"Assessing the Radiological Consequences and Risk of a 50.46 LOCA" Workshop

May 20, 2025

May 21, 2025

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

- Provide an update on the technical basis development for a potential approach to evaluating dose of fuel dispersal and using risk insights to disposition other fuel dispersal consequences.
- Provide feedback on NEI's white paper "Risk Significance of Loss of Coolant Accidents and Justification for a Risk-Informed Transition Break Size".
- Provide an opportunity for members of the public to ask questions of the NRC staff.
- The NRC is <u>not</u> looking for feedback on the Increased Enrichment (IE) Rulemaking.

Proposed Workshop Schedule

- Workshop 1 (Today)
  - Fuel Fragmentation, Relocation, and Dispersal
  - Recriticality
  - Coolability
- Workshop 2
  - Reporting
- Workshop 3
  - Transition Break Size
- Workshop 4
  - Materials
  - Inspection

Agenda – Day 1

Time	Topic	Speaker
9:00 am	Welcome	NRC
9:05 am	Opening Remarks	NRC, NEI
9:10 am	An Approach to Address Fuel Dispersal During a LOCA Using Dose Consequences and Risk Insights	NRC
10:10 am	Break	
10:20 am	Risk Significance of Loss of Coolant Accidents (LOCAs) and Justification for a Risk-Informed Transition Break Size (TBS) Methodology	NEI
11:20 am	Discussion	All
11:50 am	Public Comment Period	Public
12:00 pm	Adjourn	NRC

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

Agenda – Day 2

Time	Topic	Speaker
9:00 am	Welcome	NRC
9:05 am	Discussion	NRC
10:30 am	Break	
10:40 am	Discussion	All
11:45 am	Public Comment Period	Public
11:55 am	Closing Remarks	NRC/NEI
12:00 pm	Adjourn	NRC

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

Opening Remarks

Jen Whitman
Division Director, DSS
NRC

Al Csontos
Director of Fuels
NEI

An Approach to Address Fuel Dispersal During a LOCA Using Dose Consequences and Risk Insights

Workshop: Assessing the Radiological Consequences and Risk of a 50.46 LOCA
May 20-21, 2025

Joseph Messina
Reactor Systems Engineer
Nuclear Methods and Fuel Analysis

UNITED STATES NUCLEAR REGULATORY COMMISSION

Agenda

- Purpose
- Background
- Overview
- Preliminary Calculations

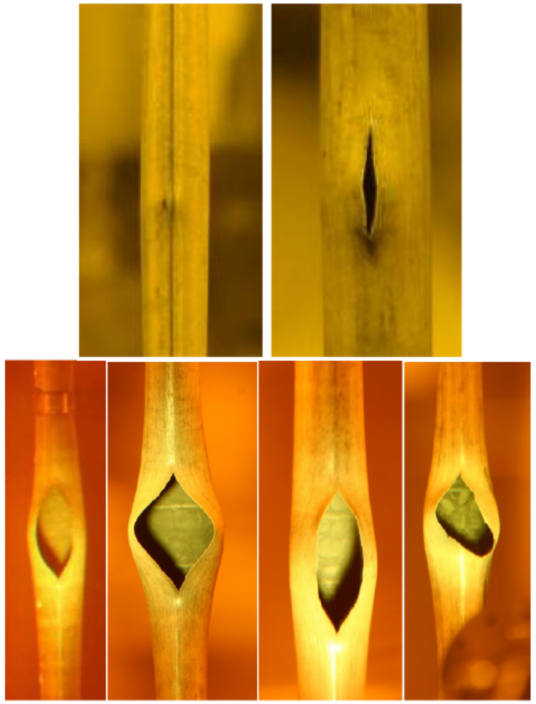
Purpose

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

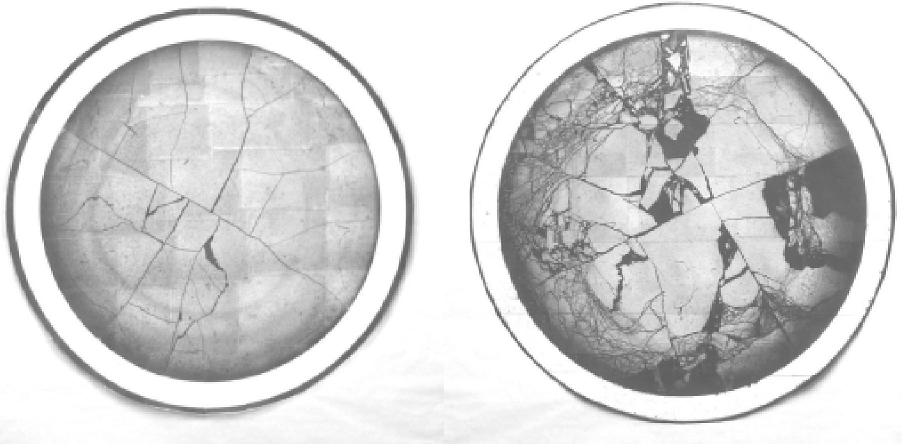
- The purpose of this meeting is to provide an update on the technical basis development for a potential approach to evaluating dose of fuel dispersal and using risk insights to disposition other fuel dispersal consequences.
- The NRC is <u>not</u> looking for feedback on Increased Enrichment (IE) Rulemaking
  - Stakeholders will have the opportunity to provide feedback during the public comment period of the IE Rulemaking.
  - Public meetings / workshops will be conducted on the IE rulemaking proposed rule during the public comment period.

Background – FFRD

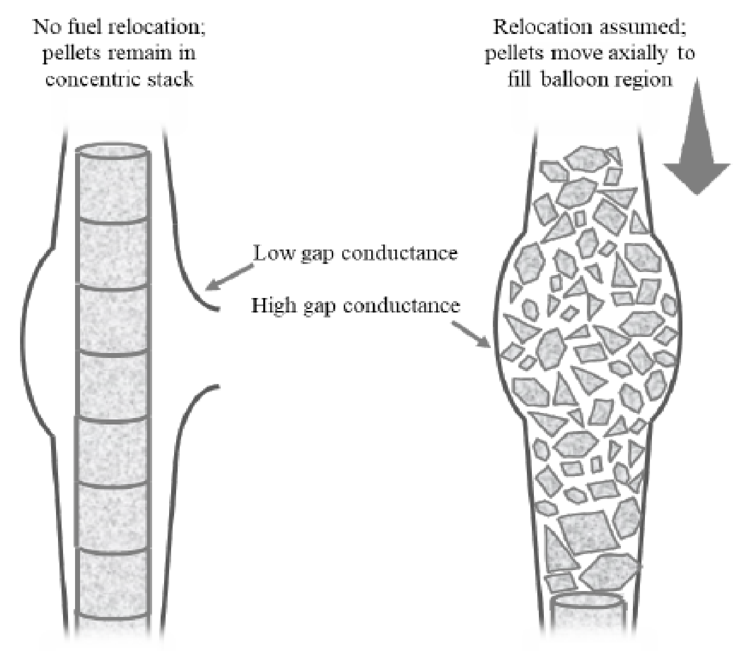
- At high burnup (HBU), experiments have shown that the fuel can fragment during a LOCA
- Differences in pressure across the cladding can lead to cladding ballooning and burst
- The fragmented fuel can relocate axially into the balloon region of the fuel rod and if burst occurs, disperse into the RCS



Burst openings from Studsvik LOCA tests (NUREG-2121)



Segment from NRC's ANL LOCA program at 55 GWd/MTU before and after testing



Background – Fuel Dispersal Consequences

Fuel dispersal has the potential to impact consequences primarily in the following areas:

- Recriticality
- Coolability
- Dose

These consequences may have impacts related to the following highlighted regulatory requirements

10 CFR 50.46:

- 2200°F peak cladding temperature (PCT) and 17% maximum local oxidation (MLO) limits must be met during loss-of-coolant accidents (LOCAs) (to a high probability)
- Core coolability must be maintained during LOCAs (to a high probability)

10 CFR 50.67:

- A dose analysis is required for design basis accidents (DBAs)

General Design Criterion (GDC)-27:

- A subcritical configuration must be maintained under postulated accident conditions

Background

The NRC staff is considering how the risk triplet can be used to address fuel dispersal:

- What can go wrong?
- How likely is it?
- What are the consequences?

The NRC is evaluating an approach that leverages a generic bounding assessment of dose and uses risk insights for post-FFRD consequences.

Approach Overview

The technical basis for this approach is currently being developed and evaluated

NRC Work: coolability calculations: NRC/RES to perform calculations to estimate the coolability of dispersed fuel as well as the effect on the 50.46 PQD requirements (i.e., PCT, MLO).

Coolability calculations: If calcs *clearly demonstrate* coolability and negligible impact on PCT and MLO, then NRC would clearly outline the applicability of the calcs, including:

- Quantity of dispersed fuel
- Geometries and reactor types
- Etc.

Coolability calculations: If calcs *do not* clearly demonstrate both coolability and little effect on PCT and MLO, then more work is needed

NRC work: NRC recriticality calculations anticipated to show that recriticality is not an issue

Industry plant-specific demonstration that NRC/RES recriticality calcs are applicable

Industry plant-specific PCT and MLO calcs, as done today

Industry plant-specific demonstration that NRC/RES coolability calcs are applicable

Viability of approach in question. To be further evaluated / considered. More work needed.

Codified Regulatory Requirements

GDC-27: Maintain subcritical configuration under postulated accident conditions

50.46: PCT and MLO limits (to a high probability)

50.46: Maintain coolability (to a high probability)

50.67: Dose analysis required for DBAs

Guidance

DG-1425 (RG 1.183, Rev. 2): FFRD DBA-LOCA dose guidance and criteria

Fuel Dispersal Consequences

Fuel dispersal has the potential to impact consequences primarily the following areas:

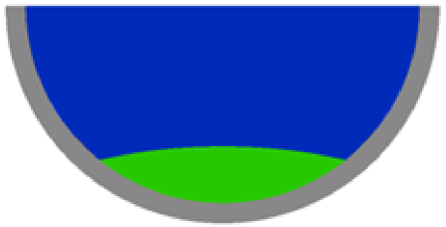
- Recriticality
  - NRC/RES work ongoing
- Coolability
  - NRC/RES work ongoing
  - Future workshop planned
- Dose
  - DG-1425 (RG 1.183, Rev. 2) section on FFRD during a DBA LOCA, as discussed in DG-1425 December ACRS Subcommittee meeting

Technical Basis Development: Recriticality

- The fuel dispersal PIRT panel believed that recriticality of dispersed fuel is not a concern (ML24155A058)
  - Panelists also stated this could be demonstrated using existing tools and engineering judgement
- Staff performed simple analysis to address the potential for recriticality (ML24319A262)
  - Focused on simplified model for the lower plenum of Westinghouse 4-loop plant
  - Assumed all fuel was at 55 GWd/MTU (fine fragmentation threshold in RIL-2021-13) and had initial U-235 enrichment of 8 weight percent
  - Did not credit soluble boron

Technical Basis Development: Recriticality

Pile Depth (cm)	K_{eff}	Fuel Mass (metric tons UO2)
20	0.725009 ± 0.000229	2.2
30	0.773866 ± 0.000261	4.0
40	0.800106 ± 0.000338	5.9
50	0.820710 ± 0.000286	7.9

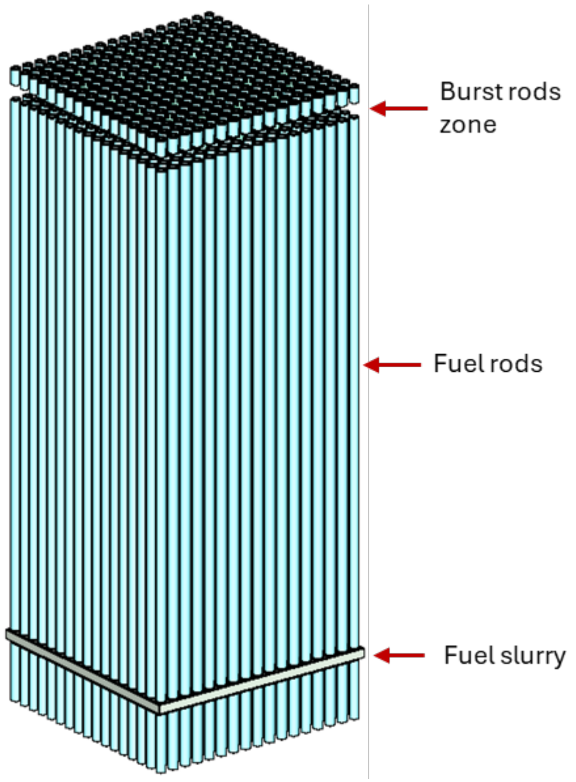


CSAS-Shift model of lower plenum/fuel mixture (grey- steel / blue- non-borated water / green- Fuel/water mixture)

- Staff calculations show mass needed for recriticality far exceeds expected dispersed mass
  - For context: if all fuel in one grid space (~10% of the rod length) from all high burnup rods (~1/2 the rods in the core) dispersed, this would result in < 5 metric tons of UO_2 (for Westinghouse 4-loop plant)

Technical Basis Development: Recriticality

- NRC/RES has recently conducted preliminary criticality calculations on fuel fragments on spacer grids
- Draft calculations show that fuel dispersal and redistribution within the reactor core region (accumulation of fuel fragments on assembly spacer grids) results in at most a negligible net positive reactivity effect
  - Largest Δk was ~100 pcm
  - In context, this is similar in magnitude to the uncertainties in some core simulator methods



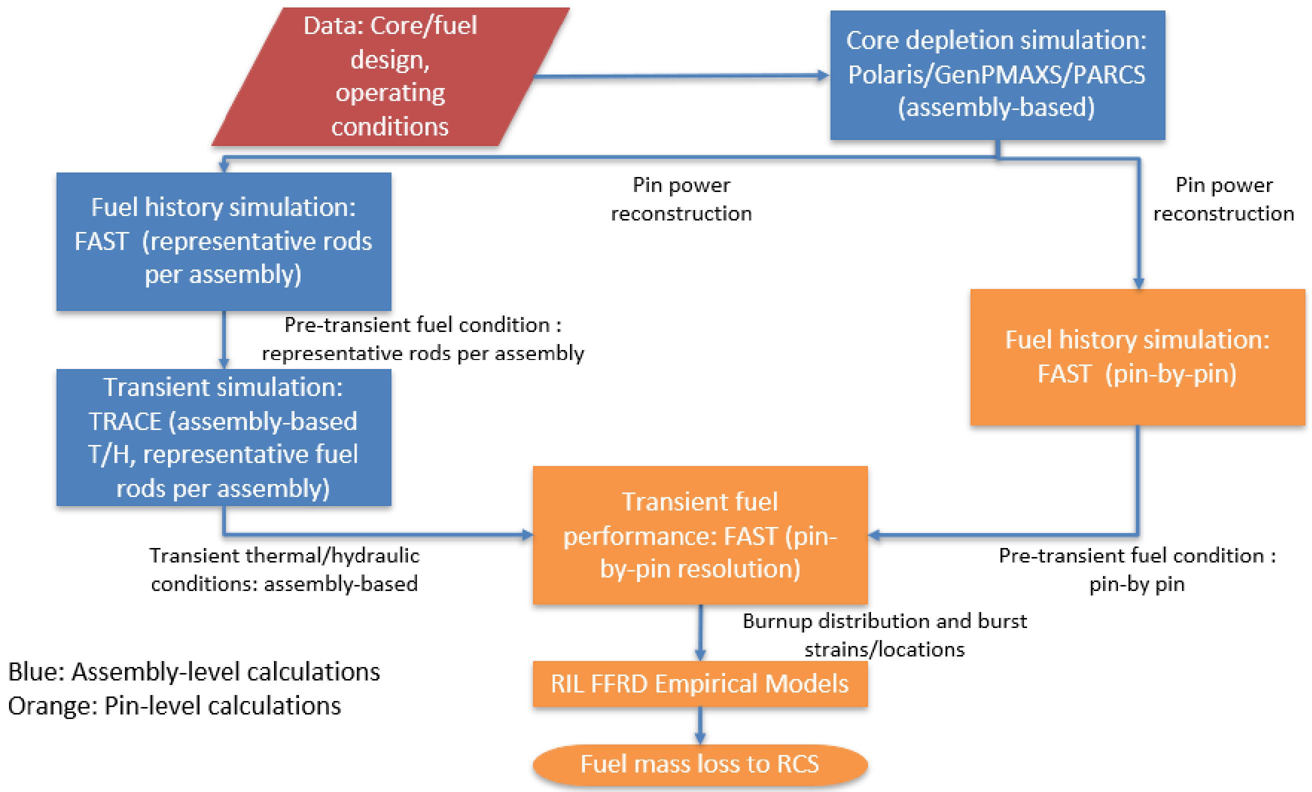
Technical Basis Development: Recriticality

- Staff analysis shows that recriticality is very unlikely
  - This is consistent with analysis performed by the OECD/NEA Working Group on Fuel Safety
  - So far, staff have only performed quantitative analysis for a few configurations, but based on engineering arguments recriticality is unlikely for other configurations
    - *NRC/RES is performing additional recriticality calculations to provide a stronger basis to resolve recriticality concern*
- Licensees should demonstrate that the potential recriticality is addressed for their plant configuration
  - Licensees can use qualitative engineering arguments if dispersed mass is significantly less than the amounts in the staff calculations

Technical Basis Development: Coolability

NRC/RES is performing work with existing tools to study dispersed fuel coolability effects

- Currently focused on realistic modeling
- Workflow for Holistic Accident Multiphysics (WHAM) Calculational methodology described in [1] to develop a range or upper bound realistic estimate of the mass of fuel dispersed.

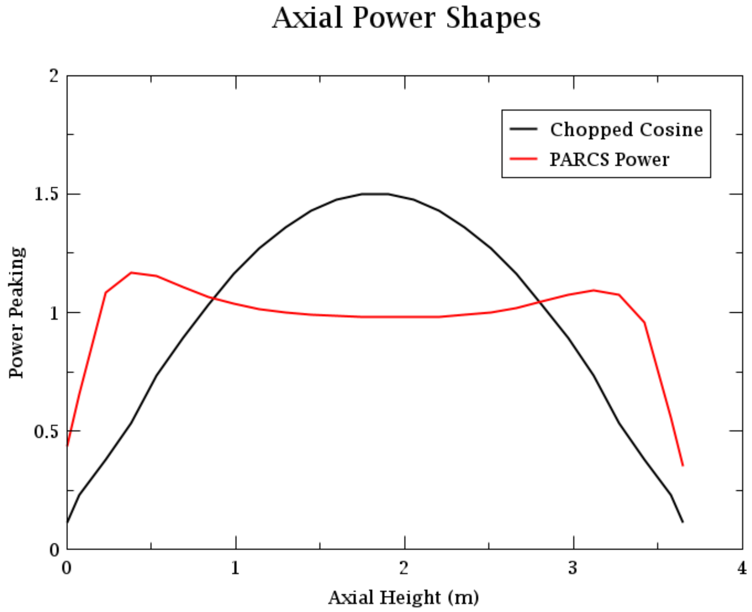


NRC Workflow for Holistic Accident Multi-Physics (from Ref. [1])

[1] A. Bielen, J. Corson, and J. Staudenmeier, "NRC's methodology to estimate fuel dispersal during a large break loss of coolant accident," Nuclear Engineering and Design, vol. 426, p. 113377, 2024.

Dispersed Mass Additional Assumptions

- Westinghouse 4-Loop plant with HBU and IE core design, as done in [2] (see figure on right).
- With and without offsite power
- Chopped cosine power shape and a power shape determined from a PARCS core depletion calculation that had power peaks near the bottom and top of core
- With 1 train of ECCS and with 2 trains of ECCS



<table>
<tr><th></th><th>H</th><th>G</th><th>F</th><th>E</th><th>D</th><th>C</th><th>B</th><th>A</th></tr>
<tr><td>8</td><td>5.95 | 200
FEED
0.0 / 37.2</td><td>5.95 | 200
H-6
37.2 / 67.6</td><td>5.95 | 200
FEED
0.0 / 37.2</td><td>5.95 | 200
N-4
35.1 / 64.6</td><td>5.95 | 200
F-4
35.1 / 64.0</td><td>5.95 | 80
B-8
35.1 / 65.4</td><td>5.95 | 200
FEED
0.0 / 33.5</td><td>5.95 | 200
K-6
36.8 / 52.4</td></tr>
<tr><td>9</td><td>5.95 | 200
F-8
37.2 / 67.6</td><td>5.95 | 200 | 24
FEED
0.0 / 34.9</td><td>5.95 | 200
J-9
34.1 / 65.4</td><td>5.95 | 200 | 24
FEED
0.0 / 34.0</td><td>6.20 | 200
F-2
32.5 / 63.5</td><td>5.95 | 200
FEED
0.0 / 37.9</td><td>5.95 | 200 | 20
FEED
0.0 / 32.0</td><td>5.95 | 200
N-9
37.9 / 52.7</td></tr>
<tr><td>10</td><td>5.95 | 200
FEED
0.0 / 37.2</td><td>5.95 | 200
M-10
35.1 / 66.1</td><td>5.95 | 200
FEED
0.0 / 36.8</td><td>5.95 | 200
E-7
34.2 / 64.2</td><td>5.95 | 200 | 24
FEED
0.0 / 34.6</td><td>6.20 | 200
B-5
29.6 / 62.9</td><td>5.95 | 104
FEED
0.0 / 33.5</td><td>5.95 | 200
P-12
55.6 / 67.4</td></tr>
<tr><td>11</td><td>5.95 | 200
D-3
35.1 / 64.6</td><td>5.95 | 200 | 24
FEED
0.0 / 33.9</td><td>5.95 | 200
G-5
34.3 / 64.2</td><td>5.95 | 200
D-13
35.1 / 63.3</td><td>5.95 | 200
G-2
32.0 / 62.6</td><td>5.95 | 200
FEED
0.0 / 37.6</td><td>6.20 | 200 | 8
FEED
0.0 / 29.0</td><td>6.20 | 200
J-4
61.1 / 69.9</td></tr>
<tr><td>12</td><td>5.95 | 200
D-10
35.1 / 64.0</td><td>6.20 | 200
P-10
32.6 / 63.5</td><td>5.95 | 200 | 24
FEED
0.0 / 34.6</td><td>5.95 | 200
P-9
32.1 / 62.7</td><td>6.60 | 128
C-13
29.5 / 62.3</td><td>5.95 | 200
FEED
0.0 / 34.8</td><td>5.95 | 200
N-5
37.4 / 55.3</td><td></td></tr>
<tr><td>13</td><td>5.95 | 80
H-14
35.1 / 65.4</td><td>5.95 | 200
FEED
0.0 / 37.9</td><td>6.20 | 200
L-14
29.6 / 62.9</td><td>5.95 | 200
FEED
0.0 / 37.6</td><td>5.95 | 200
FEED
0.0 / 34.8</td><td>6.60 | 156
FEED
0.0 / 28.9</td><td>5.95 | 200
R-9
51.9 / 61.8</td><td></td></tr>
<tr><td>14</td><td>5.95 | 200
FEED
0.0 / 33.5</td><td>5.95 | 200 | 20
FEED
0.0 / 32.0</td><td>5.95 | 104
FEED
0.0 / 33.6</td><td>6.20 | 200 | 8
FEED
0.0 / 29.1</td><td>5.95 | 200
L-3
37.4 / 55.3</td><td>5.95 | 200
G-1
52.8 / 62.7</td><td></td><td></td></tr>
<tr><td>15</td><td>5.95 | 200
F-6
36.8 / 52.4</td><td>5.95 | 200
J-13
37.9 / 52.8</td><td>5.95 | 200
D-2
55.6 / 67.5</td><td>6.60 | 156
M-4
62.0 / 71.0</td><td></td><td></td><td></td><td></td></tr>
</table>

Enr | IFBA | WABA
PrevCycLoc
BOC/EOC BU

Above: Quarter-core representation of the fuel loading pattern, with feed assemblies in red, once-burned assemblies in green, and twice-burned assemblies in blue; enrichments in the top-left corner of each assembly; number of IFBA and WABA rods in the top middle and top right; and beginning- and end-of-cycle assembly-average burnup at the bottom (from Ref. [2])

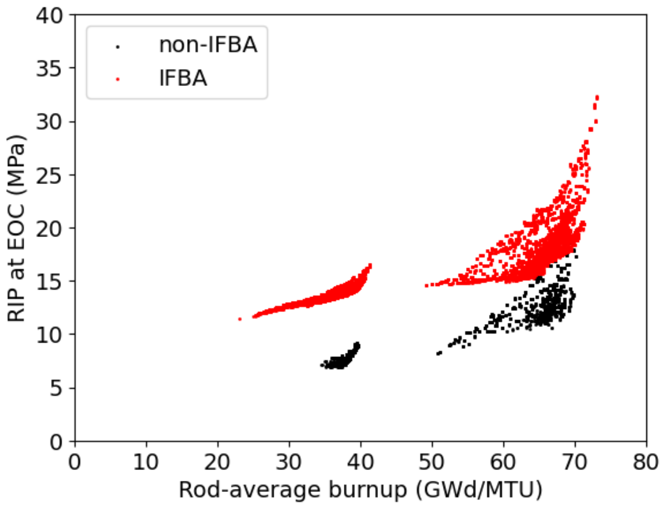
[2] N. Capps, et. Al., "Full Core LOCA Safety Analysis for a PWR Containing High Burnup Fuel (ORNL/TM-2020/1700)," Oak Ridge National Laboratory, Oak Ridge, TN, 2020

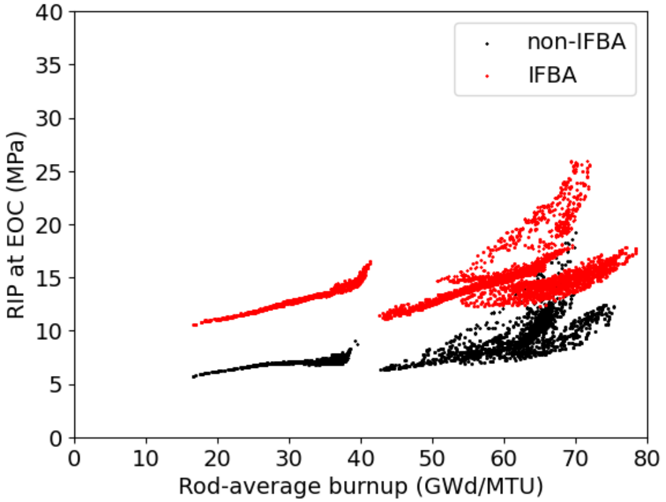
Coolability: Preliminary Results for Mass of Fuel Dispersed

	PARCS Power, Offsite Power Available, 2 ECCS Trains	PARCS Power, Offsite Power Available	PARCS Power, Offsite Power Unavailable	Chopped Cosine, Offsite Power Available	Chopped Cosine, Offsite Power Unavailable
FAST Peak Cladding Temperature (°C)	774	816	834	852	863
Burst Rods (% of total)	25	49	55	44	49
Second Cycle Burst Rods (% of total)	24	50	58	38	41
Dispersed Mass (RIL Model C) (kg UO_2)	1300	2000	2100	3400	3700
Dispersed Mass (RIL Model A) (kg UO_2)	700	940	980	2300	2500
Dispersed Mass (RIL Model A, single grid span) (kg UO_2)	380	530	540	1400	1500

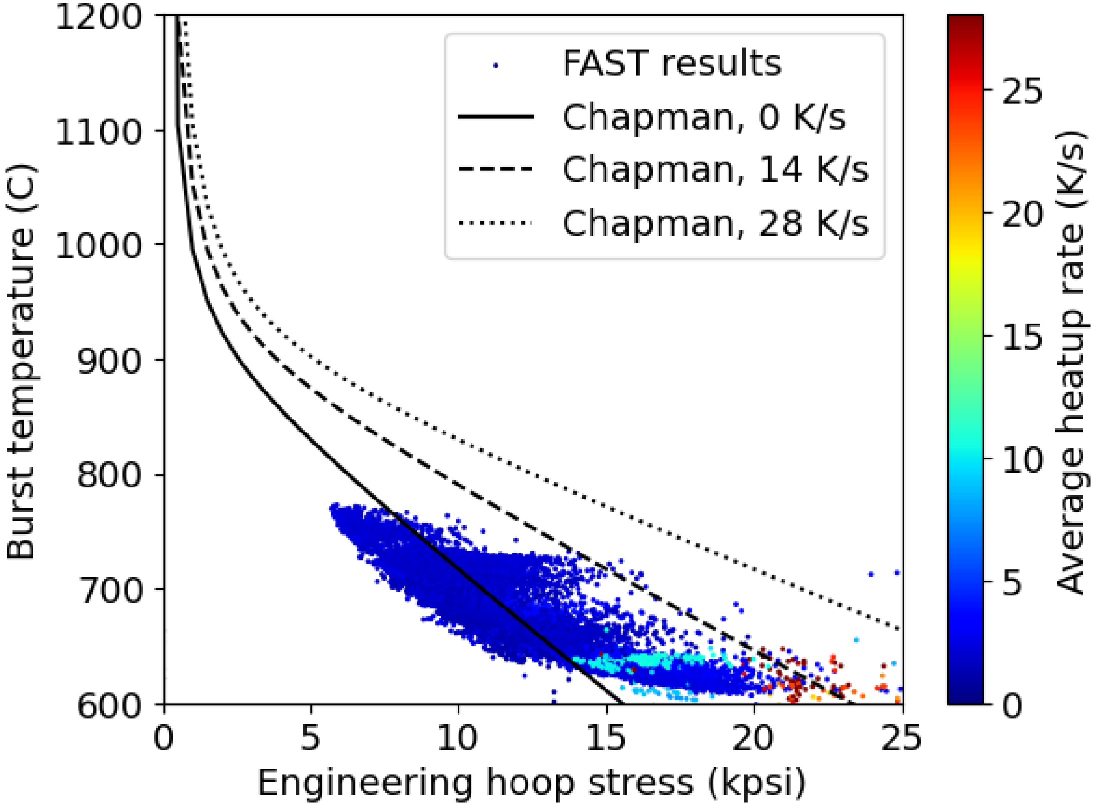
Note: these results are preliminary and represent a wide range of scenarios.

Coolability: Preliminary Results for Mass of Fuel Dispersed





Above: End-of-cycle rod internal pressures calculated by FAST for rods predicted to burst (top) and to remain intact (bottom) in the PARCS power, offsite power available case



Above: Cladding burst temperatures predicted by FAST for the PARCS power, offsite power available case, compared to the Chapman correlation

Coolability: Future Work

- The NRC plans to perform relatively simply TRACE calculations to study the projected impact that dispersed fuel on spacer grids would have on PCT and MLO.
- The calculational process that is anticipated to be established with this work would then be applied to more conservative modelling to be used in technical basis development for the approach outlined in these slides.

Conclusions

- The NRC is in the process of conducting work to develop the technical basis for an approach to use dose consequences and risk insights to disposition the consequences of FFRD.
- The focus of future work remains on coolability
  - Once a complete calculation methodology is developed to study the impact of dispersal on coolability realistically, it could be applied to conservative fuel dispersal mass predictions.

Acronyms/Abbreviations

- ANL: Argonne National Laboratory
- cm: centimeter
- DBA: design basis accident
- DG: draft regulatory guide
- ECCS: emergency core cooling system
- EQ: environmental qualification
- FFRD: fuel fragmentation, relocation, and dispersal
- GDC: general design criterion
- GWd/MTU: gigawatt-days per metric ton of uranium
- HBU: high burnup
- IE: increased enrichment
- LOCA: loss-of-coolant accident
- MLO: maximum local oxidation
- NRC: U.S. Nuclear Regulatory Commission
- OECD/NEA: Organisation for Economic Co-operation and Development / Nuclear Energy Agency
- pcm: percent mille
- PCT: peak cladding temperature
- PIRT: phenomena identification and ranking table
- PQD: post quench ductility
- RCS: reactor coolant system
- RES: NRCs Office of Nuclear Regulatory Research
- RG: regulatory guide
- RIL: research information letter
- U-235: uranium-235
- UO_2 : uranium dioxide
- WHAM: workflow for holistic accident multiphysics

Discussion

Break

Risk Significance of Loss of Coolant Accidents and Justification for a Risk-Informed Transition Break Size

Aladar Csontos, Director Fuels (NEI)
Tom Kindred, Consulting Engineer (SNC)

May 20, 2025

NUCLEAR ENERGY INSTITUTE

Background/Objectives of the Workshops

- Industry appreciates the opportunity to hold open and transparent dialogue with staff at these public workshops
- Applications for uprates and/or advanced fuels incoming
- Pragmatic implementing pathways key to near-term success while longer-term future pathways lead to new opportunities:
    - Risk mitigation for near-term objectives through improved regulatory flexibility, stability, and predictability
- Workshops align with the intent of the ADVANCE Act for the development of modern, risk-informed, and efficient processes to ensure safety while minimizing burden

Overview

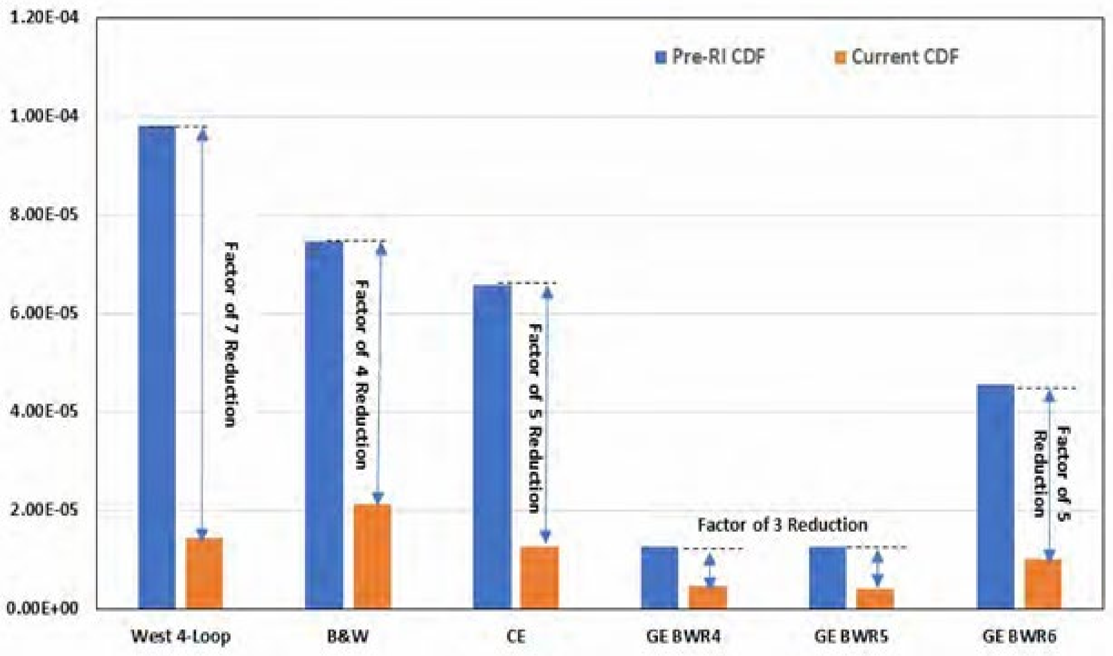
- Chronology of Risk-Informed Programs
- Plant Specific Risk from LOCA Events
- Summary of LOCA Risk Significance
- Consequences of LOCA Events
- Consequences of LOCA Events with FFRD
- Industry Prospective Pathways to Risk-Inform FFRD

Chronology of Risk Modeling and Programs

- The 1990s demonstrated a marked change in emphasis for PRA implementation into regulatory frameworks (NRC PRA Implementation Plan)
  - Initial Plant Examinations (IPEs, NUREG-1150)
  - RG-1.200 (Rev. 0, 2004) – Provided quality guidance for developing PRA models for Risk-Informed (RI) activities
  - RG-1.174 (Rev. 0, 1998) – Approach/Method for Using PRA for RI changes to a plant’s licensing basis

Chronology of Risk Modeling and Programs

- NUREG/CR-6932 Baseline Risk Index for Initiating Events
    - Extended to 2023 from INL/RPT-24-0473
    - Indicates industry trends in risk performance are continuing to improve with time
    - VSLOCAs have core damage frequencies (CDFs) on the order of 1E-10 (BWRs) and 1E-09 (PWRs) yr^{-1}
- Summary of industry risk performance trends captured in NEI-20-4
    - Consistent and re-enforces INL/RPT-24-0473
- High % of plants have approved RI programs (TSTF-425, 505, etc.), for which NRC SPAR models can be used to confirm the information in the above-mentioned publications.



NEI-20-04 Fleet Core Damage Frequency Reduction from Implementation of RI Programs

Plant/Fleet Specific Risk from LOCA Events

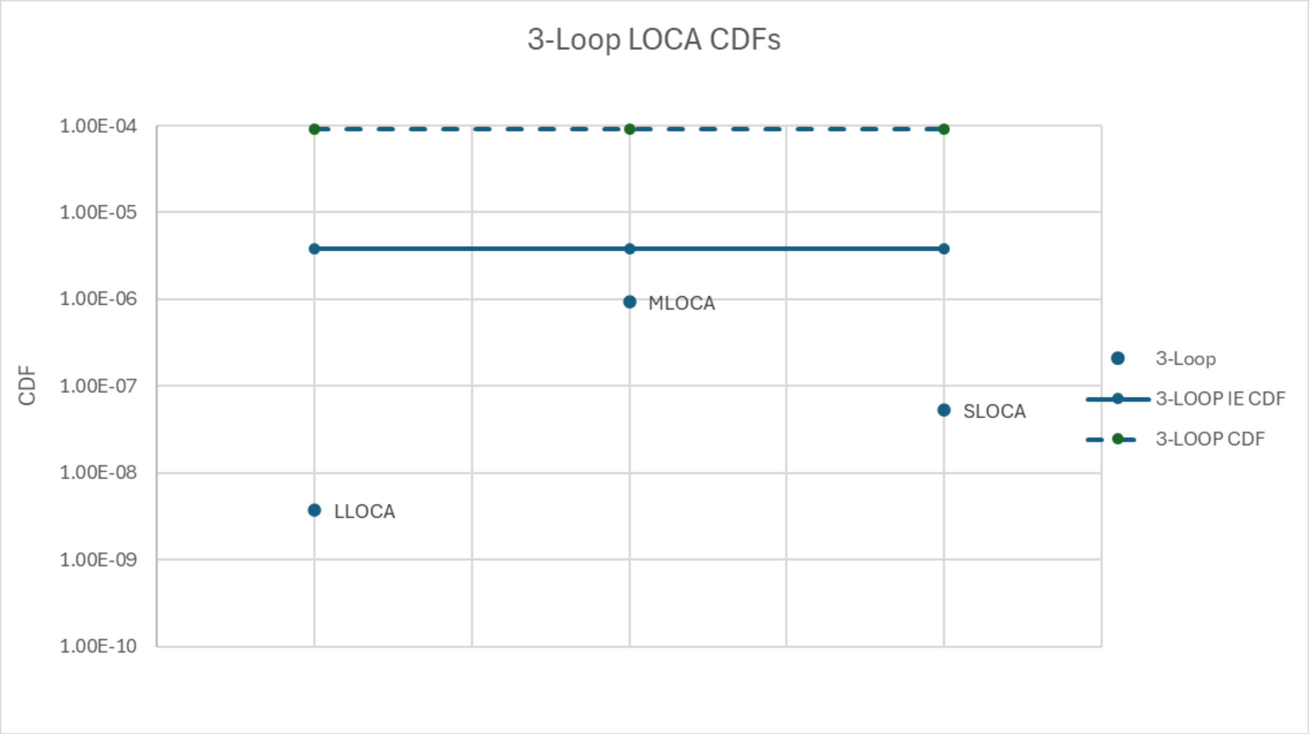
- Summary of publicly available information concludes LOCAs have a very low risk-significance
- Evaluation of a representative utility fleet and the entirety of the BWR fleet also confirms this

		Unit 1		Unit 2	
		CDF[1]	LERF[1]	CDF[1]	LERF[1]
3-Loop PWR	**Total CDF/LERF**	**3.77E-06**	**2.58E-08**	**4.13E-06**	**2.74E-08**
	LLOCA (>6")	3.69E-09	4.13E-12	4.05E-09	4.09E-12
	MLOCA (2"-6")	9.37E-07	1.23E-09	9.37E-07	1.32E-09
	SLOCA (3/8"-2")	5.32E-08	6.22E-11	5.20E-08	4.71E-11
BWR-4	**Total CDF/LERF**	**3.55E-06**	**2.26E-07**	**3.07E-06**	**2.07E-07**
	LLOCA (>6")	9.23E-09	1.41E-11	1.42E-09	1.42E-11
	MLOCA (1"-6")	3.31E-08	1.23E-10	3.30E-08	4.83E-11
	SLOCA (0.5"-1")	2.31E-10	1.01E-10	1.35E-10	2.90E-11
4-Loop PWR		**1.16E-06**	**3.48E-09**	One logic model represents both units	
	LLOCA (>6")	8.70E-09	1.25E-11		
	MLOCA (2"-6")	1.33E-07	1.97E-10		
	SLOCA (3/8"-2")	5.50E-08	7.86E-11		

[1] Internal Events Only

Plant/Fleet Specific Risk from LOCA Events

LLOCA	> 6”
MLOCA	2” – 6”
SLOCA	3/8” – 2”

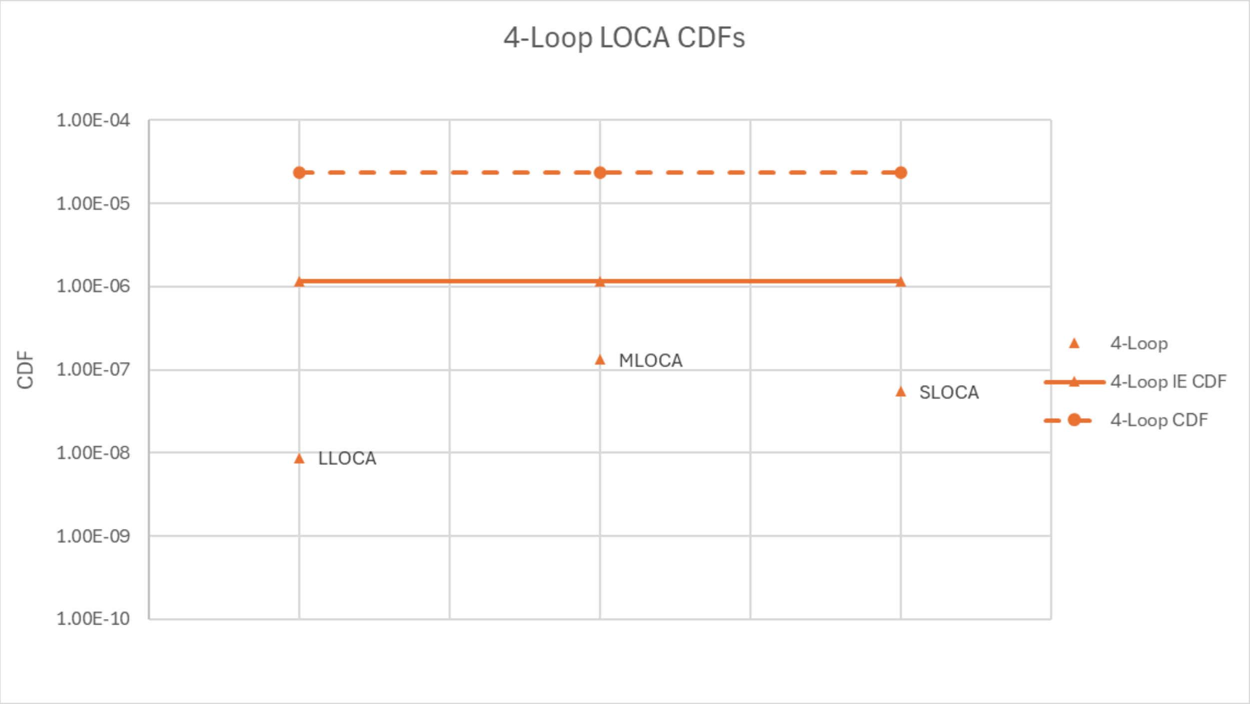


3-Loop PWR LOCA CDF

1. Inflection increase for MLOCAs
2. SLOCA Risk > 1 order of magnitude less than internal events CDF (>3 orders less when taken against overall plant CDF)
3. LLOCA Risk ~3 orders of magnitude less than internal events CDF (>4 orders less when taken against overall plant CDF)

Plant/Fleet Specific Risk from LOCA Events

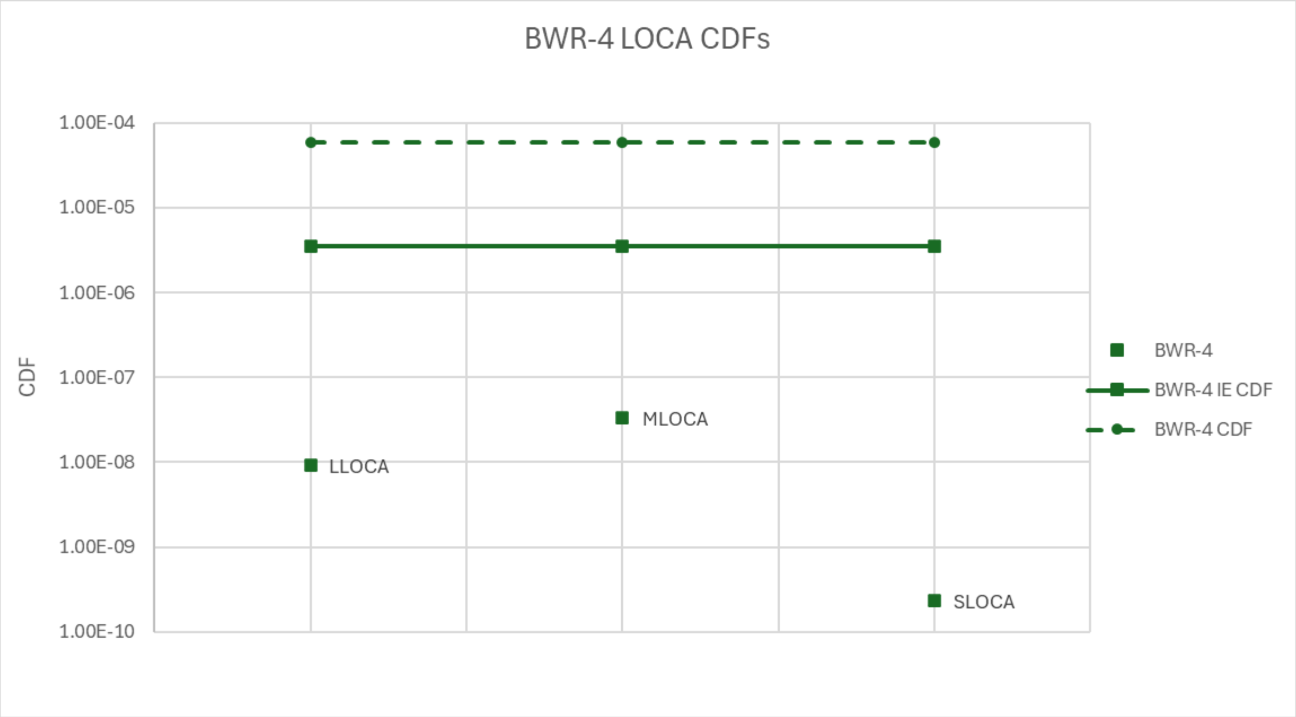
LLOCA	> 6”
MLOCA	2” – 6”
SLOCA	3/8” – 2”



4-Loop PWR LOCA CDF
1. Inflection increase for MLOCAs
2. SLOCA Risk > 1 order of magnitude less than internal events CDF (>2 orders less when taken against overall plant CDF)
3. LLOCA Risk ~2 orders of magnitude less than internal events CDF (>3 orders less when taken against overall plant CDF)

Plant/Fleet Specific Risk from LOCA Events

LLOCA	> 6”
MLOCA	2” – 6”
SLOCA	3/8” – 2”



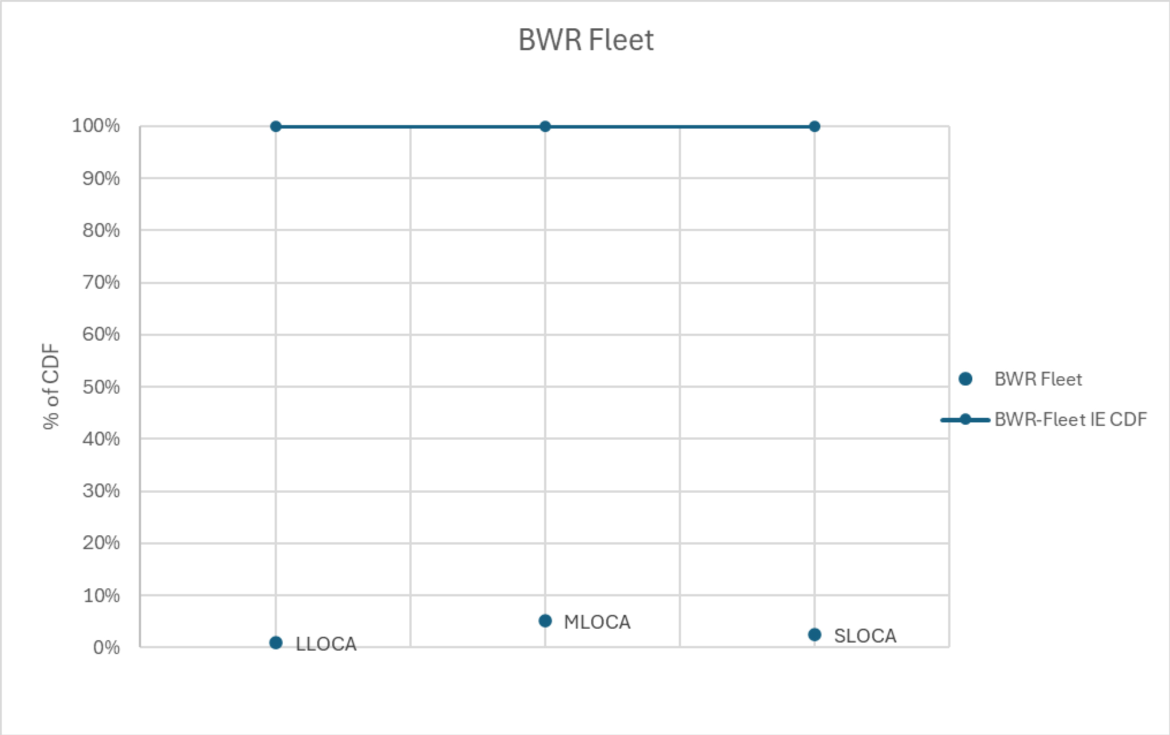
BWR-4 LOCA CDF

1. Inflection increase for MLOCAs – still ~2 orders of magnitude less than internal events CDF and ~3 orders of magnitude less than overall plant CDF
2. SLOCA Risk > 4 order of magnitude less than internal events CDF (>5 orders less when taken against overall plant CDF)
3. LLOCA Risk > 2 orders of magnitude less than internal events CDF (>3 orders less when taken against overall plant CDF)

BWRs appear to exhibit improved LOCA risk behavior because of larger reactor vessels with more inventory (greater coping time for mitigating strategies), redundant safety injection capabilities/systems (high pressure coolant injection (HPCI), Low head safety injection (LHSI) (typically 2 trains), Reactor Core Isolation Cooling (RCIC), etc.)

Plant/Fleet Specific Risk from LOCA Events

LLOCA	> 6”
MLOCA	2” – 6”
SLOCA	3/8” – 2”



BWR Fleet LOCA Risk (CDF) Performance as a function of percentage of total CDF (internal events)

1. Similar performance as exhibited on previous slides
2. Macroscopic view that encompasses the totality of the US BWR fleet

Not all plants exhibit this risk performance, but many do. For plants that have invested in improving maintenance, PRA modeling fidelity, facility modifications to improve risk and safety performance, should there be a benefit for the improved risk and safety performance?

Summary of LOCA Risk Significance

- NUREG/CR-6932 provides evidence that VSLOCAs are not risk significant (CDFs E-9 and E-10 for PWRs and BWRs respectively)
- Representative utility and fleetwide sampling of CDFs for larger break sizes indicate:
  - PWRs and some BWRs seem to have a clear demarcation between the risk significance of SLOCAs and LLOCAs as compared to MLOCAs
  - BWRs (both the BWR-4 and fleet examples) demonstrate a significantly lower risk significance for LOCA events. On a fleet wide basis, the average LOCA risk significance is > one order of magnitude less than the total internal events CDF.
  - LLOCAs across all plant types appear to be > two orders of magnitude less than total internal events CDF.

Consequences of LOCA Events

- A discussion of the Risk of LOCA events would be incomplete without a discussion of consequences
  - PRAs utilize CDF and large early release frequency (LERF) to quantify risk
    - These risk quantifiers have long been accepted in RG-1.200, RG-1.174, and the ASME PRA Standard (ASME/ANS-RA-S-1.1-2024)
    - These risk metrics serve as surrogates to satisfy the quantitative health objective (QHO) with CDFs < E-4 and LERFs < E-5 considered as meeting or satisfying the QHO (ML022120663)
    - It is important to note that the risk metrics discussed above do not credit performance of the containment in demonstrating compliance with the QHO

The previous discussion related to LOCA risk significance would indicate LOCA events are at least several orders of magnitude removed from challenging the QHO

Consequences of LOCA Events

- All operating plants are required to perform deterministic analyses for offsite dose consequences to demonstrate compliance with § 50.67 (majority of the fleet) and § 100.11
  - These analyses **do** credit the performance of containment, but **do not** credit performance of the emergency core cooling system (ECCS) in accordance with GDC 35 or § 50.46
  - Excerpt from § 50.67:

The fission product release assumed for these calculations should be based upon a major accident, hypothesized for purposes of design analyses or postulated from considerations of possible accidental events, that would result in potential hazards not exceeded by those from any accident considered credible. Such accidents have generally been assumed to result in substantial meltdown of the core with subsequent release of appreciable quantities of fission products.

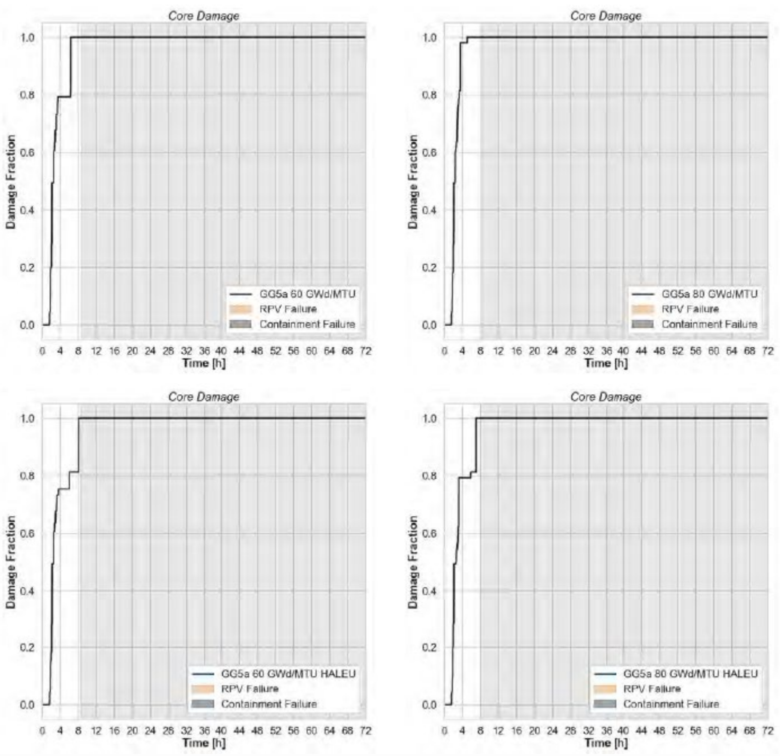
Consequences of LOCA Events

- SAND2023-01313 provides fission product release fractions and timings for use in analyzing offsite dose consequences

Table ES-1 Phase durations (in hours) and release fractions for all core variations (60 GWd/MTU, 80 GWd/MTU, LEU and HALEU).*

		Gap Release	Early In-vessel	Late In-vessel	Ex-vessel
BWR	Phase Duration	0.70	6.7	44.6	3.1
	Noble Gases	0.016	0.95	0.005	0.011
	Halogens	0.005	0.71	0.16	0.017
	Alkali Metals	0.005	0.32	0.021	0.009
	Te Group	0.003	0.56	0.19	0.003
	Ba/Sr Group	0.0006	0.005	0.002	0.038
	Ru Group	<1.0E-6	0.006	7.9E-05	<1.0E-6
	Mo Group	1.9E-05	0.12	0.002	2.3E-05
	Lanthanides	<1.0E-6	<1.0E-6	<1.0E-6	3.6E-05
	Ce Group	<1.0E-6	<1.0E-6	0.0	0.003
		Gap Release	**Early In-vessel**	**Late In-vessel**	**Ex-vessel**
PWR	Phase Duration	1.3	4.0	24.0	1.9
	Noble Gases	0.026	0.93	0.010	0.018
	Halogens	0.007	0.58	0.031	0.020
	Alkali Metals	0.003	0.50	0.013	0.015
	Te Group	0.006	0.55	0.019	0.005
	Ba/Sr Group	0.001	0.002	0.0001	0.011
	Ru Group	<1.0E-6	0.008	5.4E-05	<1.0E-6
	Mo Group	2.0E-05	0.15	0.002	0.002
	Lanthanides	<1.0E-6	<1.0E-6	<1.0E-6	1.4E-05
	Ce Group	<1.0E-6	<1.0E-6	<1.0E-6	0.0006

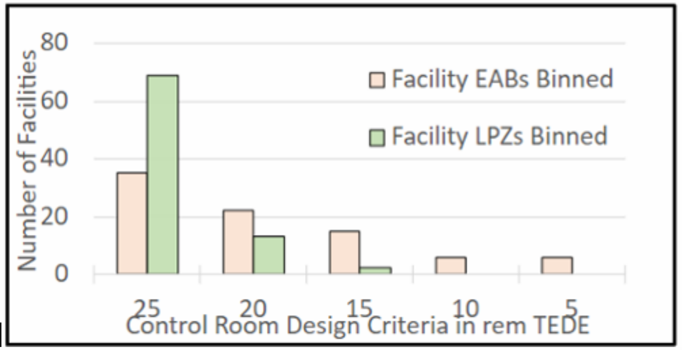
*This analysis uses the 50th percentile.



The radiological source term used for §50.67 analyses assumes the majority of the dominant dose significant nuclides are released and correspond to 100% core damage (no credit for ECCS)

Consequences of LOCA Events

- A recent polling of the US fleet § 50.67 compliant analyses demonstrate that ~97% of the fleet show doses at the low population zone would result in doses less than 10 rem TEDE
  - Dose exposures < 10 rem TEDE have been shown to result in latent and acute health effects no different from zero exposure (ML24150A080). In essence, statistically there is no difference in health effects to an individual exposed to 10 rem TEDE versus 0 rem TEDE.
  - The analysis performed to quantify the doses at the LPZ do not take credit for ECCS performance with demonstrated compliance to GDC 35 and § 50.46
  - These analyses still assume coincident worst single failures and coincident loss of offsite power (if it makes the consequences more conservative)



EAB Criteria	Design Criteria	Analysis of Record MHA-LOCA Data Binning			
		Exclusion Area Boundry		Low Population Zone	
		Facility EABs Binned	Percent of Facilities Binned	Facility LPZs Binned	of Facilities Binned
EAB ≤ 5	25	35	42%	69	82%
5 < EAB ≤ 10	20	22	26%	13	15%
10 < EAB ≤ 15	15	15	18%	2	2%
15 < EAB ≤ 20	10	6	7%	0	0%
20 < EAB	5	6	7%	0	0%

US Fleet LPZ Binning showing ~97% of plants approved to use an AST would result in LPZ doses < 10 rem TEDE (ML24066A177).

Consequences of LOCA Events

- Conclusion
  - Plant risk metrics utilizing PRAs demonstrate extremely low risk significance of LOCA events
  - CDF and LERF for LOCA events based on plant PRA data are multiple orders of magnitude away from challenging acceptability to the QHO
  - Deterministic analyses (minimum performance of safety systems, worse single failure, coincident LOSP) demonstrate the dose consequences of 100% core damage events with no credit for ECCS performance compliant with GDC 35 and § 50.46 are < 10 rem TEDE for 97% of the US fleet which results in acute and latent health effects to the public that are statistically no different from zero!

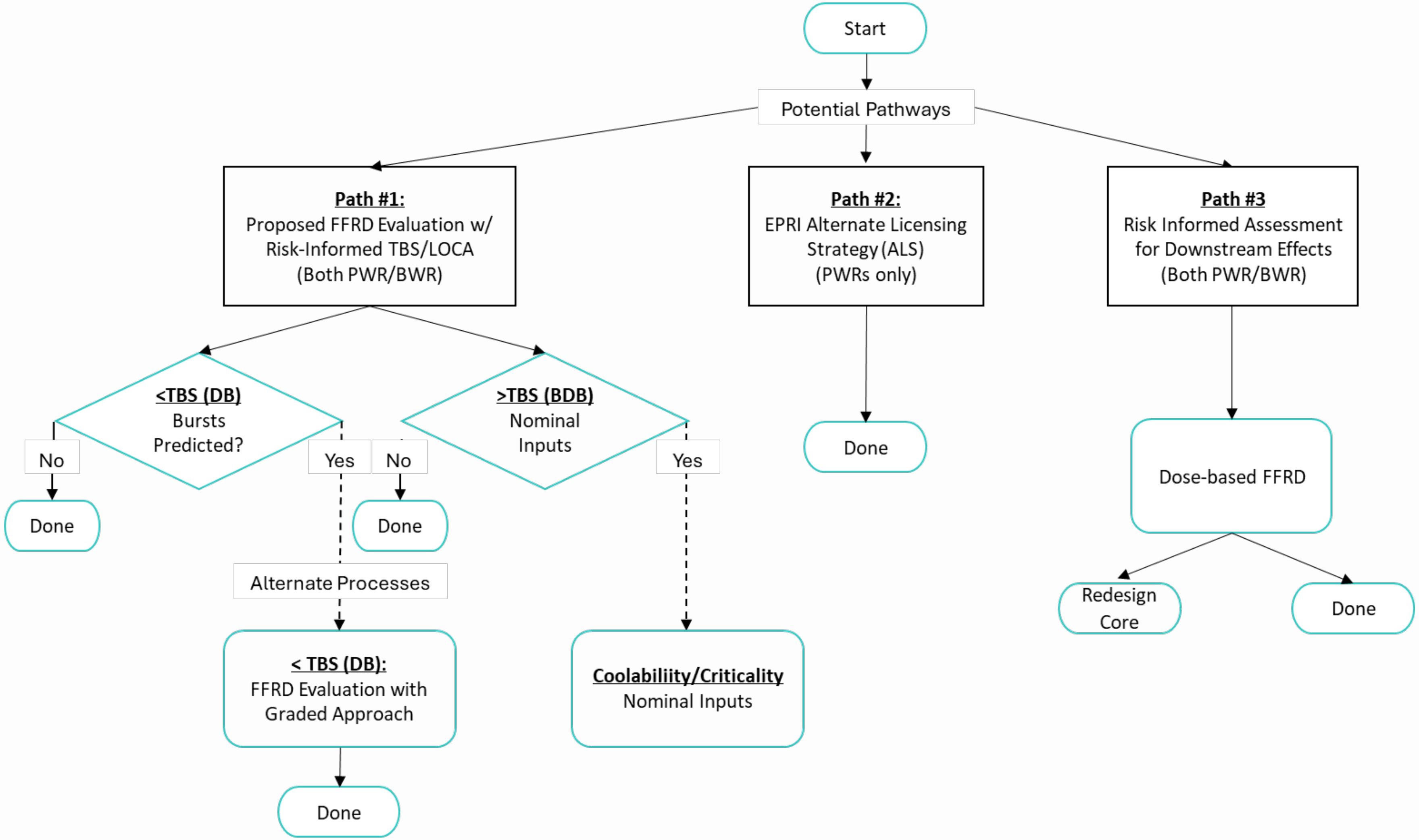
Is GDC 35 and §50.46 important for demonstrating the health and safety of the public (QHO) and environment, or is it a defense in depth rule intended to demonstrate multiple layers of protection exist? If a defense in depth rule and design criteria can it be risk informed?

Consequences of § 50.46 LOCA Events FFRD

- Radiological source terms of MHA-LOCA events performed in accordance with RG-1.183 and compliant with § 50.67 bound the dose consequences of LOCA events with FFRD (ML21197A067)

If the dose consequences of FFRD are bounded by events already analyzed and shown to have no impact on the health and safety of the public, are LOCA events with FFRD a safety issue? If FFRD is not a safety issue and doesn't challenge the QHO can it be risk-informed?

Industry Prospective Pathways to Risk-Inform FFRD



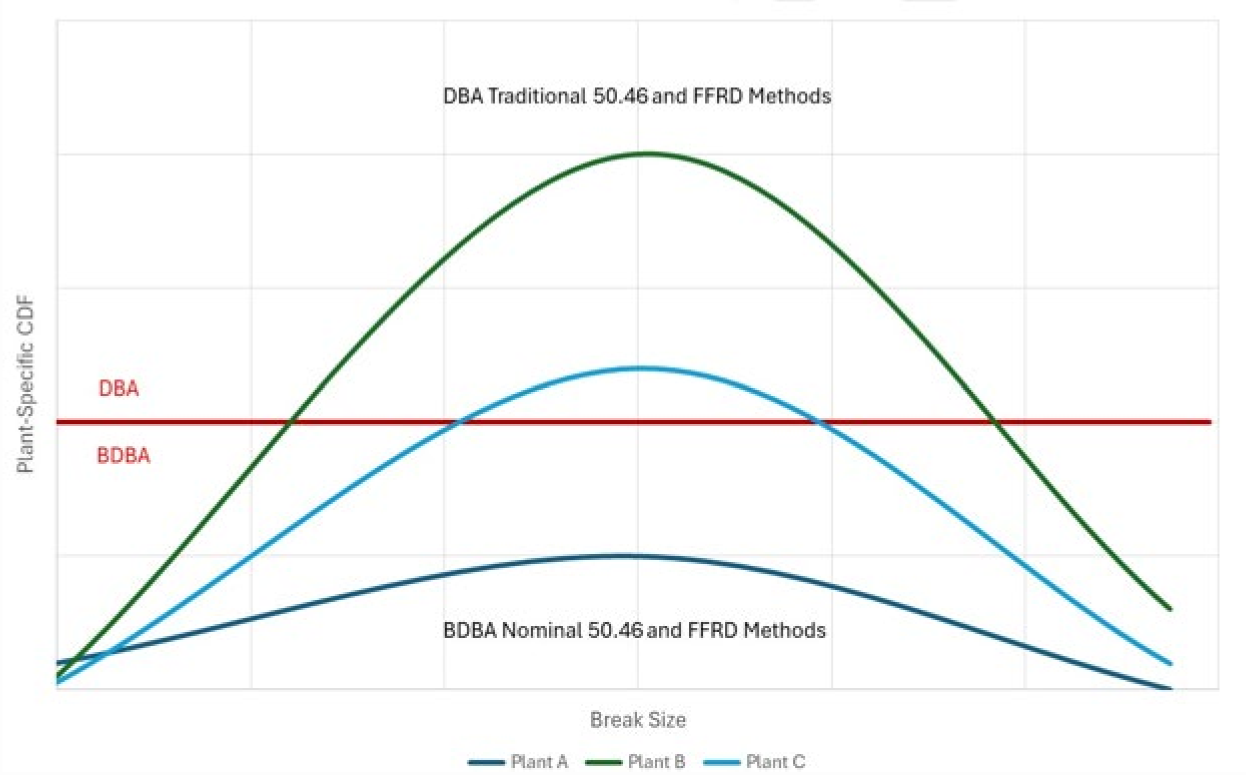
Industry Prospective Pathways to Risk-Inform FFRD

<u>Path 1:</u> Plants would submit for approval of a risk-informed TBS based on the risk significance of LOCA events in their plant PRAs

- For LOCA break sizes > RI-TBS, analyses utilizing nominal inputs would be used to demonstrate no burst of fuel rods subject to FFRD
- For LOCA break sizes < RI-TBS traditional 50.46 analyses would be performed to show no burst of fuel rods subject to FFRD
    - Given that FFRD is not a safety issue that challenges the QHO, and a plant's ability to demonstrate extremely low risk-significance of LOCA events (BWR example) to address FFRD < RI-TBS a graded approach for the no-burst analysis could also be justified

Industry Prospective Pathways to Risk-Inform FFRD

Path1: Figure demonstrating how a graded approach for addressing FFRD < RI-TBS could be justified:



Industry Prospective Pathways to Risk-Inform FFRD

Path 2: EPRI Alternative Licensing Strategy (ALS) Methodology (PWRs Only)

- Currently under review by NRC
- Industry believes only viable near-term approach to achieve strategic objectives by 2030 since it would not require a rulemaking change to implement
    - May require a policy interpretation
- Utilizes performance monitoring already in place at plants (LBB), probabilistic fracture mechanics analyses (xLPR) to demonstrate FFRD from large pipe breaks in the primary coolant loop are not credible
    - Generically addresses FFRD for small and intermediate break sizes for some PWRs

Industry Prospective Pathways to Risk-Inform FFRD

Path 3: Risk-Informed Assessment for Downstream Effects

- Innovative NRC developed concept that demonstrates acceptability for FFRD using dose consequence analyses compliant with § 50.67.
  - § 50.46 could be revised to clarify FFRD is beyond design basis and not safety significant
  - Satisfactory demonstration of dose criteria ensures the health and safety of the public is preserved
- Downstream effects including coolability and criticality could be addressed utilizing:
  - State of the Art Reactor Consequence Analyses (SOARCA) NUREG-1935, NUREG/CR-7110 Volumes 1 & 2, NUREG/BR-0359, and insights from Severe Accident Mitigation Guidelines (SAMGs)
  - Among others
- Industry believes Path 3 has many advantages from leveraging risk-informed insights and learnings that demonstrate FFRD is not risk-significant and does not represent a safety issue that would challenge the health and safety of the public and environment

Discussion Period

Break

Discussion Period

Public Comment Period

Closing Remarks

Adjourn