Higher Burnup Workshop V

September 3, 2024

Opening Remarks

Andrea Kock

Deputy Office Director

NRR

Agenda

Time	Topic	Speaker
9:00 am	Welcome	NRC
9:05 am	Opening Remarks	NRC
9:10 am	Accident Tolerant Fuel and Increased Enrichment Rulemaking Update	NRC
9:35 am	Industry Motivation for Use of Higher Burnup	Industry
9:55 am	10 CFR 50.46a/c	Industry
10:25 am	Open Discussion	All
10:35 am	Break	
10:45 am	Phenomena Identification and Ranking Tables for High Burnup Fuel Fragmentation, Relocation, Dispersal, and Its Consequences for Design-Basis Accidents	NRC
11:00 am	Proposed FFRD dose consequence guidance in DG-1425 (Draft Rev 2 of RG 1.183)	NRC
11:15 am	Environmental Evaluation of ATF with Increased Enrichment and Higher Burnup Levels	NRC
11:30 am	Spent Fuel Pool Regulatory Perspective	NRC
11:45 am	High Burnup Summary	DOE
12:05 pm	Open Discussion	All
12:15 pm	Public Comment Period	Public
12:25 pm	Closing Remarks	NRC
12:30 pm	Adjourn	NRC

Topic times are estimated based on the participation level and presentation length.

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United States Nuclear Regulatory Commission
Protecting People and the Environment

Meeting Logistics

- Meeting visuals and audio are through MS Teams.
- Participants are in listen-only mode until the discussion and public feedback period. During which, we will first allow in-person attendees to participate, then allow remote attendees to un-mute.
  - Remote attendees should utilize the hand raised feature in MS Teams, if possible.
- This is an Observation Meeting. Public participation and comments are sought during specific points during the meeting.
  - NRC will consider the input received but will not prepare written responses.
  - No regulatory decisions will be made during this meeting.
- This meeting is being recorded.

Meeting Purpose

- Exchange information between NRC and industry on higher burnup, increased enrichment, and Accident Tolerant Fuel (ATF) activities.
- Provide an opportunity for members of the public to ask questions of the NRC staff.

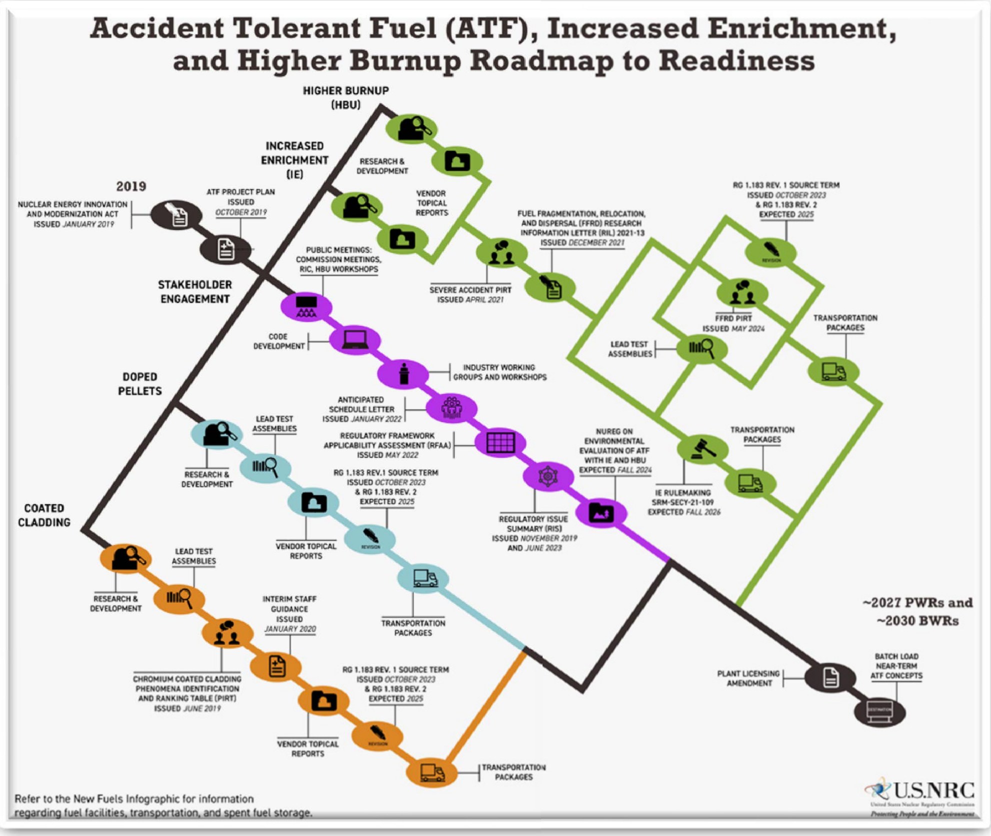
Accident Tolerant Fuel and Increased Enrichment Update

James Delosreyes, NRR/DORL

Philip Benavides, NMSS/REFS

Updates to the "Roadmap to Readiness"

- **<u>Public Meeting</u>:** May 10, 2023, (ML23138A050)
- **<u>Issued</u>:** June 28, 2023, (ML23158A288)
- **<u>Revised</u>:** January 8, 2024, (ML23353A144)
- **<u>Revised</u>:** August 2, 2024, (ML24199A170)
- **<u>Available at:</u>** https://www.nrc.gov/reactors/power/atf.html



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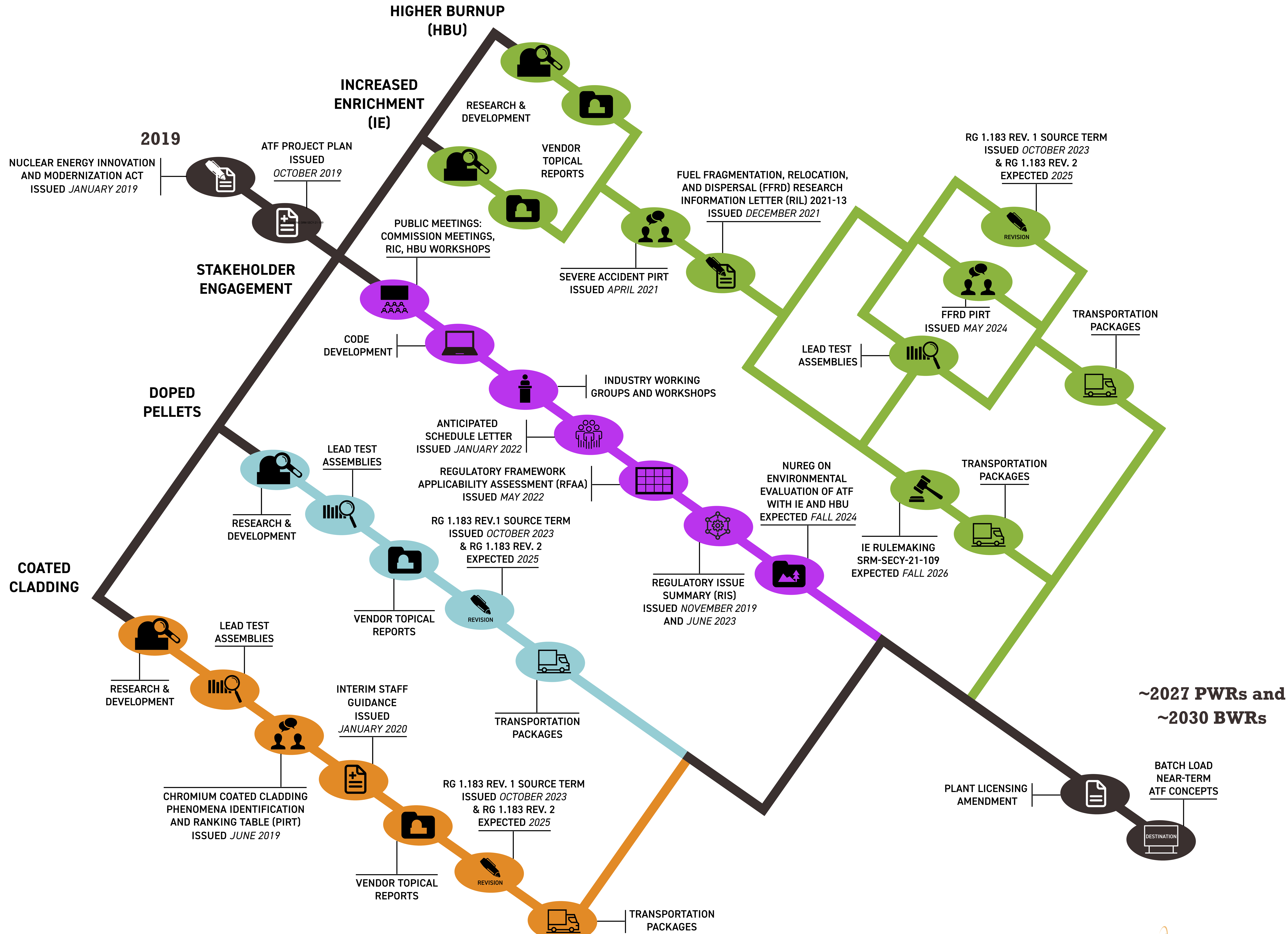
Accident Tolerant Fuel (ATF), Increased Enrichment, and Higher Burnup Roadmap to Readiness



Refer to the New Fuels Infographic for information regarding fuel facilities, transportation, and spent fuel storage.

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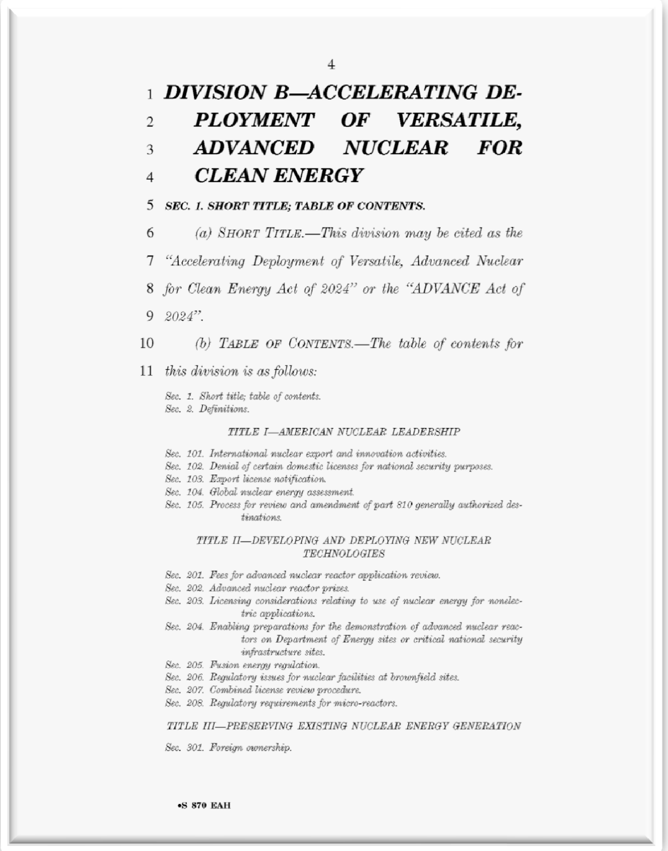
ADVANCE Act

- Signed into law on 7/19/24 with wide bipartisan support.
- Accelerating deployment of versatile, advanced nuclear for clean energy.
- Builds upon initiatives begun with NEICA in 2017 and continued with NEIMA in 2019.



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DIVISION B—ACCELERATING DEPLOYMENT OF VERSATILE, ADVANCED NUCLEAR FOR CLEAN ENERGY

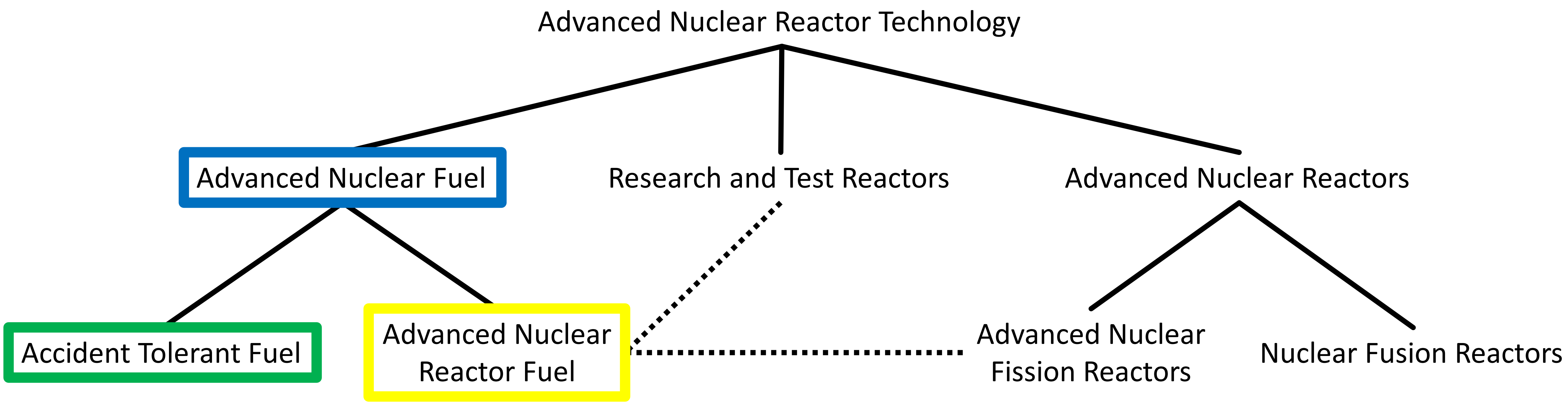
SEC. 1. SHORT TITLE; TABLE OF CONTENTS.

(a) SHORT TITLE.—This division may be cited as the "Accelerating Deployment of Versatile, Advanced Nuclear for Clean Energy Act of 2024" or the "ADVANCE Act of 2024".

(b) TABLE OF CONTENTS.—The table of contents for this division is as follows:

•S 870 EAH

ADVANCE Act


Advanced Nuclear Reactor Technology
Advanced Nuclear Fuel
Research and Test Reactors
Advanced Nuclear Reactors
Accident Tolerant Fuel
Advanced Nuclear Reactor Fuel
Advanced Nuclear Fission Reactors
Nuclear Fusion Reactors

ADVANCE Act

Accident Tolerant Fuel

Makes an existing commercial nuclear reactor more resistant to a nuclear incident

and

Lowers the cost of electricity over the licensed lifetime of an existing commercial nuclear reactor.

Advanced Nuclear Fuel

Ceramic cladding materials;

Fuels containing silicon carbide;

High-assay, low-enriched uranium fuels;

Molten-salt based liquid fuels;

Fuels derived from spent nuclear fuel or depleted uranium

and

Other related fuel concepts, as determined by the Commission

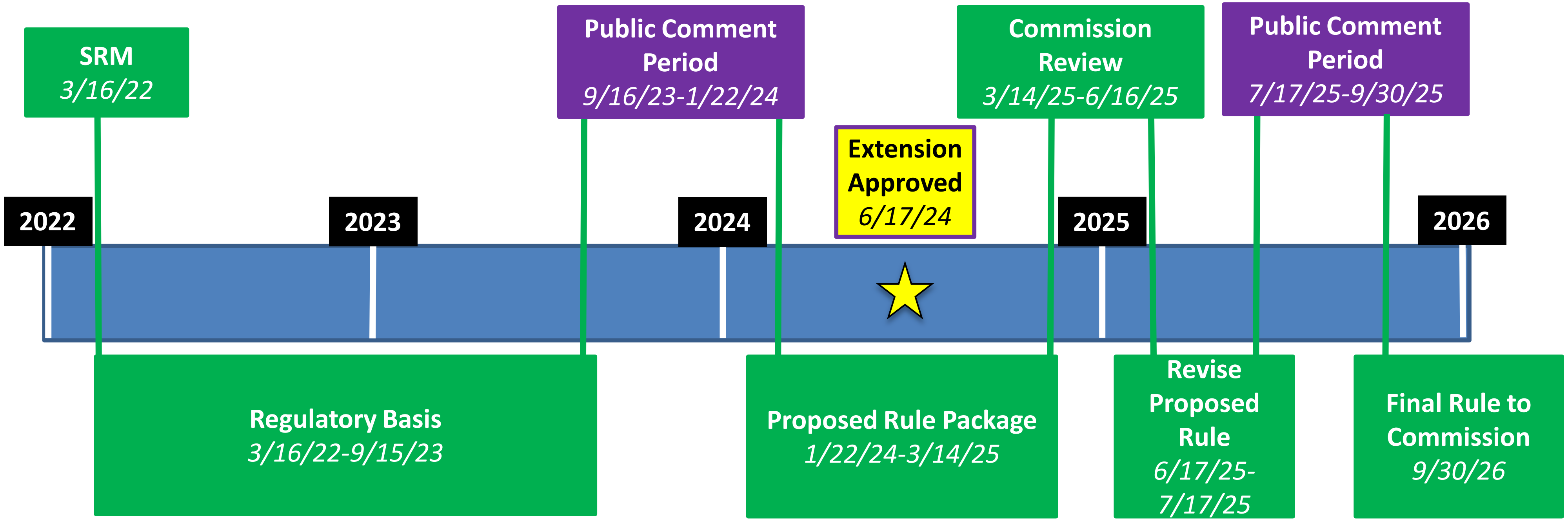
Status of Rulemaking Activity

- The NRC staff is preparing the IE proposed rule package regarding supporting industry interest in the use of fuel enriched to greater than 5.0 weight percent U-235.

- The public comment period for the regulatory basis closed on 1/22/24.

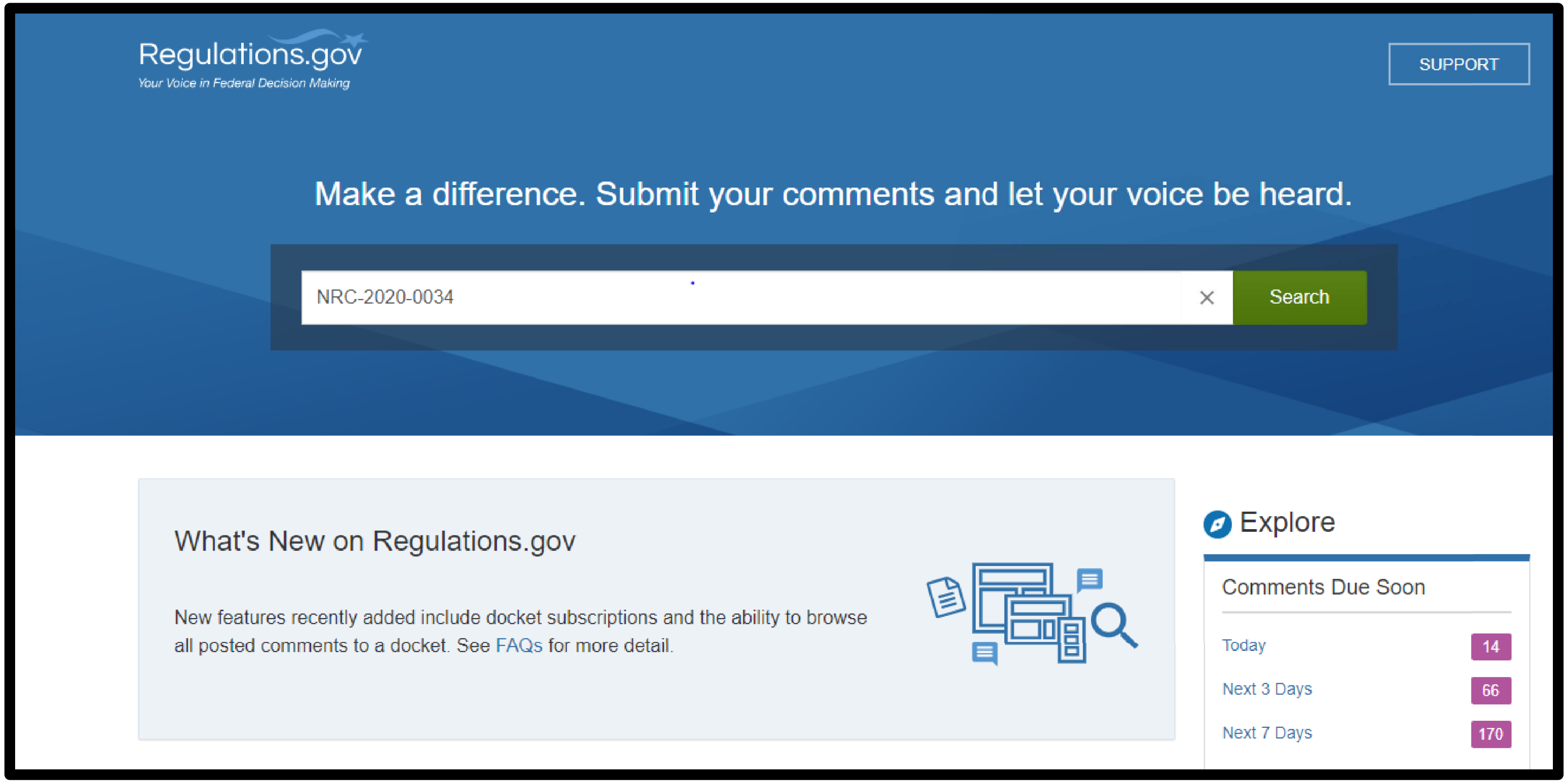
- The Commission approved a three-month extension to the IE rulemaking effort on 6/17/24 to support further work regarding Fuel Fragmentation, Release, and Dispersal (FFRD).

Status of Rulemaking Activity



Note: Dates listed are estimates only, and thus are subject to change.

Stay Updated on IE Rulemaking



Go to https://www.regulations.gov/ and search for docket ID NRC-2020-0034

FFRD and 50.46c Next Steps

2024 NRC High Burnup Workshop

Joseph Messina
Nuclear Methods and Fuel Analysis

September 3, 2024

Purpose

- The purpose of this presentation is to provide a high-level update on:
  - The activities related to FFRD in the Increased Enrichment rulemaking and
  - The NRC staff's plan to address the Commission's SRM on the 50.46c draft final rule (SRM-SECY-16-0033)
- The information presented in these slides is preliminary and has not been approved by the Commission.
  - The Commission will vote on the Increased Enrichment draft proposed rule.
- The NRC staff is not soliciting stakeholder comments in this meeting
  - There will be opportunity for public comment once the Commission votes on the draft proposed rule

IE Rulemaking Regulatory Basis FFRD Alternatives

The IE Rulemaking Regulatory Basis (ML23032A504) considered 5 licensing pathways for addressing fuel dispersal:

- **Alternative 1**: No action.
- **Alternative 2**: 50.46a-style modification of ECCS requirements.
- **Alternative 3**: Perform a safety demonstration for post-FFRD consequences.
- **Alternative 4**: Provide a generic bounding assessment of dose and use risk insights for post-FFRD consequences.
- **Alternative 5**: Use probabilistic fracture mechanics to show that leaks in large pipes will be identified before failure, precluding the need to analyze LBLOCAs.

Fuel Dispersal Path Forward

- The NRC staff plans to risk-inform LOCAs by modernizing 50.46a (based on Alternative 2) in the IE draft proposed rule to facilitate addressing fuel dispersal
  - Support for Alternative 2 expressed in many public comments
  - Smallest impact on the IE Rulemaking schedule of the alternatives that received support
    - Leveraging the technical basis and work performed in the original 50.46a
  - High level of technical maturity
- 10 CFR 50.46a was a draft final rule in 2010 that proposed to establish a transition break size (TBS), above which LOCAs would be recategorized as beyond-design-basis
  - Voluntary alternative to 50.46
  - Original philosophy being maintained with some changes
- The updated 50.46a is planned to include high-level, fuel technology neutral, performance-based Emergency Core Cooling System (ECCS) acceptance criteria

Addressing Fuel Dispersal

- The 50.46a approach is expected to facilitate safety demonstrations of fuel dispersal because true best-estimate modeling and realistic assumptions are expected to significantly reduce or eliminate the potential for fuel dispersal
- While this approach does not explicitly address non-mechanistic approaches to evaluating FFRD, as described in other alternatives in the IE Regulatory Basis, other licensing pathways exist
  - E.g., the topical report review process
  - The performance-based criteria are expected to provide relief to the prescriptive philosophy of the existing regulatory framework (including a less stringent definition of core coolability)

Highlighted Work to Support the IE Proposed Rule

- Confirmation of the transition break size (TBS)
  - NRC internal and external expert elicitation
  - xLPR runs of the NUREG-1829 bases cases
  - Evaluation of operating experience
  - Confirmation of NUREG-1903 technical basis
- Update of the following draft regulatory guides:
  - DG-1261, “Measuring Breakaway Oxidation”
  - DG-1262, “Determining Post Quench Ductility”
  - DG-1263, “Establishing Analytical Limits for Zirconium-Alloy Cladding Material”
- Development of the following draft regulatory guides:
  - DG-1426, “An Approach for a Risk-Informed Evaluation Process for Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors”
  - DG-1428, “Plant Specific Applicability of the Transition Break Size”
  - DG-1434, “Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents”

General Considerations

Consideration is being given to comments received on the Regulatory Basis regarding 50.46a and 50.46c as well as some comments on the original 50.46a rulemaking. For example, some things that are being *considered*:

- True best estimate calculations (nominal, without biases or uncertainties) of ECCS performance for LOCAs greater than the TBS
- Moving specific ECCS criteria for Zirconium-UO_2 fuel systems to guidance (DG-1263)
- Relaxation of breakaway oxidation reporting requirement in DG-1261
- Allowing low power and shutdown to be addressed with non-PRA methods
- Difficulty in demonstrating TBS applicability (i.e., DG-1428)
  - Considering periodic weld inspection requirement, which may reduce burden of DG-1428
  - Guidance on seismic considerations in DG-1428
  - Other implementation requirements may be needed

50.46C UPDATE

SRM-SECY-16-0033 (50.46c)

The Commission returned the 50.46c draft final rule package (SECY-16-0033) to the staff without Commission action and directed the staff to do the following:

1. The staff should apply an appropriate risk-informed regulatory approach to address the research findings on cladding embrittlement effects under LOCA conditions described in SECY-16-0033.
2. The staff should evaluate Item 1 with other associated technical issues being addressed, such as fuel fragmentation relocation and dispersal, and risk-informed treatment of LOCAs, including the draft final 50.46a that had been provided in SECY-10-0161.
3. The staff should evaluate whether specific emergency core cooling system criteria such as cladding temperature should be codified or instead addressed in regulatory guidance.
4. Within six months of the date of this SRM, the staff should provide, through a Commissioner Assistant's Note, an action plan for the above items.

(SRM-SECY-16-0033, ML24102A281, April 11, 2024)

50.46c Path Forward

The staff plans to include aspects of 50.46c in voluntary provisions of the Increased Enrichment (IE) proposed rule and assess the need for further action on the 50.46c rulemaking after the Commission votes on the IE final rule package (estimated date to Commission is Sept 2026).

- The staff is planning to risk inform LOCAs, as suggested in SRM-SECY-16-0033, with 50.46a in order to facilitate safety demonstrations of fuel dispersal in the IE proposed rule.
- The staff would use the public comments received on the 50.46c aspects of the IE rulemaking to inform any potential future action on the 50.46c rulemaking.
- The staff will continue to perform the annual ECCS Safety Assessments and evaluate the impacts of the cladding embrittlement research findings within the framework of existing regulatory requirements when reviewing industry submittals that could result in cladding embrittlement impacts (e.g., power uprates or burnup increases).

QUESTIONS?

Industry Motivation for Use of Higher Burnup

Fred Smith, Sr. Technical Executive
Fuel Reliability Program, EPRI

NRC HBU Workshop
September 3rd, 2024

Electricity Demand Growth Drivers

US electricity load growth forecast jumps 81% led by data centers, industry: Grid Strategies

Data from FERC Form 714 shows grid planners expect nationwide power demand to grow 4.7% over the next five years, compared to a previous estimate of 2.6%.

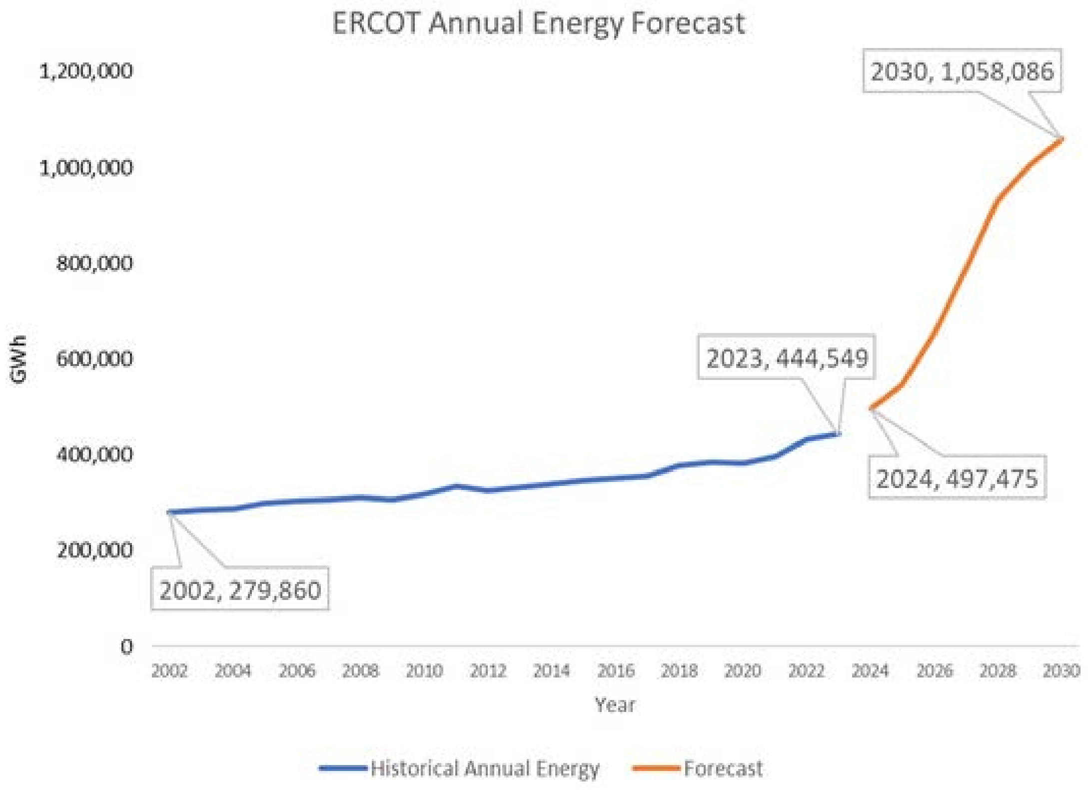
Inflation Reduction Act tax credits

Production Tax Credits for existing reactors (45U, 45Y)

Investment Tax Credit new facilities or capacity (48E)

IEA projects, Global electricity demand to rise at a faster rate over the next three years by an average of 3.4% annually through 2026. In the IEA's Net Zero 2050 scenario, electricity's share of the final energy mix nears 30% in 2030 (comparted to 20% in 2023).

https://www.iea.org/reports/electricity-2024/executive-summary



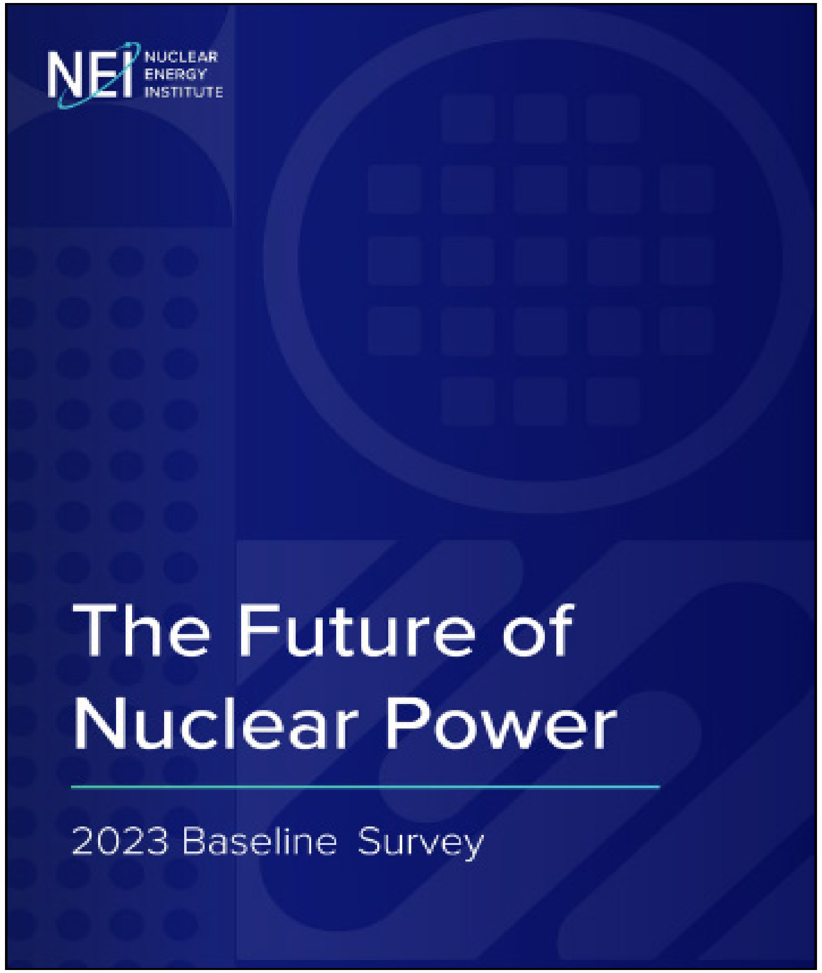
ERCOT Annual Energy

https://www.ercot.com/gridinfo/load/forecast/

NEI Survey: Power Uprate and Cycle Length Extension

- Key takeaways:
  - >55% of sites have a level of interest/planning for one or more power uprates with a combined capacity increase of 2.5 GWe
  - Nearly 33% of planned uprates are EPUs, ~50% are SPUs, and remainder are MURs
  - ~75% of PWRs interested in extended cycles

- Interests in uprates are high due in part to the IRA tax credits and projected electricity demand signals, but is ultimately about the underlying plant specific business cases



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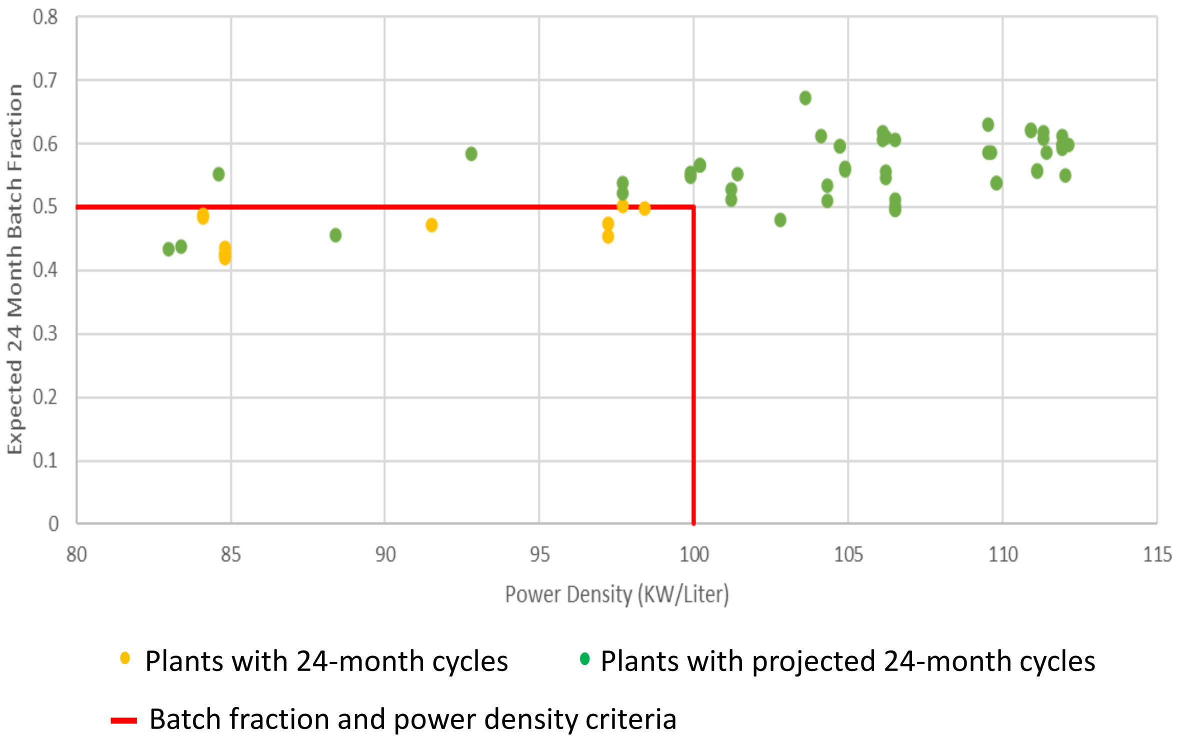
Extended Fuel Cycle Lengths

- Only low power density PWRs can transition to 24-month fuel cycles at current enrichment/burnup levels
- All BWRs can operate 24-month fuel cycles
  - All but one does so in the US
- Cycle lengths greater than 24-month fuel cycle lengths are not likely to be practical
- Worldwide utilities are considering longer fuel cycles (annual to 15/18-months or 18- to 24-months)

- Other benefits of longer cycles:
  - Better coordination of outage work force across a fleet
  - More time to plan outages
  - Improved work/life balance

Evaluation of current US PWR fleet using current burnup and enrichment limits



Most PWRs require higher burnup and increased enrichment for 24-month fuel cycles

Uncertainties in Cycle Length Extension Decision Analysis

- Factors impacting decisions to transition to longer cycles
  - Impact of Electricity Demand on Market Price
  - Geo-political impacts on Uranium Market Price
  - Inflation and aging workforce impacts on Costs
  - Seasonal variations on Electricity Demand due to weather
  - Treatment of back-end disposal costs

- Factors and associated risks vary from licensee to licensee
  - They may arrive at different decision on longer cycles

Decision Analysis for Extended Cycle Lengths

	Typical PWR Results for 4 x 18 Month Fuel Cycles compared to 3 x 24 Month Fuel Cycles*	
	With HBU/HE	**Without HBU/HE**
Fuel Cost Increase (at higher rate than the cycle energy increases)	$4.6M	$109M
Outage Cost (saves one outage in 6 years)	$-45.0M	$-45.0M
Addition Generation (1.5% CF increase)	$-28.3M	$-28.3M
Capital	$ 1M	$ 1M
Net	$-22.8M	$81M

* Based on methods employed in " The Economic Benefits and Challenges with Utilizing Increased Enrichment and Fuel Burnup for Light-Water Reactors," NEI 2019

Net benefit is only for higher burnup and high enrichment

* More information available at: Facilitating Power Uprates at Nuclear Power Plants: Feasibility Study Guideline. EPRI, Palo Alto, CA: 2023. 3002026402

Power Uprates

NEI survey of US reactors projects 2.5 GWe of nuclear capacity additions*

Technical basis to enable uprates are needed

- Existing fuel design limits can limit the uprate and make it more expensive. To address this impact:
  - Increased Enrichment and Higher Burnup Fuel
  - Advanced Cladding and other Fuel Technologies
  - Time-at-temperature T/S based operating limits
  - Updated LOCA methods

*NEI State of the Nuclear Energy Industry, 2024

Time-at-Temperature (TaT)

- Current regulation established to avoid
  - Boiling transition
  - Cladding over heating
  - Fuel failures
- Current regulation does not account for large margin between onset of boiling transition and fuel failure conditions in moderate frequency transients
- Coated cladding technologies (ATF) may provide additional margin
- CRAFT Supported PERT 2024-2025

- Several Potential Benefits of Revised TaT Criteria
  - Enable higher power uprates
  - Improved operating limits
  - Up to 5% improvements in fuel margins
  - Prompt return to operation after unexpected transient occurrences
  - Reduced dose consequences during select transients

EPRI and DOE labs are collaborating on material testing through CRAFT

Other Related Areas of Interest

- Resolution of FFRD issue
- Updated LOCA analysis framework
- Resolution of Accident Source Term Challenges RG 1.183

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Industry Perspectives on a Modernized and Combined Risk-Informed LOCA Rule (10 CFR 50.46a/c)

Aladar Csontos, Director Fuels (NEI)
Tom Kindred, Consulting Engineer (SNC)
Paul Clifford, Consultant (Framatome)

September 3rd, 2024

NUCLEAR ENERGY INSTITUTE

ADVANCE Act – Opportunities to Enable Change

Key Provisions:

- American Nuclear Leadership
- Developing and Deploying New Nuclear Technologies
- Preserving Existing Nuclear Energy Generation
- Nuclear Fuel Cycle, Supply Chain, Infrastructure, and Workforce
- Improving Commission Efficiency

Aligned with intent of recent NRC Activities:

- Reactor Accident Analysis Modernization (RAAM) Project
- SECY-21-0109: Increased Enrichment Rulemaking
- SRM-SECY-16-0033: Commission SRM (RI and combine 50.46a/c)
- Accident Tolerant Fuel and Power Uprate Project Plan/Charter

Overview

- Draft 50.46a (SECY-10-0161) Rule – Background
- Commission PRA Policy Statement
- 10 CFR 50.46 a Defense in Depth Rule?
- LOCAs not Significant Contributors to Plant Risk
- Considerations/Clarifications for Modernized 50.46a Rule
- Technology-Neutral 50.46c Requirements

Draft 50.46a Rule (Background)

- ML102210460 – Draft Final 50.46a Rule (2010)
  - Contained a thorough and well thought out rule for risk-informing LOCA analysis under 10 CFR 50.46
  - However, the rule was not fully risk-informed
  - Required a significant amount of PRA overhead
  - Plant specific analysis for breaks considered beyond design basis
  - Plant specific seismic evaluations
  - Commission approval of change control program for changes under 50.46a

In 2010, the general consensus was the large overhead required by utilities to implement a 50.46a rule was not worth the modest potential benefits due to the large overhead required.

Commission PRA Policy Statement

PRA and associated analyses should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule). The existing rules and regulations shall be complied with unless these rules and regulations are revised.

- State of the Art (SOA) for PRAs and probabilistic fracture mechanics (PFM) have changed since 2010 and 1988 (last revision to 10 CFR 50.46)
    - RG-1.245 Endorses use of xLPR (industry tool for realistically estimating PFM)
        - Confirms probability of rupture is realistically predicted by NUREG-1829
    - PRA quality and pedigree has improved with utility implementation of RI programs (Risk Informed Completion Times, 50.69 categorizations, Surveillance Frequency Control Programs, Reactor Oversight Process)
        - RG-1.200 (Quality of PRAs)
        - Endorses NEI 17-07 for quality standards for PRA Peer Reviews

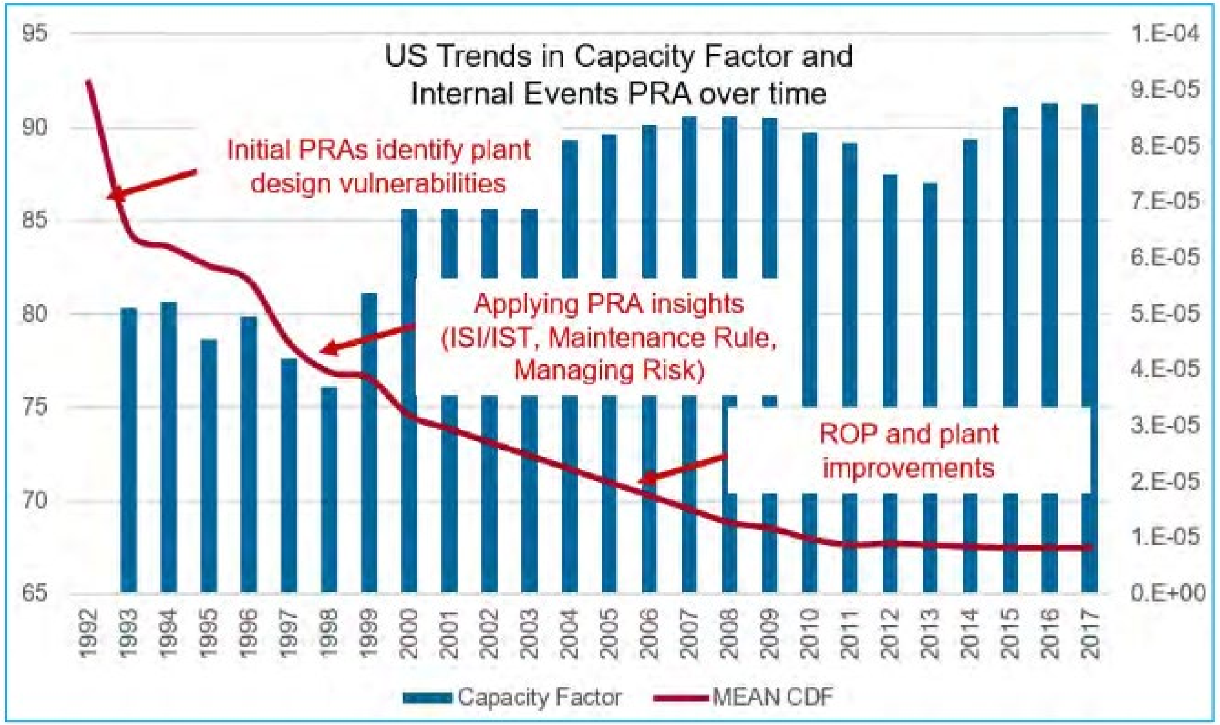
A modernized 50.46a rule is an ideal candidate for RI policy as significant changes in PRA quality and SOA of PFM can be confirmed

10 CFR 50.46 a Defense in Depth Rule?

- 10 CFR 100 and 50.67 for offsite/onsite dose consequences
  - The dose consequence analyses take no credit for ECCS performance and assumes a full core melt
    - SAND2023-01313
    - Plants demonstrate 50.67 acceptance criteria from analyses that do not credit ECCS/50.46 analyses
- Additional DiD provided by
  - 10 CFR 50.155 (SAMGs, SAWA, SAWM, FLEX)
    - Credited in NUREG-1935
  - 10 CFR 50.47 Emergency Planning

LOCAs not Significant Contributors to Overall Plant Risk

- Review of Industry Baseline Risk Index for Initiating Events (BRIIE) – NUREG/CR-6932 for Initiating Events leading to core damage (1988-2005):
  - VSLOCAs have CDFs on the order of 1E-10 (BWRs) and 1E-09 (PWRs) yr^{-1}
    - Small sample of plants confirm. MLOCAs slightly higher but on the order of E-7 to E-8 (PWR, BWRs) yr^{-1} for CDF
    - LERF values ~ 2-3 orders of magnitude smaller than CDF E-9 to E-11 yr^{-1} for LERF
  - Compared to mean CDFs of 1E-05 and 1E-06 yr^{-1}
  - OE Extended to 2020 in INL/EXT-21-63577
  - Plant risk and safety performance have continued to improve (utilities focusing on maintenance and improvements that positively impact risk, safety, and operability)



Considerations/Clarifications for Modernized 50.46a Rule

Consideration/Clarification	Justification
LOCAs > transition break size (TBS) would be beyond design basis. Additional demonstration analyses under 10 CFR 50.46 may not be needed due to extremely low probability of occurrence and associated risk?	• Consistent with intent of 50.46a (SECY-10-0161) • Considerations based on discussions on Reactor Accident Analysis Modernization Report: Item 2.4 (ML24220A292) • Do improvements in PRA quality (RG-1.200) and use of PRA acceptance criteria negate the need for a 50.46 demonstration analysis above the TBS?
LOCAs < TBS (design basis) could take credit for RI single failure, RI for crediting non-safety SSCs, and use alternate criteria to demonstrate high probability?	• LOCAs are not significant contributors to plant risk and ECCS performance is not credited to satisfy dose acceptance criteria? • Based on industry interpretation of RAAM Items 2,2, 2.3, 2.6
Selection of TBS could be based on risk criteria (CDF, LERF)?	• Draft 50.46a rule used initiation event frequency which is not a true risk metric (omits consequences)?
Separate approval of changes under 50.46a may not be required?	• Utilities that have received approval for other RI programs would receive credit for QA of RI change programs? • Evaluation of changes under RG-1.174 for RI programs and 50.59 are well vetted and established.
Site specific seismic risk demonstrations under NUREG-1903 would not be required?	• Utilities have addressed plant specific seismic risk as a part of Post-Fukushima Task Force requirements/recommendations
Implementation of a modernized 50.46a would be voluntary?	• Consistent with 2010 draft 50.46a rule

Overview of Draft Final 50.46c Rule

- Draft final 50.46c rule was a culmination of Commission directives, research findings, petitions for rulemaking, and public comments
  - Intended to be a technology-neutral, performance-based, risk-informed, modern rule
- Major changes proposed in the draft final 50.46c rule, relative to the existing 50.46 rule, shown below

Category	Item	§ 50.46	§ 50.46c
Overall ECCS Methodology	Rule Structure	Prescriptive	Performance-Based
	Applicability	Zircaloy or ZIRLO Cladding	All LWR Cladding
	Burnup Related Phenomena	None	Cladding Inner Surface Oxygen Ingress
	Corrosion Related Phenomena	None	Hydrogen-Enhanced Embrittlement
	Fabrication Related Phenomena	None	Breakaway Oxidation
	Debris Consideration	Implicit	Explicit
	LTC Regulatory Criteria	General	Explicit
	Crud Treatment	None	Explicit
Risk-Informed Alternative to Address the Effects of Debris on Long-Term Cooling	Risk-informed Debris Treatment	N/A	Allowed

Table from SECY-16-0033

Considerations for Improvements to 50.46c Rule

- Technology-neutral improvements:
  - Regulation requires rulemaking or exemptions for all new fuel technologies
- Performance-based improvements:
  - Regulation does not recognize unique performance aspects of advanced LWR design
  - Regulation does not recognize unique performance aspects of ATF designs
  - Regulation maintains prescriptive analytical limits for zirconium cladding
- Risk-informed improvements:
  - Risk-informed aspects limited to treatment of debris
  - Regulation maintains risk-ignorant, deterministic analytical requirements
  - Industry supports Commission direction in SRM-SECY-16-0033 to *apply an appropriate risk-informed regulatory approach to address the research findings on cladding embrittlement*

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SECRETARY

April 11, 2024

MEMORANDUM TO: Raymond V. Furstenau
Acting Executive Director for Operations

FROM: Carrie M. Safford, Secretary

CARRIE SAFFORD Digitally signed by CARRIE SAFFORD Date: 2024.04.11 16:20:14 -04'00'

SUBJECT: STAFF REQUIREMENTS – SECY-16-0033 – DRAFT FINAL RULE – PERFORMANCE-BASED EMERGENCY CORE COOLING SYSTEM REQUIREMENTS AND RELATED FUEL CLADDING ACCEPTANCE CRITERIA (RIN 3150-AH42)

Since SECY-16-0033 was provided to the Commission in 2016, the landscape surrounding advanced light-water reactor fuel has changed. The approaches described in the paper should be reconsidered under the current context, including consideration of topics such as accident tolerant fuel, increased enrichment, and higher burnup. Therefore, under the Internal Commission Procedures, Chapter III, page 14, the Commission is returning this paper to the staff without Commission action.

1. The staff should apply an appropriate risk-informed regulatory approach to address the research findings on cladding embrittlement effects under loss-of-coolant accident (LOCA) conditions described in SECY-16-0033.
2. The staff should evaluate Item 1 with other associated technical issues being addressed, such as fuel fragmentation relocation and dispersal, and risk-informed treatment of LOCAs, including the draft final 50.46a that had been provided in SECY-10-0161.
3. The staff should evaluate whether specific emergency core cooling system criteria such as cladding temperature should be codified or instead addressed in regulatory guidance.
4. Within six months of the date of this SRM, the staff should provide, through a Commissioner Assistant's Note, an action plan for the above items.

cc: Chair Hanson
Commissioner Wright
Commissioner Caputo
Commissioner Crowell
OGC
CFO
OCA
OPA
ODs, RAs, ACRS, ASLBP
PDR

Considerations for Improvements 50.46c Rule

- Streamlining and removing unnecessary content in the rule is essential to meet the schedule:
  - Fuel design-specific requirements on performance (50.46c(g)), analytical models (50.46c(d)(2)), and reporting (50.46c(n)) could be moved to regulatory guidance (Draft RG 1.224)
  - Alternative risk-informed approach for addressing the effects of debris on long-term core cooling (50.46c(e)) and reporting requirements (50.46c(m)(6)-(8)) no longer need to be codified since GSI-191 was successfully closed without new regulation
  - Industry focus group exercise reduced rule content from 14 pages down to 1.5 pages
- High-level ECCS performance requirements (50.46c(d)(1)) may need to be more technology-neutral and allow alternate performance metrics for ATF and LWR designs
- Level of adequate protection for compliance demonstration should be commensurate with risk to public health and safety
  - Separate performance and analytical requirements above and below transition break size

Implementation of New Requirements

- Draft final 50.46c rule was designated as adequate protection and exempt from 10 CFR 50.109, *Backfitting,* requirements
  - Risk-significance, safety margins, and implementation costs were not considered
  - Mandatory, regimented implementation schedule
- Given the low risk profile of postulated LOCAs and existing plant-specific analyses which demonstrate low safety-significance associated with 50.46c research findings, the staff's prior adequate protection exception to the Backfitting Requirements in 10 CFR 50.109 should be reconsidered
  - Adequate protection exception does not reflect latest revision to Management Directive 8.4, *Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests*
  - Implementation costs significantly underestimated
- Similar to 50.46a, final rule should be a voluntary, alternative to existing 50.46 requirements:
  - Some plants may not seek enabling changes such as ATF, LEU+, HBU, and/or power uprates and should not incur unnecessary burden to maintain their current licensing basis

Discussion Period

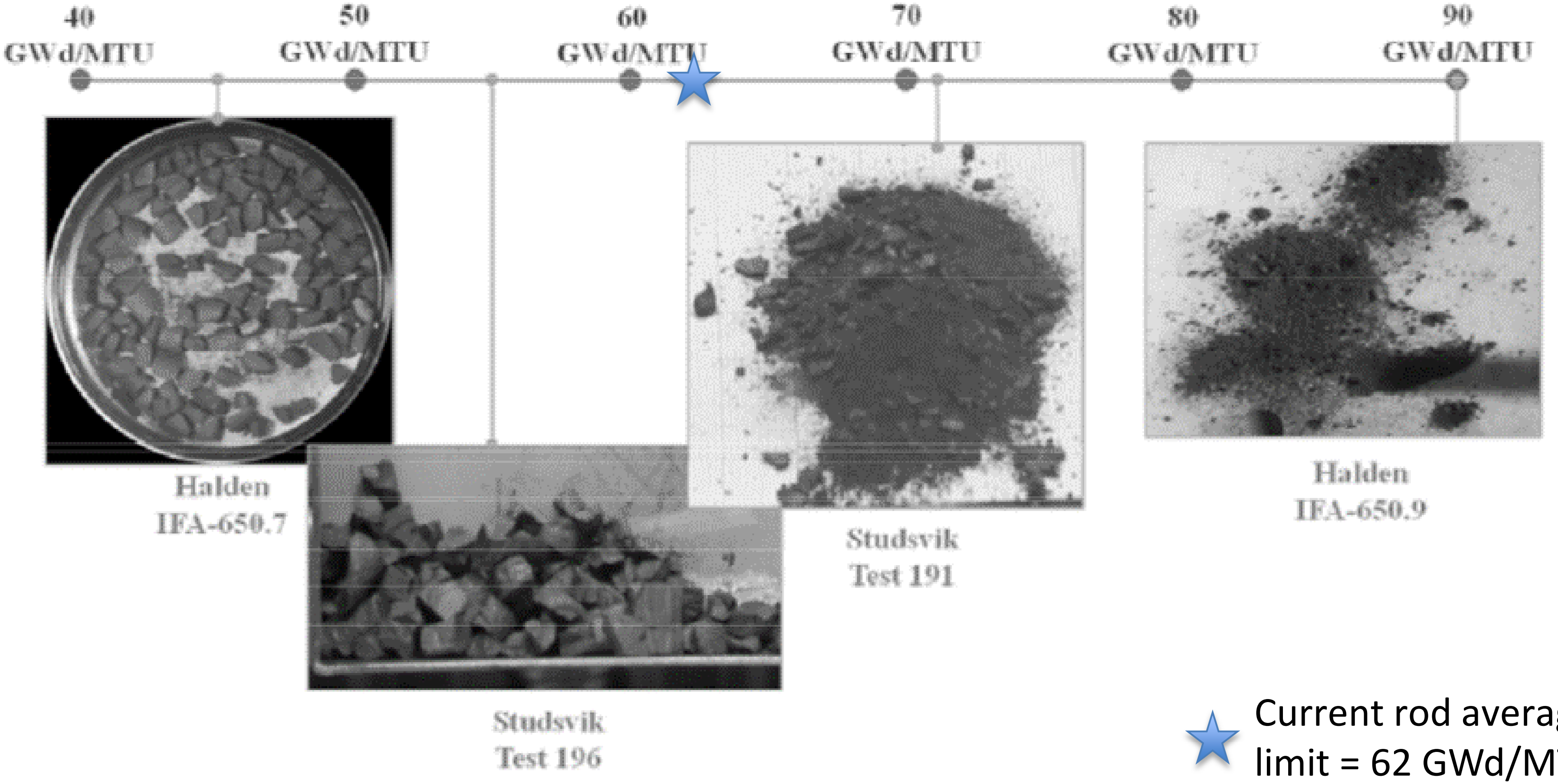
Break

NRC’s Phenomena Identification and Ranking Tables for High Burnup Fuel Fragmentation, Relocation, Dispersal, and Its Consequences for Design-Basis Accidents

James Corson

NRC Office of Nuclear Regulatory Research

Experiments have shown that fuel can fragment during Loss-of-Coolant Accident



★ Current rod average burnup limit = 62 GWd/MTU

Fuel Dispersal PIRT Overview

- NRC is sponsored a phenomena identification and ranking table exercise to address fuel dispersal and its consequences
  - PIRT will help identify areas in need of additional research
  - PIRT will also help to develop regulatory criteria related to fuel dispersal
- PIRT has been documented in NUREG/CR-7307, published in June 2024

PIRT Process

1. Define the issue that is driving the need for a PIRT
2. Define the specific objectives for the PIRT
3. Define the hardware and the scenario for the PIRT
4. Define the evaluation criterion
5. Identify, compile, and review the current knowledge base
6. Identify phenomena
7. Develop importance ranking for phenomena
8. Assess knowledge level for phenomena
9. Document PIRT results

Fuel Dispersal PIRT Panelists

- Fran Bolger, EPRI
- Nathan Capps, ORNL
- Dave Kropaczek, Veracity Nuclear
- Wade Marcum, Oregon State University
- Kurshad Muftuoglu, EPRI
- Gretar Tryggvason, Johns Hopkins University
- Wolfgang Wiesenack, Halden Reactor Project (retired)

Scope of PIRT

- Panelists evaluated impacts of extending burnup beyond 62 GWd/MTU rod-average
  - Previous NRC-sponsored PIRTs cover behavior up to 62 GWd/MTU (e.g., NUREG/CR-6742 and NUREG/CR-6744 on RIA and LOCA, respectively)
  - Goal was to consider what would change when going to higher burnups
- PIRT focused on UO2 fuel in zirconium alloy cladding

Scenarios Evaluated in the PIRT

- Large-break LOCAs
  - Scenario most likely to lead to significant fuel dispersal
  - Initiated by a double-ended guillotine break of a large pipe in the reactor coolant system
    - Typically in the recirculation loop in a BWR or the cold leg in a PWR
  - Results in rapid loss of coolant, core uncovery, and fuel heatup
    - May lead to fuel rod ductile failure due to ballooning and burst
  - Accident mitigated by emergency core cooling system injection to refill the vessel and quench the core

Scenarios Evaluated in the PIRT (2)

- Reactivity-initiated accidents (RIAs)
  - Initiated by a control rod ejection (PWR) or control blade drop (BWR)
  - Leads to a rapid power increase in nearby fuel rods, potentially leading to fuel rod failure
  - Fuel dispersal would require either very high energy deposition in the fuel (already precluded by limits in Regulatory Guide 1.236) or fuel rod ductile failure
- Fuel handling accidents
  - Initiated by dropping a fuel assembly (i.e., during refueling operations)
  - May result in fracture of the rod
  - May result in small amount of fuel dispersal, but release mechanism is very different from LOCAs and RIAs

Figures of Merit for the PIRT

- LBLOCA: fuel coolability and recriticality
  - Coolability in terms of both existing embrittlement criteria in 10 CFR 50.46 and in terms of debris bed coolability
  - Recriticality only evaluated for small number of phenomena
- RIA: fuel failure and radionuclide releases
- FHA: radionuclide releases

PIRT Rankings

- Panelists ranked phenomena according to importance, knowledge level, and uncertainty
  - 123 LBLOCA, 14 RIA, and 6 FHA phenomena ranked
  - Items are ranked High (H), Medium (M), or Low (L)
- Panelists discussed each item to aid in their rankings, but panel consensus was not required
  - Instead, tables tally the H, M, and L votes for each item

PIRT Findings: LBLOCA

- 6 phenomena assigned High importance, Low knowledge level, High uncertainty by majority of panelists
  - Transient fission gas release
  - Burst opening size relative to the fuel fragment size distribution
  - Impact of coolant mass flux on particle mobility
  - 3 phenomena related to potential for fuel accumulation on spacer grids
- Many other phenomena ranked H importance but M/H knowledge level
- Panelists concluded recriticality likely not an issue
  - Also stated that this could be demonstrated using existing neutronics codes

PIRT Findings: RIA

- Transient fission gas release was the only High importance, Low knowledge level, High uncertainty item (with respect to fuel failure)
- Several items had H importance but M/H knowledge level
  - E.g., core loading pattern and power distributions influence fuel failure but can be calculated accurately
- Overall, the panel concluded that fuel dispersal would be limited for RIA
  - Expectation is that PCMI failures dominate
  - PCMI failures characterized by axial split, rather than large openings associated with ballooning and burst

PIRT Findings: FHA

- No H, L, H items
- Panel concluded that mechanical fracture would result in minimal release of fuel fragments and additional fission gas
  - Some data exists to quantify mass of fuel fragments and fragment size distribution from high burnup fuel under FHA conditions

Conclusions from the Fuel Dispersal PIRT

- Understanding how much material disperses is crucial to demonstrating coolability
  - Key parameters influencing dispersal include transient FGR, fuel fragment size distribution, cladding burst characteristics, spacer grid characteristics, core flow patterns during the transient, and core loading pattern
  - Some parameters can be calculated fairly accurately (e.g., core loading pattern, core flow)
  - Other parameters are less well known and highly uncertain (e.g., transient FGR, fragment size distribution, burst opening size, impact of spacer grids on debris trapping)
- Dispersal of fuel fragments remains poorly understood
  - However, the panel believes it should be possible to perform simplified analyses to demonstrate coolability so long as the dispersed mass remains low

Conclusions (2)

- "Generally, the Panel members believe that FFRD during a LOCA is unlikely to cause serious coolability issues ..."
  - Conclusion based on expectation that the extent of fuel dispersal would be limited and on current understanding of particle size distribution
  - However, many important phenomena have medium or low state of knowledge and high uncertainty, which would affect the amount of fuel dispersed and its impact on coolability
  - Experimental and/or analytical work is needed to verify this conclusion

Next Steps

- NRC is participating in international experimental programs aimed at studying FFRD
  - Studsvik Cladding Integrity Project (SCIP) just kicked off its 5th phase this year
  - HBU-LOC project under the Second Framework for Irradiation Experiments (FIDES-II) will be performing tests in next couple of years
- NRC remains engaged with modeling efforts by the national laboratories that seek to quantify the mass of fuel that could be dispersed
- NRC is incorporating insights from the PIRT in guidance being developed for the increased enrichment rulemaking

High Burnup Workshop

Proposed FFRD dose consequence guidance in DG-1425 (Draft Rev 2 of RG 1.183)

Sept 3, 2024

Topics

- Public Draft Guide Excerpts File (ML24226B262) Related to Proposed FFRD Dose Consequence

- Regulatory Source Terms and FFRD
  - MHA-LOCA
  - 50.46 LOCA
  - Non-LOCA

Different Analyses for Different Defense-in-Depth Purposes

Public Draft Guide Excerpts File

- Disclaimer:

  Note: this document contains draft/predecisional guidance language to support discussions during the NRC's High Burnup Workshop and other public meetings, as needed. The NRC staff are not officially accepting comments during the public meetings and will not provide any responses to feedback provided during public meetings.

  The contents of this document are being considered for inclusion in staff guidance in DG-1425. The purpose of this document is to engage stakeholders and receive informal feedback. This document has not been subject to NRC management or legal reviews or approvals, and its contents are subject to change and should not be interpreted as official agency positions.

- Draft language being considered by staff, does not have the benefit of full internal review
- Purpose –
  - Continued opportunity for early public engagement in the regulatory process
  - Support discussions during this meeting
  - Enable decision-making
  - Support comprehensive communications with the Commission
- Staff does not intend to formally respond to comments or accept additional comments pertaining to this document, after this meeting
  - Public comment period for the DG will be aligned with the Increased Enrichment Rulemaking (Docket ID NRC-2020-0034, www.regulations.gov)

MHA-LOCA and FFRD

- **Regulations**: 10 CFR Part 100, 50 and 52.
  - Required for the purposes of licensing nuclear power plants, that radionuclide releases to reactor containments associated with a “substantial meltdown” of the reactor core be postulated.
  - Supports the evaluation of engineered safety features and barriers used to mitigate release of fission products to the environment.
  - Consequences of these radionuclide releases are evaluated assuming that the containment remains intact and leaks at the design-basis leak rate.
- **Acceptance Criteria**: 25 rem TEDE to an individual at any point on the boundary of the exclusion area boundary for any 2-hour period, 25 rem TEDE on the outer boundary of the low population zone during the entire period of the fission product release passage.
- **Conclusion**: MHA-LOCA source term bounds FFRD with no need to consider additional radionuclide releases. (Reg 1.183 Rev. 1 provides guidance)
  - NRC Internal Memorandum (ML21197A067) provides the basis for this conclusion.

Current 50.46 LOCA Analysis

- **Regulation**: (GDC – 35, 10 CFR 50.46, and Appendix K to 10 CFR 50)
  - A design basis accident which establishes a reliable long-term core cooling capability with high-capacity emergency makeup systems.
  - Ensures the facility could safely cope with a major loss-of-coolant-accident generally from a double-ended guillotine break of the largest pipe in the reactor coolant system.
  - If the amount of coolant in the reactor is insufficient to provide cooling of the reactor fuel, the fuel would be damaged, resulting in loss of fuel integrity and release of radiation.
  - If coolant is lost from the reactor coolant system and the event cannot be terminated (isolated) or the coolant is not restored by normally operating systems, it is considered an “accident” and then subject to mitigation and consideration of potential consequences.
- **Acceptance Criteria**: must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents conforms to the criteria set forth in paragraph (b) of 10 CFR 50.46 for (1) peak cladding temperature, (2) maximum cladding oxidation, (3), maximum hydrogen generation, (4), coolable geometry, and (5) long-term cooling.
- **Conclusion**: Current 10 CFR 50.46 LOCA analysis do not typically result in appreciable radiological releases. Instead, the normal operation source term is typically utilized.

Proposed 50.46 LOCA and FFRD Analysis

- Assessing radiological consequences from a 50.46 LOCA analysis which predict FFRD appears to conflict with the 10 CFR 50.46 acceptance criteria.
- FFRD-specific source term continues to be difficult to quantify but justifiably bounded by MHA-LOCA source term. NRC Internal Memorandum (ML21197A067) provides the basis for this conclusion.

Non-LOCA and FFRD

- RG 1.183, Rev. 1, Regulatory Position 3.2, states, in part:

> At the time of issuance of Revision 1 of this RG, no consensus exists on the mechanism or the computation of T_{FGR} for these events; therefore, future applicants should address this using engineering judgment or experimental data. Though not fully applicable to non-LOCA and non-reactivity-initiated DBAs, NRC Research Information Letter 2021-13, "Interpretation of Research on Fuel Fragmentation Relocation, and Dispersal at High Burnup," issued December 2021 (Ref. 29), provides data that can be used to provide a bounding estimate of T_{FGR} for high-temperature DBAs.

- **Conclusion**: State-of-knowledge for a bounding FFRD source term from a Non-LOCA DBAs has not progressed since issuance of Rev. 1

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Protecting People and the Environment

Feedback and Discussion

Environmental Evaluation of ATF with Increased Enrichment and Higher Burnup Levels

Donald Palmrose, Ph.D.

Senior Reactor Engineer

Office of Nuclear Material Safety and Safeguards (NMSS)

Key Points

- Evaluate environmental impacts of deployment and use of near-term ATF with increased enrichment (IE) and higher burnup (HBU) levels to support future licensing reviews
- Evaluation of impact in the uranium fuel cycle (UFC), transportation of fuel and waste, and decommissioning
- Up to 10 wt % U-235 and 80 GWd/MTU for UFC and Decommissioning
- Up to 8 wt% U-235 and 80 GWd/MTU for transportation of fuel and waste
- Continued Storage GEIS remains applicable
- Address release fraction uncertainty at HBU levels

NUREG-2266 Final Report

- Focus is on near-term ATF technologies
  - 1st generation: Coated cladding, doping
  - 2nd generation: FeCrAl cladding
- Longer-term ATF technologies not covered
  - UN pellets, SiC cladding, and extruded metallic fuel
- Near-term ATFs do not significantly change fuel fabrication impacts, radiological inventory, and releases fractions
- Evaluated the following enrichment and burnup levels:
  - Up to 10 wt% U-235 and 80 GWd/MTU for UFC/Decommissioning
  - Up to 8 wt% U-235 and 80 GWd/MTU for transportation of fuel and waste
- Evaluated one-third and half core reloads and provided other clarification based on public comments

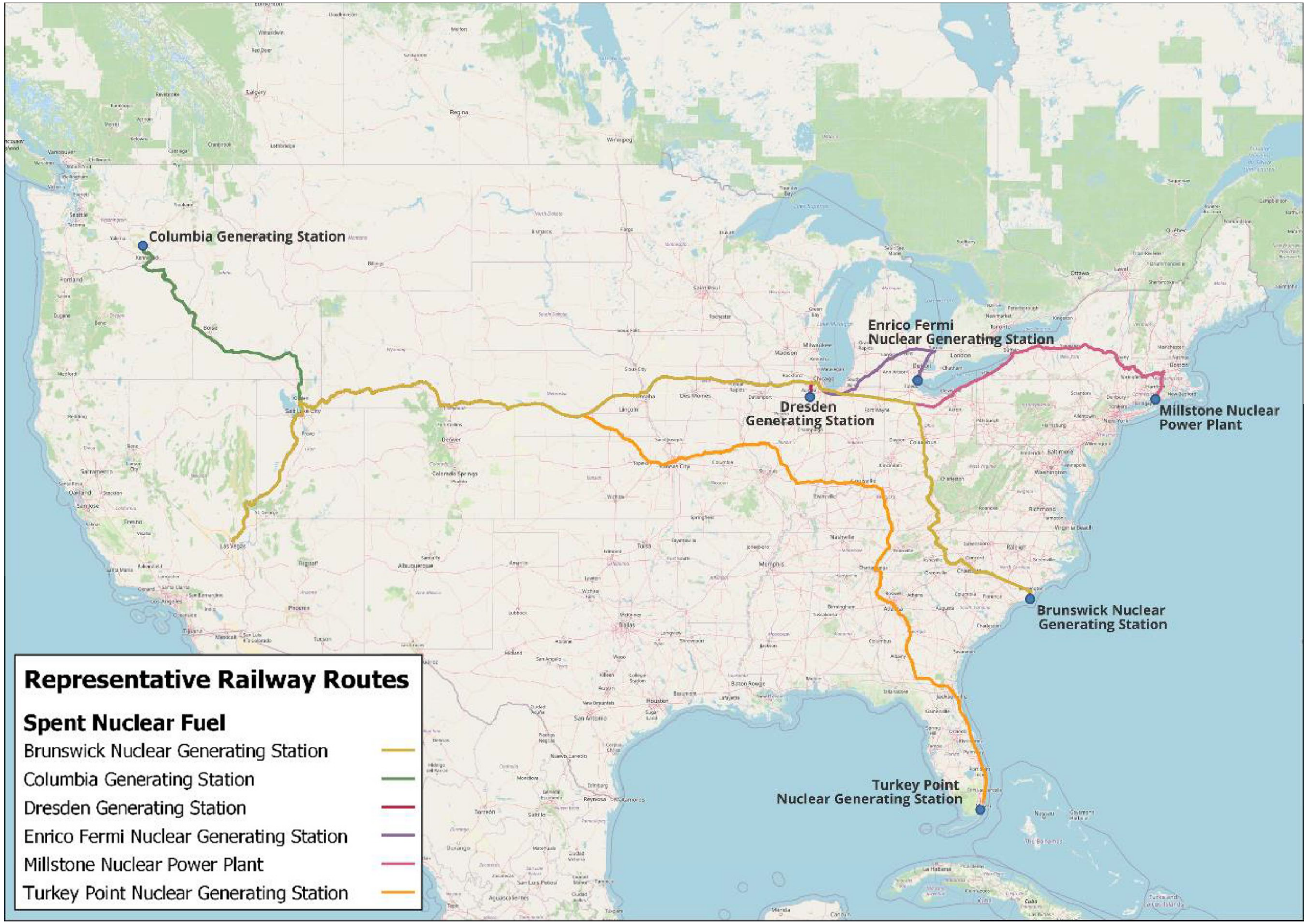
Uranium Fuel Cycle

- UFC observations and impacts were:
  - The greater amount of yellowcake to support IE would not cause a significant change in related Table S-3 impacts
  - Gaseous centrifuges use less electricity for 10 wt % U-235 than gaseous diffusion did for 4 wt % U-235
  - Longer refueling cycles would reduce the rate of spent nuclear (used) fuel generation
  - Spent ATF management would be consistent with Continued Storage (CS) GEIS (NUREG-2157) analysis
- UFC conclusion is Table S-3 and CS GEIS would bound or still apply to deployment and use of ATF with IE & HBU levels

Transportation of Fuel and Waste

- Use of DOE & NRC guidance & the code NRC-RADTRAN (radiological transportation risk) with WebTRAGIS (routing)
- Six sites selected by NRC Regions
  - Region I – Millstone Power Station (PWR)
  - Region II – Turkey Point Nuclear Generating Units (PWR), Brunswick Steam Electric Plant (BWR)
  - Region III – Enrico Fermi Nuclear Generating Station Unit 2 (BWR) and Dresden Nuclear Power Station (BWR)
  - Region IV – Columbia Generating Station (BWR)
- Standard PWR and BWR Type B packages selected for fresh fuel
- Smallest certified Type B packages selected for spent nuclear fuel


Columbia Generating Station
Framatome
Enrico Fermi
Nuclear Generating Station
Dresden
Generating Station
Millstone Nuclear
Power Plant
Brunswick Nuclear
Generating Station
Turkey Point
Nuclear Generating Station
Representative Highway Routes
Spent Nuclear Fuel
Brunswick Nuclear Generating Station
Columbia Generating Station
Dresden Generating Station
Enrico Fermi Nuclear Generating Station
Millstone Nuclear Power Plant
Turkey Point Nuclear Generating Station
Unirradiated Fuel
Framatome to Turkey Point


Columbia Generating Station
Enrico Fermi
Nuclear Generating Station
Dresden
Generating Station
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Generating Station
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Nuclear Generating Station
Representative Railway Routes
Spent Nuclear Fuel
Brunswick Nuclear Generating Station
Columbia Generating Station
Dresden Generating Station
Enrico Fermi Nuclear Generating Station
Millstone Nuclear Power Plant
Turkey Point Nuclear Generating Station

Transportation of Fuel and Waste Results

- Normalized annual truck shipments of spent ATF based on 2-yr refueling cycle for 1100 MWe reference NPP of WASH-1238 and Table S-4
  - 30 shipments for PWRs and 52 shipments for BWRs
- Normal, incident-free, conditions are bounded by Table S-4
  - Worker doses less than the 4 person-rem of Table S-4
  - Cumulative public doses, while generally higher than 3 person-rem of Table S-4 due to increases in populations along routes since WASH-1238 but not a significant impact due to the very low average individual doses
  - Average individual doses were <<1 mrem and within the Table S-4 ranges of doses for onlookers and along route populations
- Radiological accident risks still small as in Table S-4
- Non-radiological accident risk (fatalities/injuries) greater than radiological risks but also bounded by Table S-4

Decommissioning

- Decommissioning GEIS - NUREG-0586 Supplement 1
  - Extensively discussed in 2013 and 2024 License Renewal GEIS (NUREG-1437 Rev 1 and 2) and in past new reactor EISs
- ATF deployment, use, and subsequent termination of operations would only affect human health and waste management
  - All other resource areas would be the same or slightly less
- ATF deployment effects on decommissioning:
  - Effluent releases would still be lower after cessation of ops
  - Worker doses still controlled per 10 CFR Part 20
  - Would not alter the practices employed to manage the wastes
  - Would need less ISFSI capacity than staying with current fuels
- Decommissioning GEIS would bound deployment and use of ATF with IE and HBU levels

Guidance for Analysts

- If > 8 wt% U-235 enrichments, new transportation analysis could apply the methodology in Section 3 using the data sources documented in Appendix A through Appendix D
- If > 10 wt% U-235 enrichment, assess uranium fuel cycle and decommissioning impacts by applying the rationale in Sections 2 and 4, respectively
- If > 80 GWd/MTU, appropriate revised transportation radiological releases needs to be evaluated and applied in the transportation analysis along with assessing the related changes (if any) in spent fuel management to the Continued Storage and Decommissioning GEISs
- The staff encourages potential applicants to discuss the above as part of any pre-application discussions

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Protecting People and the Environment

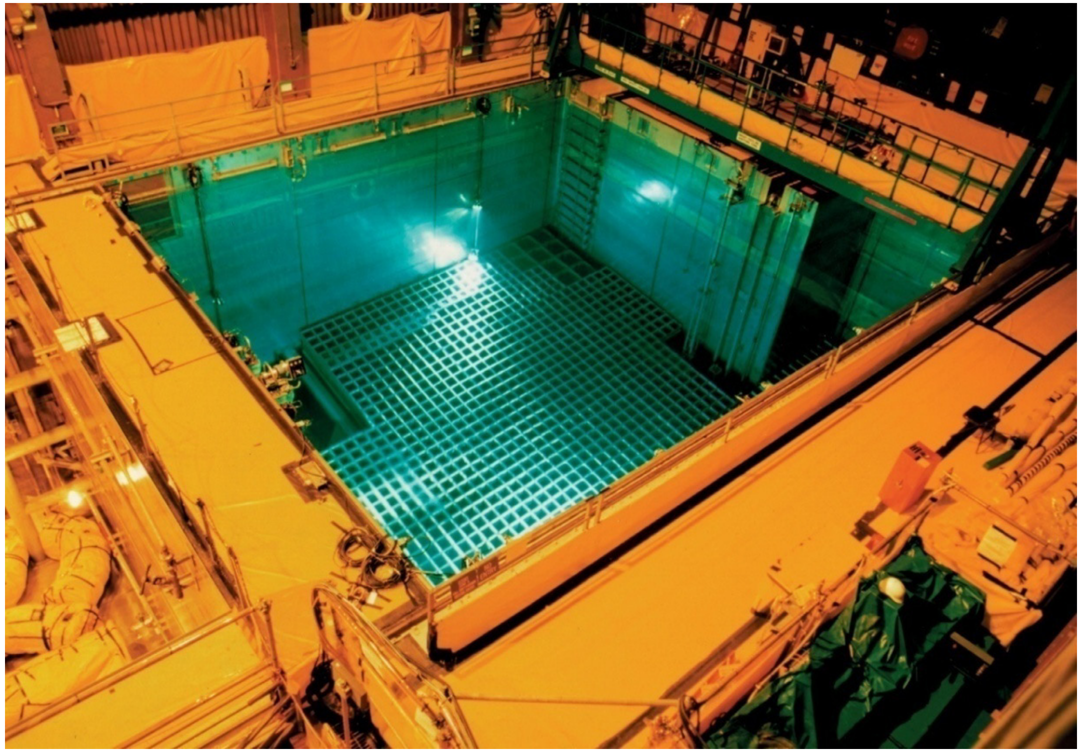
Spent Fuel Pool 2024
A Regulatory Perspective

Kent Wood
U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

High Burnup Workshop
September 3, 2024

Overview

- RegGuide 1.240
- ATF/IE/HBU
- Key Considerations



Picture: Spent Fuel Pool

Guidance

- Reg Guide 1.240 “Fresh and Spent Fuel Pool Criticality Analyses”
  - March 2021 ML20356A127
  - Endorsed NEI 12-16, R4
  - “…clarifications and exceptions…”
    - “o” guidance based on experience with LWR fuel to that point.

ATF/IE/HBU

- ATF/IE/HBU
  - ATF: which are you implementing?
  - IE: U235 up to 10 wt/% enriched?
  - HBU: up to 80 GWD/MTU?
  - What guidance from Reg Guide 1.240 and NEI 12-16 R4 applies?

ATF/IE/HBU

- LARs to date
  - Vogtle ATF/IE LTAs
    - Exemption to 10CFR50.68(b)(7): Enrichment
    - Doped Fuel =>Increased Theoretical Density => more fissile material
  - Byron and Braidwood
    - Full Batch
    - 6.5 wt/% U235

Key Considerations

- Depletion codes
  - Limitations and Conditions
- What does your future look like?
  - Transition cores
  - Equilibrium cores
- Existing work
- References/Precedents

Key Considerations

- Fuel assembly physical changes
  - – e.g. pellet diameter, fuel rod diameter…
- Changes in Operation
  - – e.g. load following…
- Pre-submittal meetings
  - – Recommended well ahead of submittal
  - – Novel and Unique situations
    - Early and maybe more than one
- High quality submittals

Oak Ridge National Laboratory: High Burnup Summary

Nathan Capps, cappsna@ornl.gov

Technical Team: Jason Harp, Peter Doyle, Yong Yan, Bob Morris, Chuck Baldwin, Mackenzie Ridley, Diego Muzquiz

Strategy to Enable Burnup Extension

Goal	Strategy	NRC RIL 2021-13
Enable and expand the safe and economic operation of the US LWR fleet beyond the current regulatory limits by developing the capabilities and performing the necessary research to support fuel performance beyond 62 GWd/tU	Perform integral and semi-integral LOCA testing	All five elements
	Develop in-situ data acquisition capabilities for performance and safety-critical phenomena (e.g., transient FGR, strain, strain rate, etc.)	Element 1, 2, and 4
	Determine the impact of irradiation conditions on microstructure changes governing fuel and cladding response at the macro-scale	Element 1, 2, and 4
	Develop characterization to model or limit dispersal mass	Element 3

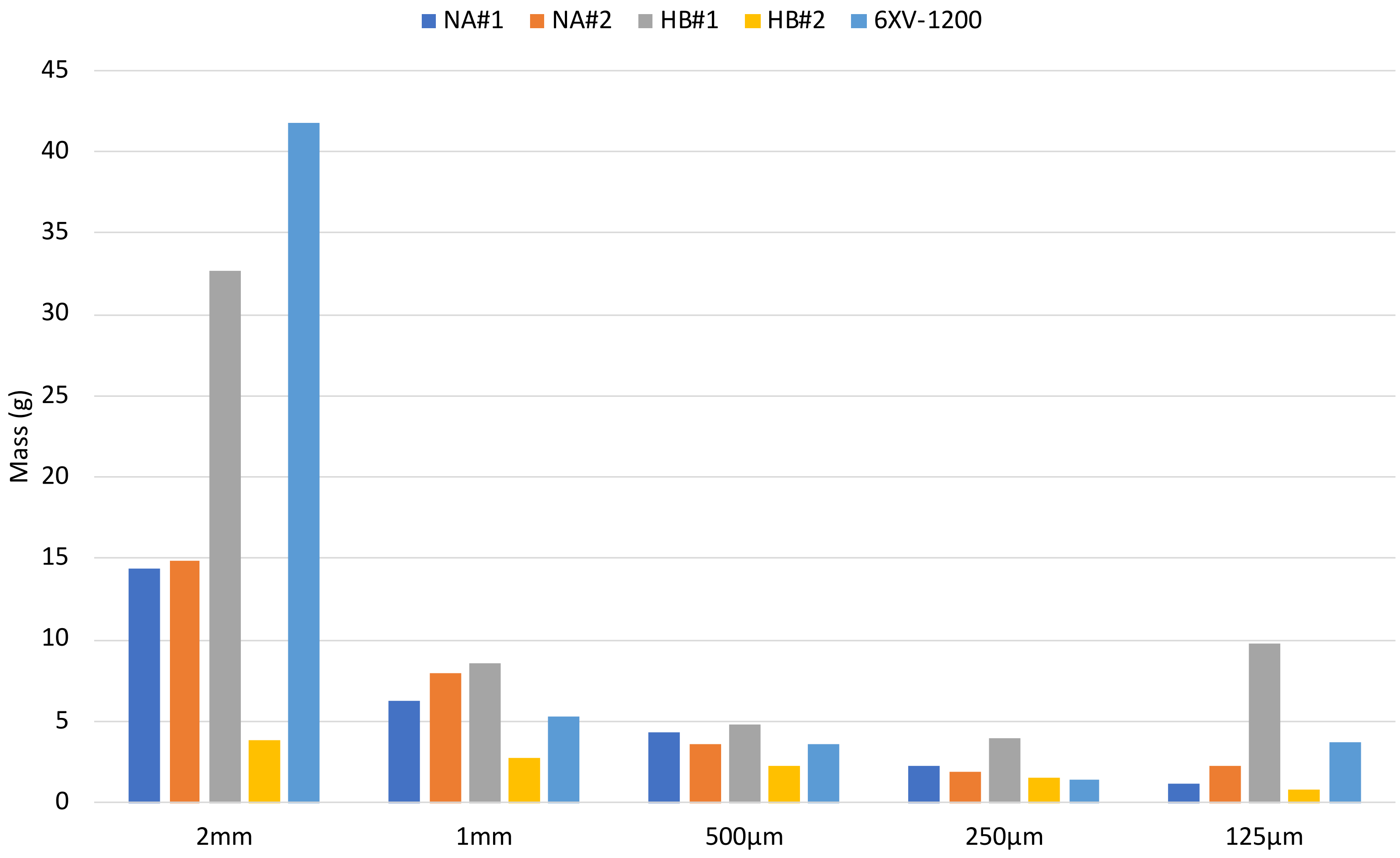
Semi-Integral LOCA Testing

Parameter	HBR#1	NA#1	NA#2	HBR#2*	NA-GS	6XV⁺⁻
Fuel	H.B. Robinson	North Anna	North Anna	H.B. Robinson	North Anna	Byron
Materials	Zry-4	M5	M5	Zry-4	M5	Protected
Father rod burnup (GWd/MTU)	66.5	63	68.5	66.5	63	73.7
Outside diameter (mm)	10.77	9.5	9.5	10.77	9.5	9.144
Wall thickness (mm)	0.76	0.57	0.57	0.76	0.57	0.575
Internal pressure at 300°C (MPa)	8.27	8.27	8.27	8.27	8.2	8.27, 11.0, 13.8
Temperature ramp from 300°C (°C/s)	5	5	5	5	5	5
Temperature at burst (°C)	770	791	816	770	861	Protected
Terminal temperature (°C)	1,000	1,200	1,000	1,000	1000	900
Burst shape	Oval	Oval	Oval	Oval	Oval/Oval	Oval
Burst length (mm)	7	16	-	8.4	4.7/5.6	24.6/TBD/22.9
Max. burst width (mm)	~2	~3	-	1.5	0.64/0.64	8.5/TBD/9.1
Max strain $(\Delta C/Cm)_{max}$ (%)	25	41	-	20	41	Protected

*Segment length = 7.9"
⁺Segment Lenth = 8.5"
⁻PIE in progress

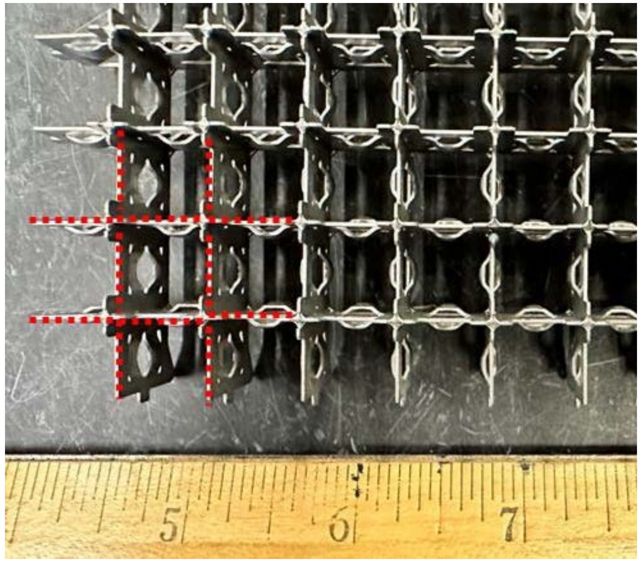
Summary: Removed Fuel Sieving

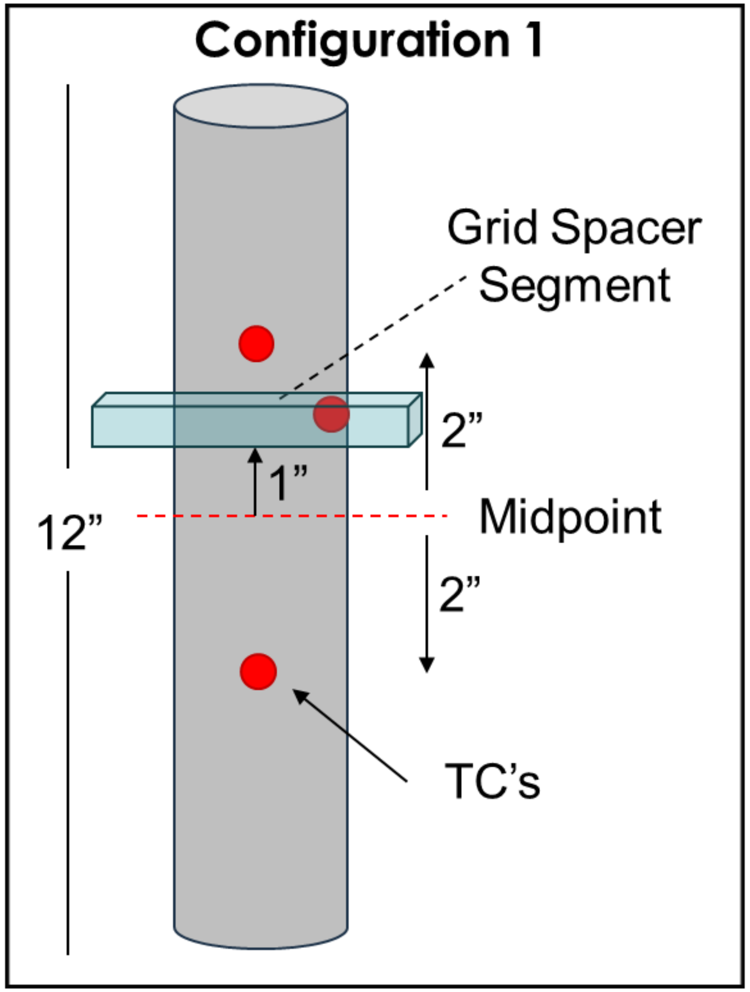
- Total fragment size distribution
  - Cut sample and dumped all mobile material
- Additional steps were taken to assess dispersal
- Ongoing PIE on remaining LOCA tests

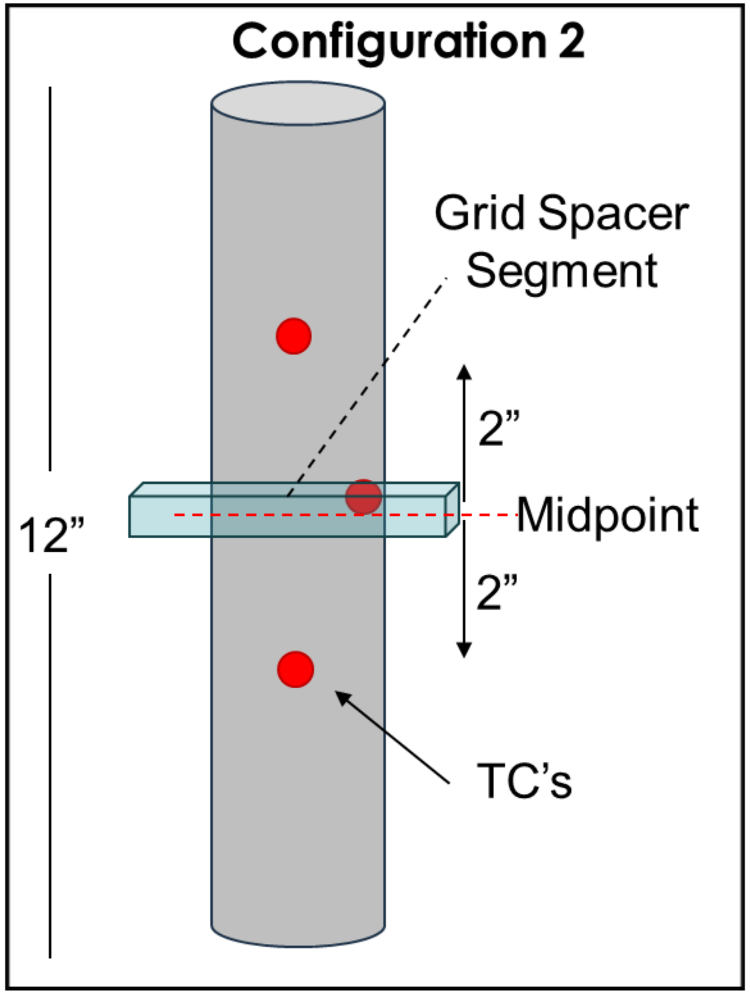


In-Cell Grid Spacer Test Conditions

- Single spacer segment fits within quartz reaction tubing
- Configuration 1: Cr/Zry & Zry
- Configuration 2: Zry
- 1200 psi, open valve
- 5°C/s ramp rate until rupture







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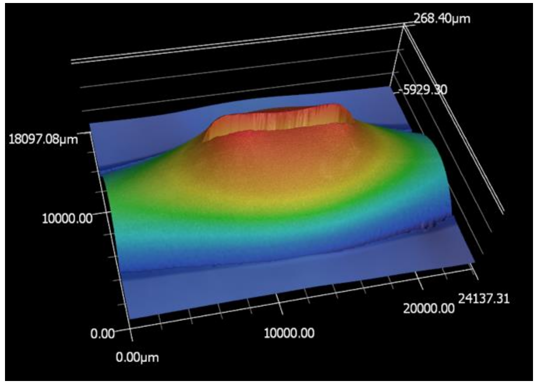
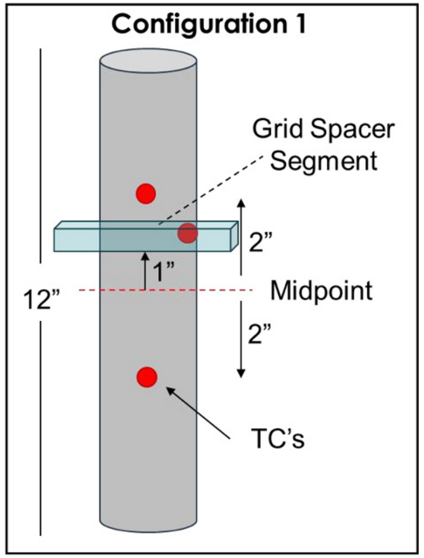
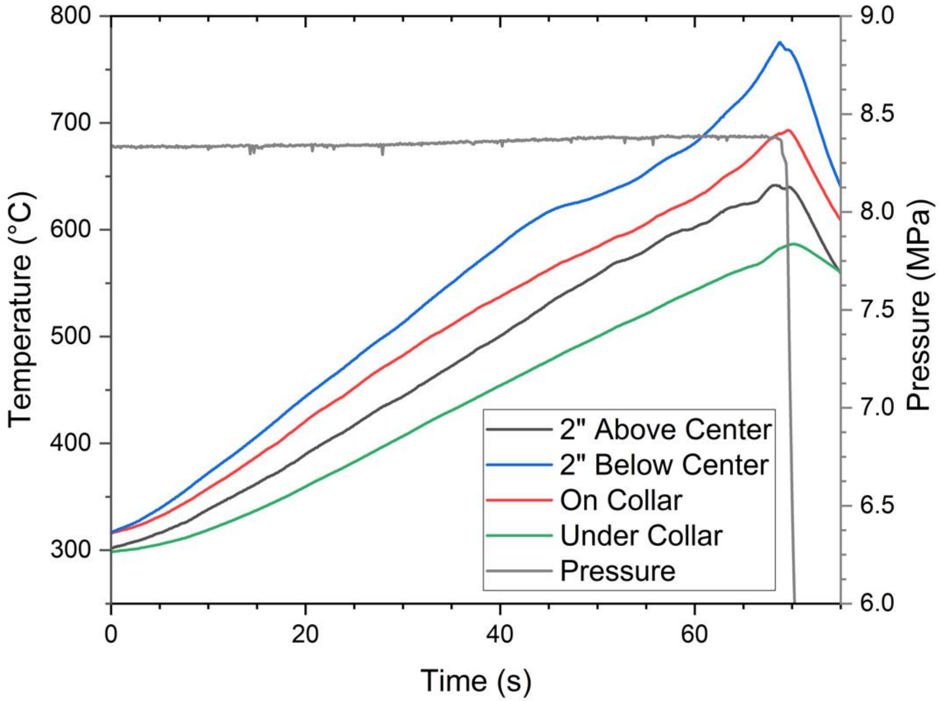
Out-of-Cell Benchmark for In-Cell Test

- Thermocouple above and below simulated spacer grid (worm clamp)
- Rupture at lower TC 2” below cladding centerline
- Estimate ~100°C delta T through the SS collar
- Simulated grid spacer restricted cladding diametric strain
- Experimental results consistent with Modeling Predictions

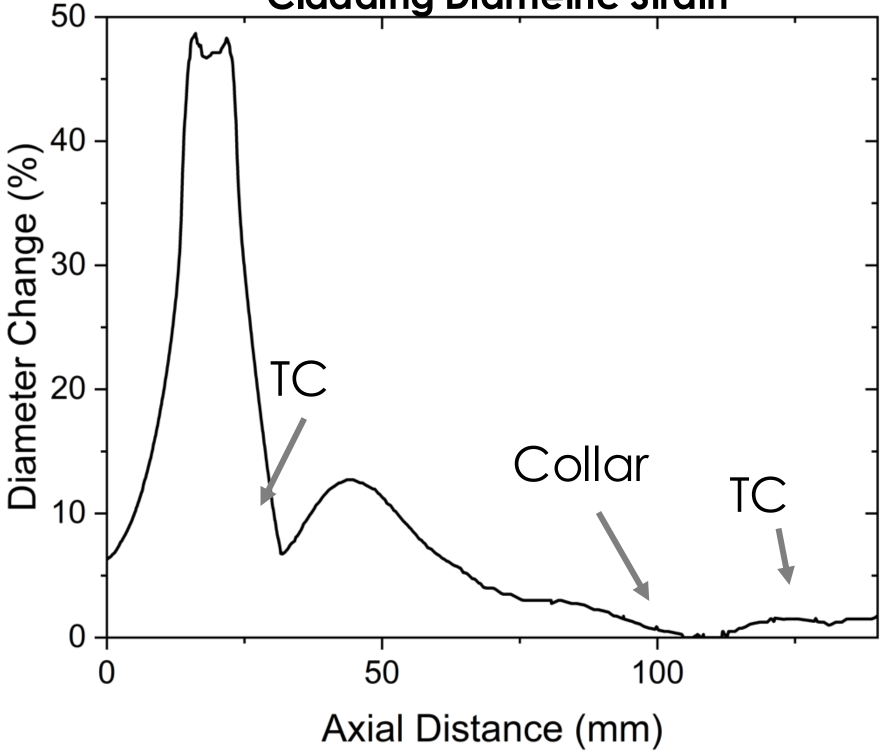
3D Scan of Rupture Opening







Cladding Diametric Strain



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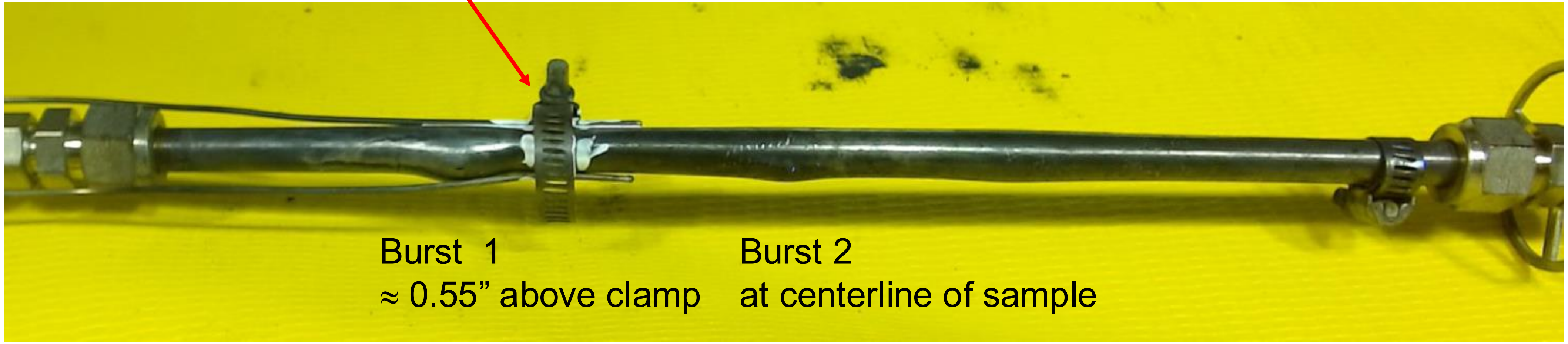
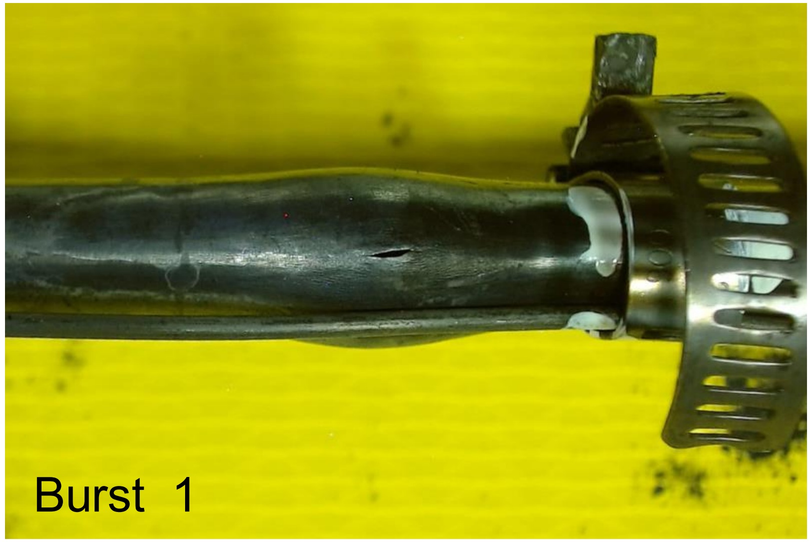
Clamp: 2” above sample centerline.



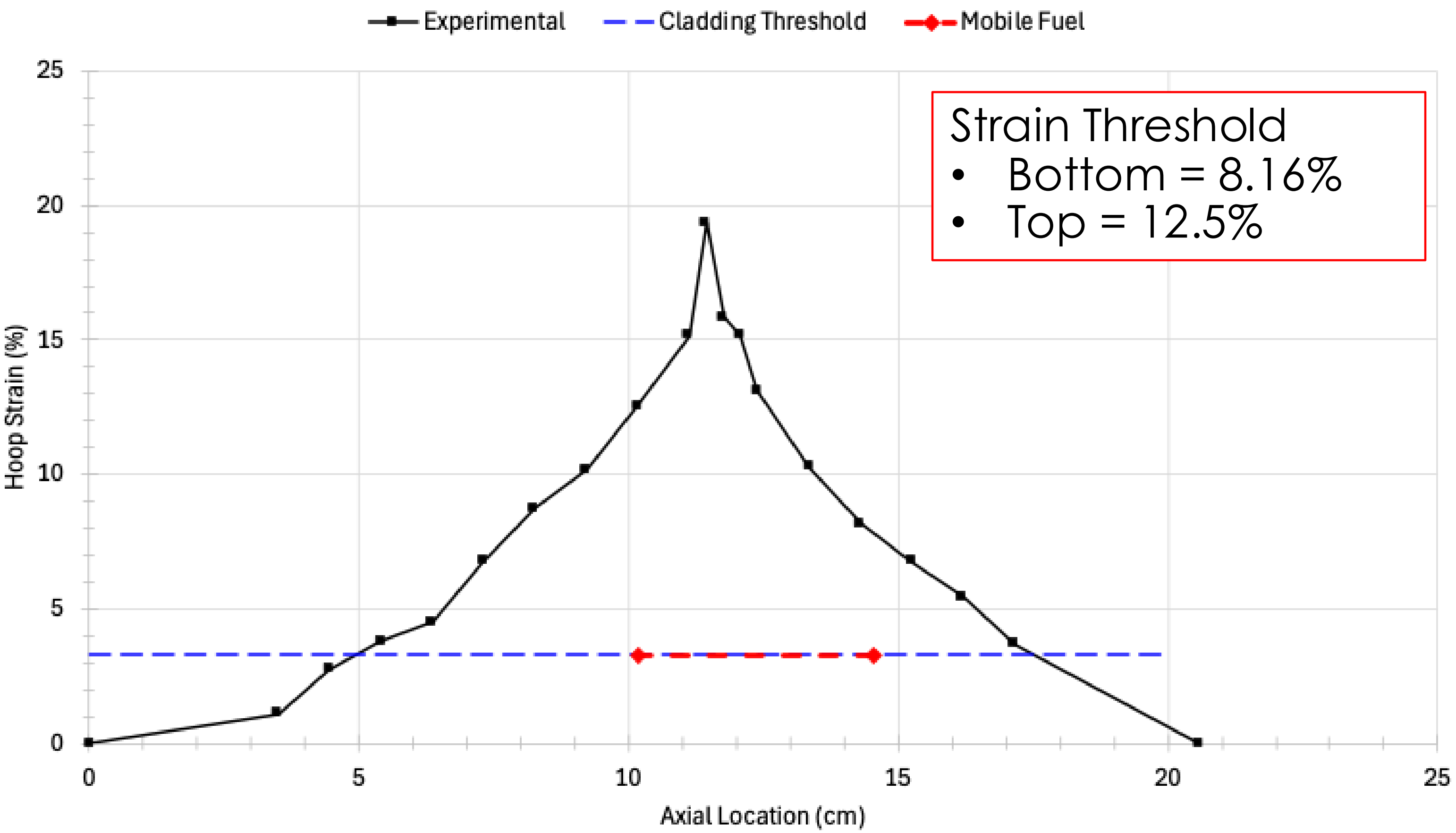


Opening Length = 0.185” Width = 0.025”
OD at 0°: 0.505” at 90°: 0.470”

Opening Length = 0.220” Width = 0.025”
OD at 0°: 0.483” at 90°: 0.458”

Cladding Strain Threshold

- Generally, agrees with the NRC RIL on FFRD
  - Key Difference: Results suggest there may be a threshold more applicable to the top and bottom
  - More data needed to verify



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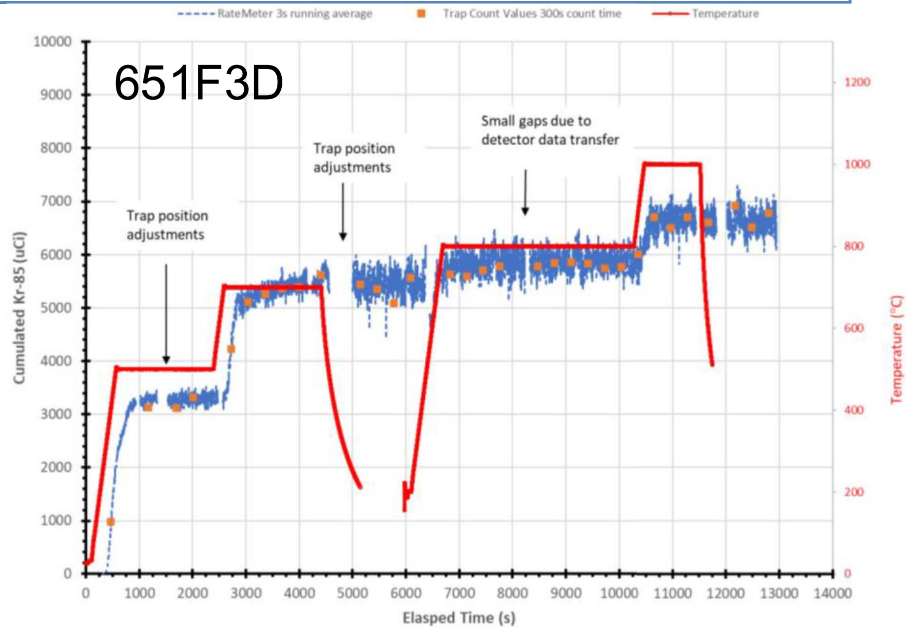
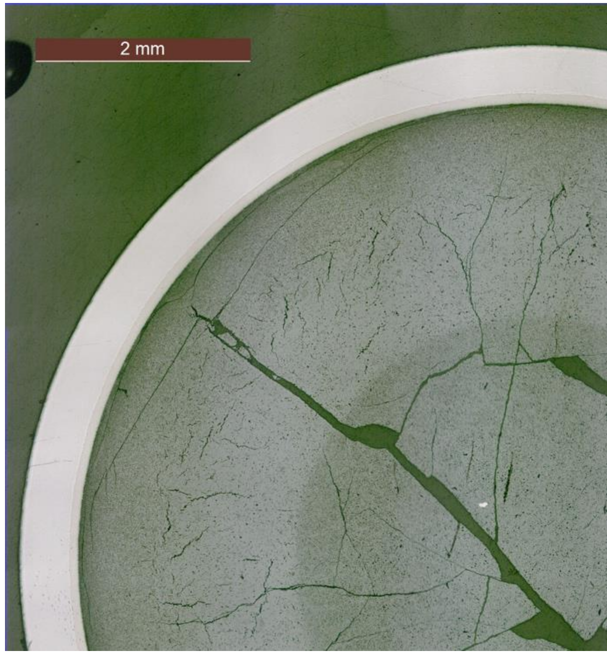
Transient Fission Gas Release

Parameter	651F3D	6XV-A3	NA-GS	Arrested FGR
Fuel	North Anna	Byron	North Anna	North Anna
Sample burnup (GWd/MTU)	68.5	75	71	68
Segment Length	~two pellets	~two pellets	12"	12"
Applied Pressure	Cladding Constraint	Cladding Constraint	8.2 MPa	6.2 MPa
Temperature ramp from 300°C (°C/s)	5	5	5	5
Temperature Conditions (°C)	Stair Step	LOCA	LOCA	Stair Step
Terminal temperature (°C)	1,000	1,000	1000	burst
Fission Gas Released (%)+	10.7	~12	5.1**	TBD

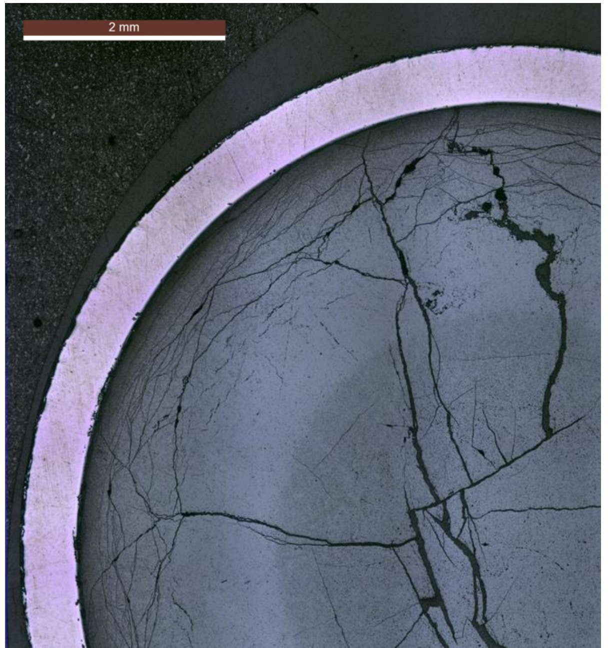
*Tentatively scheduled for October 2024
+Results are approximations and need detailed neutronics to verify
**Preliminary



651F3D : As-Irradiated

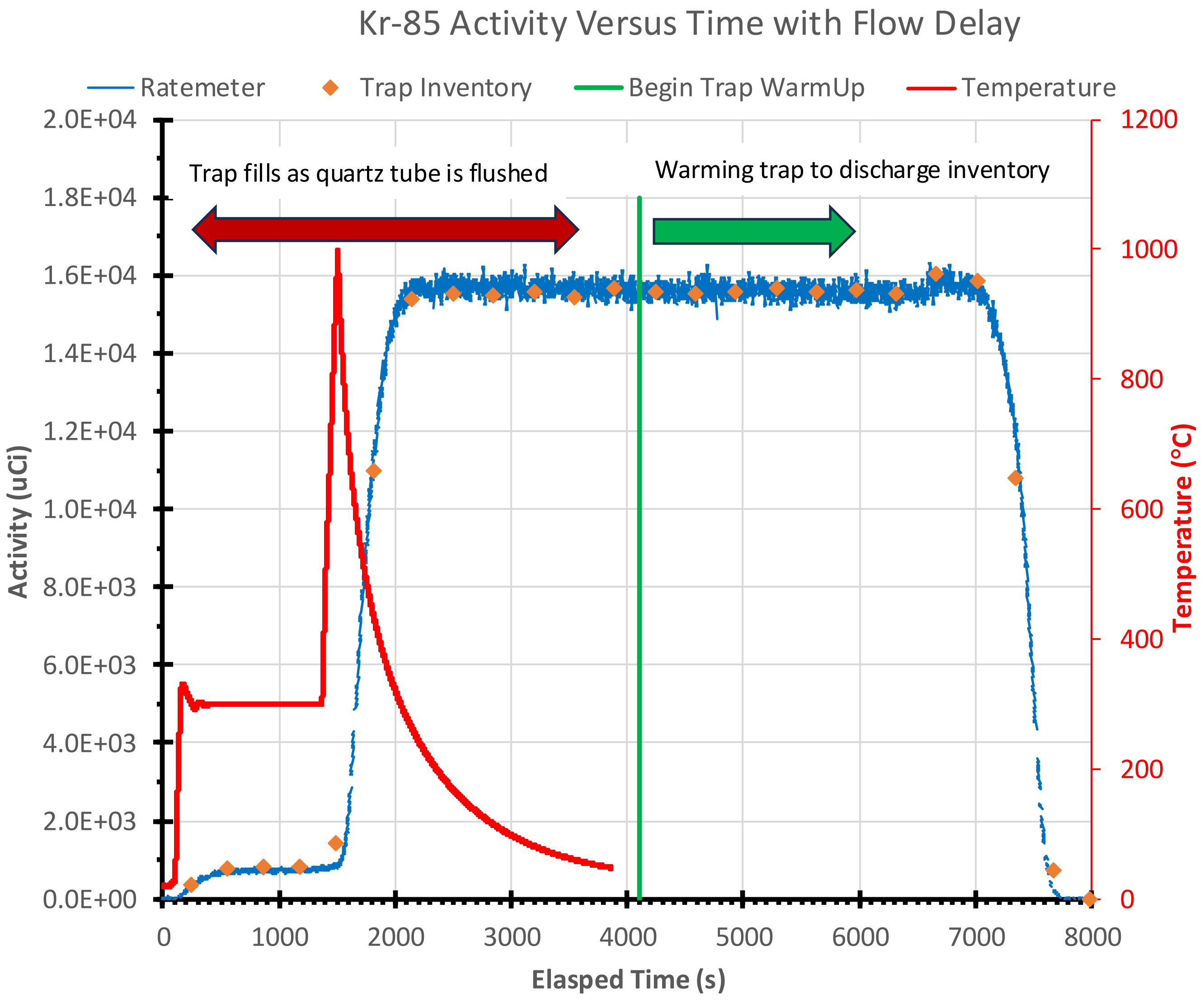


Post-Test 651F3D



Additional tFGR Test (Test 2)

- Standard LOCA Ramp on 33.5 mm segment unpressurized
- 15.6 mCi of Kr-85 Released
- Roughly 12% FGR



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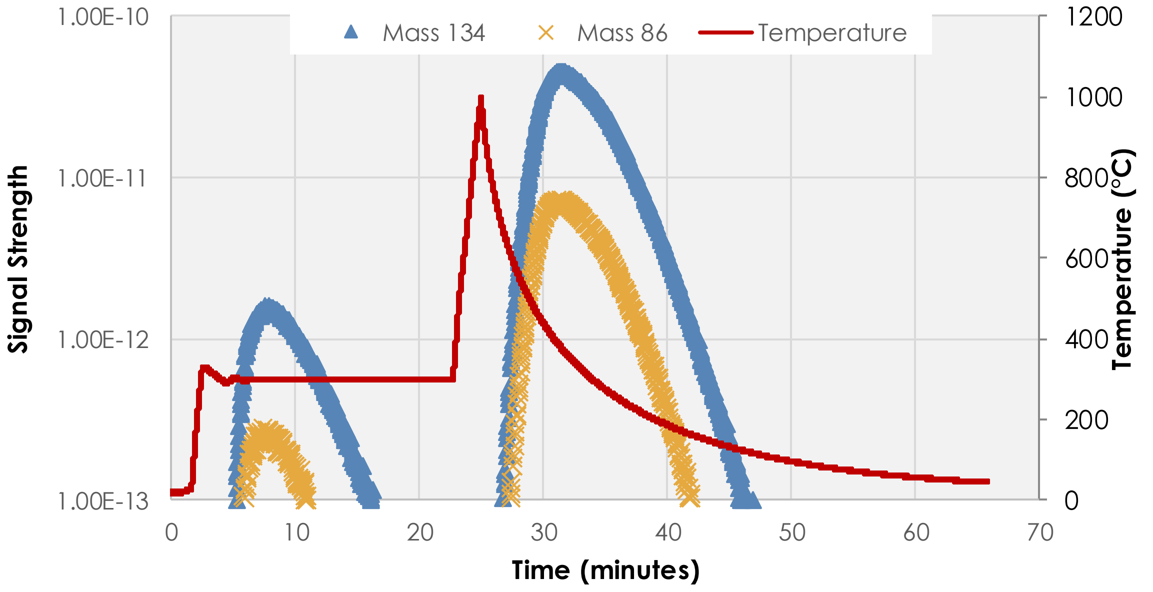
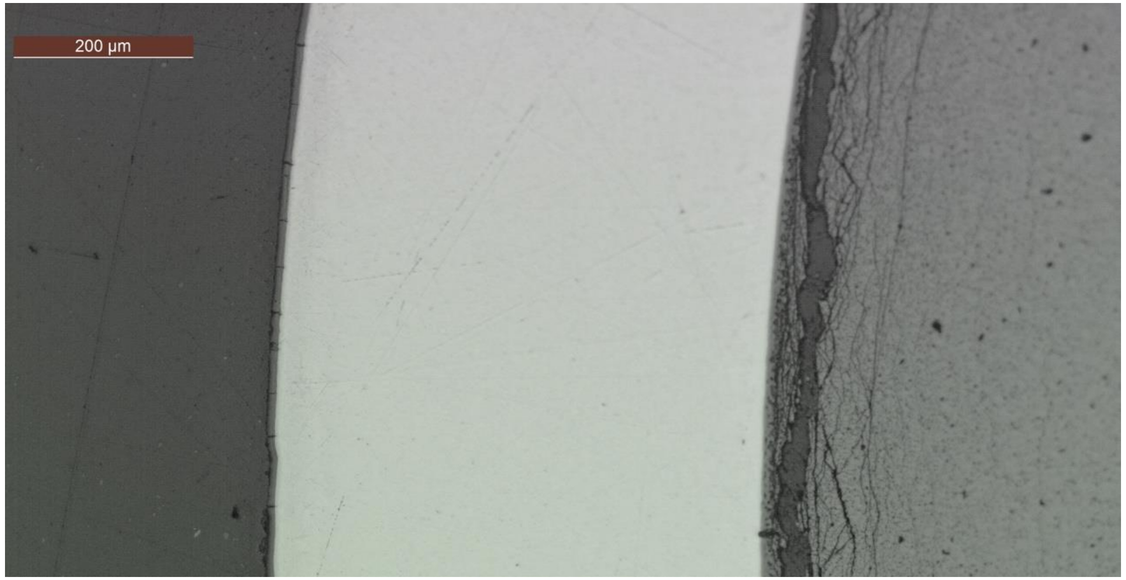
Further Analysis of Release Fission Gas (6XV-A3)

- It is possible to analyze the released Kr and Xe isotopes to determine the fraction
- The ratios collected in this test suggest most fission (~67%) was from Pu-239
  - (^{134}Xe+^{132}Xe+^{131}Xe)/(^{86}Kr+^{84}Kr+^{83}Kr)
  - Whole rod was ~50% Pu-239 Fission (73.8 MWd/kgU) from Neutronics
- Agrees with microscopy from the initial test

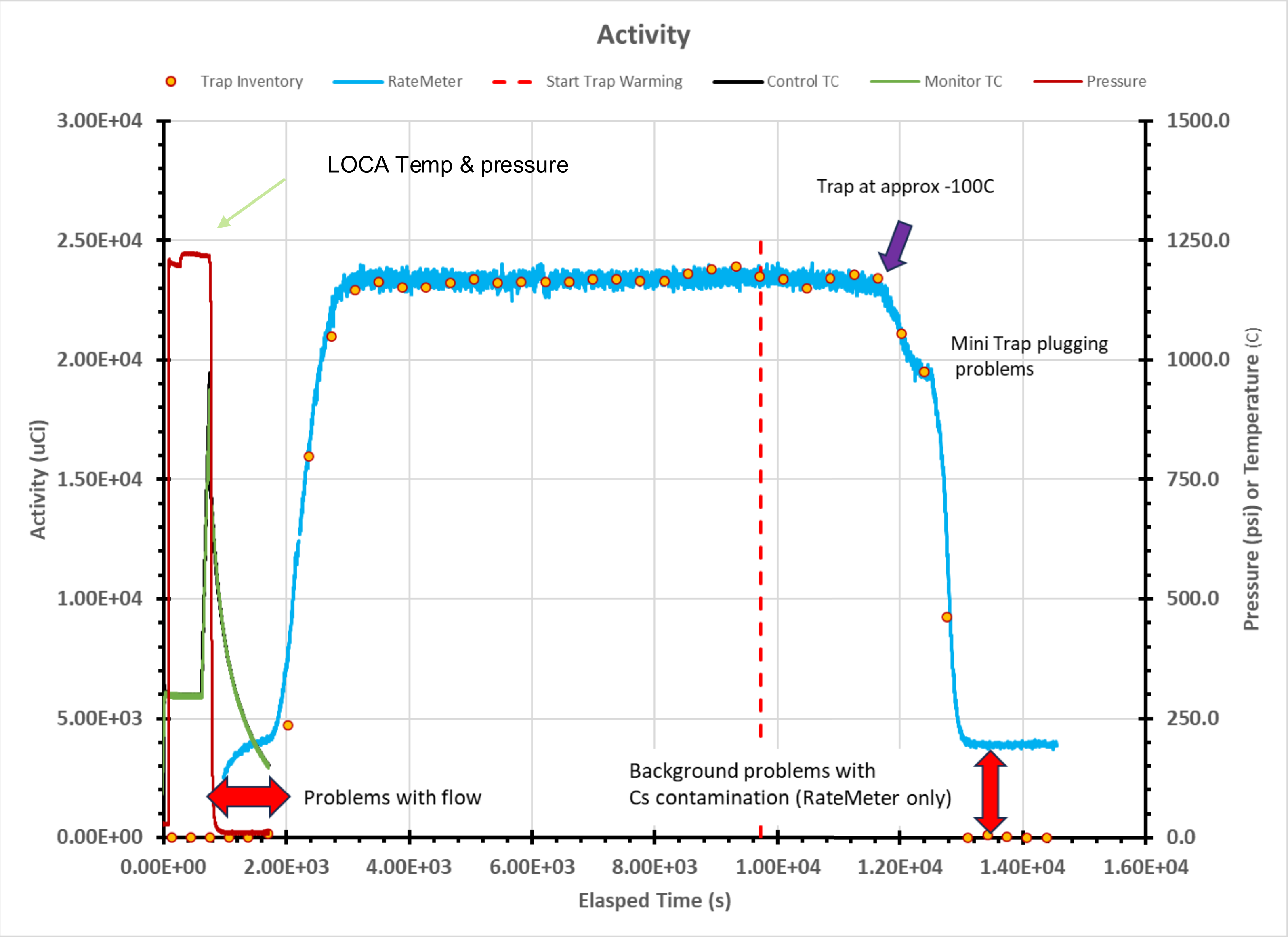




6XV-A3A

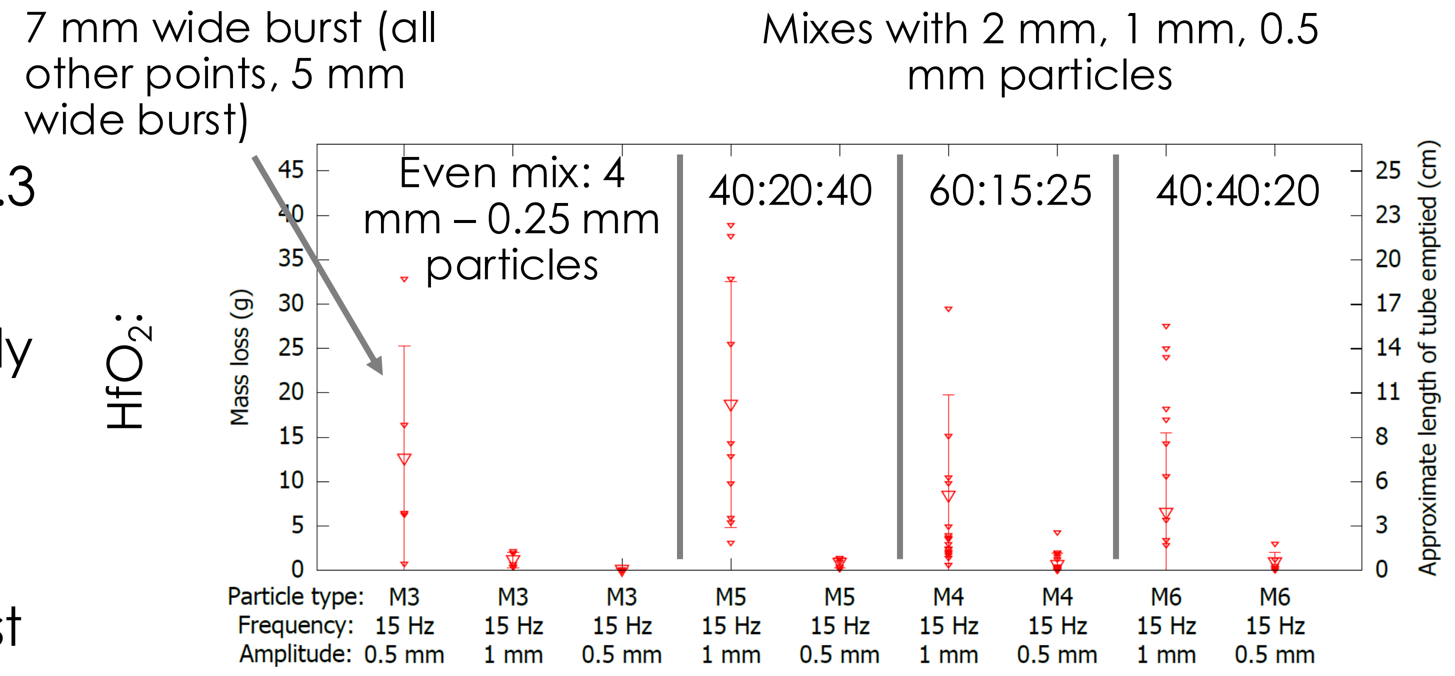
Fission gas released during the LOCA (NA-GS)

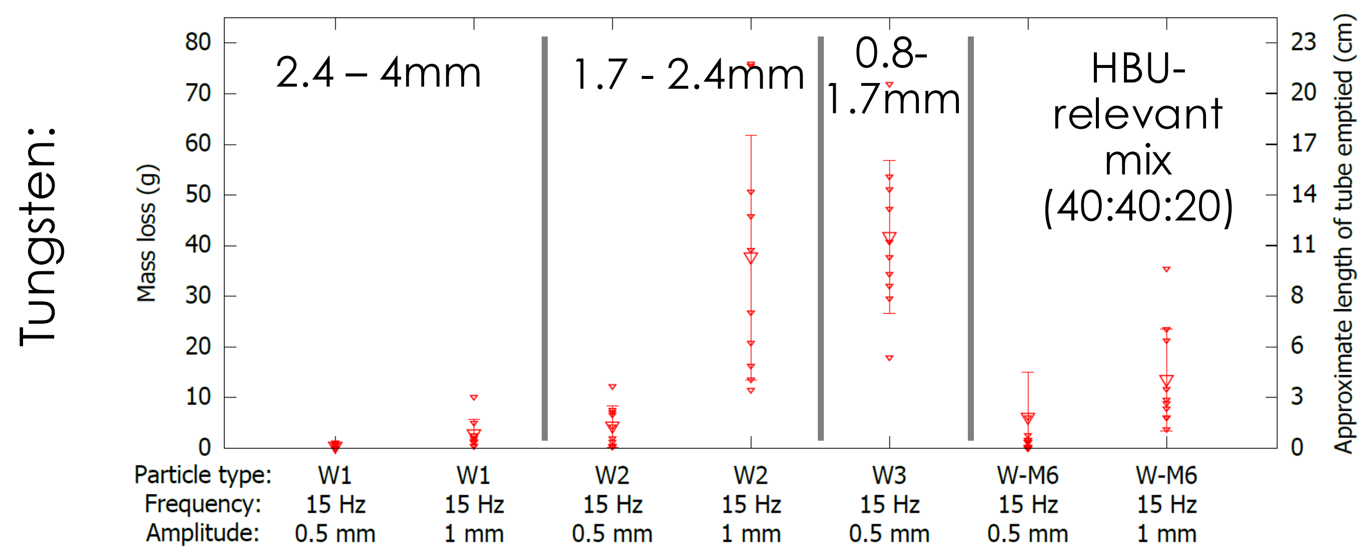
- First High Burnup LOCA test to measure FGR during a LOCA test
- Appeared to have minimal impact on cladding rupture
- Preliminary FGR Results = 5.1%
- Full length heating test suggest similar results (more to come)



Post LOCA Dispersal Behavior: Out-of-cell Test Results

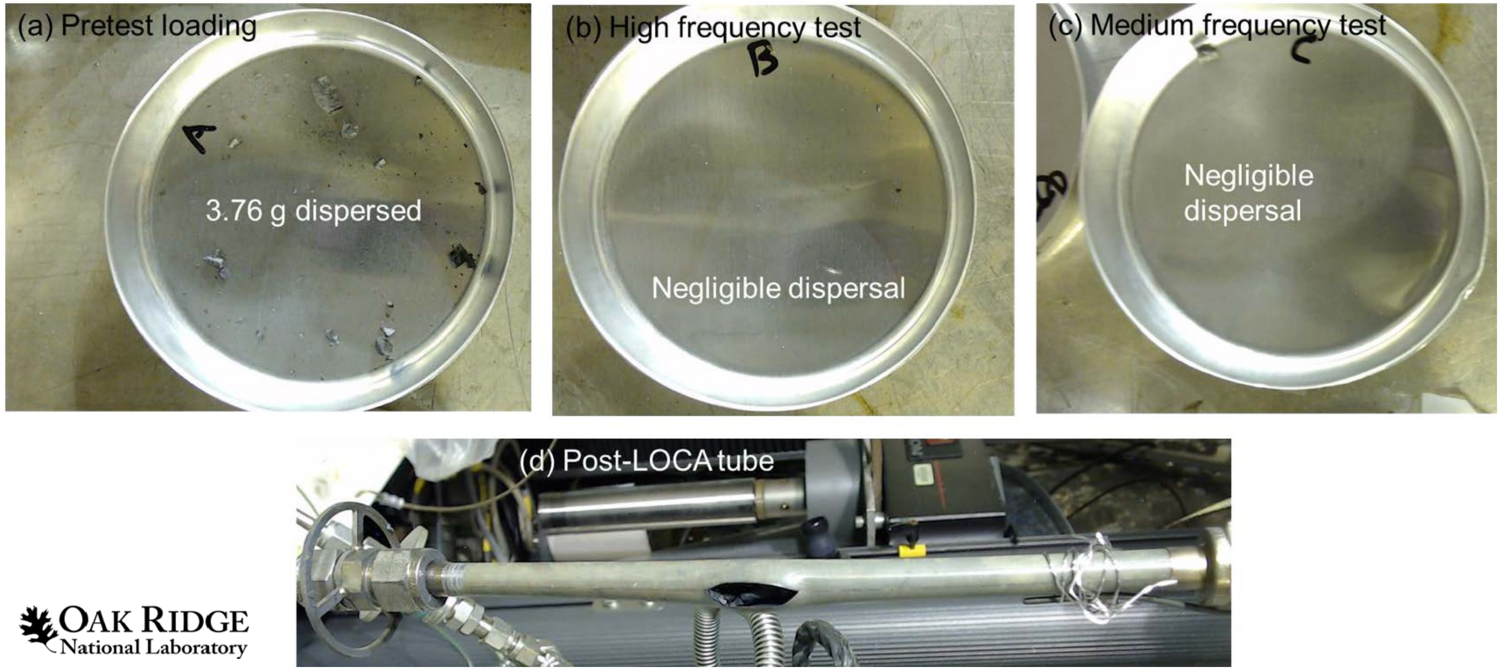
- HfO_2 and W, in various mixtures, has been tested from 2 – 25 Hz and 25 – 0.3 mm amplitude vibrations
- Dispersal limited to conditions with only small particles, very large bursts (>5 mm) or mixtures with large fractions (>25%) of small (<1 mm) particles
- Studies on particle shape effects, burst geometry, and an international collaboration are ongoing for the coming FY
- Out-of-cell testing shows limited dispersal with typical LOCA-induced burst sizes

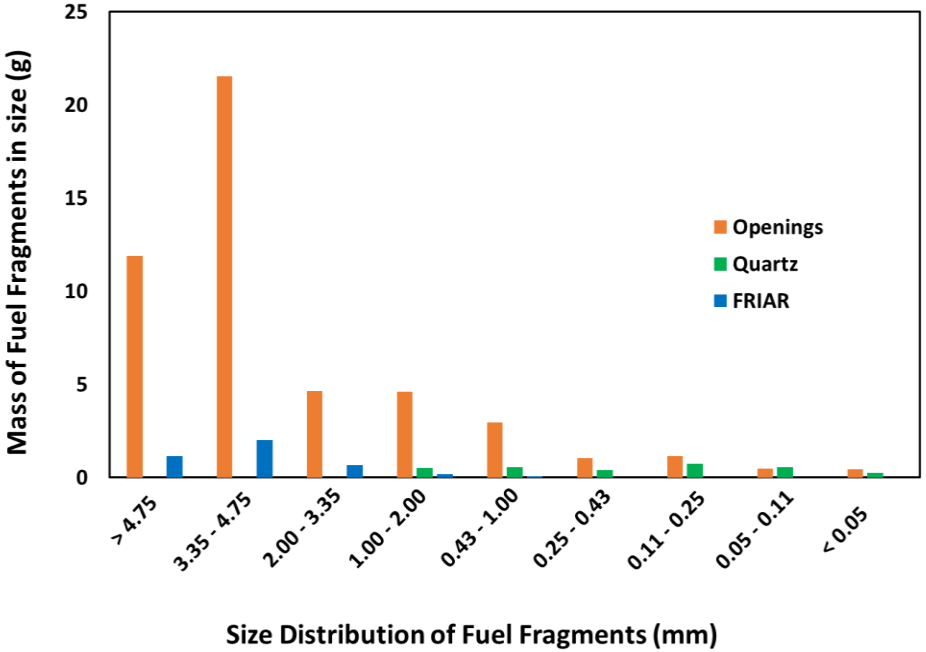




Post LOCA Dispersal Behavior: Out-of-cell vs. In-Cell Test Results

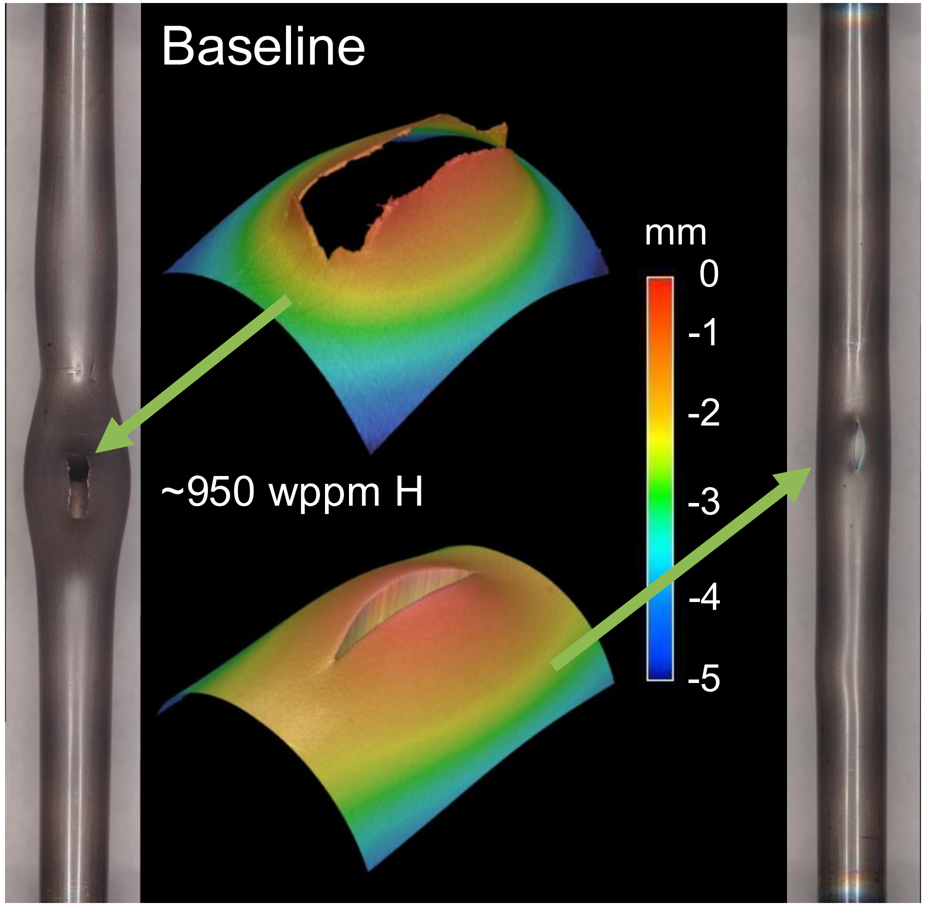
- Post-LOCA sample was tested under representative conditions intended to replicate expected LOCA forces during post burst conditions
- Sample had a relatively large burst opening and all post-burst dispersal occurred during sample loading
- Burst-dispersal was found to be similar to FRIAR dispersal and both represented <20% of dispersible, but retained, fuel fragments
- **In-cell testing showed the surrogate testing was conservative**



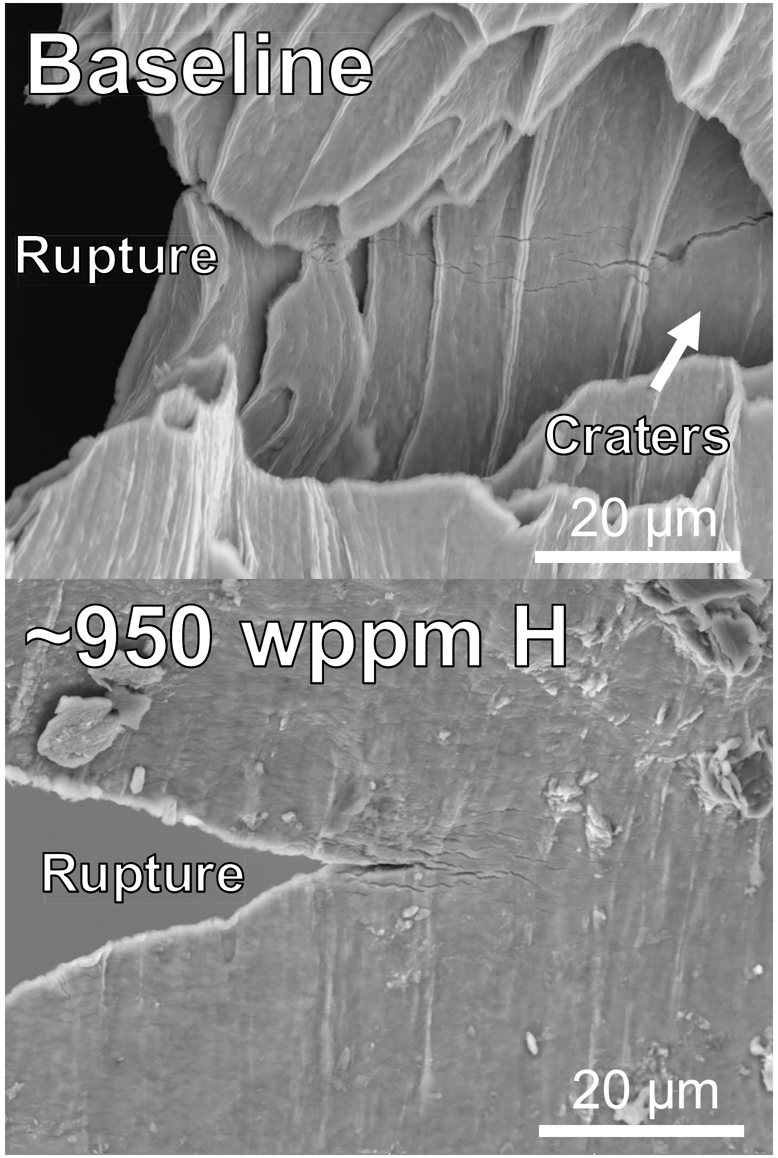


Impact of Hydrogen on LOCA Performance

3D Optical Imaging



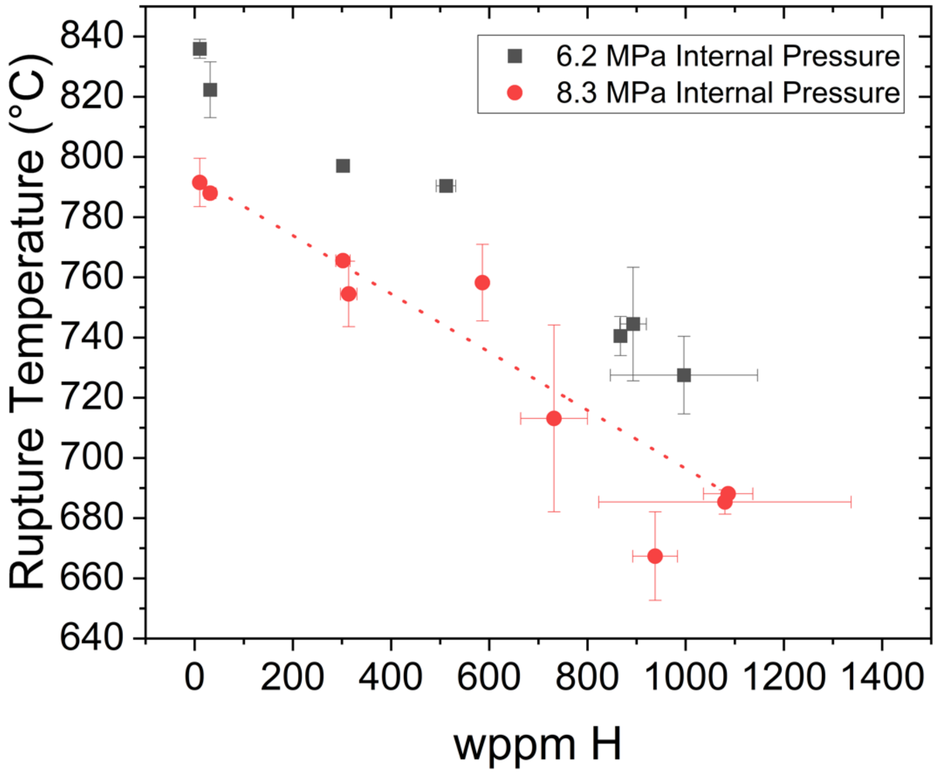
Electron Imaging

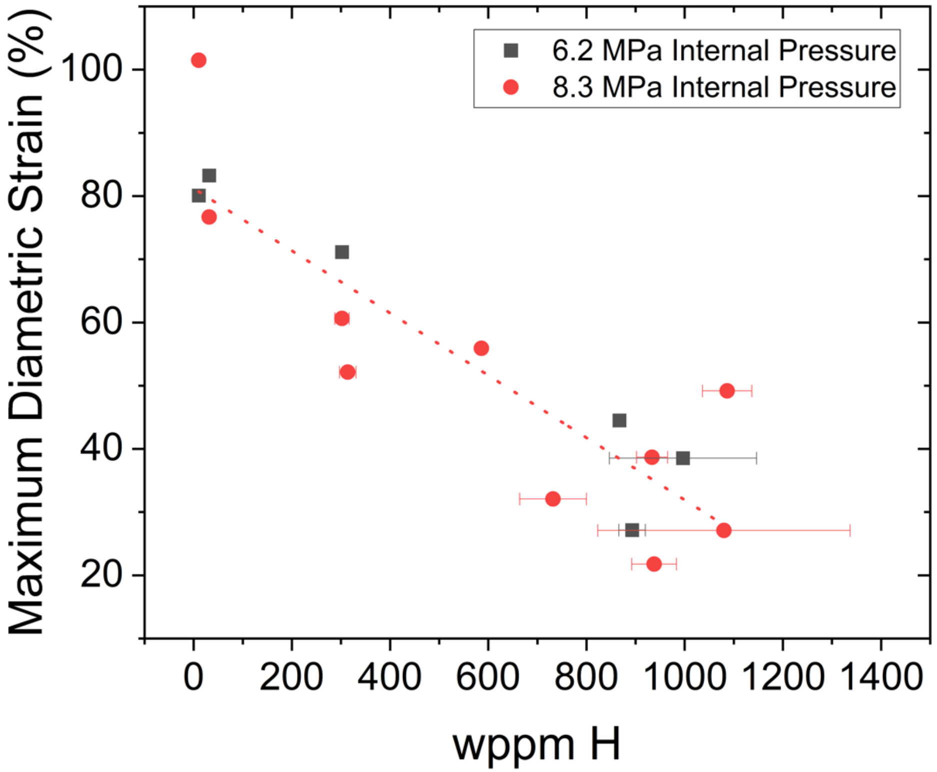


Hydrogen charged claddings show evidence of "less ductile" rupture events

Measurable decreases in rupture temperature and maximum diameter strain values with increasing H content

Material Type - Cameco Zry-4





~10°C drop in rupture temperature for every 100 wppm H added

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Future Work

- LOCA Testing
  - Complete PIE on current rods
  - Planning upwards of 10-20 test next FY across various programs
- Mechanical Testing
  - Evaluate changes in mechanical properties at higher burnups
  - Connect in-reactor to test reactor performance
- Impact of Hydrogen on Performance
  - Expand evaluation to mechanical testing
  - Assess other accident scenarios (i.e., RIA via MBT and expansion due compression
- Begin assessing high burnup ATF performance

Discussion Period

Public Comment Period

Adjourn