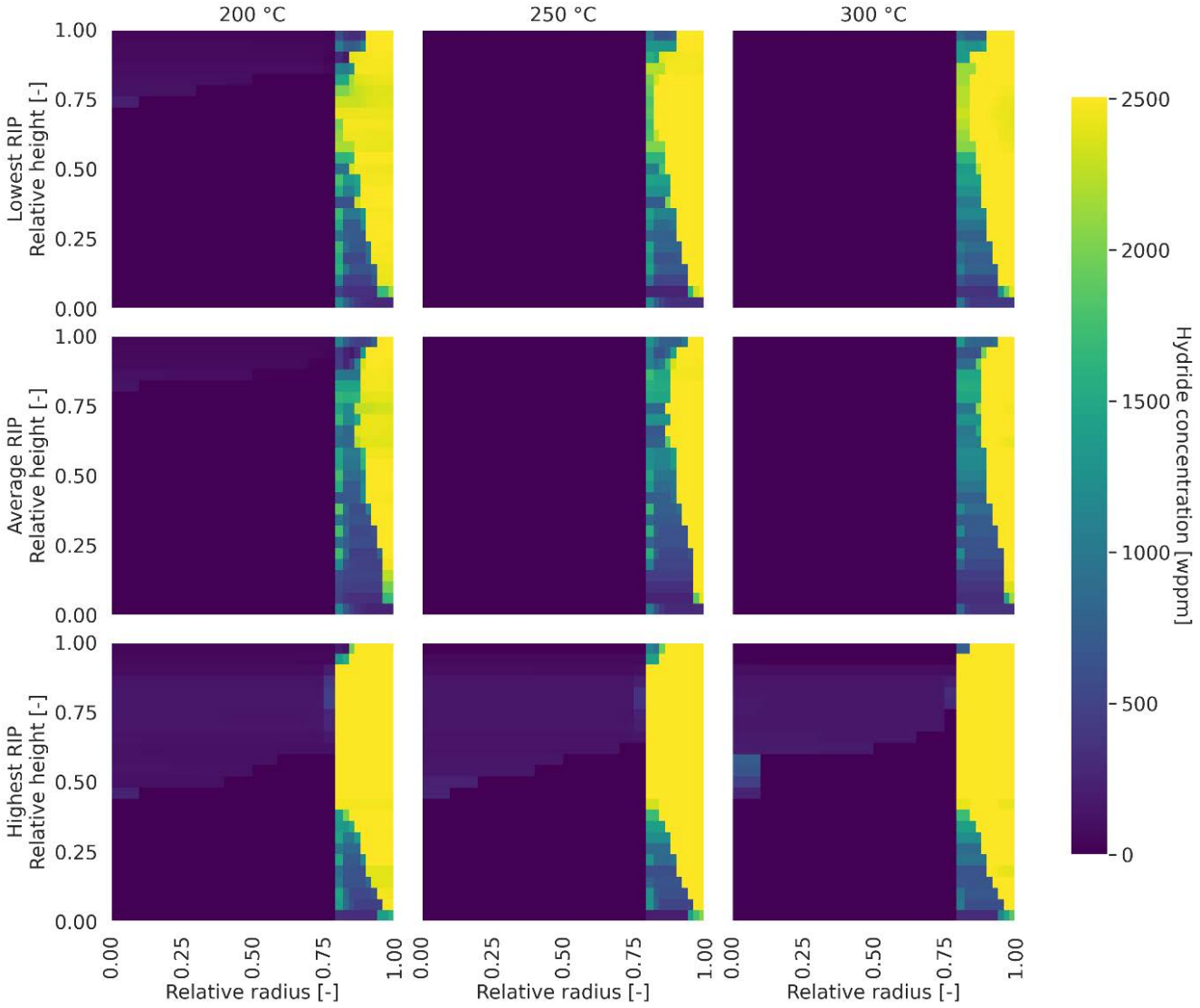


Journal paper on Wet-To-Dry-To-Wet (W2D2W)



- No impact of ‘Drying’ on the maximum stress
- No impact of ‘Transfer’ on maximum stress
- Faster stress decrease with shorter ‘Reflooding’
- Maximum stress correlated with RIP
- No significant creep

Journal paper on Wet-To-Dry-To-Wet (W2D2W)

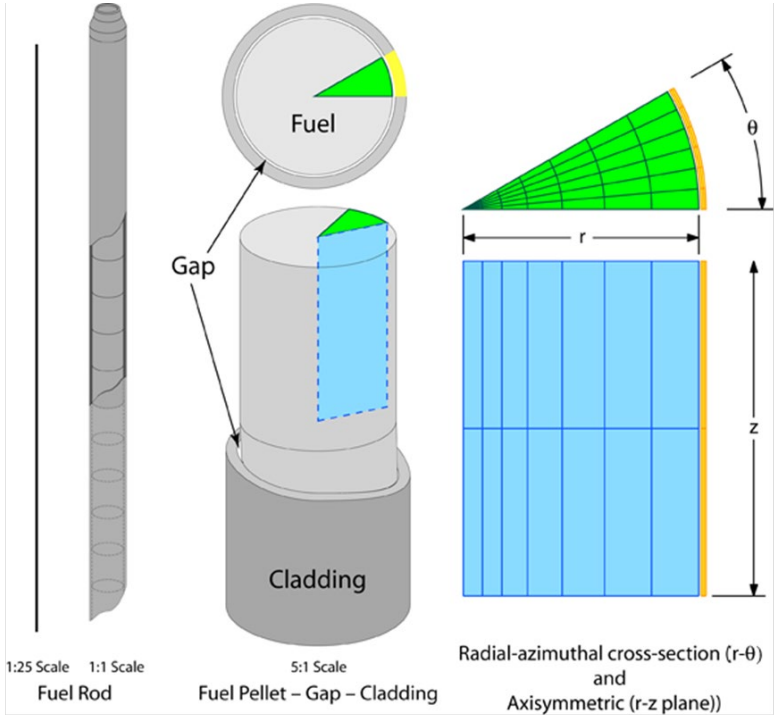


- Hydrides stay in the substrate in the top part of each rod at the maximum storage temperature 200 °C
- If the liner is saturated with hydrides, hydrogen migrates toward the inner part of the cladding driven by hydrostatic stress gradient
- More significant diffusion observed for hotter cases

Journal paper on Falcon developments

Falcon: Fuel Analysis and Licensing COde – New

- 2D single fuel rod code developed by EPRI
- PSI is part of the development team – new models and materials



Progress in Nuclear Energy 187 (2025) 105859

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Progress in Nuclear Energy

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

Implementation of dry storage models in Falcon: cladding creep, annealing of cladding irradiation damage, and fuel helium-swelling

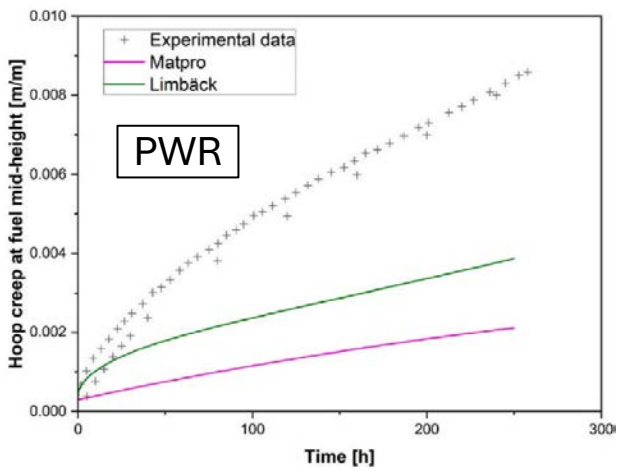
C. Cozzo [a,*], P. Konarski [a], G. Khvostov [a], B. Mervin [b], H. Ferroukhi [a]

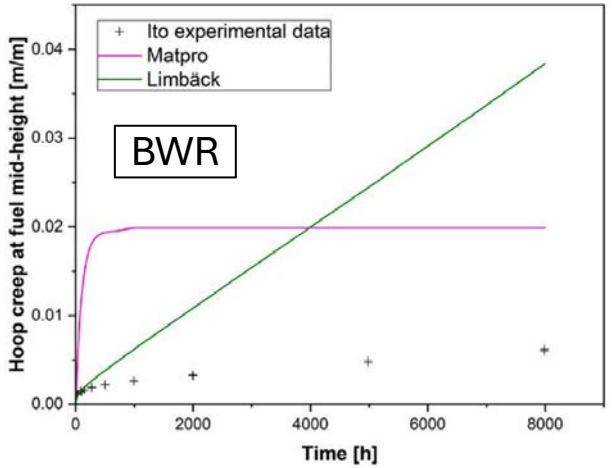
[a] PSI Center for Nuclear Engineering and Sciences, Villigen PSI, 5232, Switzerland
[b] Electric Power Research Institute, 1300 West W.T. Harris Boulevard, Charlotte, NC, 28262, USA

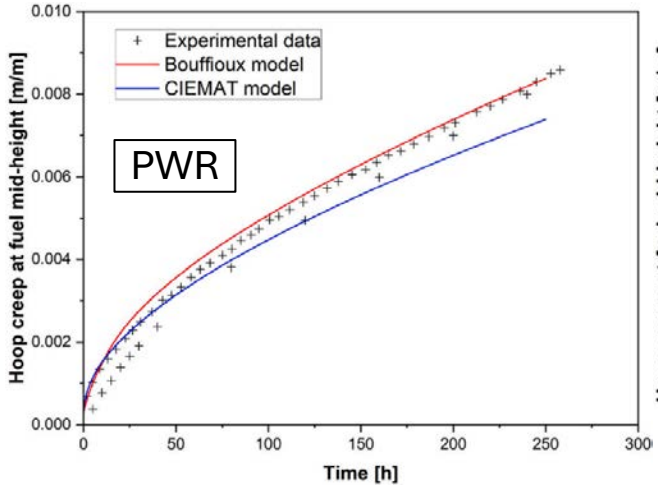
- Developing Falcon towards dry storage modelling was one of DRYstars priorities since 2019
- The following models were implemented:
  - Cladding creep
    - Bouffioux (PWR)
    - CIEMAT (PWR)
    - Ito (BWR)
  - Cladding annealing (Rashid's model)
  - Helium swelling
    - Best Estimate
    - Upper Limit

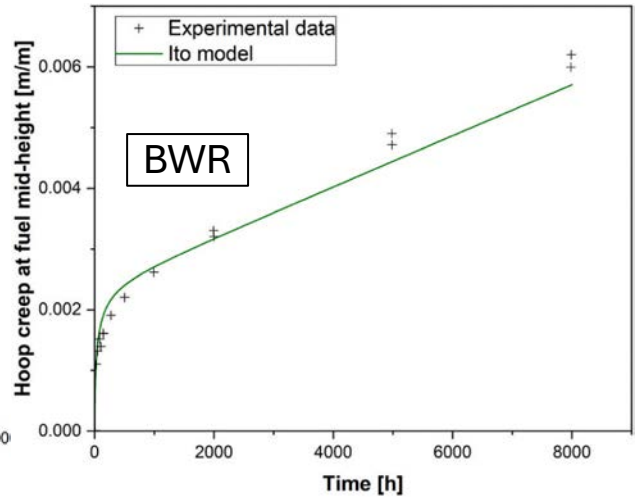
Journal paper on Falcon developments

- The need for new creep models is demonstrated by comparing Falcon's default models with experiments representative of dry storage conditions for both PWR and BWR claddings
- Models dedicated to dry storage predict cladding creep with much greater accuracy



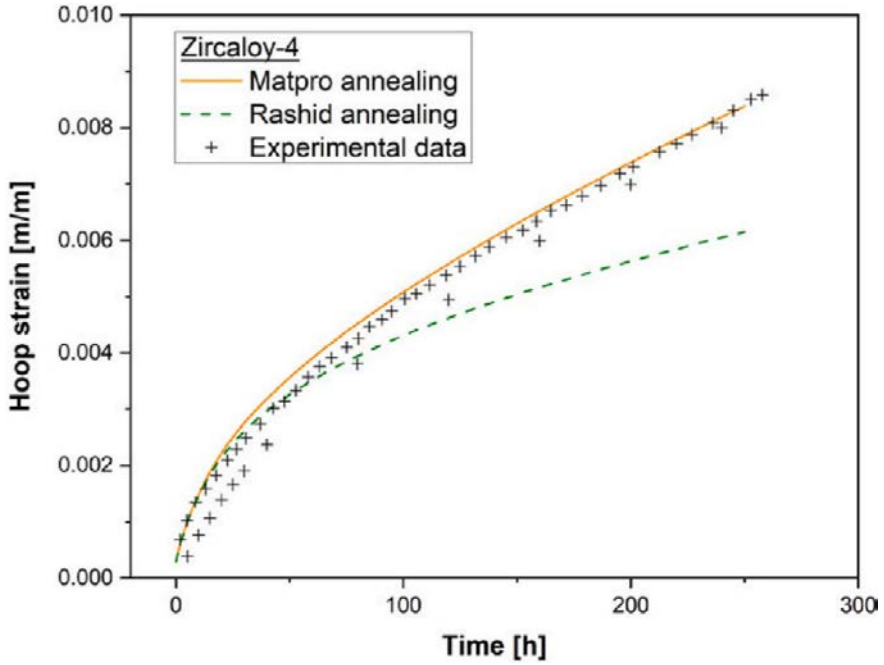




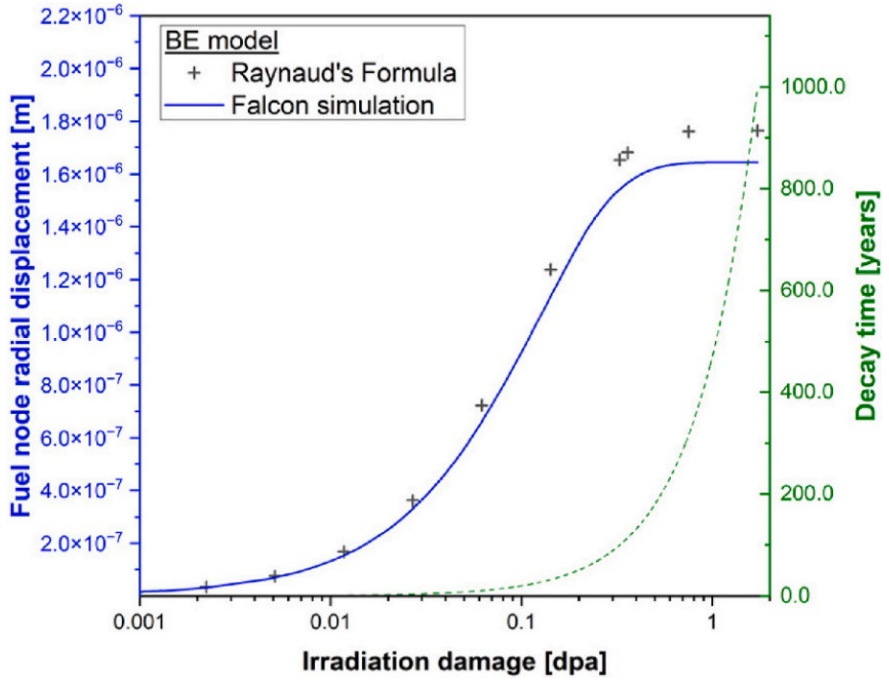


Journal paper on Falcon developments

- The impact of Rashid's annealing model verified indirectly by simulating creep experiments

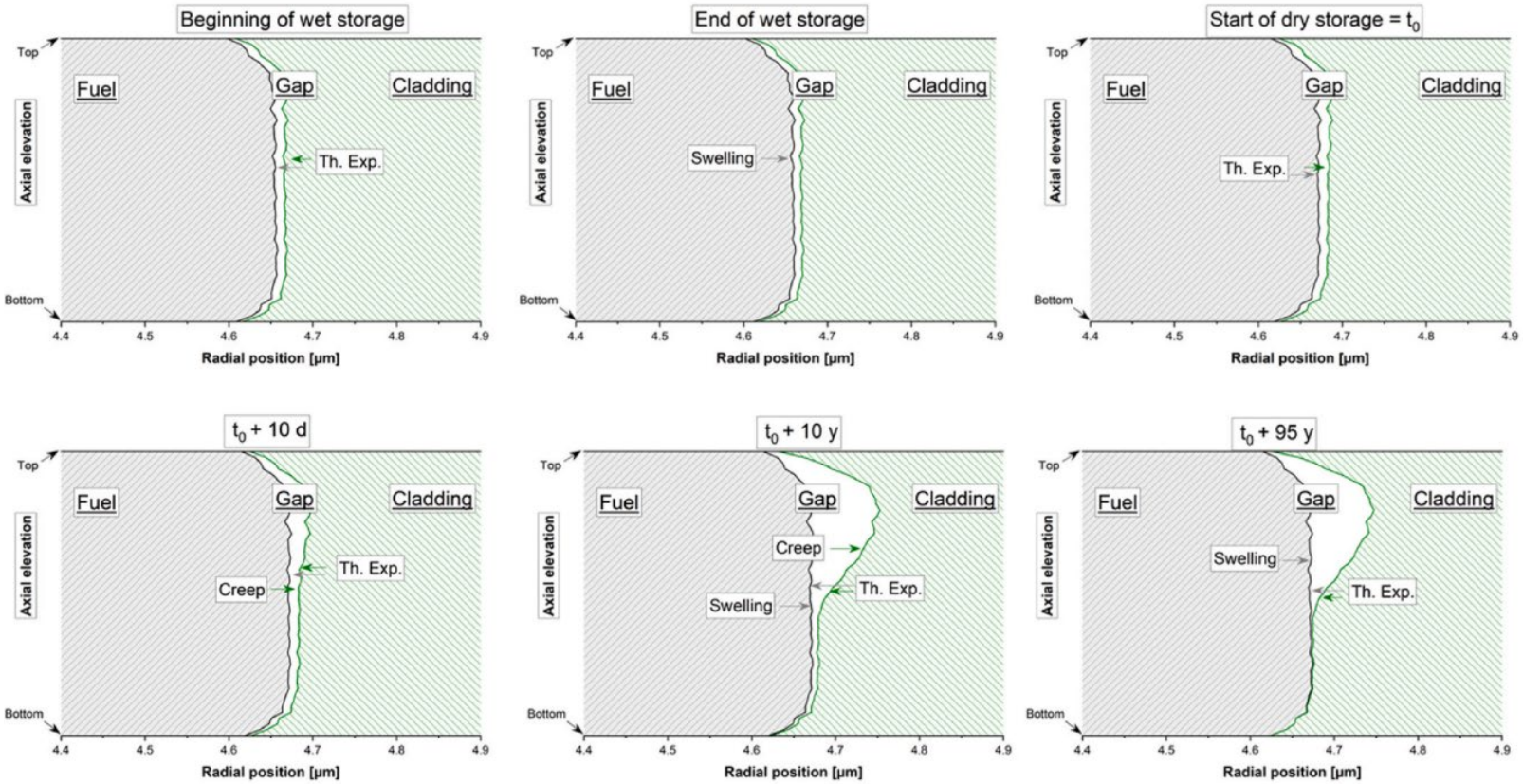


- The effect of Helium swelling model compared to the analytical solution



Journal paper on Falcon developments

- The effect of Falcon developments on dry storage studied by simulating PWR and BWR cases
- Phenomenology leading to the fuel-cladding gap axial distribution observed in dry storage is shown below

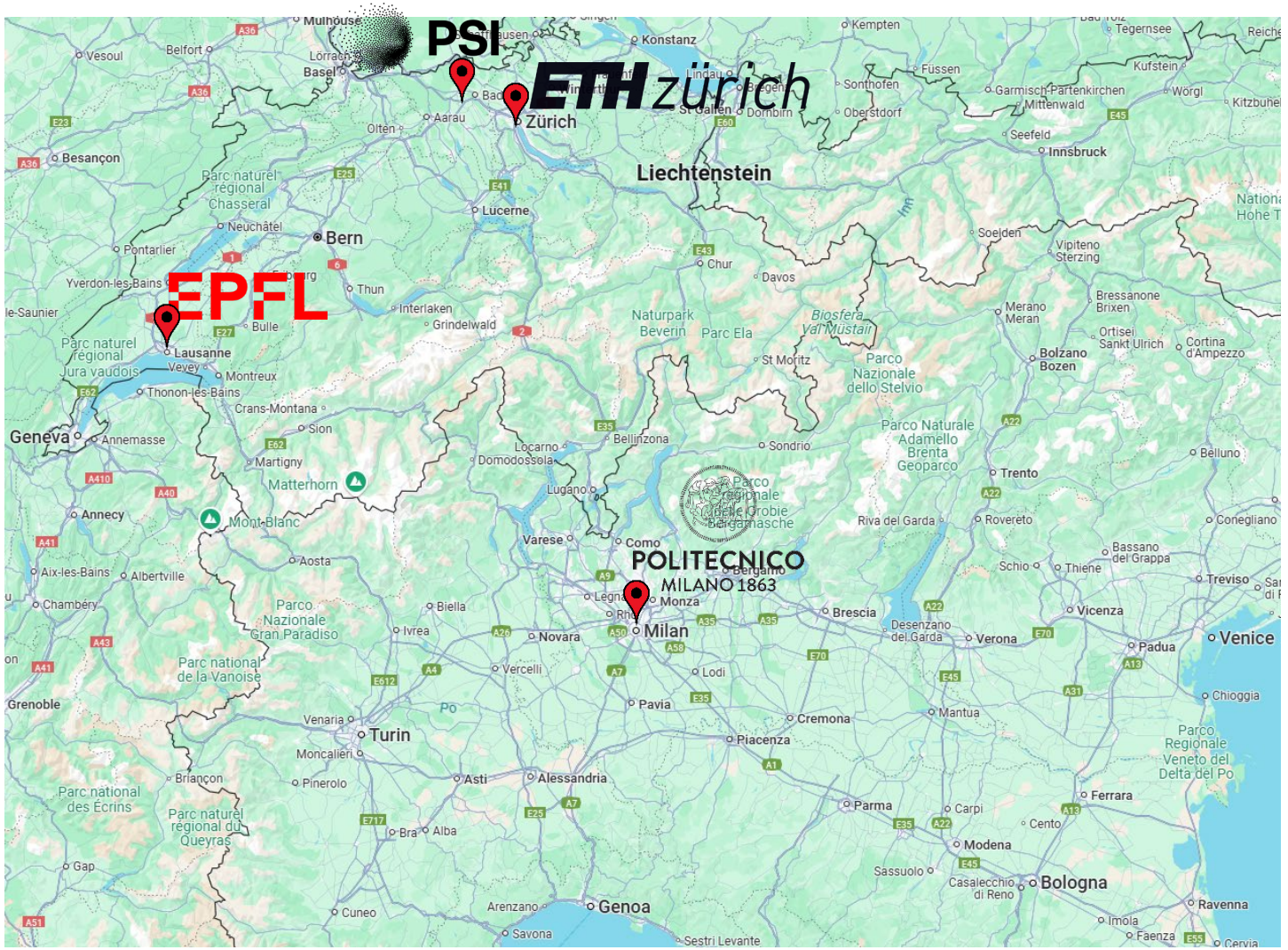


DRYstars 3 - About me

BSc: Materials Engineering and Nanotechnology at PoliMi

MSc: Nuclear Engineering at ETHZ-EPFL

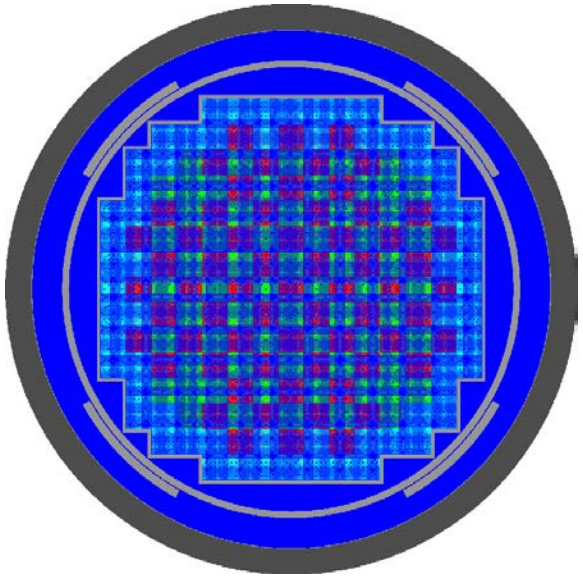
PhD: Nuclear Engineering at PSI-EPFL



DRYstars 3 – Goals

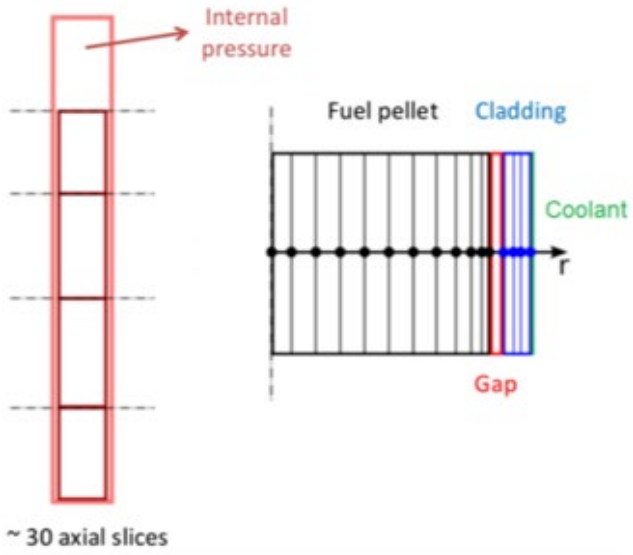
Develop an advanced methodology to perform **core-wide, pin-by-pin** calculations of all fuel rods, and to apply it as a basis for downstream dry storage analyses.

Reactor Operation



Watts Bar (WBN1)

Fuel Performance



Dry Storage

Zwilag

DRYstars 3 – Methodology

PSI

Direct coupling

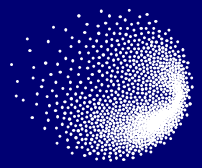
- **Integration** of the fuel performance code with the core simulator(CMSYS)
- Online or offline coupling
- Detailed physics, slower but **high-fidelity**

Reduced Order Models

- **Surrogate approach** based on high-fidelity simulations
- Large set of training data
- Near **instantaneous** solutions

Summary

- Journal paper on W2D2W was published
  - Transfer of used fuel from one pool to another studied
  - Thermo-mechanical and hydrogen behavior simulations of reactor operation, W2D2W and dry storage were performed
- Journal paper on Falcon developments was published
  - Models for cladding creep, annealing and helium swelling were implemented
  - V&V cases were proposed
  - BWR and PWR cases simulated to demonstrate the effect of developments
- DRYstars3
  - core-wide, pin-by-pin calculations of all fuel rod
  - Direct coupling and Reduced order models



PSI Center for Nuclear Engineering and Sciences

Thank you for your attention!

Evaluating the 570C limit for high Burnup

Daniel Jädernäs, Clara Anghel, VNF
Johan Ljungberg, SKB

2025-10-13

Background

- Related to safety analysis in Sweden VNF and SKB have looked into the off-normal condition in ISG-11 rev.3
- It states that fuel can be handled with peak cladding T=570 °C for 30 days.
- To us it was unclear how high burnup fuel behaves under these conditions

Max T_{clad}<400°C during normal handling
Low BU fuel OK with higher T is showing σ_{hoop}<90 Mpa
Off-normal events
$T_{kapsling}$<570°C (30 days)

Spent Fuel Project Office
Interim Staff Guidance - 11, Revision 3

Issue: **Cladding Considerations for the Transportation and Storage of Spent Fuel**

In order to assure integrity of the cladding material, the following criteria should be met:

1. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad).

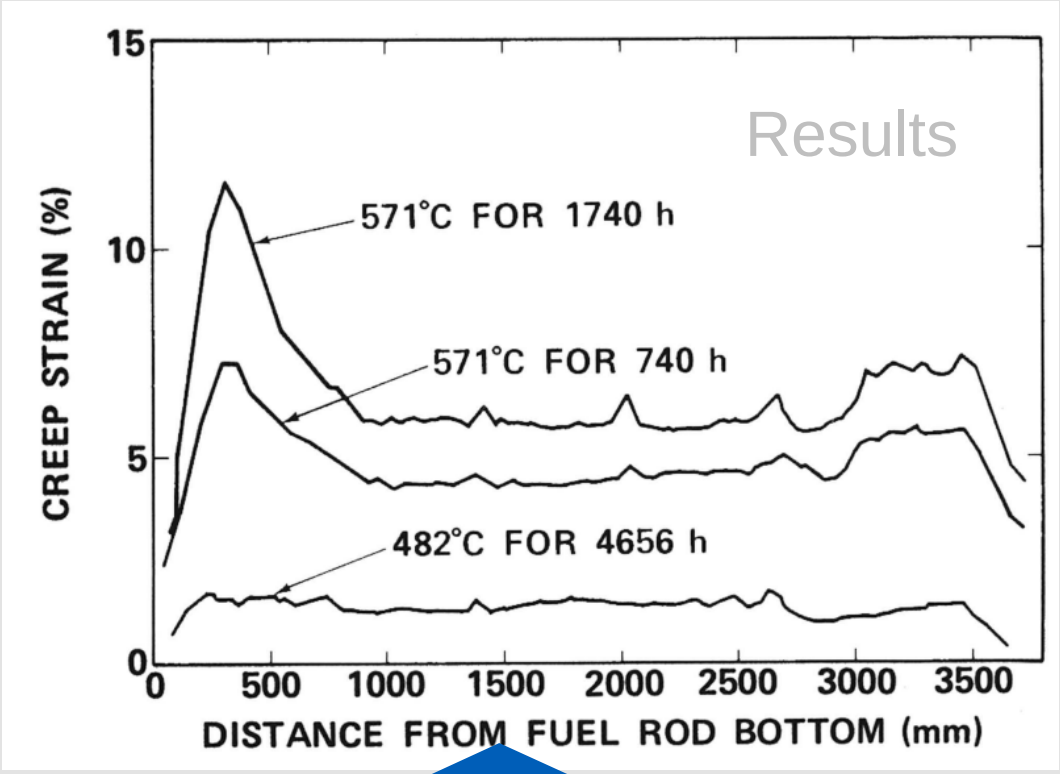
   However, for low burnup fuel, a higher short-term temperature limit may be used, if the applicant can show by calculation that the best estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed.

2. During loading operations, repeated thermal cycling (repeated heatup/cooldown cycles) may occur but should be limited to less than 10 cycles, with cladding temperature variations that are less than 65°C (117°F) each.

3. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

ISG-11 rev. 3 - background

- Guidance in ISG-11 is based on the following data
  - Einziger etal. 1982 (High Temperature Postirradiation Materials Performance of Spent Pressurized Water Reactor Fuel Rods under Dry Storage Conditions)
- Tests on PWR fuel with BU<31 MWd/kgU
  - CW SRA Zry-4, diameter: 10.72 mm, t: 0.62 mm
  - ~100 w-ppm H, 10-20 µm oxide
  - RIP up to 3.5 MPa (35 bar)
  - Inert or limited air atmosphere
- Results: No through-wall cladding damage at 571°C for 72.5 days. However, up to 12% cladding hoop strain.



High creep strain but no clad failure

Differences

ISG-11 rev. 3		Our scenario
Low BU <35 MWd/kgU	→	High BU>60 MWd/kgU
Inert or limited air	→	In air

Differences

- Higher EOL RIP
- Cladding oxidation at high T
- Possibly higher consequences if clad fails

Reality check

How many rods are at risk for rupture at a maximum T of 570 C?

Real case

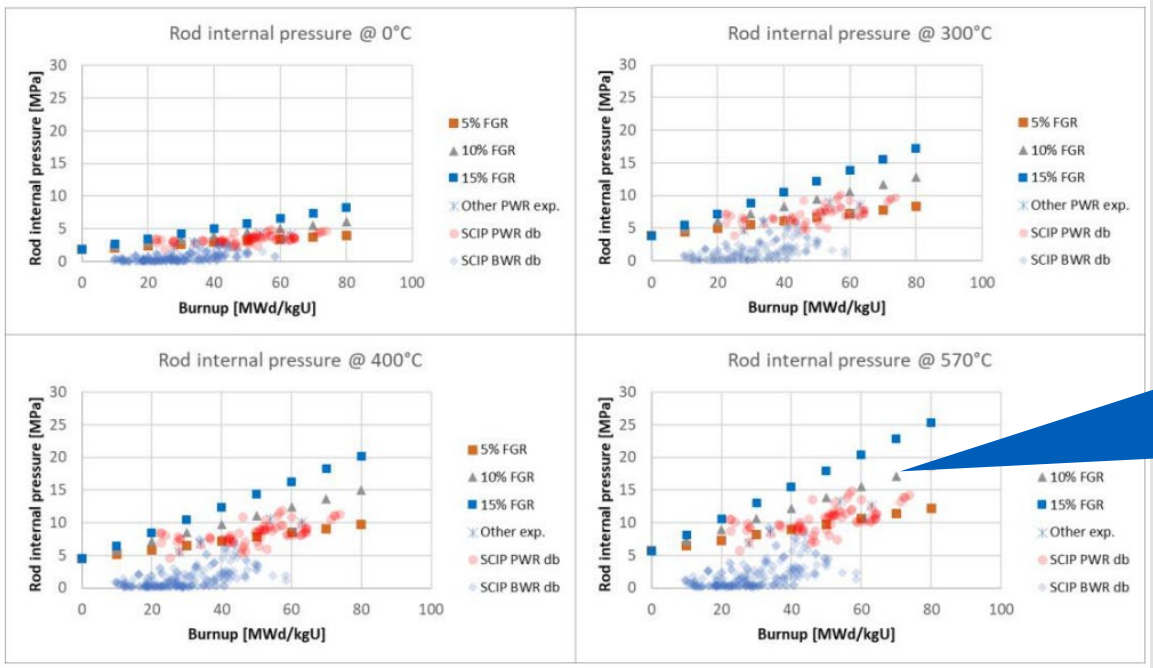
- Temperature distribution inside transport cask
- Temperature calculation conservatism
- FRP variation
- Burnup variation, mean foreseen BU 55 MWd/kgU
- Air convection

Evaluated case

- Maximum T
- Highest FRP
- Highest Burnup, 60 MWd/kgU
- Full air access to cask, no circulation of air

EOL RIP and clad hoop stress at 570°C

- PWR rods assumed to be limiting case – fill pressure ~20 bar@20°C



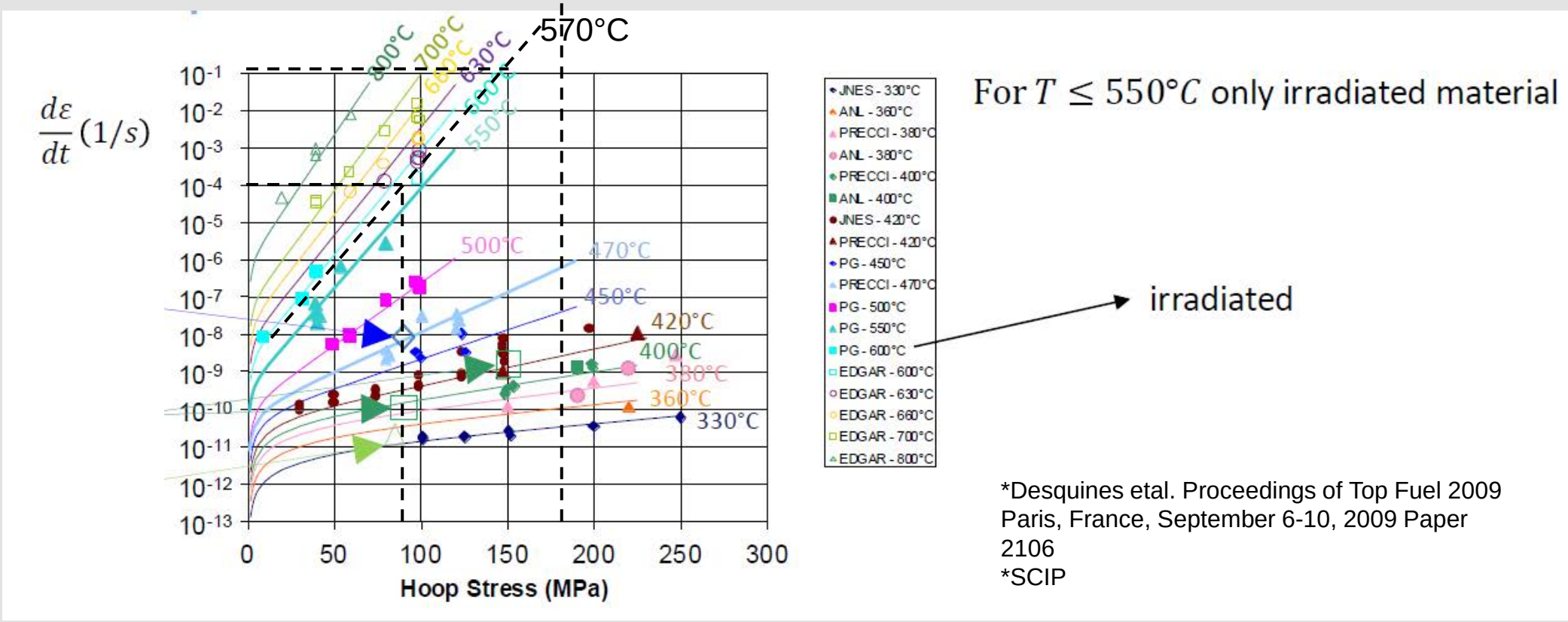
EOL RIP @ 570°C and 70 MWd/kgU is 110-230 bar
Results in high clad hoop stresses
85-180 MPa @ 570°C

$$\sigma = p*D/2t$$

*Vattenfall Nuclear Fuel AB
*SCIP

Corresponding creep rates at 570°C

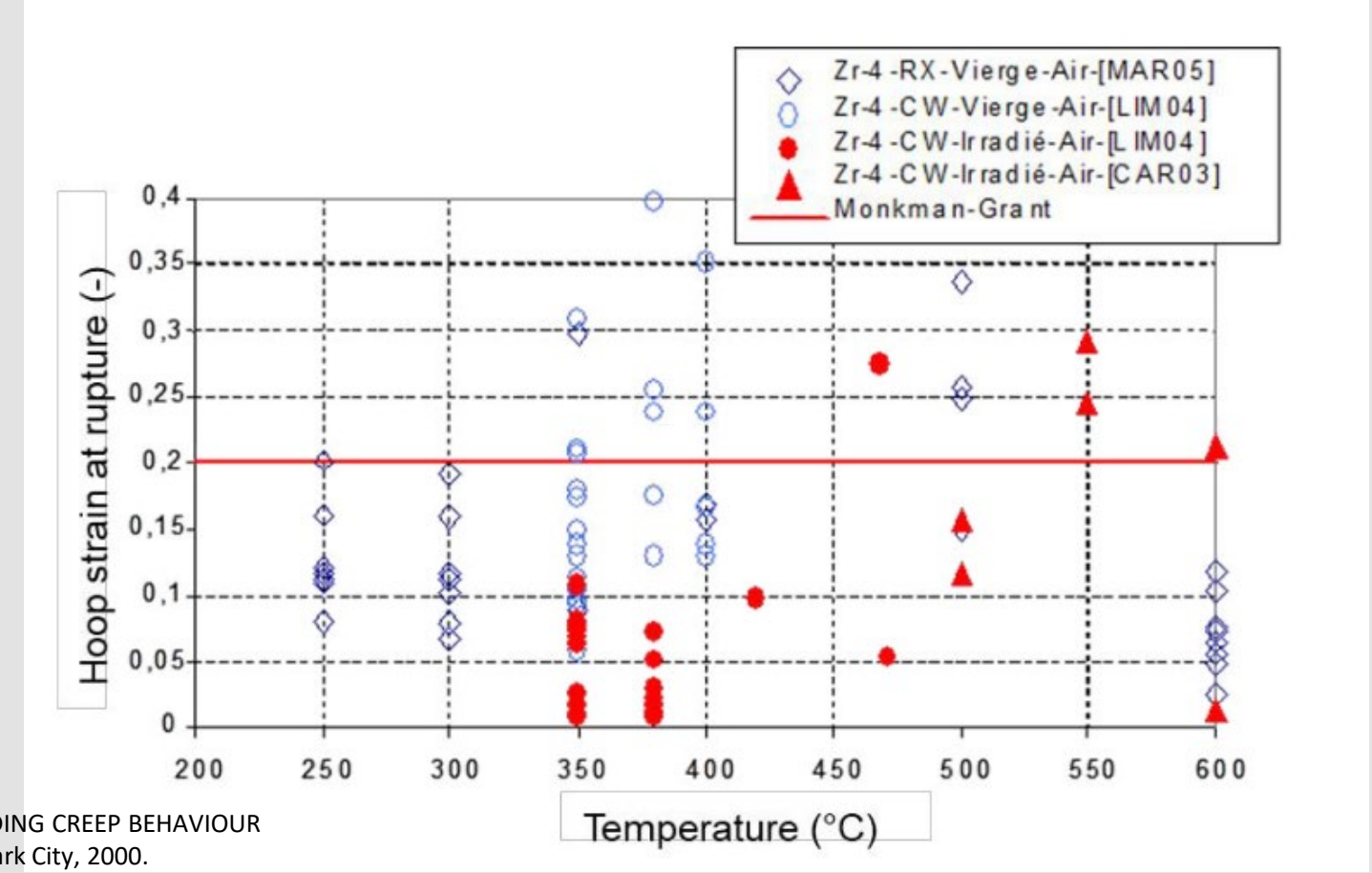
- Creep acceleration observed above ~450°C. 7 orders of magnitude high creep rate at 600°C compared to 400°C
- Estimated time to 12% (Einziger tests) strain varies between 1 s - 40 min depending on hoop stress (5-15% FGR).



*Desquines etal. Proceedings of Top Fuel 2009 Paris, France, September 6-10, 2009 Paper 2106
*SCIP

What is the clad strain-to-failure?

- Very hard to predict
- At 500-600°C strain to failure here varies between 1-34%.
- Also, the 12% strain observed in Einzigers tests is not conservative when it comes to if the clad fails or not.



*R. Limon, C. Cappelaere, T. Bredel, "A FORMULATION OF THE SPENT FUEL CLADDING CREEP BEHAVIOUR FOR LONG TERM STORAGE", Light Water Fuel Performance Meeting, April 10-13Park City, 2000.
*Brian Marple – Creep Rupture Study of Annealed Zircaloy-4 stress and temperature effects – Master of Science Thesis – University of North Carolina – 2005.

Interpretation of results

Are the limiting factor the strain…

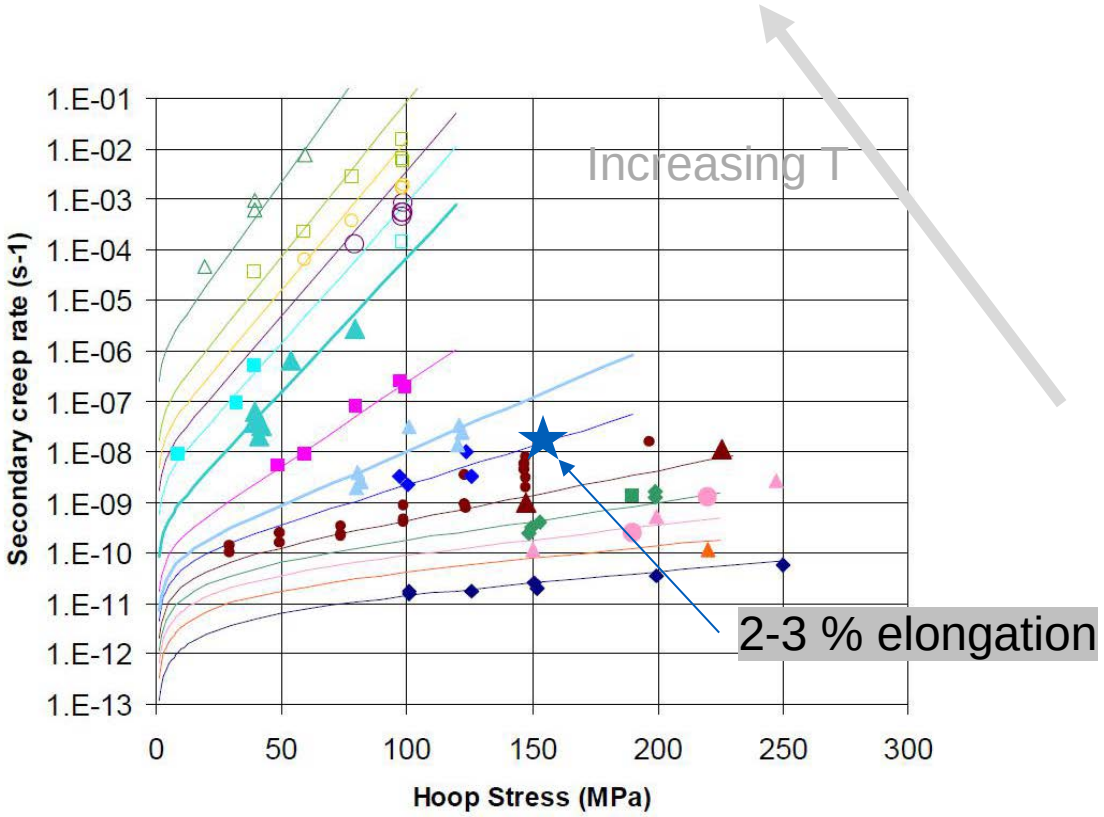
or the transition from secondary creep to tertiary creep?

- If the limiting factor is strain, there are some unknown factors affecting the variation! When the variation seems great it is hard to have a certain strain as a limit for when the rod bursts.

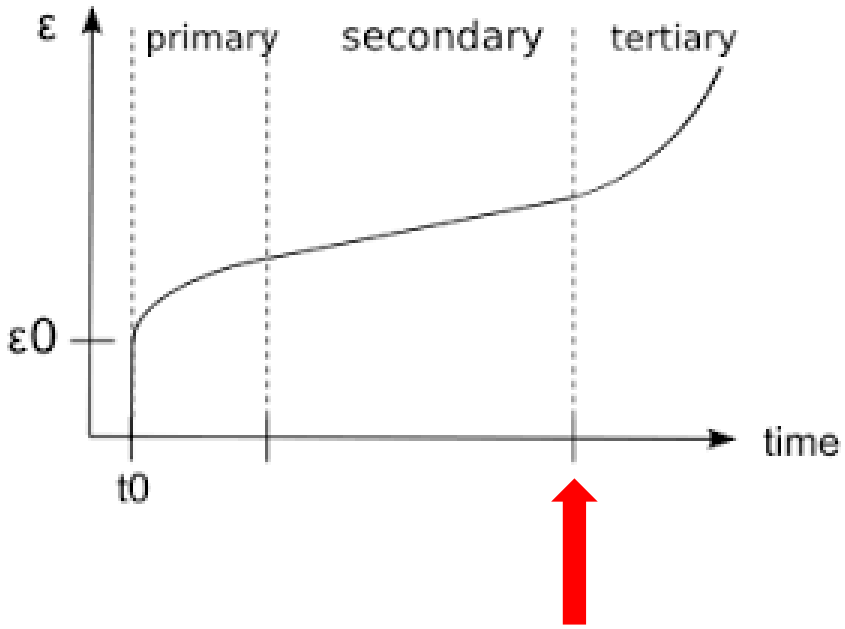
- If the limiting factor is instead the transition from secondary to tertiary creep, the underlaying factor may be deduced from time at temperature, internal pressure and other factors such as fuel rod status (Geometry, hydrides and bonding to pellet)

Interpretation of results, continued

- Time to burst or time to strain limit?
- 30 days are 2.6 e6 seconds, at 450C ~2% elongation



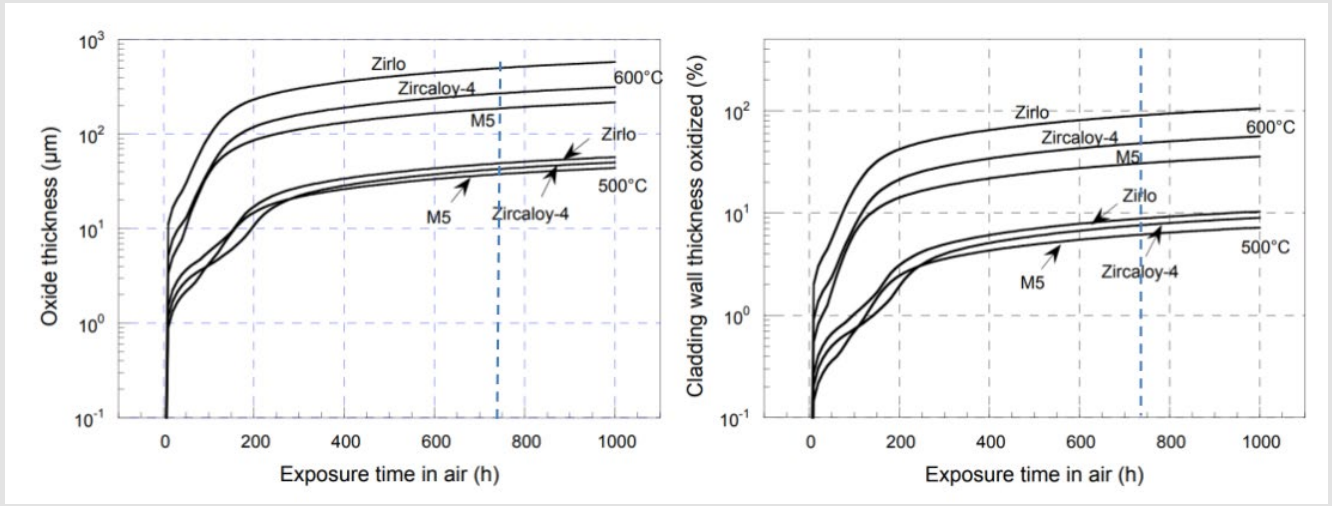
VS

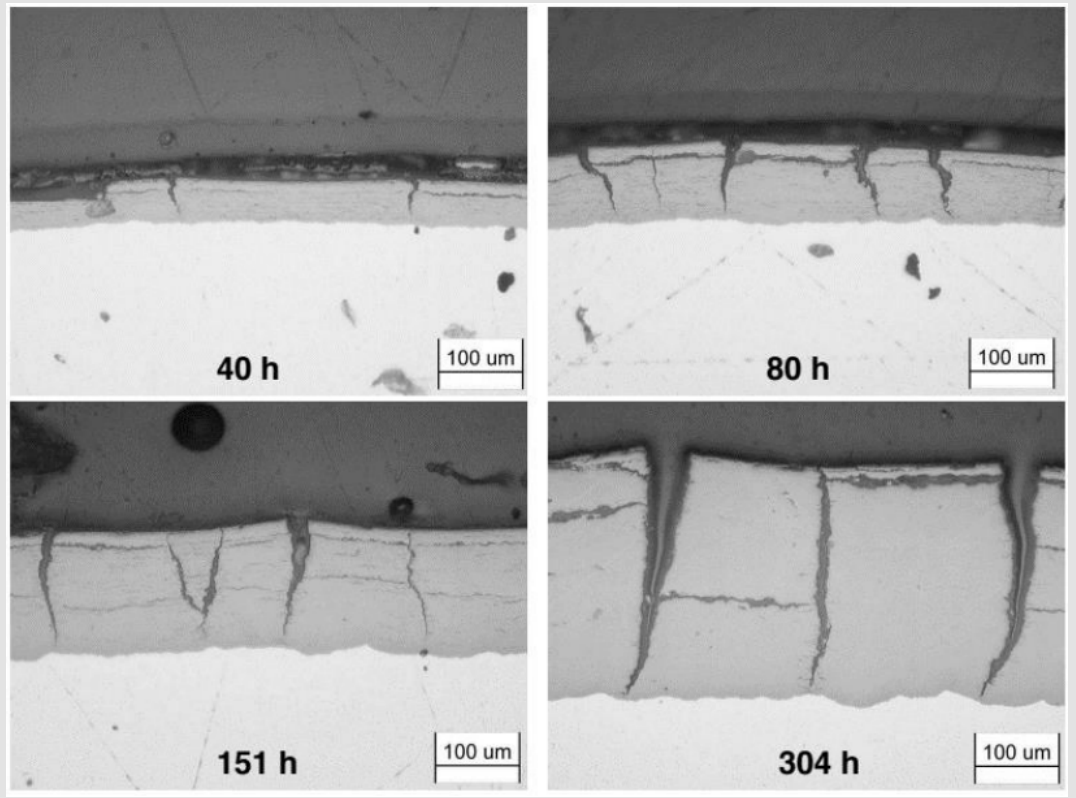


Time to burst

If in air

- With access to air during handling giving rise to elevated temperatures the tests below indicate that, depending on cladding alloy, 100% of the clad wall thickness can be oxidized after 30 days at 600°C.





*Air Oxidation Kinetics for Zr-Based Alloys NUREG/CR-6846 ANL-03/32

Causes

- Slow increase over the years (1979 – 2025) from 30 => 60 MWd/kgU

→

Increase in burnup gives higher FGR and higher internal pressure.

- Misinterpretation of ISG 11 rev 3

→

“For hypothetical accident conditions, the licensee must assure that there is no significant cladding failure.”

”…no significant cladding failure…” does not mean no cladding failure at all

Experimental backgound low burnup fuel (30 MWd/kgU) not high burnup (60 MWd/kgU)

CONCLUSIONS

=> There is a gap in the data supporting higher T above 450 C
=> There are a differences in results with only cladding tubes and results with whole fuel rods

Is there a bonding effect between cladding and pellet increasing the strength of the combined fuel rod?

SFP LOCA tests indicate that all released FG as well as significant fuel dispersal occurs if the clad bursts at 570C.

Is there an effect of increasing volume when the ballooning start resulting in a lower load?

Further investigation and tests may increase the limit above 450 C.

SKB safety assessment of transport cask

- Swedish intermediate storage is essentially a big fuel pool underground
- Storage is in deionized water
- We do wet-dry-wet transition at about 9-20 months FA out of core.

Transport conditions between NPP fuel pool and Clab:
Temperature limit is 400 C, at drying in vacuum near this limit (300+ C). During transport in reality a lot less, typically 120 C.

Accident conditions when received at Clab:
Temperature limit has been 570 C for a month at 30 days in transport cask filled open to air.

Solution:
The limit is changed from 570 to 450 C.

skb.se

Review of Technical Readiness for Fresh and Spent Fuel Storage & Transportation

Summary of EPRI ATF/LEU+/HBU Workshop

Hatice Akkurt
Used Fuel and High-Level Waste Management, EPRI

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 27-30, 2025

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EPRI ATF/LEU+/HBU Workshop - Review of Technical Readiness for Fresh and Spent Fuel Storage & Transportation

Objectives:

- **To evaluate the technical readiness for near term implementation of ATF/LEU+/HB**
  - Review of the significant amount of work already completed by numerous organizations, including EPRI, NRC, vendors, and utilities
  - Identify any remaining issues and develop a path forward

Workshop was conducted February 11-13, 2025, as hybrid meeting – in person at EPRI Washington, DC office

Desired Outcome:

- **To identify a clear and timely path forward for NRC review and implementation of ATF/LEU+/HBU fuel by industry**

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ATF/LEU+/HBU Workshop Focus Areas:

1. Overview of fuel types and specifications
   - Fuel changes and potential impact on criticality
2. Code system capabilities and validation of the codes
3. Depletion uncertainty
4. Use of RG 1.240 for new and used fuel storage (based on NEI 12-16 Criticality guidance)
5. Dry storage & Neutronic issues

Workshop Participation

- **Total number of registered attendees: 106 (37 in-person)**
- Representatives from 10 countries
  - Brazil, Czech, Canada, France, S. Korea, Spain, Sweden, Switzerland, UK, USA
- Representatives from
  - NRC
  - Utilities (US and non-US),
  - ORNL, INL, DOE
  - Fuel vendors (Westinghouse, GNF, Framatome)
  - Studsvik
  - NEI

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ATF/HE/HBU Workshop – Review of Technical Readiness for Fresh and Spent Fuel Storage (Wet and Dry) & Transportation Structure

Day 1

- **General overview** – Industry perspectives and Fuel overviews
- **NRC Updates**; SCALE Updates; Criticality code validation

Day 2

- **Depletion uncertainty** (EPRI; ORNL; Comparison of results)
- **Code Validation** (Polaris and Parcs; Casmo5 and Simulate5; Paragon2 and ANC)
- **Comparison of results from different codes** (Casmo versus Triton/Newt versus Polaris)
- **Dry Storage and Transportation** – NRC Technical Readiness

Day 3

- **24 months versus 18 months** – Absorber and burnup profiles (Industry presentations)
- **Significance of ATF/HE/HBU: Fuel changes** (fuel vendor presentations)
- **Discussions** – to determine consensus on certain items, technical gaps, if any, and path forward

Working on an EPRI report to summarize the key points and agreements from the workshop

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1. Fuel Features

1. ATF – Coatings

- Ongoing demonstrations of performance
- **Impact on reactivity slightly negative**

2. Dopants

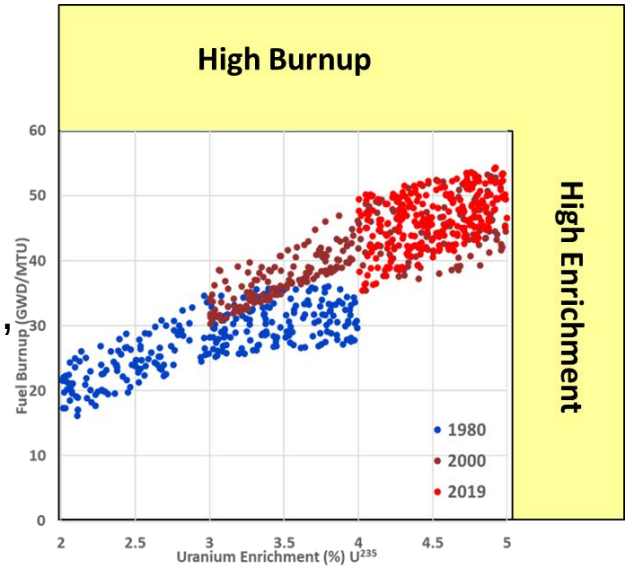
- Significant history in BWRs and ready for use
- Increased fuel density, similar trends with depletion (compared to undoped pellets), and lower fission gas release (considered in updated RG 1.183 Rev. 1)
- **Impact on reactivity: Small positive to small negative depending on the product**

3. LEU+

- Enrichments: 5%<E<8%, likely close to 6.5% for existing plants
- **Several transportation packages already approved**
- **Sufficient critical experiments for validation**

4. HBU

- 62<BU<80, likely interim limit ~68 followed by goal of reaching to ~75
- **Newer materials show smaller physical changes during depletion, supporting HBU** (see graphs from vendor presentations for grid growth, clad oxidation, etc.)
- **Improved validation basis in 7303 – with the addition of more data and removal of high uncertainty data**



Incremental improvements consistent with industry's history – no major changes for neutronics; gains for corrosion resistance

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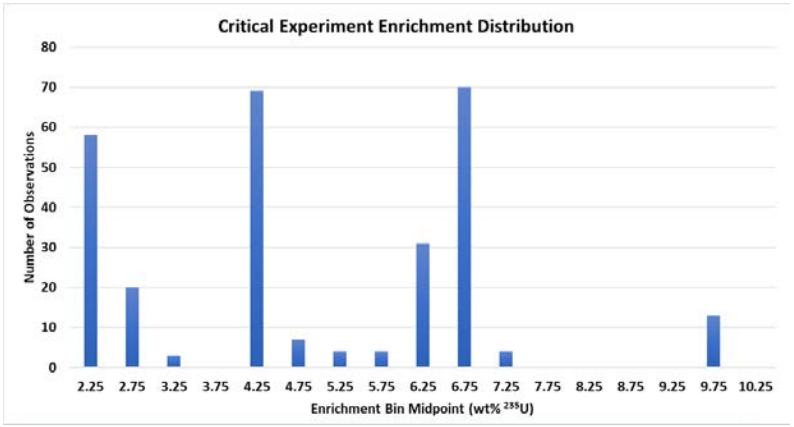
2. Readiness of Codes

1. **Criticality codes**
   - Consensus: Scale/MCNP – no challenges
   - Code validation: Enough critical experiments for LEU+ code validation



Significant number of experiments in the 5-8% range

1. **Depletion codes**
   - Approved (PARAGON2, LANCR, etc.) or submitted for approval (CMS5)
   - Confirmatory analysis by comparison to other codes (Polaris and/or Triton) – vetted/validated via NUREG/CRs

Codes and nuclear data does not show any cliff edge effects:

1. **None of the codes required any modifications for ATF/LEU+/HBU**
2. **Codes are in good agreement with each other as part of confirmatory analysis**

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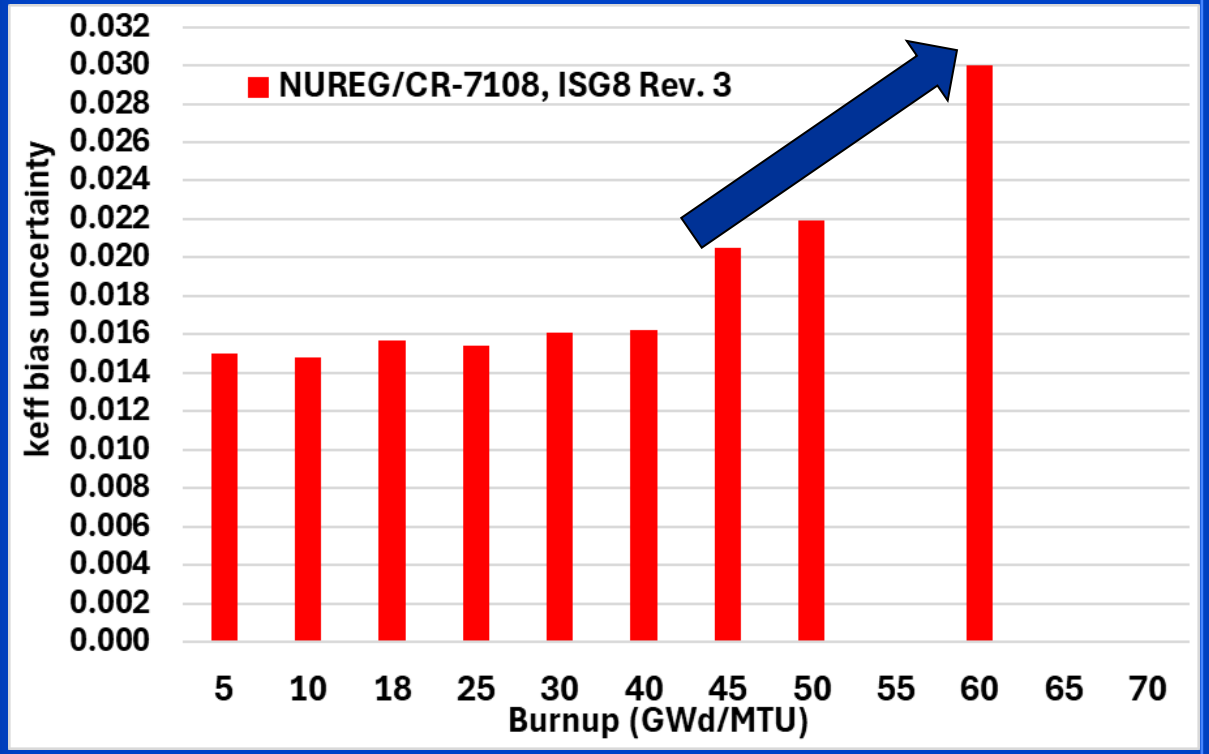
3. Depletion Uncertainty (DU) – NUREG/CR-7108

- NUREG/CR-7108 was published in 2012
- Based on chemical assay measurement ➔ Large measurement uncertainty
- **Showed increase in uncertainty with increasing burnup**
  - NRC was concerned about increased uncertainty beyond 60 GWD/MTU due to increasing trend
- **NUREG/CR-7108 was implemented via ISG-8, Rev. 3 (used for dry storage)**



Increase in depletion uncertainty with increasing burnup

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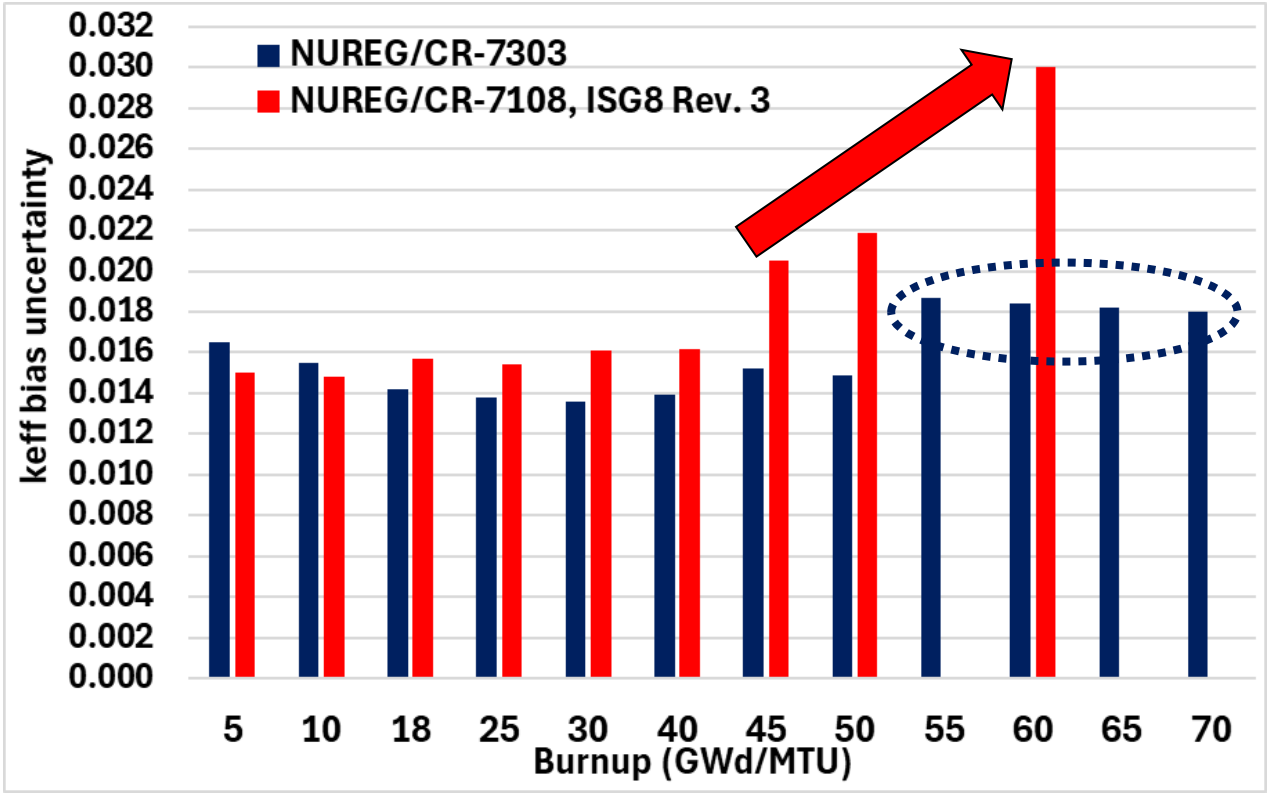
3. Depletion Uncertainty (DU) – NUREG/CR-7108 versus 7303

- **NUREG/CR-7303 published in September 2023**
  - Added improved quality measurements
  - Excluded some low-quality measurements
- **NUREG/CR 7303 versus 7108**
  - 7303 shows significant reduction in depletion uncertainty compared to 7108
  - Increased enrichment and burnup does not increase depletion uncertainty
  - Results are still conservative for estimation of the depletion uncertainty due to:
    - Lack of covariance – each nuclide distribution assumed independent of all others, which is not the real case
    - Presence of RCA measurement uncertainty
    - Inherent burnup uncertainty in RCA samples



Key conclusion from 7303: Depletion uncertainty does not increase with increasing burnup – almost flat with slight decrease trend

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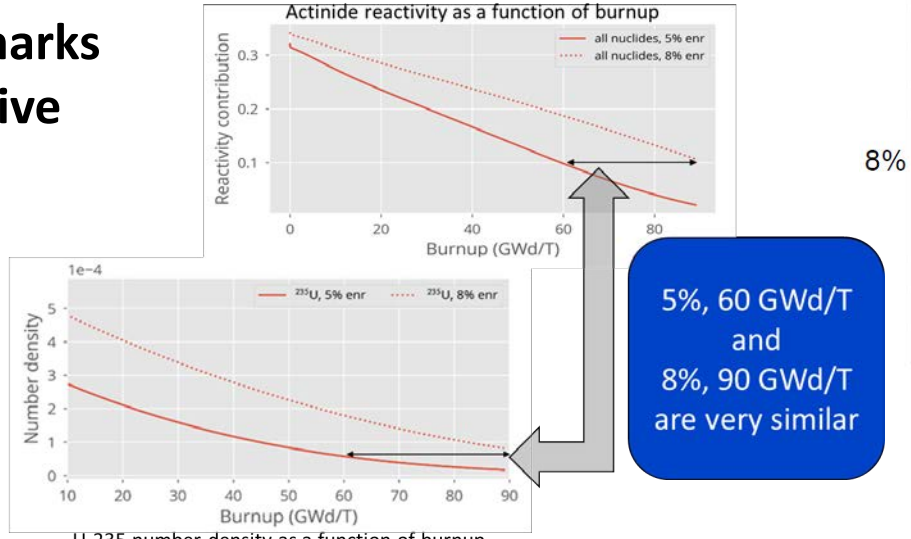
3. Depletion Uncertainty (DU) – EPRI Benchmarks and Extension of EPRI Benchmarks

Regulator approved EPRI Benchmarks showed Kopp memo is conservative

Burnup (GWd/MTU)	Uncertainty (%)	Bias (%)
10	3.05	0.0
20	2.66	0.0
30	2.33	0.0
40	2.12	0.15
50	1.95	0.35
60	1.81	0.54

Valid up to 5% enrichment and 60 GWd/MTU Burnup



U-235 number density as a function of burnup

5%

8% Bu	0	0.5	10	20	30	40	50	60	70	80	90
0	**0.99**	0.99	0.95	0.87	0.77	0.66	0.54	0.45	0.36	0.3	0.25
0.5	0.99	**0.99**	0.95	0.87	0.78	0.67	0.56	0.46	0.37	0.31	0.26
10	0.98	0.98	**0.98**	0.94	0.86	0.76	0.66	0.57	0.49	0.42	0.37
20	0.95	0.96	0.99	**0.97**	0.92	0.84	0.75	0.67	0.59	0.53	0.48
30	0.91	0.92	0.98	0.99	**0.96**	0.9	0.83	0.75	0.68	0.62	0.58
40	0.85	0.86	0.95	0.99	0.98	**0.95**	0.9	0.83	0.77	0.72	0.67
50	0.78	0.79	0.9	0.97	0.99	0.98	**0.94**	0.89	0.84	0.8	0.76
60	0.71	0.72	0.85	0.93	0.98	0.99	0.97	**0.94**	0.9	0.86	0.83
70	0.62	0.64	0.78	0.89	0.95	0.98	0.99	0.97	**0.94**	0.91	0.88
80	0.54	0.56	0.71	0.83	0.92	0.97	0.99	0.99	0.97	**0.95**	0.93
90	0.47	0.48	0.64	0.77	0.87	0.94	0.98	0.99	0.98	0.97	**0.96**

5%, 60 GWd/MTU and 8%, 90 GWd/MTU are very similar – Ck is 0.99

Extension of EPRI Benchmarks to LEU+/HBU - EPRI report, 3002026550

Burnup (GWd/MTU)	10	20	30	40	50	60	70	80	90
Bias (% of depletion reactivity)	0.64	0.62	0.62	0.62	0.61	0.59	0.57	0.56	0.54
Uncertainty (% of depletion reactivity)	3.7	3.0	2.8	2.7	2.6	2.6	2.5	2.4	2.3

Physics, similarity index does not show significant change for ATF/LEU+/HBU; hence, depletion uncertainty for LEU+/HBU << 5% Kopp memo.

Results published in publicly available EPRI report 3002026550

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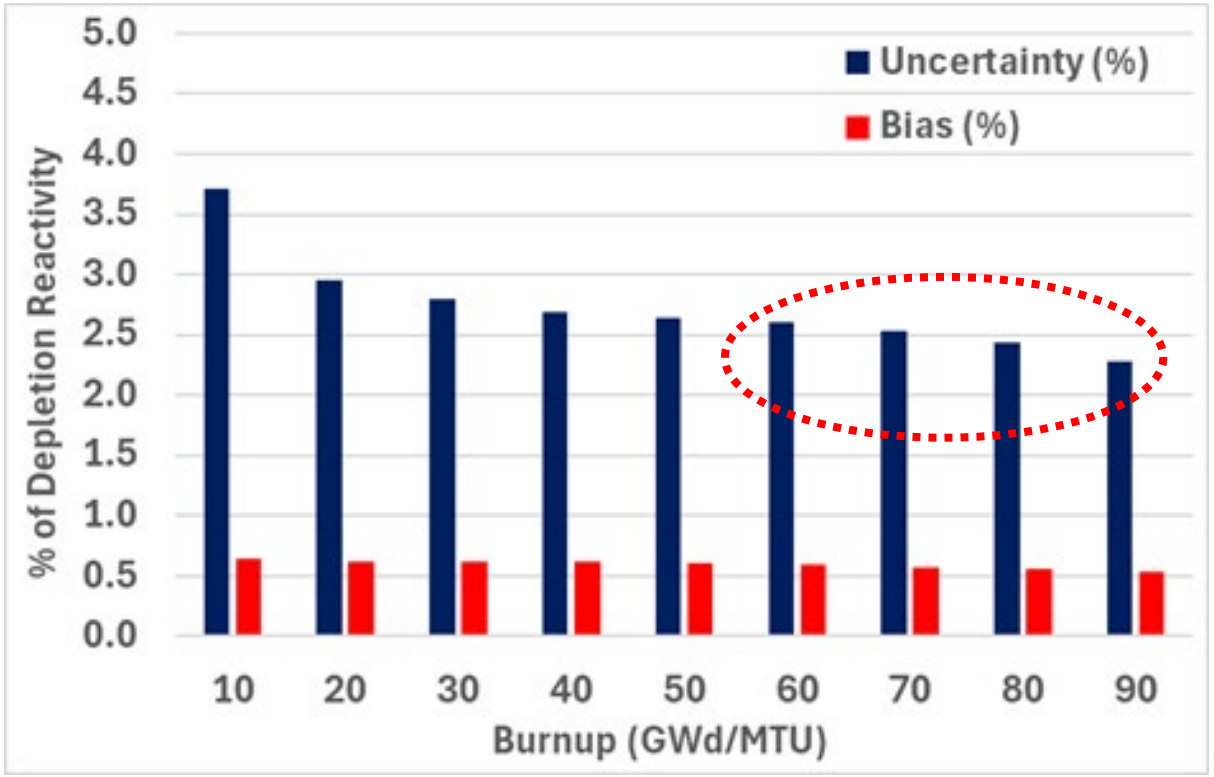
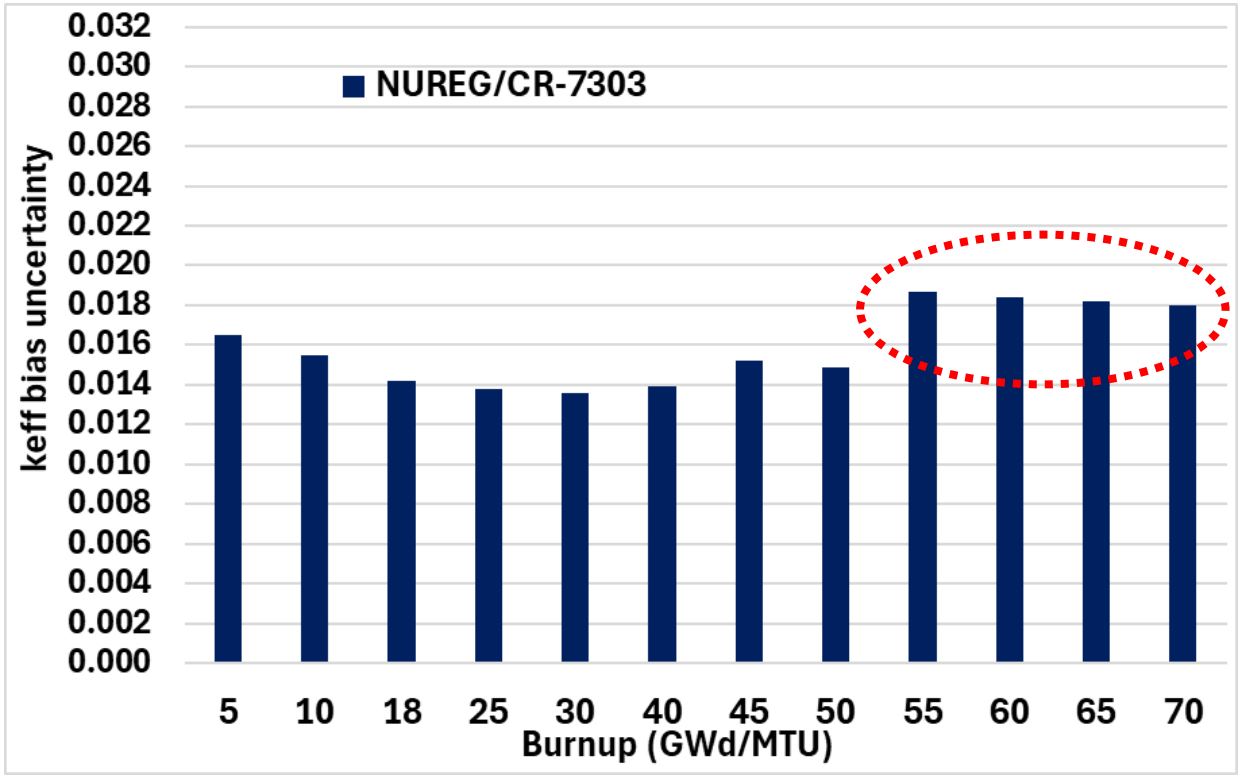
3. Depletion Uncertainty (DU) – EPRI Benchmark Extension versus NUREG/CR 7303



EPRI Benchmark Extension



NUREG/CR-7303

Multiple results (7303, EPRI Benchmarks) show similar trends (no increase in depletion uncertainty with increasing burnup) and support continued use of 5% depletion uncertainty (Kopp memo) for ATF/LEU+/HBU

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4. RG 1.240 Applicability for ATF/LEU+/HBU

- **Criticality code validation**
  - Can continue to use **NUREG/CR 6698** methodology – **Guidance still applies**
- **Depletion codes**
  - Licensing approach depends on if there is an approved TR (LEU+/HBU) – **Guidance still applies**
- **Depletion uncertainty (5% Kopp memo use)**
  - Multiple results (7303, EPRI Benchmarks) support continued use of 5% depletion uncertainty (Kopp memo) for ATF/LEU+/HBU – **Guidance still applies**
- **Unvalidated nuclides**
  - Conclusions in **NUREG/CR 7109** applies to LEU+/HBU – **Guidance still applies**
- **Fuel Assembly physical changes**
  - Values and trend may change - **Guidance still applies**
- **Axial Burnup Distributions**
  - Generic burnup shapes need to be confirmed for 24-month cycles – **Guidance still applies**
- **Eccentric positioning**
  - **Guidance continues to apply**
- **Multiple misloads**
  - **Guidance continues to apply**
- **Volatile fission products**
  - RG 1.183 Rev.1 covers doped pellets - **Guidance still applies**

There was consensus that guidance in RG 1.240 is applicable for ATF/LEU+/HBU despite the listed exception that states RG 1.240 is limited to 5% enrichment and 62 GWd/MTU

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5. Dry Storage Items & NRC Work in the Area

Recently Completed & Upcoming SNF Research

2020 - 2023

• Assessment of Existing Transportation Packages for Use with HALEU	Sept. 2020	ML21040A518
• Isotopic and Fuel Lattice Parameter Trends in EE and HBU LWR Fuel (PWR)	Feb. 2021	ML21088A336
• Isotopic and Fuel Lattice Parameter Trends in EE and HBU LWR Fuel (BWR)	Mar. 2021	ML21088A354
• Ext. Enrichment Accident Tolerant LWR Fuel Isotopic and Lattice Parameter Trends	Mar. 2021	ML21088A254
• Impacts of LEU+ and ATF on Fresh Fuel Storage Criticality Safety	Feb. 2022	ML22098A137
• Impacts of LEU+ and HBU Fuel on Decay Heat and Radiation Source Term	Apr. 2022	ML22159A191
• *Validating Actinides and Fission Products for Burnup Credit Criticality Safety Analyses - Nuclide Compositions Prediction with Extended Validation Basis**	Sept. 2023	NUREG/CR-7303

2024 - 2025

• Updated Recommendations Related to Spent Fuel Transport and Dry Storage Shielding Analyses	Feb. 2024	NUREG/CR-7302
• *Fuel Assembly and Irradiation Parametric Study for EE and HBU LWR Spent Nuclear Fuel in Dry Storage Casks and Transportation Packages**	Apr. 2024	NUREG/CR-7306
• Phenomena Identification and Ranking Table (PIRT) for Fuel and Cladding Property Changes Relevant to SNF Storage & Transportation of ATF Concepts	Oct. 2024	ML24292A170
• Assessment of Gross Breaches in Nuclear Fuel	Oct. 2024	ML24284A100
• Assessment of Alternative Criteria for Gross Ruptures in Spent Nuclear Fuel	Oct. 2024	ML24284A091
• Development of Decay Heat Sensitivity Analysis Capability in SCALE/ORIGEN	Jan. 2025	TBD
• *Validation Studies for High Burnup and Extended Enrichment Fuels in Burnup Credit Criticality Safety Analyses**	*2025*	TBD, NUREG/CR-7309
• *Sensitivity / Uncertainty Methods for Nuclear Criticality Safety Validation**	*2025*	TBD, NUREG/CR-7308
• Determination of Bias and Bias Uncertainty for Criticality Safety Computational Methods	*2025*	TBD
• *Nuclide Importance to Criticality Safety, Decay Heat, and Radiation Shielding Analyses for Transport and Storage of non-LWR Fuel (supported by DNCSH program)*	2025	TBD

*Highlighted in Today's Presentation

3

United States Nuclear Regulatory Commission
Protecting People and the Environment

Slide from the NRC presentation at the workshop

Significant amount of work done under NRC sponsorship

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Recent/Upcoming EPRI & Industry Activities

Recent Pre-submittal meetings for SFP Criticality

- **Dominio**n pre-submittal meeting was on July 17, 2025
- **Duke** pre-submittal meeting was on August 11, 2025
- Dominion & Duke used RG 1.240 along with the checklist
- No issues with the criticality code validation - enough critical experiments for LEU+
- Indicated they will use 5% depletion uncertainty (Kopp memo)
- No major challenges on the use of RG 1.240 or Kopp memo expressed by the NRC
- Approval of the CASMO5 LEU+ extension is a key component
  - CASMO5 LEU+ topical report is currently under regulatory review

Dominion and Duke pre-submittal meetings went very well.

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IHLRWM 2025 Paper Sessions on ATF/LEU+/HBU & Panel Session

Paper Session 1

Wednesday November, 12

8-9:45 am

- **Evaluation of Technical Readiness for Fresh and Spent Fuel Wet and Dry Storage and Transportation of ATF/LEU+/HBU**, Hatice Akkurt (EPRI)
- **Fuel Assembly Changes Impact on Spent Fuel Pool Criticality Analyses for LEU+ Applications**, Michael Wenner (WEC)
- **CASMO5 Analysis of Select HTC Critical Experiments Simulating Burned Fuel**, James W. Carnal (UTK), Joshua Hykes (Studsvik), Rodolfo M. Ferrer (Studsvik)
- **Considerations for KENO Benchmark for Spent Fuel Pool Criticality Safety Analyses**, David S. Orr (Duke Energy Corporation), Robert A. Hall (Duke Energy)

Paper Session 2

Wednesday November, 12

10:15-12:00 pm

- **PWR 24-Month Cycle Characteristics for SFP Criticality Analyses**, Kasey Kennett (Dominion Energy), David S. Orr (Duke Energy)
- **Application of Sampler to the LEU+ Burnup Credit Uncertainty Question**, Robert A. Hall (Duke Energy)
- **Comparison of LEU+ Reactivity Decrements for SFP Burnup Credit**, Robert A. Hall (Duke Energy)
- **Adequacy of 5% Depletion Uncertainty for LEU+/HBU SFP Burnup Credit**, Hatice Akkurt (EPRI), Robert Hall (EPRI)
- DOE Spent Nuclear Fuel Center for Applied Research, Steven Vitto (DOE), Paul E. Cantonwine (ORNL), Gordon M. Petersen (INL), Richard Wyrwas (SRNL), Mychailo Toloczko (PNNL), Paul A. Murray (DOE)

Panel

Wednesday November, 12

1-2:45 pm

- **Chair:** Hatice Akkurt (EPRI)
- **Panelists:**
- *Scott Krepel (NRC)*
- Al Csontos (NEI)
- Michael Boone (Westinghouse Electric Co.)
- David Orr (Duke Energy)
- *Andrew Barto (NRC)*

NRC may not be able to participate in panel session due to ongoing government shutdown.

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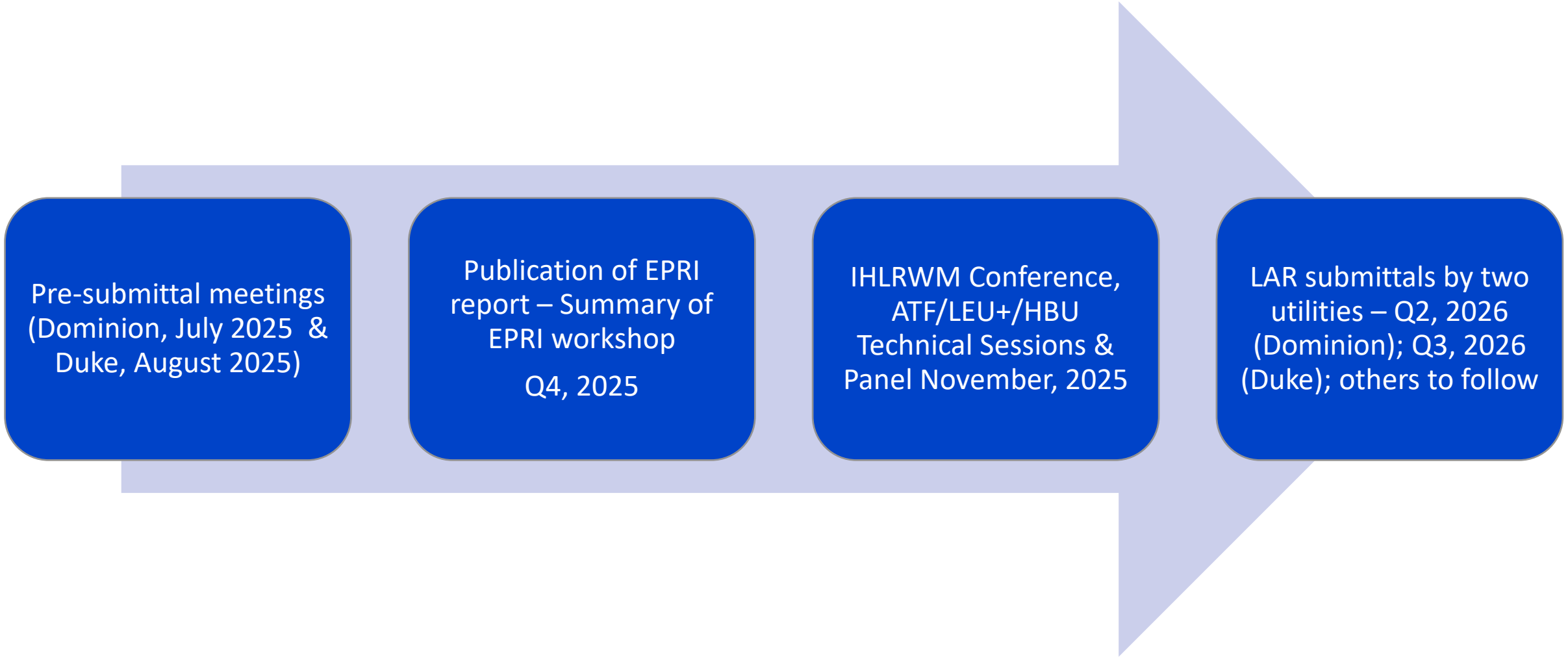
Next Steps for ATF/LEU+/HBU Implementation



EPRI workshop was instrumental in making progress for the regulatory review path

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Questions/Comments?

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Advanced Fuels Campaign

Status of PIE from Commercial LTAs at INL

Aaron Colldeweih, Karen E. Wright, Jason Schulthess, Colby Jensen, Robert Hansen, W. Alex Hanson, Katie Hawkins, Todd Jacobsen, Katelyn Baird, Phil Petersen, Jake Stockwell, Jaewoo Park, Joey Charboneau, Dahyeon Woo, David Kamerman and many more...

ESCP

October 2025

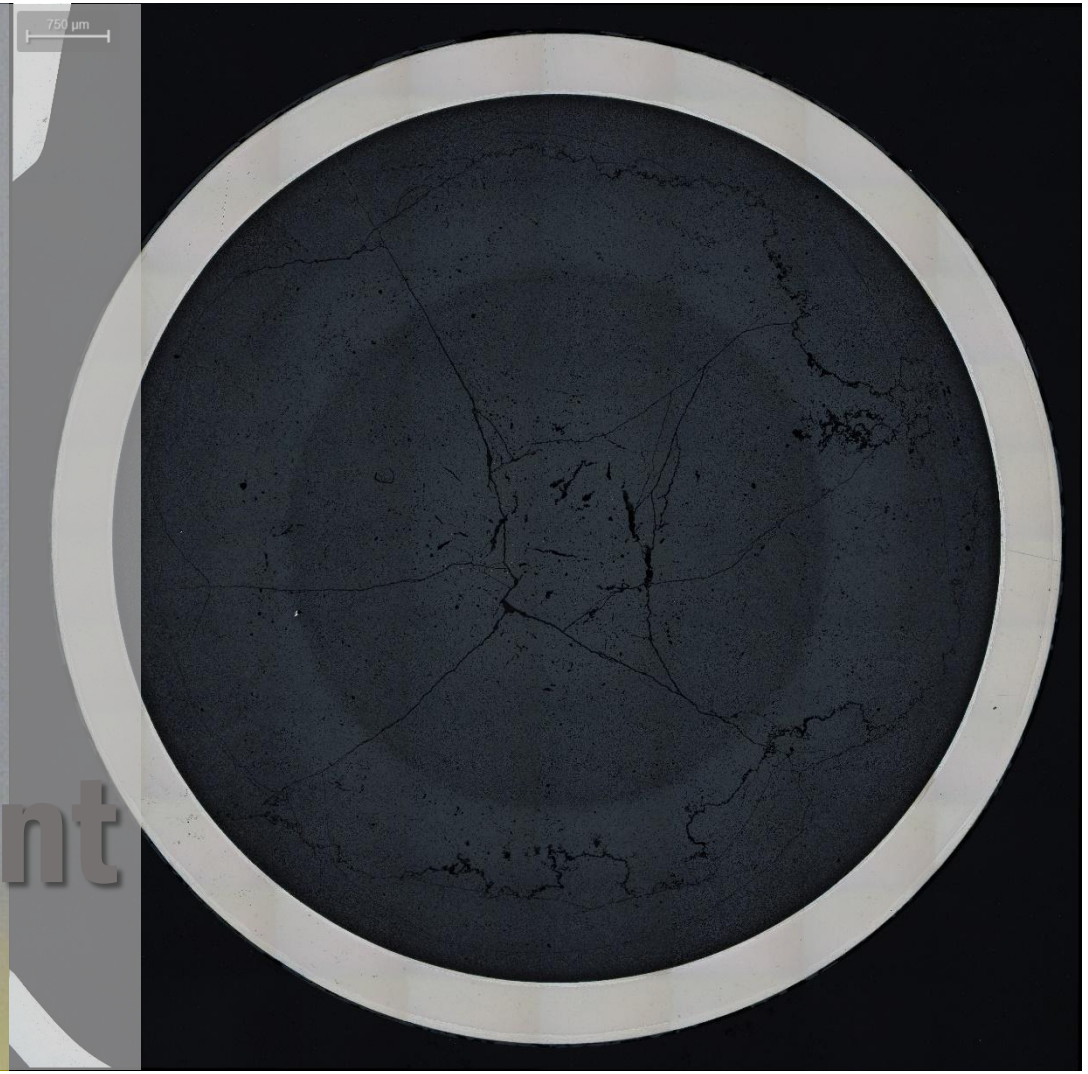
Charlote, North Carolina

U.S. DEPARTMENT OF ENERGY | Office of NUCLEAR ENERGY

Byron Shipment

PIE Results

Under Development



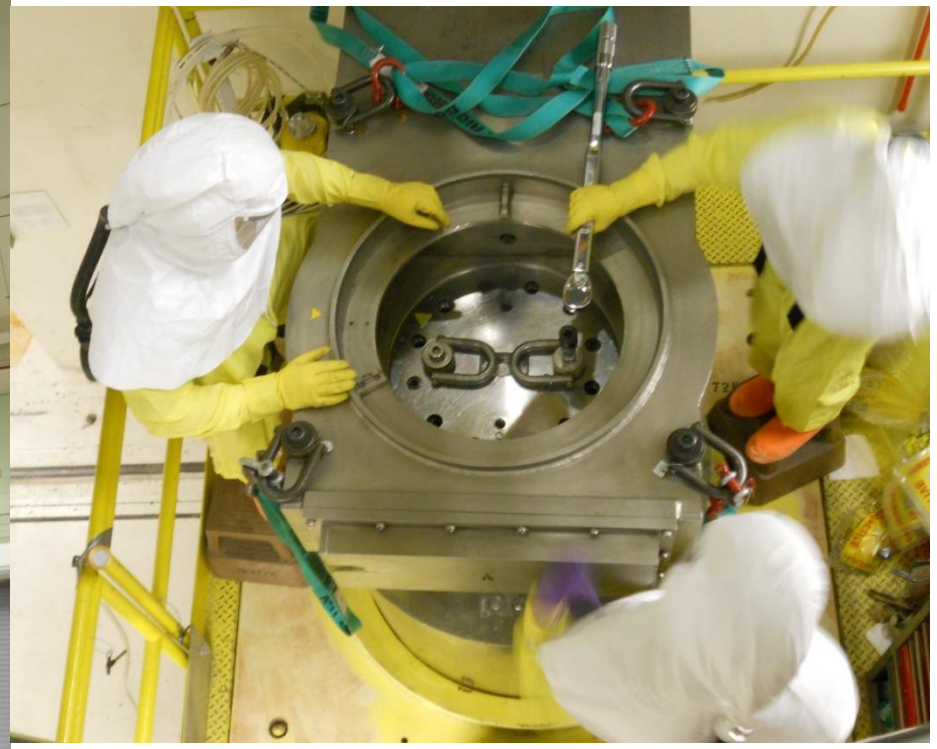
Byron – Shipment Overview

Shipments:

- NAC-LWT shipment from the Byron station received in December 2023 (ATF rods and HBU) and unloaded January 2024

Truck arriving at MFC

NAC-LWT on the high bay crane



Cask operations to prepare for unload

Byron Shipment – Inventory and Programs

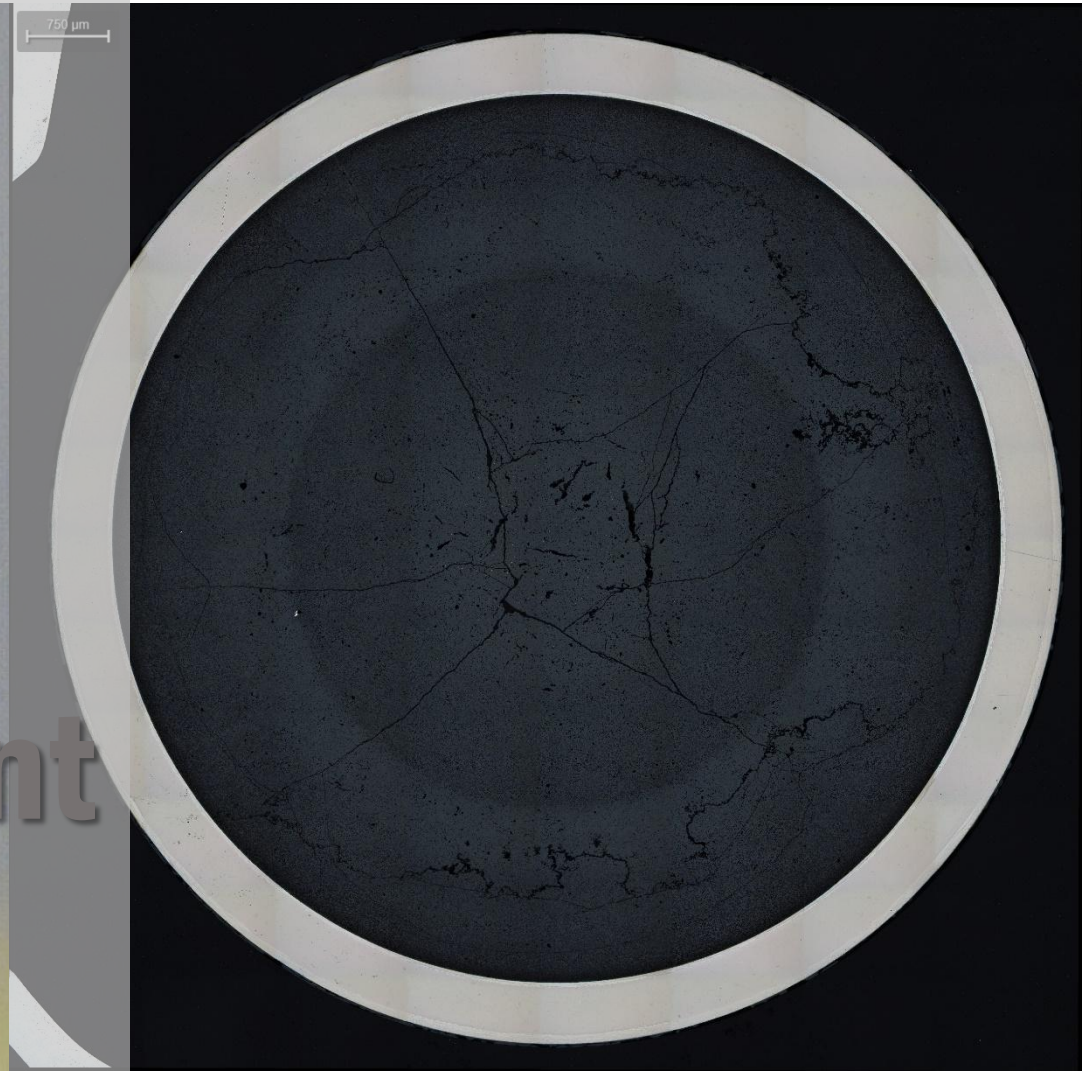
This talk!

Leading program	# Rods	Activities
AFC	5	• Comprehensive PIE pre and post testing • LOCA testing in TREAT and SATS • HERA testing in TREAT
ATF	5	• Comprehensive PIE to support WEC-led licensing efforts • HERA testing in TREAT (segments)
WEC SPP	3	• Original PIE plan being revised between parties
JFCS	12	• Limited PIE to support program objectives • Demonstration of integrated recycling test

Byron Shipment

PIE Results

Under Development



PIE Results - FY25 Overview

	Rod ID (GWd/MTU)	Full-Length NDE	Gas Puncture	Rough Section	Top rough section NDE	PIE Fine Sectioning	TREAT PREP	Cladding H content	Mech Testing	SEM + Adv char
AFC HBU	9EU (62)	Complete FY 24	Complete FY 24				HERA-HBU-1 Complete	Complete FY 25	N/A	In Progress
	HI5 (72)						HERA-HBU-2 in progress	Complete FY 25	N/A	FY 26
	SXM (71)		Complete FY 25	Complete FY 25	Complete FY 25	FY 27	N/A	N/A	N/A	N/A
	KBV (68)					FY 27	N/A	N/A	N/A	N/A
	4NB (65)					FY 27	N/A	N/A	N/A	N/A
ATF WEC	Q8V (55)	Complete FY 24	Complete FY 25	Complete FY 25	Complete FY 25	October 2025	HERA-HBU-4 in progress	October 2025	FY 26	FY 26
	4DS (55)					Complete FY 25	HERA-HBU-3 in progress	Complete FY 25	Complete FY 25	In Progress
	FM5 (55)					FY 27	N/A	FY27	FY 27	FY 27
	RK3 (29)	Complete FY 25			Gamma scan remains	FY 27	N/A	FY27	FY 27	FY 27
	1D6 (55)					FY 27	N/A	FY 27	FY 27	FY 27

PIE Results – Fission Gas Release

	Rod ID (GWd/MTU) - Fuel	Free Volume (cc)	Rod Internal Pressure (Mpa)	Fission Gas Release (%)
AFC HBU	9EU (62) – UO_2	16.7-17.5*	3.6-3.9*	3.8*
	SXM (71) – UO_2	16.4-18.5*	3.9-4.3*	3.9*
	HI5 (72) – UO_2	17.7-18.9*	3.8-4.0*	4.6*
	KBV (68) - IFBA	TBD	TBD	TBD
	4NB (65) – UO_2	TBD	TBD	TBD
ATF WEC	Q8V (55) – UO_2	17.5-18.6*	2.7-2.9*	1.2*
	4DS (55) - ADOPT	13.9-14.9*	2.8-3.0*	1.3*
	FM5 (55) – UO_2	17.0-18.2*	2.8-3.0*	1.2*
	RK3 (29) – UO_2	19.1	2.3	prelim
	1D6 (55) - IFBA	17	4.8	prelim

*based on corrected backfill and expansion pressures

PIE Results – Hydrogen analysis

Hydrogen analysis

Initiated on 2/5 AFC HBU rods

- Used for understanding the as-irradiated state of refabricated TREAT rodlets
- Hydride observations are informed with measurement values

9EU: Zr PWR Type B, UO_2 fuel, 62.7 GWd/tHM

Rod ID (GWd/MTU)	Distance from Bottom of Rod (mm)	Average Hydrogen Content (wppm)
9EU (62)	3096	565
	2850*	398
	1735	160
	1697	164
	1681	165
	1656	172
	1558	173
	1429	161
	1294	119

*Bottom of HERA-HBU-1

HI5: Zr PWR Type G, UO_2 fuel, ~72.4 GWd/tHM

Rod ID (GWd/MTU)	Distance from Bottom of Rod (mm)	Average Hydrogen Content (wppm)
HI5 (72)	2837*	161

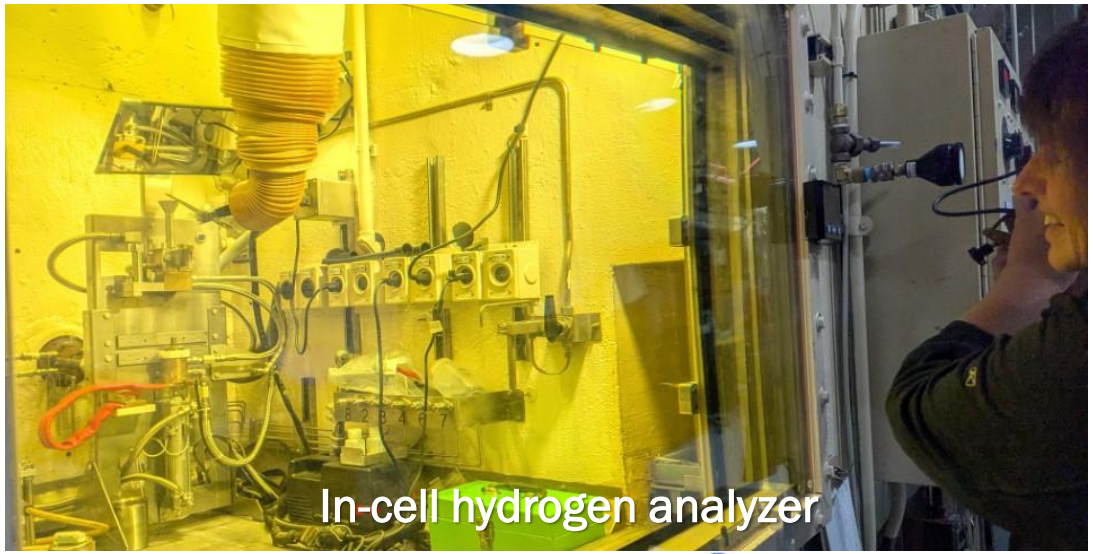
*Bottom of HERA-HBU-2

Initiated on 1/5 ATF rods

- Very little hydrogen uptake
- Corresponds well with hydride characterizations

4DS: CS Cr-coated Zr PWR Type C, ADOPT™ fuel, ~55 GWd/tHM

Rod ID (GWd/MTU)	Distance from Bottom of Rod (mm)	Average H (wppm)
4DS (55)	3082	Less than 50
	3087	
	2153	
	509	
	67	
	47	



PIE Results – HBU/ATF Microscopy

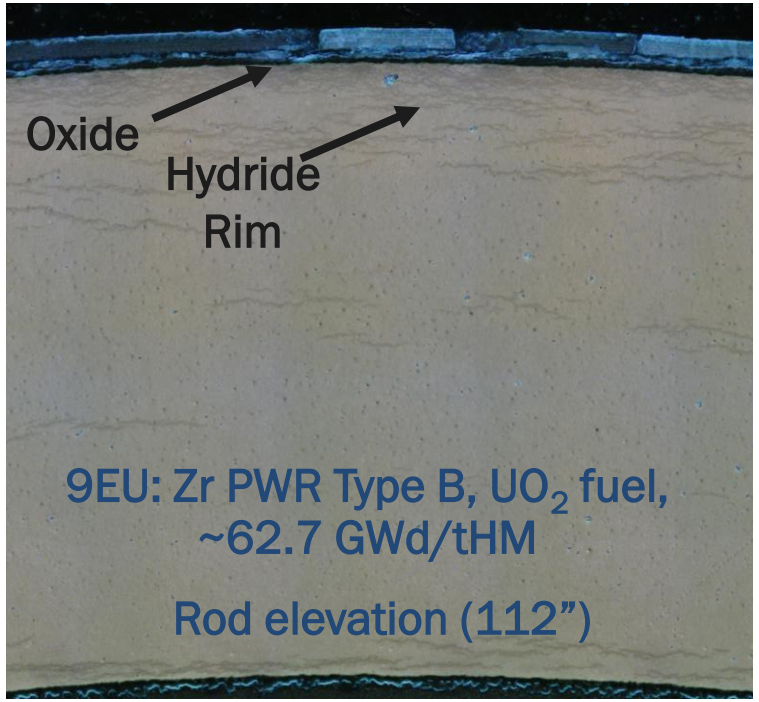
Metallography/Ceramography

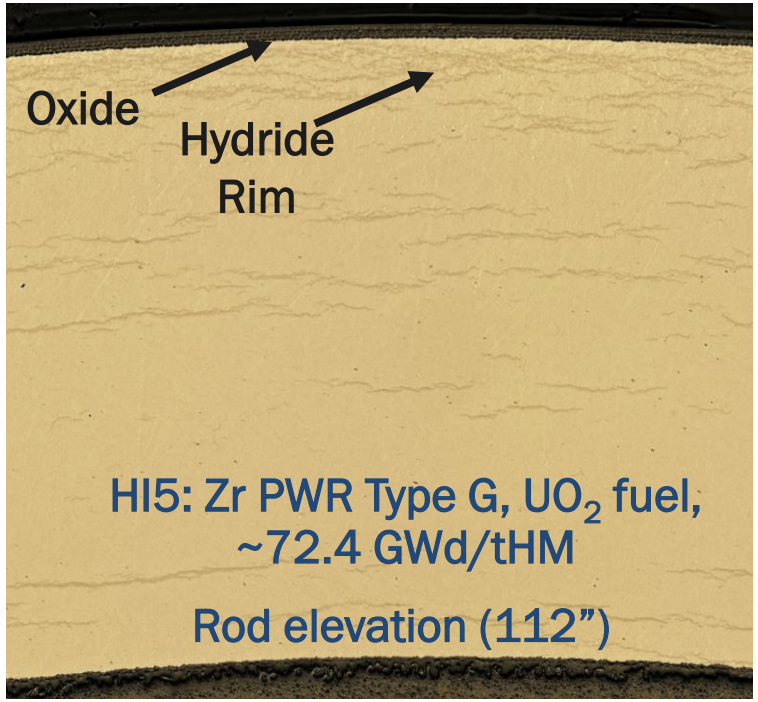
Initiated on 2/5 AFC HBU rods

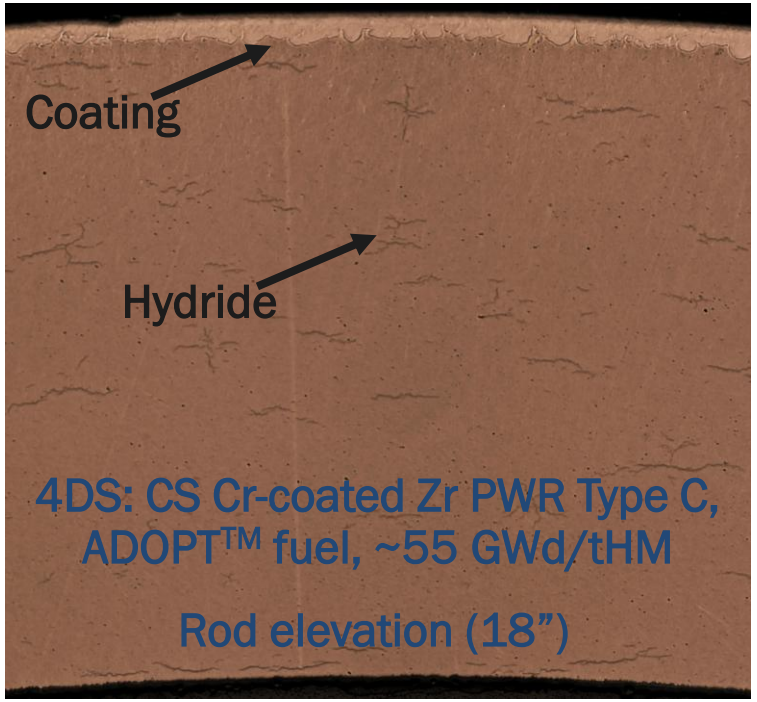
- Dimensions used as input for the refabricated TREAT rodlets
- Oxide thickness measurements have been obtained
- Hydride characterization

Initiated on 1/5 ATF rods

- Fuel structural morphology investigated
- Preliminary results show excellent adherence of the Cr-coating to the cladding
- Indication of very little hydride precipitation





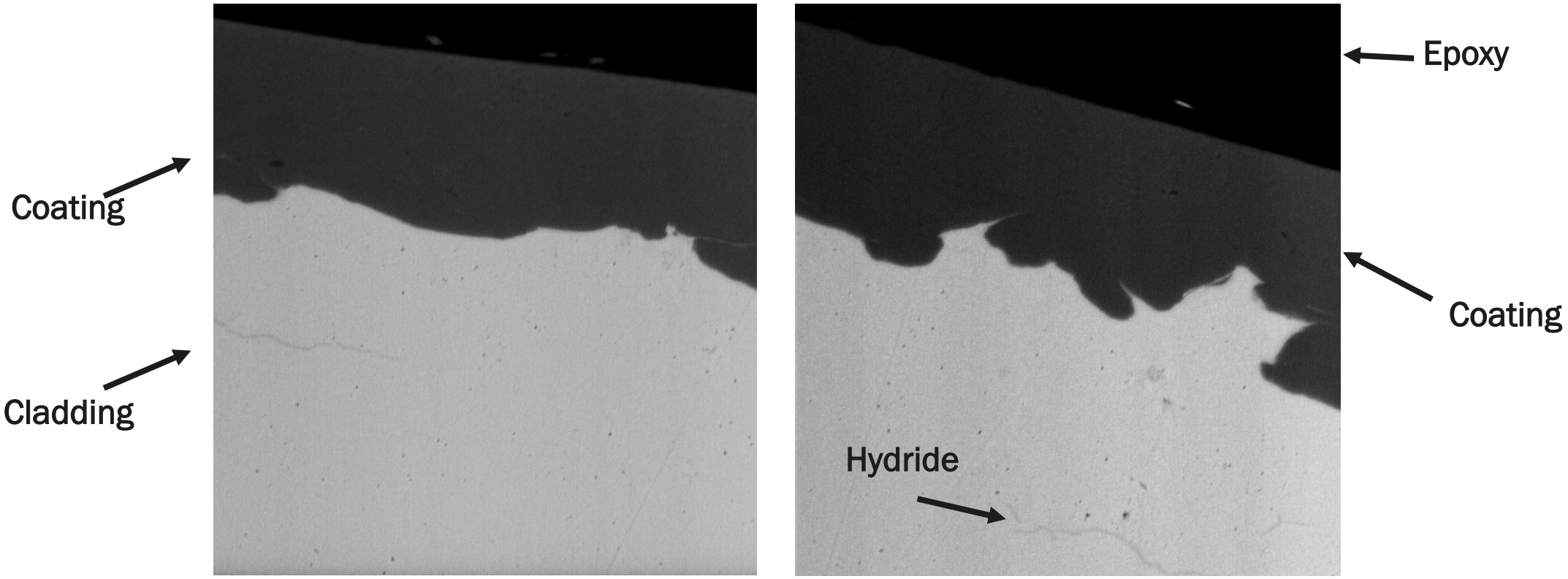


PIE Results – ATF Microscopy

Metallography

Initiated on 1/5 ATF rods

- BSE investigations on coating integrity and hydride morphology
- Preliminary results good adherence of the Cr-coating to the cladding
- Indication of very little hydride precipitation



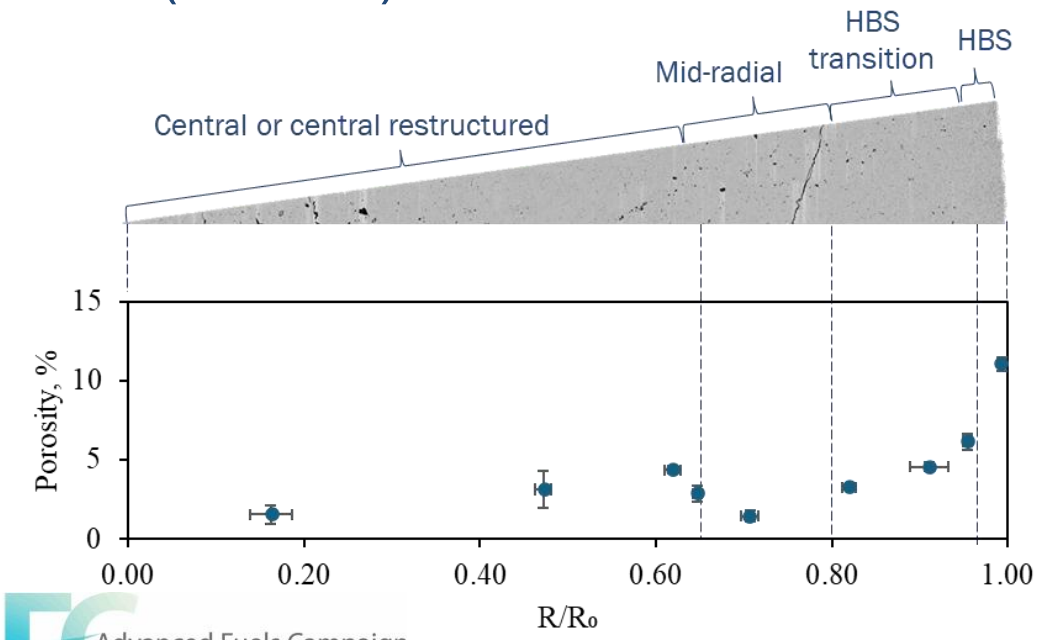
4DS: CS Cr-coated Zr PWR Type C, ADOPT™ fuel, ~55 GWd/tHM
Rod elevation (112”)

PIE Results – HBU Fuel Porosity

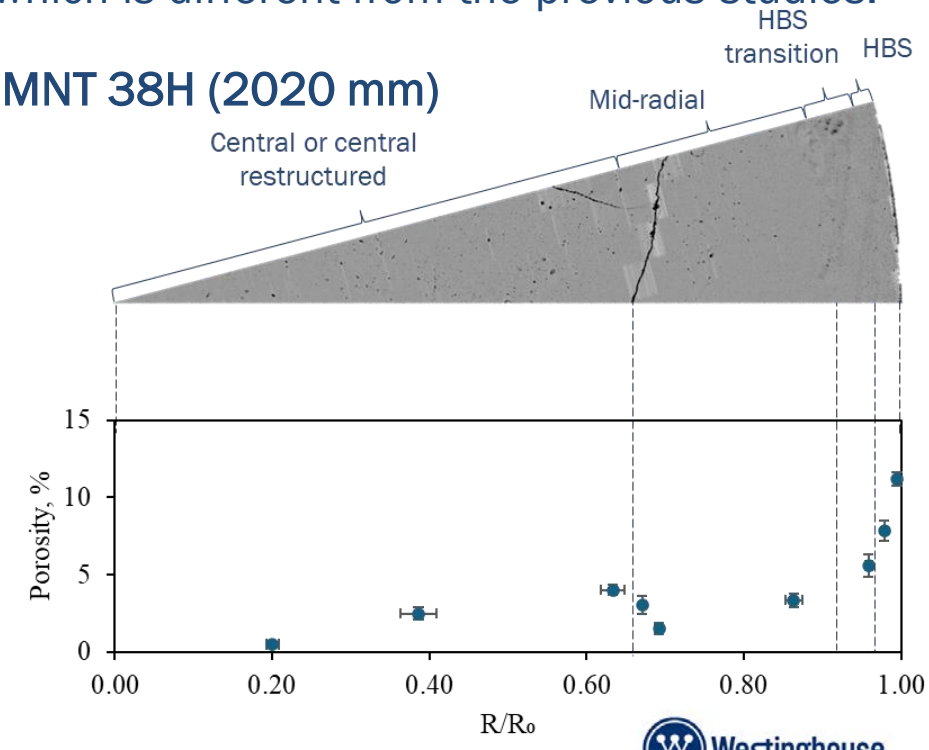
Porosity Distribution in 9EU rod

- Stitched backscattered electron images were analyzed to calculate porosity using Trainable WEKA Segmentation plugin on FIJI.
- The porosity distribution obtained from this study was comparable to the data of high burnup fuel reported by previous studies (Gerczak et al. and Cappia et al.)
- Porosity decreased near the center of fuel in this study, which is different from the previous studies.

MNT 36H (2540 mm)



MNT 38H (2020 mm)



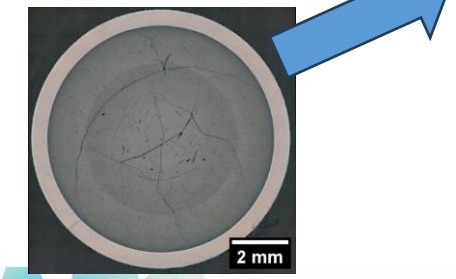
9EU: Zr PWR Type B, UO_2 fuel, ~62.7 GWd/tHM

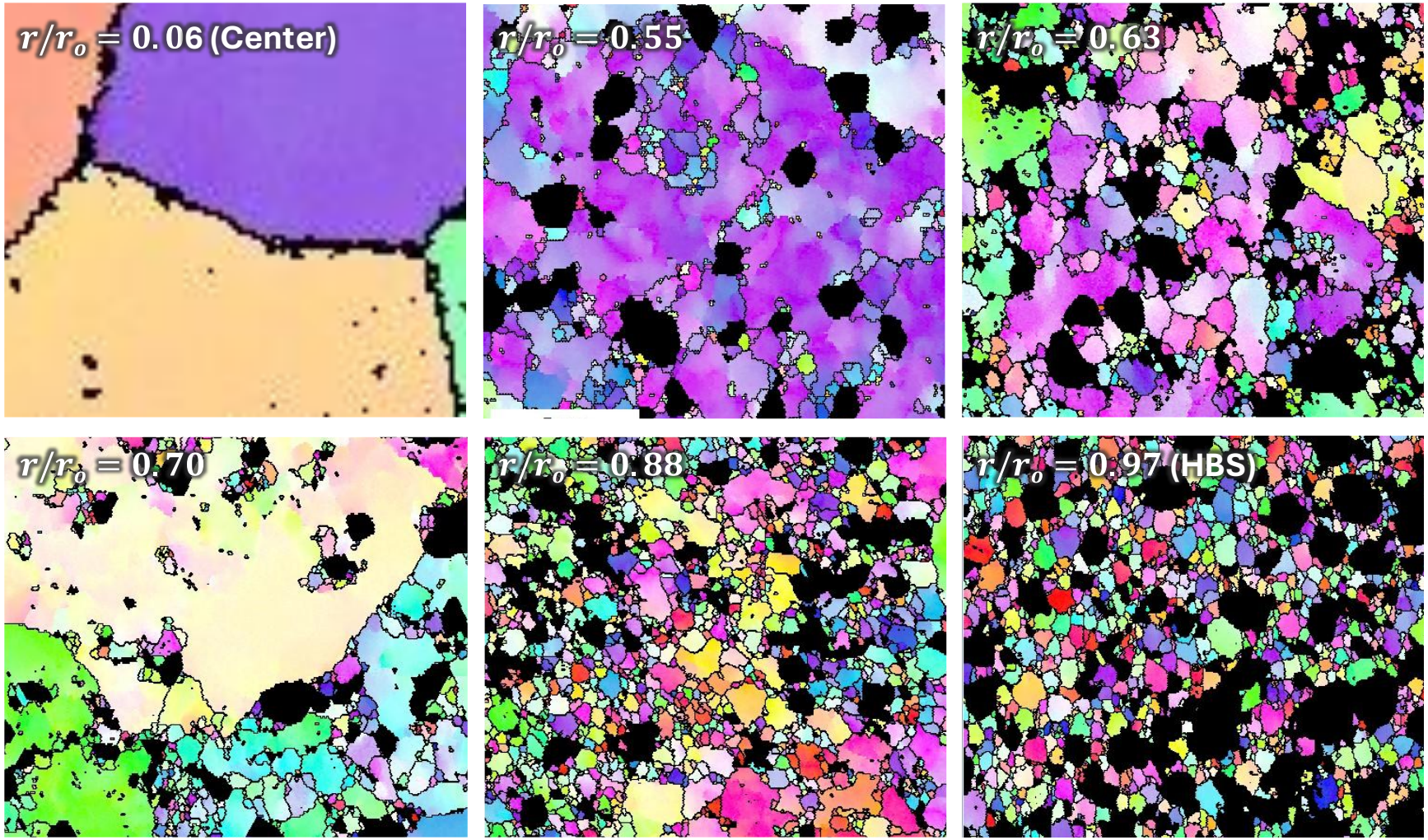
- Gerczak et al., Journal of Nuclear Materials, Vol.509 (2018)
- Cappia et al., Journal of Nuclear Materials, Vol. 569 (2022)

PIE Results – Electron Backscatter Diffraction

IPF Maps Full Fuel Cross-section

- Restructuring of grains near 0.6 r/r_o
- HBS and HBS transition is well defined
- Porosity is observed in all regions (inter- and intragranular)
- Quantitative analysis currently being processed



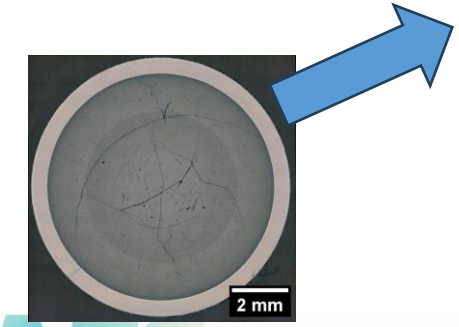


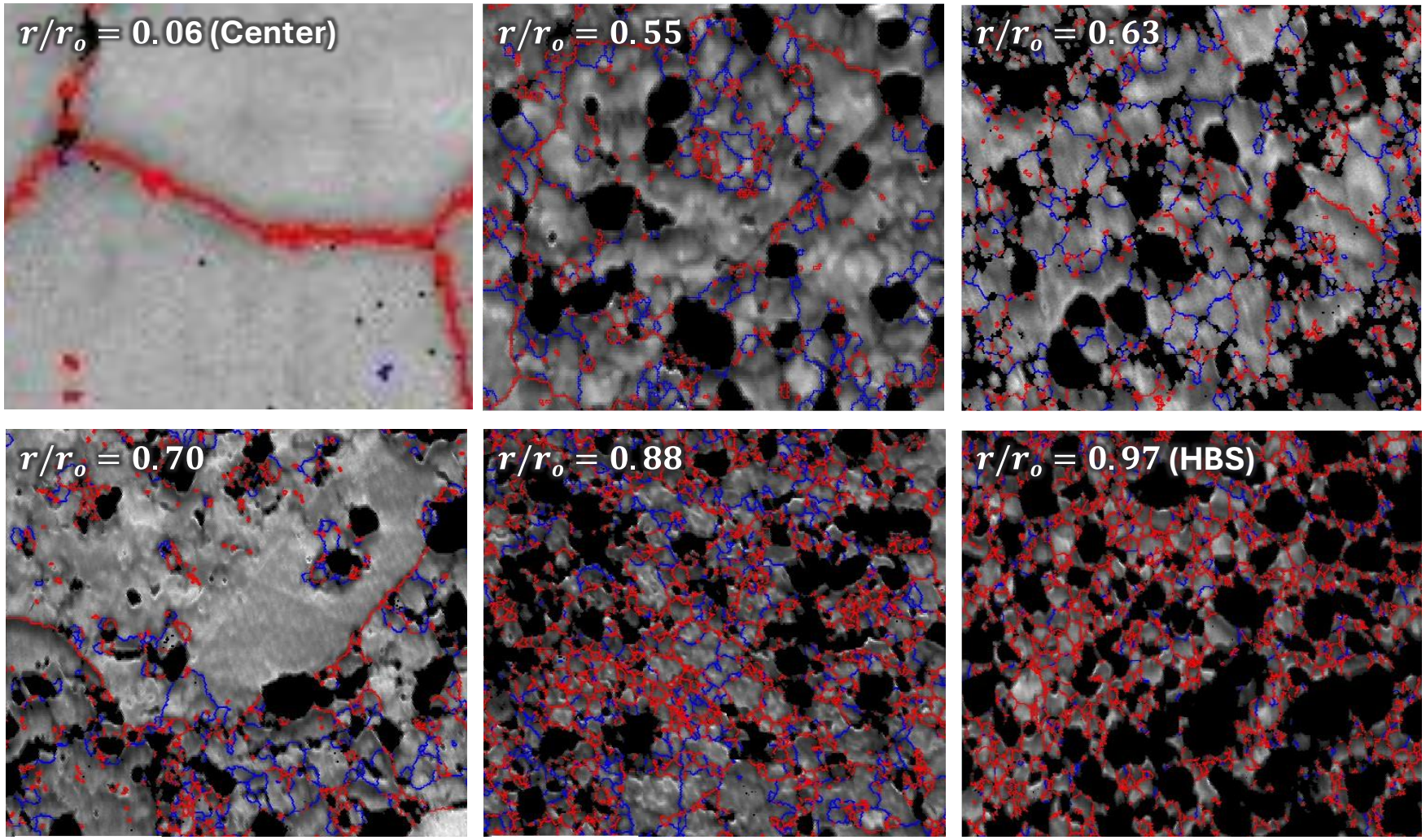
9EU: Zr PWR Type B, UO_2 fuel, ~62.7 GWd/tHM

PIE Results – Electron Backscatter Diffraction

GB Maps Full Fuel Cross-section

- HAGB present in parent grains
- Restructured zone shows the transition of LAGB to HAGB
- Clear formation of HAGB in the HBS
- Quantitative analysis currently being processed



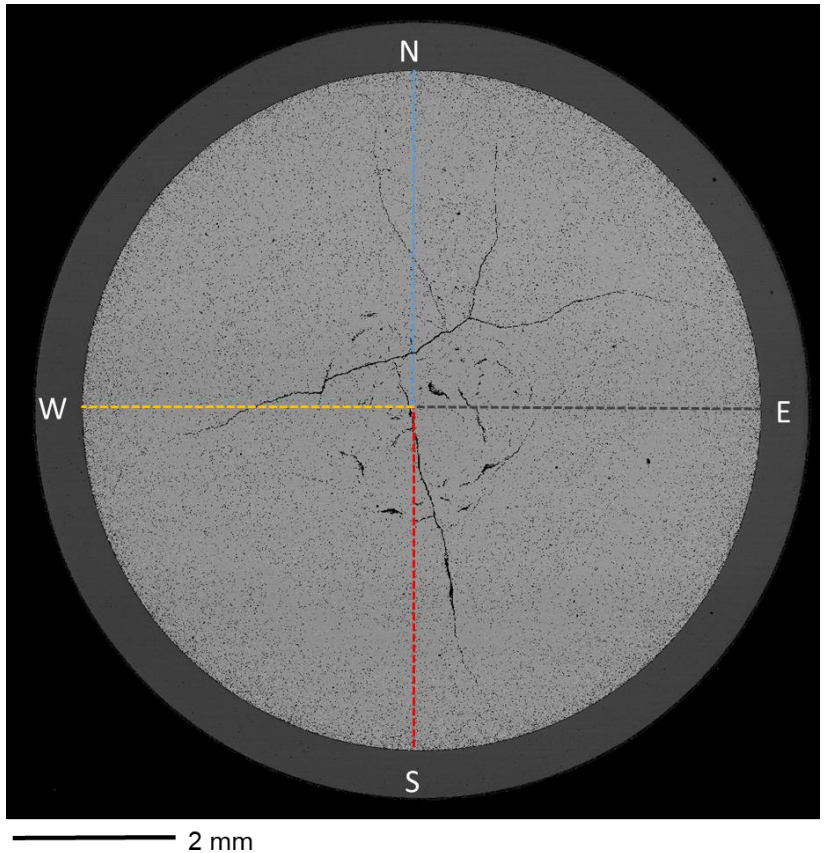


LAGB (2°–15)°
HAGB (15°–65°)

9EU: Zr PWR Type B, UO_2 fuel, ~62.7 GWd/tHM

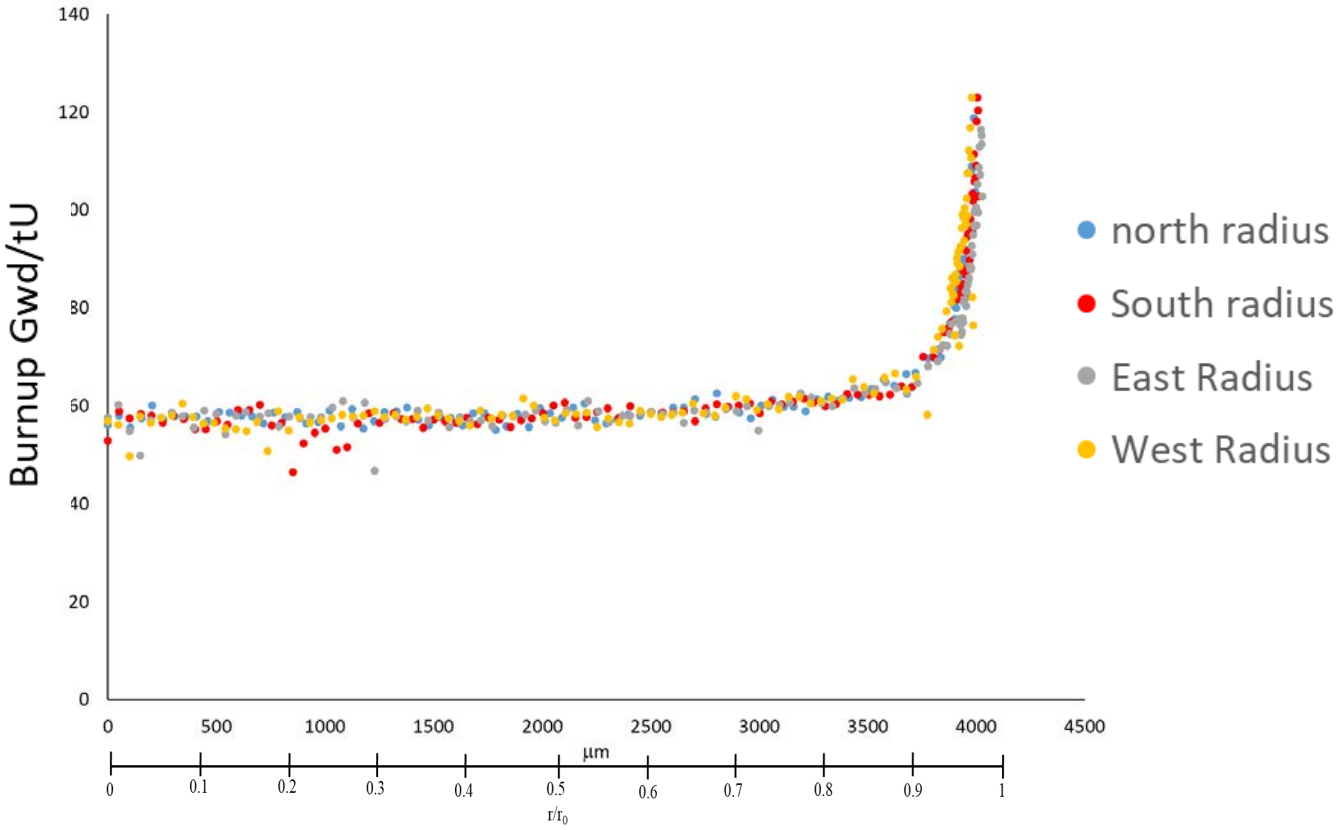
EPMA Results – Radial Burnup (30H)

Quantitative Radial Compositional Analysis Locations



9EU: Zr PWR Type B, UO_2 fuel, ~62.7 GWd/tHM

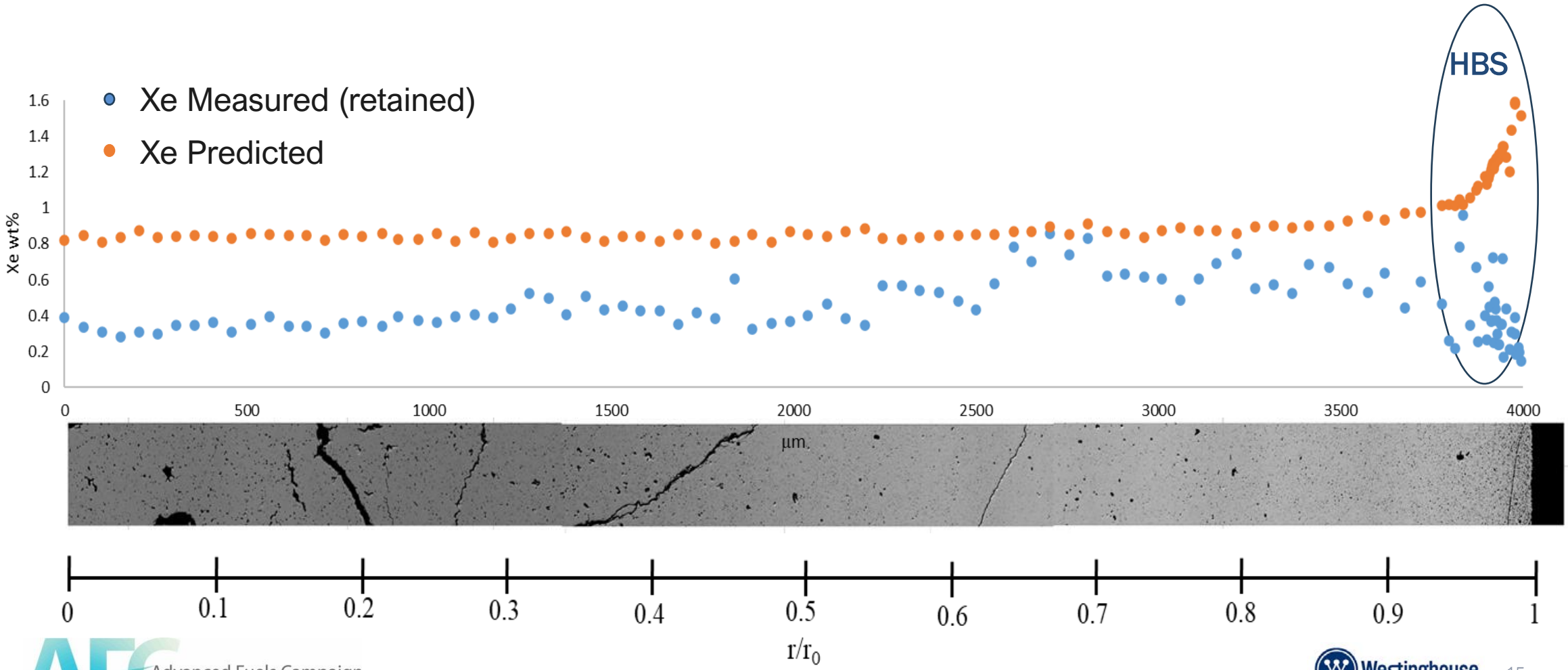
Local Radial Burnup as Calculated from Lassmann et al (1995)



Lassmann K., Walker C.T., van de Laar J., Lindstrom F. (1995), Modelling the high burnup UO2 Structure in LWR Fuel, J Nucl Mater (226) 1-8

EPMA Results – Quantitative Radial Composition (30H)

Significant Xe loss in the HBS



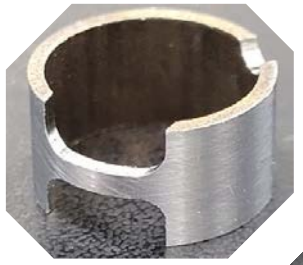
9EU: Zr PWR Type B, UO_2 fuel, ~62.7 GWd/tHM

PIE Results – Mechanical Testing

Mechanical Testing on ATF cladding

- Benchmark ATT tests complete (in-cell and out-of-cell)
  - Includes Zr PWR Type C and Cr-Coated Zr PWR Type C at room and elevated temperature
- 4DS Rod: **6/8 RTTs and 4/4 ATTs complete**
  - From regions in span 9, 5, and 1 of rod 4DS

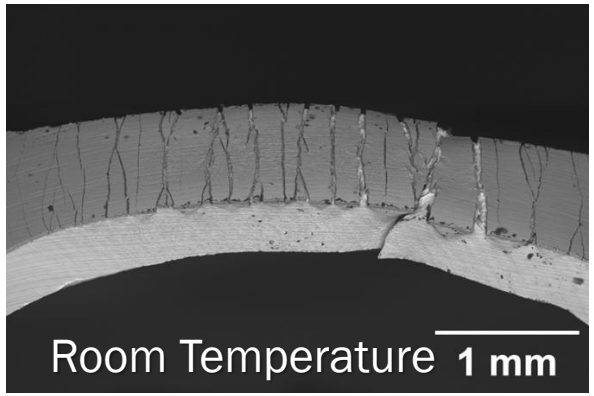
RTT

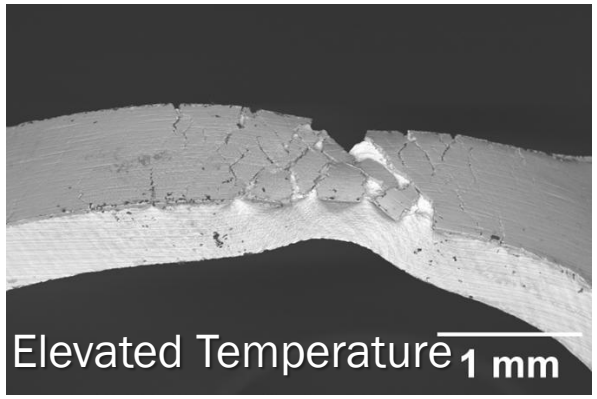


ATT

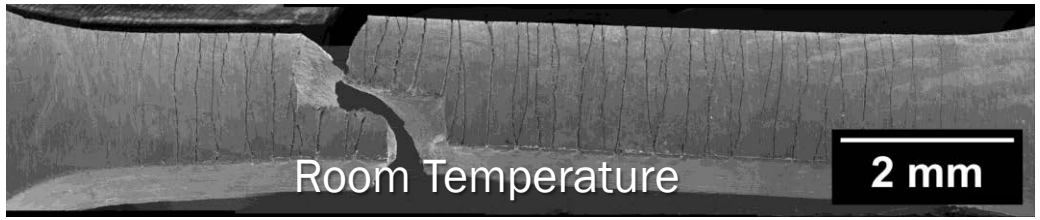
Fractography – Benchmarking Material

RTT





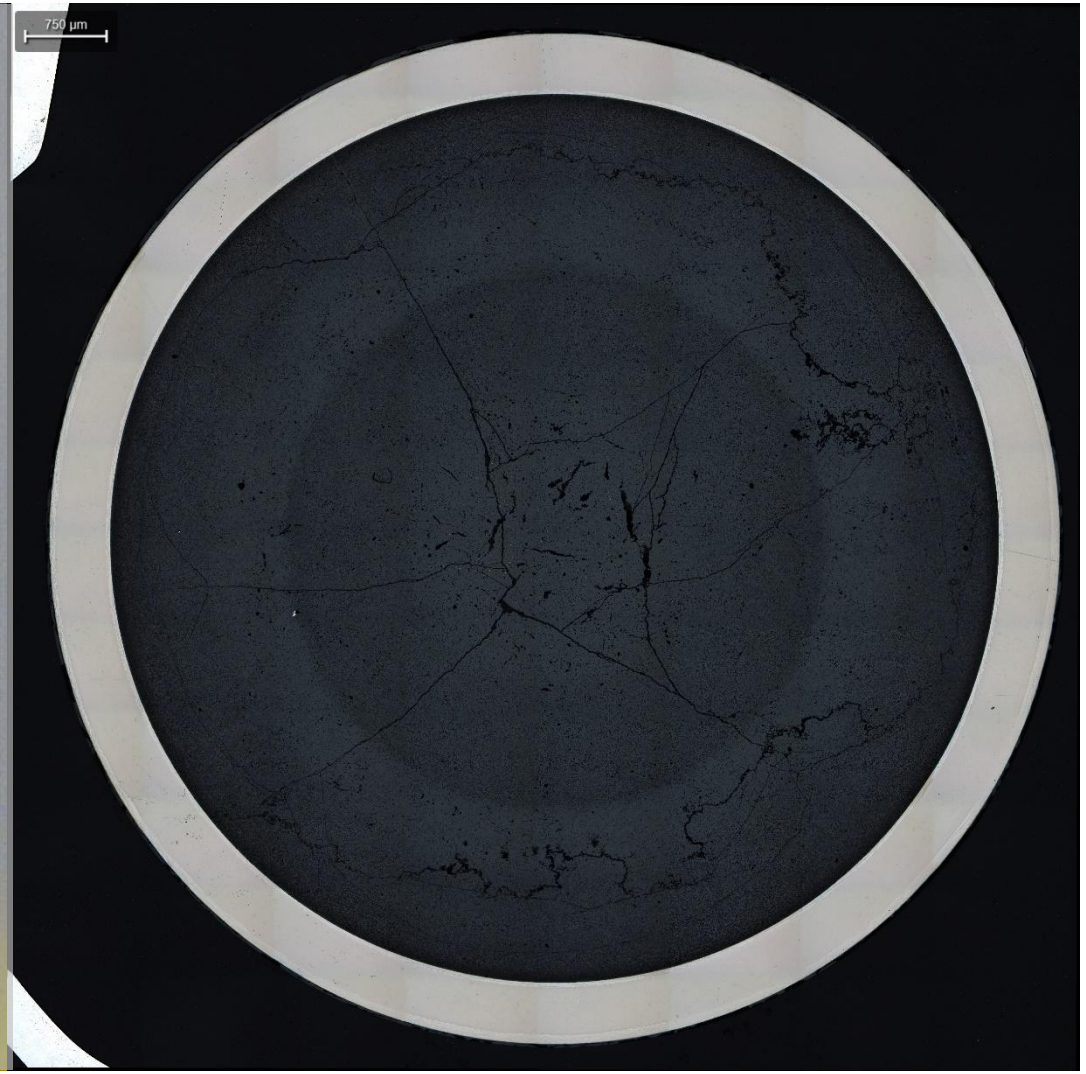
ATT



Byron Shipment

PIE Results

PIE Plans



PIE Plans – Timeline Highlights

- Receive Byron 2.0 shipment
- Advanced characterization HBU/ATF (SEM/EPMA)
- TREAT pre-characterization HERA-HBU-3 and 4

Q1 FY26

- Complete NDE on 2 HBU-ATF (75+GWd/tU) rods of Byron2.0
- TREAT refabrication for HERA-HBU-3 and 4

Q2 FY26

- Start DE of Byron2.0 (sectioning/metallography)
- BRR shipment to ORNL
- HERA HBU-3 TREAT test

Q3 FY26

- Mechanical testing (RTT/ATT) of Q8V
- Byron 2.0 Advanced characterization HBU/ATF (SEM/EPMA)
- HERA-HBU-4 TREAT test

Q4 FY26

Summary

- NDE completed on all 10 Byron rods
- FGR calculated for 6/10 rods
- Hydrogen analysis performed on 3 rods.
  - 4DS (ATF) showed little hydrogen **uptake < 50 wppm hydrogen**
- Advanced characterization underway on ***full fuel cross section***
  - SEM/BSE/EBSD/EPMA
- ***In-cell mechanical testing*** (RTT/ATT) completed on 4DS rod

Acknowledgments

- We would like to thank Westinghouse and Constellation for their support in the BNGS fuel rod shipment
- This research made use of the resources of the High-Performance Computing Center at Idaho National Laboratory, which is supported by the Office of Nuclear Energy of the U.S. Department of Energy and the Nuclear Science User Facilities under Contract No. DE-AC07-05ID14517

Questions?

Clean. **Reliable. Nuclear.**

AFC Advanced Fuels Campaign

General LTR Post-Irradiation Examination at ORNL – 2025 ESCP Meeting

Jason M. Harp, Peter Doyle, Yong Yan, Casey L. McKinney, Caleb P. Massey, Nathan A. Capps

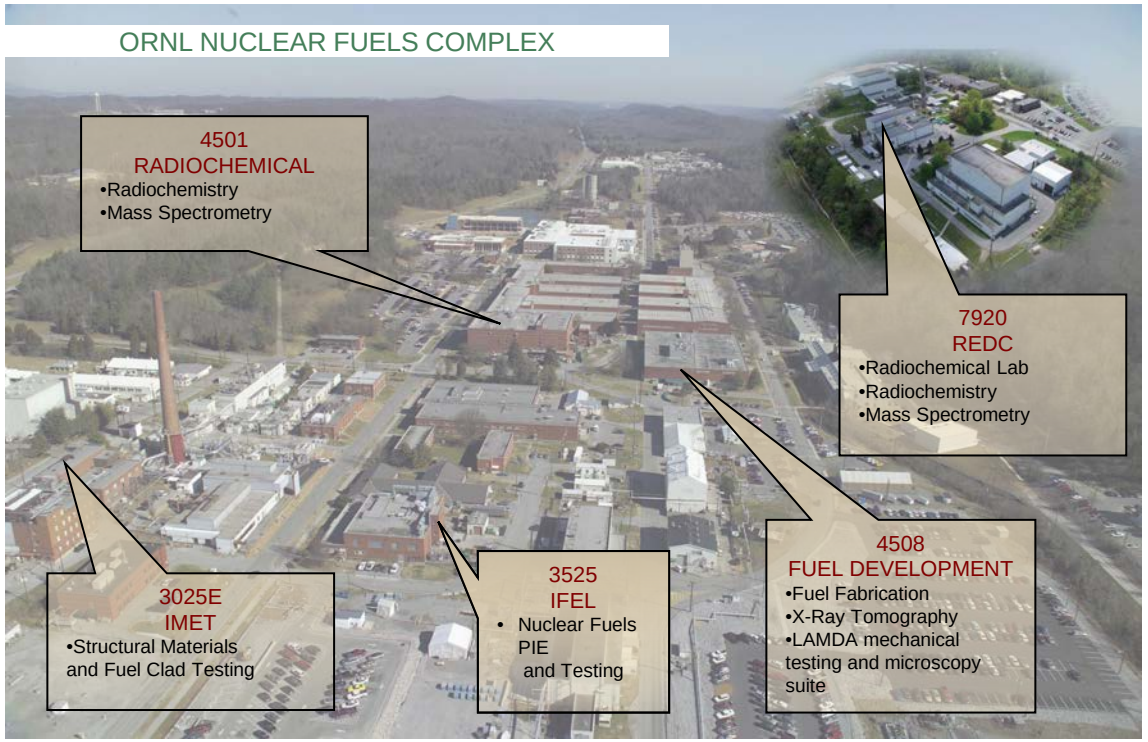
Nuclear Fuels Development Section

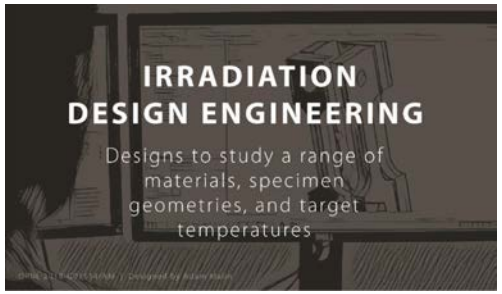
Oak Ridge National Laboratory

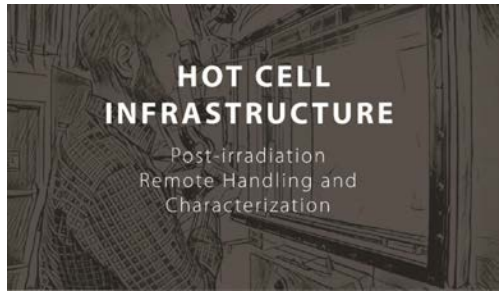
ORNL is managed by UT-Battelle LLC for the US Department of Energy

ORNL Nuclear Fuels Complex

Capabilities to meet HFIR irradiation, commercial fuel and radioisotope production needs



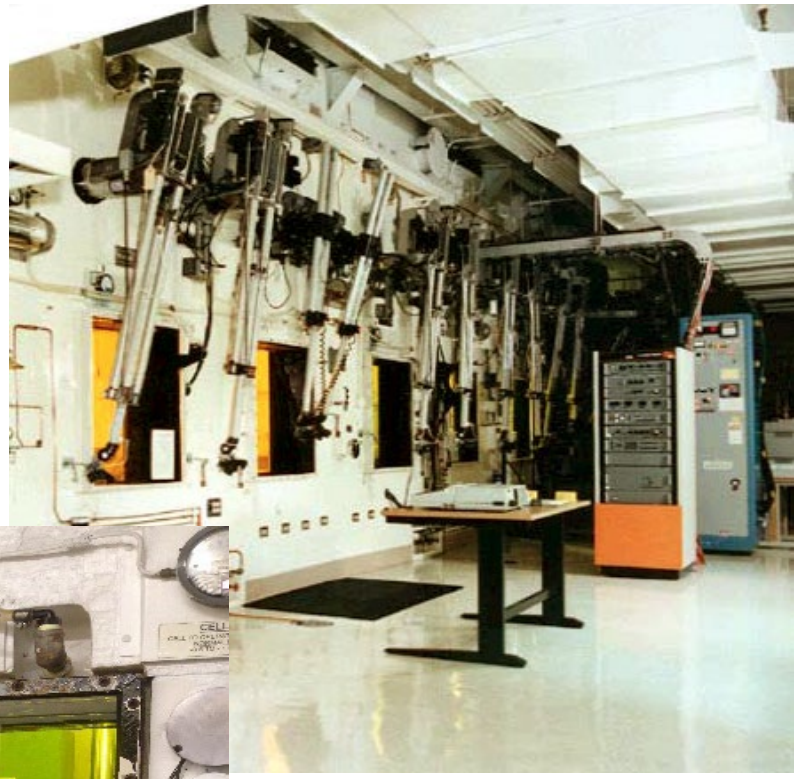


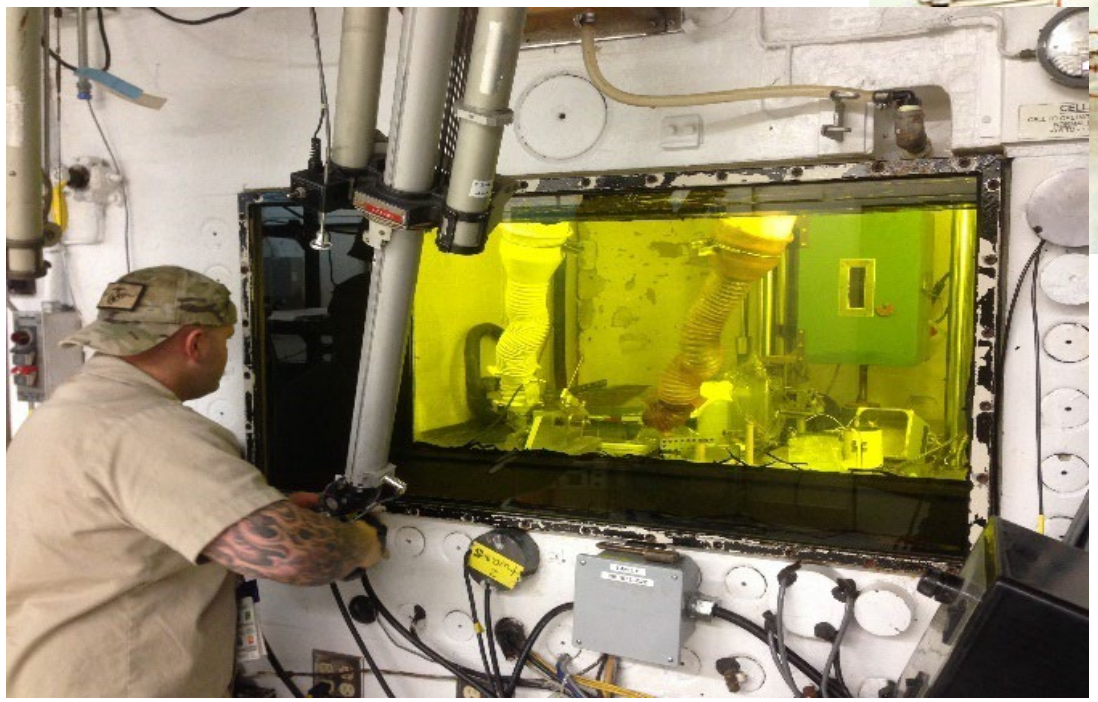


Facilities for Structural Material PIE

Irradiated Materials Examination and Testing (IMET) – 3025E

- Six interconnected steel-lined examination cells containing 30 m^2 of workspace.
- Low alpha contamination facility (<70 dpm / 100 cm^2).
- Low radiation background allows for ease of operation enabling cell entries for equipment maintenance and use of advanced tools (non-contact extensometry, in-cell SEM)
- Capabilities
  - Mechanical Load Frame Testing
  - Charpy Testing
  - Microhardness
  - In-cell SEM fractography
  - Profilometry
  - HFIR Rabbit disassembly and shipment of specimens to LAMDA

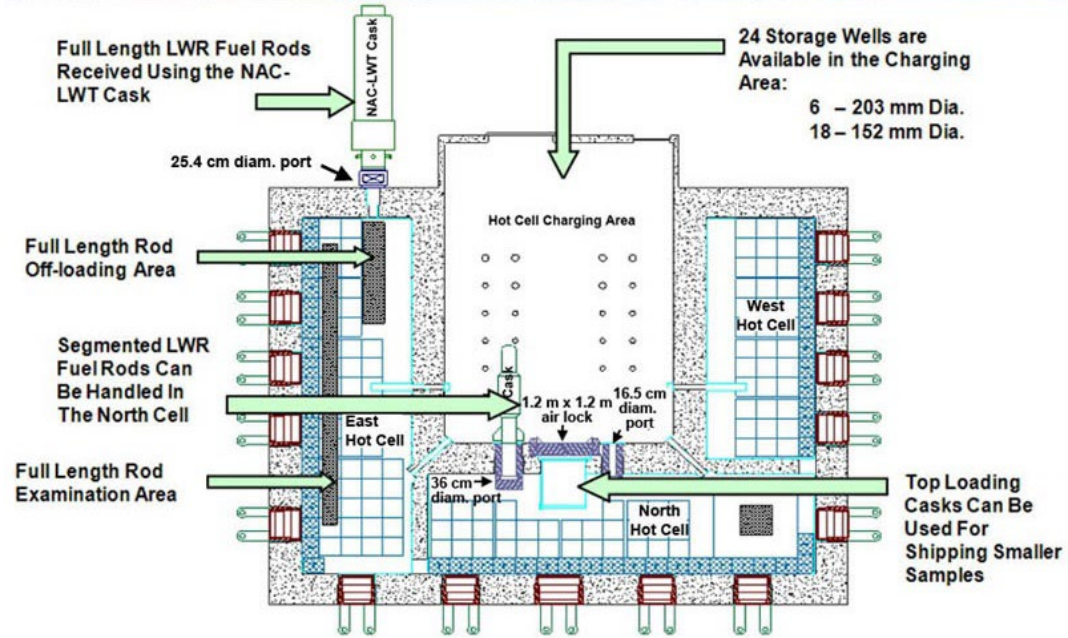




Facility for PIE of Fuel

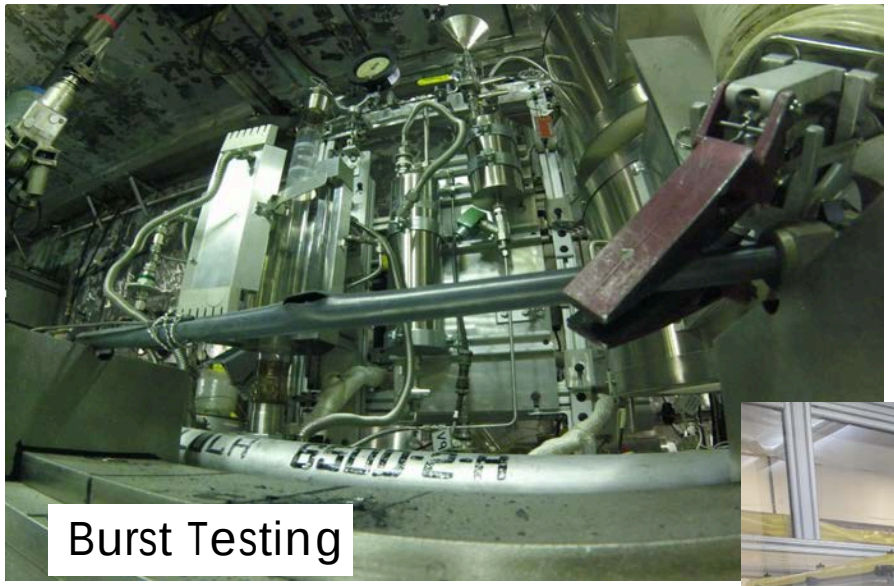
Irradiated Fuels Examination Laboratory (IFEL) - 3525

- U-shaped array of cells (East, West and North cell work areas)
- Glove box and contamination zone areas available
- Additional stand alone shielded cells for more sensitive set-up of experimental work
- Flexible and diverse cask handling (horizontal and vertical loading) including the NAC-LWT for full length commercial rods



Scope of ORNL PIE for ATF and High Burnup LWR LTR's

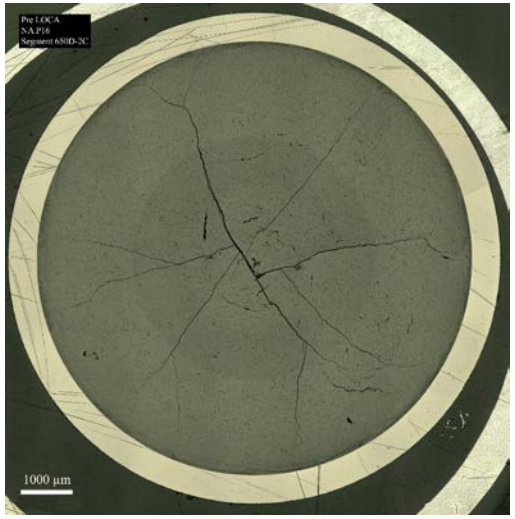
- Basic examinations on all rods
  - Visual, profilometry, gamma scanning
  - Fission gas release, basic optical microscopy, mechanical testing, chemical burnup analysis, ONH analysis
- Burnup extension enabling PIE
  - LOCA Burst Testing
    - Includes a great deal of pre and post test microscopy and mechanical testing
    - **This capability can be applied to spent fuel pool LOCA and other Used Fuel Applications**
  - High Temperature Creep testing
  - Fatigue Testing
  - **Transient Fission Gas Release**

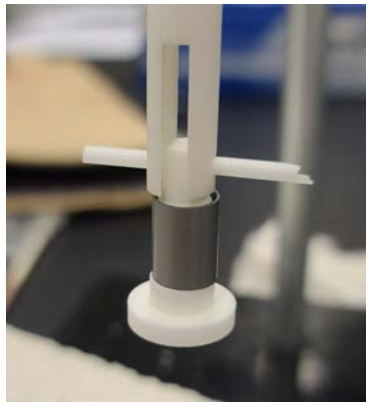


Burst Testing

O, N, H Testing

Microscopy





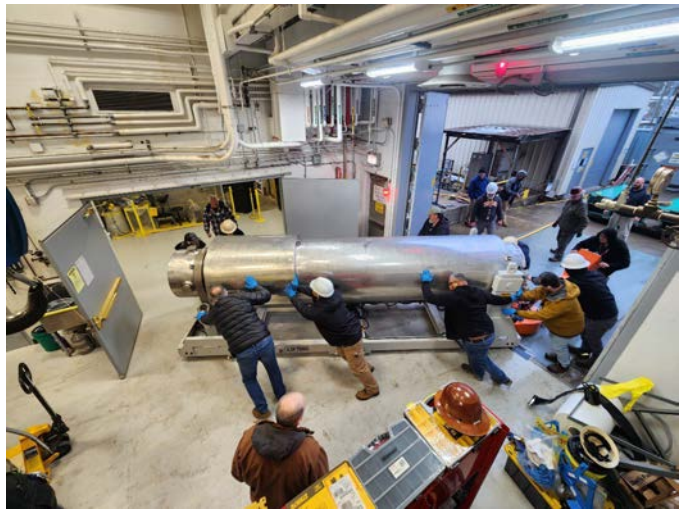
High Temperature Oxidation Testing

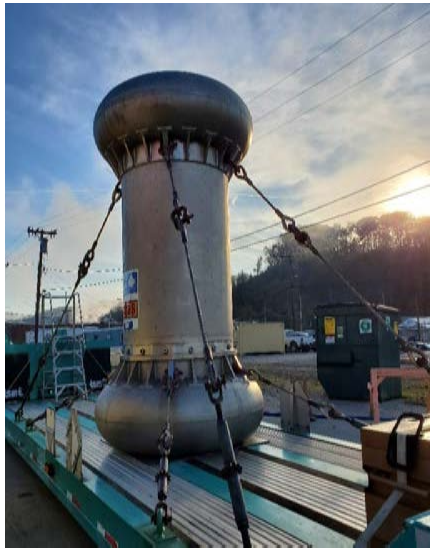
Mechanical Testing (RCT example)

Receipt of Lead Test Rods

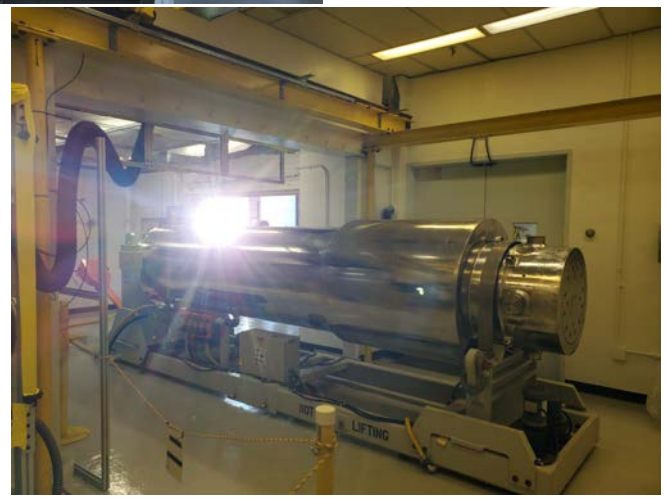
- ATF and high burnup LTR's supporting Westinghouse
  - Arrived at ORNL June 15, 2021
- NAC LWT cask Shipments
  - June 2021 – Byron, Westinghouse
  - January 2024 – Clinton, GE
- GE-2000 receipt
  - November 2020 – Hatch, GE
- ATF Concepts
  - Cr-coated Zr alloy cladding
  - Doped Fuel
  - FeCrAl
- High Burnup LTR's
  - Support Burnup Extension
  - Study FFRD and related phenomena

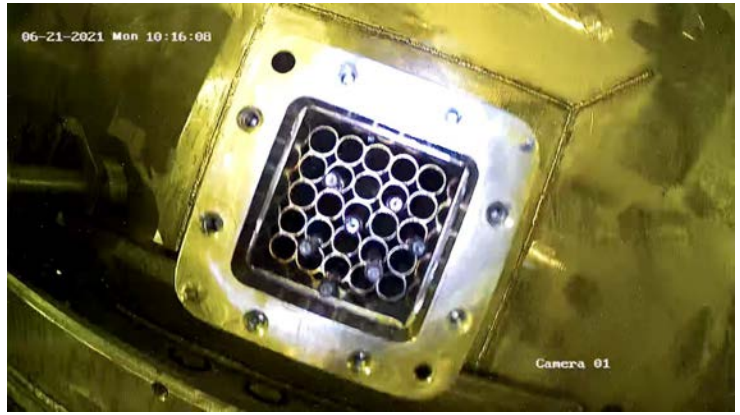
NAC Receipt





GE-2000 Receipt





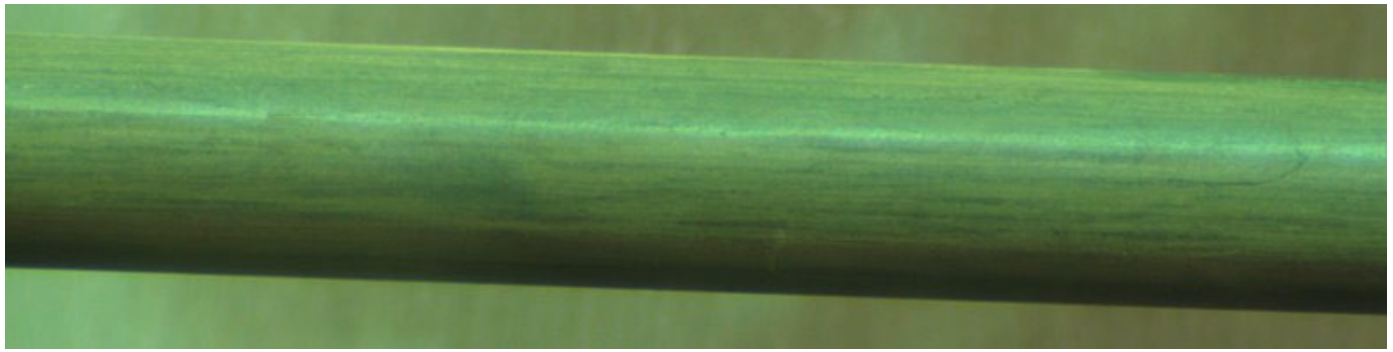
Westinghouse ATF & High Burnup LTR's ready for unloading into the ORNL Irradiated Fuel Examination Laboratory hot-cell

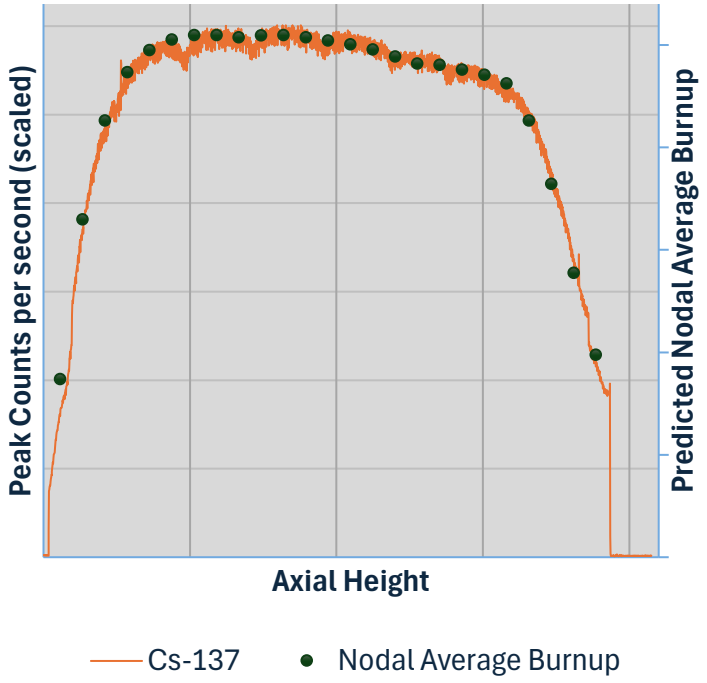
OAK RIDGE National Laboratory

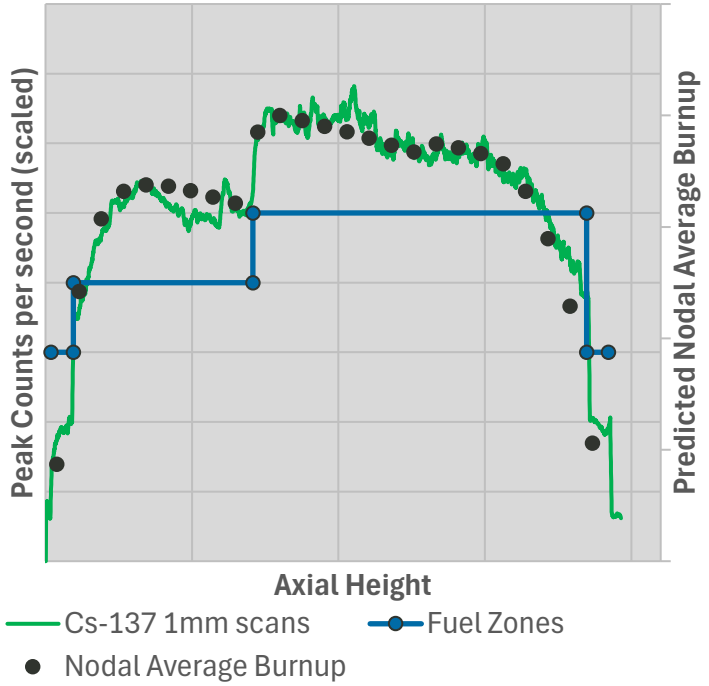
Highlights from BWR PIE to date

- Visual, Gamma scan complete
- Dimensional complete by two methods
- Fission Gas Release complete
- FSECT Completed October 2025
- Microscopy, Hydrogen pick-up, and possibly Burst Testing to follow

Surface of an Examined Rod

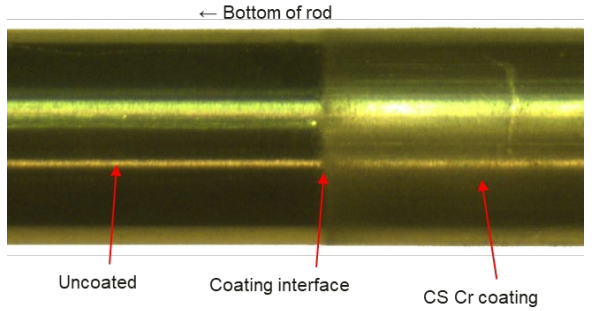


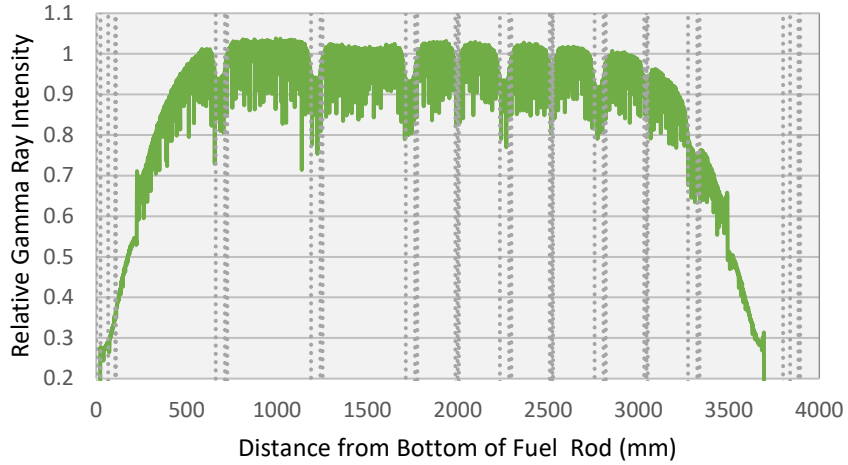


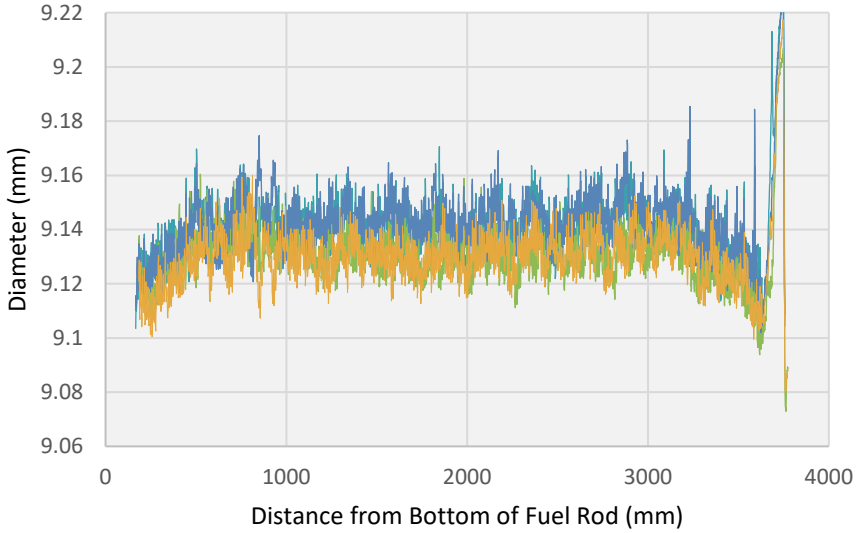


Highlights from ATF and High Burnup

- NDE Complete
- FGR Complete
  - See the TopFuel Paper www.osti.gov/biblio/2472714 for a NDE and FGR summary
- Multiple in-cell SATS tests performed and planned
  - LOCA ramp tests on ATF (3) and High Burnup (3)
  - Creep test (2)
  - Axial Gas Communication
  - High Ramp Rate tests coming soon
- Optical Microscopy largely complete for 2 of 7 rods
- Hydrogen uptake complete for first ATF rod
- Mechanical Testing initiated







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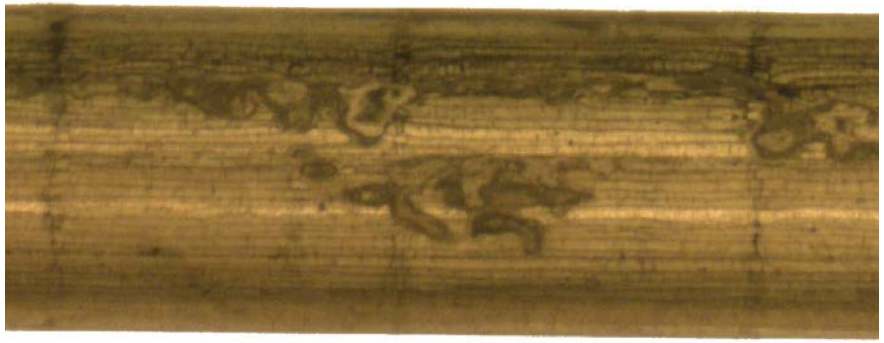
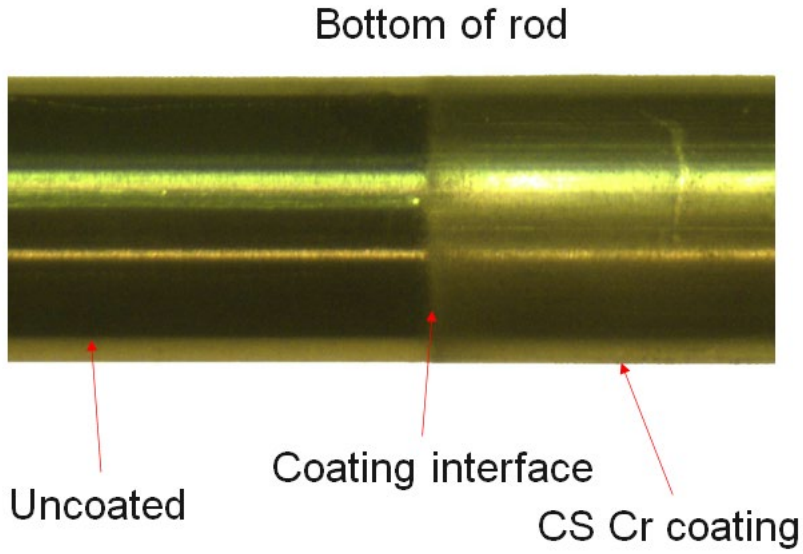
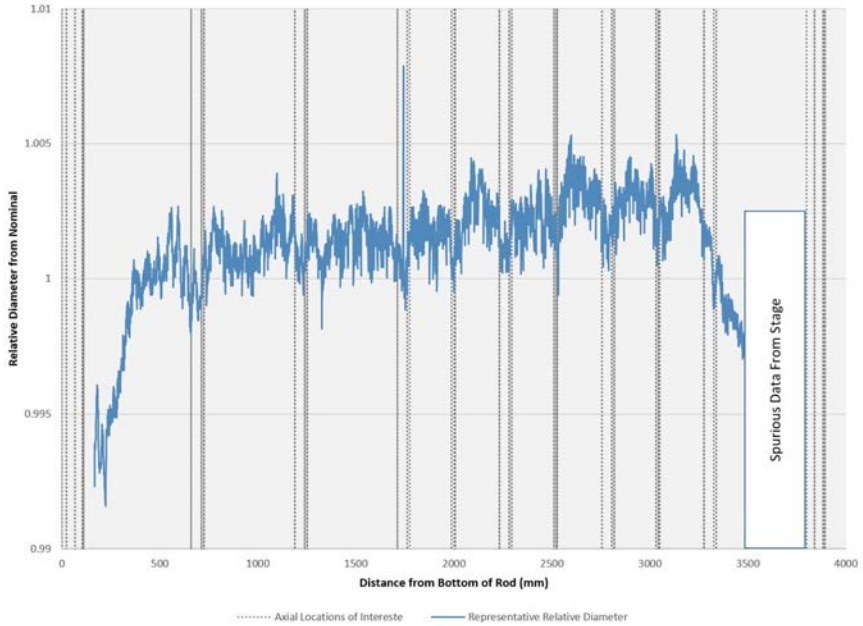
NDE of ATF and High Burnup Rods

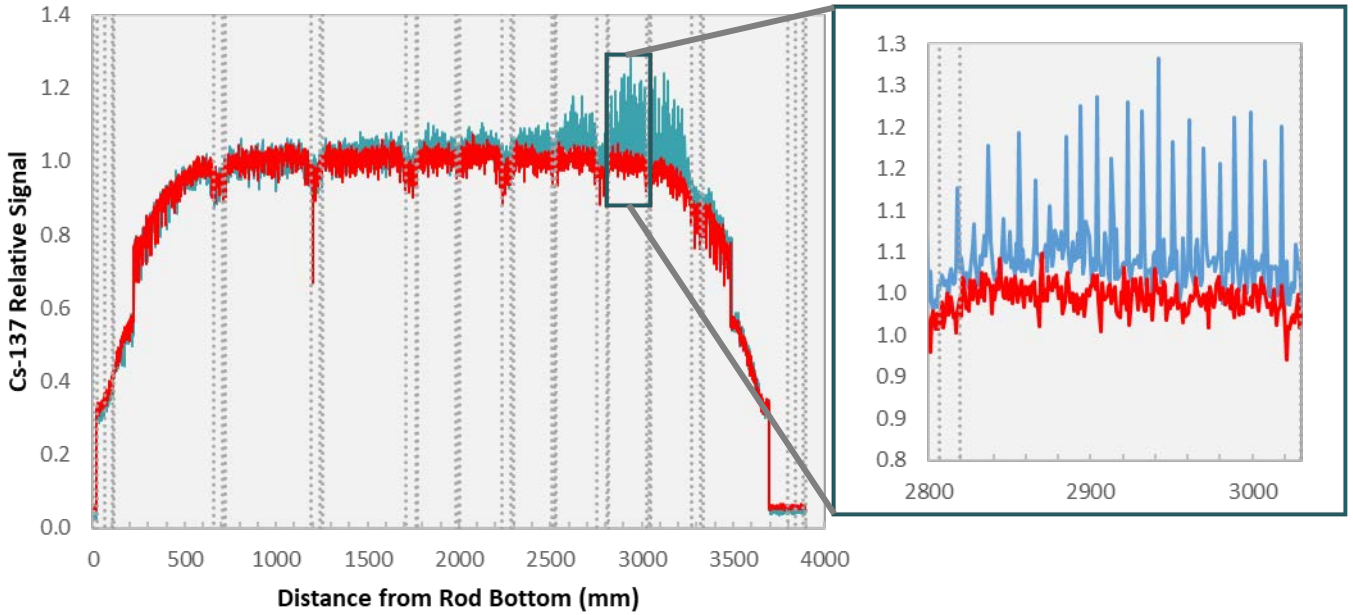


Visual Exam from a high burnup rod





Relative Diameter from a High Burnup Rod

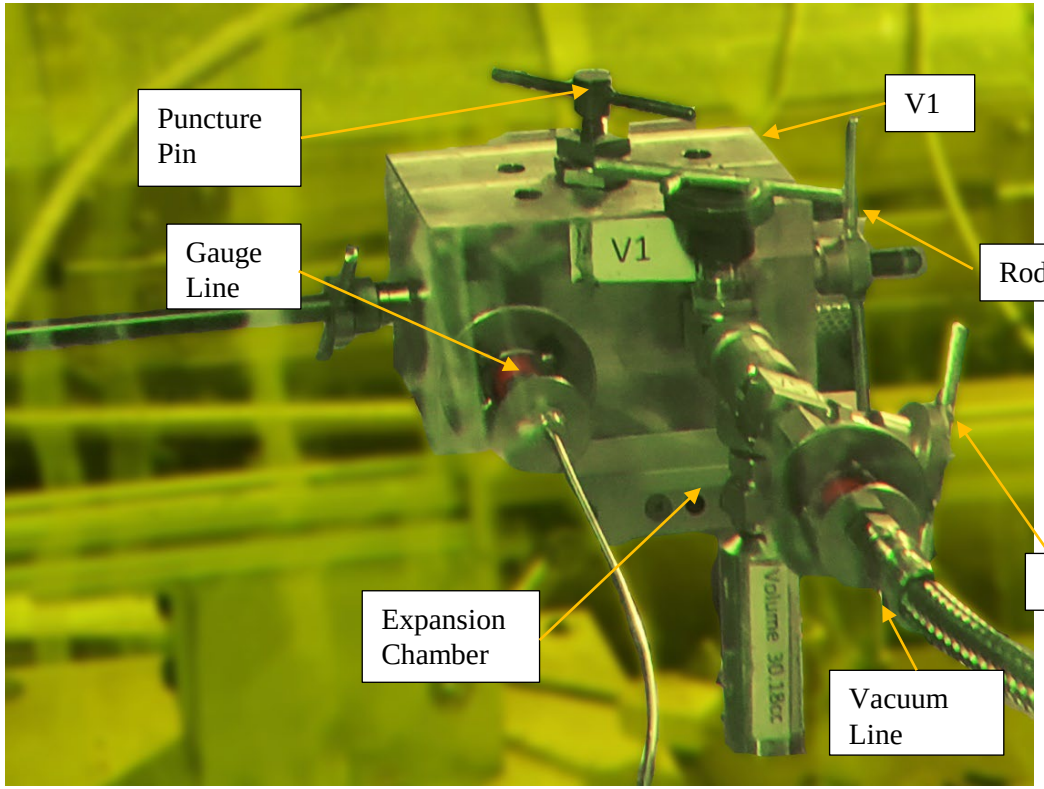


Gamma Scan from High Burnup Rods with different Irradiation Histories

OAK RIDGE National Laboratory

Fission Gas Release

Rods are punctured in the plenum region



A series of backfills are used to calculate the pin free volume

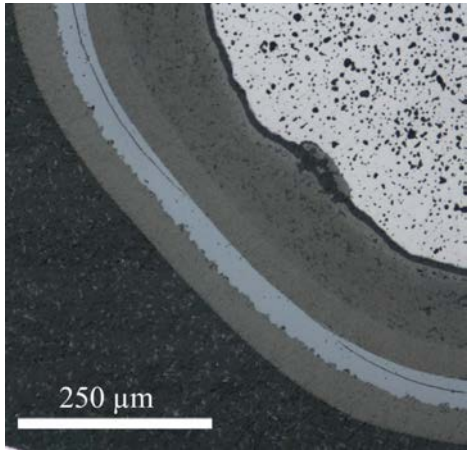
Gas is evaluated for He, Ar, Kr, and Xe content.

Transmission tests through the entire rod have also been performed.

Gas from the puncture is expanded into several known volumes and collected in a sample bottle

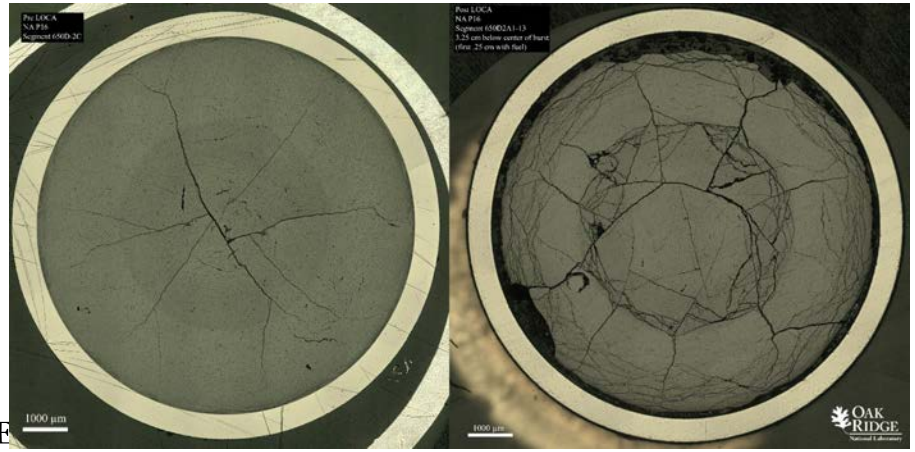
Optical Microscopy Examples

UN TRISO Optical Microscopy

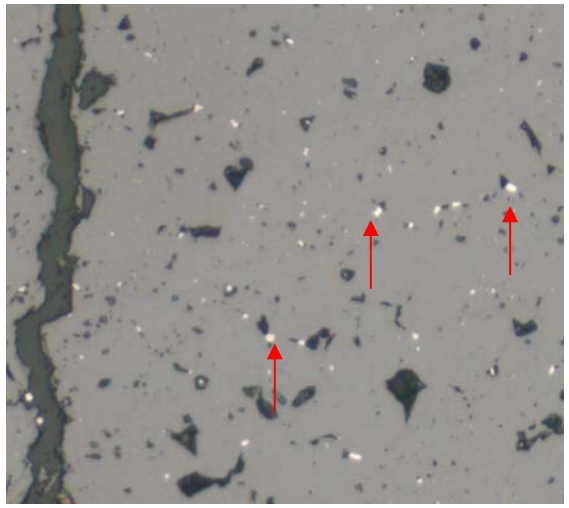


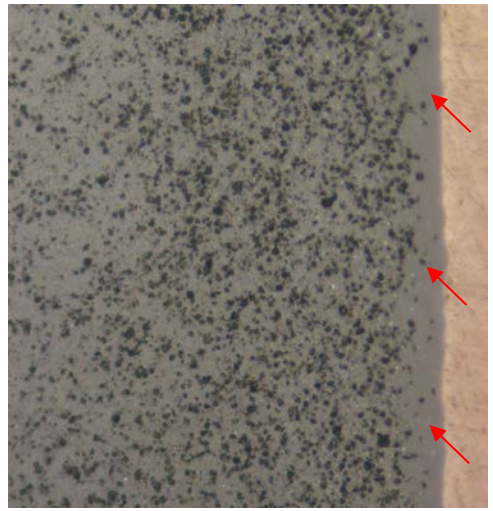
Hydrides in Zirconium Based cladding

As-Irradiated and Post-LOCA



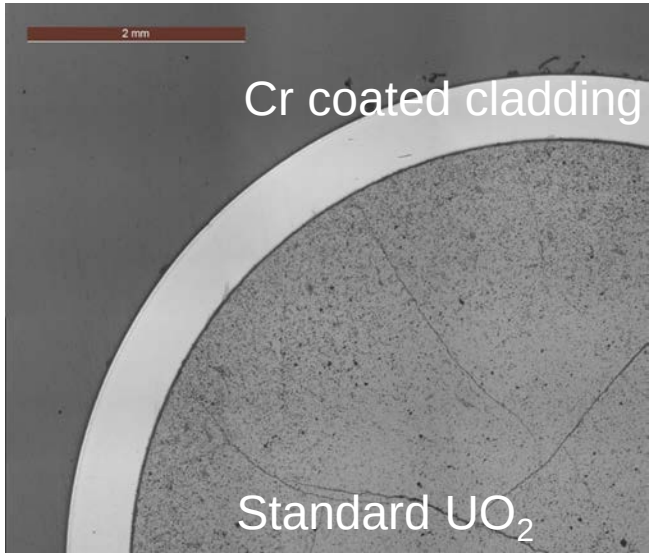
Precipitates and FCCI

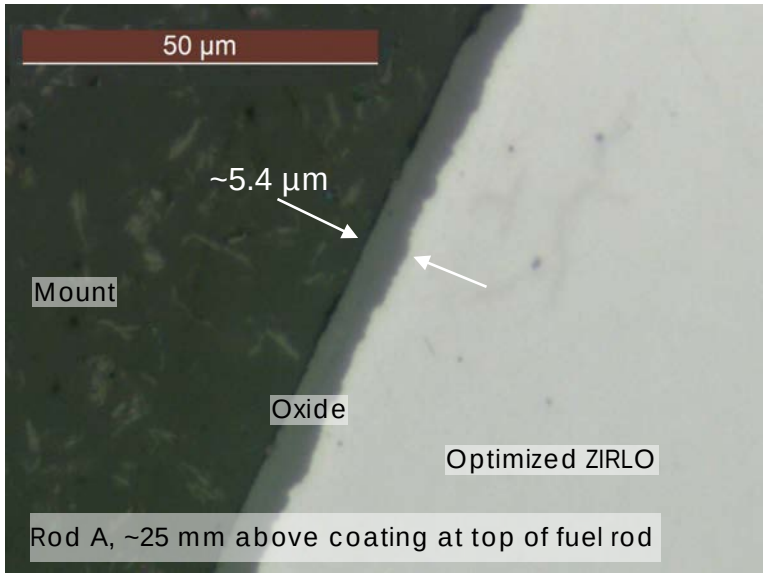


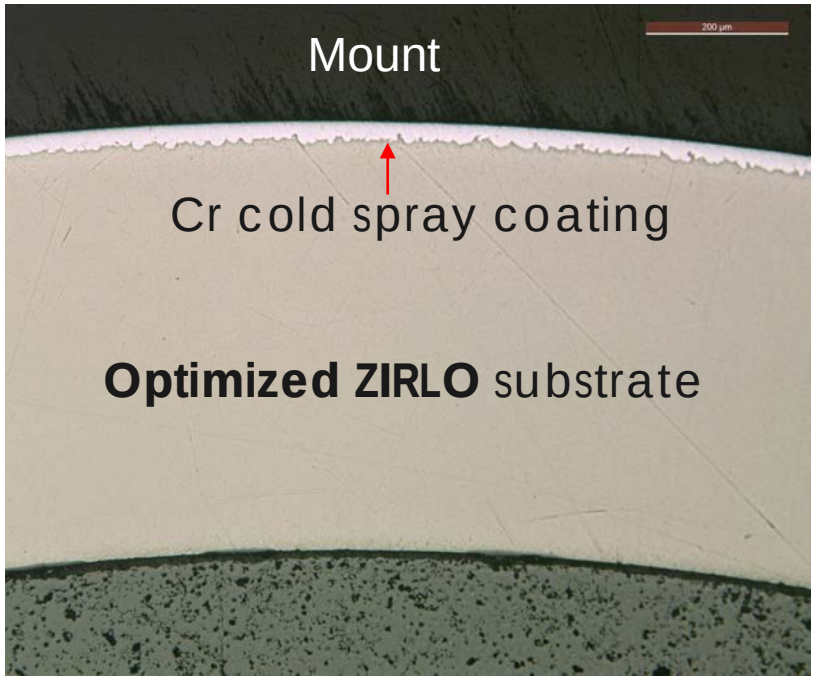


Optical Microscopy of Cr Coated Zr Alloy

- Images of Cr Coated, UO_2 fueled rod at ~31 GWD/MTU
- Fuel to cladding bonding appears to be incomplete
- Oxide on uncoated matches expectations
- No cracking observed, and coating integrity





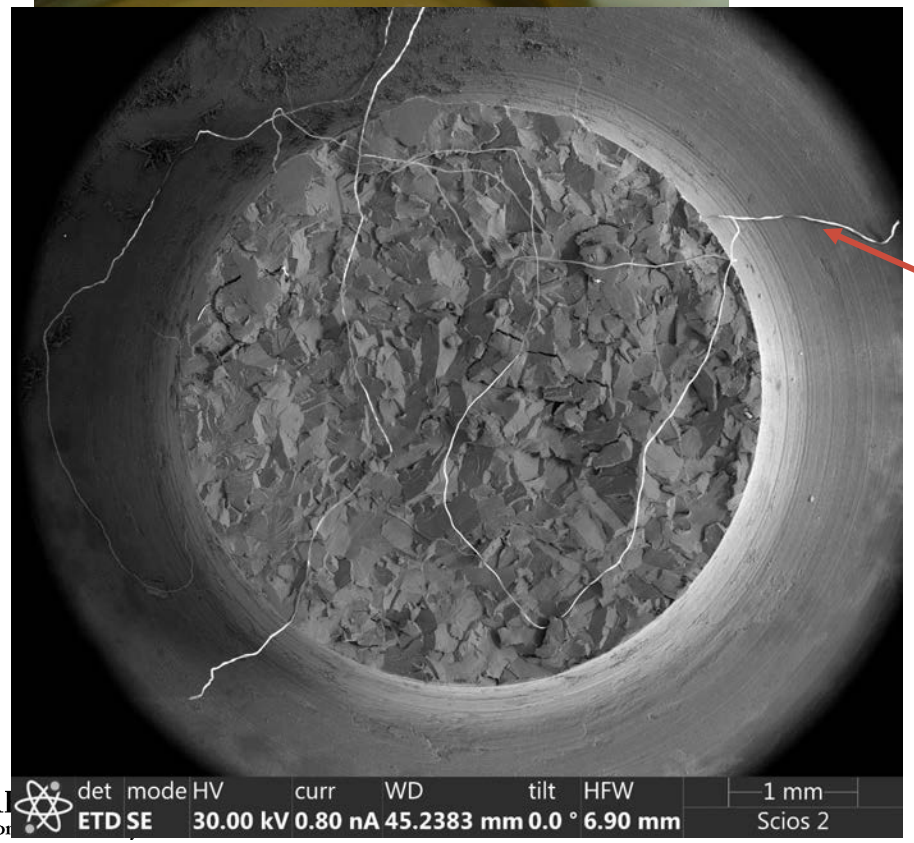


Hydride Pick-up and Etching

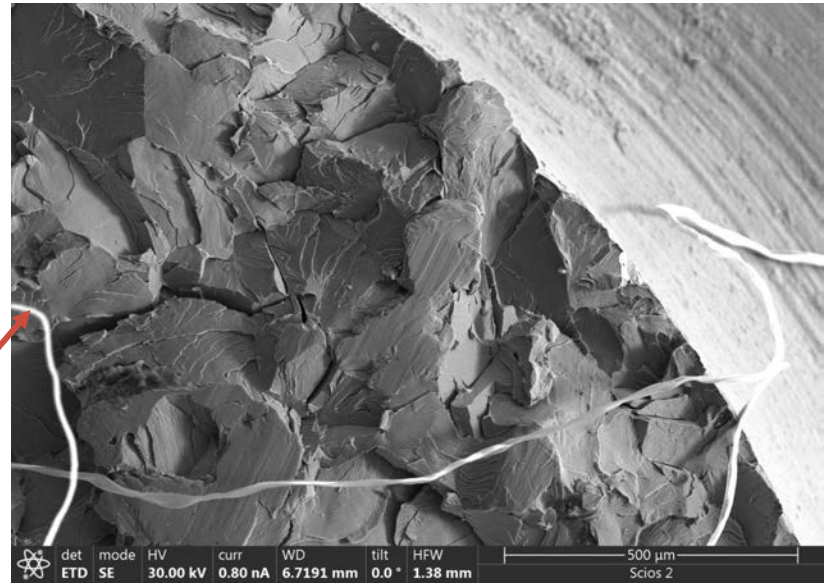
- Understanding Hydrides in Zr is critical for understanding cladding performance
- Hydrides are revealed by etching cladding with a combination of nitric acid, hydrofluoric acid, and hydrogen peroxide
- Hydrogen pick-up is also evaluated by inert gas fussion destructive testing
- Hydrogen pick-up in microscopy region is very low: HPU 11 ± 1 ppm

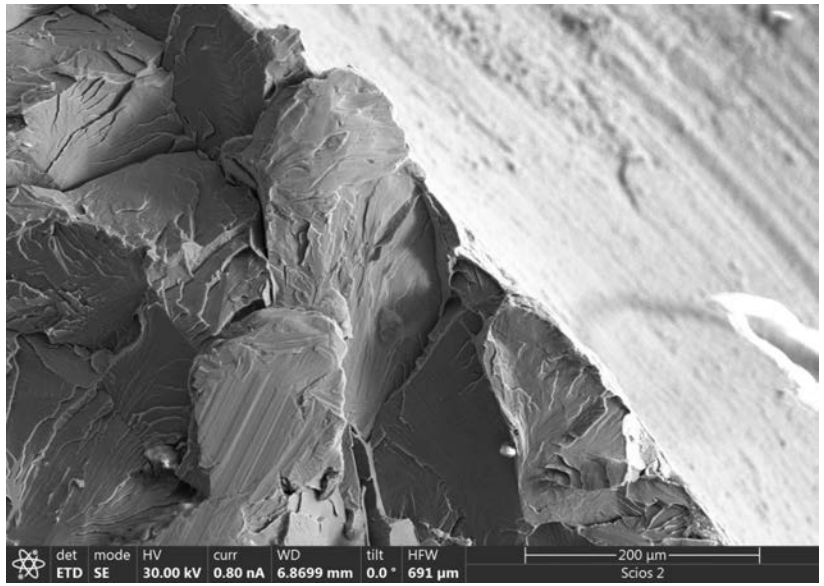
O, N, H Testing
Leco Model 836ONH

SEM Fractography



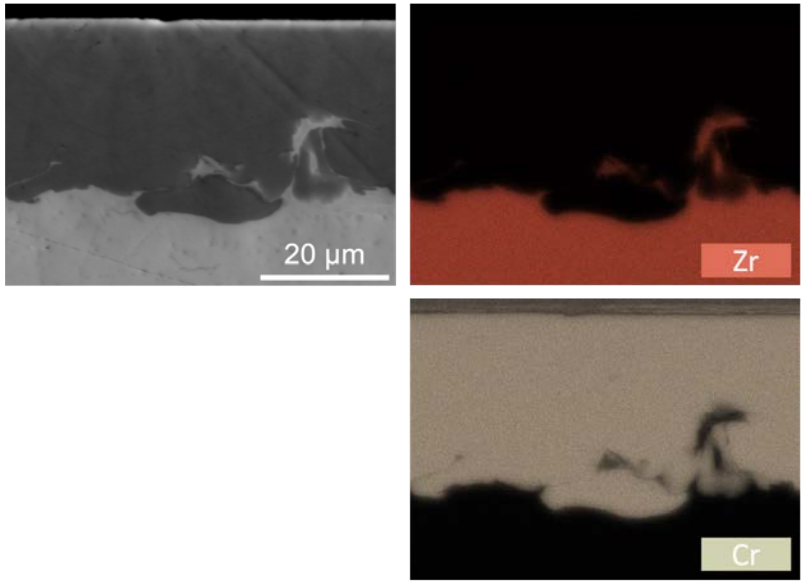
Fibers from decontamination on surface

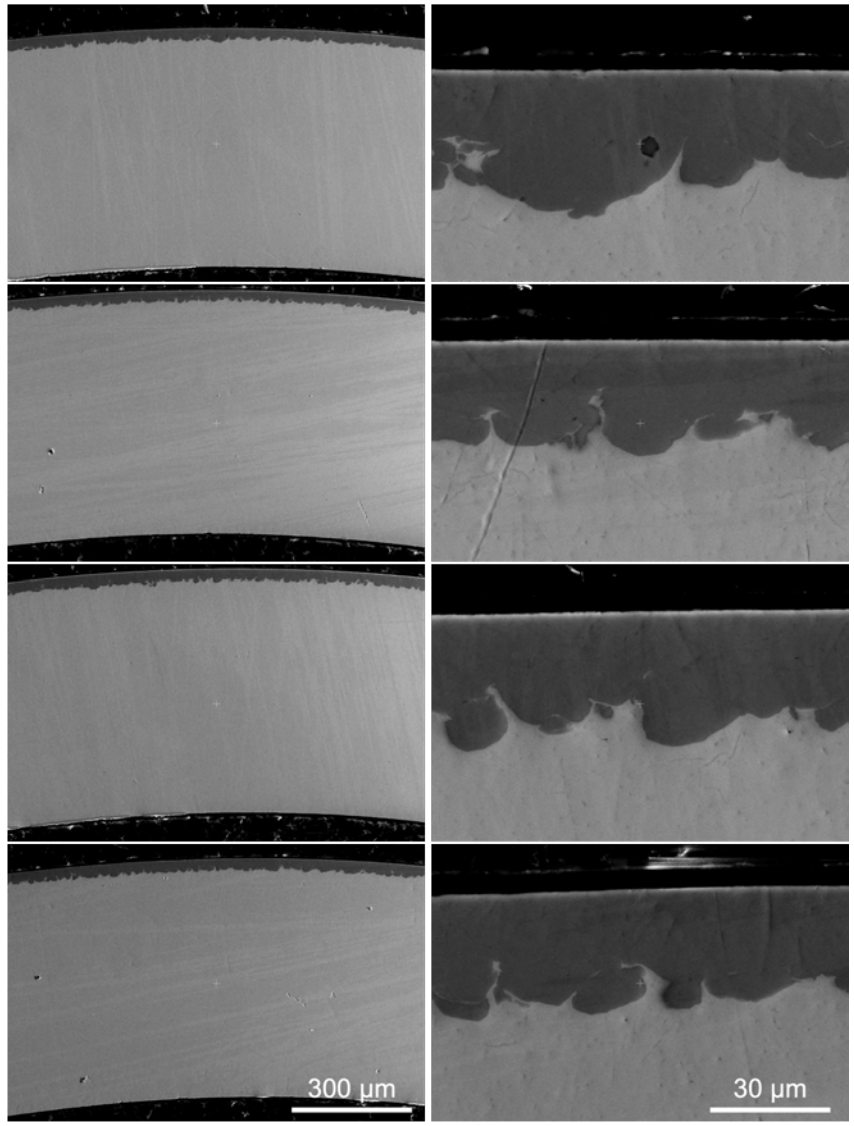




Scanning Electron Microscopy on Cr-Coated Zirlo

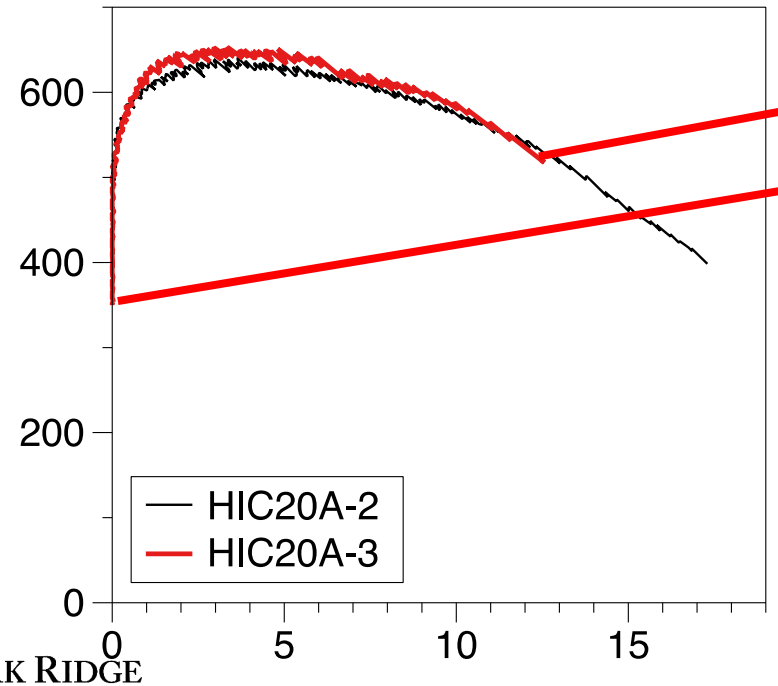
- Coating is well adhered
- No chemical interaction between Cr and Zr observed



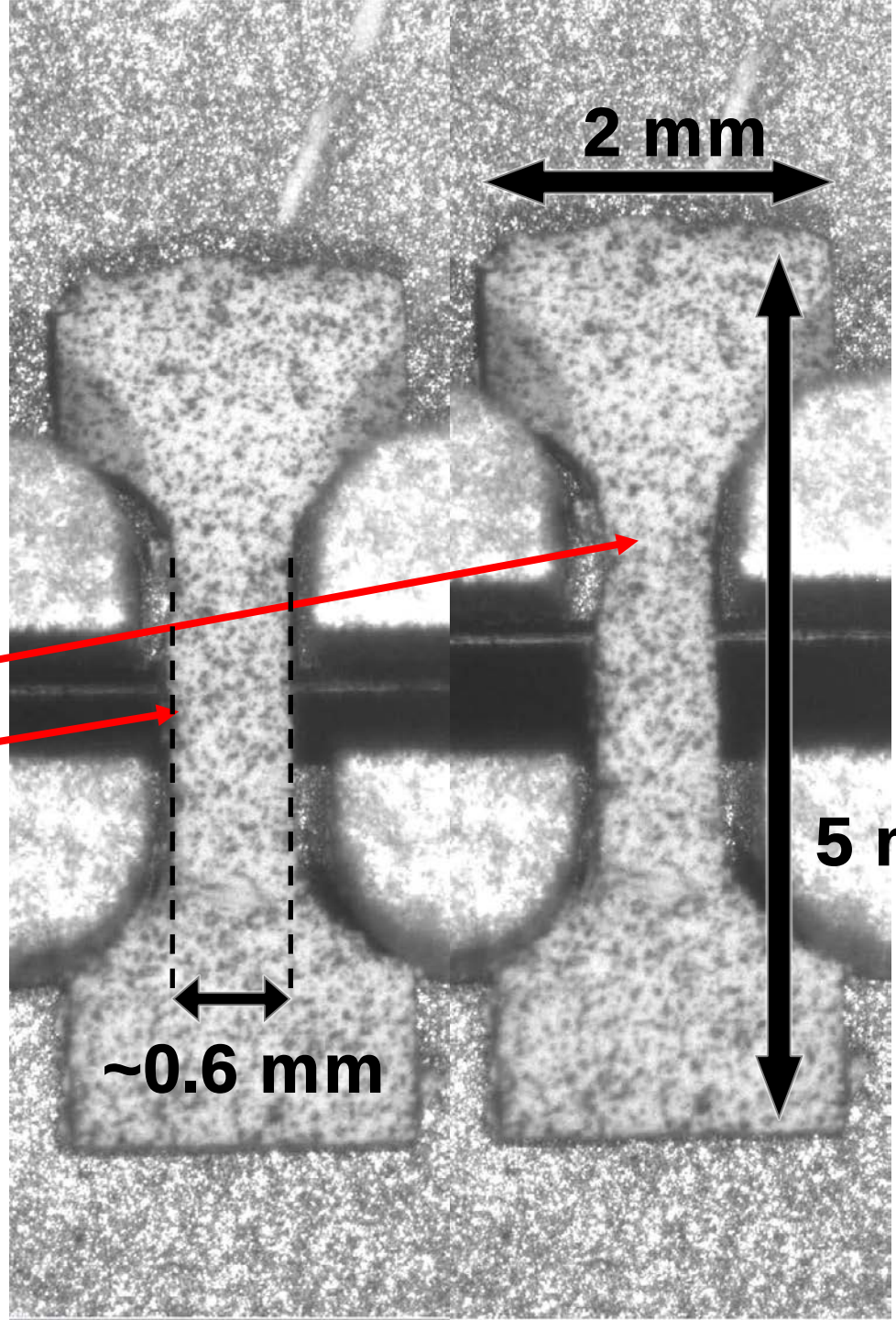


Mechanical tests of Sub-sized Samples

- Miniature tensile tests from the end plug shows some retained ductility
- DIC also available



FeCrAl (End Plug)
~3% Uniform Elongation
12-17% total Elongation
UTS ~650 MPa



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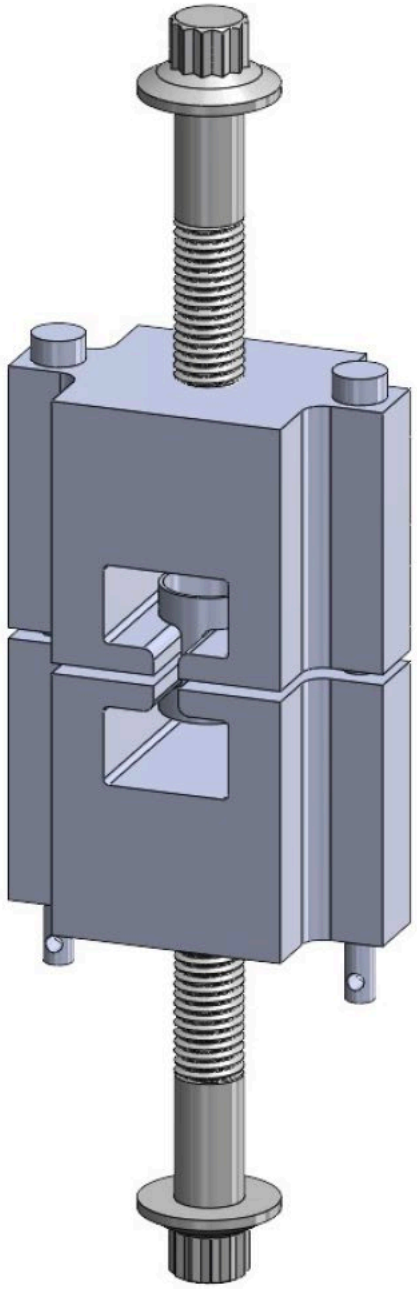
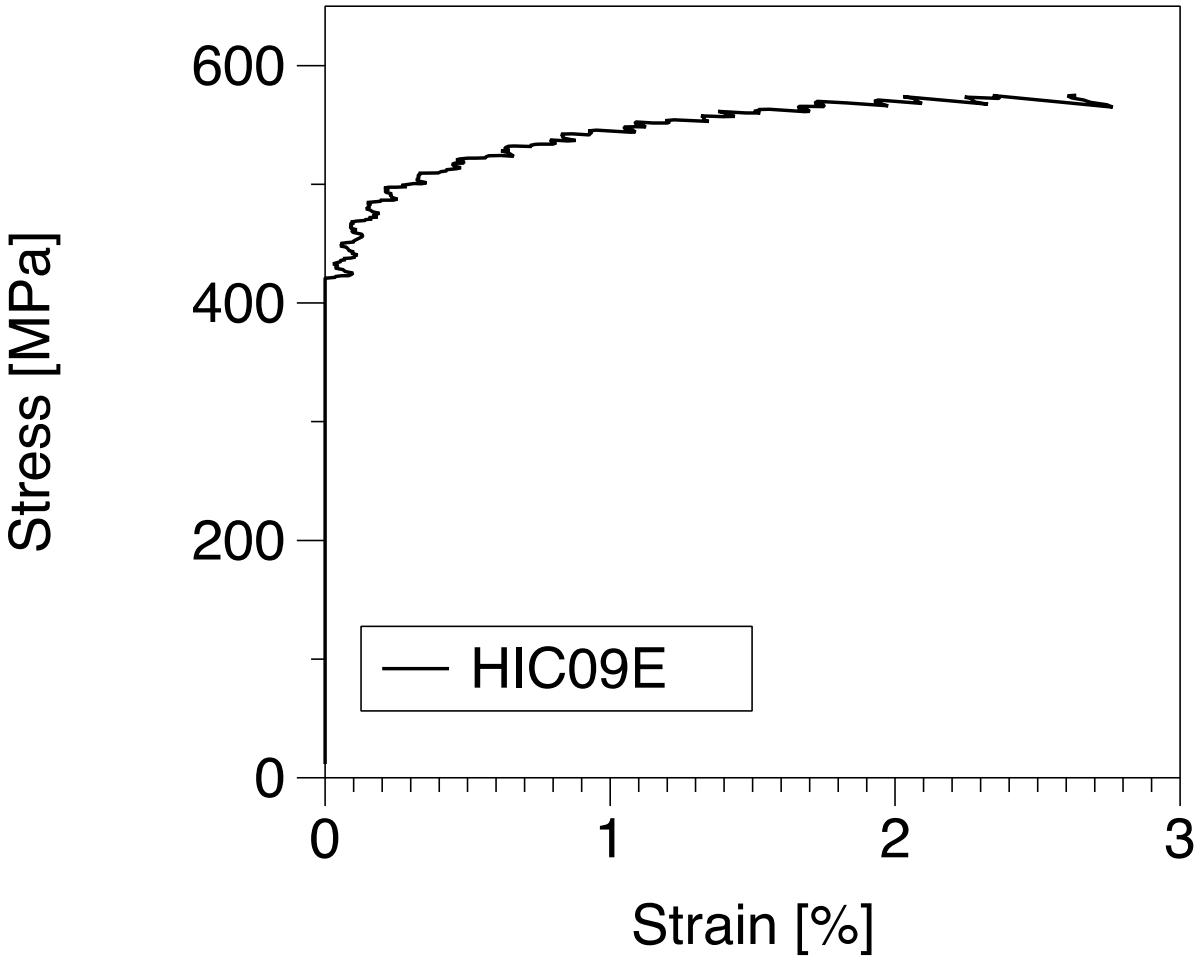
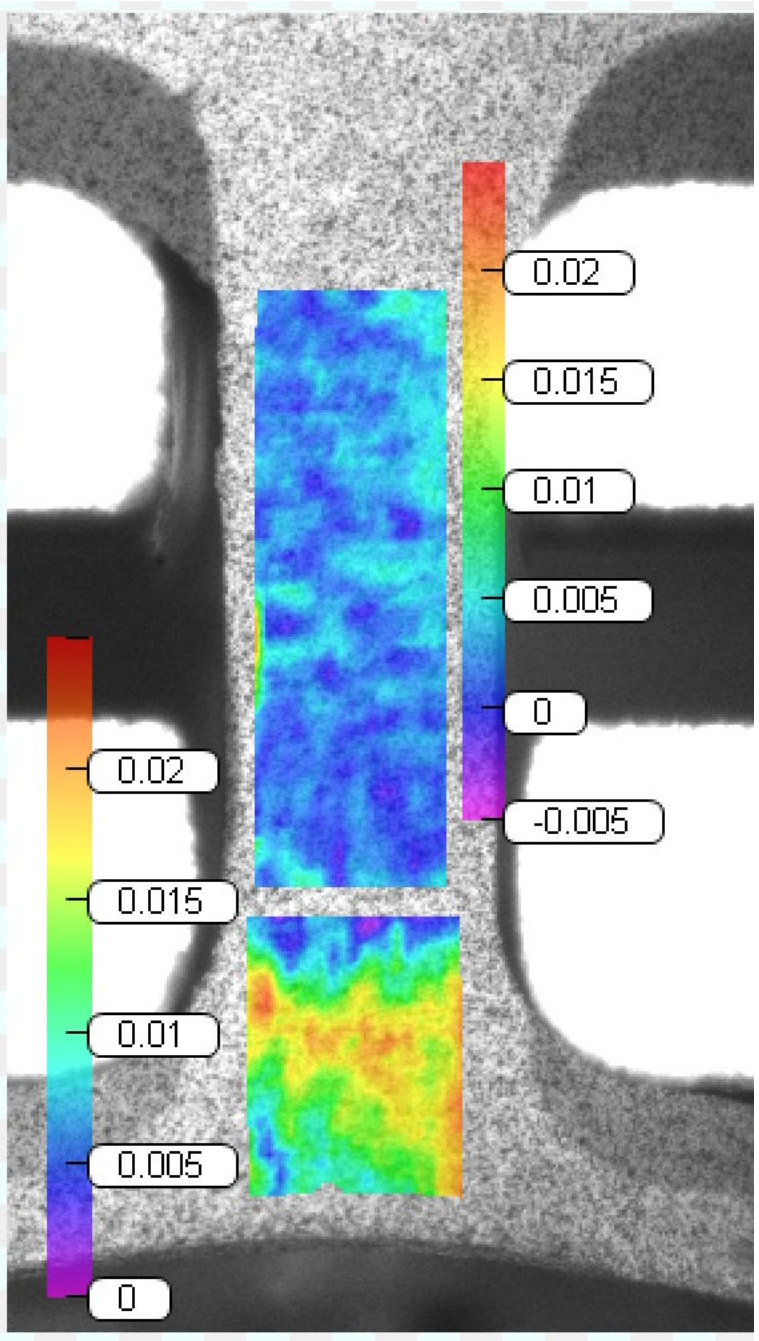
Mechanical Testing Example - FeCrAl Axial Tube Tensile Specimen Testing

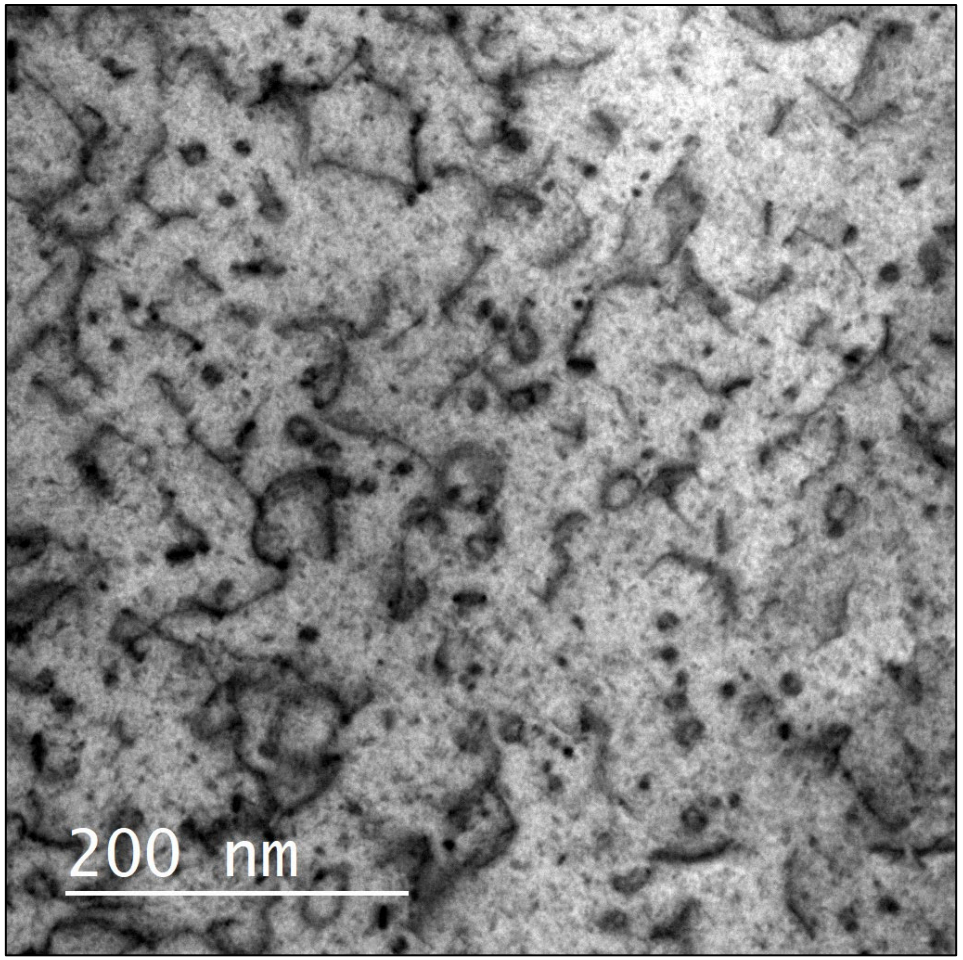


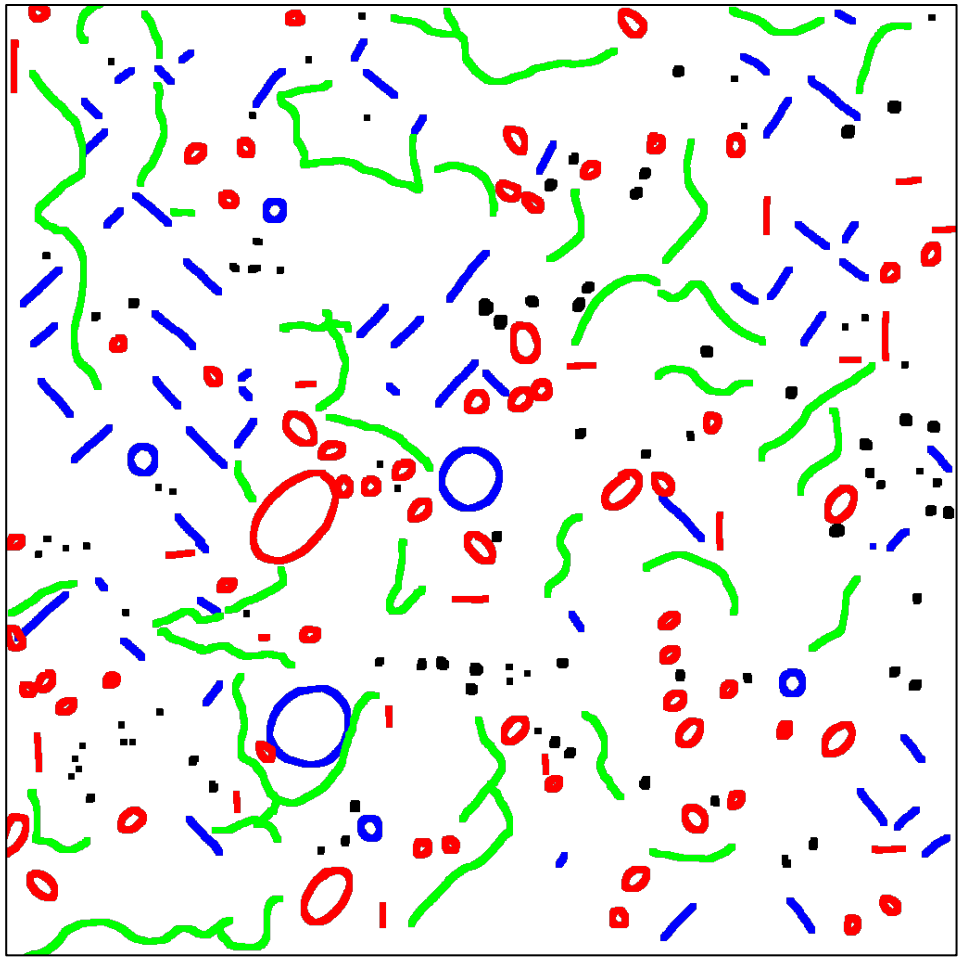




Transmission Electron Microscopy – FeCrAl alloy example

Dislocation loop densities (a{001} = 2.8e21 m^{-3} & (a/2{111} = 4.1e21 m^{-3}) in-line with prior low-dpa irradiations in HFIR for Generation 1 and Generation 2 FeCrAl alloys developed at ORNL.





Network Dislocations

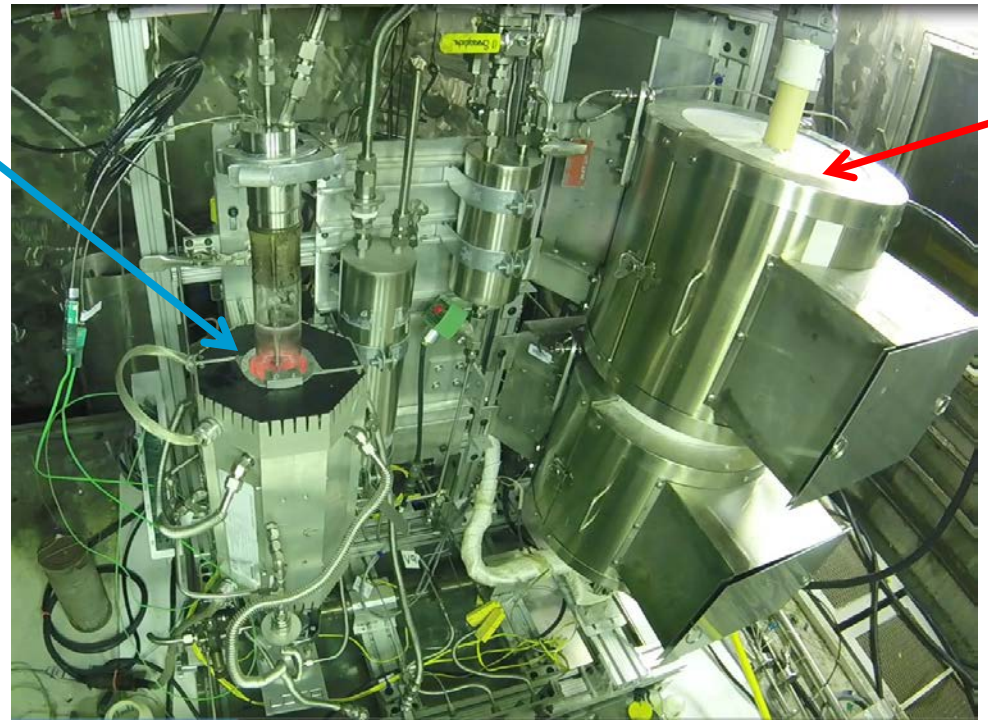
a{001} loops

a/2{111} loops

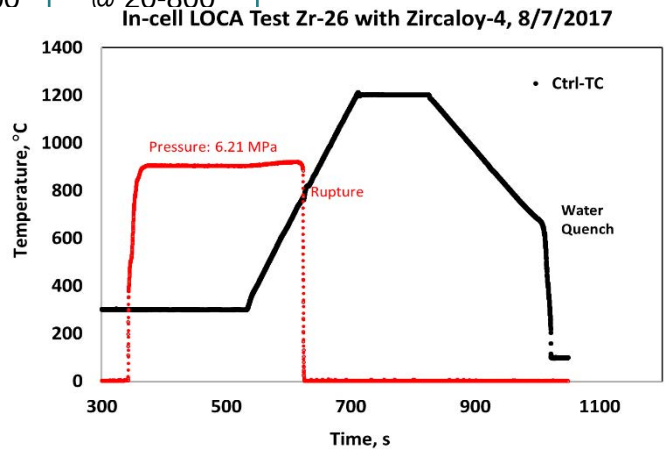
Black Dots (defects too small to resolve as loops)

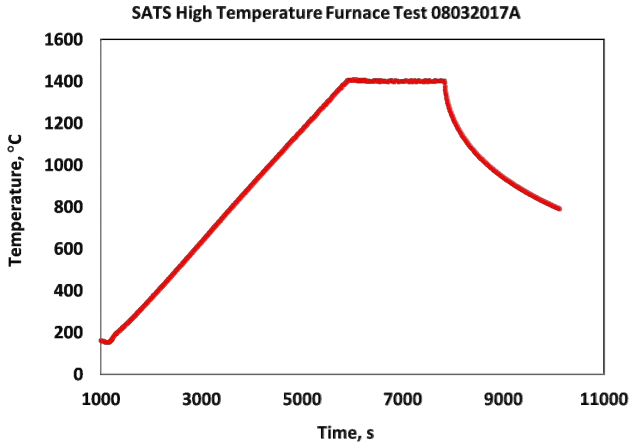
ORNL Fuel Safety: In-Cell Severe Accident Test Station

DBA (LOCA Test Stand)	LOCA Integral Test	Oxidation-Quench Test
Sample Spec	Fueled Rod	Defueled Clad
Segment (mm)	~200-300	~25-50
Pressure (MPa)	≈8, max 20	.1
Max Temp (°C)	1200	1200
Heating Rate (ºC/s)	**5** to 17	5; max 20
Steam Flow Rate (mg/cm2·s)	~5.7	~5.7
Gas Environment	Steam or argon	Steam or argon
Quench (°C)	@ 20-800	@ 20-800



BDBA High Temperature Test Station	
Sample Spec	Rod or Coupon 3mm hole needed
Segment (mm)	~25-50
Pressure (MPa)	.1
Max Temp (°C)	1700°C
Heating Rate (ºC/min)	10; max 20
Steam Flow Rate (mg/cm2.s)	3.0-7.0
Gas Environment	Steam or argon
Quench (°C)	No
Test Time	Multiple days





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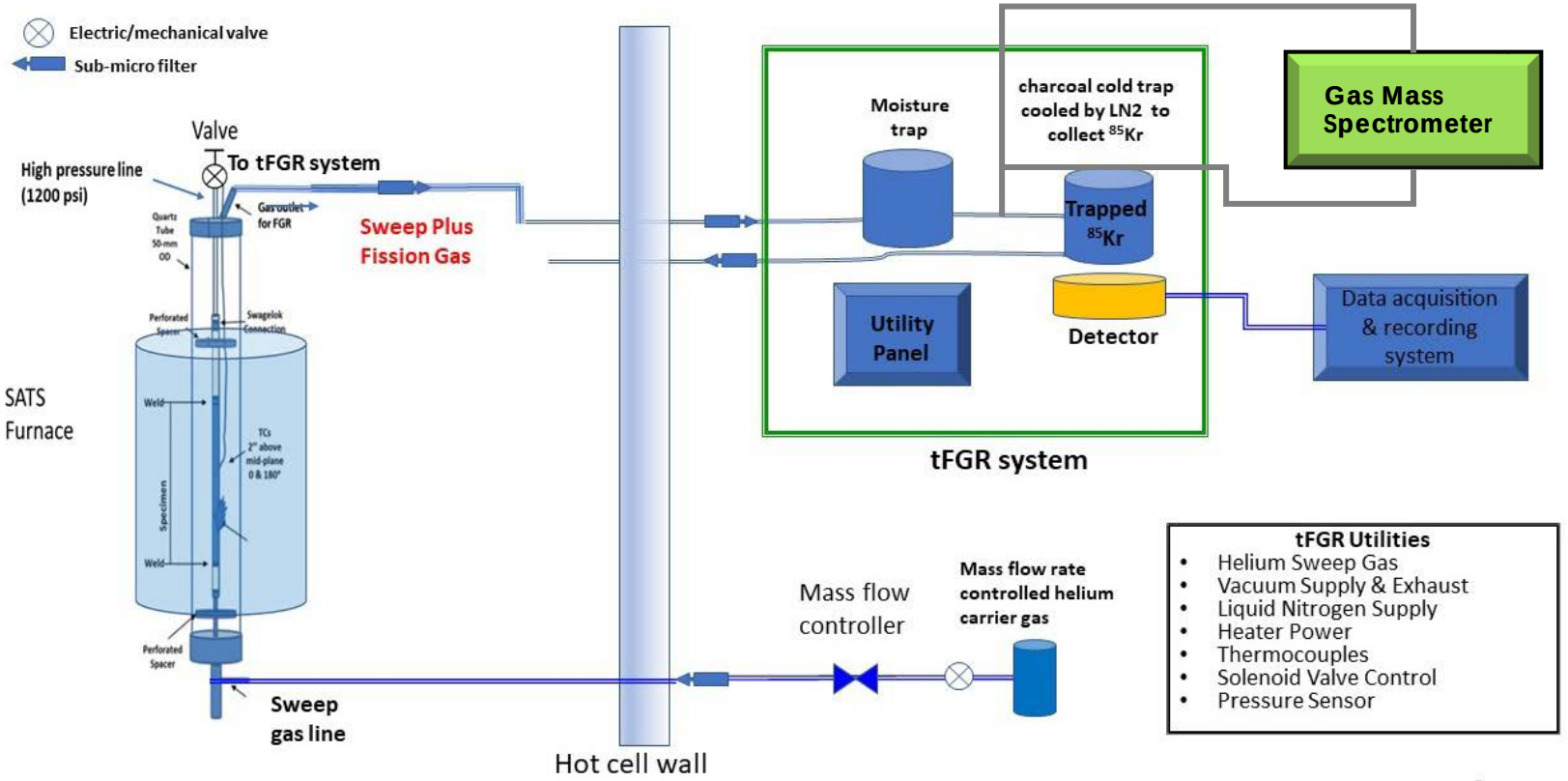
tFGR System for ORNL Severe Accident Test Station (SATS) - Concept

- The tFGR System for SATS is an additional system that takes advantage of the existing SATS infrastructure
- Short segments of irradiated fuel are tested
- Gas released during a heating transient is circulated to a cold trap to detect Kr-85
  - Can operate under PWR relevant pressures
    - Collect Kr-85 data stepwise at pressure
  - Collect fully time resolved Kr-85 release at atmospheric pressure
  - Collect integral tFGR from a burst
- Gas can also be circulated through a gas mass spectrometer to detect stable Xe and Kr

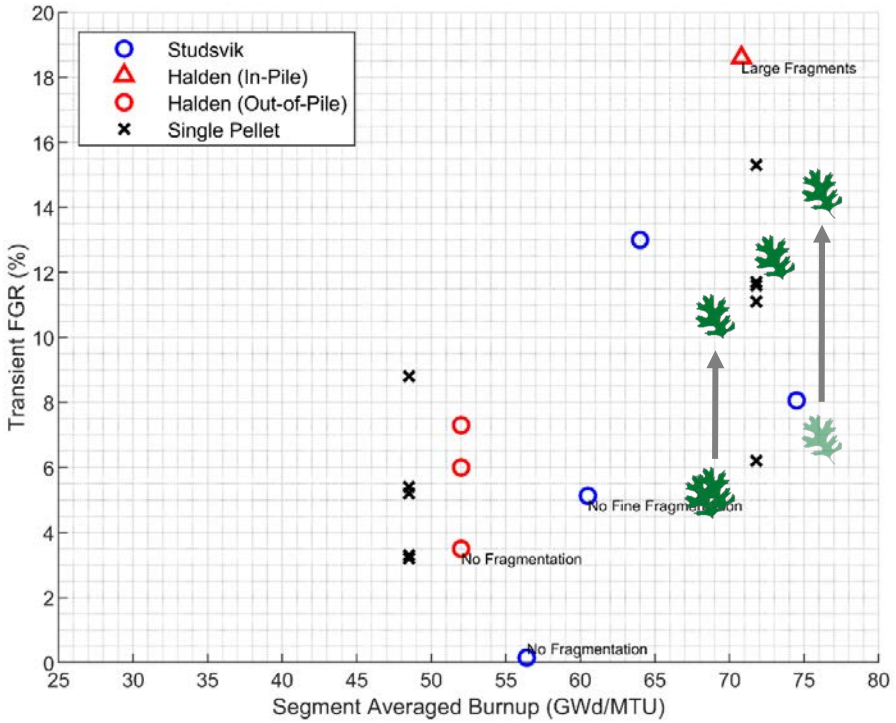


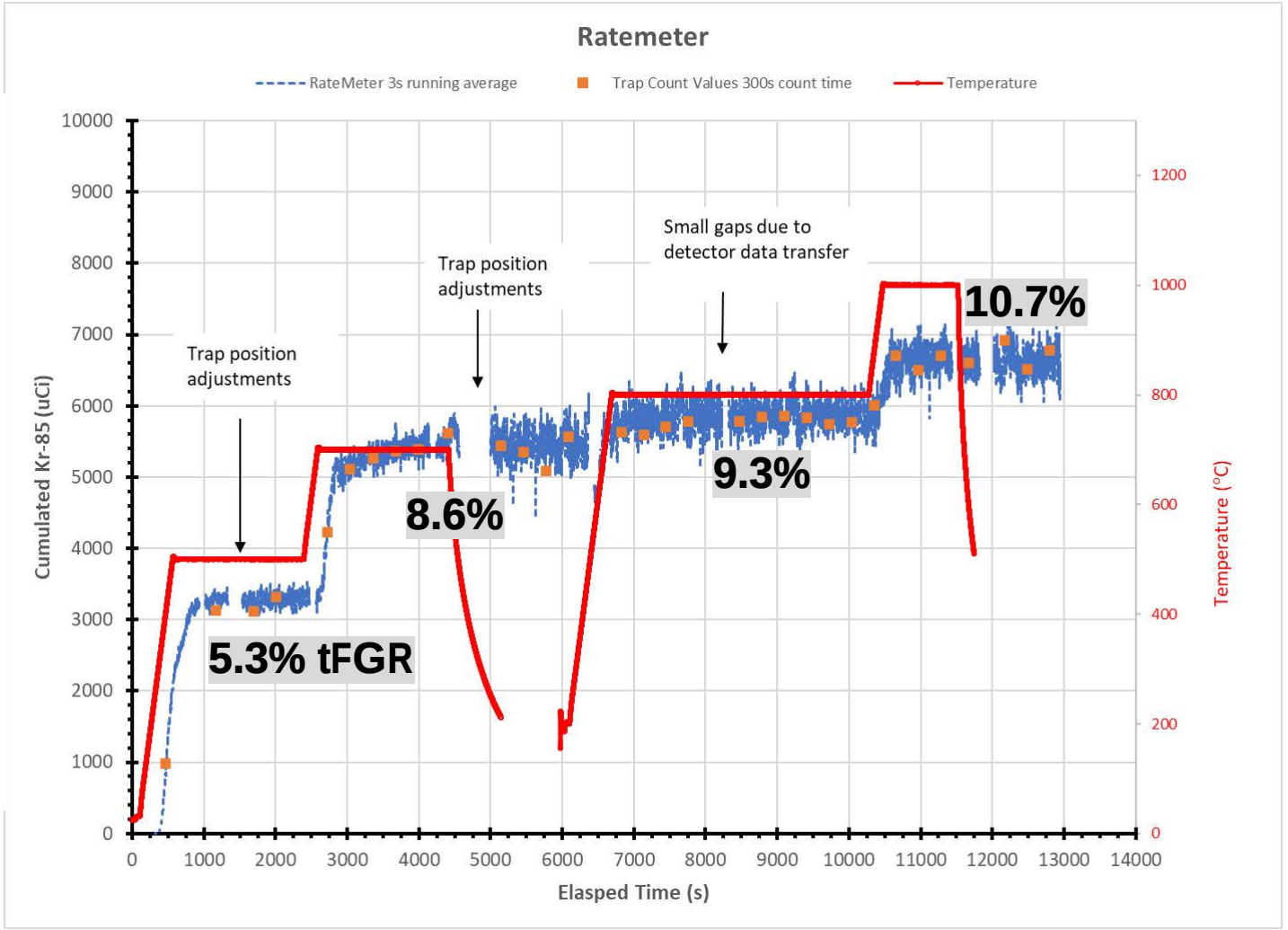
5

Transient Fission Gas Release is a lightly studied, complex phenomena that can impact LOCA behavior

Now 4+ more

~18 data points





Bales, M., Chung, A., Corson, J., & Kyriazidis, L. (2021). Interpretation of Research on Fuel Fragmentation , Relocation , and Dispersal at High Burnup. *Research Information Letter Office of Nuclear Regulatory Research*, RIL 2021-13.

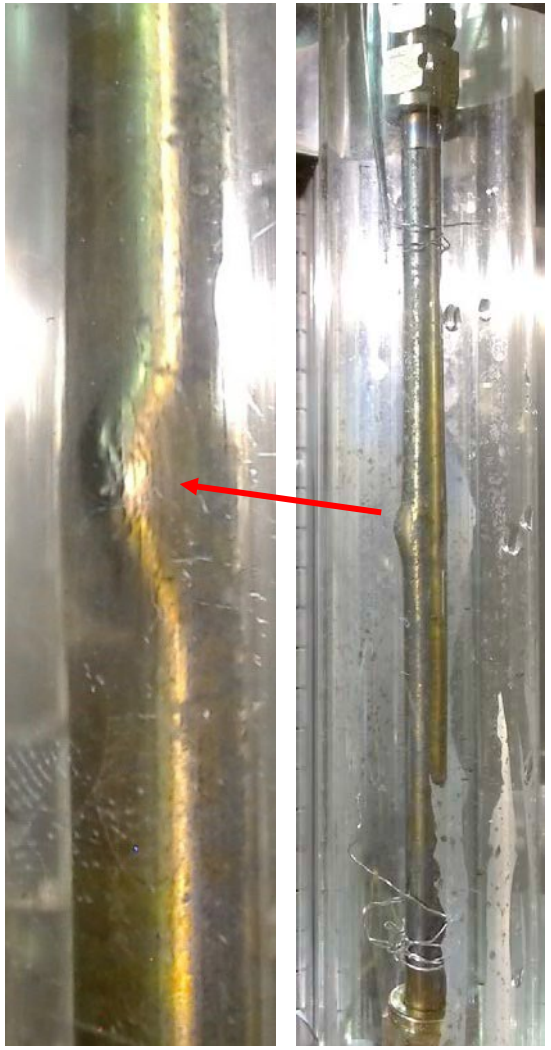
First Loss-of-Coolant Accident (LOCA) Test Performed on Commercially Irradiated Accident Tolerant Fuel (ATF) Cladding

- **Motivation:** Support licensing of ATF to enhance the safety and economics of the existing Light Water Reactor fleet
- **Problem:** No LOCA data on commercially irradiated ATF material
- **Approach:** Compare Cr-Coated Zircaloy LOCA performance between as fabricated and commercially irradiated materials

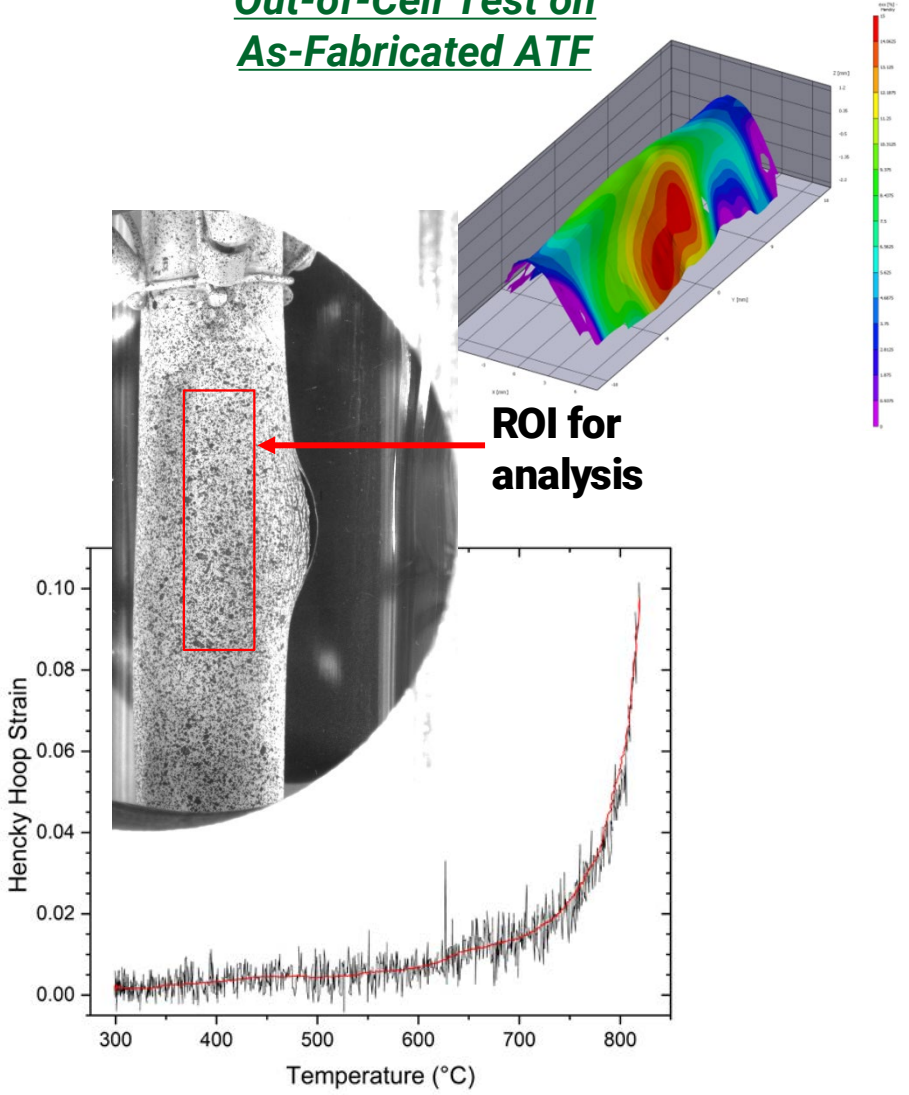
	As-Fabricated Cr-Coated Zircaloy	Commercially Irradiated Cr-Coated Zircaloy	Standard Zircaloy
Rupture Temperature (C)	>Zry	>Zry	754
Peak Strain (%)	<Zry	<Zry	58
Rupture Width (mm)	1.56	1.58	6.5
Rupture Length (mm)	9.1	8.57	19.3

- **Impacts:** ATF increases burst temperature and reduces both balloon and burst sizes compared to uncoated cladding. ATF maintains performance up to 31 GWd/tU rod average burnup.
- **Future Work**: Confirm whether the increased burst margin is sustained to high burnup conditions to validate the safety benefits, thereby supporting fuel burnup extension and power uprates.

Commercially Irradiated Cr-Coated Zircaloy (31 GWd/tU)



Out-of-Cell Test on As-Fabricated ATF



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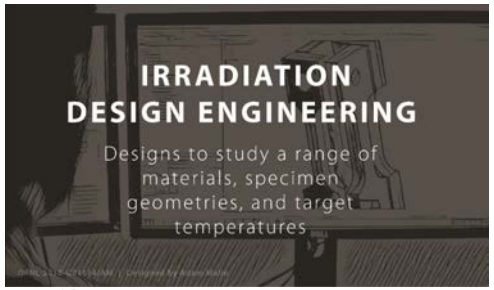
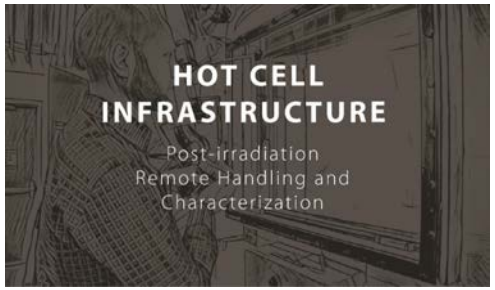
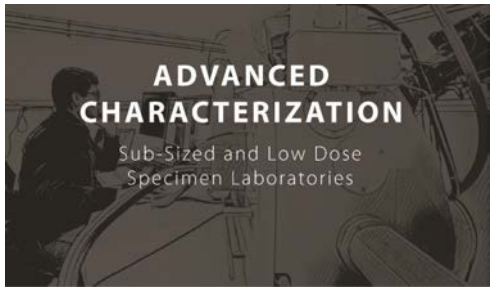
Summary

- ORNL operates several Nuclear and Radiological Facilities used to support nuclear fuel and material post-irradiation examination
- The ORNL fuels hot-cell is capable of receiving full length commercial fuel pins and HFIR irradiations
  - The β,γ hot-cell can receive HFIR irradiated and non-alpha irradiated material
- 3 shipments of commercially irradiated LTR's were recently received at the ORNL hot-cells in addition to material supporting Used Fuel Research
- PIE on these lead test rods is on-going and supports both Westinghouse and GE ATF FOA efforts
- Not covered here, but PIE on HFIR irradiations supports a wide number of external customers







ESCP Task Group Identified ATF/LEU+/HHBU Data Needs for Extended Storage and Transportation

Potential storage, extended storage, transportation, and disposal of Accident Tolerant Fuel, High Enrichment Fuel, & Higher Burnup Fuel

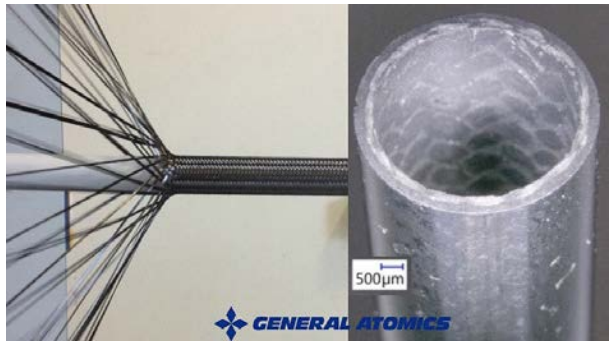
Sven Bader
Orano Federal Services, LLC

orano

EPRI ESCP
October 27, 2025

ATF/LEU+/HHBU

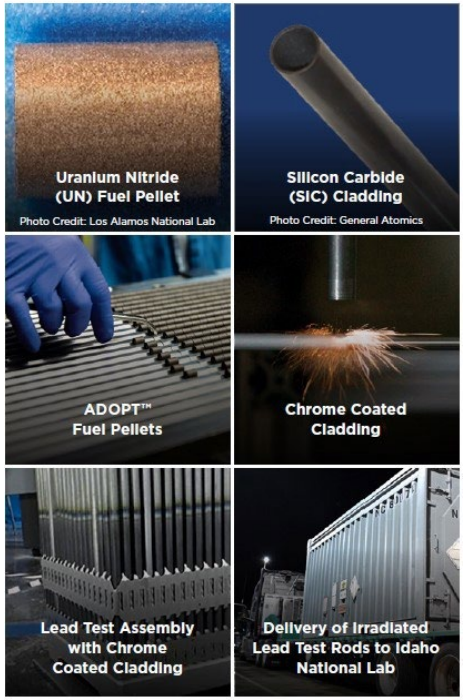
Framatome PROtect



General Atomics SiGA®

Key Benefits

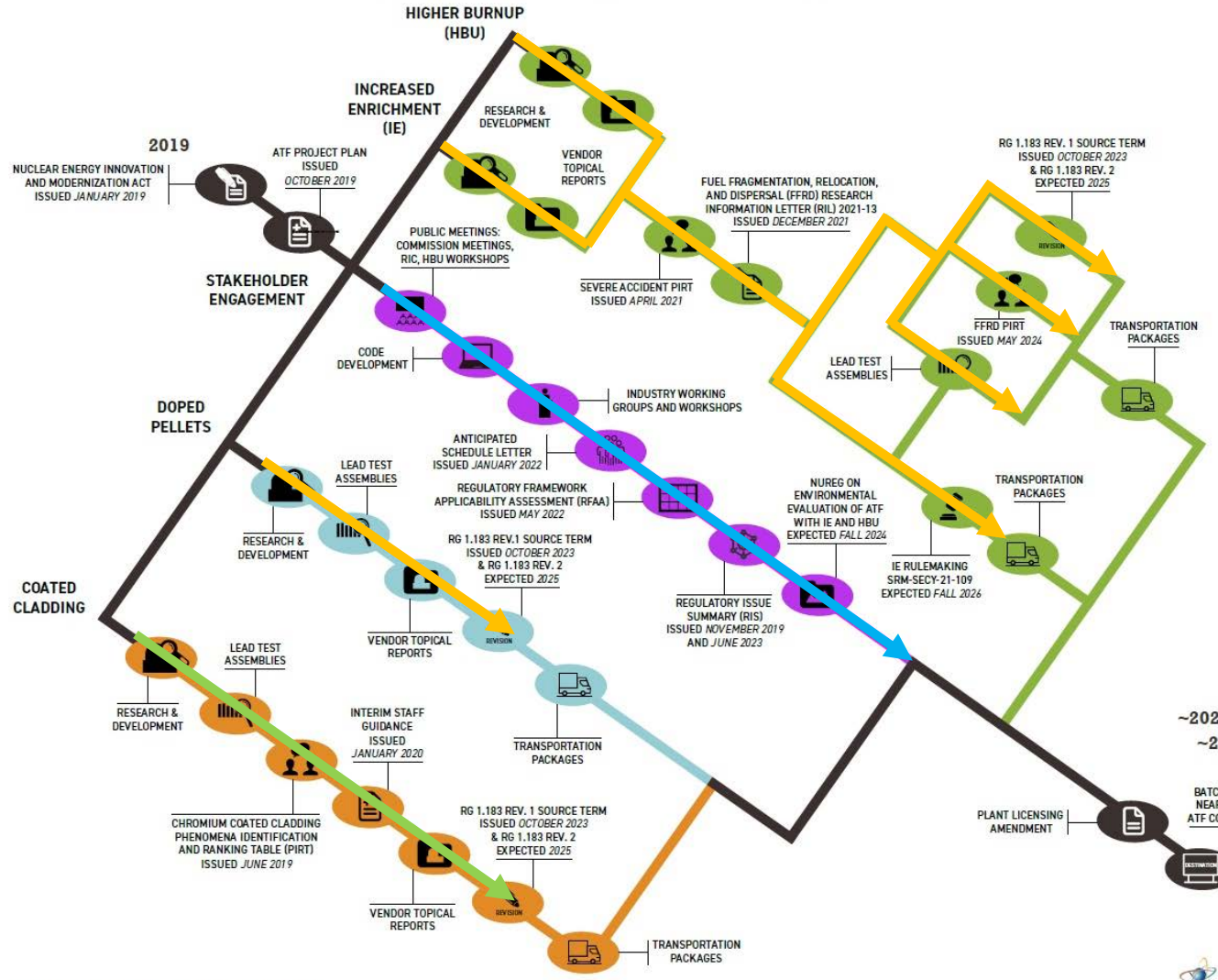
- Designed to have:
  - Higher performance (longer fuel cycle lengths)
  - Enhanced tolerance to severe, beyond DBA conditions (increase coping times)
  - Increased operational flexibility
  - Fewer fuel assemblies to purchase and to store
- Desirable performance attributes include increased power density, longer fuel cycle, and operation to higher burnup.
- Designs include:
  - Coated claddings (e.g., chromium, silicon carbide)
  - Doped pellets (e.g., uranium silicide)
  - Iron-Chromium-Aluminum (FeCrAl) cladding
- Framatome, General Atomics, GVH, Westinghouse



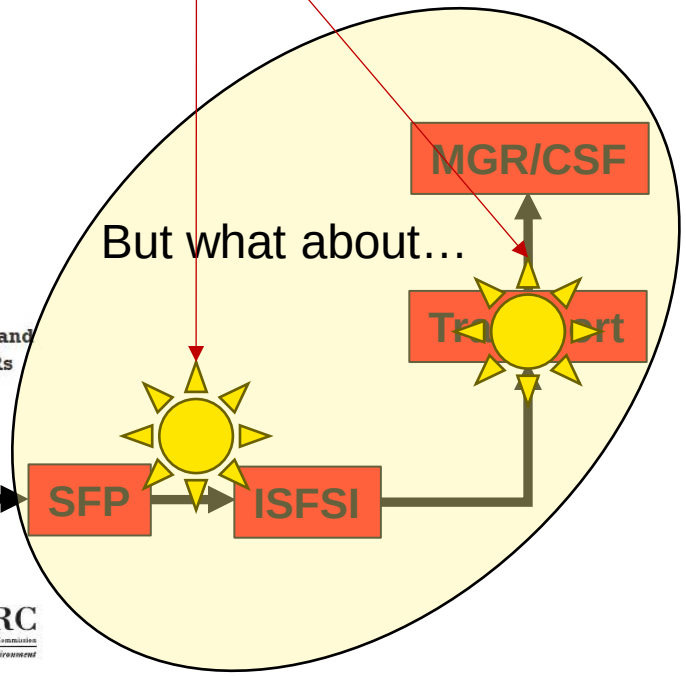
Westinghouse EnCore®

GE Vernova Hitachi (GVH)
ARMOR & IronClad

NRC Roadmap to ATF, Increased Enrichment, & Higher BU Fuels

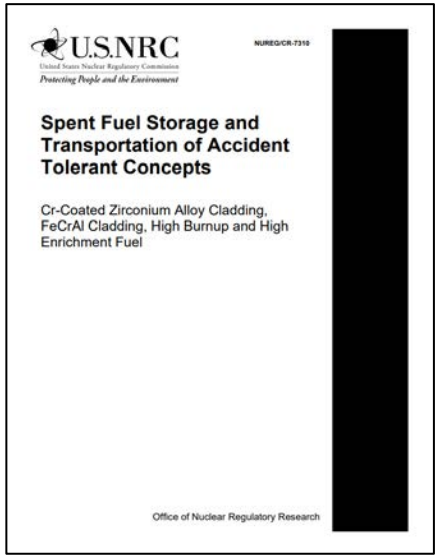


NUREG/CR-7310



NUREG/CR-7310

- **Spent Fuel Storage and Transportation of Accident Tolerant Concepts: Cr-Coated Zirconium Alloy Cladding, FeCrAl Cladding, High Burnup and High Enrichment Fuel**
- **Date Published: June 2025**
- **Author: PNNL**
- **Contents:**
  - Provides state-of-the-industry information on material properties and fuel performance
  - Covers:
    - *GNF: FeCrAl alloys & Abrasion Resistant, More Oxidation Resistant (ARMOR) cladding*
    - *Westinghouse: Cr-coated ZIRLO® standard UO_2 fuel & Cr_2O_3+Al_2O_3-doped UO_2 fuel*
    - *Framatome: Cr-coated M5® cladding & Cr_2O_3 -doped UO_2 fuel*
  - Identifies and discusses degradation and failure modes of these ATF cladding concepts, including fuel performance characteristics that may not be addressed within existing regulatory documents.
  - The implications of high burnup (> 62 GWd/MTU) and enhanced enrichment (> 5 w/o 235U) on the storage and transportation of SNF are discussed.
  - The recommendations made in this report are based on current publicly available data.
  - Identifies available data and data needed for safety evaluations



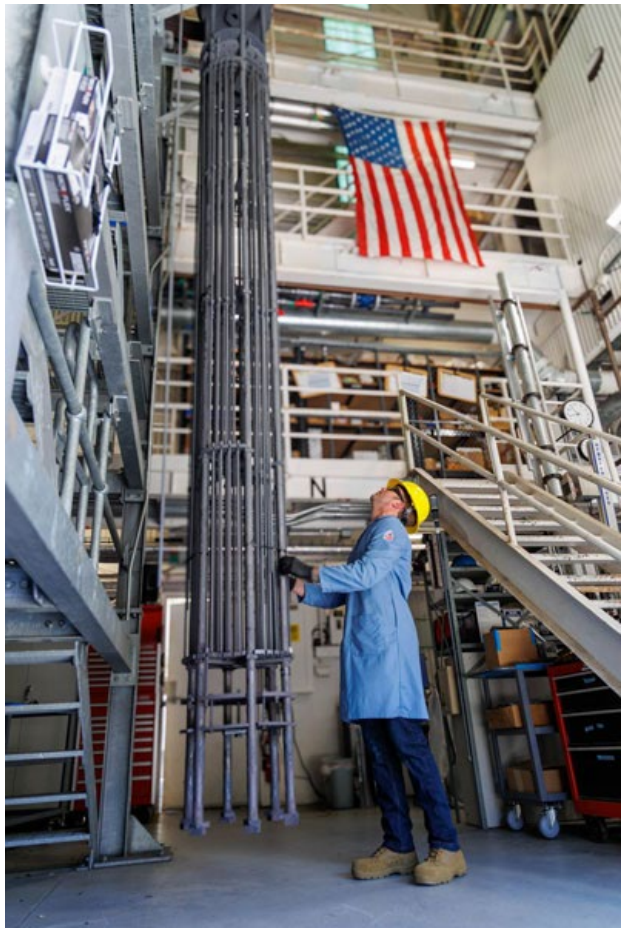
Lead Test Assembly/Rod (LTA) Testing Opportunities

- Lead test assemblies/rods/rodlets (LTAs) are currently being placed in reactors for several cycles
- After irradiation in the core, these LTAs are sent to a hot cell for NDE and DE to verify/support reactor operations
- Once the NDE and DE in support of reactor operations are completed, there is the opportunity to utilize the remaining material for additional testing in support of post-irradiation activities to:
  - Potentially resolve identified gaps
  - Identify other issues potentially impacting post-irradiation activities

Westinghouse U-Silicide Pellets in Cr-Coated Zirconium Clad (Westinghouse)

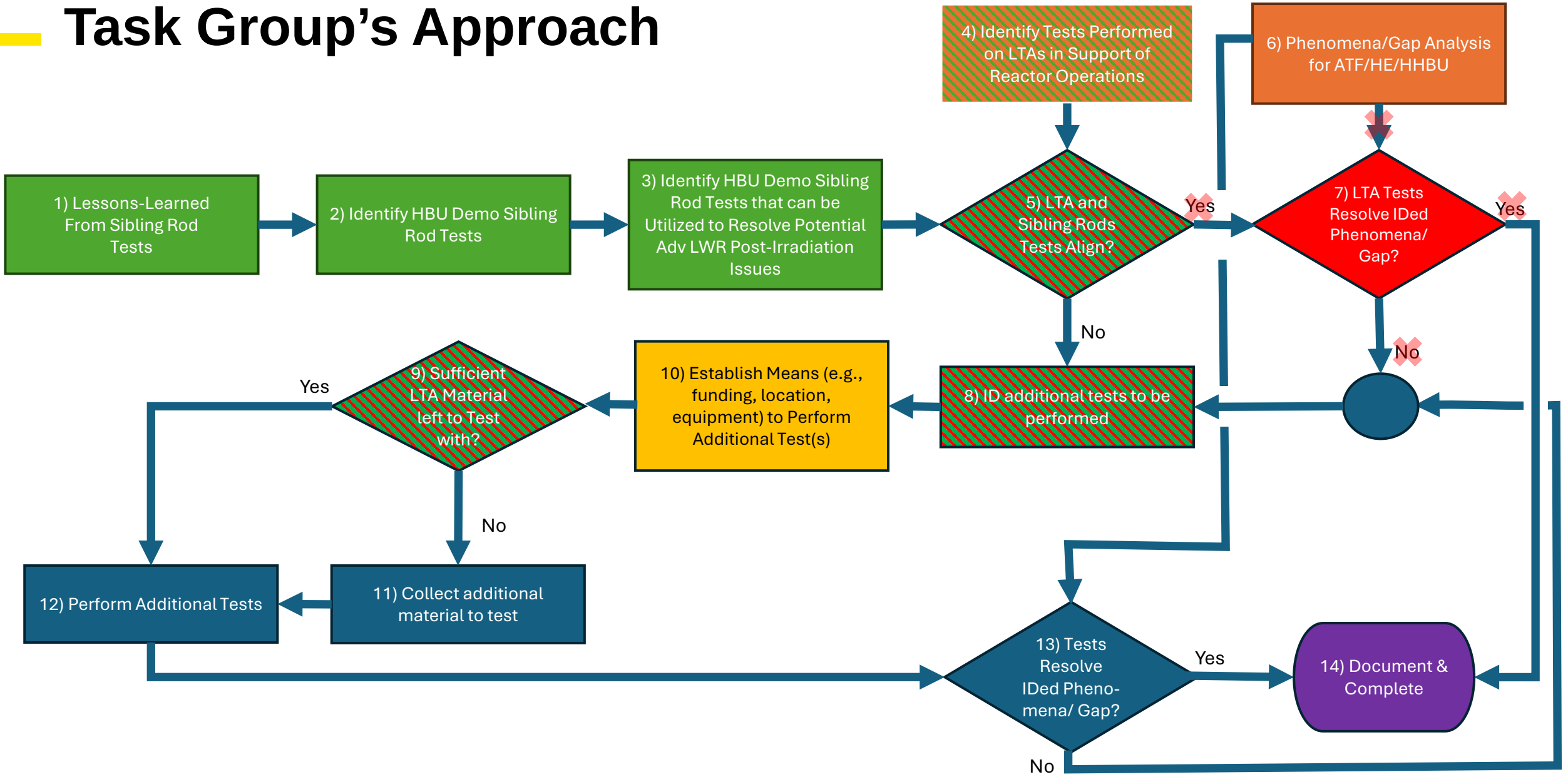
GVH ARMOR & IronClad (INL)

Framatome GAIA EATF (European Nuclear Society)



General Atomics SiGA (General Atomics)

Task Group's Approach



Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (Very & High Priority)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
Puncture & Depressurization of Rod	Very High	Very High	Rod internal pressure	Rod	$	Y	• Rod internal pressure is very important
Rod Heat Treatments	High	High	Simulates temperature excursions in dry storage (e.g., vacuum drying) – yield strength of cladding	Rod	$	N	• Important to annealing demo but needs longer duration tests (yield & ultimate strength) • Could help relax 400C limit • Cover gap of HHBU cladding properties
Ring Compression	High	High	Pinch loading	25 mm segment	$$	?	• May identify transition temperature to brittle behavior
Cladding Microscopy	High	High	Visual oxide thickness, hydrides & hydride distribution	2 mm segment	$$	?	• Easy to do and provides basic info • 1st test to be performed
Burnup/Fuel Isotopics	High	High	Isotopic distribution and burnup of one pellet	35 mm pellet	$$$$$	?	• Only needed if burnup credit is to be applied to benchmark models

Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (High Priority for a Fuel Type)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
SEM	High	N/A	Examine effects of coating	<1 mm segment	$$$$	?	• Verify cohesion of coated cladding • More relevant to ops than backend
Four-Point Bend	High	Low	Flexural strength and modulus	6” segment	$$	?	• Measurement of axial strain
Cladding Hydrogen	High	Low	Total hydrogen in cladding quadrant	4 mm segment	$$$	?	• Needed to confirm expected low hydrogen levels
Fission Gas Sampling	Low	High	Gas composition	< 50 cc	$$	?	• Understanding the impact of doped pellets on the gas composition is important
Fueled Microscopy	Low	High	Pellet: cracks, HBU rim, localized effects to cladding, pellet-pellet interfaces	2 mm segment	$$$	?	• Especially important for higher burnup fuels

Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (Medium Priority)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
Profilometry	Medium	Medium	Rod outer diameter	Rod	$	?	• Gives an idea of any damage to cladding
Axial Tension	Medium	Medium	Tensile strength, yield strength, elastic & plastic deformation	4” to 6” segment	$$	?	• Clad mechanical properties will likely need to be proprietary
Fatigue (CIRFT)	Medium	N/A	Fatigue lifetime	4 – 6” segment	$	?	• Coating can impact fatigue
Gas Communication	N/A	Medium	Relative ease of gas movement though fuel.	Rod	$	?	• Allows for comparison between doped and undoped pellets • Tests should be performed at operating temperature
Aerosol Collection	N/A	Medium	Quantity, size, and isotopic distribution of aerosols	6” segment	$$$	?	• Support design basis analysis

Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (Low Priority)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
Gamma Scanning	Low	Low	Location of pellets & length measurements	Rod	$	?	• Validation of method for establishing BU
Length Measurement	Low	Low	Overall rod length	Rod	$	Yes	• Nice to have
Visuals	Low	Low	High resolution visual of rod	Rod	$	Yes	• Good for documentation purposes
Microhardness	Low	Low	Vickers hardness of cladding, fuel pellet hardness	2 mm segment	$$	?	• Data not used in safety analyses • Used in tensile stress assessment

Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (Not Needed)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
Eddy Current	N/A	N/A	Oxide thickness, crud, cladding thickness	Rod	$$	-	• Crud pick-up and oxidation of coated clads is very low to date
Surface Temperature	N/A	No	Surface temperature	Rod	$	-	• Could help with benchmarking burnup calculations
Induced Hydride Reorientation	N/A	N/A	Hydrogen pickup of cladding	Small rod segment	$$	-	• Not expecting much hydrogen pickup of coated clads
Steam Communication	N/A	N/A	Relative ease of steam movement during drying and disposal conditions	4” to 6” segment	$$	-	• Could be of interest to verifying dryness of cask
MiniTensile	N/A	N/A	Cladding tensile strength, yield strength, elastic & plastic moduli	25 mm defueled	$$$	-	

Summary of Results for Testing for Backend Applications for Advanced LWR Fuels (Not Needed)

Test	Priority by Fuel Type		Data Obtained	SNF Material Required	Cost	LTA Test	Notes
	Coated Cladding	Doped Pellets					
Biaxial Torsion	N/A	N/A	Shear modulus	4” segment	$$$$$	-	• Test still being validated
Pellet Push Out	N/A	N/A	Cladding/pellet bonding strength	25 – 50 mm segment	$	-	
Leach	N/A	N/A	Relative release rates of fission products in water	25 mm segment	$$	-	• Could be a low priority for doped pellets for repository
TEM	N/A	N/A	Examination of radiation damage and other nanoscale particles	<1 mm segment	$$$$$	-	• Could be used selectively for coated cladding

Work Remaining to Do…

- Tie the above tests to specific regulatory needs for storage, extended storage, and transportation
- Verify testing needs fill gaps identified in NUREG/CR-7310
- Identify which of the above tests have been performed as part of the LTA testing program
- For the remaining tests establish if the remaining LTA material is sufficient to perform these test
- Establish means (e.g., funding, location, equipment) to perform additional tests
- Perform tests and identify if results resolve identified phenomena/ gaps

Interested in Joining Us?

Let Hatice (hakkurt@epri.com) or Sven (sven.bader@orano.group) know

We meet 3rd Thursday of every month from 10a – 11a (EST)

We currently have ~20 “contributors” & ~20+ “observers”

Any questions?

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Towards Optimal Vacuum Drying: Enhanced Moisture Removal and Accurate Residual Water Quantification

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TUESDAY, OCTOBER 28, 2025, 09:00 AM – 09:30 AM
EXTENDED STORAGE COLLABORATION PROGRAM

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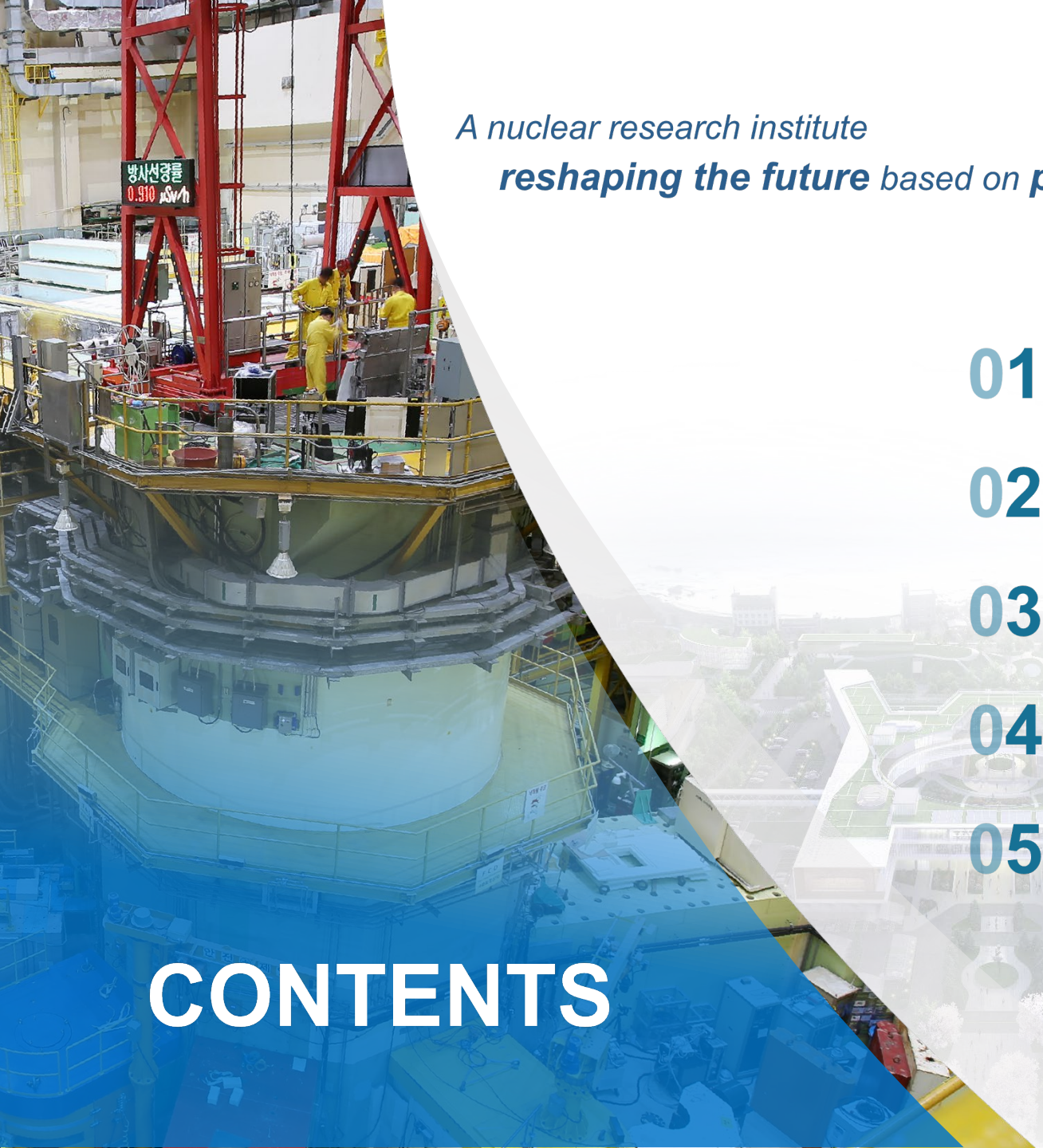

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01 Spent Nuclear Fuel Drying
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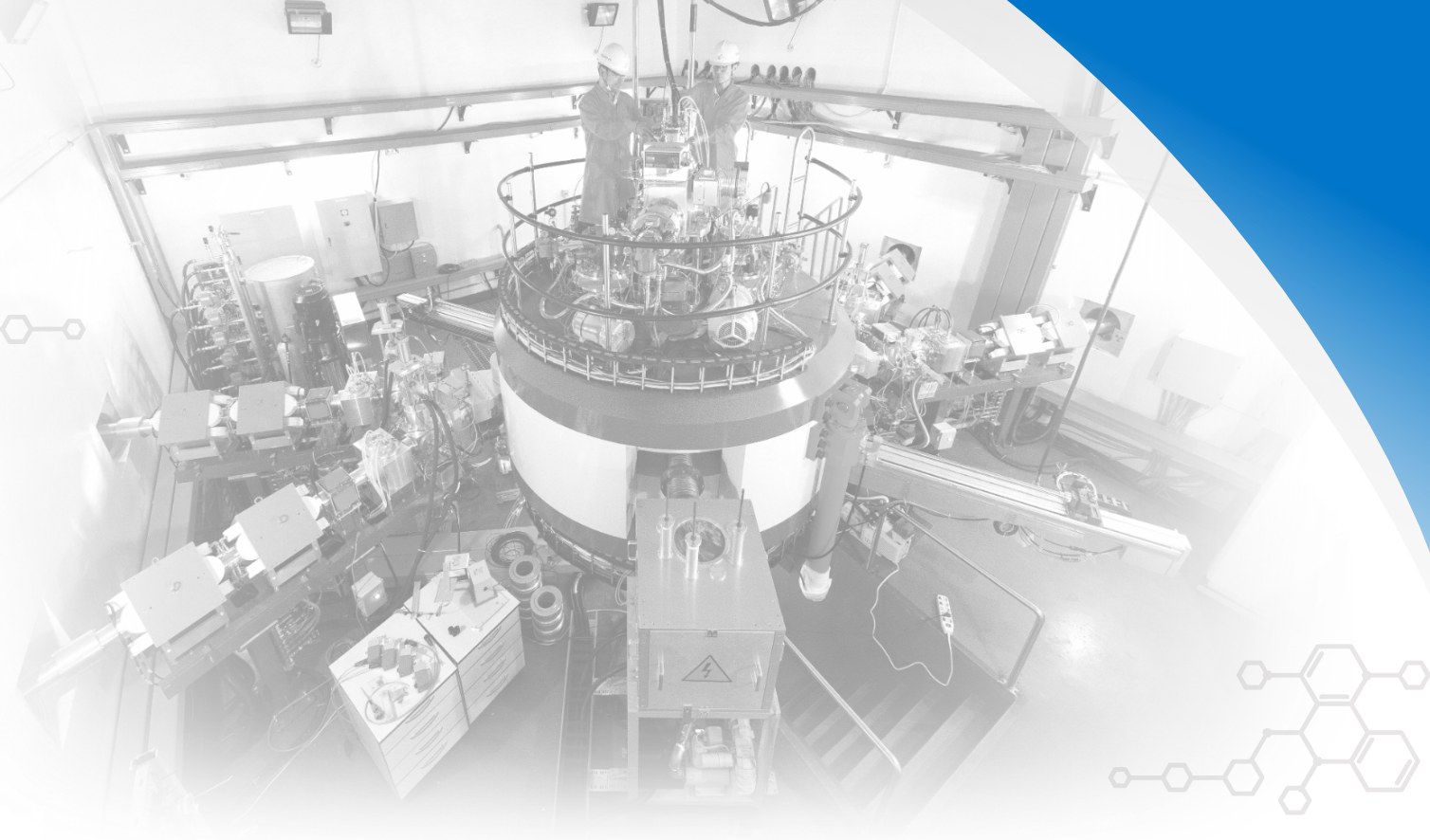
05 Conclusion and Recommendations



Towards Optimal Vacuum Drying: Enhanced Moisture Removal and Accurate Residual Water Quantification

Spent Nuclear Fuel Drying

01

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01 Background and need for research

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*NWPA: Nuclear Waste Policy Act

Objective: validate and evaluate the thermal removal and shielding performance of dry storage systems

- Thermal performance of storage casks and facilities
- Radiation dose assessment by storage procedure
- Verification of thermo-hydraulic analysis codes

Objective: collect data for prolonged licensing of dry storage systems' safety and retrievability

- Licensing extension based on NWPA
- INL: visual inspection, structural diagnosis, radiation scans, gas analysis within casks
- ANL: detailed investigation of cladding condition

Objective: renewal of licensing and necessity to extend dry storage period for high burnup fuel

- Establish aging management plan (AMP)
- Build scientific confidence for storage license extension
- Acquire confirmatory data on transport feasibility after long-term storage

01 Early demonstration projects of dry storage

Background

- Based on 1982 NWPA, conducted in collaboration between DOE and utilities at sites like Surry, Robinson, and INL

Cask verification

- Excluding WH's SSC, commercially developed storage casks used.

Code validation

- Used to validate heat transfer codes like COBRA-SFS, HYDRA (led by PNNL).→ **design criteria met, analysis results valid**

02 Dry cask storage characterization project (DCSCP)

Background

- Task jointly aimed by DOE, EPRI, NRC
- 15 years: sufficient time for degradation mechanism manifestation, expected safety for 20-40 years (Utilized 35GWd/MtU burnup)

Assumptions

- Fuel rod creep within 0.1%, no internal pressure change, no cladding annealing, sufficient creep strain margin. → **licensing extension based on data**

Conclusion

03 High burnup demonstration project

High burnup fuel

- Mechanical properties of high burnup fuel differ from low burnup: need to explore hydrogen reorientation's effect on fracture toughness, ductile to brittle transition temperature (DBTT)

Data from entire process

- Reproduce entire dry storage process and collect data at each step

Effect of vacuum drying

- **Conduct vacuum drying process; verify gas composition; quantify residual water, oxygen; assess impacts on degradation**

01 Why is drying spent nuclear fuel important?

Korea Atomic Energy Research Institute (KAERI)

- Why is it important? How much water can theoretically exist?

Canister Model	Manufacturer/Organization	Purpose	Inner Volume (liters)	Source
Holtec HI-STORM 100	Holtec International (USA)	For Dry Storage (Multi-purpose System)	6484	NRC FSAR (Holtec Design) [1]
NAC UMS (Universal MPC)	NAC International (USA)	For Dry Storage/Transport (Dual Purpose)	2100	NRC MAPS Report [2]
Orano TN-24 Series	Orano TN (France)	For Dry Storage/Transport (Dual Purpose)	10900	NRC Analysis Data [3]
GNS CASTOR V/21	GNS (Germany)	For Dry Storage/Transport (Dual Purpose)	12000	Research Report [4]
KBS-3 Disposal Canister (for PWR)	SKB/Posiva (Sweden/Finland)	For Final Disposal (Geological Deep Disposal)	1656	Academic Research [5]

Table: Maximum amount of water that can theoretically exist under 3 torr pressure by volume (Ideal Gas Law)

Volume (liters)	Moles (mol)	Mass (g)
1	1.612 x 10^{-4}	2.9 x 10^{-3}
500	0.0806	1.45
1000	0.1612	2.9
1500	0.2418	4.35
2000	0.3224	5.8
2500	0.403	7.25
3000	0.4836	8.7
3500	0.5642	10.15
4000	0.6448	11.6
4500	0.7254	13.05
5000	0.806	14.5
5500	0.8866	15.95
6000	0.9672	17.4
7000	1.1284	20.3
8000	1.2896	23.2
9000	1.4508	26.1
10000	1.612	29

PPMV Definition:

PPMV stands for "parts per million by volume." It measures the concentration of a component in a gas mixture relative to the total volume of the mixture. It is often used when dealing with trace levels of substances.

Formula:

$$\text{PPMV} = \left(\frac{\text{Volume of Component}}{\text{Total Volume of Gas Mixture}}\right) \times 10^6$$

- Residual moisture in a canister after vacuum drying of spent nuclear fuel is typically measured using gas sampling.
- It's challenging to express these measurements in grams because ppmv reflects relative volume, not mass.
- To convert ppmv to grams, the composition of other gas molecules in the sample is needed.
- This requires additional equipment to analyze the gas mixture accurately.
- PPMV is commonly used in fields like chemical research and moisture content analysis due to its effectiveness in measuring trace substances in gas.

(1) HI-STORM FSAR REPORT HI-2002444
(2) Managing Aging Processes In Storage (MAPS) Report Draft Report for Comment, NUREG-2214
(3) Effects of Residual Water on Storage Canister Internal Components, Milestone No. M3SF-20SR010207025, SRNL-STI-2020-00428
(4) Transnuclear Handouts – Part 4 of 6 - 10/07/09 Public Meeting (Draft SAR changed pages – non-proprietary)
(5) https://www.academia.edu/21247322/Spent_fuel_canister_for_geological_repository_Inner_material_requirements_and_candidates_evaluation#:~:text=used%2C%20the%20void%20space%20is,filled%20with%20groundwater%20once%20corrosion

01 Why is drying spent nuclear fuel important?

Korea Atomic Energy Research Institute (KAERI)

❖ Why is it important? How much water is actually present?

Research / Case	Drying Method and Fuel	Residual Moisture Results	Measurement Technique
High Burnup Demo (USA)	Standard vacuum drying on high burnup PWR fuel (Zircaloy cladding)	• After 12 days of drying, approximately 17,400 ppmv of steam (~0.1L) detected - 10 times more than expected. No liquid water (due to desorption/radiolysis)	• Helium backfill gas sampling followed by mass spectrometry and gamma spectroscopy

Table 5. Results of HBU Demo Gas Sample Water Analyses.

Sample #	Collection Time	Dominion ~25°C	SNL† ~25°C	SNL† 65°C
#1	5 hrs	1,633	*2,097**	
#2	5 days	8,896	~6,600*	*~10,000**
#3	12 days	8,300	11,200	17,400

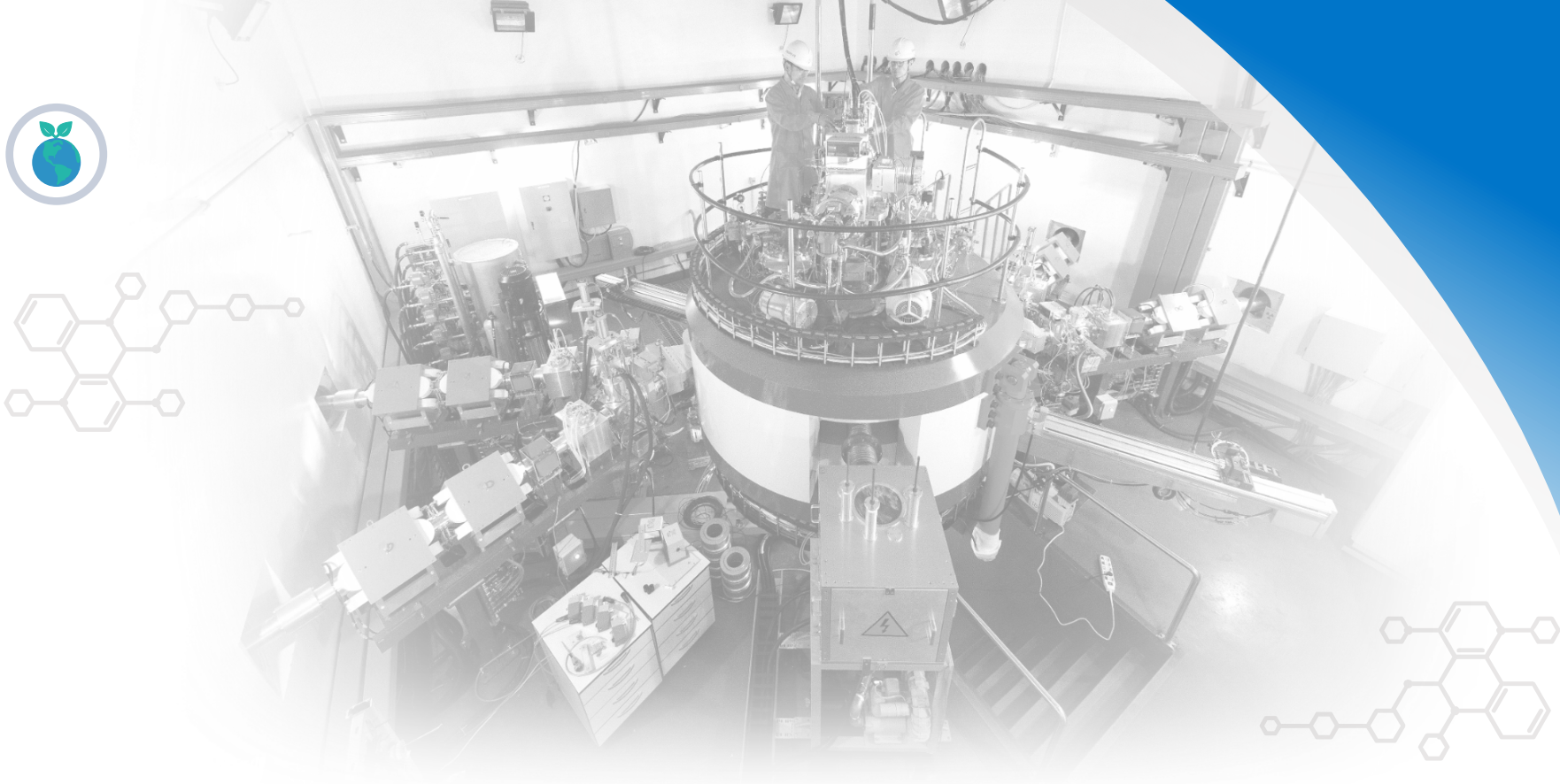
*Samples were compromised; values are minimums
†Uncertainties are estimated, based on analysis of standards, to be ±10%)

Table 4. Results of HBU Demo Gas Sample Analyses.

Gas Species	Sample 1	Sample 2	Sample 3
Sampling time	**5 hrs**	**5 days**	**12 days**
Bulk gas composition, %			
He	Confirmed balance gas		
H_2O (est.)	0.46%	1.8%	2%
N_2	0.39%	0.07%	0.07%
Measured trace gases, ppmv			
H_2	46	287	498
CH_4	< 1	9	22
CO	nm	34	25
N_2	off scale	630	680
O_2	680	38	134
Ar	49	8	9
CO_2	103	751	930

Degradation Phenomena	Effects on SNF
Hydride Reorientation (HR)	• Radial hydrides reduce ductility, increasing the risk of brittle fracture. Possibility of delayed hydride cracking (DHC)
Cladding Creep and Stress	• Though stress relaxation is possible, excessive stress can lead to wall thinning and defect enlargement. More sensitive in high burnup fuel
Cladding Oxidation (Internal/External)	• With minimal moisture, oxidation layer may increase by several microns over decades. Locally, may cause fuel swelling and corrosion
Fuel Pellet Oxidation and Swelling	• Fuel cracking and swelling may exert pressure on cladding, leading to failure. Past incidents in wet storage cases
Radiolysis and Gas Generation	• Potential for H_2 accumulation. Combustion risk if oxygen ingress occurs. Additional hydrogen release from Zr oxides over the long term

Bryan, C. R., Durbin, S., Lindgren, E. R., Ilgen, A. G., Montoya, T., Dewers, T., & Fascitelli, D. G. (2019). SNL Contribution: Consequence Analysis for Moisture Remaining in Dry Storage Canisters After Drying (No. SAND-2019-8532R). Sandia National Lab.(SNL-NM), Albuquerque, NM (United States).



Towards Optimal Vacuum Drying: Enhanced Moisture Removal and Accurate Residual Water Quantification

Experimental Method

02

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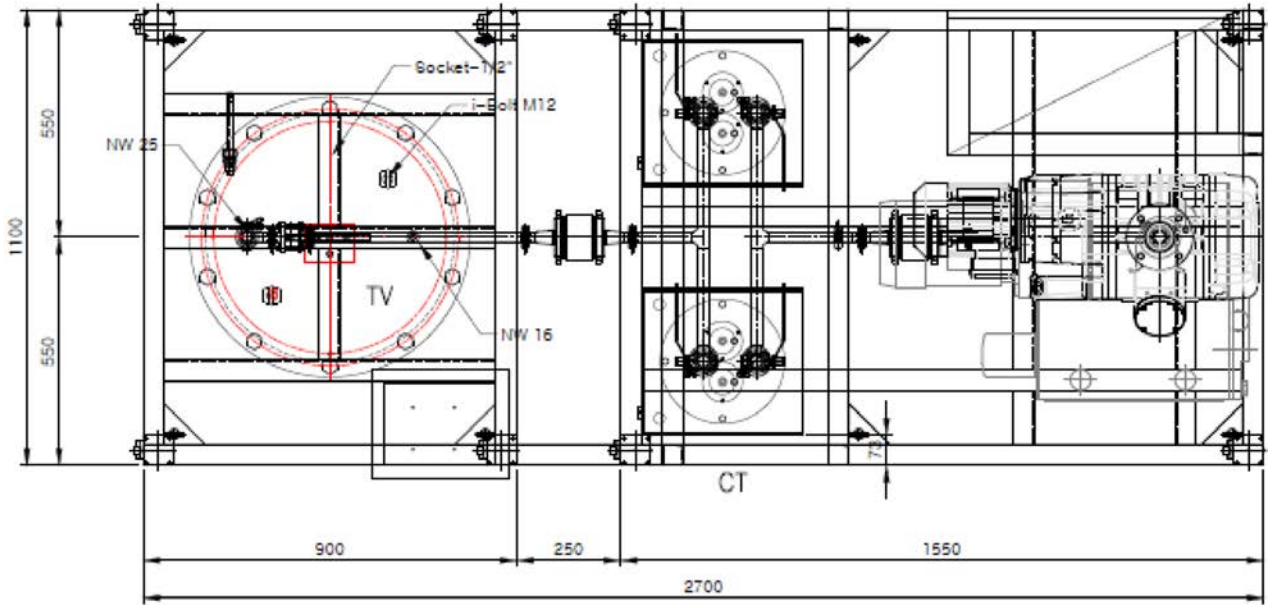
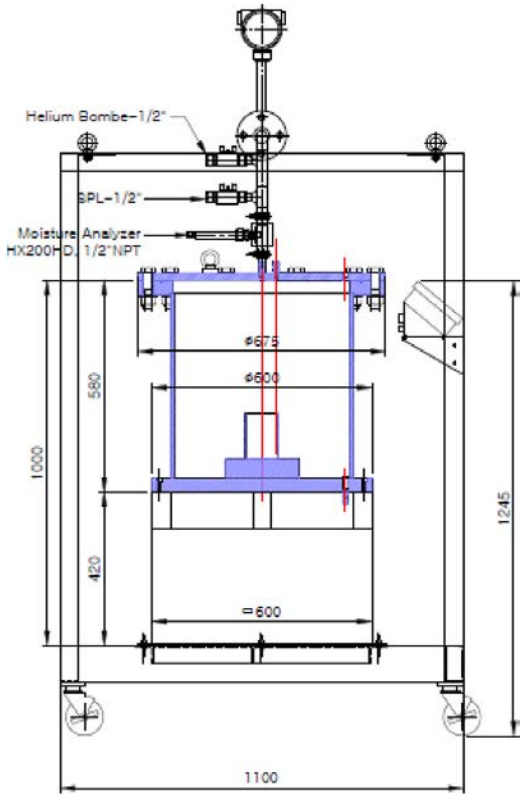
02 Vacuum drying test facility





Schematic of Vacuum Drying System

(1) **Core Components and Functionality:** The vacuum drying system consists of several key components: vacuum pumps, cooling condensation mechanisms, and cold traps. These are crucial for reducing canister pressure, extracting moisture, and ensuring precise pressure validation.

(2) **Design and Integration:** The system integrates modular components with advanced vacuum gauges, supported by the Residual Moisture Measurement (RMM) program on the LabVIEW platform. This setup ensures reliable real-time data acquisition and monitoring across experiments.

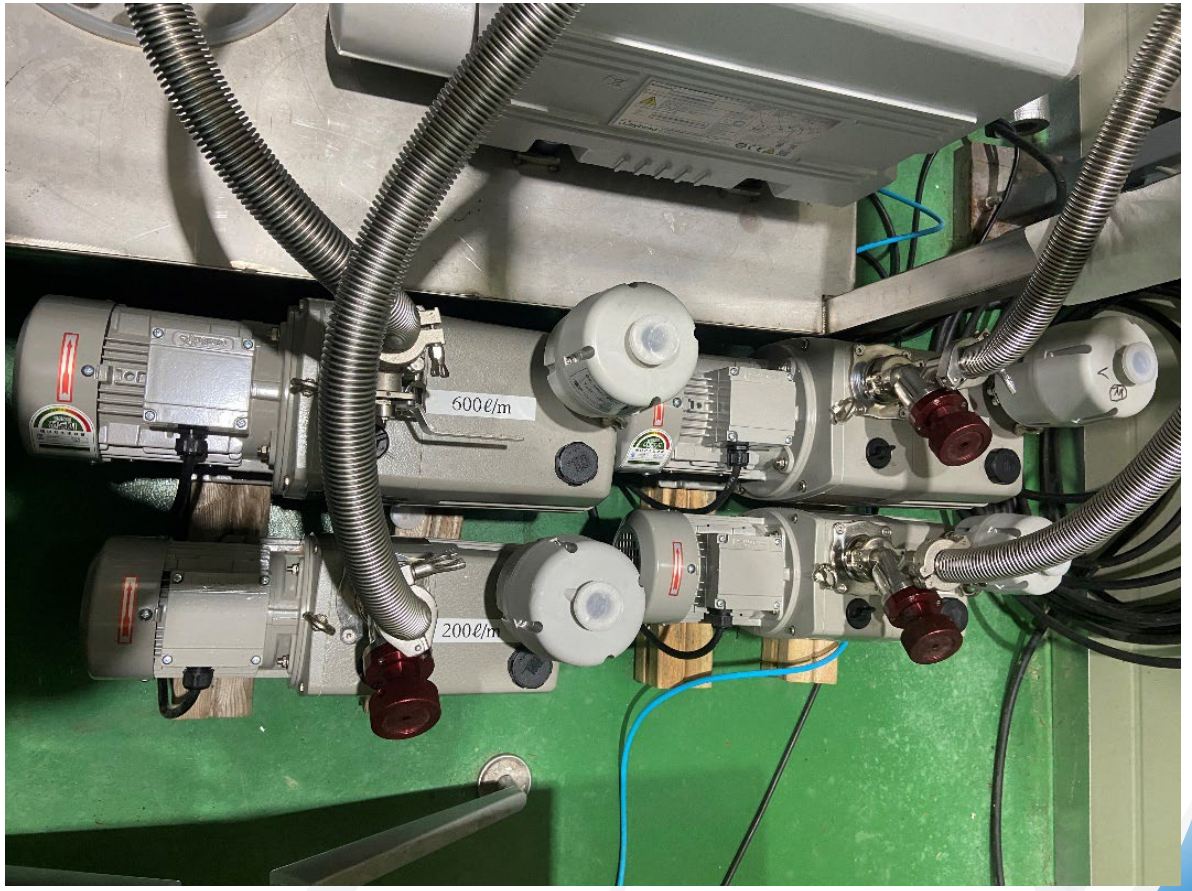
(3) **Operational Safety and Control:** The apparatus is equipped with centralized control panels to minimize radiation exposure and ensure operational precision. It includes automated recording and emergency stop functions for enhanced safety.

02 Vacuum drying test facility

Korea Atomic Energy Research Institute

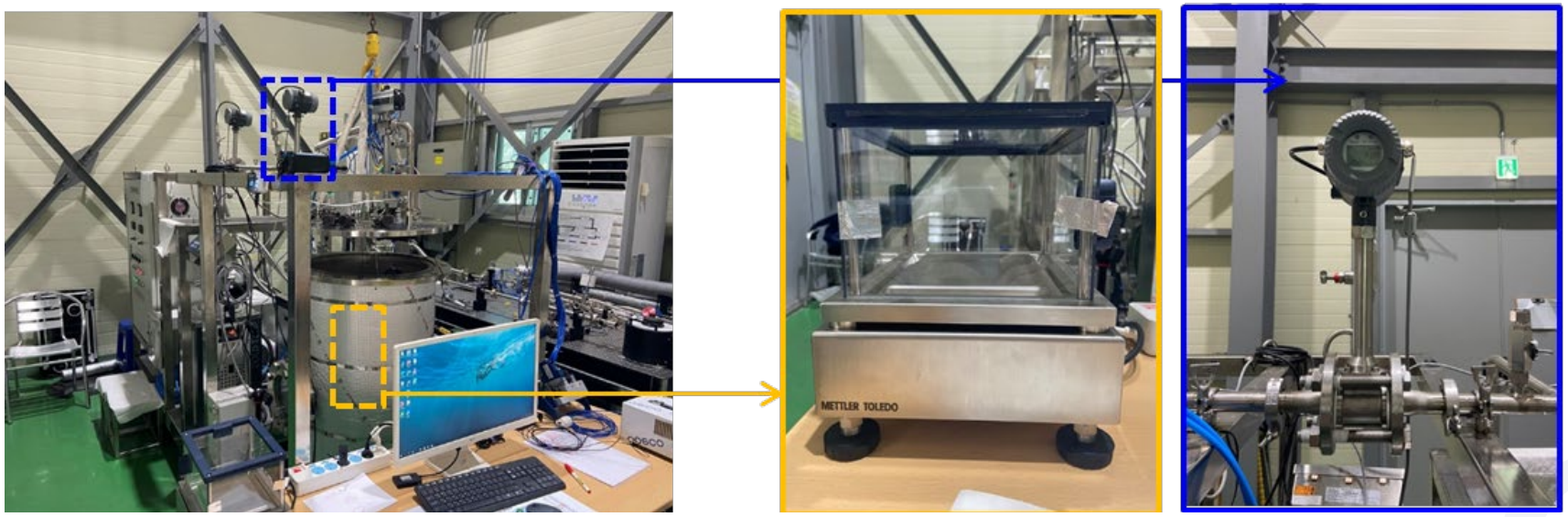
View of vacuum drying test equipment

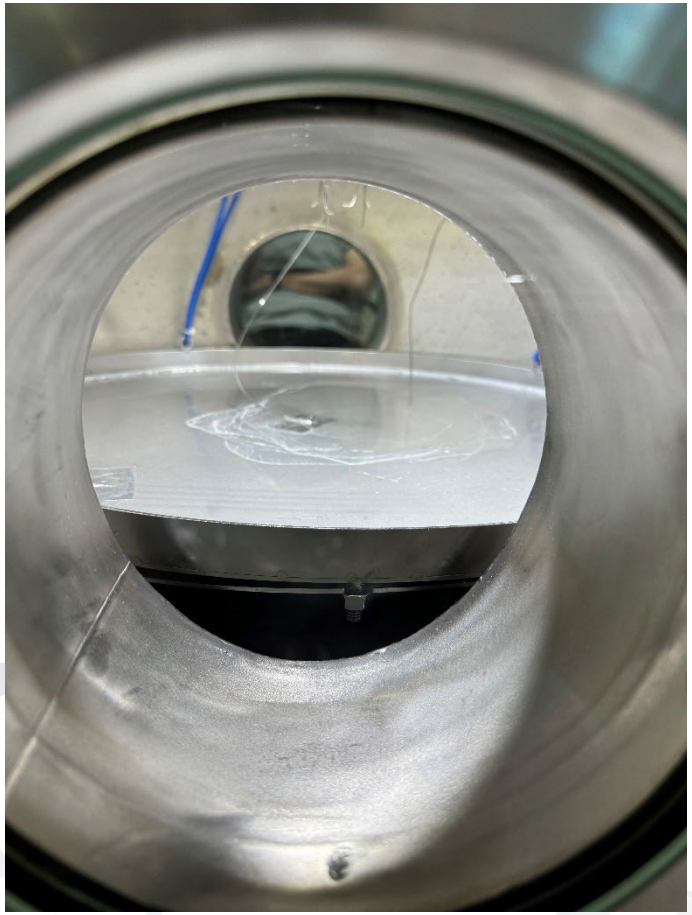
Vacuum drying pumps of various capacities



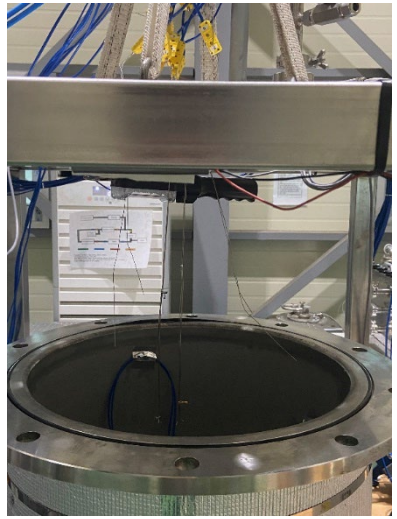
02 Vacuum drying test facility

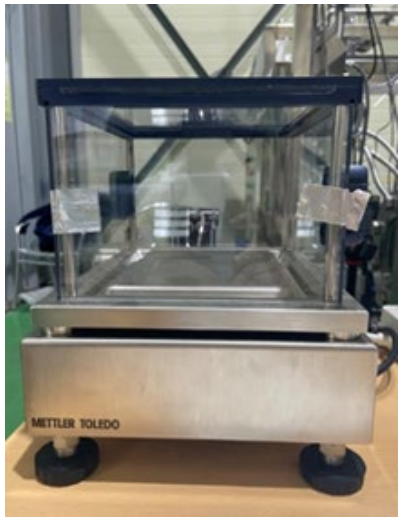
- **Canister height:** 0.5m, 1m, **Beaker capacity:** 600 mL, **TC interval**: 4.5 mm
- **Initial water height**: 1.8 cm, **Inner diameter of the beaker:** 88 mm
- **Vacuum pump capacity:** 100, 200, 400 L/min
- **Initial conditions of water:** 40℃, 100g, • **Evaporation area:** 6082.12 mm²
- **Vacuum stage presence:** presence or absence of stages, stage holding time
- Other physical properties used in calculations are based on NIST data

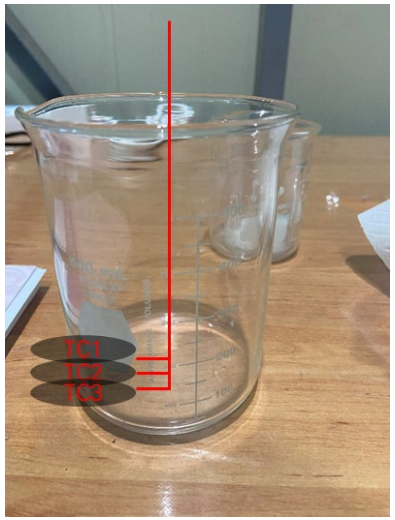


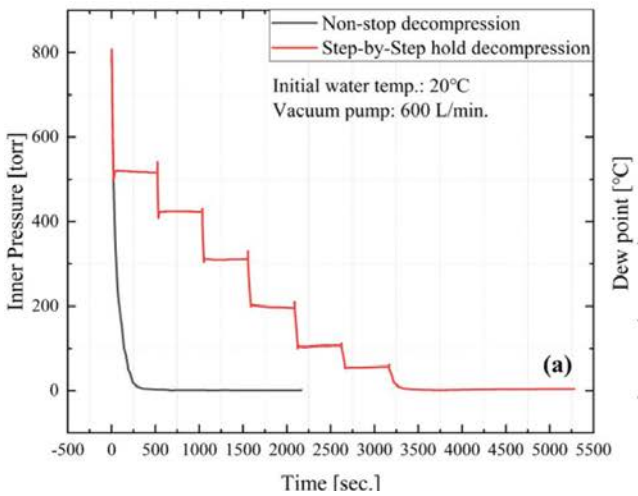


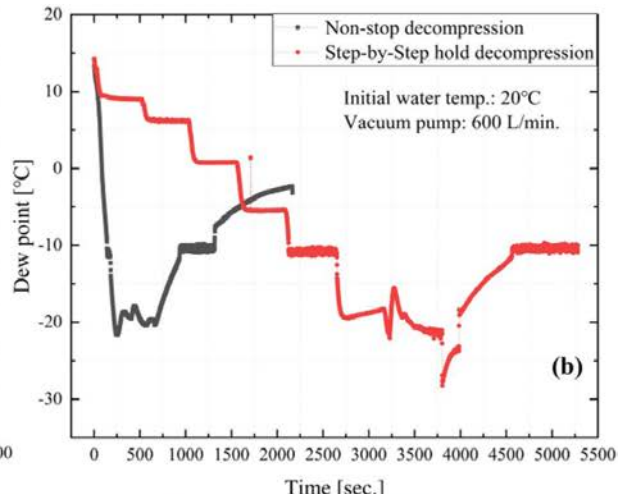
02 Vacuum drying test method and procedure











(1) Setup and Initiation: Select the appropriate canister volume and pump capacity.

- Power up the apparatus, ensuring all components are correctly initiated.
- Condition the cold trap by allowing it to cool below - 20 ◦C.

(2) Water Measurement and Equipment Preparation: Measure the water into a beaker.

- Open the canister lid and place the beaker on the micro-balance inside.
- Secure the lid and activate the measurement system.

(3) Thermocouple Placement and Monitoring: Adjust thermocouples, mounted on the canister lid, to ensure accurate placement at fixed locations within the beaker.

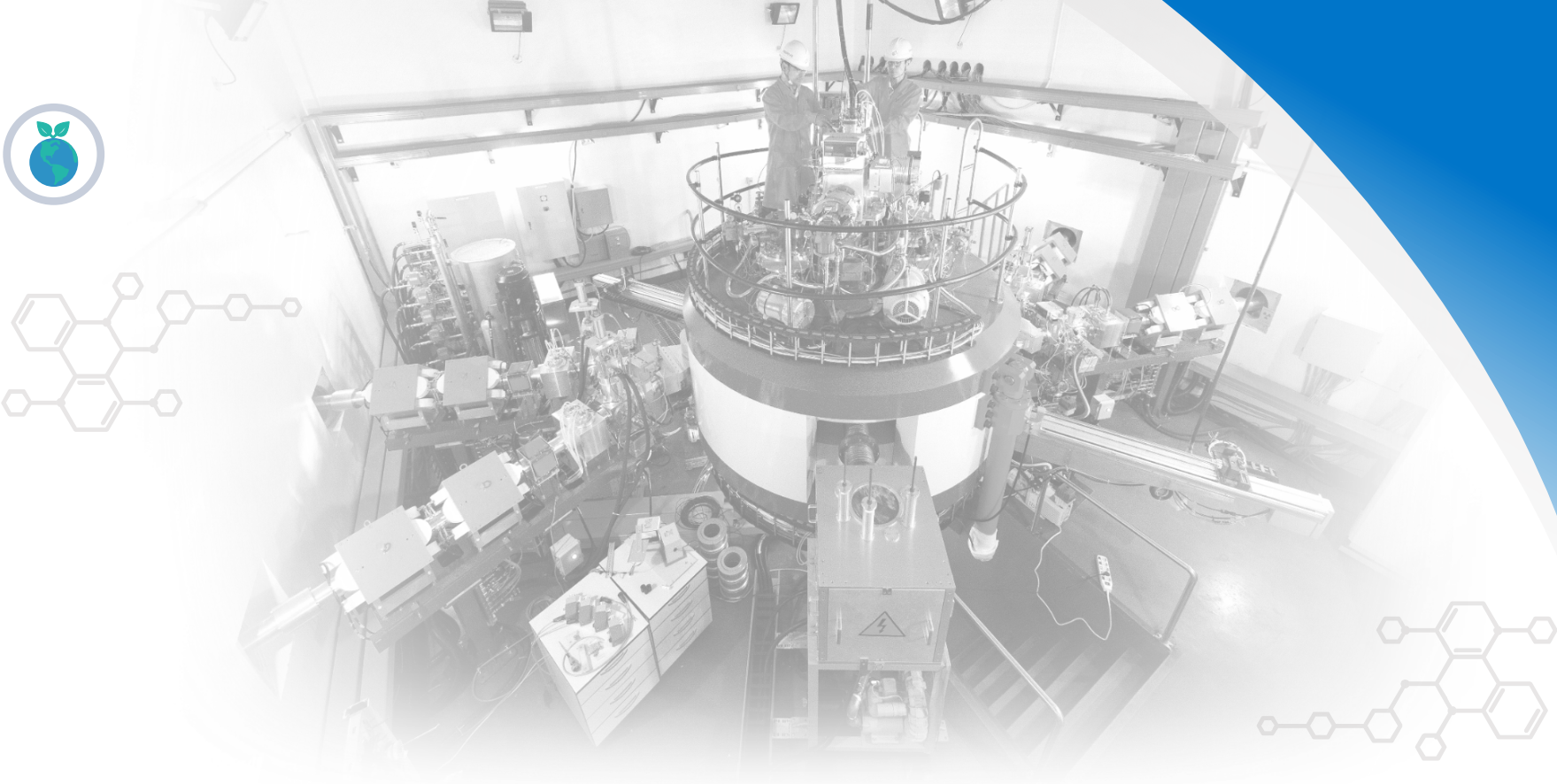
- Thermocouples are arranged linearly to measure temperatures at the water surface and at depths of 4.5 mm and 9 mm below the surface.
- Monitor temperature to detect and predict ice formation in any residual water.

(4) Measurement and System Monitoring: Start data recording as the vacuum pump achieves optimal conditions.

- Allow the temperature to stabilize at 0.1 ◦C.
- Use the thermocouple measurements to ensure the water surface temperature does not fall below 0 ◦C, preventing ice formation as a defined endpoint criterion.

(5) End of Experiment: Once standard-compliant vacuum drying criteria (<3 torr for 30 min.) or temperature criteria (preventing ice formation) are met and measurements are complete, power down the system.

- Normalize internal pressure and remove residual moisture.
- Validate no leakage and ensure full sensor functionality.



Towards Optimal Vacuum Drying: Enhanced Moisture Removal and Accurate Residual Water Quantification

Novel Vacuum drying methods

03

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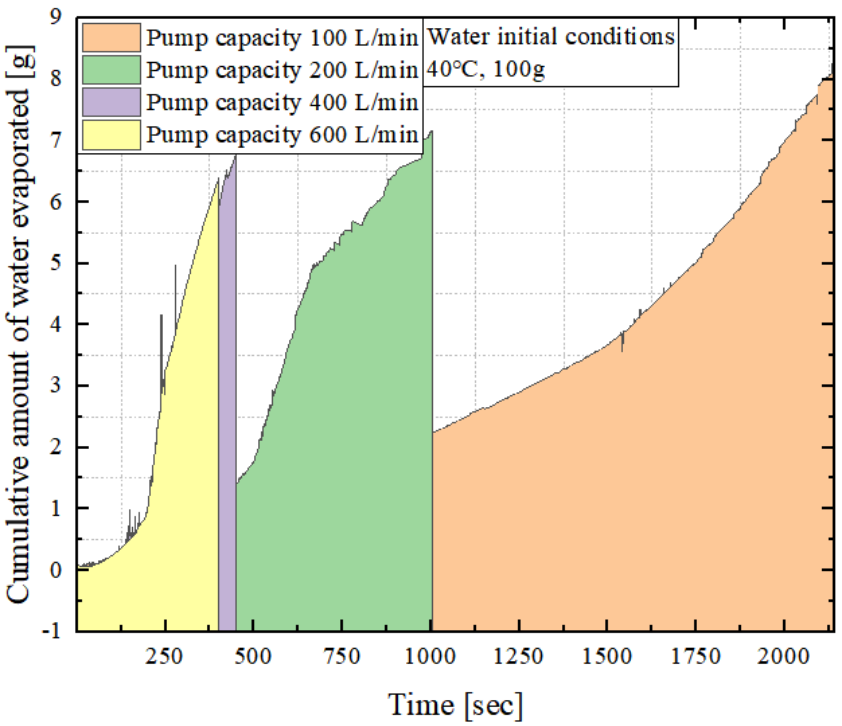
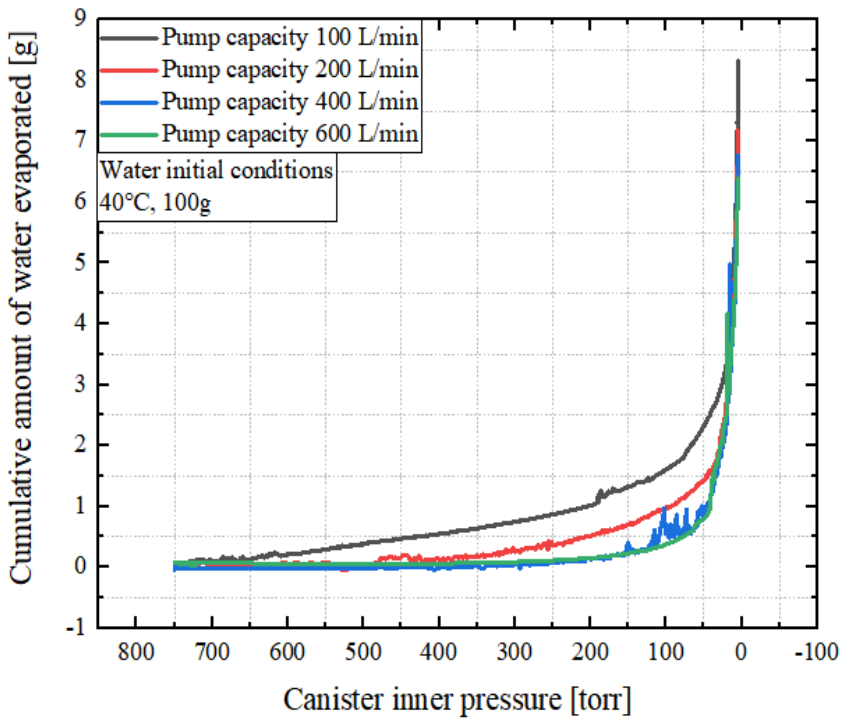
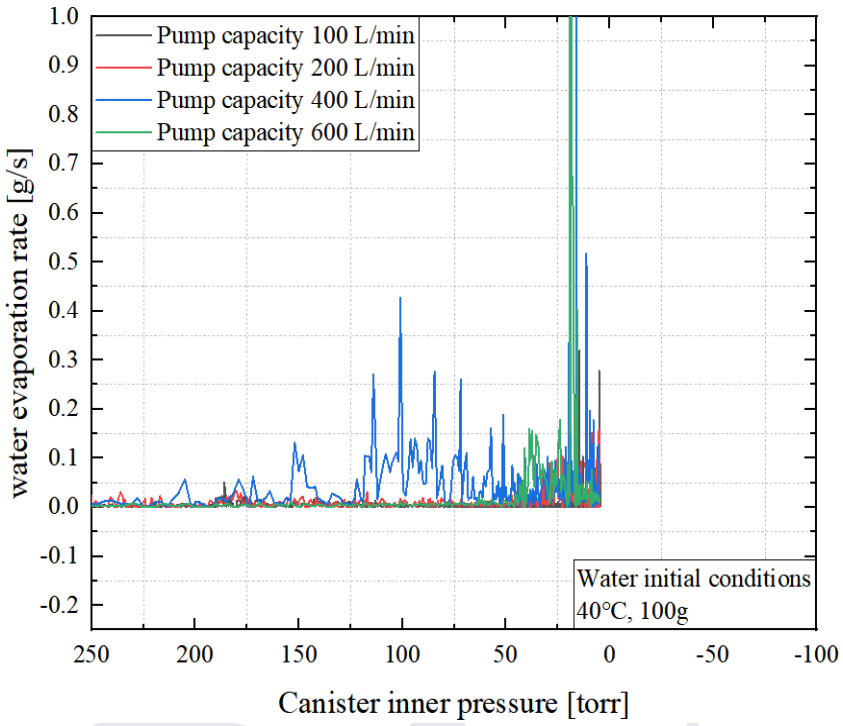
03 Evaporation-Acceleration Zone

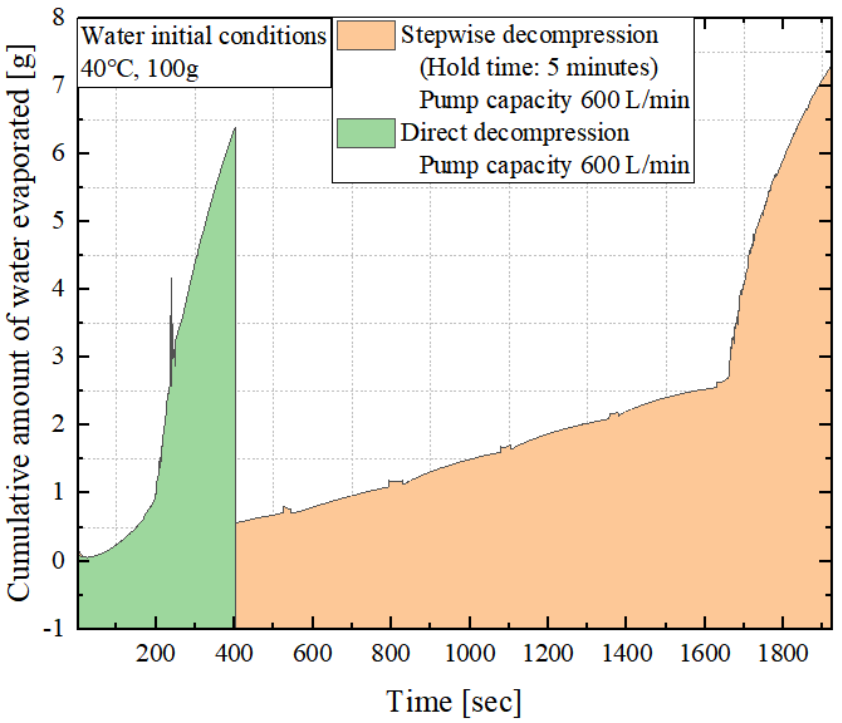


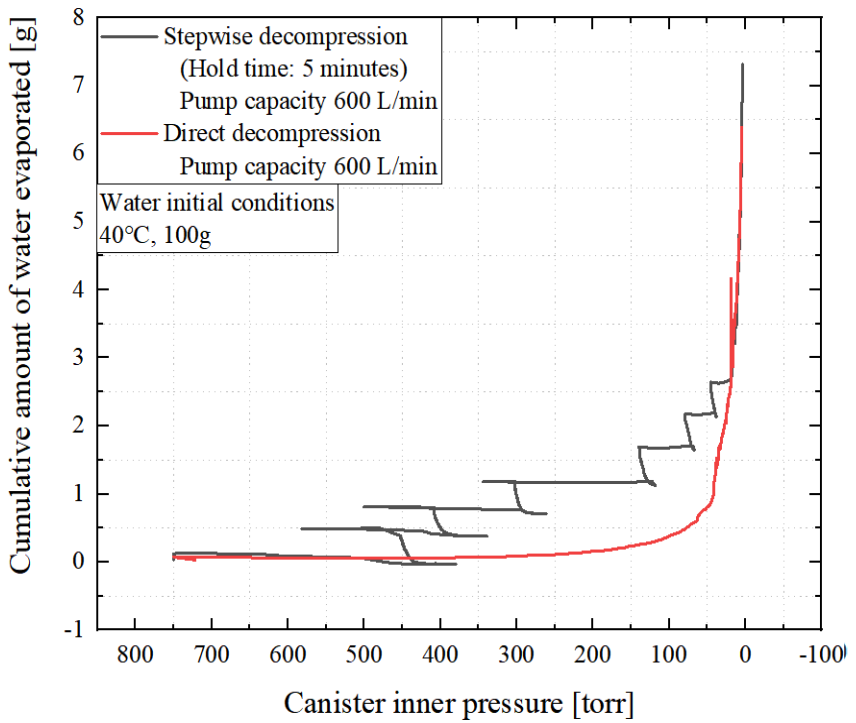


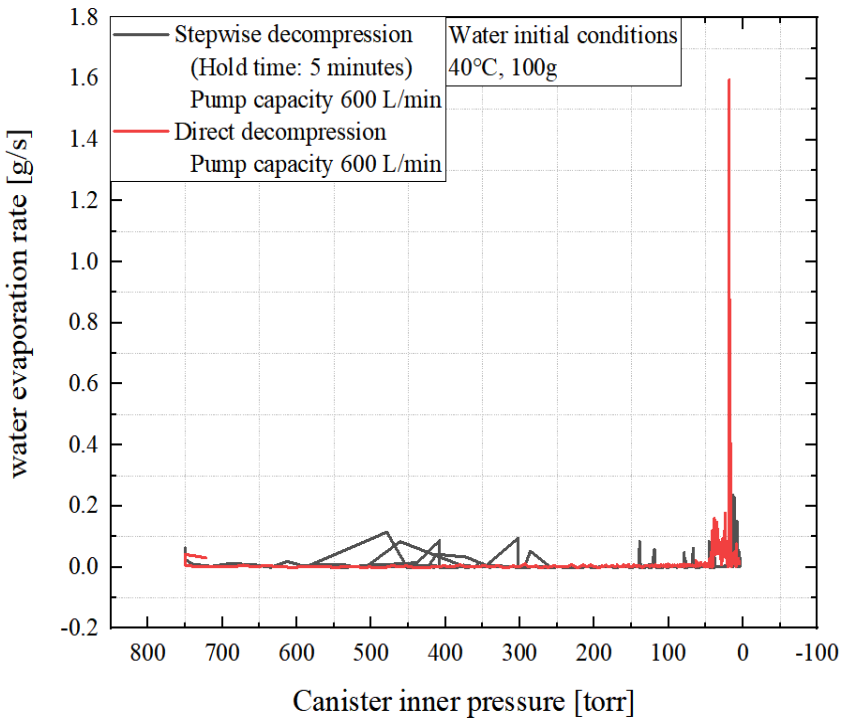


❑**Evaporation-Acceleration Zone Defined:** This zone, occurring between 40-50 torr, facilitates rapid moisture removal due to vigorous boiling, significantly enhancing evaporation rates. High-capacity pumps, like those at 600 L/min, expedite this process, highlighting their critical role in optimizing drying efficiency.

03 Evaporation-Acceleration Zone







❑**Efficiency Implications:** Entering the Evaporation-Acceleration Zone dramatically increases evaporation rates, achieving up to 1.5 to 2 g/s. However, while effective, high-capacity pumping can lead to early ice formation, necessitating careful pump rate management to optimize drying without premature phase transitions.

03 New vacuum drying method: holing time control

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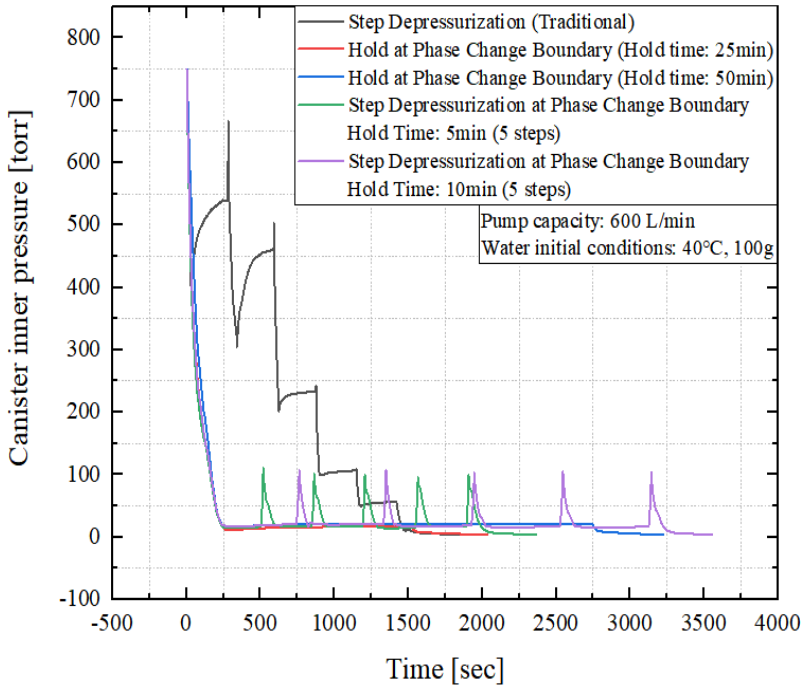
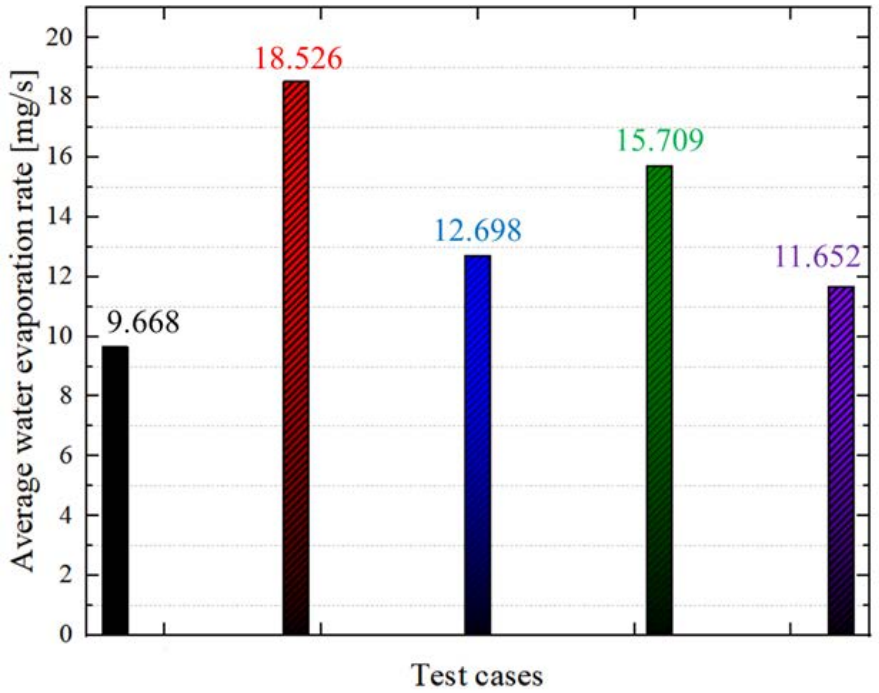
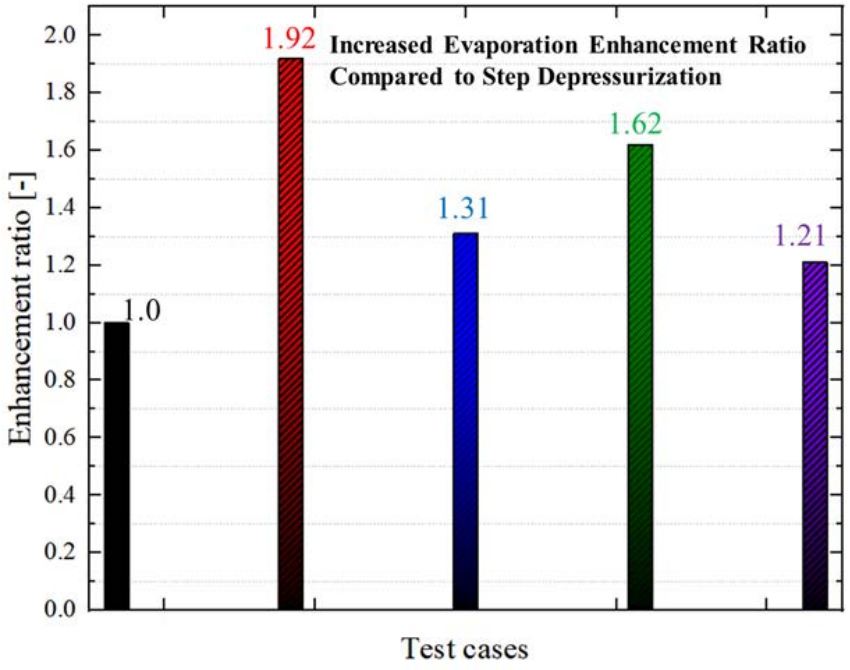
I. Decompression conditions for each strategic step to maximize evaporation during the vacuum drying process (**existing vacuum drying method**): To prevent ice formation.

II. Stepwise decompression method after entering the phase equilibrium boundary
(Hold: 25, 50 minutes): Increased efficiency by 31-91%.

III. Repeat hold and re-entry after entering the phase equilibrium boundary
(Hold: 5 minutes, 10 minutes): Increased efficiency by 20-62% compared to before.







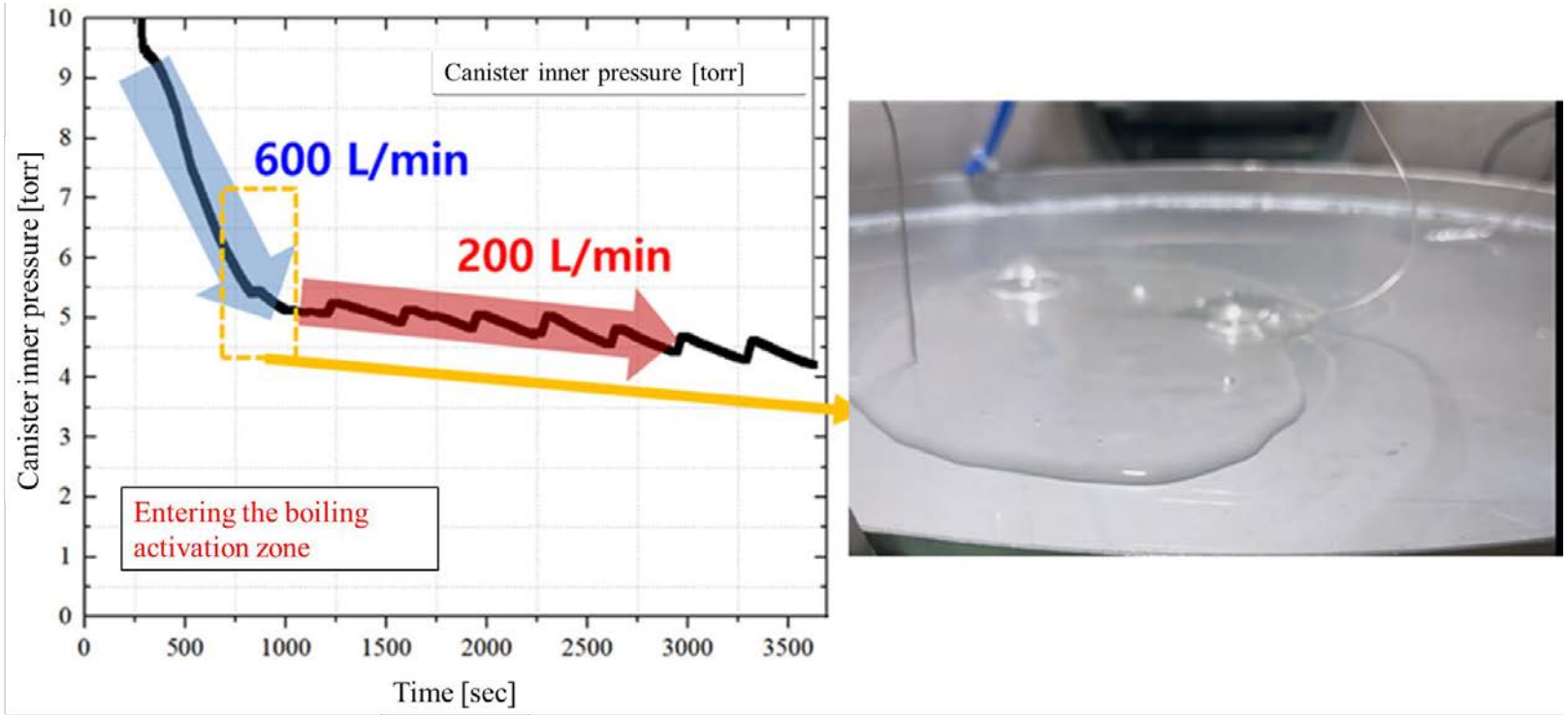
03 New vacuum drying method: holing time control

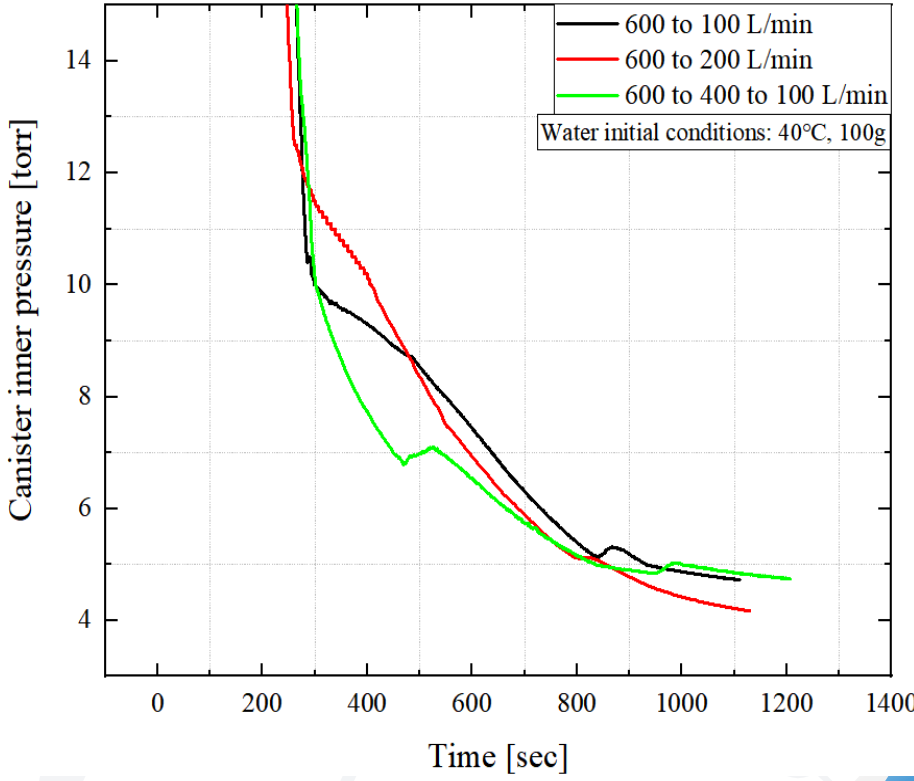
- **Enhancing Evaporation through Extended Hold Times**: Extending hold times at the Liquid Vapor phase change boundary boosts evaporation efficiency during vacuum drying.
- This approach optimizes residence time in the Evaporation Acceleration Zone, increasing moisture removal rates while minimizing ice formation. Combining direct and stepwise depressurization further refines pressure management, leading to significant improvements in drying efficiency.

	Initial hold pressure	Hold time [min]	Total time[s]	Total evaporation[g]	Eva. efficiency [g/sec.]	Enhancement ratio
Decompression conditions (conventional method)	400 torr	10	3227	31.2	9.668	**1.0**
	500 torr	10	3230	33.8	10.464	**1.08**
Stepwise decompression method after entering the phase equilibrium boundary	phase boundary	25	2035	37.7	18.526	**1.92**
		50	3213	40.8	12.698	**1.31**
Repeat hold and re-entry after entering the phase boundary		5.	2368	37.2	15.709	**1.62**
		10.	3553	41.4	11.652	**1.21**

03 New vacuum drying method: pump control

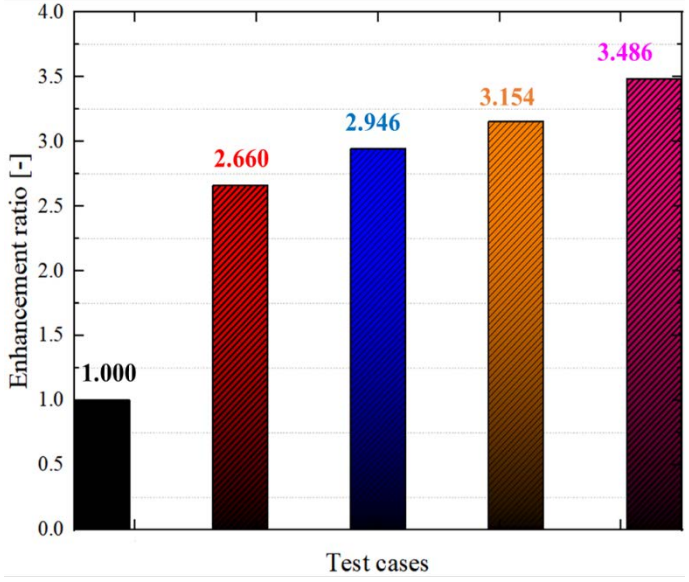
- **Pump Modulation for Enhanced Evaporation:** Adjusting pump flow rates at the Liquid Vapor phase change boundary significantly boosts evaporation efficiency. Gradually reducing from 400 L/min at ~10 torr to 100 L/min maximizes moisture removal and minimizes ice risk.

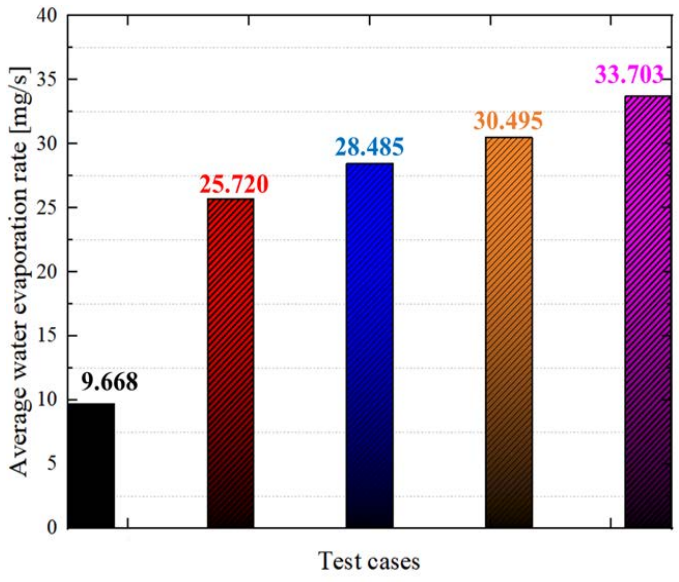


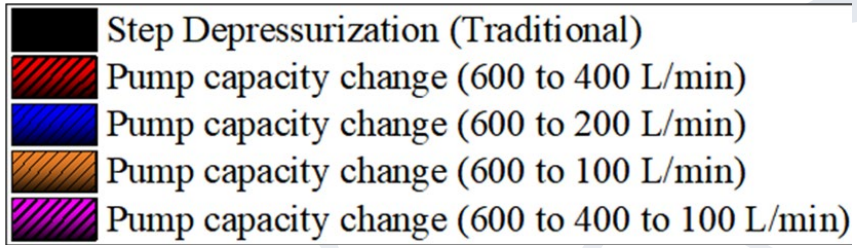


03 New vacuum drying method: pump control

	Total time [sec.]	Pump capacity [Liter/min]		Evaporation mass [g]	Eva. efficiency [g/sec.]	Enhancement ratio
Pump flow rate control during drying tests	9515	600	No change.	100	10.50	**1.08**
	860	600	400 (at 40 torr)	25.026	29.10	**3.00**
	1574	600	200 (at 40 torr)	38.575	24.51	**2.53**
	1406	600	100 (at 40 torr)	29.38	20.89	**2.16**
	1191	600	200 (at 10 torr)	33.34	27.99	**2.89**
	1215	600	400 (at 10 torr)	31.25	25.72	**2.66**
	1129	600	200 (at 10 torr)	32.16	28.48	**2.94**
	1110	600	100 (at 10 torr)	33.85	30.49	**3.15**
	1207	600	400 - 100 (at 10 torr)	40.68	33.70	**3.48**
77.9g drain hole tests	1811	600	No change.	51.46	28.41	**2.93**
	4240	600	100 (at 10 torr)	60.29	14.21	**1.47**





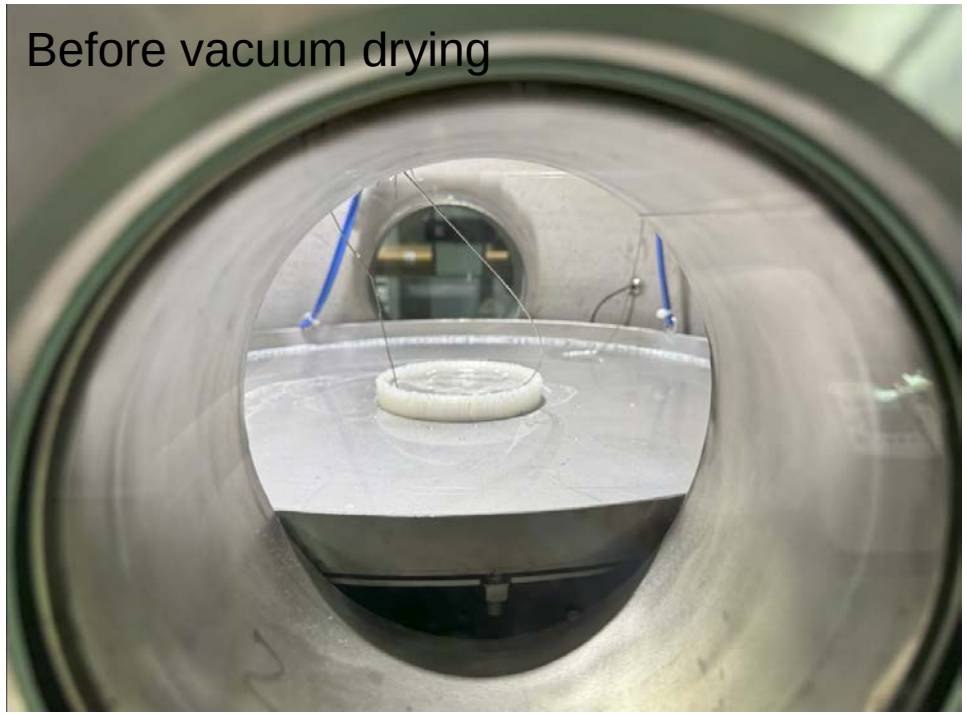


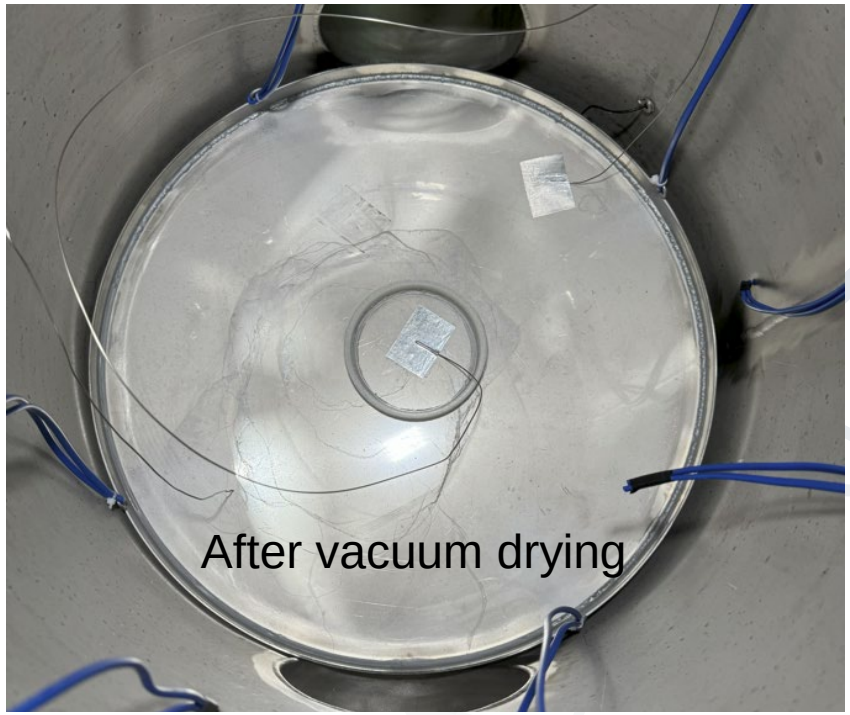
- **achieving up to 3.486 times the efficiency of traditional methods.**

03 Vacuum drying test in a situation simulating a drain hole

- Vacuum drying tests were conducted on a structure that replicates the KORAD-21 canister drain hole sizes (100mm diameter and 10mm height).
- The maximum amount of water that can theoretically exist in this structure is 77.9g.

	Total time [sec.]	Pump capacity [Liter/min]		Evaporation mass [g]	Eva. efficiency [g/sec.]	**Enhancement ratio**
77.9g drain hole tests	1811	600	No change.	51.46	28.41	**2.93**
	4240	600	100 (at 10 torr)	60.29	14.21	**1.47**

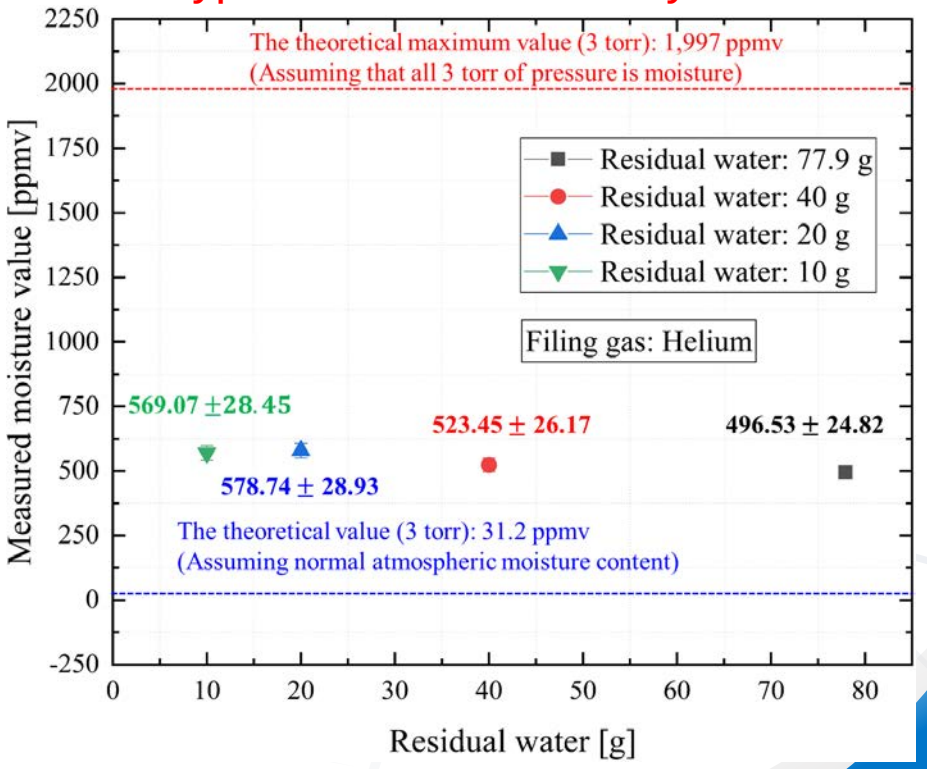


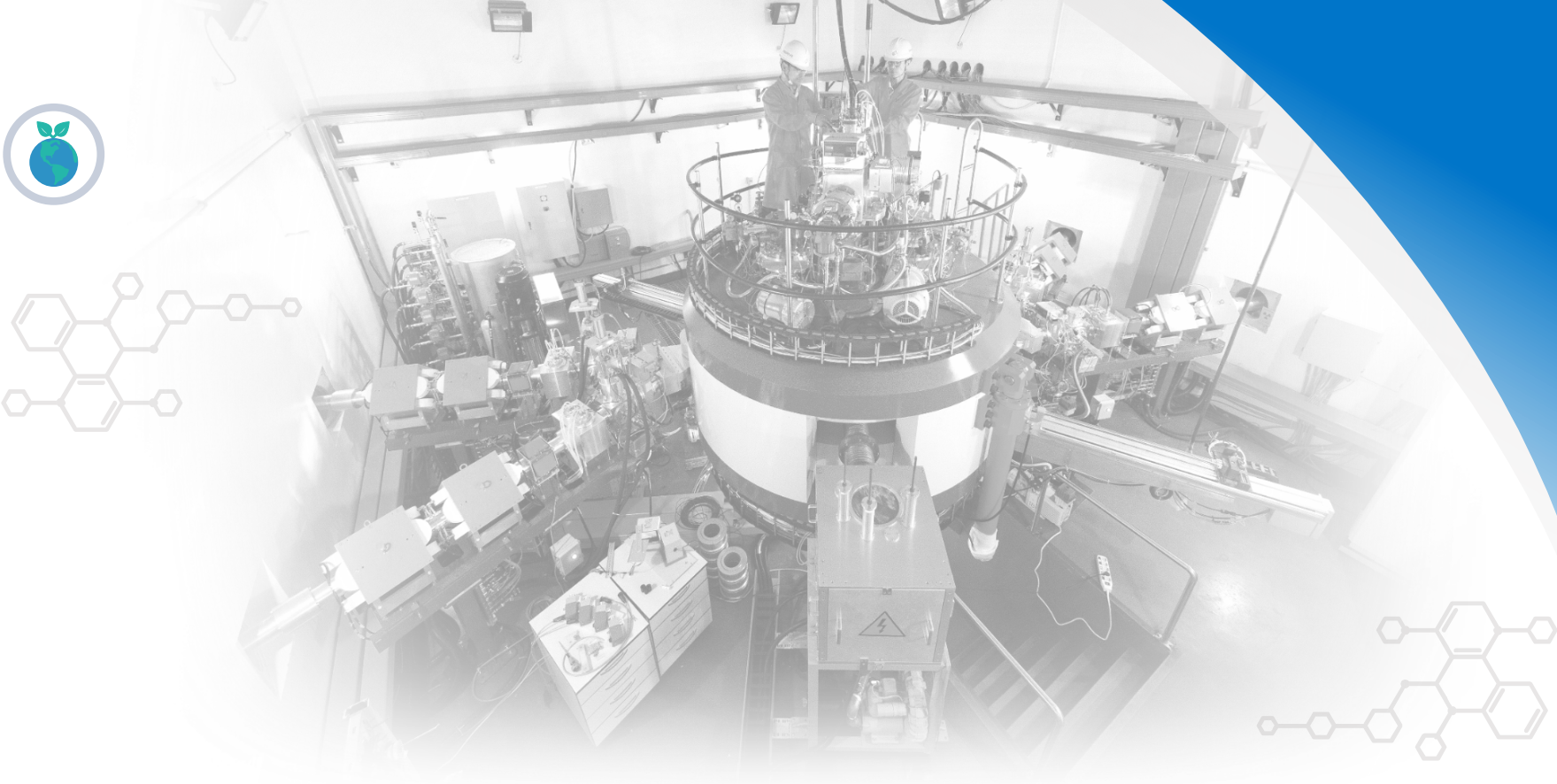


03 Quantification of residual water after vacuum drying

- Direct moisture measurements were conducted using an AquaVolt 2 electrolytic trace moisture analyzer, yielding highly resolved residual water concentrations in ppmv.
- The results revealed that despite differences in the initial water loading, the final internal moisture levels consistently stabilized within ~500–600 ppmv (corresponding to ~0.15 g), demonstrating a predictable moisture endpoint under standard-compliant drying.
- Comparative analyses showed that these values were significantly lower (by ~3–4×) than conservative worst-case assumptions but markedly higher (~15–20×) than estimates based on typical ambient humidity entrainment.

Test cases	Filing gas	Amount of water inside the canister [g]	Measured moisture value [ppmv]	The moisture content in the gas cylinder [ppmv]	The moisture content after standard-compliant vacuum dying		The theoretical maximum value (3 torr) [g]	The theoretical value (3 torr) [g]
					[ppmv]	[g]		
Case #1	Helium	10	599.10	30.03	569.07 ± 28.45	0.162 ± 0.00810	0.569 (Assuming that all 3 torr of pressure is moisture)	0.0089 (Assuming normal atmospheric moisture content)
Case #2	Helium	20	607.15	28.41	578.74 ± 28.93	0.164 ± 0.00820		
Case #3	Helium	40	552.84	29.39	523.45 ± 26.17	0.149 ± 0.00745		
Case #4	Helium	77.6	577.91	32.76	545.15 ± 27.25	0.155 ± 0.00775		
Case #5	Helium	77.9	515.82	19.29	496.53 ± 24.82	0.141 ± 0.00705		





Towards Optimal Vacuum Drying: Enhanced Moisture Removal and Accurate Residual Water Quantification

Vacuum drying of damaged fuel

04

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04 Test specimen & methodological framework

❑ The damaged nuclear fuel was fabricated using Zircaloy alloy material, with its design dimensions adopted from both PLUS7 and ACE7 fuel specifications, which share identical outer and inner diameters. The PLUS7 damaged fuel rod features an outer diameter of 9.5mm and an inner diameter of 8.357mm.

Figure. Test section of vacuum drying: simulated damaged spent fuel

04 Step-by-Step execution of vacuum drying Protocol

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(1) Preparation and Setup:

- Carefully select the vacuum pump capacity. Activate the lab-scale vacuum drying equipment and ensure the cold trap is below -20°C.
- Use a crane to open the canister lid and position the rack and simulated damaged nuclear fuel for accurate measurement.

(2) Water Infusion and Monitoring:

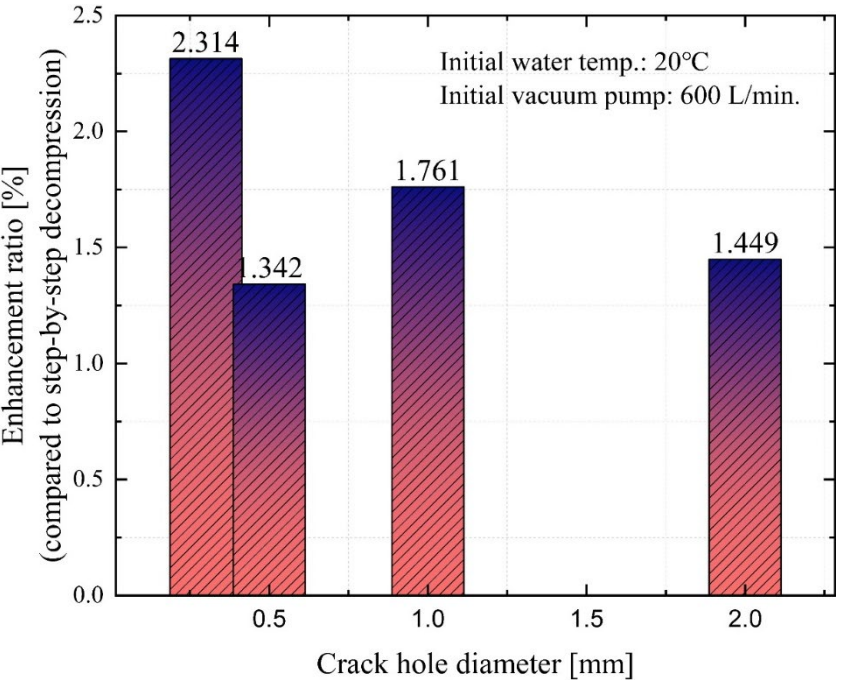
- Introduce 2 g of thermally conditioned residual water into the canister using a syringe.
- Adjust the fuel position for visibility of any cracks. Carefully place the thermocouple in the water, then secure the canister lid.

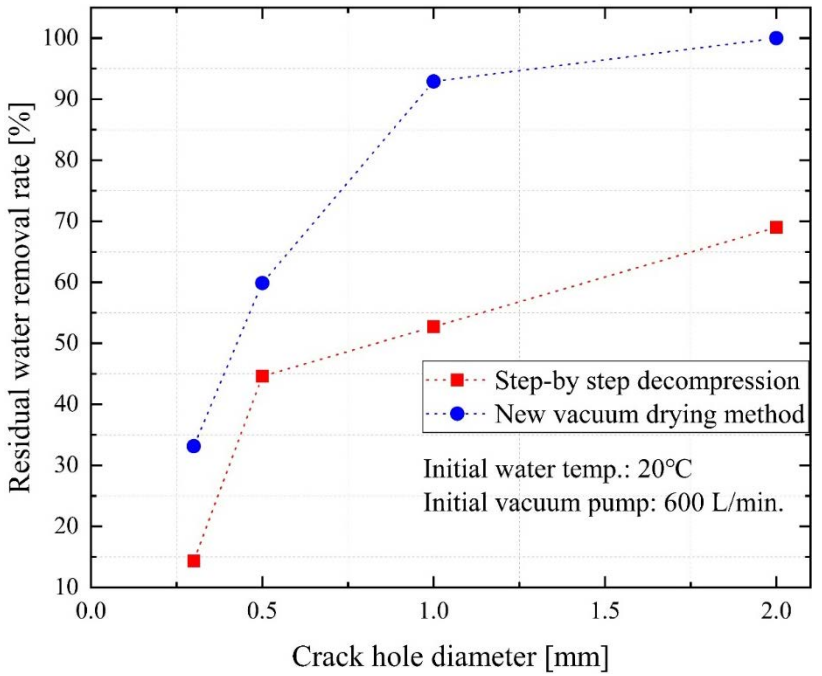
(3) Data Collection and Pressure Maintenance:

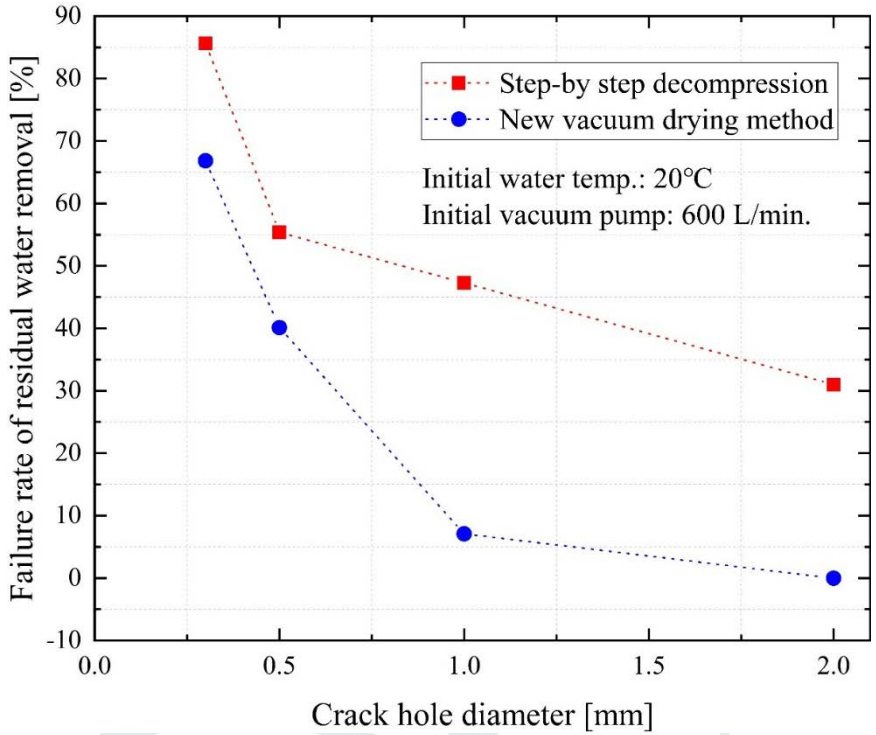
- Start the computer systems and measurement equipment, enter a unique file name, and begin data collection. Reduce canister pressure to 1.5 Torr, maintaining it at 3 Torr for 30 min.
- Once validated, end the measurement program, deactivate the equipment, and return pressure to atmospheric conditions.

Video. Test section of vacuum drying: simulated damaged spent fuel

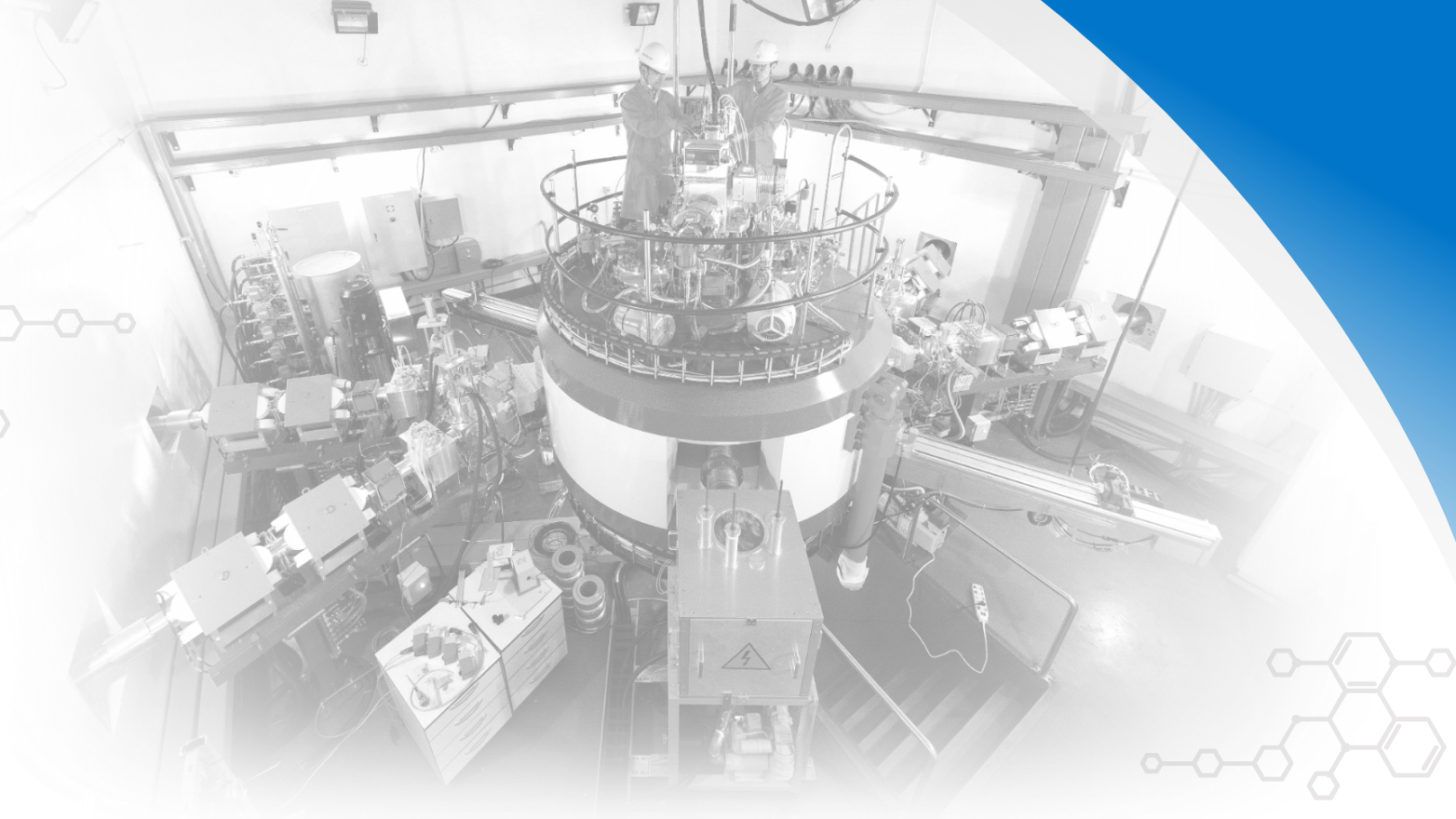
04 Advanced vacuum drying technique for residual water removal from damaged fuel







- **Novel Vacuum Drying Method:** By employing strategic pump capacity modulation and phase change boundary manipulation, the method significantly improves water removal efficiency.
- It achieves up to 2.314 times greater evaporation rates and complete water removal for 2.0mm crack cases, offering a breakthrough in addressing corrosion risks associated with residual moisture in nuclear fuel storage.



Novel strategies in vacuum drying for advanced moisture removal in spent nuclear fuel canister

Conclusion and Recommendations

05

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Summary

Conclusion and Summary

Review Related to Vacuum Drying

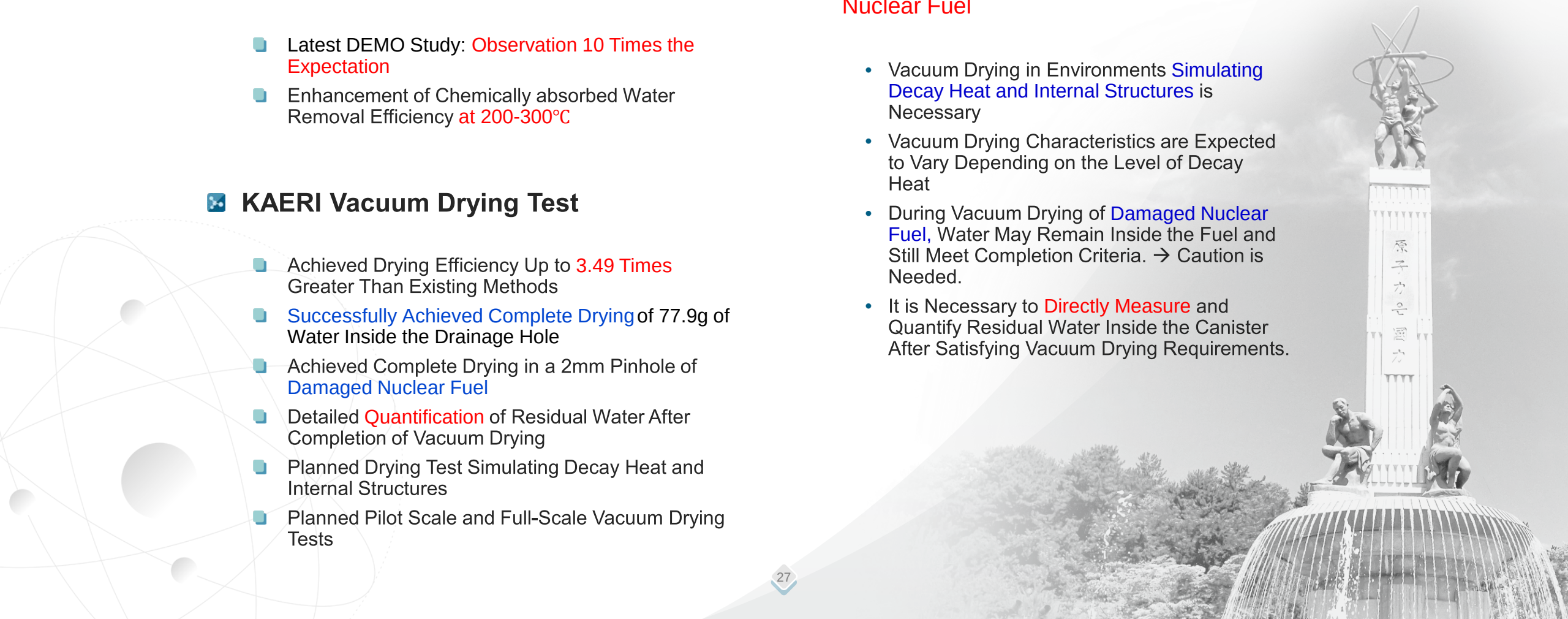
- Latest DEMO Study: Observation 10 Times the Expectation
- Enhancement of Chemically absorbed Water Removal Efficiency at 200-300℃

KAERI Vacuum Drying Test

- Achieved Drying Efficiency Up to 3.49 Times Greater Than Existing Methods
- Successfully Achieved Complete Drying of 77.9g of Water Inside the Drainage Hole
- Achieved Complete Drying in a 2mm Pinhole of Damaged Nuclear Fuel
- Detailed Quantification of Residual Water After Completion of Vacuum Drying
- Planned Drying Test Simulating Decay Heat and Internal Structures
- Planned Pilot Scale and Full-Scale Vacuum Drying Tests

Recommendations

- Vacuum Drying Simulating Actual Spent Nuclear Fuel
  - Vacuum Drying in Environments Simulating Decay Heat and Internal Structures is Necessary
  - Vacuum Drying Characteristics are Expected to Vary Depending on the Level of Decay Heat
  - During Vacuum Drying of Damaged Nuclear Fuel, Water May Remain Inside the Fuel and Still Meet Completion Criteria. → Caution is Needed.
  - It is Necessary to Directly Measure and Quantify Residual Water Inside the Canister After Satisfying Vacuum Drying Requirements.



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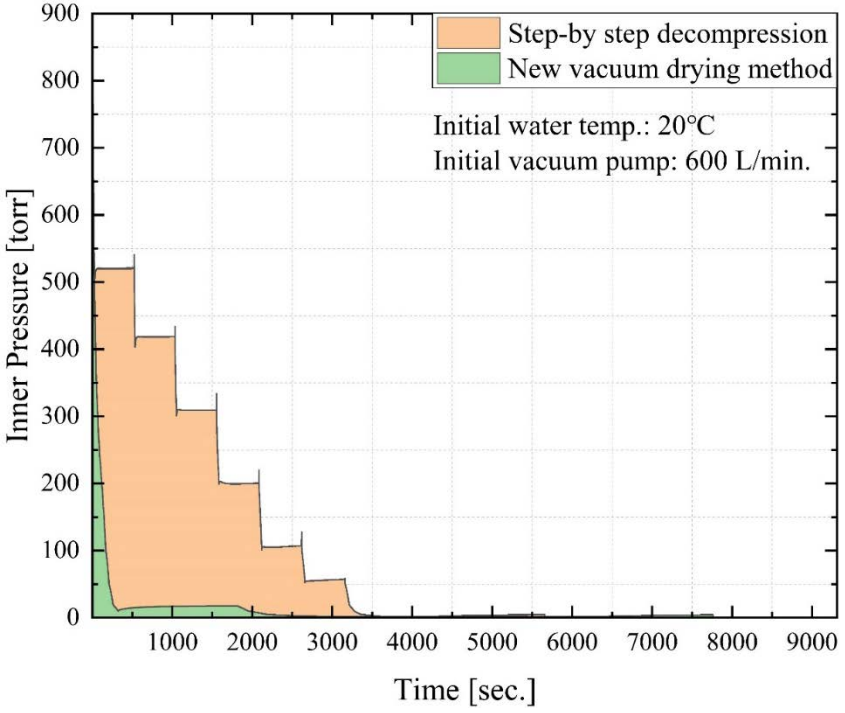
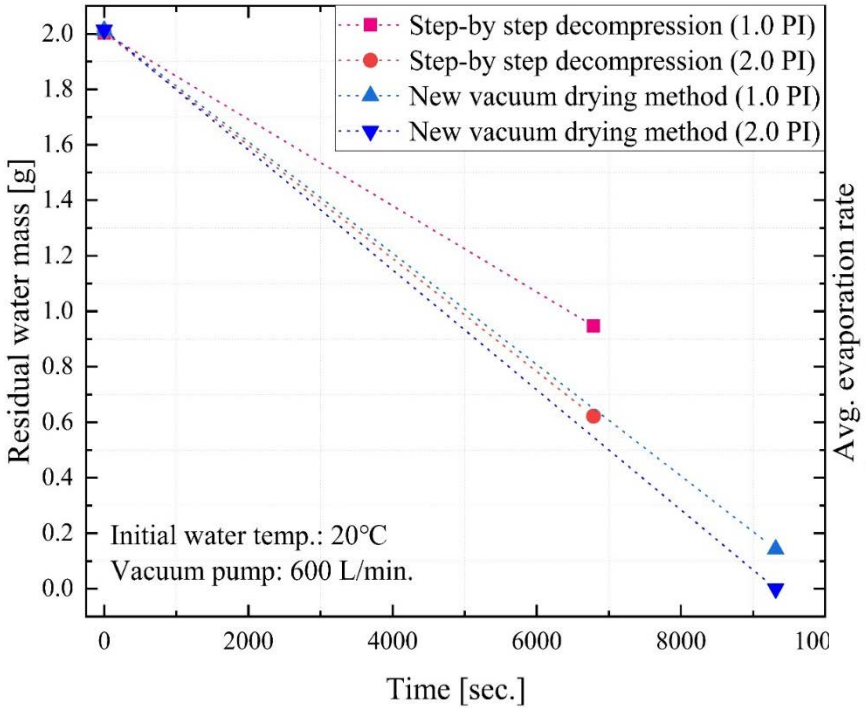
Novel strategies in vacuum drying for advanced moisture removal in spent nuclear fuel canister

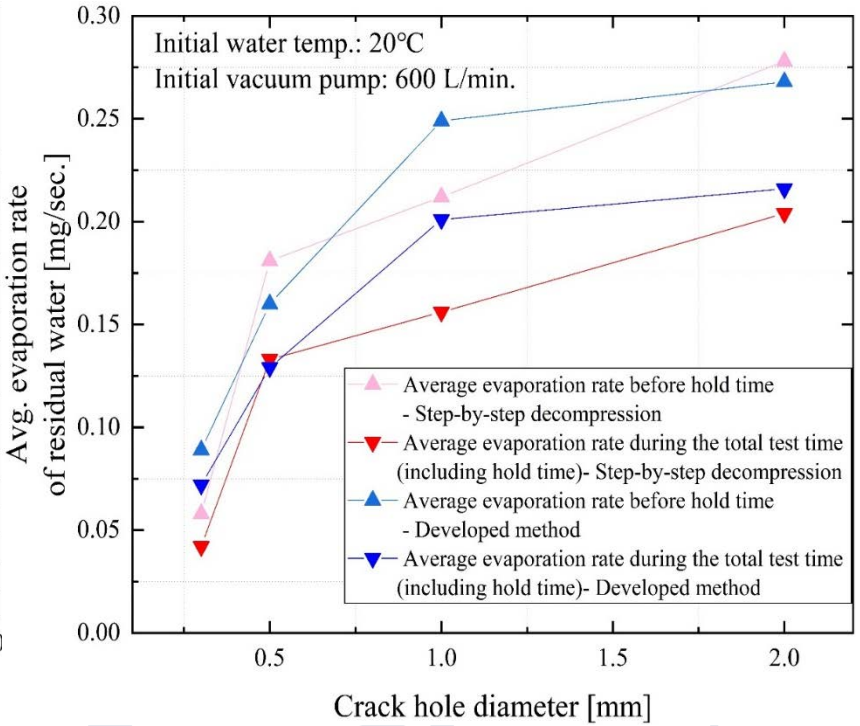
THANK YOU

This work was supported by the Institute for Korea Spent Nuclear Fuel (iKSNF) and National Research Foundation of Korea (NRF) grant funded by the Korea government (Ministry of Science and ICT, MSIT) (No. 2021M2E1A1085226).

04 Advanced vacuum drying technique for residual water removal from damaged fuel







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01 FHD vs. Vacuum drying

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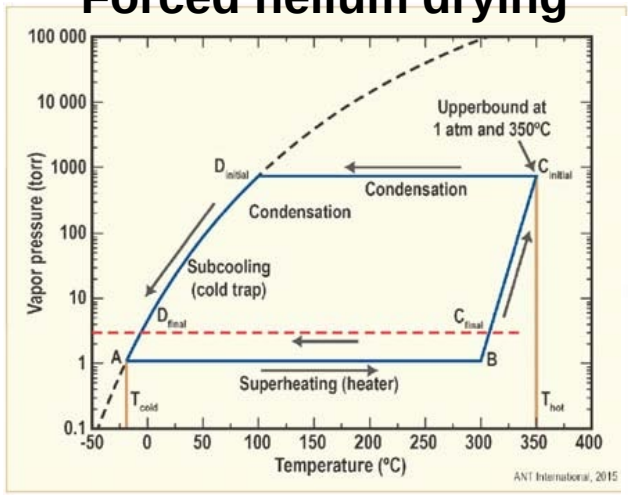
Table 2-1. Comparison of Vacuum Drying and Forced Helium Dehydration Methods

Drying Method	Vacuum Drying	Forced Helium Dehydration
Criterion	Maintain vacuum of 3 torr for 30 minutes	Exit recirculating gas temperature of −6 °C [≤ 21 °F] for ≥ 30 minutes*
Efficiency/Process Time	Estimated time range of 30 to 65 hours	Estimated to be 16 to 38 hours†
Advantage	- Industry standard and accepted method of drying‡ - Simple air-cooled vacuum drying systems eliminate need for external heating and cooling units - Less expensive to purchase and maintain	- Eliminates the possibility of ice formation in canister - Maintain cladding temperature due to heat transfer of recirculating gas
Disadvantage	- Uncertainty of possible formation of ice in canister - Difficult to judge whether there has been sufficient time for sublimation of ice - Note: At pressures below 4.7 torr, ice may form in canister requiring 3 to 4 times longer for removal§	- Internal configuration (e.g., basket structure) may prevent adequate flow or circulation of gas over all surfaces in canister - May have difficulty with pooled water in canister bottom - Requires more complicated equipment setup (e.g., helium circulator), preheat and chiller module

References
*Holtec International. "Final Safety Analysis Report for the Hi-Storm 100 Cask System." Rev. 4. USNRC Docket No. 72–1014. West Marlton, New Jersey: Holtec International. April 4. 2006.
†Bencel, K. "TAD/STC Drying and Inerting Calculation." 000–M0C–MR00–00100–000–00A. Washington, DC: Office of Civilian Radioactive Waste Management. 2007.
‡NRC. NUREG–1536, "Summary Review Plan for Dry Cask Storage Systems." Washington, DC: U.S. Nuclear Regulatory Commission. 1997.
§Kelly, J.G. "The Myths and Realities of Vacuum Drying." Dry Storage Information Forum, May 15–17, 2007. Clearwater Beach, Florida: Published on CD ROM. Washington, DC: Nuclear Energy Institute. 2007.
Note: [1 torr = 1 mm Hg]
[1 °F = 1.8 °C + 32]

Forced helium drying



Vacuum drying

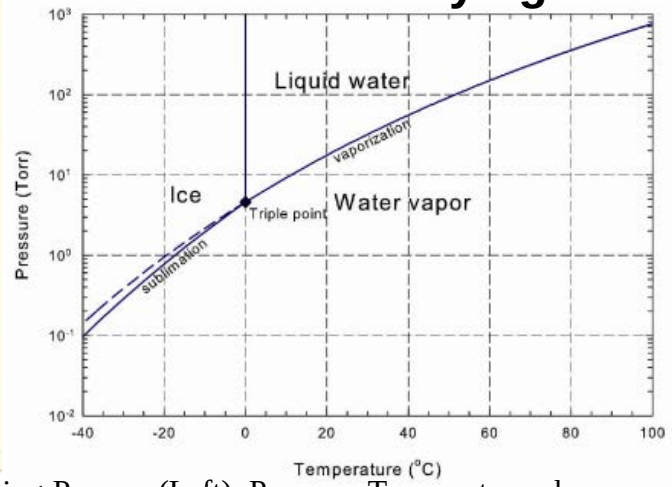


Fig. Pressure–temperature cycle of helium drying Process (Left), Pressure-Temperature phase diagram for pure water (Right)

Table 1. Multi-purpose canister (MPC) cavity drying limits

Fuel Burnup (GWd/MtU)	MPC type	MPC Heat Load (kW)	Method of Moisture Removal
All assemblies≤ 45	MPC-37	≤47.05	VDS or FHD
	MPC-89	≤46.36	
One or more assemblies > 45	MPC-37	≤29.6	VDS or FHD
	MPC-89	≤30.0	
One or more assemblies > 45	MPC-37	≤47.05	FHD
	MPC-89	≤46.36	

* VDS (Vacuum drying system), FHD (Forced helium dehydration system)

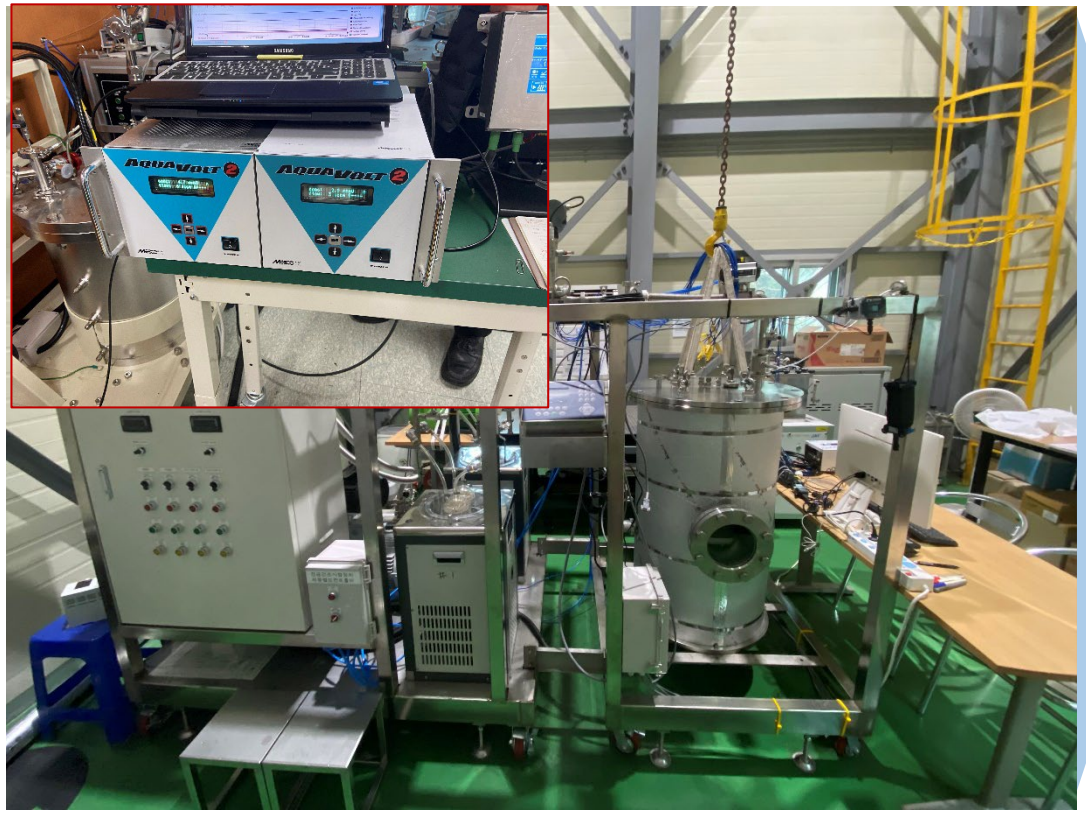
Chae, G. S., Shin, K. W., Park, B. M., Han, J. H., Lee, G. H., & Park, J. S. (2017). JNFCWT, 15(4), 403-409.

04 Appendix.

Year	Authors	Journal	Title	Summary of key contents
2023. 07	d'Entremont, Anna, et al.	Nuclear Technology	Drying of Spent Nuclear Fuel: Considerations and Examples	• Important considerations in drying include characteristics of the fuel assemblies and canister internal components, such as their areas, which can hold physisorbed and chemisorbed water. • reactor-induced breaches (~1-mm holes): small release (<1 g) • drill holes of 1 mm or 2.4 mm: small release (<1 g) • For commercial SNF, inadvertent free water, **even up to large amounts (e.g., 10+ mol), is not predicted to cause adverse corrosion degradation**, except possible fuel oxidation for breached SNF.
2023. 02	Ayanoglu, Muhammet al. (WM2023 Conference)	Oak Ridge, TN (United States), 2023.	The Effect of Vacuum Drying on Hydride Reorientation in High-Burnup Nuclear Fuel Claddings for Extended Dry Storage	• This study investigated the effect of vacuum drying on hydrogen behavior and the resulting hydride morphology in different baseline and full-length heat-treated (FHT) high-burnup (HBU) spent nuclear fuel (SNF) claddings (US DOE HBU Spent Fuel Data Project). • For this purpose, three HBU fuel rods—M5, Zircaloy alloy, and Zircaloy-4 (Zirc-4)—were heat treated at 400°C using a spent fuel rod FHT oven.
2023. 10	Bae, Junghyun et al.	No. ORNL/SPR -2023/3081.	A Vacuum Drying Study of Simulated Failed Nuclear Fuel (FY23).	• There are two related objectives of this work. The first is to develop a simple and inexpensive experiment to better understand removal of water from inside failed spent nuclear fuel (SNF) rods, • The second is to develop a model to predict drying of simulated failed fuel in the simple experiment. Once a model captures the phenomenon in the simple experiment, it can be extended to predict drying under more prototypical conditions.
2023. 01	Eidelpes, Elmar, et al.	Journal of Nuclear Materials	Technical basis for extended dry storage of aluminum-clad spent nuclear fuel	• Experimental work was conducted, including investigations of the performance of surrogate aluminum specimens and actual ASNF material featuring adherent (oxy)hydroxide layers during wet or dry storage. • Drying experiments were executed, providing better estimations of the amount of residual, chemisorbed, and physisorbed water after ASNF dry storage canister conditioning.

05 Future works

- Expected to receive Aquavolt, direct measurement equipment for residual water
- Testing is conducted by installing Aquavolt on current Pilot-scale equipment.
- Install Aquavolt in a ½ bundle canister and measure the amount of residual water after drying.



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ASNF Dry Storage Pilot

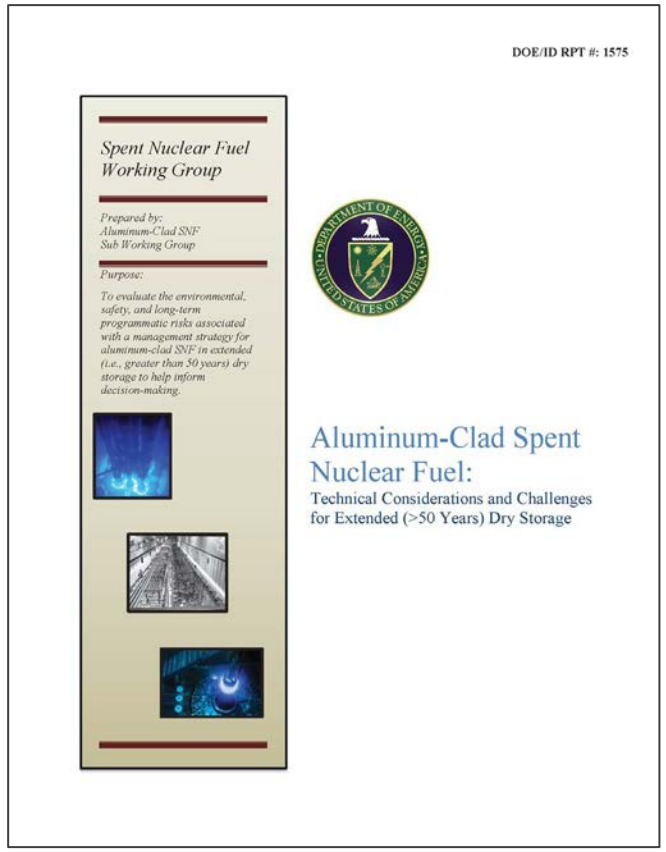
Verification & Validation of Drying and Dry Storage Performance of Aluminum-Clad, Aluminum-Based Spent Nuclear Fuel (ASNF) for Extended (> 50 Years) Safe Storage

Drew Fairchild (PM), Anna d'Entremont, Claire Morgan, Fernando Farelas, Bob Sindelar, Rich Wyrwas, Bill Bates

EPRI ESCP October 30, 2025

Background and Motivation for ASNF Dry Storage Pilot

- DOE SNF storage sites (Savannah River and Idaho) have large inventories of Research Reactor ASNF
- DOE-EM/NE SNF Working Group identified challenges to ASNF extended (> 50 years) dry storage
  - Key challenge is chemically-bound water (average up to ~ 25 μm thick) as aluminum (oxy)hydroxides (AlOOH or $Al(OH)_3$) with radiolytic H_2 gas generation therefrom
- INL/SRNL closed gaps and established the technical bases to enable vented/sealed storage
- **Verify:**
  - **Heated drying can reduce residual water via breakdown of hydrated oxides**
  - **Radiolytic H_2 will build-in and subside (with finite residual water) with insignificant O_2 generation**
- **Validate:**
  - **Drying and sealed dry storage in the multipurpose DOE standard canisters (DSCs) provides a system for safe extended storage**


DOE/ID RPT #: 1575

Spent Nuclear Fuel Working Group

Prepared by:
Aluminum-Clad SNF Sub Working Group

Purpose:

To evaluate the environmental, safety, and long-term programmatic risks associated with a management strategy for aluminum-clad SNF in extended (i.e., greater than 50 years) dry storage to help inform decision-making.

Aluminum-Clad Spent Nuclear Fuel:
Technical Considerations and Challenges for Extended (>50 Years) Dry Storage

Challenges to ASNF Dry Storage (2017)

Journal of Nuclear Materials

Technical basis for extended dry storage of aluminum-clad spent nuclear fuel

E. Eidelpes, J.J. Jarrell, T.E. Lister, G.P. Horne, E.H. Parker-Quaife, J.K. Conrad, C.D. Pilgrim, A.W. Abboud, P.L. Winston, R.E. Smith, A.L. d'Entremont, B.C. Randall, R.L. Sindelar

ARTICLE INFO

ABSTRACT

This paper presents the technical basis for extended dry storage of aluminum-clad spent nuclear fuel (ASNF). This basis addresses and closes five knowledge gaps defined by the U.S. Department of Energy (DOE) Spent Nuclear Fuel Working Group (SNFWG) in DOE-ID/RPT-1575 "Aluminum-clad Spent Nuclear Fuel: Technical Considerations and Challenges for Extended (>50 Years) Dry Storage." The SNFWG's primary concern pertaining to dry storage of ASNF is rooted in the presence of (oxy)hydroxide layers adherent to the aluminum cladding of the spent fuel that formed in service, indicating a risk of radiolytic molecular hydrogen (H_2) generation in a strong gamma radiation field—a process that can be associated with disintegration of (oxy)hydroxide and the molecular decomposition of chemisorbed and physisorbed water bonded to it. Potential consequences are overpressurization of sealed ASNF dry storage canisters and creation of a flammable atmosphere in sealed and vented canisters. Further, there is the concern that the generation of other chemical species due to chemical reactions, such as nitric acid (HNO_3), could lead to canister component corrosion. To address these issues, experimental work was conducted, including investigations of the performance of surrogate aluminum specimens and actual ASNF material featuring adherent (oxy)hydroxide layers during wet or dry storage. Further, a comprehensive characterization of the sample material was completed. Drying experiments were executed, providing better estimations of the amount of residual, chemisorbed, and physisorbed water after ASNF dry storage canister conditioning. These empirical activities allowed for the development of predictive numerical models to simulate the chemical species yield within sealed and vented dry storage canisters for 50 years. The findings of these activities lead to the main conclusion that extending current ASNF dry storage durations by 50 years is both safe and viable.

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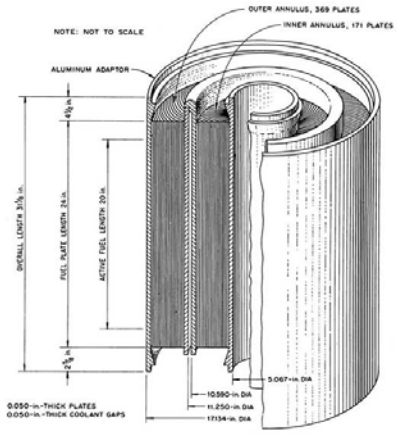
1. Introduction

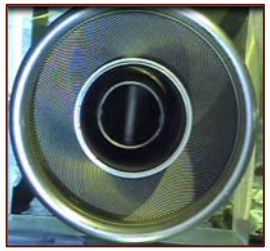
1.1. Technical knowledge gaps

ASNF Technical Bases (2023)

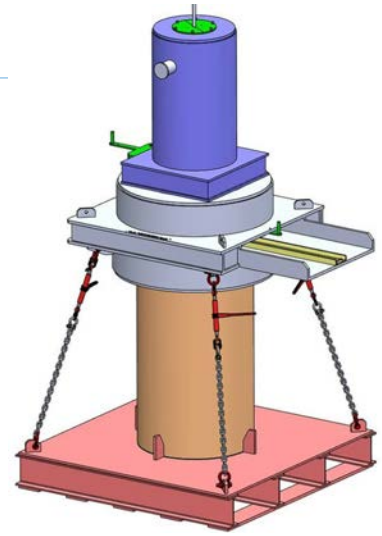
ASNF Dry Storage Pilot Concept

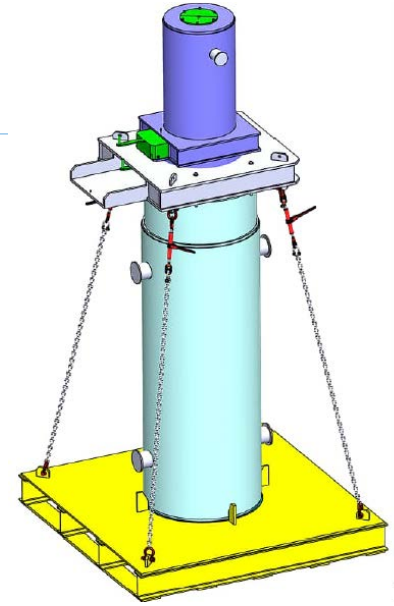
- Dry-to-dry transfer of HFIR inner core elements from GE2000 transport cask into two modified 18-in OD x 10-ft DOE Standard Canisters (DSC)
  - 3 HFIR inner fuel elements per DSC
  - Prototypic basket (no neutron absorber)
  - Instrumentation to monitor gas (H_2, O_2, N_2 to ~10 vppm in helium) and temperature internal conditions
  - Bolted-flange lid for SNF retrieval after pilot
- Two commercial drying processes: 1) vacuum drying; and 2) forced-helium dehydration (FHD). Collect water from drying.
- 1-5 years of monitoring: temperature and gas composition
- Place and store fuel in SRS L Basin after Pilot*





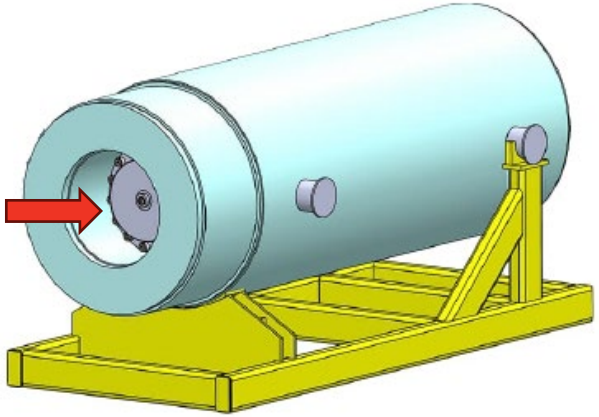
HFIR Inner Core [in L Basin]





Dry Transfer Cask on top of GE-2000 Cask (left) and on top of Shielded Modified-DSC (right) [Holtec concept]

Modified-DSC with bolted lid and instrumentation manifold (not shown)



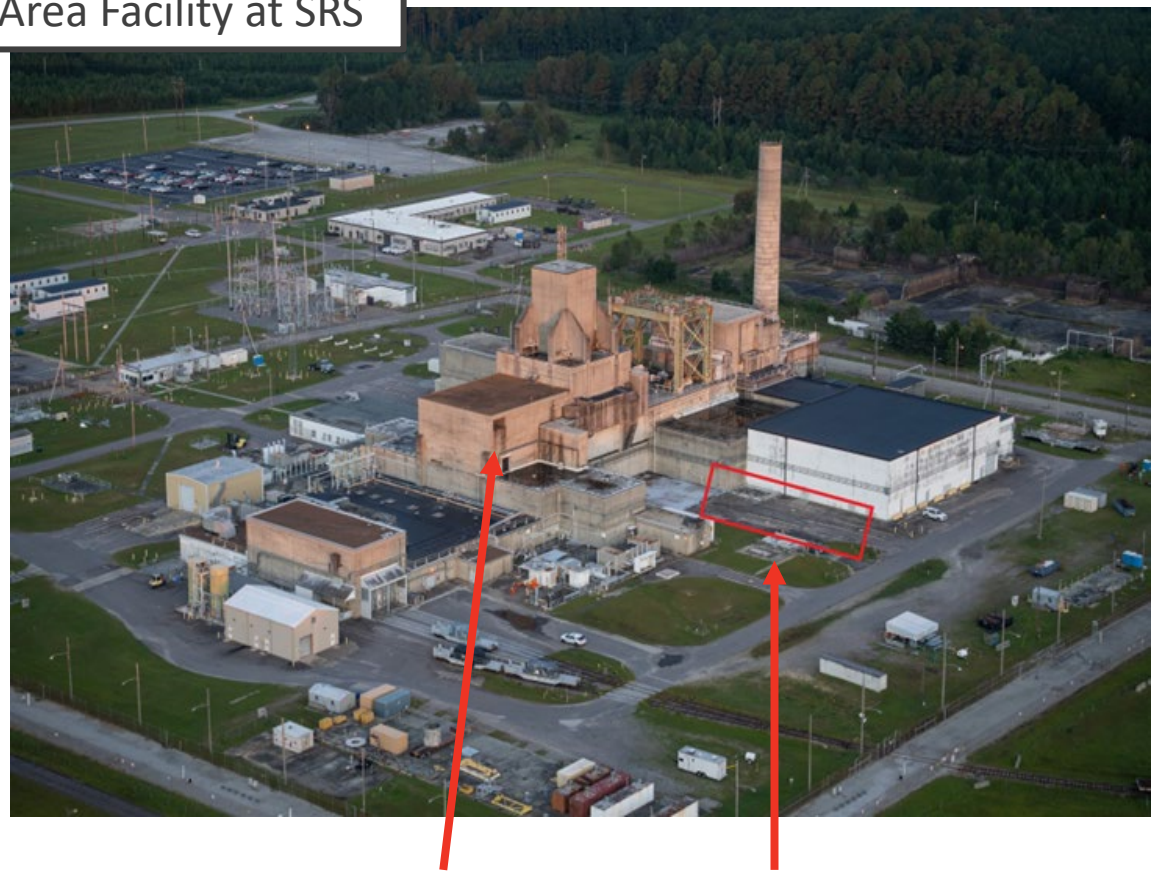
Shielded Overpack with Modified-DSC on Tilt Frame [Holtec concept]

*Consider development of "road-ready" dry storage system

Pilot Implementation and Test Plan

- SRS L Area Siting, Documented Safety Analysis Safety in progress
- Components design/build with commercial vendor (2026 – 2027)
- Task 1: Drying testing (pre-HFIR hot tests)
  - Dry runs of VD and FHD with dummy fuel
    - Demonstrate added free water removal
    - Verify equipment function
    - Train SRS operations personnel
- Task 1: Drying testing (HFIR hot tests)
  - VD and FHD with actual fuel (drying recipes TBD)
    - Collect water removed in drying
- Task 2: Gas collection and temperature monitoring (gas sampling schedule TBD)
  - Monitor (H_2, O_2, N_2 to ~10 vppm in helium) and temperature internal conditions
- Task 3: Pre-test, and post-drying and dry storage visual examination of HFIR cores

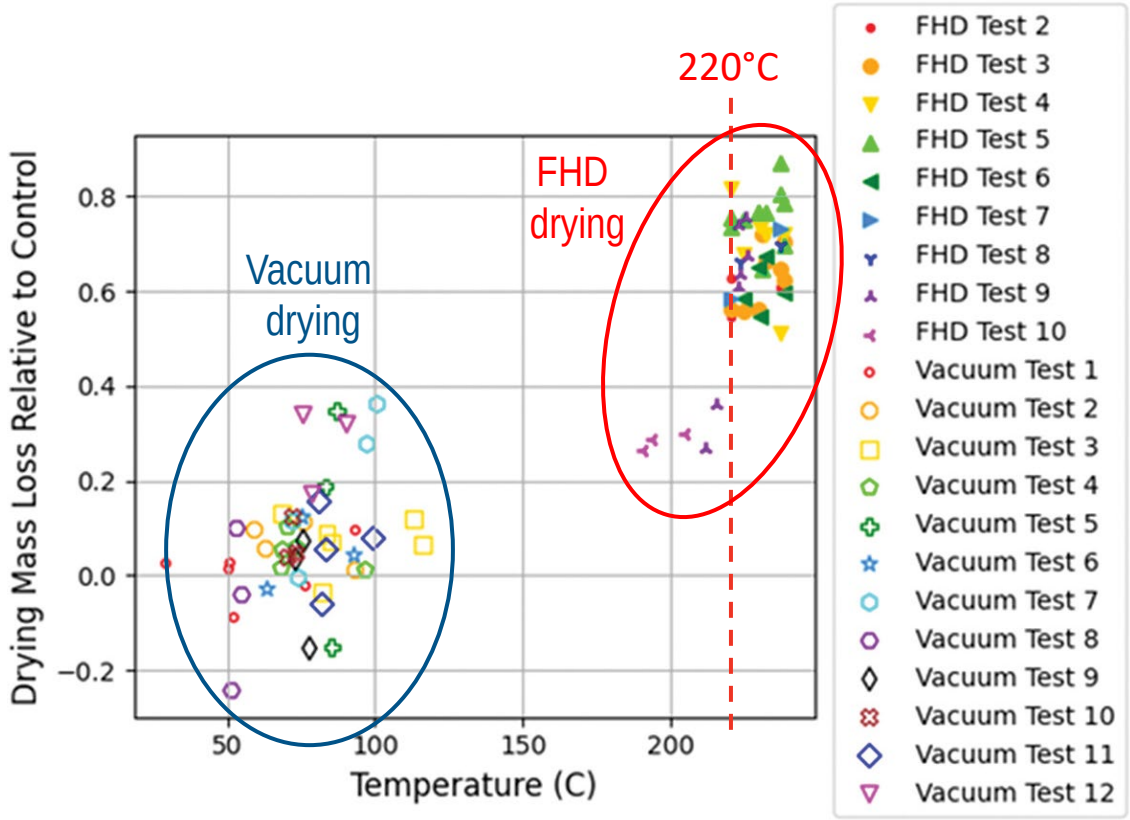
L Area Facility at SRS



L Basin Staging area planned for Pilot

Previous Drying Tests

- Lab-scale tests (TGA) showed dehydration of bayerite $Al(OH)_3$ and transformation to boehmite AlOOH starting at ~220°C
- Engineering-scale tests of vacuum drying and FHD on 1/3-height canister mockup
  - Both methods removed free water
  - FHD could bring mockup SNF to 220°C (vacuum drying did not)
  - Sharp increase in drying mass loss at 220°C threshold



- Estimated residual water in Pilot from 3 HFIR inner cores:
  - **~35 mol in undried (oxy)hydroxide (mixed hydrated oxides, 17 μm total). Reduced to ~20 mol if $Al(OH)_3$ dried to AlOOH**
  - Other H_2O <0.1 mol

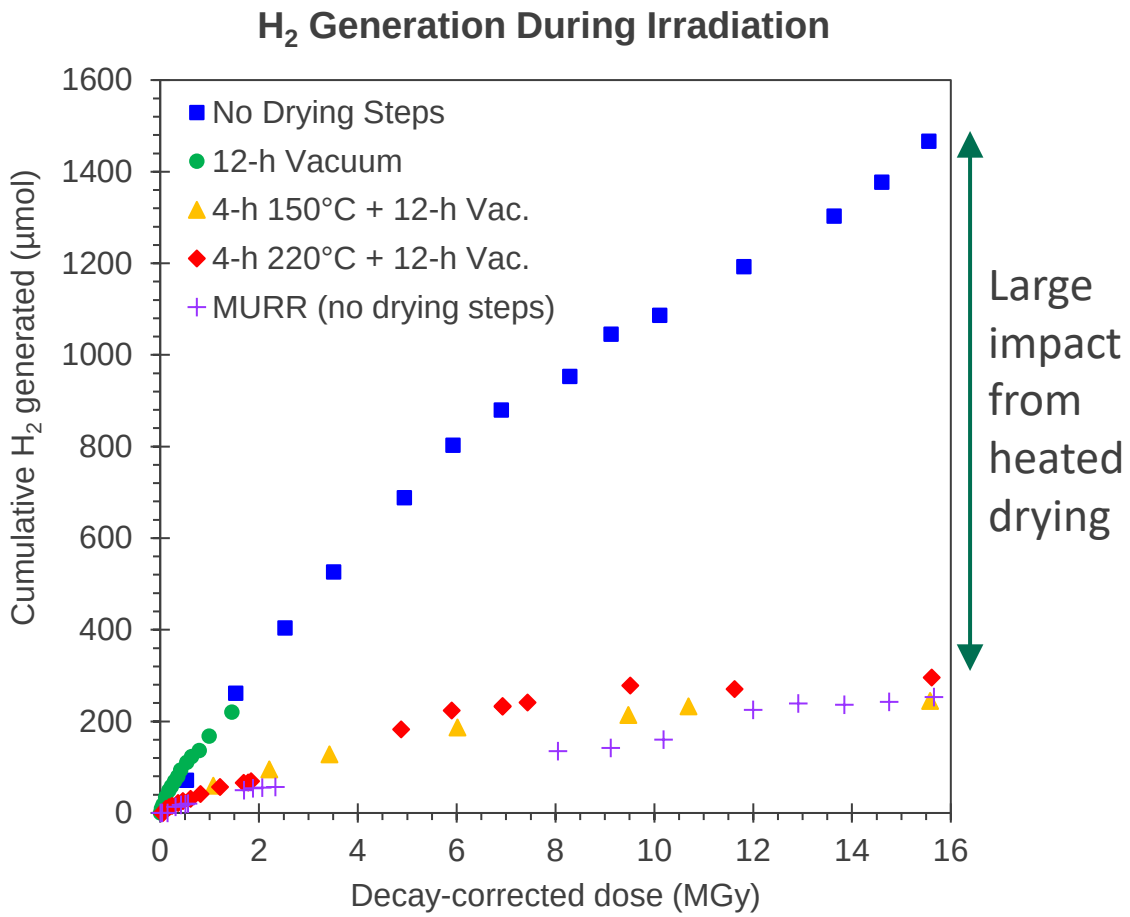
Lab-scale Precursor: mini-canister radiolysis testing

Sampling line through lid

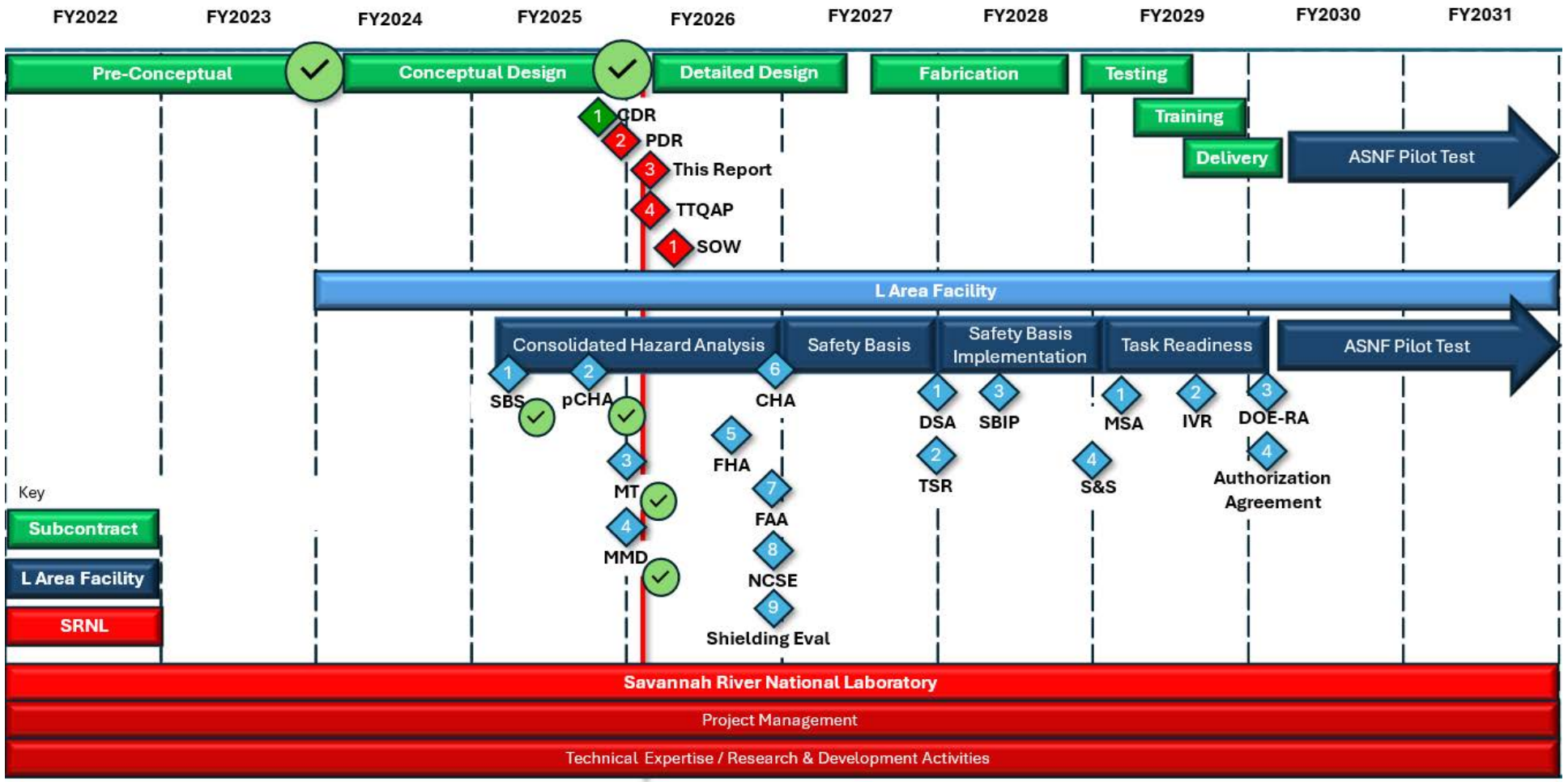
Small commercially available steel vacuum vessels

Aluminum surrogate assemblies with bayerite

MURR cropping from actual ASNF



Roadmap for ASNF Dry Storage Pilot



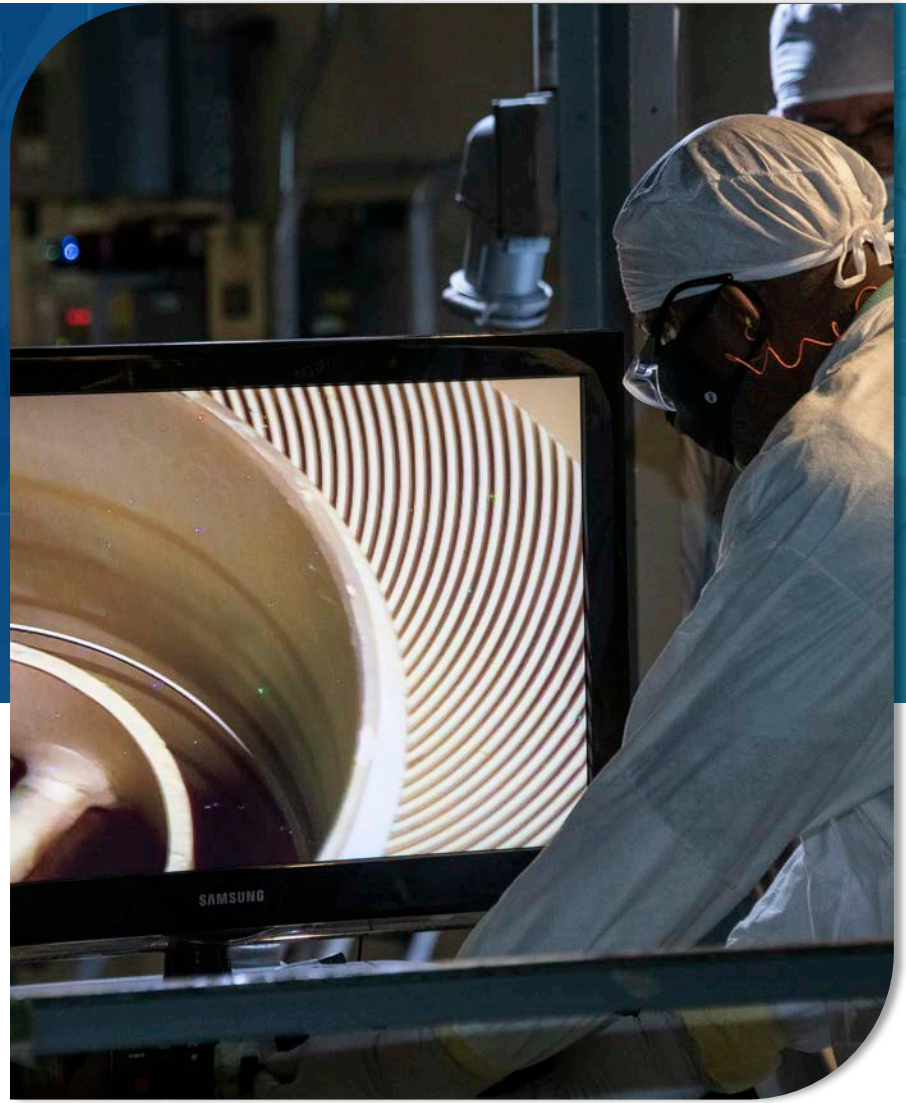
Conclusions

Dry storage is a potential path for SNF remaining in L Basin after H Canyon closure and/or for future SNF receipts to SRS

- Pilot will provide verification and validation of drying and dry storage performance for ASNF-in-DSC
- Pilot will establish SRS capability in
  - Dry-to-dry transfers (potentially applicable to future SNF receipts)
  - Use of vacuum drying and FHD processes, applicable to wet-stored SNF or SNF directly from transport casks
- Open to suggestions! – rare opportunity to collect data to understand drying and dry storage performance of SNF

Acknowledgments

- The technical bases for drying and dry storage of ASNF was developed by INL and SRNL from 2017 – 2024 under sponsorship of DOE-EM 3.21, Office of Technology Development. EM OTD is sponsoring the ASNF Pilot
- Contributors to the ASNF dry storage technical bases development:
  - University of South Carolina – Professor Travis Knight group for 1/3 Engineering Scale Tests
  - Holtec International – 1/3 Engineering Scale Tests
  - PNNL – Independent Technical Review of INL/SRNL technical bases work
- Contributors to the ASNF Pilot Project:
  - Savannah River Site Spent Fuel Project Operations and Engineering – DSA and Staging
  - Holtec International – Conceptual Design; Design/Build Special Engineered Components
  - ORNL HFIR – planning for HFIR core selection, and shipments to SRS
  - PNNL (Brady Hanson) – Independent Technical Review of Pilot testing plans



Questions?

Residual water estimates

- Estimated surface areas
  - HFIR inner core: ~19 m^2 each
  - Basket (stainless steel): ~9 m^2
  - Canister interior: ~3.8 m^2
  - Spacers currently undetermined
- Gas volume in canister: <0.323 m^3
- Drying expected to remove:
  - Free water (except <3 Torr vapor)
  - Some physisorbed
  - Some (oxy)hydroxide (FHD only)

- Estimated water
  - (Oxy)hydroxide on 3 HFIR cores
    - **For 17 µm of 50%-50% bayerite/boehmite: 35 mol H_2O**
    - **Fully dried to boehmite: 20 mol H_2O**
  - Physisorbed
    - Depends on material, humidity, microscopic surface area → # of monolayers of water
    - Canister and basket: 0.21 mmol/monolayer H_2O
    - 3 HFIR cores: 0.93 mmol/monolayer H_2O
  - <3 Torr water vapor: ≤53 mmol.

Aluminum-Clad SNF (ASNF) Pilot Test

Pilot Test Evolution

- Execute Dry-to-Dry Transfer of ASNF (HFIR Inner Cores)
- Perform Drying, Two Variants, and Collect Water:
  - Forced Helium Dehydration (FHD)
  - Vacuum Drying (VD)
- Monitor gas generation

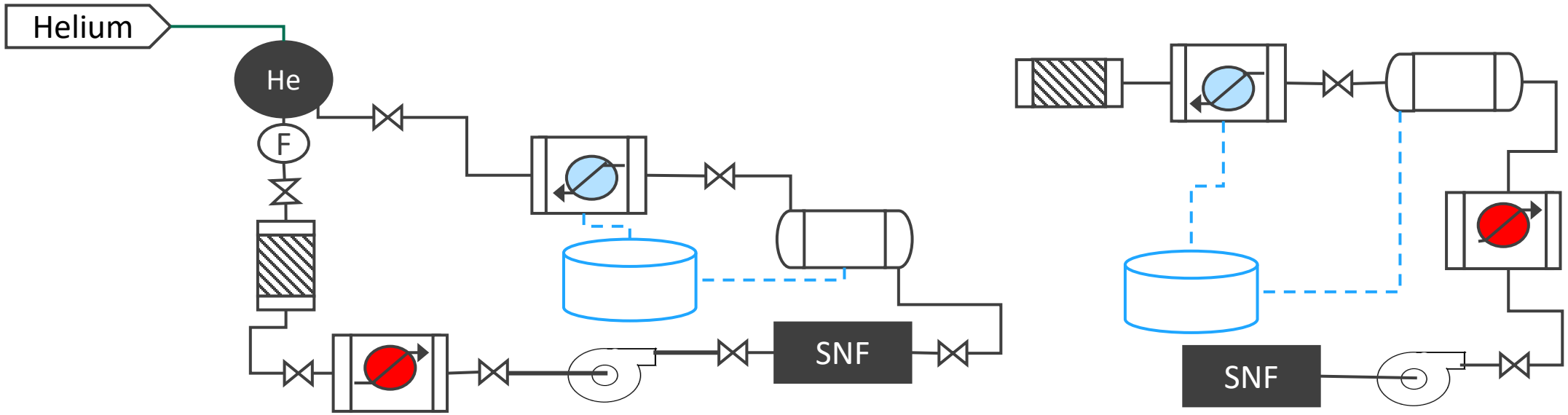


Figure 2: Conceptual Schematics - Forced Helium Drying System (Left) & Vacuum Drying System (Right)

IAEA updates on Spent Fuel Management

Christoph Gastl, Spent Fuel Management Specialist (NFCMS)

Department of Nuclear Energy, IAEA

ESCP 2025 27-30.10.2025

Content

The IAEA an Overview

- Role & Activities
- Governance & Advisory Boards
- Organization

Global Status and Trends in the Back End of the Nuclear Fuel Cycle

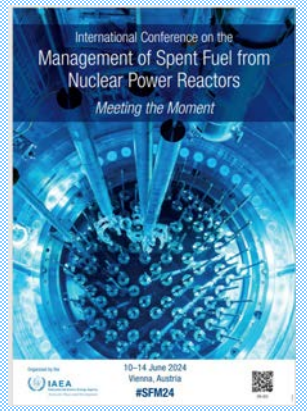
- Spent Fuel Management Strategies Worldwide
- Spent Fuel Current Global Inventory
- Spent Fuel Recycling through Reprocessing
- Challenges in Spent Fuel Management

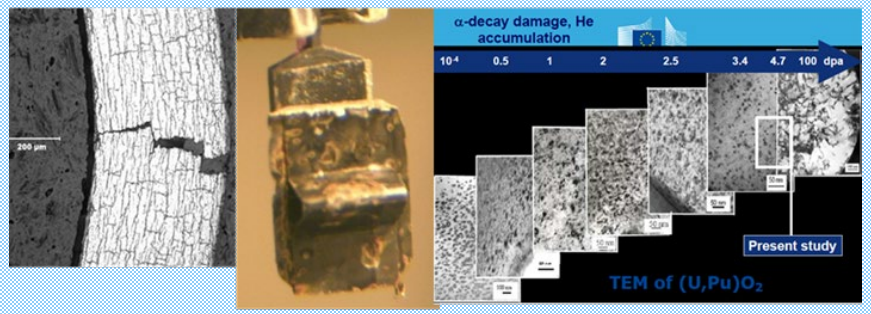
IAEA’s Activities in Support of the Backend of the Nuclear Fuel Cycles

- Publications
- Scientific/Technical Events
- Coordinated Research Projects
- e-Tools
- SFM Network

IAEA's Role and Activities

- Established in 1957, as an autonomous organization within the United Nations system, the IAEA actively supports its 180* Member States to (*inter alia*) **improve their capabilities related to Fuel Cycle technologies with the aim to reduce the waste burden** and to enhance the sustainability of nuclear power
- Through:
  - Organizing international **conferences** and **workshops**
  - Publishing **technical documents** and **reports**
  - Coordinating international **research activities** (CRPs)
  - Providing **review missions/advisory services**
  - Managing specific **databases** and **e-tools**





** as of 15 November 2024*

IAEA's Governance and Advisory Boards

- Biennial programmes (ex: 2024-2025) taking into consideration Member States' recommendations & requests expressed through the yearly adopted resolutions, during the General Conferences

  "Calls upon the Secretariat and Member States in a position to do so to investigate new reactor and fuel cycle technologies with improved utilization of natural resources, and proliferation resistance, including technologies for the recycling of spent fuel and its use in advanced reactors under appropriate controls and for the long-term disposition of remaining waste materials, taking into account economic, safety, and security factors"

  "Requests the Secretariat to continue and strengthen its efforts relating to the nuclear fuel, fuel cycle, spent fuel and radioactive waste management, as well as decommissioning, including through coordinated research projects (CRP), and to assist Member States to develop and implement adequate programmes, in accordance with relevant safety standards and security guidance."

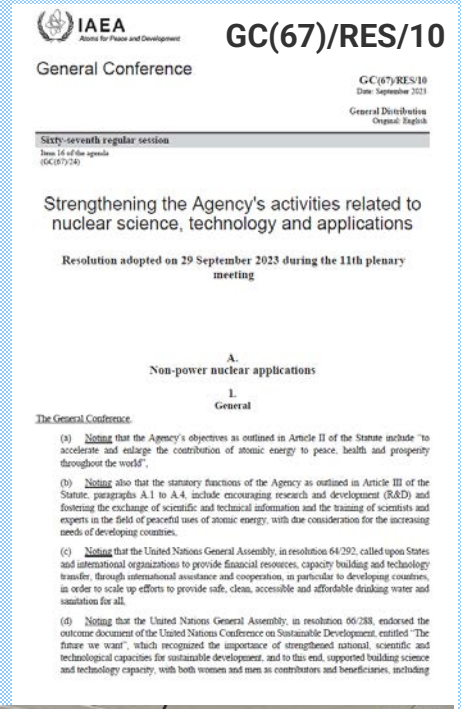
- **Standing Advisory Groups** (SAGs)

  Standing Advisory Group on Nuclear Energy (**SAGNE**): a group of international experts advising (yearly) the Director General on nuclear power, fuel cycle and nuclear science issues

- **Technical Working Groups** (TWGs)

  Groups of international experts advising (yearly) the DDG-NE on the orientation and implementation of NE programmatic activities (ex: **TWG on Nuclear Fuel Cycle Options and Spent Fuel Management**)

General Conference


IAEA
Atoms for Peace and Development

GC(67)/RES/10

General Conference

GC(67)/RES/10
Date: September 2023

General Distribution
Original: English

Sixty-seventh regular session
Item 16 of the agenda
(GC(67)/24)

Strengthening the Agency's activities related to nuclear science, technology and applications

Resolution adopted on 29 September 2023 during the 11th plenary meeting

A.
Non-power nuclear applications

1.
General

The General Conference,

(a) Noting that the Agency's objectives as outlined in Article II of the Statute include "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world",

(b) Noting also that the statutory functions of the Agency as outlined in Article III of the Statute, paragraphs A.1 to A.4, include encouraging research and development (R&D) and fostering the exchange of scientific and technical information and the training of scientists and experts in the field of peaceful uses of atomic energy, with due consideration for the increasing needs of developing countries,

(c) Noting that the United Nations General Assembly, in resolution 64/292, called upon States and international organizations to provide financial resources, capacity building and technology transfer, through international assistance and cooperation, in particular to developing countries, in order to scale up efforts to provide safe, clean, accessible and affordable drinking water and sanitation for all,

(d) Noting that the United Nations General Assembly, in resolution 66/288, endorsed the outcome document of the United Nations Conference on Sustainable Development, entitled "The future we want", which recognized the importance of strengthened national, scientific and technological capacities for sustainable development, and to this end, supported building science and technology capacity, with both women and men as contributors and beneficiaries, including

TWG on Nuclear Fuel Cycle Options and Spent Fuel Management 2025

IAEA's Organization

Department NE

Deputy Director General
Mr Mikhail Chudakov

Division NEFW

Director
Ms Olena Mykolaichuk

Section NFCMS

Section Head
Mr Clement Hill

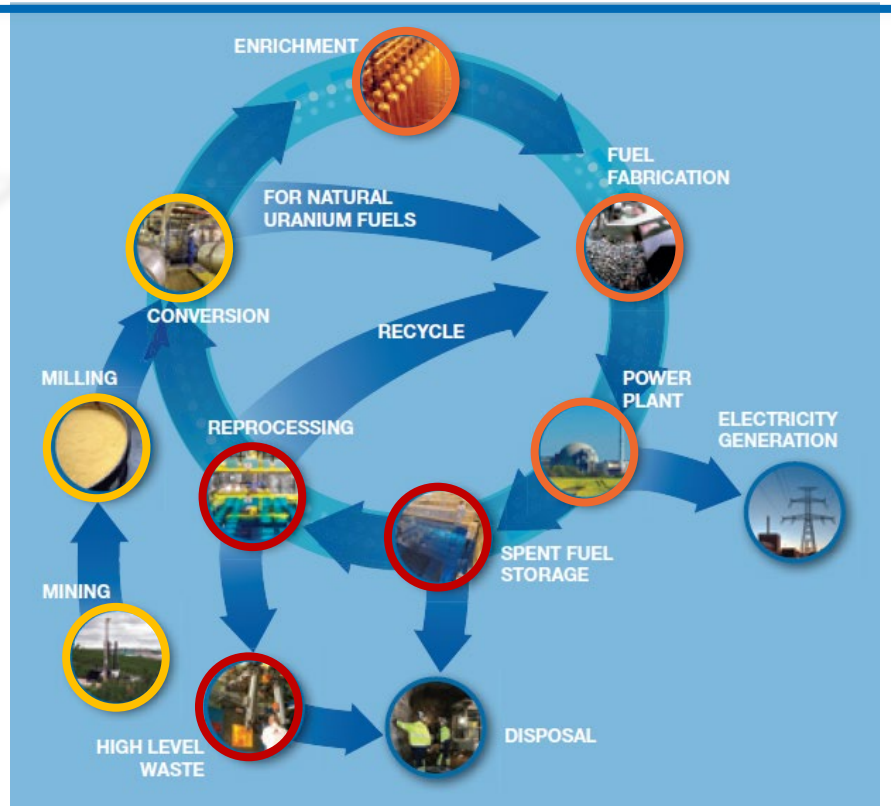
Teams in Nuclear Fuel Cycle and Materials

Nuclear Fuel Engineering — NFE

Team Lead
Anzhelika Khaperskaia

Uranium Production and Conversion — UPC

Team Lead
Adrienne Hanley



SFM

Spent Fuel Management

Advanced Fuel Cycles GEN-IV

Radioactive Materials Transport

Team Lead
Amparo Gonzalez Espartero

Spent Fuel Management Strategies Worldwide

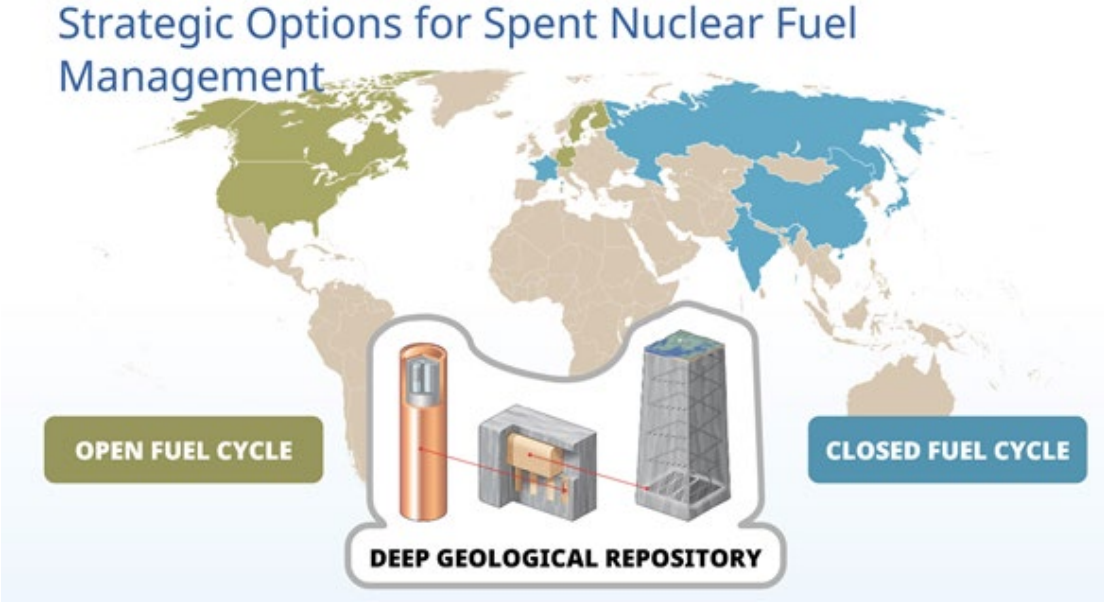
Today mainly countries with large nuclear power programmes recycle spent fuel: France, the Russian Federation, Japan, India and China.

Some countries have not yet made a final decision.

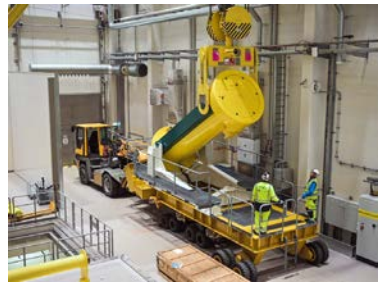
Most spent fuel is in interim storage.

Several other countries have opted for direct disposal: Finland, Sweden, Canada, Germany

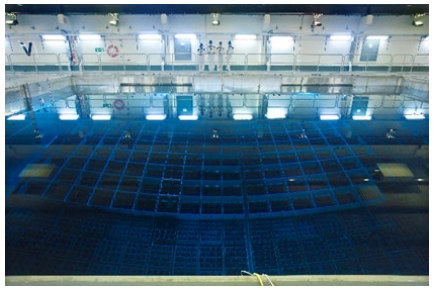
The Netherlands reprocesses SNF from Borssele NPP abroad and stores High Level Waste at HABOG facility



Posiva's encapsulation plant Trial Run on-going



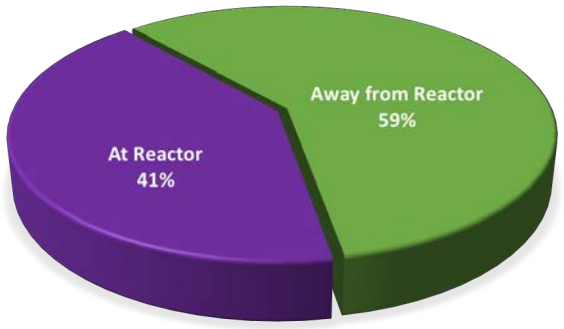
Spent Fuel Current Global Inventory



~ 416 NPPs* in 31 countries

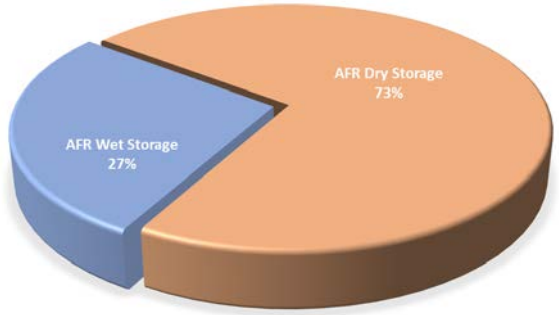
Global Inventory (discharged) by the end 2024, ~ **451 000 tHM**

At Reactor vs Away from Reactor Storage
Global Inventory (~322 300 tHM)



Away from Reactor

Away from Reactor Wet vs Dry Storage
(~189 100 tHM)



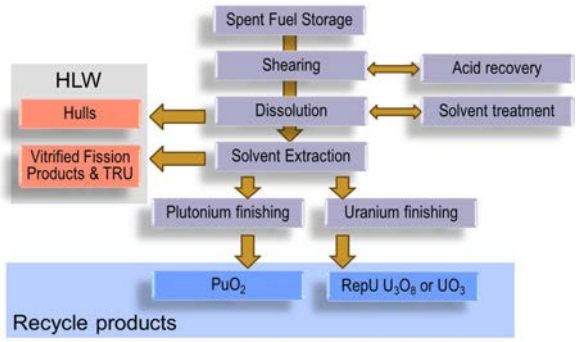
(*see PRIS database: https://pris.iaea.org/pris/)

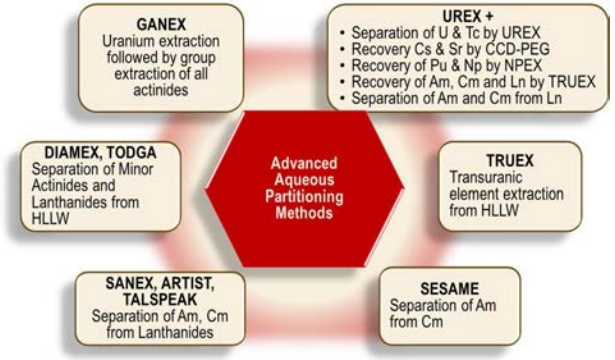
- Global Inventory of spent fuel at the end 2024: **~ 451 ktHM**, among which **~ 322 ktHM are in storage**
  - 41% at Reactor, 59% Away from Reactor (27**% Wet Storage** / **73% Dry Storage**)
- Selection of storage technology depends on many factors: Fuel, Economics, Stakeholders' preference

Spent Fuel Recycling through Reprocessing

- Reuse of **Reprocessed Pu as MOX in Light Water Reactors**
  - More than 40 years of experience worldwide (44 LWRs have used MOX fuel at industrial scale since 1986)
  - Loading cores partially with MOX (25-50%) and the remainder with UOX fuel
  - Recent reactor designs can accommodate 100% MOX cores
  - Demonstrating Multiple Recycling in LWRs contributes to transitioning to Fast Reactors
- Reuse of **Rep-Pu as MOX in Fast Reactors**: implemented in Russia in BN-800
- Reuse of **Rep-U** as **Enriched Reprocessed U (ERU)** in Thermal Reactors (PWRs, VVERs, RBMKs, AGRs, PHWRs)
  - More than 30 years of experience worldwide (TRs can accommodate 100% Rep-U cores)
- For decades, advanced (hydro/pyrometallurgaical) processes have been researched worldwide for **Minor Actinides recycling**

PUREX Process



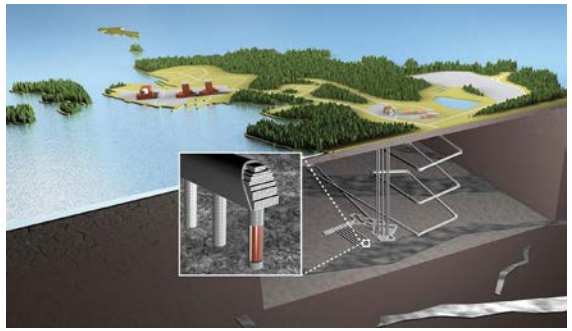


- Recycling Spent Fuel is a **mature technology** (Pu recycling in LWRs saves up to 25% of natural uranium resources)
- Reference options exist worldwide (PUREX)
- Reprocessing capacities exist in **France**, **Russia**, **India**, **Japan** and **China**

Current Challenges in Spent Fuel Management

- **Spent Nuclear Fuel Storage**
  - License renewal of storage systems
    - Confirming on-going SNF behaviour & integrity
    - Maintenance and inspection of SSCs
    - Ageing management (beyond design basis for most)
- **SNF transportability**
  - After long storage durations
  - From orphan sites
- **Spent Nuclear Fuel recycling**
  - Implementation of multi-recycling in LWRs at industrial scale
  - Demonstration and scaling-up of multi-recycling through Advanced Fuel Cycles for innovative reactors
- Accommodation of SNF from **Small and Modular Reactors**
  - New fuel types including higher enrichment
- Successful implementation of **Deep Geological Repositories**

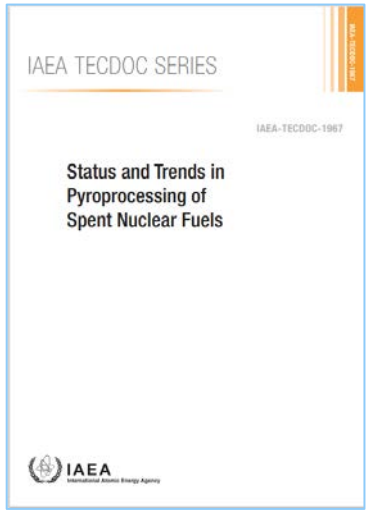
Yankee Rowe NPP (USA)

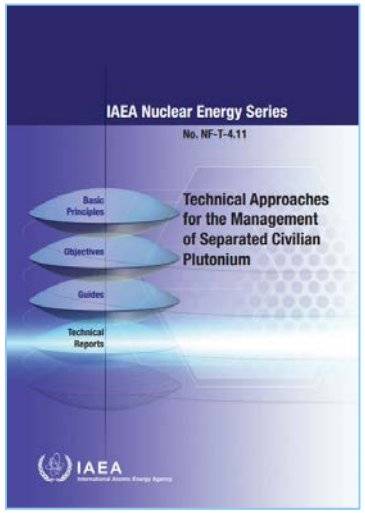


IAEA Publications on Spent Fuel Management

Translated in Arabic, Chinese & Russian

Translated in Arabic, Chinese and Russian



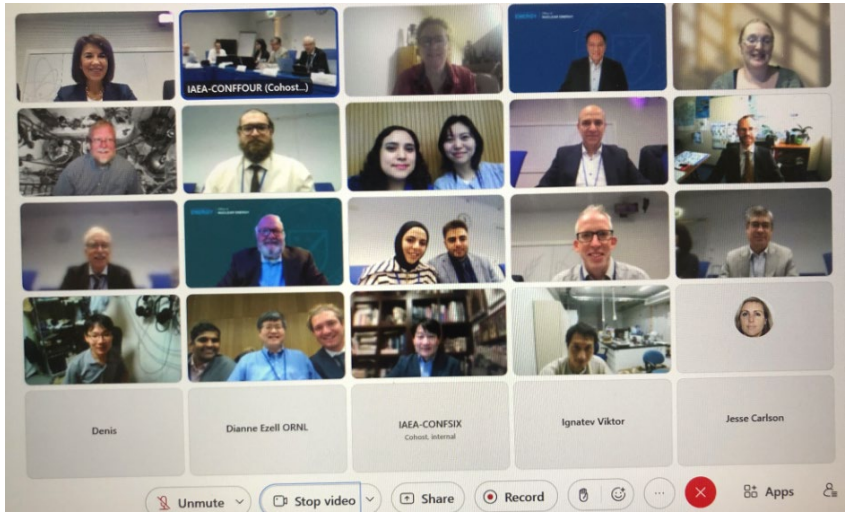


https://www.iaea.org/publications

- **TECDOCs**
- **NE Series publications**
- **Interactive Books** with pictograms, animations and downloadable pictures and charts

International Workshop on the Chemistry of Fuel Cycles for Molten Salt Reactor Technologies, 2-6 Oct. 2023, in cooperation with the OECD/NEA

- **Held in Vienna on 2-6 October 2023, co-organised by the NEA and the IAEA**
- **44 participants** (28 onsite + 16 online) including **4 developers of MSRs** (Copenhagen Atomics, Seaborg Technologies, Naarea and Terrapower), **R&D organisations, regulators and industry experts** from **13 countries** and the EU
- Several **technical sessions** including technical presentations, "thoughts from the floor" sessions and panel discussions with all the presenters
- **Breakout sessions** for the participants to engage in further discussion and work together to identify gaps and needs: experimental underpinning; irradiations; monitoring and operation; irradiated fuel treatment (material and waste streams)



Publication gathering Main Findings, Topical Discussions, Break Out Sessions, Conclusions and Recommendations is under preparation

Technical Meeting on Managing SNF (Pebbles and Compacts) from HTGRs, 7 -11 July 2025

Participation

47 participants (27 in person and 20 online) from **22** Member States

https://conferences.iaea.org/event/414/

Agenda included sessions on

- Information from IAEA activities on HTG reactors, fuels, disposal of i-graphite, and CRP SMR-COGS
- Management of irradiated graphite from HTGR spent fuels
- HALEU fuels for SMRs
- Recycling HTGR spent fuels
- Session on country experiences on managing spent fuel from HTGR (e.g, China, Germany, Japan, Kazahastan, Russia, UK, USA, EC on-going Projects)
- Session on newcomer's perspectives on HTGR spent fuel management (e.g., Egypt, Jordan, Pakistan, Poland)
- 3 Break-out Sessions to work on Scenarios Open Cycle Pebbles, Open Cycle Compacts, Closed Cycle (Pebbles and Compacts)

Recent IAEA Activities

- Technical Meeting on Operating Experience and Lessons-learned on Managing Non-standard/Exotic Legacy Power Reactor Fuels, Vienna, 18-21 February 2025

- Third Research Coordination Meeting CRP SFERA, Vienna, 24-28 March 2025

- Joint Workshop CRPs PASSED and SFERA, Seoul, 23-27 June 2025

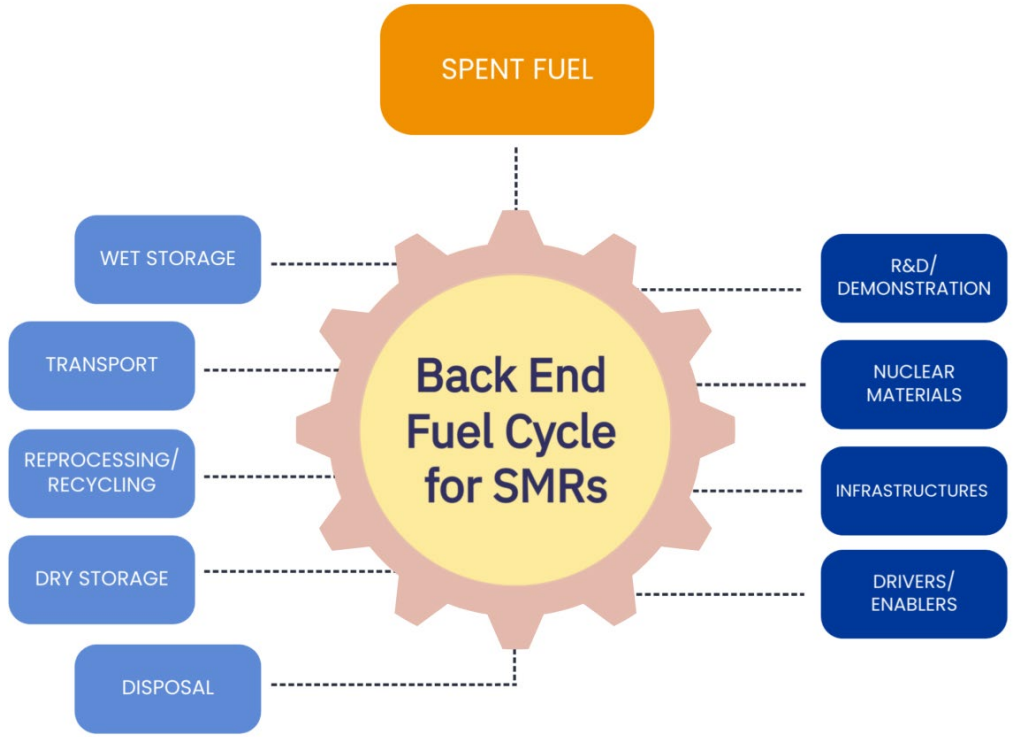
CRP on *Challenges, Gaps and Opportunities for Managing Spent Fuel from SMRs*

Understanding the implications of the management of new spent fuels is paramount to make informed decisions

MAIN OBJECTIVES:

- To identify viable nuclear fuel cycle options for the different SMR technologies
- To identify common technologies/similarities for various reactor types and/or significant differences
- To prepare a list of generic key parameters for countries to perform their analysis incorporating their specific context

SMR-COGS, CRP T13021



Coordinated Research Project on *Challenges, Gaps and Opportunities for Managing Spent Fuel from SMRs*

MAIN OUTPUTS

- Development of ***specific roadmaps*** for **managing spent fuel from the different SMR technologies** (e.g. LWR, HTGR, LMFR, MSR), identifying what can be derived, optimized or adapted from existing practices, or what needs to be fully developed
- To compare various SMR systems, in terms of efforts required to develop and implement an SFM strategy
  - Nuclear fuel cycle facilities
  - Technology readiness level
  - Nuclear materials involved
  - Infrastructures (e.g., human resources, financing)
  - R&D / Demonstrations
  - Enablers/Synergies

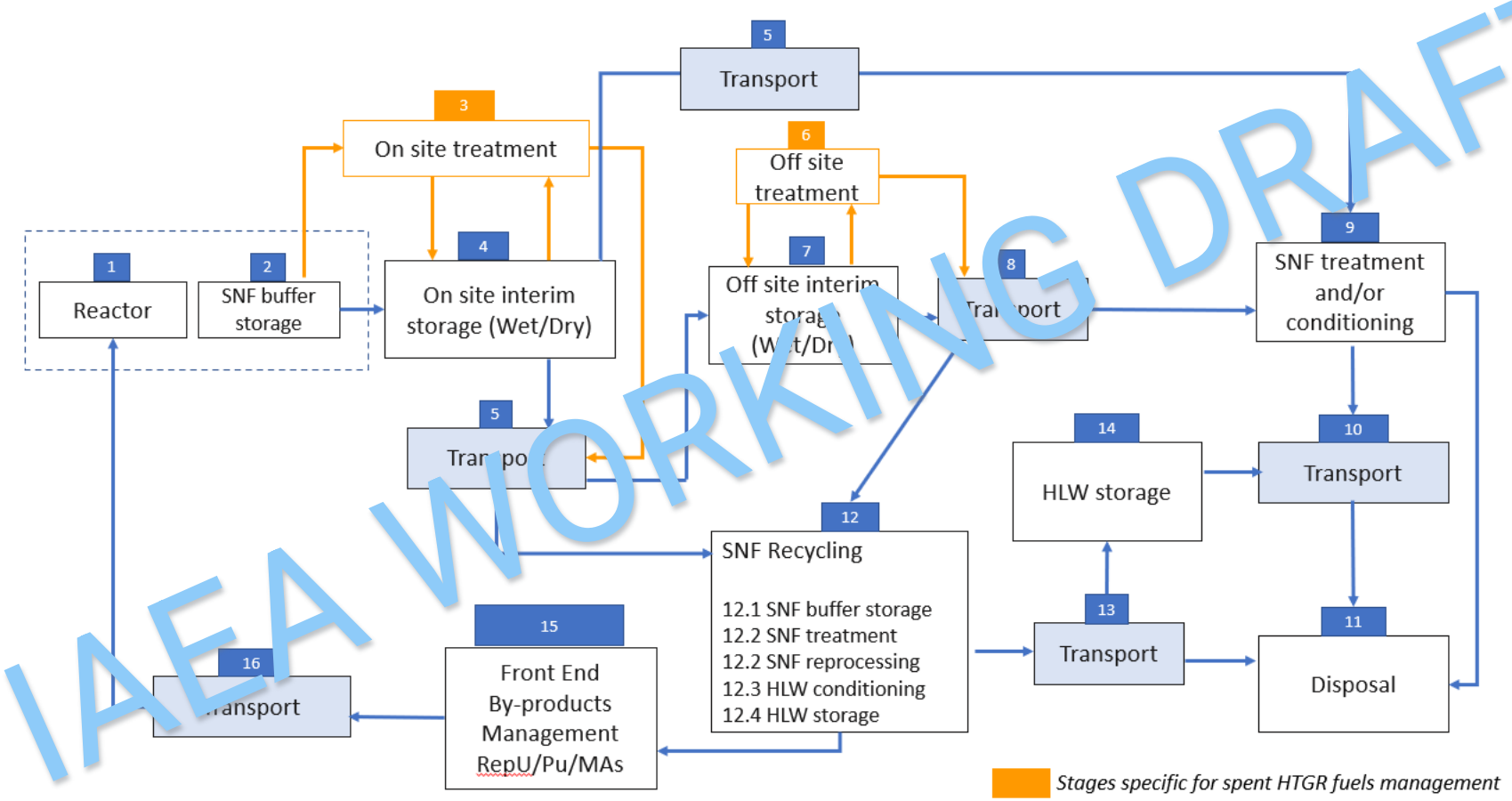
STATUS of the Coordinated Research Project SMR-COGS

- 14 Research Contracts from ARG, ARM, CPR, CZR, EGY, INS, LIT, MEX, POL, ROM(2), UKR(3)
- 18 Research Agreements from CAN(2), CPR, DEN, EGY, JOR, NOR, SIN, SPA, SWE, TUR, UK(2), USA(5)

from Industry, Operators, Researchers, Regulators, etc.

First Research Coordination Meeting held on 11 to 15 November 2024 in Vienna

WORKING DRAFT OF GENERIC STEPS FOR DIFFERENT FUEL CYCLE SCENARIOS



IAEA WORKING DRAFT

Timeline: Short, Medium, Long Term

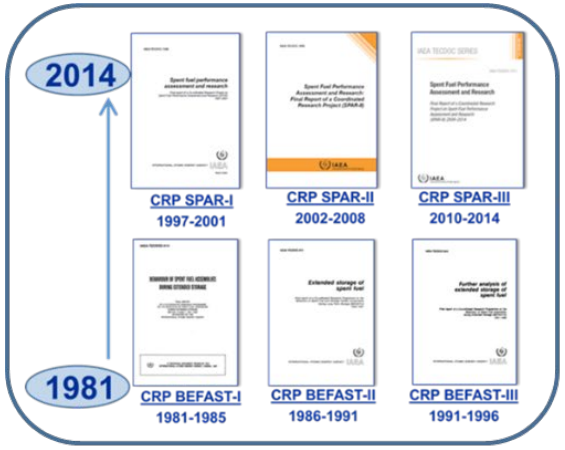
- High Level elements for the scenario
  - NFC facilities
  - Nuclear materials involved
  - RW generated
  - Gaps/Challenges/Opportunities/ R&D/ Demonstrators
  - Technology readiness level
  - Infrastructures needed
  - Potential enablers
  - …
- Time scale for development and implementation
- Needs for development and implementation
- What is needed and when
- Decision points

CRPs on Spent Fuel Behaviour
(+40y Operational Experiences and Research Worldwide)

Main Objective

- To sustain and improve the IAEA's Member States **technical knowledge base on the long-term behaviour of power reactor spent fuel** through sharing and disseminating information, reporting topical research carried-out in participating Member States, and by documenting ongoing spent fuel performance

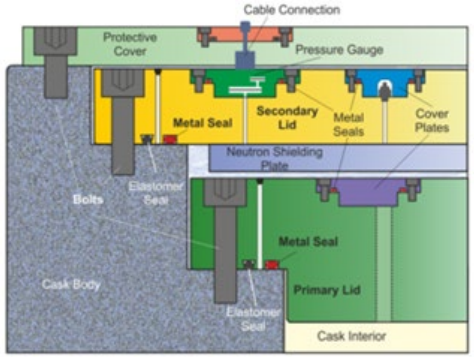
- **Series of Coordinated Research Projects** BEFAST and SPAR (1981-2020)
  - Covering all power reactor fuels: MAGNOX, RBMK, WWER, AGR, BWR, PWR, HWR, PHWR
  - Spent Fuel performance in Wet and Dry storage

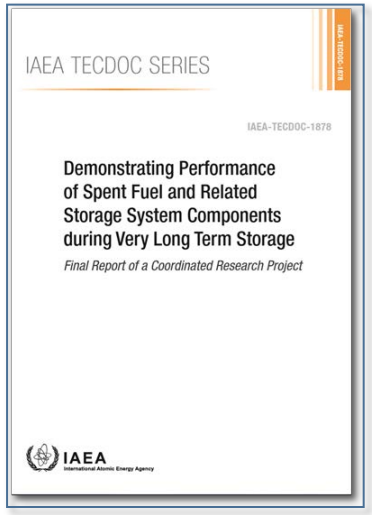


CRPs on SNF Dry Storage Systems Behaviour

- **Demonstrating Performance of Spent Fuel and Related Storage System Components During Very Long Term Storage"** (DEMO) CRP T13014 (**Closed**, 2012-2016)

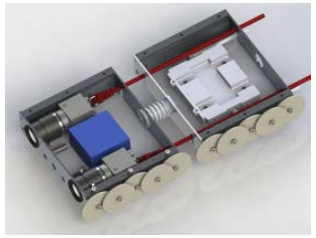
  Linked to Extended Storage Collaboration Programme (ESCP) International Sub-committee of EPRI





- **Ageing Management Programmes for Dry Storage Systems"** (AMP) CRP T21028 (**Closed**, 2016-2020) *(Final Report in Drafting)*

  CRP Overall Objective: To develop the technical basis and methodology to enable guidance to be provided to Member States on how to generate an ageing management programme for spent fuel dry storage systems



Overview about development of inspection technologies

CRPs on SNF Storage Systems Behaviour

"Performance Assessment of Storage Systems for Extended Durations" **(PASSED)**
T13019 (2022-2026)

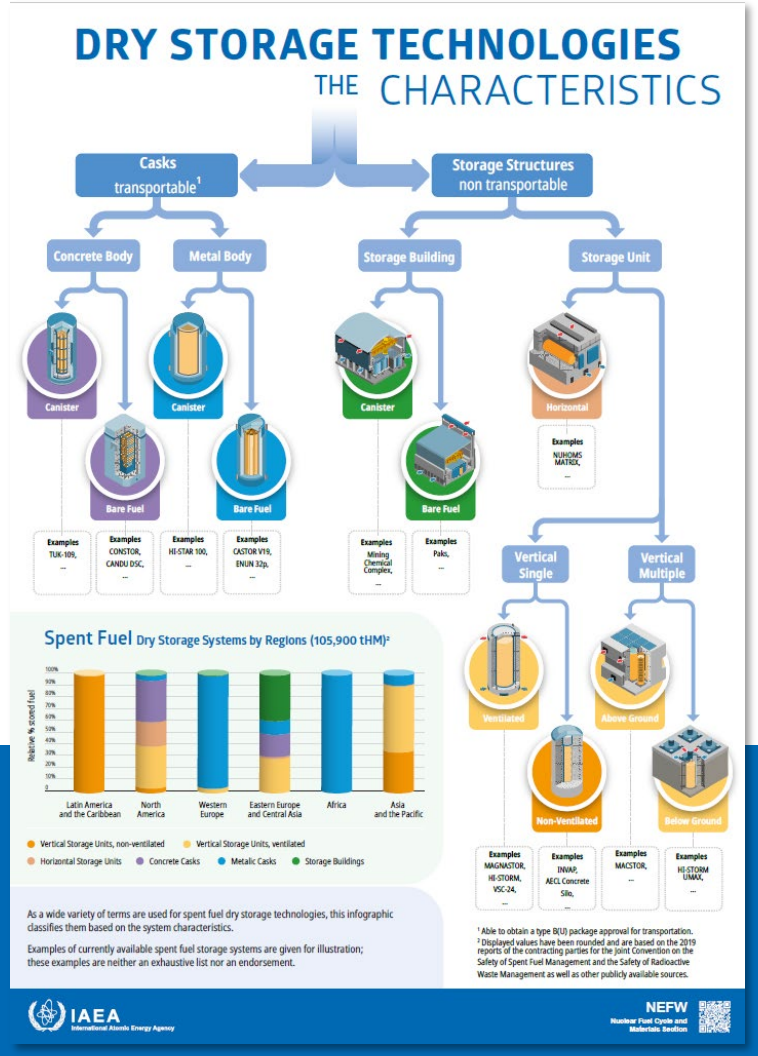
Main Objective:

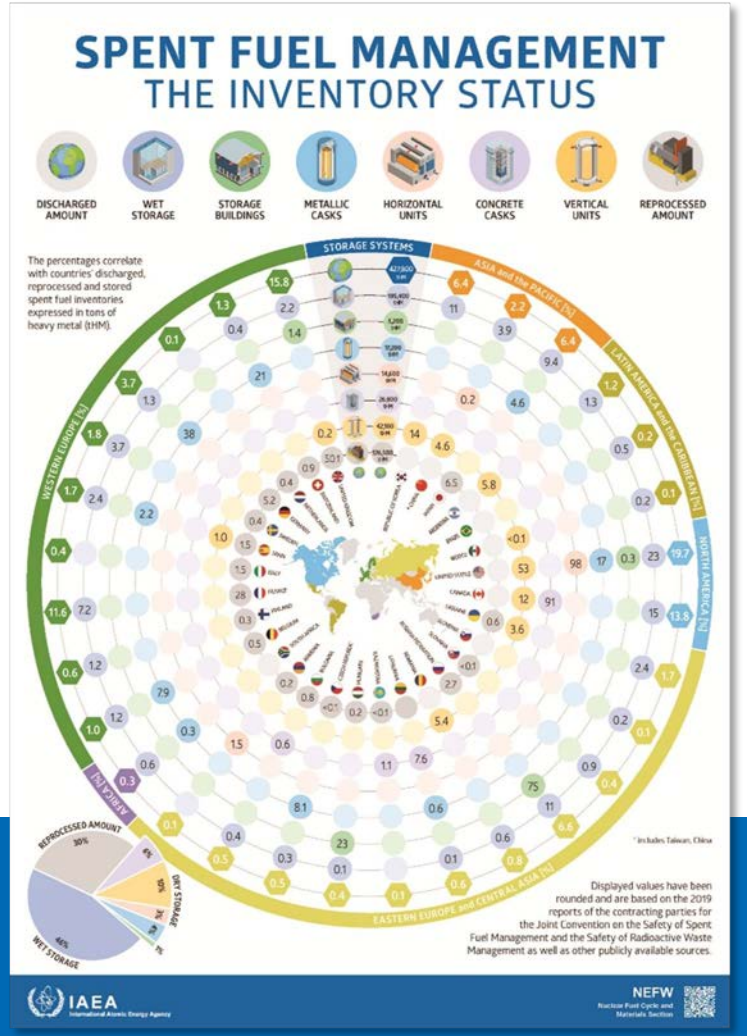
- To sustain and improve IAEA Member States' technical knowledge base on the **long term behaviour of spent fuel storage systems, inspections possibilities and monitoring technologies**, through the sharing and disseminating of technical information, the reporting on topical researches carried-out and the documentation of on-going storage systems' performance

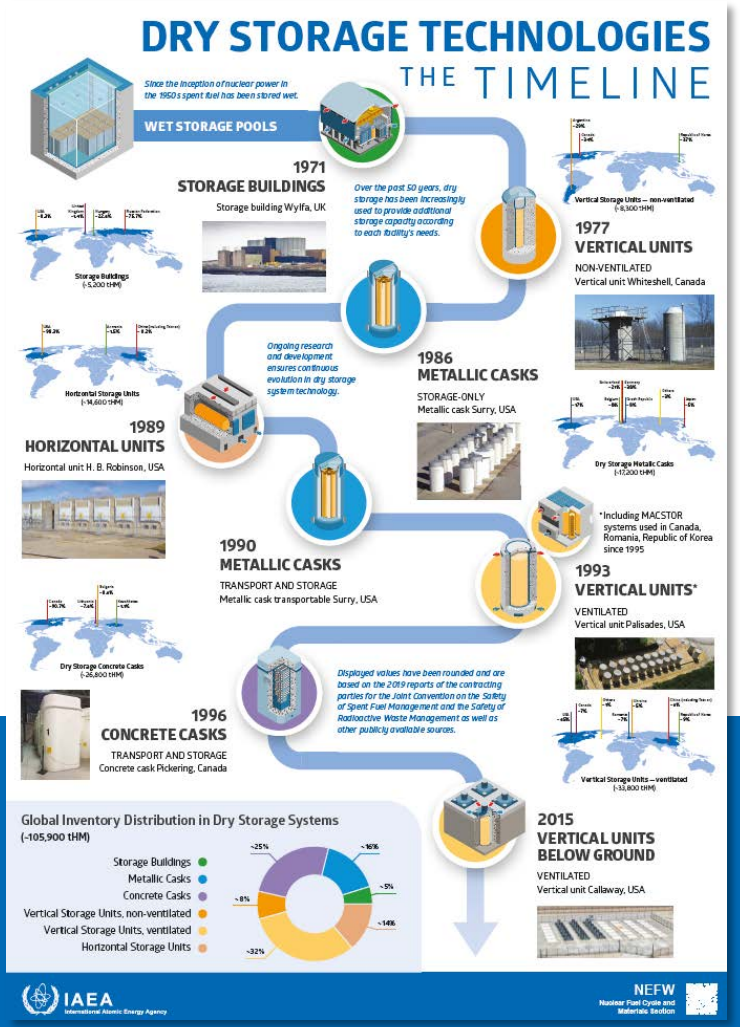
Covers wet & dry spent fuel storage systems

- operational experiences storage system inspections
- new/novel techniques for monitoring
- predictions of spent fuel storage system behaviour over long periods
- documenting the technical basis for spent fuel storage system performance assessment
- predictions of spent fuel storage system behaviour

IAEA Infographics on Spent Fuel Management

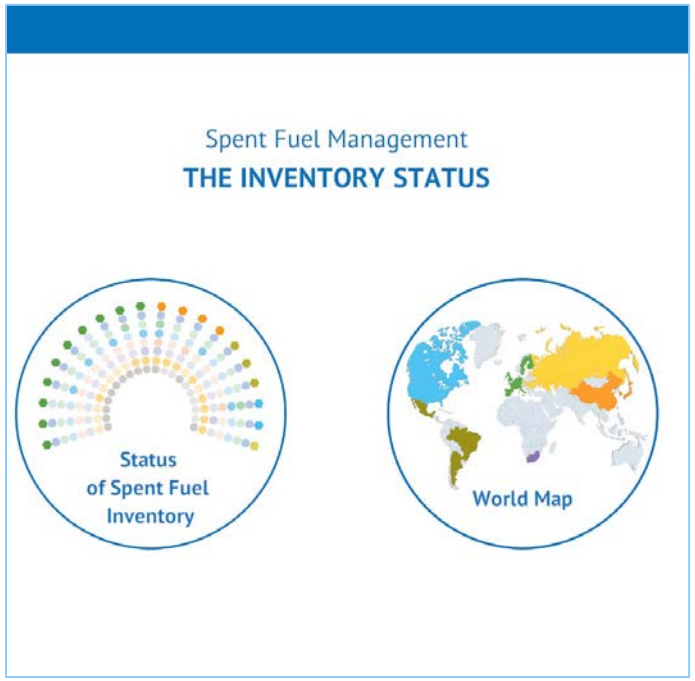


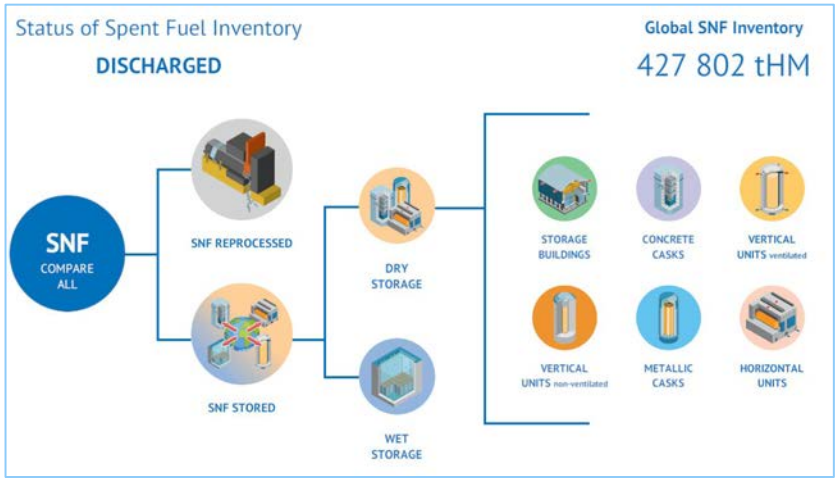


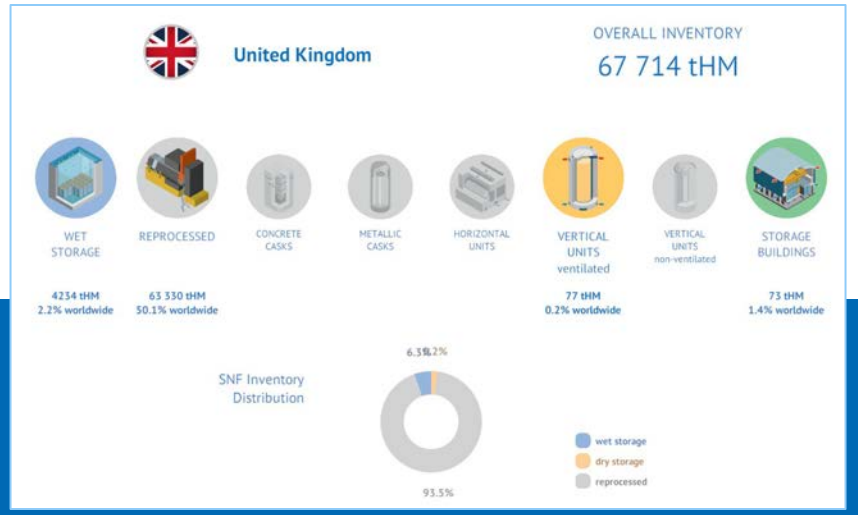


https://www.iaea.org/topics/spent-fuel-management

IAEA Interactive Infographics on Spent Fuel Management



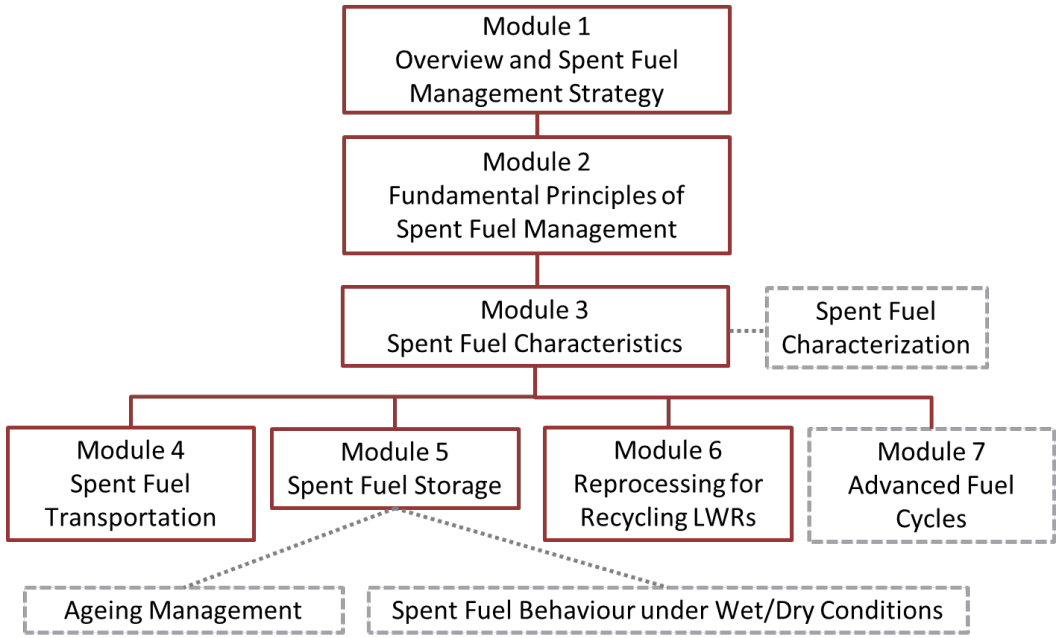




Currently developed on the 2019 data
2024 inventories will be added soon.

IAEA e-Learnings on Spent Fuel Management

6 Modules (11 Lectures) already available in



https://www.iaea.org/services/education-and-training/online-learning

Up-coming Activities

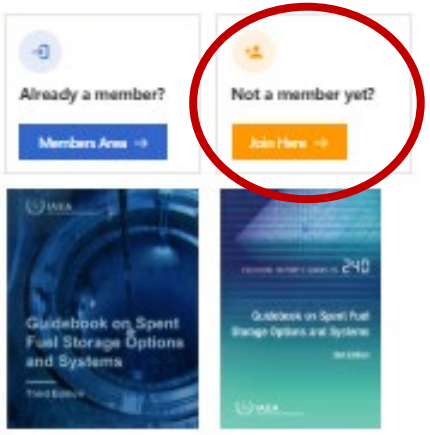
- International Workshop IAEA-NEA-EC/JRC on Molten Salt Reactor Fuel Cycle Taxonomy and Related Terminology, 3 – 7 November 2025, Vienna
- CRP PASSED Consultancy Meeting (virtual), 18 – 19 November 2025, to discuss progress on research activities
- CRP COGS Consultancy Meeting (virtual), 24 – 28 November 2025, to discuss progress on research activities
- RCM-3 PASSED, 27. April – 01. May 2026, Vienna
- RCM-1 Spent Fuel Characterization 2nd phase, 7. – 11. September 2026, Vienna
- International Workshop on the Challenges in Managing Spent Evolutionary Advanced Technology Fuels, 02. – 06. November 2026, Vienna
- CRP SFERA 2 Consultancy Meeting, 28 – 30. November 2026, to finalize the design of the follow up

International Conferences in 2026

- International Conference on **Fast Reactors and Related Fuel Cycles** (FR26) in Beijing (China) from **18 to 21 May 2026** (In collaboration with IAEA/NPTDS)

- International Conference on **Fuel Supply Chain for Sustainable Nuclear Power Development** in Vienna from **13 to 15 October 2026** (NFCMS)
  - Topic 1. Industry Prospects and Challenges Facing Raising Fuel Supply Demand
  - Topic 2. Supply and demand of raw materials for nuclear fuel supply
  - Topic 3. Advanced nuclear fuels for innovative reactor technologies
  - Topic 4. Industrial and Innovative technologies for recycling nuclear materials

Spent Fuel Management Network



Welcome to the IAEA International Network on Spent Fuel Management - SFM Net

The spent fuel management (SFM) network is a forum for the sharing of practical experience and international developments on spent fuel management.

Its main objectives are to facilitate the efficient exchange of information, communication and cooperation amongst professionals working in the back end of the fuel cycle – from its removal from a reactor core to its final disposition (i.e. SNF wet and dry storage, transportation, handling and retrieval, reprocessing and recycling, economics of the back-end of nuclear fuel cycle, damaged SNF management, stakeholder involvement, communication issues, etc.).

The establishment of the SFM Net is aimed at fostering safe, sustainable and efficient spent nuclear fuel management practices across all IAEA Member States.

For further information or questions please contact SFM.Contact-Point@iaea.org

Featured Publications

SEE ALL

New infographics now available!

Events 2025

- Technical Meeting on Operating Experience and Lessons Learned on Managing Non-Standard Legacy Spent Fuels from Power and Research Reactors (18 – 21 February 2025) *EVT1004638*
- Third Coordination Research Meeting on Spent Fuel Research and Assessment (24 – 28 March 2025) *EVT1004557*
- 43rd Meeting of the Technical Working Group on Nuclear Fuel Cycle Options and Spent Fuel Management (31 – 04 April 2025) *EVT1003074*
- Technical Meeting on the Behaviour of Spent Fuel and Cladding During Storage and the Performance of Spent Fuel Storage Systems (25 – 27 June 2025) *EVT1004873*
- Technical Meeting on the Management of Spent Fuel Pebble Beds and Prismatic Fuel from High Temperature Gas Cooled Reactors (7 – 11 July 2025) *EVT1004559*
- Joint IAEA-NEA Workshop on the Taxonomy and Related Terminology of Fuel Cycles for Molten Salt Reactors (03 November – 07 November 2025) *EVT1004560*
- Workshop on the Challenges in Managing Spent Evolutionary Advanced Technology Fuels (10 – 14 November 2025) *EVT1003196*
- Technical Meeting on the Operation and Maintenance of Storage and Transportation Casks (09 – 12 December 2025) *EVT1004556*

Latest News

SFM.Contact-Point@iaea.org

Thank You

C.Gastl@iaea.org

NEA
NUCLEAR ENERGY AGENCY

NEA Division of Radioactive Waste Management and Decommissioning: Activities on SMRs and Generation IV Systems

Rebecca TADESSE

Head of Division, NEA-RWMD

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NEA
NUCLEAR ENERGY AGENCY

NEA Project on Waste Integration for Small and Advanced Reactor Designs (WISARD)

NEA RWMD-SCI

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NEA WISARD Joint Project – Objectives

Waste Integration for Small and Advanced Reactor Designs

Innovative solutions for innovative systems

WISARD project objectives:

- *Understand* effects of fuel cycle options & new technology choices
- *Include* waste management perspective in design decisions
- *Improve* design optimisation & robustness by creating opportunities to address future issues early.

WISARD will investigate:

1. Is waste from these innovative systems compatible with current waste management solutions?
2. If not, what are the technology needs for a sustainable future back-end strategy?
3. How will these innovative systems require innovative solutions?

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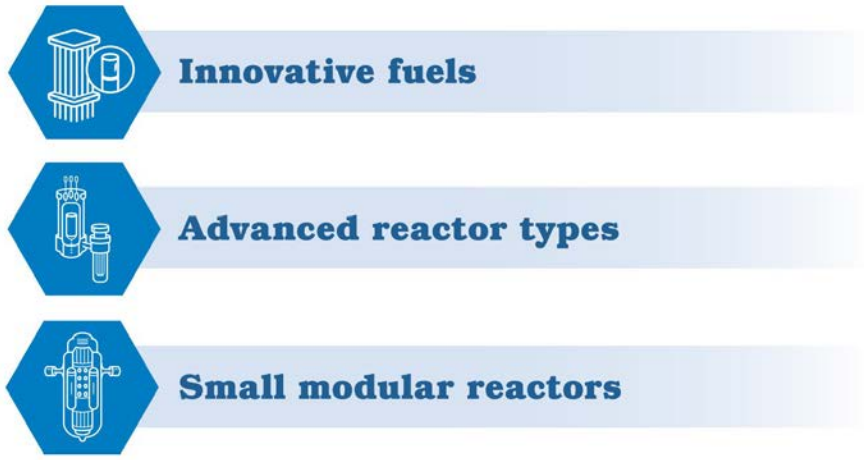
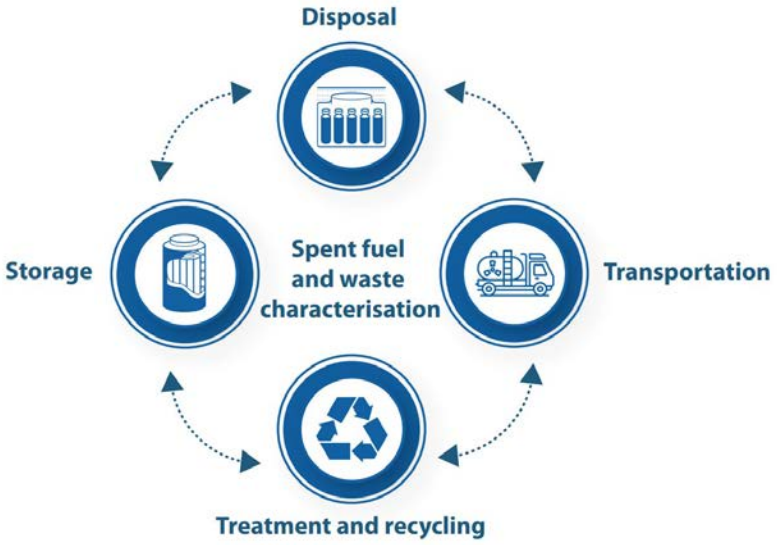
NEA WISARD Joint Project – Focus Topics

Waste Integration for Small and Advanced Reactor Designs





Front-end & design-phase decisions will have an impact on back-end waste management strategies:

- Increased fuel enrichment (e.g. HALEU)
- Irradiated graphite-matrix TRISO fuel management
- Activated coolant disposal (e.g. liquid metal, salt)
- Liquid fuel waste management strategies
- Location limitations (e.g. waste transport for remote SMRs)

WISARD Task focus areas:

1. Irradiated fuel & coolant characterisation
2. Treatment & recycling
3. Storage
4. Transportation
5. Disposal (DGR & borehole)
6. Cross-cutting integration and interdependencies

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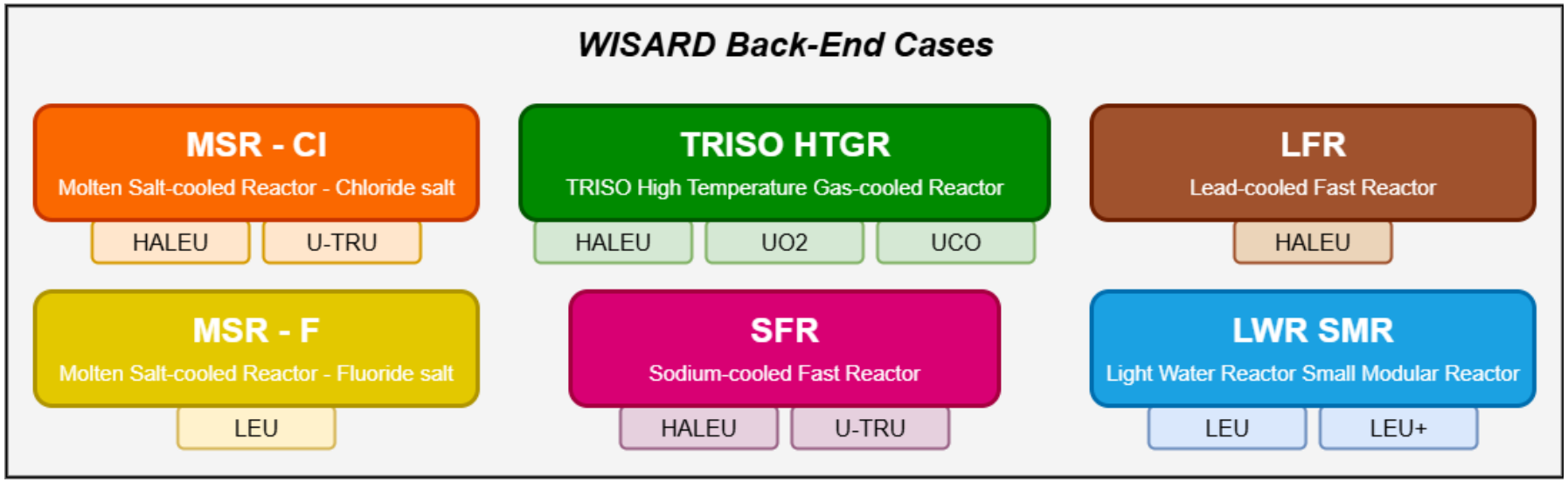
NEA WISARD Joint Project – SMR/AR Systems

Waste Integration for Small and Advanced Reactor Designs

Engaging with developers to provide the most accurate back-end information

- WISARD focuses on selected systems and fuels of interest to project participants
- SMR/AR designers and vendors participating in WISARD will provide the most up-to-date information, allowing detailed characterisation of used fuel and waste materials
- WISARD is a closed project and all participants retain ownership of IP shared within the activities
- Used fuel and waste characterisation data will be collected into Back-End Cases (BECs) and assessed for compatibility with different waste management strategies under the Task focus topics



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NEA WISARD Joint Project – Participants

Waste Integration for Small and Advanced Reactor Designs

Signatories

Canadian Nuclear Laboratories

SALTFOSS

VTT

naarea

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Nuclear Waste Services

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KINGS

Environment Agency

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NEA WISARD Joint Project – Tasks

Waste Integration for Small and Advanced Reactor Designs

TASK 1
Irradiated fuel & coolant characterisation

TASK 2
Treatment & recycling

TASK 3
Storage

TASK 4
Transportation

TASK 5
Disposal

TASK 6
Cross-cutting issues & interdependencies

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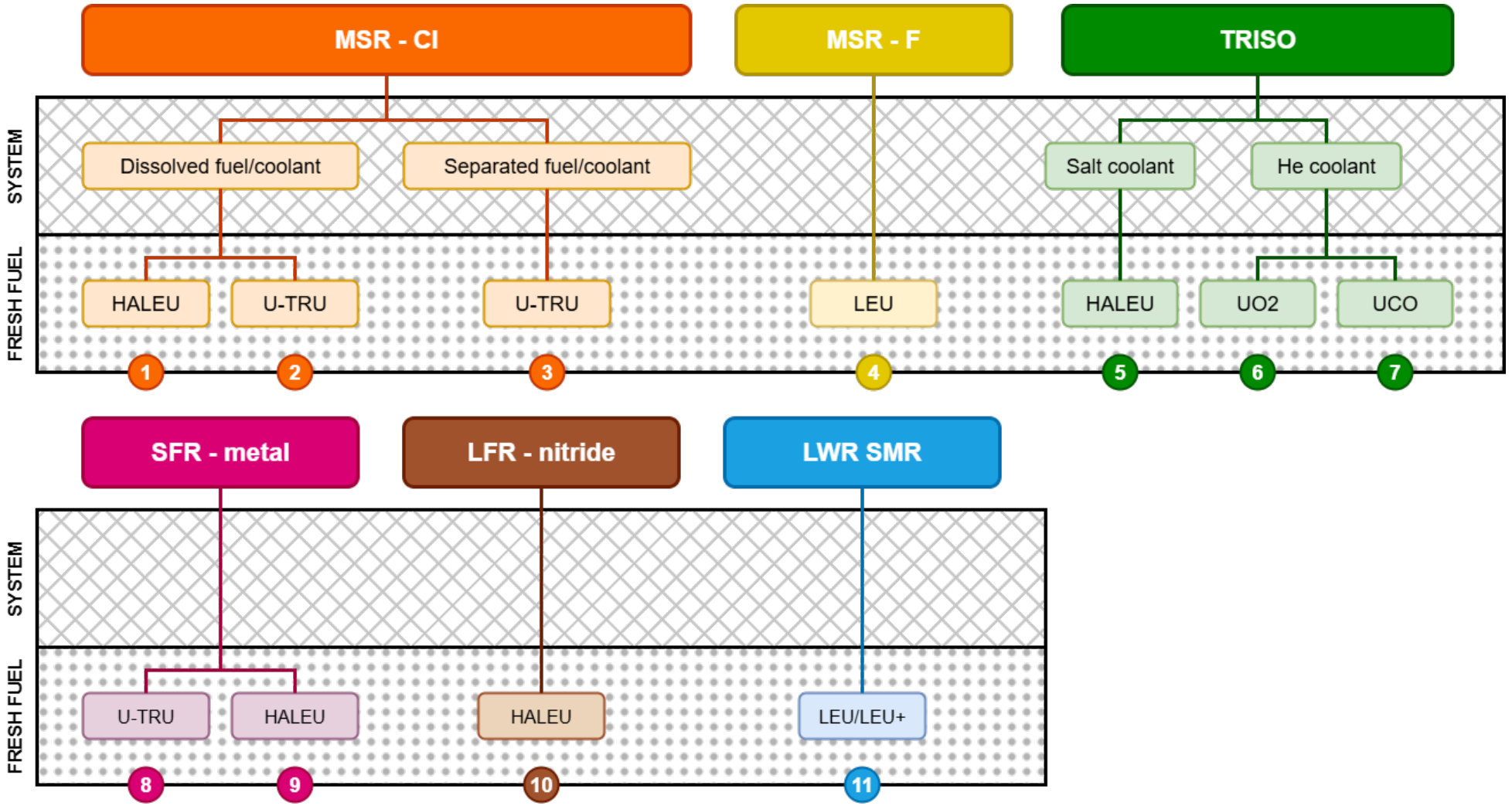
NEA WISARD Joint Project – Task 1

Waste Integration for Small and Advanced Reactor Designs



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NEA WISARD Joint Project – Tasks 2-5

Waste Integration for Small and Advanced Reactor Designs

TASK 2
Treatment & Recycling

- Assessment of the need for treatment and evaluation of the compatibility with existing treatment processes
- Re-characterisation of waste destined for final disposal after treatment/recycling
- Assessment of additional waste streams generated by treatment/recycling
- Identification of cost drivers and regulatory challenges

TASK 3
Storage

- Assessment of technological challenges
- Comparison of wet storage densities
- Feasibility study and gap analysis for dry storage
- Identification of cost drivers and regulatory challenges

TASK 4
Transportation

- Assessment of novel fuel forms/waste streams under accident conditions
- Cask compatibility assessments
- Assessment of global transport capacity and suitability of current transport infrastructure
- Identification of cost drivers and regulatory challenges

TASK 5
Disposal

- Systems analysis of disposal system performance for novel waste
- UQ/SA to identify key impacts on repository design and operation
- Assessment of waste suitability for direct disposal
- Identification of cost drivers and regulatory challenges

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NEA WISARD Joint Project – Task 6

Waste Integration for Small and Advanced Reactor Designs

Task 6 Overview

Task focus topic:	Continuous assessment of interdependencies between Tasks to allow integrated consideration of waste management strategies.
Assessment method:	Review of Task findings and other assessments as defined using methods employed in other Tasks.
Key results:	Additional calculations, where needed in other Tasks. High-level cost estimation analysis.

- As WISARD is an integration-focused project, it is crucial to capture the interdependencies between Tasks to provide a full assessment of different waste management strategies for SMRs/ARs.
- Task 6 will:
  - Review results from all Tasks continuously throughout the Project, ensuring cohesion between different working groups through periodic meetings.
  - Provide recommendations for additional calculations where needed to address interdependencies between Task results.
  - Collect high-level cost drivers from Tasks and co-ordinate with NEA expert groups to provide cost estimation analysis.

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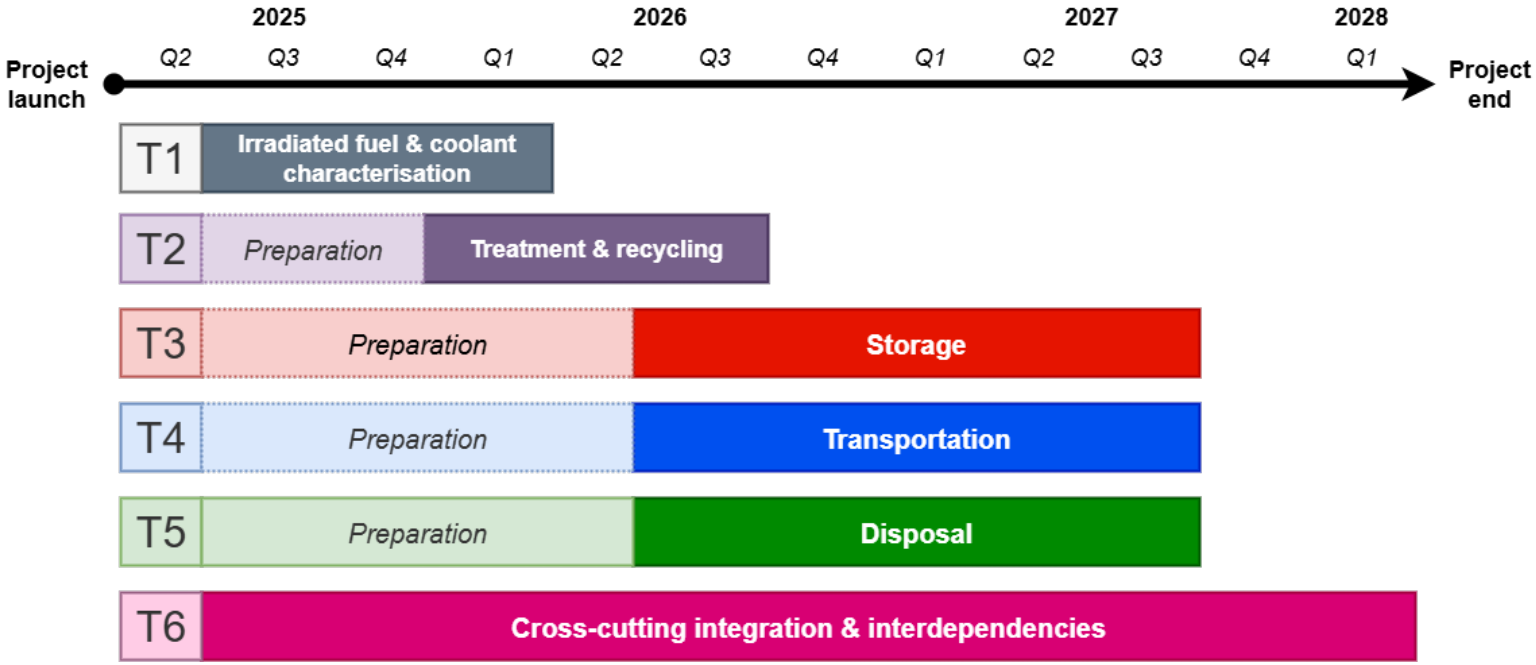
NEA WISARD Joint Project – Timeline

Waste Integration for Small and Advanced Reactor Designs



- WISARD will run for three years: Q2 2025 – Q1 2028
- As WISARD is a closed project, results will be shared only with the participants
- New participants may join throughout the project:
  - Subject to approval by the WISARD Management Board
  - Contact the NEA Secretariat for further information

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NEA
NUCLEAR ENERGY AGENCY

NEA Focus Group on the Back-End of Advanced Reactors (FG-BEAR)

NEA RWMD

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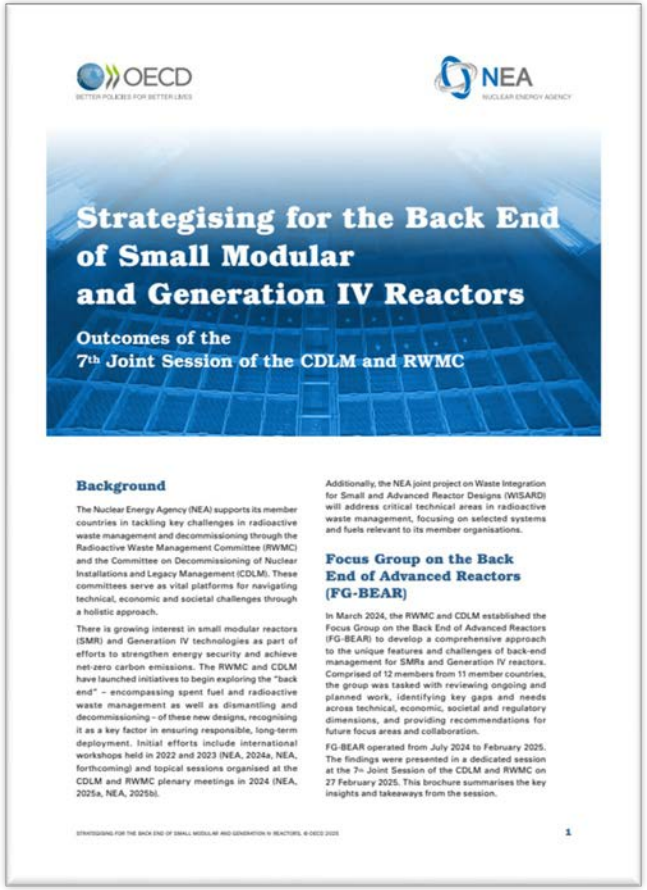
Focus Group on the Back End of Advanced Reactors (FG-BEAR)

Objectives

- Review relevant work, including planned activities and other NEA or external initiatives
- Identify key gaps and needs
- Provide considerations and recommendations for future actions

Membership

- 12 members from 11 countries
- Operated from July 2024 to February 2025


OECD

NEA

Strategising for the Back End of Small Modular and Generation IV Reactors

Outcomes of the 7th Joint Session of the CDLM and RWMC

Background

The Nuclear Energy Agency (NEA) supports its member countries in tackling key challenges in radioactive waste management and decommissioning through the Radioactive Waste Management Committee (RWMC) and the Committee on Decommissioning of Nuclear Installations and Legacy Management (CDLM). These committees serve as vital platforms for navigating technical, economic and societal challenges through a holistic approach.

There is growing interest in small modular reactors (SMR) and Generation IV technologies as part of efforts to strengthen energy security and achieve net-zero carbon emissions. The RWMC and CDLM have launched initiatives to begin exploring the "back end" – encompassing spent fuel and radioactive waste management as well as dismantling and decommissioning – of these new designs, recognising it as a key factor in ensuring responsible, long-term deployment. Initial efforts include international workshops held in 2022 and 2023 (NEA, 2024a; NEA, forthcoming) and topical sessions organised at the CDLM and RWMC plenary meetings in 2024 (NEA, 2025a; NEA, 2025b).

Additionally, the NEA joint project on Waste Integration for Small and Advanced Reactor Designs (WISARD) will address critical technical areas in radioactive waste management, focusing on selected systems and fuels relevant to its member organisations.

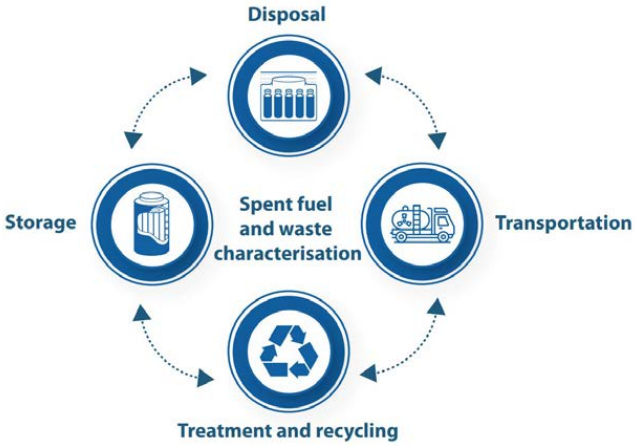
Focus Group on the Back End of Advanced Reactors (FG-BEAR)

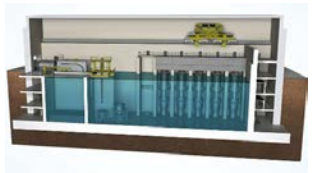
In March 2024, the RWMC and CDLM established the Focus Group on the Back End of Advanced Reactors (FG-BEAR) to develop a comprehensive approach to the unique features and challenges of back-end management for SMRs and Generation IV reactors. Comprised of 12 members from 11 member countries, the group was tasked with reviewing ongoing and planned work, identifying key gaps and needs across technical, economic, societal and regulatory dimensions, and providing recommendations for future focus areas and collaboration.

FG-BEAR operated from July 2024 to February 2025. The findings were presented in a dedicated session at the 7th Joint Session of the CDLM and RWMC on 27 February 2025. This brochure summarises the key insights and takeaways from the session.

[Link to flyer publication]

Review Phase: RWMD-supported Activities





Images: NUWARD, Terrestrial Energy, Rolls-Royce SMR, NuScale Power.

- Materials from NEA Workshops in 2022-2023
  - SMR workshop – Canada, Nov. 2022
  - Extended storage and transportation workshop – US, Dec. 2023
- Materials from Topical Sessions at Back-End Week in 2024
  - RF: *"Recent activities in the development of SMRs (non-light water) and advanced reactors and regulatory impacts including waste-related issues, recycling, reprocessing and decommissioning consideration"*
  - CDLM: *"Advanced Reactor Decommissioning and Opportunities for Learning"*
  - RWMC: *"Challenges in SMR and Advanced Reactor Back-End Management"*
- Additional information from CDLM and RWMC
  - CDLM Matrix
  - Expert Group on Costing for Decommissioning of Nuclear Installations and Legacy Management (EGCDL)
- Public information about WISARD Joint Project
  - Preparatory phase throughout 2024

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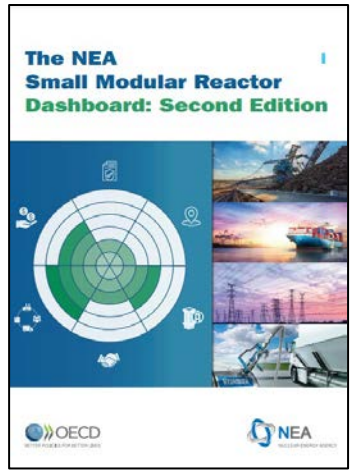
Review Phase: Further Materials

Other NEA Activities

- NEA initiative on “Accelerating SMRs for Net Zero”
- NEA SMR Dashboard
- CSNI Technical Opinion Paper No. 21 – Research Recommendations to Support the Safe Deployment of Small Modular Reactors
- CRPPH Task Force on SMRs: “High-level Analysis of Radiological Protection Challenges Related to the Deployment of Small Modular Reactors”
- *International Workshop on SMRs: Bridging Law and Technology for International Deployment (8-10 December 2025, Stockholm – Sweden)*
- Presentations from NEA Divisions at the CDLM/RWMC Joint Bureau meeting in Sept. 2024



IAEA Activities

- Waste from Innovative Types of Reactors and Fuel Cycles
- Benefits and Challenges of Small Modular Fast Reactors
- Considerations for the Back End of the Fuel Cycle of Small Modular Reactors

Further material

- Information from the ETSON Report – Chapter on Decommissioning of SMRs and modules

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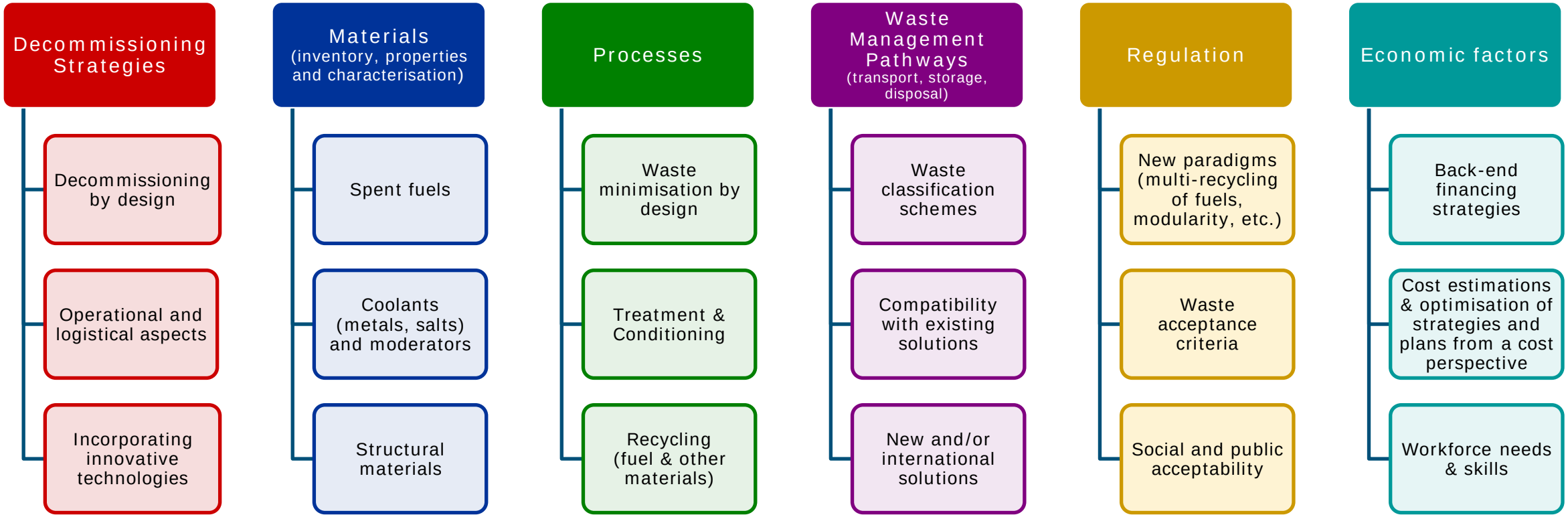
FG-BEAR – Topic Areas and Subareas of Interest



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FG-BEAR – Work Methodology & “BEAR Session”

Establish the state of the art for each topic subarea

Identify key gaps and needs

Based on review phase

Two group members for each topic area

Share findings and overall orientations

BEAR Presentation

Discuss input from RWMC/CDLM and relevant “stakeholders”

Panel Discussion

Identify considerations and recommendations for future actions

Session Takeaways

Capture key insights and takeaways in a brochure

Session Outcomes

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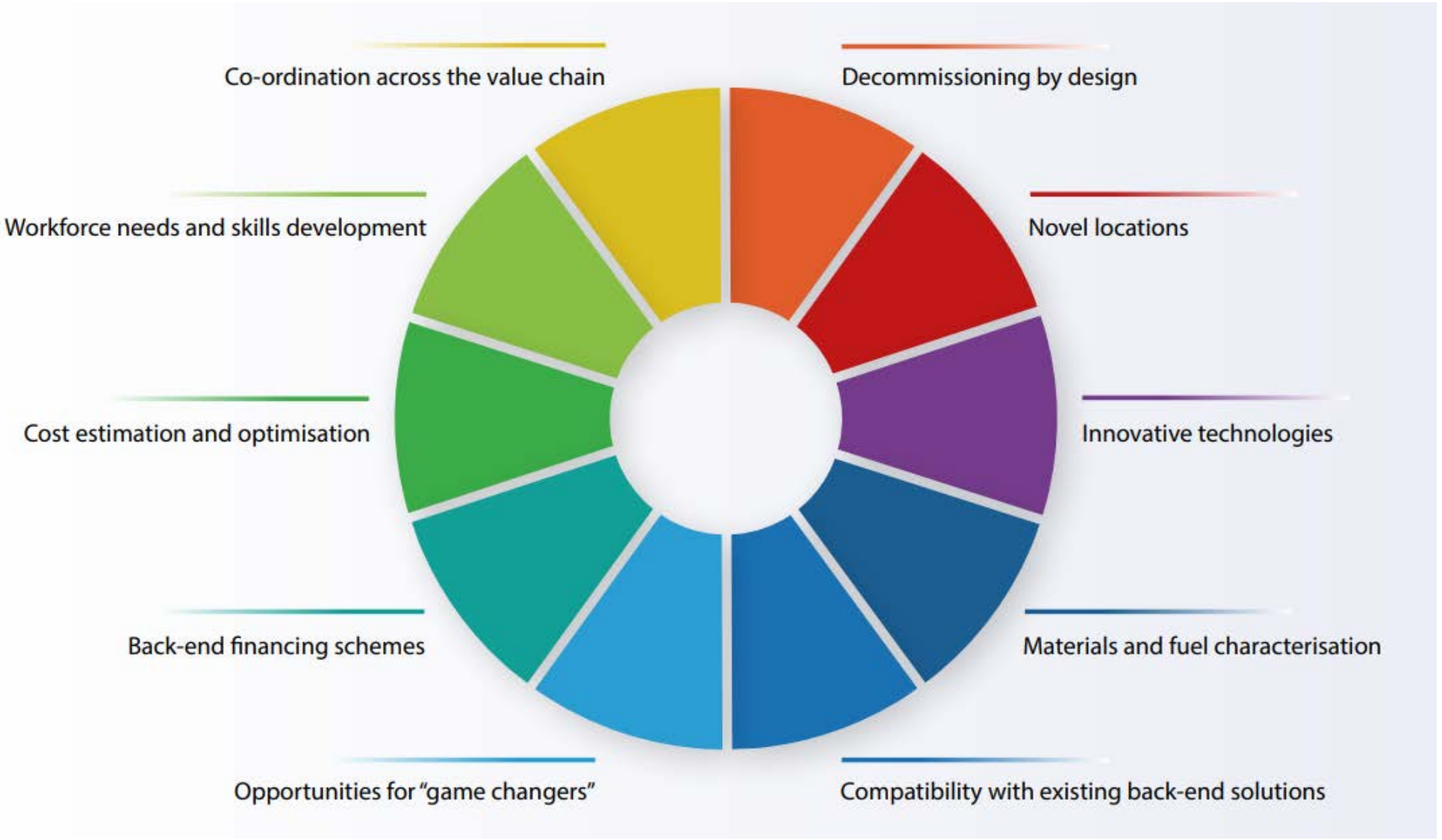
FG-BEAR – Key Focus Areas for Further Consideration



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FG-BEAR – Key Focus Areas for Further Consideration

Decommissioning by design

- SMR, nuclear island, and facility decommissioning timelines
- Dismantling scenarios linked to initial decommissioning plans
- Immediate dismantling and waste management

Novel locations

Remote, industrial, urban, maritime settings

- Tailored decommissioning and waste logistics
- Flexible waste management infrastructure
- Integrated siting and regulatory planning

Innovative technologies

AI, digital twins, robotics, etc.

- Automation and AI-driven solutions to increase decommissioning efficiency
- Digital twins for lifecycle tracking and waste management
- Advanced monitoring and robotics to improve waste treatment

Materials and fuel characterisation

Experimental data and modelling capabilities

- Radionuclide inventory calculations
- Criticality
- Unconventional waste forms like graphite, salts, and liquid metals

Compatibility with existing back-end solutions

Storage, transportation, treatment, recycling and disposal

- Treatment and disposal options for new reactor waste
- Potential for modular treatment units for SMR waste processing
- Regulatory compatibility of new fuels and materials

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FG-BEAR – Key Focus Areas for Further Consideration

Opportunities for "game changers"

Circular economy and international solutions

- Recycling technologies
- Global policies and regulations to expand the nuclear material market
- Waste minimisation criteria across different reactor types

Back-end financing schemes

- Cost uncertainties in advanced fuel cycles
- Funding for new waste treatment and storage solutions
- Long-term investment strategies

Cost estimation and optimisation

- SMR-specific cost evaluation requirements
- Waste management costs across reactor designs
- Economic feasibility of new vs. extended storage facilities

Workforce needs and skills development

- SMR-specific workforce and skill requirements
- Long-term education and training initiatives
- Workforce demand for new reactor technologies

Co-ordination across the value chain

Reactor designers, RWMO, regulators, R&D bodies

- Public concerns on SMR waste management
- Stakeholder engagement on nuclear waste policy decisions
- Back-end considerations in reactor design

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Thank you for your attention!

Visit the WISARD website or contact WISARD@oecd-nea.org for more information

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BGZ Gesellschaft für Zwischenlagerung mbH

Dry Storage of HTR Spent Fuel in Germany

Dr. Peter Kaufholz, Dr. Maik Stuke, Ralf Schneider-Eickhoff

EPRI-ESCP Winter Meeting

October 28th 2025

BGZ's task: Safe interim storage of LLW and HLW

Federal Ministry for the Environment, Climate Action, Nature Conservation and Nuclear Safety

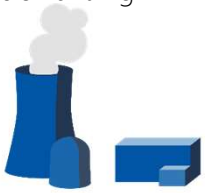
Sole shareholder of the BGZ:
Federal Republic of Germany, represented by the BMUKN

100 %

BGZ Gesellschaft für Zwischenlagerung mbH

Our mandate: **Safe interim storage**

Decommissioning & dismantling



NPP

Packaging of different types of waste

Nuclear fuel and waste from reprocessing (high-level waste)

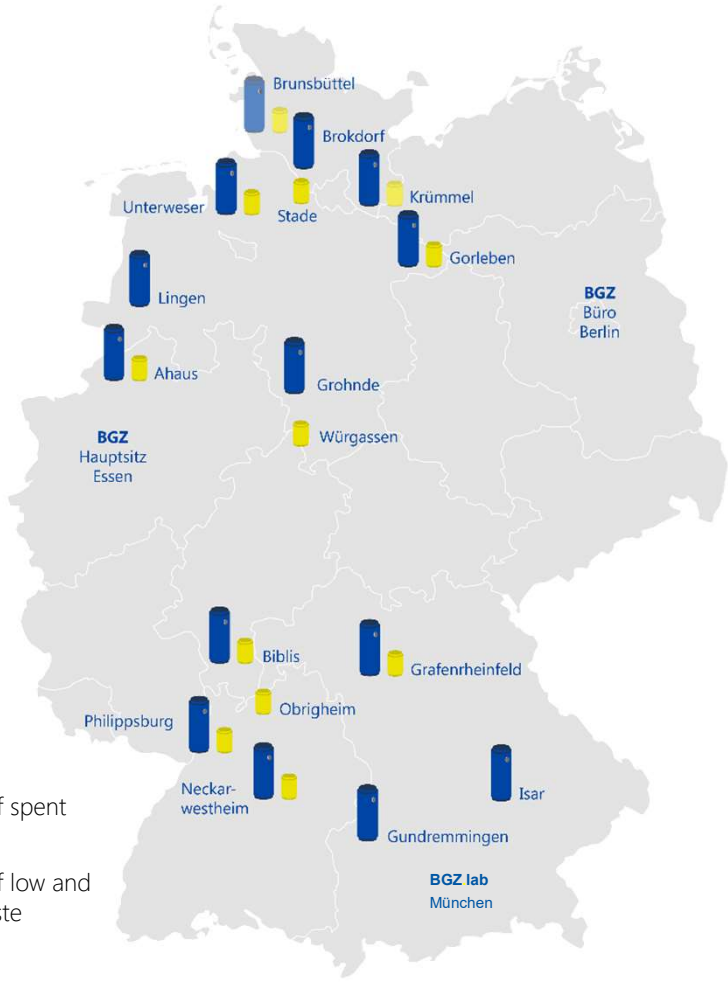
BGZ

Waste from operation and decommissioning (low and medium radioactive)

SNF/HLW geological repository (site selection act)

BGE

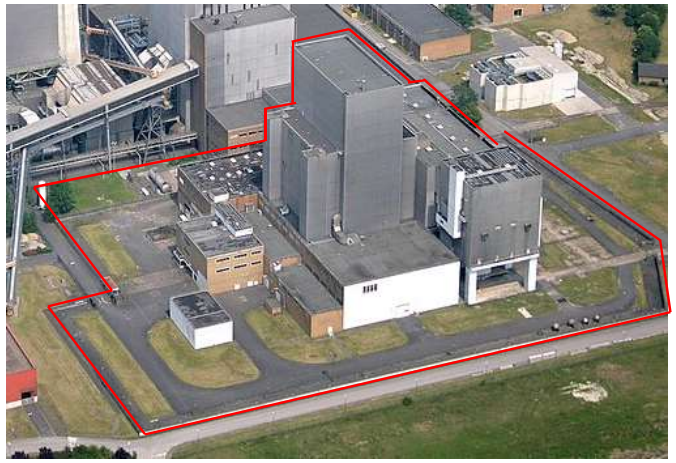
Konrad repository



Interim storage of spent nuclear fuel

Interim storage of low and medium level waste

High Temperature Reactors operated in Germany



AVR

Jülich (NRW)

Operation: 1967-1988

Type: Pebble-Bed reactor

Focus: Demonstration of PB Technology

Thermal Power: 46 MW_{th}

Net. electrical Power: 15 MW_{el}

Core with 100,000 Operating Elements (OE)

Coolant temperature (He): up to 810 °C

THTR-300

Hamm-Uentrop (NRW)

Operation: 1987-1993

Type: Pebble-Bed reactor

Focus: Power generation

Thermal Power: 760 MW_{th}

Electrical Power: 308 MW_{el}

Core with 670,000 Operating Elements (OE)

Coolant temperature (He): 750 °C

THTR operating elements

- The THTR-300 reactor core contained approx. 670,000 operating elements, of which approx. 85% were fuel elements:
  - 563,000 fuel elements
  - 76,000 graphite elements
  - 31,000 absorber elements
- Diameter of the operating elements: 60 mm
- Unirradiated THTR fuel elements (weight ca. 200 g) contain approx.
  - 1 g of highly enriched uranium (93% U 235) and
  - 10 g of thorium
  - (Th,U)O_2 kernel surrounded by a HTI-BISO coating
- Absorber elements contain hafnium and boron in the graphite matrix
- Graphite elements consist of pure graphite

PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

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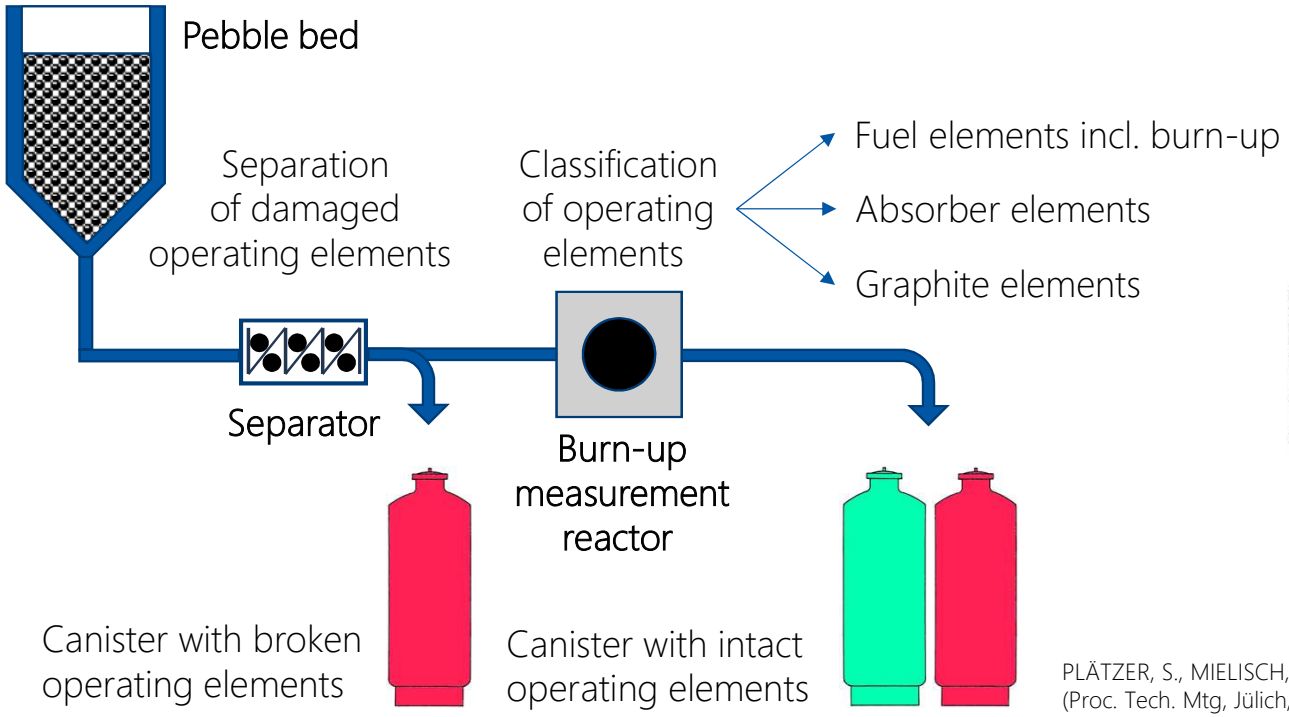
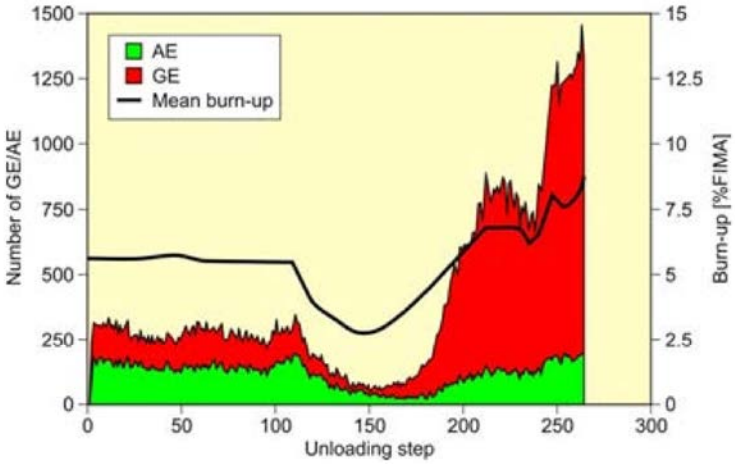
Management of unloading & packaging of the THTR fuel



Number of graphite/absorber elements and mean burn-up per unloading step with a total of 2.100 fuel elements



PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

Management of unloading & packaging of the THTR fuel

- Stepwise unloading:
  - Separation of damaged fuel during normal operation
  - Unloading of the pebble bed reactor
  - Dismantling of the burn-up measuring reactor (AMR)
- Filling the steel canisters with intact only or mixed intact/defective operating elements
- Excess graphite/absorber elements are sorted out and treated as LLW
- Packaging of AMR fuel plates in 4 specific canisters (2 per cask)
- Buffer storage of the loaded canisters in cavities within the operating element storage facility
- Loading of the casks on demand:
  - Inside the shielded loading station, insertion of the canister into the cask cavity and preliminary bolting of the primary lid, both remotely operated by a manipulator
  - Outside the loading station, application of full torque to the primary lid bolts, back-filling the cask cavity with helium, fitting of the secondary lid and leak testing of both lid systems

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Shipment of the casks

- Between June 1992 and April 1995, a total number of approx. 620,000 spent fuel elements had been transported in 305 CASTOR casks from THTR to BZA in 57 shipments, usually 6 casks on 2 railway wagons per shipment.
- Unloading operation (December 1993 - October 1994)
  - Achievement of a maximum processing rate of 11 CASTOR casks per week in 3-shift operation, 6 days a week, with optimized handling

PLÄTZER, S., MIELISCH, M., Unloading of the Reactor Core and Spent Fuel Management of THTR-300 (Proc. Tech. Mtg, Jülich, 8-10 Sept, 1997), IAEATECDOC-1043, IAEA, Vienna (1998)

Year	No. of loaded/shipped casks
1992	14
1993	6
1994	278
1995	7

Storage of SNF at BGZ storage facility in Ahaus

- First licensing of Ahaus interim storage facility: April 10, 1987
- Licensing of CASTOR THTR/AVR for storage of THTR fuel elements in Ahaus: March 17, 1992
- Commissioning of the interim storage facility with the first storage of CASTOR THTR/AVR casks: June 25, 1992

- Approved / max. total heat load of the ISF: 17 MW / < 0.1 MW
- Approved / used number of storage positions: 420 / 50

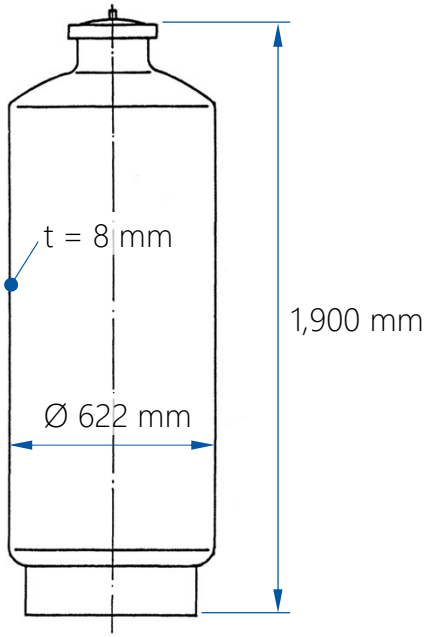
- Current inventory:
  - 305 CASTOR THTR/AVR
    with THTR fuel (operating elements / burn-up measuring reactor)
  - 3 CASTOR V/19
  - 3 CASTOR V/52

    } with LWR fuel assemblies from NPPs
  - 18 CASTOR MTR 2
    with fuel elements from the Rossendorf VVR-SM research reactor

CASTOR THTR/AVR in storage configuration

- The monolithic cask body of the CASTOR THTR/AVR is made of ductile cast iron, without additional moderator or cooling fins
  - Wall thickness (side/bottom): 370 mm
- Overall length: ca. 2.8 m
- Outer diameter: ca. 1.4 m
- Weight cask body: 23.0 t
- Gross weight loaded cask: 26.1 t
- Weight steel cannister, loaded: 770 kg
- Weight steel cannister, empty: 350 kg
- Loading: 1 steel canister with approx. 2,100 operating elements
  - Cannister is helium filled (1 bar) and sealed (< 10^{-4} Pa·m^3/s)

Steel canister for intact THTR-OE



Operational experinces

- The design and approved heat load is approx. 200 watts per cask
  - Real burn-up (max. 8.8% fima per cask, mean < 6% fima per cask) significant lower than design burn-up (mean 11.4% fima per cask)
  - Actual heat load per cask approx. 100 W at the time of loading
- Due to the lower heat load, risk of falling below the dew point
  - Moisture may cause corrosion at unprotected cask components
- Countermeasures:
  - Closure of the vents to reduce the amount of moisture entering with the outside air and to increase the minimum temperature inside
  - Additional corrosion protection measures at the outer cask surfaces → Maintenance campaign in the early 2000s

- Regular 10-year inspection carried out on six casks to date
  - No unexpected effects, except for normal signs of wear and aging

Outlook / News

- Decision was taken in August 2025 to transport AVR OE to BGZ

- BGZ will receive 152 CASTOR THTR/AVR from JEN
  - OE Packed in two ss canisters per cask
  - BGZ storage site will be TBL Ahaus (as for THTR)
  - In total 288,161 OE from AVR
  - Only intact OE were loaded to the casks

- Operation Elements for AVR are of different types Ziermann et al. Jül 3448 (1997)
  - Different kernel types (BISO/TRISO)
  - Different fuel chemical form (Carbide/Oxide)
  - LEU fuel (U_{nat}; 15 %) OE

- First Transport is planned for late 2025

Summary

- BGZ, by far the largest operator of interim storage facilities in Germany, has extensive operational experience, including with dual-purpose casks for spent THTR fuel.
- The entire high-level radioactive inventory of the THTR-300 nuclear power plant was packed into a total of 305 CASTOR THTR/AVR casks and transported to the Ahaus interim storage facility, where it has been safely stored for more than 30 years.
- The long-term safety of the inventory was assessed as part of the licensing procedure. Due to the low loads during storage compared to normal operation in the reactor core and the high safety margins in the cask design, no safety-relevant ageing effects are expected.
- BGZ expects 152 CASTOR THTR/AVR from AVR-operation to be transported to TBL-Ahaus

BGZ Gesellschaft für Zwischen-lagerung mbH

Thank you for your interest !

BGZ's research program

TRISO Storage and Transportation

Consideration of NUREG 2215 Guidance

Dan Moneghan, PhD
Senior Technical Leader, EPRI

Extended Storage Collaboration Program
October 28 2025

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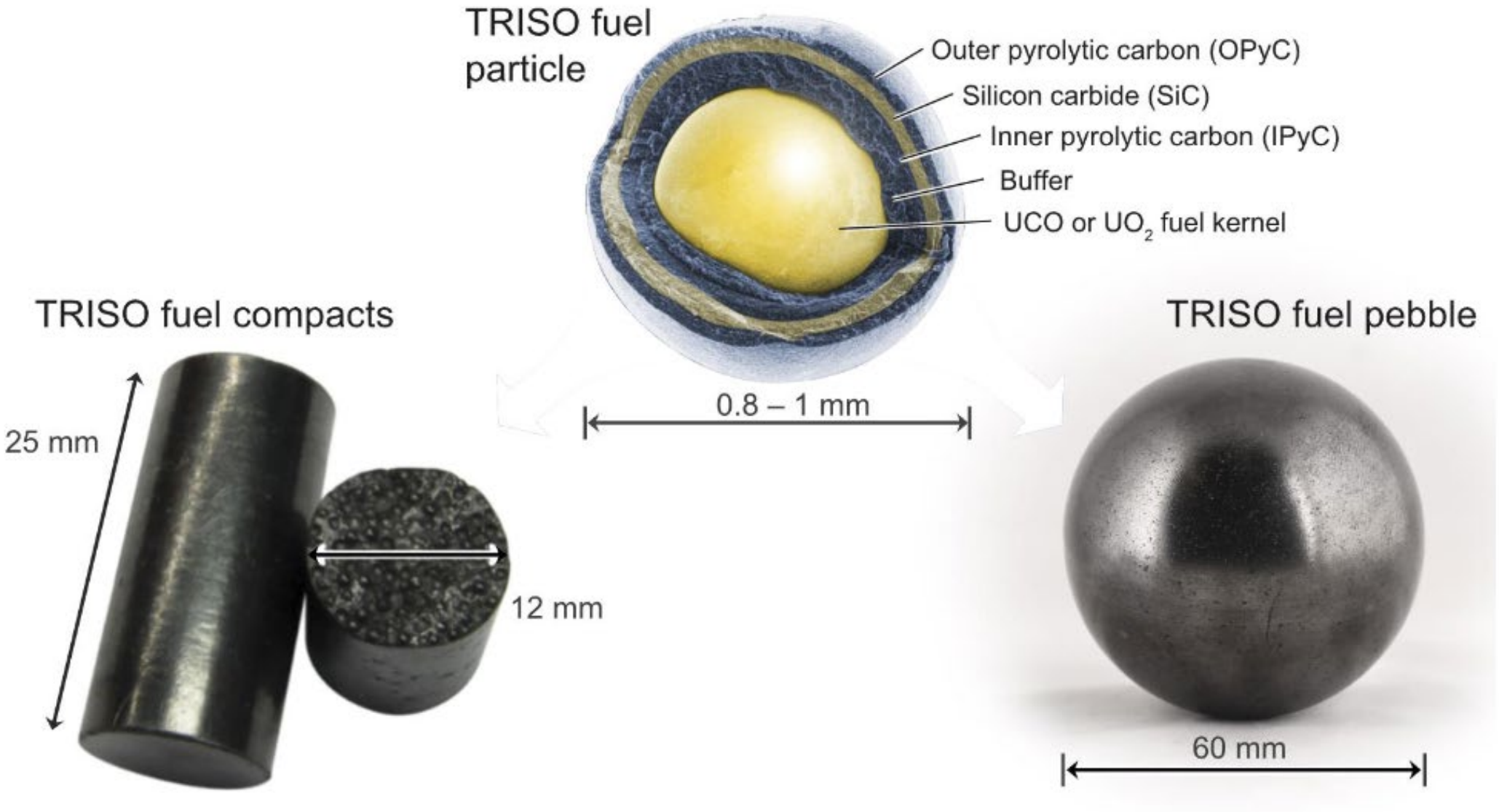
Tristructural Isotropic Coated Particle Fuel



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Applicability of NUREG-2215 and 2216 to TRISO

Assessment Categories

- ***Cat 1**: Regulatory limits that are established based on LWR performance:* Various parts of the regulatory guidance rely on fuel performance characteristics of LWR fuel. For instance, the maximum storage temperature of used fuel under normal conditions is 400°C, which was established to ensure zirconium cladding integrity.
- ***Cat 2**: Descriptions of fuel criteria are LWR specific:* Various parts of the regulatory guidance either list characteristics of fuel that must be outlined that are LWR specific (e.g., assembly size) or are LWR-specific in nature (e.g., the use of maximum enrichment fuel pins as a basis for license applications).
- ***Cat 3**: Guidance provided for LWR fuels but not for non-LWR fuels*: Certain parts of the license application guidance documents present methods with which regulatory requirements can be satisfied for LWR fuel.
- ***Cat 4**: Additional considerations are often LWR-specific*: Some additional considerations that need to be incorporated into license applications are LWR-reactor specific in nature. For example, the consideration of cobalt-containing crud in radiation source term considerations.

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Applicability of NUREG-2215 to TRISO

Chapter	Subsection	Guidance Category
Chapter 1 (General Information Evaluation)	Section 1.5.2 (DSS or DSF Description and Operational Features)	Cat 2
	Sections 1.3.4/1.5.4 (Contents)	Cat 2
Chapter 3 (Principal Design Criteria Evaluation)	Sections 3.4.2/3.5.2 (Design Bases for Structures, Systems, and Components Important to Safety)	Cat 1, Cat 2, Cat 4
	Table 3-2 (Outline of Design Criteria and Bases)	Cat 2
Chapter 4 (Structural Evaluation)	Sections 4.3.1/4.5.1 (Structures, Systems, and Components Important to Safety)	Cat 2
	Sections 4.3.2/4.5.1 (Design Criteria)	Cat 1, Cat 2
	Section 4.5.5 (Normal and Off-Normal Conditions)	Cat 1, Cat 2, Cat 4
	Section 4.5.6 (Accident Conditions)	Cat 1, Cat 2, Cat 4

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Chapter	Subsection	Guidance Category
Chapter 5 (Thermal Evaluation)	Section 5.1 (Review Objective)	Cat 1
	Sections 5.4.2/5.5.2 (Material and Design Limits)	Cat 1, Cat 2, Cat 3, Cat 4
	Sections 5.4.3/5.5.3 (Thermal Loads and Environmental Conditions)	Cat 2
	Sections 5.4.4/5.5.4 (Analytical Methods, Models, and Calculations)	Cat 2, Cat 3
Chapter 6 (Shielding Evaluation)	Sections 6.4.2/6.5.2 (Radiation Source Definition)	Cat 2, Cat 3
	Sections 6.4.4/6.5.4 (Shielding Analyses)	Cat 2, Cat 3
Chapter 7 (Criticality Evaluation)	Section 7.5.2 (Fuel Specification)	Cat 2, Cat 3
	Section 7.5.3 (Model Specification)	Cat 2, Cat 3
	Section 7.5.5 (Burnup Credit)	Cat 3
Chapter 8 (Materials Evaluation)	Section 8.5.15 (Spent Fuel)	Cat 1, Cat 2, Cat 4
Chapter 9 (Confinement Evaluation)	Section 9.5.3 (Nuclides with Potential for Release)	Cat 2, Cat 3, Cat 4

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5.4.2 Material and Design Limits

An application to store spent fuel or reactor-related greater-than-Class-C (GTCC) waste in an ISFSI or to store spent fuel, HLW, or reactor-related GTCC waste in an MRS must include the design criteria for the proposed storage installation (10 CFR 72.120(a)). The ISFSI or MRS must be designed, made of materials, and constructed to ensure that there will be no significant chemical, galvanic, or other reactions between or among the storage system components, spent fuel, reactor-related GTCC waste, and/or high level waste including possible reaction with water

uranium (MTU)), minimum acceptable cooling time of the spent fuel before storage in the spent fuel storage cask, maximum heat designed to be dissipated, maximum spent fuel loading limit, condition of the spent fuel (i.e., intact as
ting atmosphere
requirements (10 CFR 72.236(a)). Des
vided for SSCs
important to safety (10 CFR 72.236(b))

Cat 1 – Numerous TRISO reactors intend to implement a dry spent fuel route with no pool storage period.

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To guarantee the integrity of zirconium-based alloy cladding, the maximum calculated fuel-cladding temperature should not exceed 40 enheit (°F)) for normal conditions of storage and short- ing and backfilling. A higher temperature limit may only SNF) (less than 45 gigawatt days MTU), as long as th estimate cladding hoop stress is equal to or less nd pounds per square inch (ksi)) for the proposed f s, repeated thermal cycling should be limited to less than 10 cycles when the cladding temperature difference exceeds 65 °C (149 °F). For off-normal and accident conditions, the maximum zirconium-based cladding temperature should not exceed 570 °C (1,058 °F).

Cat 1 – No zirconium-based alloy cladding

Cat 1 – Thermal cycling limits are based on LWR fuel limits

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The maximum internal pressure of the fuel container should remain within its design pressures for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the initial fill gas and 30 percent of the fission product gases generated within the fuel rods during operation.

Cat 1 – rupture limits derived from LWR fuel performance

Cat 2 – No fuel rods present in TRISO

Cat 1 – No fill gas present in TRISO particles or fuel elements

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2025 Advanced Reactor Fuel Workshop Summary

Overview for the Extended Storage Collaboration Program

Aaron Totemeier, PhD
Principal Technical Leader, Nuclear Fuels

October 28, 2025

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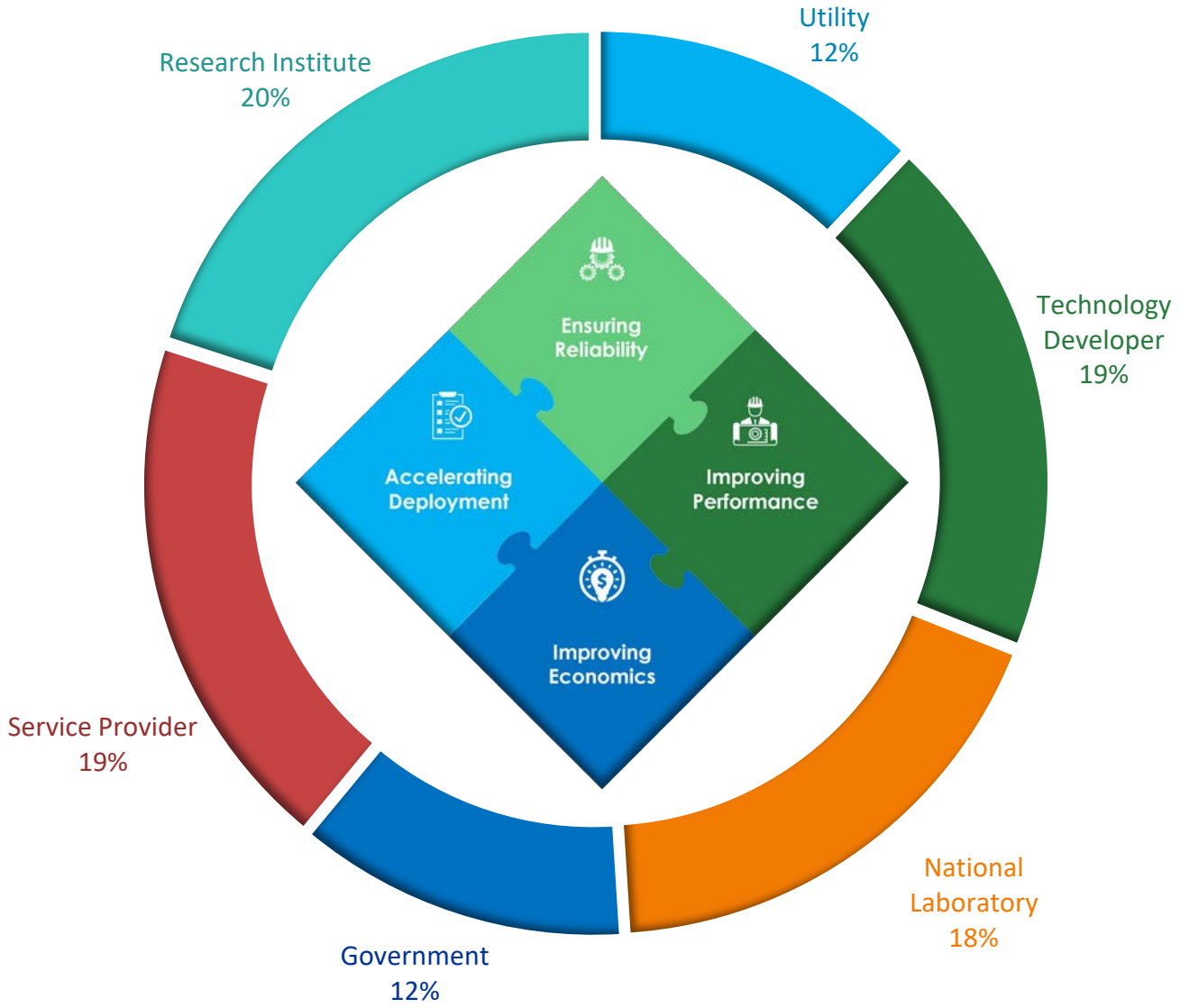
Workshop Summary

- On June 3 – 5, 2025 EPRI hosted the Advanced Reactor Fuels Workshop.
- The workshop brought together over 160 stakeholders from across the nuclear sector to identify and prioritize remaining gaps in the path to first deployment and eventual commercial-scale deployment of advanced reactor fuels.
- Participants submitted gaps via a pre-workshop survey, an in-workshop submission process, and through presentations and discussions.
- Workshop participants were asked to rank their top 3 gaps across several topic areas.

Participant Organizations



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Session Topics:

- Uranium Fuel Supply
- Solid Fuel:
  - Supply Chain & Manufacturing
  - Transportation & Handling
  - On-site & In-core Operations
- Liquid Fuel:
  - Synthesis, Transportation & Handling
  - On-site & In-core Operations
- Fuel Qualification & AFQ
- LWR Lessons Learned Panel
- Used Fuel Storage & Handling

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Workshop Sessions & Highest Ranked Gaps

Session Title Highest Ranked Gap	Gap Category
Uranium Fuel Supply (tie) Lack of firm market demand for HALEU is delaying investment in large-scale production infrastructure.	Execution
Insufficient U.S. capacity for uranium mining, conversion, and enrichment to meet projected HALEU needs.	Infrastructure
Solid Fuel: Supply Chain & Manufacturing Limited manufacturing capability and throughput for advanced nuclear fuels.	Infrastructure
Solid Fuel: Transportation & Handling Limited availability of fresh fuel shipping containers with broad and flexible licenses for solid fuels.	Infrastructure
Liquid Fuel: Synthesis, Transportation & Handling Commercial-scale production capacity for 100 kg-scale batches of constituent salts are not yet established.	Technical Knowledge
Liquid Fuel: On-site & In-Core Operation The behavior and evolution of fission products (e.g. solubility, volatility) in molten salt systems is not well understood.	Technical Knowledge
Solid Fuel: On-Site & In-core Operation Greater flexibility is needed in applying Technical Specification Requirements or Limiting Operating Conditions when test data is unavailable for reactor licensing.	Regulatory
Fuel Qualification Establish the validity range and uncertainty in fuel performance models.	Technical Knowledge

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EPRI 2025 Advanced Reactor Fuels Workshop Poll

Used Fuel Storage & Handling (Post-Session)

Total Votes = 29
Highest Possible Result = 8.0

		Category	# of Votes	Result
1	Transportation requirements for used salt fuels containing fission products (FP) and TRU are not fully defined.	Regulatory	19	4.17
2	Final disposition pathways for sodium-bonded metal fuels remain uncertain.	Regulatory	16	3.72
3	There is a lack of licensed Consolidated Interim Storage Facilities (CISFs) for advanced reactor fuels.	Infrastructure	15	3.55
4	Regulatory limitations for on-site storage of uranium-transuranic (U-TRU) fuels are undefined.	Regulatory	13	2.79
5	No standard contract for used nuclear fuel (UNF) exists for advanced reactors.	Legal	12	2.76

# of Votes = Number of total voting participants who selected the gap for ranking.
Result = Average score of rankings for the gap.

Used Fuel Storage & Handling Gaps (Top 5)

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AR Fuel Development Near-Term Focus Areas

- Solid Fuels
  - Manufacturing cost reduction (U-10Zr; TRISO)
    - process efficiency improvements, waste reduction, alternative process development
  - Expanding the performance envelope for TRISO fuels
  - Developing reliability frameworks for solid fuels based on existing LWR fuel experience
- Liquid Fuels
  - Scale-up of fuel component production
  - Thermophysical property measurement & evolution
  - Developing transportation requirements
- Accelerating fuel qualification (all fuels)
  - AFQ framework is in place; application and standardization of the methodologies is needed

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Future of EPRI AR Fuel Workshops

- Feedback from the first ARFW demonstrated clear need for more cross-technology engagement
- Potential future workshop structure:
  - Day 1: general cross-cutting fuel topics & issues
  - Day 2 & 3: fuel-specific breakouts focused on technical deep-dive

2026 ARFW planned for Q3 … formal announcement to come.

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Duke Energy - 2025 Aging Management Inspections

EPRI ESCP – October 29, 2025

Discussion Topics

- McGuire NAC-UMS Baseline AMP Inspection
- Second Oconee TN NUHOMS AMP Inspection (5 years after Baseline Inspection)
- Robinson TN NUHOMS Baseline AMP Inspection
- Overall Summary/Conclusions from Duke Energy AMP Inspections

Background - McGuire ISFSI

- TN-32
  - Ten (10) systems loaded - 2001-2003
  - First system in-service – February 2001
  - Baseline AMP inspection - October 2022
- NAC-UMS
  - 28 systems loaded – 2004-2009
  - First system in-service - December 2004
  - Entered PEO – December 2024
  - Baseline AMP inspection – March 2025
- NAC-MAGNASTOR
  - 43 systems loaded to date
  - First system in-service – August 2013
  - Enter PEO – August 2033

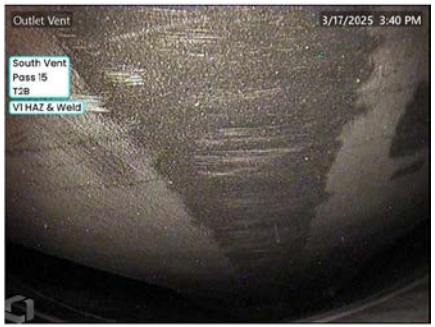
Baseline Inspection Scope

- Completed scope
  - TSC accessible external surfaces
  - TSC accessible welds and heat affected zones
  - Internal VCC accessible metallic components including inlets/outlets
  - External VCC accessible metallic components
  - Reinforced VCC concrete structure
  - ISFSI Pad & groundwater monitoring

- Not required at this time
  - VCC Lid and shield plug were not removed
  - Transfer Cask & Adapters – prior to next use (no longer loading UMS)
  - High Burnup Fuel AMP - First Duke Energy HBU tollgate for Robinson TN NUHOMS 24PTH in 2030

McGuire UMS Inspection Results

- TSC & VCC were observed to be in satisfactory condition with no signs of aging-related degradation
- The inspection did not identify any indications exceeding the established acceptance criteria
- Water stains from rain infiltration were observed on TSC surface & VCC liner
- White deposit streaks/scuff marks on TSC from initial dry runs where TSC rubbed against painted internals of transfer cask



McGuire UMS Inspection Results (continued)

- VCC liner – limited surface discoloration, areas of damaged coating, deposits of calcium carbonate
- Minimal amount of foreign material was present (bugs, spider webbing and small amounts of other unidentified debris)

Background – Oconee ISFSI

- Site Specific ISFSI first loaded in 1990
- Site Specific License renewed in 2009. Separate aging management program
- General License ISFSI (TN NUHOMS CoC 1004) first loaded in February 1999
- 184 total systems loaded to date (TN NUHOMS 24P, 24PHB & 24PTH-S-LC)
- Baseline CoC 1004 AMP inspection - July 2020
- Second AMP inspection - April 2025

Oconee NUHOMS Internal Inspection Results

- DSC & HSM were in satisfactory condition with no evidence of aging-related degradation
- Scratches from initial DSC insertion were present with depths less than 0.016” and no signs of aging-related degradation
- Water blemishes from rain observed on DSC and heat shields
- Manufacturing marks observed
- Heavy dust deposits from 11:00-1:00 positions on DSC (horizontally inserted)
- Coating damage on support structure legs
- Debris on floor of HSM

Oconee NUHOMS External & Transfer Cask (TC) Inspection Results

- HSM Exterior was in good condition with no significant degradation
- Leaching on the HSM door and front face of HSM
- Corrosion on the HSM door and docking ring
- No evidence of spalling or differential settling
- Compared 2020 & 2025 inspection photos – “essentially the same”

- TC in good overall condition. No signs of damage or defects
  - General wear consistent with normal use

Robinson Background

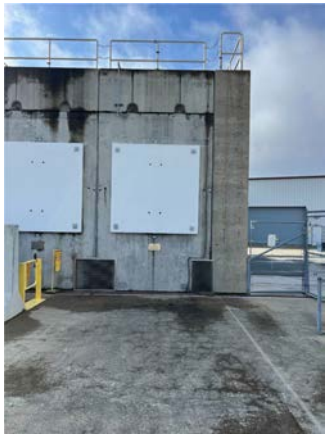
- Site Specific NUHOMS 7P ISFSI first loaded in 1989
- Site Specific License was renewed in 2005. Separate aging management program
- General License (TN NUHOMS CoC 1004) first loaded August 2005
  - First NUHOMS 24PTH and HSM-H loadings
- 45 total systems loaded to date (NUHOMS 7P and 24PTH)
- Baseline AMP inspection performed in May and July 2025

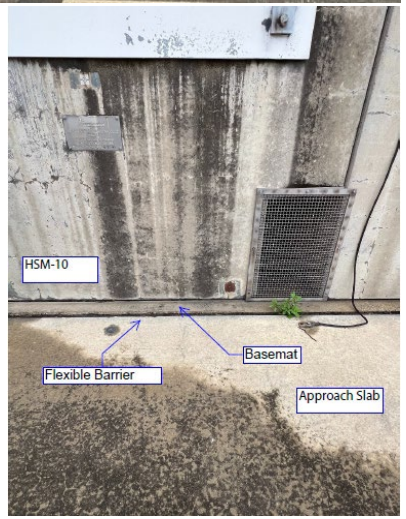
Robinson NUHOMS Interior Inspection Results

- DSC & HSM found to be in satisfactory condition. No evidence of aging-related degradation.
- Scratching on DSC from initial insertion was present but less than 0.002” in depth and showed no evidence of aging-related degradation
- Dust/ hardened deposit buildups on DSC between 10:00-2:00 position observed
  - Same condition in backup HSM
  - Attributed to water intrusion through louvred heat shields onto dust layer
- Some DSC support structure paint flaking
- Water blemishes from rain infiltration
- No corrosion identified

Robinson NUHOMS Exterior Inspection Results

- Good condition with no indications that exceeded acceptance criteria
- No evidence of spalling or differential settlement
- Door had areas where topcoat had peeled/worn off but primer in good condition and no evidence of corrosion
- Minor corrosion on bolts/washers
- Roof in good condition. No indications identified
- Transfer cask AMP inspection was not required at this time
  - In service less than 10 years





Duke Energy AMP Inspections - Overall Summary/ Conclusions

- All equipment inspected (canisters, overpacks, ISFSI pads and transfer casks) were in satisfactory condition with no evidence of aging-related degradation

- No issues were identified during Duke Energy AMP inspections that require further evaluation
- We continue to demonstrate that there are no aging-related degradation concerns with in-service dry storage systems
- Second AMP inspection results virtually identical to baseline inspection results

DUKE
ENERGY®

Davis-Besse

Aging Management Program Inspection #2

29 October 2025

Glossary

AMP – Aging Management Program

CISCC – Chloride-Induced Stress Corrosion Cracking

CoC – Certificate of Compliance

DSC – Dry Shielded Canister

HSM – Horizontal Storage Module

ISFSI – Independent Spent Fuel Storage Installation

Objectives

- Briefly discuss the background AMP inspections
- Summarize the inspections and preliminary results
- State Vistra actions going forward
- Provide visuals from inspection

Background Information

- DSCs and HSMs loaded under CoC 1004 in 1996
- First AMP inspection performed in September 2020 with no findings exceeding the acceptance criteria
  - HSM #1 with DSC *24P-R01* chosen as most limiting canister
- CISCC Tollgate Assessment #2 performed in calendar year 2021
- CISCC Tollgate Assessment #3 performed in calendar year 2024
- AMP Inspection #2 performed in August 2025 with no findings exceeding the acceptance criteria

AMP Inspection Details

What needs to be inspected?

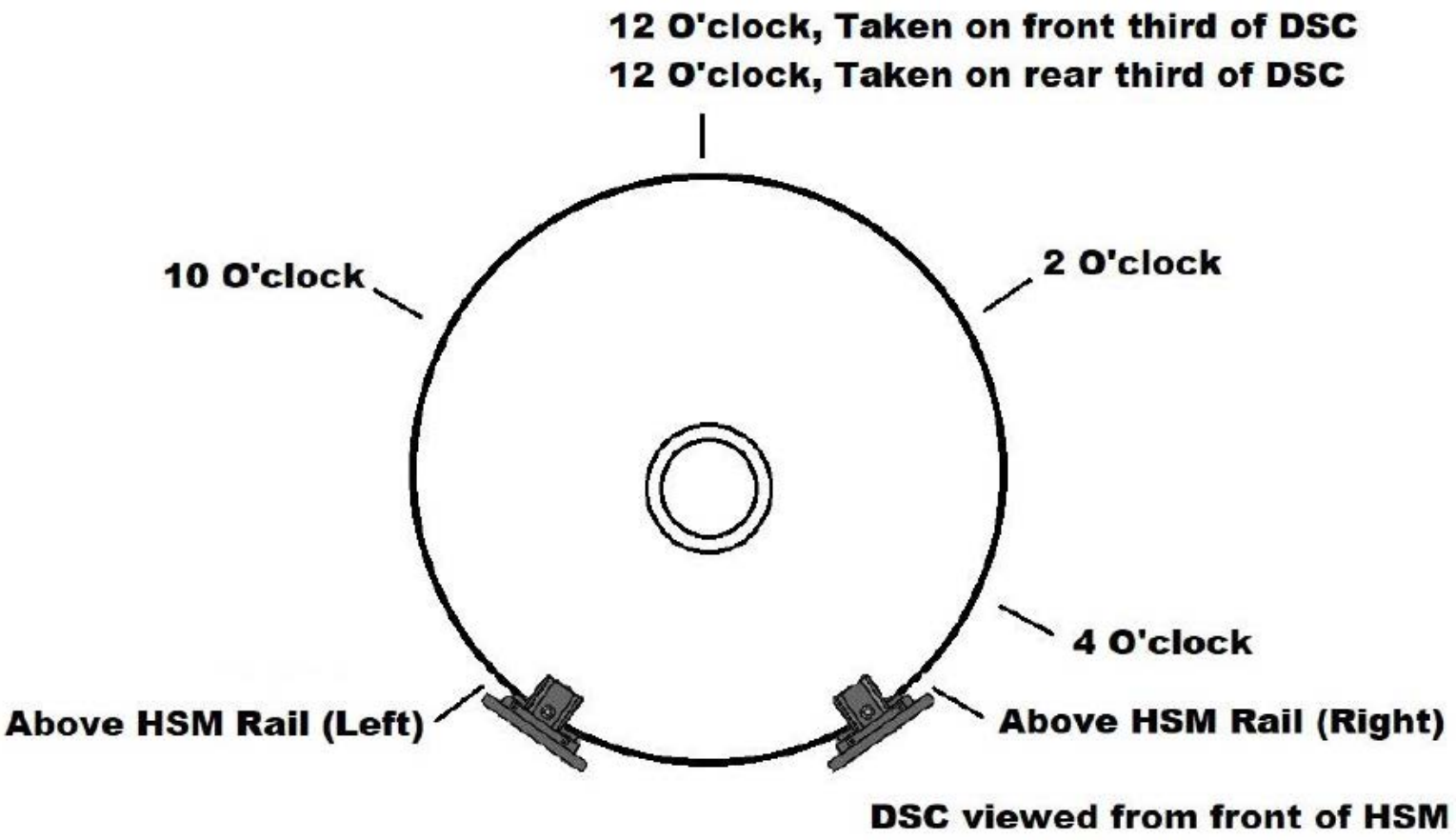
- HSM Internals
- HSM Externals
- DSC Externals
- Chloride Samples from DSC Exterior (5 samples minimum)
- HSM Basemat Levelness (*pending*)
- ISFSI Groundwater Sampling (*pending*)

What are we looking for?

- Concrete (HSM - Externals & Internals)
  - Detection of aging effects, as described in ACI 349.3R (cracking, spalling, etc.)
- Metallic Surfaces (DSC Externals & Support Structures)
  - DSC surfaces, welds and heat affected zones (HAZs), DSC support structures
  - Looking for loss of material, discoloration, cracking, water staining
- Chlorides (Samples collected from DSC exterior)
  - Presence of chlorides on DSC surface or inside gouges/scrapes
  - Total of 6 samples taken from DSC surface – right rail, 2, 4, 10, 12 (x2) o'clock position
- Base Mat (Immediate area around the HSM)
  - Differential settlement of the base mat for the HSM shall not exceed 2 inches (front-to-back or side-to-side)
- Ground Water (ISFSI Groundwater Sampling)
  - pH $\geq$ 5.5, Chlorides $\leq$ 500 ppm, Sulfates $\leq$ 1500 ppm

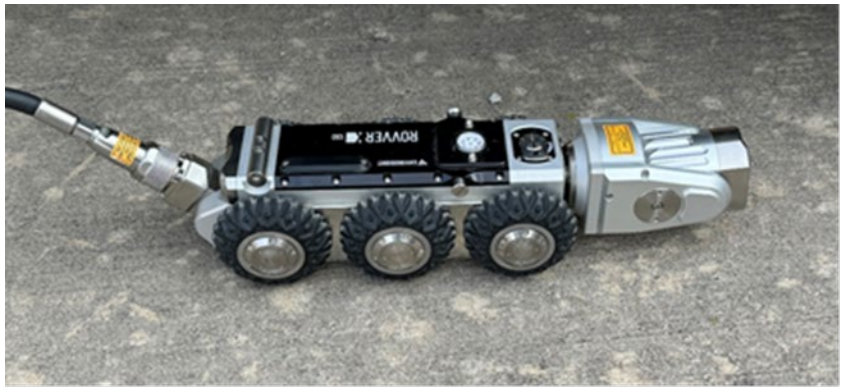
AMP Inspection Details

Figure 9 - Deposit Sampling Locations



AMP Inspection Details

- Orano TN performed the inspections
- Utilized small robotic crawlers (supplied by sub-vendor RTT)
  - Crawlers fitted with VT-3 and VT-1 cameras
  - Also fitted with ability to collect chloride samples from DSC surface



AMP Inspection Findings

- Chloride sample results

Table 1. Results in Total Micrograms

	Cl	NO_3	PO_4	SO_4
10 o'clock	15.34	12.6	7.66	127.1
12 o'clock Left	7.48	6.45	3.78	61.07
2 o'clock	7.27	6.12	4.70	43.32
12 o'clock	17.46	20.57	17.1	184.6
4 o'clock	1.50	<0.20	<1.00	1.30
Rail Right	1.53	<0.20	<1.00	1.348

Table 2. Results in mg/m^2

	Cl	NO_3	PO_4	SO_4
10 o'clock	7.93	6.51	3.96	65.68
12 o'clock Left	3.87	3.33	1.95	31.56
2 o'clock	3.76	3.16	2.43	22.39
12 o'clock	9.02	10.63	8.86	95.40
4 o'clock	0.78	<0.10	<0.52	0.67
Rail Right	0.79	<0.10	<0.52	0.70

Note: Using a sampling area of 19.35 cm^2

AMP Inspection Findings

- HSM Externals and Internals
  - No findings except minor spalling on the roof under the west tie rod that did NOT exceed acceptance criteria (pictured later)
- DSC Externals
  - No findings that exceeded acceptance criteria
  - Various areas of discoloration/debris but upon VT-1 inspection, found no loss of material (pictured later)
  - Gouges/scratches from 2020 had not changed in 5 years
- Actual dose for campaign: **13.2 mrem** (versus 17.1 in 2020)
- Basemat levelness survey and groundwater sampling est. completion by year's end

AMP Inspection Findings

- NRC on-site to observe inspections
  - Overall, NO FINDINGS from the NRC just general observations
  - NRC took issue with wording in DB AMP Basis Document

5.7.1 HSM AMP–Corrective Actions Background

Conditions that do not meet the acceptance criteria in Section 5.6 are entered into the site's corrective action program for evaluation and determination of any actions to be taken.

5.5.2 HSM AMP–Monitoring and Trending Davis-Besse Specific Implementation Information

All deficiencies identified during the inspections will be documented in accordance with procedure NOP-LP-2001 (Ref. 2.10).

Trending of the deficiencies identified during the inspections is performed in accordance with procedure NOP-LP-2001 (Ref. 2.10).

  - NRC read 5.5.2 as DB having to enter every deficiency into the CAP whereas Vistra only entered items into the CAP that exceeded the acceptance criteria (IAW 5.7.1)
  - In the end, Vistra wrote a CR to alter language in AMP BD
  - NRC was happy with communication in the field, level of professionalism, and stopping work to discuss future actions/observations/etc.

AMP Inspection Next Steps

- Receive and review final report from Orano
  - Upon acceptance, incorporate into DB site AMP basis document
  - Ensure final report and 2024 CISCC Tollgate Assessment #3 are entered into AMID
- Complete ISFSI basemat levelness survey and review results
  - If results are consistent with last year's, consider altering frequency of surveys from yearly to every 5 years
- Complete groundwater sampling and review results
  - Revise DB site AMP basis document with groundwater sampling results
- If final results of AMP inspection are satisfactory (i.e. no findings exceeding acceptance criteria), consider 72.48 to move AMP inspections from 5-year periodicity to 10-year periodicity?

AMP Inspection - Appendix

- Spalling on roof observed again (found in 2020 inspection, initially found in 2013)
  - Dispositioned as mechanical spalling in 2020, not age-related
  - Due to position beneath tie rod, unable to measure the spalling but no rebar is showing, so it was dispositioned as acceptable

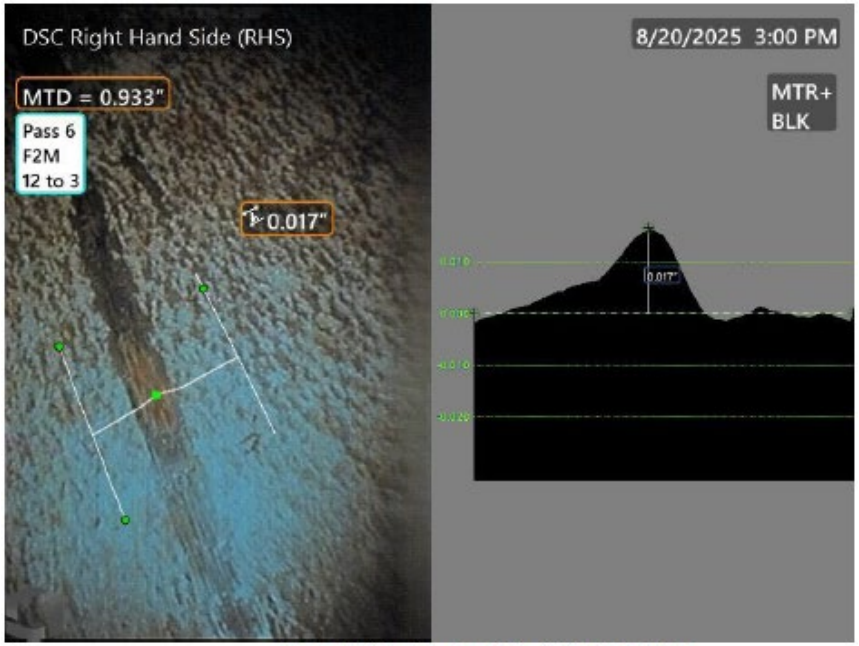
AMP Inspection - Appendix

HSM_Floor015.JPG

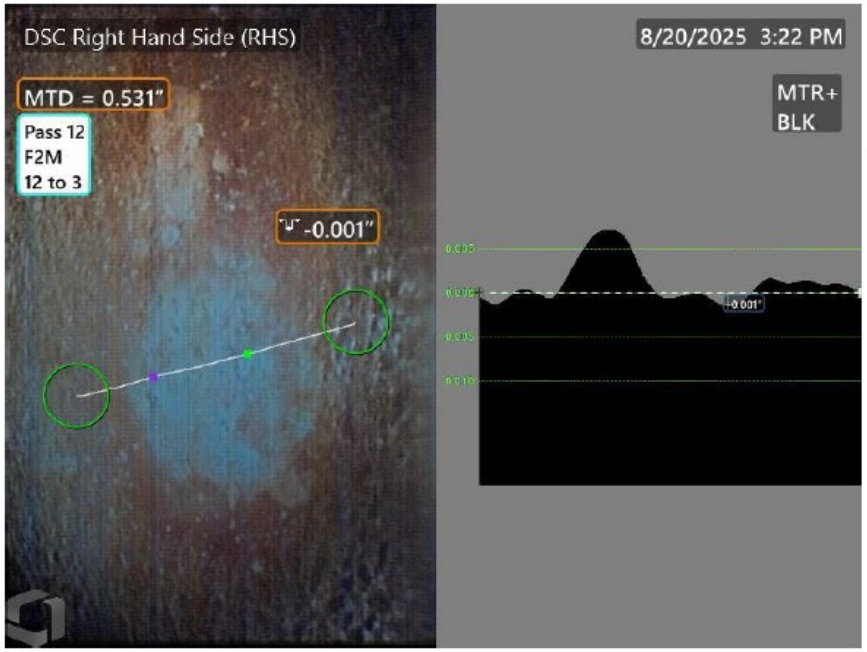
Label1	HSM
Label2	Floor
Comments	Debris

- Debris in corner of HSM
  - Causes no issue, believed to be dust/dirt/dead mayflies

AMP Inspection - Appendix



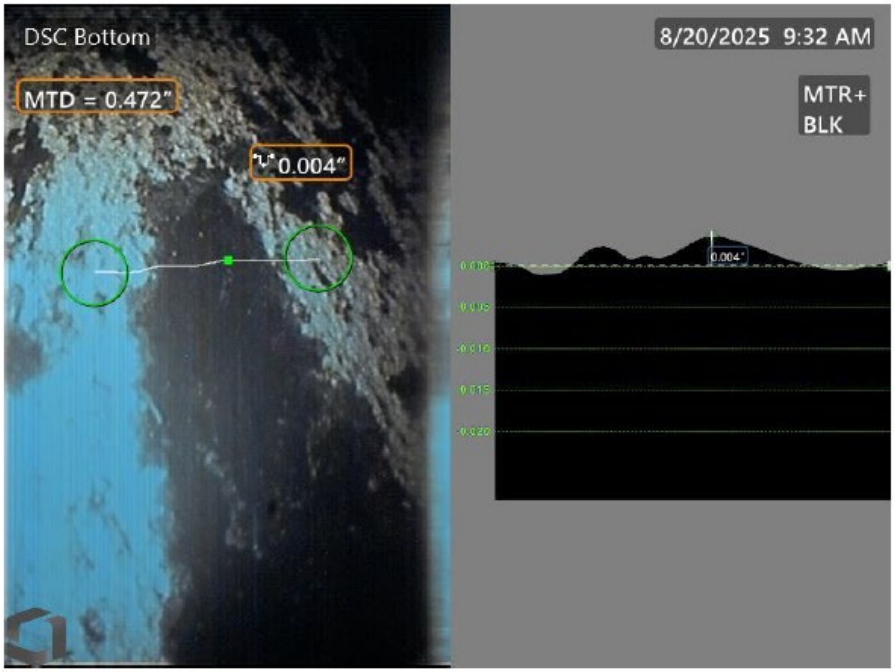
DSC_Right_Hand_Side_(RHS)004.JPG



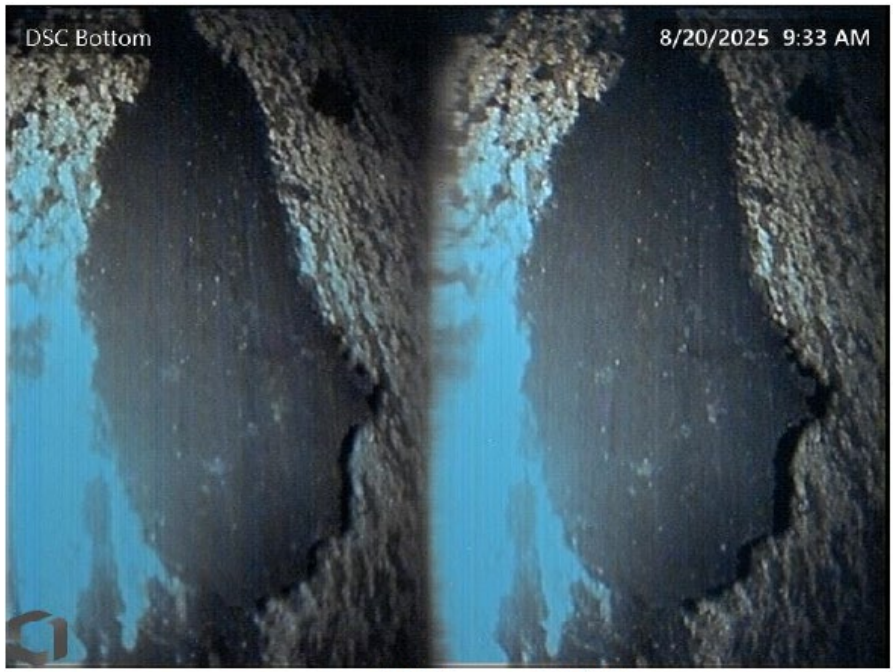
DSC Right Hand Side (RHS)006.JPG

- Scratches and discoloration
  - No loss of material from either one

AMP Inspection - Appendix



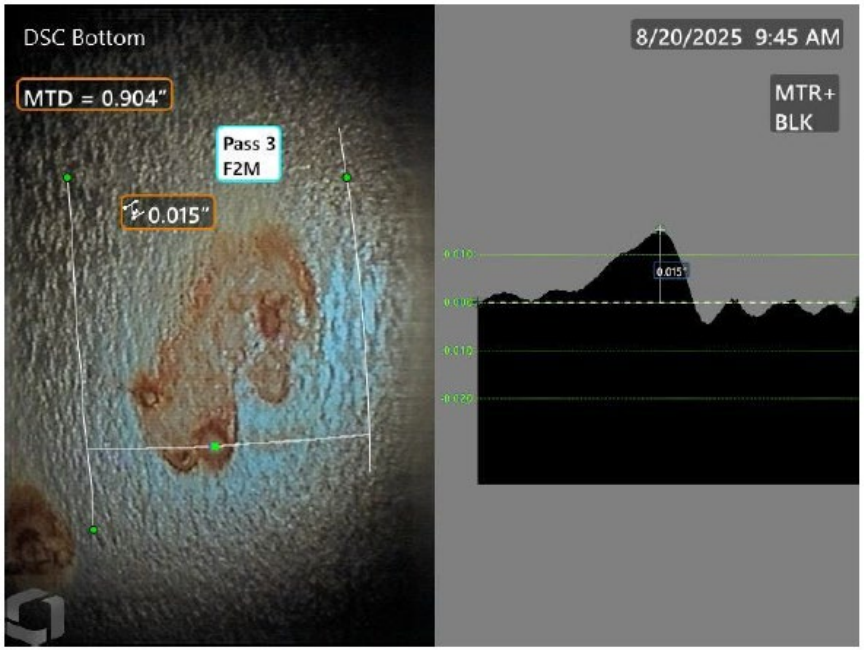
DSC_Bottom001.JPG



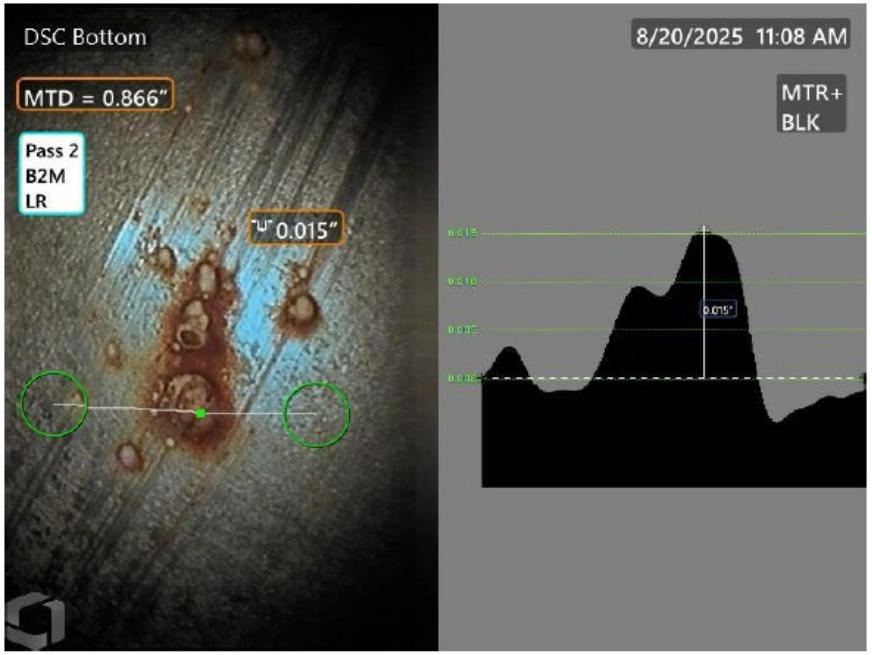
DSC_Bottom003.JPG

- Discoloration (deposit) on bottom of DSC
  - Dispositioned as not exceeding acceptance criteria and not a threat to DSC surface integrity

AMP Inspection - Appendix



DSC_Bottom006.JPG



DSC_Bottom010.JPG

- Discoloration (deposits)
  - Dispositioned as no loss of material, no threat to DSC integrity, and did not exceed acceptance criteria

The End

Questions?

NAC INTERNATIONAL | Kanadevia Group

Results from the Palo Verde and Connecticut Yankee Aging Management Inspections

Presented by Dan Hogan, Project Engineer, NAC International

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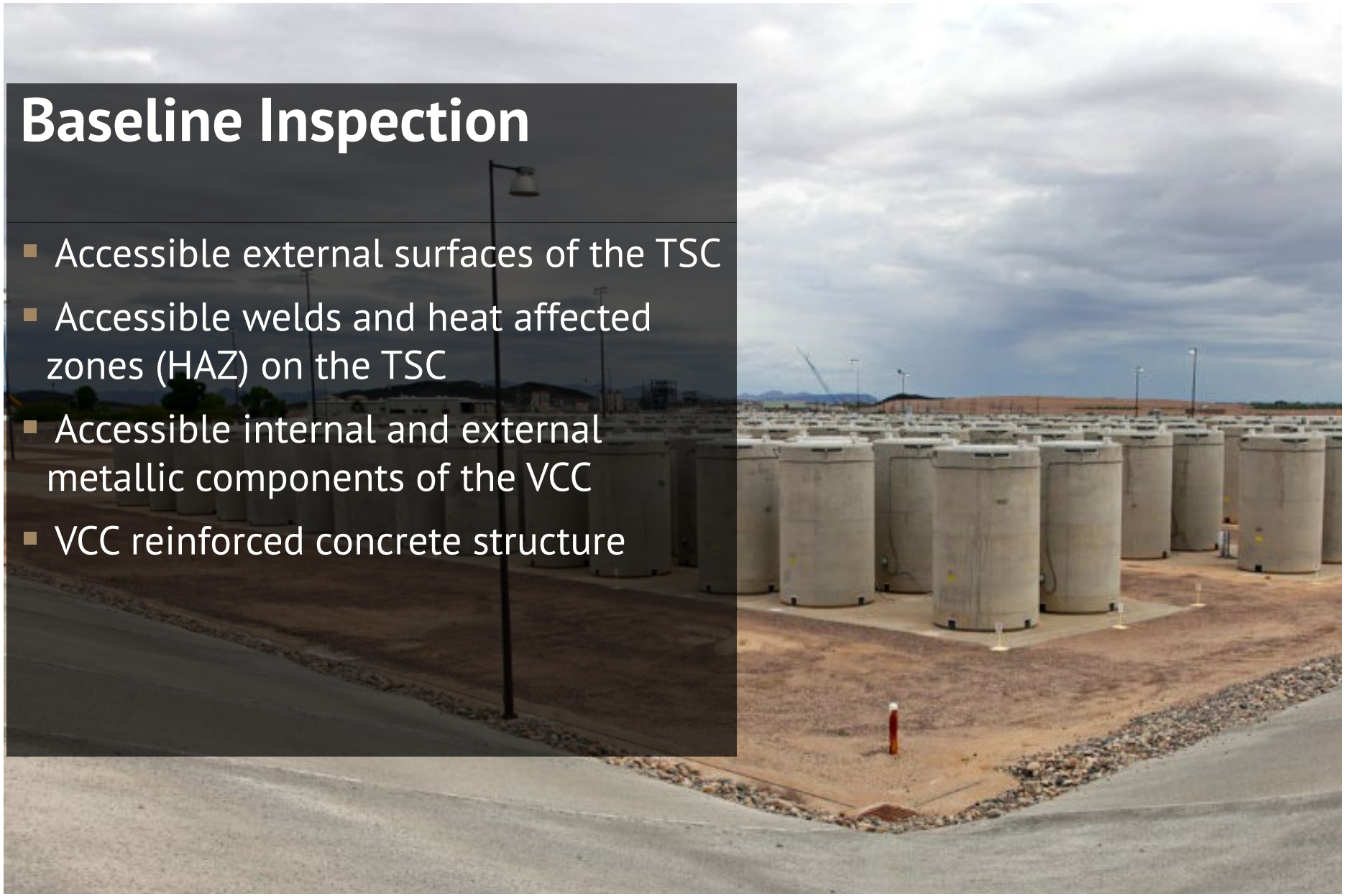
Baseline Inspection

- Accessible external surfaces of the TSC
- Accessible welds and heat affected zones (HAZ) on the TSC
- Accessible internal and external metallic components of the VCC
- VCC reinforced concrete structure



Inspection methods

- VT-3 and VT-1 visual inspection via robotic crawler
- Qualified Inspectors
- Qualified video equipment
- Procedures
- Dust sampling

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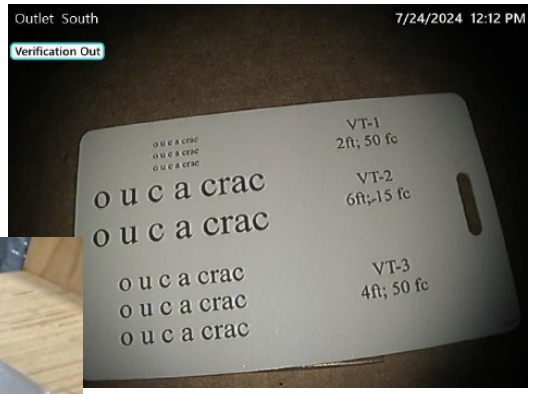
Inspection methods (cont'd)

- Robot Crawler
- Visual Exam Resolution Cards
  - EPRI cards per ASME Section XI, Table IWA-2210-1
- Camera Equipment
  - GE Mentor Visual iQ system with 84100-Quartz Fiber Probe
- Light meter
  - Spectroline XDS-1000 (calibrated)
- Inspection Team
  - Robotic Technologies of Tennessee – robot crawler and operators
  - Integrity Assessment Group – NDE Inspectors (VT-3)
  - Applied Technical Services – NDE Inspector (VT-1)
  - NAC – oversight, quality assurance, engineering



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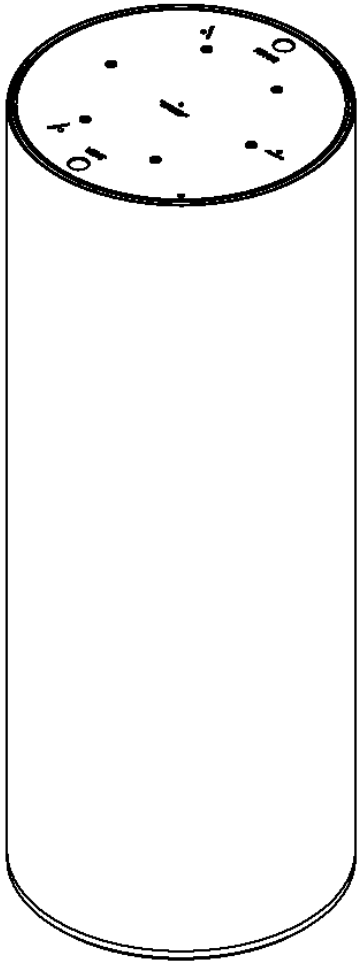
TSC Inspection Criteria

- TSC (Containment Boundary)
  - *No evidence of cracking of any size*
  - *No evidence of general corrosion or pitting corrosion resulting in an obvious loss of base metal*
  - *No corrosion products having a linear or branching appearance*



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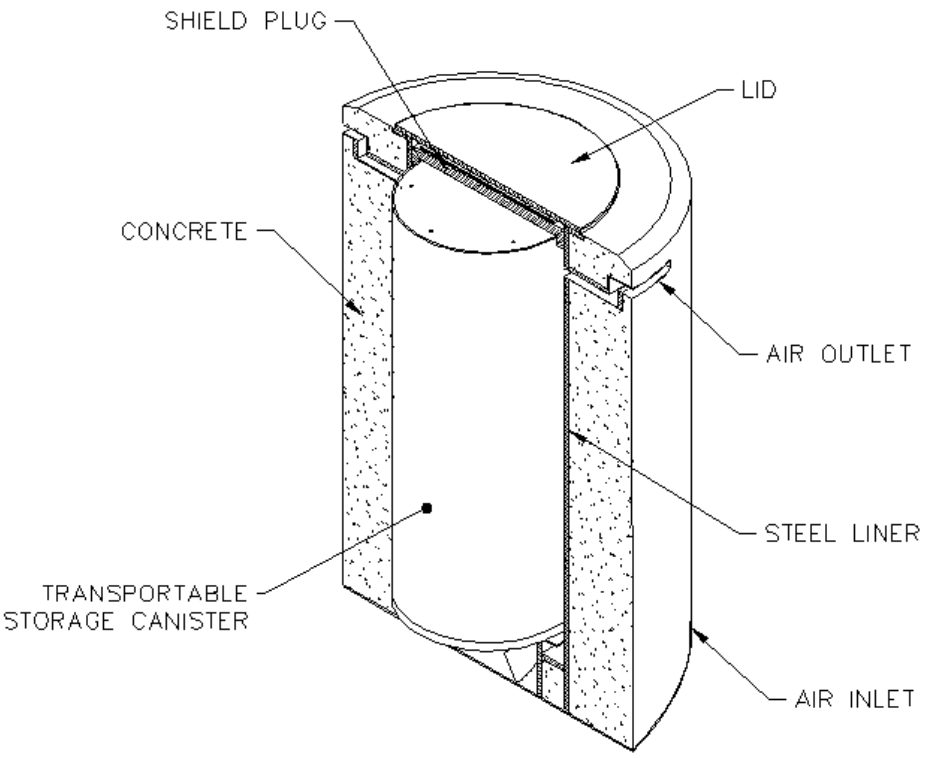
VCC Inspection Criteria

- VCC (metallic components)
  - *Active corrosion resulting in obvious loss of base metal*
  - *Indication of displaced or degraded components*
  - *Damaged bolts or bolt holes*
- VCC (concrete structure)
  - *Pop outs and voids*
  - *Scaling, Spalling*
  - *Cracks exceeding Tier 2 criteria (ACI 349.3R-02)*



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Palo Verde Generating Station (PVGS) Results

- NAC-UMS storage system
- Performed in February 2025
- Initial Heat Load of 7.59 kW, in service for 21 years at time of inspection
- Inspection found the VCC and TSC in very good condition with no signs of aging related degradation
- CoC renewed in July 2024 for an additional 40 years

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PVGS Inspection Summary - Photos

TSC – General Condition

TSC Shell to Bottom Plate circumferential weld and HAZ

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PVGS Inspection Summary – Photos

TSC Shell circumferential weld and HAZ

TSC Shell vertical weld and HAZ

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PVGS Inspection Summary - Photos

TSC Shell HAZ at closure lid weld

TSC closure lid weld and HAZ

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PVGS Inspection Summary - Photos

VCC outlet vent

VCC inlet vent
(image is inverted)

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PVGS Inspection Summary - Photos

VCC pedestal
(image is inverted)

VCC liner

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PVGS Inspection Summary - Photos

VCC lid

VCC outlet vent (external)

PVGS performed the VCC reinforced concrete structure inspection. There are no photos of it as it had no degradation requiring monitoring (i.e., nothing exceeding ACI 349.R3 tier 2 criteria).

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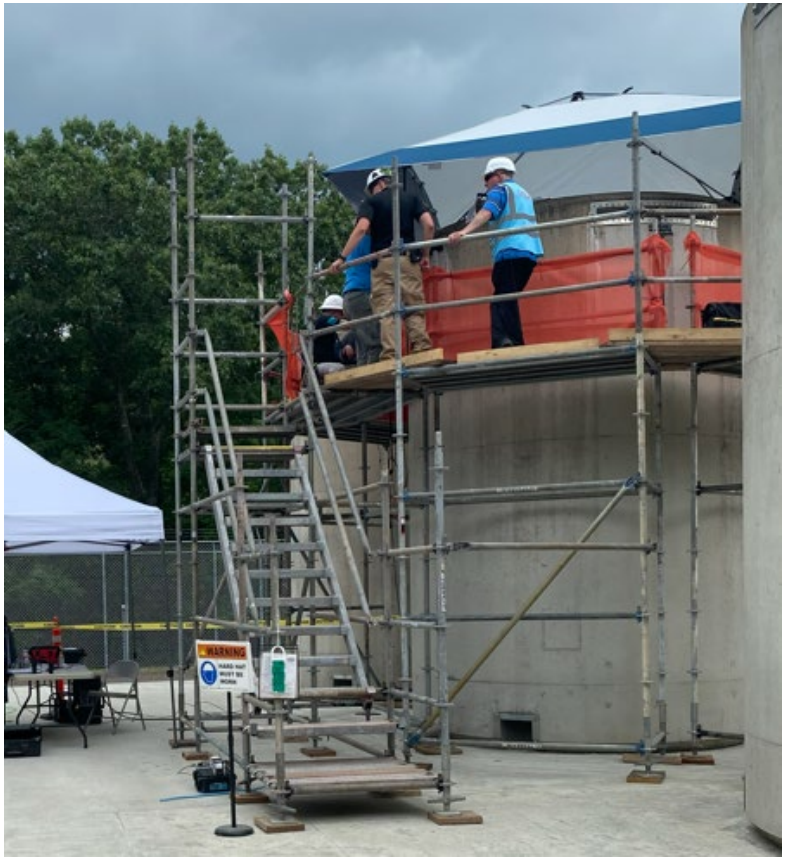
Connecticut Yankee (CY) Results



- NAC-MPC storage system
- Performed in July 2024
- Initial Heat Load of 6.13 kW, in service for 19 years at time of inspection
- Inspection found the VCC and TSC in very good condition with no signs of aging related degradation
- CoC renewed in October 2023 for an additional 40 years

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CY Inspection Summary - Photos

TSC general condition

TSC Shell to Bottom Plate circumferential weld and HAZ

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CY Inspection Summary - Photos

TSC Shell vertical weld and HAZ

TSC Shell vertical weld and bottom circumferential weld intersection

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CY Inspection Summary - Photos

TSC Shell HAZ at closure lid

Outlet__East_025.JPG

TSC Shell HAZ at Closure Lid (water streaking, deposits)

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CY Inspection Summary - Photos

TSC Closure Lid weld

VCC outlet vent

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CY Inspection Summary - Photos

VCC Inlet Vent
(image is inverted)

VCC Pedestal and Inlet Vent
(image is inverted)

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CY Inspection Summary - Photos

VCC Liner

VCC Liner

- CY performed the VCC external metallic component inspection, and noted the presence of localized coating damage but no obvious loss of base metal. All components are in satisfactory condition, and a work order was initiated for coating repair.
- CY also performed the VCC reinforced concrete structure inspection, and noted presence of tier 2 cracking coming from the bolt holes at outlet vents. These cracks are acceptable with no action required.

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Lessons Learned

- Water streaking/staining was observed during the CY screening examination (VT-3)
- Water streaks were brownish in color, and were present in the HAZ but no evidence of corrosion in the area
- Dust samples indicated an absence of chlorides
- By the AMP-1 definition this was conservatively considered a "minor indication of corrosion"
  - *"…evidence of water intrusion stained the color of corrosion products"*
- This is an undefined visual anomaly per ASME CC N-860

Outlet__East_025.JPG

TSC Shell HAZ at Closure Lid (water stains and deposits)

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Lessons Learned

- Figure 1 of CC N-860 was utilized to arrive at the conclusion that current inspection intervals and population should be maintained.
  - *Visual anomaly not defined in CC N-860*
  - *Not 3rd successive consistent inspection*
- However, AMP-1 requires an assessment examination (meeting the requirements of VT-1) to be performed since the water streaking was present in the HAZ.
  - *"…required if the screening examination identifies <u>any</u> visual anomaly that is not consistent with prior results or is identified for the first time"*
- CC N-860 Section -2300 (VT-1) only defines visual anomalies related to degradation.
  - *Surface discontinuities*
  - *Corrosion or corrosion products*
  - *Flaws (e.g., cracking, pitting)*
  - *Loose, missing, damaged systems, structures, or components*
- The discrepancy between the AMP-1 requirements for a visual anomaly and CC N-860 visual anomaly definitions and VT-1 inspections caused confusion.
  - *VT-1 implies there is at least some level of corrosion or corrosion products, which in this case there were none. Note that <u>any</u> corrosion in a weld region invokes Supplemental Examination.*
  - *CC N-860 does not give a clear "off ramp" for visual anomalies that are not corrosion based.*
- The combination of no observed corrosion, in conjunction with the absence of chlorides in the dust samples nullifies the AMP-1 Section 6.4 supplemental examination requirements which are invoked by "minor indications of corrosion". The licensee is not required to implement more frequent inspection intervals.

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Lessons Learned

- CY personnel performed a re-examination of the area in June 2025. The area had not changed in appearance.
- Four dust samples were taken from the water streak area using a long reach tool and gauze pads soaked in demineralized water.
- Upon collecting the fourth dust sample, it was observed that the water streaking had been removed.
- Key laboratory analysis results of the dust samples:
  - *Undetectable levels of salts (chlorides)*
  - *No particles consistent with corrosion*

TSC Shell HAZ at Closure Lid
(appearance at re-examination)

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Lessons Learned

- A memorandum was written to supplement the inspection report. The memo concluded that the minor indication does not adversely affect the TSC and can be downgraded to surface contamination. It also noted that the water intrusion does not affect the function of the VCC or the TSC.
- Future inspections should consider having tools available during the inspection that can reach the upper portion of the TSC shell diameter. The tools would be used for collecting additional dust/contaminant samples if desired, and potentially to remove surface contamination.
- Removal of surface contamination in conjunction with laboratory analysis of the contaminants can effectively mitigate concerns of the contaminants being corrosion products.

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Questions?

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Robotic Guided Wave Scanning Technologies for Spent Fuel Dry Storage Canister Evaluation

EPRI | ELECTRIC POWER RESEARCH INSTITUTE

EPRI Extended Storage Collaboration Program (ESCP) Meeting
27 Oct 2025

Presented by:
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GUIDEDWAVE

RTT
Robotic Technologies of Tennessee LLC

PennState

OUR TEAM

DR. CODY BORIGO
VP & Dir. of Engineering

- Leading experts in guided wave and AE technologies with extensive real-world experience
- 20+ years of R&D and product development
- Custom solutions for most challenging NDE problems with a focus on bringing new technology to the field
- Our business is built on technology licensing and inspection services partnerships

DR. JAMIE BEARD
President

- Robotics company focused on inspection and manufacturing
- Extensive experience with DSC inspections and robotic sensor deployment
- Have inspected 44 vertical and 16 horizontal storage canisters to-date at more than two dozen facilities
- Partnered with Guidedwave on SBIR program to develop DSC inspection technology

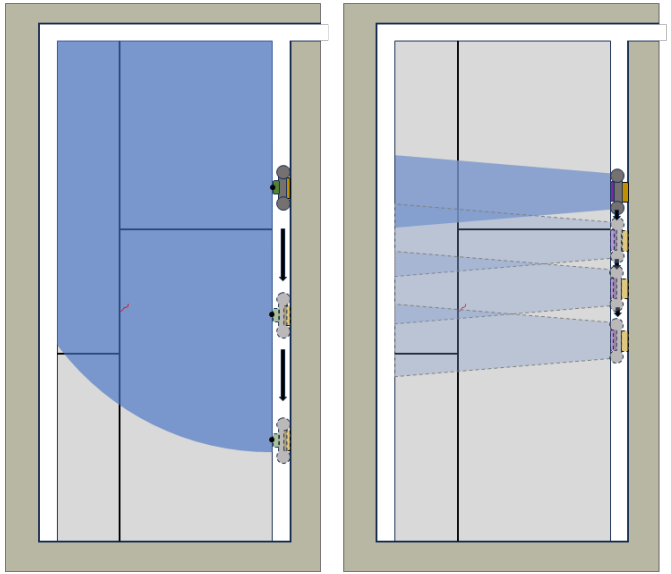
PROF. CLIFF LISSENDEN
Dept. of Engineering Science & Mechanics

- 30 years of NDE and related engineering experience
- Knowledgeable in the requirements and challenges for this application
- Developed EMAT technologies for spent nuclear fuel canister inspection
- Extensive history of working with GW

ROBOTICALLY DELIVERED GW SENSOR TECH

OBJECTIVE: Provide a practical field-ready inspection technology for detecting and locating CI-SCC in spent fuel dry storage canisters.

APPROACH: Utilize proven guided wave scanning technologies that can cover large areas of a canister from a limited access regions (no need to scan directly adjacent to the CI-SCC).

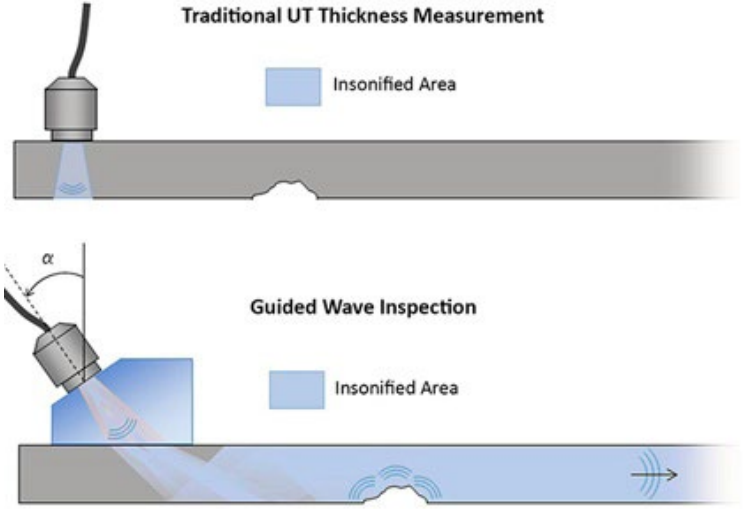


GWPA DCMS

- Guided wave sensors are capable of remote flaw detection, which is advantageous for DSC CI-SCC inspection
- Based on GW's extensive experience with guided wave testing, two potential technologies were selected as candidates
- GWPA and DCMS use SH guided waves in the 100-150 kHz range and have been successfully used in many applications
- Facilitating robotic deployment and coupling of these sensors is the primary challenge for this application, followed by optimizing the sensors and data analysis processes

GUIDED WAVE ULTRASOUND

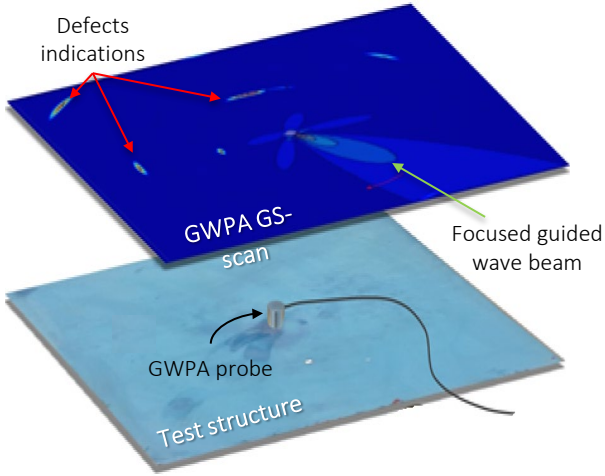
Guided wave ultrasound uses lower-frequency ultrasonic waves (typically 20-200 kHz) to screen large or inaccessible areas for defects.

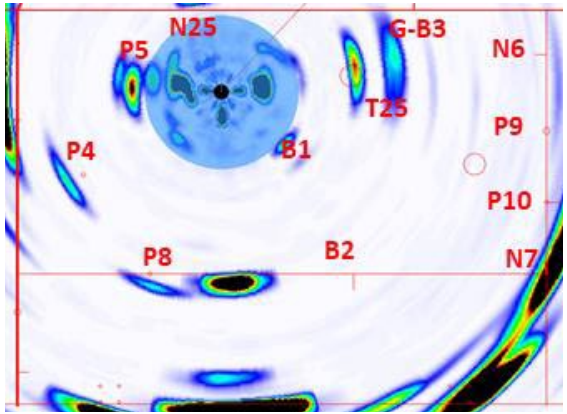


- Detect corrosion, cracking, and other damage mechanisms
- Inspect inaccessible regions or screen large areas
- Medium- to long-range inspection beyond sensor location
- Focusing and beam steering capable
- Can be robotically integrated for scanning
- Various types of sensors and inspection methods possible
- Can be permanently-installed for long-term monitoring

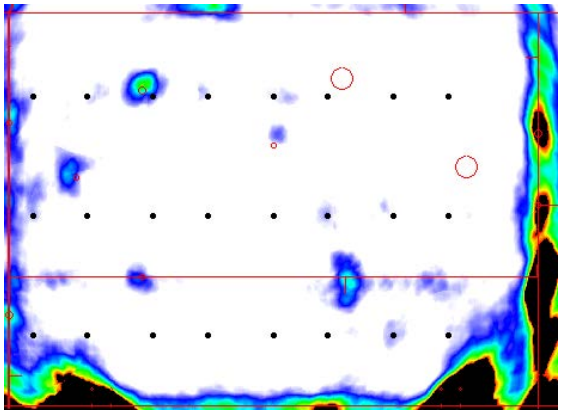
GUIDED WAVE PHASED ARRAY (GWPA) TECHNOLOGY

- The GWPA technology electronically steers a beam of ultrasonic guided wave energy around the transducer to generate a “radar-like” scan of a large area of a plate.
- GWPA is offers the benefits of LRUT/guided wave pipe inspection for plates.
- Efficient, large-area screening for corrosion and cracking under insulation, soil, or other obstructions with limited access requirements and minimal surface preparation.



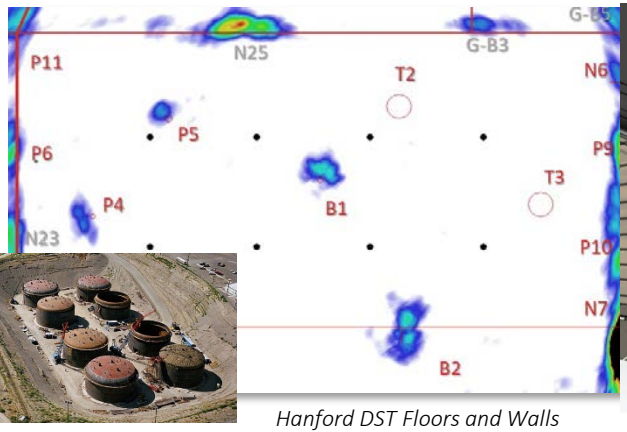


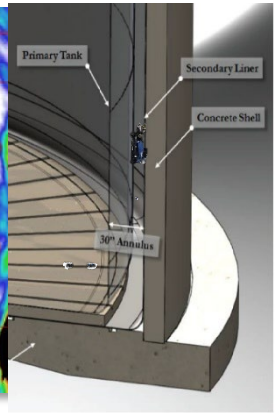
Individual GS-scan



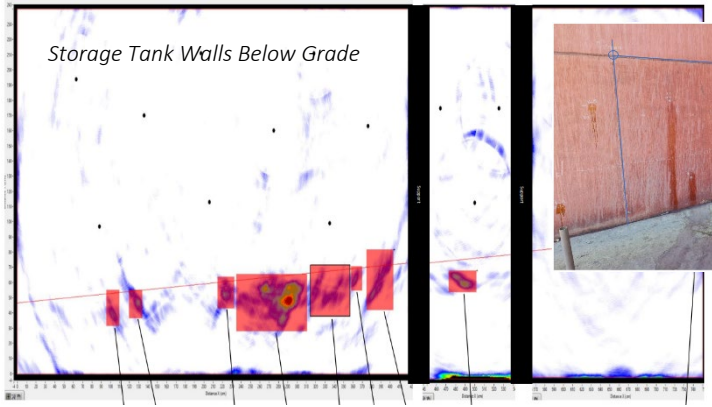
Multiple GS-scans compiled into a composite image

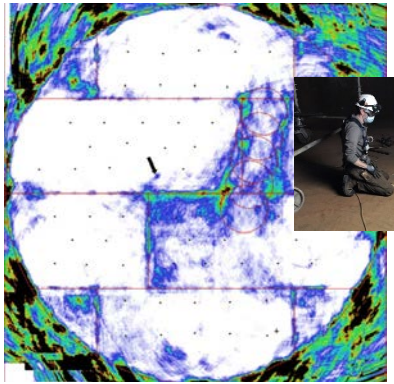
SUCCESSFUL GWPA APPLICATIONS

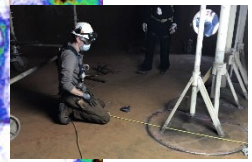




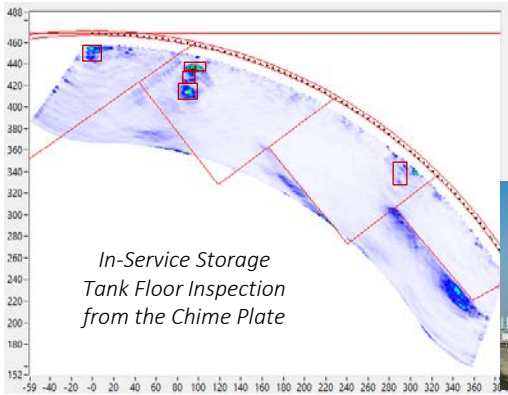
Hanford DST Floors and Walls



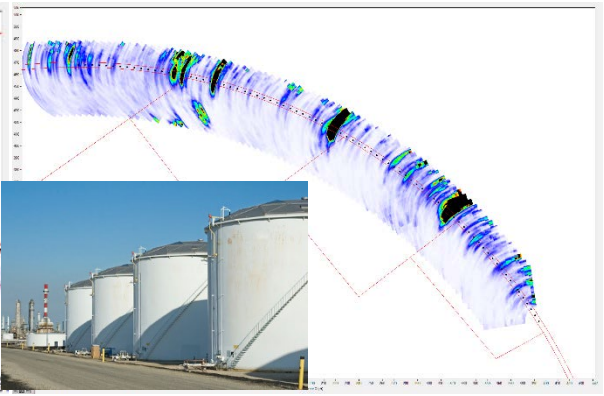




Tank Floor Screening

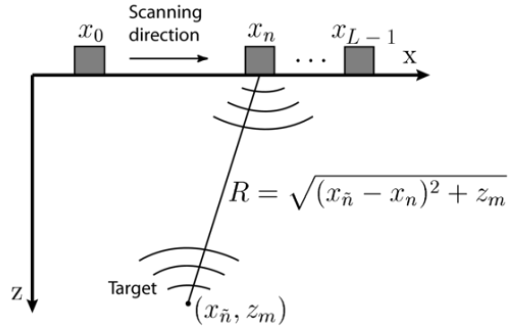


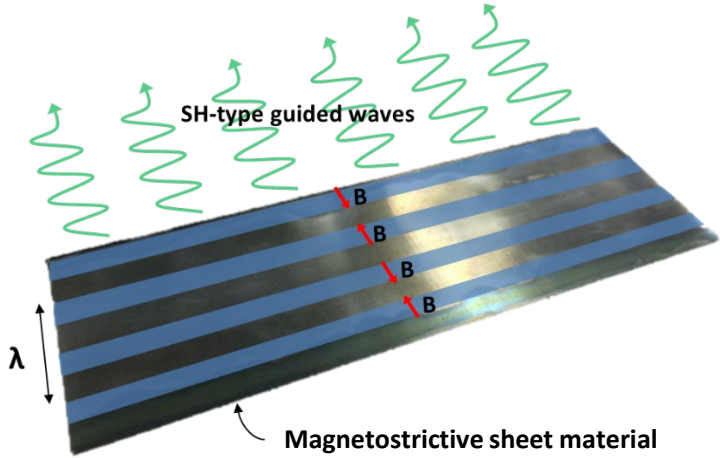
In-Service Storage Tank Floor Inspection from the Chime Plate

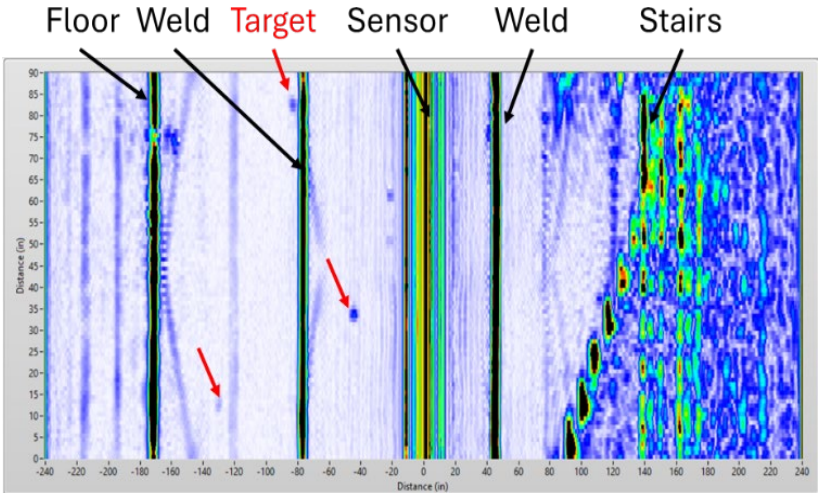


DRY-COUPLED MAGNETOSTRICTIVE (DCMS) TECHNOLOGY

- Dry-coupled magnetostrictive scanning (DCMS) technology uses an electromagnetic-acoustic transducer with high-efficiency MS material
- Long-range guided wave beam is emitted perpendicular to (or at an oblique angle to) the scan direction
- Scan data is collected in approximately ½" increments and then recombined using Synthetic Aperture Focusing Technique (SAFT)
- SAFT is used in ultrasonic and radar scanning
- Generates a focused 2D image of a large area from a single line scan

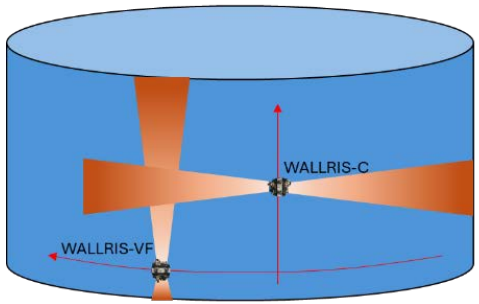


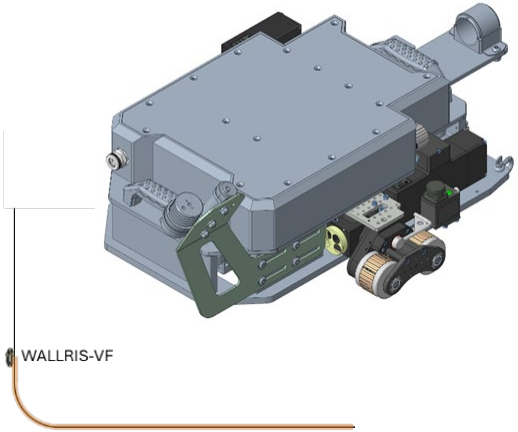




DCMS - HANFORD DST EXPERIENCE

- Adapted DCMS technology for Hanford DST wall inspections
- Dry-coupled magnetostrictive scanning technology with SAFT focusing
- Long-range guided wave flaw detection and large-area coverage
- These scans can reach inaccessible regions of the tank walls and provide 100% screening in an order of magnitude less time than conventional UT scanning
- Successful test deployment in Hanford Tank Farm in Aug 2025

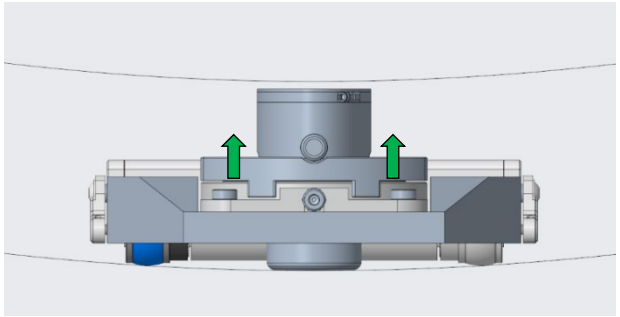


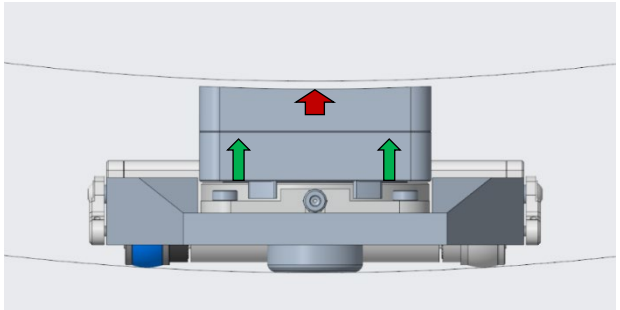


DOE PHASE I SBIR – GW INSPECTION OF SNF CANISTERS

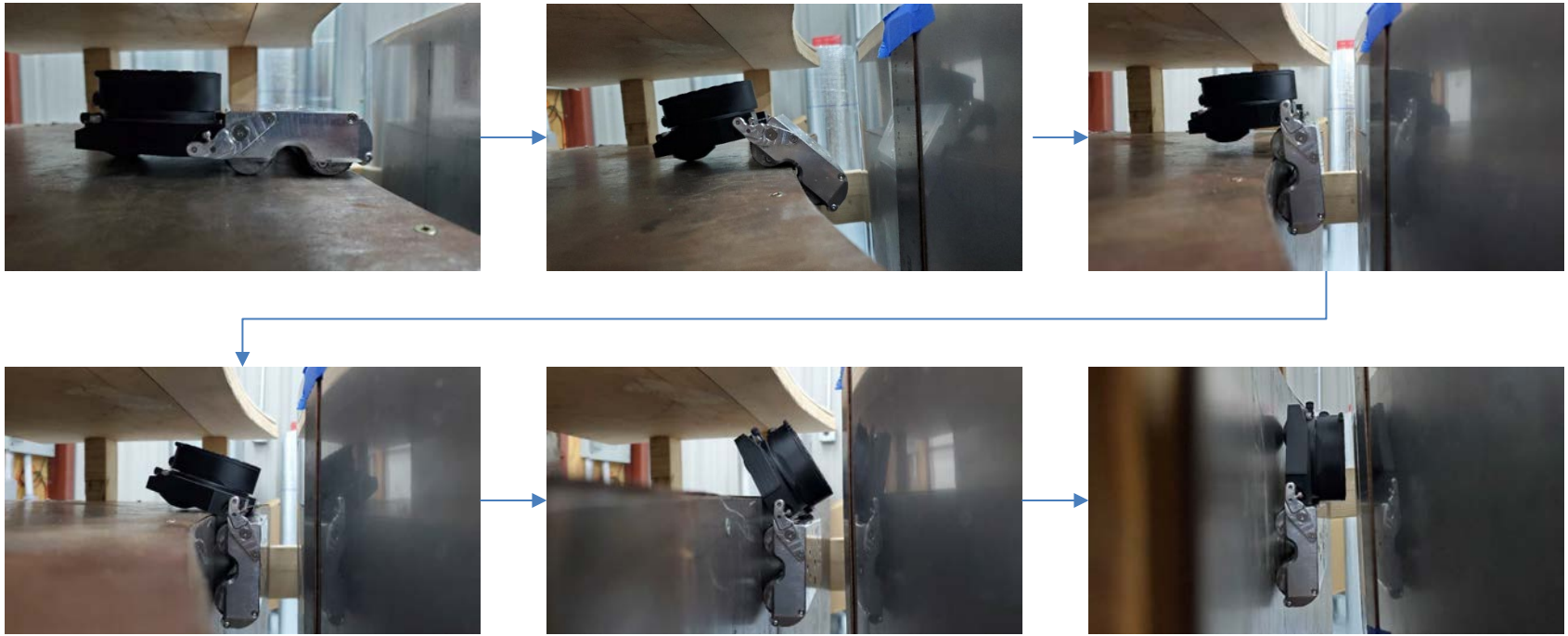
GW recently completed a Phase I DOE SBIR project (Aug ‘24 – Jul ‘25) to evaluate the **feasibility of using several of our guided wave inspection technologies for minimally-invasive robotic scanning of DSCs**

1. Collaborated with RTT to develop feasibility prototype robotically deployed GW sensors
2. Demonstrated GWPA and DCMS sensor deployment on a mockup
3. Performed flaw detection study online stainless-steel mockup with machined flaws
4. Collaborated with EPRI and UNCC to collect GW and AE data on a DCS canister



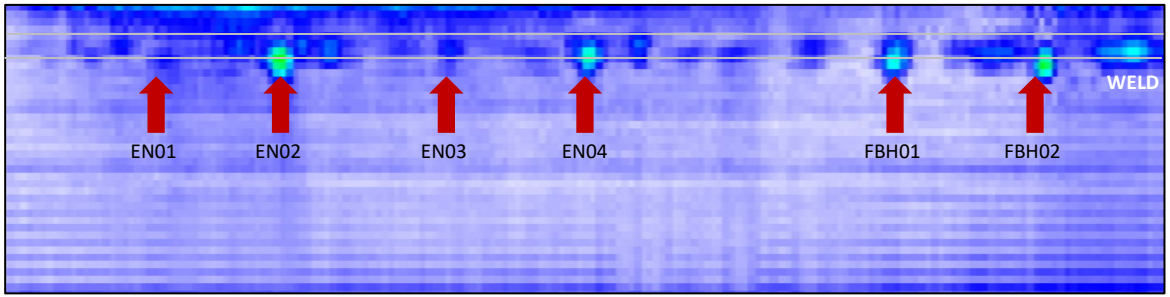


DOE PHASE I SBIR – GW INSPECTION OF SNF CANISTERS

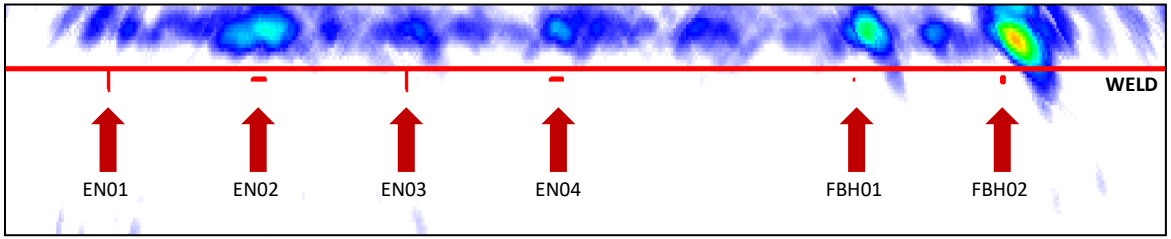


Successful Air Vent Deployment Test with RTT Robot and GW Sensor

GUIDED WAVE FLAW SENSITIVITY EVALUATION



DCMS flaw sensitivity scan on GW's stainless-steel mockup



GWPA flaw sensitivity scan on GW's stainless-steel mockup

*Additional indications in weld generally correlate with internal weld flaws as assessed with PAUT

Flaw ID	Type	Width (in)	Depth (in)	X (in)	Y (in)	Angle (deg)
EN01	EN	1.00	0.25	48	24	90
EN02	EN	1.00	0.25	48	36	0
EN03	EN	0.42	0.20	48	48	90
EN04	EN	0.42	0.20	48	60	0
FBH01	FBH	0.125	0.25	48	84	N/A
FBH02	FBH	0.25	0.25	48	96	N/A

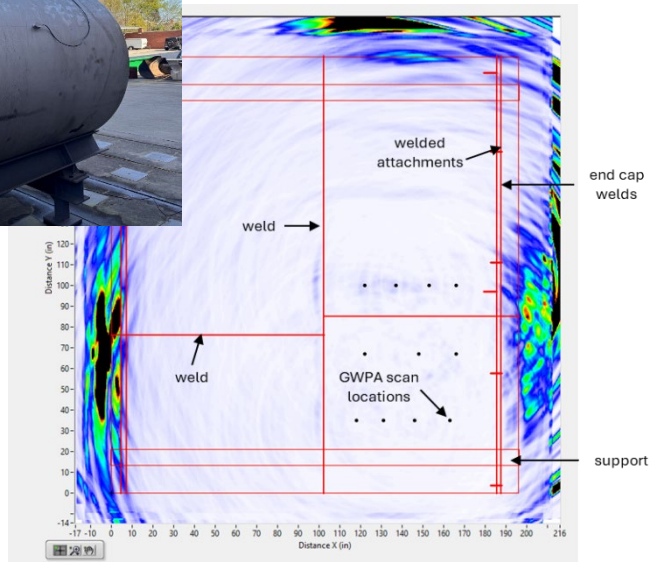
EN = Elliptical Notch **FBH = Flat-Bottom Hole**

TASK 3: ROBOTIC PROTOTYPE – MOCKUP DEPLOYMENT

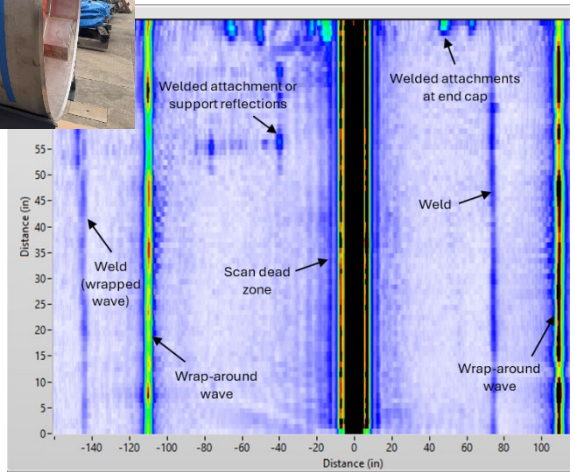
Successful Annulus Deployment and Coupling Test with RTT Robot (and DCMS Sensor)

GW SCANNING FEASIBILITY TESTING ON DSC

- Collaborated with EPRI and UNCC to perform GW testing on full-scale DSC in April 2025
- Performed manual guided wave scans (GWPA and DCMS)
- Demonstrated that signal quality is high, attenuation is low, and both sensor types are compatible



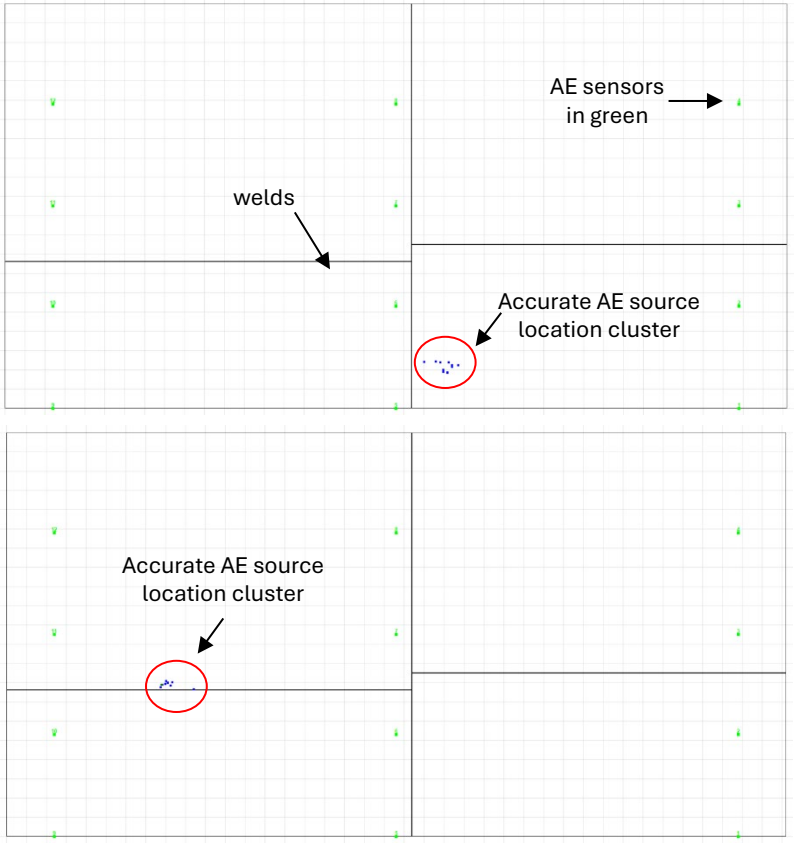
GWPA



DCMS

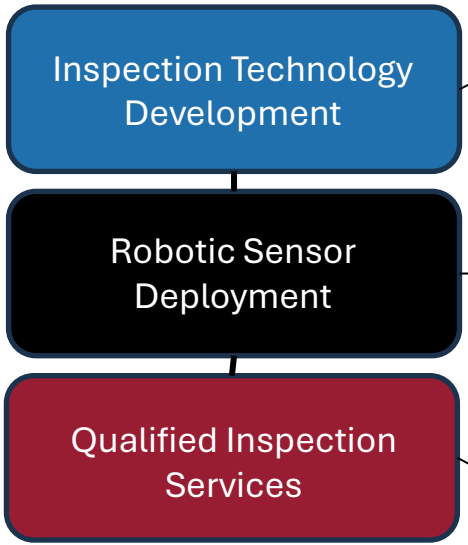
AE FEASIBILITY TESTING ON DSC

- Collaborated with EPRI and UNCC to perform AE testing on full-scale DSC in April 2025
- Performed AE source location and attenuation tests
- Demonstrated that AE signal quality and range is excellent – as few as 4-6 sensors could be sufficient



PRACTICAL TECHNOLOGY IMPLEMENTATION STRATEGY

Successful implementation will require far more than technology development and demonstration. Our team has been assembled to support the full technology implementation cycle, from development to long-term inspection services.



GUIDEDWAVE

- Team lead
- Sensor/software development
- Inspection services support

RTT
Robotic Technologies of Tennessee LLC

- Robotic system lead
- Experienced in DSC inspections

Structural Integrity

- Inspection services lead
- Nuclear NDE program
- Qualified personnel and scalability
- Structural health monitoring infrastructure

SUMMARY OF PROGRESS

OBJECTIVE: Provide a practical field-ready inspection technology for detecting and locating CI-SCC in spent fuel dry storage canisters.

APPROACH: Utilize proven guided wave scanning technologies that can cover large areas of a canister from a limited access regions (no need to scan directly adjacent to the CI-SCC).

PROGRESS: Phase I R&D was funded through a DOE SBIR – hoping to continue this work with a Phase II award in 2026

- Collaborated with RTT to integrate two GW scanning technologies on field-proven crawlers - PROTOTYPES
- Collected scans on small-scale mockups with manufactured flaws
- Collected manual GW scan data on a full-scale empty canister owned by EPRI at UNC Charlotte
- Established partnership with Structural Integrity to provide GW inspection services

NEXT STEPS (TIMELINE ASSUMES DEVELOPMENT FUNDING PROVIDED):

- Mature the robotics scanning technologies to produce field-ready prototypes (2026)
- Fabricate CI-SCC mockups for high-fidelity flaw sensitivity evaluation and sensor optimization (2026)
- Conduct full-scale system prototype testing and refine system and procedure (2027)
- Conduct first qualified field inspections (2028)

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IAG/RTT ECA Test Results Updates

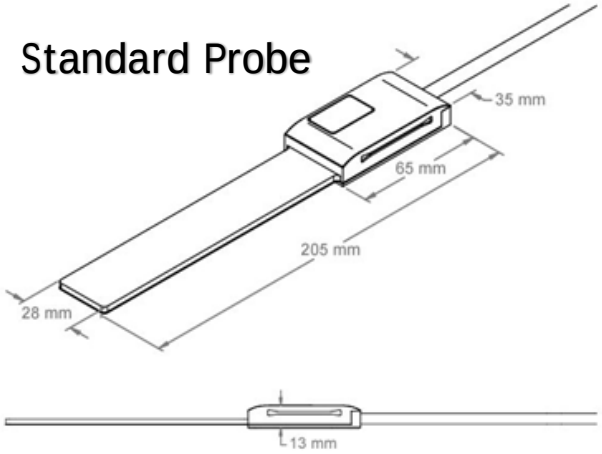
Updates in Equipment, Techniques, & Improvements

ECA Probe Selection

- OEM Tested ALL Versions of ECA Probe Coil Matrix Configurations on the EPRI Calibration Plate in Spring of 2024
- Customized “Off The Shelf” Medium I-Flex ECA Probe Selected based on performance
- Customization of cable length, and casing exit orientation achieved

Body	I-Flex
Casing	Medium
Topologies	Impedance Long, single driver Short, double driver
Cable	3 m (9.8 ft)
Coverage	56 mm (2.21 in)
Central frequency	250 kHz
Frequency range	50–525 kHz
Coils (diameter × number)	3.5 mm × 48 (0.14 in × 48)
Channels (according to topology)	32, 59, 60
Minimum channel requirement	32 or 64
Penetration (stainless steel/aluminum)	Up to 2 mm (0.079 in)
Minimum detectable crack length	1 mm (0.039 in)

Standard Probe



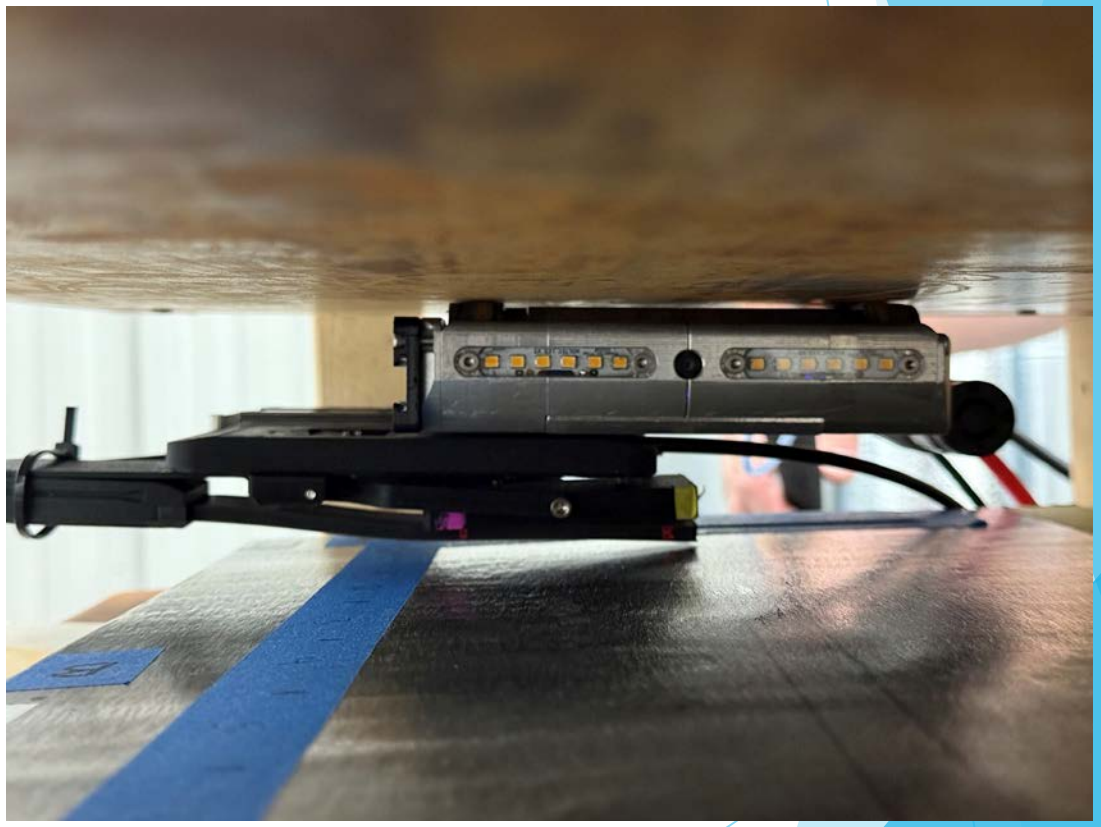
Custom Probe

ECA Probe Selection Advantages

- Medium I-Flex Probe is flexible, yet rugged enough for test conditions
- Medium I-Flex Probe coil matrix can be expanded to increase footprint
- Ability to Scan with all four (4) topologies active
- Proven capability to detect all design flaws, in all orientations
- Coil diameter and frequency combination can resolve target design flaws
  - Smallest Design Flaws: EDM Notches 3.175mm L x 0.127mm W x 1.24mm D
  - EDM Notches oriented Axially, Transverse, Skewed at 45°
- Capable of integration with robotic crawlers, and external encoders

Integrated ECA Technology Capabilities

- Integrated with RTT Robotics
- Calibration Needs Identified
- Work Procedure Framework Built
- Sizing Curve Process Validated
- Analysis Process Created
- Blind Testing Completed July 2025
- Results to be issued fall 2025



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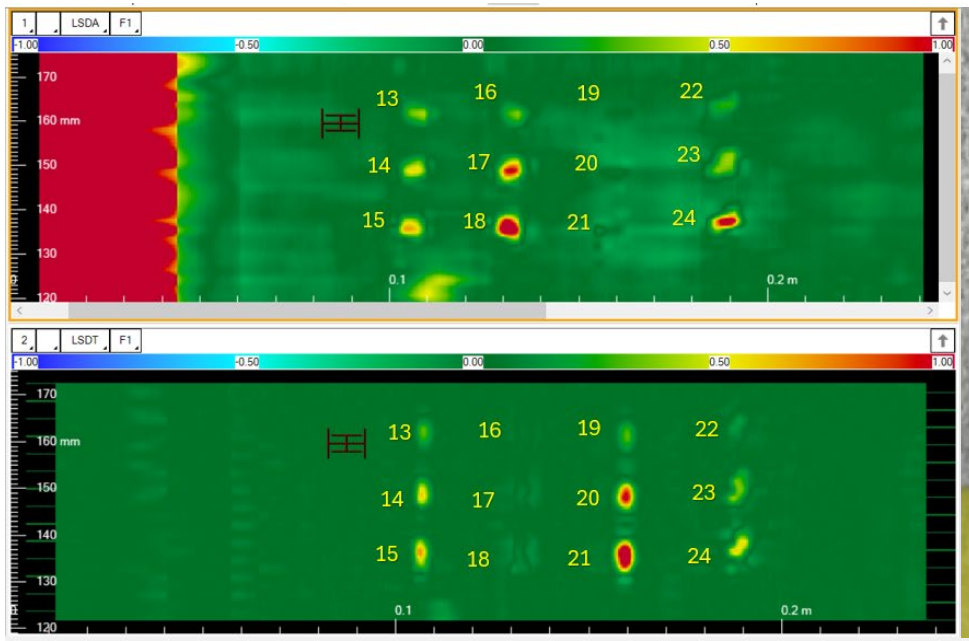
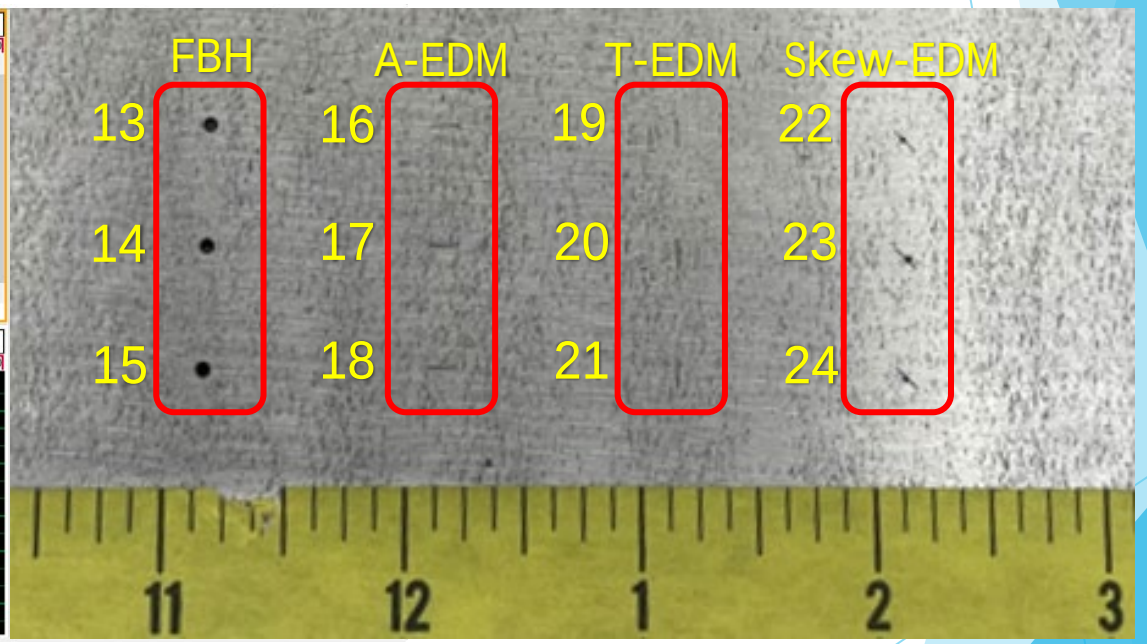
Robotic ECA Calibration Development

- Eddy Current Array (ECA) Calibration Scan Example





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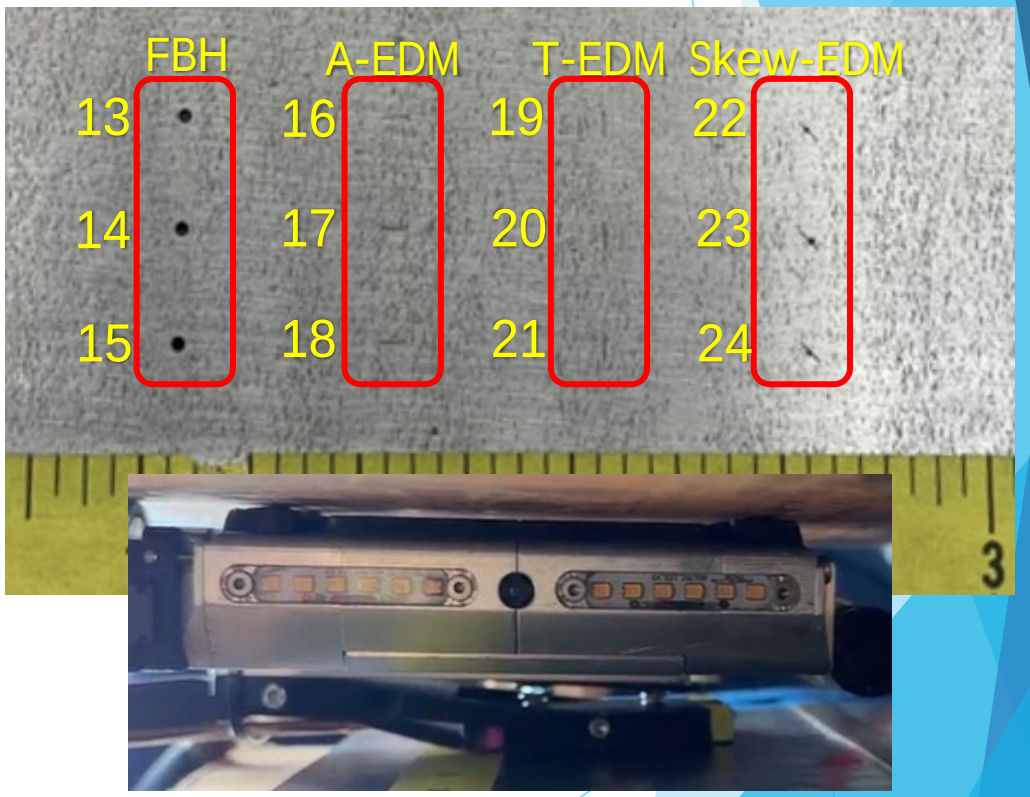
Robotic ECA Sizing Capabilities

- Sizing Curves Built Using Design Flaws
- Ensures High Detection Confidence
- Confirms Scan Speeds
- Resulted in 100% Detection in all Tests

Lessons Learned for Future

- Limited Depth Curve in Design Flaws
- Build Calibration Block for Field Tests
- Representative Flaw Range is Best



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Conforming to Weld Cap Contours

- Redesigned Probe Backing to Match Contour of Probe on Cap
- Down Pressure by Robotically Mounted Probe
- Delivered Repeatable Results
- Axial Scanning Only!

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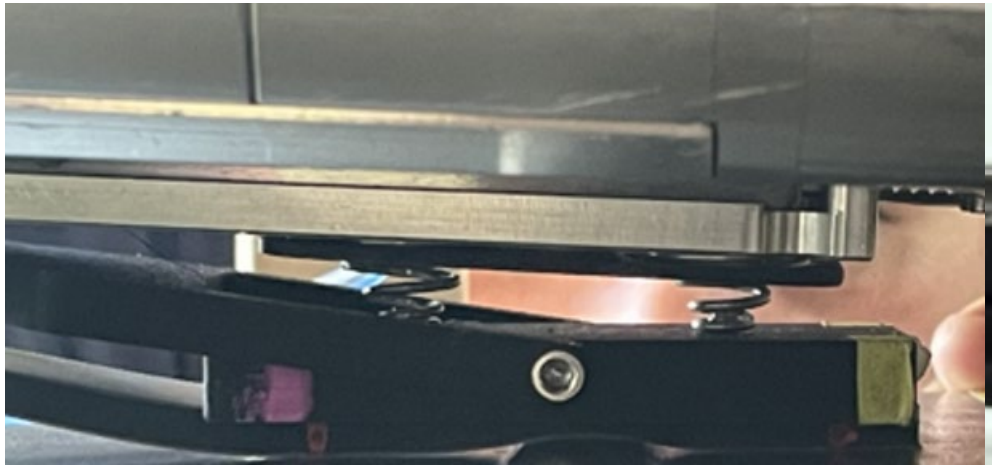
Spring Mount to Fully Pneumatic Mount

- Spring Mounted Probe had Limited Nulling Lift Off
- Pneumatic Probe Mount Enabled Full Nulling Range
  - Helps Limit Wear on Probe Face When Not Scanning



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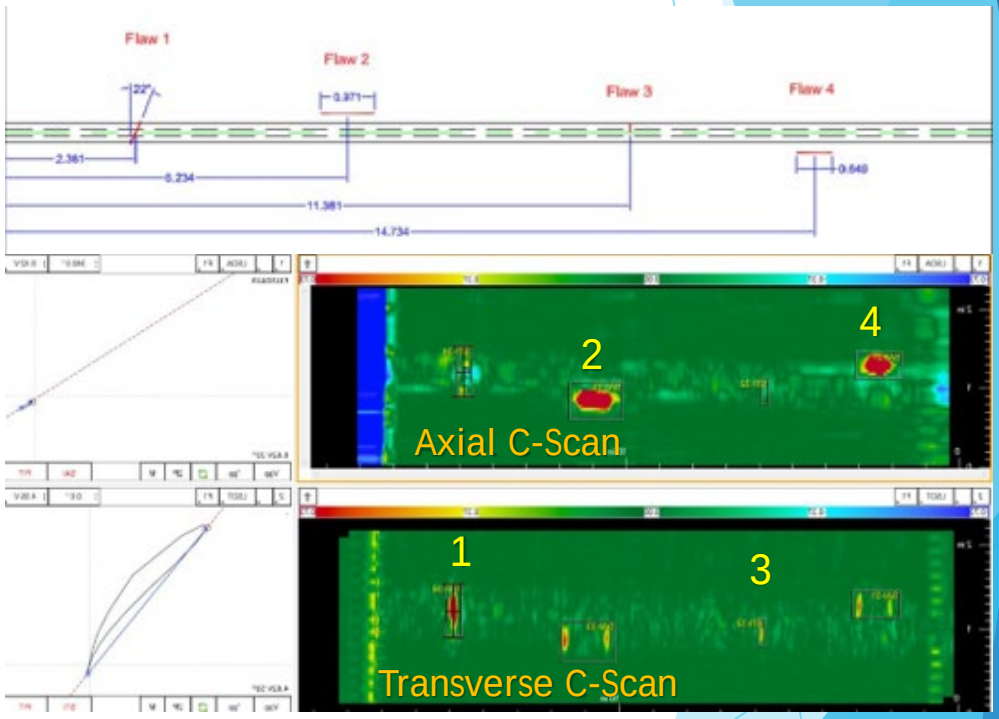
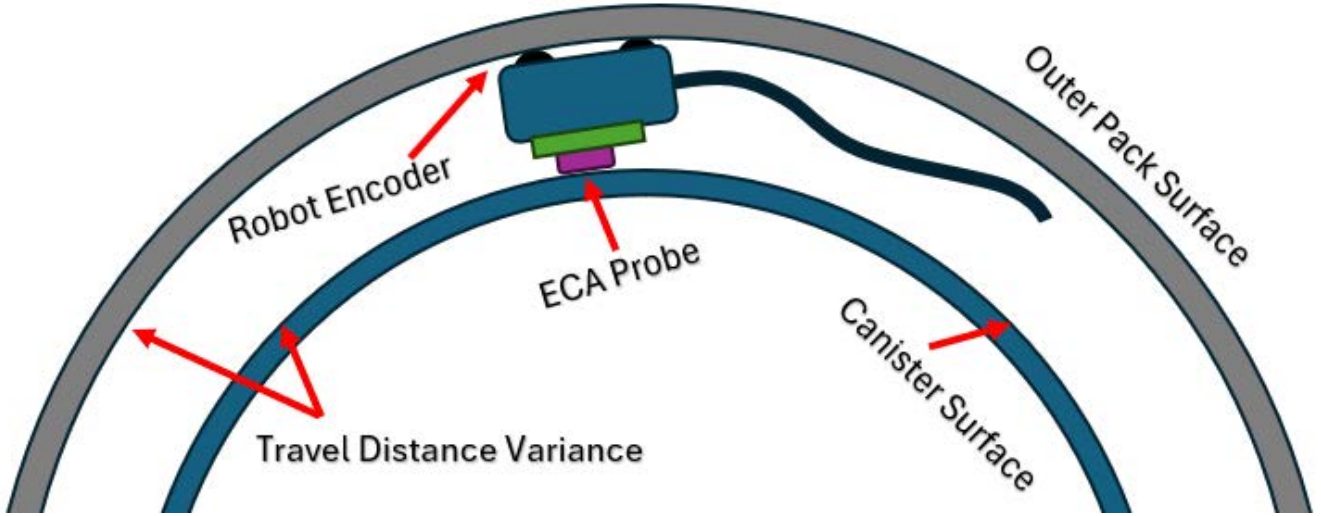
Open Plate ECA Testing Capabilities

- Open Test Plate = 4 x Design Flaws
- All 4 x Design Flaws Detected
- Lengths at +7% variance to actual
- Widths (too small to measure)
- Depths are Maxed out at 2.6mm
  - Due to Design Flaw Depths used in sizing curve creation



ECA Blind Testing Setup & Scan Plan

- ECA Robot Drives on Outer Pack Surface
- 2nd Robot Operates as Positional Validation
- Calculated the Encoder offset in Distance for Flaws



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Outcomes & Lessons Learned

- Remote ECA is a Viable Inspection Approach ✅
- All Design Flaws were Detected and Sized ✅
- Calibration Standard MUST Represent Flaw Range
- Sizing Curves are Limited to Design Flaws Used
- ECA Coil Configurations Built to Inspection Need
- Canister Dimensions are Critical to Scan Plan Design

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Robotic UT Technology Enhancing Drum Scans

IAG's 2025 Emerging Solutions Team

New Tech Capabilities – R.U.D.I. System

- Robotic Ultrasonic Drum Inspection
- Automate a repetitive task
- Improve repeatability & coverage
- Reduce risk & exposure of people
- Provide validation of condition
- 3,000 Drums inspected since March
- (Video for those interested)

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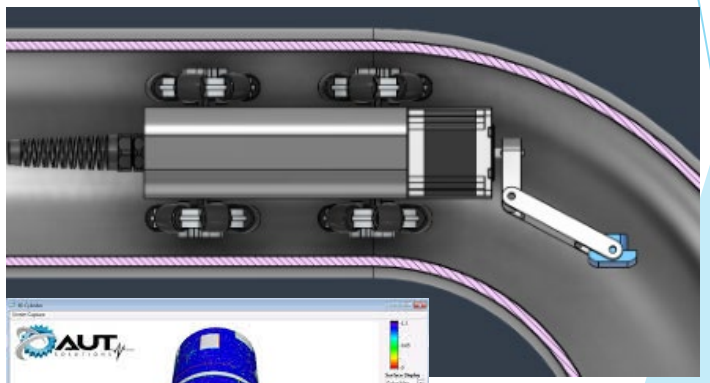
ROBOTIC TECHNOLOGIES OF TENNESSEE LLC

Robots Enhancing ID Corrosion Mapping

IAG's 2025 Emerging Solutions Team

New Tech Capabilities – ID Ultrasonic Robot

- Robotic Ultrasonic ID Pipe Crawler
- Automated Corrosion Mapping
- Improve repeatability & coverage
- Reduce risk & exposure of people
- Provide validation of condition
- Solution did NOT exist!
- (Videos available)



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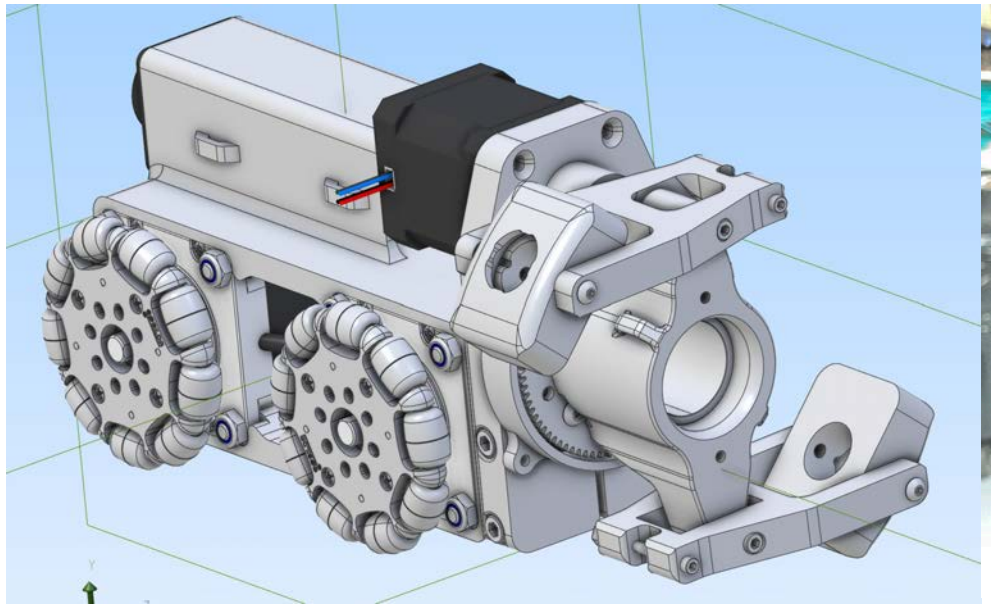
New Tech Capabilities – ID Ultrasonic Robot

From Design to Deployment



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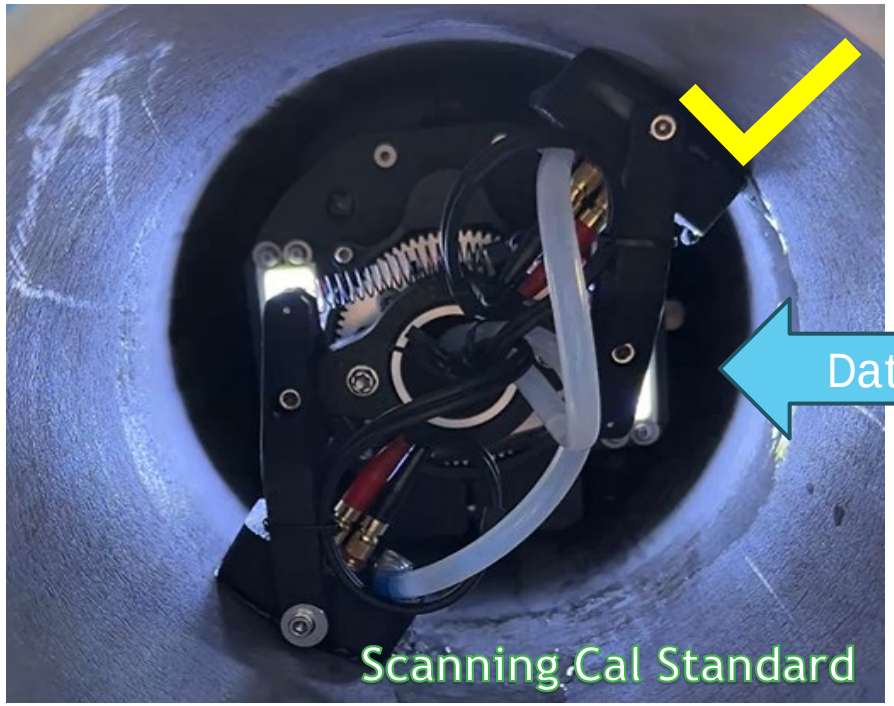
Delivering Design Results – ID Ultrasonic Robot

Acceptance Testing - Detection & Resolution



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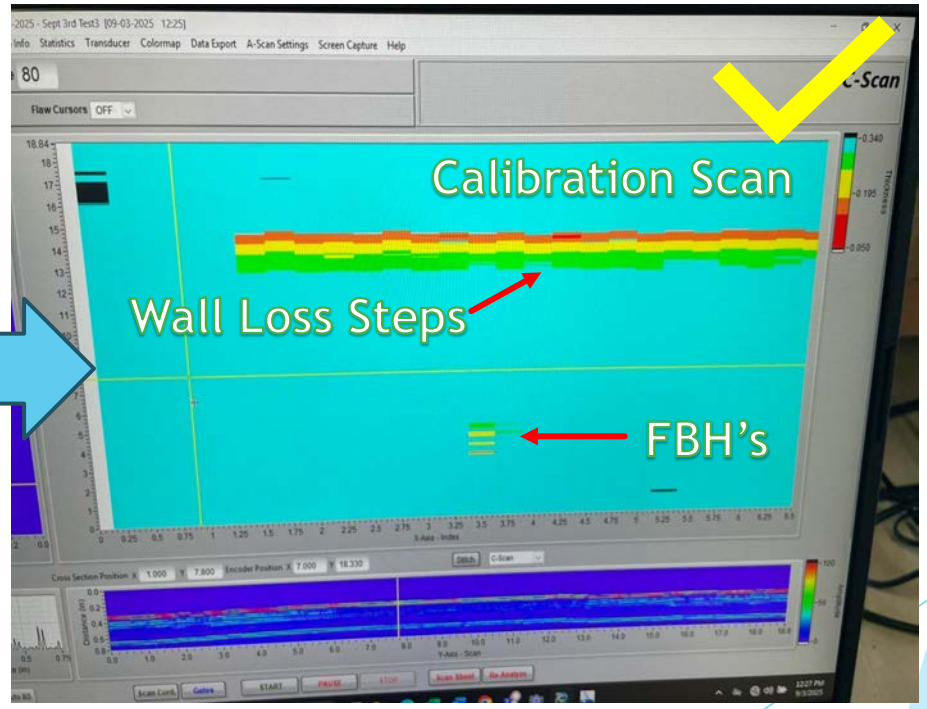

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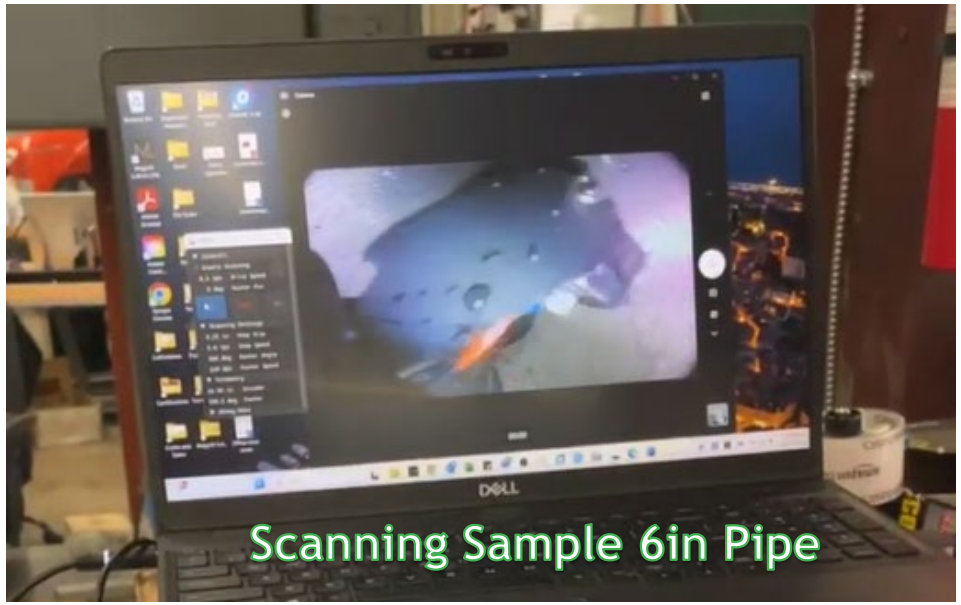
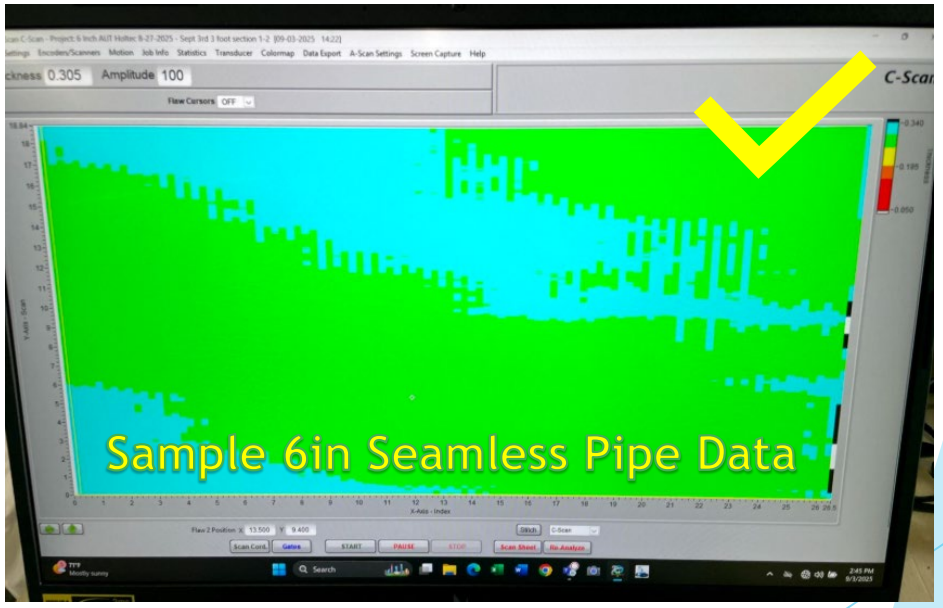
Delivering Design Results – ID Ultrasonic Robot

Acceptance Testing - 100% Coverage





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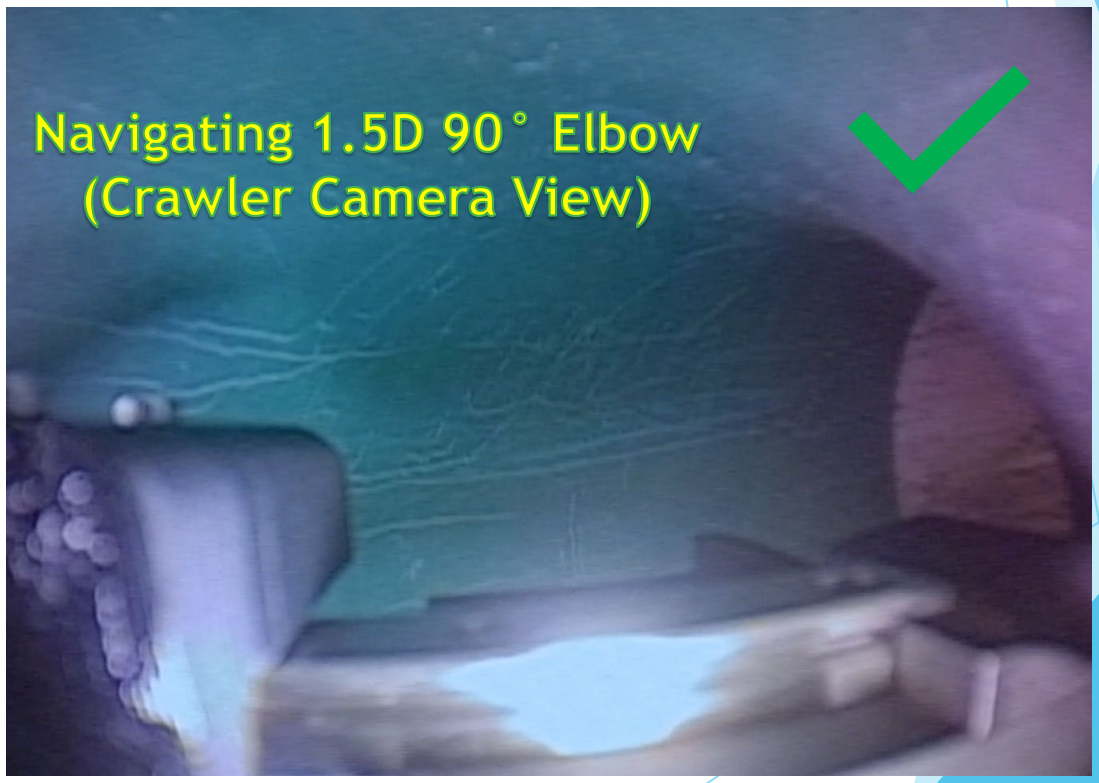
Delivering Design Results – ID Ultrasonic Robot

Acceptance Testing - Navigate 1.5D 90° Elbow



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In-Field Challenges
ID Ultrasonic Robot

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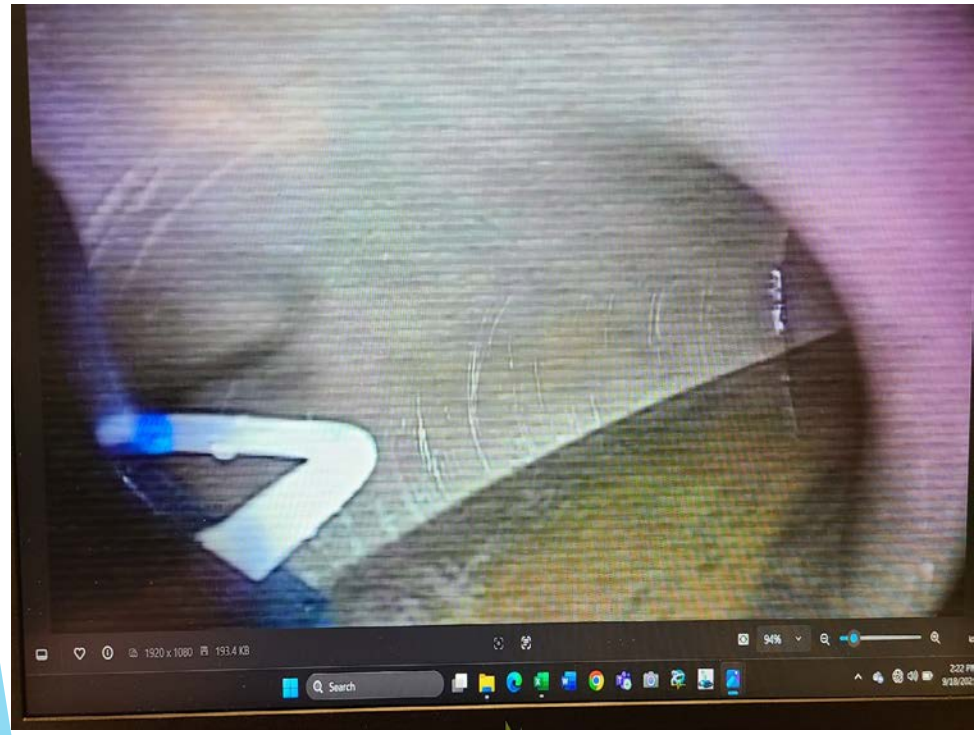
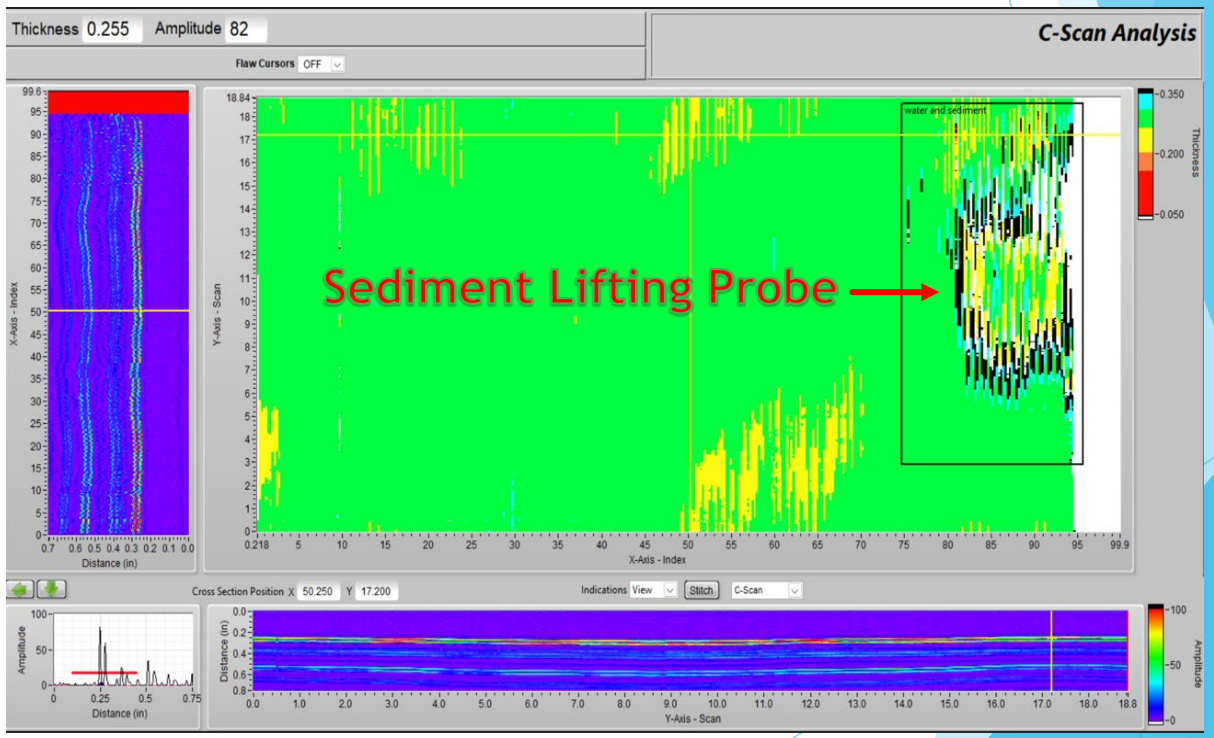
Field Inspection Challenges – ID Ultrasonic Robot

Flooded Pipe / Sediment / 1D 90° Elbow





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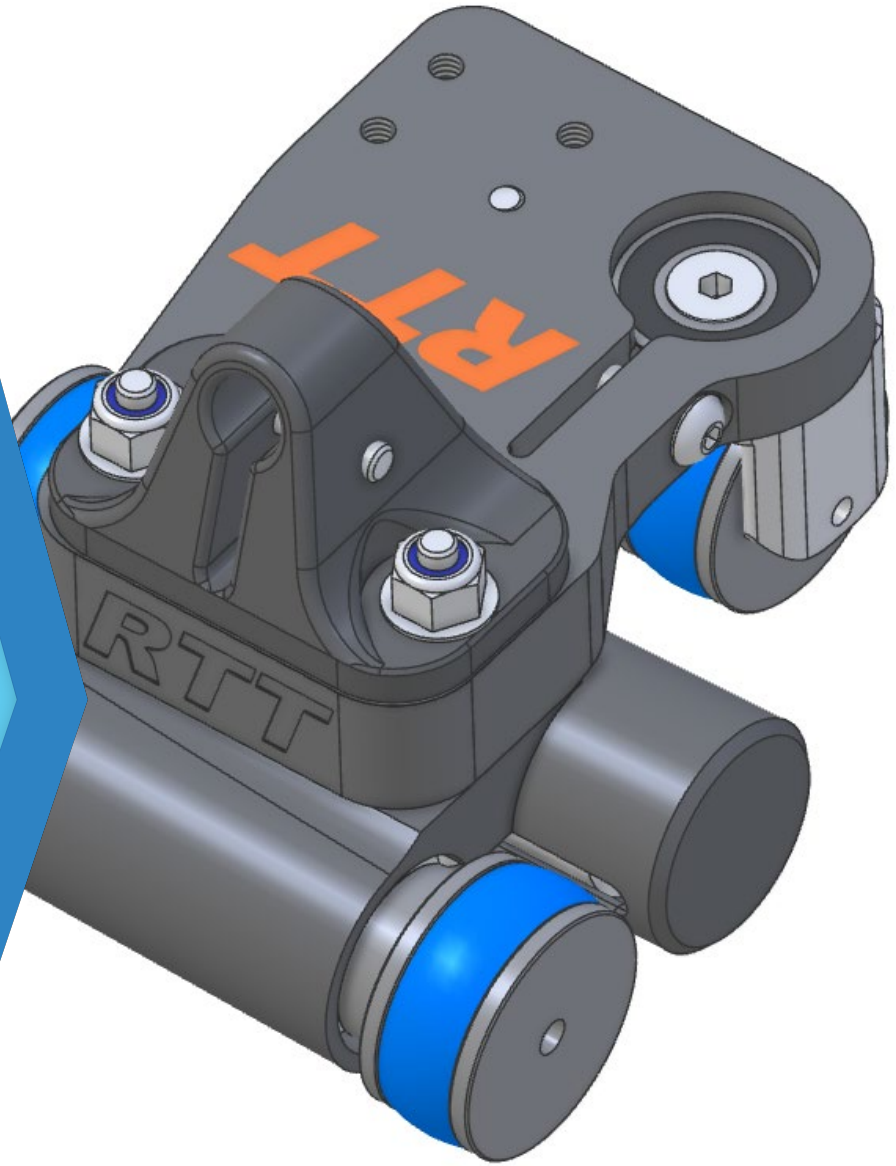
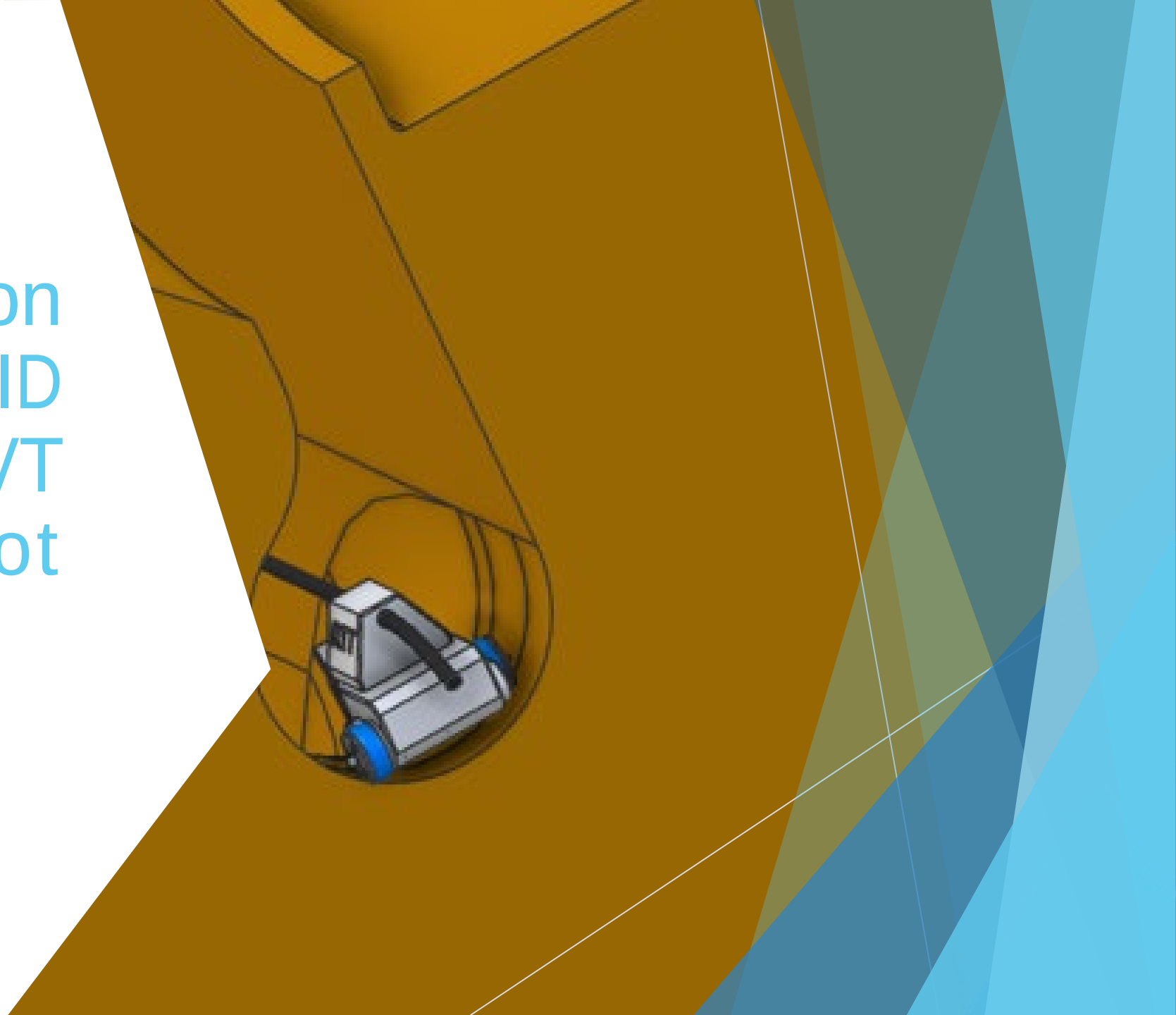
RTT
ROBOTIC TECHNOLOGIES OF TENNESSEE LLC

Customized Robots for Unique Inspections

IAG's 2025 Emerging Solutions Team



Field Inspection Challenges – ID Turbine VT Robot



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Field Inspection Challenges – ID Turbine VT Robot

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Nuclear Waste Sampling – Off Riser Sampling System

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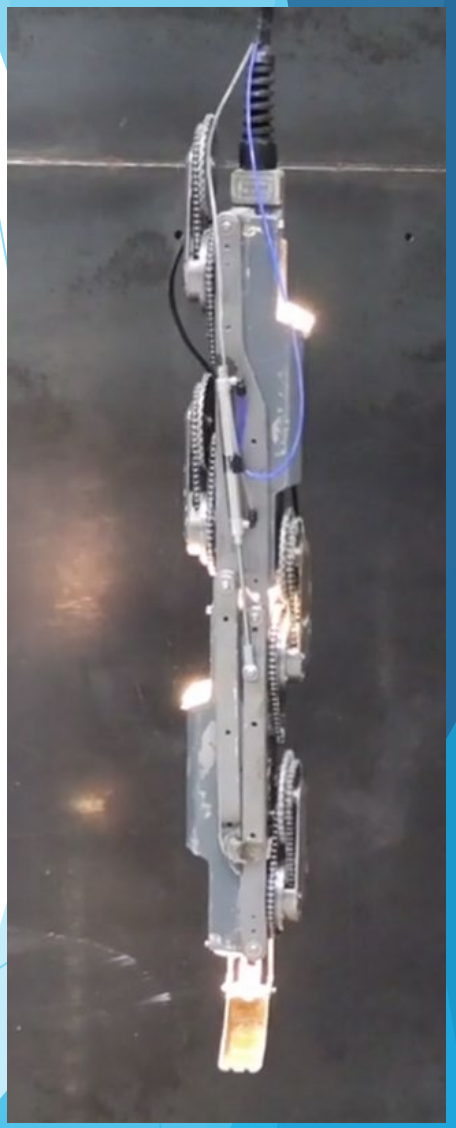
Nuclear Waste Sampling – Off Riser Sampling System

- Off Riser Sampling System (ORSS)
- Access waste tank through 6" riser
- Collect waste samples for testing
- (Videos available)



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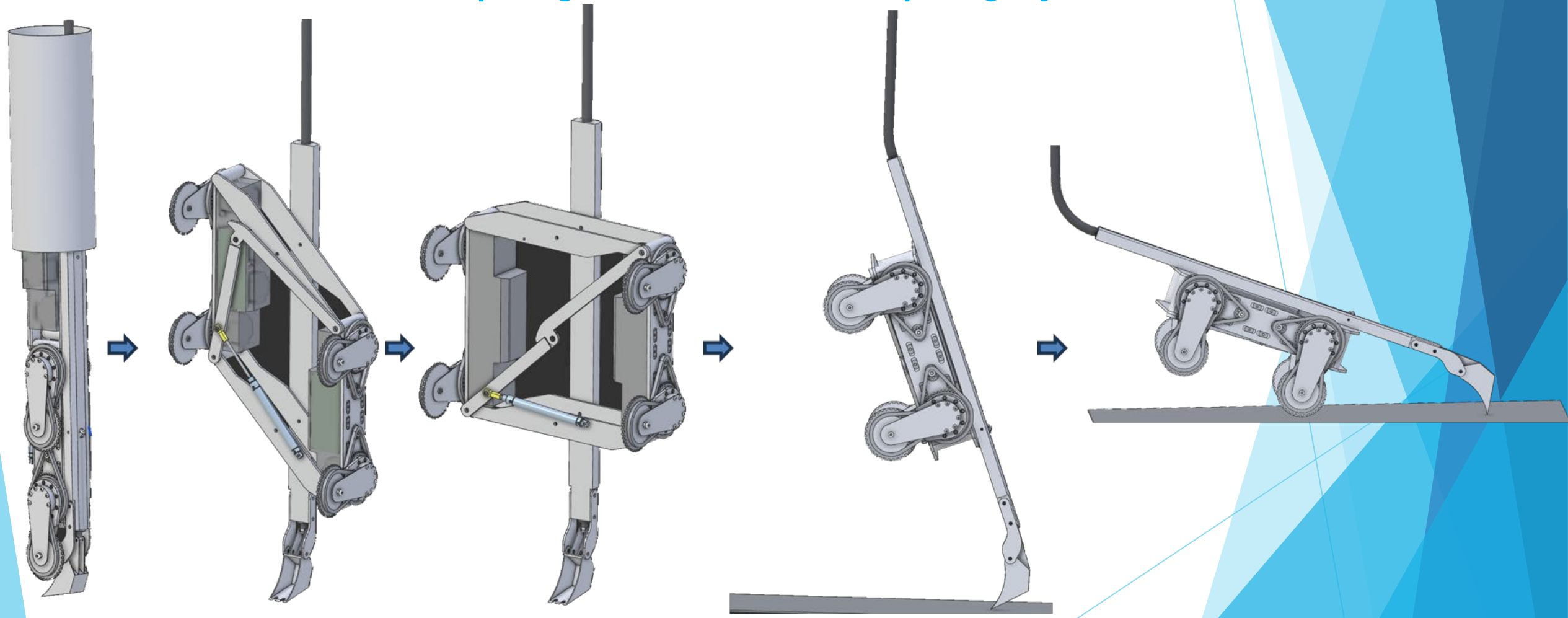
Nuclear Waste Sampling – Off Riser Sampling System



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Questions?

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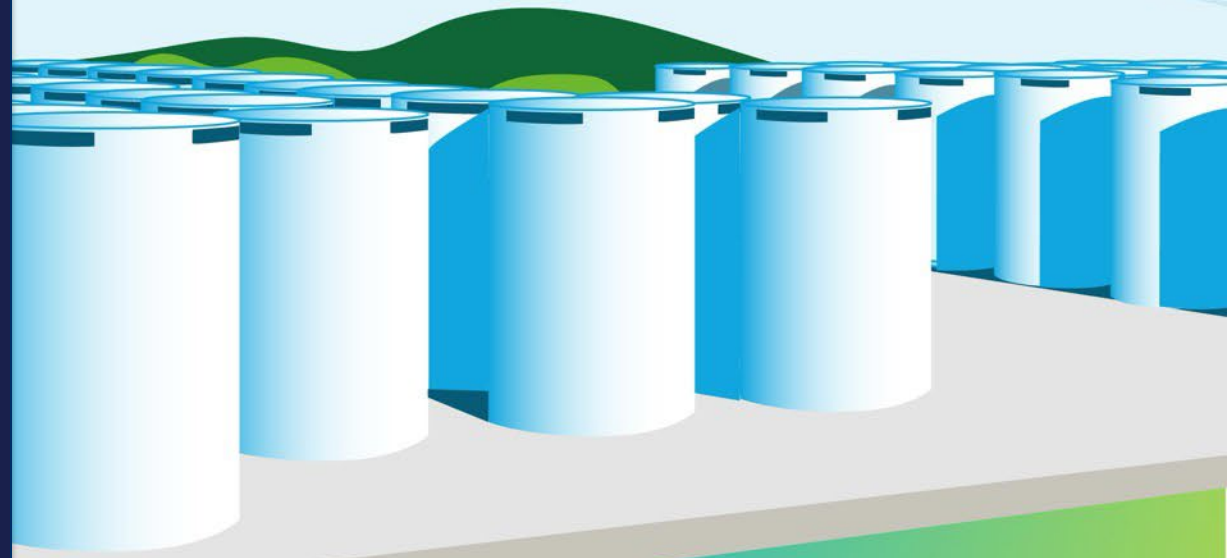
Chloride-Induced Stress Corrosion Crack Growth Rate Testing Using Salts Relevant to Near-Marine Environments

M.B Toloczko, D.D. Kling
Pacific Northwest National Laboratory

2025 EPRI ESCP Meeting
October 27-30
Charlotte, NC

PNNL-SA-217410



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Spent Fuel and High-Level Waste Disposition

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961)

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and constructing of a spent nuclear fuel repository.

Introduction

- **Conducting chloride-induced stress corrosion cracking (CISCC) tests to inform on aging management practices for independent spent fuel storage installations (ISFSIs) and future consolidated storage facilities (CSFs).**

- **Prior CISCC testing focused on:**
  - **CISCC growth rate tests of 304L exposed to simple salts.**
    - **Obtained a baseline understanding of CISCC and compared to literature information.**
  - **CISCC tests relevant to inland sites.**
    - **304L exhibits very low susceptibility.**

- **FY25 focused on CISCC tests relevant to near-marine sites.**
  - **304L is exhibiting very low CISCC growth rates when exposed to simulated near-marine site salts.**

Experimental Setup

Environmental Conditions for Tests (1/2)

- **Near-marine salt conditions.**
- **2–20 km from the ocean shoreline.**
- **SNL dust sample analysis of Hope Creek and Maine Yankee indicated NaCl, nitrates, and sulfates as dominant salts.**
  - **Higher salt loading than inland.**
  - **Very low levels of observed $MgCl_2$.**
- **Importance of nitrates:**
  - **Literature data generally indicate that nitrates suppress CISCC.**
  - **There is essentially no NO_3 in seasalt.**
  - **Nitrates are present in atmosphere due to industrial pollution and farming.**
  - **Canister dust sampling indicates NO_3 to Cl molar ratios ranging from 1:10 to 1:1.**
- **Little literature data are mixed on whether sulfates enhance or suppress CISCC with chloride salts.**

Environmental Conditions for Tests (2/2)

- **Near-marine relevant salts evaluated:**
  - **NaCl – Baseline behavior.**
  - **4:1 NaCl : $NaNO_3$ – Low-level nitrate additions.**
  - **5:1:1 NaCl : $NaNO_3$: Na_2SO_4 – Low-level nitrate & sulfate additions.**
  - **1:1:1 NaCl : $NaNO_3$: Na_2SO_4 – High-level nitrate & sulfate additions.**
  - **Simulated Hope Creek Salt – At the mouth of Delaware River (40% of ocean salinity)**
  - **Simulated Maine Yankee Salt – River estuary with brackish water (50% of ocean salinity)**

- **Controlled humidity tests.**
- **Baseline temperature = 40ºC.**
- **CISCC growth rate testing was started with RH 3–5% above the deliquescence relative humidity (DRH) of the salt.**

Materials (1/2)

- **Materials evaluated:**
  - **Three heats of 304L – Assumed standard material for dual-purpose canisters (DPCs).**
  - **Three heats of 304 (304H) – When used in a sensitized condition, it serves as a reference material to show that low CISCCGRs in 304L are not due to improper experimental setup.**

			Elemental Composition (wt%)													
Matrl	Heat/ID	**Short Name**	C	Co	Cr	**Cu**	Mn	**Mo**	N	Nb	Ni	P	**S**	Si	Ti	Fe
304L	*ASTM A240 Specification*		*0.03*	---	*18-20*	---	*2.0*	---	*0.10*	---	*8-11*	*0.045*	***0.030***	*0.75*	---	*bal*
	04E28VAA	**304L-P1**	0.017	0.23	18.1	**0.41**	1.78	**0.41**	0.080	0.014	8.0	0.037	**0.001**	0.24	0.002	bal
	SD41059	**304L-P2**	0.019	0.21	18.2	**0.34**	1.63	**0.10**	0.073	0.010	8.2	0.028	**0.008**	0.39	0.003	bal
	N85426	**304L-N8**	0.024	NR	18.8	**0.26**	1.53	**0.27**	0.060	NR	8.0	0.035	**0.009**	0.61	NR	bal
304	*ASTM A240 Specification*		*0.08*	---	*18-20*	---	*2.0*	---	*0.10*	---	*8-11*	*0.045*	***0.030***	*0.75*	---	*bal*
	863541	**304H-P1**	0.060	NR	18.3	**0.43**	1.19	**0.53**	0.06	NR	8.1	0.031	**<0.001**	0.49	NR	bal
	74901	**304H-P2**	0.056	0.13	18.1	**0.32**	1.81	**0.36**	0.09	0.013	8.1	0.029	**0.001**	0.29	0.000	bal
	SD44996	**304H-P3**	0.056	0.22	18.4	**0.31**	0.92	**0.15**	0.03	0.01	8.1	0.028	**0.005**	0.49	0.003	bal

Materials (2/2)

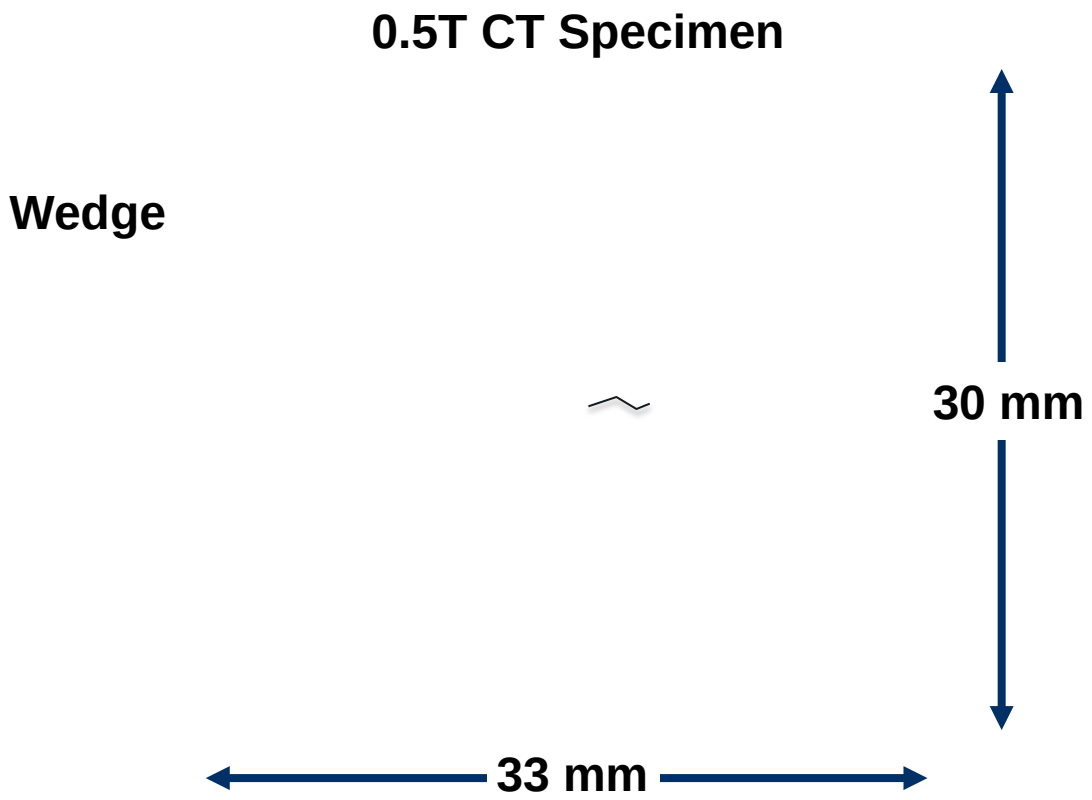
- **304L was tested in various thermomechanical conditions.**
  - **Mill annealed (MA) condition - Typical condition of as-purchased 304L plate.**
  - **MA + 10% cold strained - Simulate deformation due to construction and handling.**
  - **Sensitization treatment applied - Meant to represent 304L heat affected zone (HAZ) regions by welds.**

- **304L "S2" sensitization treatment:**
  - **800ºC/10min +**
  - **controlled cooling from 800→650ºC over 30 min +**
  - **controlled cooling from 650→500ºC over 14 hrs +**
  - **air cool**

Specimen Preparation for Controlled Humidity Tests

- **Specimen preparation:**
  - Specimens were air-fatigue precracked for 1 mm without salt.
  - Wedge placed in the specimen mouth at peak load to hold the crack open (creating 5–10 µm crack opening).
  - Side grooves sealed with silicone sealant.
  - Sonicated in a conc. salt solution for 20 min.
  - Dried at 40ºC.
  - Salt wiped from specimen exterior except notch.
  - Installed in a CISCC test system.

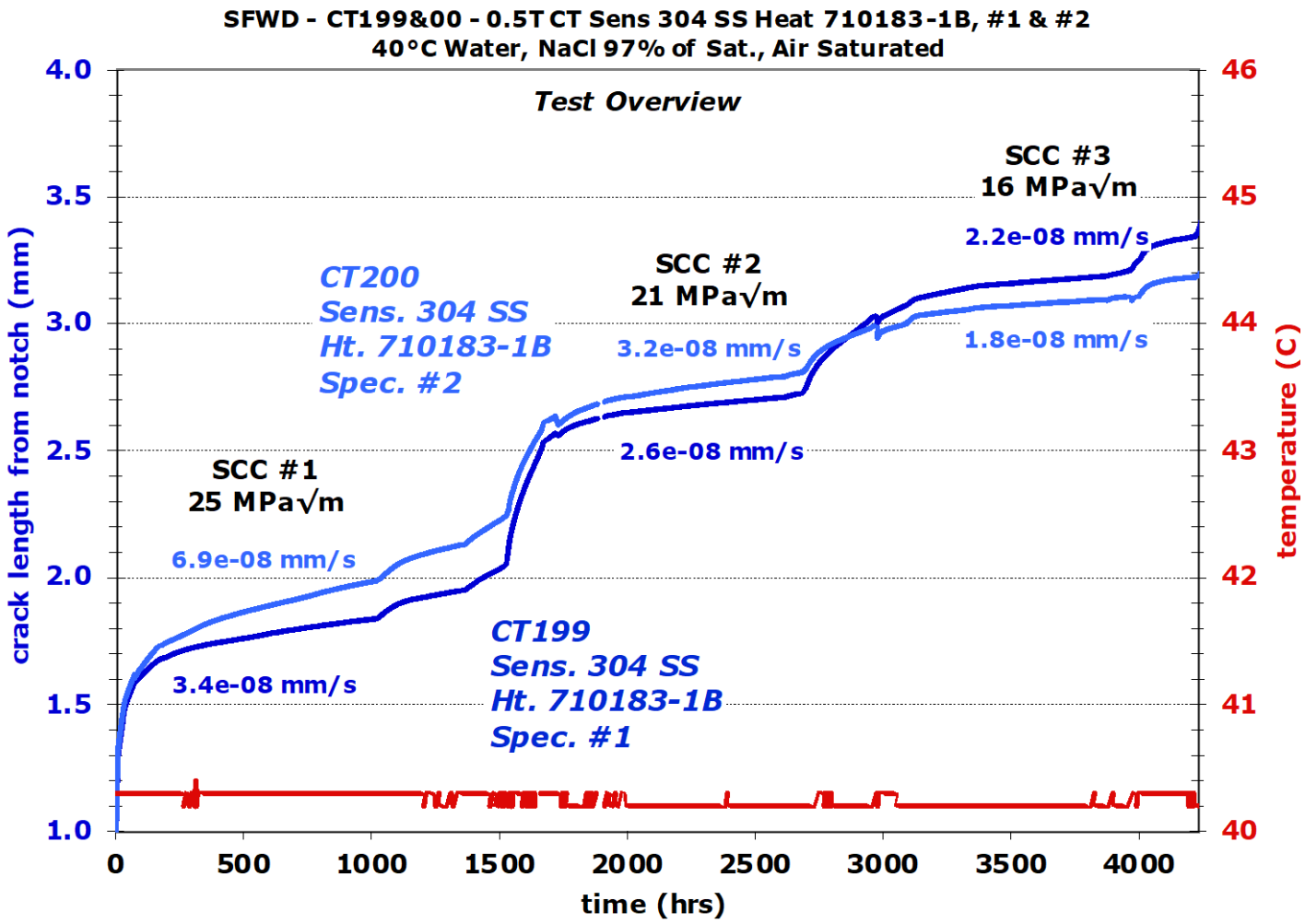
- **<u>HIGH salt loading.</u>**
  - **<u>Not canister-realistic.</u>**
  - ***<u>Provides bounding behavior</u>*.**



Testing Approach (1/2)

- Five test systems.
- Each test system can test up to 6 specimens simultaneously.
- In-situ measurement of crack length by direct current potential drop (DCPD).
- On-the-fly control of:
  - Load or stress intensity
  - Application of cyclic loading
  - Temperature
  - RH
- ≥500 hours of observation of CISCC behavior at each condition.

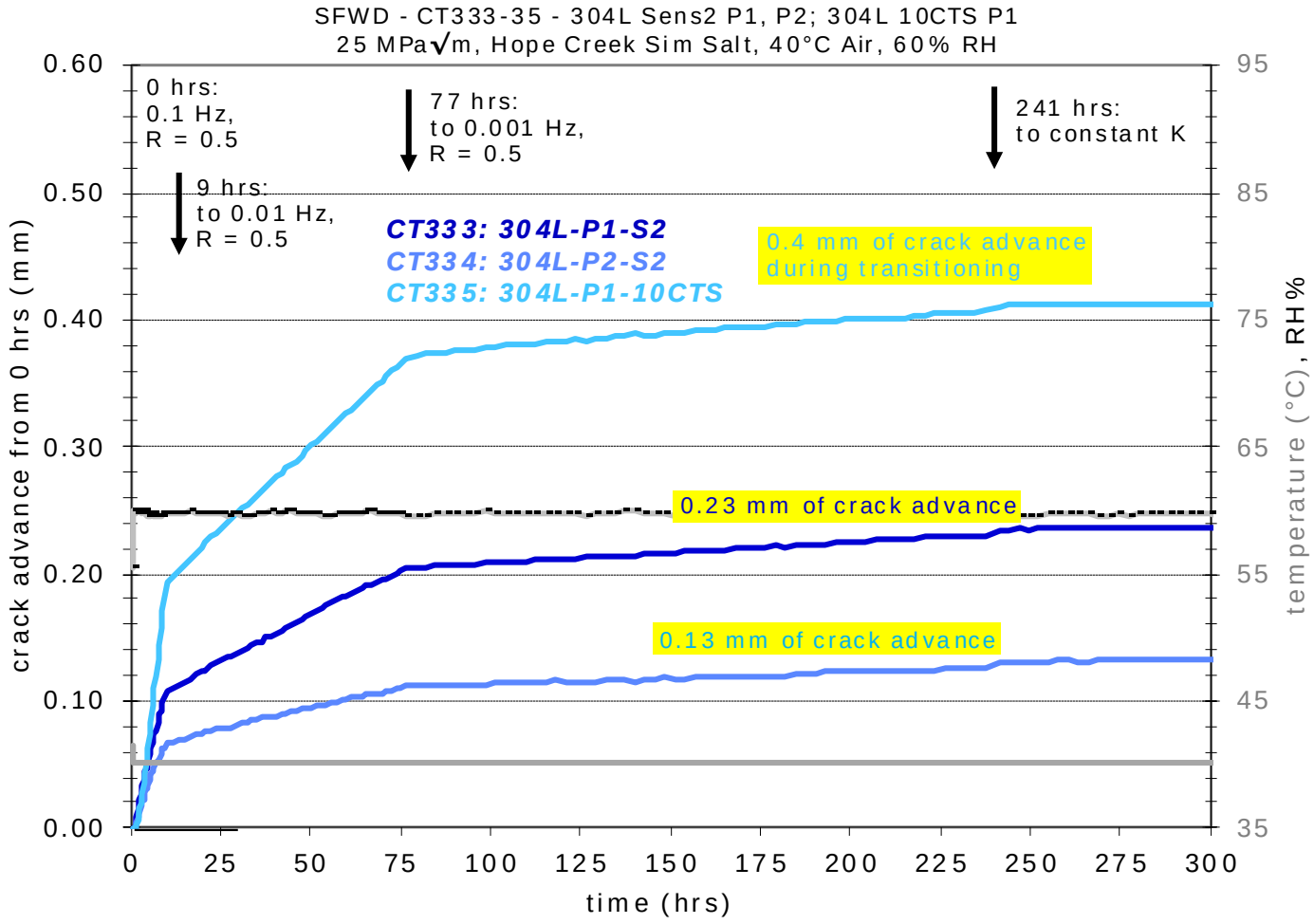
Example of a 2-specimen test



Testing Approach (2/2)

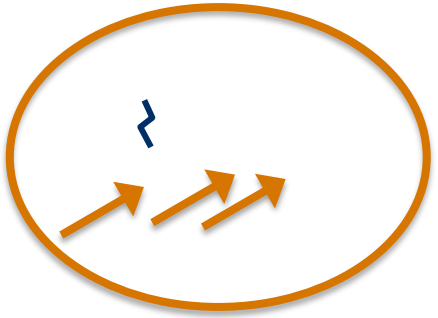
- A series of progressively more gentle load cycling steps are applied at the onset of environmental testing to transition the air fatigue precrack to constant load CISCC conditions.
- Typically target 0.3–0.5 mm of crack advance during transitioning.

Example of a 3-specimen test



Post-Test Specimen Examination Approach

- **Tests typically ended at constant load or constant stress intensity factor (K), so that final crack morphology is representative of constant load/K.**
- **Half of a specimen is set aside for crack surface examinations.**
- **The other half is sliced into three pieces for cross section examinations.**
- ***Cross-sections are typically more useful for CISCC exams due to complex crack morphology.***

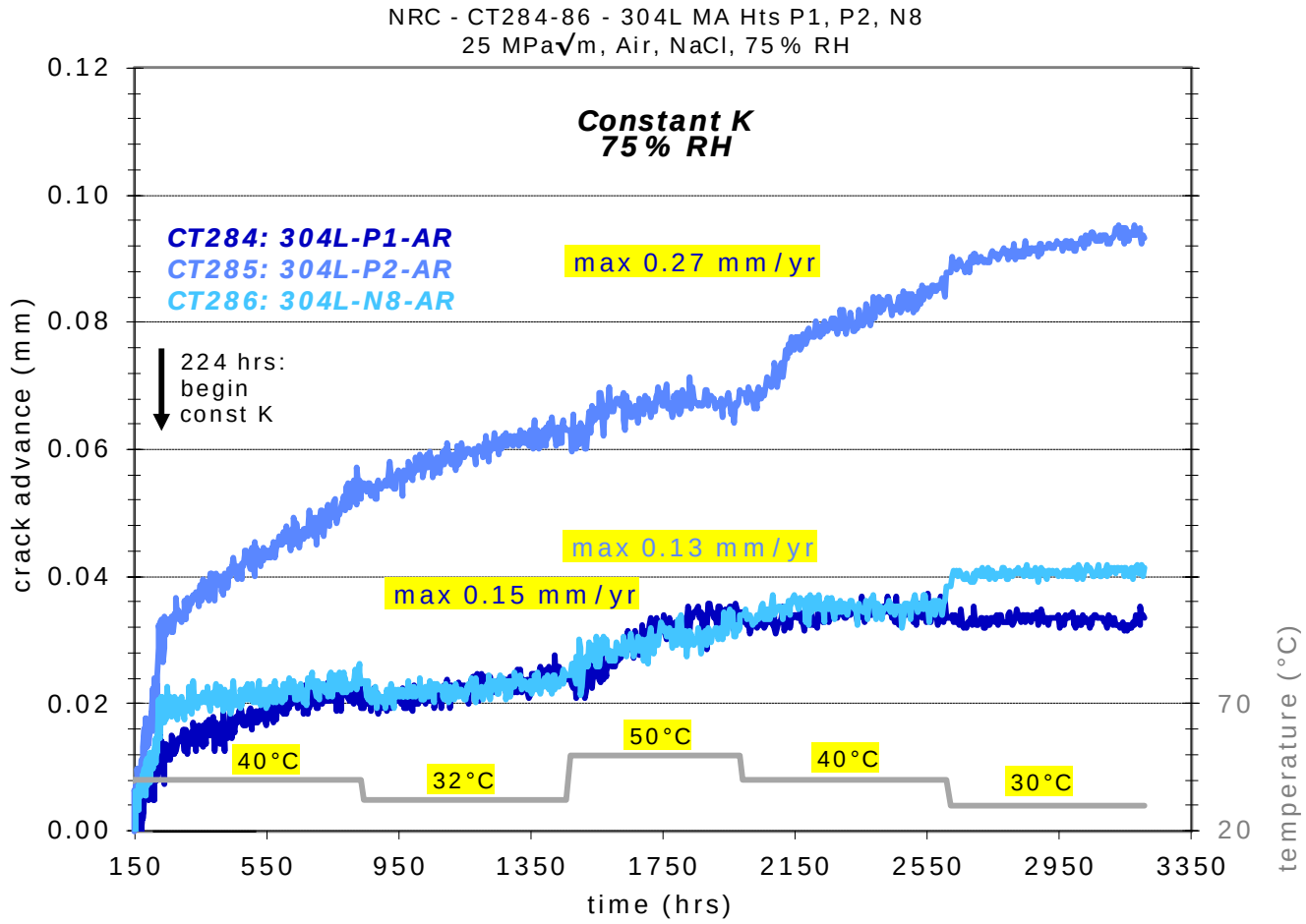


Test Results

304L Exposed to NaCl – CISCC Growth Rate Response

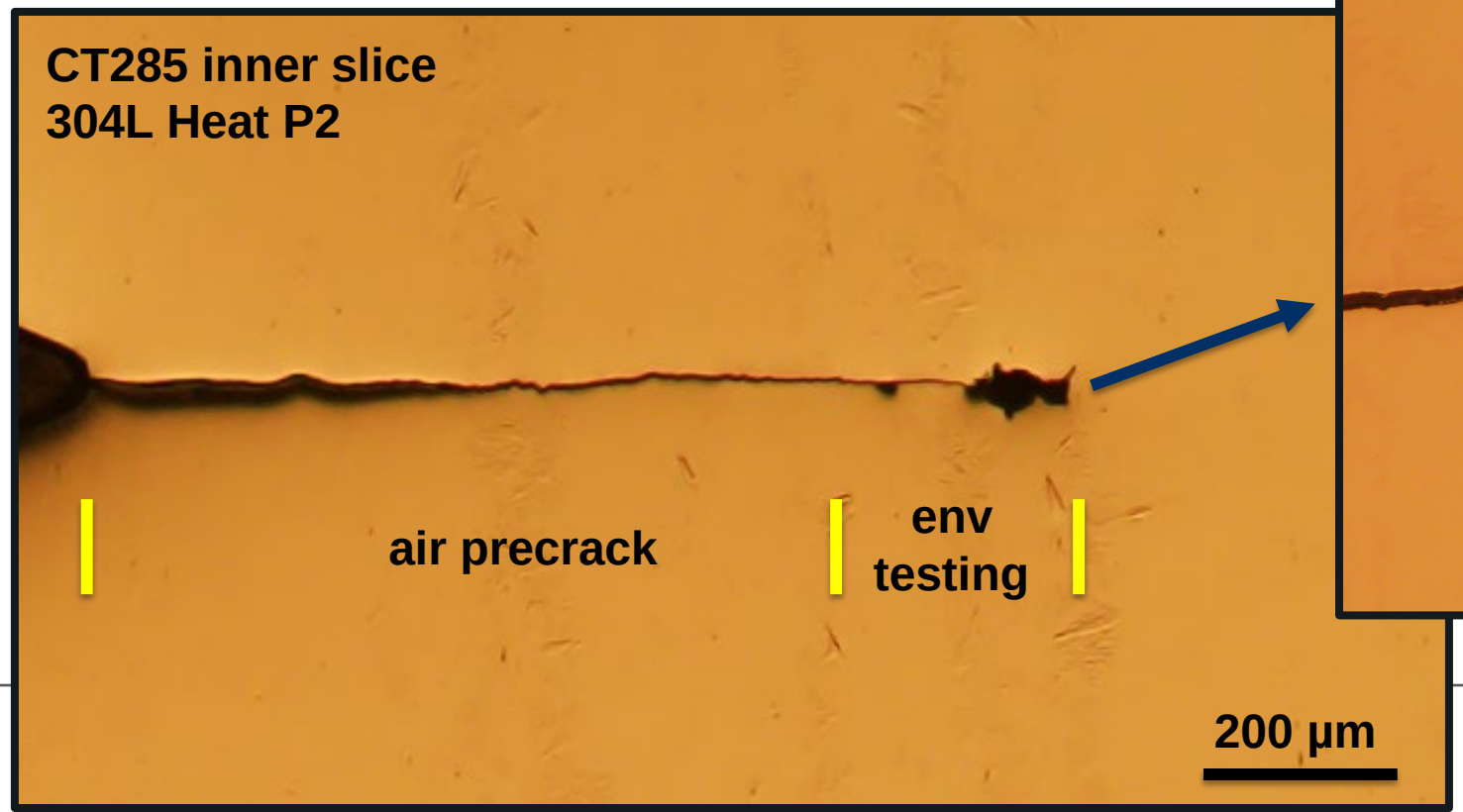
- **MA 304L exhibits low CISCC growth susceptibility to NaCl at canister-relevant temperatures where deliquescence is possible.**
- **The same results were observed in multiple tests with concentrated NaCl at various RH levels and temperatures.**

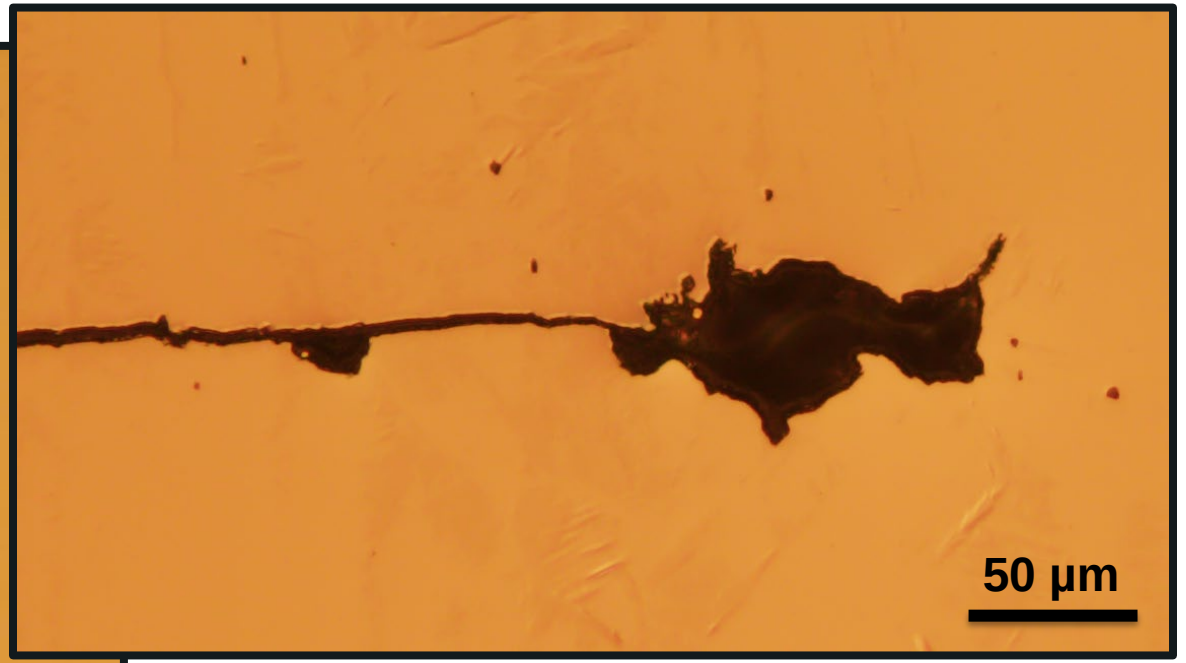
Temperature Effects Evaluation



304L Exposed to NaCl - Post-test Crack Morphology

- Showing one cross section out of three for the 304L Heat P2 specimen.
- Crack wall corrosion with a blunted crack front is common.
  - But some cross sections have uncorroded cracks.
- Fatigue transitioning steps account for crack advance during environmental testing.

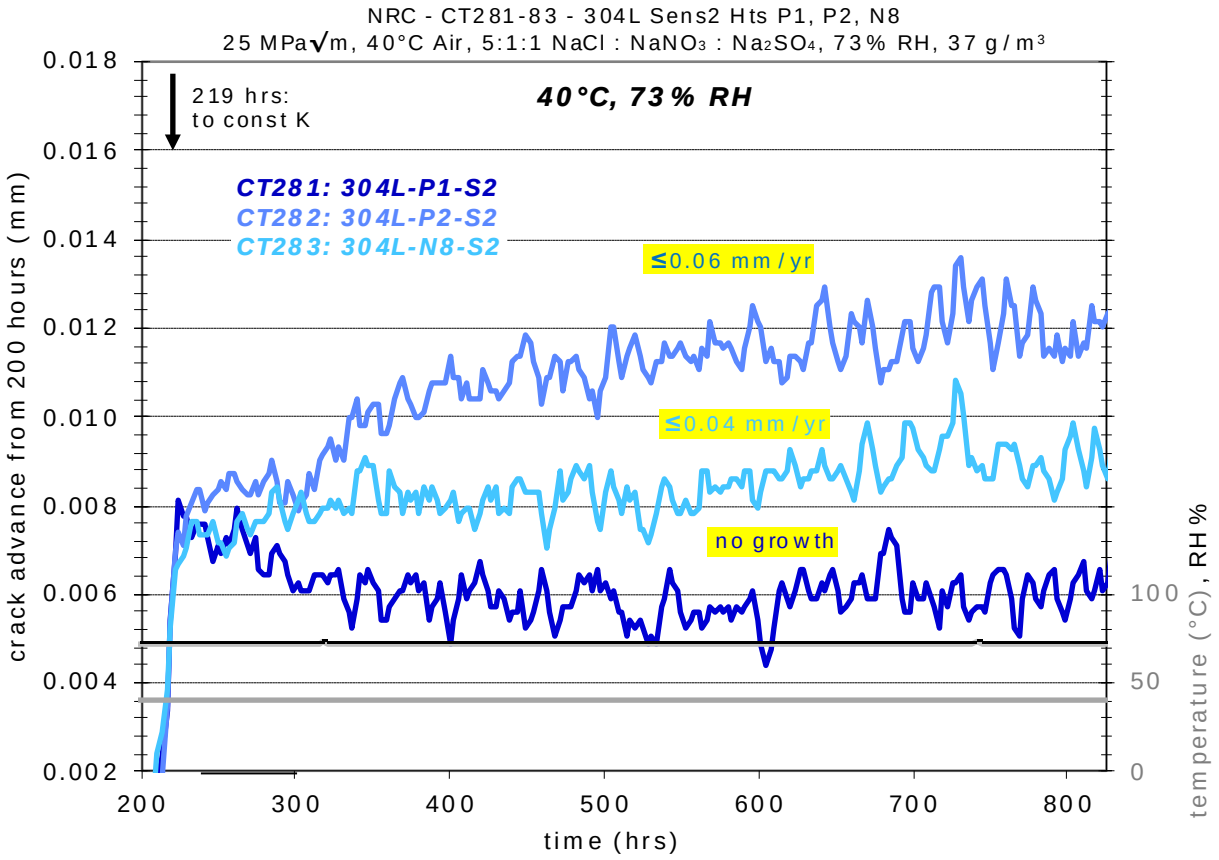




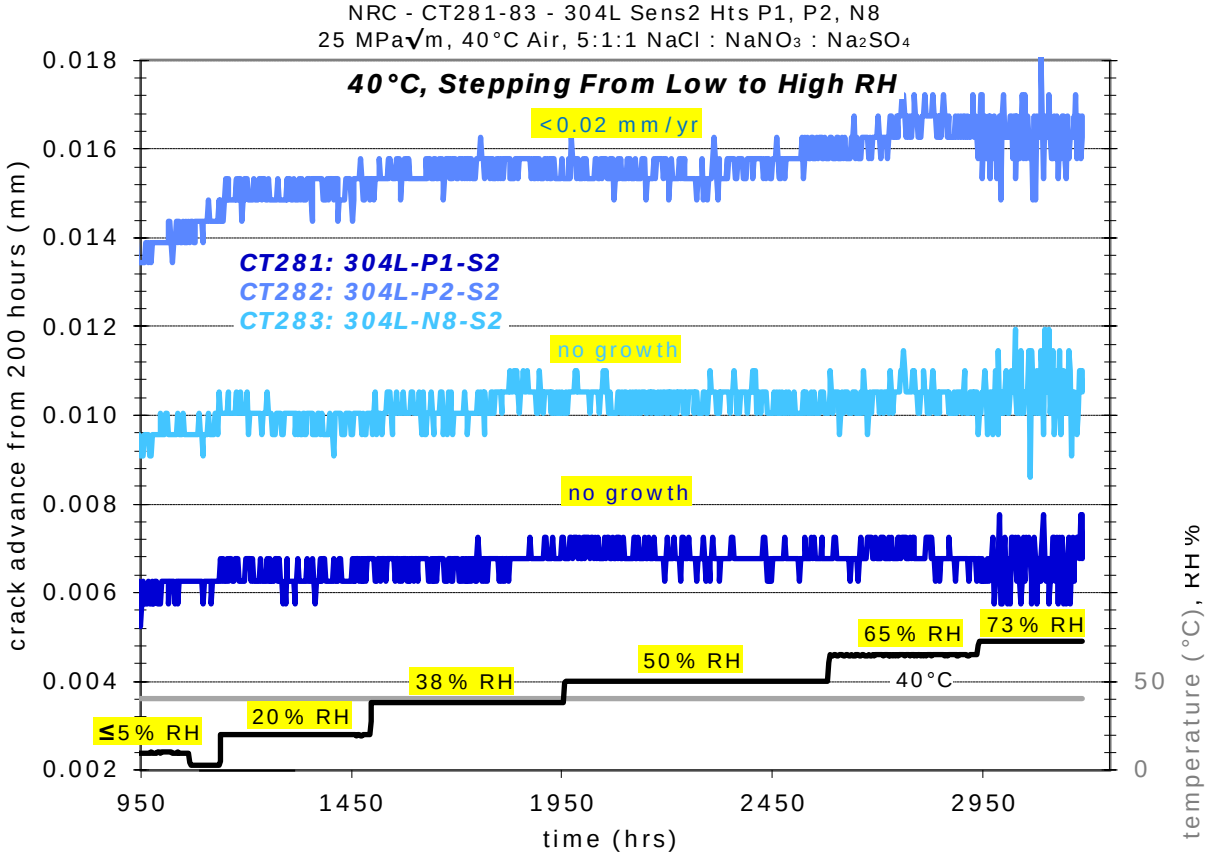
304L Exposed to 5:1:1 NaCl : $NaNO_3$: Na_2SO_4 – CISCC Growth Rate Response

- 304L with sensitization treatment applied.
- Relatively small additions of nitrate and sulfate.
- Low CISCC growth rates regardless of the RH level. Same as for pure NaCl.

First portion of test at 73% RH

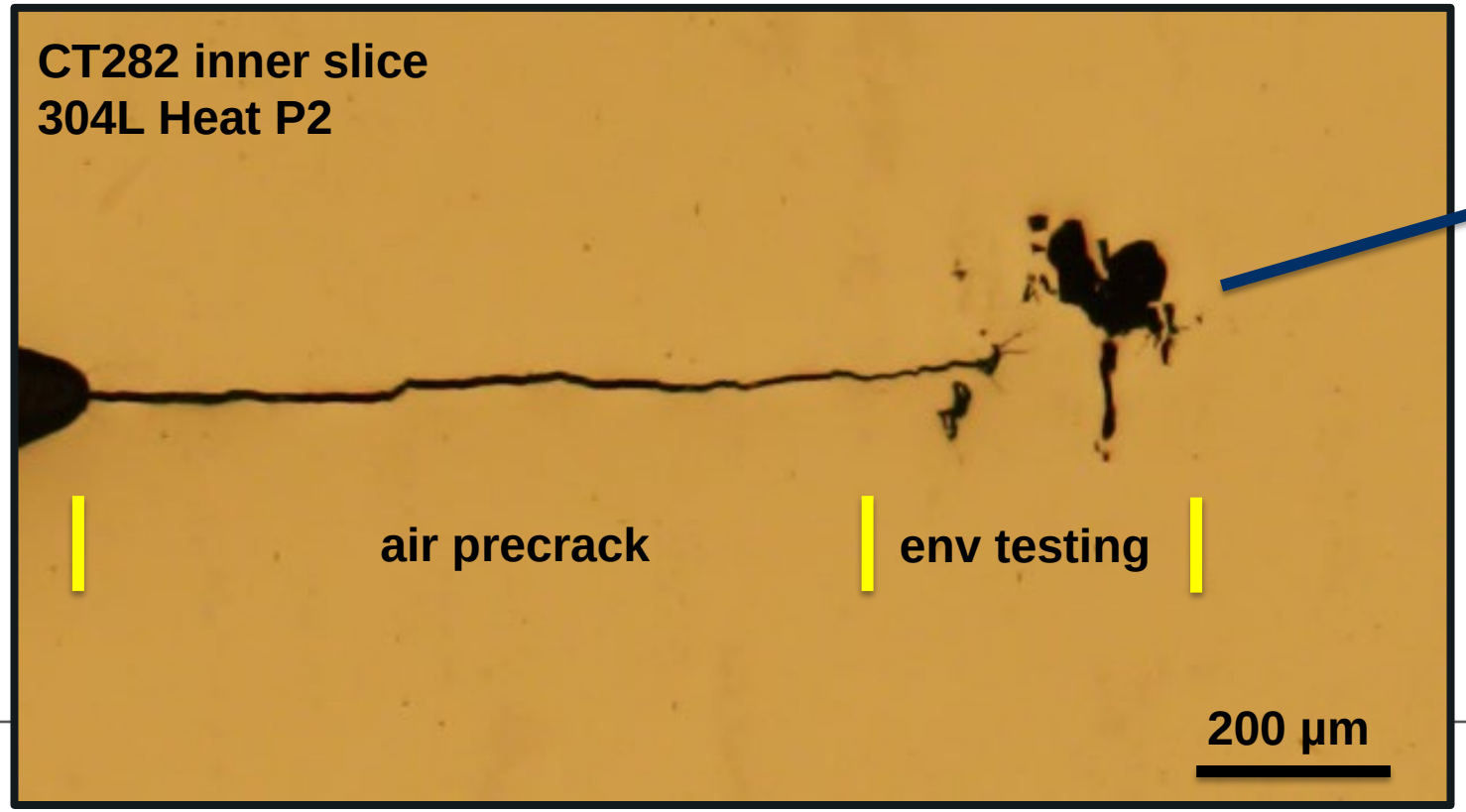


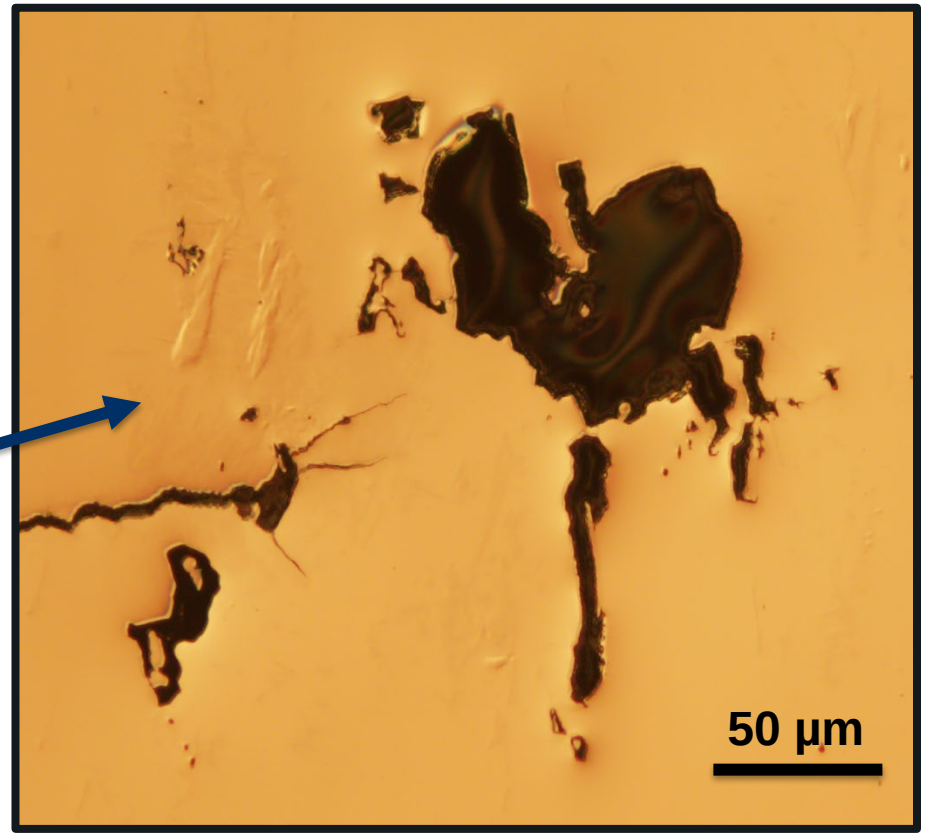
Second portion of test, stepping RH



304L Exposed to 5:1:1 NaCl : $NaNO_3$: Na_2SO_4 – Post-test Crack Morphology

- Typical crack morphology shown.
- No obvious CISCC growth.





304L Exposed to Simulated Maine Yankee Canister Salt - Salt Composition

- **Salt composition based on Maine Yankee DPC dust sample analysis by SNL[1].**
- **"Selected Composition" adjusted to reduce concentrations of ions associated with insoluble species at relevant RH.**

	Composition, (Cl- Normalized)	
Dissolved Species	From Dust Samples	**Selected Composition**
Na+	2.0670	2.0676
K+	0.7267	0.7270
Ca+2	10.0405	0.0805
Mg+2	0.5488	0.0275
Cl-	**1.0000**	**1.0000**
NO3-	**1.0796**	**1.0804**
SO4-2	**1.2371**	**0.4624**
HCO3-	19.4186	0.0055

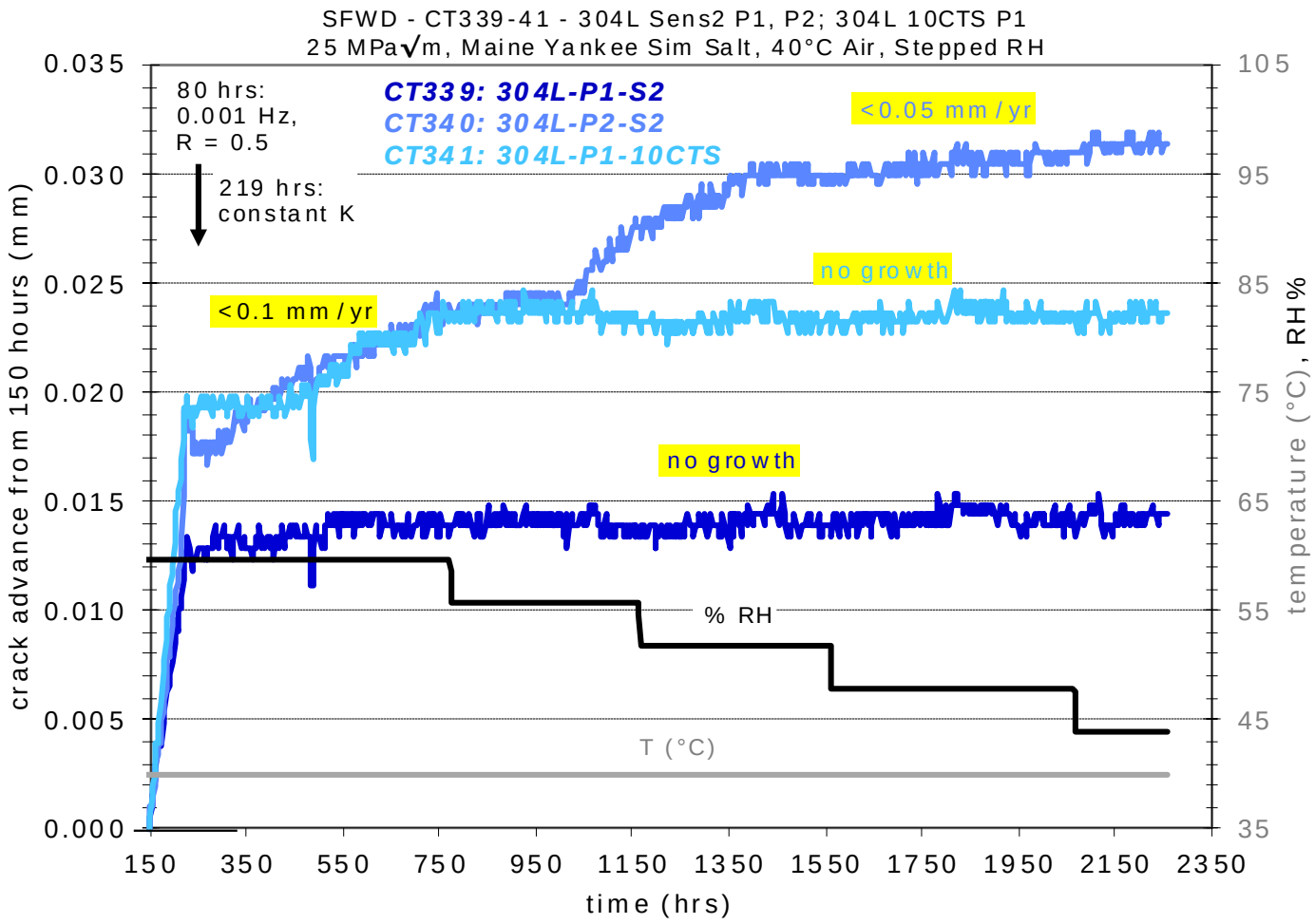
Deliquescence begins at 55% RH. Almost fully deliquesced at 70% RH.

Site is ~10 km from ocean. Located on a brackish inlet.

[1] R. Schaller, et al., "FY20 STATUS REPORT: SNF INTERIM STORAGE CANISTER CORROSION AND SURFACE ENVIRONMENT INVESTIGATIONS", SAND20-12683 R, 2020. Available in OSTI.

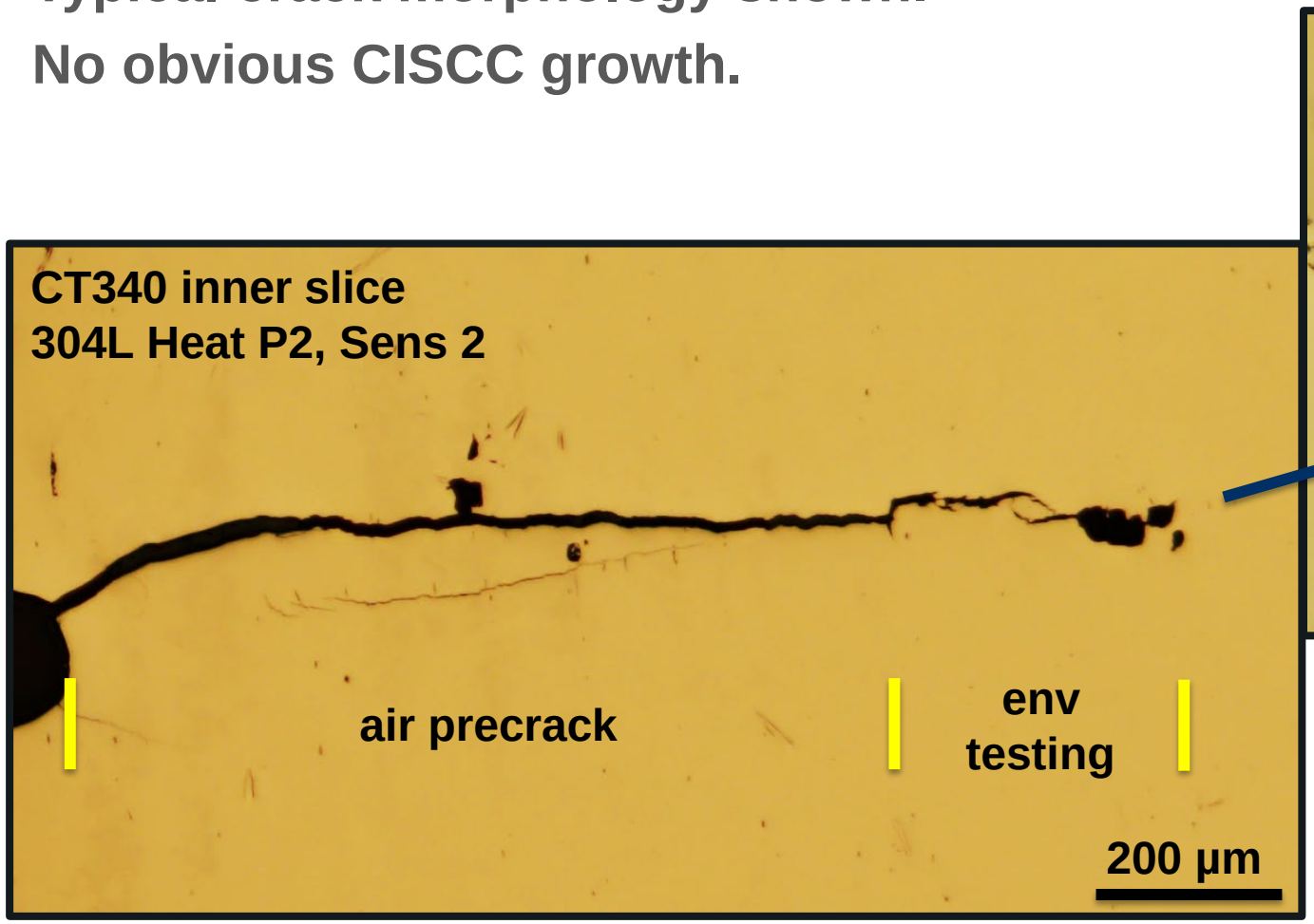
304L Exposed to Simulated Maine Yankee Canister Salt – CISCC Growth Rate Response

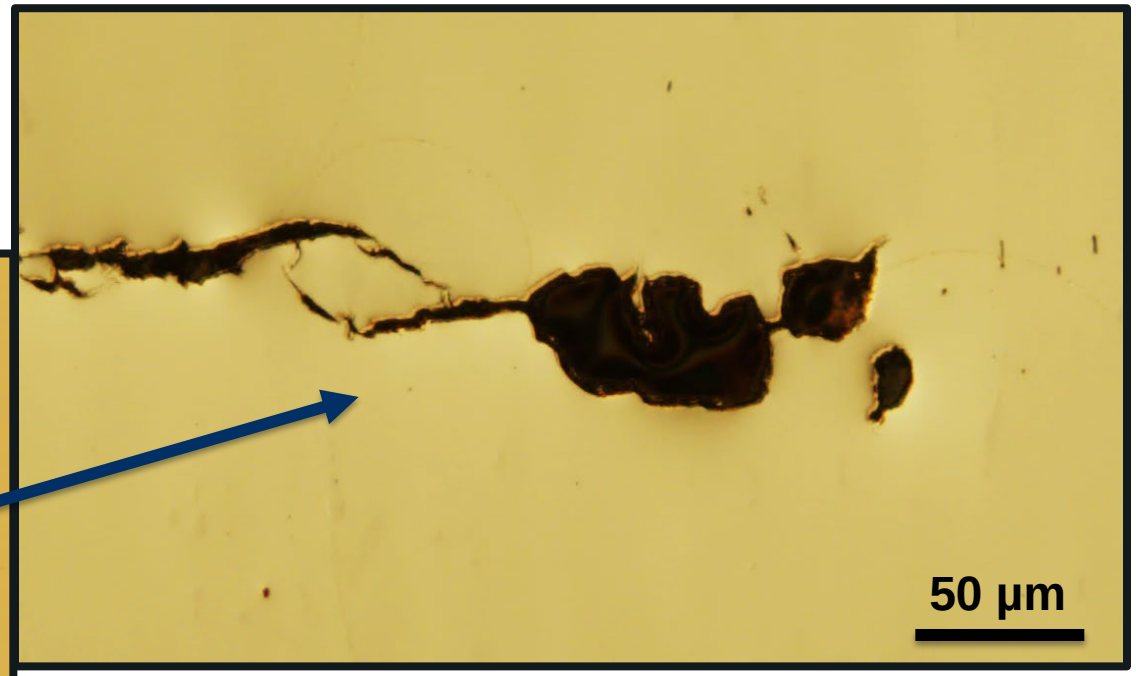
- Deliquescence begins at 55% RH. Almost fully deliquesced at 70% RH.
- Test started at 60% RH.
  - Step down by 4% RH every 350–500 hours.
  - Final RH of 44%.
- Very low or no CISCC advance.



304L Exposed to Simulated Maine Yankee Canister Salt – Post-test Crack Morphology

- Typical crack morphology shown.
- No obvious CISCC growth.





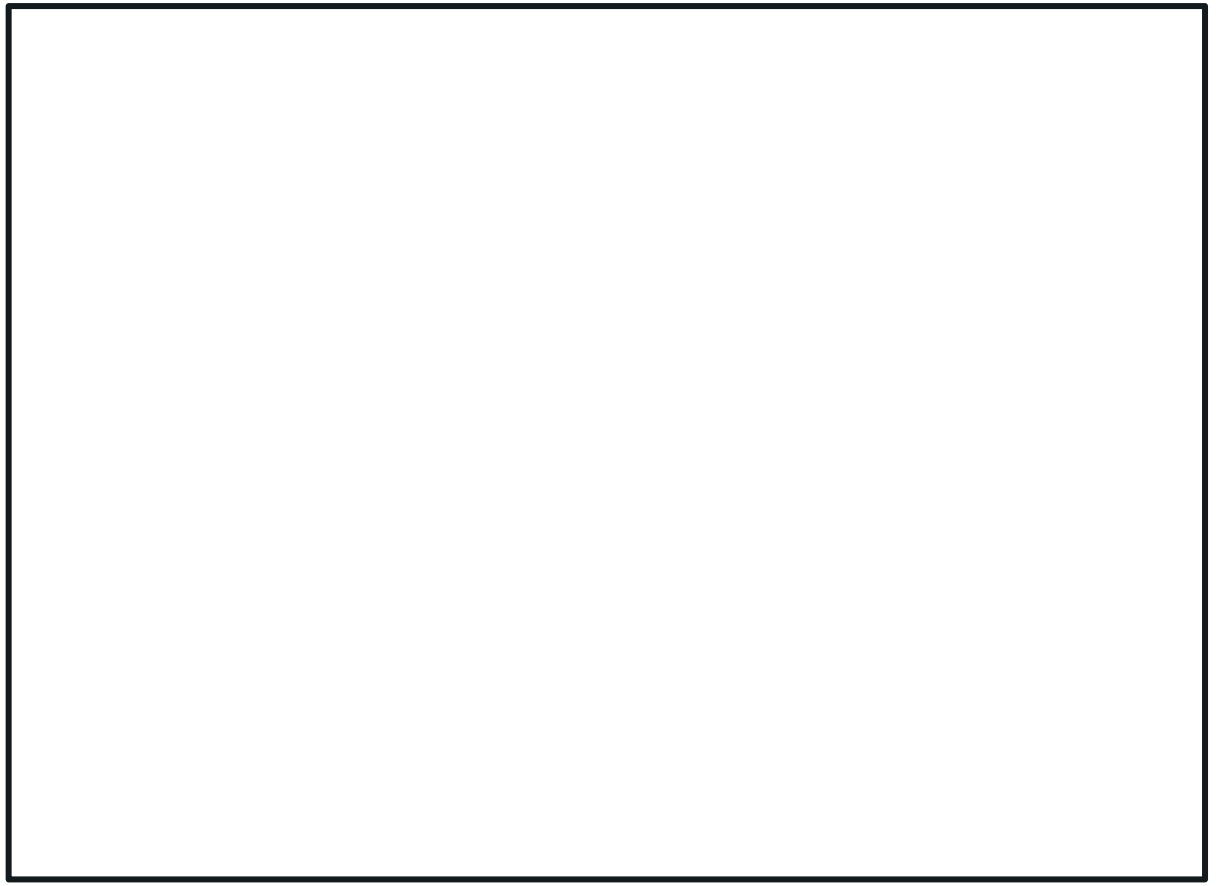
304L Exposed to Simulated Hope Creek Canister Salt – Salt Composition

- **Salt composition based on Hope Creep DPC dust sample analysis by SNL[2].**
- **"Selected Composition" adjusted to reduce concentrations of ions associated with insoluble species at relevant RH.**

	Composition, (Cl- Normalized)	
Dissolved Species	From Dust Samples	**Selected Composition**
Na+	2.7314	4.025
K+	0.6971	1.076
Ca+2	3.8095	0.067
Mg+2	2.0263	0.007
Cl-	**1.0000**	**1.000**
NO3-	**0.4537**	**0.651**
SO4-2	**1.5093**	**1.780**
HCO3-	10.6278	0.037

Deliquescence begins at 55% RH. Almost fully deliquesced at 75% RH.

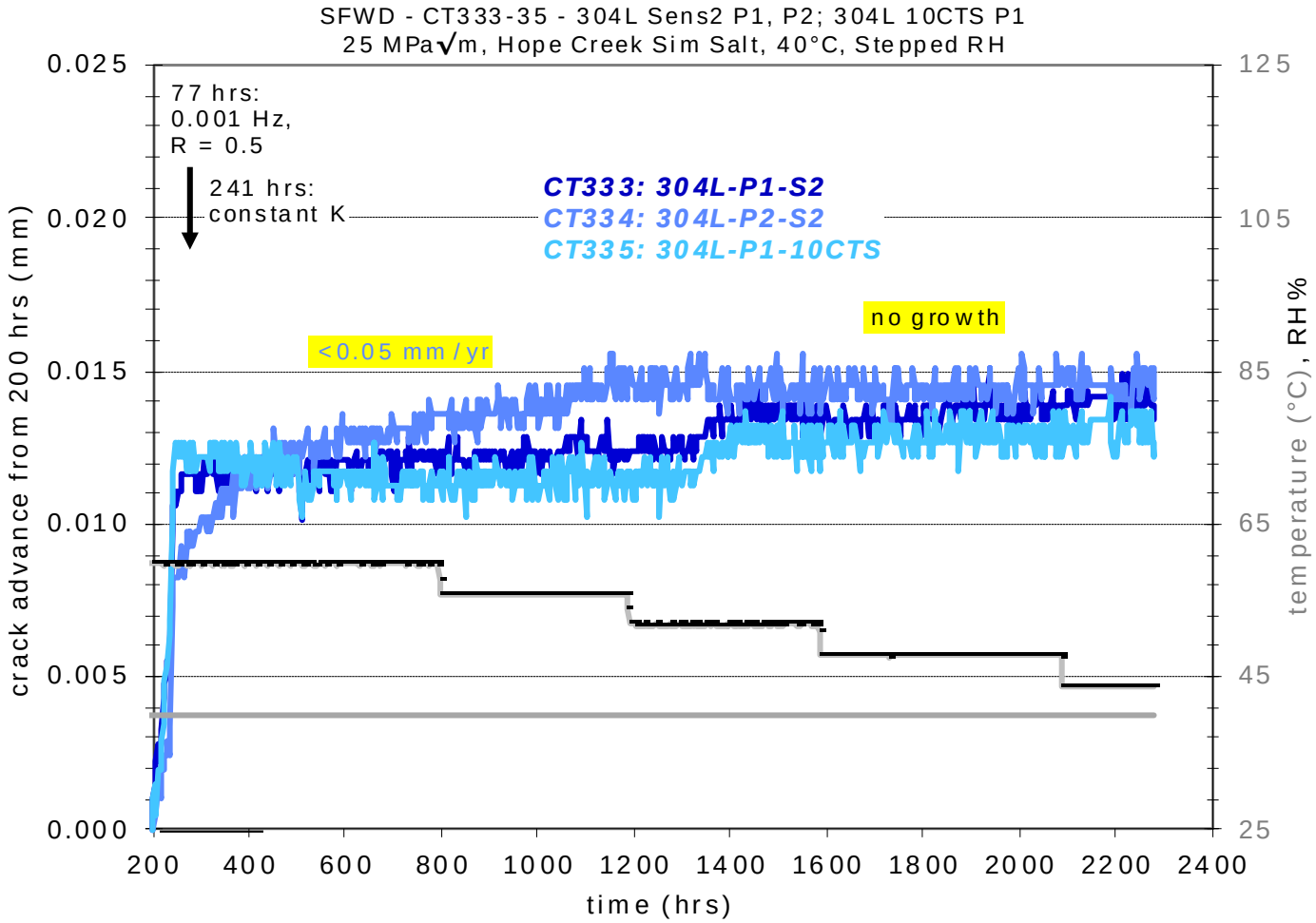
Site is 10-30 km from ocean shoreline.



[2] R. Schaller, et al., "FY20 STATUS REPORT: SNF INTERIM STORAGE CANISTER CORROSION AND SURFACE ENVIRONMENT INVESTIGATIONS", SAND20-12683 R, 2020. Available in OSTI.

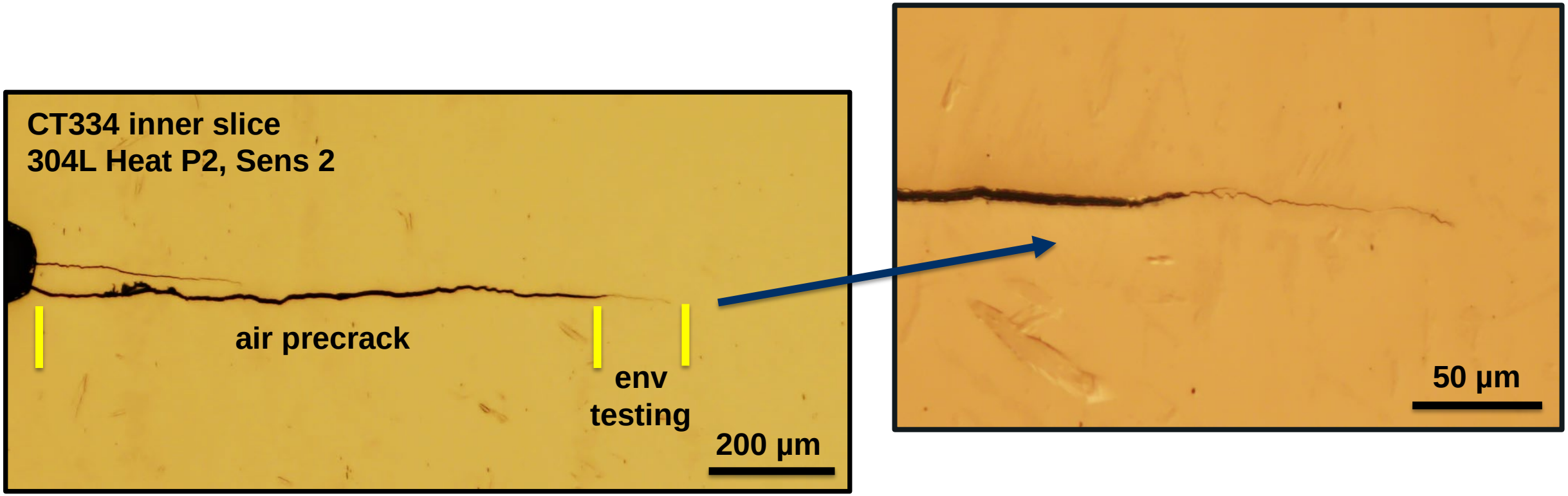
304L Exposed to Simulated Hope Creek Canister Salt – CISCC Growth Rate Response

- Deliquescence begins at 55% RH. Almost fully deliquesced at 75% RH.
- Test started at 60% RH.
  - Step down by 4% RH every 350–500 hours.
  - Final RH of 44%
- Very low or no CISCC advance.



304L Exposed to Simulated Hope Creek Canister Salt – Post-test Crack Morphology

- Typical crack morphology shown.
- Environmental crack advance due to corrosion fatigue steps.



Summary of Near-Marine-Relevant 304L Testing

- **NaCl is the dominant chloride salt at all geographic locations.**
- **304L is not susceptible to CISCC when exposed to NaCl at canister-relevant conditions.**
- **Near-marine ISFSI dust sampling indicates low $MgCl_2$.**
- **Small additions of nitrate and sulfate to NaCl did not induce CISCC susceptibility.**
- **Simulated Hope Creek and Maine Yankee salts (large amounts of nitrate and sulfate) did not induce CISCC susceptibility.**
- **For 304L, no effect of sensitization treatment or small amounts of cold work on CISCC behavior.**
- **Marine-relevant CISCC testing planned in FY26.**

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Spent Fuel and High-Level Waste Disposition

Backup Slides

Kennebec Estuary (Maine Yankee area) River Salinity

- **Estuary water salinity is 50% of ocean water in vicinity of Maine Yankee ISFSI.**

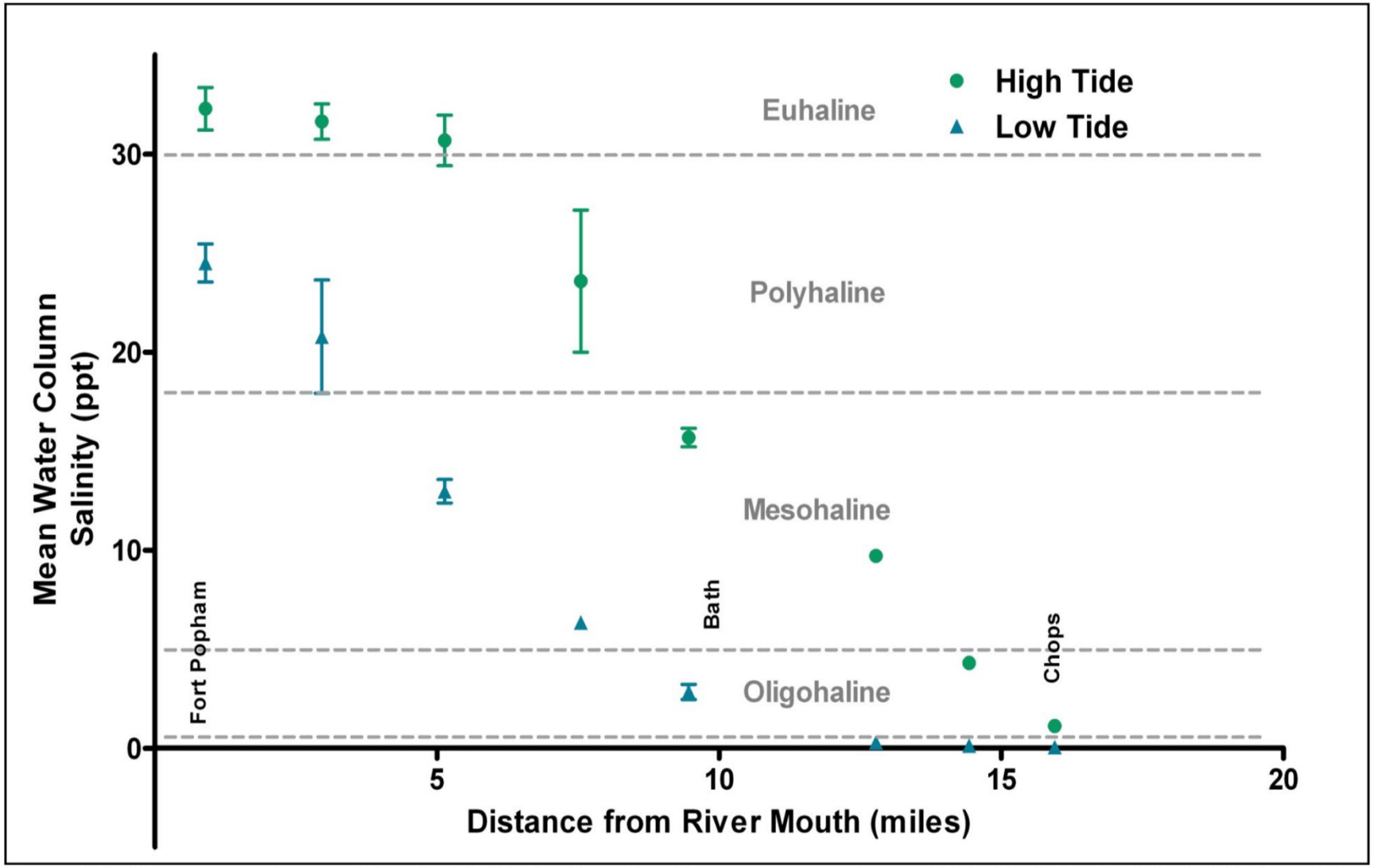
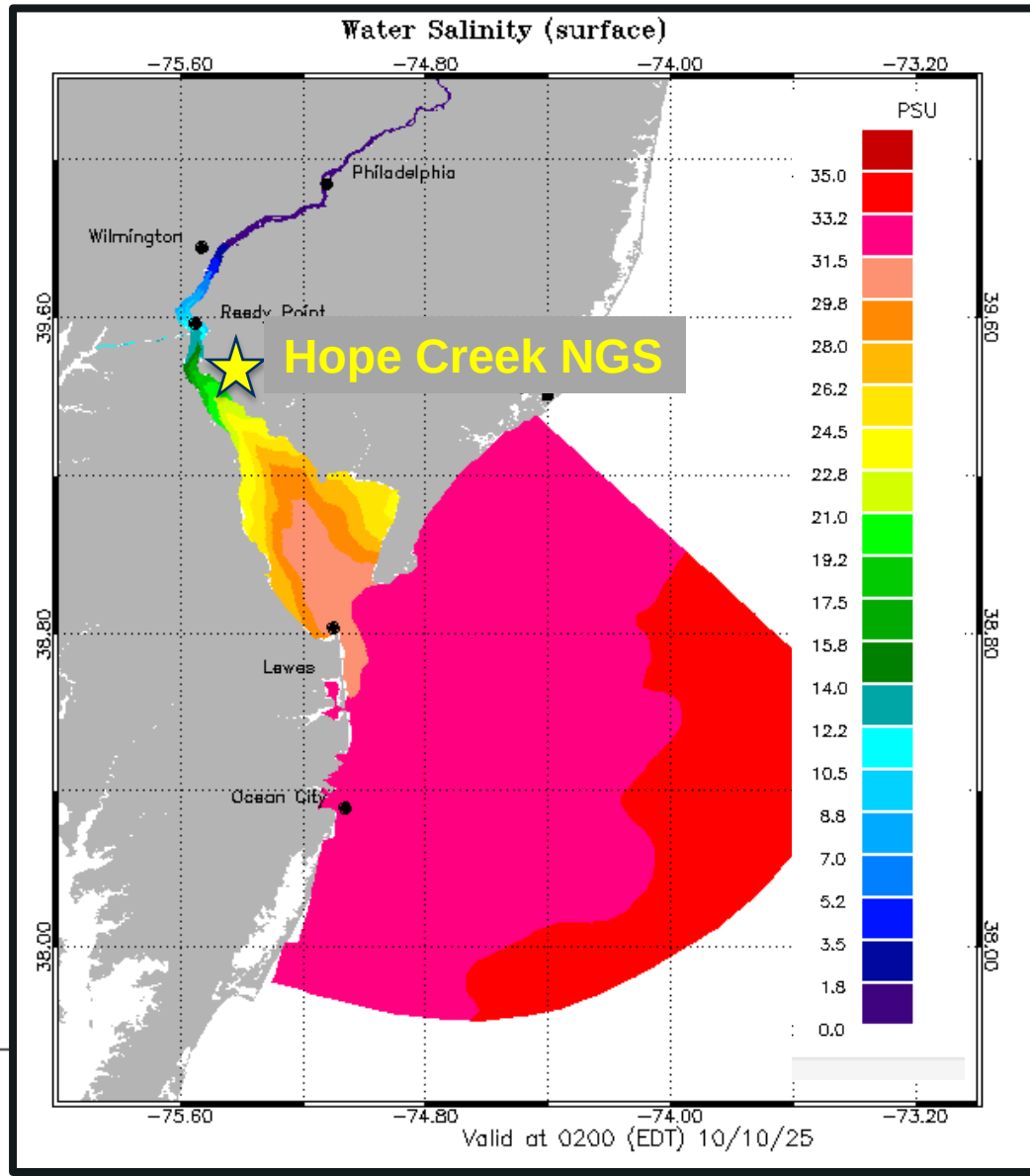


Figure 1-2. Mean water column salinities (± 1 SD) in the Kennebec Estuary at or near low and high tide during the fall of 2007. Boundaries between salinity zones are delineated according to Odum (1988). Adapted from J. Reblin and J. Lichter (unpublished data).

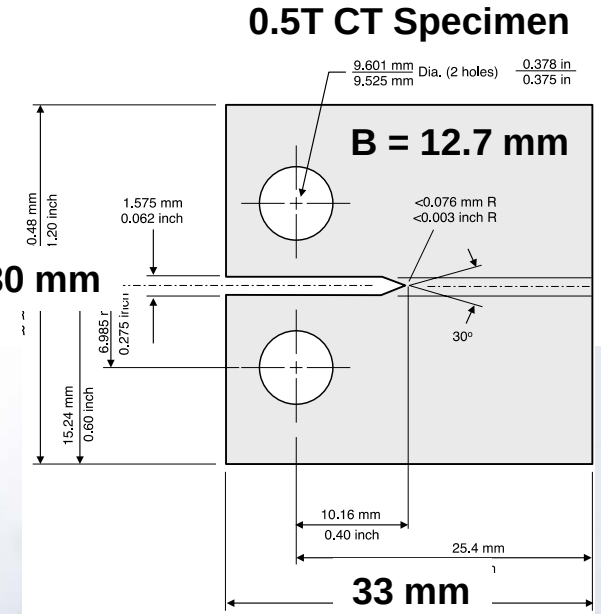
Delaware Bay Salinity

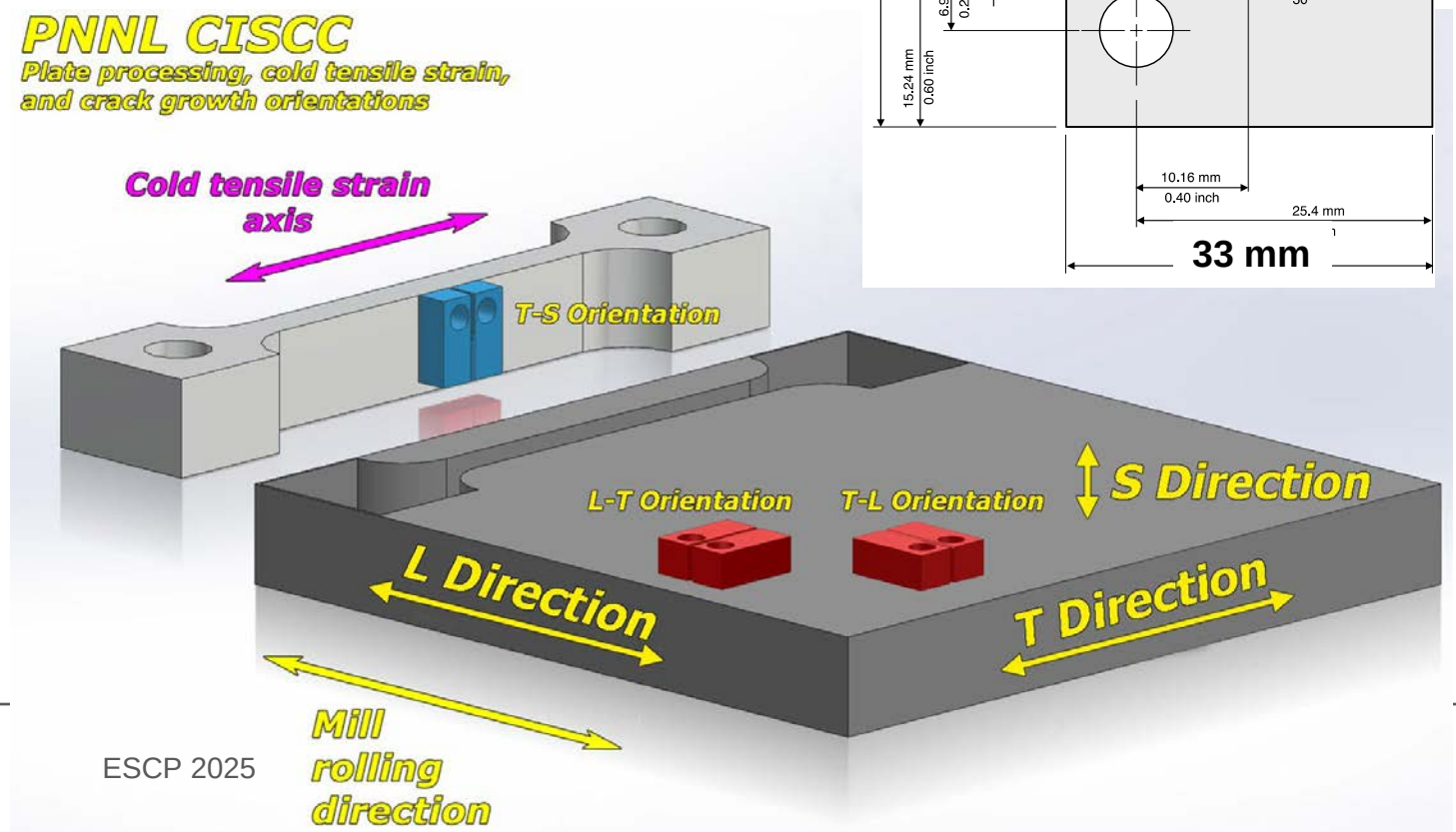
- **River water salinity is 40% of ocean water in vicinity of Hope Creek Nuclear Generating Station (NGS).**



Specimens

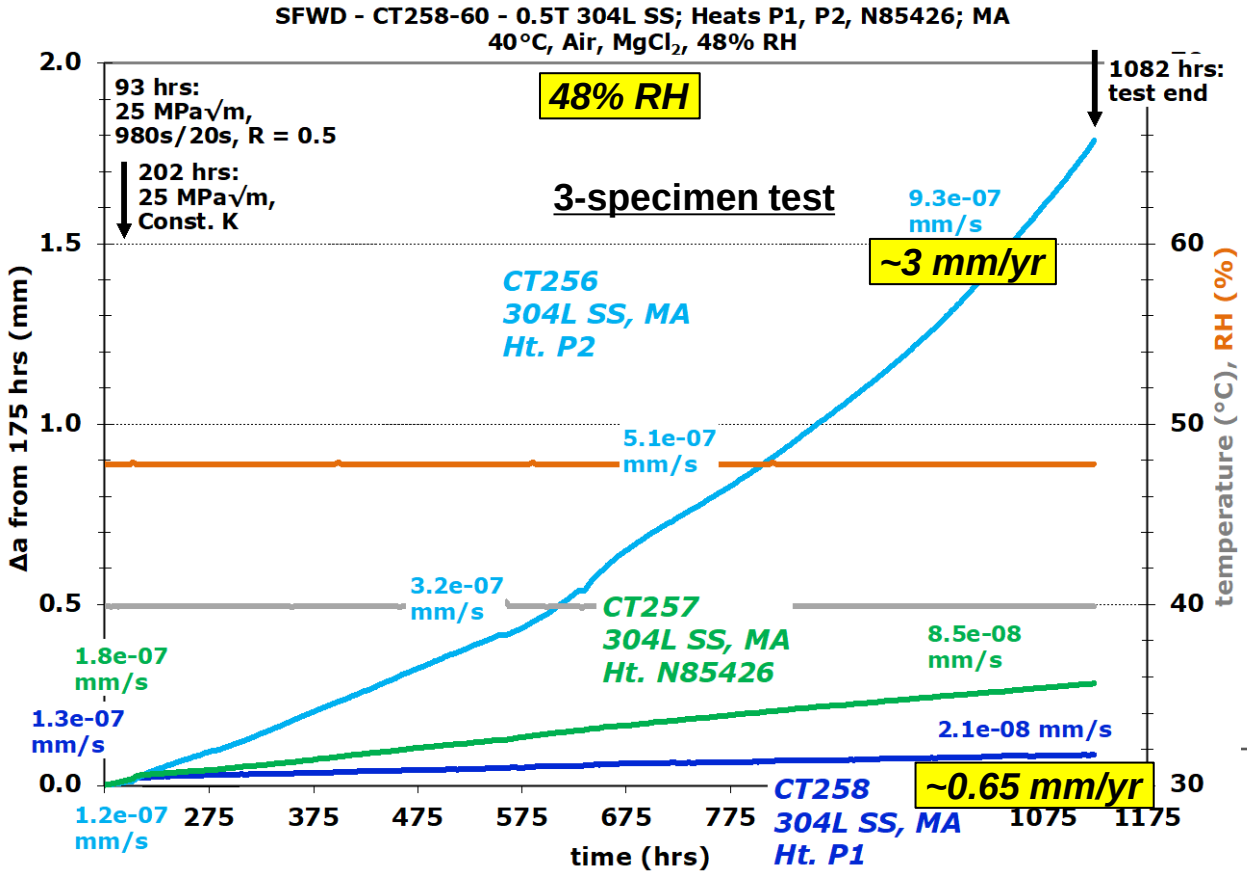
- Side-grooved 0.5T CT specimens used for these tests.
- Specimens extracted from plate material.
  - Most specimens are in a T-S orientation orientation.
    - Realistic orientation for through-wall cracking of plate material.
- Side-grooves sealed with silicone sealant prior to testing to create a more realistic crack environment.



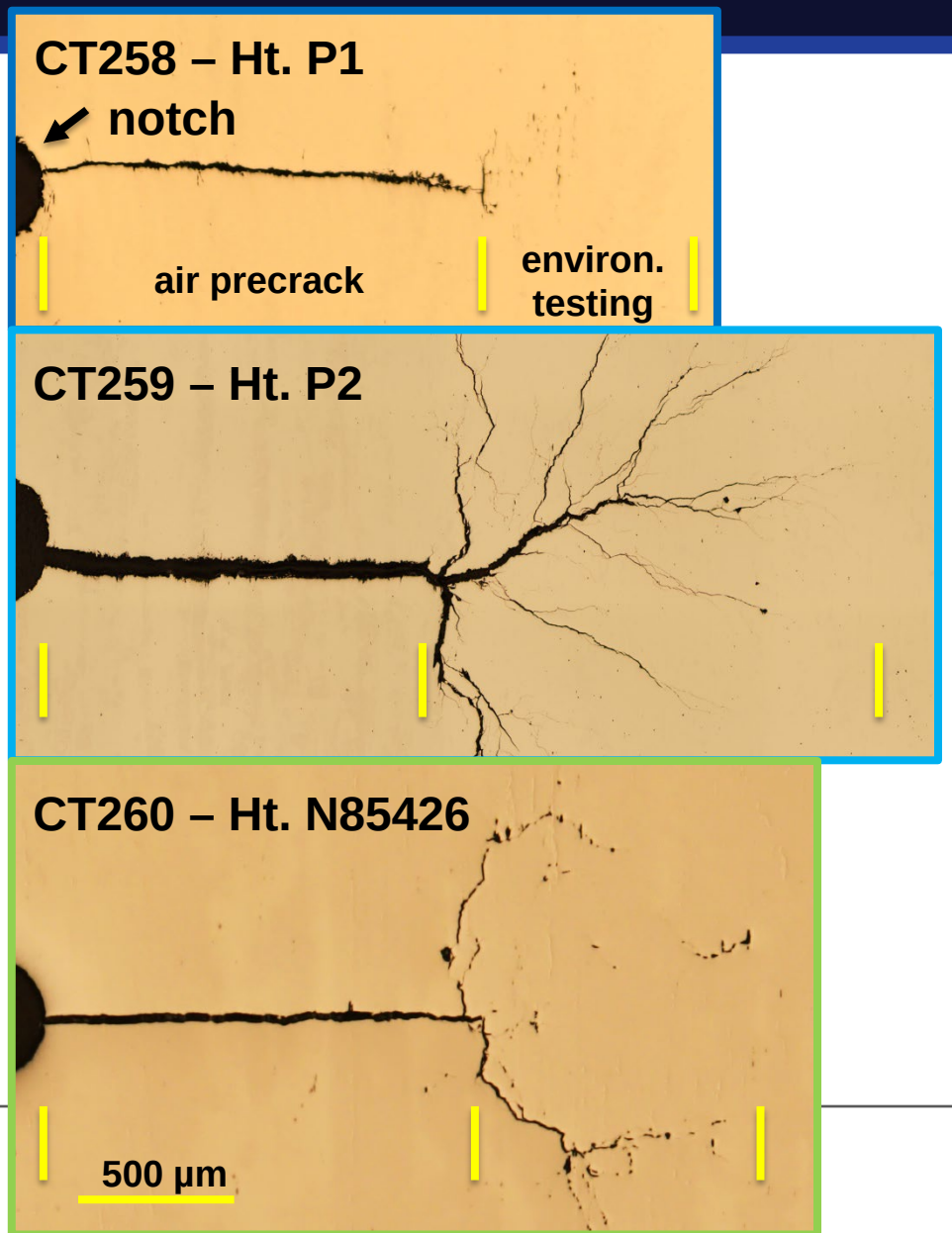


304L Exposed to $MgCl_2$

- 3-specimen test – 1x each of heat P1, P2, N8.
- Steady, moderate-to-high CISCC growth rates (CISCCGRs) observed for all three heats of 304L.
- Multiple tests using $MgCl_2$ have been conducted with the same result each time.
- Susceptibility is consistent with the literature on the effect of $MgCl_2$ on CISCC of many 300-series stainless steels.
- Effect of heat-to-heat composition variation on CISCC.
  - In this test and in others, heats P2 and N8 exhibit high CISCCGRs.
    - These have higher S, and lower Mo and Cu.



304L Exposed to $MgCl_2$



- Post-test cross section optical image of each specimen shown.

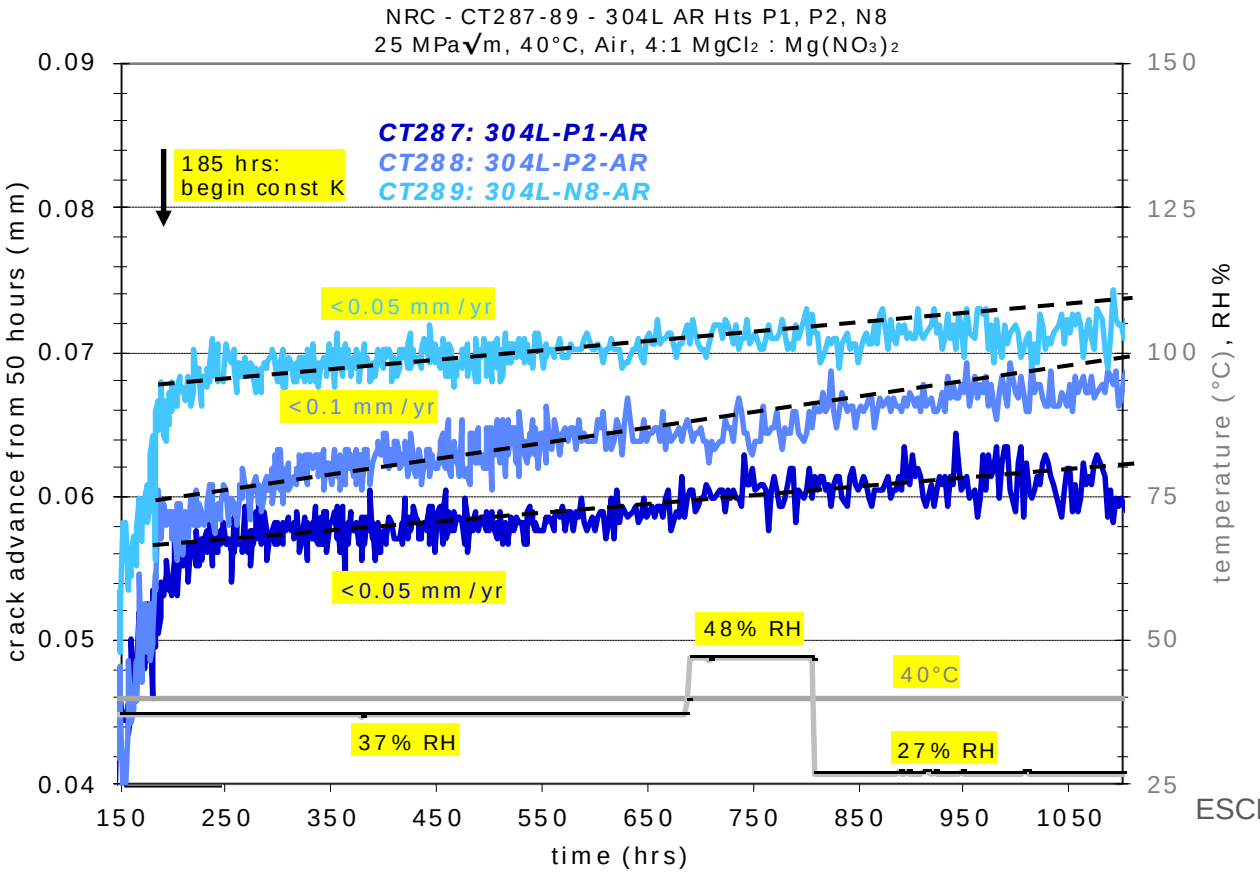
- Transgranular (TG) cracking.
- Sharp, branched cracks.
- Heat P1 specimen (ultra-low S, elevated Mo and Cu) exhibited the least amount of cracking and lowest CISCCGR.
- Heat P2 specimen exhibited the most extensive cracking and highest CISCCGR.

- Cracking morphology is consistent with the literature on $MgCl_2$ CISCC.

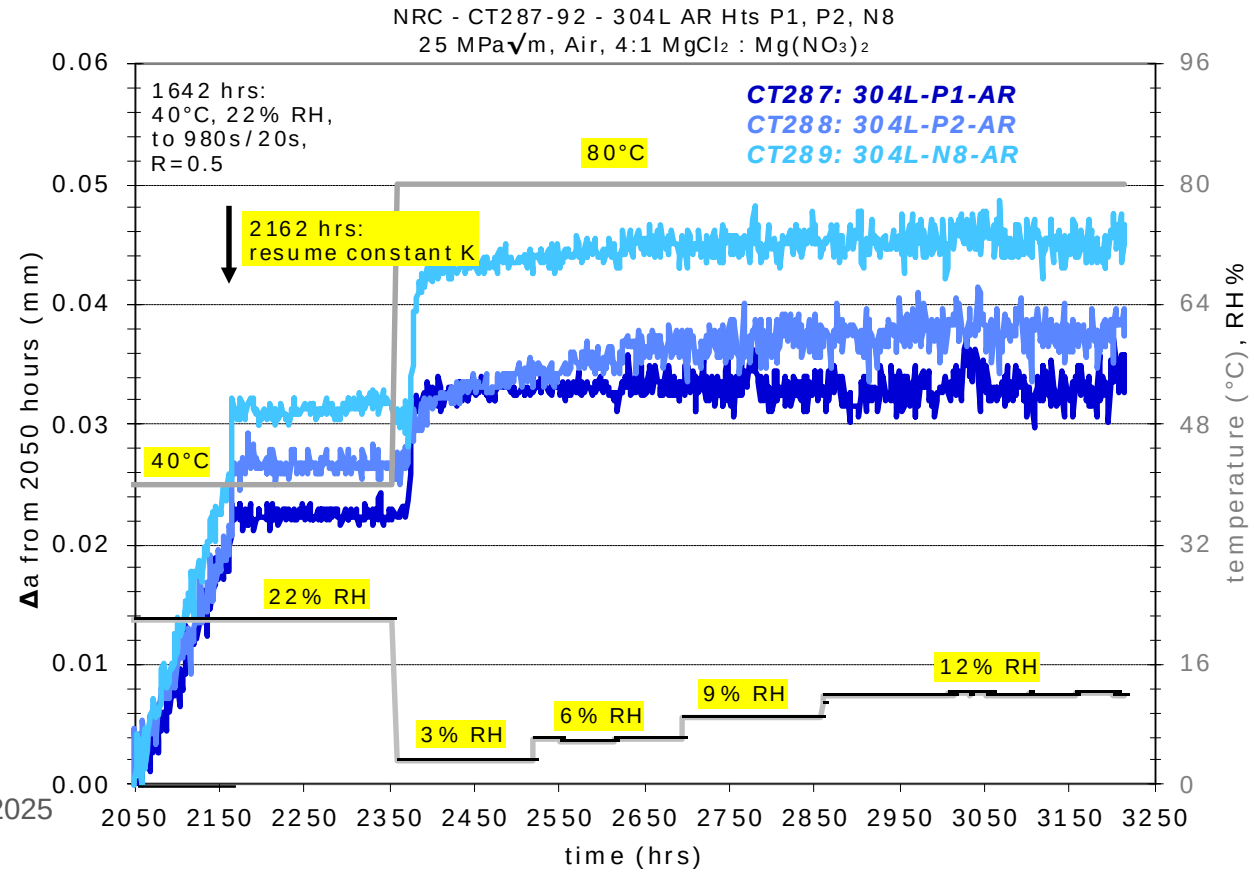
304L Exposed to 4:1 $MgCl_2$: $Mg(NO_3)_2$

- At 40C, deliquescence at 20% RH. Low / no growth at 27, 37, 48% RH.
- At 80C, deliquescence at ~0% RH. Low / no growth at 3, 6, 9, 12% RH.

40C - 37, 48, 27% RH

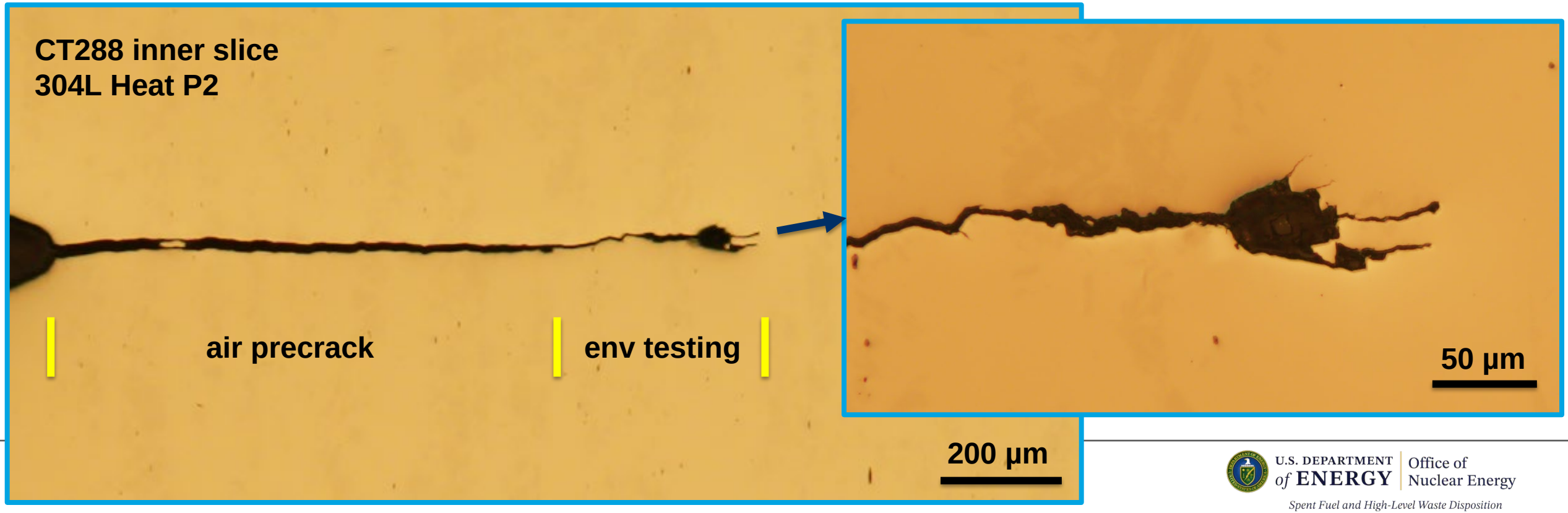


40C - 22% RH 80C - 3, 6, 9, 12% RH



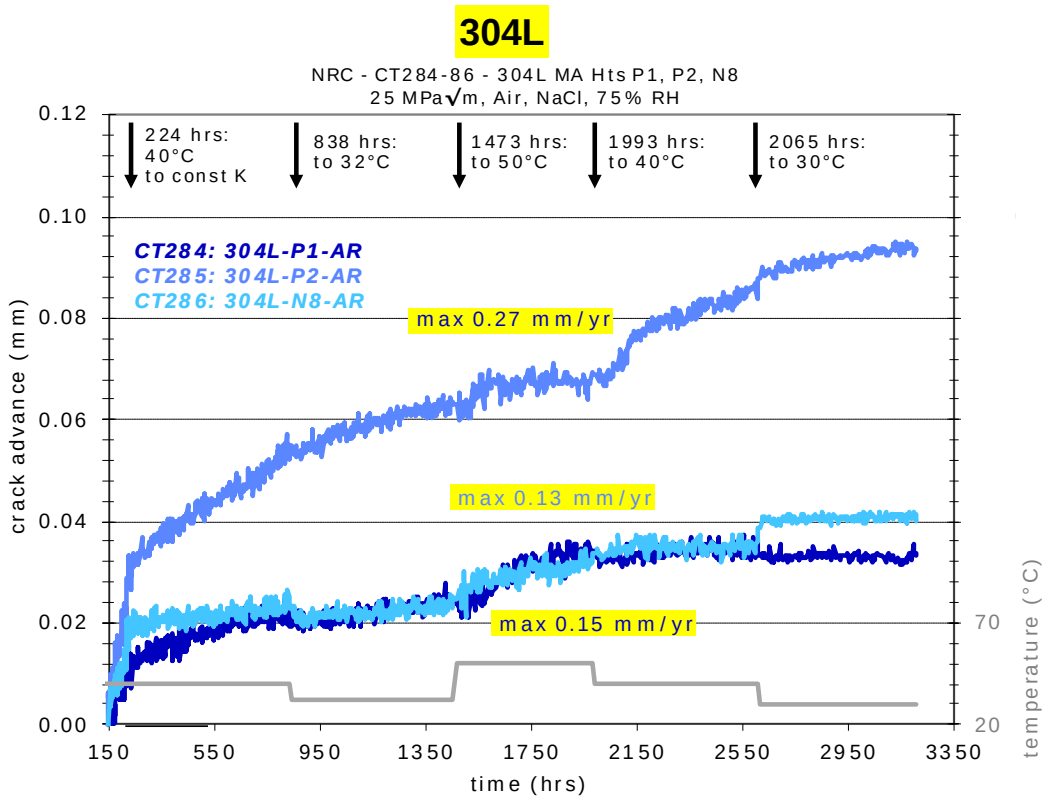
304L Exposed to 4:1 $MgCl_2$: $Mg(NO_3)_2$

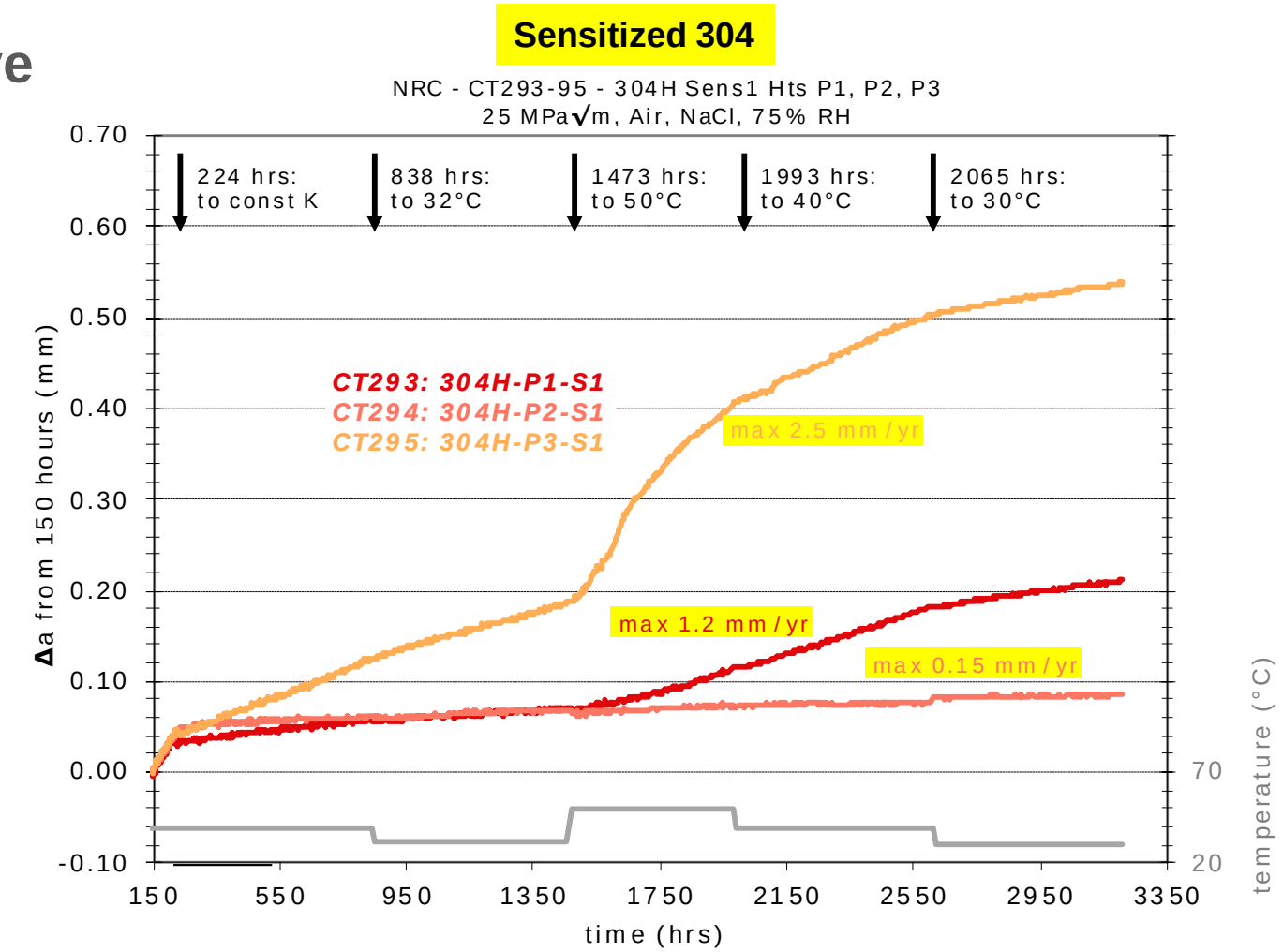
- Typical crack morphology.
- No CISCC growth.
- Short, sharp branch cracks likely due to cyclic loading applied at ~2000 hours.



Sensitized 304 as a Reference Material.

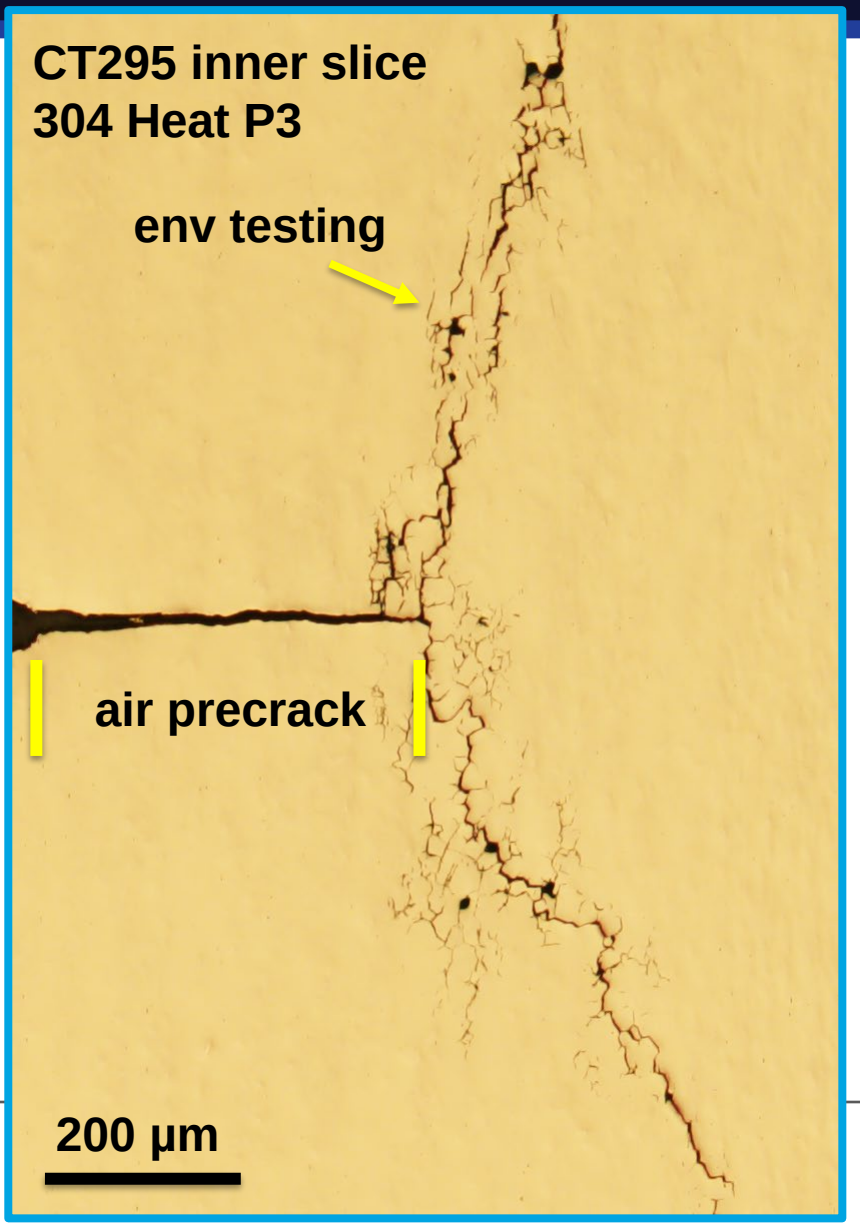
- **Sensitized 304 found to be susceptible to CISCC for all salts evaluated.**
- **Most tests include sensitized 304 specimens to serve as proof of effective specimen preparation.**





Sensitized 304 Exposed to NaCl

- Typical crack morphology of sensitized 304 shown.
- Intergranular (IG) cracking confirmed by SEM.
- Crack is following microstructure inhomogeneity.
  - Responsible microstructure under investigation.



PNNL CISCC Testing Laboratory

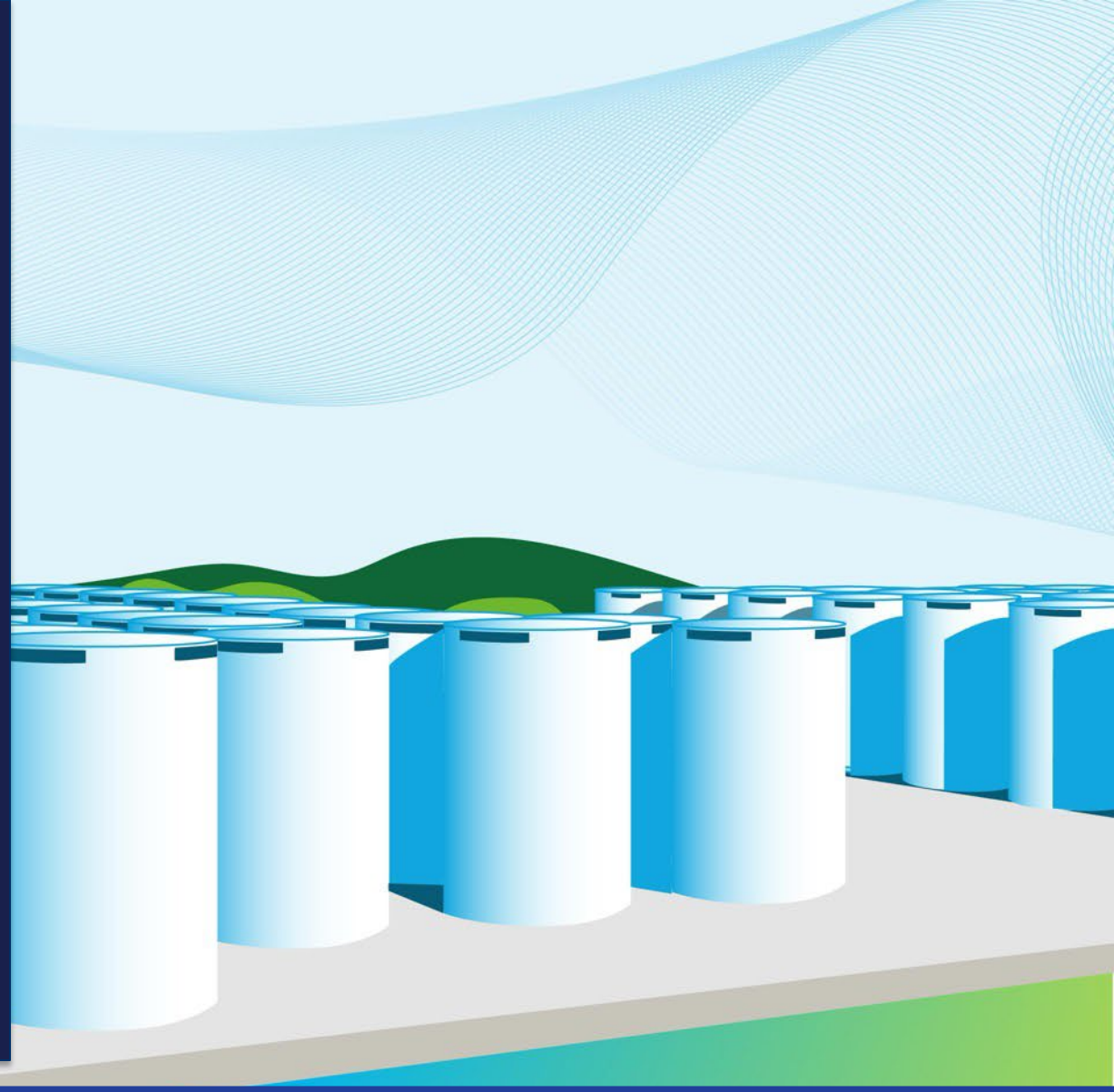
- Five CISCC test systems.
- Capable of testing in concentrated salt water or controlled humidity conditions.

Evaluation of Potential CISCC Locations and Morphologies in Dry Storage Canisters

Nathan Bailey, David Richmond,
Mychailo Toloczko (presenting)
Pacific Northwest National Laboratory

2025 EPRI ESCP Meeting
October 27-30, Charlotte, NC

PNNL-SA-217252



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Spent Fuel and High-Level Waste Disposition

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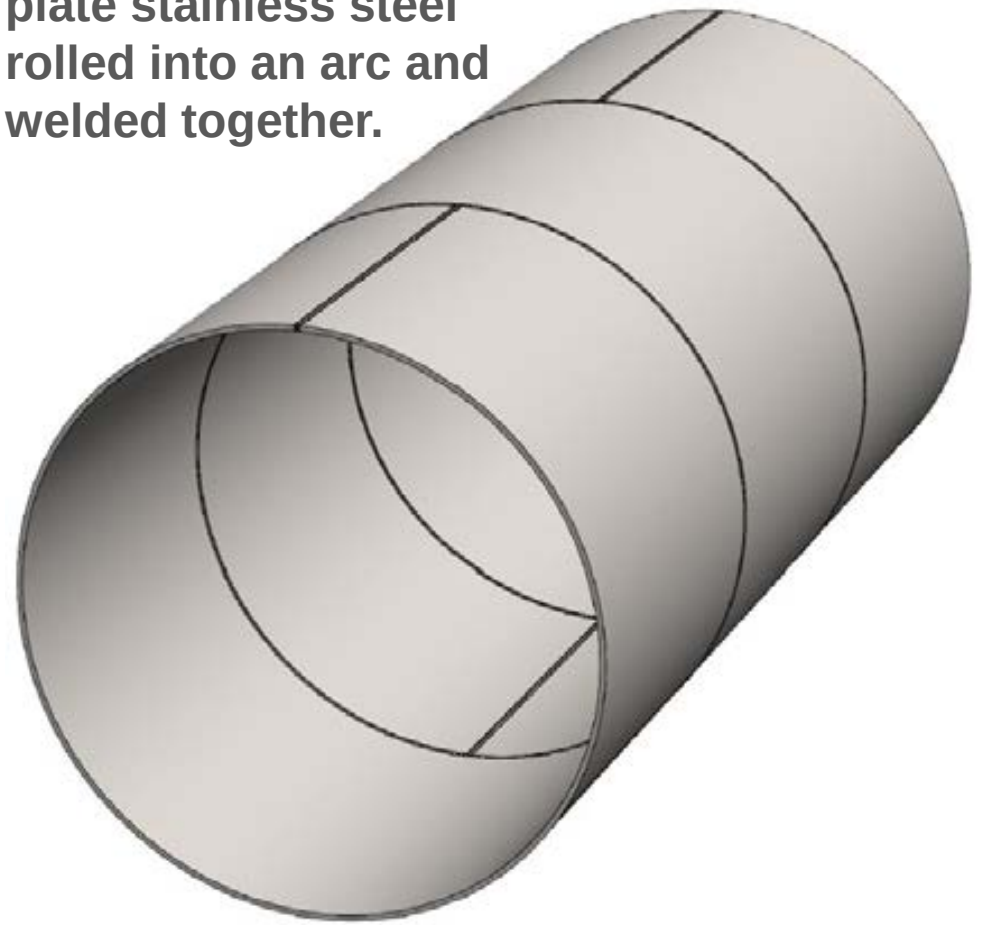
Overview

- Document the current understanding of factors that would affect the location and morphology of potential CISCC in dual purpose canisters (DPCs).*
- Attempt to summarize all factors.
- Informs on:
  - Visual inspection locations.
  - Placement of non-destructive evaluation (NDE) equipment.
  - Consequence analysis.
  - DPC design.
- Many of the factors are generally understood and covered in bits and pieces across the literature.
- Assembling this information in one place to assist in current and future DOE-NE dry storage system management.

DPCs are made from plate stainless steel rolled into an arc and welded together.



* This task is not meant to indicate a strong concern for CISCC or discuss the probability of CISCC. Current research indicates that there is a very low probability at inland sites, and research is in-progress to assess CISCC probability at other geographic locations.

CISCC Susceptibility Factors

Canister and Overpack Characteristics

1. Orientation
2. Weld orientation
3. Construction method
4. Overpack design

Environmental

1. Chloride loading
2. Salt type
3. Temperature
4. Relative humidity

Material

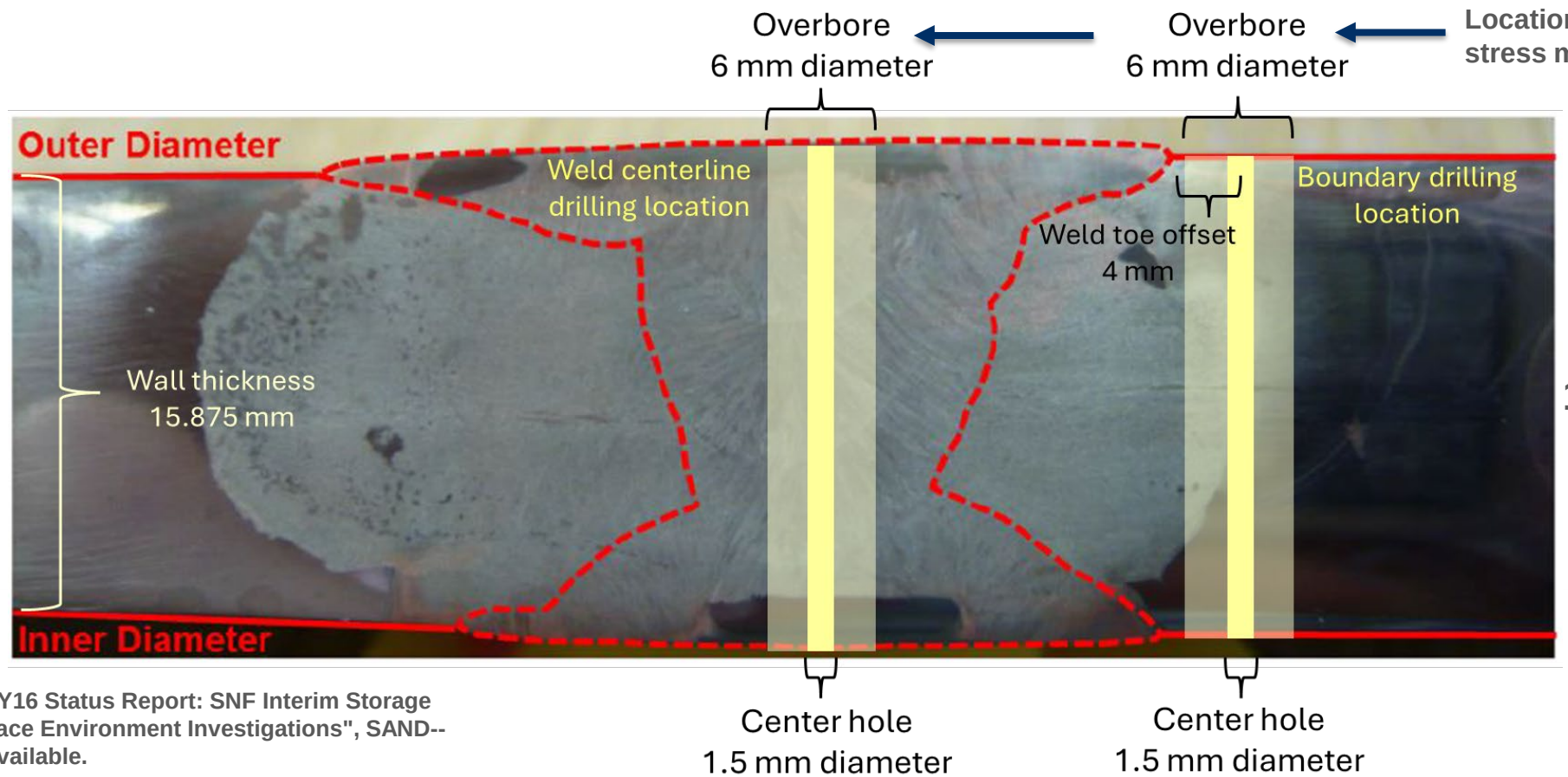
1. Composition
2. Sensitization
3. Degree of cold work

These factors affect potential crack location and morphology.

SNL Mockup Canister Weld in Cross Section

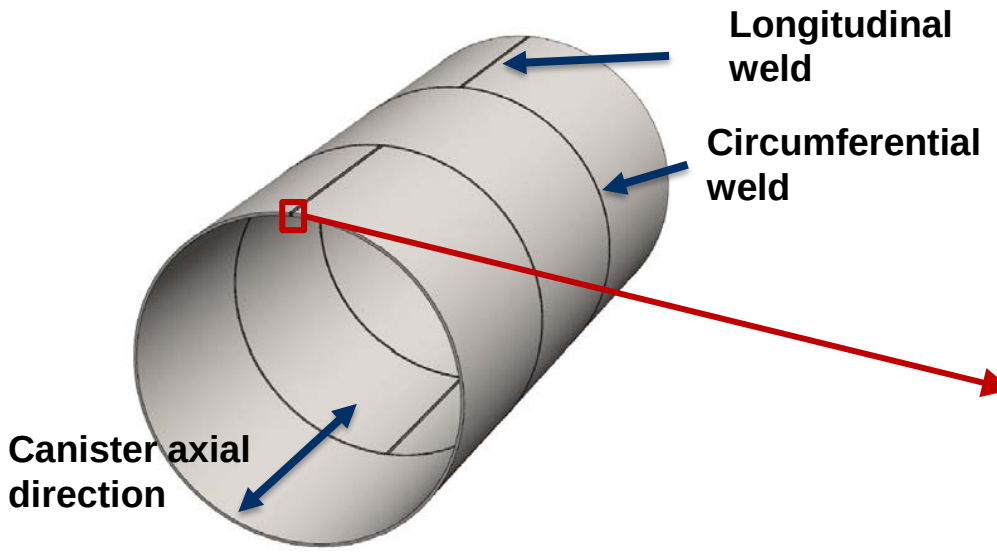
- This weld is representative of typical DPC longitudinal and circumferential welds.
- 5/8" (16 mm) thick plate.
- Double bevel geometry - common, if not ubiquitous for canister shell construction.

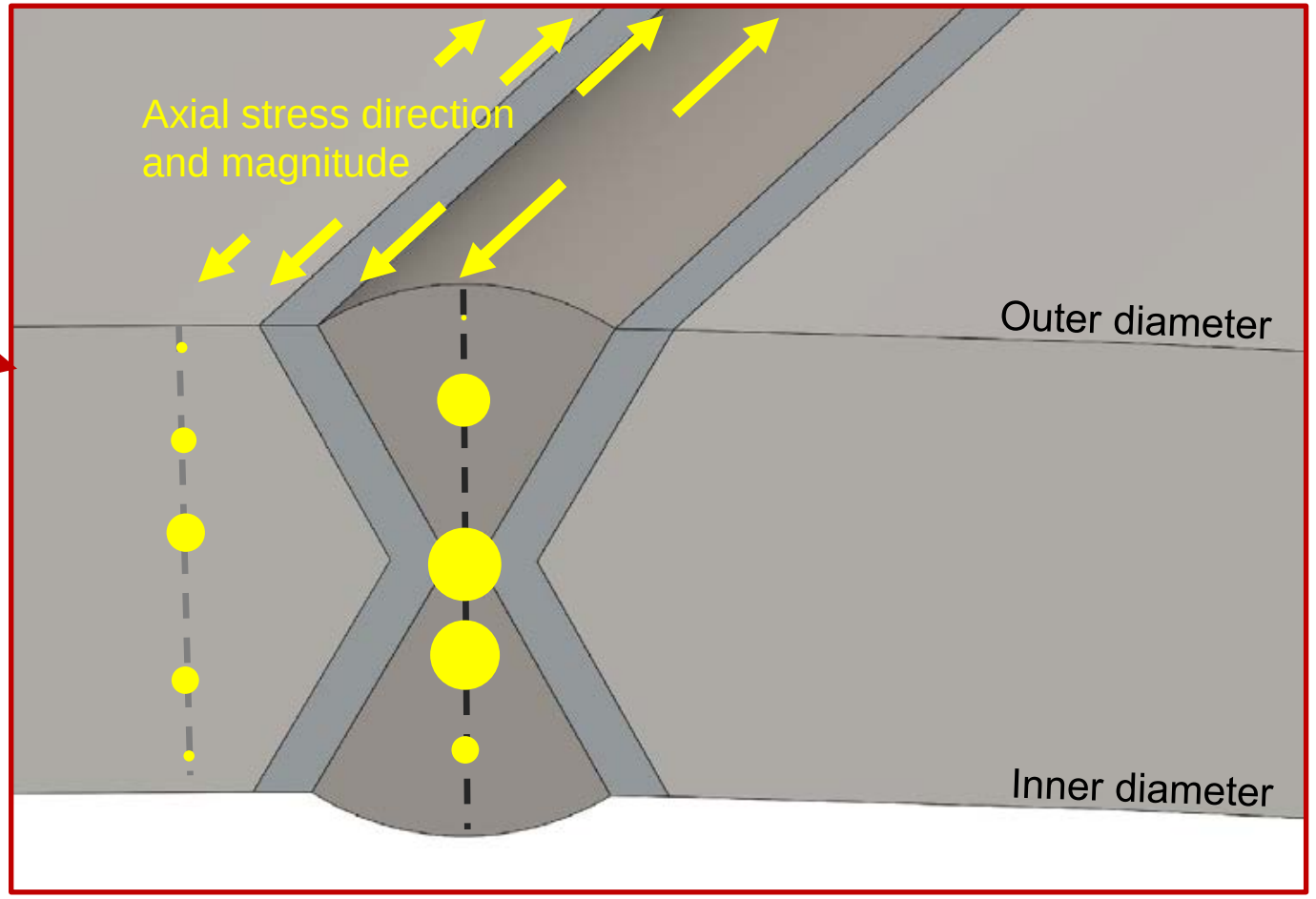
SNL-funded canister weld mockup [*]



* C.R. Bryan and D. Enos, "FY16 Status Report: SNF Interim Storage Canister Corrosion and Surface Environment Investigations", SAND--2017-2498R, 2017, Publicly available.

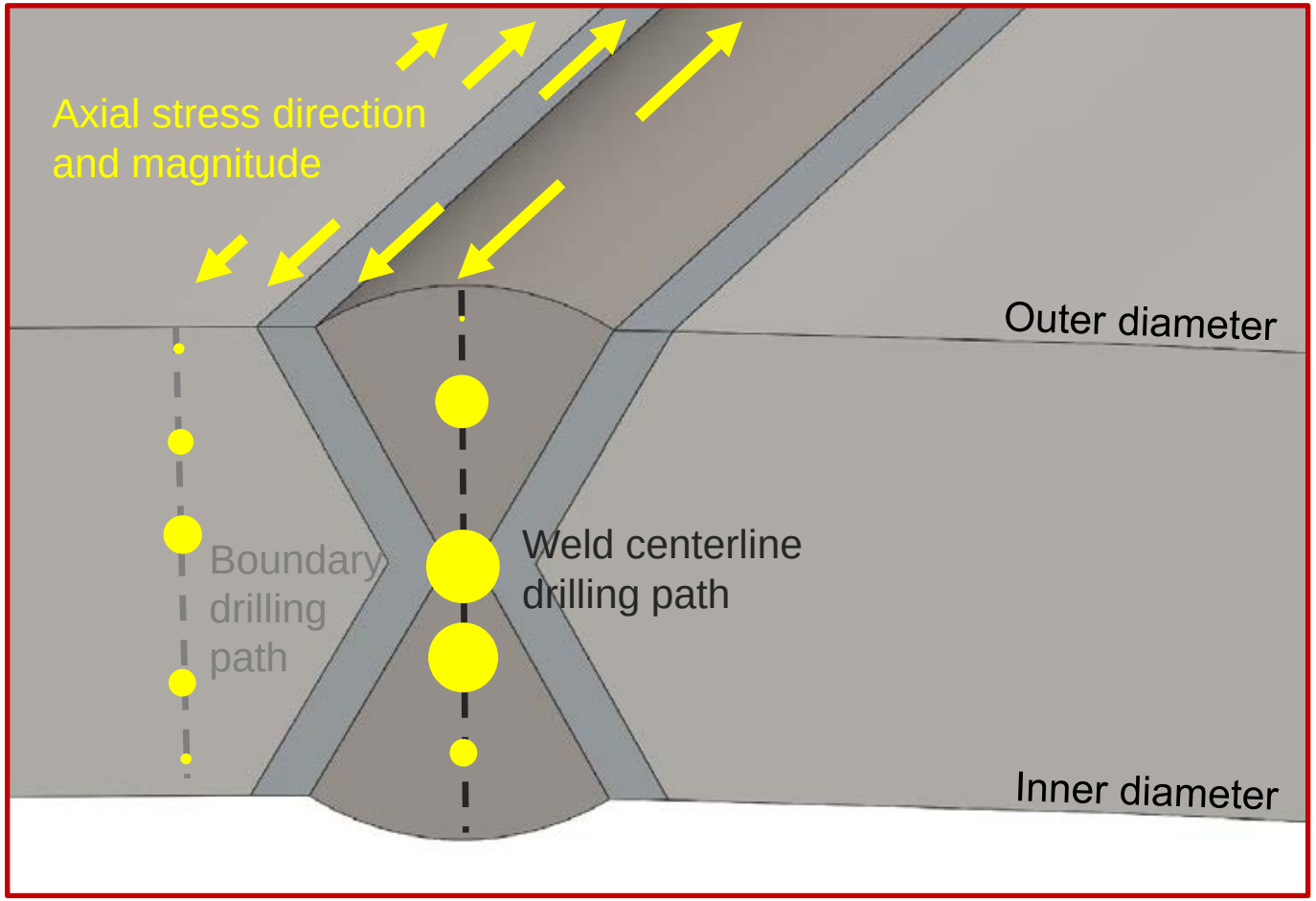
Through-Wall Tensile Stresses Only Near Welds



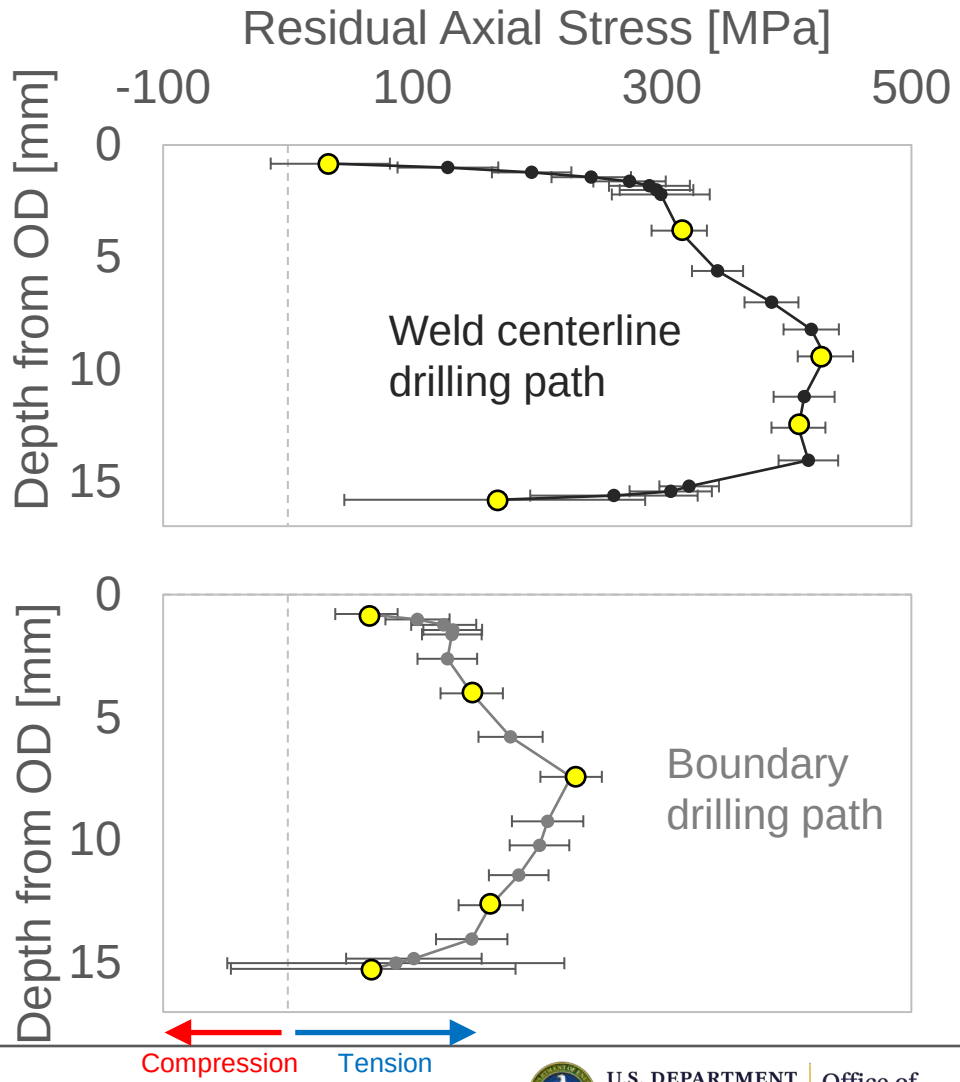


- Example is for longitudinal seam welds in canisters.
- Circumferential seam welds similar but lower peak stresses.
- Plate bending stresses are compressive at inner diameter for at locations away from welds.

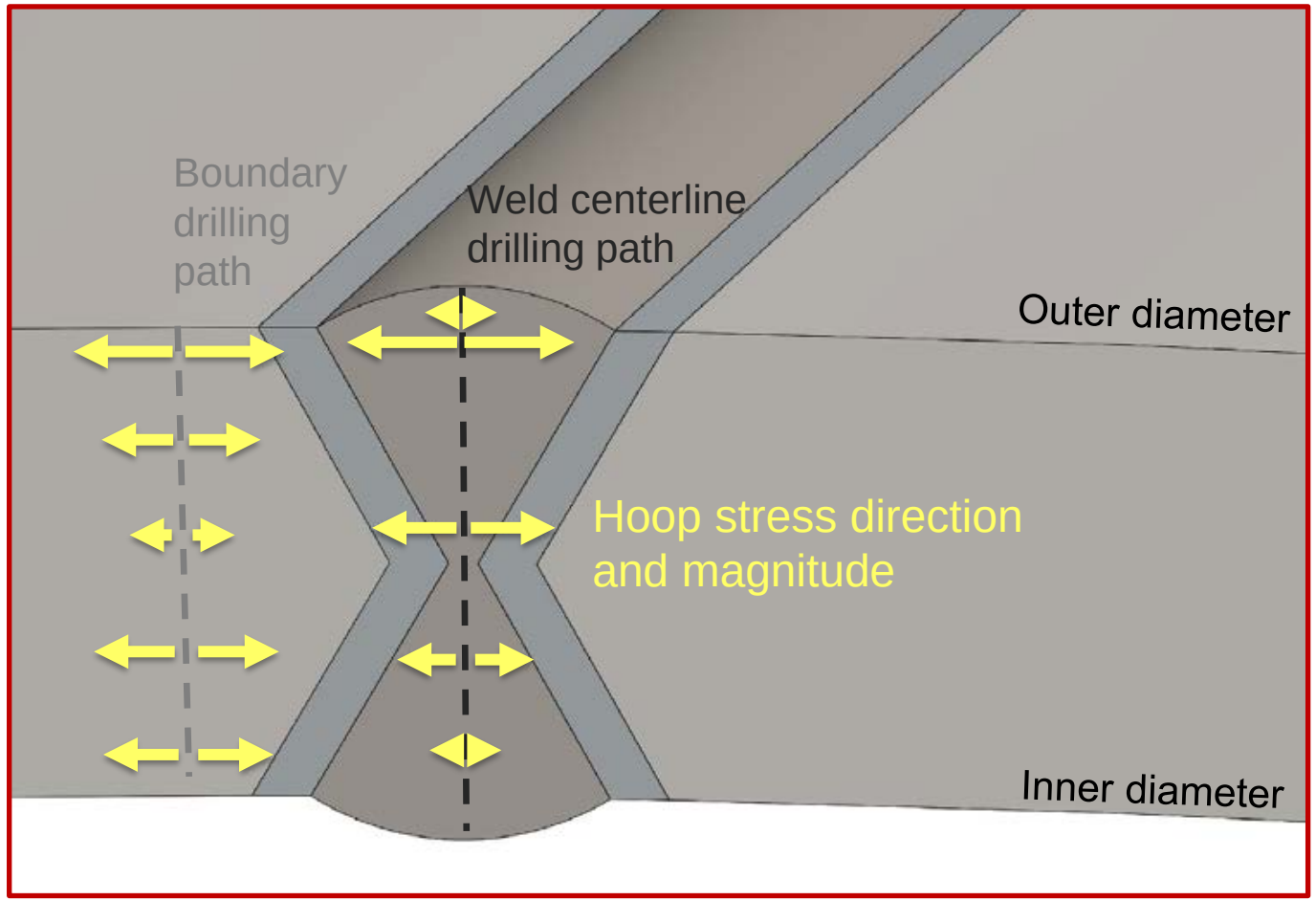
Axial Stresses at Longitudinal Welds

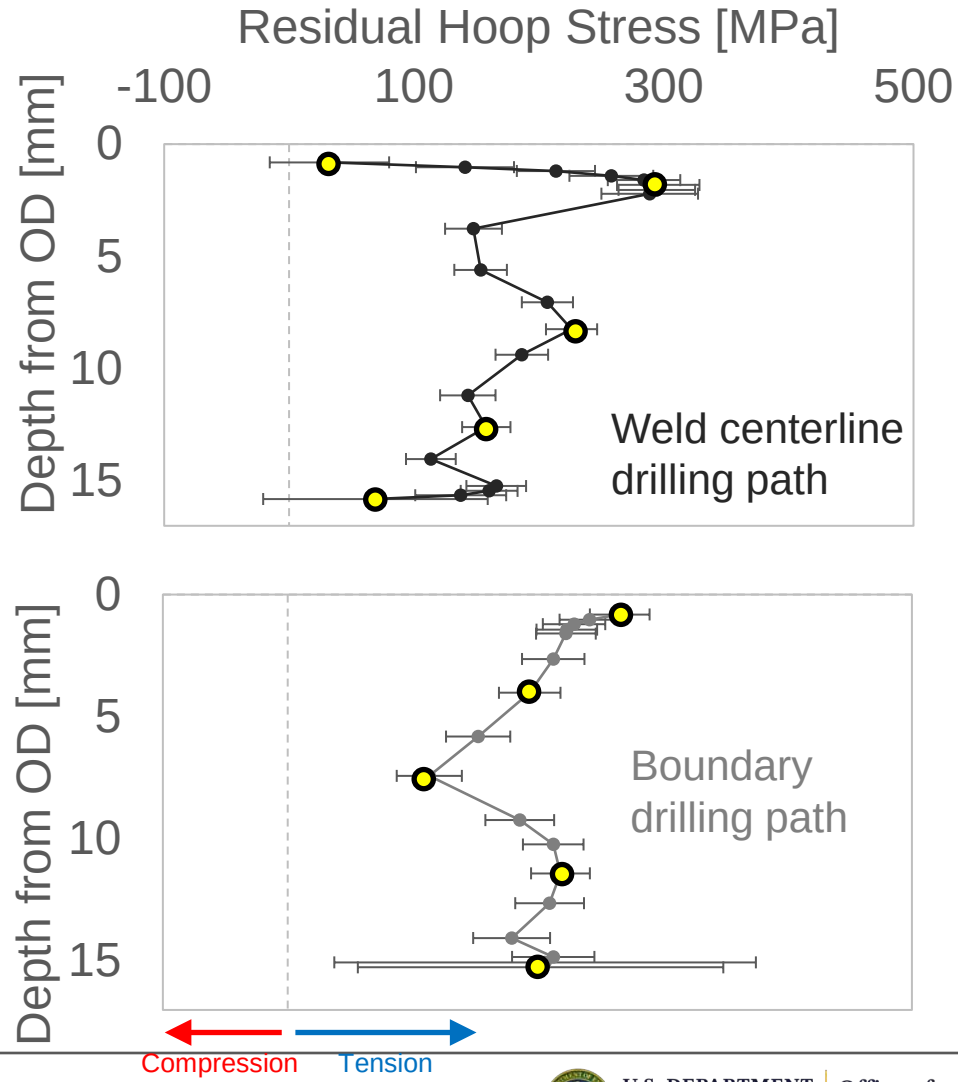


Data from: C. Bryan, D. Enos, FY16 Status Report on Canisters, SNL, FCRD-UFD-2016-000429.



Hoop Stresses at Longitudinal Welds



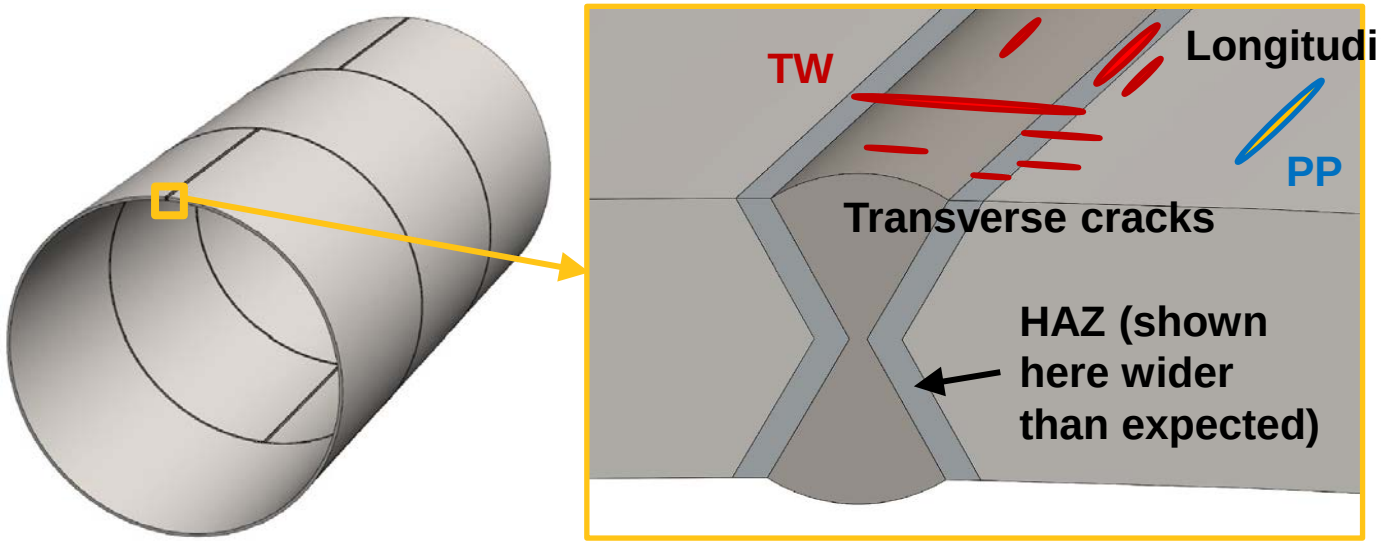


Data from: C. Bryan, D. Enos, FY16 Status Report on Canisters, SNL, FCRD-UFD-2016-000429.

Material Factors

- Alloy type:
  - 304L plate is most prevalently used.
  - Corresponding weld metals are 308L and 309L. Differences in composition may have some effect on corrosion and stress corrosion susceptibility.
- Alloy thermo-mechanical condition:
  - Significant plastic deformation increases CISCC susceptibility – plate bending (very small), product handling, weld residual strains.
  - Welding-induced thermal exposure can increase CISCC susceptibility in non-L grade stainless where carbon level is > 0.03 wt%.
- Primary factor here is weld residual strains which could increase the through-wall CISCC susceptibility in the vicinity of welds.
- Weld metal CISCC susceptibility in relevant environments not well-studied. Could be more or less susceptible than surrounding plate material.

Potential CISCC Crack Orientation



Typical double-bevel weld with indicated heat-affected zone (HAZ).

Through-Wall (TW) Cracks

- Longitudinal and transverse orientations possible.
- Crack location and orientation dependent on weld susceptibility relative to base metal.
- The width of transverse crack can only extend a short distance beyond the weld due to WRS distribution.

Partial Penetration (PP) Cracks

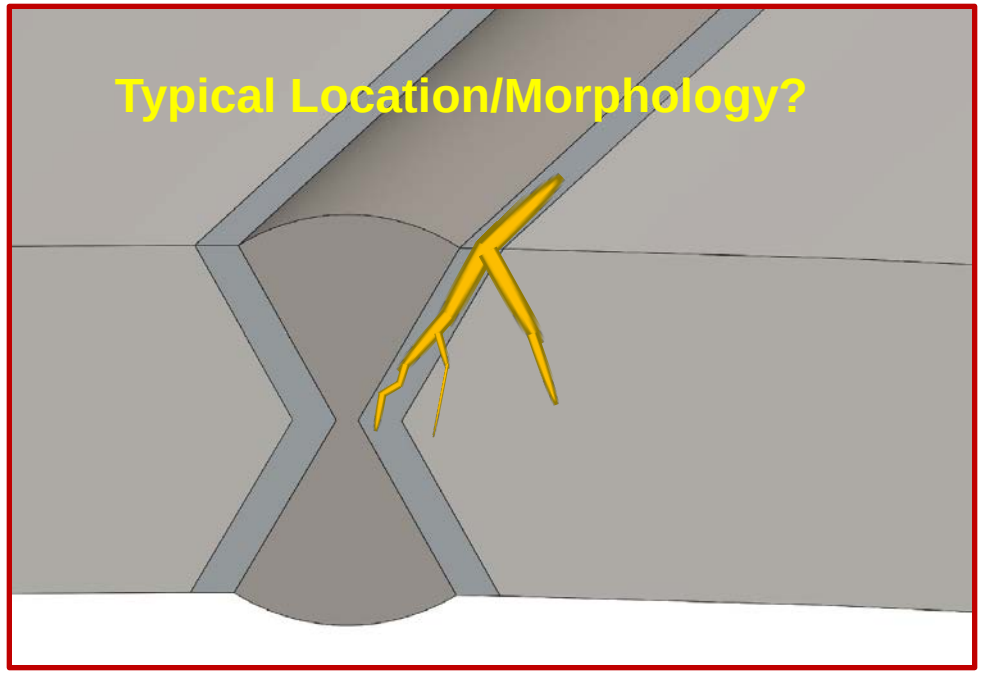
- Plate stresses away from weld are due only to bending and handling.
- Stress are not through-wall.

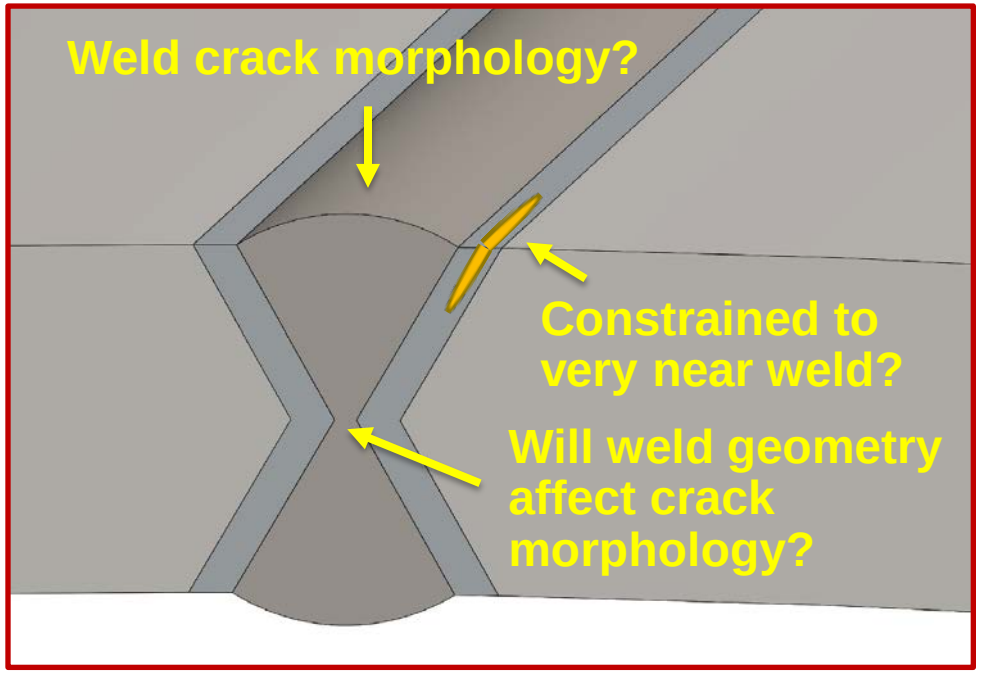
Potential CISCC Crack Morphologies (1/2)

- Affected by salt type.
- The only salts observed to cause significant CISCC of 304L in canister-relevant environments are those containing large amounts of $MgCl_2$.
  - Typical morphology - multiple primary cracks from point of initiation.

From: J.P. Basson and C. Wicker, "Environmentally induced transgranular stress-corrosion cracking of 304L stainless steel components at Koeberg", Fontevraud 5 Symposium, Fontevraud, France, 2002.

Potential CISCC Crack Morphologies (2/2)





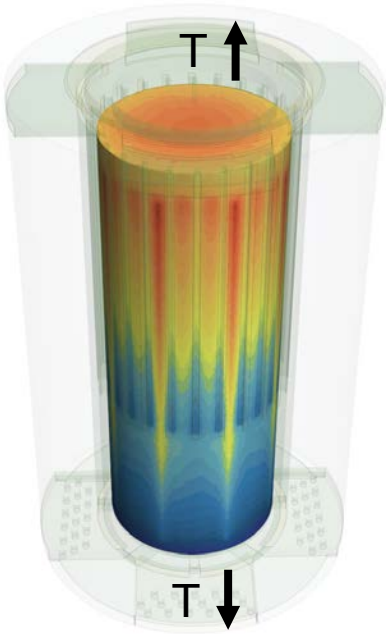
Will local microstructure affect crack morphology?

Material type and material condition may affect crack morphology.

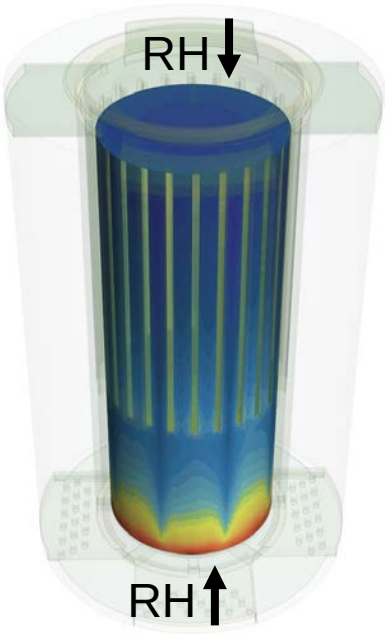
Environmental Factors - Vertical Canister Orientation

Variations in canister temperature and salt load factor into susceptible locations on canisters.

Magnastor TSC-37



Surface temperature



Relative humidity

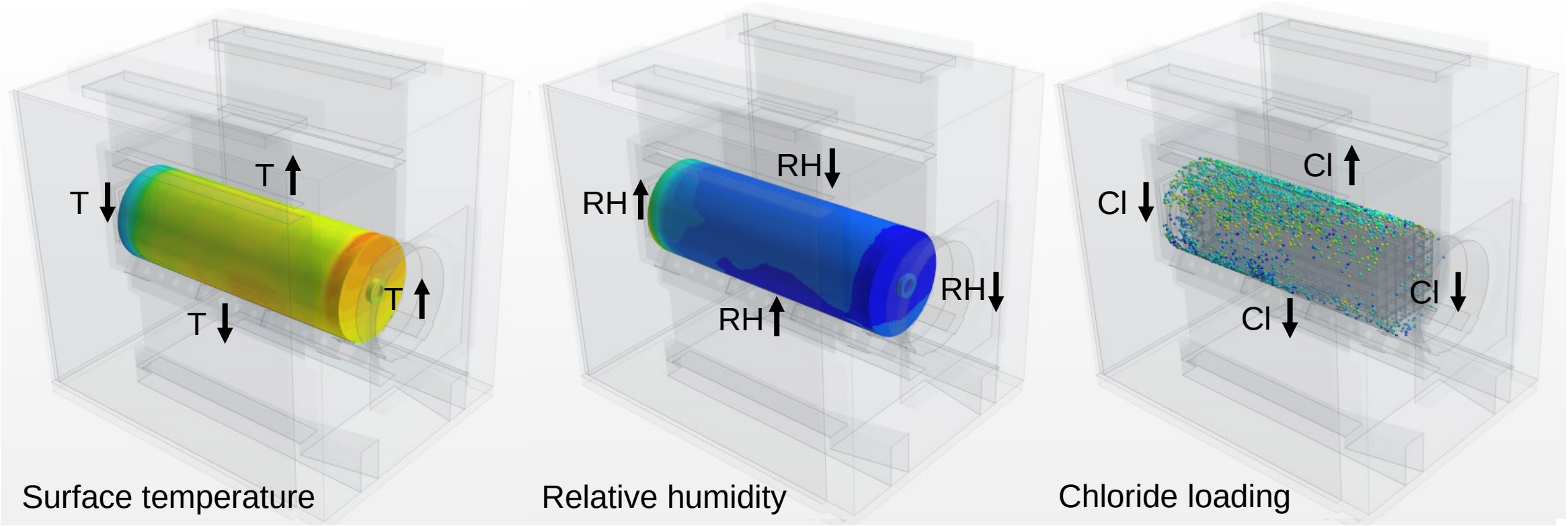
Chloride loading

[3] Suffield, S. R., Perkins, W. A., Jensen, B. J., Hanson, B. D., & Ross, S. B. (2024). Turbulent Particle Flow in Spent Nuclear Fuel Storage Systems. *Nuclear Technology*, *211*(2), 241–257. https://doi.org/10.1080/00295450.2024.2324213

Environmental Factors - Horizontal Canister Orientation

Variations in canister temperature and salt load factor into susceptible locations on canisters.

NUHOMS 32 PTH2



Surface temperature | Relative humidity | Chloride loading

Suffield, S. R., Perkins, W. A., Jensen, B. J., Hanson, B. D., & Ross, S. B. (2024). Turbulent Particle Flow in Spent Nuclear Fuel Storage Systems. *Nuclear Technology*, *211*(2), 241–257. https://doi.org/10.1080/00295450.2024.2324213

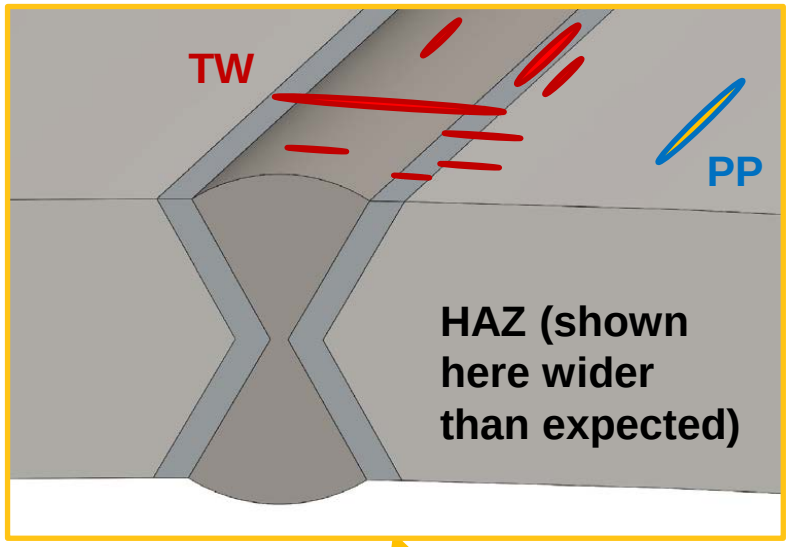
Summary

Multiple factors influence assessment

- Canister and overpack characteristics
- Environmental factors
- Material properties

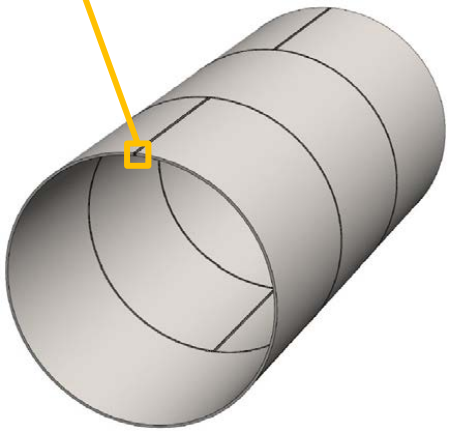
Applicability

- Visible inspection locations
- NDE equipment and location/orientation
- Consequence analysis



Through-wall crack location/morphology assessment

- Location
  - Positions on canister with sufficient salt and RH.
  - In and/or immediately adjacent to welds.
- Morphology
  - Transverse and longitudinal both possible.
  - Branched cracking, but possibly affected by local microstructure and weld configuration.



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Spent Fuel and High-Level Waste Disposition

The Technical Position for the Reduced Risk of Chloride-Induced Stress Corrosion Cracking (CISCC) at Inland Independent Spent Fuel Storage Installation (ISFSI) Sites

Rebecca Schaller

Sandia National Laboratories

Sandia National Laboratories is a multimission laboratory managed and operated by National Technology and Engineering Solutions of Sandia, LLC., a wholly owned subsidiary of Honeywell International, Inc., for the U.S. Department of Energy's National Nuclear Security Administration under contract DE-NA0003525

SAND2025-13450PE

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Technical Position for the Reduced Risk of CISCC at Inland ISFSIs

Objective:

- Document the NE-8 technical position for the reduced risk of CISCC at inland ISFSIs.

Scope

- Perform an up-to-date evaluation of the propensity for CISCC occurrence at inland ISFSIs.
- This effort will produce a Technical Position Report.

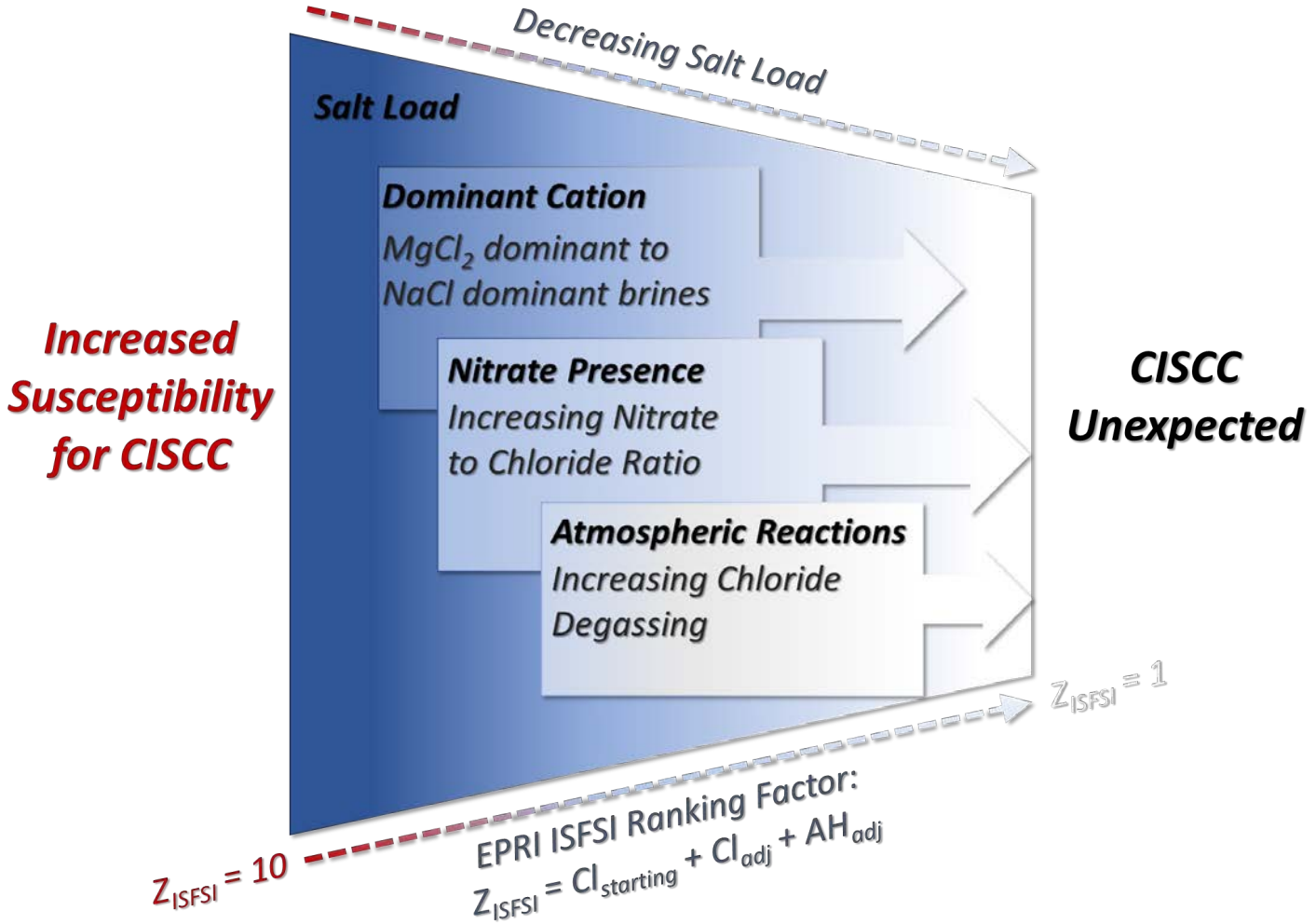
Value Statement

- Supports the NRC update to inspection intervals based on the conclusion that CISCC is not likely at sites with $Z_{ISFSI} < 3$ (see NRC Reg Guide 3.78) [inland sites in this category].
- Sets the technical foundation to potentially extend similar arguments to near-marine and even marine sites.
- Factors into Federal Consolidated Storage Facility (CSF) location preference and CSF aging management practices.

Environmental Influences on CISCC

Technical Position Report:

- *Supports NRC/EPRI ISFSI ranking factors*
- *Data and interpretation for reduced CISCC susceptibility at inland sites (with enhanced arguments involving environment)*



Inland Sites Argument Based on Environment

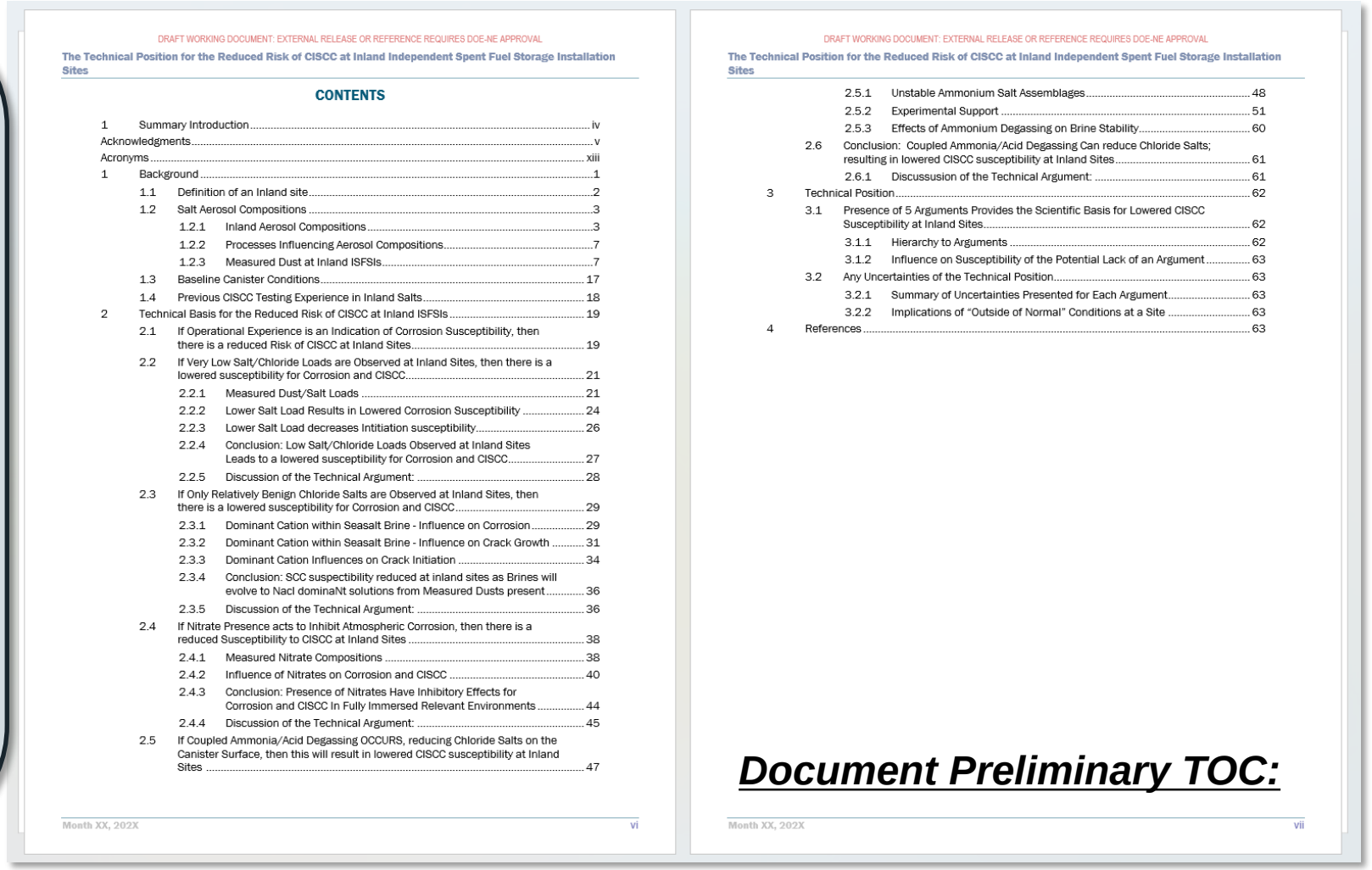
Criteria for CISCC:

- *All three criteria need to be met for CISCC to occur*
- *Reduced susceptibility to CISCC at inland sites: an aggressive environment is not present*

Document Development

Developed Outline

- *Background:*
  - *Define Inland Site*
  - *Salt Aerosols*
  - *Baseline Canister (report focus: non-sensitized stainless steel)*
  - *Previous Testing*
- *Technical Basis: 5 arguments*
  - *OPEX*
  - *Salt Load*
  - *Dominant Cation*
  - *Nitrate Presence*
  - *Atmospheric Reactions*
- *Technical Position*



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The Technical Position for the Reduced Risk of CISCC at Inland Independent Spent Fuel Storage Installation Sites

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The Technical Position for the Reduced Risk of CISCC at Inland Independent Spent Fuel Storage Installation Sites

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Document Preliminary TOC:

Background

- ***Approach and Relation to Existing Assessments:***
  - *EPRI ISFSI ranking factor*
  - *NRC RG 3.78*
- *Salt Aerosols:*
  - *Comparison of literature/collected data of available inland aerosols*
- *Baseline Canister*
- *Previous Testing*
  - *Summary of NRC tests*

EPRI ISFSI ranking factor does not account for compositional arguments:

$$Z_{ISFSI} = Cl_{starting} + Cl_{adj} + AH_{adj}$$

$Cl_{starting}$	Cl_{adj}	AH_{adj}
Chloride load based on distance to shoreline	Chloride adjustments: elevation, road salts, cooling towers	Absolute humidity from site or near site weather data

NRC arguments in RG 3.78 include some compositional factors (nitrate effects).

Inland site report provides detailed evidence supporting composition-based arguments.

Background

- *Approach and Relation to Existing Assessments:*
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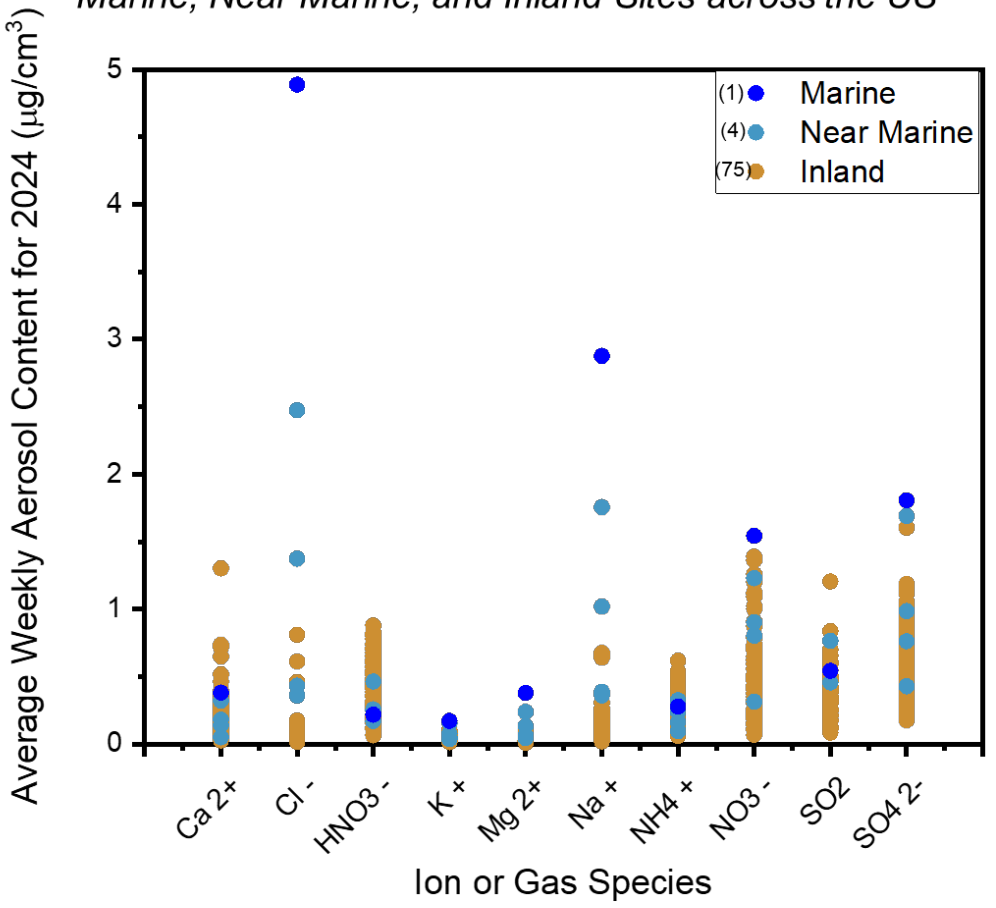
Selected CASTNET EPA Aerosol Sampling data:

Weekly average aerosol content (over 2024) for comparable Marine, Near Marine, and Inland Sites across the US



Background

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  - *NRC RG 3.78*
- ***Salt Aerosols:***
  - *Comparison of literature/collected data of available inland aerosols*
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Predicted Chemical Evolution of Measured Inland ISFSI Dusts:

Table 4. Predicted deliquescent mineral assemblages for Site A and Site B soluble salt samples: Removal of NO_3^- first, then remainder as Cl^-.

Sample	DRH (%)	Deliquescent pH	Deliquescent Mineral Assemblage	Sulfate Salts	Comments
A-1	58.9	7.60	NaCl-KNO_3-$NaNO_3$	anhydrite, glauberite	
A-2	58.9	7.60	NaCl-KNO_3-$NaNO_3$	anhydrite, glauberite	
A-3	61.5	9.92	NaCl-KNO_3-$NaHCO_3$	thenardite, glaserite	
A-4	72.1	9.79	NaCl-$Na_2Ca(CO_3)_2{:}2H_2O$	thenardite, glaserite	All NO_3^- removed
A-5	61.6	7.18	NaCl-KNO_3	thenardite, glaserite, glauberite, bloedite	
A-7	64.5	7.98	NaCl-KNO_3	glaserite, glauberite, syngenite	
A-8	72.1	9.79	NaCl-$Na_2Ca(CO_3)_2{:}2H_2O$	thenardite, glaserite	All NO_3^- removed
AbIk-1	61.5	8.18	NaCl-KNO_3	thenardite, glaserite, glauberite	
AbIk-2	58.9	7.60	NaCl-KNO_3-$NaNO_3$	anhydrite, glauberite	
AbIk-3	72.0	9.81	NaCl-$NaHCO_3$	thenardite, glaserite	All NO_3^- removed
AbIk-4	57.5	9.92	NaCl-KNO_3-$NaNO_3$-$NaHCO_3$	thenardite	
B (all Samples)	58.9	7.60	NaCl-KNO_3-$NaNO_3$	Anhydrite, glauberite	

Gypsum—$CaSO_4{:}2H_2O$; anhydrite—$CaSO_4$; glauberite—$Na_2Ca(SO_4)_2$; glaserite—$(K,Na)_3Na(SO_4)_2$; thenardite—Na_2SO_4; syngenite—$K_2Ca(SO_4)_2{:}H_2O$

- Even for the bounding measured inland dust cases, the brines did not evolve into Mg-chloride-rich brines
- NaCl-containing brines with high nitrate are predicted to be present in all cases except for a few of the worst-case scenarios
  - Samples that suffered complete loss of nitrate and evolved to NaCl brines also evolved to carbonate-rich pH 9.8 solutions
- All brines were predicted to contain high concentrations of sulfate, a potential corrosion inhibitor.

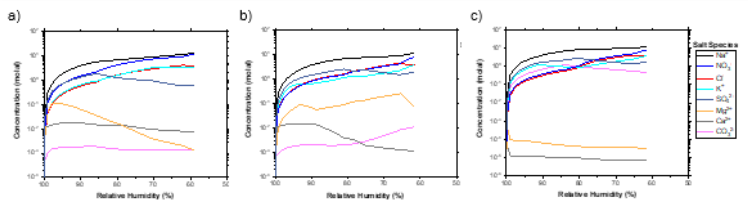

Figure 9. The predicted evolution of the Site A deliquescent salt assemblage (Proportional Removal of Cl^-/NO_3^-) for three observed deliquescent mineral groups a) NaCl-KNO_3-$NaNO_3$ (sample A-1), b) NaCl-KNO_3 (Sample A-5), and c) NaCl-KNO_3-$NaHCO_3$ (Sample A-3) (note difference in y-axis for c).

Background

- *Approach and Relation to Existing Assessments:*
  - *EPRI ISFSI ranking factor*
  - *NRC RG 3.78*
- *Salt Aerosols:*
  - *Comparison of literature/collected data of available inland aerosols*
- *Baseline Canister*
- ***Previous Testing***
  - *Summary of NRC tests*

Prior CISCC Testing Summary

- Discussion of tests carried out by the NRC and Southwest Research Institute in marine salts and a suite of salts that were developed to represent non-marine (inland) salts.
- References discussed:
  - NRC, *Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments*. 2010, U.S. Nuclear Regulatory Commission: Washington D.C. p. 84.
  - IMPROVE. *Interagency Monitoring of Protected Visual Environments*. 2025 1/7/2025]; Available from: https://vista.cira.colostate.edu/Improve/.
  - NRC, *Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Atmospheric Chloride and Non-Chloride Salts*. 2014, U.S. Nuclear Regulatory Commission: Washington D.C. p. 173.

Technical Basis

Five arguments:

- ***OPEX***
- *Salt Load*
- *Dominant Cation*
- *Nitrate Presence*
- *Atmospheric Reactions*

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- No CISCC has been identified on SNF interim storage canisters at any ISFSI site.
- OPEX provides summary of SCC in other components at nuclear reactor sites.
  - Occurrences observed due to crevicing and/or contamination other than atmospheric dusts

where abundant chloride-rich marine aerosols are present.

Table 5. Instances of atmospheric CISCC at marine sites on reactor site components.

Plant	Distance to salt water, m	Material and Component	Thickness, or crack depth, mm	Time in Service, years	Average crack growth rate, mm/yr
Koeberg (South Africa)	100	304L refueling water storage tank	5.0 to 15.5(?)[a]	17[b]	0.29 to 0.91(?)[a]
Ohi (Japan)	200	304L refueling water storage tank	1.5 to 7.5	30	0.17 to 0.25
St Lucie (FL, USA)	800	304 refueling water storage tank pipe	6.2	16	0.39

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Turkey Point (FL, USA)	400	304 pipe	3.7	33	0.11
San Onofre (CA, USA)	150	304 pipe	3.4 to 6.2	25	0.14 to 0.25

[a]Cracks were observed in the thickest components, but it is not clear if penetration occurred.
[b]Some thin-walled components apparently failed within as little as 11 years, but details are unavailable.
Source: Bryan and Enos [41], modified from Dunn [40].

Of these sites, the most documented example occurred at the Koeberg Nuclear Power Station in South Africa. At this facility, CISCC occurred in several different 304L components of a borated water storage tank system and associated piping. The CISCC that occurred on the storage tank was most commonly observed in areas where attachment fillet welds and butt plate welds created high residual stresses. In the piping, CISCC occurred due to high stresses imparted during pipe and elbow fabrication [42]. The water tanks were not pressurized, and the system was at ambient temperature. They were originally constructed in the open, but were enclosed in specially-built rooms in 1990; which reduced salt deposition however, it was speculated that enclosing the tanks, which eliminated rain-washing of salts from the surface of the tanks and piping, led to CISCC [42, 43]. A few pinhole leaks were identified as early as 1990, but were attributed to weld flaws. Several more leaks were identified in 1995 and 2001, through an inspection program. Inspection of the piping and the tank indicated that the metal was extensively pitted, and that the pitted areas served as initiation points for CISCC cracks which tended to initiate from the base of pits.

At inland reactor sites, CISCC has been identified on stainless steel piping in plants, however all known cases were due to chloride build-up in crevices [44, 45]. Various sources have been attributed for the presence of chlorides (in many cases unknown) and are documented in Table 6.

Table 6. Instances CISCC at inland sites on reactor site components.

Location	Years in Service	Component	Description of Failure/ Likely Cause	Source
Callaway Plant, Portland, MO	24	SS304 pressurizer auxiliary spray line	Leak due to cracking underneath a painted pipe support clamp ODSCC due to chloride build up in crevice conditions between the pipe and clamp, chloride source unknown	CAR 200809886
Wolf Creek Generating Station, Burlington, KS	24	SS304 pressurizer auxiliary spray line	Leak due to cracking underneath a painted pipe support clamp ODSCC due to chloride build up in crevice conditions between the pipe and clamp, chloride source unknown	OE 30001
Sequoyah Nuclear Power Plant, Soddy-Daisy, TN	3	CVCS BA Transfer Line (stainless steel piping)	ODSCC due to buildup in crevice, Chloride contaminant source unknown	OE 1845
River Bend Station, St. Francisville, LA	8	CRD hydraulic system piping (stainless steel)	ODSCC due to buildup in crevice, pressurized pipe, and chloride from a historical spill	LER #94-027-00

Technical Basis

Five arguments:

- *OPEX*
- ***Salt Load***
- *Dominant Cation*
- *Nitrate Presence*
- *Atmospheric Reactions*

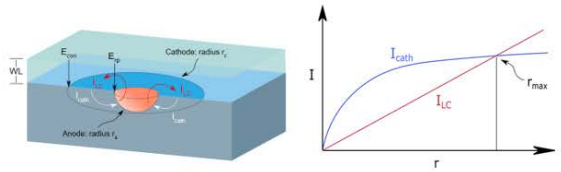

Figure 13. Schematic of maximum pit model.

The anode demand is given by the Galvele pit stability product [52], which dictates the current required at any given depth to maintain a critical environment at the surface, allowing for pit propagation to occur. The cathodic current relies on electrochemical parameters, anode size, conductivity, and water layer (WL) thickness. The anodic current must be equal and opposite of the cathodic current as the pit (anode) is inherently galvanically coupled to the surrounding cathode. It is the intersection of the anodic current and cathodic current (that both are a function of the anode size) that yields the maximum pit size (Figure 13). A pit cannot grow greater than this intersection as it cannot maintain the critical environment in the pit and will repassivate.

As previously noted in Section 1.2.3.3, the RH will dictate the equilibrium chloride concentration which in combination with salt load and composition will dictate the WL thickness. The WL (shown in Figure 13) was assumed to be conformal and given by:

Example: Lower salt loads lead to smaller maximum predicted pit sizes, thus lower likelihood for crack initiation (Kondo criteria)

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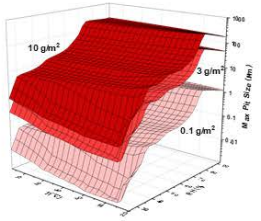

Figure 14. Influence of salt loading density on the predicted maximum pit size for sea salt vs T and RH.

While the previous arguments presented a computational framework for assessing severity of corrosion with salt load, experiments have also addressed this. Observations of pit sizes on aluminum increased with droplet sizes which increased with salt loading density [54]. Mass gain, a measure for total corrosion, showed that increased LD increased mass gain for Aluminum alloys [55-57]. In general, for SS304L, increasing LD with a $MgCl_2$ salt increased the corroded area [58]. Additionally, in unpublished data collected at Sandia, samples that were coated with 10 µg/cm² sea salt (ASW) and exposed in the same manner to 300 µg/cm² [59, 60] displayed no visible signs of corrosion on the samples. Similar observations of correlation of corrosion damage to salt load have been found for atmospheric exposures. One example, compared materials exposed over multiple years at a 70 m (oceanfront) and 190 m (near ocean) exposure lots at Kure Beach, NC [61]. The average chloride loads at each were ~176 and 54 mg/m²/day, respectively. Exposure over time resulted in roughly an order of magnitude greater corrosion rate of steel at the higher chloride load (70 m) lot [61]. While both lots are what would be considered marine exposure sites, these demonstrate the influence of salt load on corrosivity. Overall, there is a significant amount of experimental research that indicates that lower salt loads decrease the overall amount/severity of corrosion.

2.2.3 LOWER SALT LOAD DECREASES INTITIATION SUSCEPTIBILITY

From the previous section, it was noted that lowering salt LD decreases the pitting corrosion size. The Sandia probabilistic stress corrosion cracking model [51] utilizes the Kondo criteria [62], a competing rate theory that compares the corrosion rate of a pit to the crack growth rate. When the crack growth rate exceeds that of the pit growth rate, a critical pit size is identified where a crack will nucleate and become dominate, i.e. it defines the critical pit size for crack initiation.

The stress intensity of a pit (K_{pit}) is proportional to the square root of the pit size (i.e., $K_{pit} \sim \sigma\sqrt{a}$) (where σ is the applied stress and a is the pit or crack depth). Thus, smaller pits will have a lower stress intensity. Given that the threshold stress for stress corrosion crack nucleation, K_{iscc}, is independent of salt load, this would indicate that lower salt loads would lead to a lower stress intensity and a lower propensity for crack nucleation as $K_{pit\,small} < K_{iscc}$. The calculated stress intensity vs. crack depth (also pit depth) is shown in Figure 15 and indicates that smaller cracks (pits)

Technical Basis

Five arguments:

- *OPEX*
- *Salt Load*
- ***Dominant Cation***
- *Nitrate Presence*
- *Atmospheric Reactions*

Figure 15. Effect of temperature and orientation on resultant crack morphologies of SS304L exposed to saturated NaCl. It is noted that crack growth is from the left to the right.

Overall, NaCl dominant brines have a slower crack growth rate and a higher K_{ISCC} in comparison to the $MgCl_2$ dominant brines indicating that NaCl brines are less severe. If dust found at inland ISFSI locations is dominant in NaCl, once deliquesced (and as mentioned this would occur at much longer time frames), the CGR data indicates that the brines that may form display a lower severity for CISCC.

2.3.3 DOMINANT CATION INFLUENCES ON CRACK INITIATION

As previously noted in Section 2.3.1, micro-cracking was observed in pits formed on samples with the #4 surface finish exposed to $MgCl_2$ dominant ASW while no or very few micro-cracks were observed in NaCl dominant ASW. This suggests that both a low RH leading to an $MgCl_2$-dominated environment and a rough surface containing significant residual stress are necessary for micro-cracking [52, 60] but also indicates that $MgCl_2$ dominant solutions are more prone to crack nucleation.

In CISCC testing, rising K testing methodologies provide K_{ISCC} and have been demonstrated to be similar to decreasing frequency testing [68]. The range of K_{ISCC} values for $MgCl_2$ (red band) and NaCl (blue band) electrolytes were presented in Figure 10, indicating that the K_{ISCC} (determined from rising K testing) is lower for $MgCl_2$ electrolytes in comparison to NaCl. Further testing in ASW (Figure 13) [68] noted a lower K_{ISCC} for $MgCl_2$ dominant ASW in comparison to NaCl dominant ASW. That is NaCl-dominant brines have a higher K_{ISCC} in comparison to $MgCl_2$ dominant brines. This indicates that crack initiation is more difficult in NaCl dominant brines.

Controlled nucleation testing was performed by Sandia on thick SS304L tensile bars. Samples were loaded at yield strength and subjected to solutions heated to 55 °C [67]. In this study, saturated $MgCl_2$ and saturated NaCl were utilized for the electrolyte solutions. The sample exposed in saturated $MgCl_2$ electrolyte failed in 71 days, cracking completely in half. However, the sample exposed to saturated NaCl electrolyte did not fail even after 139 days. As can be seen in the optical images in Figure 16(a) and (b), numerous cracks were observed on the sample exposed to saturated $MgCl_2$ while no cracks were observed in saturated NaCl. It is noted that a large pit (radius = 600 µm) was present on the SS304L surface post-exposure to saturated NaCl as indicated by the corrosion staining near the top of the sample, however, this pit did not nucleate a crack (and no other cracks were observed across the sample surface).

Example: No crack initiation observed in NaCl in double the time of exposure as compared to $MgCl_2$

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Figure 16. Optical images post exposure displaying the influence of brine on crack initiation. Many cracks were observed pos-exposure in saturated $MgCl_2$ (left) while no cracks were observed for saturated NaCl, instead only one large pit (right), both exposures at 55 °C.

While the comparison of NaCl vs $MgCl_2$ dominant brines provides some insight for inland site CISCC initiation behavior, to date, crack initiation studies through rising K methodologies have not been performed for inland site compositions nor in sulfate/nitrate containing brines.

Other studies have investigated the influence of chemistry and/or RH (therefore changing dominant ions) on CISCC initiation. One study observed in U-bend testing of 304 with ASW that the susceptibility for CISCC tended to increase with increasing relative humidity (only up to 44 % RH) [58]. While an increase was noted, the dominant soluble ions in solution over the RH range tested are still Mg. Increasing RH will slightly decrease the overall soluble chloride concentration while increasing the WL thickness on the surface. It is believed that the increase in initiation due to increasing RH was due to the WL thickness rather than a change in chemistry. In a separate study, 304L and 316L were investigated with Mg, ASW, and site-specific ASW at 70 % RH. At 70 % RH, ASW becomes dominant in Na^+ in comparison to Mg^{2+} [51]. Both 304L and 316L grades did not exhibit any optically visible cracks when exposed to custom, site-specific sea-salt (Na^+ = 32.7; Mg^{2+} = 3.8; K^+ = 1.2; Ca^{2+} = 4.1; NH_4^+ = 6.6; Cl^- = 39.2; SO_4^{2-} = 3.6 NO_3^- = 9.1 all in wt. %) that contained nitrate and ammonium ions suggesting some influence of these chemical species on impeding [51]. Interestingly, pitting was more prominent for site-specific salts in comparison to $MgCl_2$ and ASW, but the observed pits were typically smaller. This was potentially linked to ammonium and the influence on pitting. In a comparison of the $MgCl_2$ and ASW testing, similar values were calculated for pit density and crack growth rate [51].

2.3.4 CONCLUSION: SCC SUSPECTIBILITY REDUCED AT INLAND SITES DUE TO PRESENCE OF NACL DOMINANT CHLORIDES WITHIN MEASURED DUSTS

As dust measurements at inland sites have indicated that the majority of chlorides present are NaCl, the potential brines that could form are more likely NaCl dominant. NaCl dominant brines, once formed, and as shown, this would occur at much longer times than $MgCl_2$ dominant brines, are less aggressive with respect to CISCC. In NaCl dominant brines, it is difficult to initiate a crack; these require a higher K_{ISCC} and need a larger pit (or initiating feature), hence, as discussed in the previous section (2.2), these brines necessitate a higher salt load to form pits (according to the maximum pit size model), and also result in slower crack growth rates.

Technical Basis

Five arguments:

- *OPEX*
- *Salt Load*
- *Dominant Cation*
- ***Nitrate Presence***
- *Atmospheric Reactions*

Figure 28. (a) Pits per droplet and (b) pit area for various nitrate to chloride ratios as a function of cation. Samples were exposed for one week under atmospheric diurnal cyclic conditions (constant T and varying RH from 20-80 %RH).

SANDIA EXPERIMENTAL OBSERVATIONS – CISCC INITATION AND CGR TESTING

Nitrate influences on crack growth rates were also examined and are shown in Figure 29 for an SS304L compact tension sample. The sample was fully immersed in $MgCl_2$ dominant (40%) ASW with an applied constant K of 50 MPa√m. The initial portion (to time = 0) had only the $MgCl_2$ dominant ASW brine, then at time t=0, the first addition of 463.4 g of $Mg(NO_3)_2$ was introduced into the brine and at time t = 1.26 x 10^8 s, the second addition was of 257.9 g (total of 721.25 g) of $Mg(NO_3)_2$ was made. The crack growth rate decreased initially to roughly $1.5{\cdot}10^{-7}$ mm/sec upon the first addition of nitrates. After roughly $2{\cdot}10^8$ s, the crack growth rate decreased again to roughly $4{\cdot}10^{-8}$ mm/sec. After the second addition of nitrates (for a total nitrate to chloride ratio of 1:11), crack growth rate ceased (below the resolution limit of the test frames used). After no crack growth was observed with a 1:11 ratio of nitrate:chloride at 55 ˚C, the temperature was increased to 75˚C. After temperature stabilization, measurable crack extension was observed. The temperature was subsequently decreased back to 55 ˚C and there was no indicated crack growth. These testing results indicate that in full immersion testing, nitrates can influence CISCC susceptibility and lead to crack arrest however the critical ratio of nitrate to chloride will be dependent upon temperature.

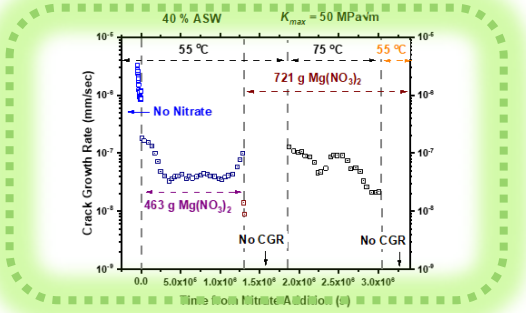

Figure 29. Influence of nitrate additions on crack growth rate for 40% ASW at various temperatures and 50 MPa√m.

As crack growth rate stopped at a 1:11 ratio of nitrate:chloride in full immersion exposure at 55 ˚C, this would indicate that in nitrate rich brines, K_{iscc} would be greater than pure $MgCl_2$ dominant brines and less
nitrate/
M Mg(N
radius)
cracks

exposed to the same environment and temperature for a total of 116 days with no failure. Again, no cracks were observed over the sample. This further indicates the potential inhibitory influences of nitrate on not only crack growth rates, but also CISCC initiation.

PNNL EXPERIMENTAL OBSERVATIONS – CGR TESTING

Similar observations of the influence of nitrates under fully immersed cracking conditions were observed at PNNL. Studies to-date in immersed conditions reveal significant reductions in CISCCGRs at lower temperatures for tests using $MgCl_2$ as shown in the compilation in Figure 30. The addition of nitrates however, may not be as effective in reducing CISCC susceptibility at higher temperatures. However, this establishes that nitrates may be beneficial in some conditions, such as inland sites, where there is little to no $MgCl_2$ present.

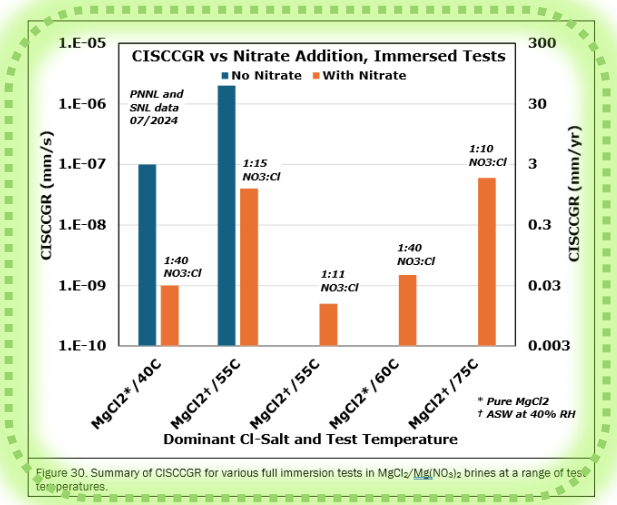

Figure 30. Summary of CISCCGR for various full immersion tests in $MgCl_2$/$Mg(NO_3)_2$ brines at a range of test temperatures.

Example: Presence of nitrates reduces and if sufficient concentration, stops detectable CISCC growth

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Technical Basis

Five arguments:

- *OPEX*
- *Salt Load*
- *Dominant Cation*
- *Nitrate Presence*
- ***Atmospheric Reactions***

$(NH_4)_2SO_4$: NACL MIXTURES

The experiments performed with single salt exposures of NH_4NO_3, NH_4Cl, and $(NH_4)_2SO_4$ indicated that the first two phases cannot accumulate in their pure form on the surface of a canister at temperatures above ambient. At 50°C, even a relatively heavy salt load (1 g/m²), present as either dry or deliquesced NH_4NO_3 or NH_4Cl, would degas from the canister surface within days and at 30°C, complete degassing would occur within several weeks. This rate of loss is greater than typically observed atmospheric deposition rates for these salts, indicating that accumulation is unlikely and they could not persist on a canister surface at slightly elevated temperatures. However, $(NH_4)_2SO_4$ is relatively stable and could accumulate/persist for years on a canister surface. If $(NH_4)_2SO_4$ is present with other thermally stable chloride salts (e.g. NaCl), there may be potential for a corrosive ammonium-bearing chloride-rich brine to form on a canister surface once deliquescence occurs.

While NH_4NO_3 and NH_4Cl decompose from the solid form and ammonia degasses along with the associated anionic species, once deliquescence occurs, the source of the ammonium (the identity of the original ammonium mineral) is unimportant. Ammonia and HNO_3 or HCl will be lost from the deliquescent brine in proportion to the partial pressure of each generated by the solution. The partial pressure of each in the ambient environment will also play a role, but because the canister surface is always hotter than the ambient condition, it is likely that the brine partial pressure will always be higher than the ambient partial pressure. Once a deliquescent brine of $(NH_4)_2SO_4$ and a chloride or nitrate salt forms, loss of ammonia and HCl/HNO_3 from that brine will occur, until either the ammonium or the (chloride + nitrate) is depleted. Therefore, Sandia performed experiments to further explore these mixed salts

Degassing experiments were run with a mixture of $(NH_4)_2SO_4$ and NaCl on a QCM wafer and two stainless steel coupons (one control). In this test, the two salts were deposited separately to minimize reaction prior to the beginning of the test. At the end of the experiment, the salts were leached off both coupons with deionized water, analyzed, and compared to determine if degassing had occurred, the control sample providing information on the initial salt distributions and ion concentrations. The ammonium sulfate did not deposit evenly, but instead beaded up to form large (200-300 µm) droplets which crystallized into coarse crystals. The NaCl formed a finely-dispersed coating over both the metal surface and the coarse aggregates of ammonium sulfate. While the two-step deposition procedure limited reaction between the two salt phases, some reaction did occur (Figure 37). The chemical analyses of the samples suggest that this was negligible.

Simultaneous ammonium/acid degassing can reduce chlorides present, thus lowering CISCC susceptibility

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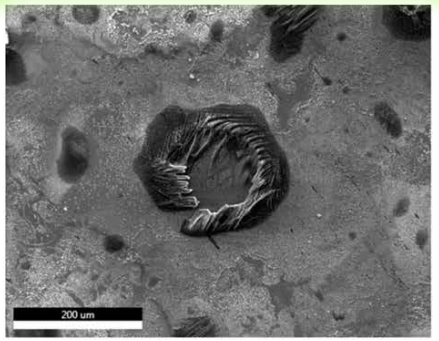
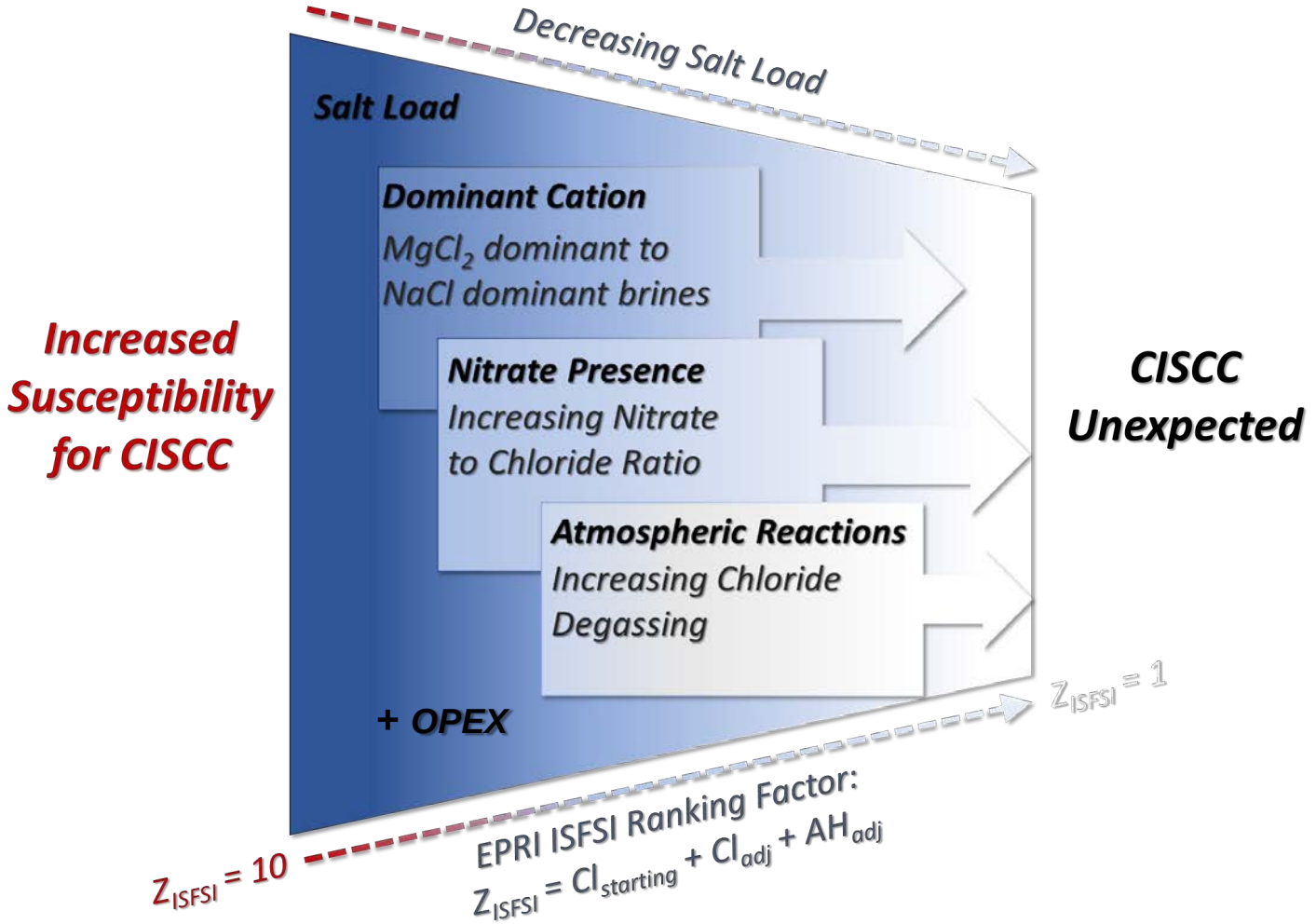


Figure 37. SEM image and element maps of a crystallized droplet of ammonium sulfate on the control sample, showing depletion of ammonium and chlorine in areas where the two deposited salts (NaCl and $(NH_4)_2SO_4$) overlap.

In the experiment, the test coupon was placed in an environmental chamber at 50°C at 10%RH for a few days to ensure dryness. Then, the RH was increased to 75% RH, sufficiently high for the salts to deliquesce at least partially and held for 24 h. The sample was then dried at 10% RH overnight. The sample was then examined by SEM (Figure 38). The two salts had deliquesced and mixed, forming

Goal of the Technical Position Milestone Report

- *Provide stronger scientific basis for NRC Reg Guide 3.78*
- *Enhance/further develop understanding of CISCC susceptibility at inland sites:*
  - *5 technical arguments (including OPEX) that each reduce susceptibility; sum together to further reduce susceptibility*



FY26 Next Steps

Development of Technical Position Report:

- *Introduction and Background – draft complete*
  - *Summary of Aerosol data from other available sources (CASTNET) – data collected and summarized in draft report.*
- *5 Technical Arguments – draft underway*
  - *Initial argument drafts developed*
  - *Currently working to incorporate collaborative research from PNNL*
- *Technical Position Summary – initiated draft*
  - *Summary of arguments underway with conclusions being developed for technical position.*

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Spent Fuel and High-Level Waste Disposition

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Spent Fuel and High-Level Waste Disposition

In-Service Spent Fuel Canister Surface Sampling

Laura McManniman (EPRI) and Rebecca Schaller (SNL)

EPRI ESCP
October 2025

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DOE Disclaimer

DISCLAIMER

This is a technical presentation that does not take into account contractual limitations or obligations under the Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste (Standard Contract) (10 CFR Part 961).

To the extent discussions or recommendations in this presentation conflict with the provisions of the Standard Contract, the Standard Contract governs the obligations of the parties, and this presentation in no manner supersedes, overrides, or amends the Standard Contract.

This presentation reflects technical work which could support future decision making by the U.S. Department of Energy (DOE or Department). No inferences should be drawn from this presentation regarding future actions by DOE, which are limited both by the terms of the Standard Contract and Congressional appropriations for the Department to fulfill its obligations under the Nuclear Waste Policy Act including licensing and construction of a spent nuclear fuel repository.

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Electric Power Research Institute (EPRI) – Department of Energy (DOE) Memorandum of Understanding (MOU)

← Press Releases

EPRI, DOE Sign MOU on Used Nuclear Fuel Research

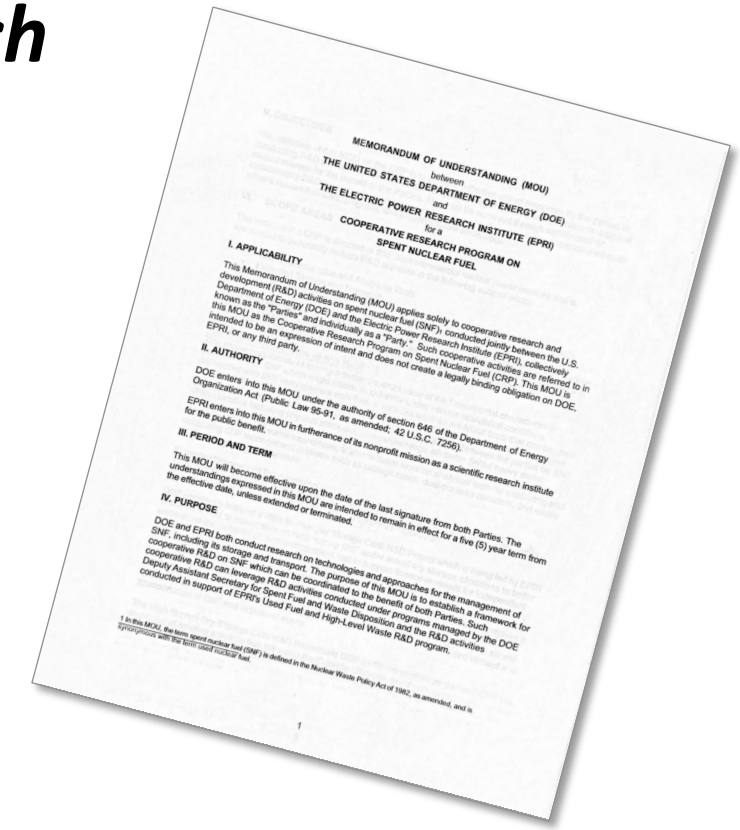
Jul 11, 2024

PALO ALTO, Calif. – July 11, 2024 – EPRI and the U.S. Department of Energy (DOE) today announced a memorandum of understanding (MOU) with the goal of establishing a cooperative research program focused on technologies and approaches for the management of used nuclear fuel including its storage and transport. Finding solutions for the safe, long-term management of used nuclear fuel will help to ensure the sustainability of nuclear energy as a carbon-free resource.

Used nuclear fuel is fuel that that has been used in the production of electricity that is no longer capable of efficiently generating energy. It is radioactive and is safely stored in either large pools of water or in large steel-reinforced concrete containers. The MOU will allow cooperative research & development activities to be jointly conducted between DOE and EPRI. The five-year MOU may include research and development activities in the following subject areas:

- Used Nuclear Fuel Characteristics and Analytical Tools – R&D activities providing a better understanding of used nuclear fuel's fundamental characteristics to inform its potential behavior during fuel management activities. This also includes creating analytical tools for evaluating and optimizing used nuclear fuel management systems such as storage casks, dual-purpose canisters, and onsite fuel management approaches.
- Extended Storage and Transport – R&D activities focused on technologies and operations related to used nuclear fuel extended storage and/or transportation.
- Disposal Considerations – R&D activities involved with the development of technologies and approaches which could facilitate eventual disposal of used nuclear fuel.

Project conducted under the EPRI-DOE MOU on Used Fuel Research



MEMORANDUM OF UNDERSTANDING (MOU)
between
THE UNITED STATES DEPARTMENT OF ENERGY (DOE)
and
THE ELECTRIC POWER RESEARCH INSTITUTE (EPRI)
for a
COOPERATIVE RESEARCH PROGRAM ON
SPENT NUCLEAR FUEL

I. APPLICABILITY

This Memorandum of Understanding (MOU) applies solely to cooperative research and development (R&D) activities on spent nuclear fuel (SNF)[1] conducted jointly between the U.S. Department of Energy (DOE) and the Electric Power Research Institute (EPRI), collectively known as the "Parties," and individually as a "Party." Such cooperative activities, collectively this MOU as the Cooperative Research Program on Spent Nuclear Fuel (CRP). This MOU is intended to be an expression of intent and does not create a legally binding obligation on DOE, EPRI, or any third party.

II. AUTHORITY

DOE enters into this MOU under the authority of section 646 of the Department of Energy Organization Act (Public Law 95-91, as amended; 42 U.S.C. 7256).

EPRI enters into this MOU in furtherance of its nonprofit mission as a scientific research institute for the public benefit.

III. PERIOD AND TERM

This MOU will become effective upon the date of the last signature from both Parties. The understandings expressed in this MOU are intended to remain in effect for a five (5) year term from the effective date, unless extended or terminated.

IV. PURPOSE

DOE and EPRI both conduct research on technologies and approaches for the management of SNF, including its storage and transport. The purpose of this MOU is to establish a framework for cooperative R&D on SNF which can be coordinated to the benefit of both Parties. Such cooperative R&D can leverage R&D activities conducted under programs managed by the DOE Deputy Assistant Secretary for Spent Fuel and Waste Disposition and the R&D activities conducted in support of EPRI's Used Fuel and High-Level Waste R&D program.

[1] In this MOU, the term spent nuclear fuel (SNF) is defined in the Nuclear Waste Policy Act of 1982, as amended, and is synonymous with the term used nuclear fuel.

1

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Canister Chloride Induced Stress Corrosion Cracking (CISCC)

- Identified as a potential degradation mechanism for welded stainless steel canisters
- Desire to maintain awareness of canister environment
- ***Current focus is developing a standardized method of dust sampling DPCs***
  - Captured experiences, learning, and developments from surface sampling campaigns completed to date
  - Recommends a methodology and supporting work going forwards to ensure consistency of approach and comparability of results

Conditions for SCC to Occur

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Collaborative Approach to Methodology Development

- DOE Office of Spent Fuel & High-Level Waste Disposition – Providing their experiences and opinions on dust sample acquisition and measurement.
- EPRI – Using this information to create a standardized dust sampling methodology.

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Dust Sampling Techniques

- Since 2012, several sampling campaigns at multiple ISFSIs have been carried out (often combined with inspections).

Year	Location	Canister Orientation	Collection Method	Sample Type
2012	Calvert Cliffs	Horizontal, the lid to the overpack was removed	Hand Tool	Wet: SaltSmart™ and Dry: Scotch-Brite™
2014	Hope Creek and Diablo Canyon	Vertical	Magnetic wheeled robot with extensible arm, long rod for hand sampling	Wet: SaltSmart™ and Dry: Scotch-Brite™ (without abrasive hard particles)
2016	Maine Yankee	Vertical canister containing greater than class C waste	Robotic and hand sampling	Wet: Sponges or Filter papers
2017	Calvert Cliffs	Horizontal	Hand Tool	Wet: SaltSmart™ and Dry: Scotch-Brite™
2018	Maine Yankee	Vertical	Hand sampling	Wet: sponge and Dry: Scotch-Brite™
2019	Maine Yankee	Vertical	Hand collection of witness samples	Exposed glass coupons
2020	Inland Site A & B, Near Marine Site C	Horizontal	RTT robotic crawler	Dry: Scotch-Brite™
2023	Maine Yankee	Vertical	Hand collection of witness samples	Exposed glass coupons (and entire sample blocks with 4-point bends retrieved)

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Dust Sampling Techniques

- While sampling campaigns were successful in many ways, multiple lessons learned as well as pros/cons for various techniques were established.
- Example pros/cons & lessons learned:
  - In general, across all campaigns, sampling efficiencies are not known
  - Hand Sampling tool – 10 ft long rod influences ability to effectively sample
  - Temperature issues with SaltSmarts™ (could not be used > 70°C)
  - Sample Transport – ambient conditions can play a role in shipment (resulted in mold growth)
  - Robotic sampling – fit of sampling filters/abrasive pads influences collection

Mold on Sample

Undersized Filter/Abrasive

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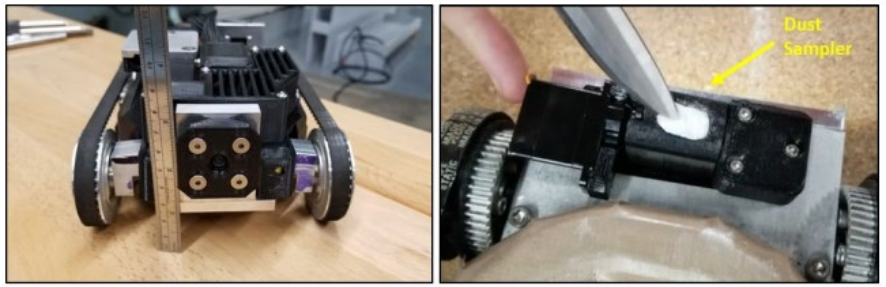
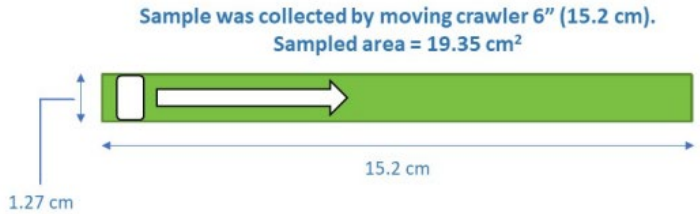
Dust Sampling Techniques

- Throughout campaigns, as lessons were learned, some mitigations were applied to alleviate issues.
- Example pros/cons & lessons learned:
  - Robotic sampling with venturi method applied to alleviate issued from hand-sampling (could sample known area)
  - SaltSmarts™ use discontinued (due to temperature issues and manufacturer going out of business)
  - Sample Transport – recommendations for transport with desiccant applied
  - Robotic sampling – fit of sample enhanced in future campaigns (still issues with oversized filter/abrasive)

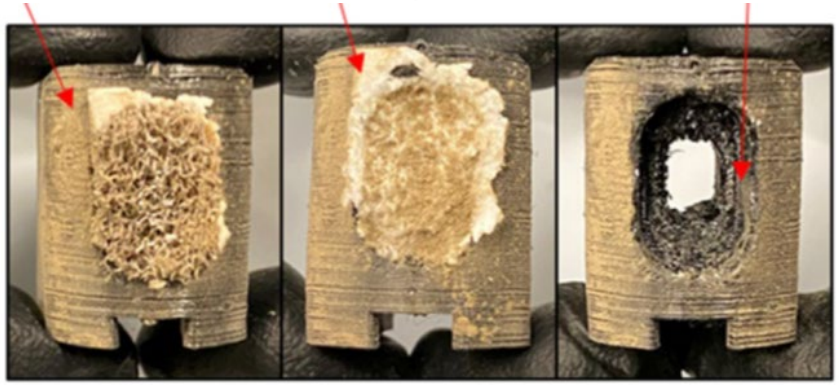
Initial Robotic Crawler and Sampling Pattern





Undersized Filter/Abrasive

Oversized Filter/Abrasive



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Dust Analysis

Methods applied across previous sampling campaigns:

- *Chloride Concentration/ Deposition Density Analysis:*
  - **Conductivity Measurements** – Determine surface salt concentrations/ deposition density

- *Dust Chemical Analysis:*
  - **XRD** *(X-ray Diffraction)* – Identify and quantify potential crystalline phases
  - **XRF** *(X-ray Fluorescence)* – Identify composition of dust
  - **GCMS** *(Gas Chromatography-Mass Spectrometry)* – Identify composition of wet samples (from SaltSmart™)
  - **IC** *(Ion Chromatography)* – Identify leachable anions and cations from dust
  - **ICP-OES** *(Inductively Coupled Plasma Optical Emission Spectrometry)* – Measurement of composition of soluble leached dust
  - **Raman** – Identify organic and inorganic molecular species
  - **FTIR** *(Fourier Transform Infrared Spectroscopy)* – determine volatile and semi-volatile organics
  - **SEM/EDS** *(Scanning Electron Microscopy/ Energy Dispersive Spectroscopy)* – Identify mineralogical and morphological details of dust present

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Dust Analysis Methodology

Lessons learned from previous methods applied :

- *Chloride Concentration/ Deposition Density Analysis:*
  - **Conductivity Measurements** – calibration of measurement to seawater may not be applicable at inland sites

- *Dust Chemical Analysis:*
  - Measurements were heavily influenced by the sampling method and material used to collect them.
    - Influenced collection efficiency: if too little sample collected, many methods could not detect dust above background.
    - Canister surface abrasion: influenced identified phases.
    - Porosity of sampling material: influenced detection ability in some cases (i.e. XRD).
    - Sample material contamination: Whatman filters contained nitrates and chlorides, amides influenced FTIR.
    - For image analysis, samples often biased based on collection locations/distribution of samples.

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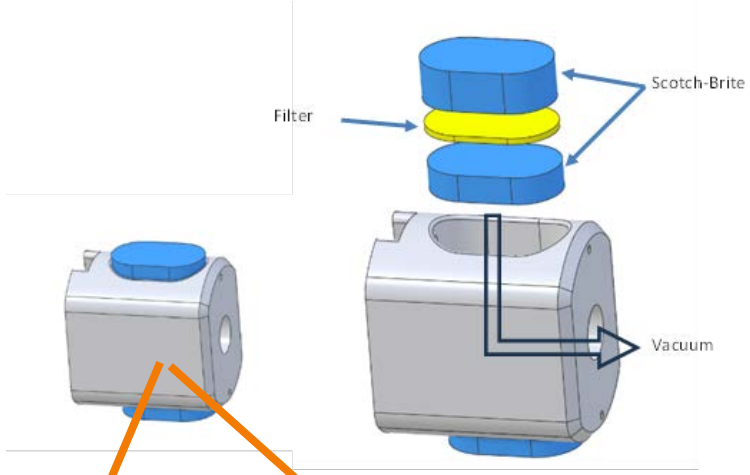
Dust Sampling Methodology

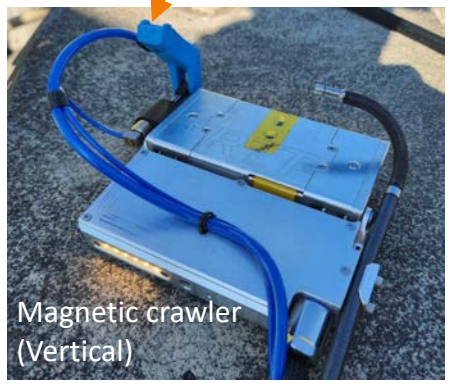
- Current process is combination of suction and mechanical scrubbing
- Standardized sampling head used for collection
- Sample handling process developed based on learning from earlier campaigns

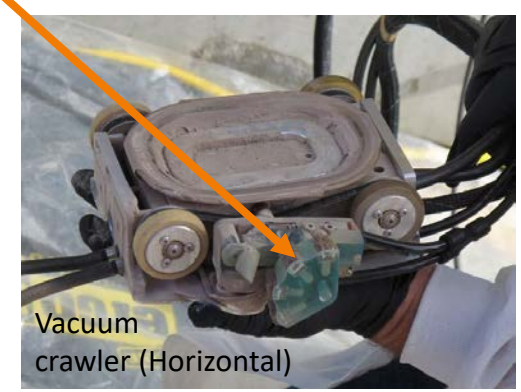
Canister Type	Sampling Location
Horizontal	12:00*, 2:00, 4:00, at support rail*
Vertical (body)	At the four outlet vent clock locations 2 ft from top of canister
Vertical (lid)	At the four outlet vent clock locations 50-100mm in from edge of lid

Sample Filter Area	13 mm x 9.5 mm
Area Sampled	~2000 mm^2 (13 mm x 154 mm)
Sample Vacuum Flow Rate	78 LPM
Ample Vacuum Pressure	48 kPa
Mechanical Scrubbing Element	Aluminum oxide embedded nylon non-woven, open-web (3M 07440, Scotch-Brite™)
Filter Element	Inert, high temperature quartz-fiber non-woven filter



Sampling head designed by RTT


Magnetic crawler (Vertical)


Vacuum crawler (Horizontal)

Can be deployed using modified inspection crawlers

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Dust Analysis Methodology

Currently summarizing previous sampling campaigns to develop inputs for sampling strategy.

Will be informed by:

- *Lessons Learned*
- *Desired measurements (deposition density, composition, etc.)*
- *Value proposition of measurements*

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Method Efficiency Testing

- Next steps to test sampling method efficiency
  - Support comparison of sample results from different sites
- Testing could be used to:
  - Indicate the reliability & repeatability of sample results
  - Demonstrate the effectiveness of the sample head and repeatability of its assembly
  - Trial alternate sample head arrangements to determine optimum Scotch-Brite / filter paper arrangements

Canister mock-up at EPRI Charlotte Lab

Continuing EPRI-DOE collaboration under MOU

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Collaborative Activities

- Dust deposition:
  - Apply previously developed methodologies for quantitative dust deposition of salts and/or salts with inert particulate to mimic expected ISFSI dust depositions on sampling test plates
- Sampling:
  - Use the developed canister/overpack mock-up and dust deposited sampling plates, perform remote robotic sampling with proposed RTT robotic crawler (test varying sample material configurations/arrangements)
- Analysis:
  - Once suggested sampling strategy is developed, quantify and determine sampling efficiency

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Spent Fuel and High-Level Waste Disposition

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SCC of Stainless Steel

Combined Effects of Chloride and Nitrate

Jared Smith
Sr. Principal Technical Leader, EPRI International Materials Research

EPRI Extended Storage Collaboration Program (ESCP)
EPRI Charlotte Office
October 27-30, 2025

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Outline

- Introduction – CANDU Moderator System and Calandria Relief Ducts
- Background – Bruce Unit 7 Operating Experience
- R&D for SCC Mitigation
  - Electrochemical Testing
  - Stress Corrosion Cracking Testing
- Conclusions

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INTRODUCTION

Moderator System - Overview

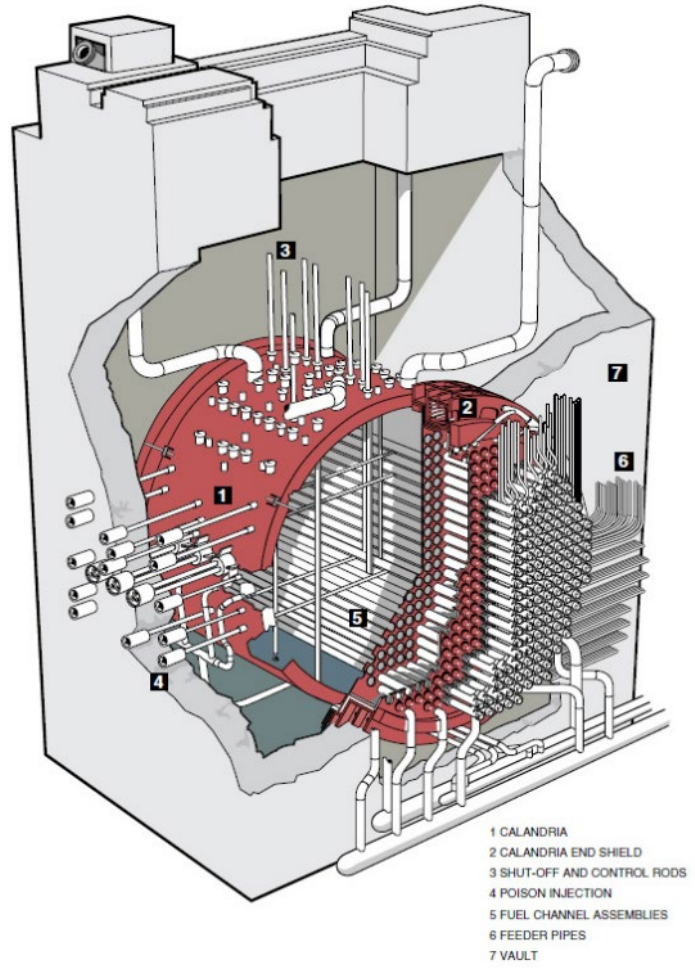
- In CANDUs neutron moderation and reactivity control is performed separate from the primary coolant → moderator system.

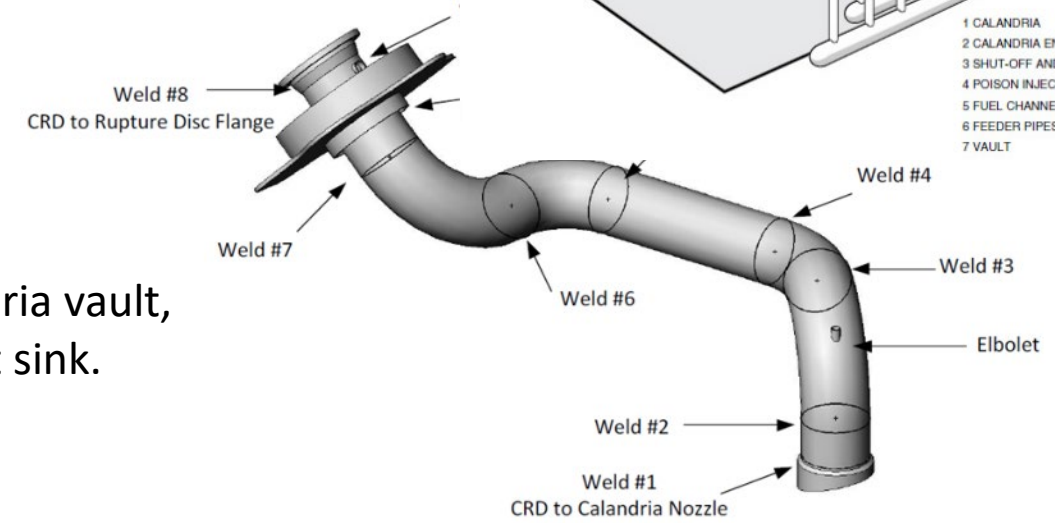
- Moderator System:

  <u>Recirc aqueous system (calandria vessel)</u>
  - Horizontal welded 304L/308L tank with tubesheets
  - D_2O, $Gd(NO_3)_3$
  - pH 4.5-7.5
  - 50-70°C
  - Net radiolysis conditions, ECP ~500 mV_{SHE}

  <u>Recirc He cover gas system (relief ducts)</u>
  - 18" diameter, 3/8" thick, welded 304L pipe/elbows
  - He with radiolytically produced H_2 and O_2
  - Vapor/condensate

- Calandria vessel and ductwork sit within shield tank or calandria vault, immersed in light water → biological shield & secondary heat sink.





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Bruce, Unit 7, CRD Cracking Experience

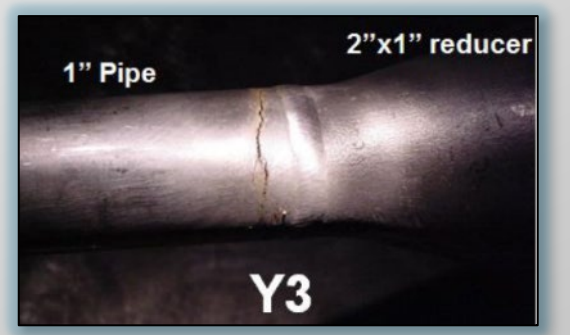
Bruce NGS B, Unit 7, CRD Cracking Experience

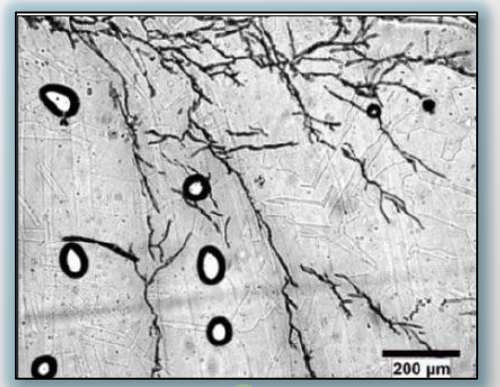
2005

- He leak in tie lines connecting CRDs.
- Prompted visual examination of CRDs.
- Fe contam & pitting observed on CRD ID.
- Limited UT inspection, 25 %tw flaws on outlet CRDs (ID initiated).
- Failure analysis on 1" tie line.
- TGSCC → Chloride-induced SCC

2008

- Further proof of TGSCC by inspection





2011

- Phased array UT, LPI, replication, visuals.
- SCC is actively propagating.

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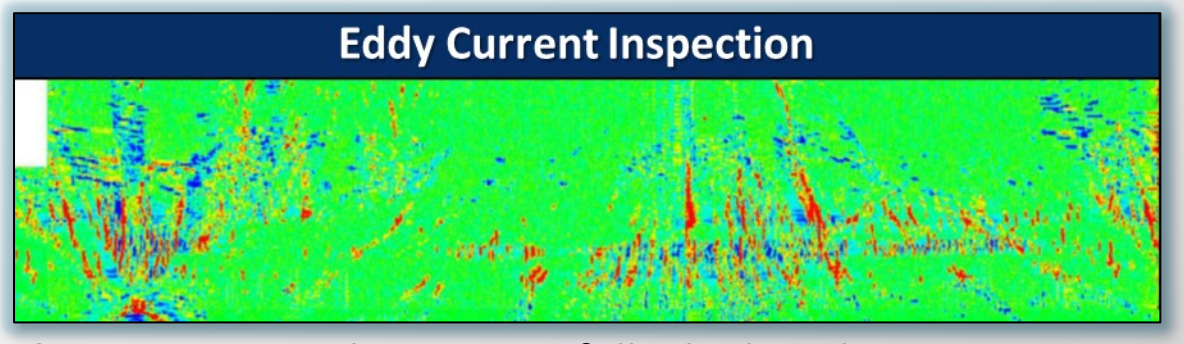
Bruce NGS B, Unit 7, CRD Cracking Experience



- Cleaning & inspection tooling successfully deployed.
- Cleaning removed lightly embedded corrosion product but some deeply embedded particles still evident at welds.
- Inspection system successfully detected and characterized surface breaking TGSCC allowing for monitoring program.

2014

- Further inspection revealed continued propagation of flaws.
- Project undertaken to design and deploy tooling for full volumetric inspection and cleaning by high-pressure water jet.

2016

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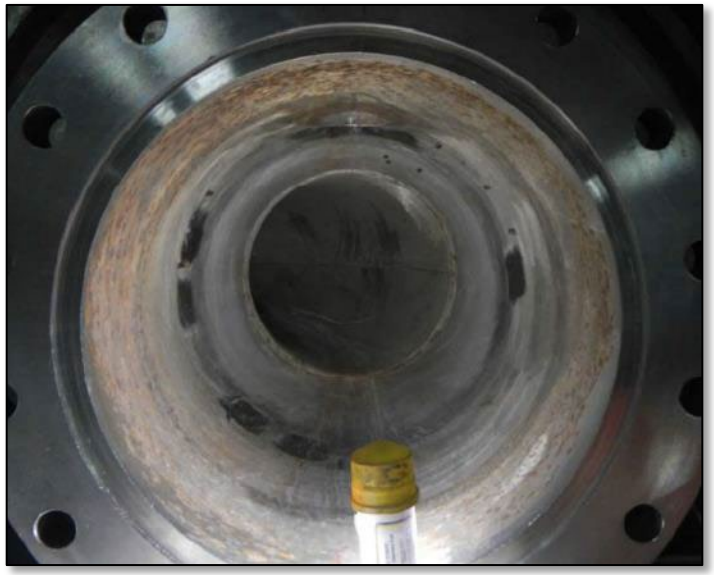
CRD Cracking - Extent of Condition

- Not a safety concern, but potentially life limiting:
  - Through-wall cracks → in-leakage of light water from shield tank → downgrading D_2O.
  - Calandria not refurbished and weld repair is challenging.
- Reported root cause: Cl-SCC in weld HAZ exacerbated by Fe contam.
- Fe contamination → grinding of ID surface during fabrication with non-stainless steel tooling.
- SCC initiation not necessarily at Fe inclusions → different than typical mechanism.
- Source of chloride unknown (moderator chemistry good, $[Cl^-]< 2$ ppb), acute or chronic? concentration factor?
- Cleaning by water jet:
  - Effective at removing most, but not all, Fe contamination.
  - Effective at mitigating crack propagation, to date.
- Inspections of CRD at other units, Bruce NGS Units 2&5, Darlington NGS, Unit 2:
  - No SCC.
  - Fe-based surface staining (rouging), minor corrosion, but no pitting.



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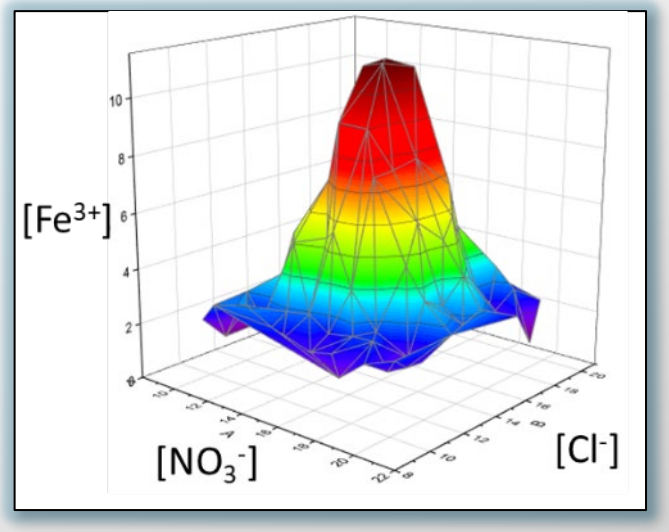
R&D for CRD Cracking Mitigation

Research Objectives

- **Goal: Preventative mitigation for the CANDU fleet.**

- Questions to be resolved:
  - Degradation mechanism?
  - Domains of susceptibility?
  - Detection prior to the onset of cracking?
  - When to clean?
  - Was cleaning effective?



Understand the Degradation

- Baseline mechanistic tests
- E-chem to define domains of susceptibility

Proof of Concept

- Validation SCC test on standardized specimen

Actionable Solution

- Propose feasible means of inspection / mitigation

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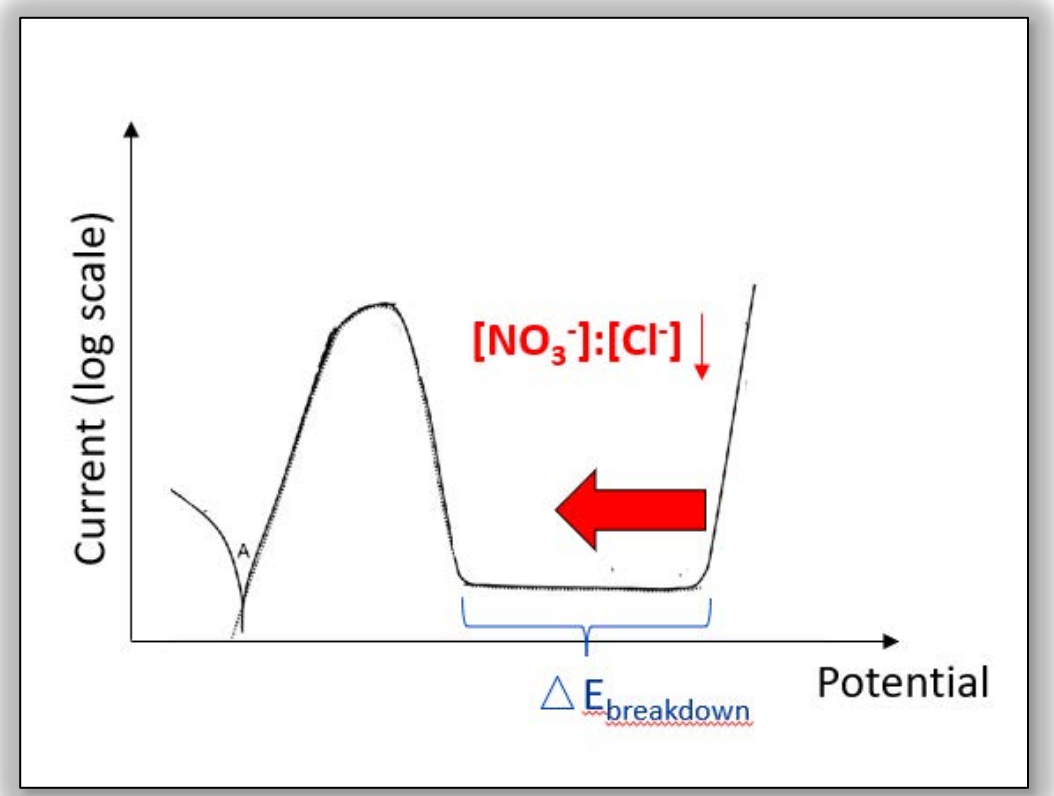
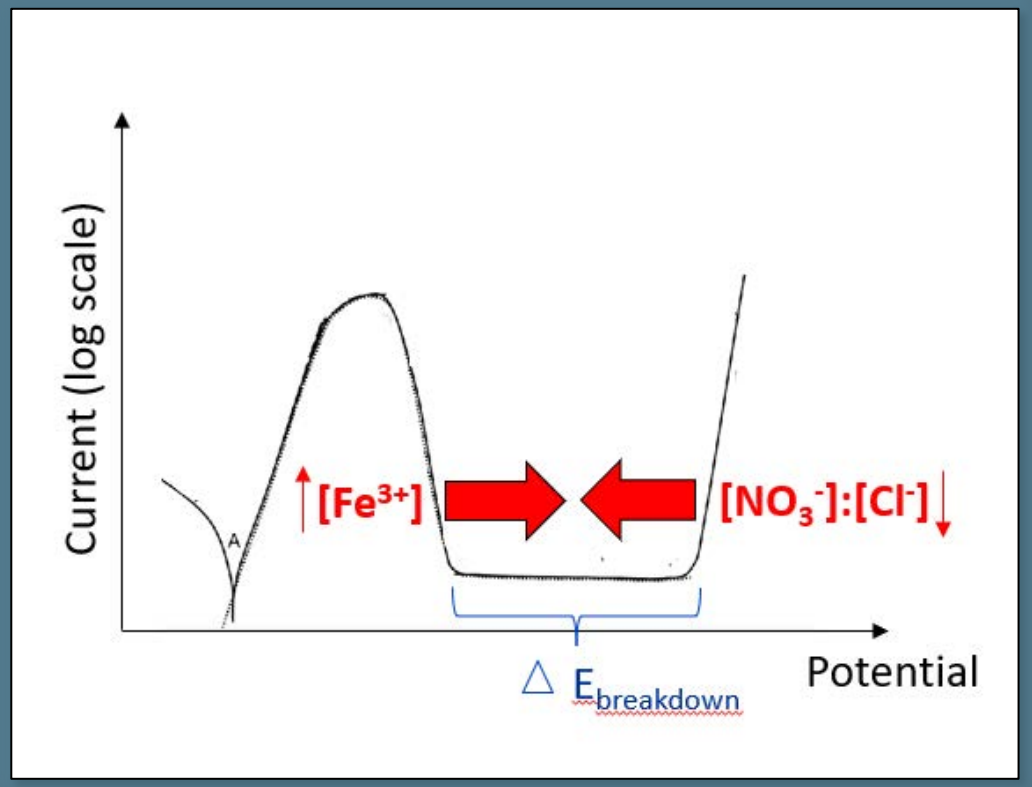
Mechanistic Investigations

The Role of Chloride & Nitrate



- Chloride detrimental, but nitrate has a mitigating effect.
- Region of passivity is diminished as [NO3$^-$]:[Cl$^-$] decreases.
- If chloride is present at far greater concentrations, passivity is lost completely.

The Role of Iron Contamination



- Iron surface contamination is oxidized to soluble Fe^{3+} by nitrate → mobile!
- Increases the potential of the stainless steel (globally, not locally).
- Passive window is diminished from both sides.

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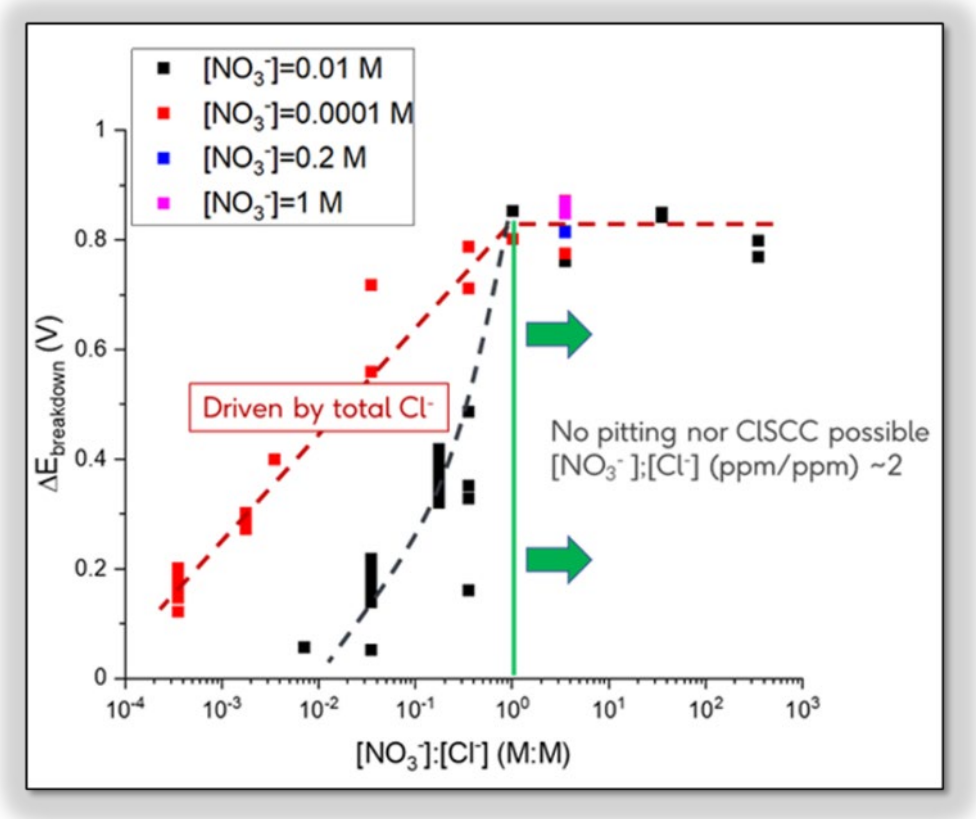
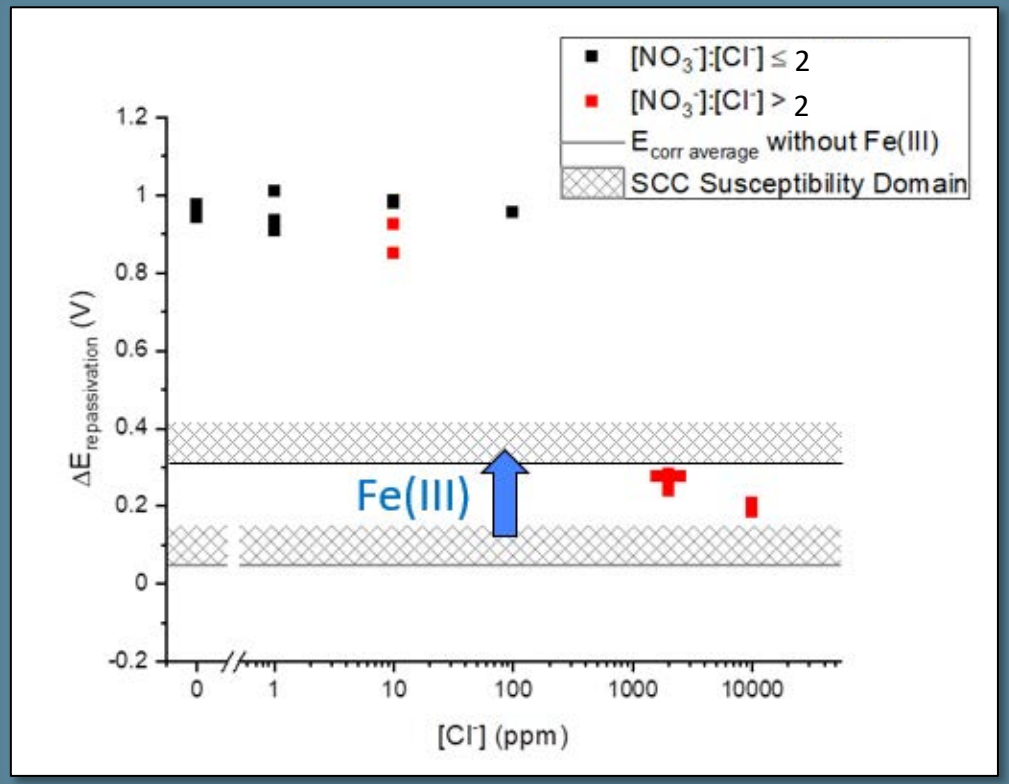
Domains of Susceptibility



- Domain of susceptibility exists at a $[NO_3^-]$:$[Cl^-]$ of 2.
- $[NO_3^-]$:$[Cl^-]$ < 2 susceptible to pitting and the degree of susceptibility corresponds to total $[Cl^-]$.
- $[NO_3^-]$:$[Cl^-]$ > 2, no pitting.
- For pitting, this relationship holds true, regardless of degree of iron contamination. What about SCC?



- Affect of Fe^{3+} is similar to that of O_2.
- Raises the ECP and erodes the margin of pitting / SCC resistance.
- Electrochemical data suggests CRD cracking is unlikely in the absence of iron contamination, but not impossible.

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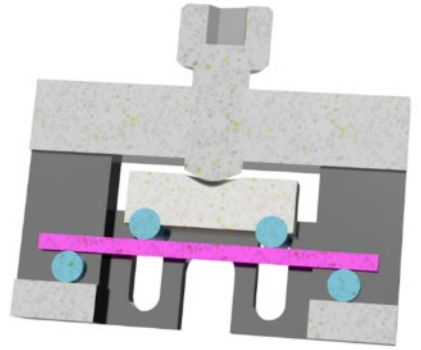
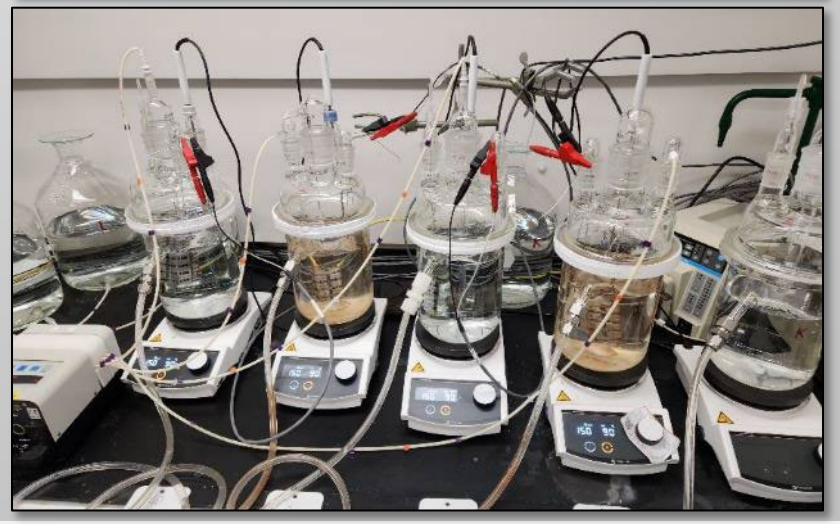
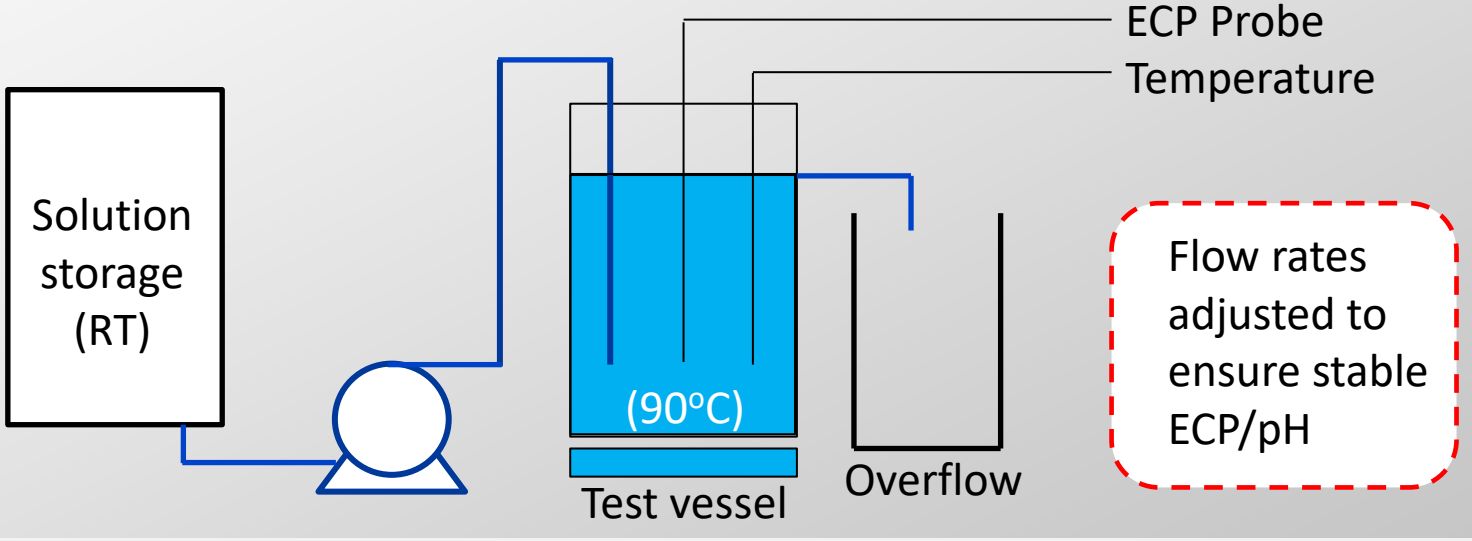
SCC Testing



- Test Material:
  - ASTM A240 Type 304L SS.
  - Cut from welded calandria tubesheet mockup: Fabricated to 1960s specs applicable to Bruce & Darlington (material, geometry, grinding, surface finish, welding procedures/parameters/apparatus, etc).
- Test Specimens
  - 32 x 1.25 x 4 mm 4PB specimens
  - Surface finish: as manufactured or polished to 1200 grit targeting ground and non-ground CRD internal surfaces.
- Tests: ASTM G39, four-point bend tests, static load ~ 1.1xYS.
- Total of 60 specimen tested





Flow rates adjusted to ensure stable ECP/pH

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SCC Test Matrix

Domain where $[NO_3^-]:[Cl^-]$ allows SCC initiation

pH	NO_3^- (ppm)	Cl- (ppm)	NO_3^-:Cl- (ppm/ppm)	Fe^{3+} (M)	Surface condition
2	620	2000	0.31	-	as received
					#1200
				$5x10^{-6}$	as received
					#1200
				$1x10^{-5}$	as received
					#1200
		10000	0.062	-	as received
					#1200
				$1x10^{-6}$	as received
					#1200
				$5x10^{-6}$	as received
					#1200

pH	NO3- (ppm)	Cl- (ppm)	NO_3^-:Cl- (ppm/ppm)	Fe^{3+} (M)	Surface condition
4	6.2	2000	0.0031	-	as received
					#1200
				$2x10^{-4}$	as received
					#1200
	620	2000	0.31		as received
					#1200
	6.2	10000	0.00062	-	as received
					#1200
				$1.5x10^{-4}$	as received
					#1200
	80000	10000	8	-	as received
					#1200

Condensate pH Fe^{3+} is stable

Condensate pH Fe^{3+} is unstable (Fe_2O_3)

Domain where $[NO_3^-]:[Cl^-]$ does not allow SCC initiation

- 2 pH values: one for bulk liquid moderator; one for drying condensate drop (different Fe^{III} stability).
- Moderate chloride and high chloride.
- Systematically increase $[Fe^{3+}]$ until ECP moves into SCC susceptibility regime.

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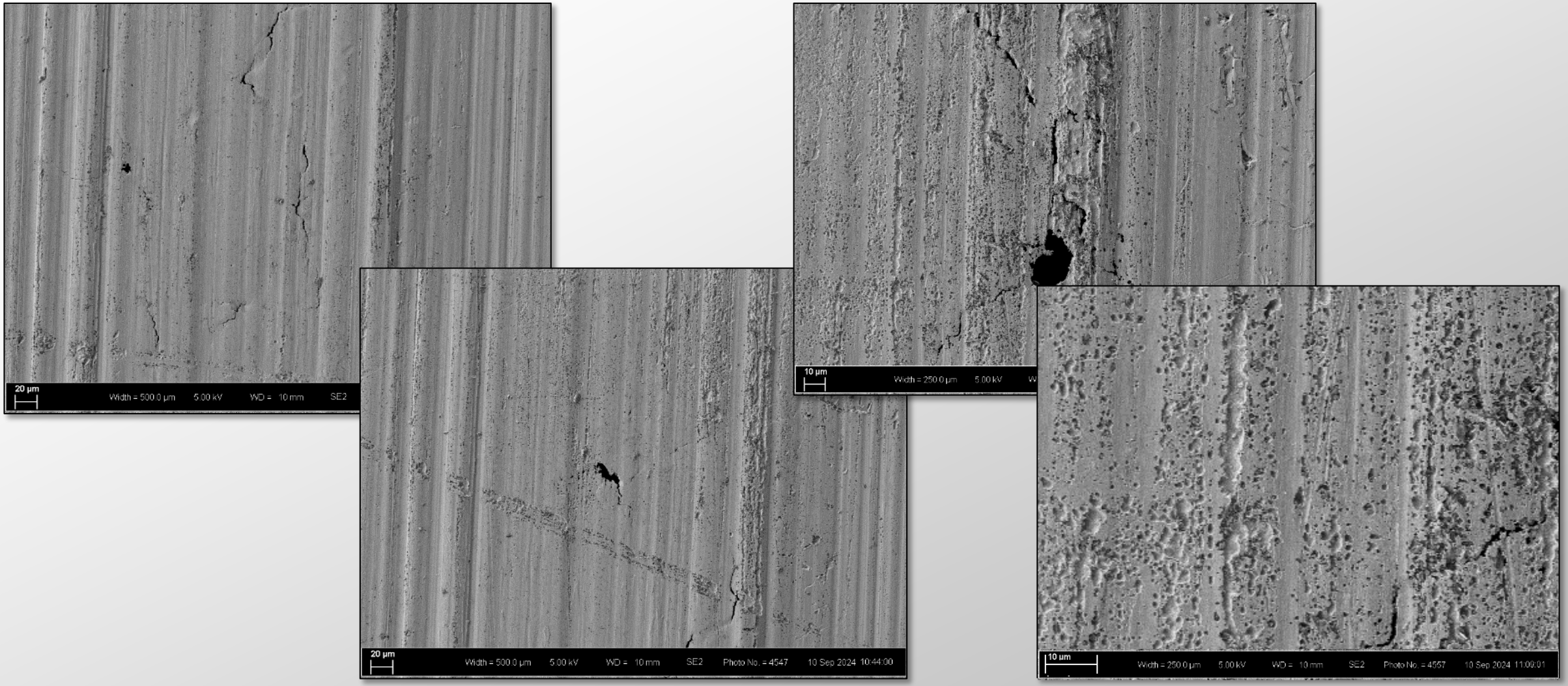
SCC Tests – pH 2

- pH = 2 (Fe^{3+} stable); 2000 ppm Cl^- (moderate Cl^-); $[NO_3^-]:[Cl^-] < 2$ (SCC initiation allowed); $1x10^{-5}$ M Fe^{3+} (highest Fe); 400 hrs.



- SCC tests appear consistent with electrochemical predictions.

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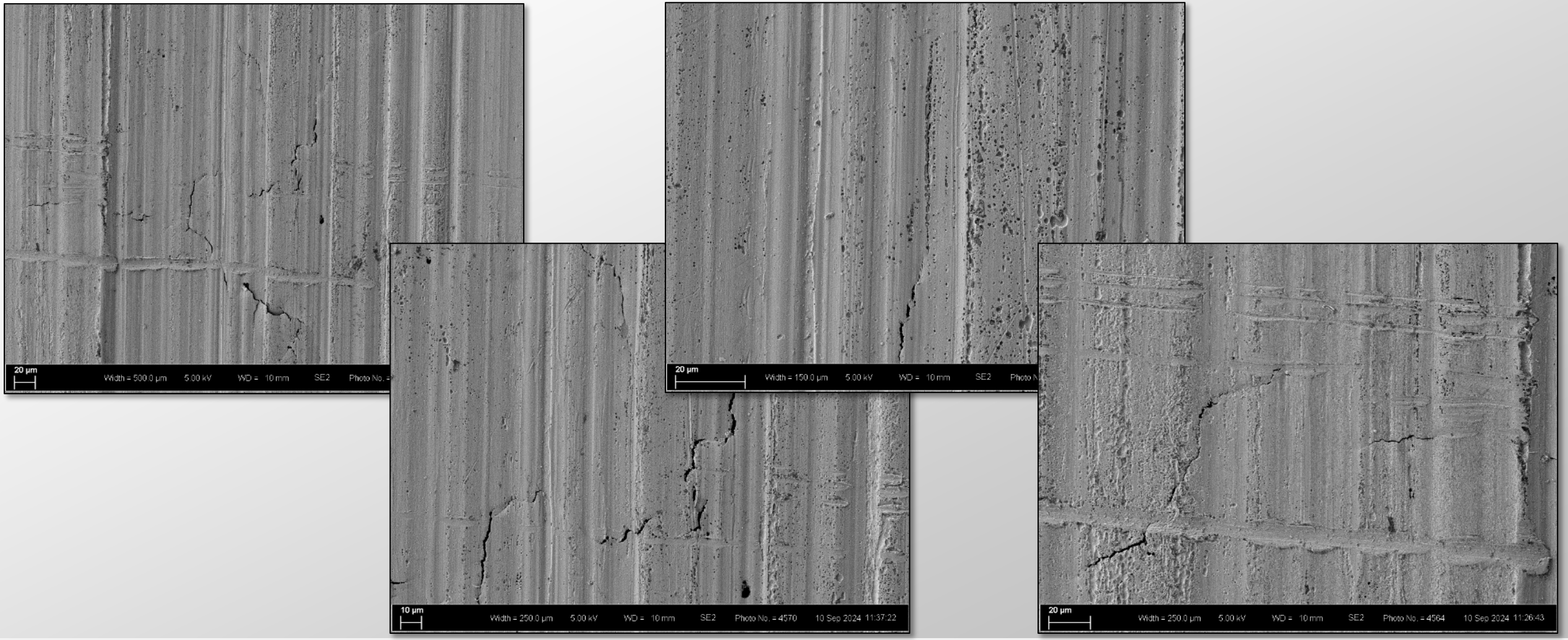
SCC Tests – pH 2

- Specimen re-immersed in solution with lower [Fe^{3+}].
- $1x10^{-5}$ Fe^{3+} 400h → $5x10^{-6}$ M Fe^{3+} 700h.
- Propagation of existing cracks without initiation of new cracks.



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SCC Tests – pH 2 Summary

pH	NO_3^- (ppm)	Cl- (ppm)	NO_3^-:Cl^- (ppm/ppm)	Fe^{3+} (M)	Surface condition	Test duration (h)	Result
2	620	2000	0.31	-	as received	1500	No SCC
					#1200		
				$5x10^{-6}$	as received		
					#1200		
				$1x10^{-5}$	as received	400	SCC
					#1200		
				$1x10^{-5} \rightarrow 5x10^{-6}$	as received	400 + 700	SCC propagation No new SCC initiation
					#1200		
		10000	0.062	-	as received	1500	No SCC
					#1200		
				$1x10^{-6}$	as received	900	
					#1200		
				$5x10^{-6}$	as received	1500	SCC
					#1200		

- The presence of sufficient [Fe^{3+}] is required for SCC initiation (not explicit from electrochemistry).
- SCC not guaranteed even in the presence of high Cl & oxidizing conditions (mitigating effect of NO_3).
  - Requires favorable [NO_3^-]:[Cl^-] ratio and sufficient Fe^{3+}.
- [Fe^{3+}] threshold to SCC initiation does not mitigate crack propagation.

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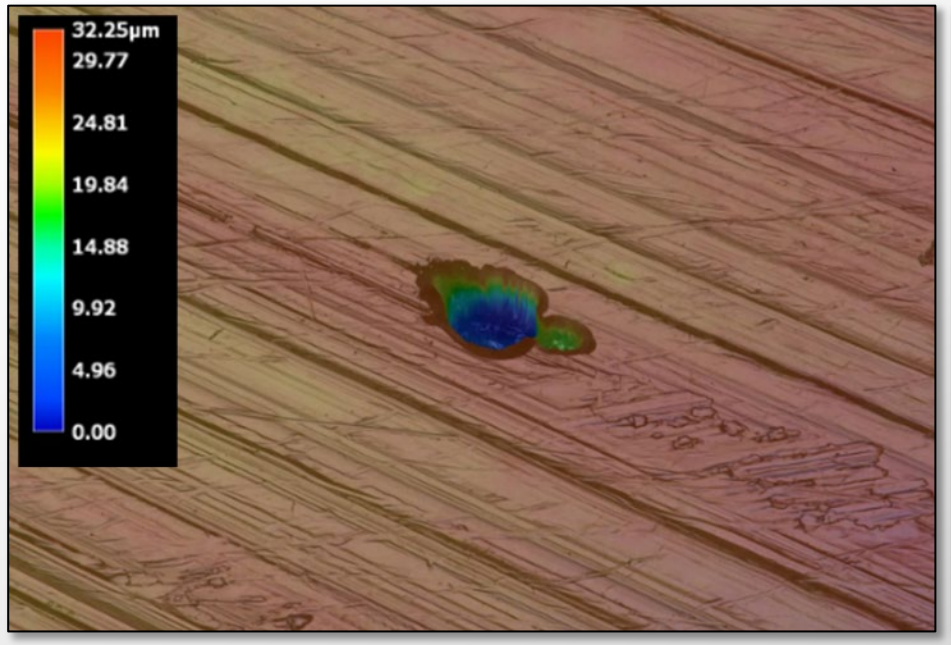
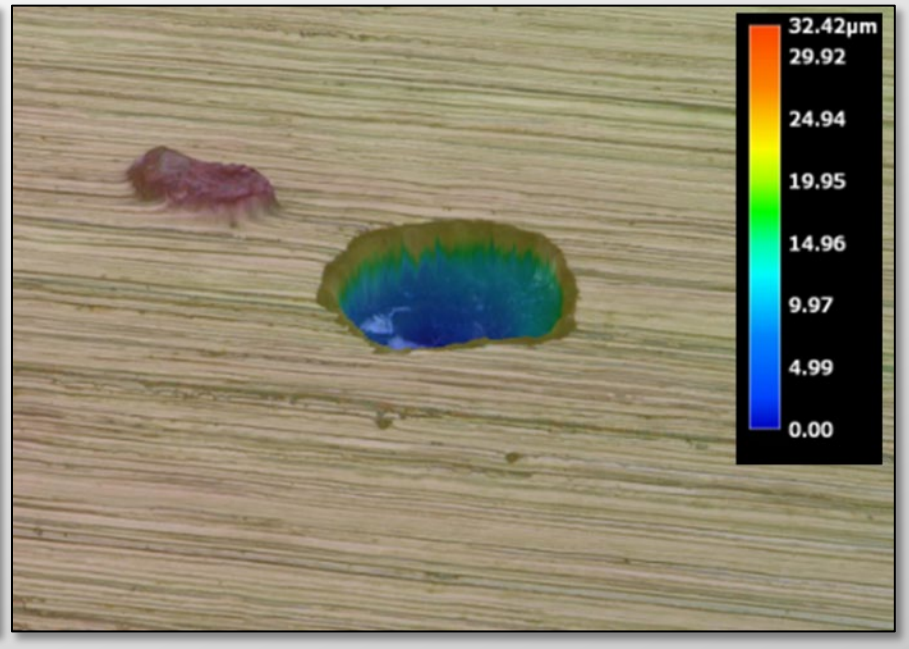
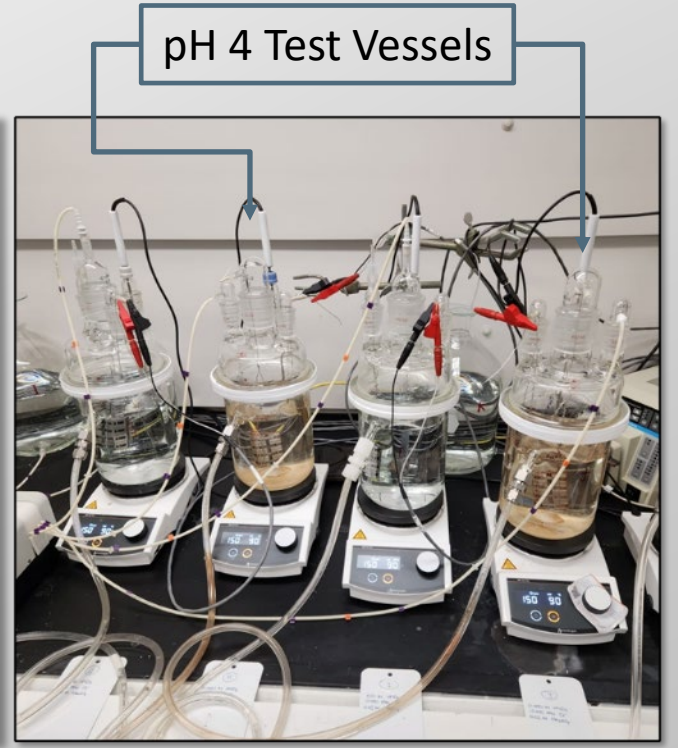
SCC Tests – pH 4

- pH = 4 (Fe^{3+} not stable).
- 2000 and 10000 ppm Cl^- (Moderate and high Cl^-).
- $[NO_3^-]$:$[Cl^-]$ < 2 (SCC initiation allowed).
- 1.5-2.0x10^{-4} M Fe^{3+} (All Fe concentrations).







- Only pitting observed. No SCC.

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SCC Tests – pH 4 Summary

pH	NO3- (ppm)	Cl- (ppm)	NO_3^-:Cl^- (ppm/ppm)	Fe^{3+} (M)	Test duration (h)	Result
4	6.2	2000	0.0031	-	1500	Pitting
				$2x10^{-4}$	900	
	620	2000	0.31			
	6.2	10000	0.00062	-	2500	
				$1.5x10^{-4}$	900	
	80000	10000	8	-	2500	No indications

- Regardless of test conditions, Fe^{3+} precipitates as Fe_2O_3 (<2 hrs).
- No Fe^{3+} → No elevation of ECP → No SCC initiation.
- Only pitting observed (provided $[NO_3^-]$:$[Cl^-]$ <2).

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Conclusions

Summary of Mitigation R&D

Understand the Degradation

- Competing effects of Cl^- and NO_3^-
- Deleterious effect of Fe^{3+}
- Pitting threshold $[NO_3^-]:[Cl^-]$ 2.

Proof of Concept

- $[NO_3^-]:[Cl^-] < 2$ requirement for SCC.
- Additional Fe^{3+} threshold requirement for SCC initiation.
- Once SCC initiated, Fe^{3+} not required for propagation.

Actionable Solution

- High chloride ≠ high SCC risk.
- SCC propagation when Fe contam removed.
- Assess SCC risk by surface chemical analysis.
- Inform if/when cleaning required and effectiveness of cleaning.
- Potential means of analysis:
  - UT couplant analysis,
  - Deposit scrape analysis.

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Mitigating Stress Corrosion Cracking In Spent Nuclear Fuel Dry Storage Casks

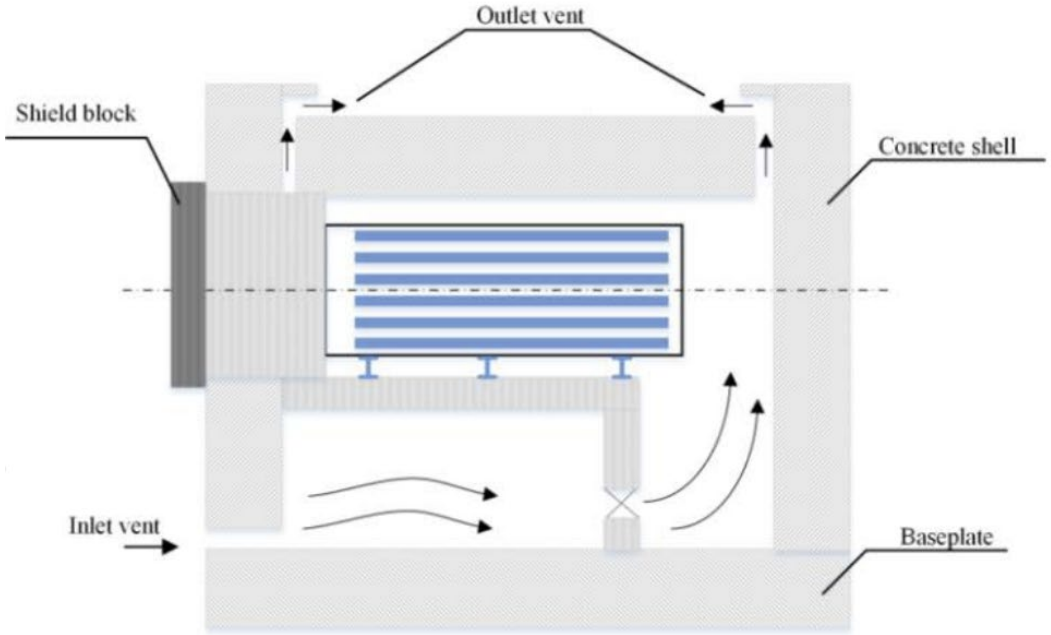
October 29, 2025

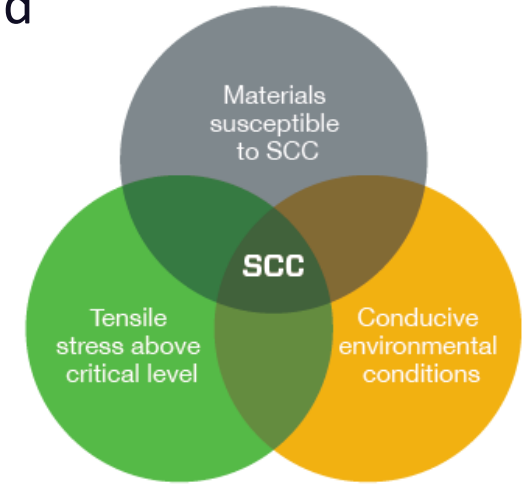
Lauryn Baranowski, Brian France, Trudy Scholten,
Justin Feis, Sven Bader, Prakash Narayanan,
Muhammet Ayanoglu, Jamie Beard

TDA Research Inc. | Golden, CO 80403 | www.tda.com

Introduction

- TDA is working to develop a radiation- and heat-resistant coating that can mitigate stress corrosion cracking (SCC) in spent nuclear fuel storage canisters
- TDA had significant experience in developing inhibitors and coatings to prevent SCC in aluminum alloys
- Work funded under DOE SBIR program starting in 2020
- Partnered with Orano and RTT for demonstration and large-scale testing activities
- Current work is focused on demonstration efforts:
  - Remote application of coating
  - Integration with RTT robotic inspection crawler
  - Coating application on Orano demo canister





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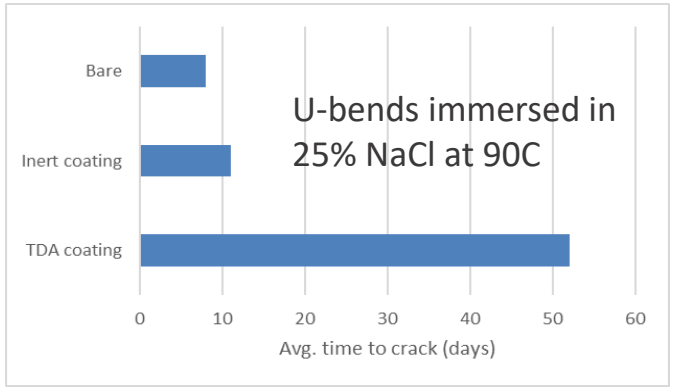
Previous Work

Initial feasibility

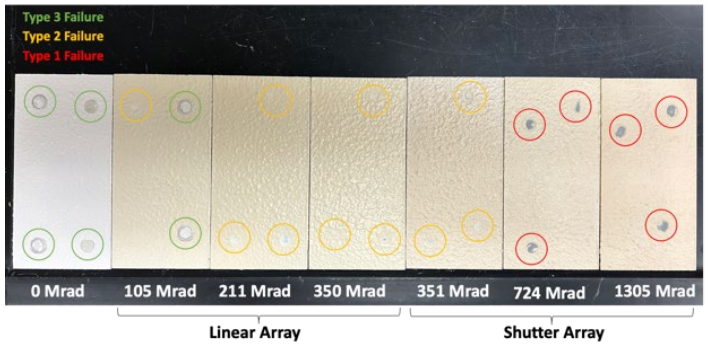
- Identified promising inhibitors and combinations of inhibitors for stainless steel
- Demonstrated inhibitor effectiveness when integrated into a coating



5-6x extension in time to cracking

Development

- Integrated inhibitor packages into a commercial heat- and radiation-resistant epoxy
- Validated inhibitor effectiveness after heat and gamma radiation exposures
- Demonstrated minimal changes to coating physical properties after realistic exposures



Validation

Performed validation testing to determine suitability for use as a mitigation solution for DCSS:

- Good adhesion when applied at surface temperatures up to 120°C
- Evaluated cleaning requirements
- Minimal off-gassing
- Compatible thermal properties
- Passed He leak test

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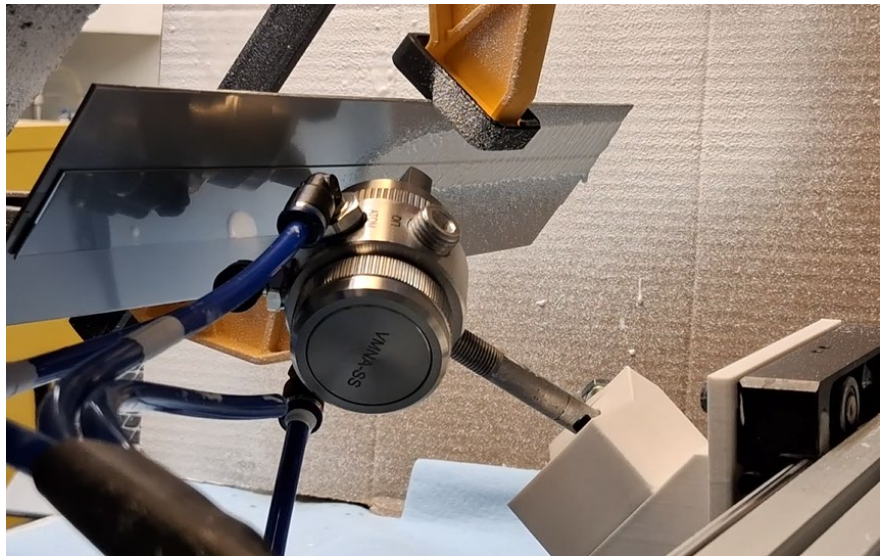
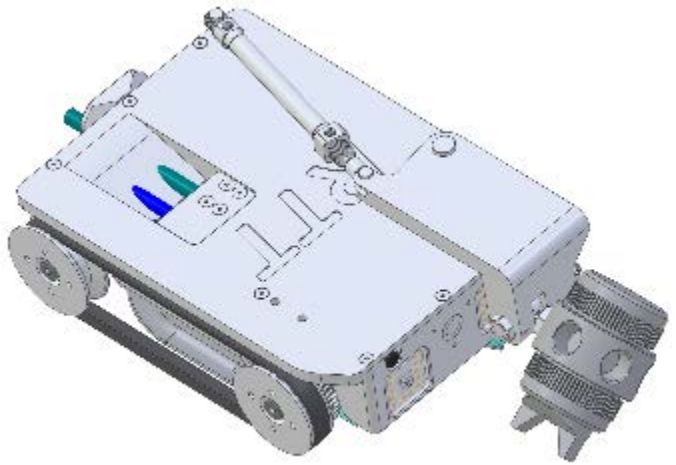
Results



Remote application of coating



Integration of remote spray head with RTT inspection crawler

Test plan for first demonstration

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Remote application at TDA

- TDA built a remote application system for our coating using a commercial spray head attached to a programmable articulating arm
- Initial demonstration used one-axis motion
- Air and coating were managed from a stationary supply
- Variable application parameters:
  - Horizontal speed of spray head
  - Air pressure
  - Feed (coating) pressure
  - Distance of spray head from substrate

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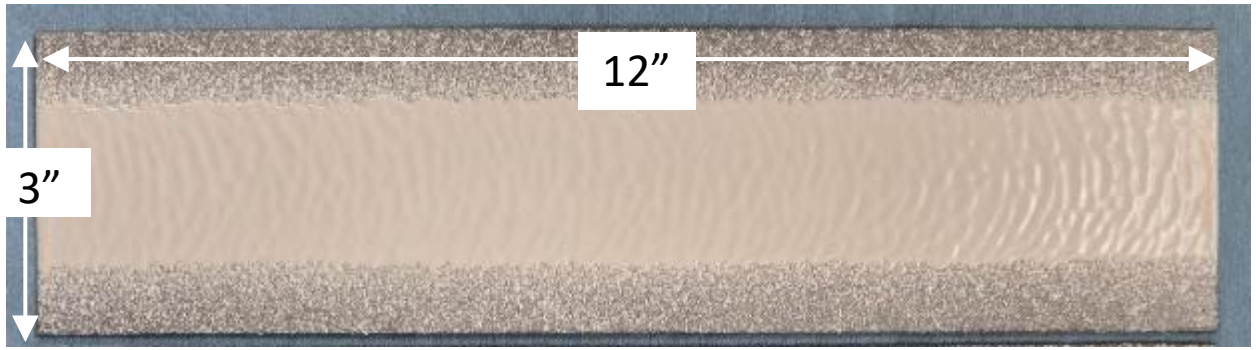
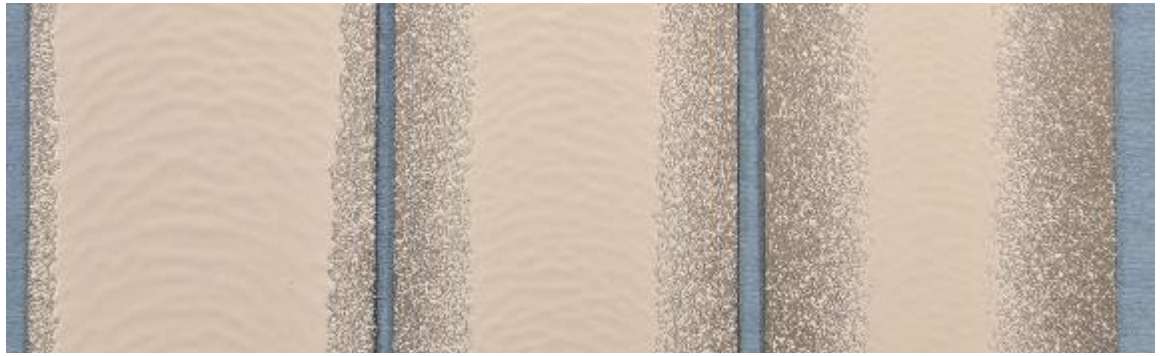
Coating thickness and uniformity

- Coating thickness and width of coating strip are both controlled by:
  - Spray head speed
  - Feed (coating) pressure

	Hand spray	Remote application
Coating thickness	7.6-8.8 mil (193-223 µm)	6.4-11.8 mil (163-300 µm)
Thickness variation	≤13%	≤8%
Coating width	36-48 mm	25-50 mm
Width variation	≤7%	≤5%





150 spm 300 spm 500 spm

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Physical property comparison

- No major detrimental changes to physical properties with remote application
- Improved adhesion with exception of one sample

Property	Hand-sprayed coatings (avg.)	Remotely applied coatings (avg.)	Change compared to hand-sprayed
Abrasion resistance	34.9 mg/1000 rev	33.5 mg/1000 rev	-4%
Pull-off adhesion	4.12 MPa (3.38-4.82 MPa)	5.89 MPa (3.96-7.05 MPa)	+43%
Scratch hardness	4H	4H	x
Gouge hardness	5H	5H	x

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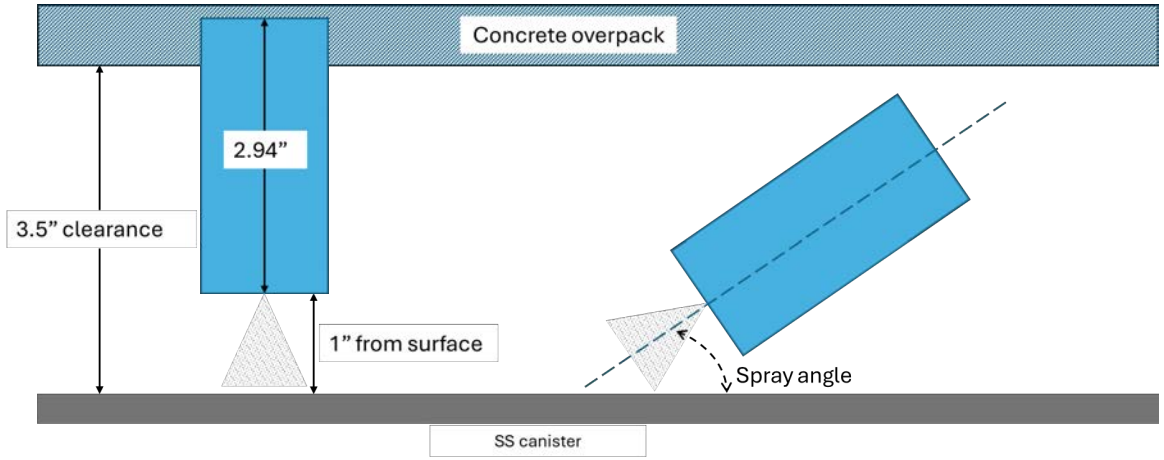
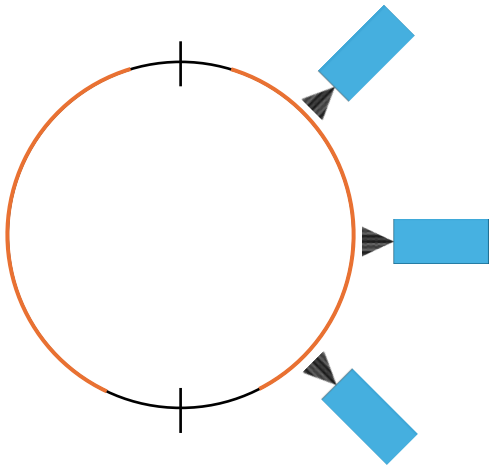
Testing with smaller spray head

- Due to space restrictions inside overpack, we selected and purchased a smaller spray head (old: 4.1” long, new: 2.94” long)
- Performed testing to explore geometric constraints:
  - Spray angle <90°
  - <1” from surface (0.8” target)
  - Varying application angle due to positions of longitudinal welds





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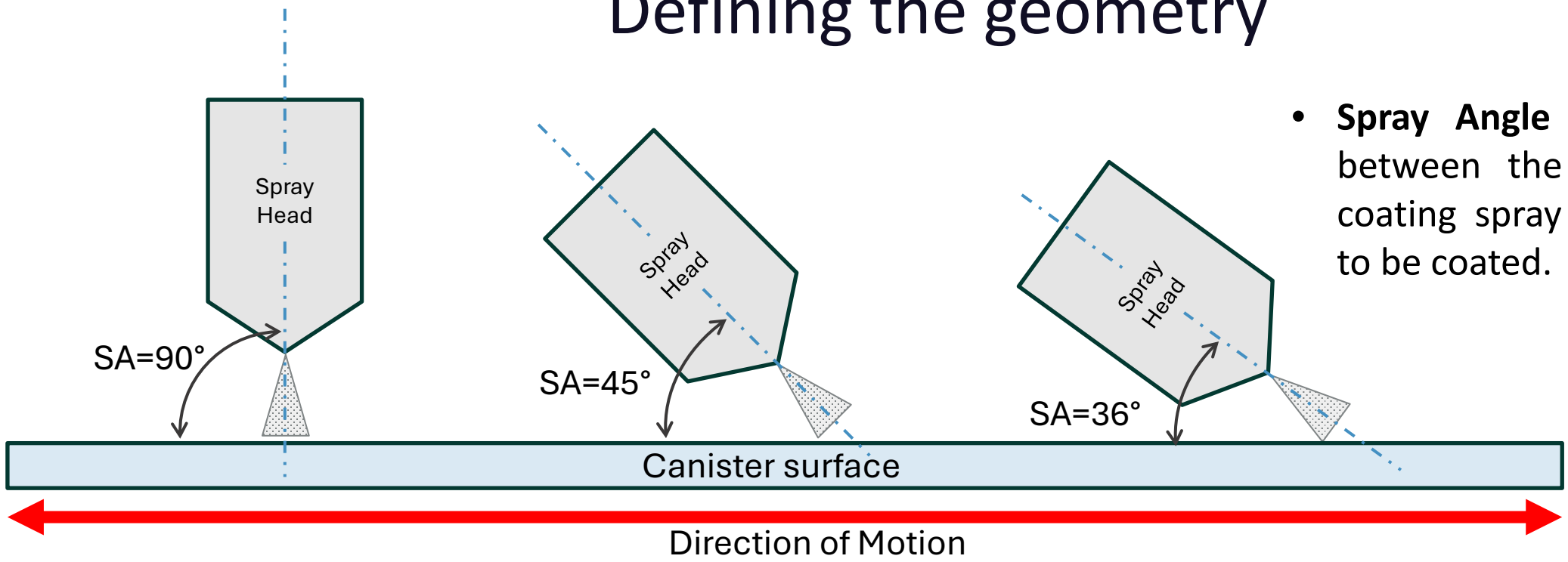
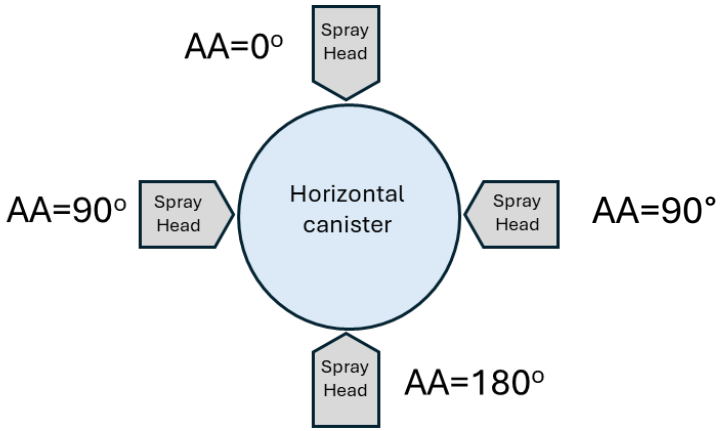
Defining the geometry



- **Spray Angle (SA):** the angle between the midline of the coating spray and the surface to be coated.

- **Application Angle (AA):** the angle of the spray head relative to the typically perceived vertical and horizontal. Orthogonal to spray angle.



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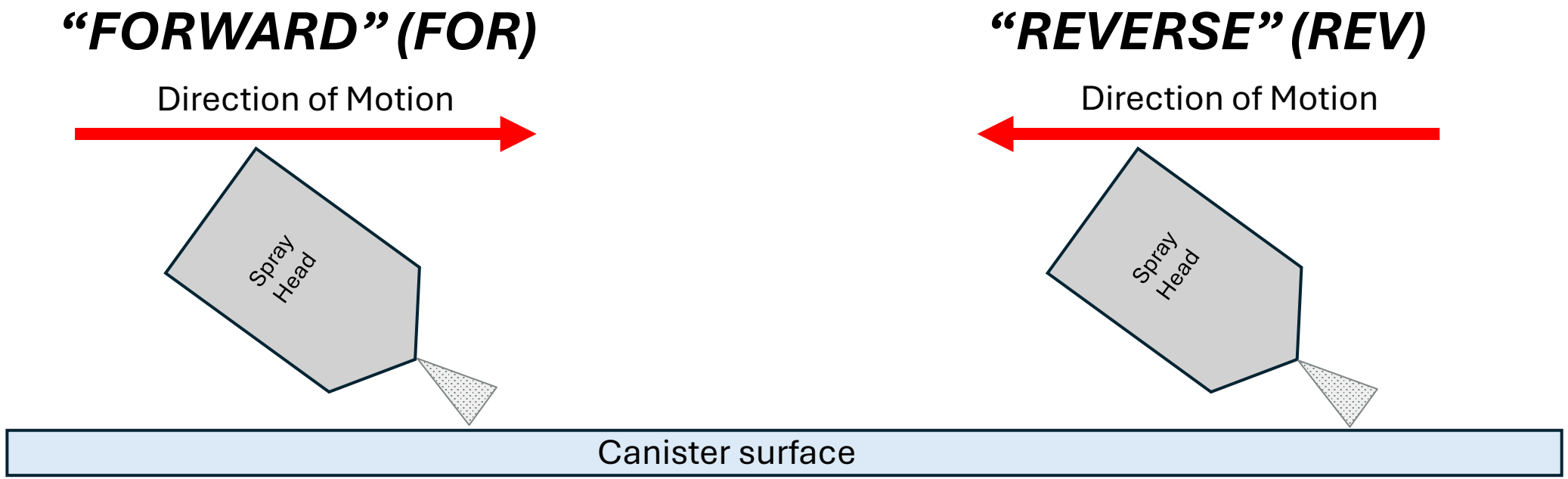
Defining the geometry

Tested two different directions of motion of the spray head during coating:

- **“Forward” (FOR)** – direction of spray head motion is the SAME as the spray direction
- **“Reverse” (REV)** – direction of spray head motion is OPPOSITE the spray direction



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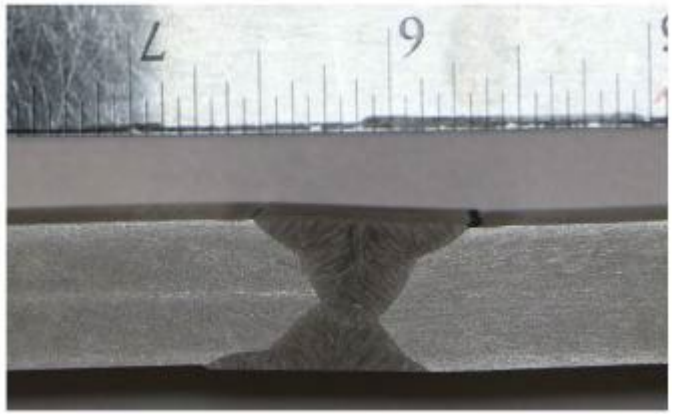
Optimizing application parameters

Target values

- Spray width: 1.5”
- Coating thickness: 5-10 mil (125-250 microns)



Weld cross-section

Setting/parameter	Range explored	Recommended for use
Air pressure	5-15 psi	15 psi
Coating pressure	10-30 psi	20 psi
Distance from nozzle to surface	0.8”-3”	0.8”
Spray angle	12°-90°	36°
Spray head direction of motion	FOR/REV	REV
Application angle	0°-135°	0°-135°
Spray head speed	0.27”/sec – 1”/sec	For AA≤45°, 1”/sec For AA>45°, 0.5”/sec
Spray pattern	One pass / two pass	Two pass

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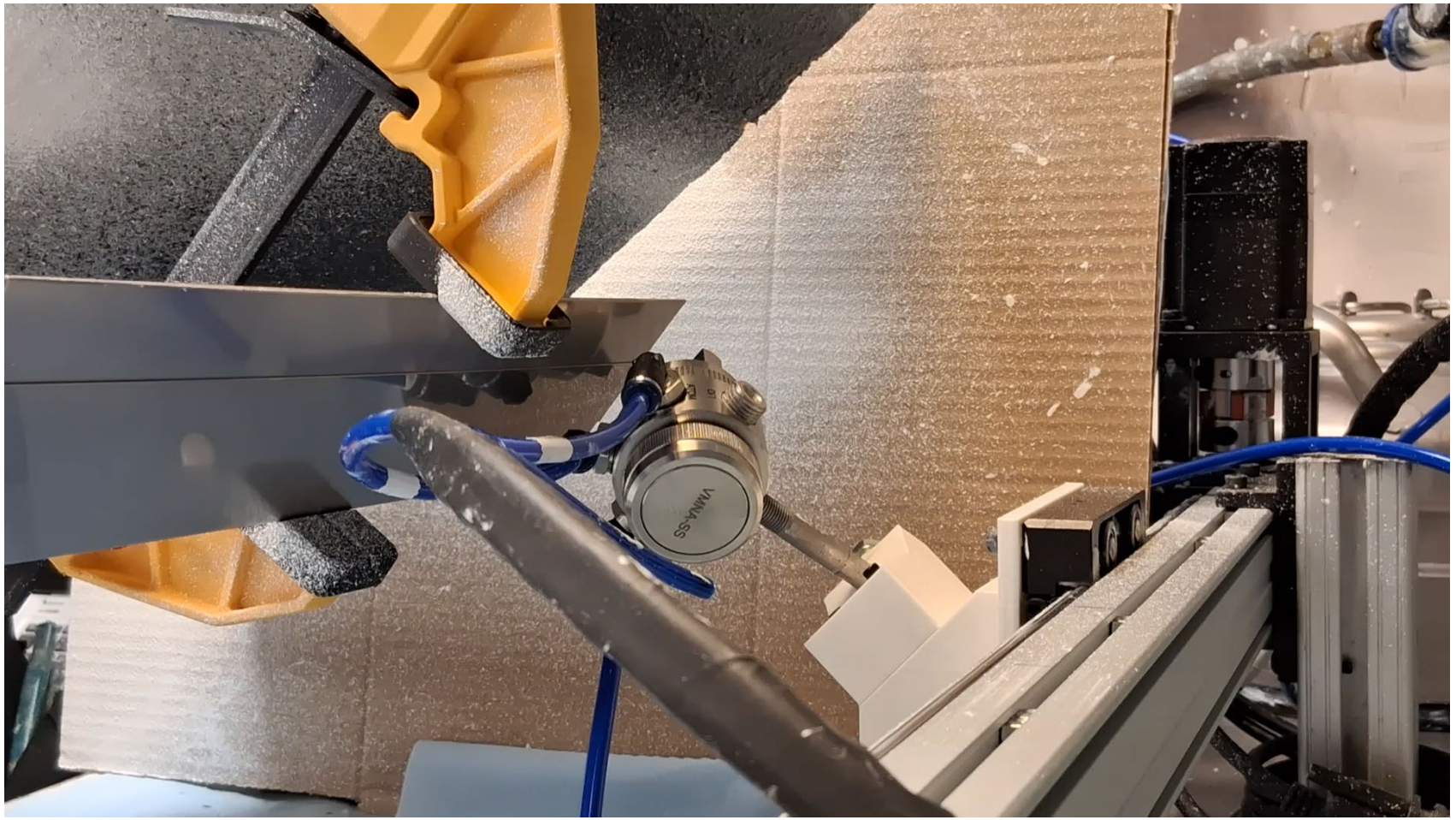
Putting it all together



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Integration of remote applicator with RTT crawler

- Since 2020, RTT has fielded crawlers to support NDE in over 50 dry storage units
  - Vertical and horizontal canisters
  - Orano, Holtec, and NAC systems
- For the Orano HSM, RTT developed a custom suction crawler for inspection and in-situ repair and mitigation

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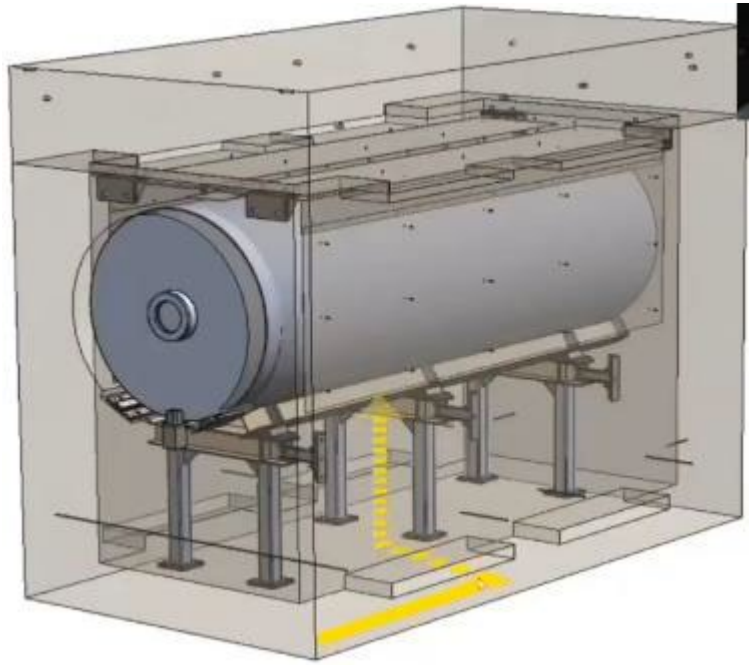
Accessing the canister

- Secondary deployment crawler platform is used to carry the suction crawler through the HSM inlet vent and lift unit up to canister shell
- Space constraints on crawler:
  - 6” wide alley
  - 5” tall vent opening
  - Low height profile (<3”) to clear internal heat shields of HSM



Deployment crawler placing inspection crawler onto dummy DSC

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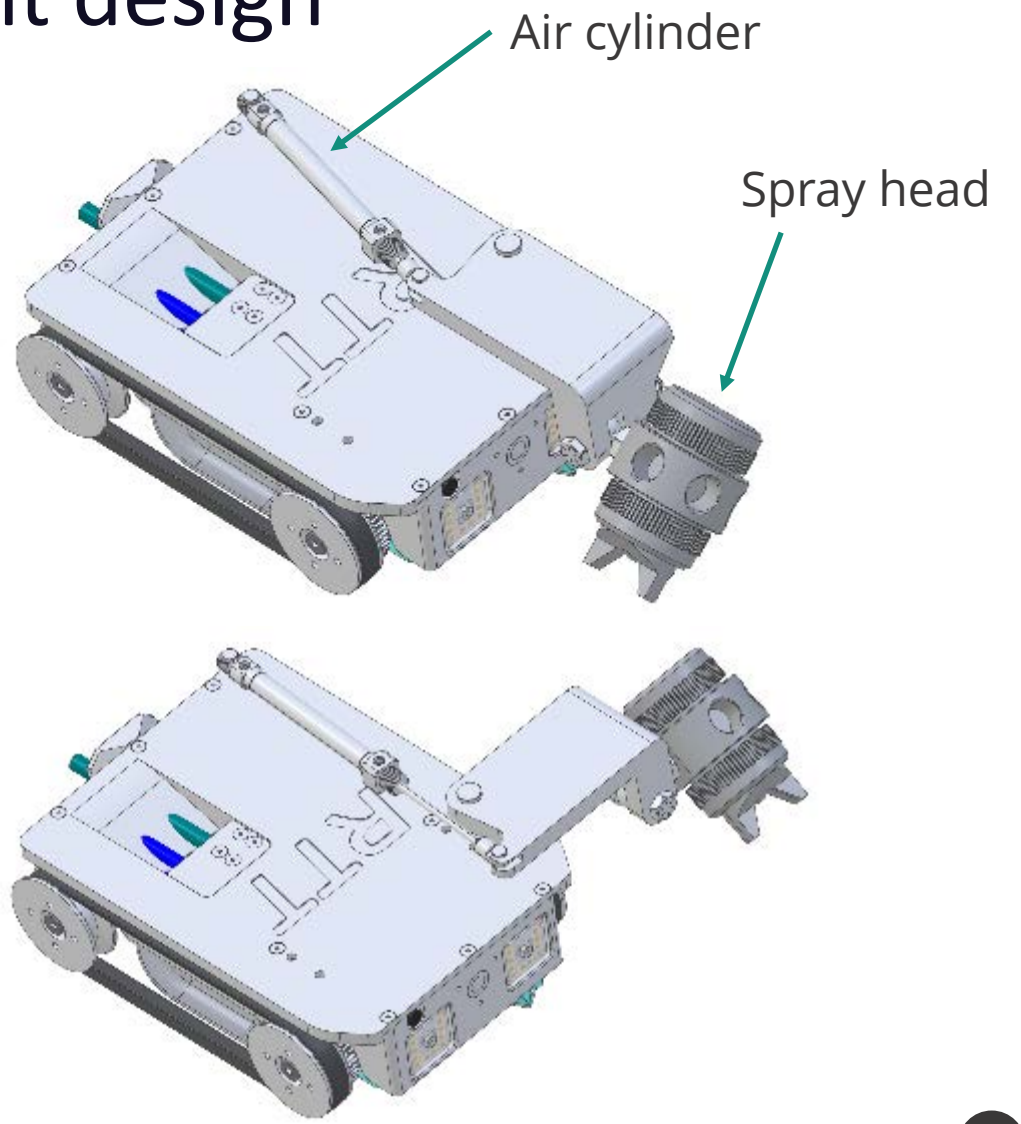
Spray head mount design

- Active mount stows spray head in front of crawler during transition through inlet vents and up to the canister
- Once on the canister, an air cylinder pivots the mount to the side of the suction crawler
- Design does not require any changes to deployment crawler or deployment procedure



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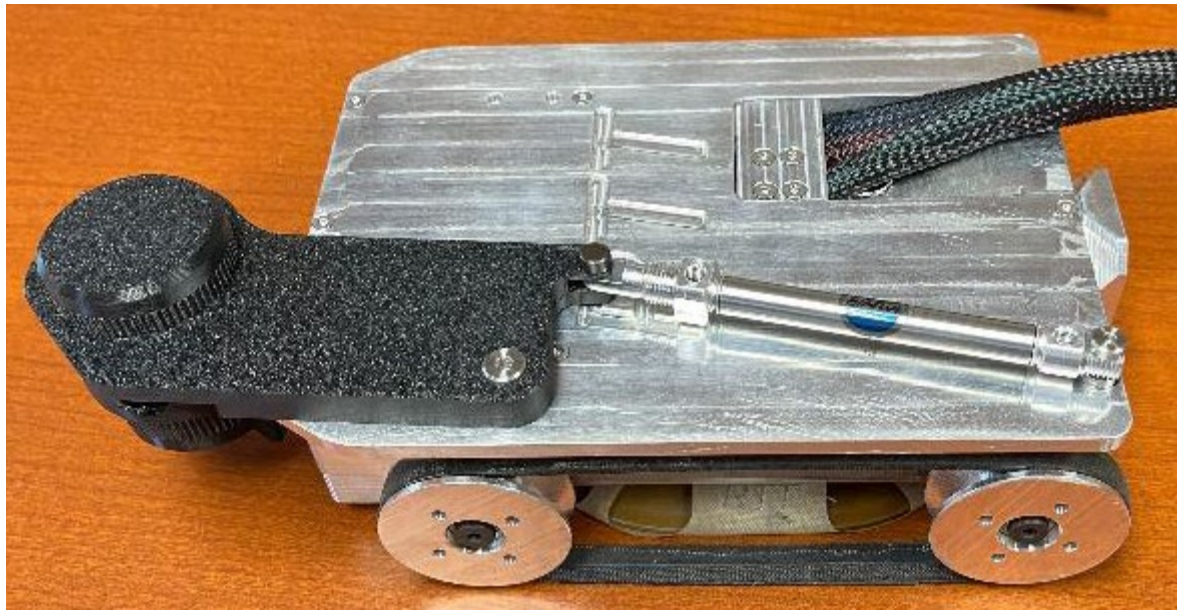
Prototype



Stowed for entry into overpack and deployment onto canister

Deployed for coating application

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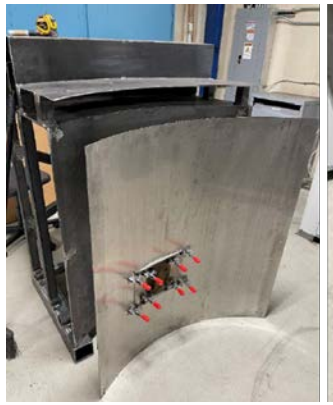
Test plan for first demonstration (Jan 2026)

- Navigate modified crawler with spray head through inlet vent and deploy onto demonstration canister
- Apply coating using two-pass technique at application angles of 45° and 135°
- Apply coating to canister mockup section with removable panel to generate samples for characterization at TDA
- Main goals:
  - Validate modified crawler entry into overpack and deployment onto canister
  - Assess maneuverability of crawler with paint sprayer and additional cables/tubing
  - Ensure that paint can be carried through tube without clogging or other flow issues
  - Generate samples of paint applied with crawler for post-demo testing



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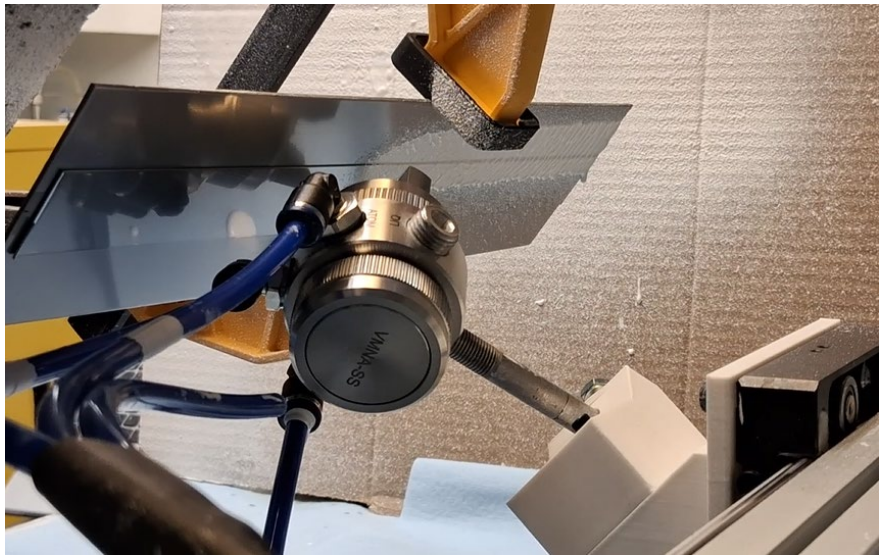
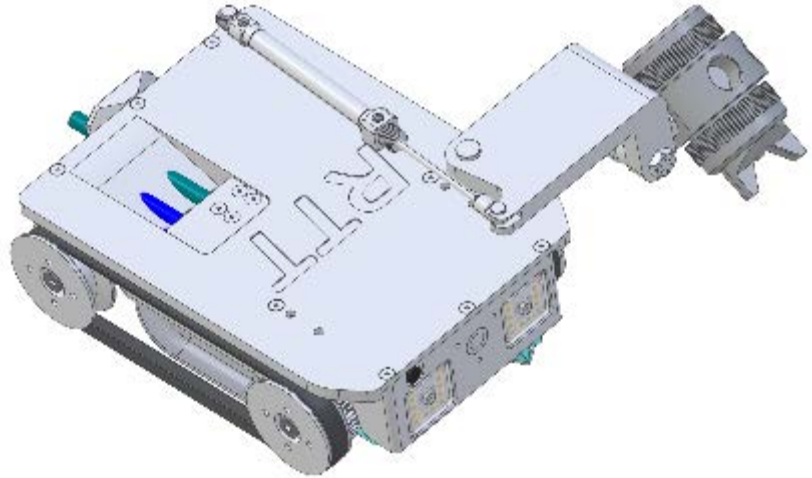
Summary

- TDA Research has developed a coating that can significantly delay SCC in stainless steel
- Coating can withstand the expected heat and radiation exposures experienced in a mitigation scenario
- Explored and optimized parameters for remote application
- RTT robotic crawler modified with automatic paint sprayer and deployment system
- First demonstration planned for January 2026





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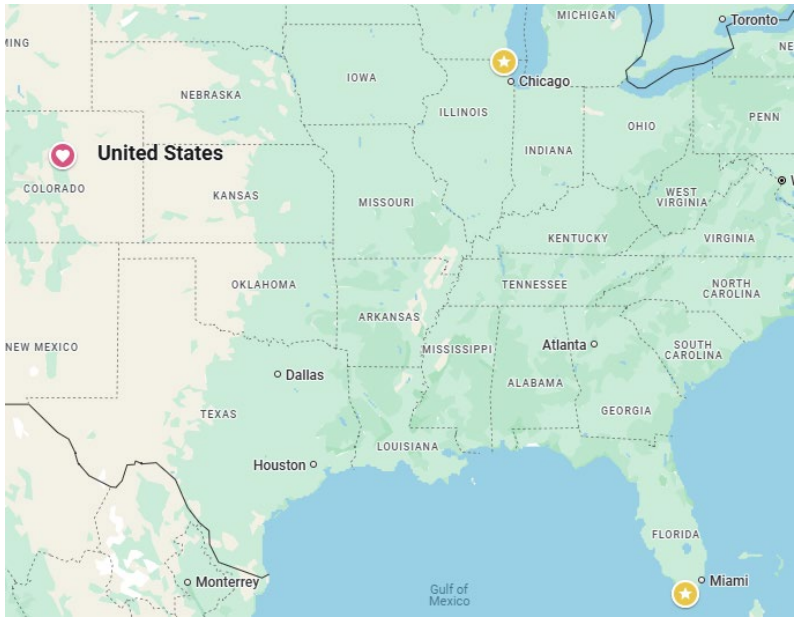
Backup slides

Test coating under realistic conditions

- Outdoor exposure testing of bare and coated U-bends
- Total of 57 U-bends
- Selected two test sites:
  - Mt. Prospect, IL (suburban area of upper Midwest where road salt is used)
  - Layton Key, FL (coastal)
- Additional U-bends to be tested in TDA salt fog shed
  - Replicates humidity and salinity of a coastal site
- Testing duration runs from 6-18 months



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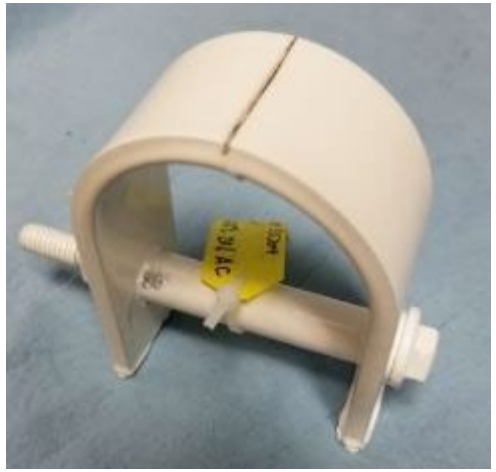
Sample matrix for Task 4

- Samples placed at test sites in February 2025
- 6-month samples have been received at TDA and evaluated



Site	Coating	Flaw?	Quantity	Notes
Coastal	Bare	x	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Coastal	Inert	Y	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Coastal	W/ inhib	Y	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Coastal	Inert	N	3	Return at 18 mo
Coastal	W/ inhib	N	3	Return at 18 mo
Inland	Bare	x	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Inland	Inert	Y	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Inland	W/ inhib	Y	5	Return one ea. at 6, 9, 12, 15, and 18 mo
Inland	Inert	N	3	Return at 18 mo
Inland	W/ inhib	N	3	Return at 18 mo
Salt Fog shed	Bare	x	3	
Salt Fog shed	Inert	Y	3	
Salt Fog shed	W/ inhib	Y	3	
Salt Fog shed	Inert	N	3	
Salt Fog shed	W/ inhib	N	3	

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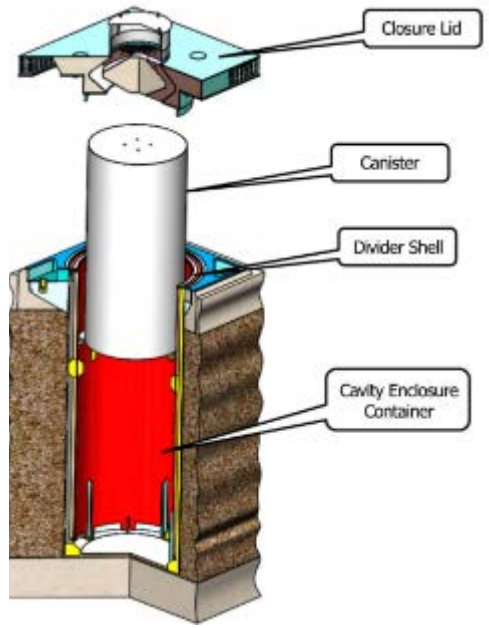
Vertical storage canisters



- Work with Orano has focused on mitigation scenario for canisters in HSM
- Over twice as much SNF worldwide stored in vertical canisters vs. horizontal
- Basic coating req's are the same, but in some cases, vertical overpacks are stored underground
  - E.g., Holtec HI-STORM 100U
- Underground storage adds more requirements for coating

Acceptance Criteria for the Selection of Coatings	
Rank	**Criteria**
1	suitable for immersion and/or below grade service
2a	compatible with the ICCPS (if used) • adequate dielectric strength • adequate resistance to cathodic disbondment
2b	compatible with concrete encasement (if used) • adequate resistance to high alkalinity
3	adequate radiation resistance
4	adequate adhesion to steel
5	adequate bendability/ductility/cracking resistance/abrasion resistance
6	adequate strength to resist handling abuse and substrate stress

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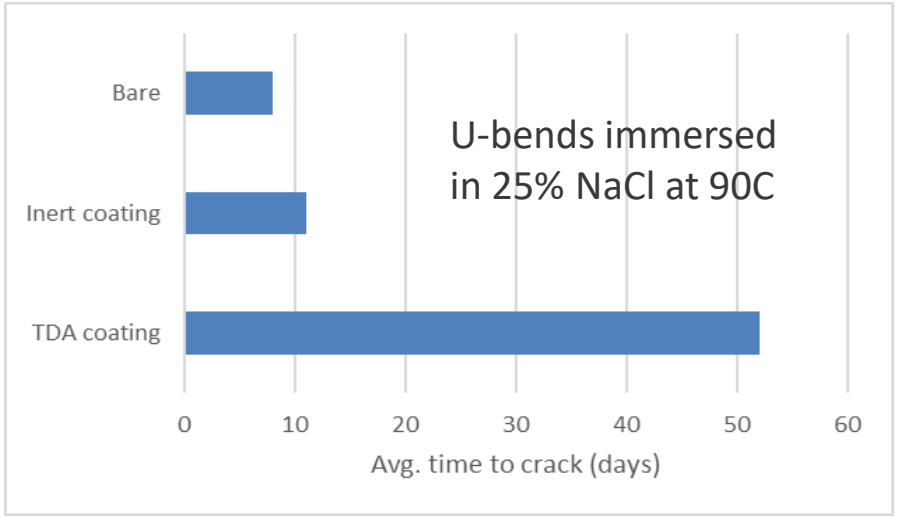
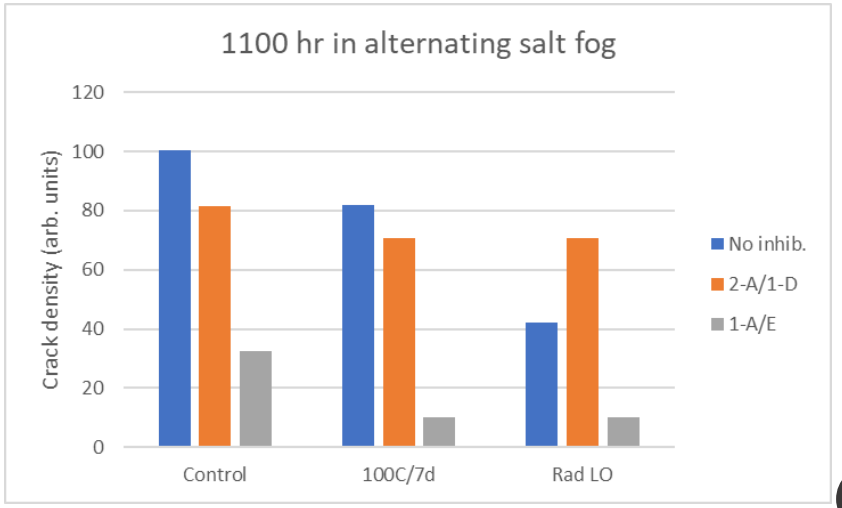
Previous work

- Demonstrated that inhibitors extend time to cracking vs. bare (6x) and inert coating (5x)
- Integrated top inhibitor packages into a commercial high temperature epoxy with many desirable properties:
  - Application and operation temperatures up to 120°C
  - Certified to ASTM D4082 ($1x10^9$ rad gamma radiation)
- Identified one inhibitor package that maintained SCC prevention properties after gamma radiation exposure (~80°C) OR exposure to 100°C for 7 days





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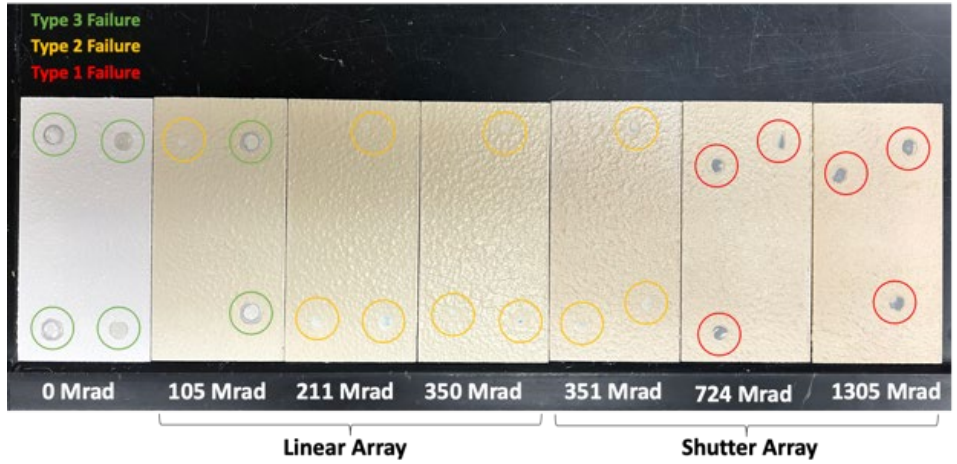
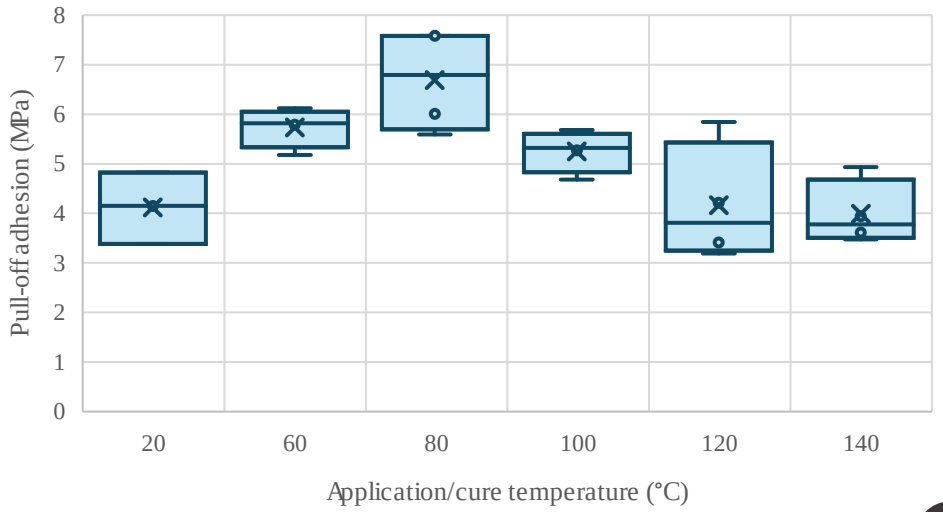
Previous Work

- Most physical properties were unaffected by heat or radiation (abrasion resistance, hardness); adhesion is reduced by ~50% at highest levels of irradiation
- Coating could be applied to hot (up to 120°C) or corroded substrates without loss of adhesion
- In-depth validation testing indicated suitability of coating for use with DCSS:
  - Minimal off-gassing
  - Compatible thermal properties
  - Passed He leak test





Nation, B., et al. FY23 Status: Corrosion-Resistant Coatings on Spent Nuclear Fuel Canisters to Mitigate and Repair Potential Stress Corrosion Cracking. Sandia National Laboratories. August 10, 2023.

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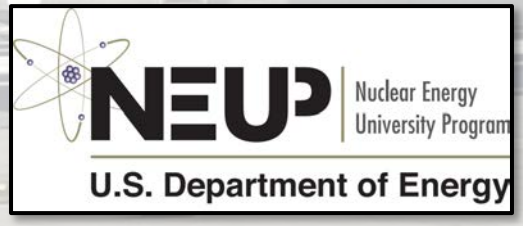
Validating Aerosol Deposition Models with Experiments in a Lab-Scale Dry Storage Cask

Andrew R. Metcalf, Ren Garity, James Henry, Moein Mousavi

John Saylor, Prasad Rangaraju, Nigel Kaye

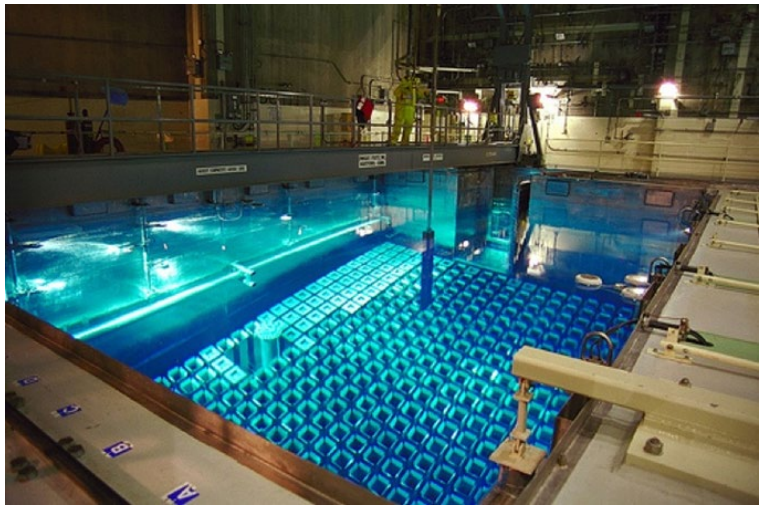
Sarah Suffield, Megan Higley, Dina Carpenter-Graffey



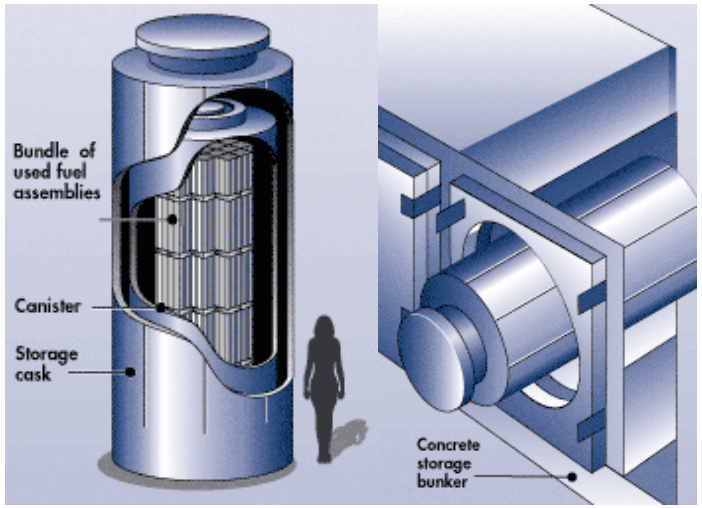
Award # DE-NE0009276

Background and Motivation

Why are aerosols relevant to nuclear waste storage?

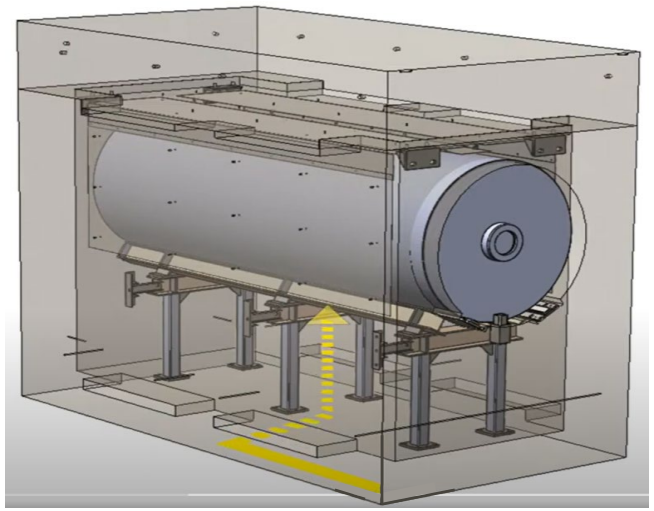


Short-term storage of spent nuclear fuel (SNF);
spent fuel pools
(five to ten years)



Long-term storage; SNF moved to **dry storage casks (DSCs)**
(decades or longer)

Orano NUHOMS System

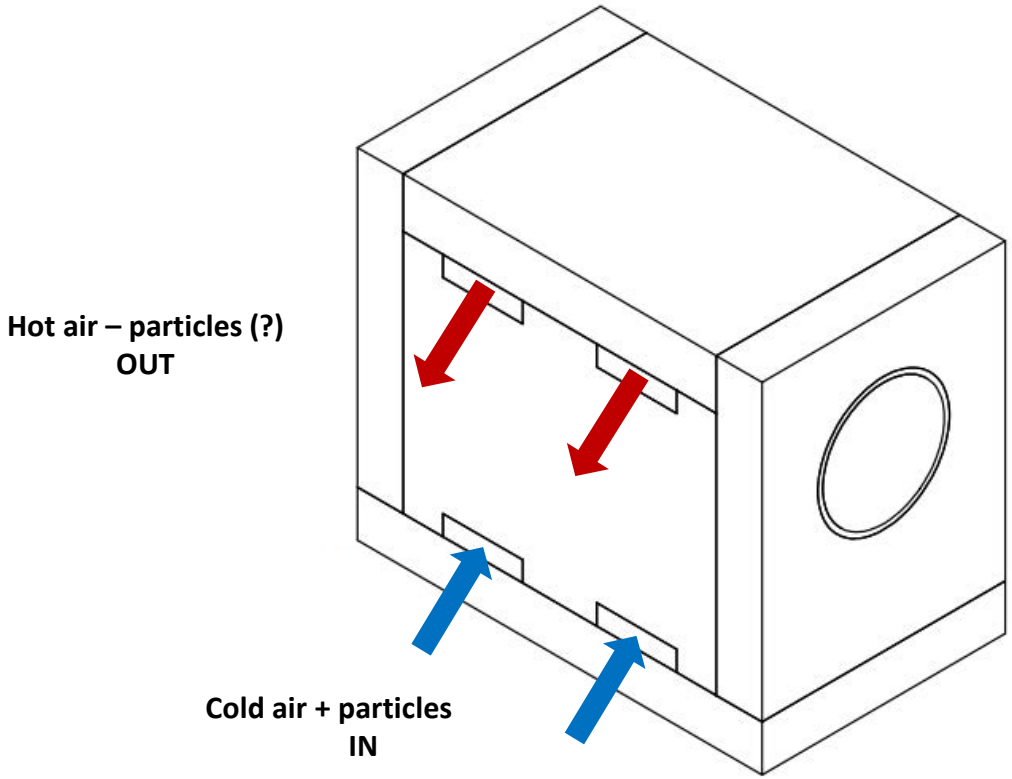


Ambient air cools contained SNF—**aerosols** allowed to **deposit** on stainless steel cylinder; effects to be determined

- High **temperatures**
- Variable **flow**
- Changes in local **weather**

Background and Motivation

Full-scale dry storage cask at Oconee Nuclear Station



Science Questions

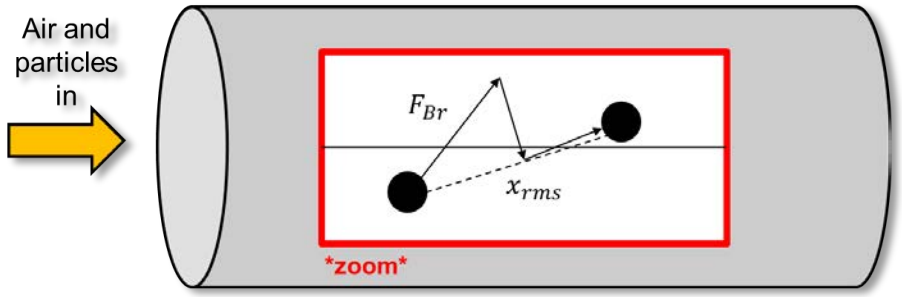
- How does aerosol deposition inside spent nuclear fuel dry storage casks affect long-term storage plans?
- Can CFD modeling accurately represent aerosol deposition mechanisms in complex environments?

What modeling has been done?

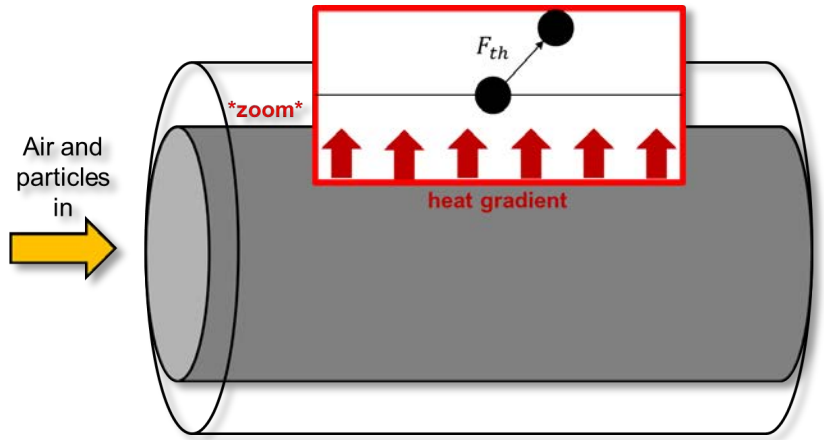
- Model of heat, turbulent flow, and aerosol deposition developed at PNNL using STAR-CCM+ model platform.
- Results indicate enhanced aerosol deposition at the top of the canister.

Suffield, et al. 2023, Report #PNNL-34772

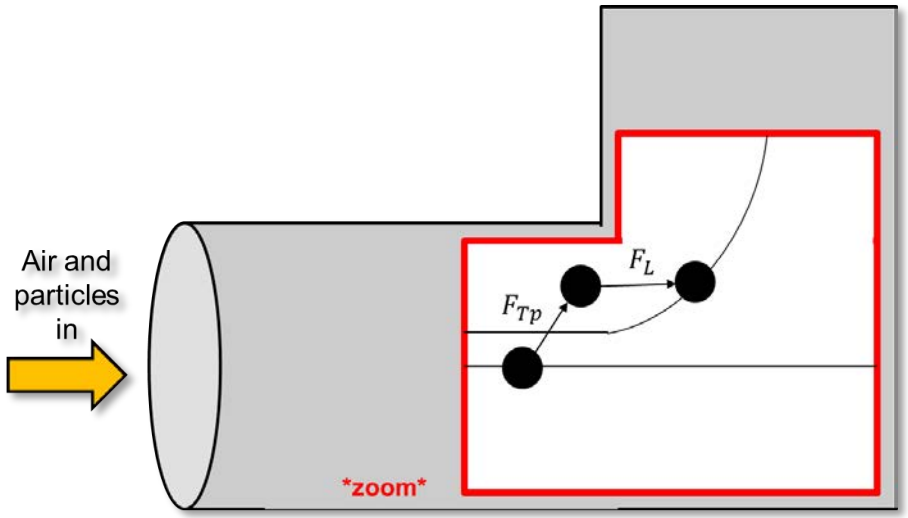
Relevant Deposition Mechanisms



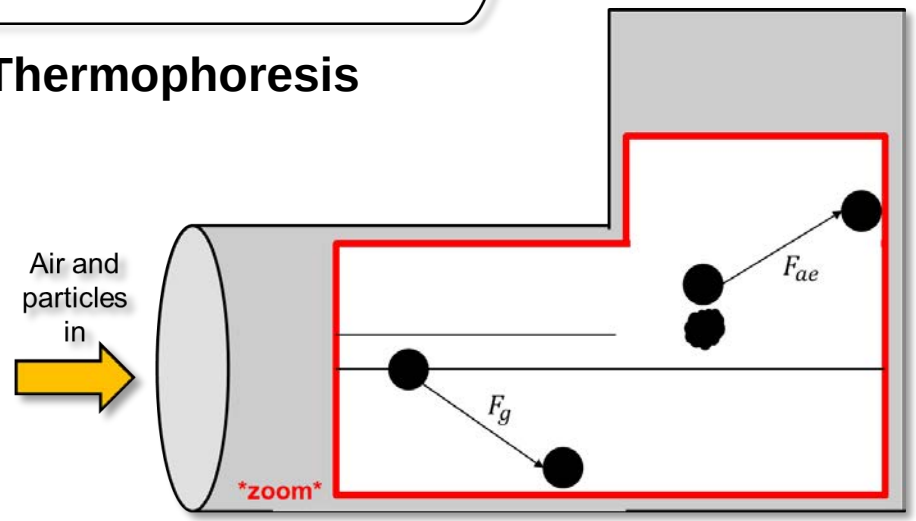
Brownian Motion



Thermophoresis



Turbophoresis and Saffman Lift

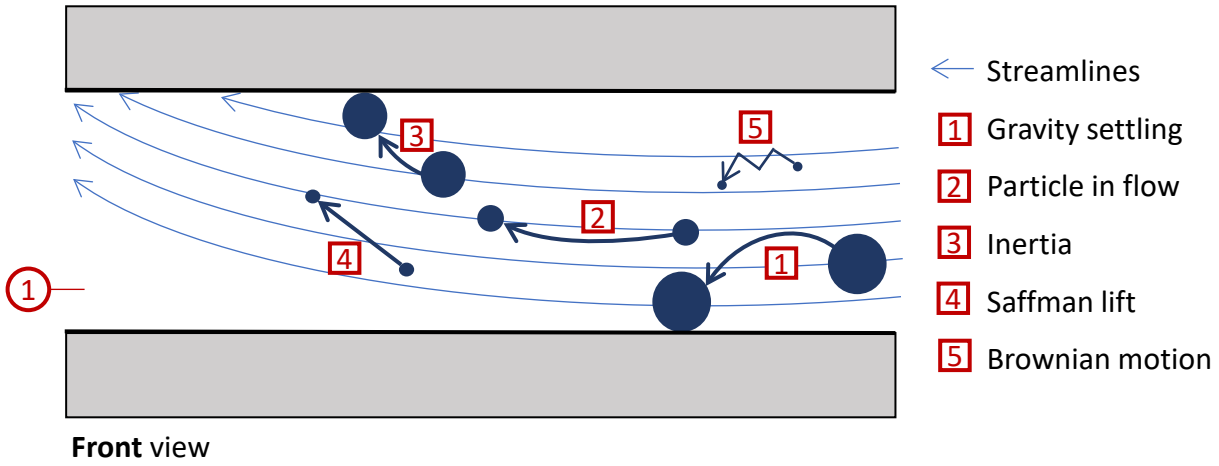


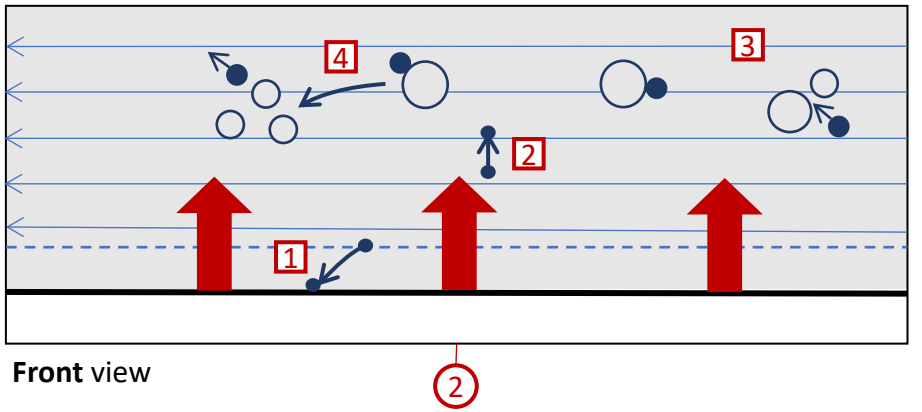
Gravitational Settling and Inertial Impaction

Garity, et al. (in prep)

Environmental Variables

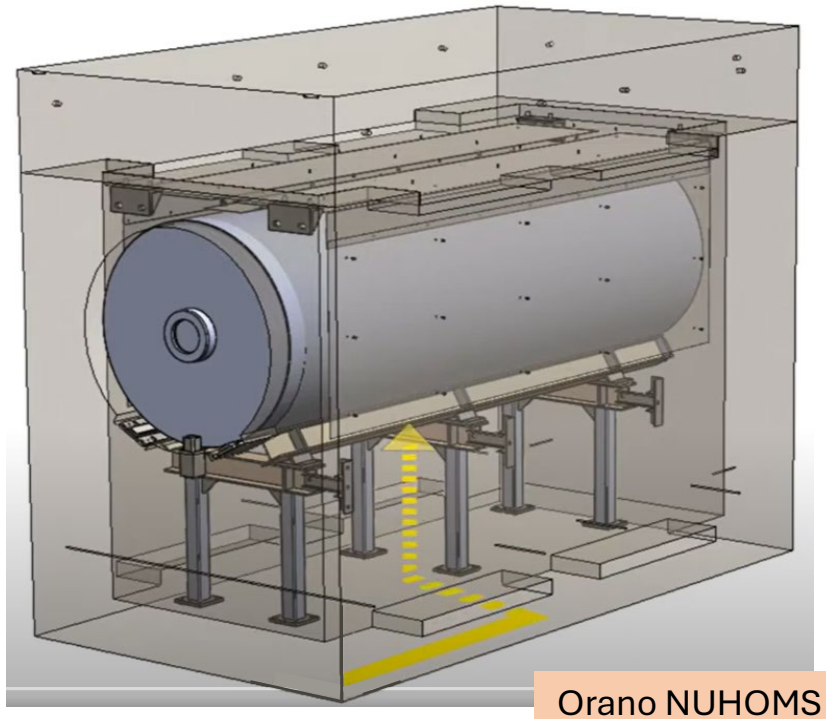
- **Temperature**
  - Thermophoresis
  - Brownian motion
- **Humidity**
  - Stefan flow/diffusiophoresis
- **Flow**
  - Saffman lift
  - Turbophoresis





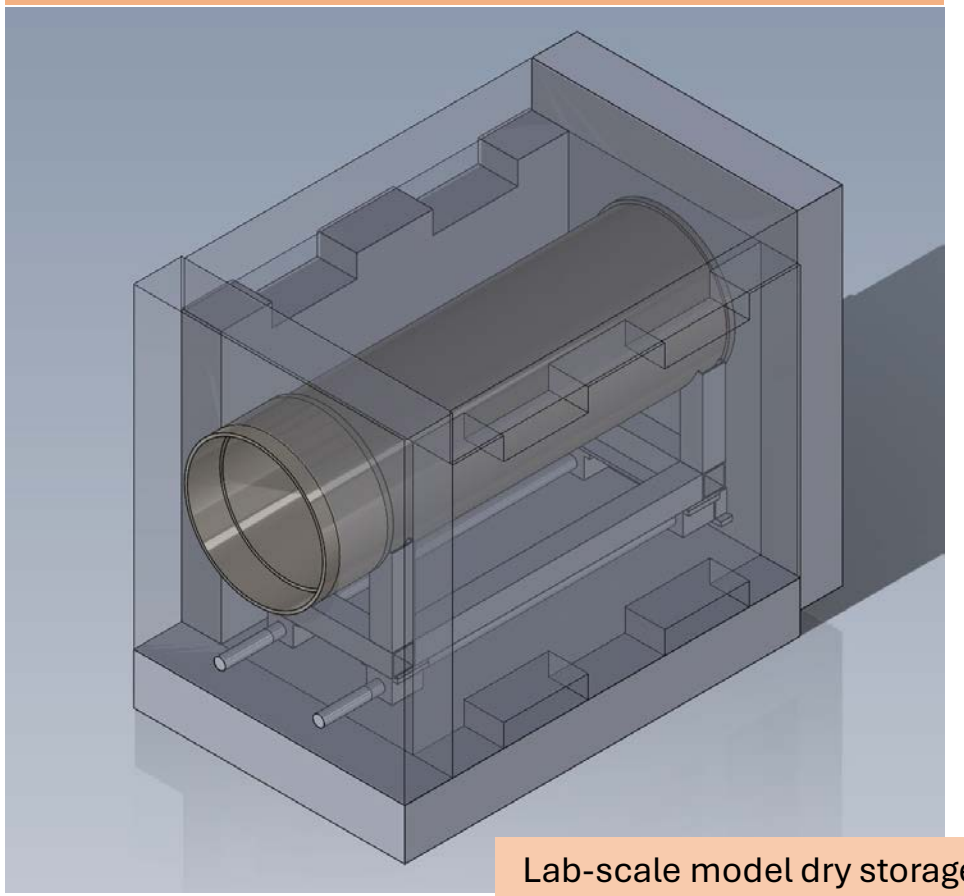
Lab-scale Dry Storage Cask

Full-scale (cylinder length 18 feet / 6 meters)



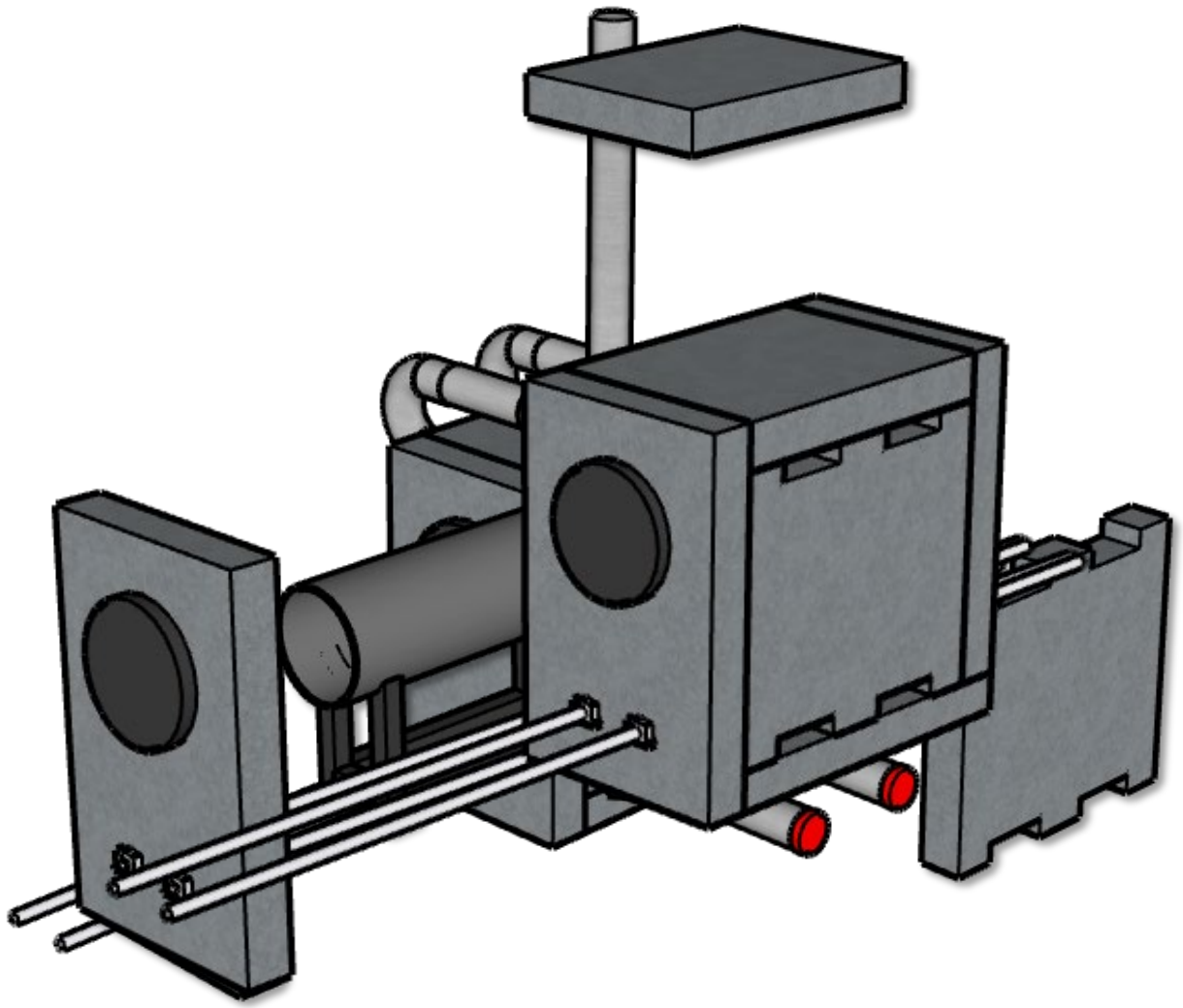
Orano NUHOMS System

Lab-scale (cylinder length 3 feet / 1 meter)

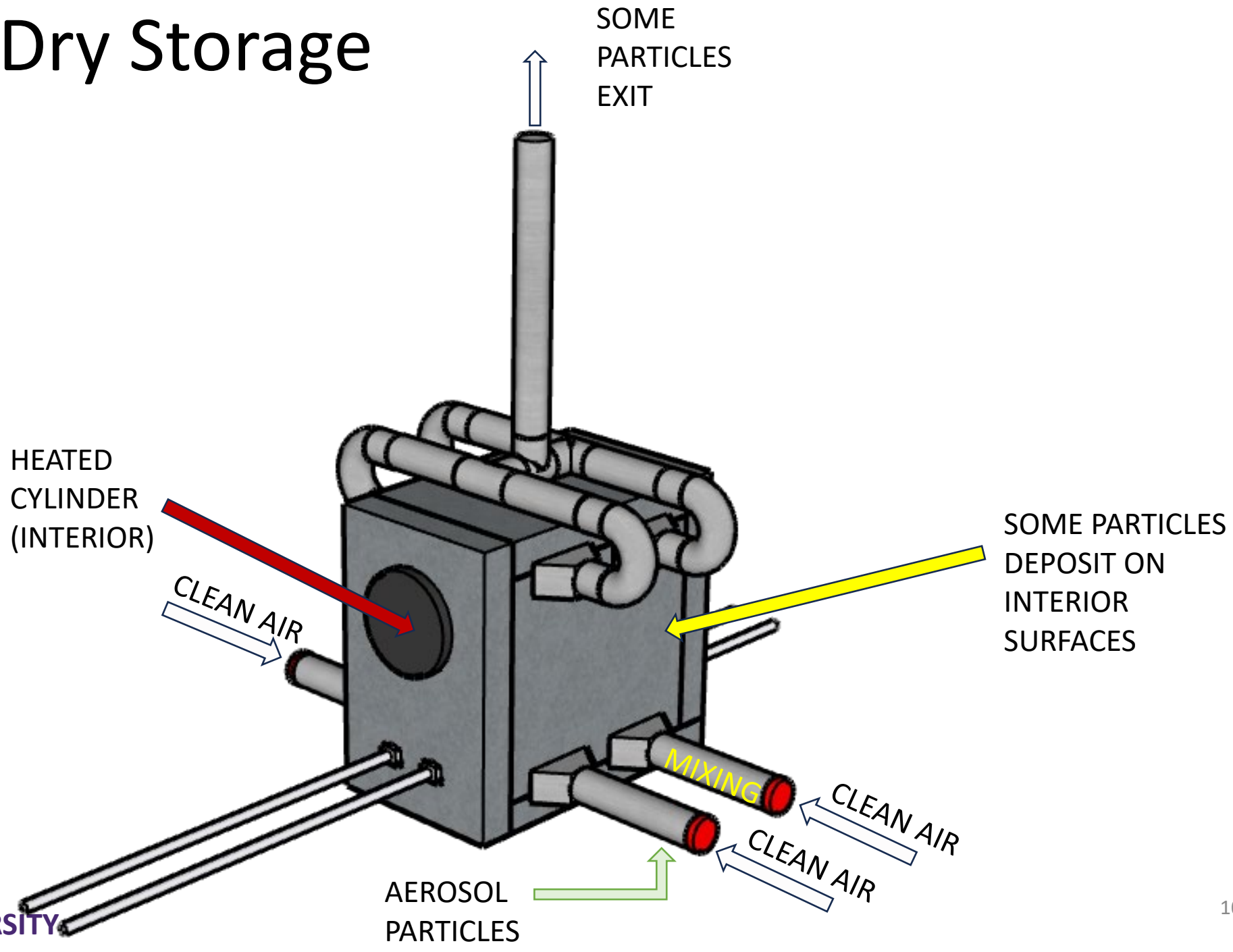


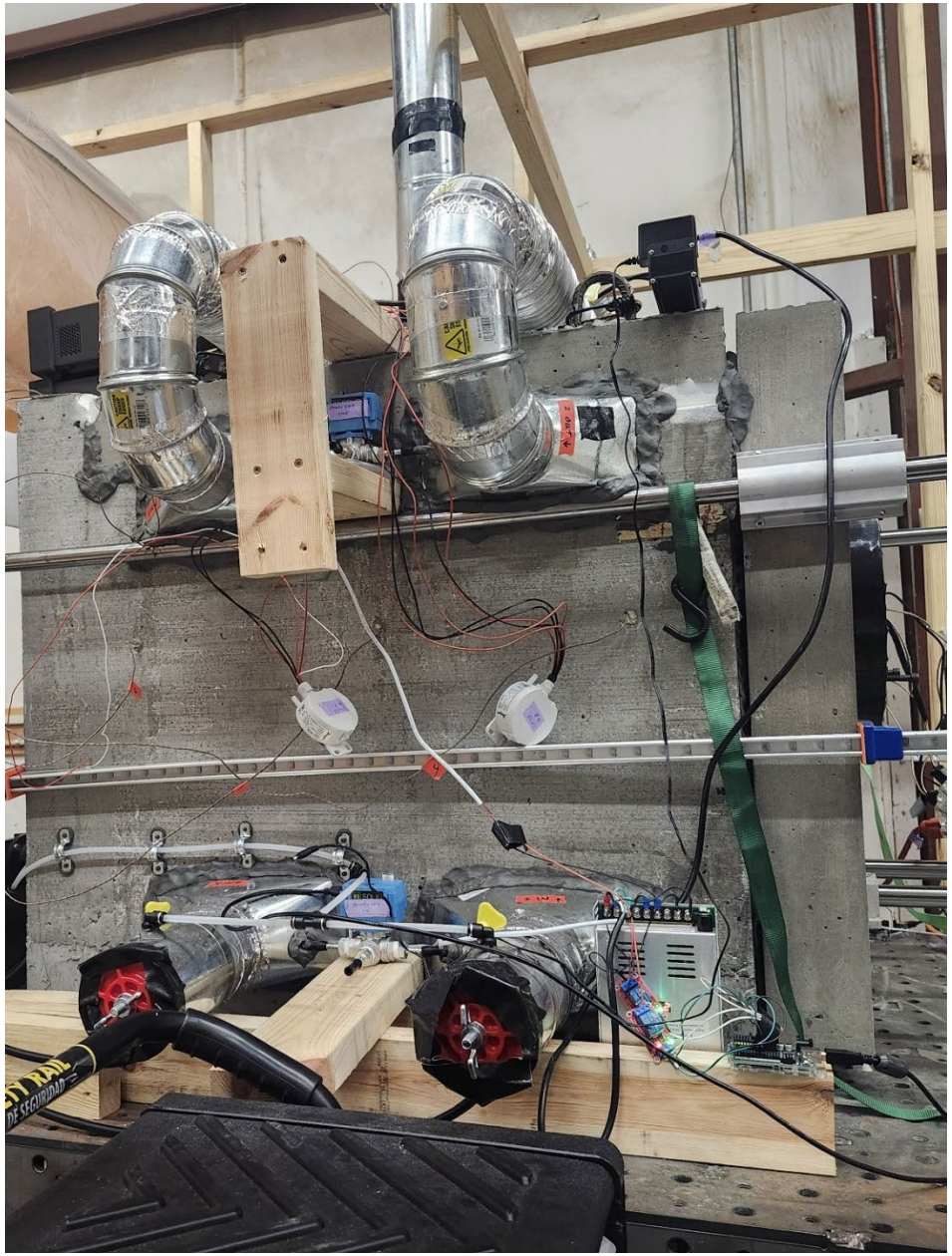
Lab-scale model dry storage cask
(Image captured from SolidWorks)

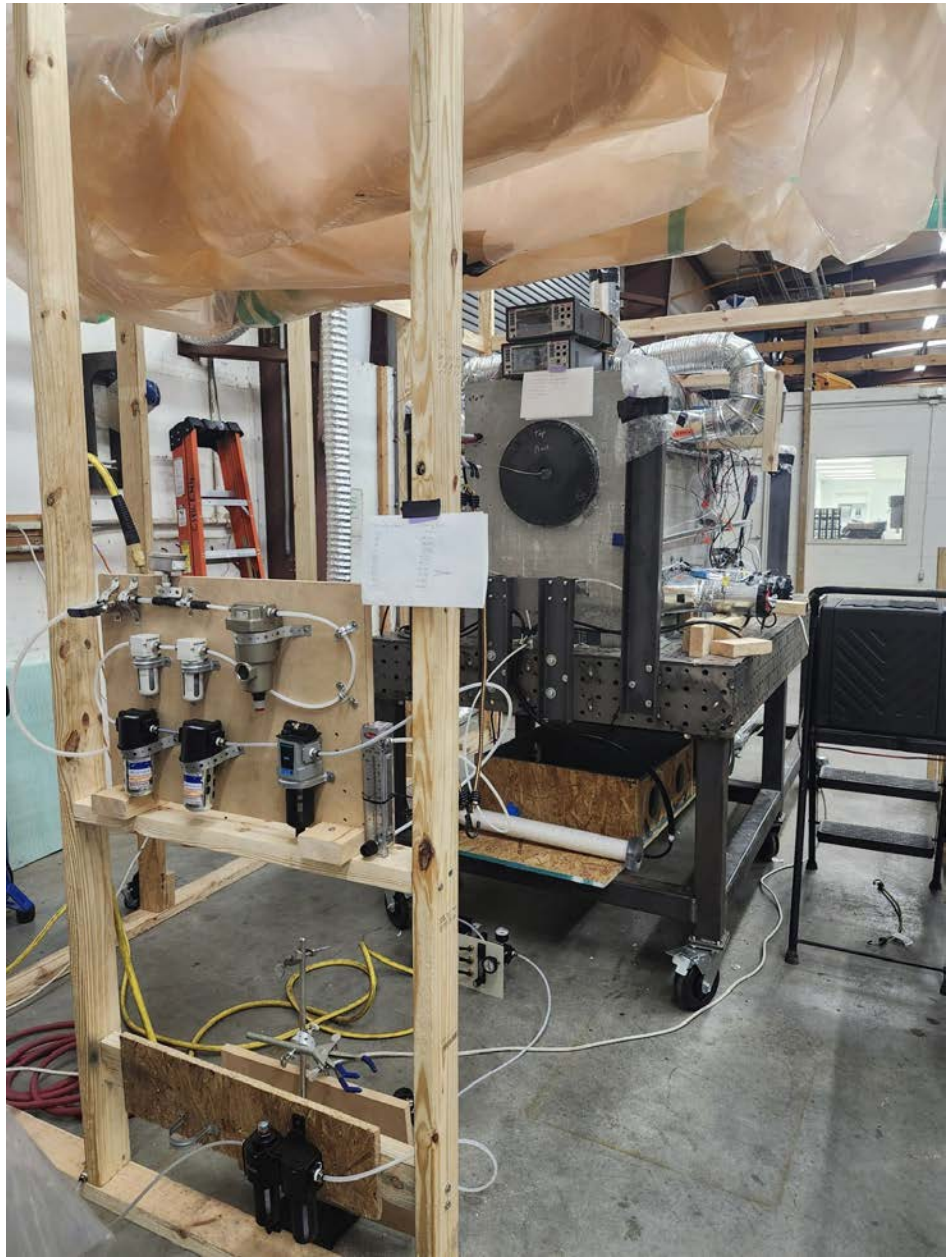
Lab-scale Dry Storage Cask



Lab-scale Dry Storage Cask







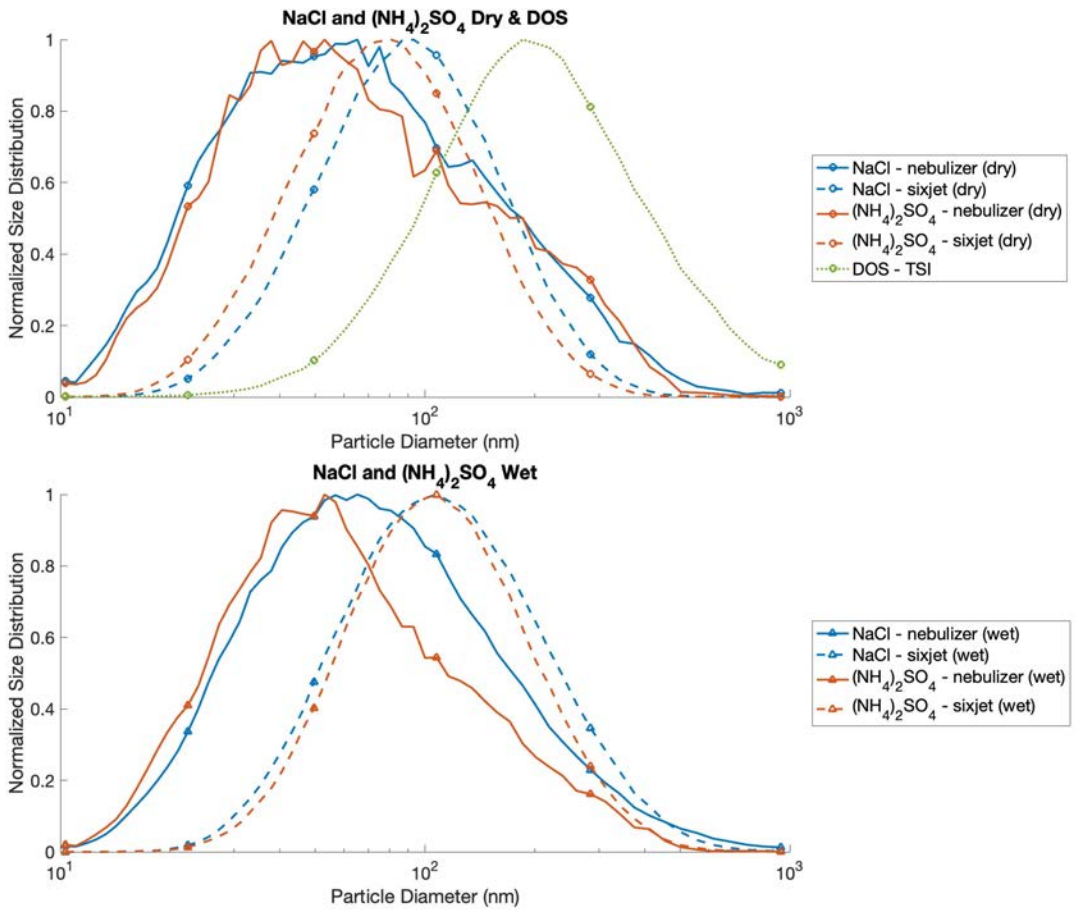
Lab Experiments

Cylinder temperature [degC]	Flow rate [lpm]	Residence time [min]	Inlet/outlet sampling time [min]	Particle counter sampling time [s]
20, 40, 60	20, 40, 60	7.5, 3.75, 2.5	90, 45, 30	90, 45, 30

Flow → residence time ($\boldsymbol{\tau}$) → scavenging coefficient ($\boldsymbol{\lambda}$)

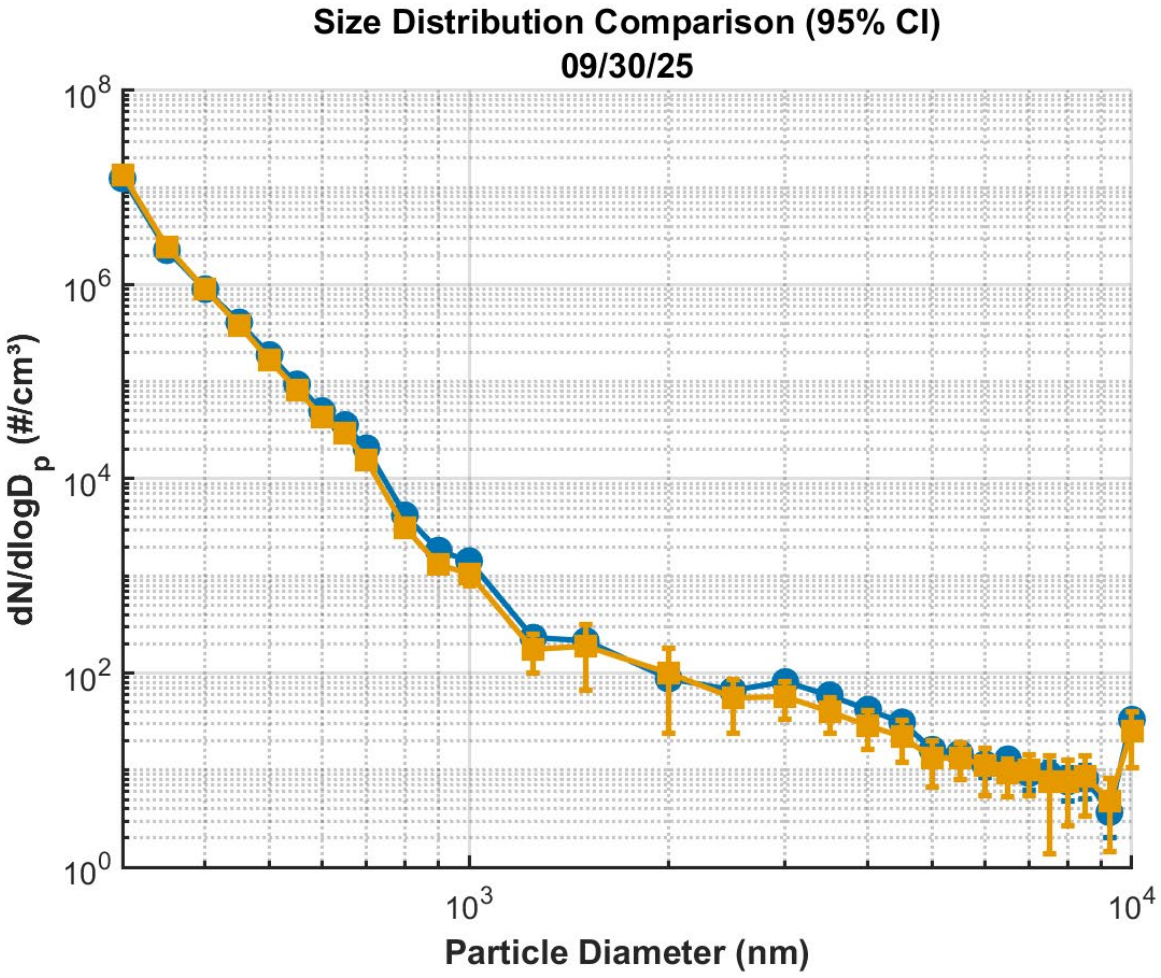
$$\lambda = \ln(C_{out}/C_{in})/\tau$$

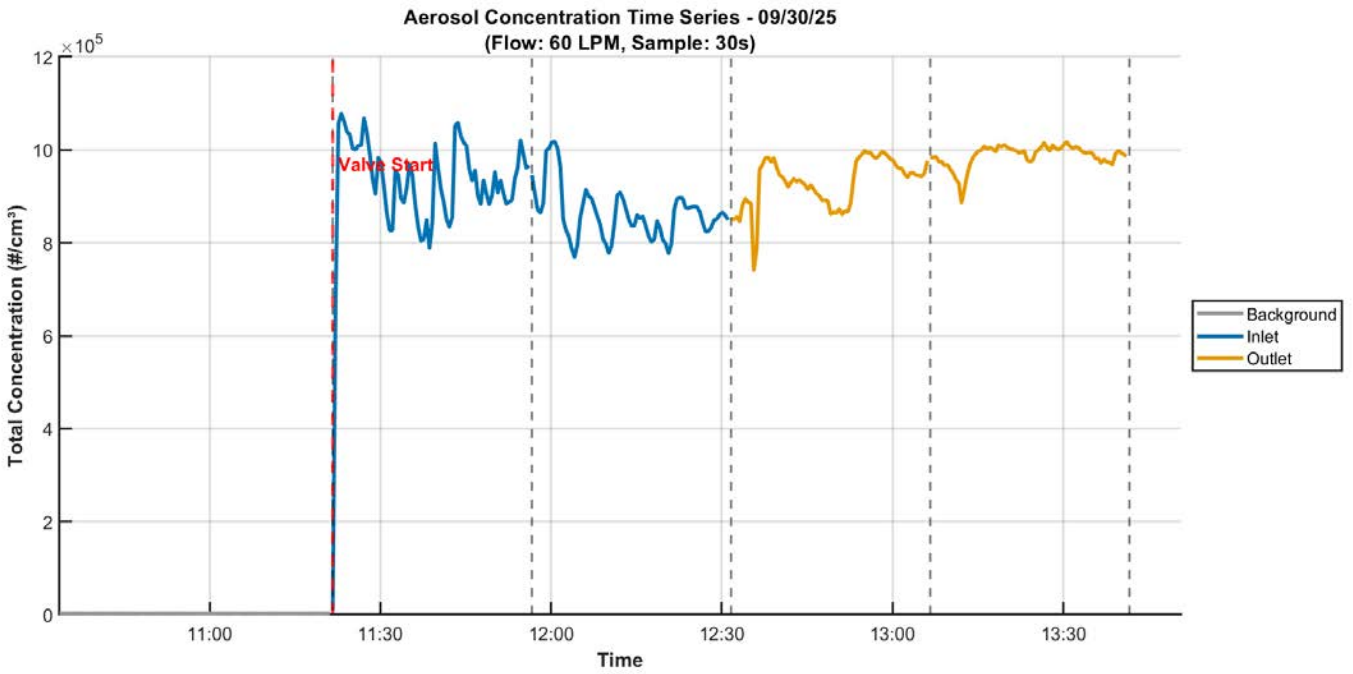
Aerosol Size Distributions



- Submicron aerosol
- Variations in size depending on salt content, composition, and generation device used

Results



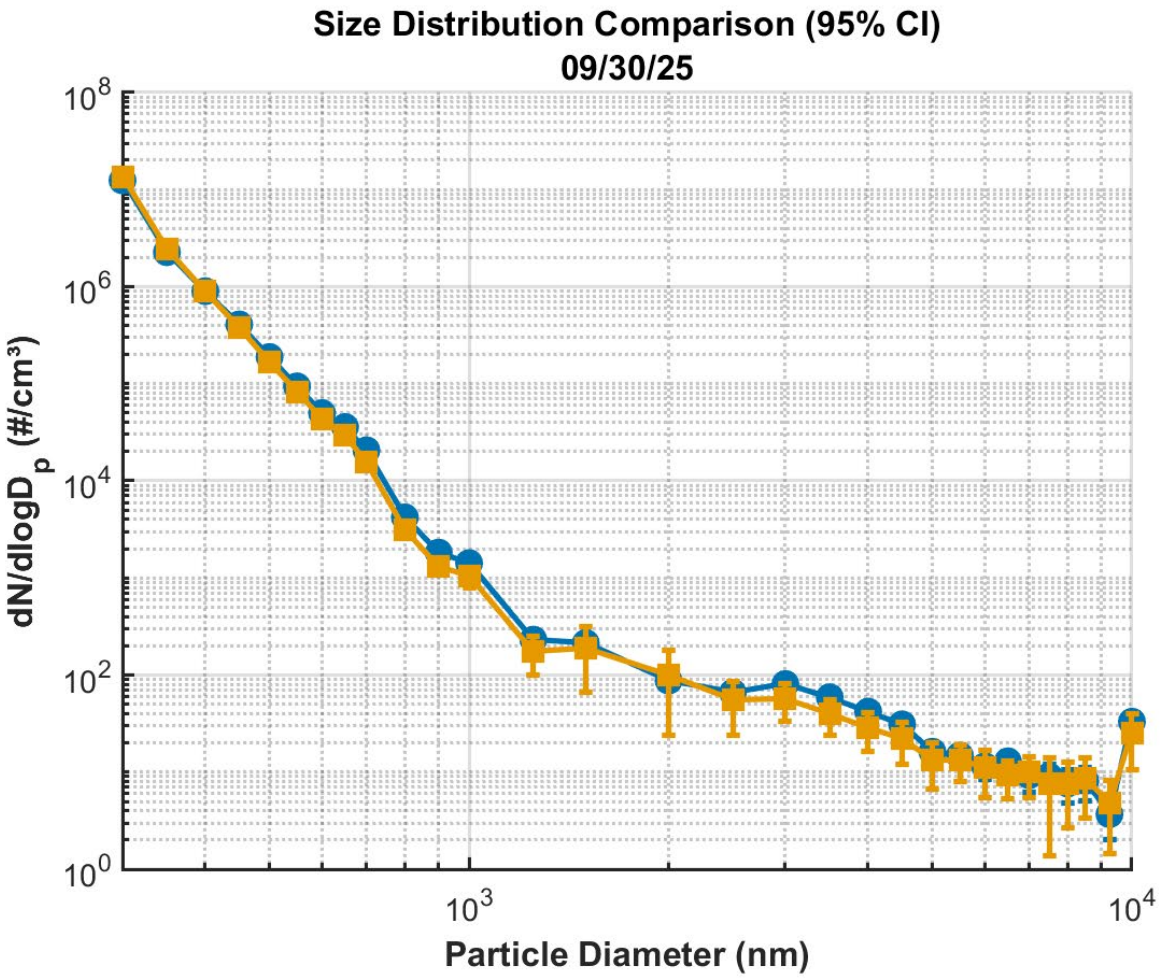


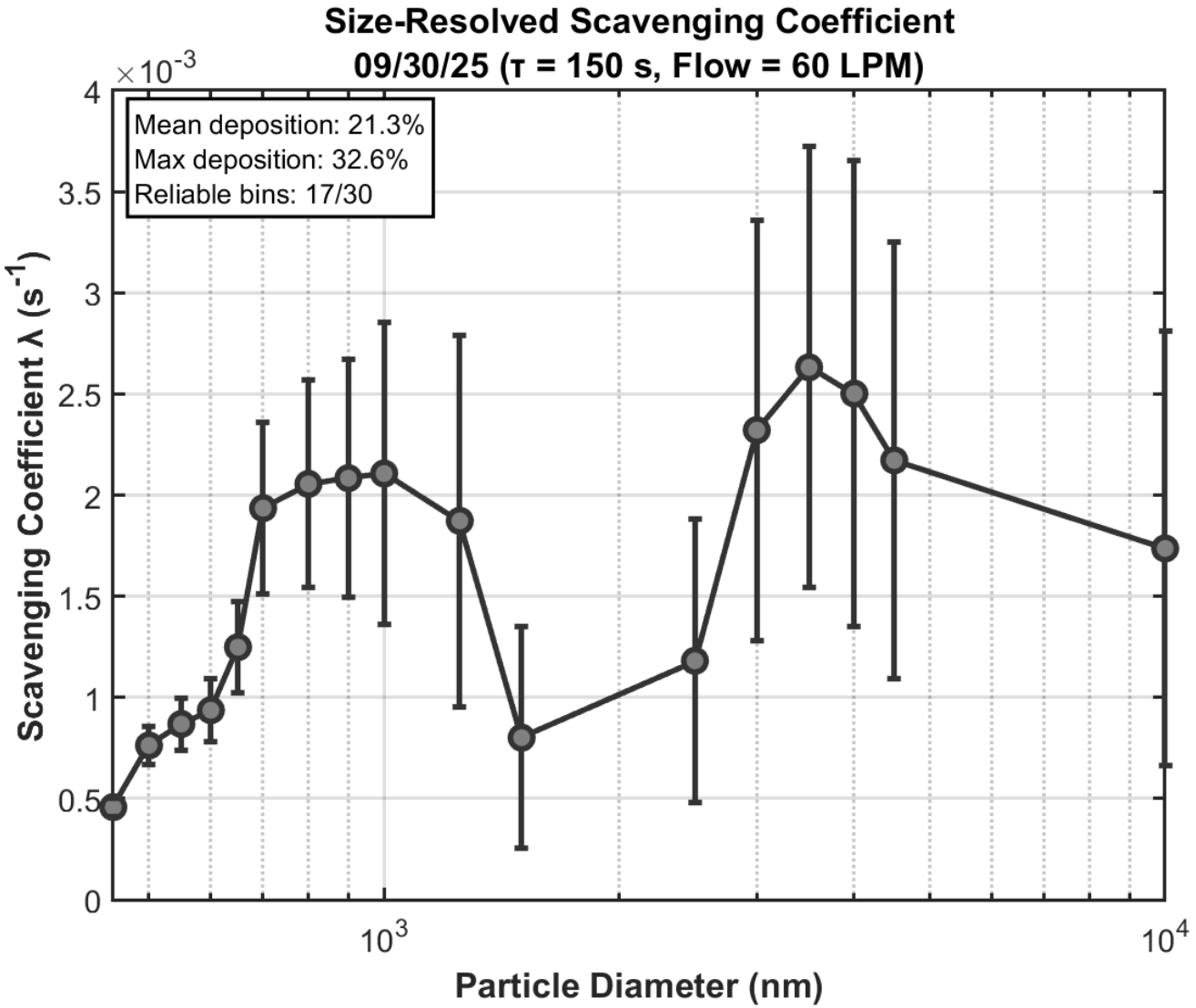
Statistics:

Inlet: n=140 samples
Outlet: n=140 samples
Flow: 60 LPM

Flow rate: 60 lpm
Cylinder temperature: 20 degC

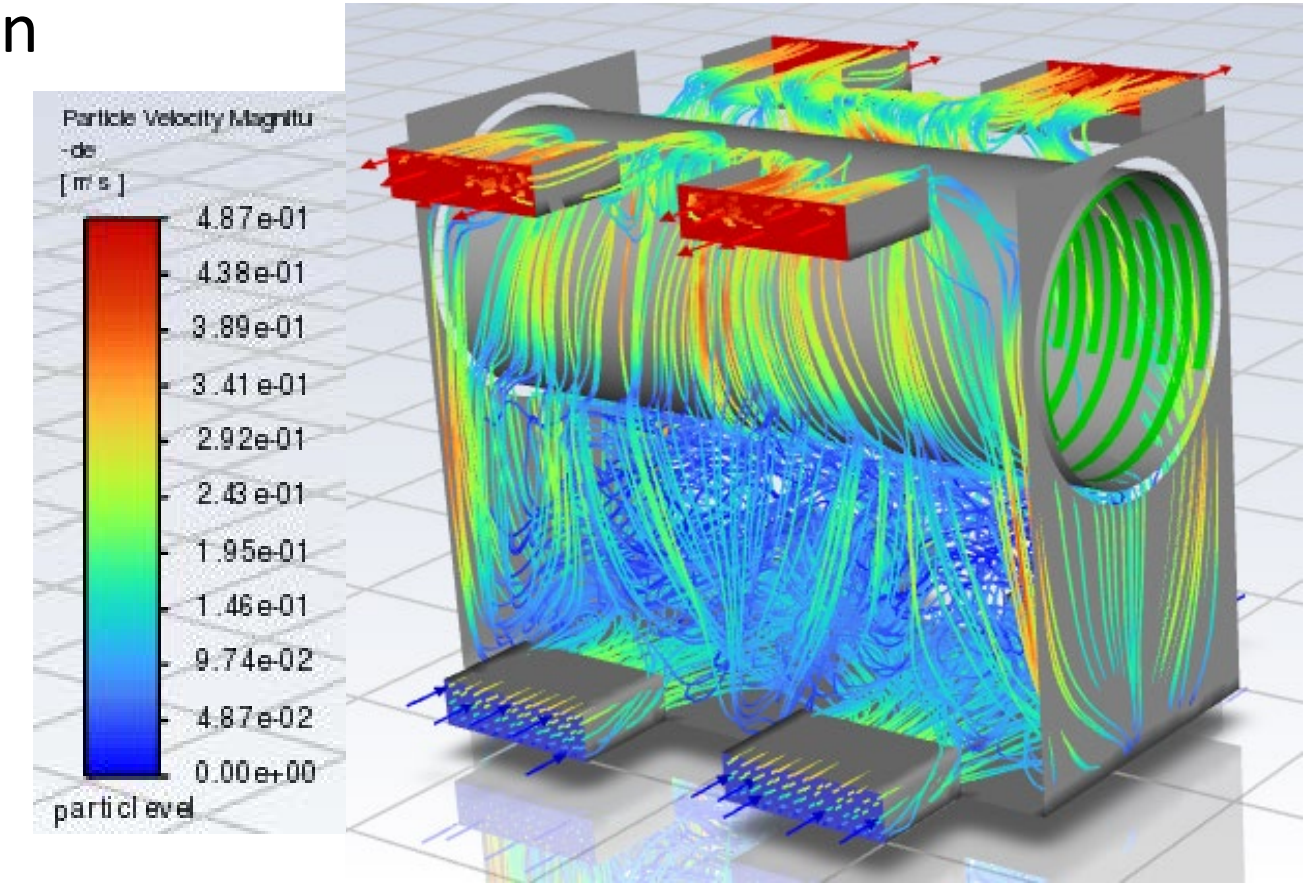
Results


Size Distribution Comparison (95% CI)
09/30/25
dN/dlogD$_p$ (#/cm^3)
10^8
10^6
10^4
10^2
10^0
10^3
10^4
Particle Diameter (nm)


Size-Resolved Scavenging Coefficient
09/30/25 (τ = 150 s, Flow = 60 LPM)
×10^{-3}
Mean deposition: 21.3%
Max deposition: 32.6%
Reliable bins: 17/30
Scavenging Coefficient λ (s^{-1})
4
3.5
3
2.5
2
1.5
1
0.5
0
10^3
10^4
Particle Diameter (nm)

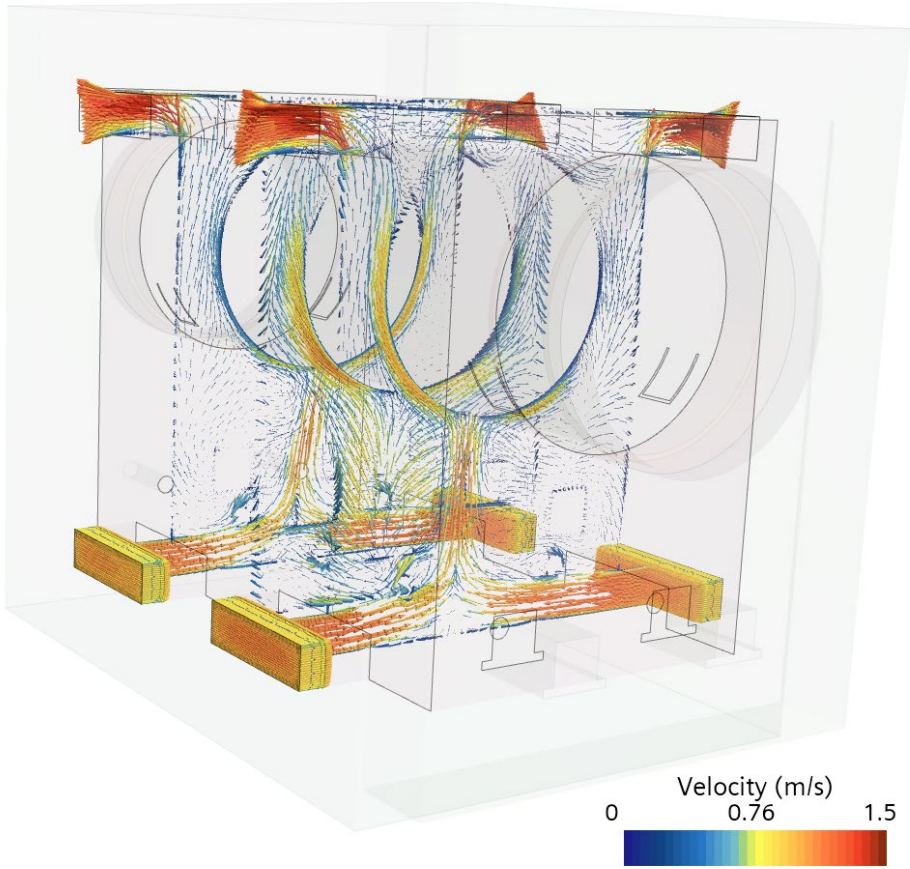
Experimental Challenges

- Stability of aerosol generation
- Generating coarse-mode aerosol
- Measurement accuracy and precision
- Number of simultaneous aerosol measurements

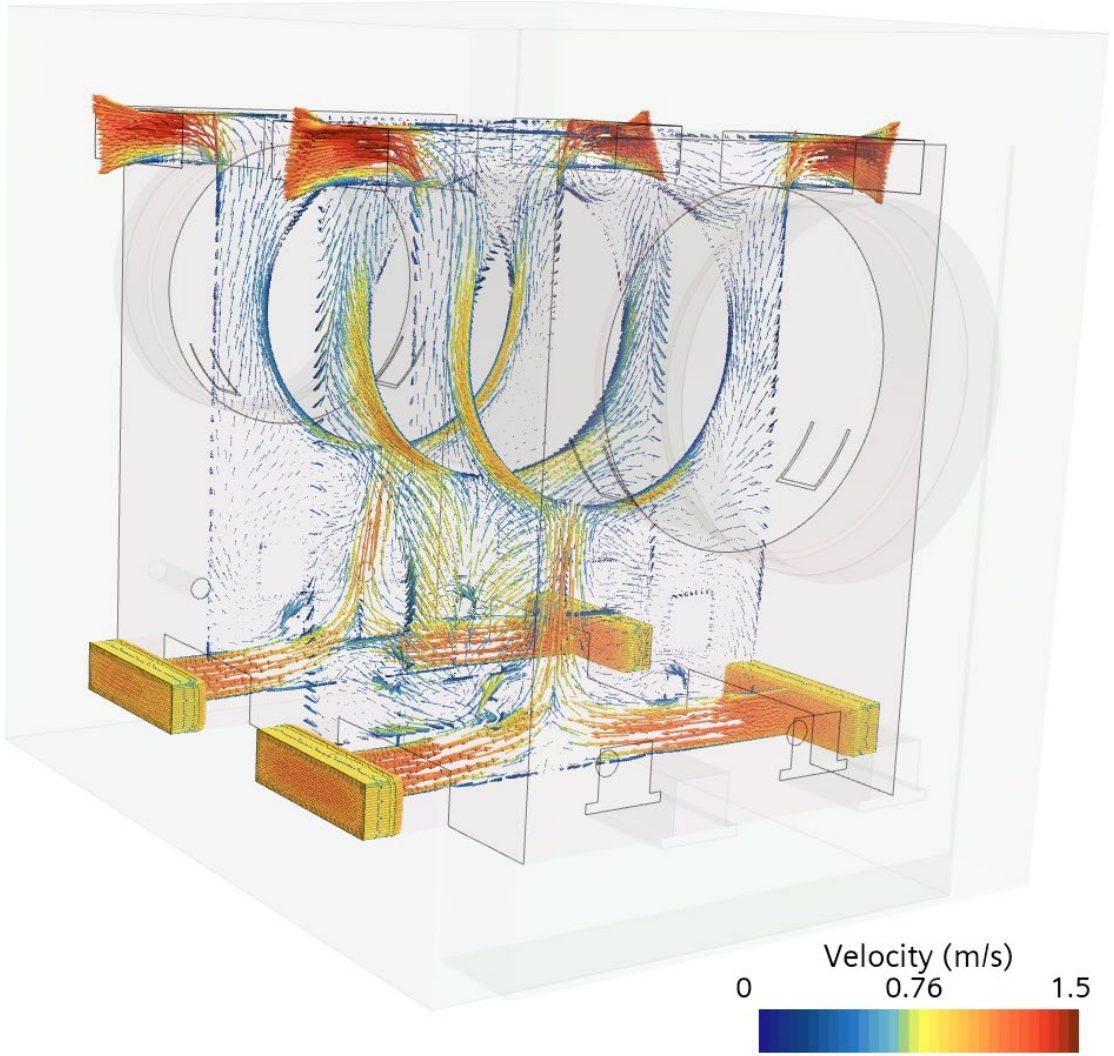


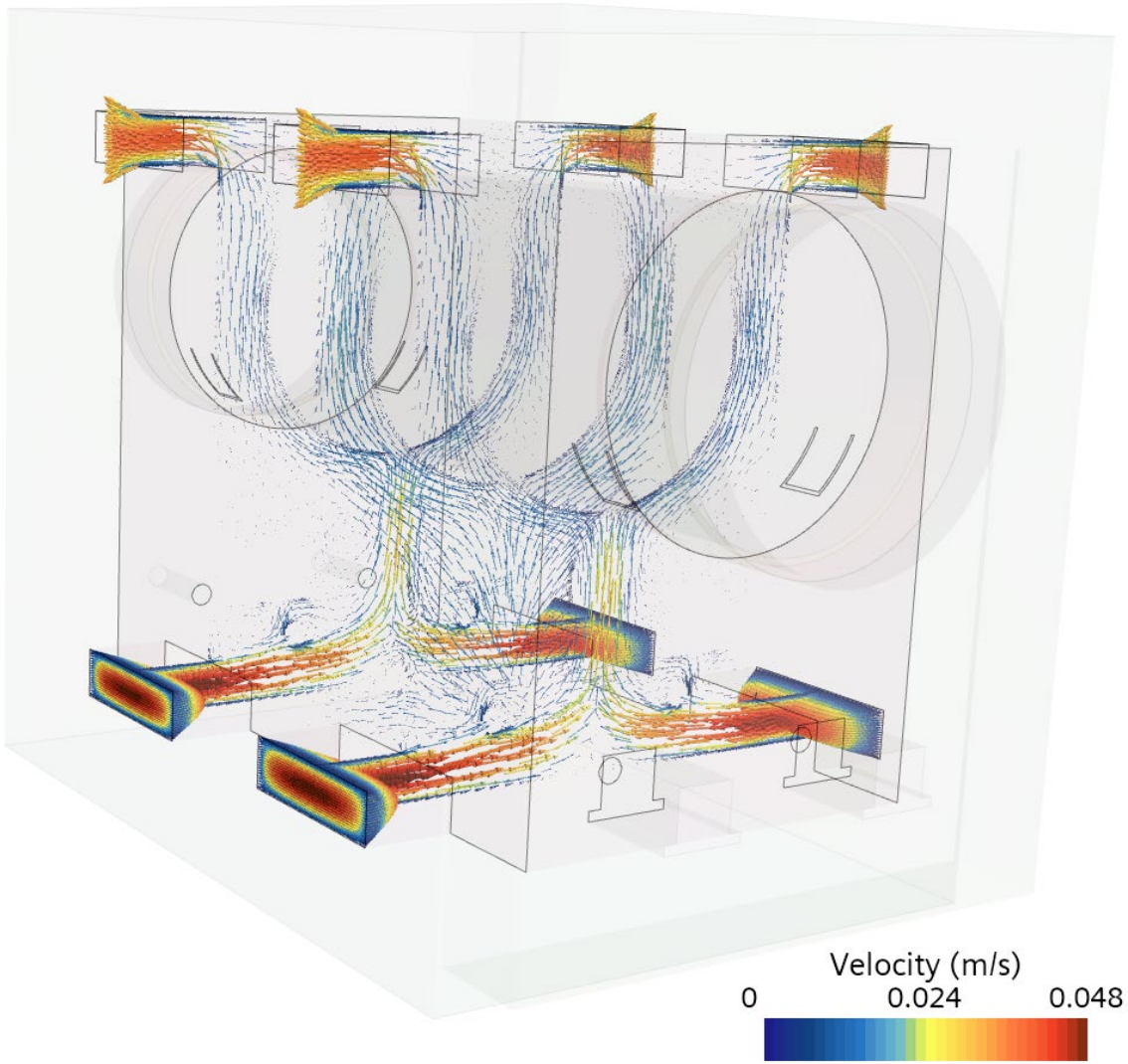
PNNL Deposition Model

- RANS and k-omega SST Turbulence Model
  - Gas: modeled as multicomponent gas (air & water vapor for relative humidity)
  - Particle injection: Lagrangian multiphase model with solid particulates
- Deposition Forces & Mechanisms:
  - Aerodynamic deposition
  - Gravitational settling
  - Saffman lift
  - Turbulent dispersion
  - Thermophoresis
  - Wall correction force
- Model geometry based on CAD from Clemson
- Boundary Conditions:
  - Inlets: stagnation
  - Outlets: velocity
  - No heat generation, T_{amb} = 300 K



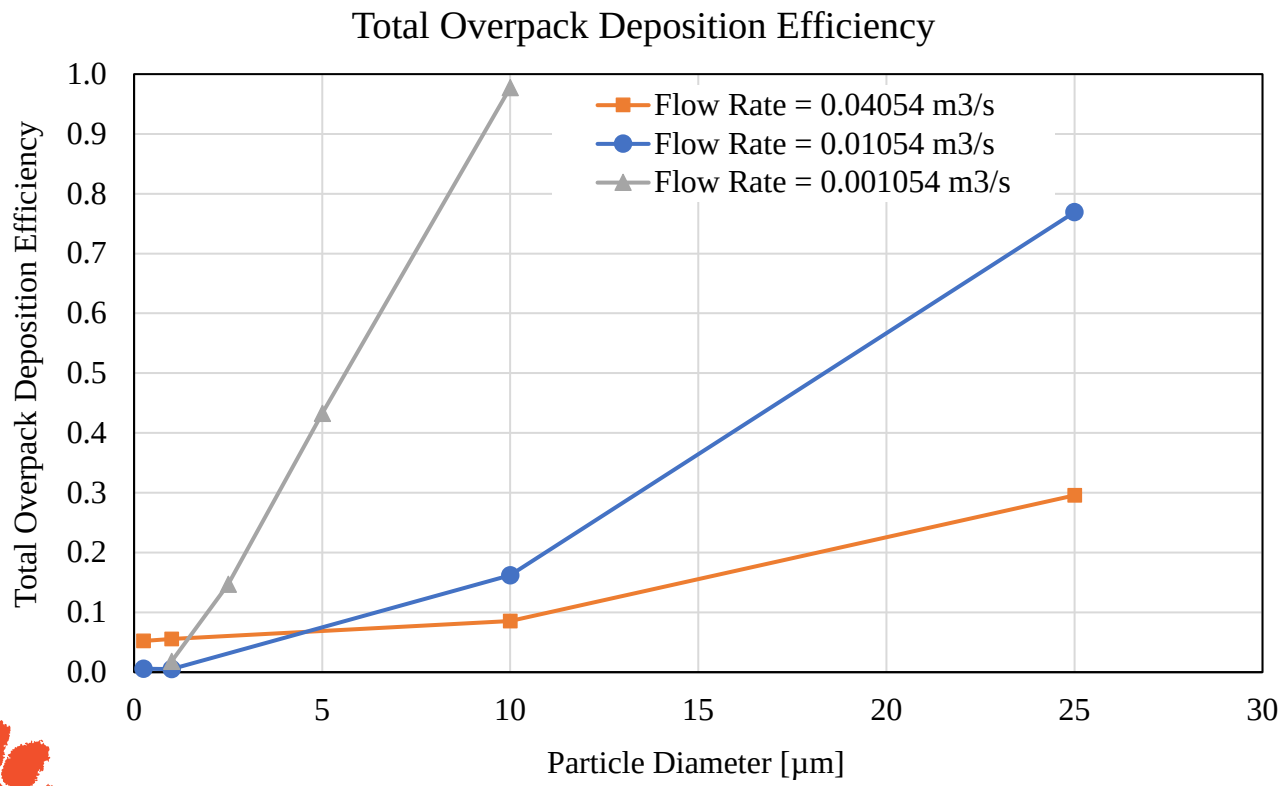
Modeled Velocity

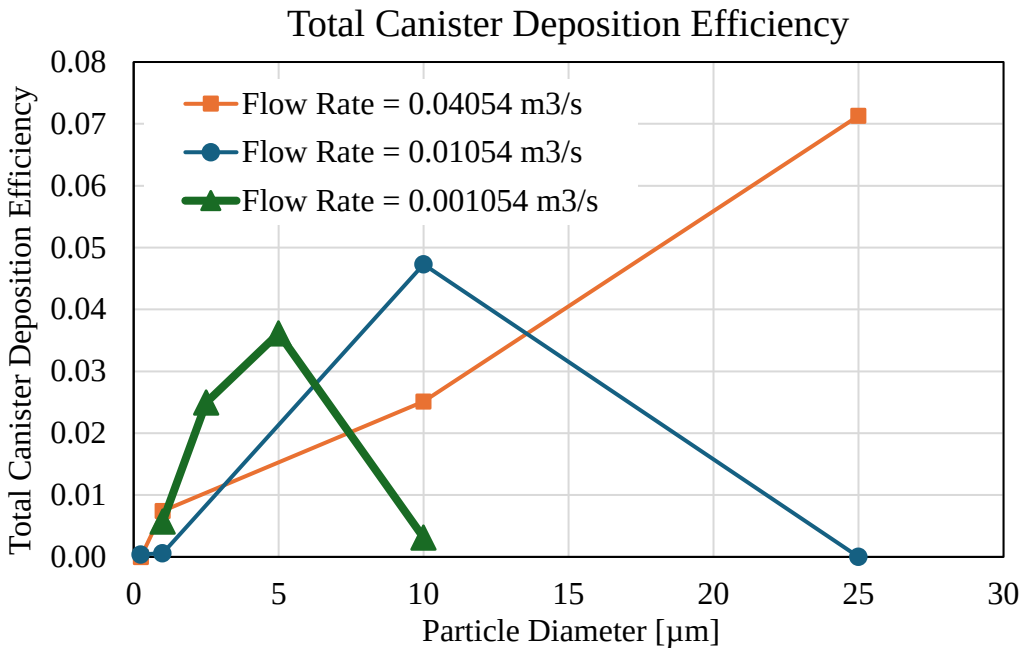


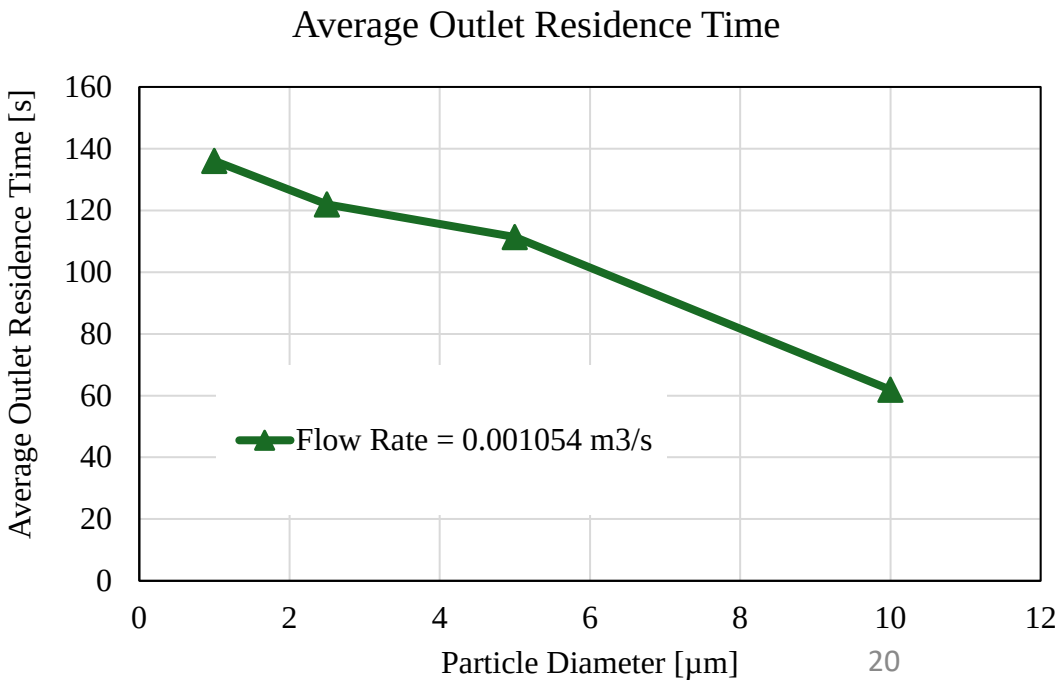


Initial Results

- Deposition within overpack tends to increase with particle size
- Deposition on canister peaks at moderate particle diameters for moderate flow rates
- Average outlet residence time decreases with increasing particle diameter at lowest flow rate



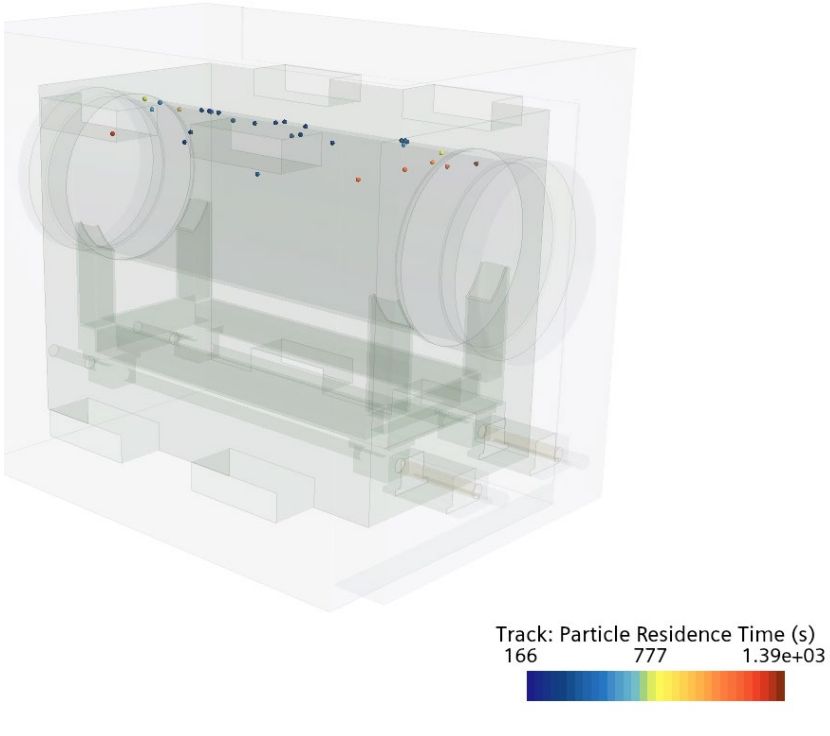




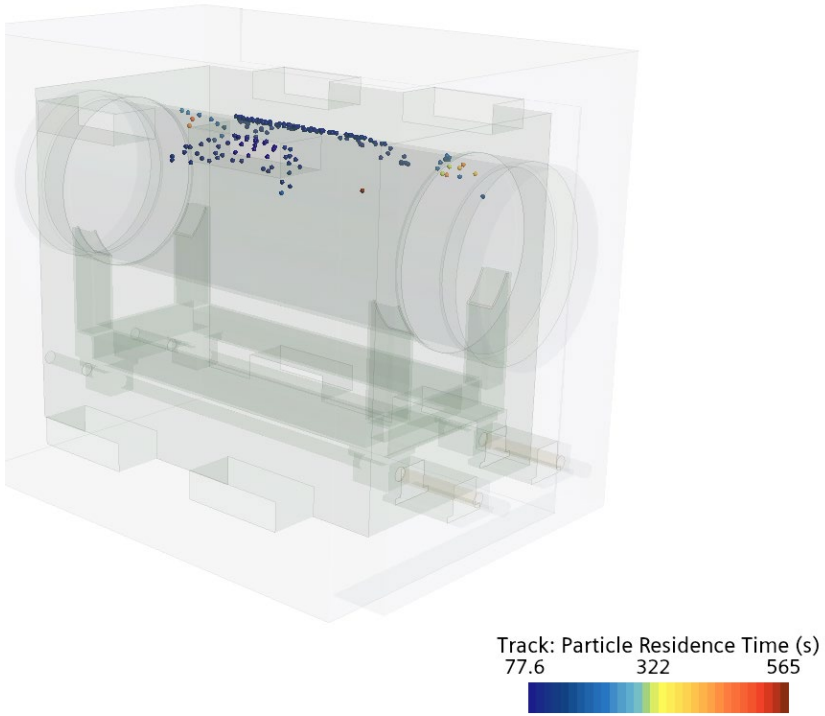
Initial Results

- Location and timing of deposition can indicate dominant deposition mechanism
- More and faster settling on top of canister = gravitational settling dominant

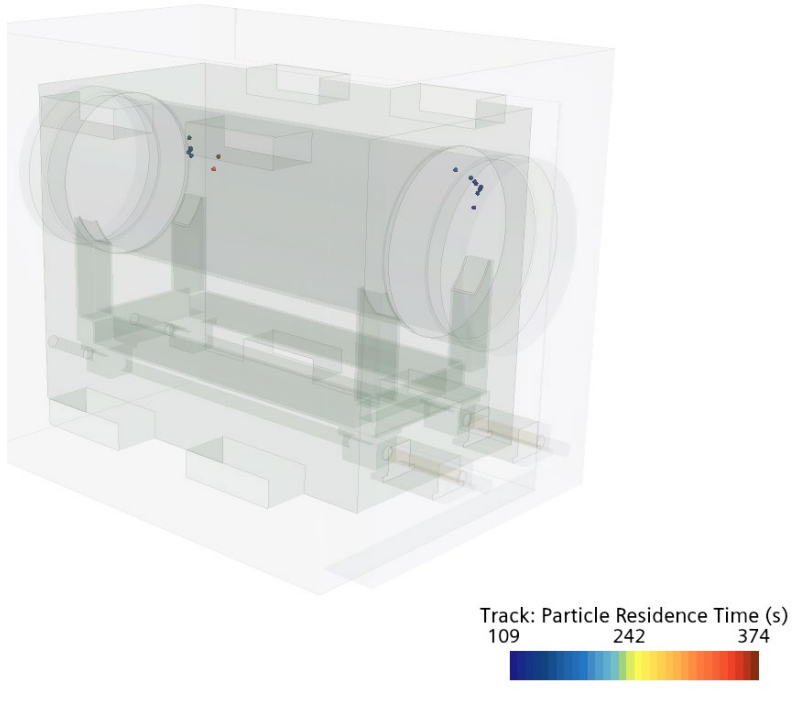
Flow rate = 0.001054 m^3/s



Particle diameter = 1 µm



Particle diameter = 5 µm



Particle diameter = 10 µm

Modeling Conclusions and Future Work

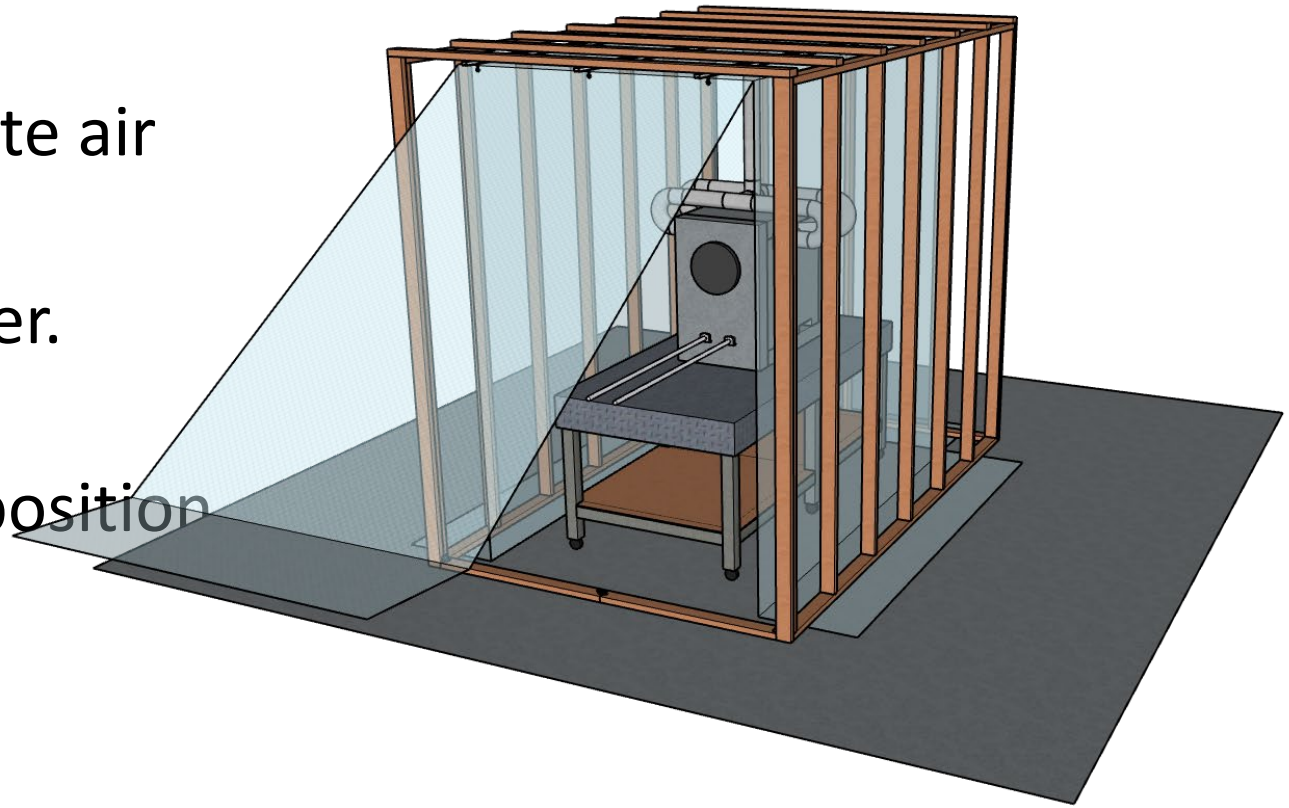
- Distinguishable, measurable overpack deposition results are produced at larger particle sizes and lower flow rates
  - Gravitational deposition is dominant in overpack deposition
- Highest canister deposition efficiency is ~7% at high flow rate and large particle sizes
  - Momentum and gravitational deposition
  - Significant difference in canister deposition efficiency at 1 micron vs 10 microns at the moderate flow rate, lowest flow rate canister deposition efficiency peaks at 5 microns
- Update model to reflect experimental conditions:
  - Polydisperse MgCl particles with measured particle size distribution and concentration
  - Particle diameter growth and depletion due to relative humidity and temperature

Summary

- Lab-scale dry storage cask facility is built and operational.
  - Controlled temperature and relative humidity.
  - Variable aerosol generation: type, size, amount.
- Preliminary data suggests aerosol deposition is small.
  - Measurement challenges to address.
  - Work-in-progress to validate “digital twin.”
- Ongoing work for NEUP project:
  - Experiments with larger (coarse-mode) aerosol.
  - Vary air flow rates, canister heating.
  - Measure spatial distribution of aerosol deposition inside cask.

Future Projects?

- Testbed for new cask designs.
- Heat/convection profiles to validate air flow predictions.
- Aerosol escape from inside canister.
- Material properties' (e.g., surface roughness) impact on aerosol deposition and air flow.



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- Megan Higley, Dina Carpenter-Graffey

Award # DE-NE0009276

Thank you!

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Prediction models for the long-term performance of metal seals in dual-purpose casks

Ralf Schneider-Eickhoff, Dr. Ghina Jezdan

EPRI-ESCP 2025
27th to 30th of October 2025, Charlotte, NC

Outline

I. MSTOR project and experimental data

II. Time-temperature equivalence

III. Prediction models and ageing scenario

IV. Conclusions and outlook

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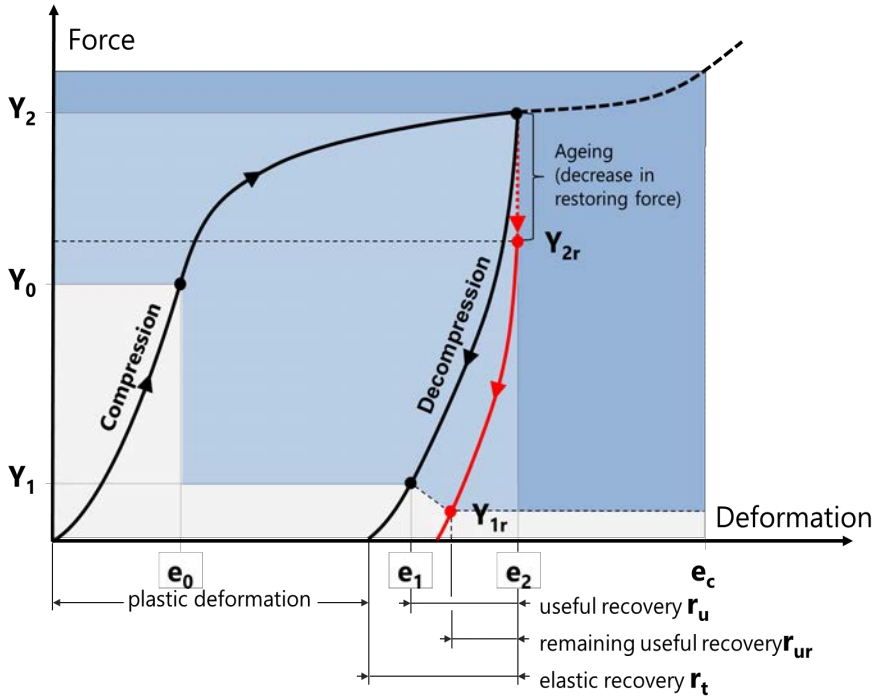
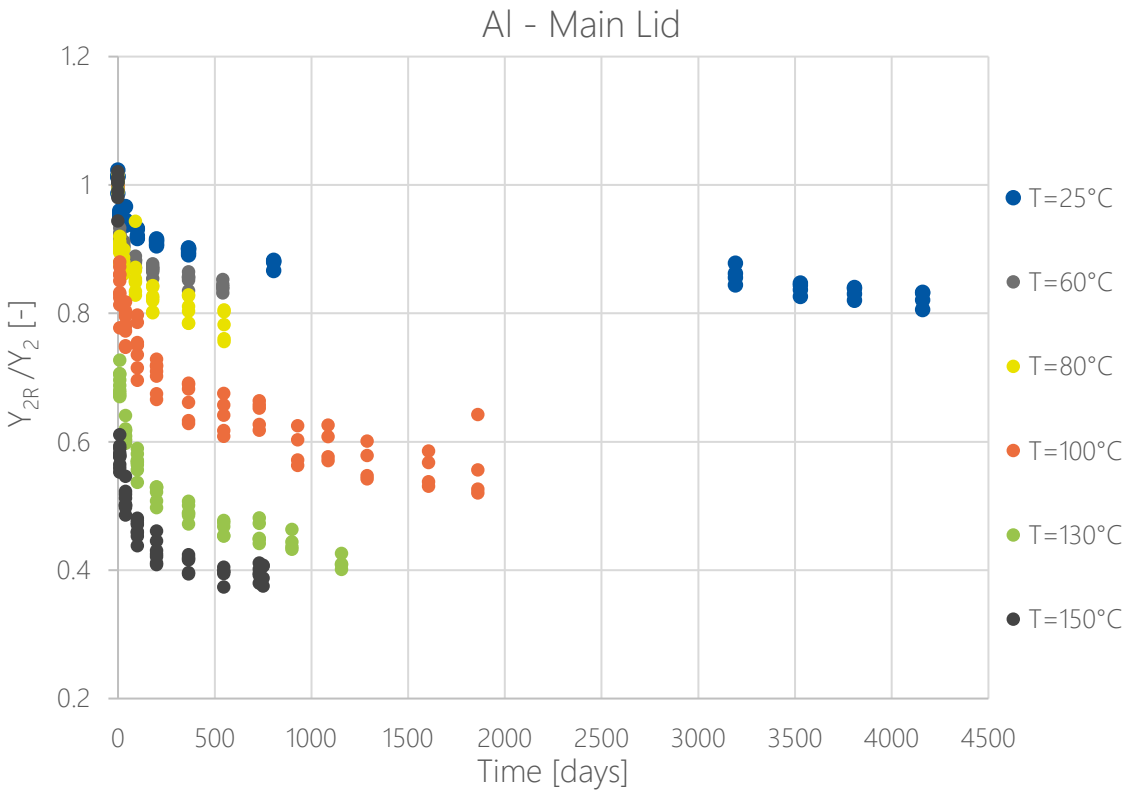
I. MSTOR project and experimental data

MSTOR test program

- Evaluation of the long-term behavior of HELICOFLEX® spring-energized metal seals, covering the assembling and operating condition at BGZ
- Within the MSTOR test program, several working packages are involved:
  - Isothermal ageing tests at 60°C, 80°C, 100°C, 130°C, 150°C, and RT,
  - Seals with 4 different torus diameters (acc. to small/main lids),
  - Seals with 2 different outer liner materials: aluminum / silver,
  - Tests with 2 different flange combinations (SST/SST and SST/DCI) to represent the different sealing surfaces found in original casks,
  - Test series at varying temperatures as a basis for validating the prediction models to be developed within the MSTOR program.
- Measurement of restoring force Y_2 and r_u depending on ageing time and temperature → more than 200 seals tested
  → more than 1.000 data points so far



Preliminary results



- Tests at elevated temperatures at 130°C/150°C are finished after 2 and 3 years of ageing, respectively.
- Tests at 100°C have reached a duration of 6 years and will be continued.
- Additional tests at lower temperatures (60°C/80°C) were started 2 years ago and are still ongoing.
- All isothermally determined measurement data show the same trend for the development of the restoring force:
  - Strong degradation at the beginning,
  - followed by an asymptotic decay.
- Ageing of the metal seals is disproportionately determined by the high temperatures at the beginning of the operation.

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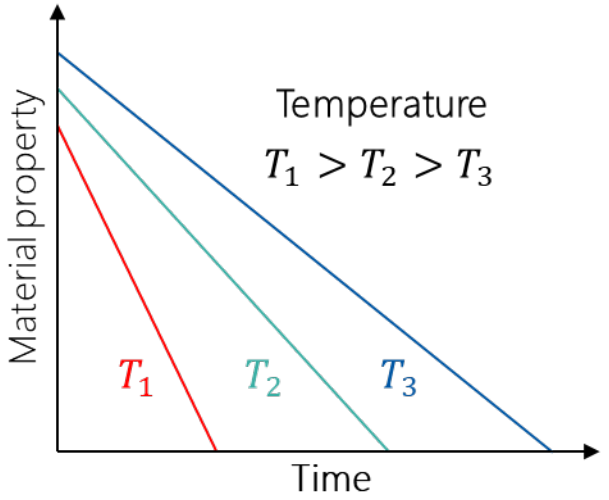
II. Time-temperature equivalence

Time-temperature equivalence

- The measured parameters show a time-temperature equivalence: long-term ageing under low temperatures results in comparable ageing effects as reached at higher temperatures with shorter ageing periods.

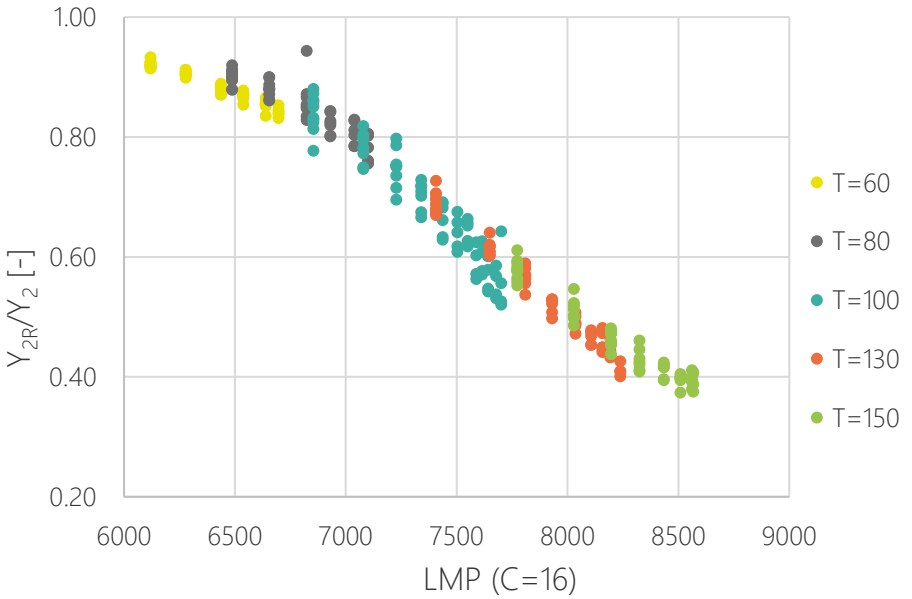
- The time-temperature equivalence is represented by two different classes of prediction models:
  - Time-Temperature parameter
    - Larson-Miller Parameter (LMP) → only one constant "C" for the temperature dependence
    - Orr-Sherby-Dorn Parameter,
    - …
  - Time-Temperature Superposition (TTS)
    - Time and temperature effects can be treated separately.
    - There is a large selection of suitable prediction function:
      - Mathematically → Regression analysis (linear, non-linear)
      - Physically → Phenomenological functions (e.g. Arrhenius)
        → Rheological Modes (e.g. Maxwell, Kelvin-Voigt, …)



Larson-Miller-Parameter (LMP)

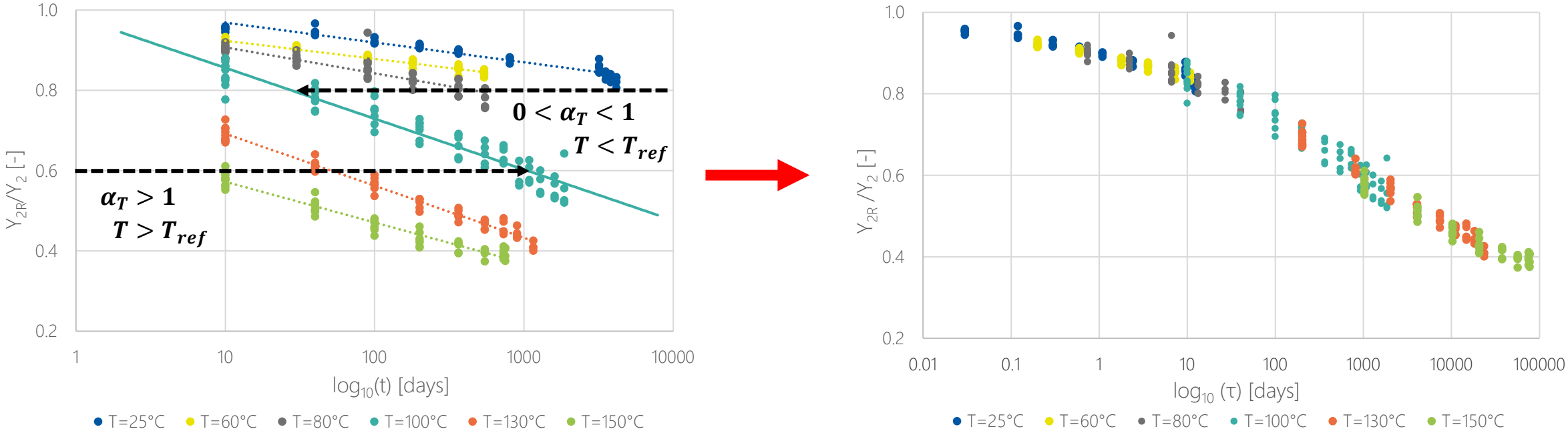
- Developed to describe creep behavior, but can also be used to predict relaxation due to similar microstructural processes.
- LMP is based on Arrhenius type equation for thermally activated processes:
  - $r = \frac{\Delta l}{\Delta t} = A \cdot e^{-\frac{Q}{RT}}$
    r: strain rate, A: constant, Q: activation energy, R: Universal gas constant, T: Temperature
  - $ln\left(\frac{\Delta l}{\Delta t}\right) = ln(A) - \frac{Q}{RT} \rightarrow \frac{Q}{R} = T \cdot \left(\ln(A) - \ln\left(\frac{\Delta l}{\Delta t}\right)\right)$

    $\boldsymbol{LMP = T \cdot (C + \log(t))}$ with $C = \frac{ln\left(\frac{A}{\Delta l}\right)}{ln(10)}$
- Application requirements:
  - In addition to accelerated ageing experiments, the data base shall contain measurements at realistic operating temperatures.
  - Due to its linear trend curve, LMP is not realistic for predictions beyond the range covered by measurements (extrapolations).
  - Both the constant "C" as well as its regression quality depend on the temperature range included in the analysis.



Time-Temperature Superposition (TTS)

- The time- and temperature-dependence is given by a function: $G(t,T) \xrightarrow{TTS} G(\tau, T_{ref})$ "Master function"
  - Reference temperature: in the middle of the examined temperature range, e.g $T_{ref} = 100\ °C$
  - Master time: $\tau = \alpha_T \cdot t$ with shift factor α_T, e.g. $\alpha_T = \frac{t\,[Y_{2R}(T_{ref})]}{t[Y_{2R}\,(T)]}$



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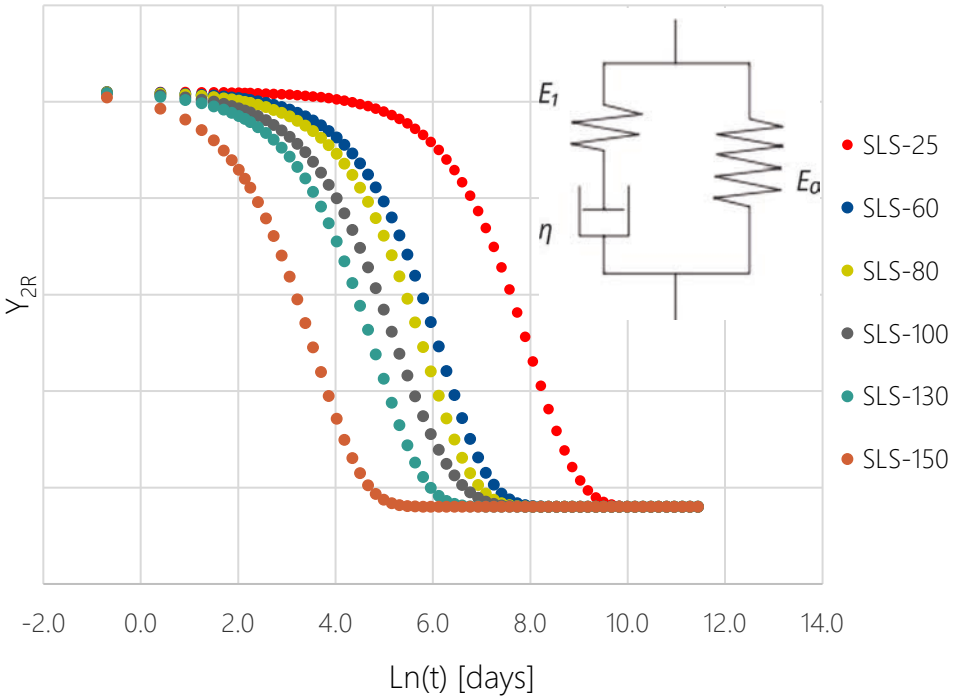
III. Prediction models and ageing scenario

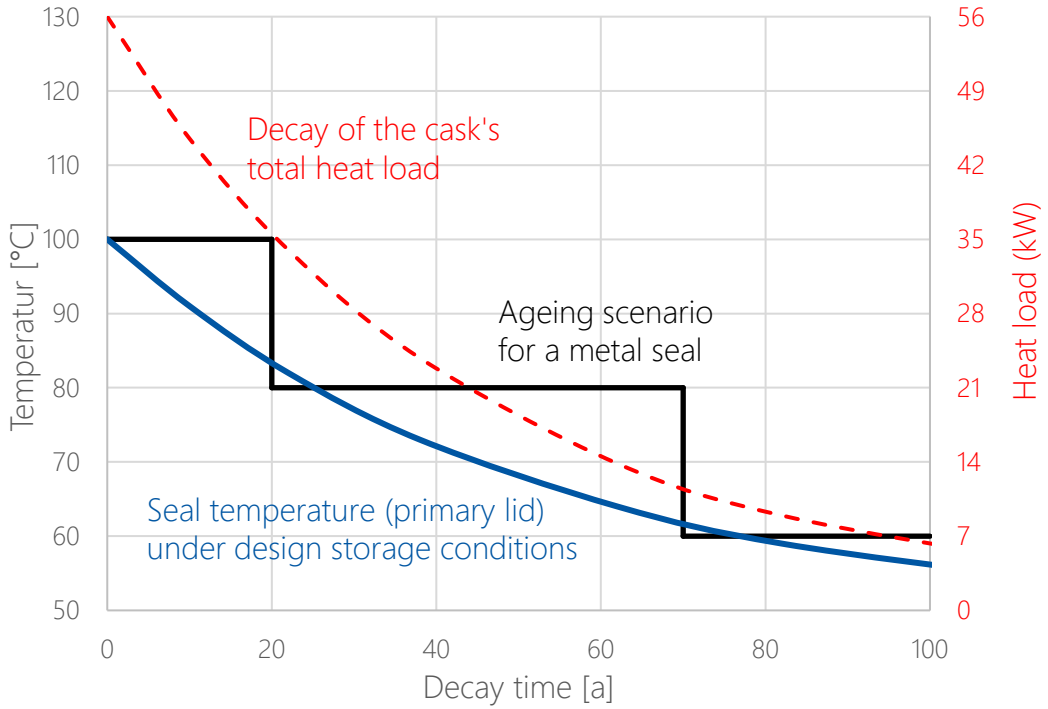
TTS master functions

- Based on regression analysis:
  - Ln function: $Y_{2R} = x_1 + x_2 \cdot Ln(\alpha \cdot \tau + b)$
  - Sigmoid function: $Y_{2R} = x_1 + \frac{x_2}{1+e^{(-m \cdot \tau + n)}}$
- Based on rheological models (e.g. Standard Linear Solid, SLS):
  - SLS-Function: $\sigma(t) = \left(E_0 + E_1 e^{\frac{-E_1}{\eta} t}\right) \varepsilon_0$
  - Parameter identification (E_0, E_1, η) using the dataset
- Separate determination of the shift factors
  - Direct scaling: $\alpha_T = \frac{t\,[Y_{2R}(T_{ref})]}{t[Y_{2R}\,(T)]}$
  - Modified Arrhenius equation (Nakamura et al., Qiao et al.): $\alpha_T = e^{\frac{-Q}{RT_{ref}\left[1-\left(\frac{T}{T_{ref}}\right)^n\right]^{1/n}}}$
  - SLS: Scaling by means of viscous parameter: $\alpha_T = \frac{\eta_{T_{ref}}}{\eta_T}$



Ageing scenario

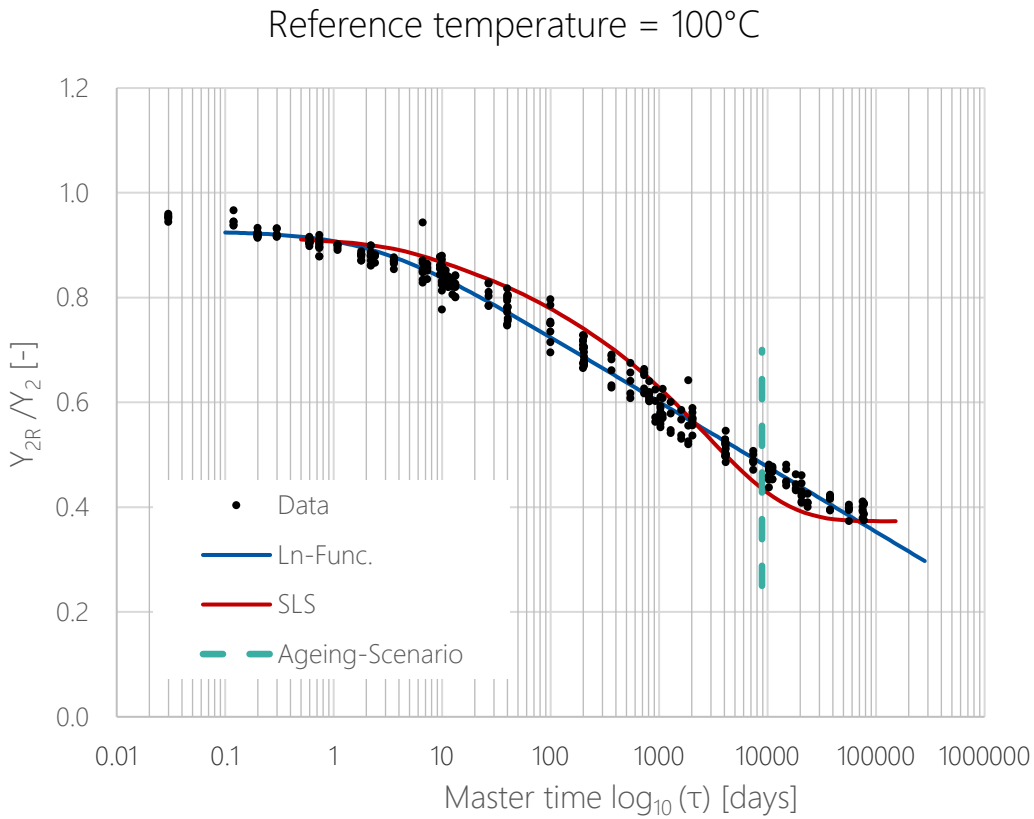
- Step-wise temperature curve is chosen as ageing scenario, because the prediction models used here
  - are not able to calculate a continuous temperature change,
  - are dependent on the measurement data (@ 100°C, 80°C, and 60°C).
- The ageing scenario is representative for the metal seals within the CASTOR HAW28M, the cask type with the highest design thermal load of 56 kW.
- Ageing scenario covers the real conditions to be expected.



Comparison of the pretiction results

Prediction model	LMP	TTS		
Master function	w/o	w/o	Ln	SLS
Restoring force, Y_{2R}/Y_2 (-)	0.47 – 0.50	0.45 – 0.51	0.48	0.44
Master time, τ (years)	23.02	24.30		

- All prediction values lie within the experimentally obtained dataset, although the selected ageing scenario is very conservative.
- TTS und LMP models deliver very close prediction values.
- A significant restoring force is predicted to remain also after 100a.
- Both SLS and Ln are suitable as master function using TTS method:
  - Ln shows steady degradation beyond the data range,
  - SLS prognosis corresponds better to the physically expected asymptotic trend, but requires parameter identification.



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IV. Conclusions and outlook

Conclusions and Outlook

- In the MSTOR program extensive seal ageing data are gathered and used to develop prediction models for the long-term behavior of metal seals in DPCs under storage conditions.
- The models, which are based on the LMP and TTS methods, return plausible and very close prediction results for the restoring force of the seal.
- TTS method can be recommended as a framework for the further development of prediction models, as it offers a wide range of possibilities to adjust the master curve, by implementing
  - Physically based parameters that allow extrapolations beyond the data base,
  - Continuous non-linear shift factors for precise modeling of the temperature change over time.
- LMP requires careful selection of the constant "C" and is not suitable for predictions beyond the experimentally validated data range. However, its ease of use recommends application for test planning.

- Our next goals are to improve the derivation of the shift factors and master functions used for the TTS method and to extend the investigations to cover all other seal parameters and configurations analyzed in the MSTOR program.

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Many thanks to all colleagues involved

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BGZ Gesellschaft für Zwischen-lagerung mbH

Thank you for your attention!

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bgz.de/en/research-program/

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NDE Capability Demonstration for Dry Storage System Welded Canister

Byungsik Yoon
Principal Technical Leader

2025 Extended Storage Collaboration Program
10/29/2025

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Background

- Permanent repositories are not available to accept used fuel
  - Spent Fuel Pools are approaching capacity at many sites around the world and increased need for dry storage systems
  - ~200 systems are added per year

01 NDE inspections of welded canisters used in DSSs are necessary to fulfill aging management commitments made during the Independent Spent Fuel Storage Installation (ISFSI) license renewal process.

02 Boiler and Pressure Vessel Code Case N-860 was approved in 2020 and provides recommended requirements for welded stainless steel canister inspections and evaluation criteria if cracks are found.

03 This Code Case requires performance demonstrations for surface and volumetric examinations. Intermediate Rigor (Section V, Article 14) and Section XI, Mandatory Appendix VIII are called out for use in performance demonstrations.

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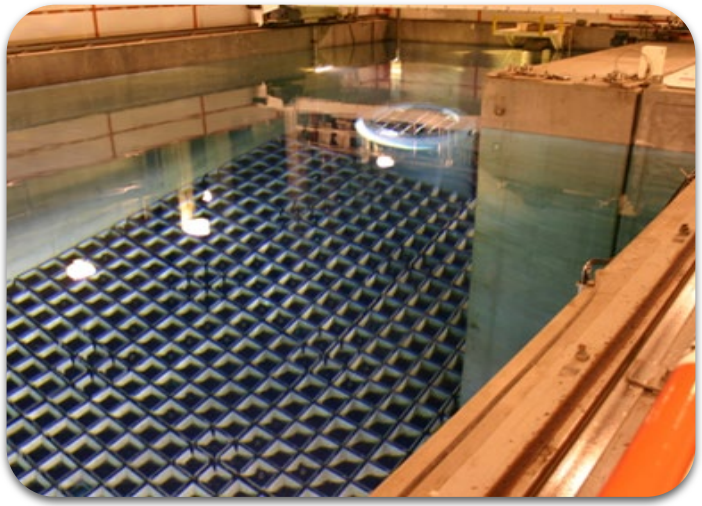
Objective of Project

Objectives

- Capability study of surface and volumetric inspection technologies for in-situ application to stainless steel dry storage canisters.
- Focus on NDE methods that do not require use of a liquid couplant and develop blind specimens for demonstration
- Perform Capability Demonstration using Selected NDE Method and Procedure



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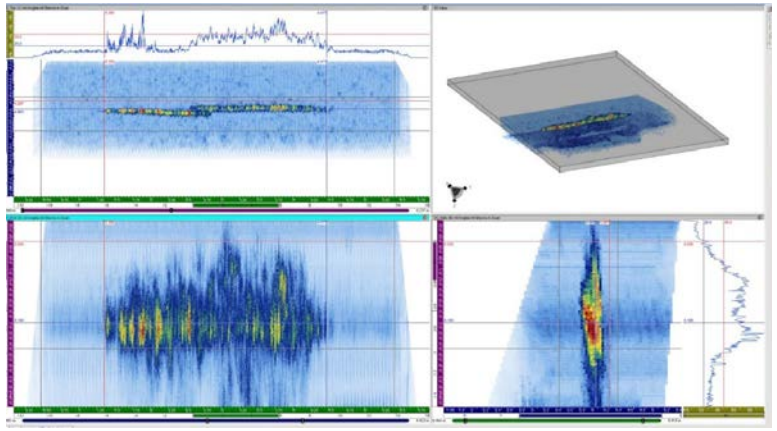
DCSS Environment Presents Challenges for Inspection

1 Temperature

- Originally estimated to be up to ~180°C
- Measured temperatures have ranged from 21-118°C (70-245°F)

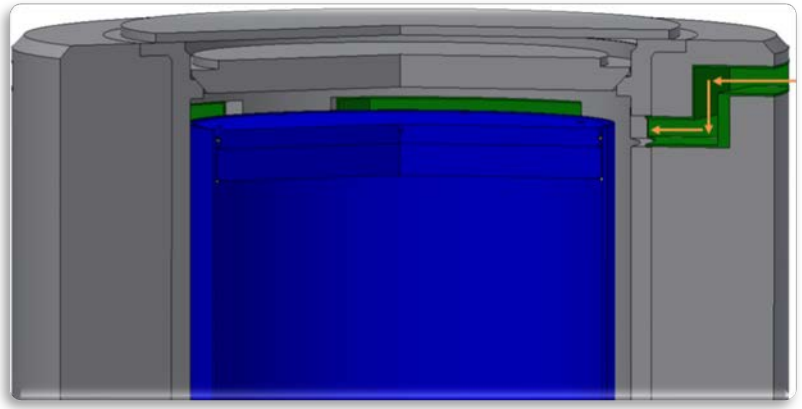


2 Radiation Dose

- Original estimate was ~1000-10,000 R/hr
- Max measured to date is ~100 R/hr

3 Challenging Access

- Confined entry and exit pathways
- Narrow annulus between canister and overpack with obstructions



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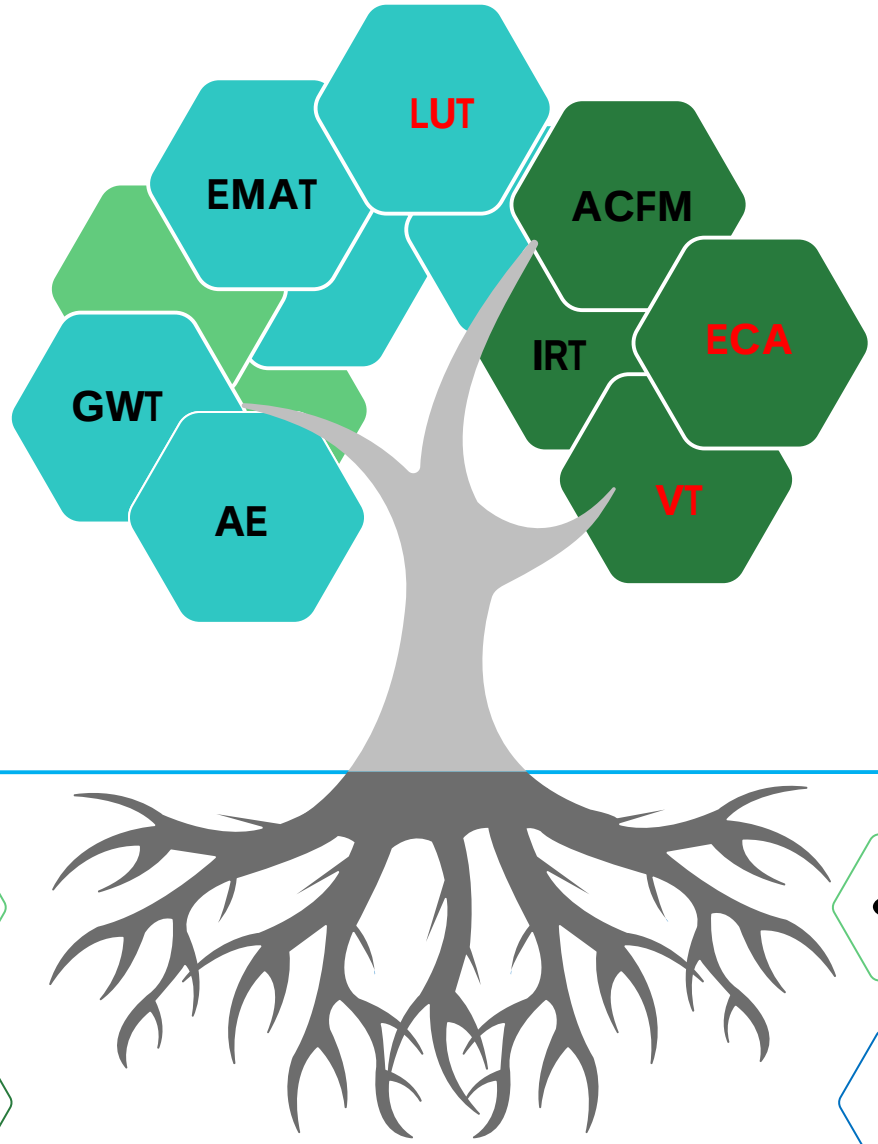
Candidate NDT Techniques

- **ACFM**: Alternating Current Field Measurement
- **AE:** Acoustic Emission
- **ECA**: Eddy Current Array
- **EMAT**: Electromagnetic Acoustic Transducer
- **GWT**: Guided Wave Testing
- **IRT:** Infrared Thermography
- **LUT:** Laser Ultrasonic Testing
- **VT:** Visual Testing



Surface NDE Technique

Volumetric NDE Technique

Technology Readiness

Is the technology commercialized, or can it be commercialized?

Applicability

Can the technology be deployed in confined spaces without couplant?

Resistance

Is the technique resistant to radiation and temperature?

Inspection

Can the detected defect be characterized?

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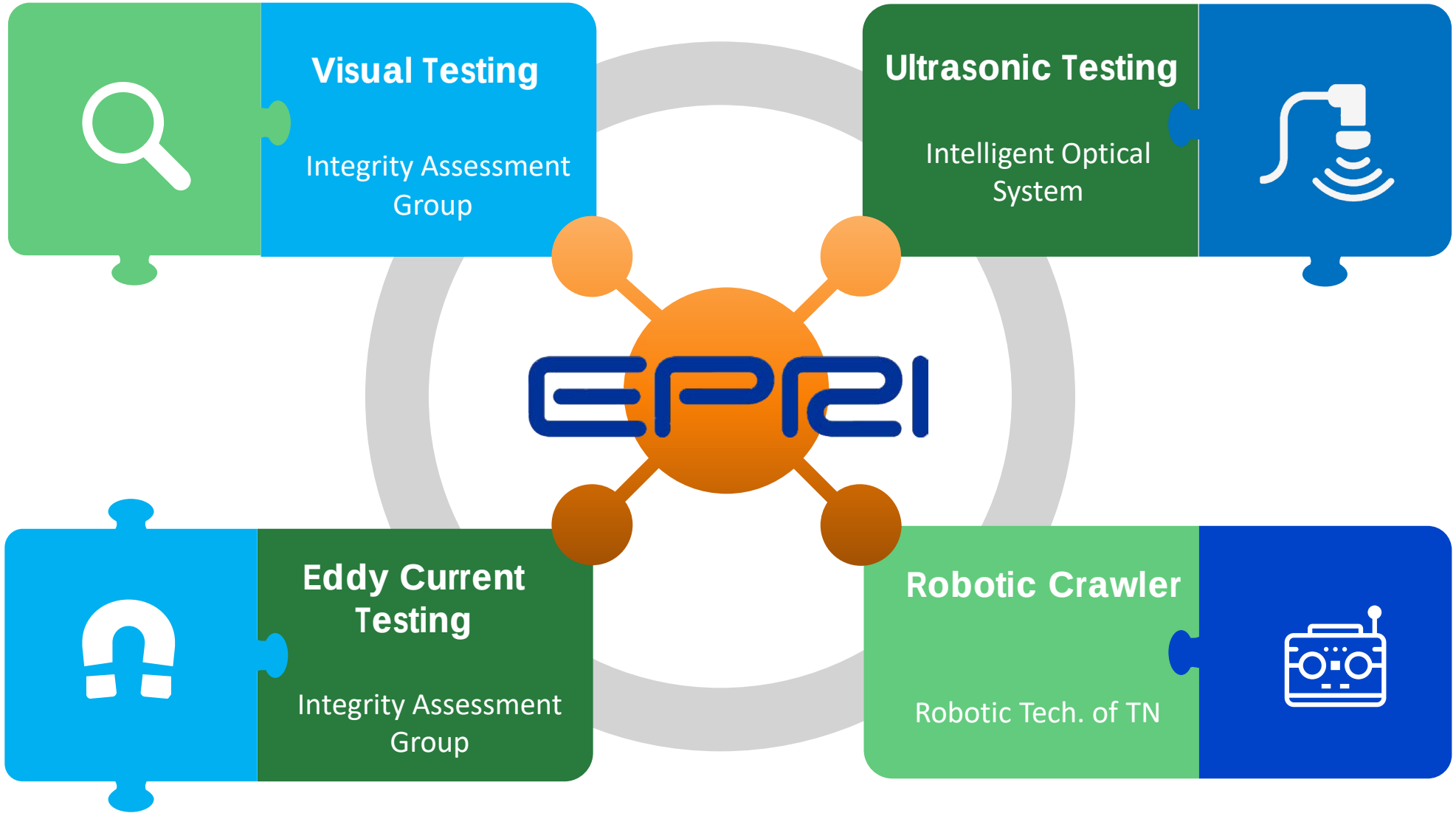
Industry Collaboration Structure



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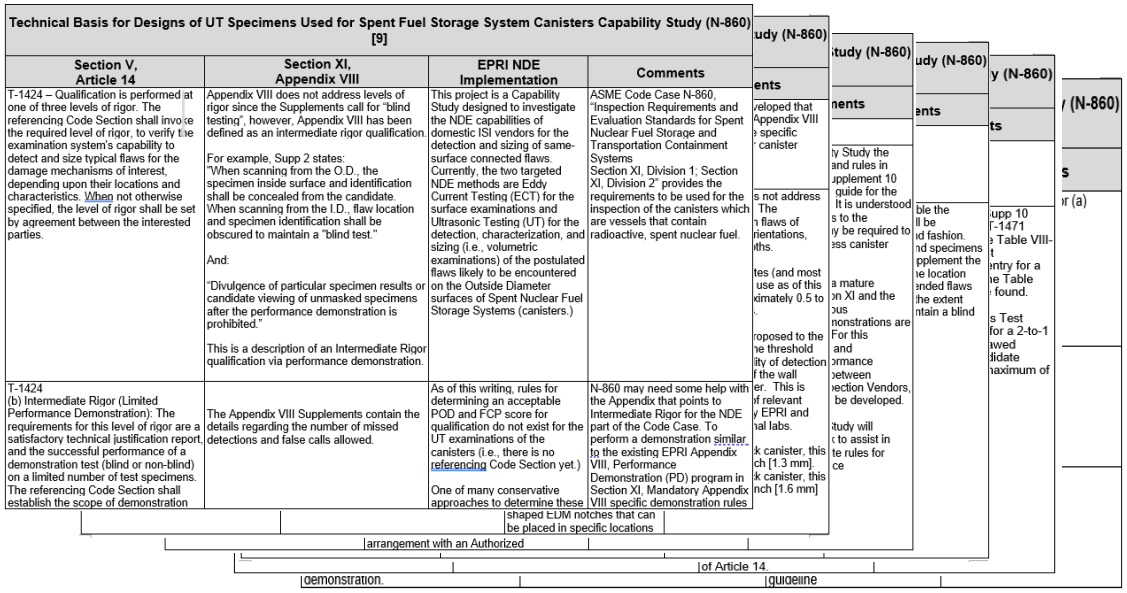
Technical Basis for Designs of Specimens

- EPRI developed a technical guidelines
  - Design of specimens for the capability assessment
  - Performance demonstration implementing for the UT examination



Technical Basis for Designs of UT Specimens Used for Spent Fuel Storage System Canisters Capability Study (N-860) [9]

Section V, Article 14	Section XI, Appendix VIII	EPRI NDE Implementation	Comments
T-1424 – Qualification is performed at one of three levels of rigor. The referencing Code Section shall invoke the required level of rigor, to verify the examination system's capability to detect and size typical flaws for the damage mechanisms of interest, depending upon their locations and characteristics. When not otherwise specified, the level of rigor shall be set by agreement between the interested parties.	Appendix VIII does not address levels of rigor since the Supplements call for "blind testing", however, Appendix VIII has been defined as an intermediate rigor qualification. For example, Supp 2 states: "When scanning from the O.D., the specimen inside surface and identification shall be concealed from the candidate. When scanning from the I.D., flaw location and specimen identification shall be obscured to maintain a "blind test." And: "Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited." This is a description of an Intermediate Rigor qualification via performance demonstration.	This project is a Capability Study designed to investigate the NDE capabilities of domestic ISI vendors for the detection and sizing of same-surface connected flaws. Currently, the two targeted NDE methods are Eddy Current Testing (ECT) for the surface examinations and Ultrasonic Testing (UT) for the detection, characterization, and sizing (i.e., volumetric examinations) of the postulated flaws likely to be encountered on the Outside Diameter surfaces of Spent Nuclear Fuel Storage Systems (canisters.)	ASME Code Case N-860, "Inspection Requirements and Evaluation Standards for Spent Nuclear Fuel Storage and Transportation Containment Systems Section XI, Division 1; Section XI, Division 2" provides the requirements to be used for the inspection of the canisters which are vessels that contain radioactive, spent nuclear fuel.
T-1424 (b) Intermediate Rigor (Limited Performance Demonstration): The requirements for this level of rigor are a satisfactory technical justification report, and the successful performance of a demonstration test (blind or non-blind) on a limited number of test specimens. The referencing Code Section shall establish the scope of demonstration	The Appendix VIII Supplements contain the details regarding the number of missed detections and false calls allowed.	As of this writing, rules for determining an acceptable POD and FCP score for qualification do not exist for the UT examinations of the canisters (i.e., there is no referencing Code Section yet.) One of many conservative approaches to determine these	N-860 may need some help with the Appendix that points to Intermediate Rigor for the NDE part of the Code Case. To perform a demonstration similar to the existing EPRI Appendix VIII, Performance Demonstration (PD) program in Section XI, Mandatory Appendix VIII specific demonstration rules

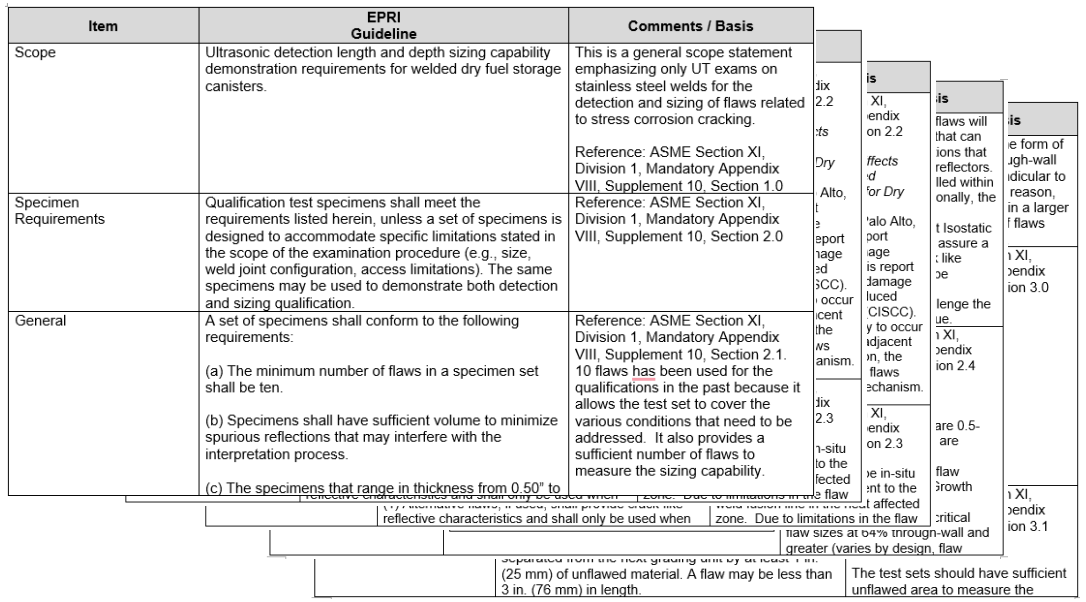
shaped EDM notches that can be placed in specific locations

arrangement with an Authorized

of Article 14.

demonstration. guideline

[Specimen design technical guideline]



Item	EPRI Guideline	Comments / Basis
Scope	Ultrasonic detection length and depth sizing capability demonstration requirements for welded dry fuel storage canisters.	This is a general scope statement emphasizing only UT exams on stainless steel welds for the detection and sizing of flaws related to stress corrosion cracking. Reference: ASME Section XI, Division 1, Mandatory Appendix VIII, Supplement 10, Section 1.0
Specimen Requirements	Qualification test specimens shall meet the requirements listed herein, unless a set of specimens is designed to accommodate specific limitations stated in the scope of the examination procedure (e.g., size, weld joint configuration, access limitations). The same specimens may be used to demonstrate both detection and sizing qualification.	Reference: ASME Section XI, Division 1, Mandatory Appendix VIII, Supplement 10, Section 2.0
General	A set of specimens shall conform to the following requirements: (a) The minimum number of flaws in a specimen set shall be ten. (b) Specimens shall have sufficient volume to minimize spurious reflections that may interfere with the interpretation process. (c) The specimens that range in thickness from 0.50" to	Reference: ASME Section XI, Division 1, Mandatory Appendix VIII, Supplement 10, Section 2.1. 10 flaws has been used for the qualifications in the past because it allows the test set to cover the various conditions that need to be addressed. It also provides a sufficient number of flaws to measure the sizing capability.

reflective characteristics and shall only be used when zone. Due to limitations in the flaw

flaw sizes at 64% through-wall and greater (varies by design, flaw

(25 mm) of unflawed material. A flaw may be less than 3 in. (76 mm) in length.

The test sets should have sufficient unflawed area to measure the

[PD Implementing technical guideline]

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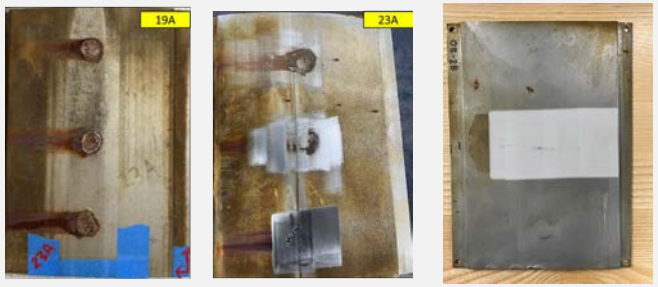
Specimens

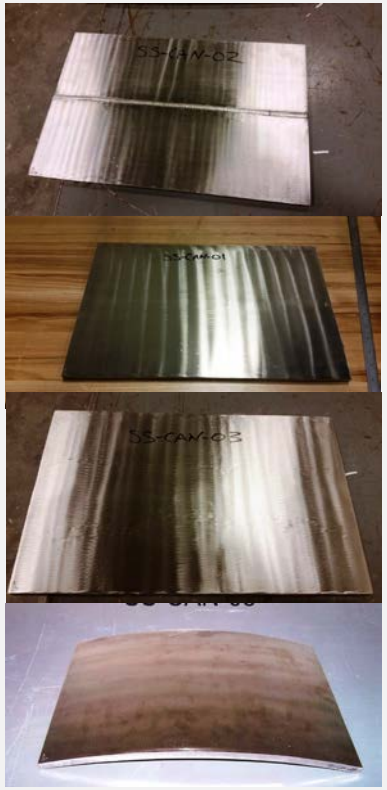
01 Open Specimens

- Four (4) ET specimens
- Four (4) VT specimens
- Three (3) corrosion specimens

[VT specimens]



[Corrosion specimens]



[ET specimens]

02 Blind Specimens

- Three specimens for capability assessment and one for practice

[Blind Specimens]

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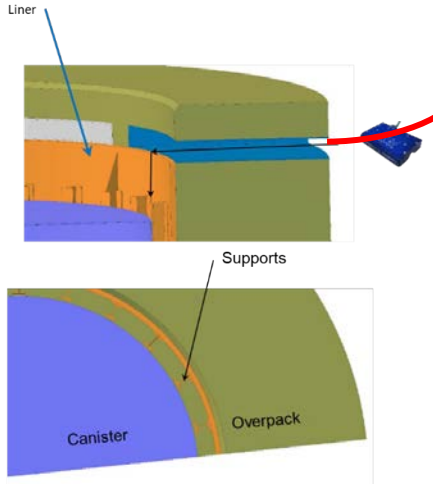
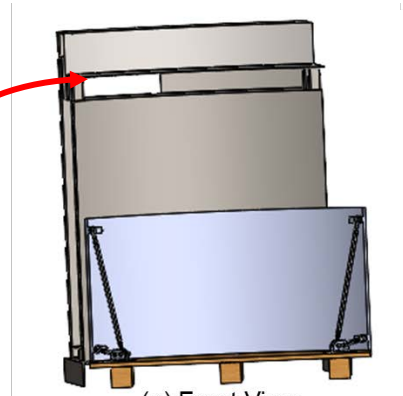
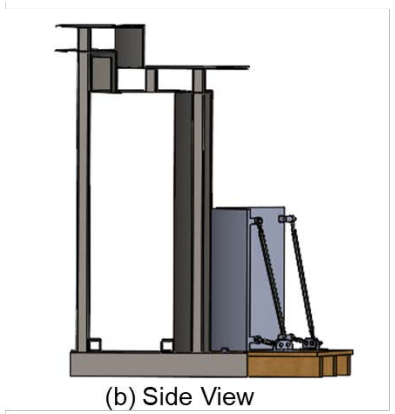
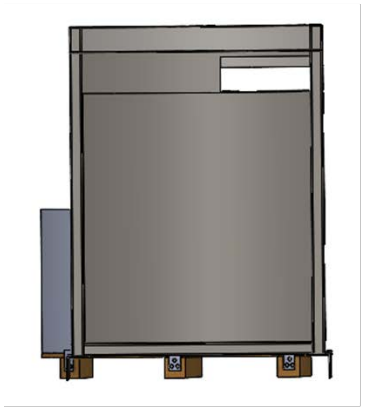
Canister and Overpack Partial Mockup

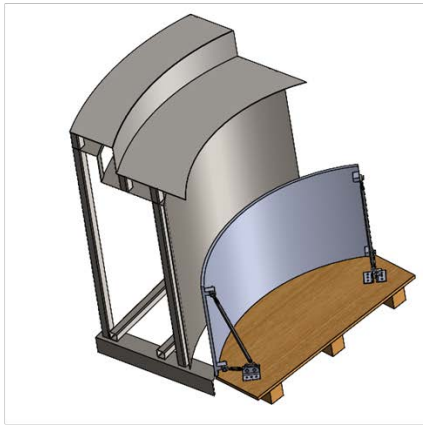
- Designed canister-overpack partial mockup




(a) Front View


(b) Side View


(c) Rear View


(d) 45° View

- Fabricated mockup for demonstration

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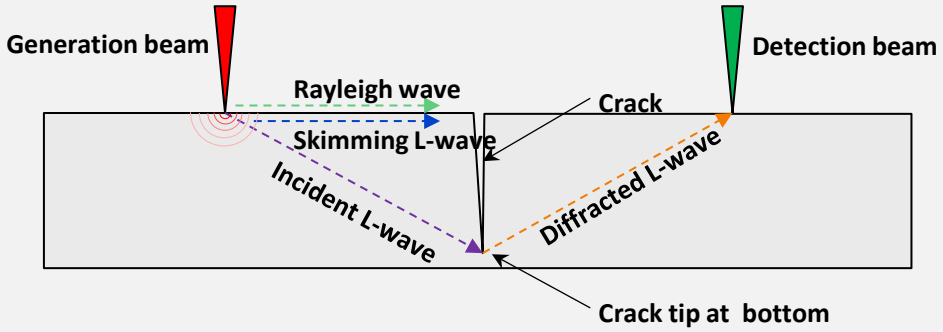
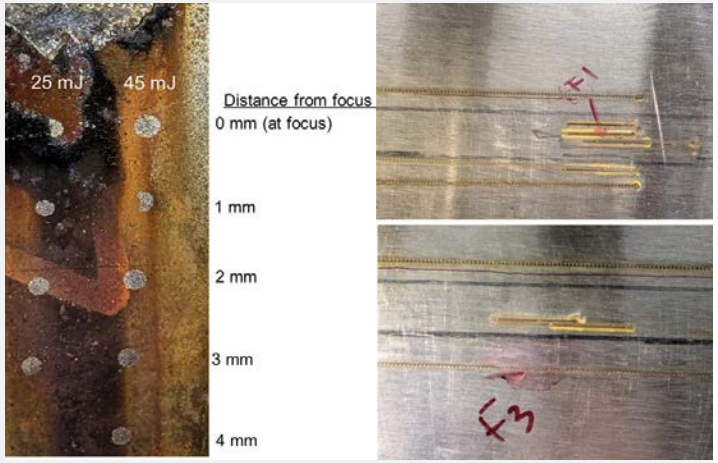
LUT Technique Development

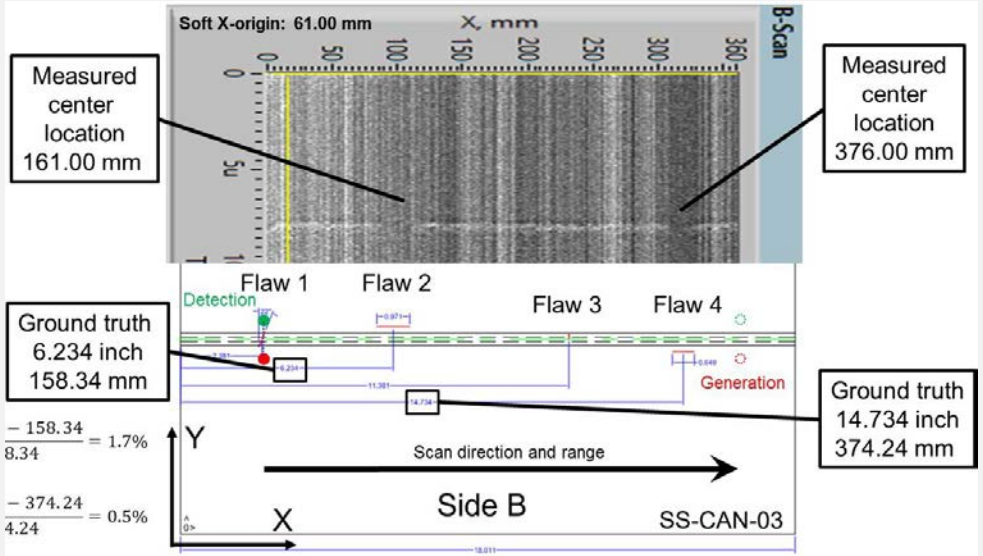
- Laser Ultrasonic Testing (LUT)



- Evaluated the effects of laser parameters
  - Optimized the laser: power, number of pulses, standoff
  - Heat tolerance
  - Surface effect



- Experiment on defect characterization



- Results
  - Capable of length sizing
  - No tip diffracted signal observed from the EPRI specimen
  - Permanent ablation marks on the testing surface

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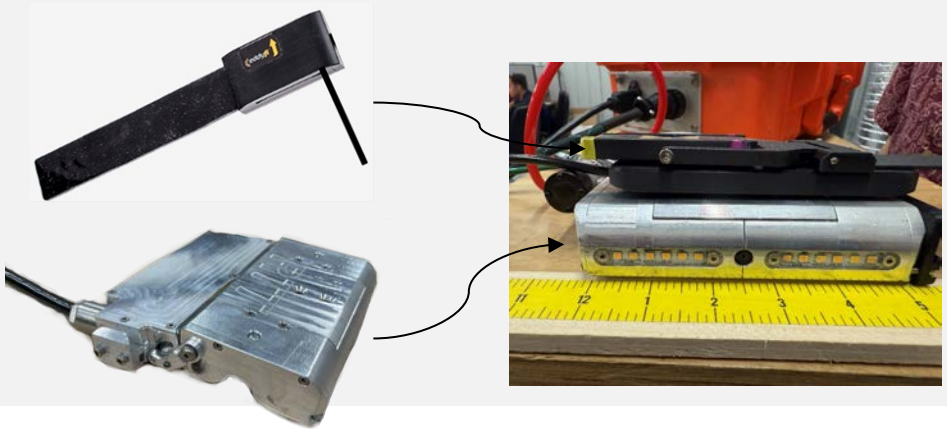
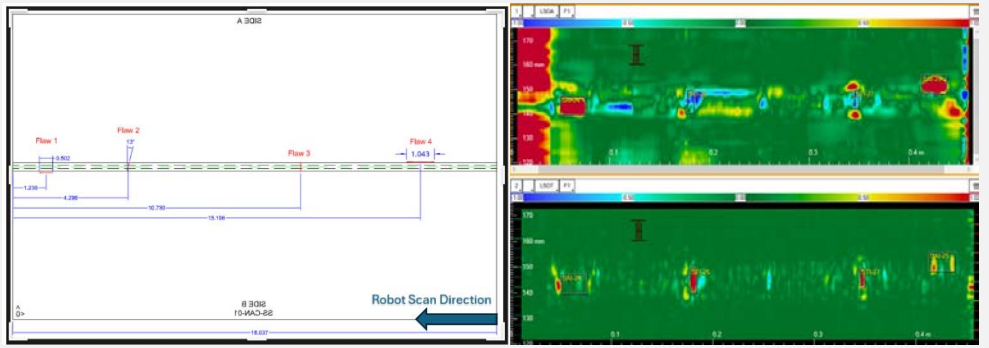
ECA Technique Development

- Eddy current array (ECA) technique

- Technique development
  - Customized the standard I-Flex probe
  - Generate depth sizing curves.
  - Manual scanning vs. remote scanning.
  - Weld cap effect and scanning directions.



- Experiment on defect characterization
  - Use EPRI open specimens and tested in various field conditions to develop procedure.



[Welded open specimen signal acquired from robotic crawler]

- Results
  - Excellent in flaw detection and length sizing.
  - ECA inspection technology and delivery robot have proven to be an effective solution for detecting defects in parent metal, HAZ and welded area.
  - ECA procedure developed for field inspection.

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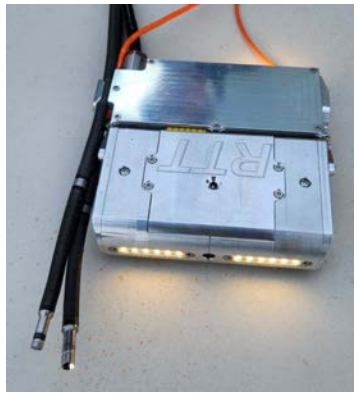
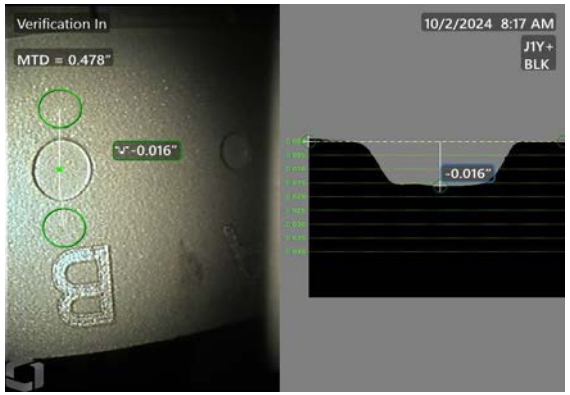
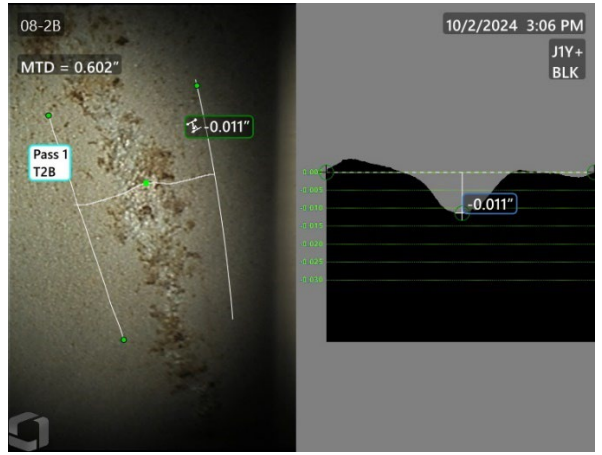
Visual Testing (VT) Capability Assessment

- Visual testing:
  - Conducted to detect discontinuities and imperfections on the surface of components such as cracks, wear, corrosion, or erosion.
- Selected 6 VT specimens
  - 5 with wide weld areas, 1 ground flush
  - 16 fatigue flaws included in specimens
    - 9 have COD > 25 microns
- Two video probes with stereo & phase viewing tips





- VT capability test



- Results
  - Visual inspections meeting ASME Section XI VT-3 and VT-1 criteria.
  - The smallest width measured was 0.076 mm (0.003 in).
  - Tight defect with a ratios of 5:1, 10:1 and 20:1 were not accurately measured the depth.

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Capability Demonstration

- ECA capability assessment
  - Participant: IAG
  - Date: July 07- 09, 2025
  - EPRI PD facility: secure area
  - Using the developed ECA procedure
- Secure data management
  - The mockup was covered by blanket during the testing
  - The laptop used to acquire the ECA data was swept three (3) times before the equipment was removed from the PD facility.

[Cover canister-overpack partial mockup for testing]

[Remote control of the robotic crawler]

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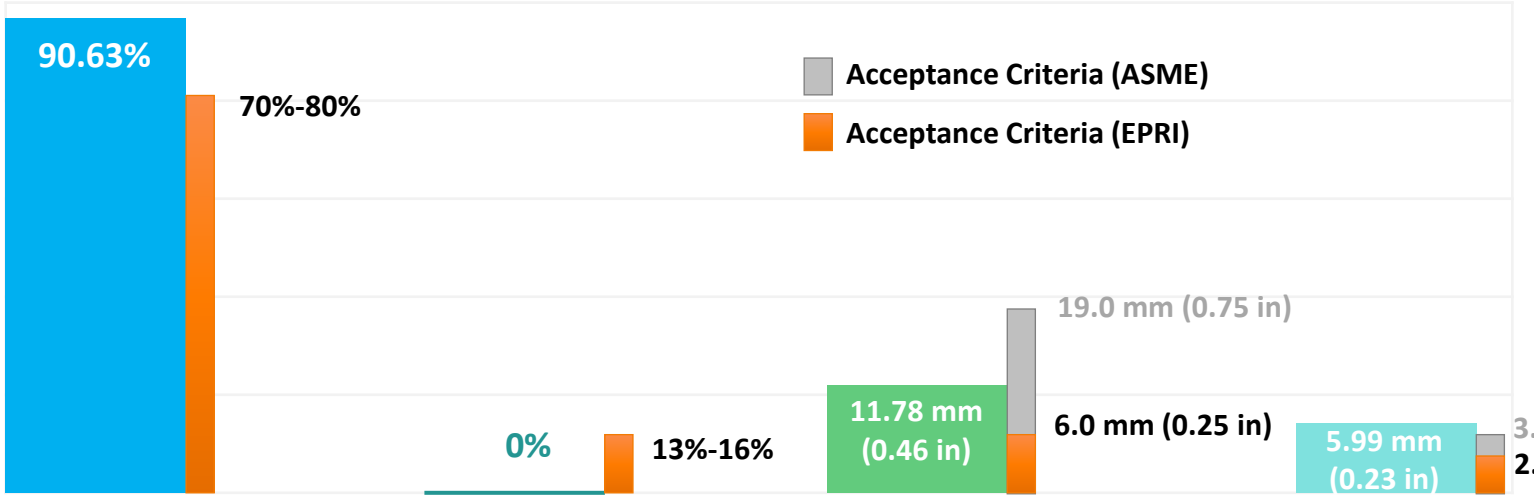
Demonstration Results

- Vendor submitted the results to administrator
- Grading results



Spent Fuel Canister Detection and Sizing Data Sheet (OD)

Candidate's Name:			Company:			
Block ID:		Test Set ID:	Scan Area:			
File Name:						
Ind. No.	Surface (ID/OD)	Depth to Indication (OD Flaw Height / ID Flaw RL) (in.)	End Point 1		End Point 2	
			Y1 (in.)	X1 (°)	Y2 (in.)	X2 (°)

[Detection report sheet]

※ *The flaws missed were all oriented perpendicular to the weld and at the toe of the weld.*

※ **Acceptable detection rate based on requirement of ASME Section XI, Appendix VIII. Length and depth sizing did not meet the acceptance criteria**

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Summary and Conclusion

EPRI Developed Technical Guideline

- Established a framework for administering the NDE inspection capability demonstration
- Technical guideline of DSS specimen design

Mockup Fabrication

- Four (4) new mockups have been developed
- Canister and overpack partial mockup fabricated for capability demonstration

NDE Technique Development

- VT capability assessment was conducted using EPRI open specifications
- ECA inspection technique and procedure were developed and demonstrated excellent defect detection and sizing capabilities
- LUT revealed limitations in defect detection due to defect orientation and challenges in identifying the tip-diffraction signal.

ECA capability demonstration conducted at the EPRI site: showed an acceptable detection rate but did not meet the defect sizing requirements

Final report will be available in November: 3002031987

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Updates on ASME Code Cases

Code Case N-860 Rev 1

- Revision to Code Case N-860 required to support inspection requirements for cold spray deposits
- Opportunity to make revisions to clarify intent of corrosion degradation for expanded examination requirements
  - Class C corrosion requires supplemental examination using surface (eddy current) or volumetric (ultrasonic) methods OR “analysis in lieu”
  - Include language in degradation classification section to clarify water staining types of deposits from corrosion degradation
- Include 40-year inspection interval concept from RG 3.78 acceptance of CC N-860 Rev 0

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Cold Spray Code Case – Background

- ASME Section XI Code Case N-860 defines inspection requirements
  - Prepare for inspection results that may need to be “fixed”
- Draft CC N-927 being developed
  - Define requirements for cold spray mitigation of canister degradation
- Draft Code Case and technical basis document passed by ASME canister groups August 2024
  - Demonstration-based approach using witness coupons
  - Focus on porosity and adherence for performance

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Cold Spray Code Case – Status Update

- Letter ballot at WG Welding and SG RRA September 2024
  - Small number of “Disapprove” votes
  - Large number of comments associated with those votes
- Code Case revised to address comments and Disapprove votes
  - Based in part on technical requirements for qualification of DOE canisters
- September 2025 letter ballot resulted in even more comments and Disapprove votes
- Canister groups continue working to address comments in 2025
  - Educate reviewers on demonstration-based approach vs one that specifies the essential parameters

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ESCP Aging Management Working Group Meeting: Key Recommendations and Status Update

Hatice Akkurt
Used Fuel and High-Level Waste Management, EPRI

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 27-30, 2025

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ESCP Aging Management Working Group Meeting

Objectives:

1. **Evaluation of the current status of the aging management research** - what has been done to date and what can be concluded from the results.
2. **Development of a roadmap for the next 2 years** to conclude key items (2 years not calendar time – assuming respurces are in place).

Working Group Meeting:

1. Participation by invitation only. List of invited guests developed with input from Subcommittee and Task group leads. Number of registered attendees: 35
2. Attendees include participants from utilities, vendors, regulator, DOE, National laboratories, NEI, university.
3. Developed questionnaire and distributed prior to the meeting.
4. Parallel sessions and joint sessions were scheduled.

Documentation of what has been done in the past >10 years and development of roadmap for closure of key items within the next 2 years were the key objectives

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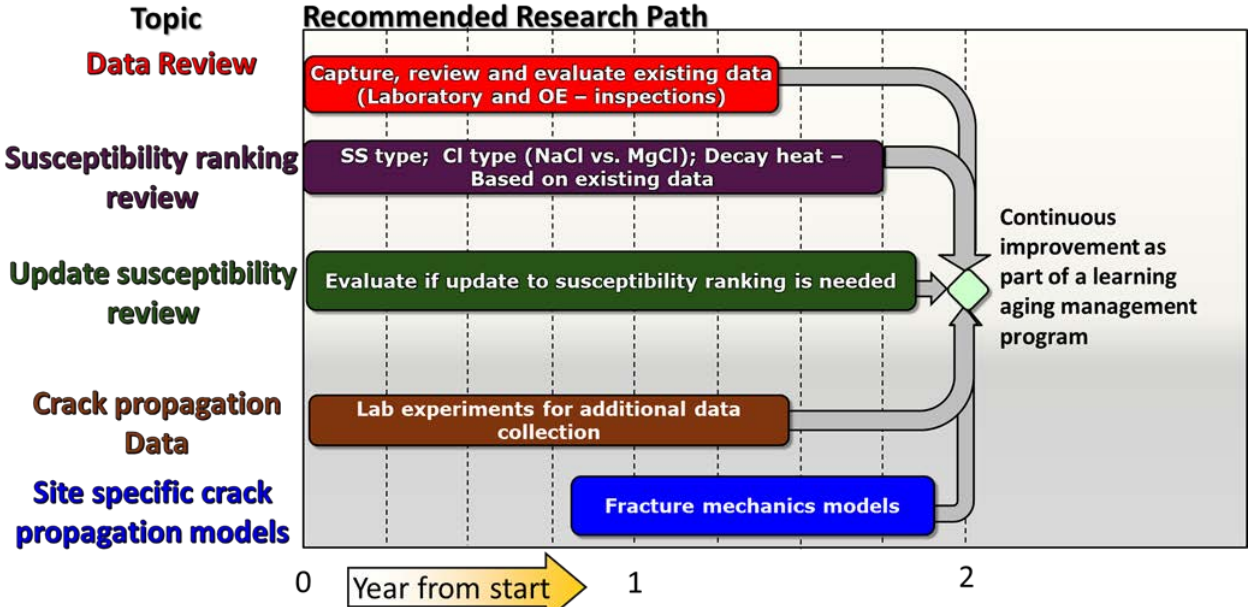
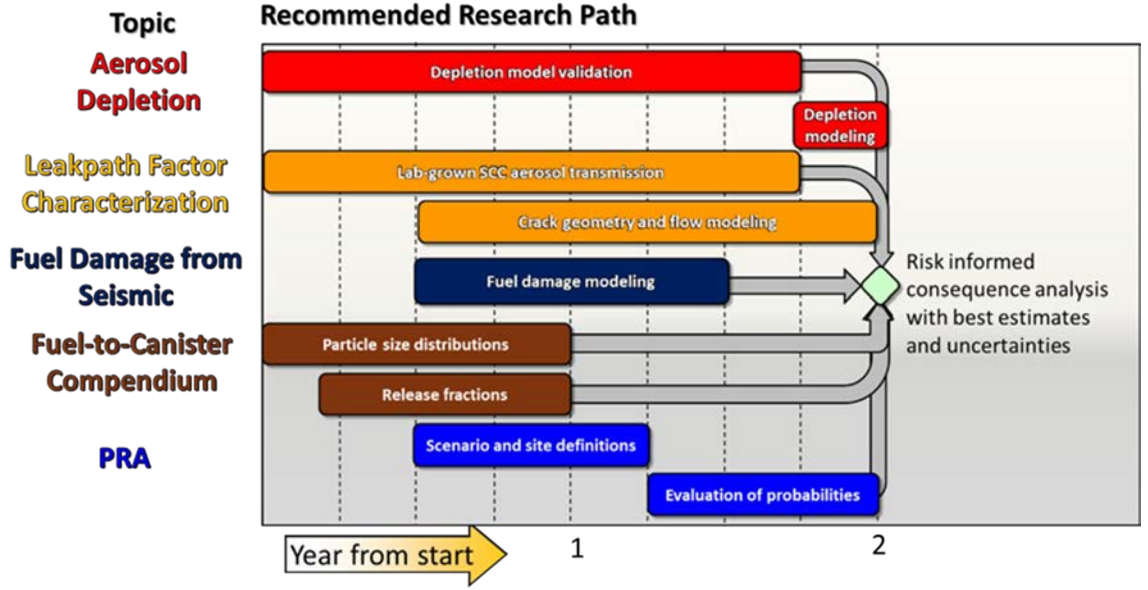
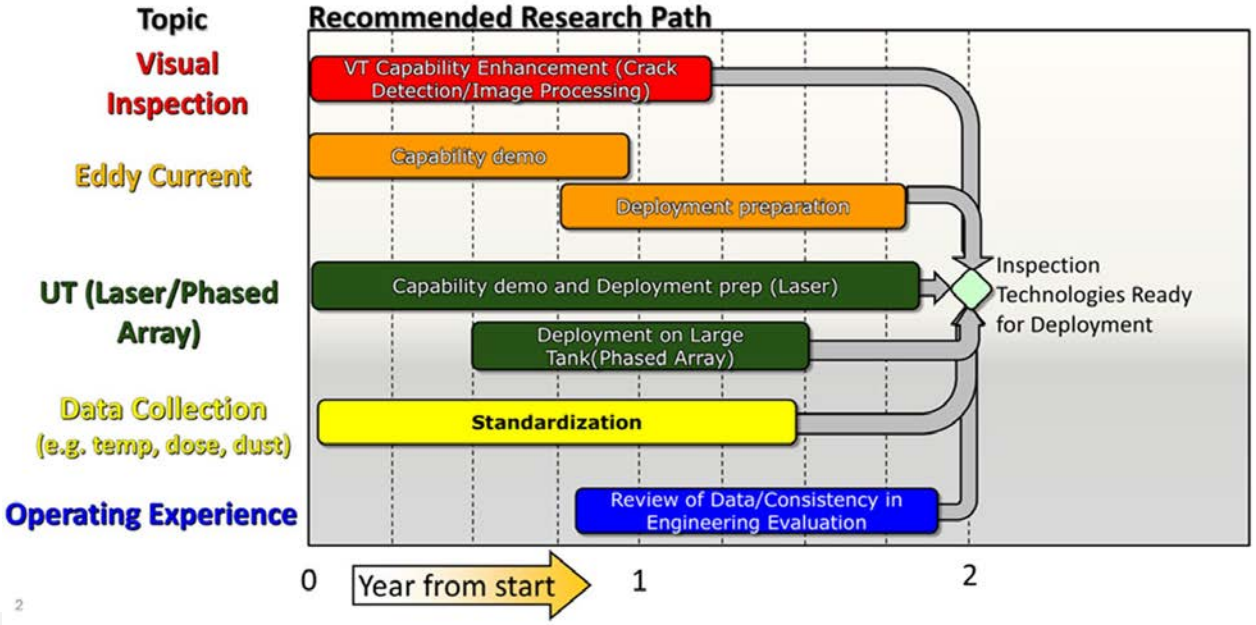
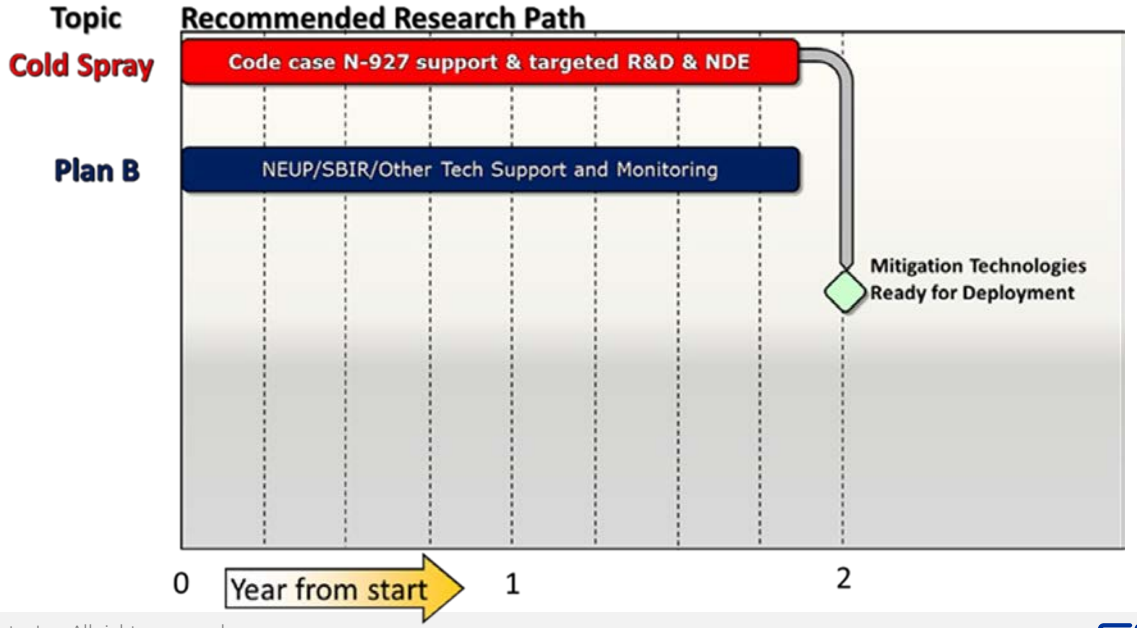
Aging Management Workshop & Recommended Roadmaps









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Working group key recommendations and the pillars supported by the key recommendations

Key Recommendation	Pillars Supported
Evaluation of all the laboratory data, related to aging mechanisms and consequence, to date	Probability; Consequence; Inspection
Evaluation of inspection data to date with the aim of improvements	Probability; Inspection; Mitigation
Collection of temperature/dose data during inspections and analysis/database	Probability; Inspection; Mitigation
Collection of surface deposit samples during inspection	Probability; Inspection; Mitigation
Standardization of sample collection and analysis, including independent analysis at multiple labs	Probability; Inspection; Mitigation
Collection of crack growth data for enhancements for the model used in ASME N-860 and xLPR type prediction tools	Probability; Inspection
Evaluation of surface and volumetric inspection tools and determination of path forward based on demonstration	Probability; Inspection; Mitigation
Development of Plan B for mitigation, besides cold spray	Mitigation
Collection of particle release fractions and sizes from analog experiments	Consequence; Inspection
Generating validation data for GOTHIC/MELCOR for consequence studies	Consequence; Mitigation

Each recommendation supports multiple pillars

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Key Recommendations & Lead Performers: Discussion with ESCP Steering Committee Members

Key Recommendation	Lead Organization*
Evaluation of all the laboratory data, related to aging mechanisms and consequence, to date	EPRI/ESCP Task Groups**
Evaluation of inspection data to date with the aim of improvements	EPRI
Collection of temperature/dose data during inspections and analysis/database	Utilities during scheduled inspections
Collection of surface deposit samples during inspection	Utilities during scheduled inspections – EPRI/Labs
Standardization of sample collection and analysis, including independent analysis at multiple labs	EPRI/Labs
Collection of crack growth data for enhancements for the model used in ASME N-860 and xLPR type prediction tools	NRC/DOE/NL
Evaluation of surface and volumetric inspection tools and determination of path forward based on demonstration	EPRI (ongoing activity)
Development of Plan B for mitigation, besides cold spray	DOE/NEUP & SBIR
Collection of particle release fractions and sizes	EPRI/ESCP
Generating validation data for GOTHIC/MELCOR for consequence studies	Joint collaborative effort

*Tentative list/lead organization for SC discussions

**Provided access to the data is granted

Many of the key recommendations support multiple pillars. Recommendations will require collaboration between different stakeholders

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RG 3.78 versus ASME Code Case N-860

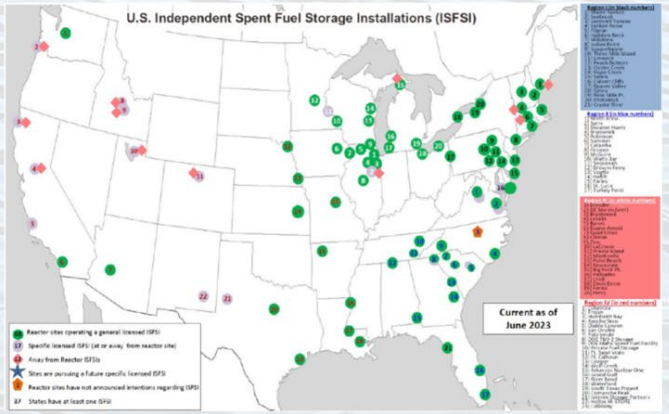
ISFSI locations and Z_{ISFSI} values

71 operating ISFSI sites with 3755 welded austenitic stainless steel DSS in use as of December 2023

$$Z_{ISFSI} = Cl_{starting} + Cl_{adj} + AH_{adj}$$

EPRI–3002005371

Z_{ISFSI}	ISFSI sites (estimated)	DSSs (estimated)
1-3	60	3,148
4-7	6	295
8-10	5	312



Composition of salt deposits at ISFSI sites with $Z_{ISFSI} \leq 3$ are not expected to promote CISCC

Assuming CISCC could occur, CISCC propagation at $Z_{ISFSI} \leq 3$ sites would proceed slowly

Both ASME CC N-860 & RG. 3.78 uses EPRI–**3002005371**, Susceptibility Assessment Criteria for CISCC to set initial inspection population

$$Z_{ISFSI} = Cl_{starting} + Cl_{adj} + Ah_{adj}$$

High rank site: $Z_{ISFSI} \geq 8$; Low rank site: $Z_{ISFSI} \leq 7$

ASME CC N-860: Maximum inspection interval can be increased from 10 to 20 years for low rank sites (ZISFSI ≤ 7) after a minimum of 3 inspections that show no indications of CISCC

RG 3.78: Alternative inspection frequency

- Maximum inspection interval of 40 years
- Applicable to ISFSI sites with a CISCC susceptibility ranking of 3 or below (EPRI-3002005371)
- Does not require a minimum of 3 inspections
- Contingent on prior inspection results showing no indications of corrosion or CISCC

AMPs start conservative and refined as more data/OE becomes available

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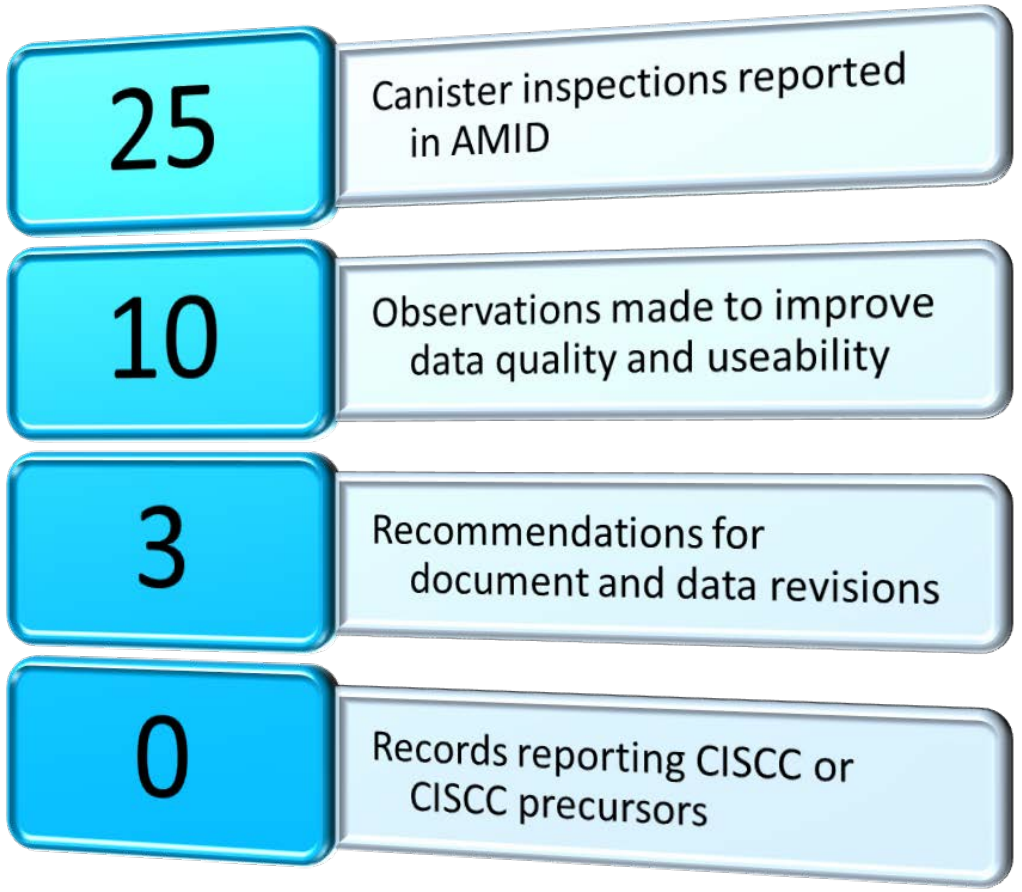
Status Update on Key Recommendation 2: Evaluation of the Inspection Data to Date - EPRI Review of AMID Entries

- Holtec, NAC, Orano, and Westinghouse agreed to allow EPRI to review AMID data for tracking and trending
- **EPRI thoroughly reviewed the data**
  - Gathered data for tracking and trending
  - Found inconsistencies in data entries
- **EPRI recommendations:**
  - Update existing entries to correct records
  - Update NEI 16-10 guidance document
  - Modify AMID to facilitate data entry and collect additional information based on recent advancements in CISCC research
- **NEI issued interim recommendation to AMID users**



EPRI Recommendations will improve AMID for meeting AMP requirements

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Key Recommendations 3: Collection of Temperature/Radiation Dose Data During Inspections – Analysis & Database

- Utilities are collecting temperature/dose data during inspections
- EPRI will start database formation efforts next year

Key Recommendations 4-5: Sampling and Standardization of Sample Collection

- Number of samples already collected from different sites
- Efforts underway for standardization and best practices (earlier presentation by EPRI/DOE

Status Update on Key Recommendation 7: Evaluation of surface and volumetric inspection tools and determination of path forward based on demonstration

- Based on the demonstration studies, presented earlier, EC is ready for deployment
- Laser UT still has challenges
- UT with couplant can be used

Agreement at the working group to use Cold Spray if VT followed with EC show indications

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Key Recommendation 1: Evaluation of all the laboratory data, related to aging mechanisms and consequence, to date

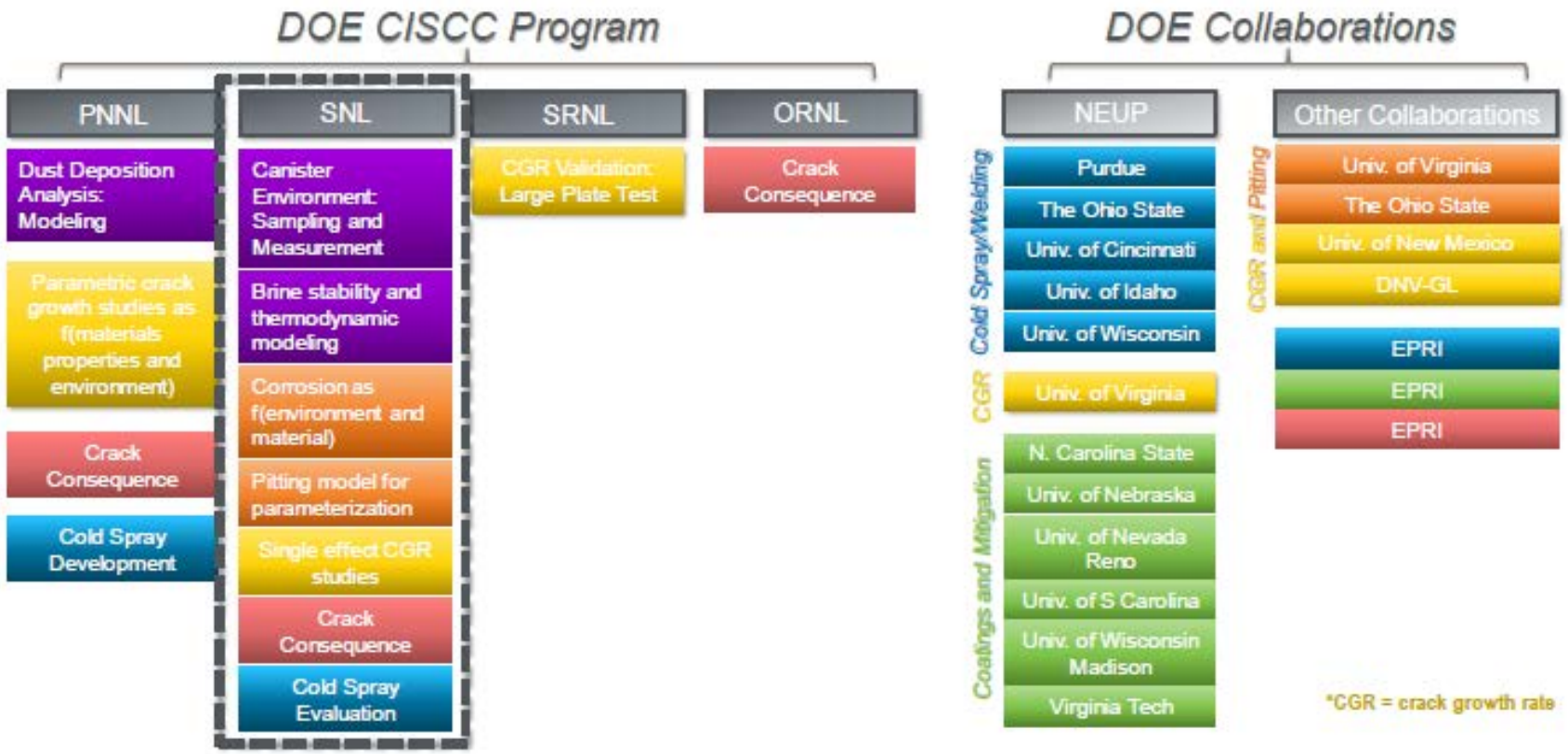
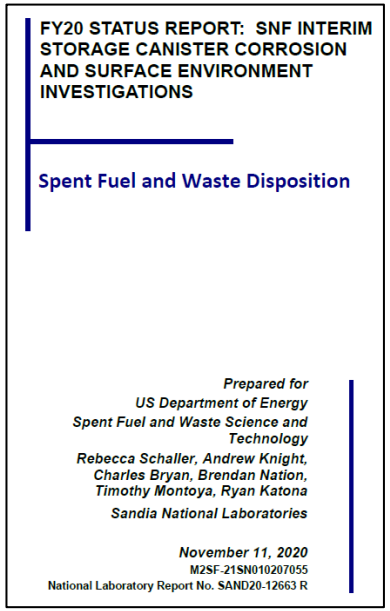


Figure 1. Experimental work on canister SCC currently being carried out by SNL and other national laboratories and collaborators.



FY20 STATUS REPORT: SNF INTERIM STORAGE CANISTER CORROSION AND SURFACE ENVIRONMENT INVESTIGATIONS

Spent Fuel and Waste Disposition

Prepared for
US Department of Energy
Spent Fuel and Waste Science and Technology
Rebecca Schaller, Andrew Knight, Charles Bryan, Brendan Nation, Timothy Montoya, Ryan Katona
Sandia National Laboratories

November 11, 2020
M2SF-21SN010207055
National Laboratory Report No. SAND20-12663 R

Incomplete list as it does not include non-US tests (i.e., CRIEPI, and others)

Significant amount of work across many organizations – capturing and evaluating this data would be very beneficial

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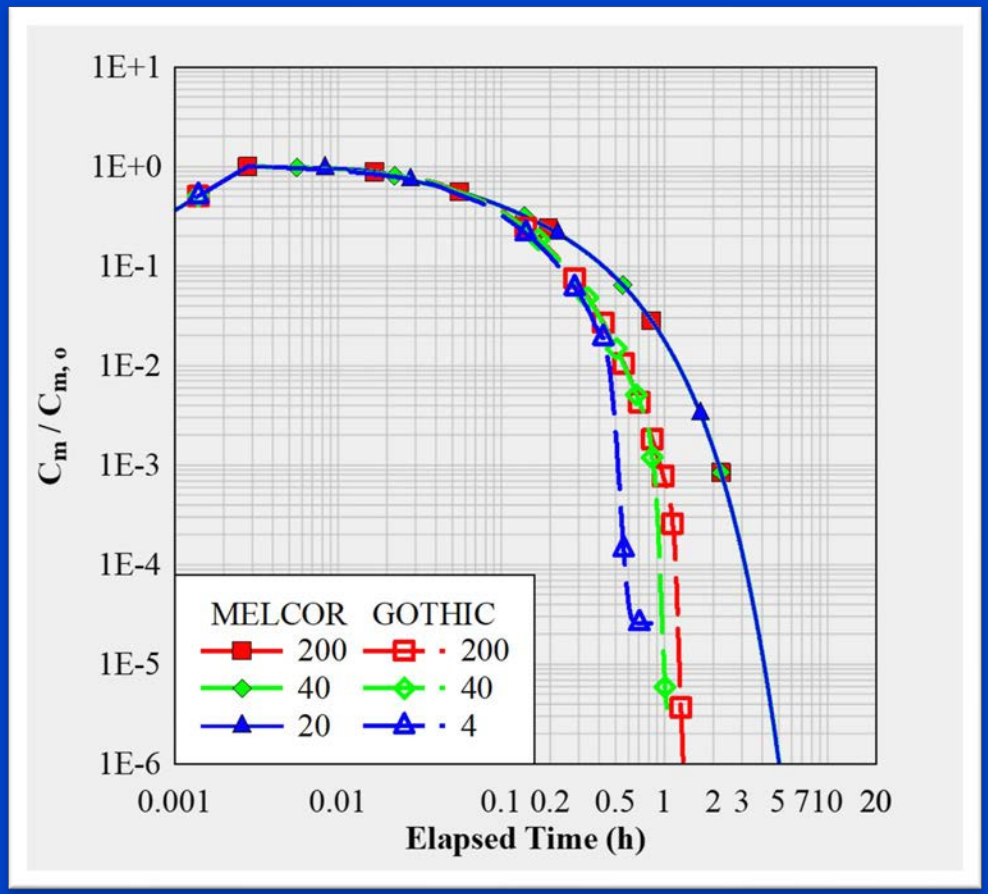
Status Update on Key Recommendation 9: Collection of Particle Release Fractions & Particle Size

ESCP Fuel Subcommittee will generate a report in 2026. Report will include contributed data from

- JRC – agreed to provide the data
- ORNL (Sibling Rod Tests)?
- SCIP?
- Other sources?

Realistic approximations will improve the models and **reduce excess conservatism** in consequence studies

Key Recommendation 10: Generating Validation Data for GOTHIC/MELCOR for Consequence Studies



6 orders of magnitude reduction in normalized aerosol mass depletion in order of 1-5 hours

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Summary on the Status of the Key Recommendations

In Progress

1. Evaluation of inspection data to date with the aim of improvements
2. Evaluation of surface and volumetric inspection tools and determination of path forward based on demonstration
3. Collection of temperature/dose data during inspections and analysis/database
4. Standardization of sample collection and analysis, including independent analysis at multiple labs and sample collection during inspections
5. Collection of crack growth data for enhancements for the model used in ASME N-860 and **xLPR type prediction tools**
6. Collection of particle release fractions and sizes from analog experiments
7. Development of Plan B for mitigation, besides cold spray

Not Started

1. Evaluation of all the laboratory data, related to aging mechanisms and consequence, to date
2. Generating validation data for GOTHIC/MELCOR for consequence studies

The two items in "No Progress to Date" category will require collaboration across multiple organizations

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Questions/Comments?

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ESCP Task Group: Bolted Cask Aging Management

Under ESCP Aging Management Subcommittee

D. McGinnes
Axpo Power

R. Schneider-Eickhoff
BGZ

29.10.2025

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1st subgroup meeting (21.10.2025)

- Clarification of the subgroup's objectives and the participants' expectations of the collaboration. Are both compatible, or do adjustments need to be made? ✓
- Informing participants about the framework conditions for the collaboration. ✓
- Establishing the roadmap. ✓
- Agreement on how the group will work, how often it will meet, ✓ and who is responsible for which tasks. ✗

1st meeting 21.10.2025 (CET 13.00 to 14.30 MS Teams)

2nd meeting 29.10.2025 (ESCP AM Subcommittee Meeting 16.40 to 17.00)

3rd meeting 20.01.2026 (CET 14.00 to 15.30 MS Teams)

4th meeting 24.03.2026 (CET 14.00 to 15.30 MS Teams)

5th meeting 02.06.2026 (CET 14.00 to 15.30 MS Teams)

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2nd subgroup meeting (29.10.2025)

- Nomination of a person to be responsible for moving each task forward
- Nomination of one person per country to coordinate the collection and collation of information (France, Germany, Japan, Spain, Switzerland, USA)
- Agreement of next steps

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Objectives of the subgroup

- Until now, the EPRI-ECSP program has mainly focused on aging management and the long-term behavior of canister-based storage systems and inventory. With the new sub-group, the focus will be on bolted cask-specific topics.
- If comparable boundary conditions can be observed, it would be desirable to harmonize the scope and cycles of bolted cask inspections internationally, or to make appropriate recommendations.
- It should be examined whether the inspection methods used to date are considered sufficient for aging management; any need for further research / improvement should be identified.
- Preliminary breakdown of tasks:

  Task 1 Bolted Cask Data Collection

  Task 2 Aging Management

  Task 2.1 Aging Effects Relevant for Bolted Casks

  Task 2.2 Ageing Management Programs

  Task 3 Tests / research programs (on the TN 32 cask)

  (Reflooding -> ESCP Fuel Grp.)

For each Task,

- a responsible person must be appointed and
- an EPRI report must be created.

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Task 1 - Bolted Cask Data Collection

The first step is to gain an overview of the subject matter, i.e., the casks and their storage conditions (similarities/differences).

1) Identification of bolted cask types in use for storage
   - Cask type, manufacturer, normal storage/transport configuration, number of casks, design lifetimes,
   - Design principles/features relevant to ageing (materials, corrosion protection, etc.)
   - Repair concepts (for the lid system)
   - Content & amount (BWR, PWR, HLW, etc.), thermal load at the time of loading
2) Storage location & environment
   - Internal (storage hall) / external (storage pad) - details environment, etc.
   - Monitoring measures (leak-tightness, temperatures, radiation level)
   - Operational activities (storage, transports),
   - Infrastructure of the ISF (service station, hot cell, overhead crane, trolley, etc. …)
3) Regulatory requirements
   - International: IAEA SSR-6, IAEA Draft Safety Standard DS546 etc.
   - National: (France, Germany, Japan, Spain, Switzerland, USA …)

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Task 2.1 - Aging effects relevant for bolted casks

Development of recommendations for the implementation of inspections on bolted casks during the storage period and the monitoring of critical cask parameters (temperatures, leak rate, dose rate values, etc.)

1) Relevant Ageing mechanisms & potentially affected component (based on the outcome of Task 1)

- Relaxation – bolts, metal seals
- Corrosion – external surfaces
- Thermal aging – organic components
- Radiation aging – organic components, absorber materials

2) Identification of suitable monitoring and inspection methods

- Which inspection methods are available/suitable for use on bolted casks (to detect aging effects or monitor safety objectives)?
- What is needed / nice to have ?
- Anything EPRI can do that is not being done elsewhere.
- Application of canister (thin wall) monitoring methods to thick-walled bolted casks ?

➔ GAP Review [what needs to be done and what currently can be done?]

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Task 2.2 Ageing Management Programs

Development of recommendations for the implementation of inspections on bolted casks during the storage period and the monitoring of critical cask parameters (temperatures, leak rate, dose rate values, etc.)

1) Ageing Management Programs
   - Compilation/evaluation of existing regulations/publications related to aging management of bolted casks
   - Use MAPS NUREG-2214 AMP format?
   - Implemented / planned AM programs
   - Maintenance programs / repair concepts
   - Spare part management

2) Develop the basis for the scope and frequency of inspections (level - good practice)
   - Environmental considerations, dew point, lifetimes of coatings e.g. Zn/Al, etc.
   - Statistical basis (sample size methodologies - established for NPP components)
   - Scope and form of the documentation of inspection results (for later understanding / further processing, e.g. statistical evaluation of findings)
   - Use of reference cask(s) (representative and accessible for periodic inspection)

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Task 3 - Tests / research programs (on the TN 32 cask)

1) Short term research work **on the TN 32 cask** to autumn 2026 (Lead: Keith Waldrop EPRI)
   - Measurement of He concentration in the OP System and Cask Cavity: Any change after 20 years storage without fuel?
   - Replace bolts one at a time effect on seal?
   - Leaking quick connection – how to deal with this if it does not reseal after gas sample is taken?
2) HBU – Dry Run (2027?)
3) Potential future research (from 2028?) subject to availability of the cask (Lead Samuel Trost INL).
   - Benchmarking of thermal models: correlation of surface temperatures with component temperatures, e.g. primary lid seals
   - Remote monitoring (through wall): selection of most promising technology and facilitating its further development (cavity gas composition, dose rate, temperatures, etc.)
   - Testing of technology to remotely inspect cask surfaces where access between casks is restricted (full cask storage halls), presence of cables (cask monitoring)
   - In-situ measurement of the status of bolts, e.g. relaxation
   - Demonstration of the feasibility of exchanging primary lid bolts without effecting the safety case for transport (performance of seals under accident conditions) and storage (handling accidents without shock absorbers).
   - Trunnion bolts, long term storage, requirements to perform load tests before final transport
   - Exchanging of primary lid seals using a mobile facility
   - Effect of ageing of lubricant on friction coefficient and its effect on retorquing (deviation from require torque)

General: To be able to propose research programs it is necessary to understand any intellectual property restrictions and how they can be managed. Is the level of design and manufacturing (as built) information sufficient to facilitate benchmarking?

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Next Steps

- Task 1 - Agreement of the content of a Bolted Cask Data Collection questionnaire (Goal 3rd meeting)
- Task 1 – Collection of Bolted Cask data (Goal presentation and review at 4th meeting)
- Task 2.1 – Start of task 2.1 (Focus 5th meeting)
- Task 2.2 – Start of task 2.2 (Focus 6th meeting)

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Questions/Comments?

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CRIEPI's activities on Interim Dry Storage

Central Research Institute of Electric Power Industry

Masumi WATARU

Oct 30, 2025

CRIEPI
Central Research Institute of Electric Power Industry

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Contents

- 1. Current Situation of Spent Fuel Management in Japan
- 2.Studies on the Interim Dry Storage

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Japanese Spent Fuel Management Policy

- 6th Strategic Energy Plan (2021)
  - Measures for spent nuclear fuel:
    - promotion of construction/utilization of interim storage facilities and dry storage facilities, etc. to increase storage capacity;
    - and technology development for reducing the volume and harmfulness of radioactive waste

(Ref: https://www.enecho.meti.go.jp/en/category/others/basic_plan/pdf/6th_outline.pdf)

- 7th Strategic Energy Plan (2025)
  - Promotion of the nuclear fuel cycle
    - We will strengthen its effort for facilitating construction and utilization of new intermediate storage facilities and dry storage facilities while studying a wide range of locations as possible sites, regardless of whether they are inside or outside the premises of a power plant.
    - The Government will also stand on the front lines and respond proactively to these spent fuel issues and will make best efforts to ensure the understanding of all parties involved.
    - The Government will promote the stable and long-term use of spent fuel stored in intermediate storage facilities and other sites under the policy that spent fuel stored in such facilities will be transported to the Rokkasho Reprocessing Plant, ensuring the safety of the plant, which is necessary for this purpose.

(Ref: https://www.enecho.meti.go.jp/category/others/basic_plan/pdf/2025_strategic_energy_plan.pdf)

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Operational Status

As of Sep/2025

Contents

- 1. Current Situation of Spent Fuel Management in Japan
- 2.Studies on the Interim Dry Storage

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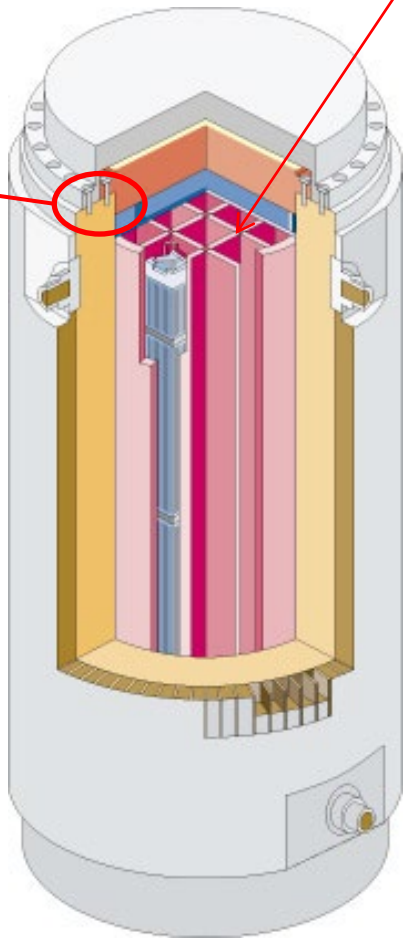
2. Studies on the Interim Dry Storage in CRIEPI

③ Ageing effect of aluminum alloy for the basket

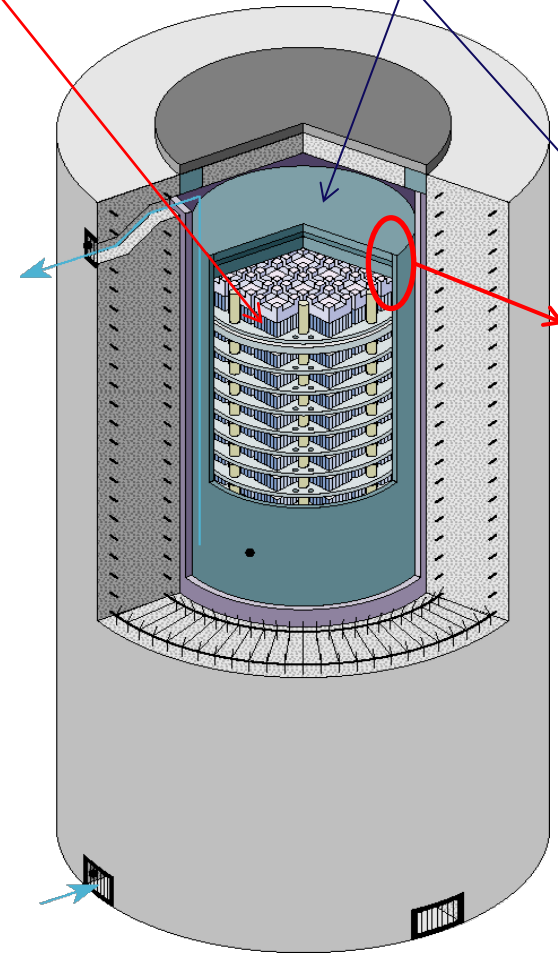
Canister

Metal Gasket

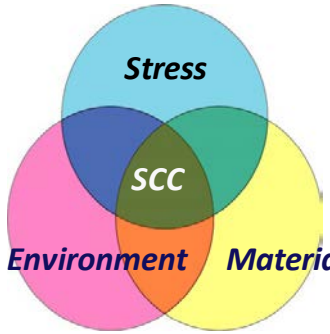
① Seal Performance of Metal Gasket



Metal Cask

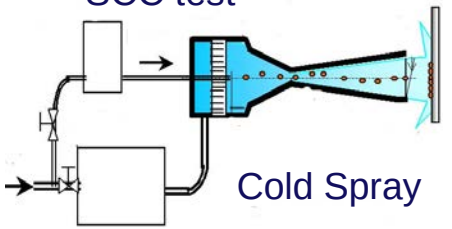


Concrete Cask



Three Factors of Stress Corrosion Cracking (SCC)

SCC test



Cold Spray

② Evaluation, Countermeasure and Repair for SCC of the Canister

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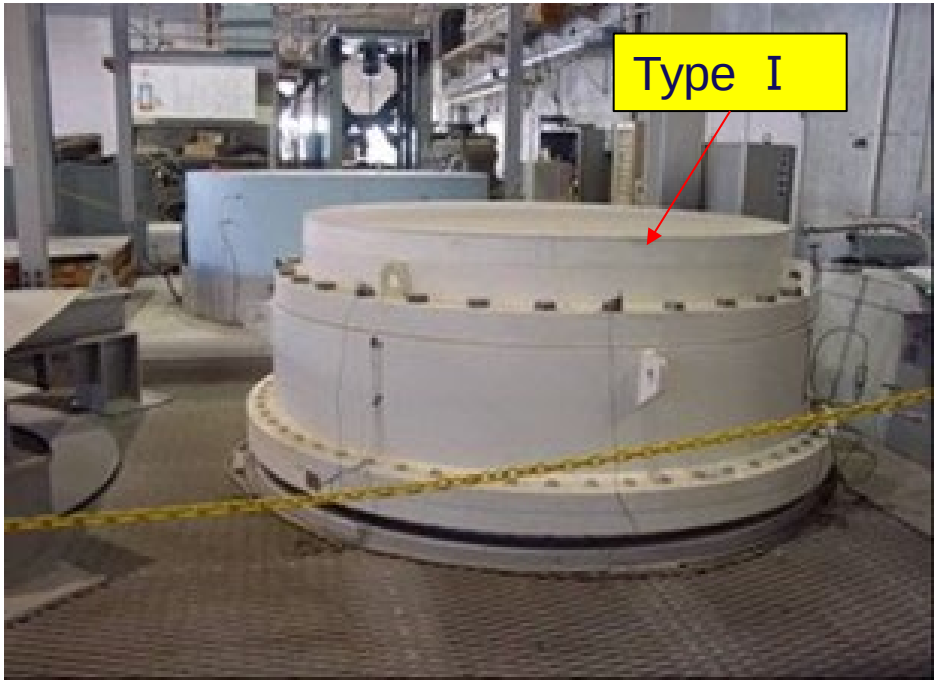
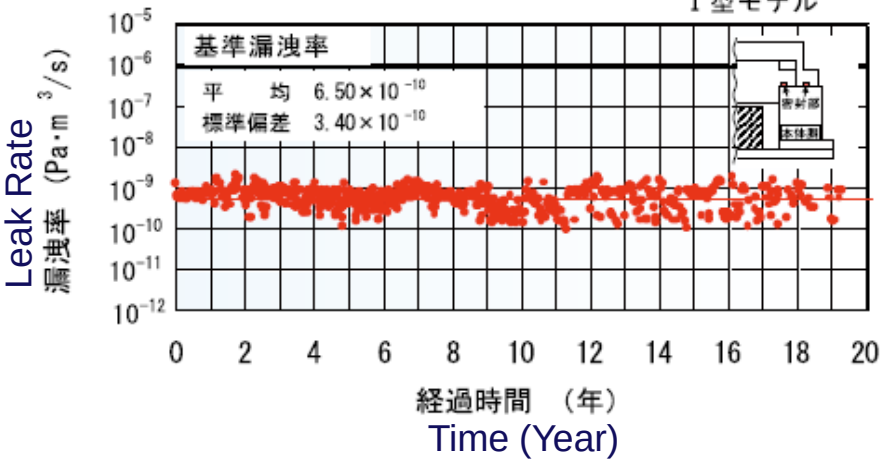
Seal Performance Test using a Full-scale Mock-up of Metal Cask



Type Ⅰ

Ⅰ型モデル



Test results using metal gaskets made by French Company

Long term seal performance test using the same mock-up with metal gaskets made by Japanese Company

Ref: https://criepi.denken.or.jp/koho/topics/201211vol14.pdf

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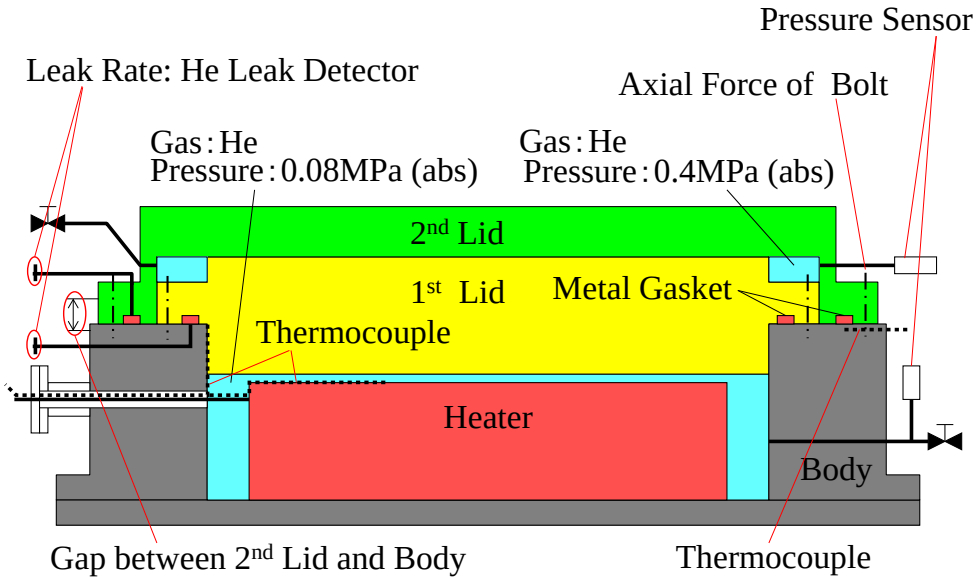
Re-machining of the cask lid model

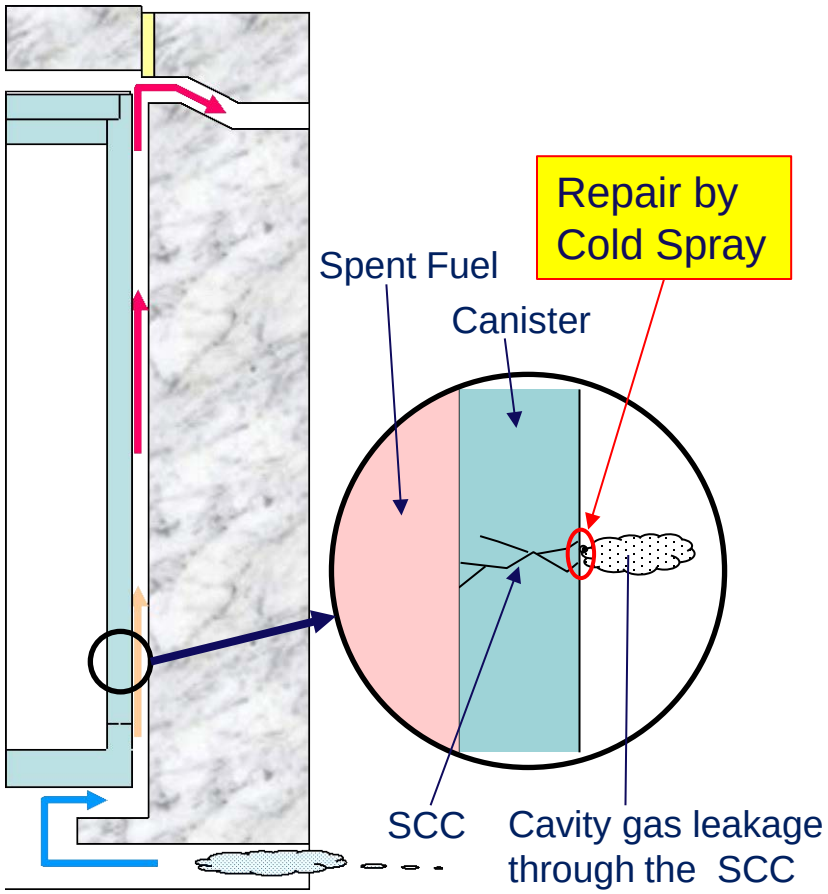
Cask Lid model after re-machining



Measurement Items

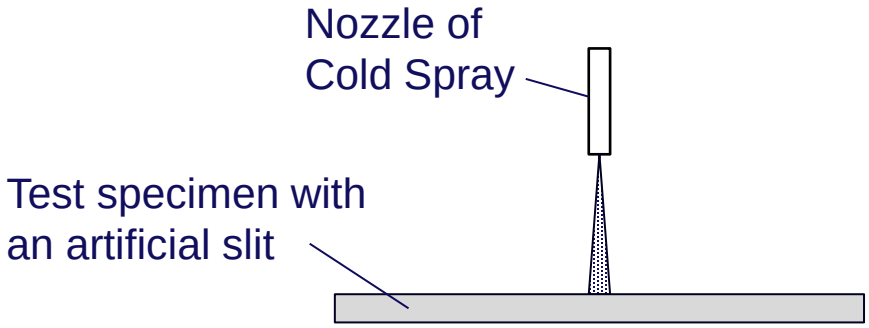
- The leak rates of secondary lid gasket have been maintained at an order of 10^{-10} Pa･m^3･s for 600 days.
- The test will continue for at least the next three years.

Repair for a Canister



Test Facility and Equipment of Plasma Giken

http://plasma.co.jp/en/



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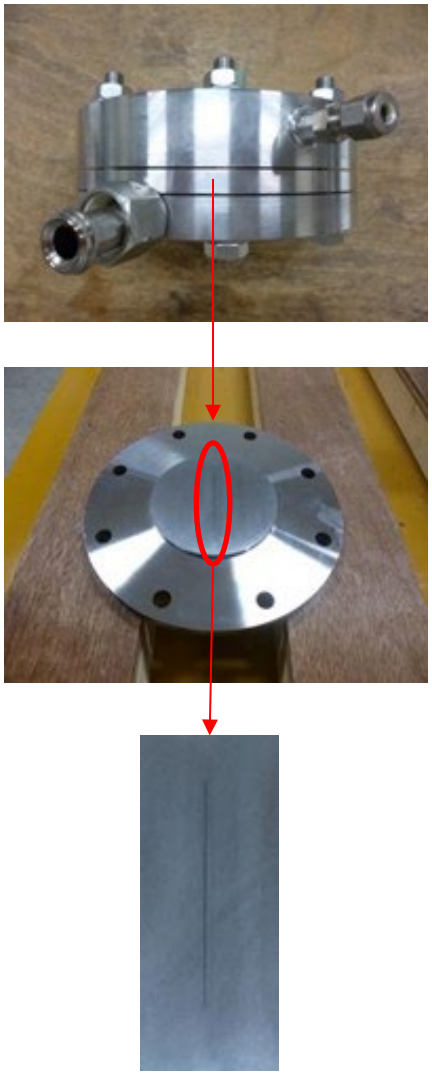
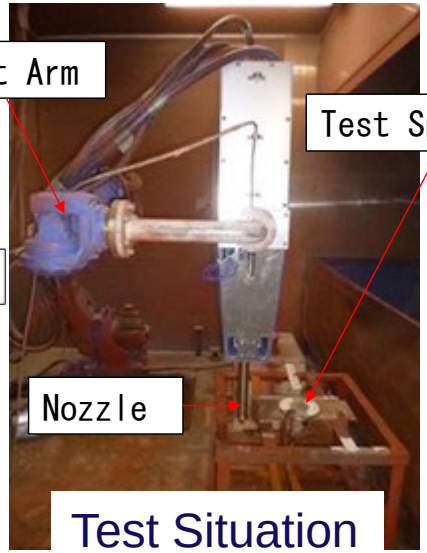
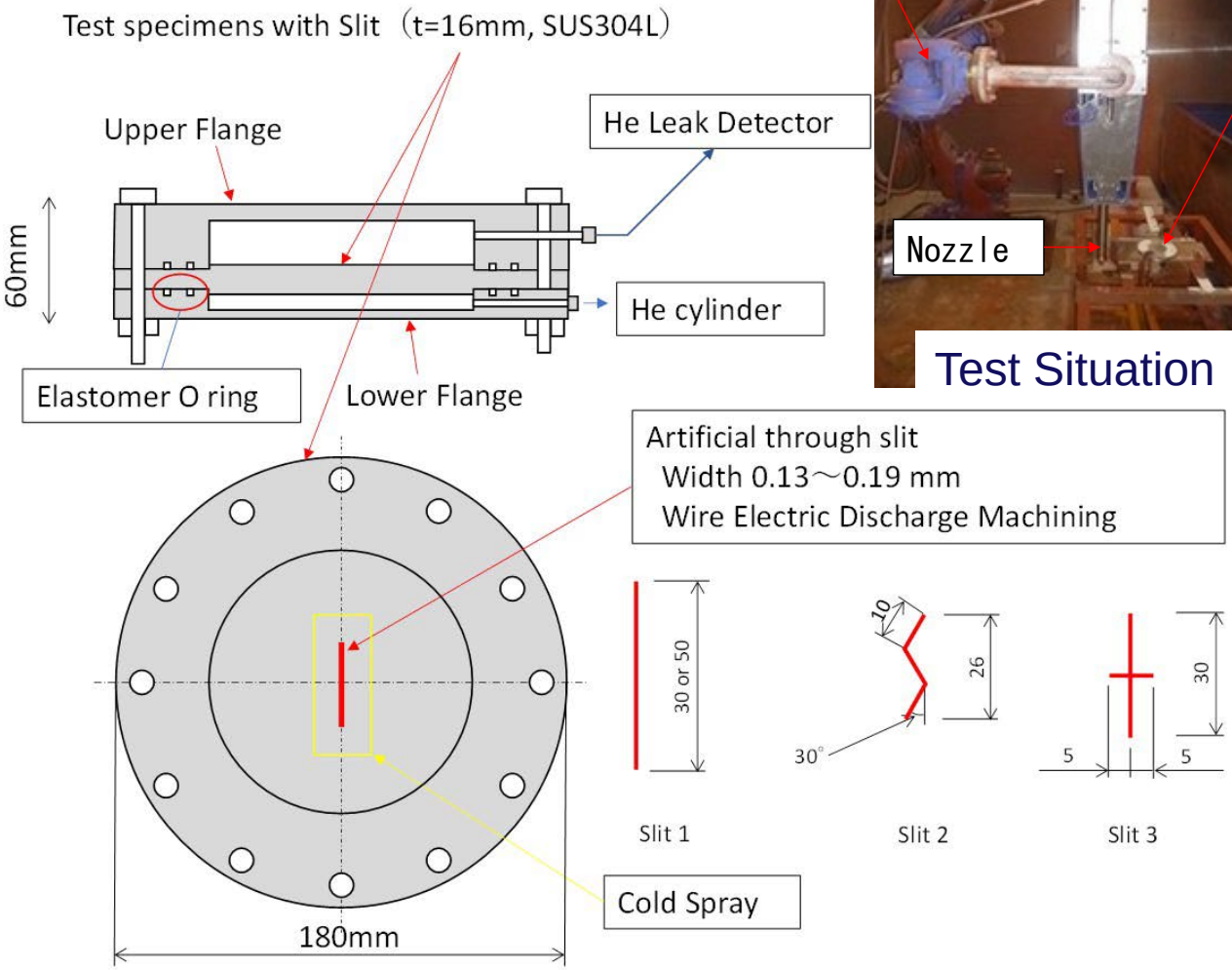
Cold spray tests



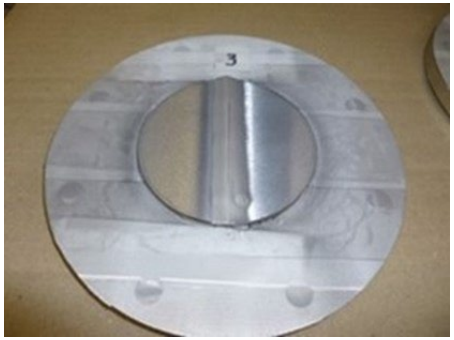




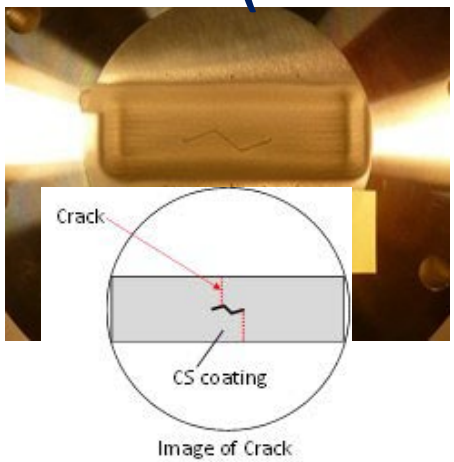
Test Specimens

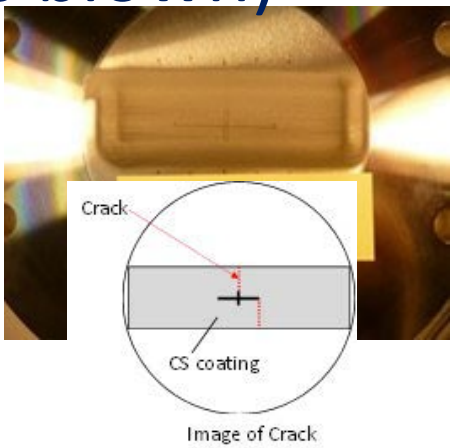
Ref: M.Wataru, A STUDY ON REPAIR USING COLD SPRAY OF THROUGH CRACKS OF CANISTERS CAUSED BY STRESS CORROSION CRACKING, Proceedings of the 21st PATRAM2025

Test Results (without N_2 gas blown)

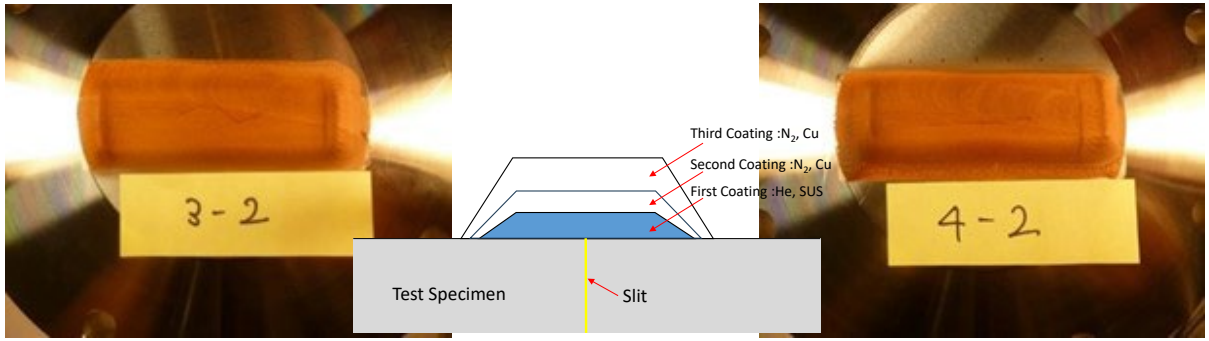


(a) Slit 1





Test Specimen after CS using He and Stainless Steel Particles (First coating)



Test Specimen after CS using N_2 and Cu Particles (Third coating)

(b) Slit 2

(c) Slit 3

(Slit 1)

Test No.	9	10
Leak Rate ($Pa \cdot m^3/s$)	$1.5*10^{-11}$	$1.9*10^{-11}$
Test No.	11	12
Leak Rate ($Pa \cdot m^3/s$)	$4.6*10^{-11}$	$2.0*10^{-5}$

Test No.	14(Slit 2)	15(Slit 2)	17(Slit 3)
Leak Rate ($Pa \cdot m^3/s$)	$3.2*10^{-5}$	$1.3*10^{-5}$	$5.2*10^{-5}$

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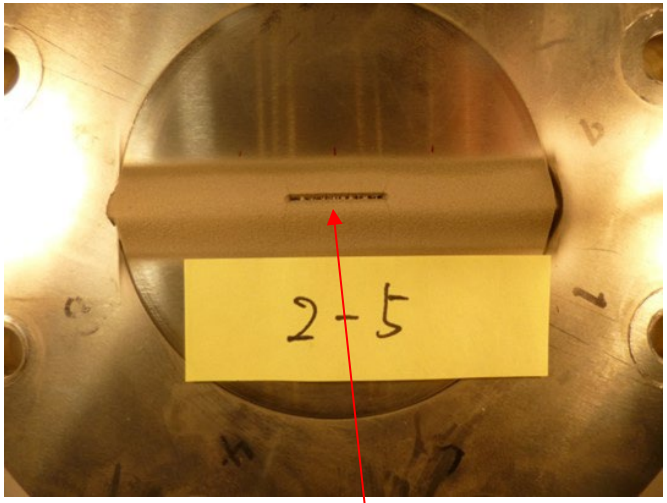
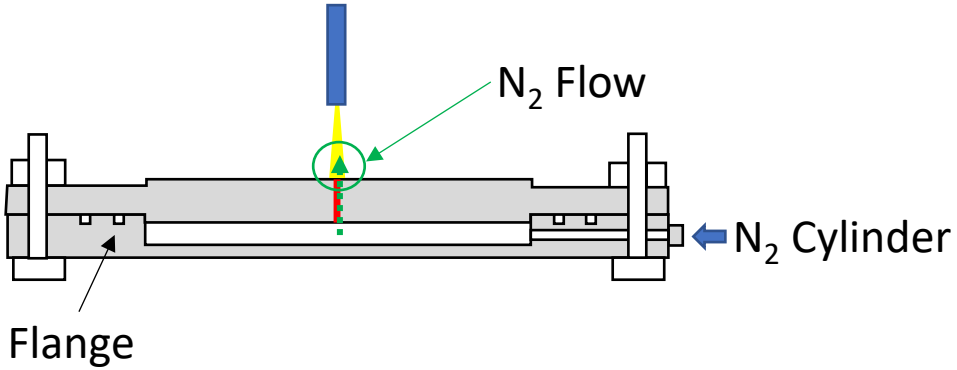
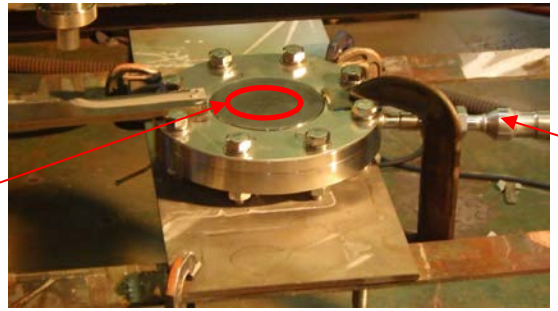
Ref: M.Wataru, A STUDY ON REPAIR USING COLD SPRAY OF THROUGH CRACKS OF CANISTERS CAUSED BY STRESS CORROSION CRACKING, Proceedings of the 21st PATRAM2025

Test Results (with N_2 gas blown)





Slit



Piping connected to N_2 Cylinder

CS test setup

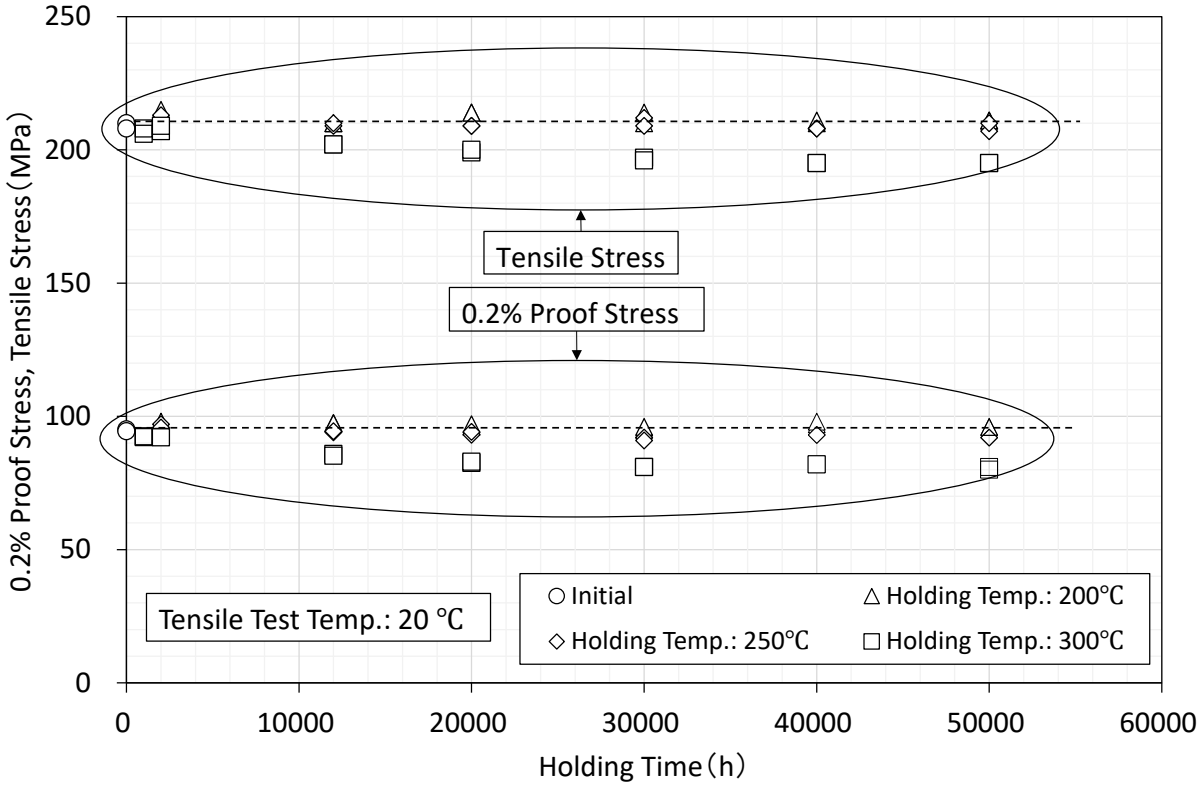
CS applied while gas was being blown through the slit did not plug the through slit.

Ref: M.Wataru, A STUDY ON REPAIR USING COLD SPRAY OF THROUGH CRACKS OF CANISTERS CAUSED BY STRESS CORROSION CRACKING, Proceedings of the 21st PATRAM2025

Ageing effect of Aluminum Alloy for the basket

Aluminum Alloy Plate
(Aluminum Alloy based on JIS A3004)
(Length:725 mm, Width: 250 mm
Thickness: 15 mm)

Temperature Chamber



Tensile Test Results

Summary

- In Japan, projects of dry storage facility are increasing gradually.
- CRIEPI has finished the following items.
  - Canister repair by cold spray
- CRIEPI is studying the following items.
  - Long term confinement of a metal cask using full-scale mock-up with Japanese metal gasket
  - Aging of mechanical properties of Aluminum alloy for a basket

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TopFuel 2026:
Nuclear Reactor Fuel Performance Conference

October 12-16, 2026
Tokyo, Japan
Shinagawa Intercity Hall & Conference

TopFuel 2026 TOKYO

I. Operation and Experience
II. Advances in Designs, Materials, and Manufacturing
III. Evolutionary and Innovative Advanced Technology Fuels (ATF)
IV. Modeling, Analysis, and Methods
V. Transient Fuel Behavior and Safety Related Issues
VI. Used Fuel Storage, Transportation, and Re-Use
VII. Advanced Reactor Fuel Designs and Performance

Important dates

Deadline for abstract submission: January 16, 2026.
Author notification for acceptance: February 27, 2026.
Deadline for draft paper submission: April 24, 2026.
Review of papers and author notification: June 19, 2026.
Deadline for final paper: July 17, 2026.

https://web.apollon.nta.co.jp/topfuel2026/index.html

U.S. DEPARTMENT *of* ENERGY

Office of Environmental Management

Hanford Field Office

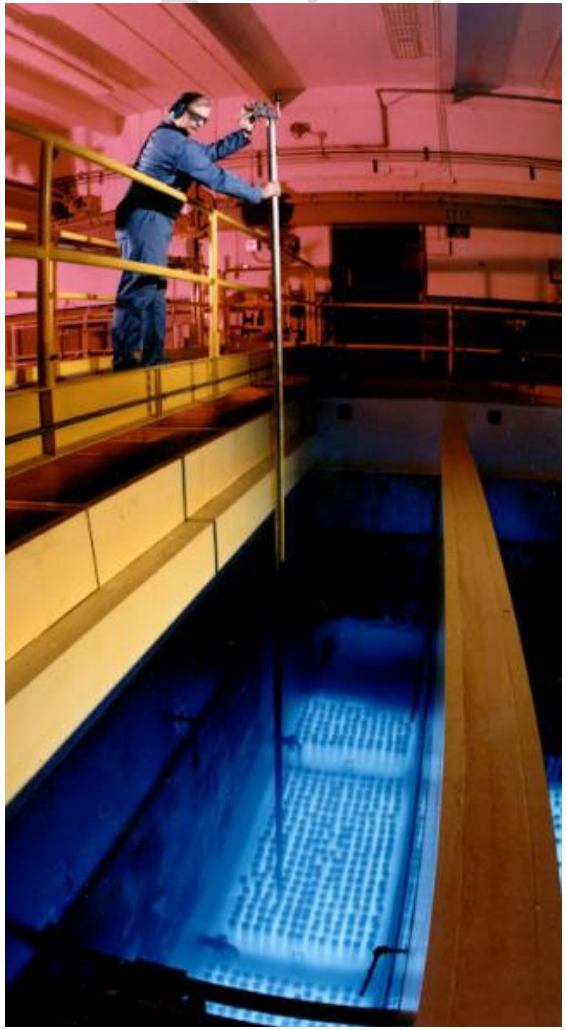
CPCCo
Central Plateau
Cleanup Company

Hanford Lead Canister and Aging Management Activities Update

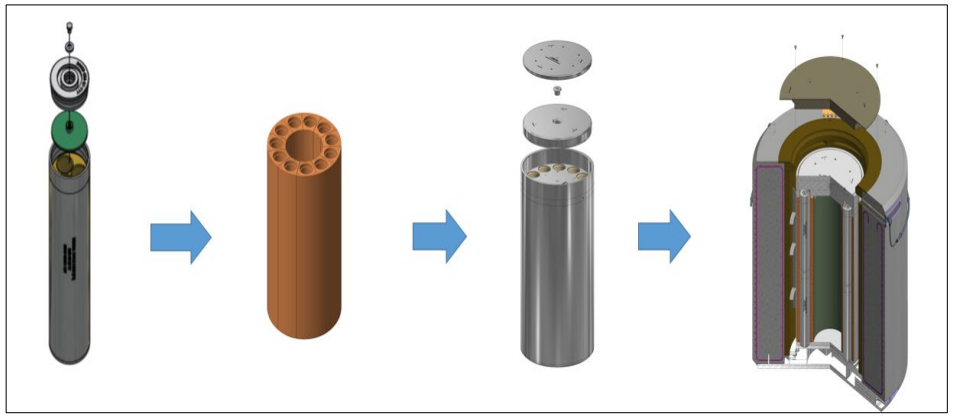
Marie T. Gillespie – *Central Plateau Cleanup Company (Fluor Affiliate)*

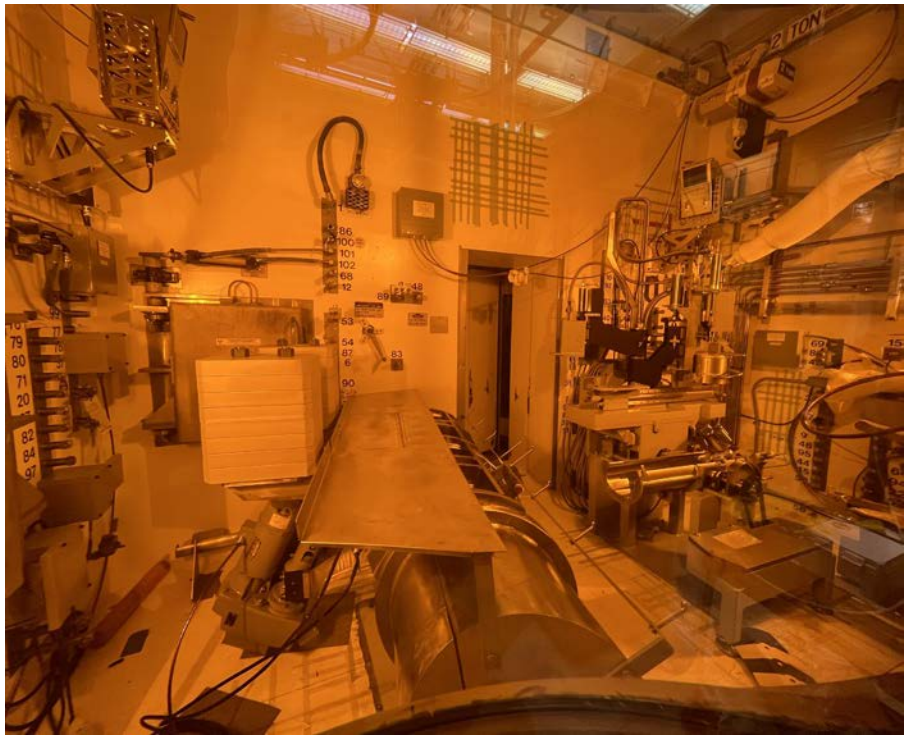
Gary R. Cannell – *Fluor Nuclear Power*

Hanford Cesium Strontium Capsule Dry Storage



- Unique design and fabrication of radioactive materials storage containers
  - Hanford Cs-Sr capsules will transfer from wet pool to dry storage
  - Capsules will be packaged into dry-storage canisters based on NAC MPC design
  - Storage term: 300 years



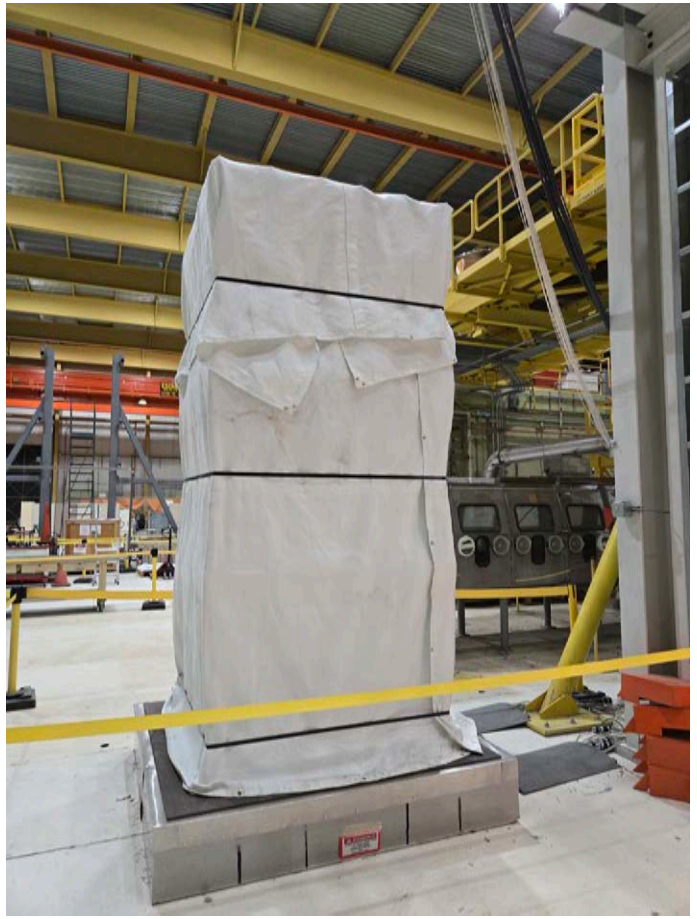


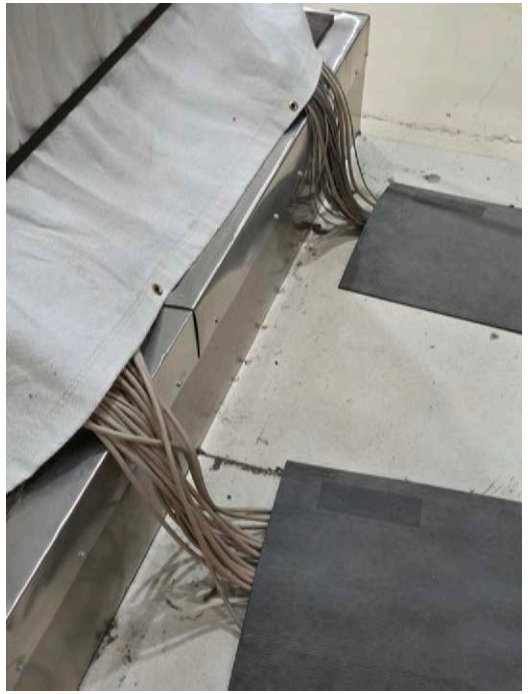
Hanford Lead Canister

- First article, full quality DCSS
  - Supports project integrated testing, training, and operations (2024-2030)
  - Performs leading indicator function upon conclusion of capsule transfer with no dose consequences (2030)
  - Include internal heating to represent worst case (coolest) cask
- Secondary functions – mockup for:
  - Mitigation & Repair (M&R) technology development / demonstration
  - In Service Inspection and M&R practice and procedure development
  - Long- and short-term field service data collection
- HLC Value
  - Support aging management of storage containers subjected to extended storage
  - Evaluate materials and fabrication practices for next generation of canisters
  - DOE asset available to DCSS community
  - Evaluate monitoring technologies and strategies

Hanford Lead Canister Activities

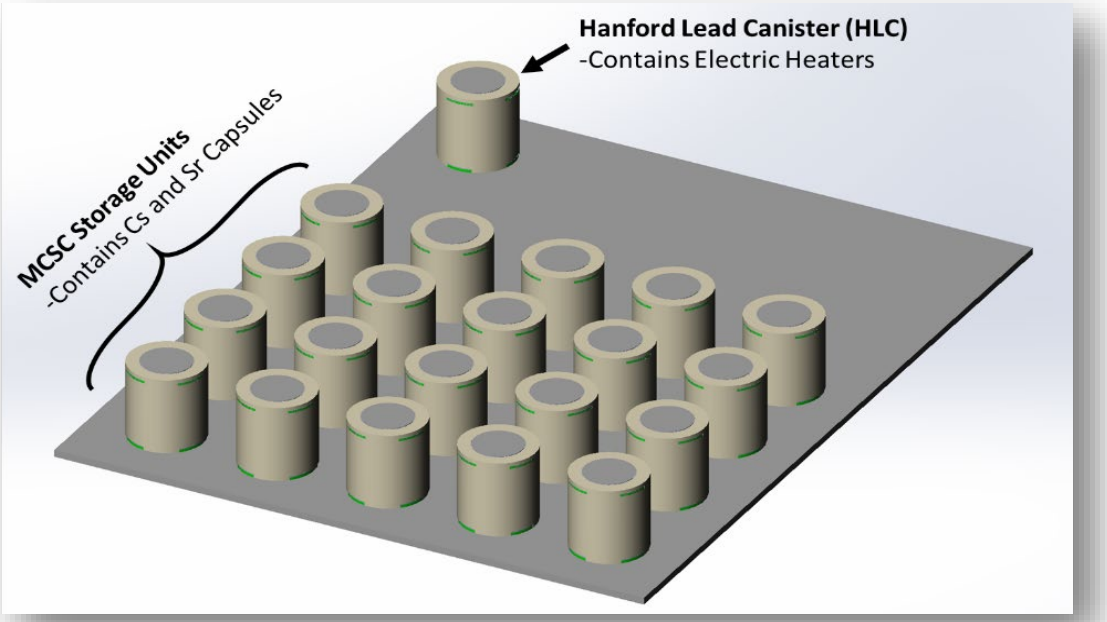
- HLC internal heater fabrication and testing
  - Pacific Northwest National Lab (PNNL) designed thermal controls and strategy
  - Bench testing complete
  - Design for routing of heater wires out of CSS in progress.





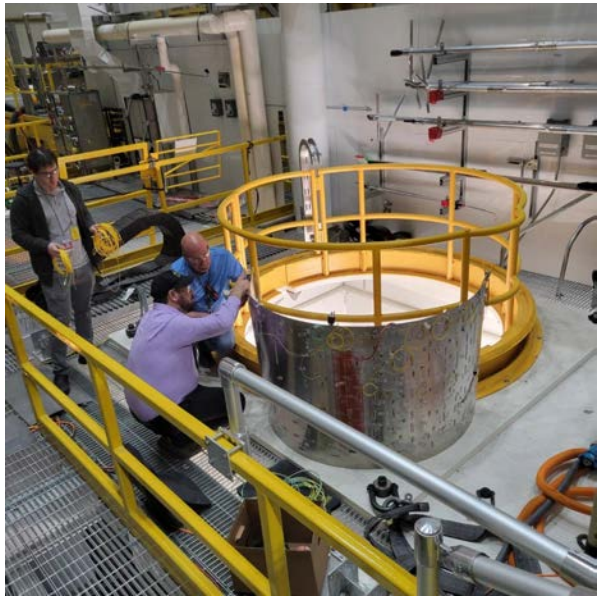
Hanford Lead Canister Activities

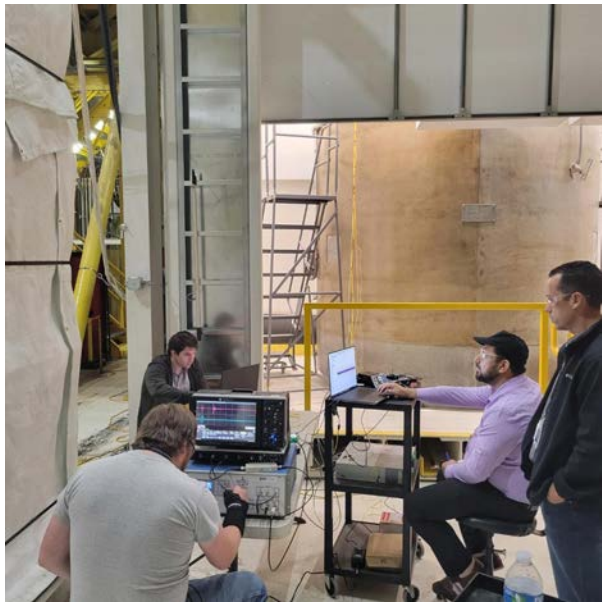
- Mitigation and Repair panels and strategy
  - EPRI temporary panels for short term data collection to evaluate tech and delivery in situ
  - Long-Term Panels
    - Evaluate materials, fabrication practices and monitoring technologies
  - Available to the DCSS community for R&D and demonstrations



Hanford Lead Canister Activities

- University of Pittsburgh (Nuclear Energy University Program)
- Fiber optic sensors measure temperature, static and dynamic strain, and acoustic crack detection

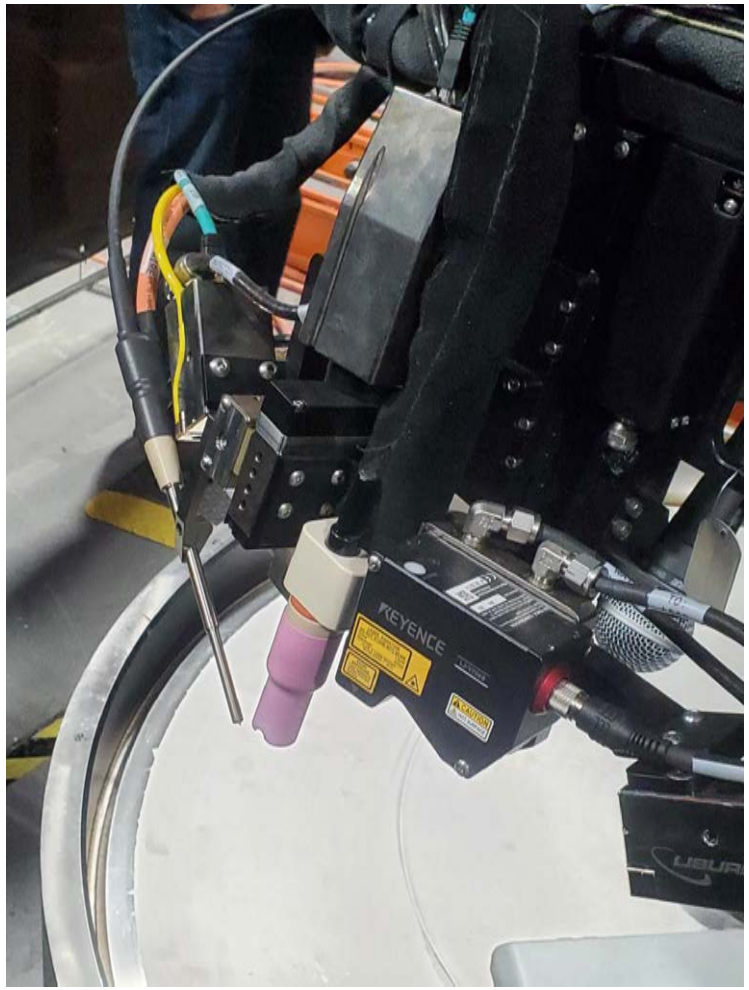




Aging Management Activities

- Department of Energy coordination
  - DOE Hanford Field Office
  - Nuclear Energy-8 Spent Fuel and High-Level Waste Disposition
  - Technology roadmap in development for canister aging and HLC role
  - SNF Center for Applied Research (High Burnup Cask) – HLC testbed?

Energy Facilities Contractors Group (EFCOG) Container Aging Management and Closure Welding



- **Subgroup Operations:**
  - Quarterly Teams calls
  - Annual face-to-face meeting, rotated between the Sites
- **Membership:**
  - Responsibilities for container management
  - Technical background or experience with containers and closure welding
  - Representation from each of the Sites with containers and National Labs that support radioactive material management.
  - Current members: **Hanford, ANL, INL, NAC Intl, ORNL, PNNL, SRNL**

Questions?

BGZ Gesellschaft für Zwischenlagerung mbH

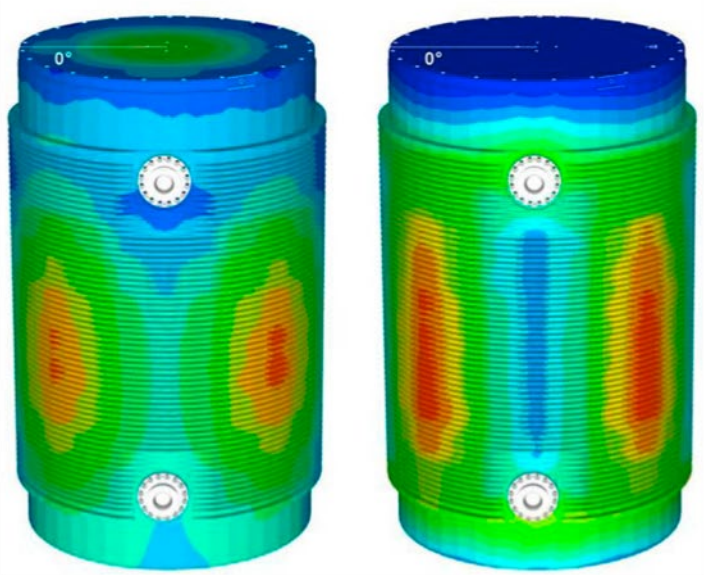
Towards high-resolution Muography

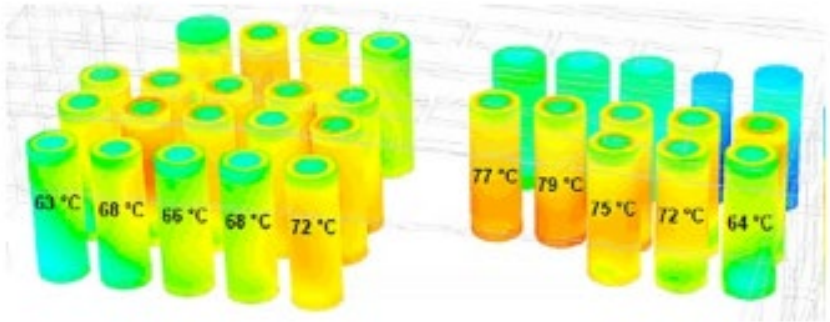
Maik Stuke, Julia Niedermeier

EPRI-ESCP, Charlotte, NC

29th Oct 2025

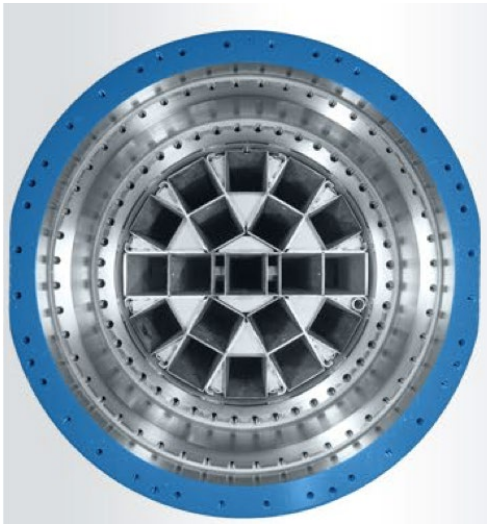
How to get information on the LWR fuel w/o opening a cask?



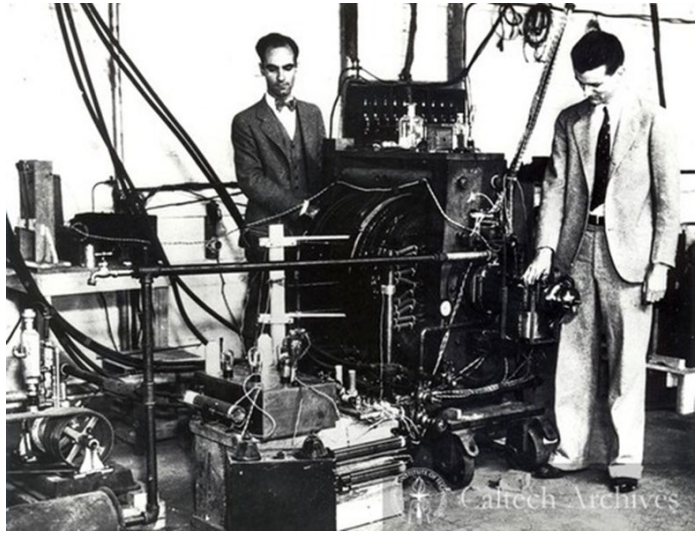


Collect and interpret information from inside via radiation
- n, gamma, temperature
- Difficult or impossible to get information on inner fuel assemblies

Radiation passing through the cask….?

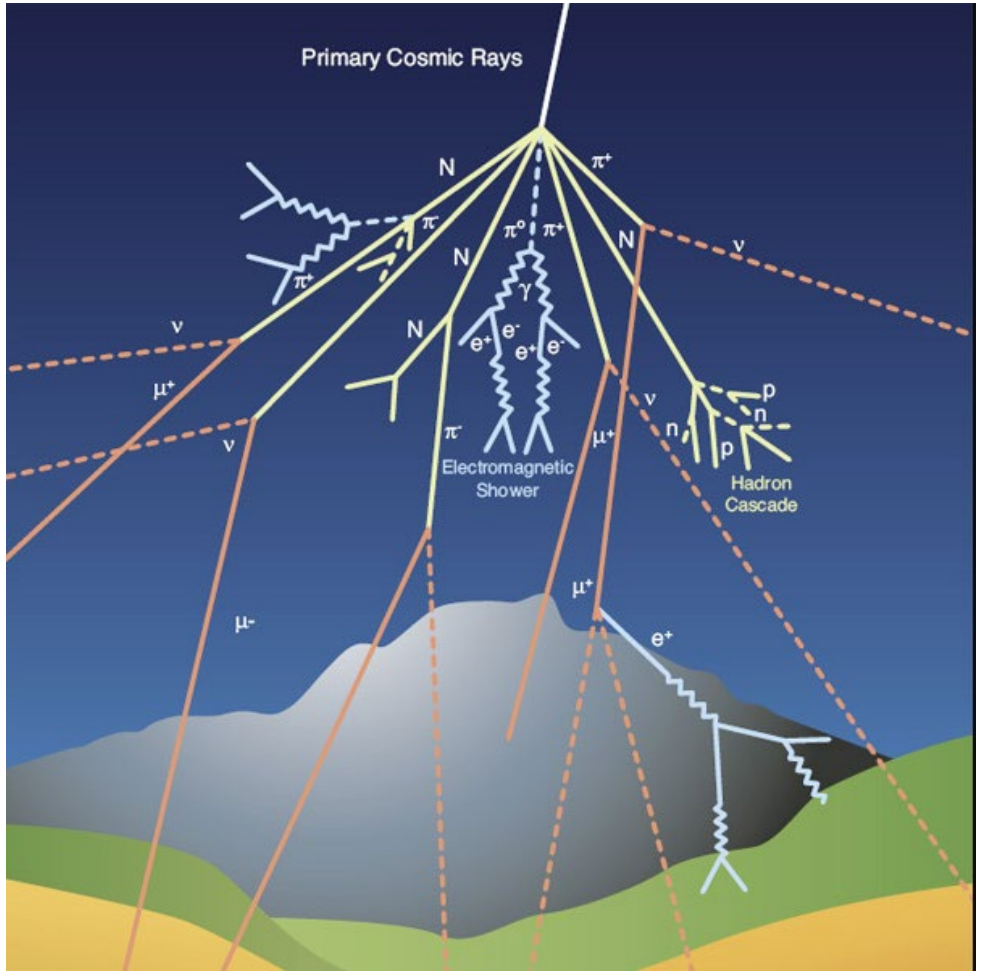


Muons and Muography



Discovered by Carl D. Anderson and Seth Neddermeyer in 1936
They detected muons while studying cosmic radiation with a **magnet cloud chamber**: one particle bent **less than an electron** but **more than a proton**

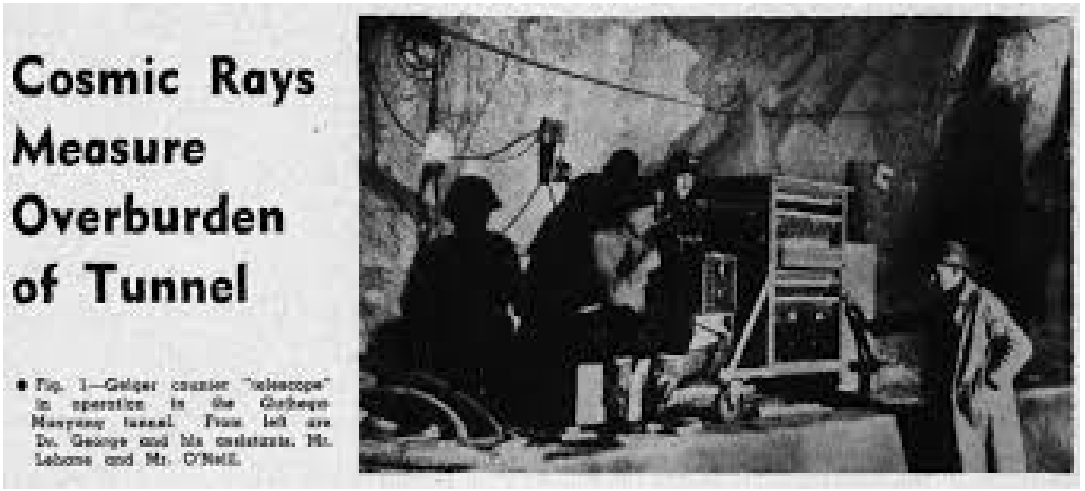
Cosmic muons are produced as the end-result of nuclear interactions between **cosmic rays** and **atoms in the upper atmosphere (app. 15 km)**



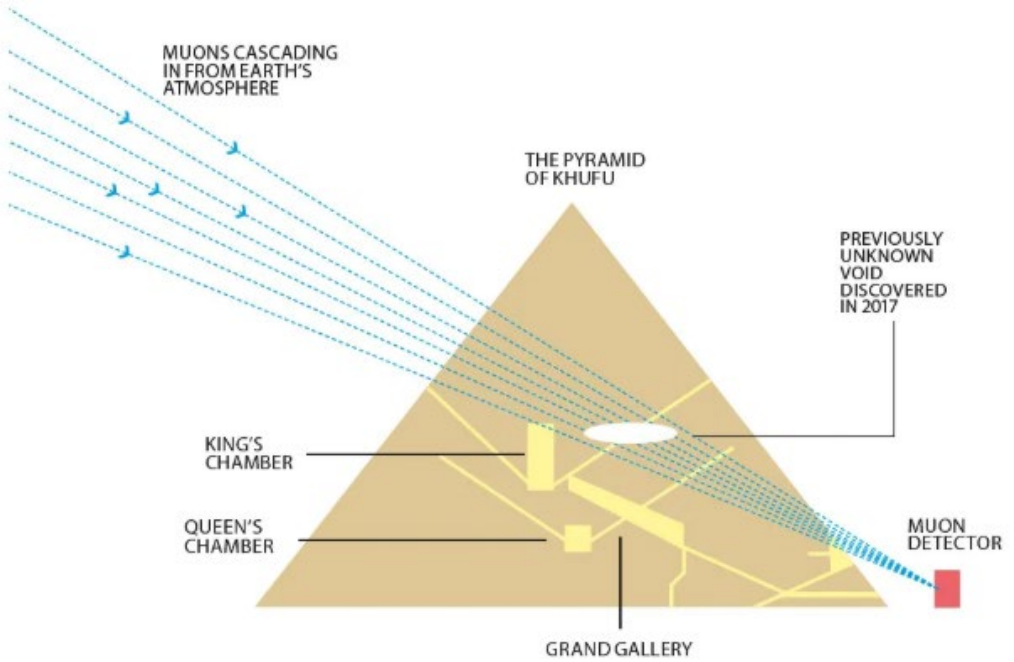
Muons and Muography Absorption

Muons loose energy during interaction and get absorbed (decay). Density of material muons pass through has impact on flux

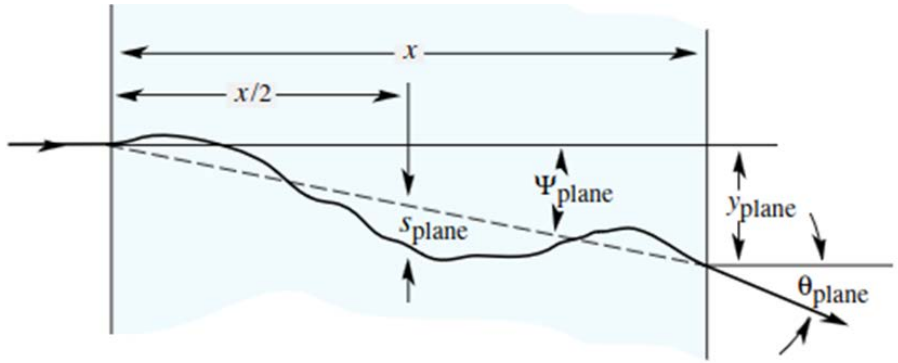
- Eric George was the first one to apply cosmic muons to measure the thickness above a tunnel in 1955
- 14 years later, Luis Alvarez tried to search for hidden chambers in the Pyramid of Khafre in Egypt


Cosmic Rays Measure Overburden of Tunnel

George EP. 1955 Cosmic rays measure overburden of tunnel. Commonwealth Engineer 43, 455–457



Muons and Muography
Scattering Angle Analysis

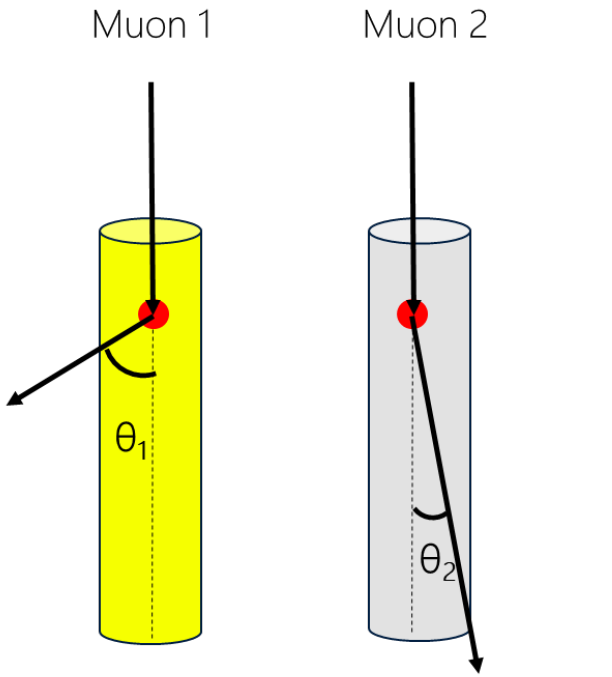


Scattering behaviour of muons in matter:

Two important factors:
- density and
- atomic number Z

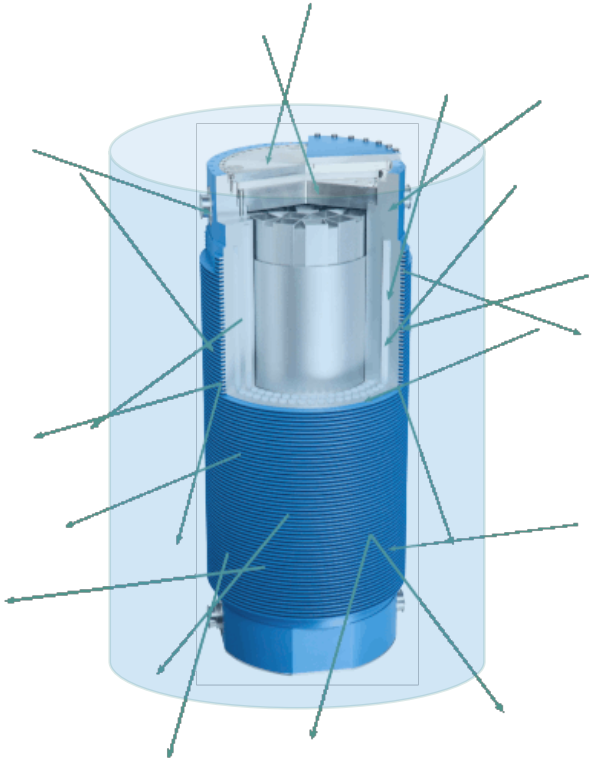
The angular distribution of scattering follows a gaussian distribution

The root mean square (RMS) value of the scattering angle depends on the radiation length.



$\theta_1 >> \theta_2$

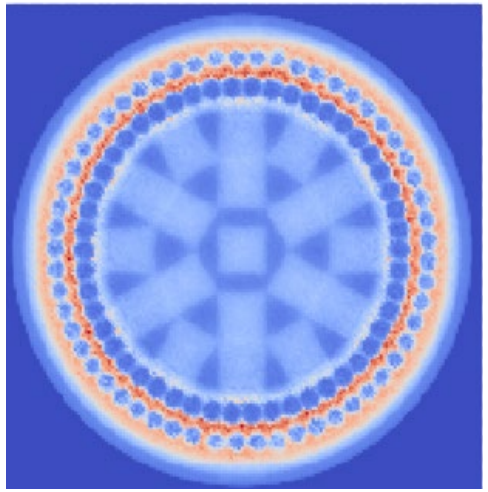
Muons and Muography
Scattering Angle Analysis



Propterties of incoming muons

Properties of outgoing muon

Calculation of the scattering angle

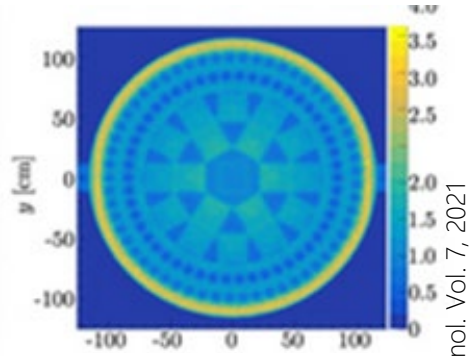


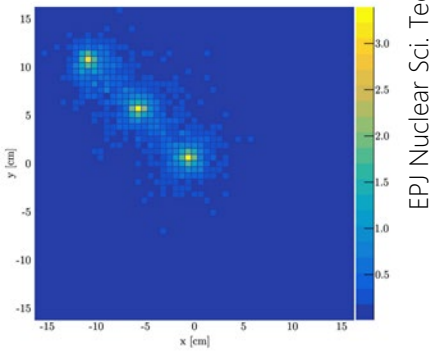
Exemplary heat map;

Why would I do this? For academic pupose?

Muon Tomography

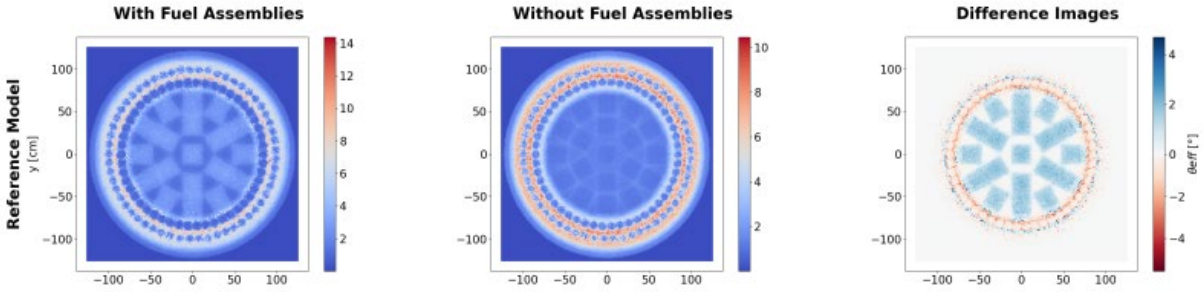
Subject Matters:	• Development of competencies in the experimental implementation and theoretical evaluation of non-destructive inventory monitoring methods • Objective to examine a cask using muon radiography
Project Partners:	Various partners: TU Munich, INFN (IT), Universities Brescia and Padova, Research Center Jülich and DLR
Project Period:	2021 - 2030

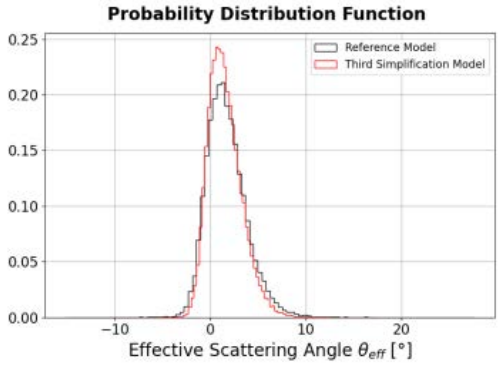


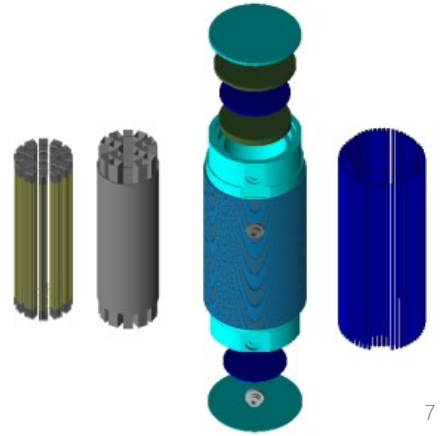


EPJ Nuclear Sci. Technol. Vol. 7, 2021

- How are measurement time and resolution related? Is a single rod resolution feasible?
- Goal: Experimental campaign in 3-5 years







MuTomCa
A field test @ BGZ

Muography for re-verification of DPC's

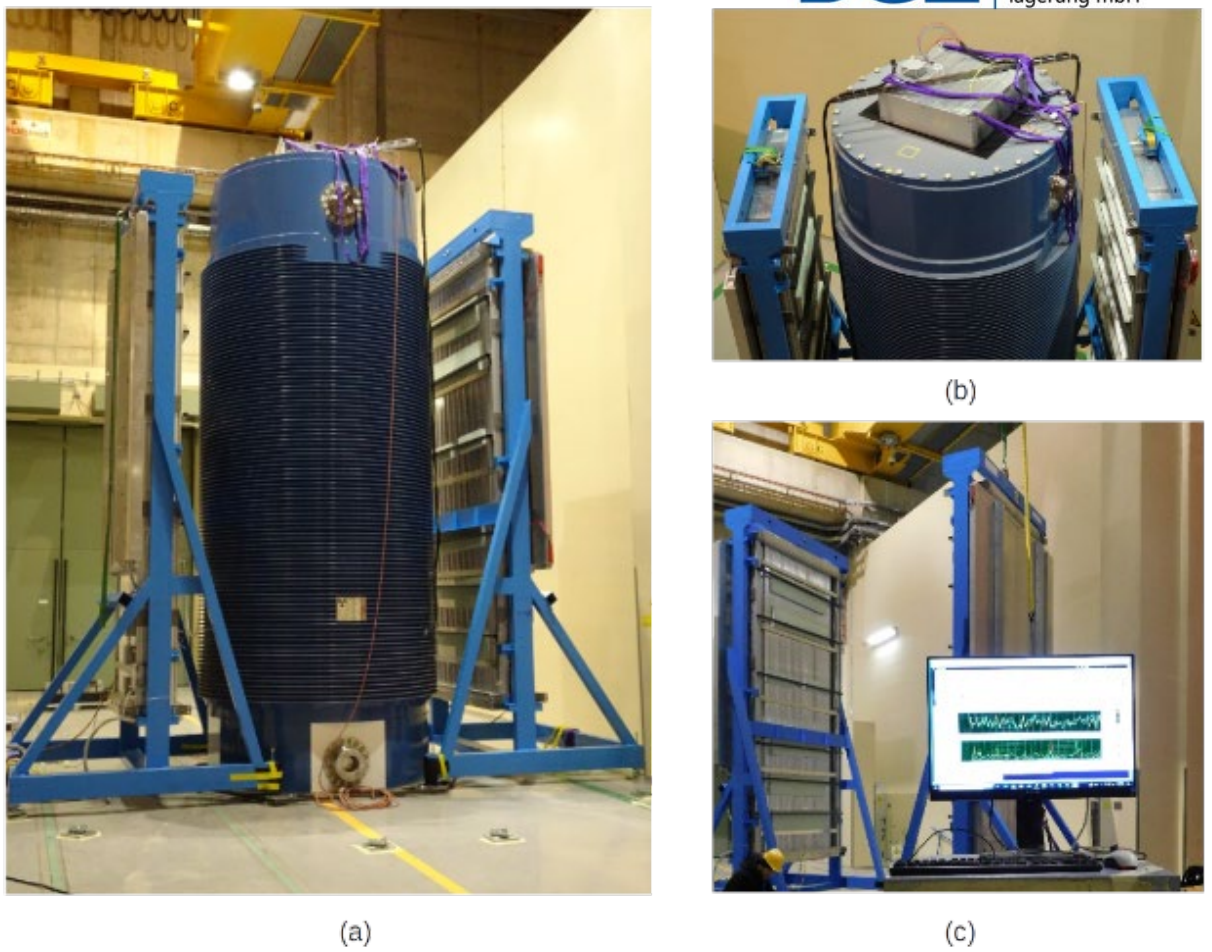
Differentiation between dummy elements and fuel assemblies

Where? Interim storage Grafenrheinfeld

Planning phase: 2020 – 2022
Experimental phase: 16.01.2023 bis 03.03.2023
Analysing phase: ongoing

Project partners:

- Forschungszentrum Jülich GmbH, Jülich
- Gesellschaft für Zwischenlagerung mbH (BGZ), Essen
- National Institute for Nuclear Physics (INFN), Padova Division, Italy
- Euratom, DG ENER-Safeguards, Luxemburg



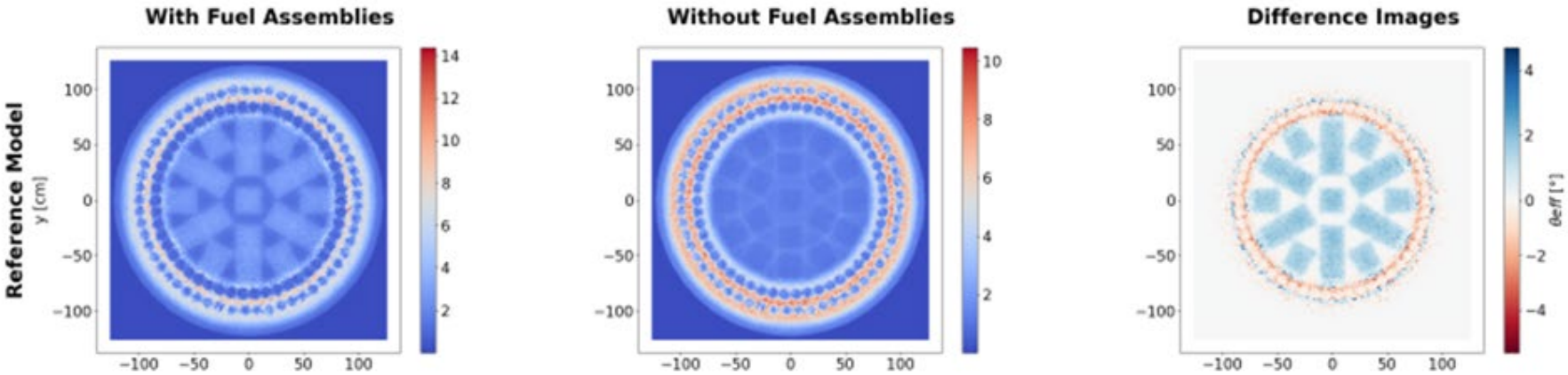
(a) Whole setup. (b) Top view of additional drift tube chamber (c) Exemplary measured muon flux without a cask.

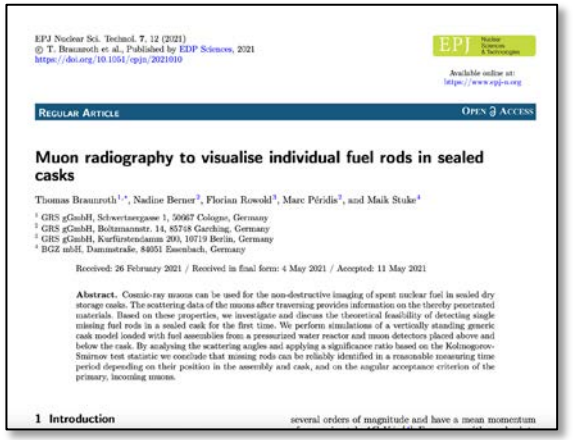
Further details: P. Andreetto et al, J. Appl. Phys. 138, 124903 (2025)

Lorenzon et al. The MUTOMCA Project
Niedermeier et al. MUTOMCA – An overview

BGZ Gesellschaft für Zwischenlagerung mbH

Can we get to resolutions at order of individual fuel rods?





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https://doi.org/10.1051/epjn/2021010

Regular Article — Open Access

Muon radiography to visualise individual fuel rods in sealed casks

Thomas Braunroth[1,*], Nadine Berner[2], Florian Rowold[3], Marc Péridis[2], and Maik Stuke[4]

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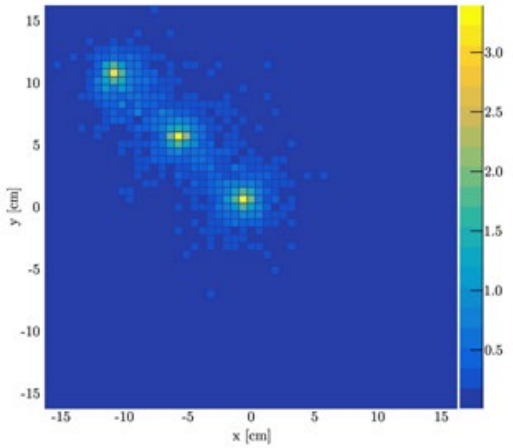
Received: 26 February 2021 / Received in final form: 4 May 2021 / Accepted: 11 May 2021

Abstract. Cosmic-ray muons can be used for the non-destructive imaging of spent nuclear fuel in sealed dry storage casks. The scattering data of the muons after traversing provides information on the thereby penetrated materials. Based on these properties, we investigate and discuss the theoretical feasibility of detecting single missing fuel rods in a sealed cask for the first time. We perform simulations of a vertically standing generic cask model loaded with fuel assemblies from a pressurized water reactor and muon detectors placed above and below the cask. By analysing the scattering angles and applying a significance ratio based on the Kolmogorov-Smirnov test statistic we conclude that missing rods can be reliably identified in a reasonable measuring time period depending on their position in the assembly and cask, and on the angular acceptance criterion of the primary, incoming muons.

1 Introduction

several orders of magnitude and have a mean momentum



Theoretically possible, but experimentally?
→ More detailed modelling before we march in our ISF….

Analysis of the Scattering Behavior: Simulation Setup

Detector Configuration:

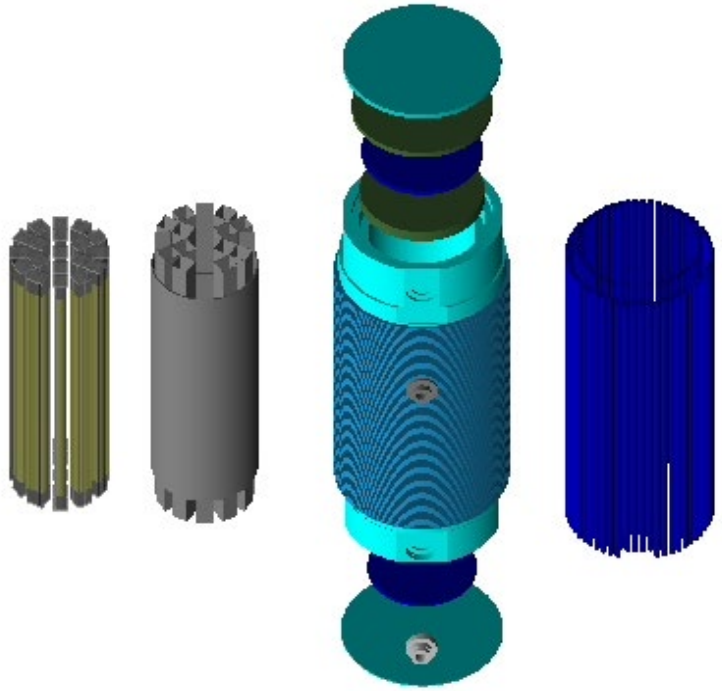
- **Ideal Detectors:** 100% resolution
- **Dimensions:** 2600 mm
- **Setup:**
  - **Incoming Detector:**
    measures incoming properties
  - **Outgoing Detectors:**
    measure outgoing properties

Simulation Setup:

- **Generator:** EcoMug
- **Muons Generated:** 50 and 25 million
- **Generation Area:** 2500 mm × 2500 mm
- **Incident Angle:** 0° – 25°
- **Incident Energy:** no limitations
- **Absorption Rate:** ~ 60 %

Analysis of the Scattering Behavior: Modelling

Geant4 Model

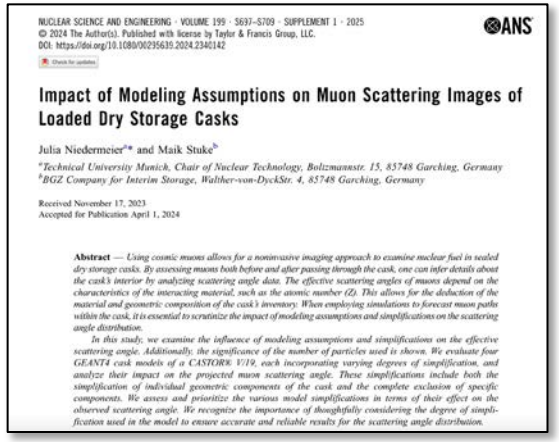


19 fuel assemblies, fuel basket, monolithic body with integrated lids, four trunnions, protection plates and polyethylene absorber plates + neutron moderator rods

CASTOR V/19

© GNS

Influence of model simplifications on scattering behavior



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DOI: https://doi.org/10.1080/00295639.2024.2340142

ANS

Impact of Modeling Assumptions on Muon Scattering Images of Loaded Dry Storage Casks

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[b]*BGZ Company for Interim Storage, Walther-von-Dyck5tr. 4, 85748 Garching, Germany*

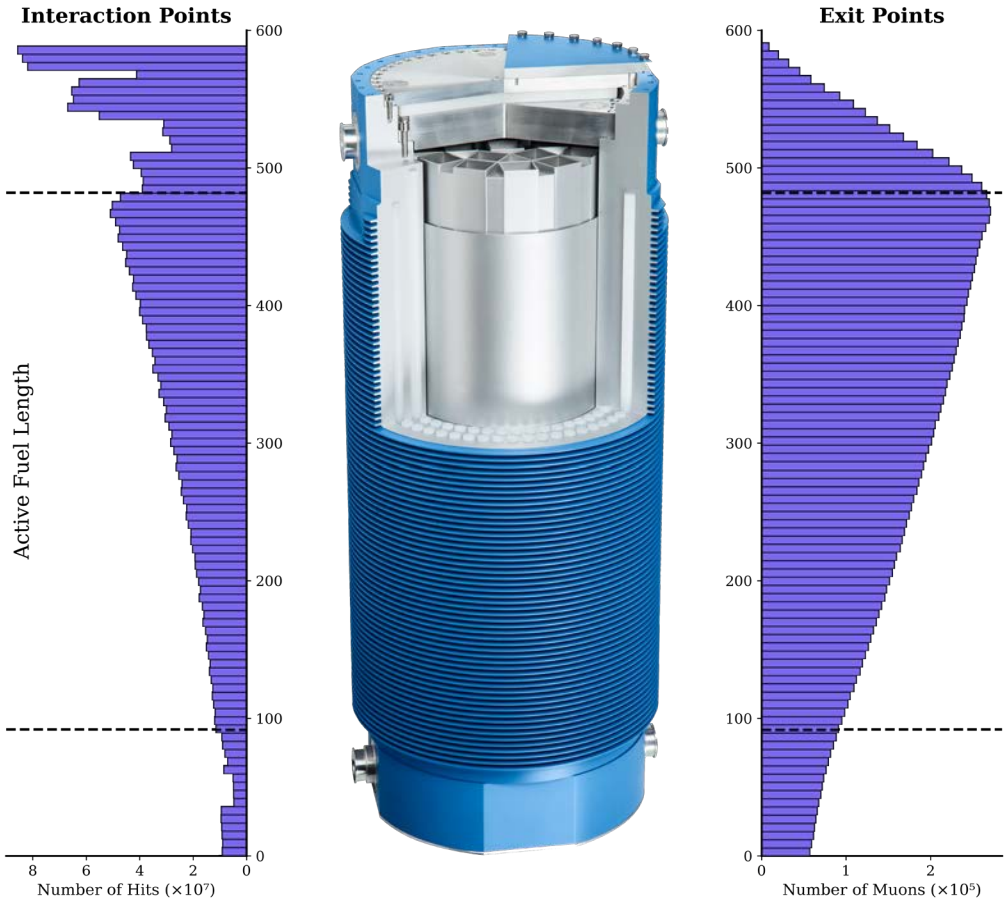
Received November 17, 2023
Accepted for Publication April 1, 2024

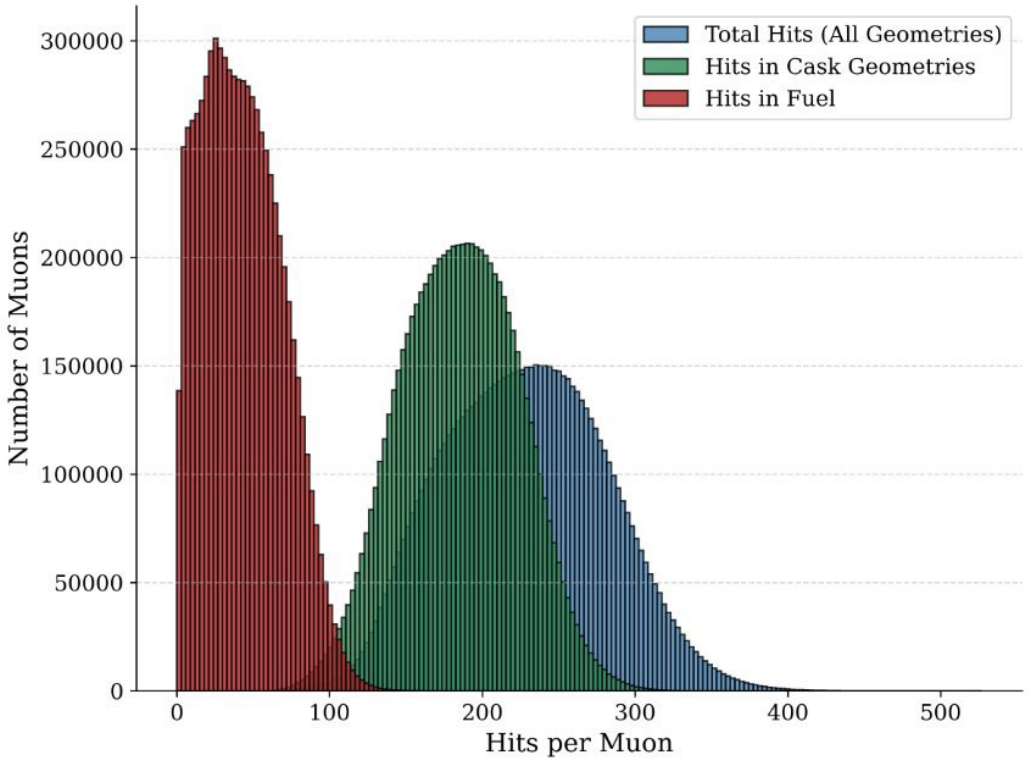
Abstract — *Using cosmic muons allows for a noninvasive imaging approach to examine nuclear fuel in sealed dry storage casks. By assessing muons both before and after passing through the cask, one can infer details about the cask's interior by analyzing scattering angle data. The effective scattering angles of muons depend on the characteristics of the interacting material, such as the atomic number (Z). This allows for the deduction of the material and geometric composition of the cask's inventory. When employing simulations to forecast muon paths within the cask, it is essential to scrutinize the impact of modeling assumptions and simplifications on the scattering angle distribution.*

In this study, we examine the influence of modeling assumptions and simplifications on the effective scattering angle. Additionally, the significance of the number of particles used is shown. We evaluate four GEANT4 cask models of a CASTOR® V/19, each incorporating varying degrees of simplification, and analyze their impact on the projected muon scattering angle. These simplifications include both the simplification of individual geometric components of the cask and the complete exclusion of specific components. We assess and prioritize the various model simplifications in terms of their effect on the observed scattering angle. We recognize the importance of thoughtfully considering the degree of simplification used in the model to ensure accurate and reliable results for the scattering angle distribution.

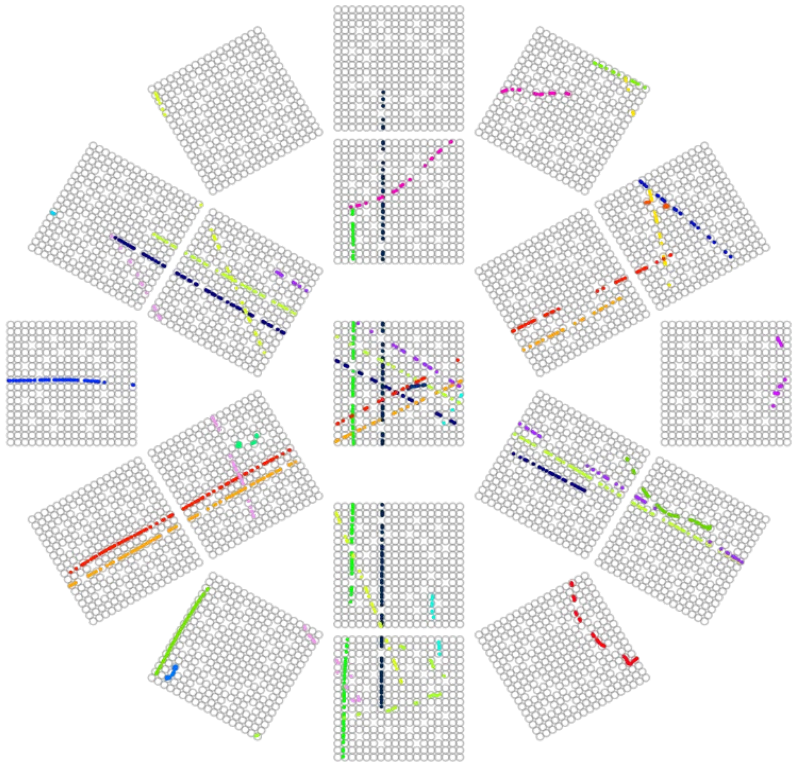
Short answer: There is an influence of model simplifications,
but the impact is weakening for FA/FR further away from the simplification

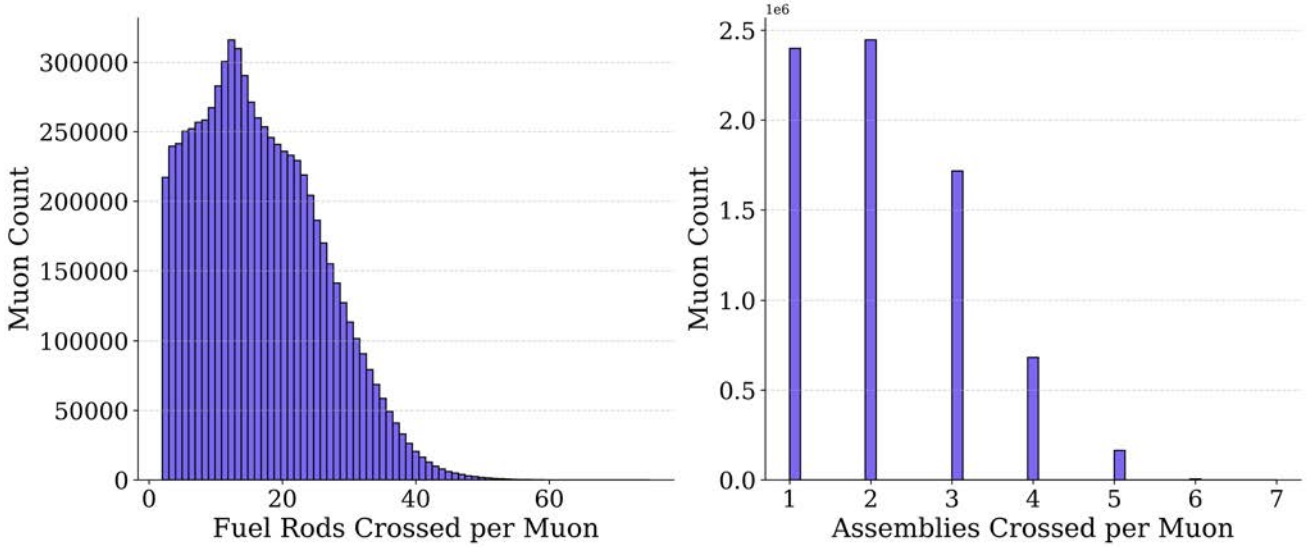
Analysis of the Scattering Behavior: Results (detailed model)





Analysis of the Scattering Behavior: Results



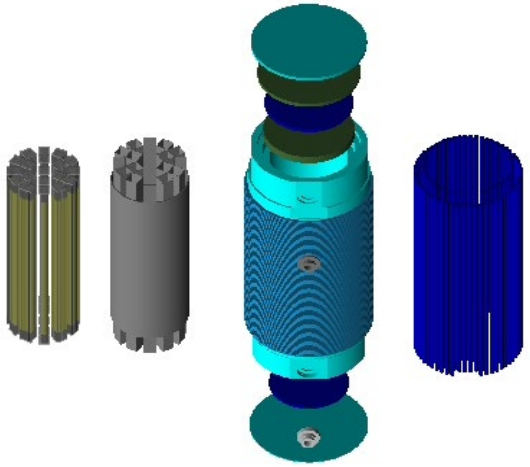


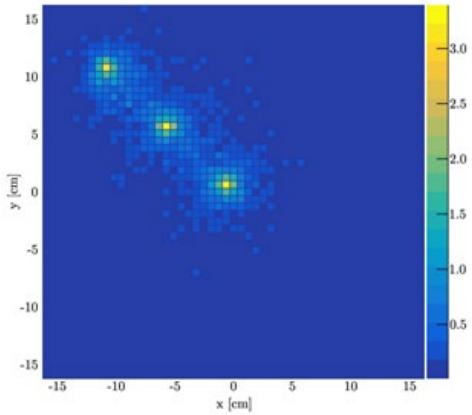
Detailed analysis of muon exit points and number of interactions with fuel helps **designing experimental setup** and **determining suitable reconstruction algorithms**

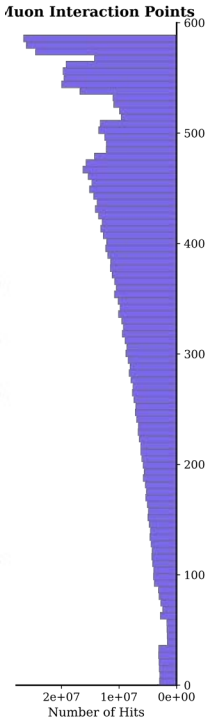
Summary

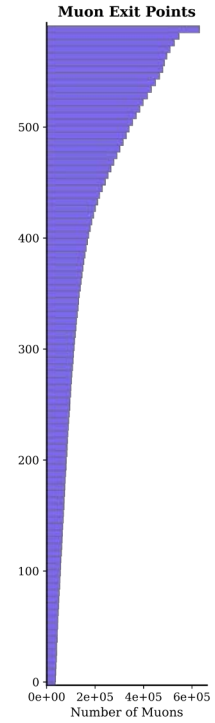
- Muography is a good complement to neutron, gamma, and temperature analysis of loaded DPC's
- Variety of results on both, experimental and simulation indicate good resolution

- Next step: Finding reconstruction algorithms and designing experimental setup









BGZ Gesellschaft für Zwischenlagerung mbH

Thank you for your attention

Updates on GOTHIC and Cask Modeling

YJ Choi, GOTHIC PM
Principal Technical Leader

October 30, 2025
Extended Storage Collaboration Program (ESCP)

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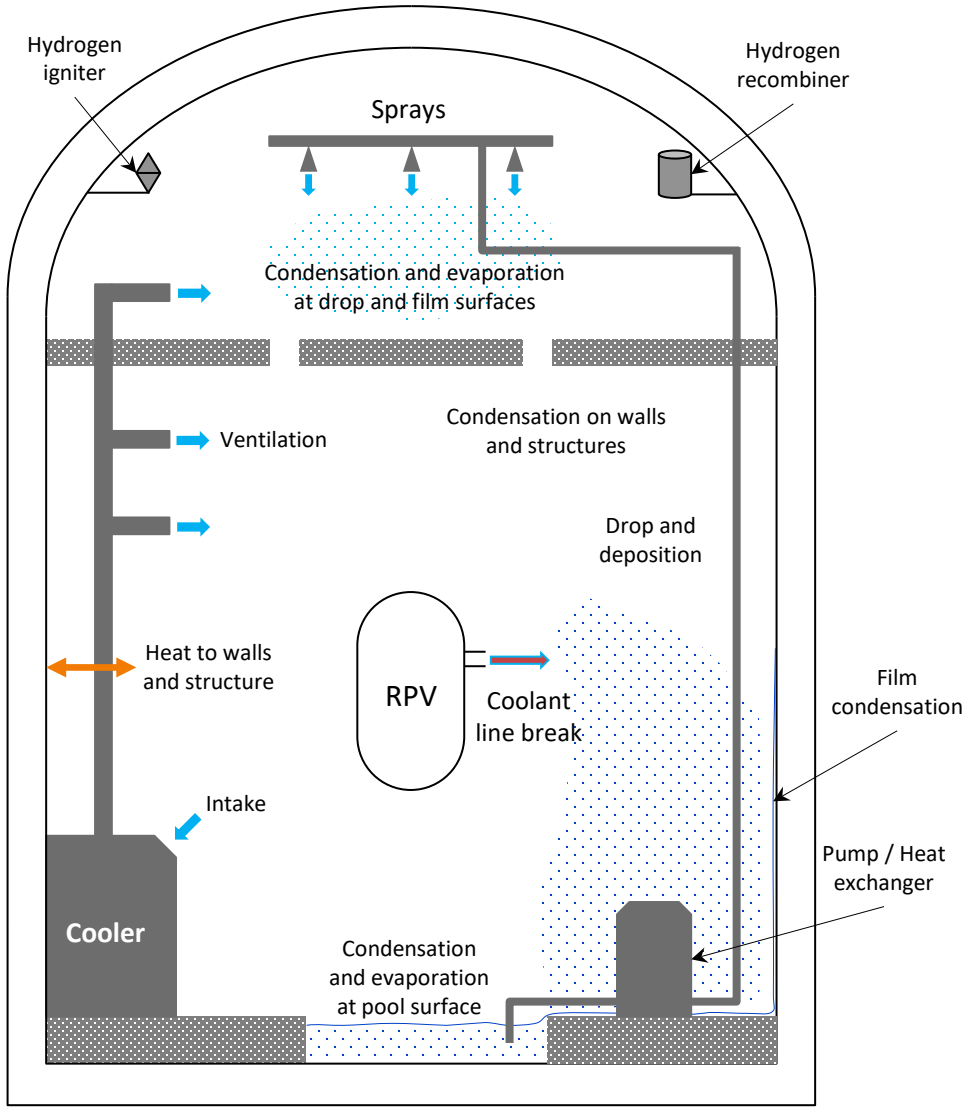
Overview of GOTHIC

- GOTHIC is fully validated best-estimate nuclear thermal hydraulics (TH) analysis software
  - Coarse-mesh multi-phase CFD and system level nuclear TH
  - Originated from COBRA-TF subchannel analysis code 40 years ago
  - Strong user community supported by EPRI since 1993
- GOTHIC analyzes NPP containment TH, in principle
  - Analyze relative physics in NPP buildings and components
  - Applicable to both LWR and non-LWR designs
  - Versatile general-purpose multi-physics engineering tool
- GOTHIC supports licensing of NPP world widely
  - Compliance with U.S. 10CFR21, 10CFR50 Appendix B, ASME NQA-1
  - Extensive benchmark, validation and qualifications
- **Latest version is GOTHIC v8.5(QA), released Oct. 2024**
  - **Including updated features related to DCS modeling**



Major phenomena of containment analysis

EPRI Controls Quality and License of GOTHIC

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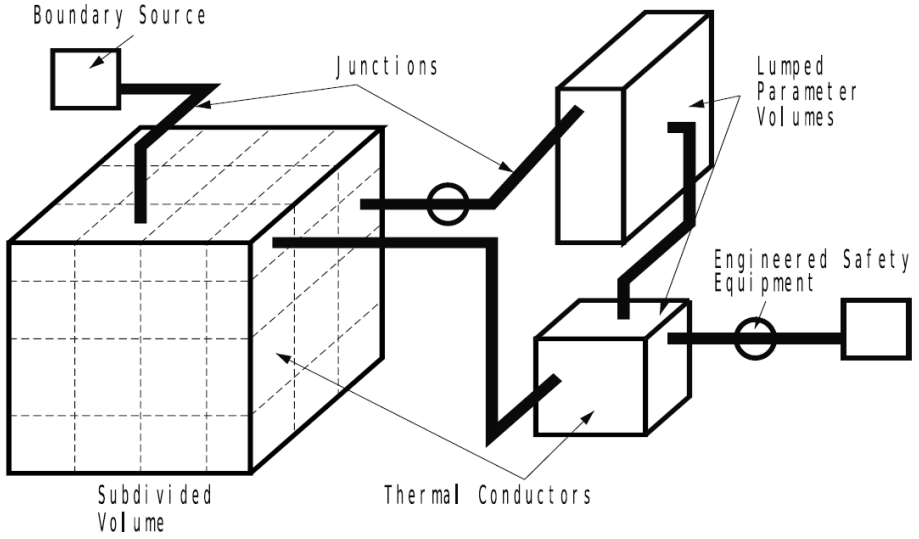
Fundamental of GOTHIC

- Multiphase and multicomponent
  - Underlying conservation equations
  - Interphase heat, mass, and momentum transfer
  - Thermal conductors and wall heat transfer
- CFD-like capabilities
  - Coarse-grid CFD (CGCFD)
    - Combined lumped and subdivided volumes
    - Super-imposed heat conductors
  - Semi-implicit finite volume discretization
    - Reynolds averaged Navier-Stokes equations
- Engineered equipment and components
  - Including control systems
  - Main and aux. buildings / Spent fuel storages
- User interface and post-processing
- Integrated training modules



Concept of coarse-grid analysis of GOTHIC

Feature	GOTHIC	CFD
Grid/Mesh	Rectangular with volume and area porosity	Body fitted
Molecular diffusion (Mass, energy, momentum)	Yes	Yes
Turbulent diffusion (Mass, energy, momentum)	Yes	Yes
Boundary layer wall effected	Correlations (heat transfer coefficient, friction factors)	Detailed
Phase change	Condensation, evaporation, boiling, gas dissolve	Limited
Equipment or component models	Yes	No

Feature comparison between GOTHIC and CFD

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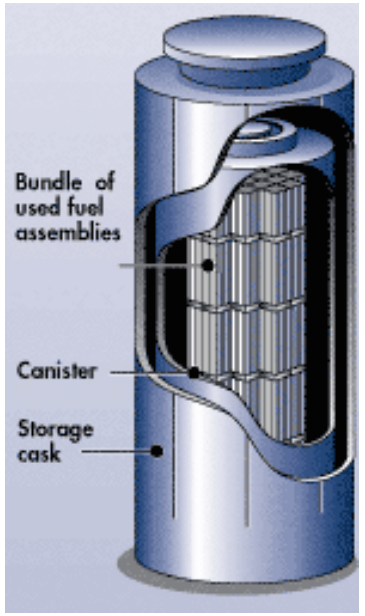
GOTHIC for Spent Fuel Cask Analysis

- GOTHIC has been used to model spent fuel casks to examine:
  - Internal circulation patterns
  - Internal pressure
  - Steady state & transient response
    - Fuel and cask walls
    - Concrete in region below cask
    - Surrounding environmental conditions
- Various evolutions and phases
  - Drain/Dry/Fill during initial loadings
  - Steady state during long term storage
  - Reflood scenarios
  - Time-to-boil
- Challenges in multi-scale problem
  - Internals of a single cask interaction with surrounding environments
  - Independent Spent Fuel Storage Installation (ISFSI)



https://www.nrc.gov/waste/spent-fuel-storage/diagram-typical-dry-cask-system.html

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GOTHIC's Relevant Capabilities

- Flexible nodalization with 1D and/or 3D models
  - 1D modeling of lumped fuel pins → Simplified modeling
  - 3D modeling of individual fuel pins → To view local phenomena
- Buoyancy driven natural convection heat transfer
  - Mixed convection, conduction and radiation heat transfer
- Flexible radiation heat transfer
  - Wall ⇔ Wall (Transparent media) / Wall ⇔ Steam
- Transient conduction in both the fluid and structures
  - Includes conduction within the fluid itself
  - Axial and radial power profiles of the fuel
  - Some capability for 2D conduction with structures
- Mechanistic aerosol modeling
  - Used extensively for drop behavior and spray cooling
    - Loss of coolant accident scenarios
  - Enhanced capabilities in dry aerosols modeling
    - Fully dry aerosols
    - Entrainment of solid particles in drop/aerosol fields
    - Tracking for radioactive isotopes
    - Decay heat release from radioactive particulates

Phenomena	GOTHIC Modeling
Aerosol formation	• Entrainment from films and pools caused by faster moving vapor flow • Specified source • Spray nozzle component • Water to drop conversion at cell faces • Fog formation for water droplets
Growth of aerosol particles	Growth and shrinkage due to condensation and evaporation are included Agglomeration due to: • Thermal Diffusion – Brownian motion • Turbulent Diffusion – relative motion of drops in turbulent eddies • Gravitational Collection – relative motion caused by gravity GOTHIC assumes the periphery of the drop is water when estimating the agglomeration rates.
Shape of particles	Assumes deformable spherical shape for estimating the droplet drag coefficient.
Deposition	Deposition due to: • Impaction - Bulk velocity towards a wall or surface • Gravitational Settling – gravitational forces • Thermal Diffusion – Brownian motion • Turbulent Diffusion – eddy velocity towards a wall or surface • Thermophoresis – motion due to a temperature gradient • Diffusiophoresis – mass diffusion within the vapor phase (Stefan flow for condensation)
Engineered Removal Equipment	• Generalized filter model with size dependent efficiency. • Charcoal filter and dryer/demister models are also available • Spray wash out is automatically included in the agglomeration modeling
Gas phase flow pattern	• 3D flows in complex geometries • Turbulence and jet mixing effects • Buoyancy driven flows • Leakage to adjacent rooms and environment
Liquid phase flow pattern	• Transport of deposited aerosol in films and pool – 3D flow in complex geometries
Radioactivity	• Decay and daughter formation • Decay heat • Retention in filters

Aerosol Modeling Phenomena and GOTHIC Capabilities*

*State of the Art Report on Nuclear Aerosols, NEA/CSNI/R(2009)5, 2009.

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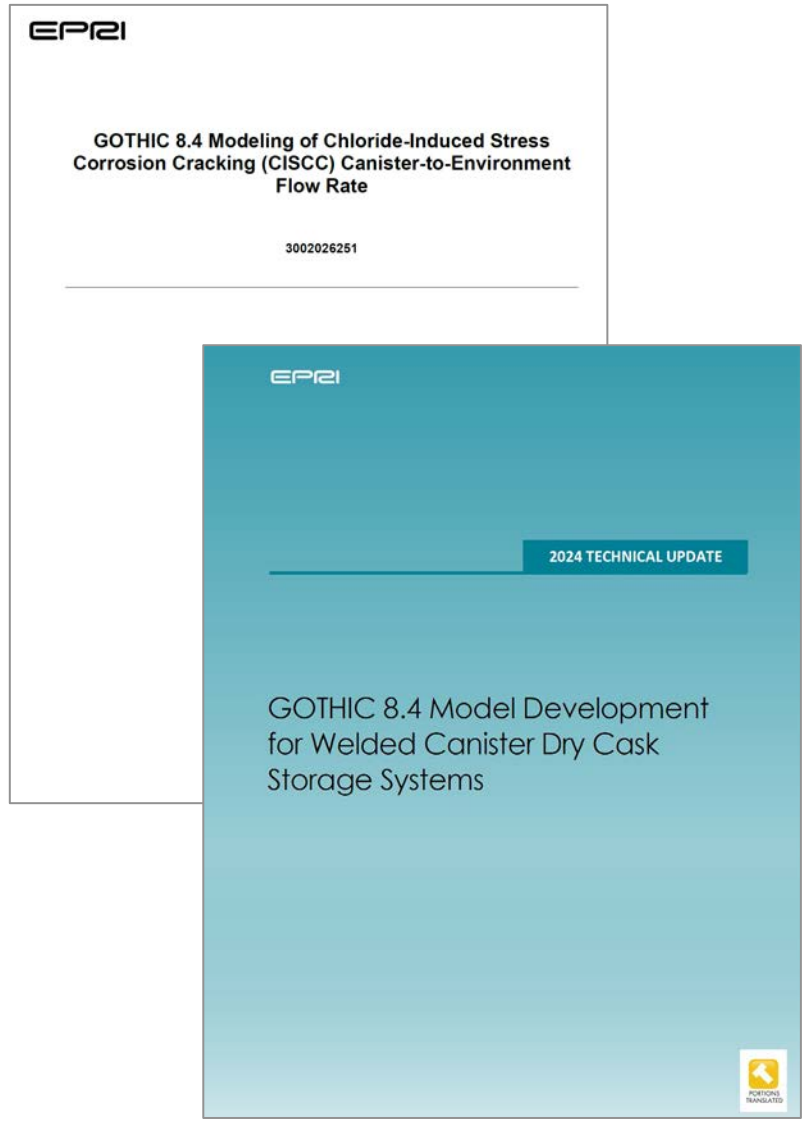
Related Activities

- TN-24P cask benchmark for model validation (Result shown at ESCP 2024)
  - GOTHIC Benchmarking for iPWR Aerosol Deposition and Spent Fuel Cask Tests, EPRI 3002020724, 2021.
  - Spent Fuel Cask and Independent Spent Fuel Storage Installation (ISFSI) Analysis Using GOTHIC, J. W. Lane, J. M. Link, P. C. Skelton, T. L. George and J. R. Harrell, Paper 21258, Waste Management Symposium, Phoenix, AZ, March 7-11, 2021. (Publicly available)
- HI-STORM cask benchmark for particle transport
  - GOTHIC 8.4 Modeling of Chloride-Induced Stress Corrosion Cracking (CISCC) Canister-to-Environment Flow Rate, EPRI 3002026251, 2023. (Publicly available)
    - Use lumped model for fuel and cask internals
  - GOTHIC 8.4 Model Development for Welded Canister Dry Cask Storage Systems, EPRI 3002028999, 2024. (Publicly available)
    - Update from previous CISCC model by using 3D sub-volume for fuel and cask internals
- Particle transportation model benchmark during fuel failure, fragmentation & relocation → On-going



Updated benchmark is needed with latest GOTHIC version

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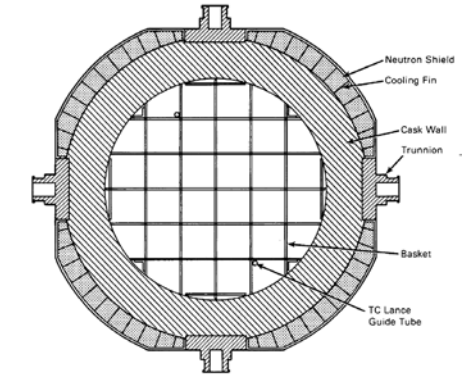
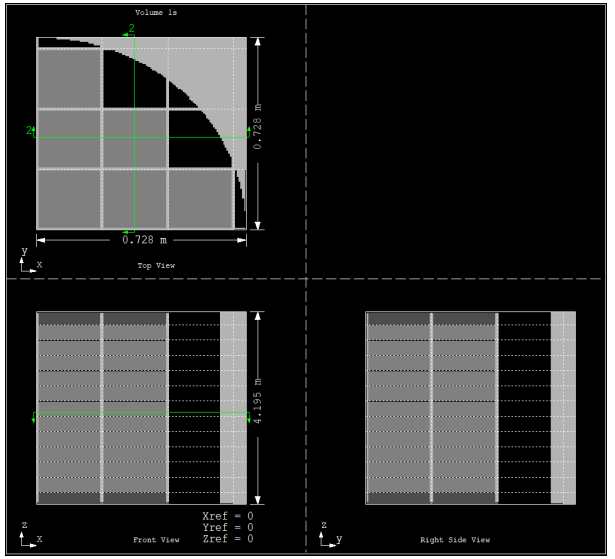
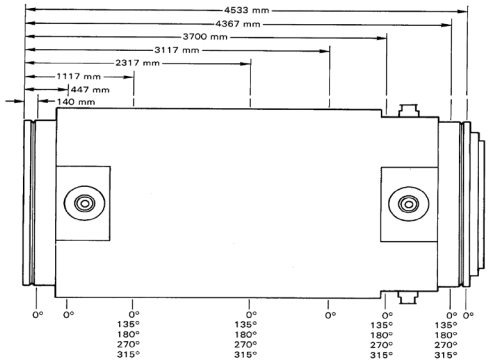
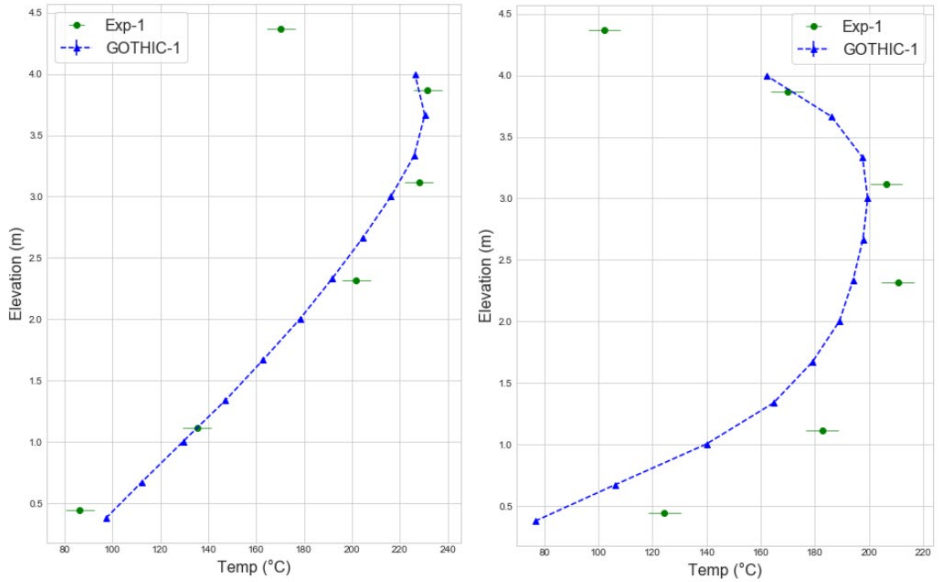
TN-24P Cask Benchmark

- EPRI experiments with TN-24P cask
  - Research for temperatures behavior of spent fuel storage cask
- GOTHIC model for the vertical temperature profile
  - 1D fuel assembly model
  - Sub-divided volume represent cask and basket walls
  - Open cell faces at the top and bottom for natural circulation path
  - Thermal conductors for fuel pins, basket plates, and cask body
  - Uniform internal heat generation for fuel modeling
  - Radiation heat transfer considered
- Results
  - Very good agreement for nitrogen fill gas throughout the cask
  - Helium fill gas shows underprediction for inner fuel assemblies and overprediction of outer fuel assemblies
  - 3+ orders of magnitude less than comparable CFD simulation
- Other notable discussions (Detail in EPRI report)
  - Vapor flow characteristics in and outside of cask
  - Sensitivity studies were performed on the Nitrogen case
  - Propose additional simulation considering atmospheric conditions









(a) Nitrogen (b) Helium

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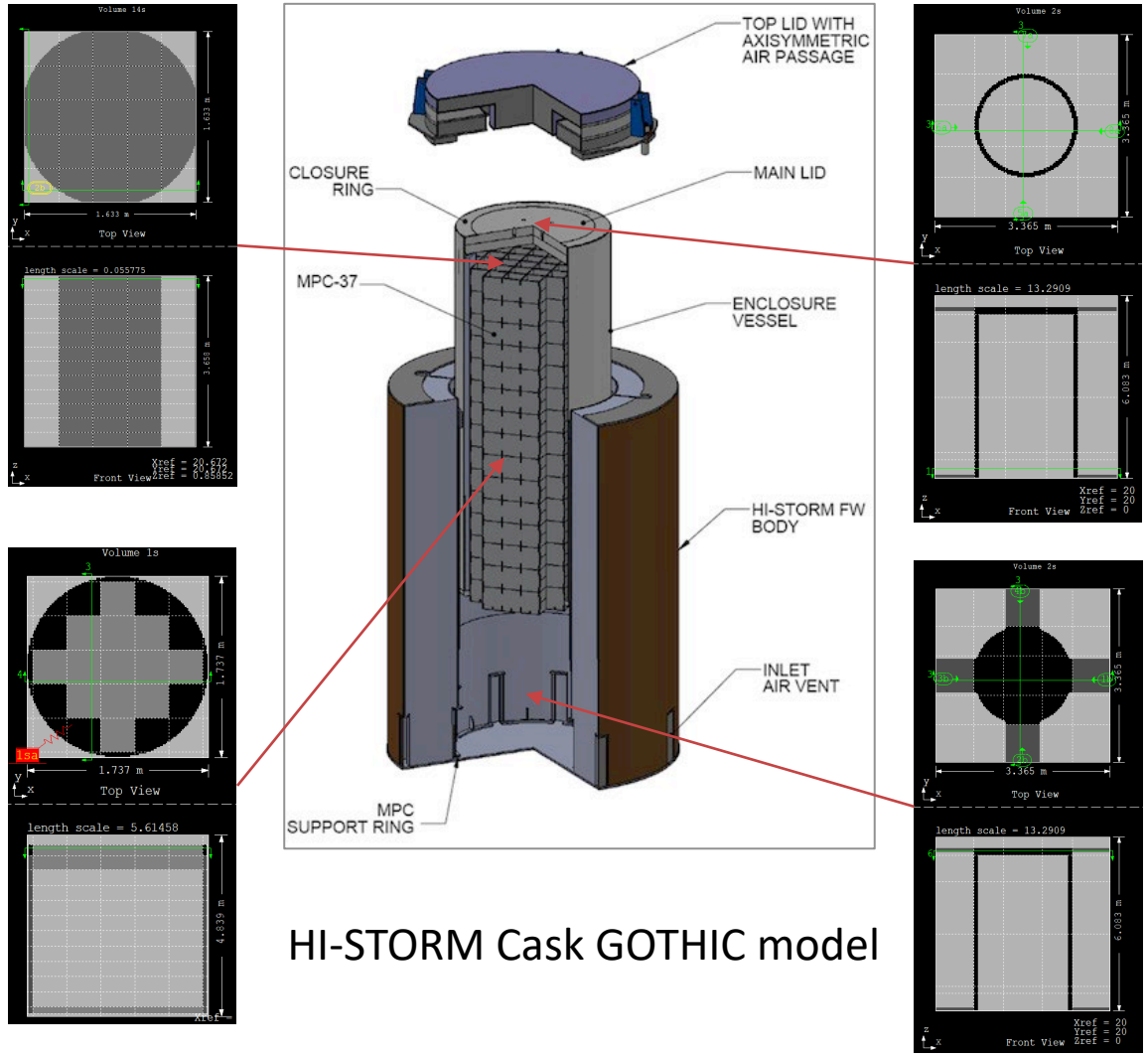
HI-STORM Cask Benchmark for Particle Transport Analysis

- Initial model development with GOTHIC 8.4
  - 3D modeling of canister internals by using sub-divided volume
  - Decay power applied
  - Mixed heat transfer in/out canister
  - Focus on fission product particle transportation behavior
  - Conclusion suggest model update
- New benchmark with GOTHIC 8.5
  - Validate updated models in GOTHIC 8.5
  - 6 cases with different storage time before crack
    - 1, 5, 10, 20, 40, and 80 years



HI-STORM Cask GOTHIC model

New benchmark aims to complete in 2026

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Update of GOTHIC 8.5 Related to DCS Analysis

- Remove mass error from artificial steam in dry condition
  - For very dry and closed condition, GOTHIC may artificially generate very small amount of steam
  - In case of very long simulation time steps (e.g., >1E6), this artificial steam amount noticeable
  - For 80 yrs simulation case of HI-STORM DCS, a mass error occurred
  - Modification of code was made to reduces the amount of artificial steam generation
- Improve code performance for dry aerosol tracking at low concentrations.
  - Goal is to conservation of very small amount dry aerosols for low liquid volume fractions (<1e-8)
  - It is important feature to simulate DCS applications
  - HISTORM model was used to verify the conservation of dry aerosols
  - Add new capability to plot the liquid component mass in the liquid phase

New feature is available in GOTHIC 8.5

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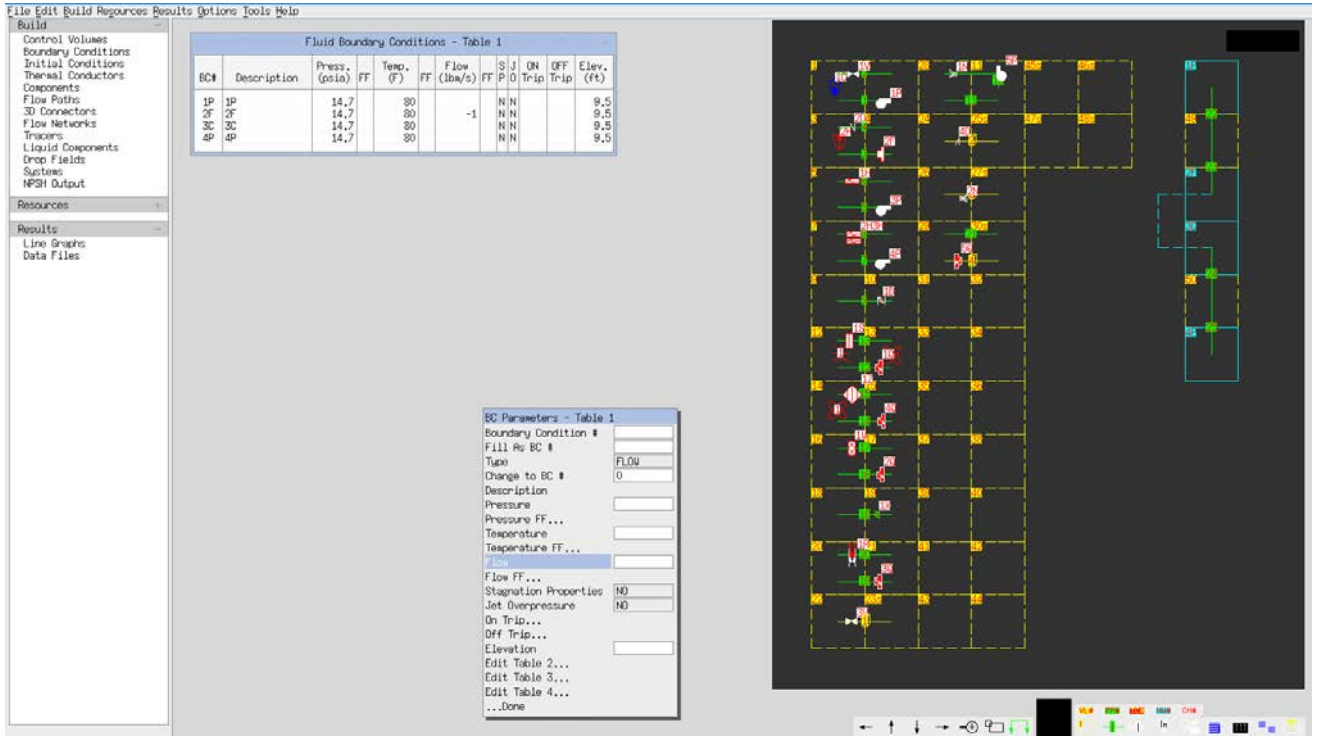
Development of GOTHIC v9.0

- Major upgrade in GUI
  - Multiple tables and tree navigation
  - User friendly design and easy customization
  - Dynamic input data checking
  - Classic option also available
- New physics models to be added
  - Drops and aerosol models / aerosol nucleation / spanned conductor, etc
  - Uncertainty analysis capability
- New models under consideration for DCS modeling
  - 3D conduction / Sol-air modeling



Updated GOTHIC v9.0 GUI

GOTHIC v9.0 Beta version release in late 2026

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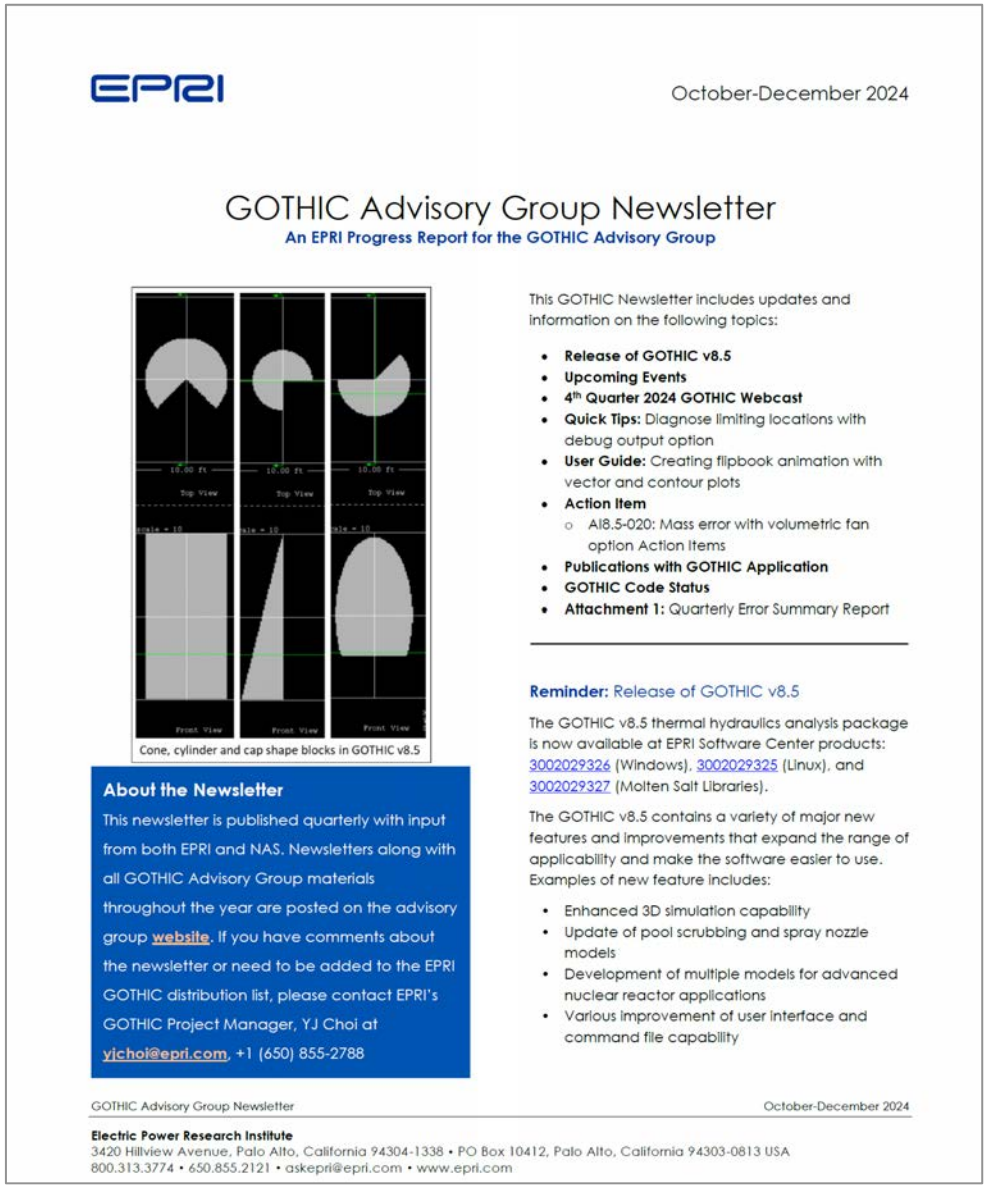
GOTHIC Advisory Group

- EPRI supports GOTHIC users
  - Maintain code quality
  - Build worldwide network
  - Provide technical support
  - Collect feedback and error
  - Develop and release new versions and add-ons
  - Organize annual Safety Technology Week
  - Provide training, webcast and workshop
  - Publish quarterly newsletter and error report
    - Events / User guide / Action items / Publications
  - Archive documents and previous versions
  - Participate in various international initiatives and science and engineering communities

Please contact gothic@epri.com for the membership



EPRI

October-December 2024

GOTHIC Advisory Group Newsletter

An EPRI Progress Report for the GOTHIC Advisory Group

Cone, cylinder and cap shape blocks in GOTHIC v8.5

This GOTHIC Newsletter includes updates and information on the following topics:

- **Release of GOTHIC v8.5**
- **Upcoming Events**
- **4th Quarter 2024 GOTHIC Webcast**
- **Quick Tips:** Diagnose limiting locations with debug output option
- **User Guide:** Creating flipbook animation with vector and contour plots
- **Action Item**
  - AI8.5-020: Mass error with volumetric fan option Action Items
- **Publications with GOTHIC Application**
- **GOTHIC Code Status**
- **Attachment 1:** Quarterly Error Summary Report

About the Newsletter

This newsletter is published quarterly with input from both EPRI and NAS. Newsletters along with all GOTHIC Advisory Group materials throughout the year are posted on the advisory group website. If you have comments about the newsletter or need to be added to the EPRI GOTHIC distribution list, please contact EPRI's GOTHIC Project Manager, YJ Choi at yjchoi@epri.com, +1 (650) 855-2788

Reminder: Release of GOTHIC v8.5

The GOTHIC v8.5 thermal hydraulics analysis package is now available at EPRI Software Center products: 3002029326 (Windows), 3002029325 (Linux), and 3002029327 (Molten Salt Libraries).

The GOTHIC v8.5 contains a variety of major new features and improvements that expand the range of applicability and make the software easier to use. Examples of new feature includes:

- Enhanced 3D simulation capability
- Update of pool scrubbing and spray nozzle models
- Development of multiple models for advanced nuclear reactor applications
- Various improvement of user interface and command file capability

GOTHIC Advisory Group Newsletter October-December 2024

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GOTHIC Advisory Group Newsletter

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Concluding Remarks

- EPRI GOTHIC code continuously supports spent fuel storage modeling
  - Evaluate potential and perform benchmarks
  - Update and develop relevant models in heat transfer and particle transport and deposition
- Upcoming GOTHIC events
  - Dec. 4, 11am EST: 4Q Webcast, Registration
  - Jun. 15-19, 2026: Safety Technology Week @ EPRI Charlotte offices
- Possible training and technical support
- Contacts
  - YJ Choi, yjchoi@epri.com, 208-821-4526
  - Software User Support & Licenses
  - GOTHIC User Support
    - gothic@epri.com / gothic@numerical.com / support@numerical.com
  - GOTHIC License Requests
    - https://www.numerical.com/gothic_license

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Preliminary Decay Heat Validation Results

EPRI-SKB Collaborative Project on Decay Heat

Fredrik Johansson[1] and Hatice Akkurt[2]
[1]SKB
[2]EPRI

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 27-30, 2025

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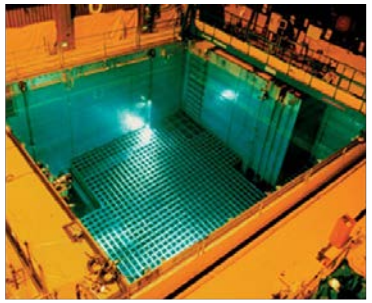
Decay Heat is an Important Parameter That Impacts Entire Back-end Operation



Spent Fuel Pool (SFP)

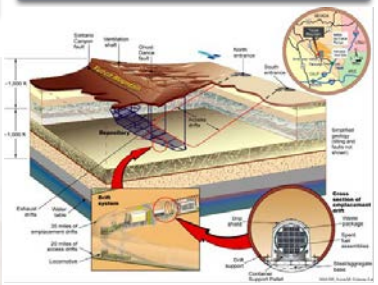
- SFP Heat Management
- Available storage capacity
- Outage management

Dry Storage & Centralized

- Loading
- Fuel/Cladding Integrity
- Canister Integrity

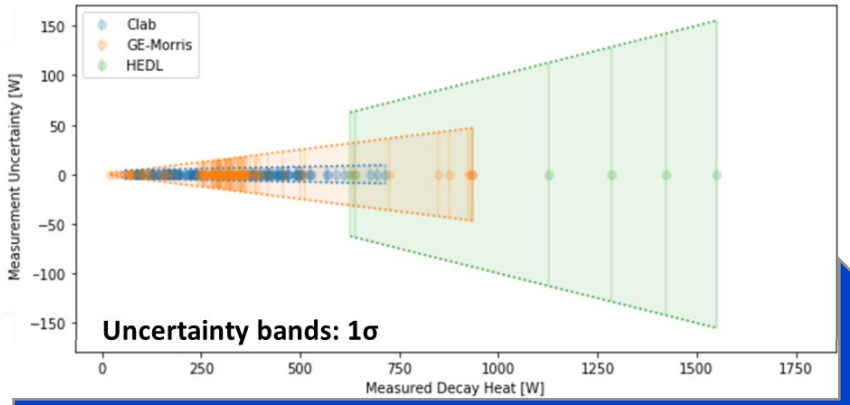
Transportation

- Transportation limits

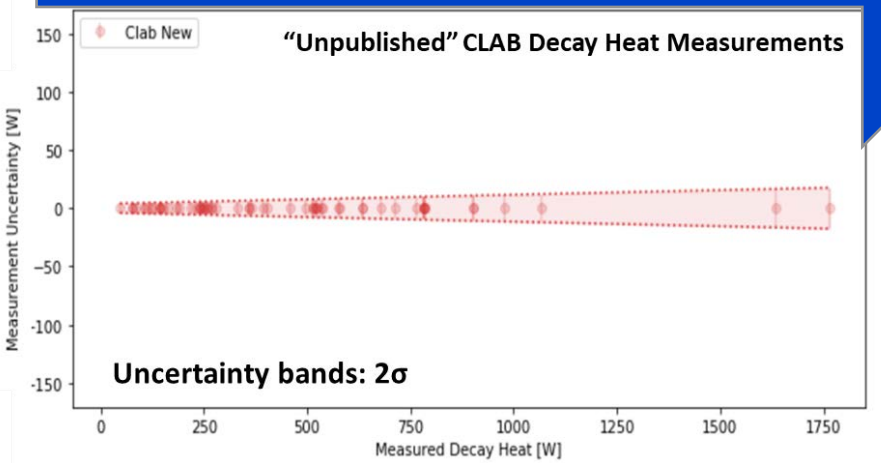


Disposal

- Heat load management
- Dictates number of casks or canisters that can be stored in repository



Reasonably accurate estimation of decay heat, with low uncertainty, is important for the entire back-end operation



EPRI report, 3002026549, published October 2024 and publicly available from epri.com

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Clab Decay Heat Measurements Performed Between 2003-2021

Summary of the report published in Progress in Nuclear Energy

Progress in Nuclear Energy 191 (2026) 106001

Contents lists available at ScienceDirect

Progress in Nuclear Energy

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

Evaluation of clab decay heat measurements from 2003 to 2021 under EPRI-SKB collaboration

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[b] Swedish Nuclear Fuel and Waste Management Company (SKB), Evenemangsgatan 13, SE-169 79 Solna, Sweden
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ARTICLE INFO

Keywords:
Decay heat
Decay heat uncertainty
Validation
Clab

ABSTRACT

Accurate estimation of decay heat for spent nuclear fuel is important for wet storage (spent fuel pool heat management), cask/canister loading campaigns (ensuring that temperature limits are met), dry storage system aging management, transportation, and eventual disposal. Clab decay heat measurements performed between 2003 and 2004 were published in 2006 and used across the globe for computer code validation purposes. Currently, Clab is the only facility in the world that performs full assembly decay heat measurements. Recently, EPRI and SKB entered into a collaborative agreement to evaluate and publish all Clab decay heat measurements performed between 2003 and 2021. The publicly available publication has not only extended the decay heat validation range significantly but has also improved characterization and evaluation of the measurements, including associated uncertainties.

In this paper, an overview of Clab decay heat measurements and evaluation of the measurement uncertainties for the decay heat measurements performed between 2003 and 2021 are presented. Clab measurement uncertainty is re-evaluated to include previously unpublished calibration data, to assess the impact of data quality screening, and to review and revise certain prior conservative assumptions. The re-evaluation relies primarily on the accumulated population of Clab measured data. Finally, lessons learned and recommendations for future decay heat measurements are provided.

Journal article in PNE is published in September 2025 and available as open access article

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Overview of Clab Decay Heat Measurements & Validation Suite

Reactor	Type	Fuel Array Size	Number of Meas.
Barsebäck 1	BWR	8x8	11
Barsebäck 2	BWR	8x8, 10x10	6
Forsmark 1	BWR	8x8, 9x9, 10x10	11
Forsmark 2	BWR	8x8	20
Forsmark 3	BWR	10x10	13
Oskarshamn 2	BWR	8x8	13
Oskarshamn 3	BWR	8x8, 10x10	30
Ringhals 1	BWR	8x8, 10x10	48
Total BWR			**152**
Ringhals 2	PWR	15x15	44
Ringhals 3	PWR	17x17	57
Ringhals 4	PWR	17x17	17
Total PWR			**118**
Total Number of Measurements			**270**

ID	Code/Method	Cross Sections	Method	Axial Detail	# of Cases
1	ORIGEN (SCALE 6.2)	ENDF 7.1	Master	2-D	270
2	ORIGEN (SCALE 6.3)	ENDF 7.1	Master	2-D	270
3	ORIGEN (SCALE 6.2)	ENDF 7.1	Master*	“3-D”	236
4	Polaris (SCALE 6.3)	ENDF 7.1	Master	2-D	270
5	Polaris (SCALE 6.3)	ENDF 8.0	Master	2-D	270
6	Polaris (SCALE 6.3)	ENDF 7.1	Master*	“3-D”	236
7	Polaris (SCALE 6.3)	ENDF 8.0	Master*	“3-D”	236
8	SNF 1.6 C4	JEFF 2.2	C4/S5	3-D	236
9	SNF 1.8 C5	ENDF 7.1	C4/S5	3-D	234
10	SNF 1.8 C5	ENDF 7.1	Master	2-D	224

The goal is to evaluate the results using different codes (ORIGEN, Polaris, SNF), cross sections libraries (ENDF7.1; ENDF8; JEFF 2.2)and approaches (2D versus 3D) - Master* refers to the Excel file attached to the EPRI report

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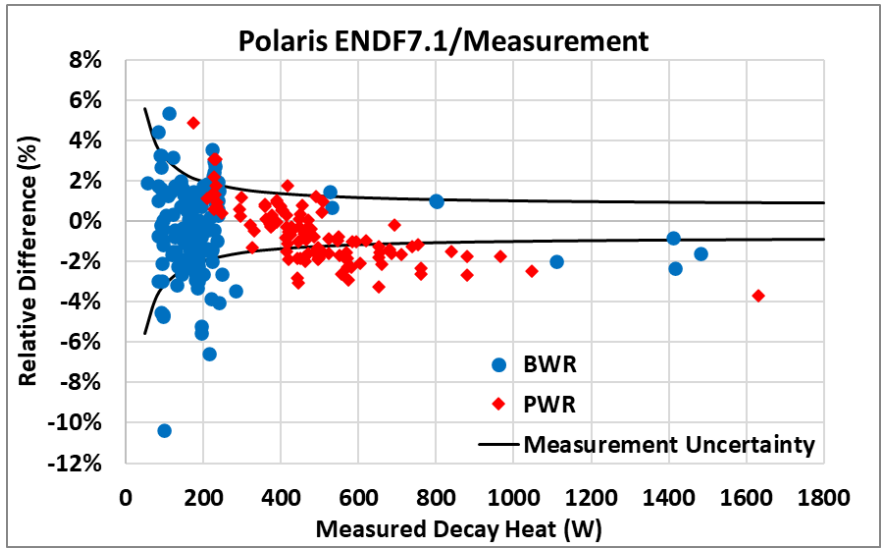
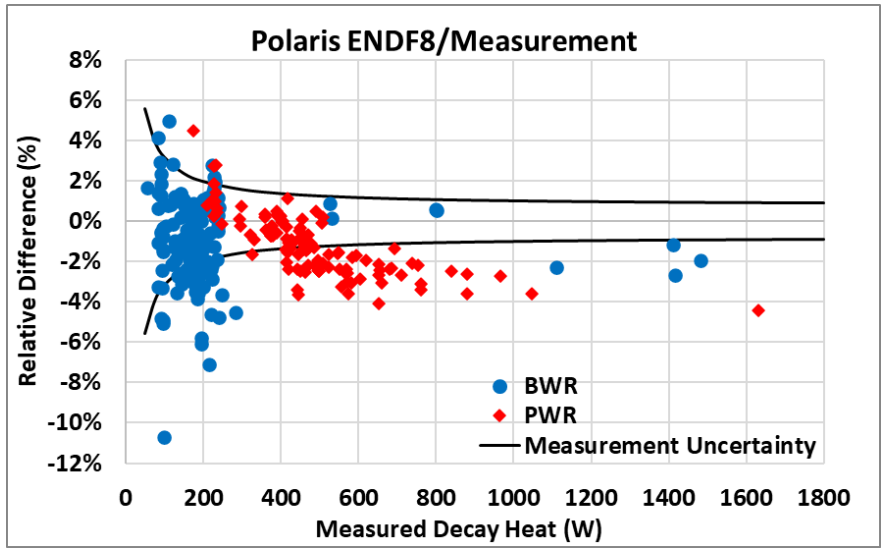
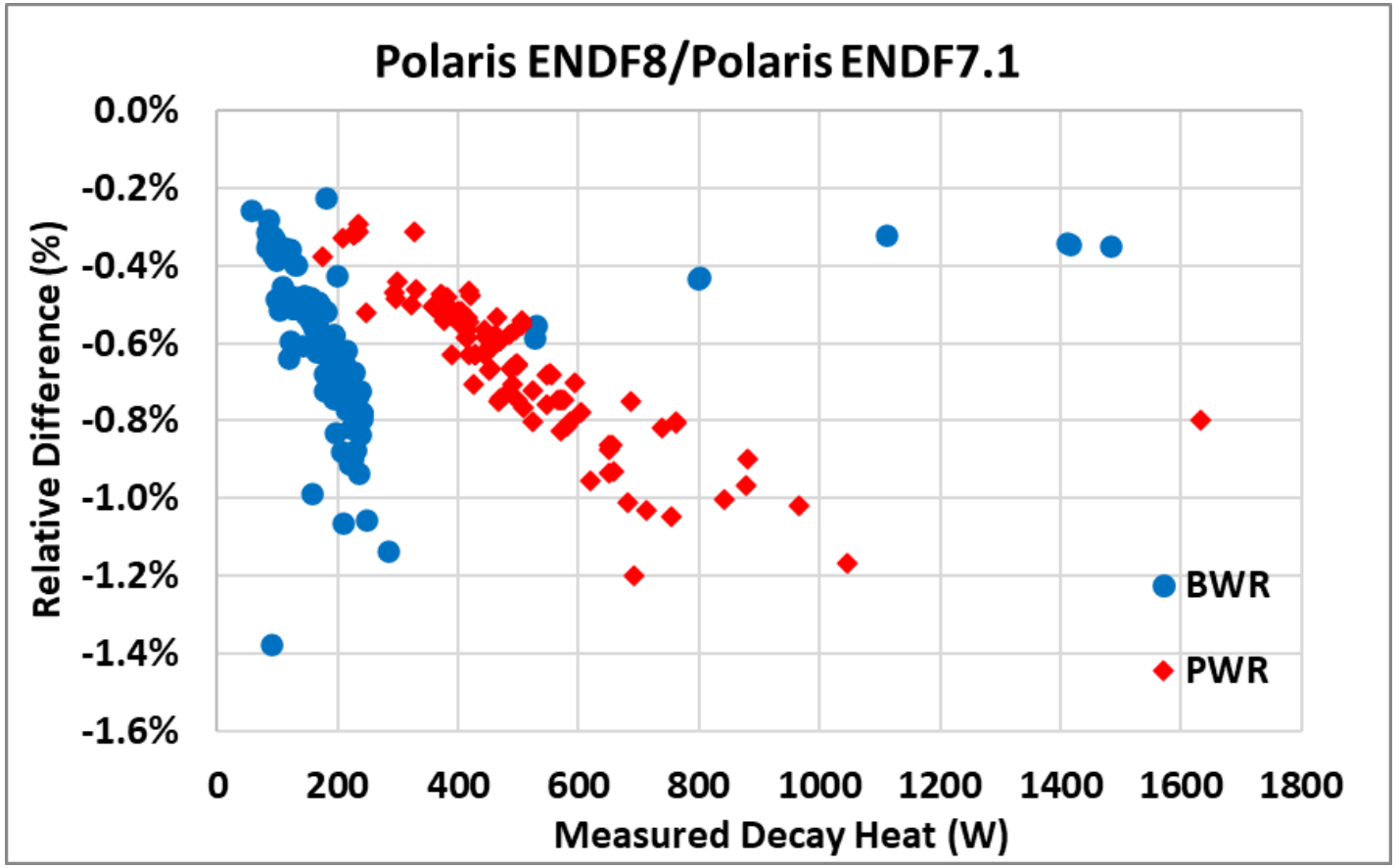
Cross Section Library Impact – Polaris ENDF7.1 versus ENDF8







Trend with the measured decay heat for both BWR and PWR measurements

ENDF8 results are consistently lower than ENDF7.1 results using Polaris

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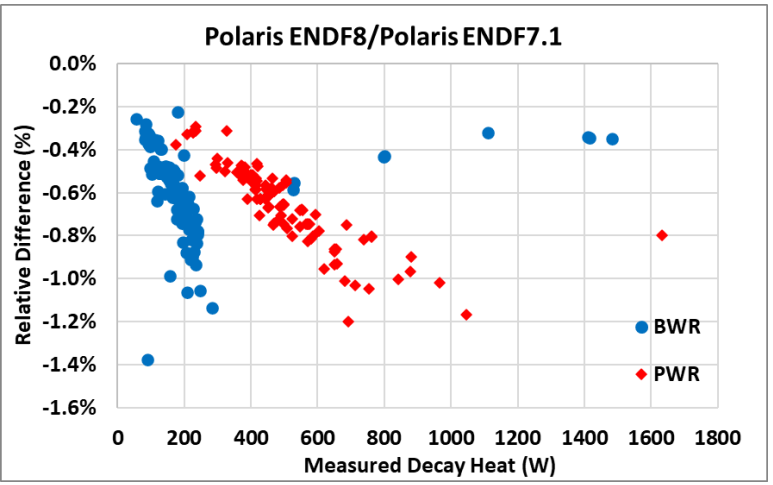
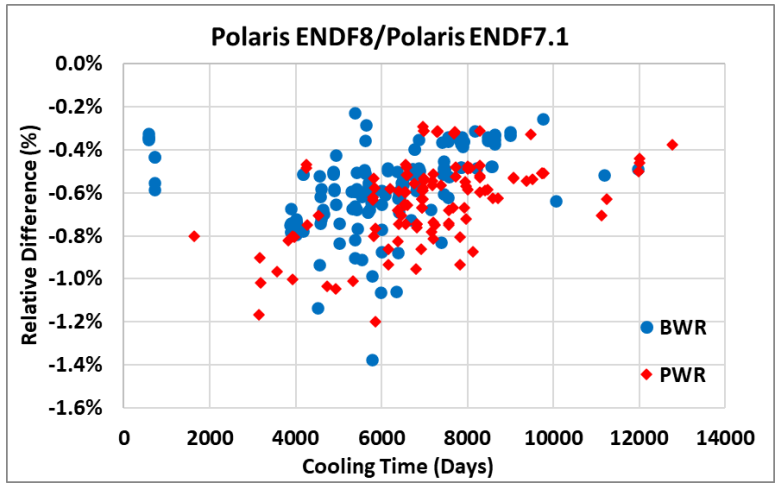
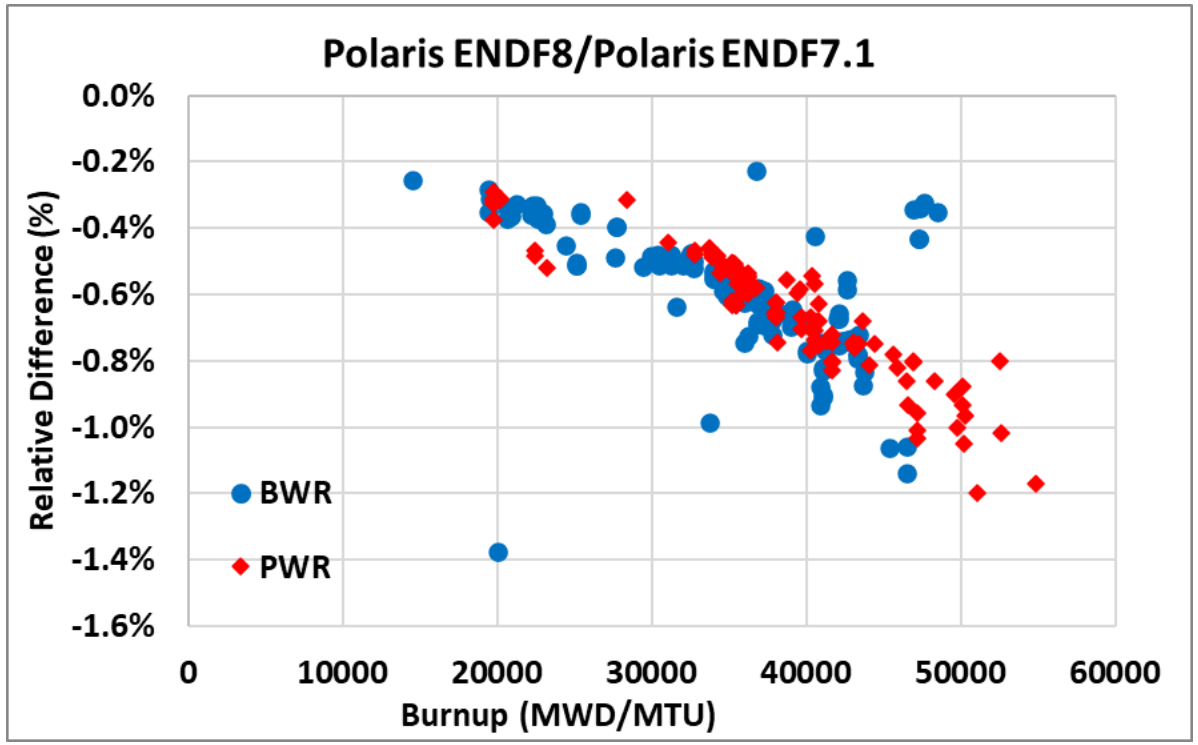
Cross Section Library Impact – Polaris ENDF7.1 versus ENDF8



Trend with the measured decay heat for both BWR and PWR measurements



No clear trend with cooling time



Clear trend with burnup for both PWR & BWR

While no clear trend as a function of cooling time, clear trend as a function of burnup

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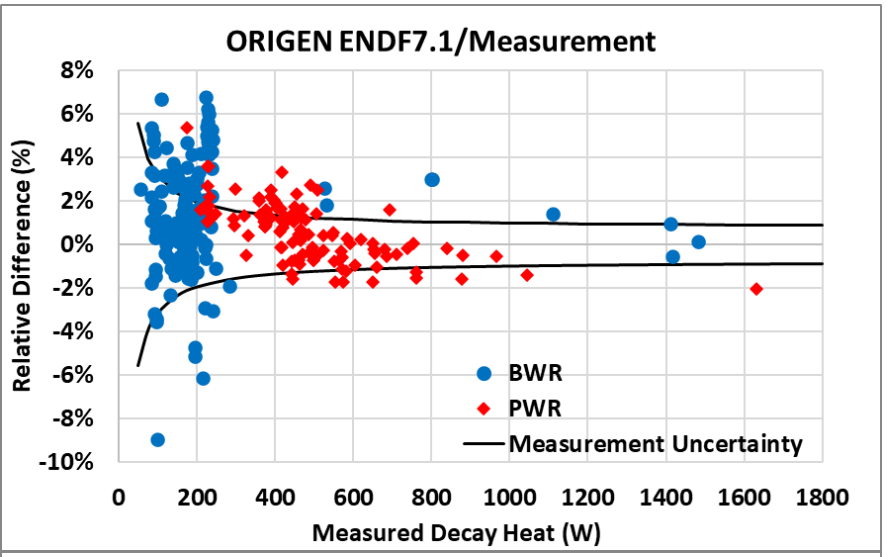
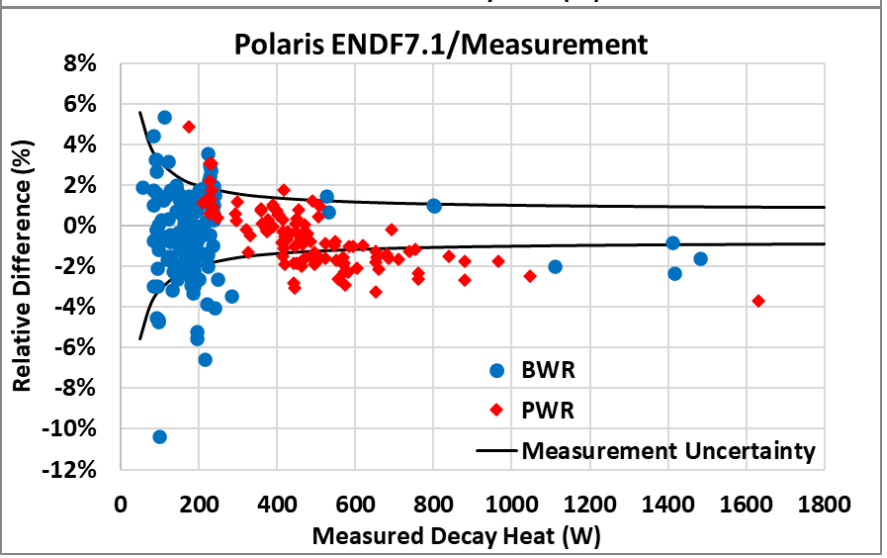
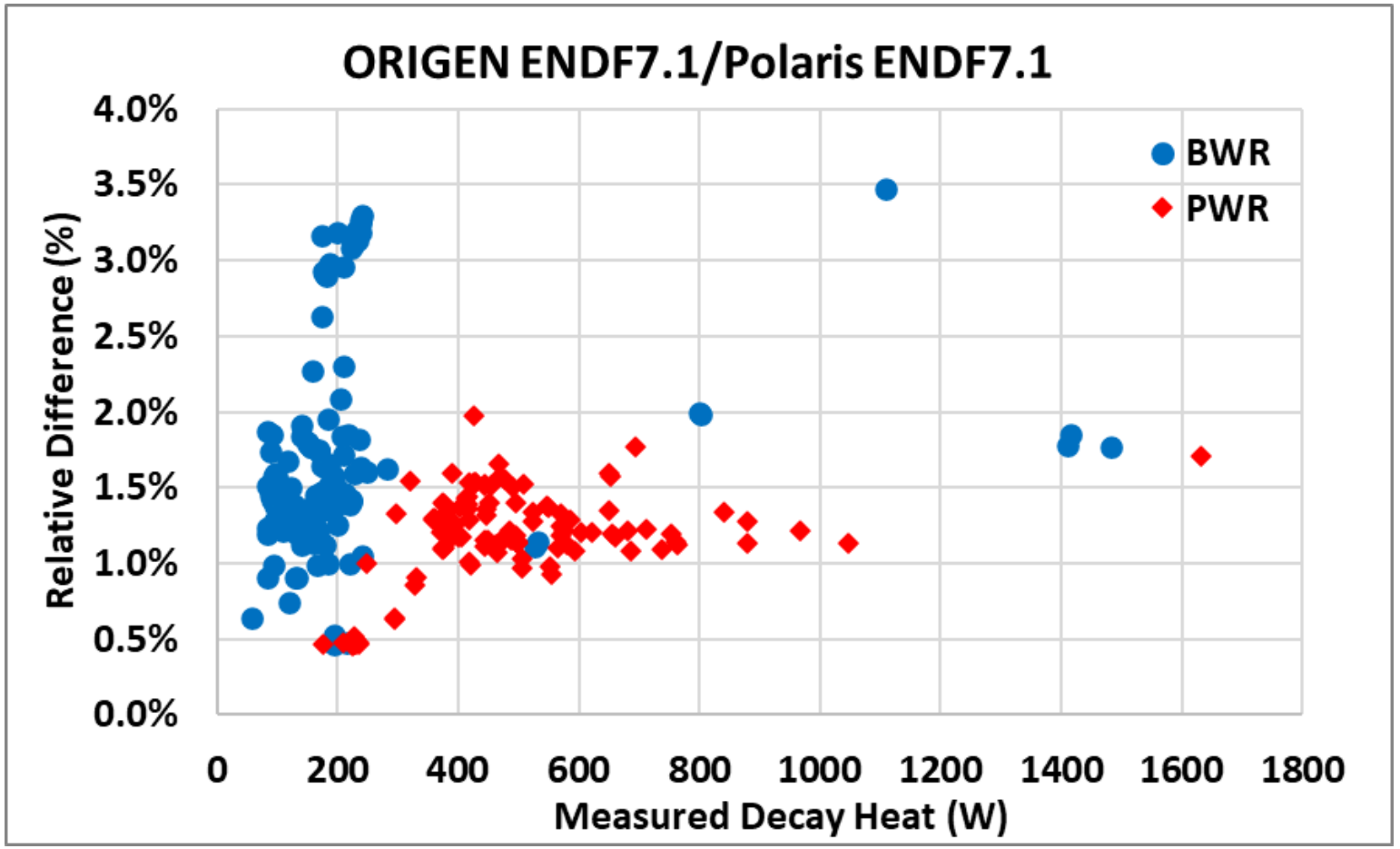
ORIGEN versus Polaris Using ENDF7.1







ORIGEN results are consistently higher than Polaris results. ORIGEN calculations performed using pre-generated cross section libraries

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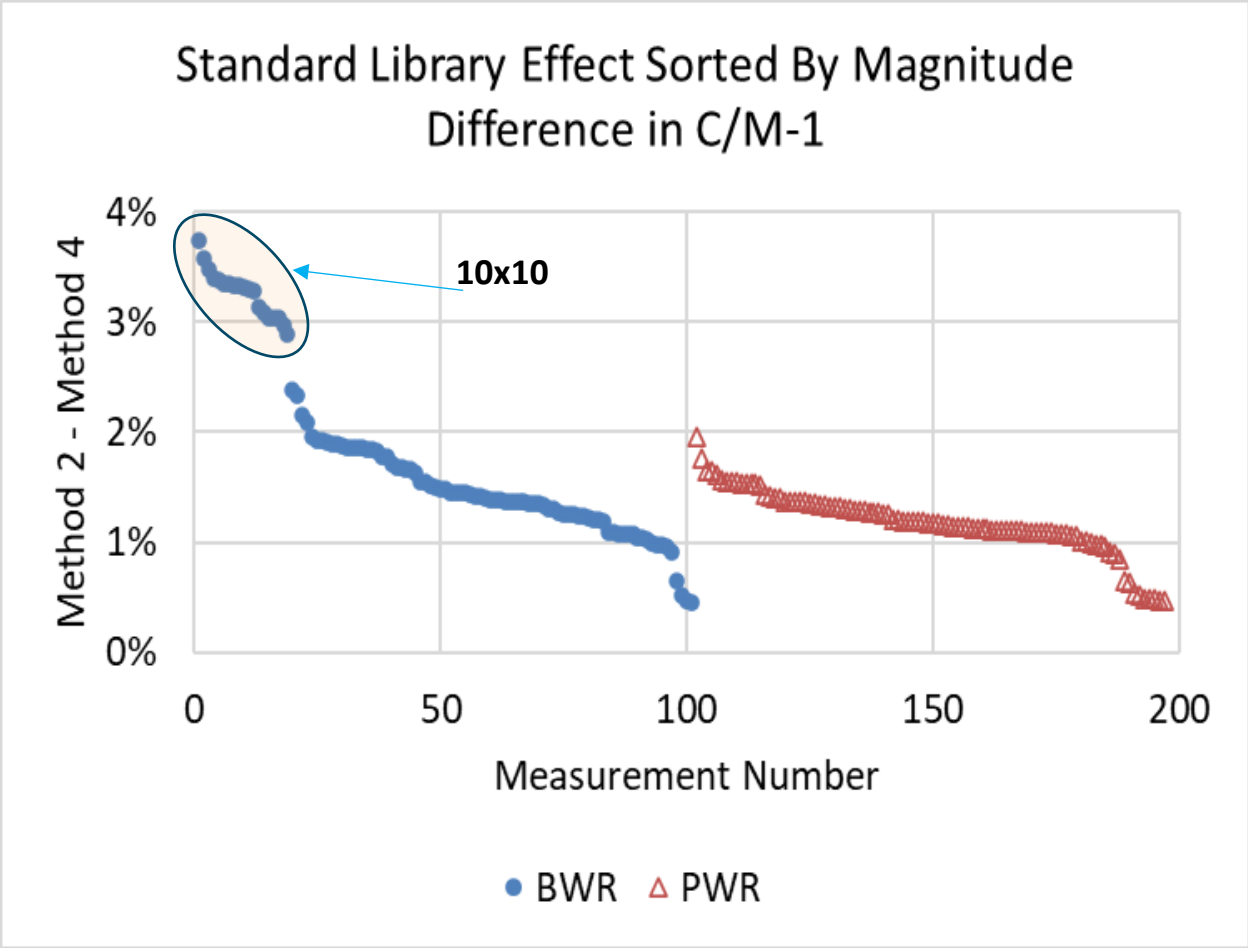
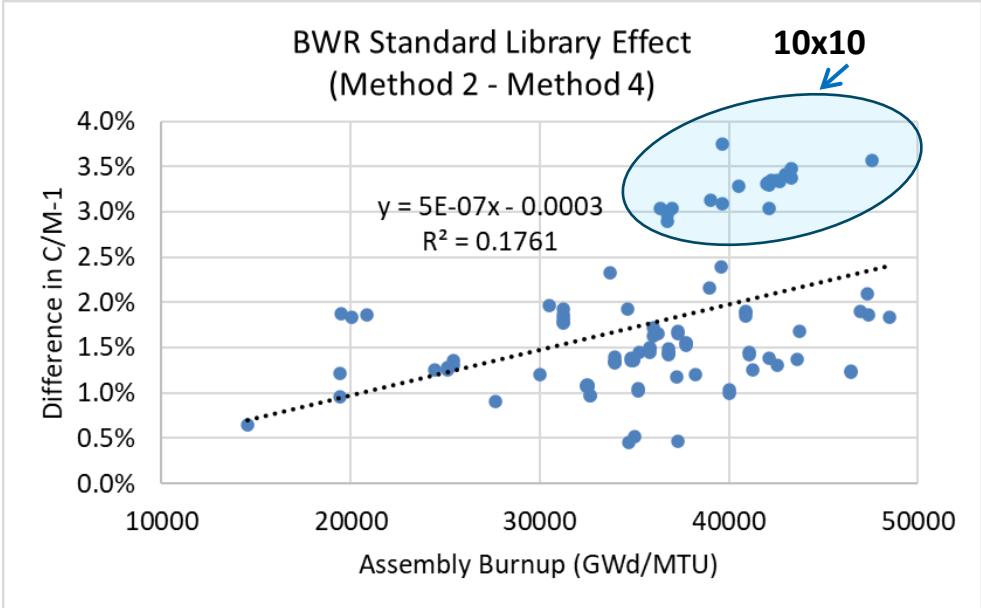
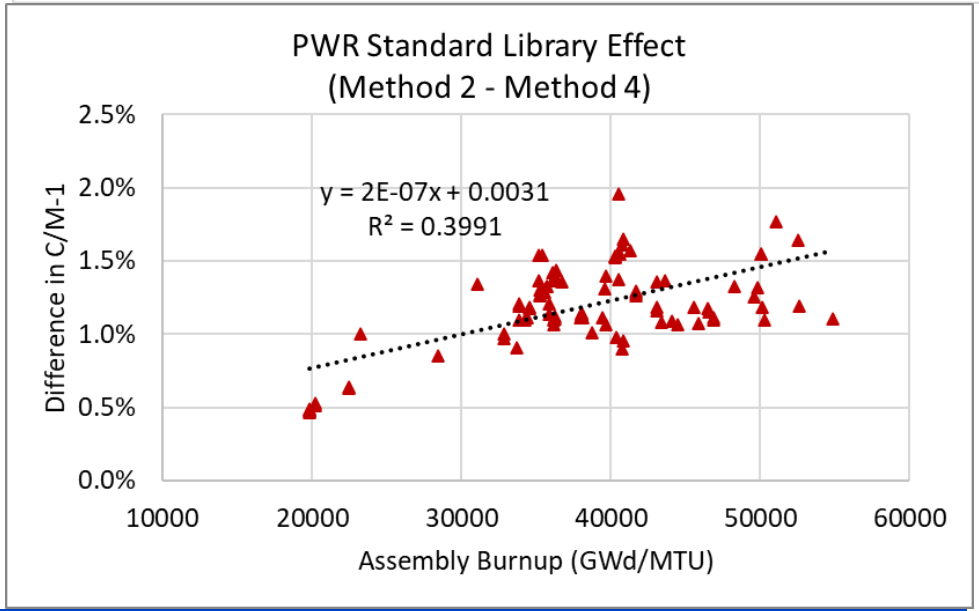
ORIGEN versus Polaris Using ENDF7.1







ORIGEN results are consistently higher than Polaris results – BWR 10x10 shows higher differences

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Preliminary Validation Results Using ORIGEN and POLARIS

- For the majority, computed results are within ±4% of the measured decay heat
- There is sensitivity to cross section library: ENDF7.1 versus ENDF8 when the same code is used
- There are differences between ORIGEN and Polaris results even when the same libraries are used
  - ORIGEN results use pre-generated libraries
  - 10x10 results show larger differences (+4% versus +2% for the others)
- In general, computational results show higher uncertainty compared to measurement uncertainty
  - Even with the cross-section library change, larger deviations between the results (using the same code)
  - Sensitivity analyses will be carried
  - Especially burnup sensitivity

While the analysis is ongoing, recommend separating BWR and PWR validation suites

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SNF Decay Heat Computations

SNF Decay Heat Computations

- **Approach 1:** SNF v1.6, CASMO4, JEFF 2.2, detailed 3-D history
- **Approach 2:** SNF v1.8, CASMO5, ENDF 7.1, detailed 3-D history
- **Approach 3:** SNF v1.8, CASMO5, ENDF 7.1, 2-D cycle-average using “Master” file

Understand effects of:

- Nuclear data libraries (JEFF 2.2 vs ENDF 7.1), including different code versions - SNF 1.6/CASMO4/JEFF 2.2 vs SNF 1.8/CASMO5/ENDF 7.1
- Modeling detail (3-D vs 2-D histories)
- Identify trends in the calculation (C) versus measurement (M) data with key variables (BWR vs PWR, burnup, cooling time, etc.)

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Validation Set, Codes, and Versions used

- **Limitations in the validation set:** The SNF version is tied to the corresponding CASMO version.
- Only a limited number of fuels were calculated with nuclear data library changes as the sole parameter.
- There are no limitations on the combination between SNF/CASMO versions and the SIMULATE version.

	C4/S3	SNF 1.6 - Casmo 4	SNF 1.8 - Casmo 5
Approach 1:	JEFF 2.2	236	0
Approach 2:	ENDF 7.1	11	234
	2-D cycle-average		
Approach 3:	ENDF 7.1	0	224

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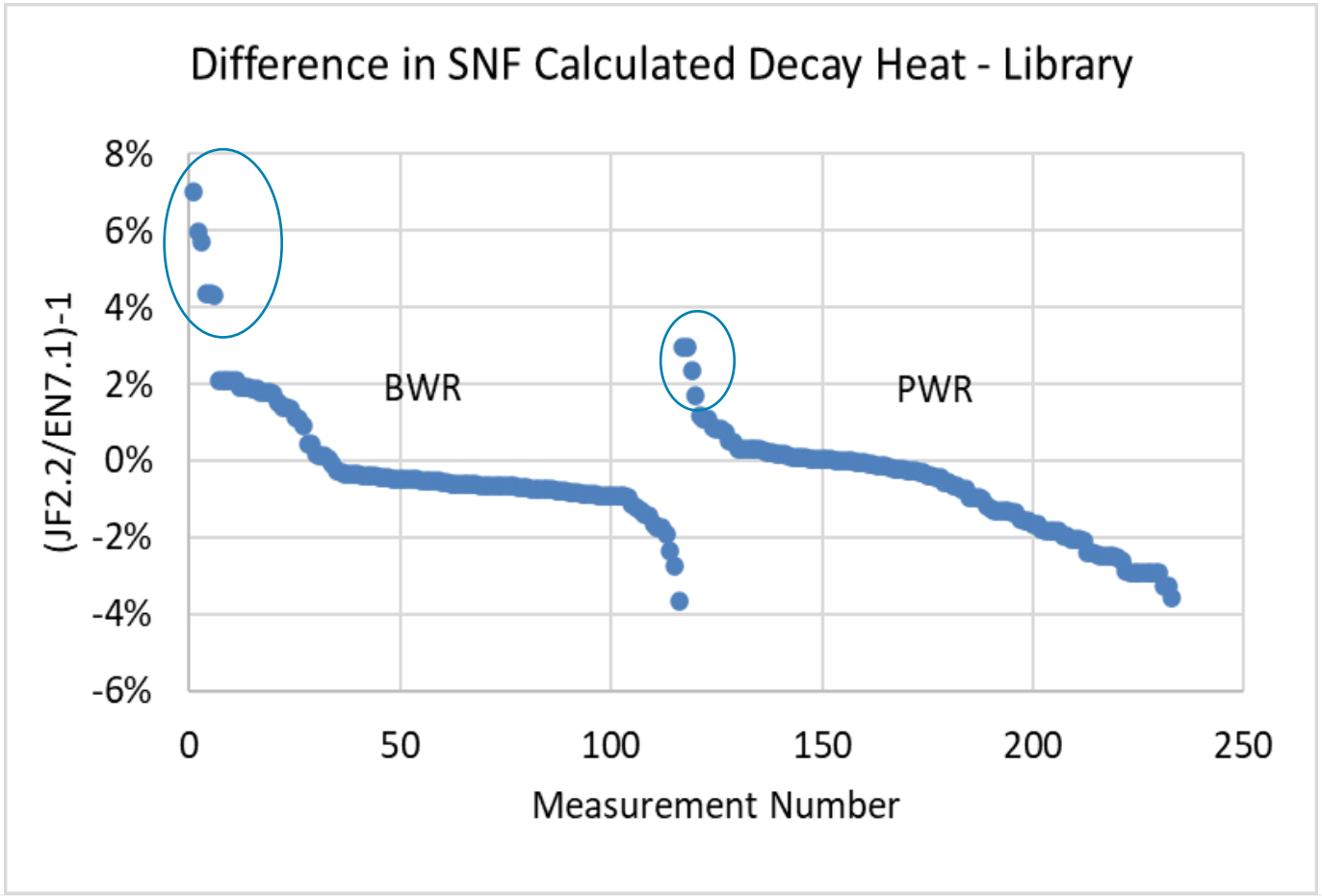
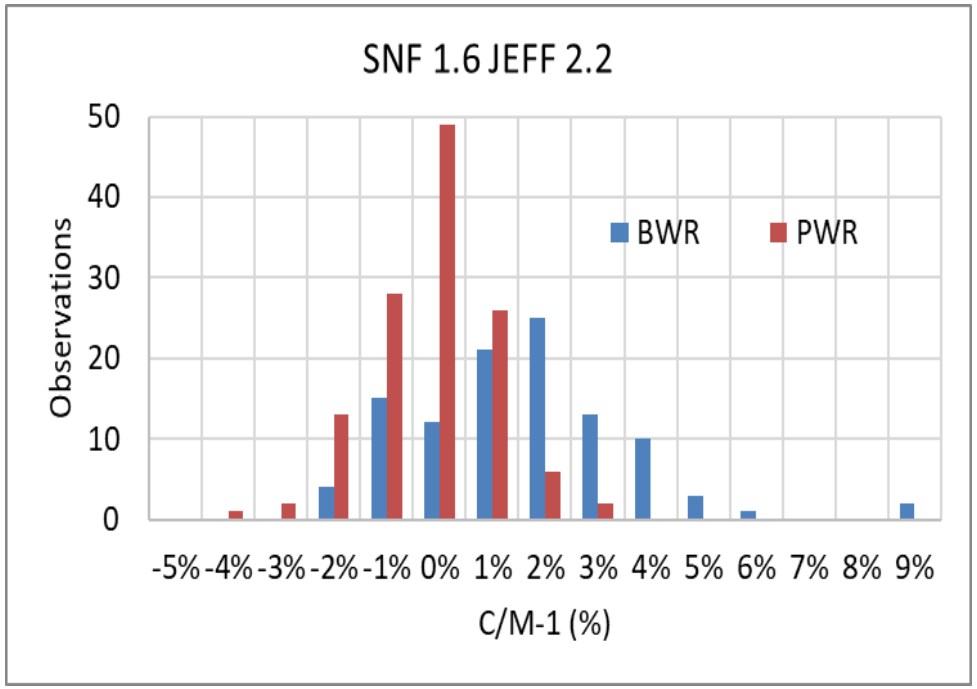
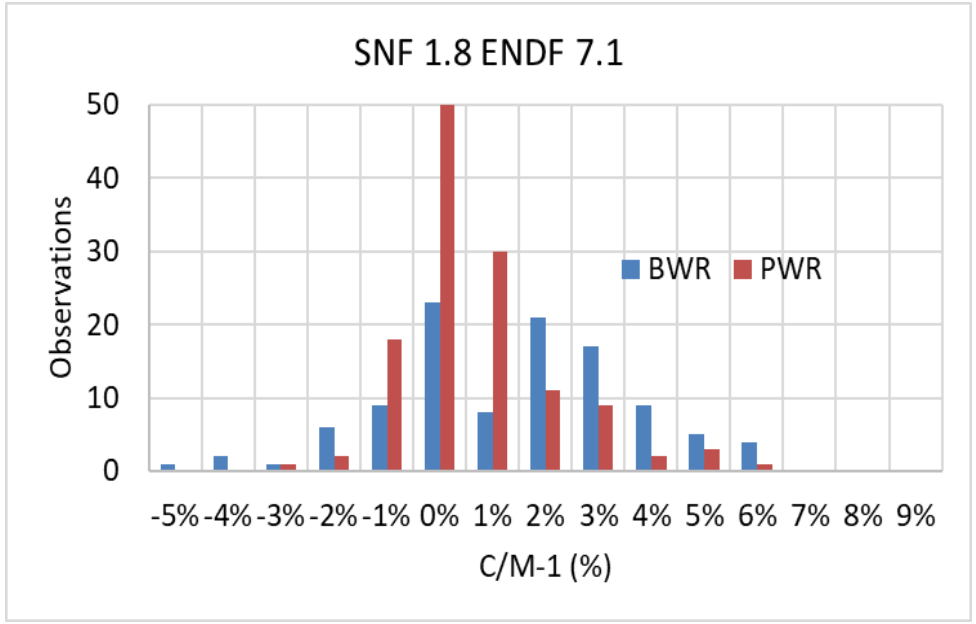
Cross-Section Library Comparison

(Compare Approach 1 vs Approach 2)

- 116 BWR + 117 PWR measurements
- **SNF 1.6 C4 +** C4/S3 **(JEFF 2.2)** vs **SNF 1.8 C5 +** C4/S3 **(ENDF 7.1)**







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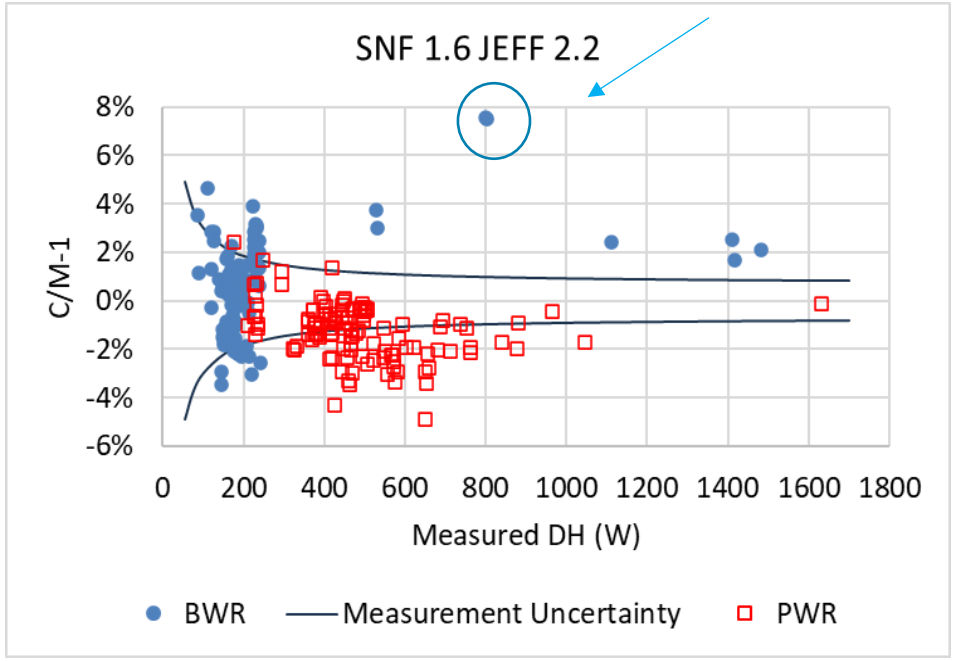
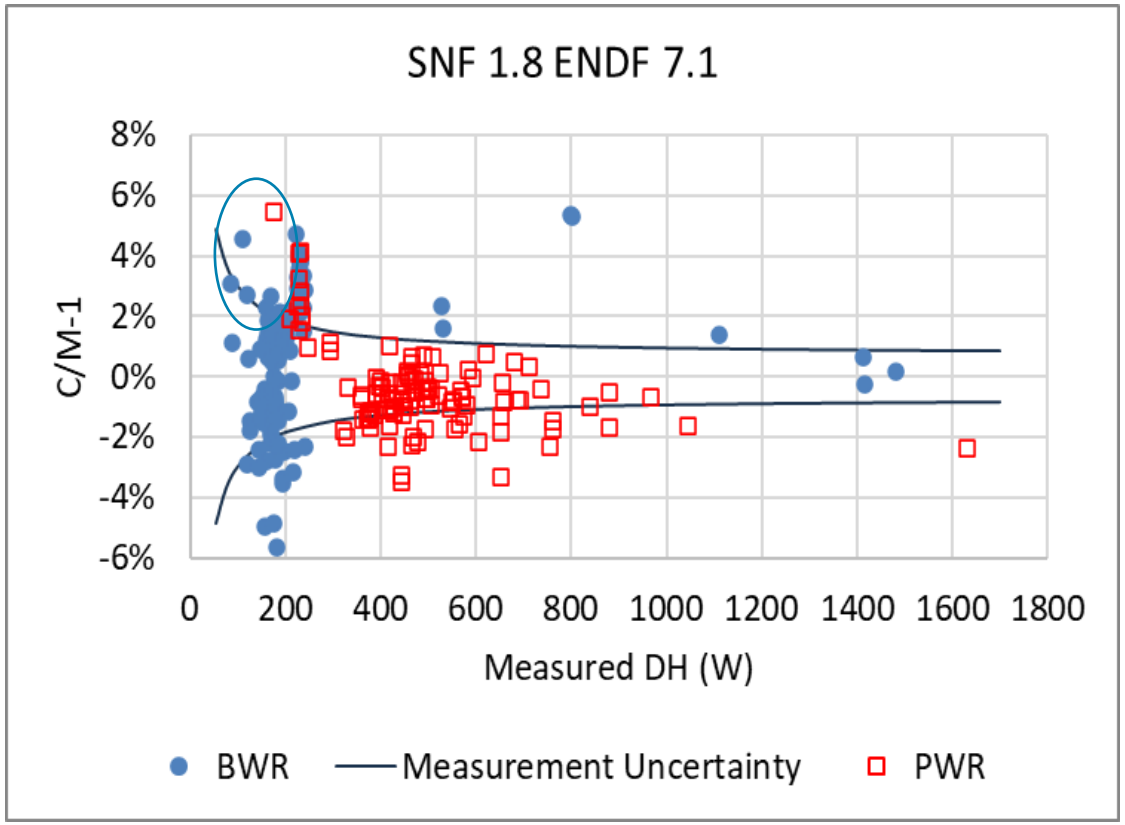
JEFF 2.2 C/M computed results as a function of measured decay heat along with approximate measurement uncertainty bands

2 Forsmark fuels at 800 W



The measurement uncertainty bands bound 57% of the C/M differences.

ENDF 7.1 C/M computed results as a function of measured decay heat along with approximate measurement uncertainty bands



The measurement uncertainty bands bound 58% of the C/M differences.

- **Individual calculated DH values changed as much as 7% due to the cross section library change.**
- **97% of the JEFF 2.2 C/M population and 95% of the ENDF 7.1 C/M population are within ±4%.**

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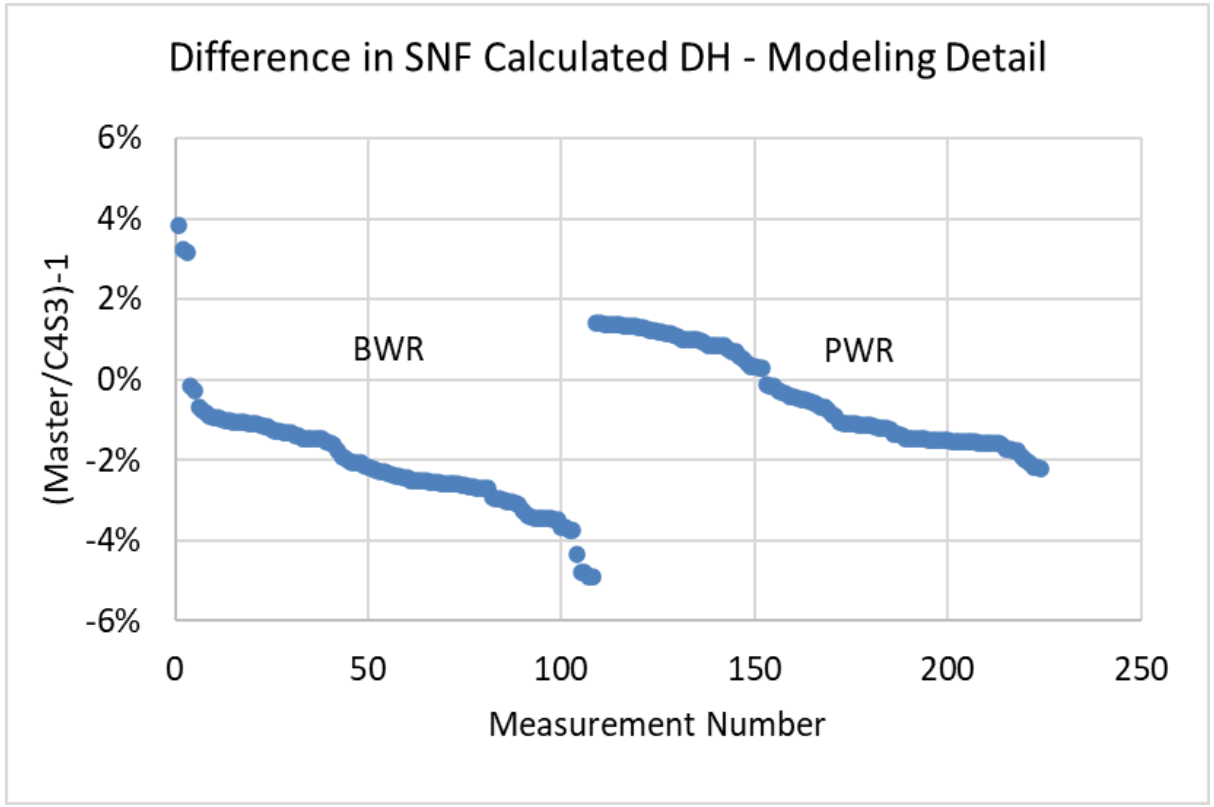
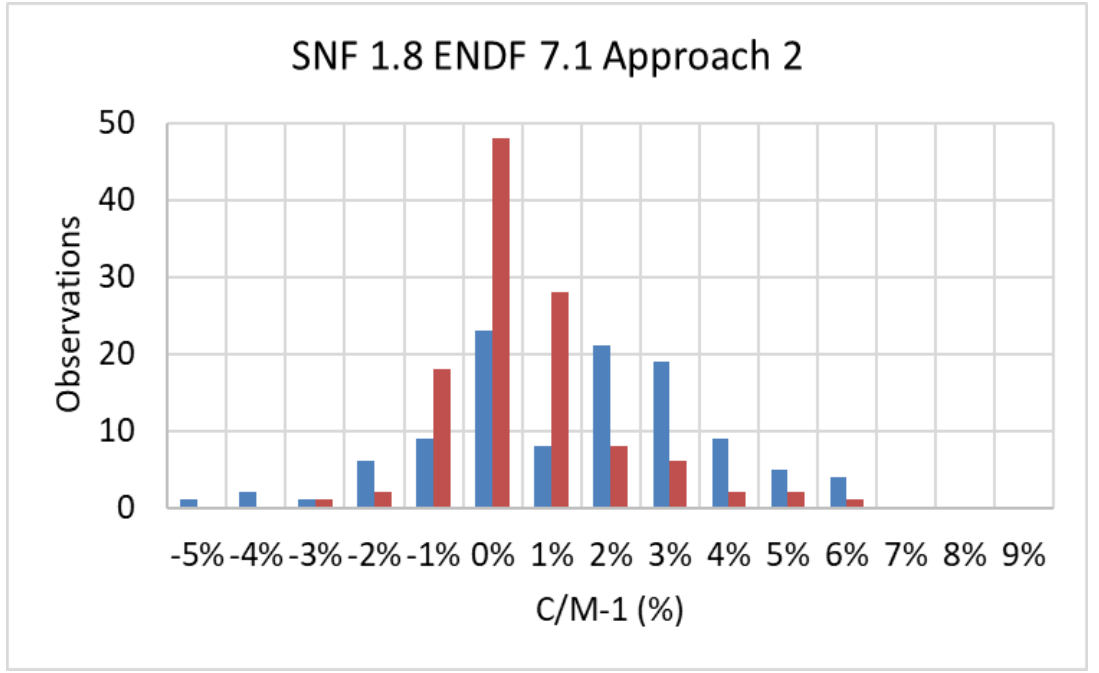
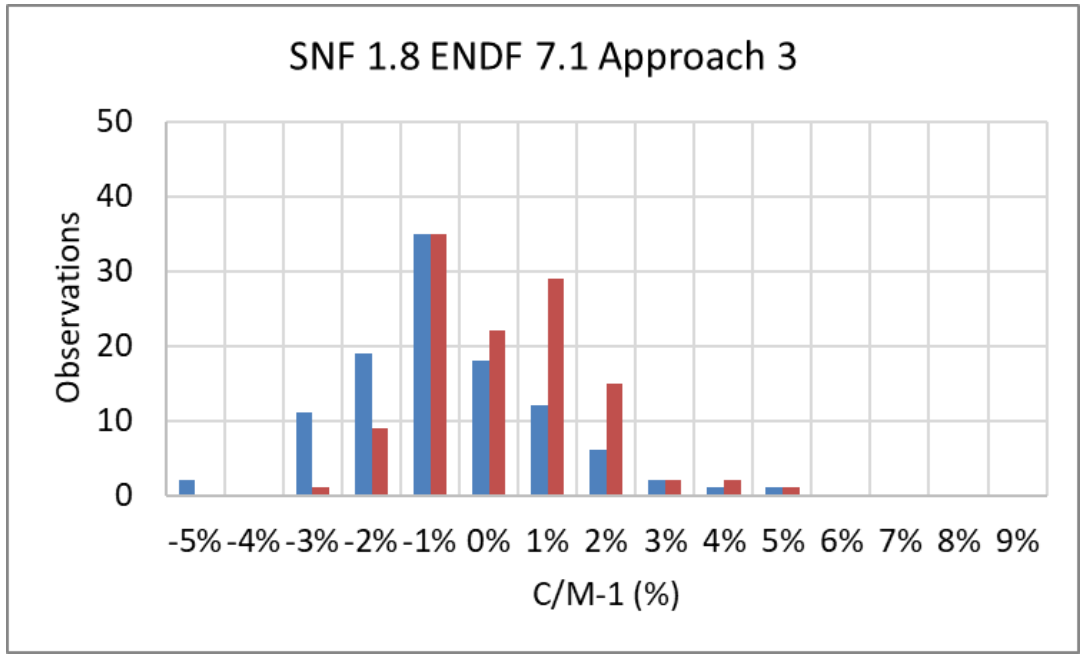
Modeling Detail Comparison

(Compare Approach 2; detailed 3-D history vs Approach 3; 2-D cycle-average)

- 108 BWR + 116 PWR measurements
- Same library (ENDF), different modeling detail







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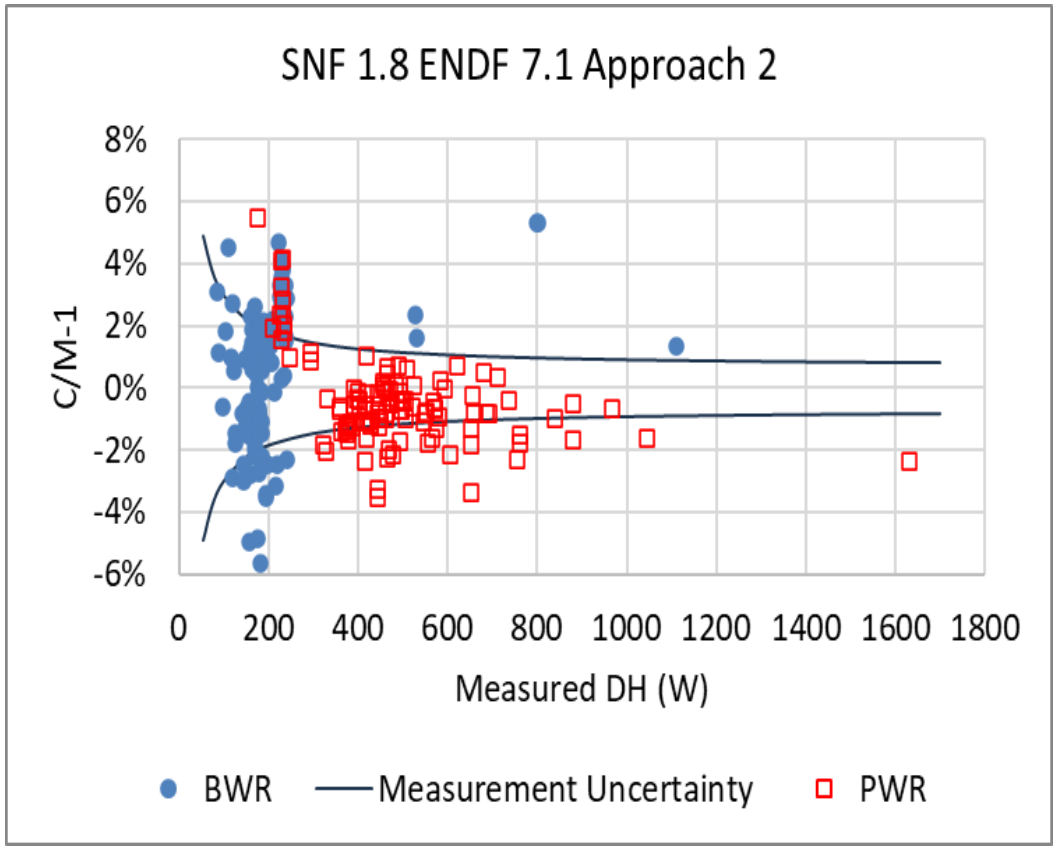
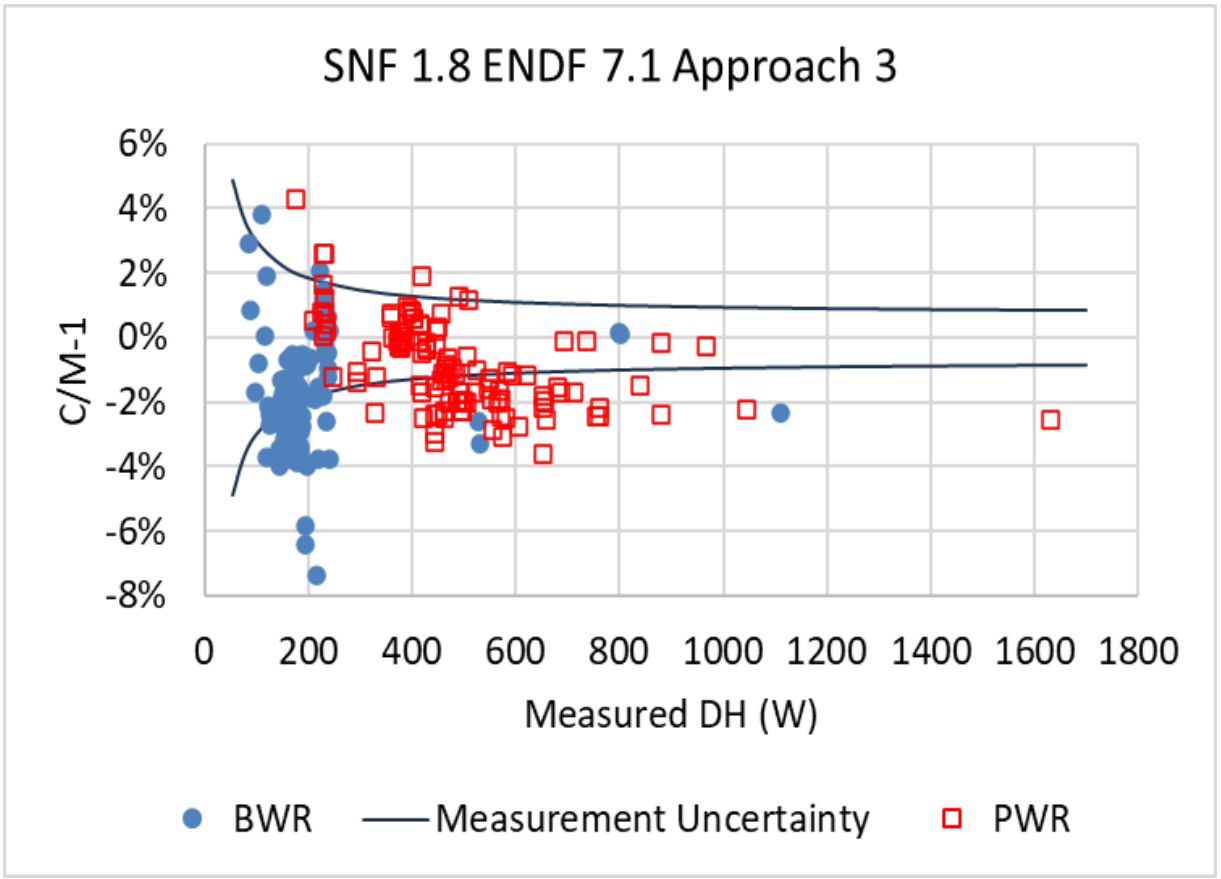
Detailed 3-D history



The uncertainty bands bound 56% of the C/M differences

2-D cycle-average (Master)



The uncertainty bands bound 46% of the C/M differences

With reduced modeling detail, even with significantly negative population bias, 98% of the Method 3 C/M population is within ±4%.

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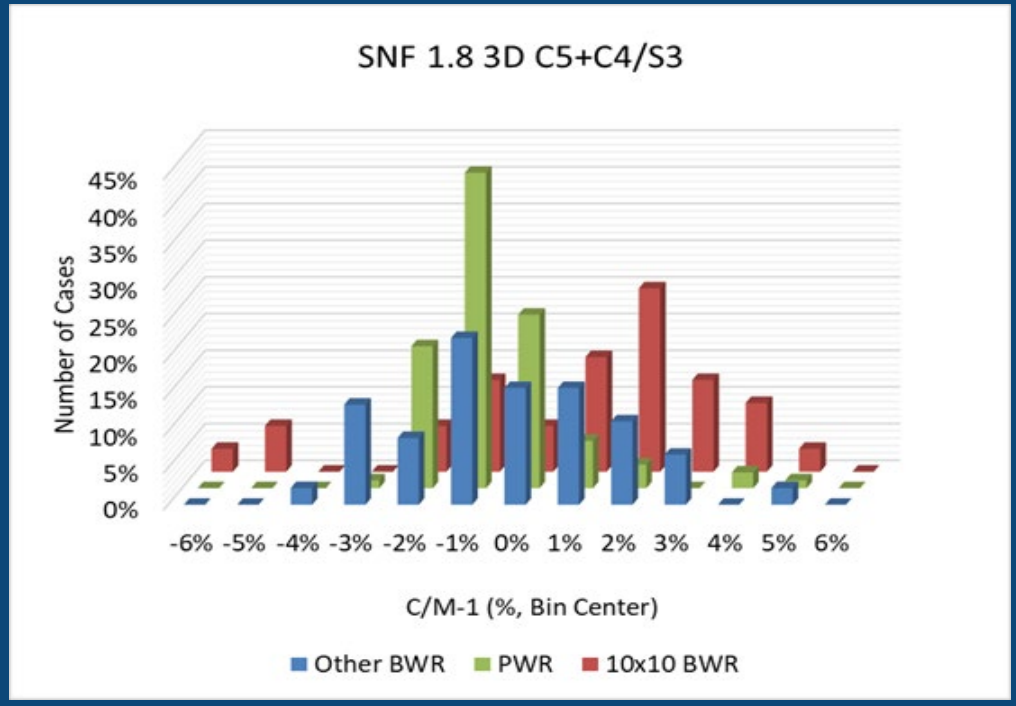
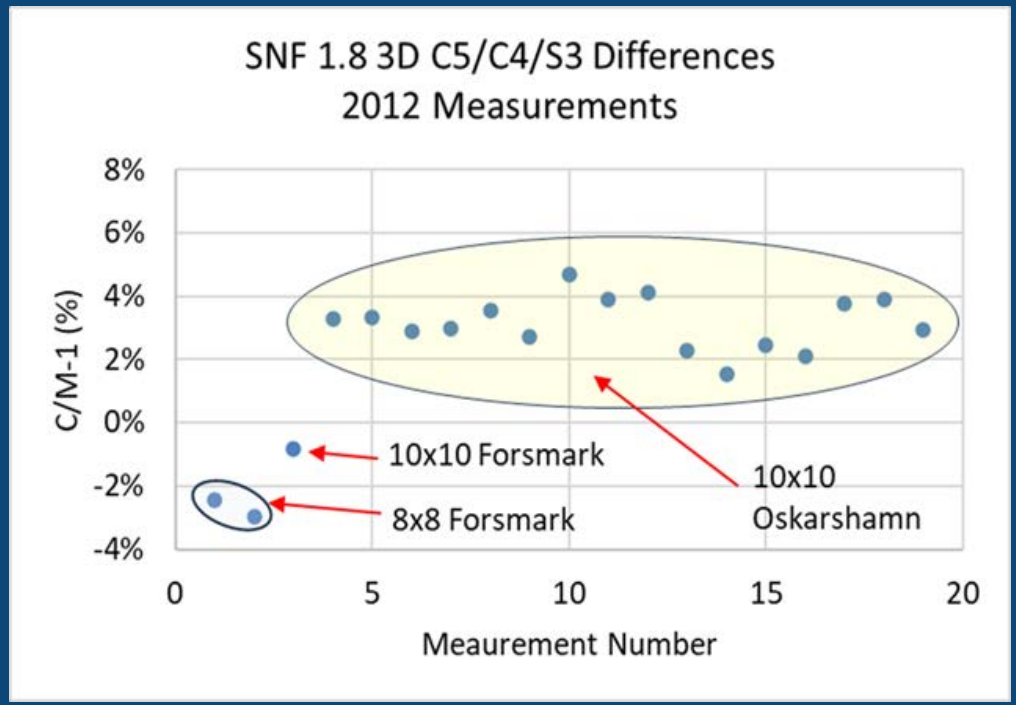
BWR Assembly Differences

When breaking down the results by assembly type, we see clear differences among BWRs.

The majority of cases correspond to O3 assemblies, originating from the same cycle and representing the first Svea-96 type, located in rim or semi-rim positions during the last cycle — likely contributing to the observed spread.

Forsmark 8×8 and 10×10 assemblies show a small, consistent deviation (≈ −2 to −3 %), while Oskarshamn 10×10 assemblies cluster closer to 4%.





Summary of Preliminary SNF Results

- In general SNF calculations shows good agreement with measurements.
- There are, however, many factors in the calculations that can affect the results, such as code version, cross-section libraries, nuclear data, incomplete handling of fuel or cycle data.
- These factors have to be carefully considered in the validation work.
- More than 95% of C/E are within 4%.
- 2D input gives reasonable results, but with a negative bias, especially evident for BWR.
- Difference between 10x10 and 8x8 BWR needs further investigations.

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Summary and Ongoing Activities

EPRI-SKB Collaboration: Summary and Forward Looking

Evaluation of Clab Decay Heat Measurements

- All measurements, from 2003-2021, evaluated for quality
- Developed screening criteria for data quality, which will be beneficial in future campaigns
- Excluded 6 calibration and 5 fuel assembly measurements
- Measurement uncertainty below 1% (2 sigma) for high decay heat
  - Significant gain in operational margins for entire back-end
- EPRI report reviewed by experts from global community and published in October 2024
- A journal article is published in Progress in Nuclear Energy

Validation of Clab Decay Heat Measurements

- Calculations using SNF, ORIGEN, Polaris
- Evaluating sensitivity to cross section libraries
- Evaluating sensitivity to cooling time
- Evaluating sensitivity to model details
- Validation report will include key lessons learned and will make recommendations
- Validation report will be published in 2026 and will be publicly available

2026 and Beyond

- Finishing calorimeter upgrades
- Performing repeat measurements
  - Using developed screening metrics for data quality
- Evaluation of repeat measurements
- Performing measurements – targeted sampling to close technical gaps
- Validation of new measurements
- Publicly available EPRI report for new measurements and validation results

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Questions/Comments?

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October 29, 2025

Kaushik Banerjee
Senior Technical Advisor
EPRI ESCP 2025 Meeting

EPRI Modeling and Benchmarking Subcommittee: Radiation dose modeling

Battelle Energy Alliance manages INL for the
U.S. Department of Energy's Office of Nuclear Energy

Idaho National Laboratory

Shielding model benchmarking by comparing with measured dose rates

- The objective of this task group is to perform dose analysis of spent nuclear fuel systems using code of choice, compare results, and compare results with the measured dose values
  - Not many spent fuel dose benchmarks available
  - Generally, dose calculations are conservative
    - Comparing with measurements can provide insights on dose margin

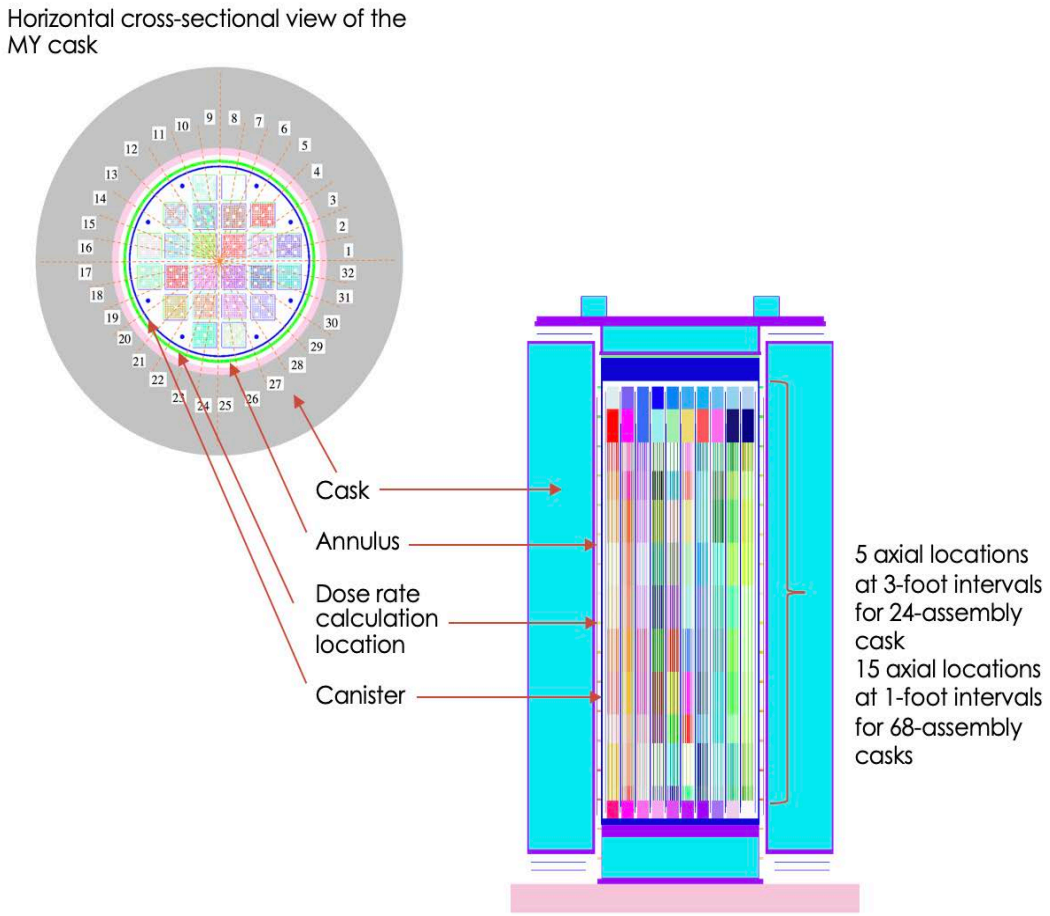
A loaded Holtec International's MPC-68 canister inside a storage overpack (HI-STORM 100) has been selected for this exercise

				1 19950 9/22/79	2 20543 9/16/78				
		3 20581 9/16/78	4 20449 9/22/79	5 20469 9/22/79	6 20088 9/22/79	7 19923 9/22/79	8 20334 9/22/79		
	9 20425 9/22/79	10 20497 9/22/79	11 32777 9/6/96	12 32777 9/6/96	13 32873 9/6/96	14 32784 9/6/96	15 20055 9/22/79	16 20308 9/22/79	
	17 20409 9/22/79	18 32791 9/6/96	19 32934 9/6/96	20 35405 8/27/93	21 35510 8/27/93	22 35376 8/27/93	23 32825 9/6/96	24 20457 9/22/79	
25 19952 9/22/79	26 20176 9/16/78	27 32896 9/6/96	28 35540 8/27/93	29 33050 9/6/96	30 33046 9/6/96	31 33046 8/27/93	32 35266 8/27/93	33 20487 9/22/79	34 19931 9/22/79
35 19951 9/22/79	36 20448 9/22/79	37 35314 8/27/93	38 32945 9/6/96	39 33047 9/6/96	40 33042 9/6/96	41 35456 35456	42 32873 9/6/96	43 20824 9/16/78	44 20769 9/22/79
	45 20436 9/22/79	46 32798 9/6/96	47 32913 9/6/96	48 35552 8/27/93	49 32922 9/6/96	50 35356 8/27/93	51 32785 9/6/96	52 20441 9/22/79	Drain tube
	53 20835 9/22/79	54 20044 9/22/79	55 35160 8/27/93	56 32869 9/6/96	57 35300 8/27/93	58 32810 9/6/96	59 19959 9/22/79	60 20631 9/16/78	
		61 19839 10/16/81	62 19991 9/22/79	63 20840 9/16/78	64 20469 9/22/79	65 20425 9/22/79	66 20318 9/22/79		
				67 19934 9/22/79	68 20904 9/22/79				

BU ~ 20 GWd/MTU
Cooling time ~ 39 & 40 years
32.8 < BU < 35.6 GWd/MTU
Cooling time ~ 22 & 25 years

Benchmark description

- Data provided
  - Loading maps with assembly characteristics at each location inside the canister including assembly type, burnup, initial enrichment, cooling time, and assembly condition
  - Depletion history (unknown)
    - Participants were recommended to use specific SCALE reactor libraries
  - Burnup profiles (not actual as actual unknown)
  - Co-59 content of non-fuel hardware regions (not actual as actual unknown)
  - Publicly available information for modeling the canisters and overpacks
  - Dose measurement locations and dates
  - Measured dose rates will be provided later (blind benchmarking)



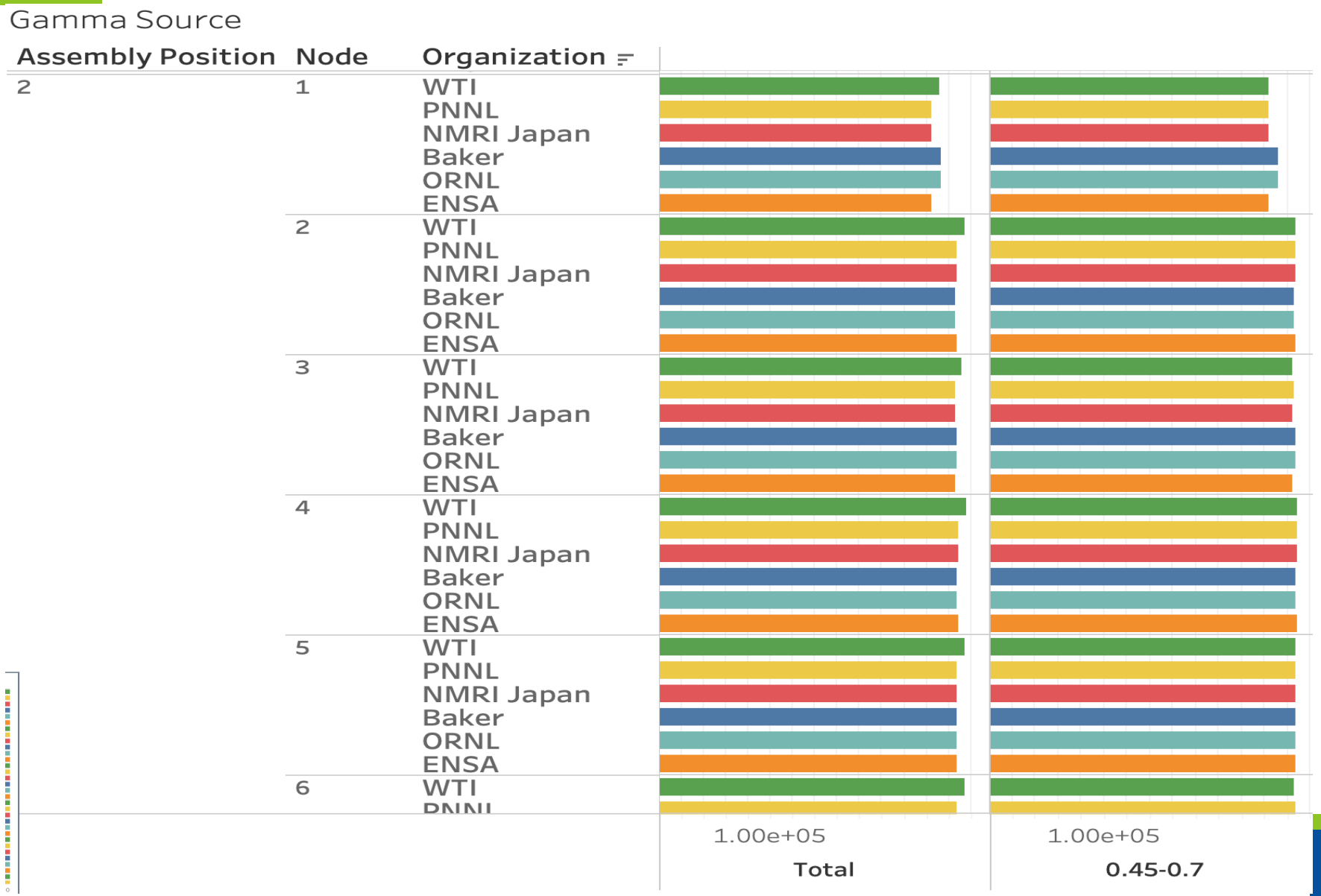
Loading map example

Position	Assembly ID	Assembly Type	Initial Enrichment	Initial Uranium (kg)	Discharge Burnup (MWD/MTU)	Cycle Start Date	Cycle End Date
1	325313	G4608G4B	2.2	183.388	19950	12/12/74	6/19/76
1	325313	G4608G4B	2.2	183.388	19950	8/7/76	8/20/77
1	325313	G4608G4B	2.2	183.388	19950	4/3/79	9/22/79
2	325196	G4608G4B	2.2	183.536	20543	12/12/74	6/19/76
2	325196	G4608G4B	2.2	183.536	20543	10/5/77	9/16/78
3	325163	G4608G4B	2.2	183.346	20581	12/12/74	6/19/76
3	325163	G4608G4B	2.2	183.346	20581	10/5/77	9/16/78
4	325305	G4608G4B	2.2	183.501	20449	12/12/74	6/19/76
4	325305	G4608G4B	2.2	183.501	20449	8/7/76	8/20/77
4	325305	G4608G4B	2.2	183.501	20449	4/3/79	9/22/79
5	325336	G4608G4B	2.21	183.813	20469	12/12/74	6/19/76
5	325336	G4608G4B	2.21	183.813	20469	8/7/76	8/20/77
5	325336	G4608G4B	2.21	183.813	20469	10/10/78	3/17/79
5	325336	G4608G4B	2.21	183.813	20469	4/3/79	9/22/79
6	325260	G4607G3B	2.31	188.431	20088	11/18/73	10/12/74
6	325260	G4607G3B	2.31	188.431	20088	12/12/74	6/19/76
6	325260	G4607G3B	2.31	188.431	20088	8/7/76	8/20/77
6	325260	G4607G3B	2.31	188.431	20088	10/5/77	9/16/78
6	325260	G4607G3B	2.31	188.431	20088	10/10/78	3/17/79
6	325260	G4607G3B	2.31	188.431	20088	4/3/79	9/22/79
7	325272	G4607G3B	2.31	188.402	19923	11/18/73	10/12/74
7	325272	G4607G3B	2.31	188.402	19923	12/12/74	6/19/76
7	325272	G4607G3B	2.31	188.402	19923	8/7/76	8/20/77

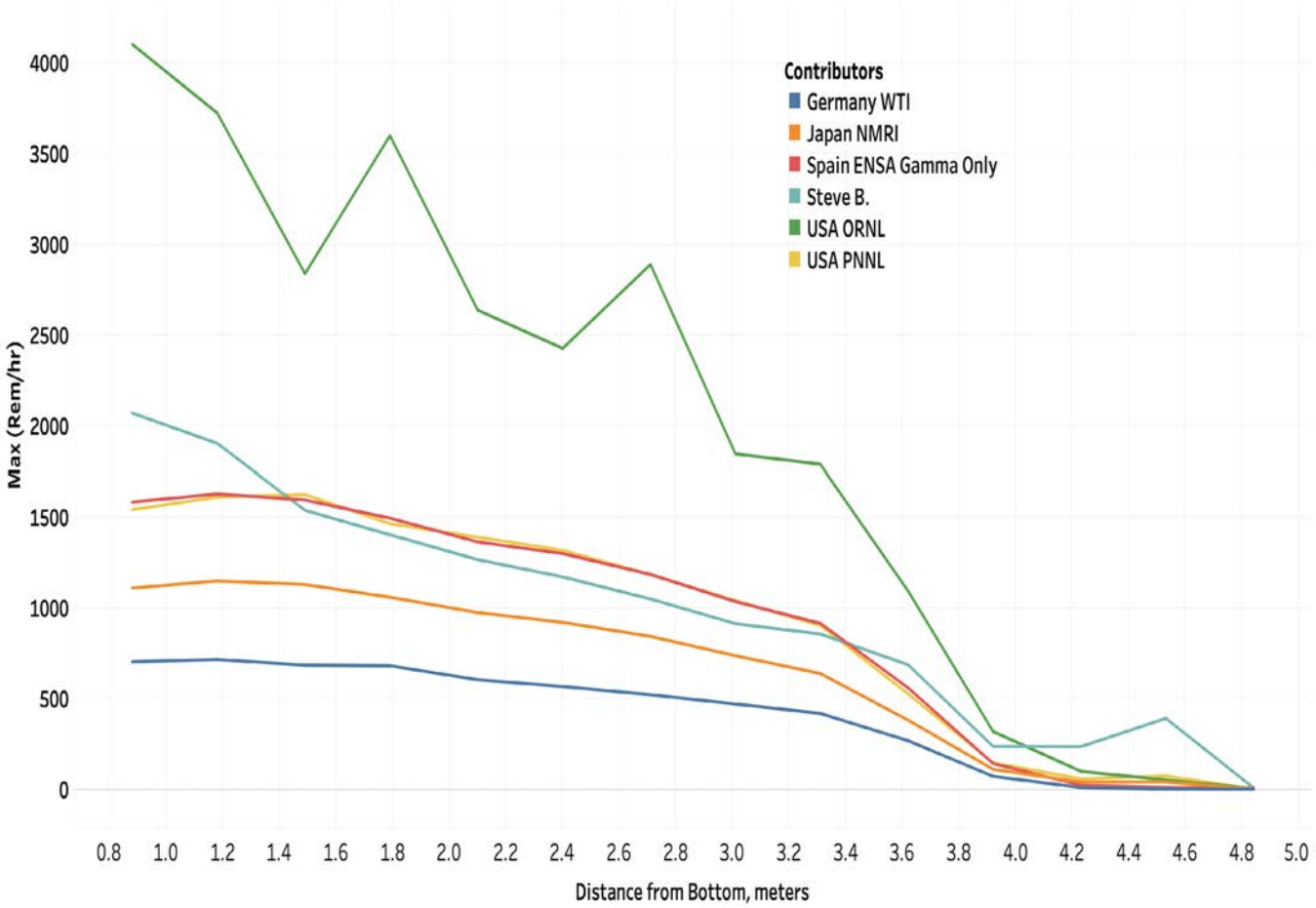
Task group participants

- Participants
  - USA: ORNL, PNNL, Steve Baker
  - Japan: NMRI
  - Germany: WTI/BGZ
  - Spain: ENSA

Status: source calculations and comparison



Status: Dose calculation



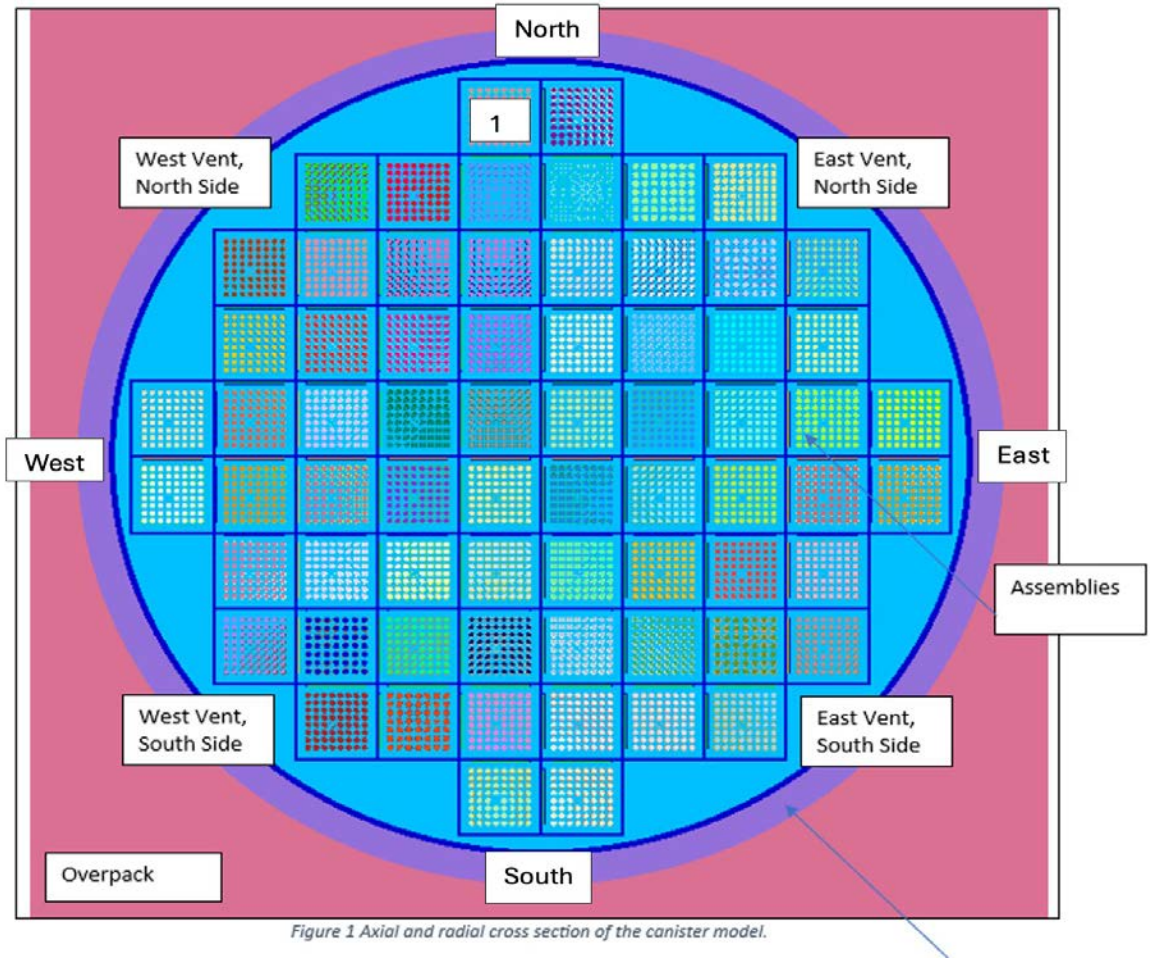
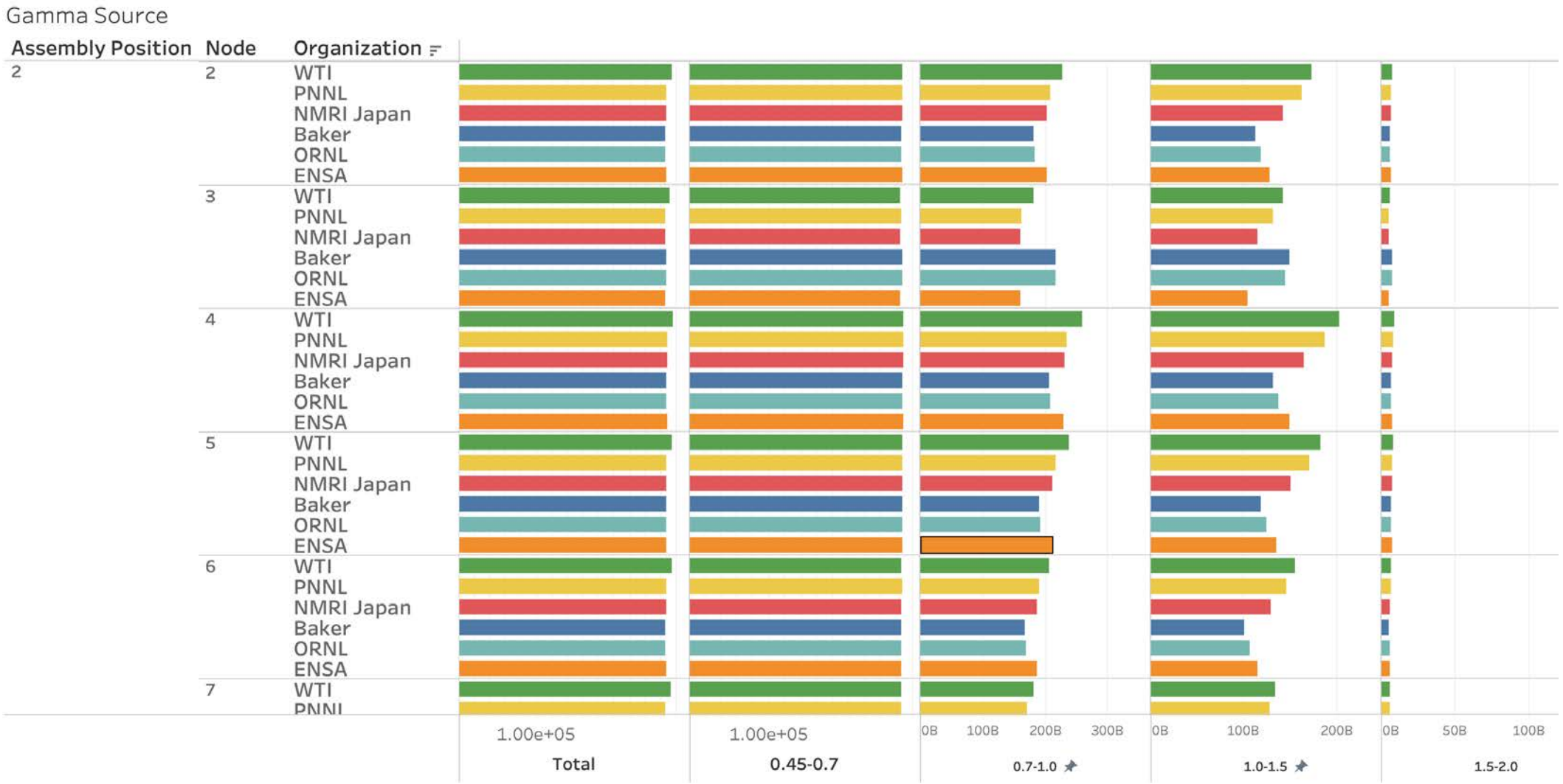


Figure 1 Axial and radial cross section of the canister model.

Source comparison with additional gamma groups



Next steps

- Understanding the differences in source terms (0.7-1.0, and 1.0-1.5 MeV groups)
  - May be due to the differences in the non-fuel composition?
- Understanding the uncertainties in the results
- Sensitivity analysis
- Expected timeline
  - Try to understand and resolve the differences (By end of CY 2025)
  - Start Compiling a report (Ongoing)
  - Publish report (Q1 CY2026)

Idaho National Laboratory

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ESCP International Thermal Modeling Project Updates

Hatice Akkurt[1] and Maik Stuke[2]
[1]Used Fuel and High-Level Waste Management, EPRI
[2]BGZ

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 27-30, 2025

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Acknowledgements

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Gesellschaft für Anlagen- und Reaktorsicherheit (GRS, Germany)	Marc Péridis, Daniel Nahm*
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Thank you to all the participants and participating organizations!

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ESCP International Thermal Modeling

Key Research Questions

1. How accurately are thermal modeling codes predicting temperatures – **accuracy of predictions**?

2. How do the estimated results change between different codes (**code-to-code variation**)?

3. Are there any variations between different users and organizations when the same code(s) are used (**user-to-user variation**)?

4. What parameters are important for accurate estimation of the results – **evaluation of the sensitivity to input parameters**?

5. What is the **impact of using only *publicly available* system design data** and approximations/estimations on the estimated temperature values?

EPRI ESCP report, 3002032045, published on October 23, 2025 and the report is publicly available

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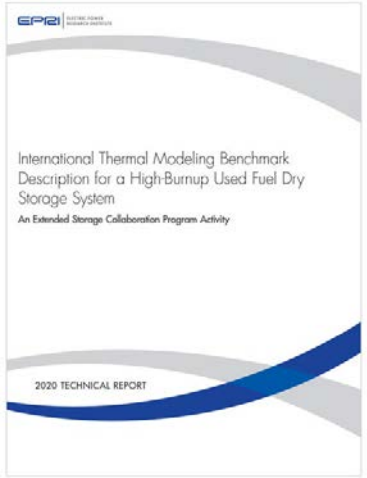
EPRI report, **3002018498**, provides a description of the benchmark:

- Based on publicly available information
- Includes a recording of the description
- Participants were asked to follow description for easier comparison of the results
- Report is publicly available from epri.com



Participating Organizations and Codes

Solution Method	Code(s)	Organization(s)
Finite Element Method (FEM)	ANSYS Mechanical	CNAT ENSA WTI
	ABAQUS	UJV
	SYRTHES	EDF
Finite Volume Method (FVM)	ANSYS Fluent	UNR
	GOTHIC	CNAT
	STAR-CCM	PNNL
Finite Difference Method (FDM)	COBRA-SFS	GRS PNNL BGZ
FEM & FVM	MOOSE	INL

10 organizations, 8 codes, and 12 submissions with 4 different solution approaches

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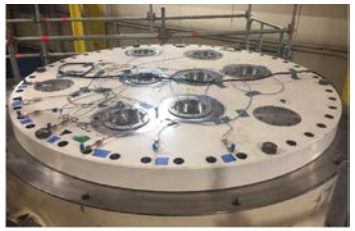
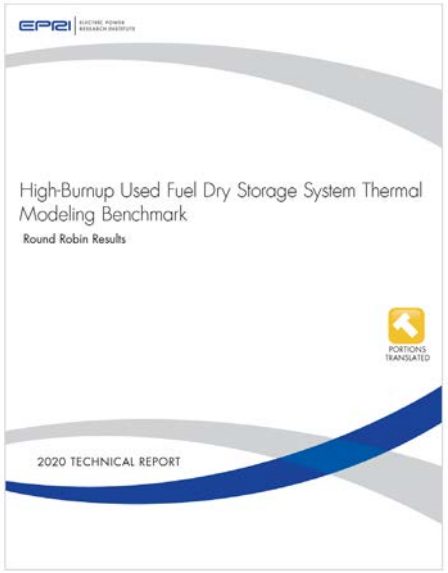
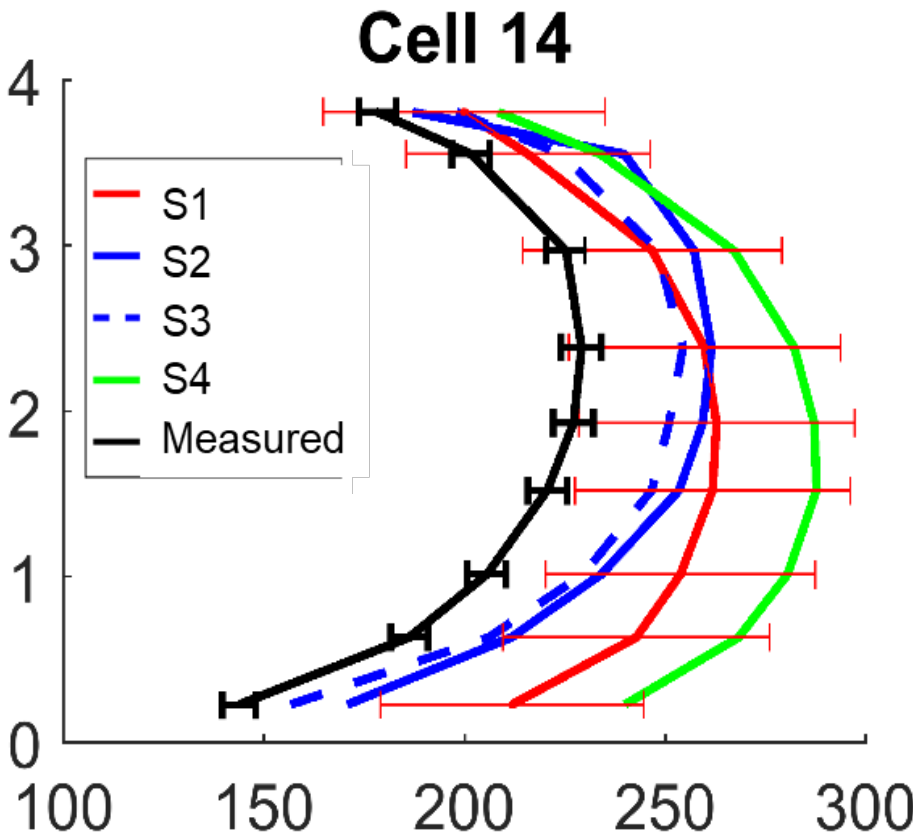
HBU Demo Showed Temperature Predictions are Overestimated



Parameter	FSAR	LAR	Best-Estimate	HBU Cask Meas.
PCT	348°C	318°C	254-288°C	229°C
Total Heat Load (kW)	36.96	32.934	30.456	30.456
Ambient Temperature	100°F	93.5°F	75°F	75°F
Design Specifics	Gaps	Gaps	Gaps	No Gaps?



EPRI report, **3002013124,** published in April 2020 and publicly available – includes blind benchmark results



Modeler	Code
S1	ANSYS Fluent
S2	STAR-CCM+
S3	COBRA
S4	ANSYS APDL

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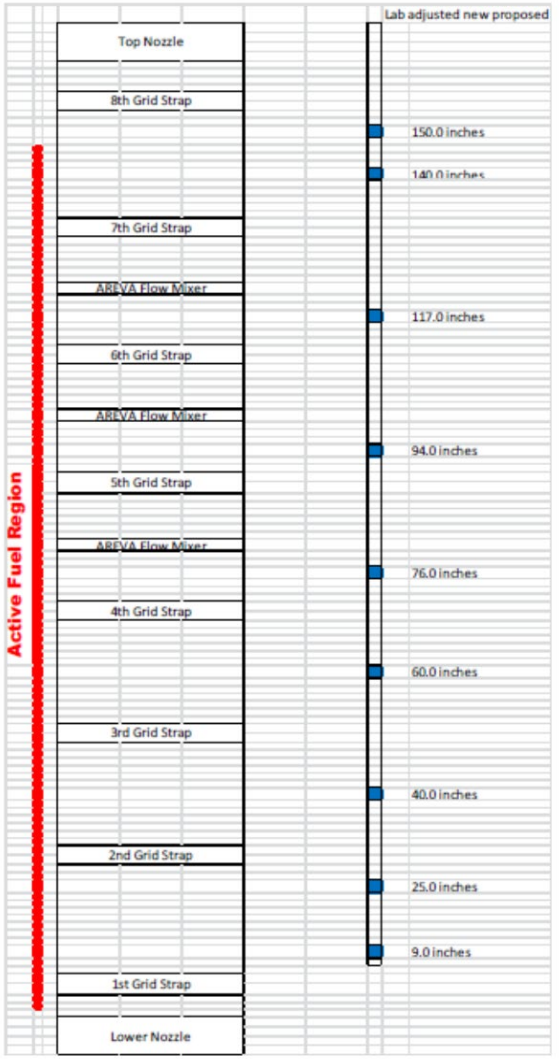
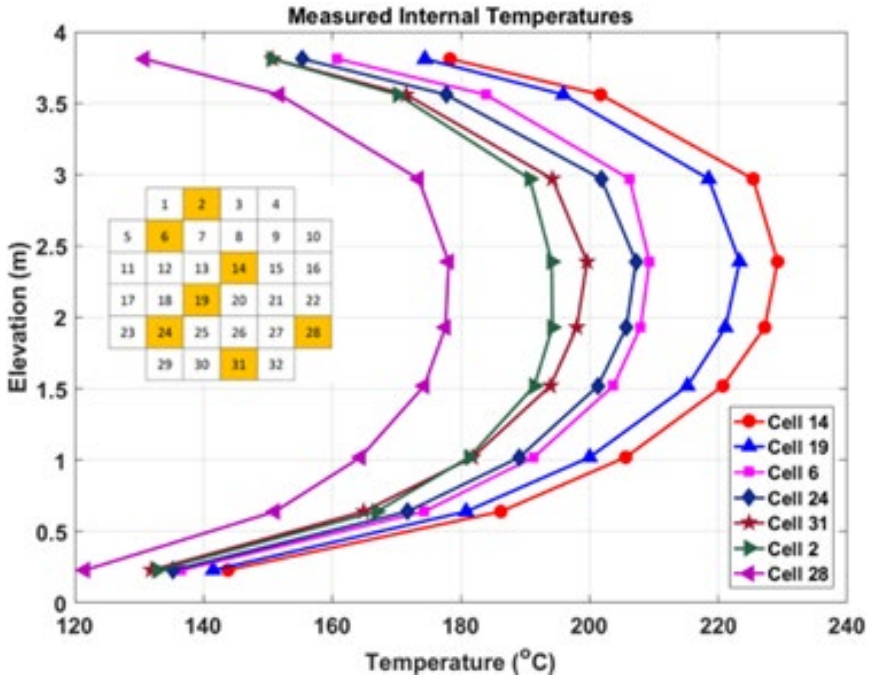
Fuel Loading Map & Thermocouple Locations

	1 6T0 NAIF/P+Z Zirlo, 54.2 GWd 4.25%/3 cy/12.1 yr 912.2 W	2 (TC Lance) 3K7 AMBW M5, 53.4 GWd 4.55%/3 cy/8.7 yr 978.2 W	3 3T6 NAIF/P+Z Zirlo, 54.3 GWd 4.25%/3 cy/12.1 yr 914.4 W	4 6F2 NAIF/P+Z Zirlo, 51.9 GWd 4.25%/3 cy/13.5 yr 799.5 W	Drain Port
5 3F6 NAIF/P+Z Zirlo, 52.1 GWd 4.25%/3 cy/13.5 yr 800.9 W	6 (TC Lance) 30A AMBW M5, 52.0 GWd 4.55%/3 cy/7.2 yr 1008.6 W	7 22B AMBW M5, 51.2 GWd 4.55%/3 cy/5.7 yr 1142.4 W	8 (PRA) 20B AMBW M5, 50.5 GWd 4.55%/3 cy/5.7 yr 1121.2 W	9 5K6 AMBW M5, 53.3 GWd 4.55%/3 cy/8.7 yr 975.1 W	10 5D5 NAIF/P+Z Zirlo, 55.5 GWd 4.20%/3 cy/17.7 yr 814.5 W
11 (Vent Port) 5D9 NAIF/P+Z Zirlo, 54.6 GWd 4.20%/3 cy/17.7 yr 802.6 W	12 28B AMBW M5, 51.0 GWd 4.55%/3 cy/5.7 yr 1135.0 W	13 (PRA) F40 LOPAR Zry-4, 50.6 GWd 3.59%/3 cy/30.6 yr 573.8 W	14 (TC Lance) 57A AMBW M5, 52.2GWd 4.55%/3 cy/7.2 yr 1037.0 W	15 (PRA) 30B AMBW M5, 50.6 GWd 4.55%/3 cy/5.7 yr 1124.8 W	16 3K4 AMBW M5, 51.8 GWd 4.55%/3 cy/8.7 yr 941.3 W
17 5K7 AMBW M5, 53.3 GWd 4.55%/3 cy/8.7 yr 961.7 W	18 (PRA) 50B AMBW M5, 50.9 GWd 4.55%/3 cy/5.7 yr 1131.1 W	19 (TC Lance) 3U9 NAIF/P+Z Zirlo, 53.1 GWd 4.45%/3 cy/10.6 yr 920.2 W	20 (PRA) 0A4 NAIF Low-Sn Zry-4, 50.0 GWd 4.00%/2 cy/23.2 yr 646.2 W	21 15B AMBW M5, 51.0 GWd 4.55%/3 cy/5.7 yr 1135.8 W	22 6K4 AMBW M5, 51.9 GWd 4.55%/3 cy/8.7 yr 941.2 W
23 3T2 NAIF/P+Z Zirlo, 55.1 GWd 4.25%/3 cy/12.1 yr 934.7 W	24 (TC Lance) 3U4 NAIF/P+Z Zirlo, 52.9 GWd 4.45%/3 cy/10.6 yr 914.2 W	25 (PRA) 56B AMBW M5, 51.0 GWd 4.55%/3 cy/5.7 yr 1133.7 W	26 54B AMBW M5, 51.3 GWd 4.55%/3 cy/5.7 yr 1136.3 W	27 6V0 AMBW M5, 53.5 GWd 4.40%/3 cy/8.7 yr 988.2 W	28 (TC Lance) 3U6 NAIF/P+Z Zirlo, 53.0 GWd 4.45%/3 cy/10.6 yr 916.9 W
	29 4V4 AMBW M5, 51.2 GWd 4.40%/3 cy/9.1 yr 914.2 W	30 5K1 AMBW M5, 53.0 GWd 4.55%/3 cy/8.7 yr 968.0 W	31 (TC Lance) 5T9 NAIF/P+Z Zirlo, 54.9 GWd 4.25%/3 cy/12.1 yr 927.7 W	32 4F1 NAIF/P+Z Zirlo, 52.3 GWd 4.25%/3 cy/13.5 yr 804.3 W	





Temperature measurement locations: Cells 2, 6, 14, 19, 24, 28, 31

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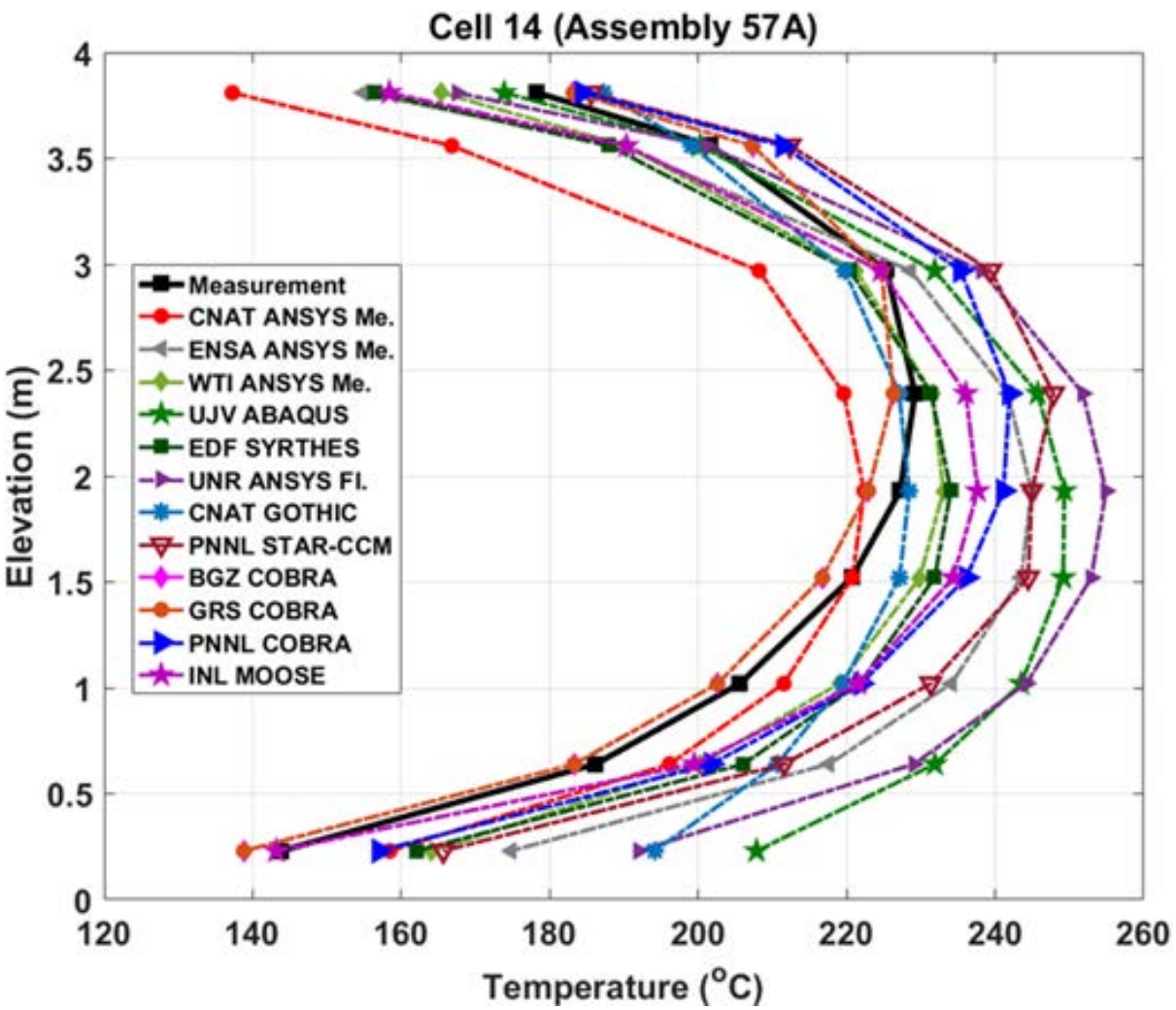
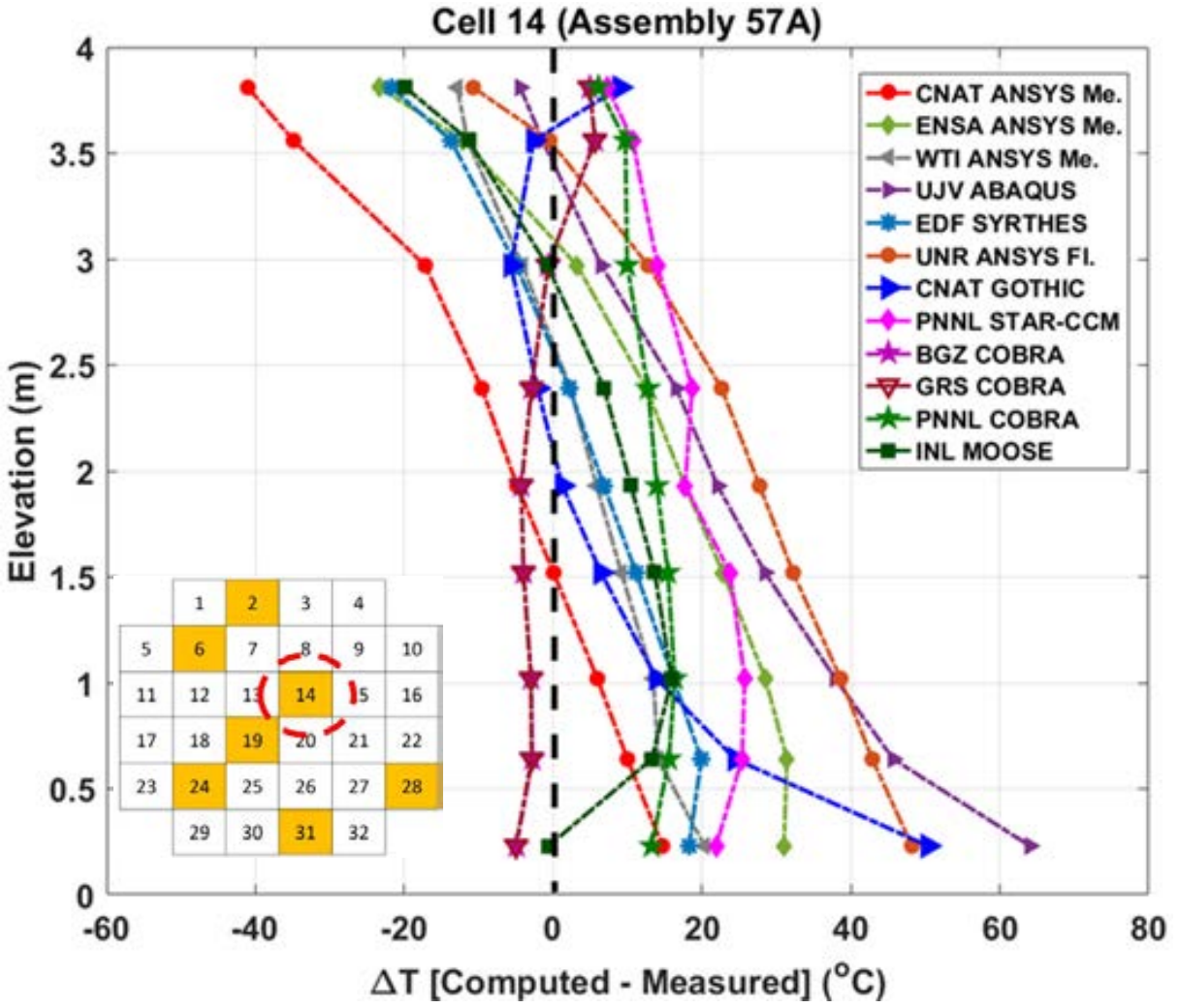
Cell 14 - Interior Cell





Shape is in general predicted well; however, mostly underprediction for top and overprediction for bottom

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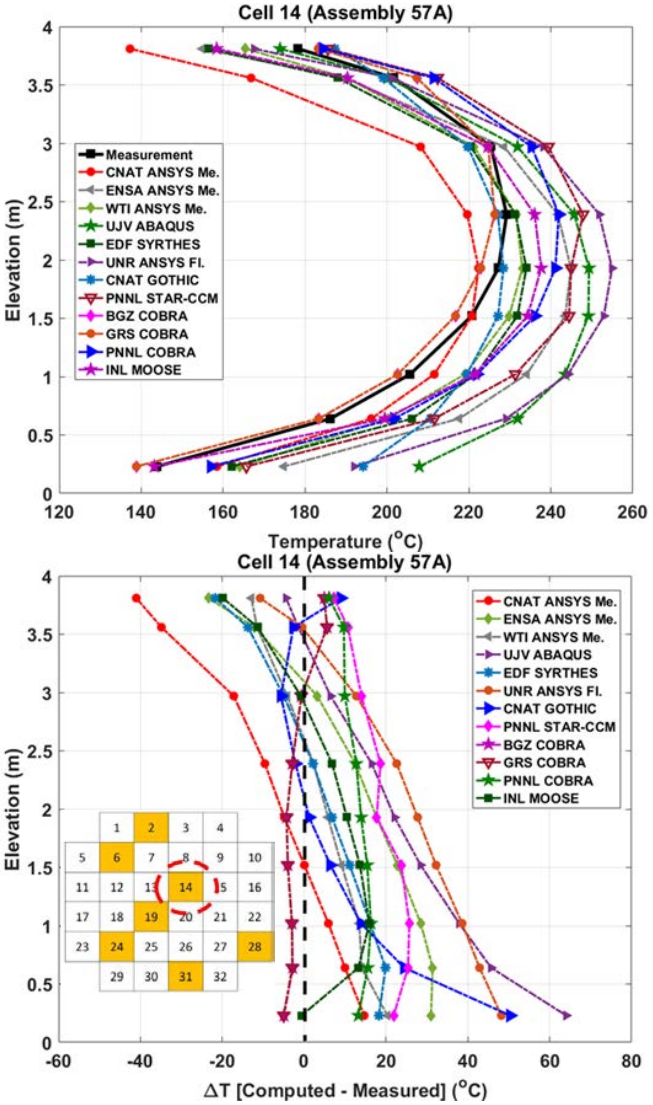
Cell 14 - Interior Cell



Submission	Min ΔT (°C) [Computed – Measured]	Max ΔT (°C) [Computed – Measured]
CNAT ANSYS Mechanical	-41.0	+17.0
ENSA ANSYS Mechanical	-23.4	+31.4
WTI ANSYS Mechanical	-12.9	+20.3
UJV ABAQUS	-4.4	+64.1
EDF SYRTHES	-21.7	+19.9
UNR ANSYS Fluent	-10.8	+48.2
CNAT GOTHIC	-5.7	+50.4
PNNL STAR-CCM+	+7.3	+25.7
BGZ COBRA-SFS	-4.9	+5.6
GRS COBRA-SFS	-5.0	+5.6
PNNL COBRA-SFS	+6.1	+16.3
INL-MOOSE	-19.9	+15.9

Shape is in general predicted well; however, mostly underprediction for top and overprediction for bottom

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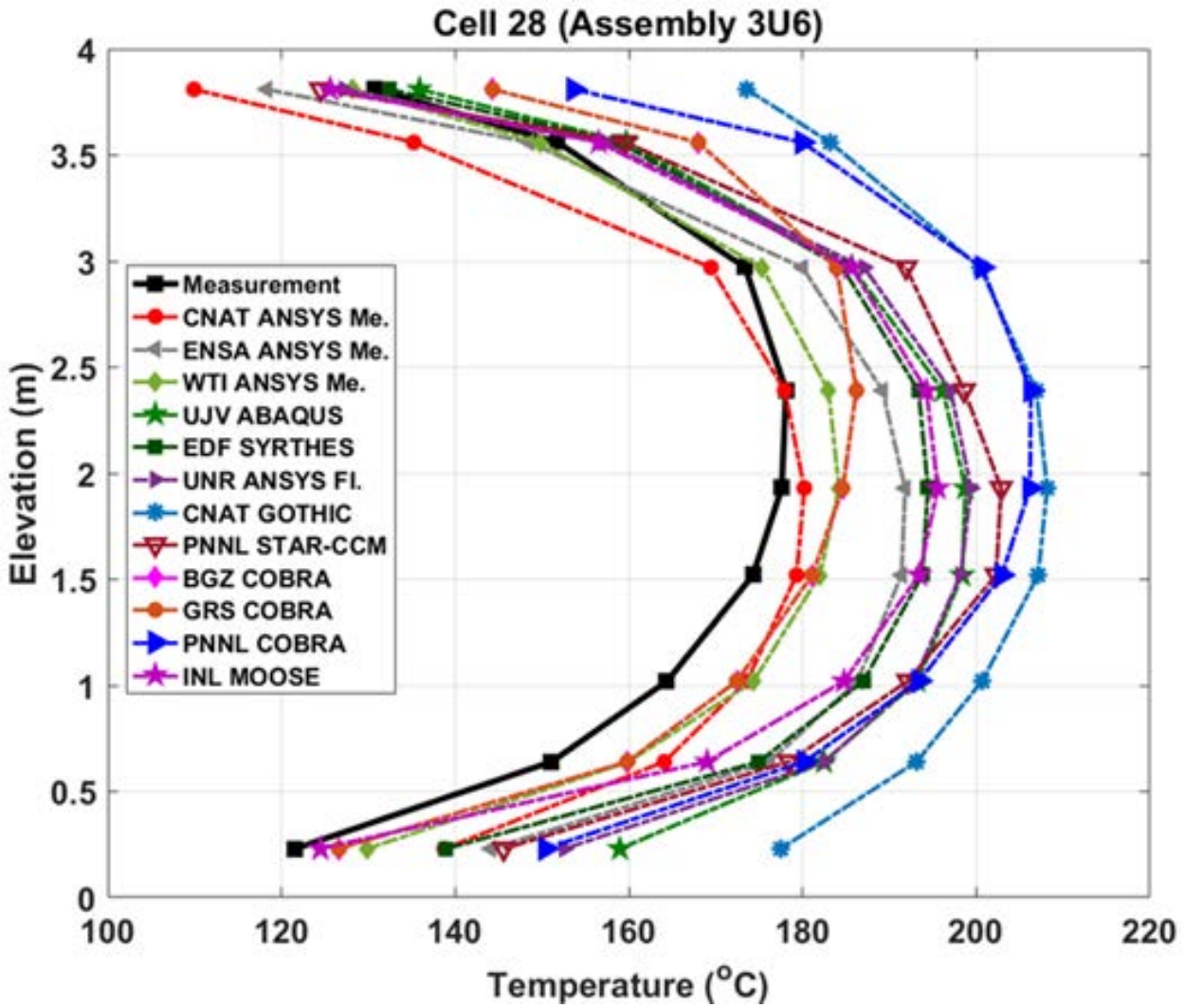
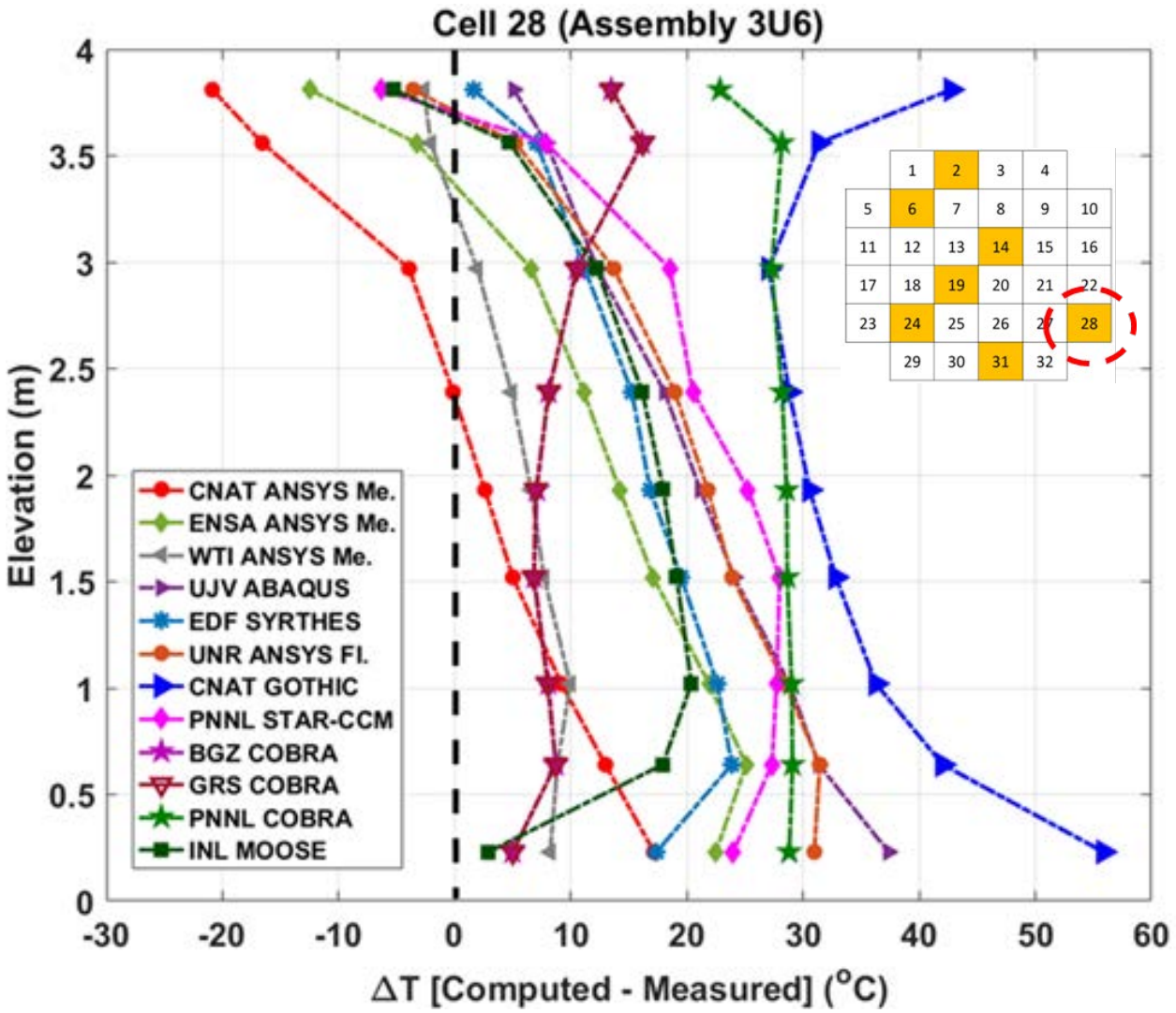
Cell 28 – Exterior Cell





Mostly overprediction for exterior cell

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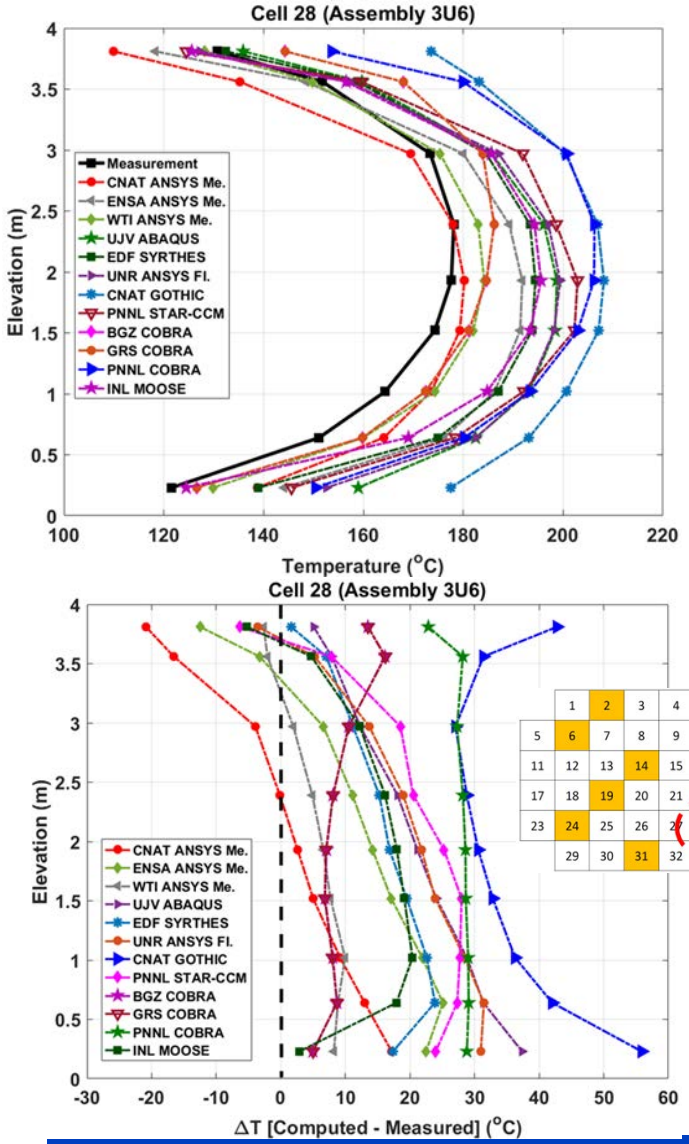
Cell 28 - Interior Cell



Submission	Min ΔT (°C) [Computed – Measured]	Max ΔT (°C) [Computed – Measured]
CNAT ANSYS Mechanical	-20.9	+17.2
ENSA ANSYS Mechanical	-12.5	+25.1
WTI ANSYS Mechanical	-2.6	+9.9
UJV ABAQUS	+5.1	+37.3
EDF SYRTHES	+1.6	+23.9
UNR ANSYS Fluent	-3.6	+31.4
CNAT GOTHIC	+27.1	+55.9
PNNL STAR-CCM+	-6.3	+28.0
BGZ COBRA-SFS	+5.0	+16.1
GRS COBRA-SFS	+5.0	+16.2
PNNL COBRA-SFS	+22.8	+29.1
INL-MOOSE	-5.1	+20.4

Mostly overprediction for exterior cell

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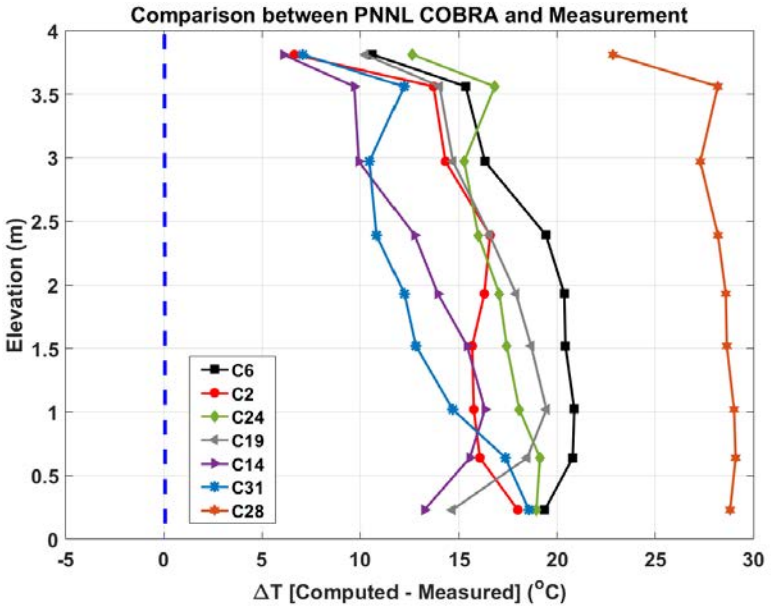
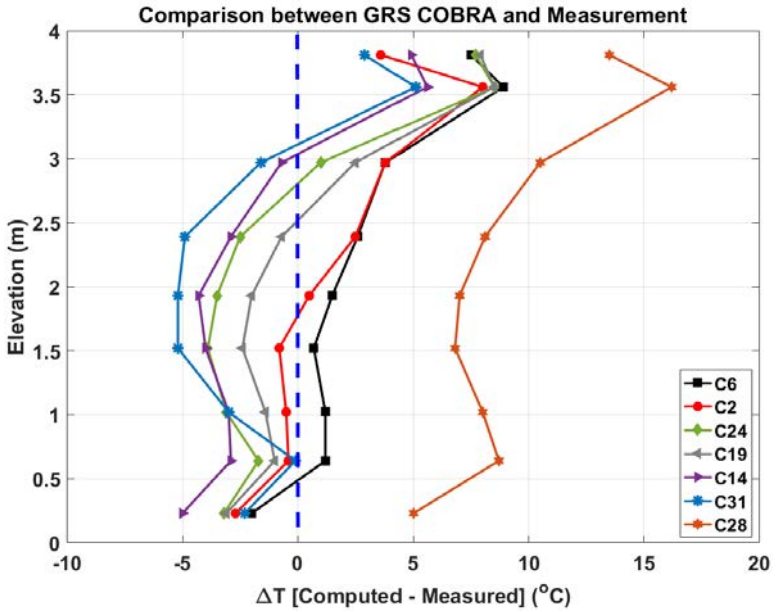
Same Code, Different Users: COBRA-SFS

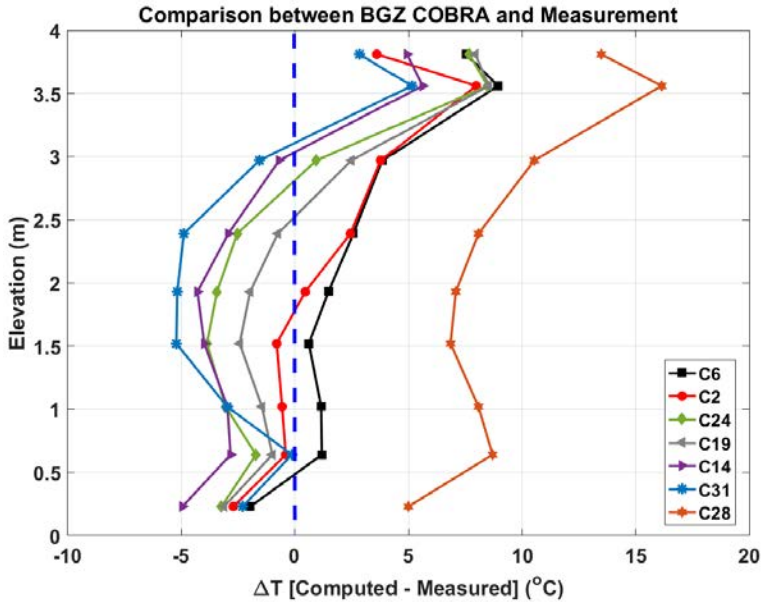
	1	2	3	4	
5	6	7	8	9	10
11	12	13	14	15	16
17	18	19	20	21	22
23	24	25	26	27	28
	29	30	31	32	



PNNL Results



GRS Results



BGZ Results

PNNL results show consistent overprediction while BGZ and GRS results, mostly, shows mixture, except Cell 28. BGZ and GRS COBRA-SFS results are almost identical.

For all three organizations, Cell 28 is significantly overestimated.

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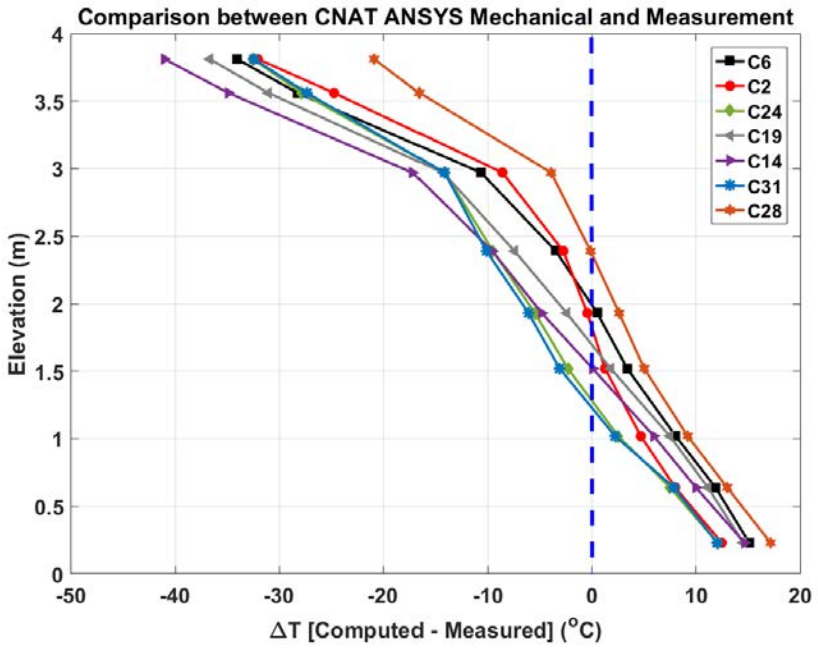
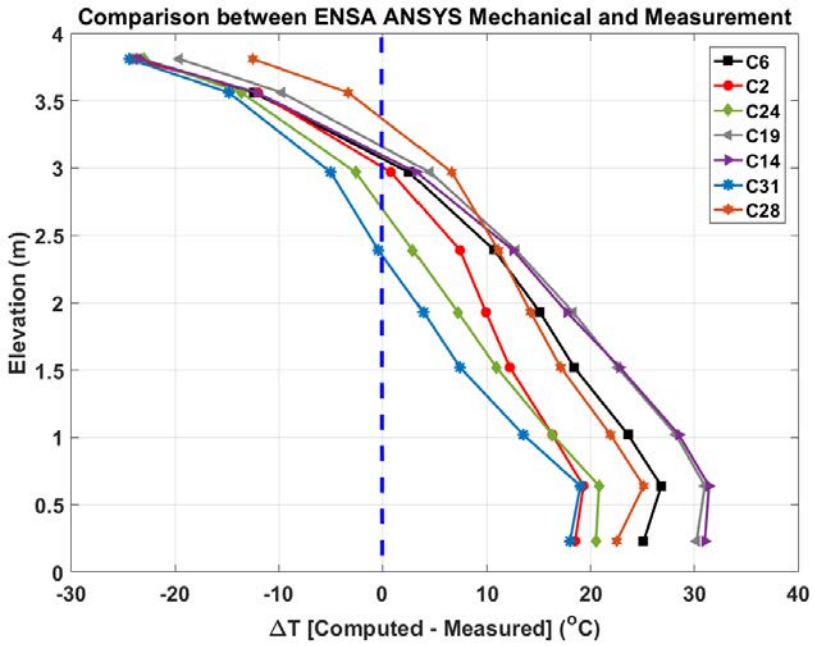
Same Code, Different Users: ANSYS Mechanical

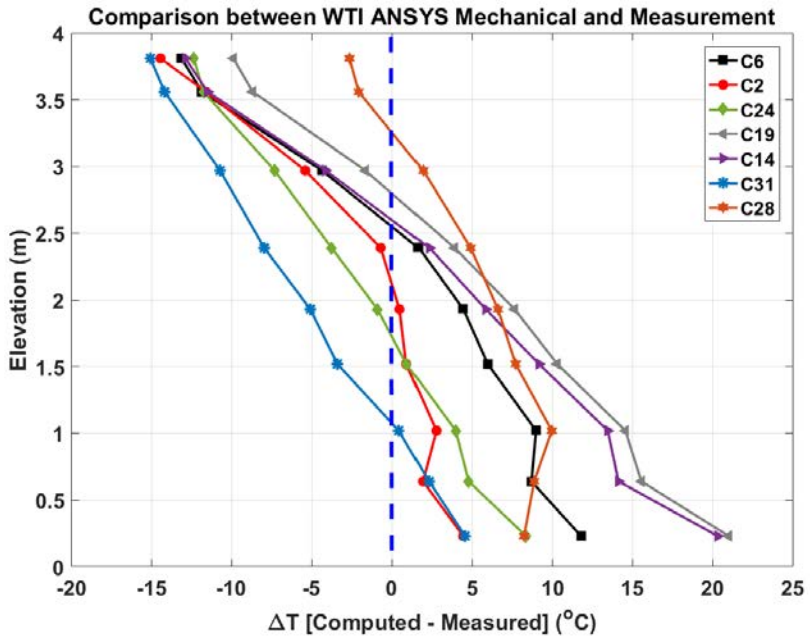
	1	2	3	4	
5	6	7	8	9	10
11	12	13	14	15	16
17	18	19	20	21	22
23	24	25	26	27	28
	29	30	31	32	



CNAT Results



ENSA Results



WTI Results

Varying results between different organizations using the same code.

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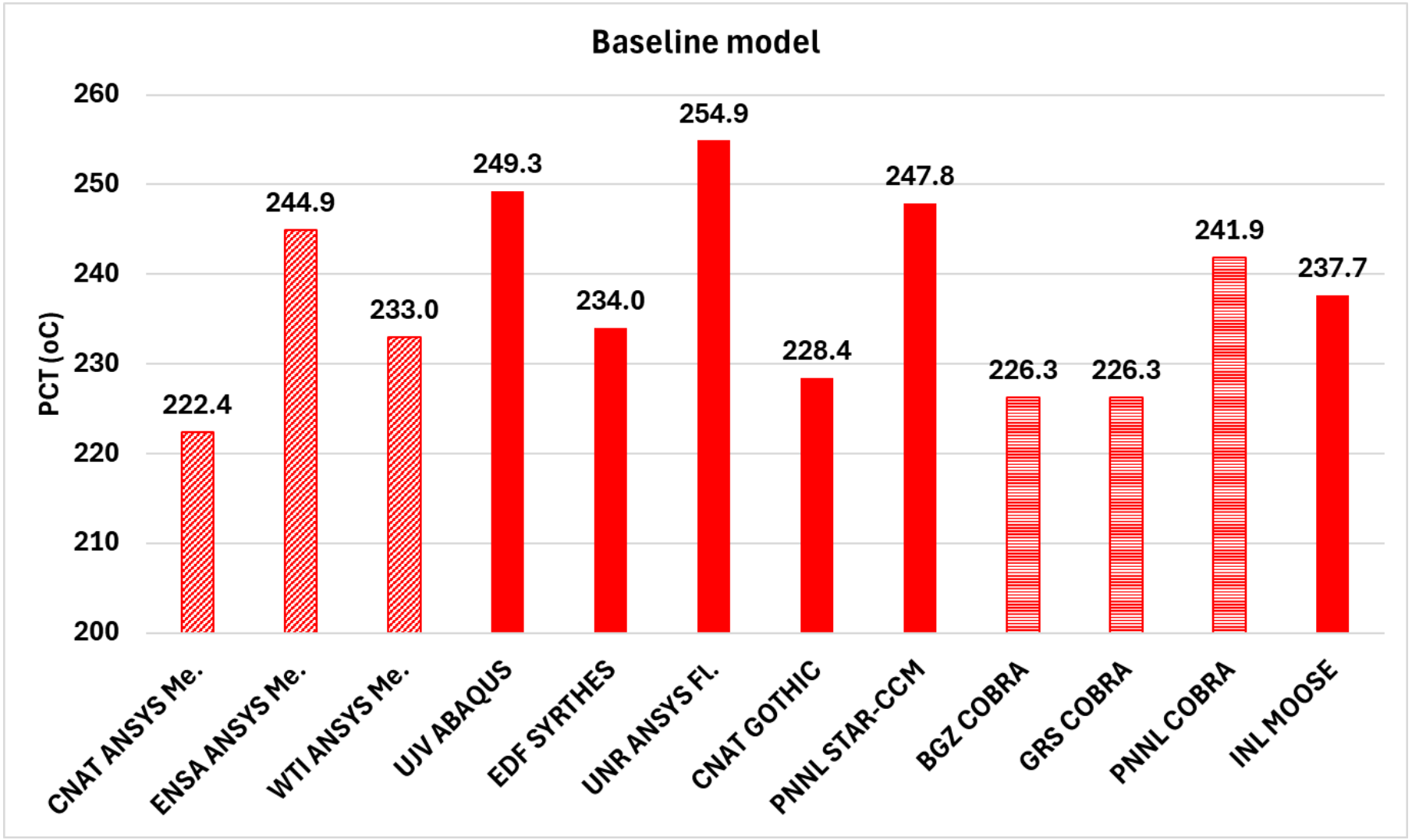
Correlation Between Codes and Results



- **ANSYS Mechanical** results provided by: CNAT; ENSA; and WTI
- **COBRA-SFS** results provided by: BGZ; GRS; and PNNL

Notes:

1. In this report, PCT is referring to peak TC temperature for internal cells (Cell 14) for consistency
2. Some participants provided actual PCT, difference is 2-4°C

Significant variations in the results submitted by different organizations, even using the same code

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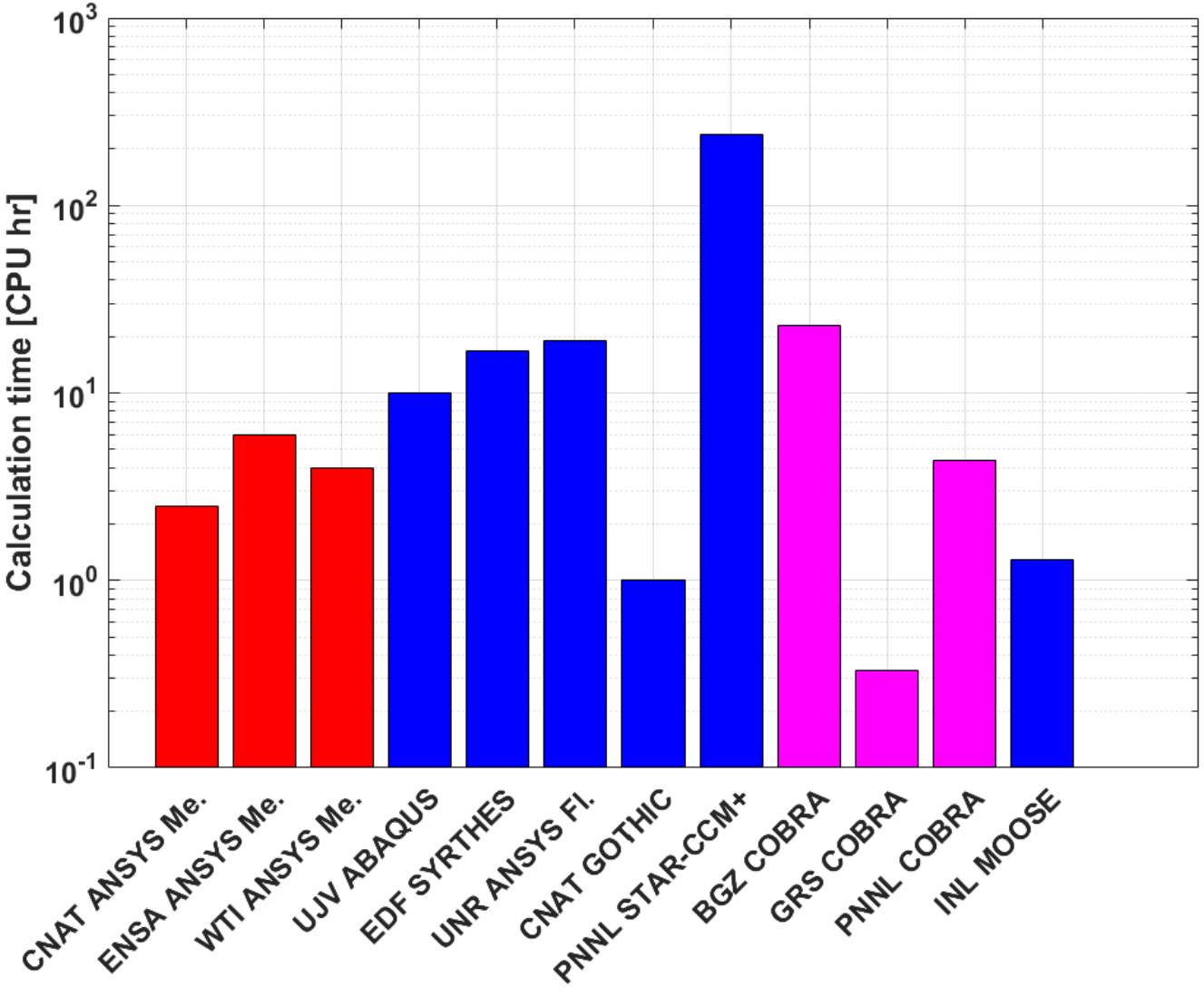
Computational Time Comparisons



- While computational time as provided by the participants is included in the report, it should also be noted that computational time is an imperfect metric to judge.
- Participants are performing computations on different systems with different performance.
- Some codes may parallelize more effectively than others making the impact of computational time on wall time difficult to judge.
- The comparison is provided in the report to provide a general idea.

Computational time varied from approximately 0.3 CPU h to more than 200 CPU

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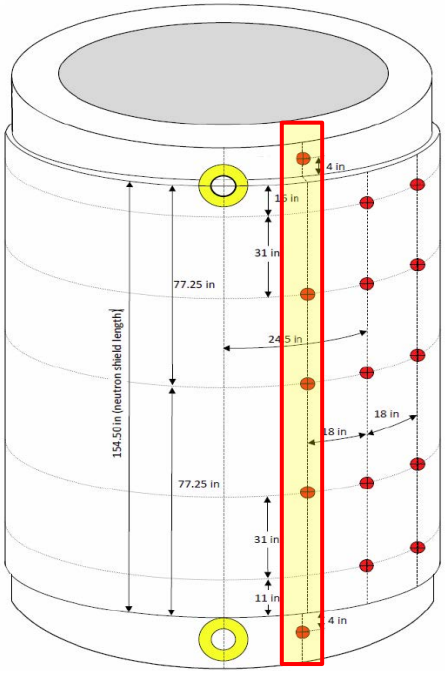
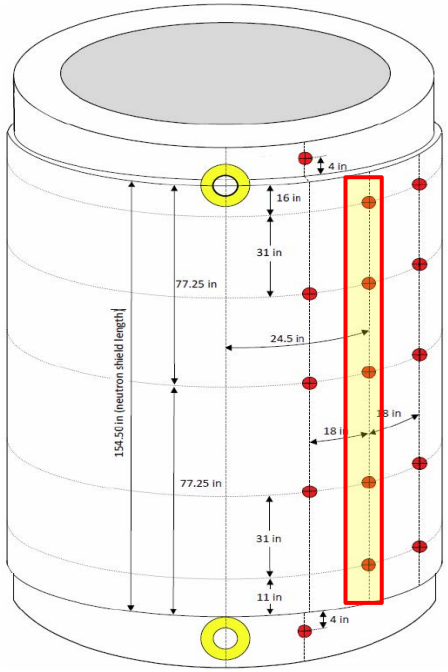
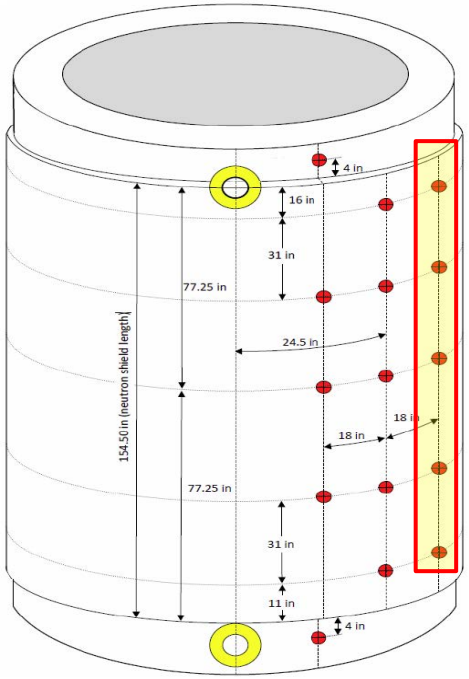
External Temperature Measurements



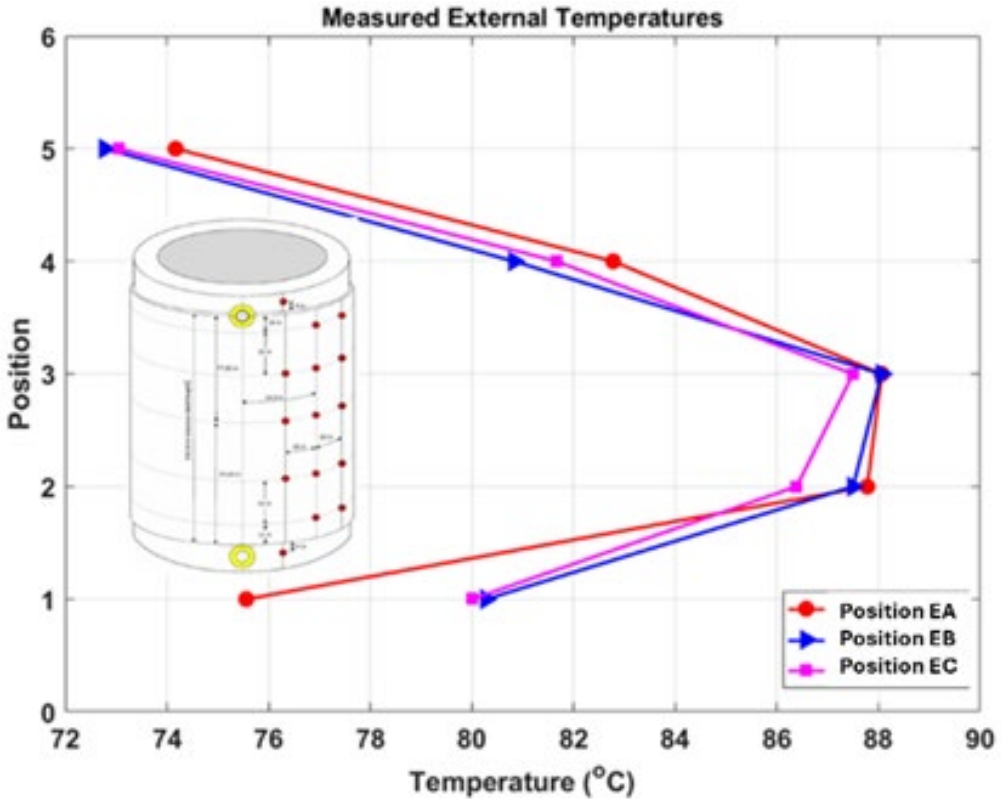
EA (left)



EB (center)



EC (right)



External temperature measurement locations and measured temperatures

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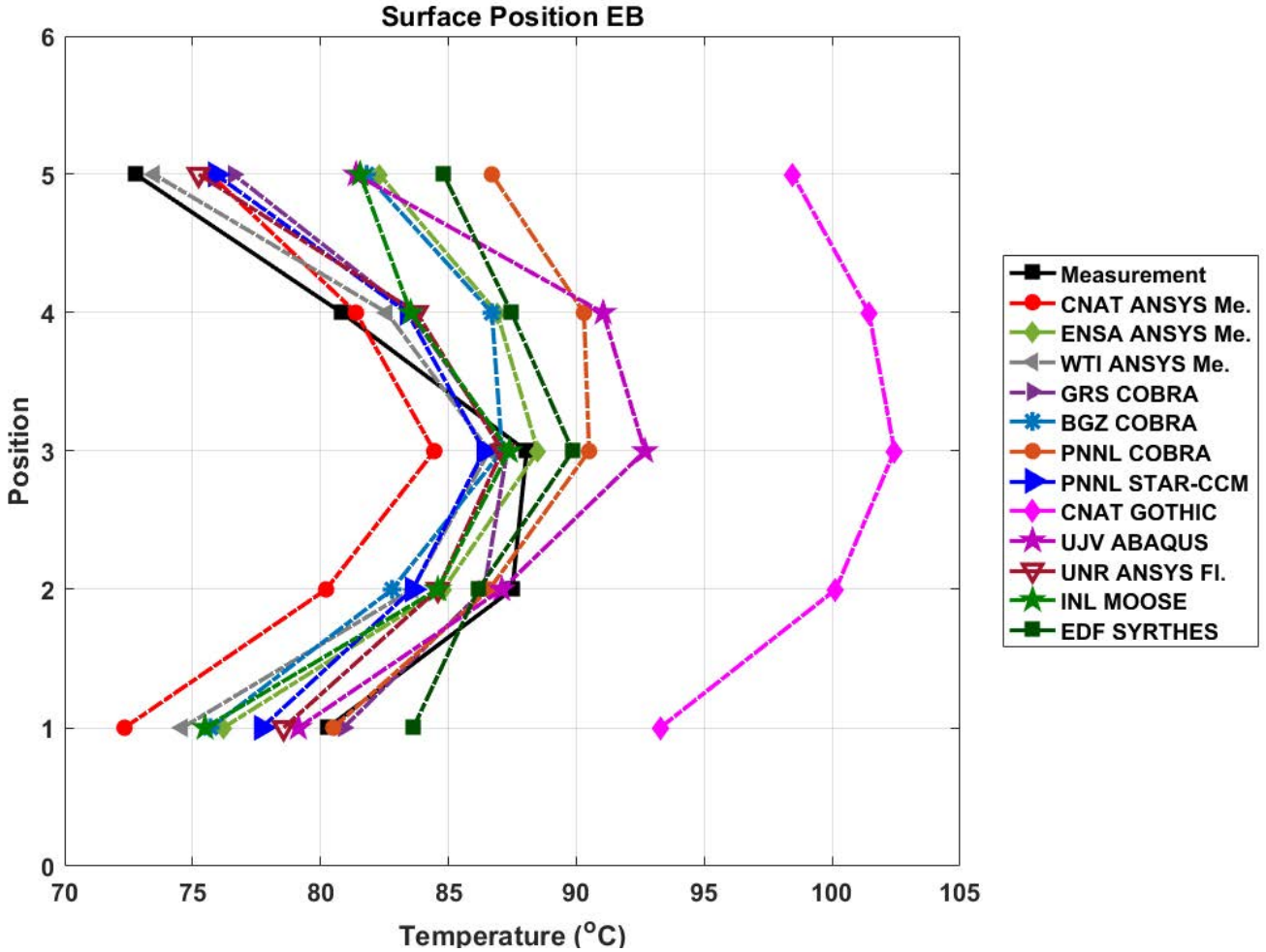
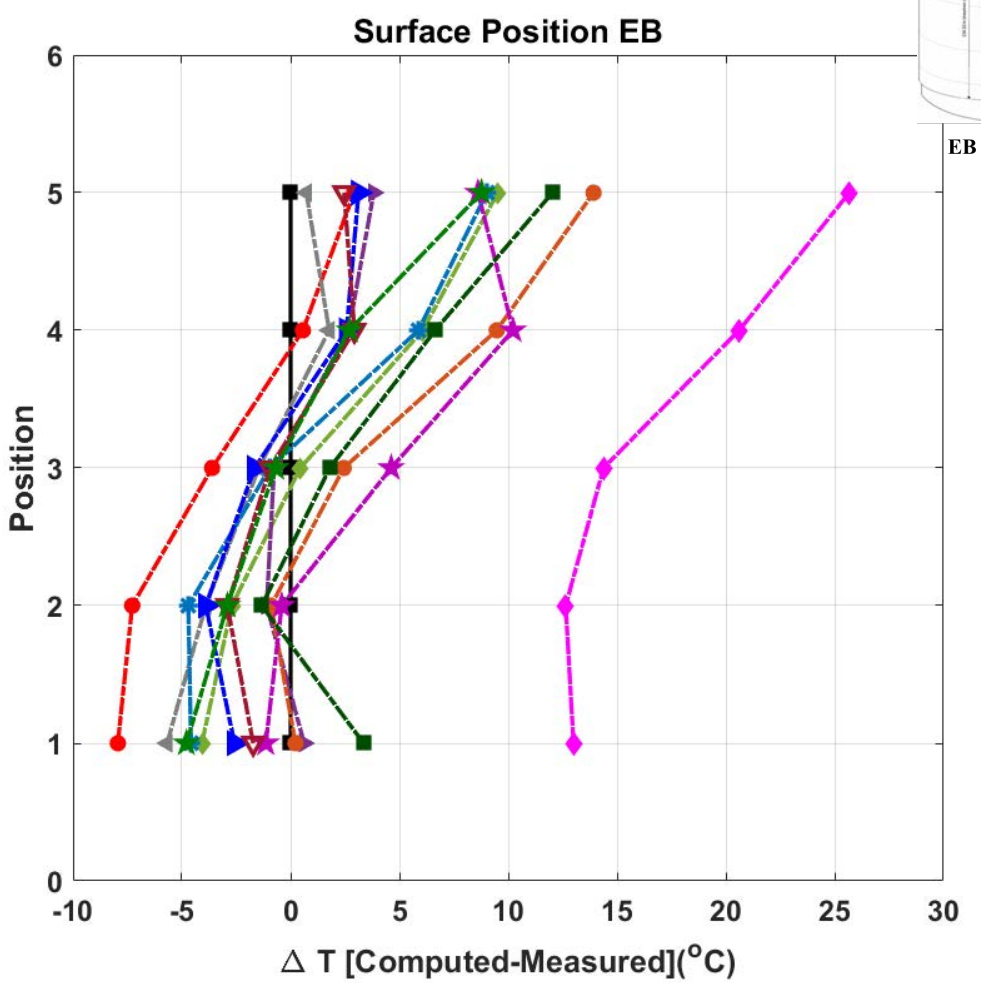
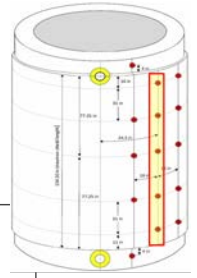
External Temperatures – Surface Position EB







Gothic code shows the highest overestimation for position EB

Large variations in the predicted external surface temperature is the reason for the request to take temperature measurements during inspections

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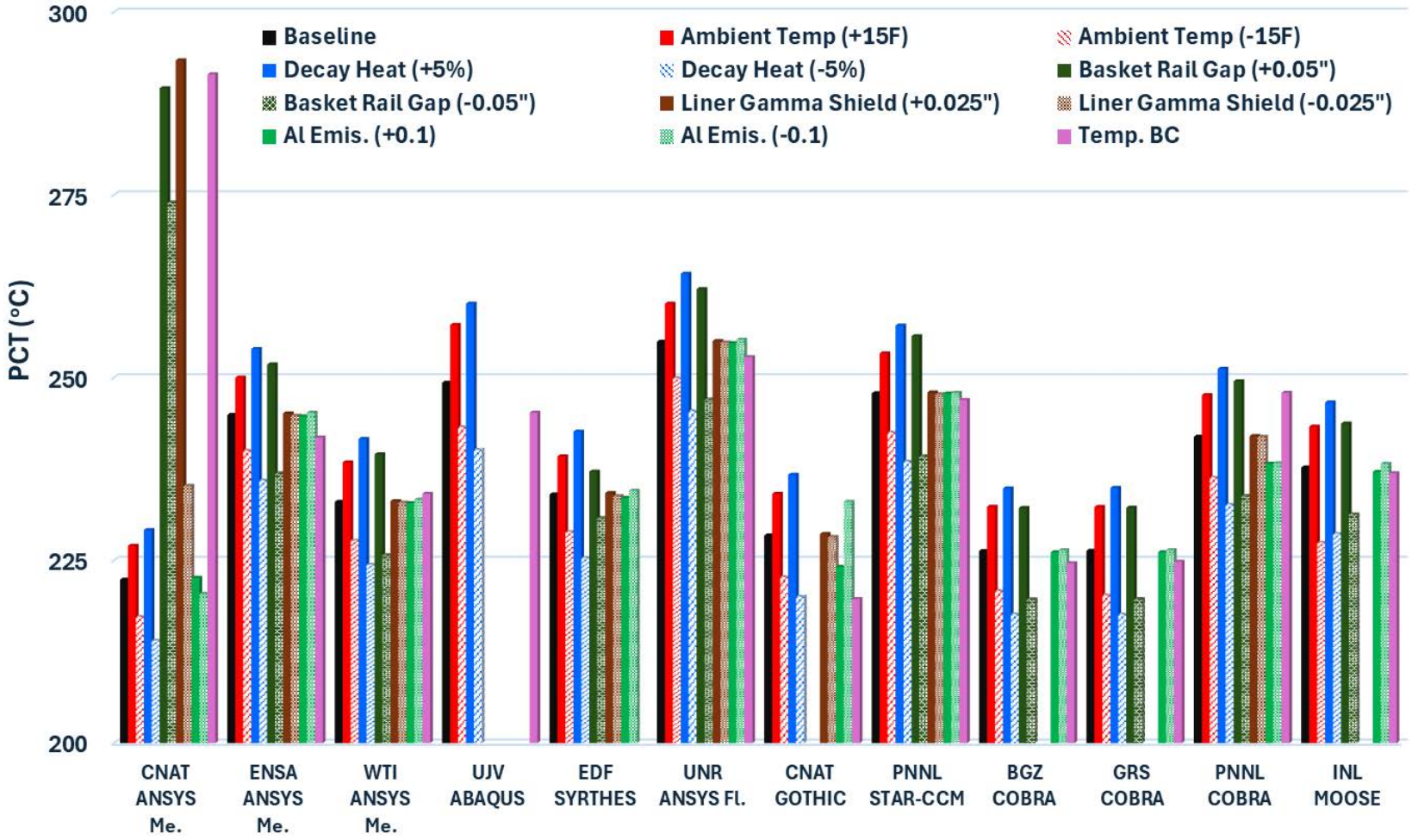
Phase II Sensitivity Parameters

- **Ambient Temperature:** 75°F ± 15°F ambient (90°F, 60°F)
- **Decay Heat:** ±5% of base decay heat – baseline: 30.456 kW
- **Gap between transition rail and basket** -0.05 to 0.15 (0.1 ±0.05)
- **Gap between inner liner base and gamma shield** – base 0.125 (range: ±0.025)
- **Emissivity of aluminum** (current spec 0.2 adjust ±0.1)
- Baseline: **Specified surface temperature boundary condition**

Received sensitivity calculation results from all the participants. Analysis is in final stages. Results will be published in an updated publicly available EPRI report (Phase I and Phase II results combined).

Sensitivity of PCT to Varying Parameters



CNAT ANSYS Mechanical shows the largest sensitivities

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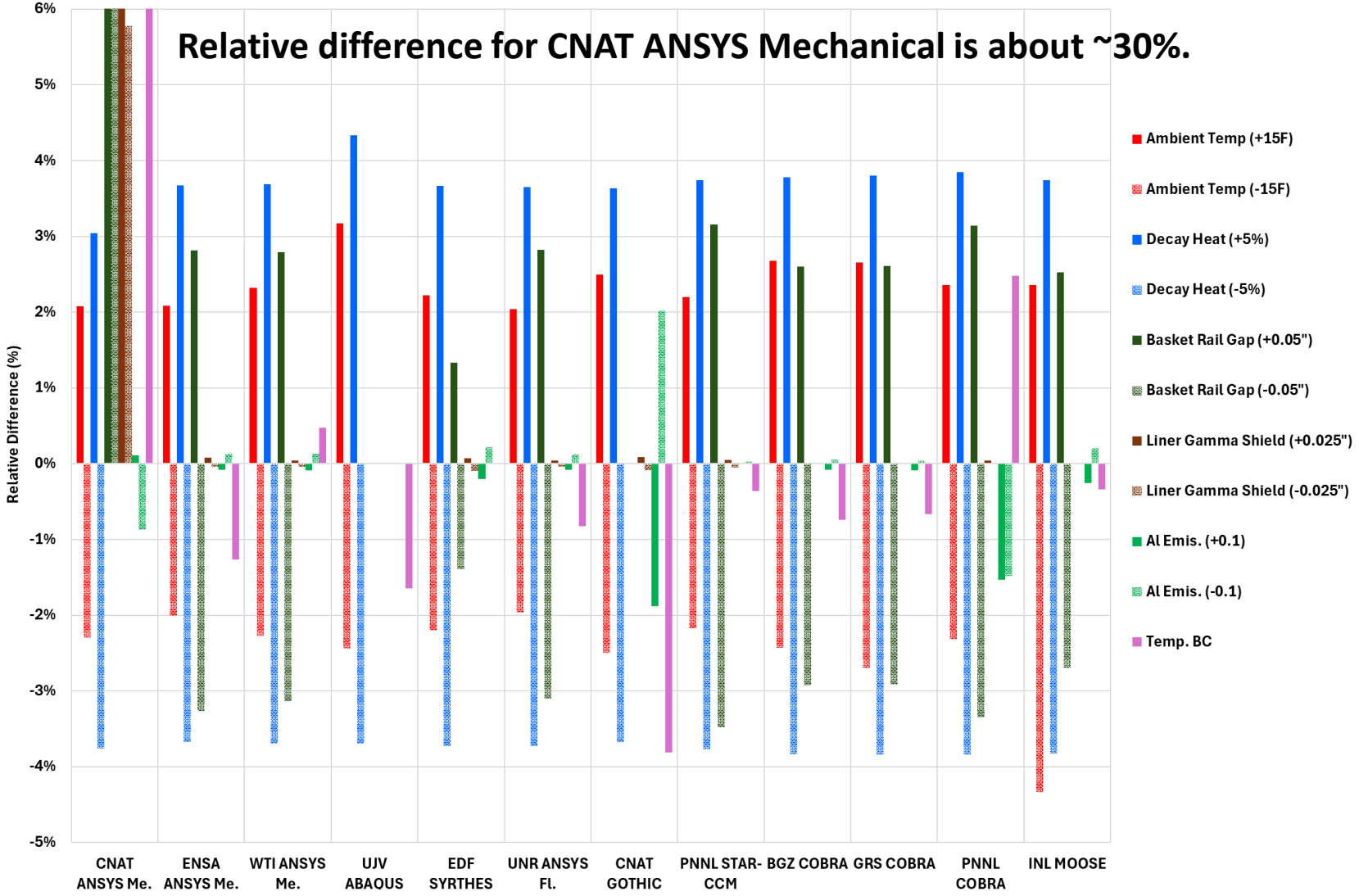
Sensitivity of PCT to Varying Parameters



In general, highest sensitivity is due to decay heat followed with ambient temperature variations

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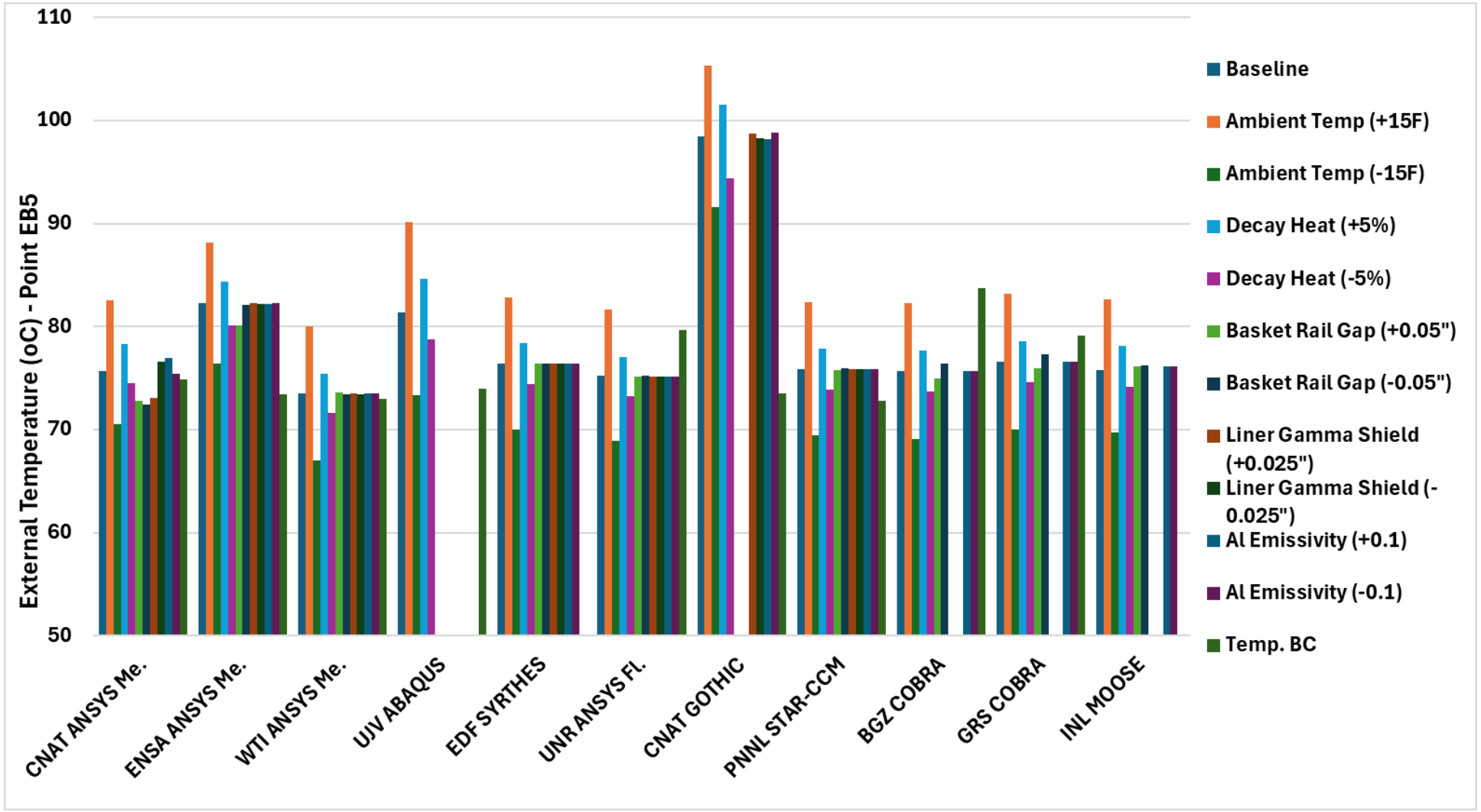
Sensitivity of External Temperature to Varying Parameters



CNAT GOTHIC shows the largest sensitivities

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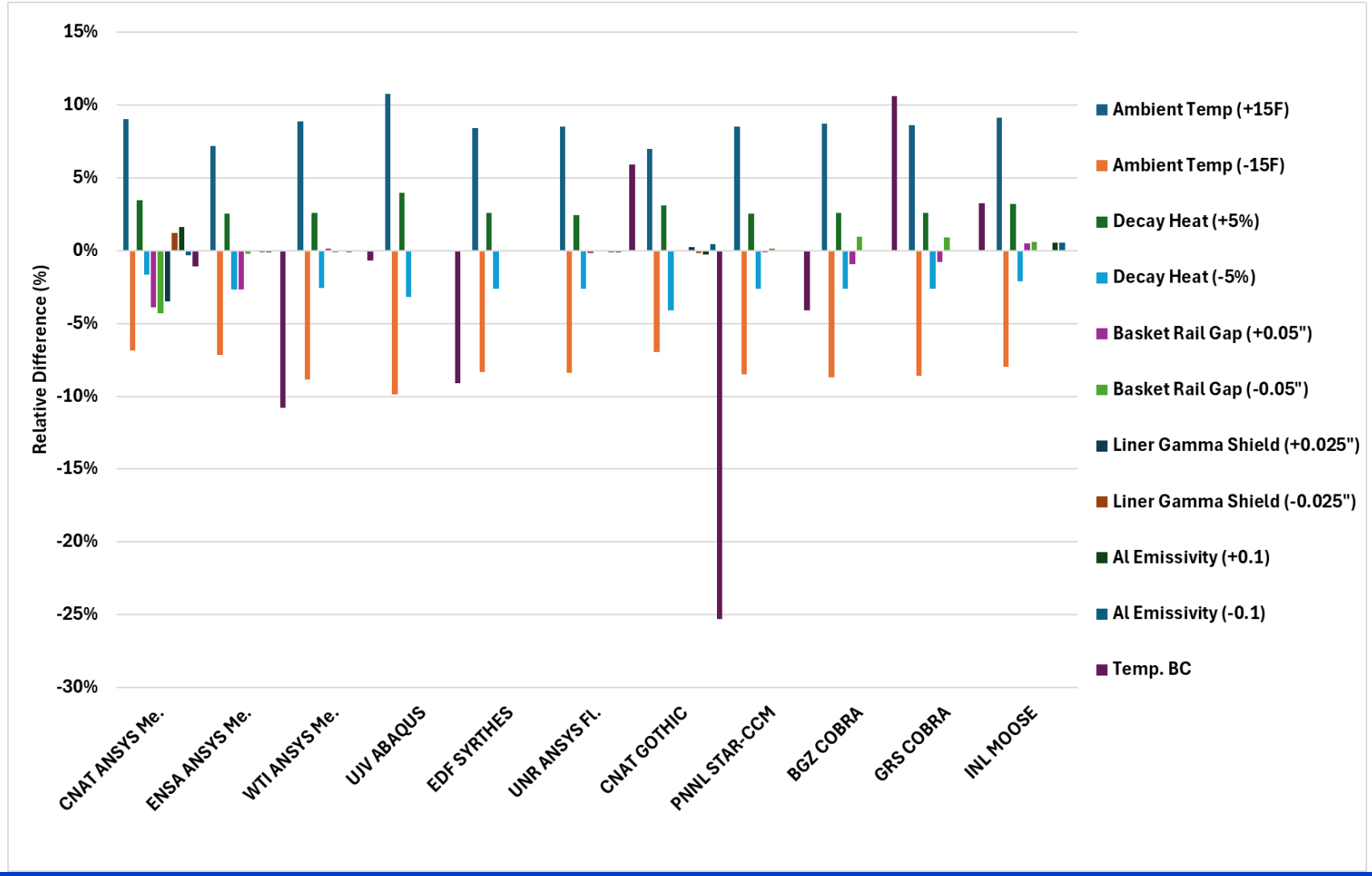
Sensitivity of External Temperature to Varying Parameters



In general, highest sensitivity is due to ambient temperature followed with decay heat

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Summary and Conclusions

- International thermal modeling Phase I and Phase II results received from
  - 10 organizations using 8 different codes
  - In total 12 submissions with 4 different solution approaches
- **Baseline model analysis show**
  - Large variations between codes for both internal cells and external surface temperatures
  - Large variations, even when the same code is used by different organizations
- **Phase II sensitivity analysis show significant sensitivity to changes in**
  - Decay heat, especially for internal cells
  - Ambient temperature, especially for external surface temperatures
  - Basket gap
- No sensitivity to Al emissivity, with some exceptions

Final report published in October 2025 as a publicly available EPRI report

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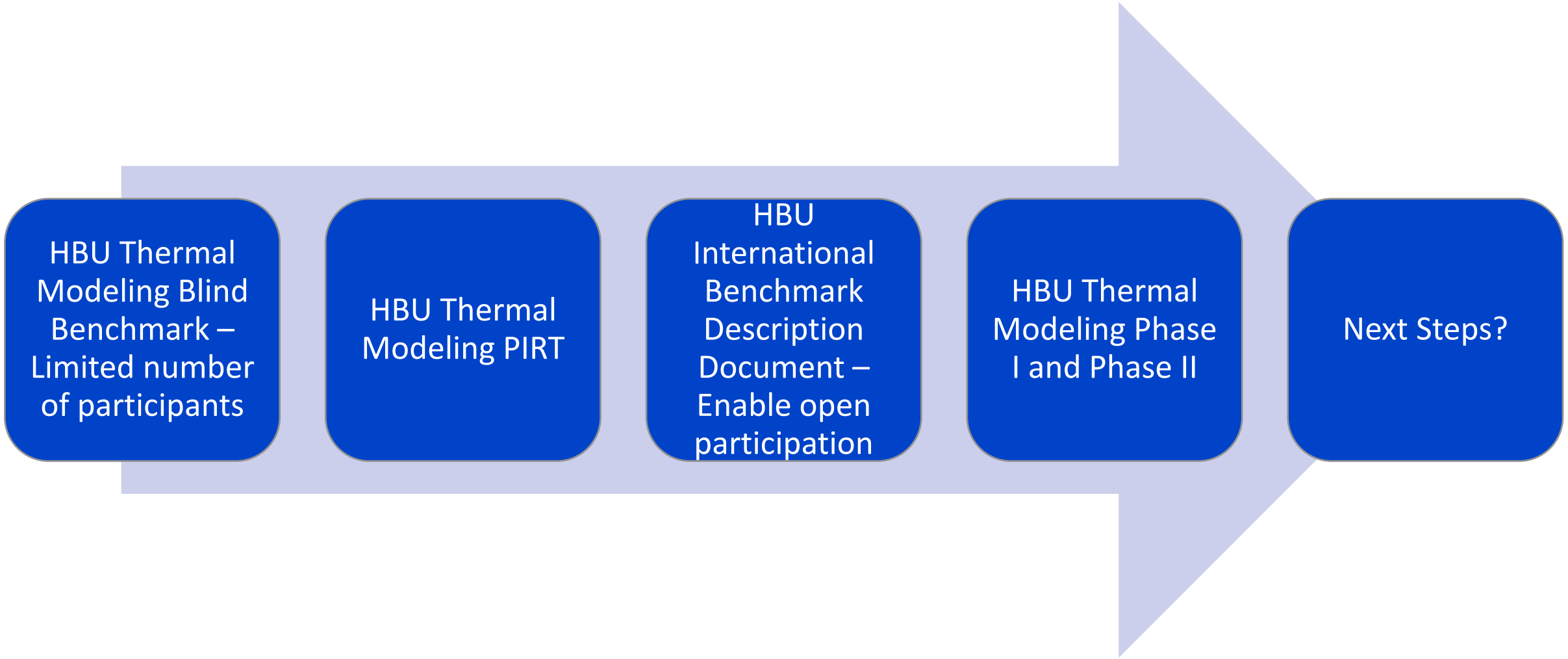
ESCP Thermal Modeling



Each step is resulting with more questions than answers?
Potential next steps were discussed with the participants

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Potential Next Steps Discussion with the Participants

- **Preparing and submitting a journal article on combined results**
  - List all participants from all organizations as co-authors
- **What is next for this group?**
  1. **Understanding the differences between the results when the same code is used by different organizations?**
     - ANSYS Mechanical used by three organizations
     - COBRA-SFS used by three organizations, although BGZ and GRS results are almost identical
  2. Understanding the differences between the results when different codes are used?
  3. Temperature profile of the fuel assemblies/rods
  4. **Other potential benchmarks?**
     - Surface temperature measurements from inspections
     - Coordinating a benchmark with EPRI/DOE canisters using heater for welded system?

There is interest, by all participants, to continue this task group. There is a need for task group lead, which will be approved by ESCP Steering Committee.

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Questions/Comments?

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Fuels Subcommittee Status

Contacts:

- David Hambley, NNL – david.i.hambley@uknnl.com
- Ricardo Torres, EPRI – rtorres@epri.com
- Hatice Akkurt, EPRI – hakkurt@epri.com

Task Group	Task Group Lead	Status
SNF Release Fractions	David Hambley, NNL david.i.hambley@uknnl.com	**Starting activity** **Joint Kickoff – December 2025** **Joint Monthly Meetings**
Annealing Effects & Structural Metrics	Ricardo Torres, EPRI rtorres@epri.com	
Fuel Performance Modeling	TBD	**Scope TBD**
Advanced LWR Fuels	Sven Bader, Orano sven.bader@orano.group	**In Progress** **Monthly Meetings**
Advanced Non-LWR fuels	Dan Moneghan, EPRI dmoneghan@epri.com	**In Progress** **Monthly Meetings**

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Participation

Seeking broad participation on conventional LWR TGs

- U.S. and International research laboratories, cask vendors, utilities, analytical and modeling entities, regulators
- Please contact SC contacts or TG leads, if interested.

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Extended Storage Collaboration Program (ESCP): Collaborative Forum for Addressing Global Technical Challenges Around Used Fuel

Closing Remarks

Hatice Akkurt
Used Fuel and High Level Waste Management, EPRI

Extended Storage Collaboration Program (ESCP) 2025 Meeting
October 30, 2025

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ESCP 2025 Meeting

- **Total number of registered attendees: 225 (99 in-person)**
- Representatives from 13 countries
  - Austria, Belgium, France, Germany, Japan, Lithuania, S. Korea, Spain, Sweden, Switzerland, Taiwan, UK, USA
- Representatives from
  - Utilities, Regulators, Universities, DOE and National Laboratories, Research Organizations, Vendors (Fuel and Cask), NWTRB, IAEA, OECD/NEA
- More than 50 presentations, presented by contributors from numerous countries and organizations
  - 2 panel sessions
  - Subcommittee and task group meetings

Comments on the ESCP 2025 meeting, suggestions for ESCP in general and improvements for future ESCP meetings are welcome

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ESCP 2025 Meeting Materials

- **Meeting materials (presentation files, agenda, attendee list, recordings)will be available via box:** Link is:
  - https://epri.box.com/s/iy3at5g794eo1crgusflj1gfi9ywfai5
    - **It is the same Box folder that contains ESCP meeting materials since 2013**
  - If you do not have Box account, can create a free account from https://www.box.com/pricing/individual
- Will grant access to ESCP Box folder to the first time ESCP attendees

Expecting to release meeting materials in ~2 weeks.

For presentation slides, will release the ones we have permission to release.

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Save the Dates: ESCP 2026 Meeting

October 26-29, 2026

EPRI Charlotte, NC office

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Thank you!

- **All the presenters** – in person and virtual presenters
  - Especially this year with all the schedule changes due to government shutdown
- **All the attendees** – in person and virtual attendees

Special thanks goes to:

Kathy Melton

For all the support. You make ESCP possible!

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Questions/comments/suggestions:
Hatice Akkurt (hakkurt@epri.com)

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