Proposed Rule: Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

December 17-18, 2024

Opening Remarks

Theresa Clark
Director
Division of Safety Systems

Overview of
Increased Enrichment Rulemaking

Philip Benavides
Project Manager
Reactor Rulemaking & Project Management Branch

Issue Identification

- **Regulatory Issue:**
  - Current licensing framework allows for the use of ≤ 5 weight percent uranium-235; however, technology developments may require numerous exemptions to utilize fuel enriched above 5 weight percent.
- **Proposed Solution:**
  - Rulemaking would provide for a generically applicable standard informed by public input, providing consistent and transparent communication, rather than individual licensing requests as discussed in SECY-21-0109, "Rulemaking Plan on Use of Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors."
- **Commission Rulemaking Plan Approval:**
  - Staff request to pursue rulemaking and develop a regulatory basis was approved by the Commission via SRM-SECY-21-0109.

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

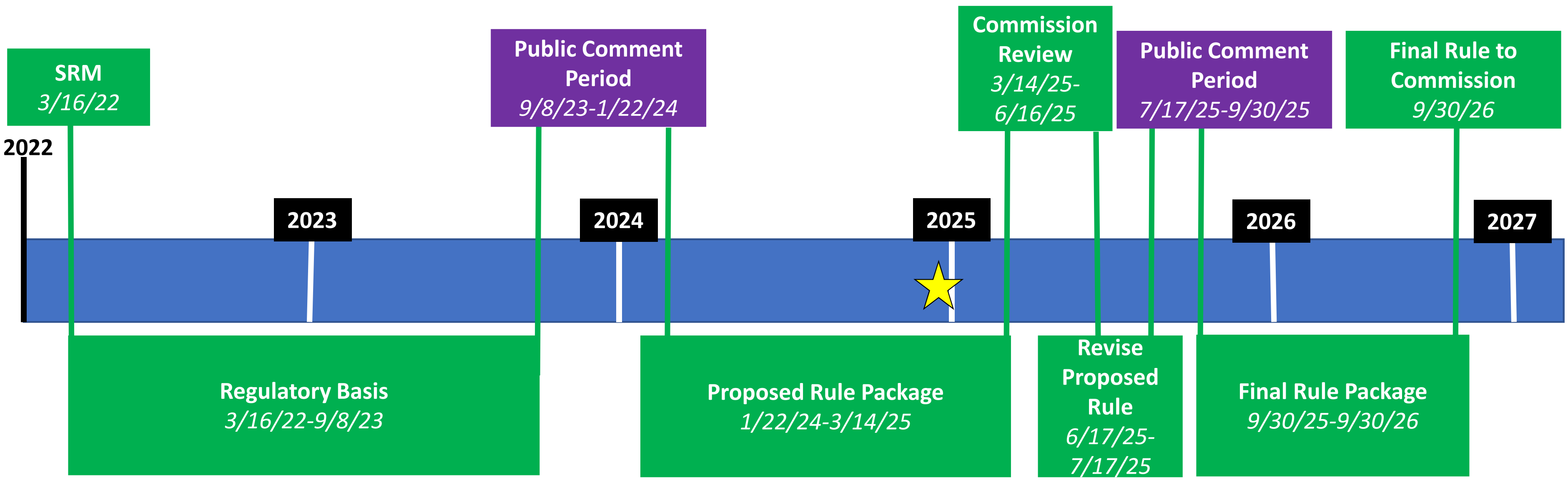
SRM-SECY-21-0109 Overview

- SRM-SECY-21-0109 was issued on 3/16/22, in response to SECY-21-0109.
  - The Commission approved the staff's proposal to initiate a rulemaking to amend requirements for the use of light water reactor fuel containing uranium enriched to greater than 5.0 weight percent uranium-235.
  - Provisions to the rule should only apply to High-Assay Low Enriched Uranium (HALEU).
  - Fuel Fragmentation, Relocation, and Dispersal (FFRD) should be appropriately addressed.
  - Staff directed by the Commission to take a risk-informed approach.

Status of Rulemaking Activity

- **The NRC staff issued a regulatory basis on September 8, 2023 (ADAMS Accession No. ML23032A504)**
- **Stakeholder Involvement:**
  - Before Regulatory Basis Issued:
    - Public Meeting on June 22, 2022 (ML22208A001)
  - Post Regulatory Basis Issued:
    - Public Meeting on October 25, 2023 (ML23319A259)
    - Comment Period closed on January 22, 2024
    - Publicly shared Fuel Dispersal insights at the NRC's Annual Higher Burnup Workshop on September 3, 2024 (ML24277A161)
- **The Increased Enrichment proposed rule package is in concurrence.**
  - Proposed rule due to the Commission: March 2025

Status of Rulemaking Activity



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

Rulemaking Topics

- **The IE rulemaking addresses the following topics:**
  - Criticality Accident Requirements (10 CFR 50.68)*
  - Uranium Fuel Cycle Environmental Data - Table S-3 (10 CFR 51.51)
  - Environmental Effects of Transportation of Fuel and Waste - Table S-4 (10 CFR 51.52)
  - Packaging Requirements for Fissile Material Transportation (10 CFR 71.55)*
  - Control Room Design Requirements (10 CFR 50.67 and GDC-19)*
  - Fuel Fragmentation, Relocation, and Dispersal*

*ACRS Subcommittee Meeting Topics

Associated Guidance

- **Control Room Design Requirements (10 CFR 50.67 and GDC-19)**
  - DG-1425, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- **Fuel Fragmentation, Relocation, and Dispersal**
  - DG-1261, Revision 1, "Measuring Breakaway Oxidation Behavior"
  - DG-1262, Revision 1, "Determining Post-Quench Ductility"
  - DG-1263, Revision 1, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding"
  - DG-1426, "An Approach for Risk-Informed Evaluation Process Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors"
  - *DG-1428, "Plant-Specific Applicability of Transition Break Size"*
  - DG-1434, "Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents"

Note: DG-1428 to be presented in January 2025

Agenda (December 17)

Topic	Presenter	Organization
NRR Staff Leadership	Theresa Clark, Director	NRR/DSS
IE Overview, Status of Rulemaking	Philip Benavides, Rulemaking PM	NMSS/REFS/RRPB
Criticality Accident Requirements (10 CFR 50.68)	Charley Peabody	NRR/DSS/SNSB
Fissile Packaging Requirements (10 CFR 71.55)	Jason Piotter	NMSS/DFM/NF
Fuel Fragmentation, Relocation, and Dispersal	Joseph Messina Robert Tregoning David Rudland Kristy Bucholtz	NRR/DSS/SFNB RES/DE NRR/DNRL NRR/DRA/APOB

Agenda (December 18)

Topic	Presenter	Organization
NRR Staff Leadership	Theresa Clark, Director	NRR/DSS
Fuel Fragmentation, Relocation, and Dispersal (Continued)	James Corson	RES/DSA/FSCB
Control Room Design (10 CFR 50.67 and General Design Criteria 19)	Elijah Dickson	NRR/DRA/ARCB

Questions

Criticality Accident Requirements (10 CFR 50.68)

Charley Peabody
Nuclear Engineer
Nuclear Systems Performance Branch

10 CFR 50.68(b) Historical Overview

- Final Rule issued in 1998
- Provides alternative to 10 CFR 70.24
- Implements k-effective safety limits
- Limits enrichments of ≤ 5% by weight U-235
- Industry desires to increase enrichment beyond 5% at operating LWRs

10 CFR 50.68(b) Recent Developments

- Single License Amendment for Lead Test Assemblies (LTA) utilizing enrichments beyond 5% has been approved to date
- NRC Contracted Research Study
  - ORNL/TM-2024/3350 “Scoping Studies on the Impacts of Increased Enrichment on Nuclear Criticality Safety,” May 2024 (ADAMS Accession Number ML24163A016)
- Reconfirmed Regulatory Basis Decision not to Update Regulatory Guide 1.240 “Fresh and Spent Fuel Pool Criticality Analysis” March 2021

10 CFR 50.68(b) Proposed Rule Unofficial Redline

- §50.68(b)(7) would be changed to state "The maximum nominal U-235 enrichment of the fresh fuel assemblies is limited to five (5.0) percent by weight or to the value specified in the operating license."
- No changes to any other paragraphs.

10 CFR 50.68(b) Proposed Rule Benefits

- The k-effective safety limits specified in 10 CFR 50.68(b)(2), (3), and (4) are maintained
- Increased enrichment criticality safety impacts will be evaluated as part of the fuel transition LAR review process
- Allows for the entire range of high-assay low-enriched uranium (HALEU) to be utilized
- This alternative preserves 10 CFR 50.68(b) compliance for all LWRs currently operating in the United States

10 CFR 50.68(b) Proposed Rule

Member questions or comments for the staff?

Packaging Requirements for Fissile Material Transportation 10 CFR 71.55

Jason Piotter
Team Leader - New Fuels Team
Division of Fuel Management - NMSS

Packaging Requirements of 10 CFR 71.55: Summary of Regulatory Issue

Current Regulations

- § 71.55(b) applicants evaluate a single package, optimally moderated and reflected
- § 71.55(g) Provides an exception to § 71.55(b) for packages containing UF_6
- § 71.55(g)(4) Specifies that enrichment of UF_6 cannot exceed 5 weight percent U-235

Regulatory History

Proposed rule (§ 71.55(g)) issued 67 FR 21390, April 30, 2002, Final Rule issued 69 FR 3698, January 26, 2004

Codified NRC longstanding practice to provide an exception to § 71.55(b)

10 CFR 71.55: Certificate of Compliance (CoC) Options for Enrichments Greater than 5 weight percent U-235

1

Evaluate UF_6 packages with optimum moderation per §71.55(b), or

2

Request an exemption to §71.55(b), or

3

Request approval under §71.55(c) for an exception to the optimum moderation requirement in §71.55(b).

Regulatory Basis - Rulemaking Alternatives

Original Recommendation

Option 1: No Action	71.55(b), 71.55(c), exemptions (Also applies to Options 2 and 3)
Option 2: Increase Enrichment Limit	20 weight percent U-235
Option 3:	Remove Enrichment Limit

Proposed Rule - Rulemaking alternatives

Current Recommendation → Option 2

Option 1: No Action	71.55(b), 71.55(c), exemptions (Also applies to Options 2 and 3)
Option 2: Increase Enrichment Limit	10 weight percent U-235, *defense in depth requirements*
Option 3: Increase Enrichment Limit	20 weight percent U-235, *defense in depth requirements*

10 CFR 71.55(g)(4): Updated Recommended Alternative

Alternative 2

Increase enrichment limit to 10.0% wt U-235, with prescriptive defense in depth requirement for additional protection of fill valve or other device

10 CFR 71.55(g): Updated Rule Text

(4) The uranium is enriched to not more than *10 weight percent* uranium-235; and

(5) A design feature is incorporated to protect the valve or other fill device from impact for contents with uranium-235 enriched above 5 weight percent and up to 10 weight percent.

Fissile Material Transportation Packages - Summary

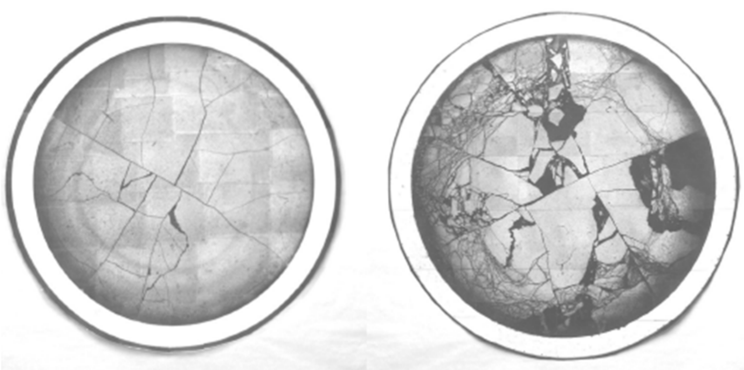
- Transportation packages for UF_6 have a certification pathway with enrichments up to 20 weight percent U235. While our current regulations as written are sufficient to transport higher enriched UF_6, we are providing a non mandatory modification of the current enrichment limit that allows for more regulatory certainty while maintaining safety

Fuel Dispersal and 10 CFR 50.46a Overview

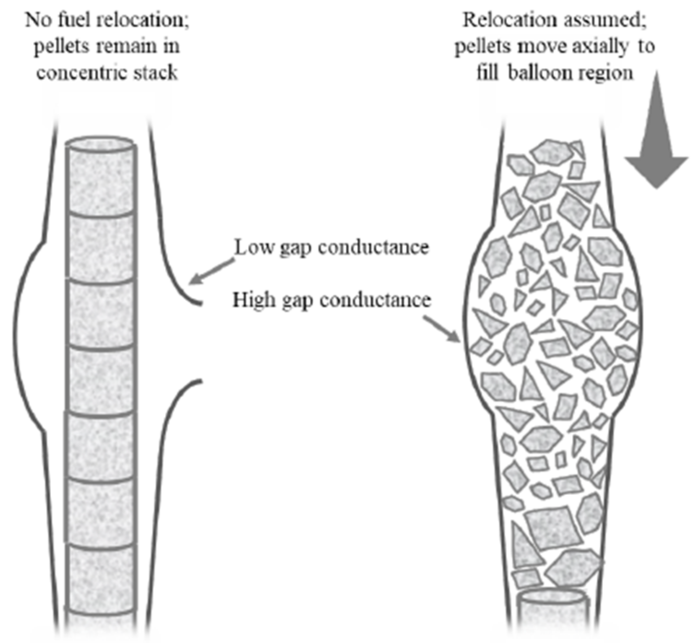
Joseph Messina
Reactor Systems Engineer
Nuclear Methods and Fuel Analysis

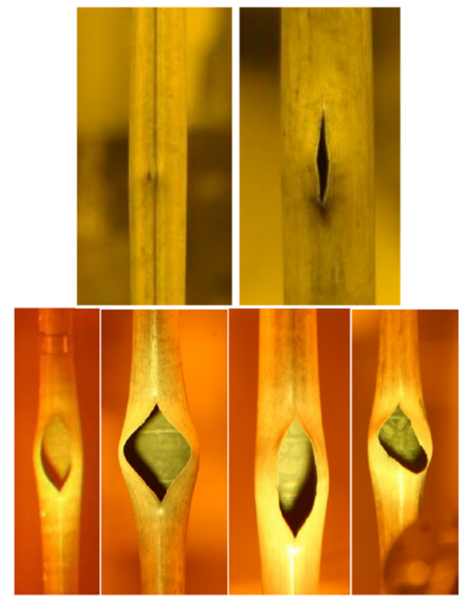
Fuel Fragmentation, Relocation, and Dispersal (FFRD)

- At 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



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





Burst openings from Studsvik LOCA tests (NUREG-2121)

Fuel Dispersal: Background and Regulatory Issue

- The 50.46 acceptance criteria date to 1974 when FFRD were not known phenomena
- Acceptable approaches to demonstrate compliance with the regulations have ensured that catastrophic failure of the fuel rod structure and loss of fuel bundle configuration are precluded
  - Fuel dispersal would be a departure of precedent
- Fuel dispersal is not explicitly addressed within the current regulations
  - Draft proposed rule language allows for some flexibility regarding fuel dispersal
  - DG-1434 provides guidance for addressing fuel dispersal within the 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.

Public Comments Overview, Part 1 of 4

- No unanimous alternative recommended by industry
- Industry recommendations were strongly based on the qualitative schedule impacts published in the Regulatory Basis.
  - These were quick and crude estimates.
  - Staff has learned more, and accuracy of these estimates has improved.
- UCS and two members of public:
  - Do not support any alternative that allows for fuel dispersal
- One member of public:
  - Recommends waiting until more research and analysis is performed for fuel dispersal

Public Comments Overview, Part 2 of 4

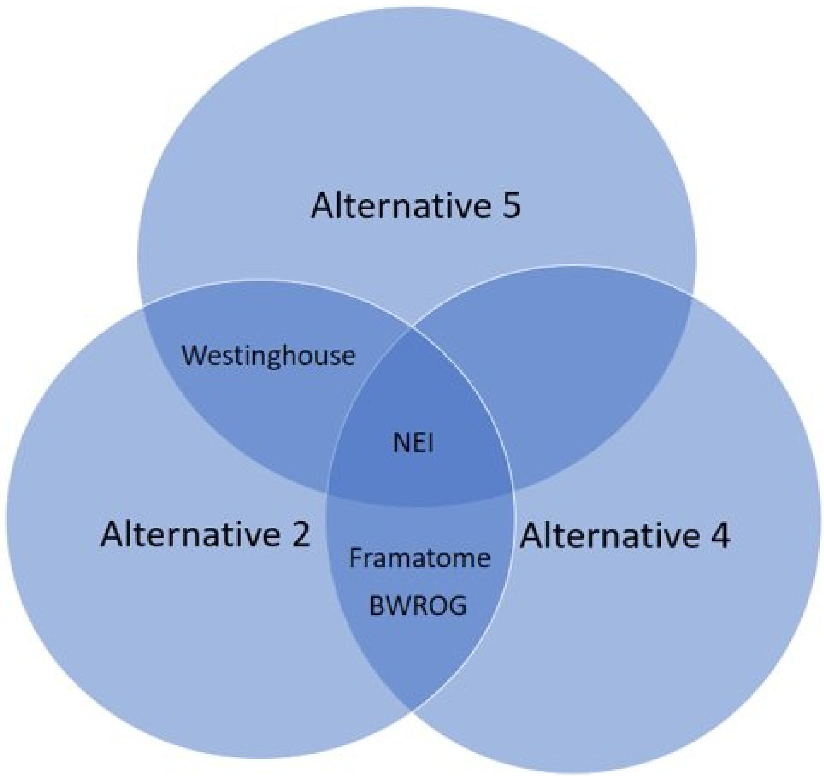
Alternatives 1 and 3:	Alternative 4:	Alternative 5:	Alternate Approaches:
• No support	• NEI and BWROG supported Alternative 4 • BWROG does not see Alternative 5 as a solution	• NEI and Westinghouse supported a modified Alternative 5 (ALS)	• Framatome and PWROG suggest using integrated decision making as done to disposition in-vessel downstream effects (IVDEs) associated with GSI-191

Public Comments Overview, Part 3 of 4

Alternative 2:

- NEI, BWROG, and Westinghouse support Alternative 2 combined with an updated 50.46c as a separate rulemaking due to perceived schedule or as a backup
- Aspects of Framatome's response and their 2023 white paper align with Alternative 2
- NEI, Westinghouse, and Framatome stated that even with a no-dispersal criterion, Alternative 2 would be reasonable with true best-estimate calculations

Public Comments Overview, Part 4 of 4



Previous 50.46a History

- **Commission SRMs on rule development**:
  - SRM-SECY-02-0057 (ML030910476)
  - SRM-SECY-04-0037 (ML041830412)
  - SRM-SECY-05-0052 (ML052100416)
  - SRM-SECY-07-0082 (ML072220595)
- **ACRS letters**:
  - ACRS letter on initial draft final rule (ML063190465): 11/16/2006
  - ACRS letter on NUREG-1829 and draft NUREG-1903 (ML073440143): 12/20/2007
  - ACRS letter on draft final rule (ML102850279): 10/20/2010
    - Sept. 2010 ACRS Subcommittee meeting transcript: ML102910759
    - Oct. 2010 ACRS Full Committee meeting transcript: ML102860120
- **SECY-10-0161 (ML102300252)**: Draft final rule was submitted to the Commission – 12/10/2010
- **Email from Greg Bowman to SECY** requesting to withdraw 10 CFR 50.46a rulemaking (ML121500380; submitted in response to verbal direction by Chairman Jaczko) – 4/20/2012
- **SRM-SECY-10-0161**: Commission approved staff's request to withdraw SECY-10-0161 and re-evaluate it to ensure compatibility with future Commission direction related to recommendations following Fukushima – 4/26/2012
- **SECY-16-0009 (ML16028A189)**: staff recommended stopping 10 CFR 50.46a rulemaking as a part of prioritization and re-baselining of agency activities – 1/31/2016
- **SRM-SECY-16-0009 (ML16104A158)**: Commission approved staff recommendation to stop 50.46a rulemaking – 4/13/2016

Fuel Dispersal Path Forward

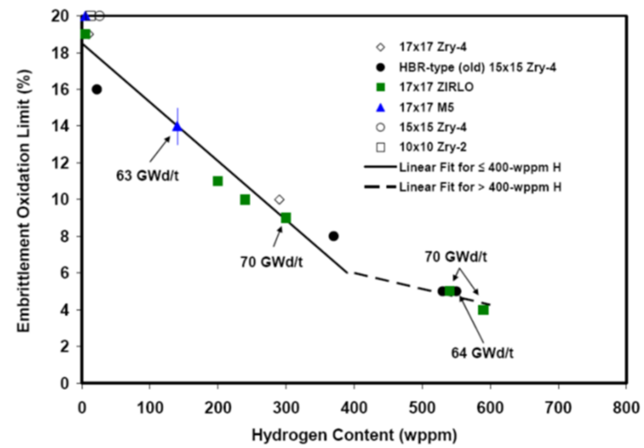
- The NRC staff plans to risk-inform LOCAs by modernizing 50.46a (based on Alternative 2) in the IE draft proposed rule to facilitate addressing fuel dispersal
  - Support for Alternative 2 expressed in many public comments
  - Smallest impact on the IE Rulemaking schedule of the alternatives that received support
    - Leveraged 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
  - LOCAs below the TBS will not be affected by this rule
- 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 prescriptive definition of core coolability)

50.46c Background

- 50.46c was a draft final rule that revised the ECCS acceptance criteria to be performance-based and reflect research findings on embrittlement of zirconium alloy cladding under LOCA conditions
  – Submitted to the Commission via SECY-16-0033 in March 2016
  – 1997-2016 NRC LOCA research program is documented in NUREG/CR-7219
- Substantial ACRS interactions on 50.46c:
  – ACRS letter on draft final rule issued February 2016: ML16048A522
- The SECY-16-0033 research findings show that under the current regulations (17% MLO and 2200°F PCT), post-quench ductility is not assured following a postulated LOCA
- New embrittlement mechanisms discussed in SECY-16-0033:
  – Hydrogen-enhanced beta layer embrittlement
  – Cladding ID oxidation
  – Breakaway oxidation



SRM-SECY-16-0033 (50.46c)

The Commission returned the 50.46c draft final rule package (SECY-16-0033) to the staff in April 2024 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.

- 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.
- Entities that elect to adopt 50.46a would be expected to address the embrittlement research findings
- 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).

Scope of Work Associated with updated 50.46a 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 (previously part of the 50.46c rulemaking):
  - 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”

Adequate Protection

While this rule relaxes the regulatory treatment of LOCAs above the TBS, the NRC staff believe that it maintains the adequate protection of public health and safety because:

- The initiating event frequency for such events are very low and the NRC will ensure that it is low and remains low on a plant-specific basis
- The NRC will ensure that risk increases from changes due to this rule are minimal and that there are not large increases in the overall plant risk
- The NRC will maintain regulatory control over such LOCAs, continuing to review ECCS evaluation models and plant-specific LOCA analyses, as done to date

50.46a and Fuel Dispersal Team

NRR/DSS:

- Joseph Messina
- Ashley Smith
- John Lehning
- Scott Krepel

NRR/DNRL:

- David Rudland
- David Dijamco
- Seung Min
- Eric Palmer

NRR/DRA:

- Kristy Bucholtz
- Michelle Kichline

NRR/DEX:

- Se-Kwon Jung

RES/DE:

- Robert Tregoning
- Matthew Homiack
- Christopher Nellis

RES/DSA:

- James Corson
- Andrew Bielen

Technical Basis of Original 10 CFR 50.46a Transition Break Size

Rob Tregoning
Senior Advisor for Materials
Office of Nuclear Regulatory Research, Division of Engineering

Presentation Objective and Outline

- **Objective:** Summarize the historical (i.e., pre-2024) technical basis use to develop the transition break size (TBS)

- **Outline:** TBS development
  - LOCA frequency assessment (NUREG-1829)
  - TBS selection
  - Confirmation of seismic integrity (NUREG-1903)

NUREG-1829: Scope and Objectives

- Develop piping and non-piping passive system LOCA frequencies as a function of leak rate and operating time up to the end of the license extension period (i.e., 60 years) using expert elicitation
  - LOCAs which initiate in unisolable portion of reactor coolant system
  - LOCAs related to passive component aging, tempered by mitigation measures
- Determine LOCA frequency distributions for typical plant operational cycle and history
- Assume that no significant changes will occur in future plant operating profiles

NUREG-1829: Historical LOCA Frequency Evaluation

- LOCA frequencies previously developed from operating history
- Notable Previous Evaluations:
  - WASH-1400 (1975): Estimates largely based on experience in other industries
  - NUREG-1150 (1987): Updated the WASH-1400 distributions to account for the additional service since WASH-1400
  - NUREG/CR-5750, Appendix J (1998): Updated original WASH-1400 study for SB LOCAs while MB and LB LOCA frequencies were calculated from precursor leaks in class 1 systems
- Operating history, by itself, may not accurately reflect future performance and requires significant extrapolation for MB and LB LOCA frequencies

NUREG-1829: Expert Elicitation Process

- Classical approaches
  - Operating experience: LOCA events are rare
  - Plant modeling: Number and diversity of possible failure modes is too complex to accurately model
- Expert elicitation is a formal process for providing quantitative estimates for the frequency of physical phenomena when the required data is sparse and when the subject is too complex to accurately model
- Elicitation has been used often at NRC
  - Development of seismic hazard curves
  - Performance assessments for high-level radioactive waste repository
  - Determination of reactor pressure vessel flaw distributions

NUREG-1829: Elicitation Approach

- Conduct pilot elicitation
- Select panel and facilitation team
- Develop technical issues
- Quantify base case estimates
  - Develop quantitative estimates for well-defined piping conditions
  - Quantify non-piping precursors and targeted failure scenarios
- Formulate elicitation questions
- Conduct individual elicitations
- Analyze quantitative results and qualitative rationale
- Summarize and document results
- Conduct internal and external review of process and results

NUREG-1829: Pilot and Elicitation Team

- **Pilot Elicitation**
  - Conducted using 11 internal (NRC) experts with broad knowledge-base
  - Provided interim results for rulemaking development
  - Developed possible framework for subsequent elicitation and its strengths and weaknesses
  - Identified technical issues for subsequent consideration

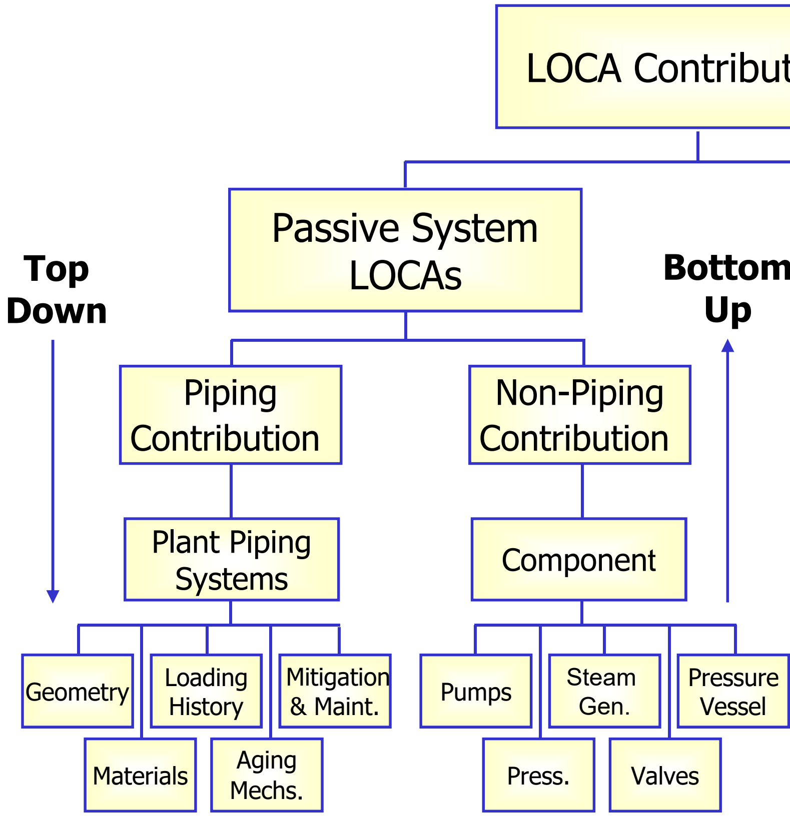
- **Panel and Facilitation Team**
  - Individual elicitations conducted for each expert, led and monitored by a facilitation team
  - Twelve external experts assembled from nuclear industry, DOE laboratories, consultants, and international regulatory agencies with broad knowledge-base
  - Facilitation team comprised largely of NRC subject matter experts

NUREG-1829: LOCA Size Classification

- LOCA sizes based on flow rate to group plant system response characteristics
  - First three categories similar to NUREG-1150 and NUREG/CR-5750
  - Three additional LBLOCA categories used to determine larger break frequencies
- Correlations developed to relate flow rate to effective break area
- Three time periods evaluated
  - Current day ~ 2004 (average 25 years of operation)
  - End of design life (average 40 years of operation)
  - End of first life extension (average 60 years of operation)

Category	Flow Rate Threshold (gpm)	LOCA Size
1	> 100	SB
2	> 1500	MB
3	> 5000	LB
4	> 25,000	LB a
5	> 100,000	LB b
6	> 500,000	LB c

NUREG-1829: General Issue Classification

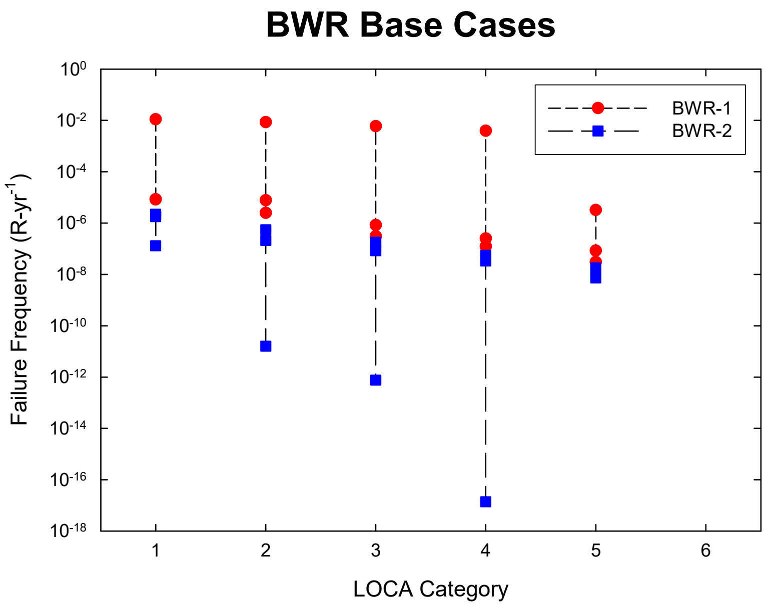


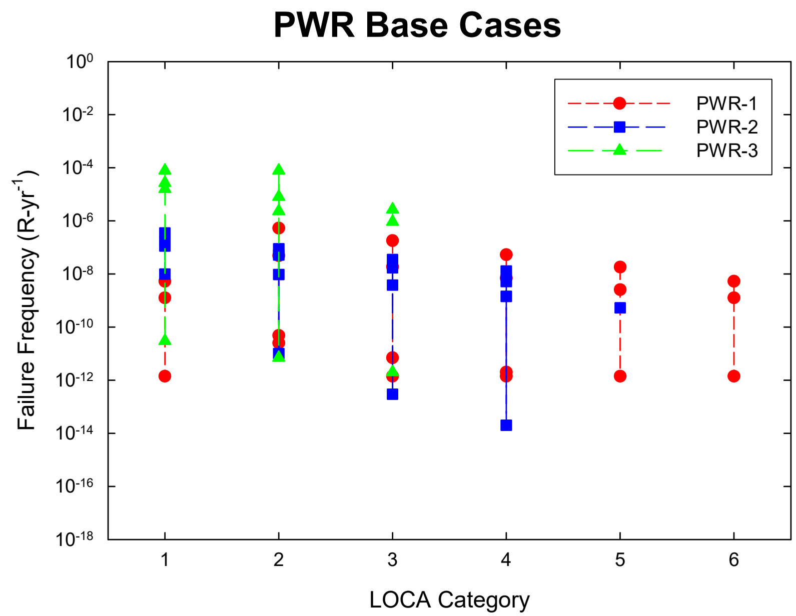
- Elicitation focuses on passive system LOCAs
- Important piping and non-piping attributes identified
- Elicitation structure supported top-down and bottom-up analysis

NUREG-1829: Piping Base Cases

- The base cases were available for anchoring the elicitation responses.
- Base case conditions specify the piping system, piping size, material, loading, degradation mechanism(s), and mitigation procedures
- Five base cases defined
  - BWR
    - Recirculation System (BWR-1)
    - Feedwater System (BWR-2)
  - PWR
    - Hot Leg (PWR-1)
    - Surge Line (PWR-2)
    - High Pressure Injection makeup (PWR-3)
- The LOCA frequency for each base case condition is calculated as a function of flow rate and operating time
- Four panel members individually estimated frequencies: two using operating experience and two using probabilistic fracture mechanics

NUREG-1829: Piping Base Cases Summary Results





- Large variability due to inconsistencies in both the conditions evaluated and differences in approaches
- Each base case participant presented their approach and results to entire panel
- Each panel member was asked to critique approaches & results during their elicitation session

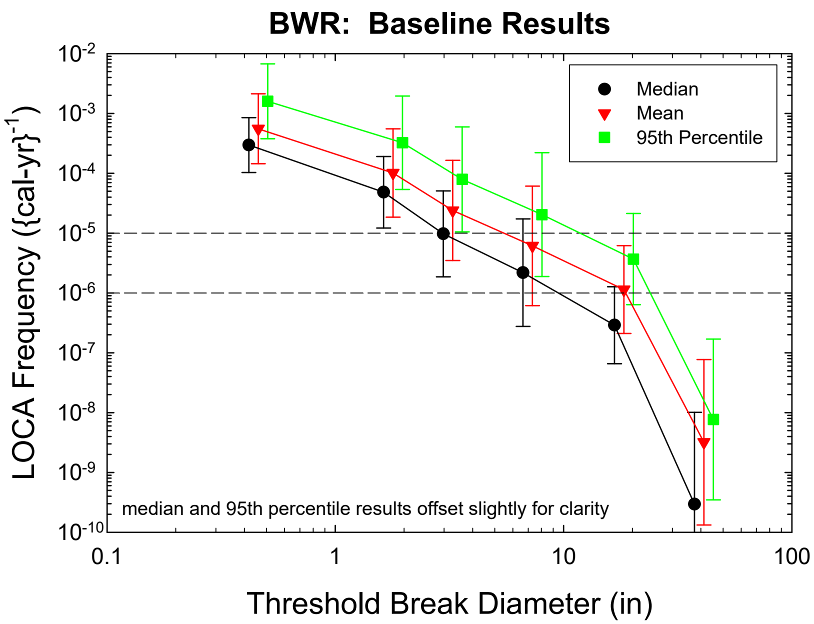
NUREG-1829: Non-Piping Base Cases

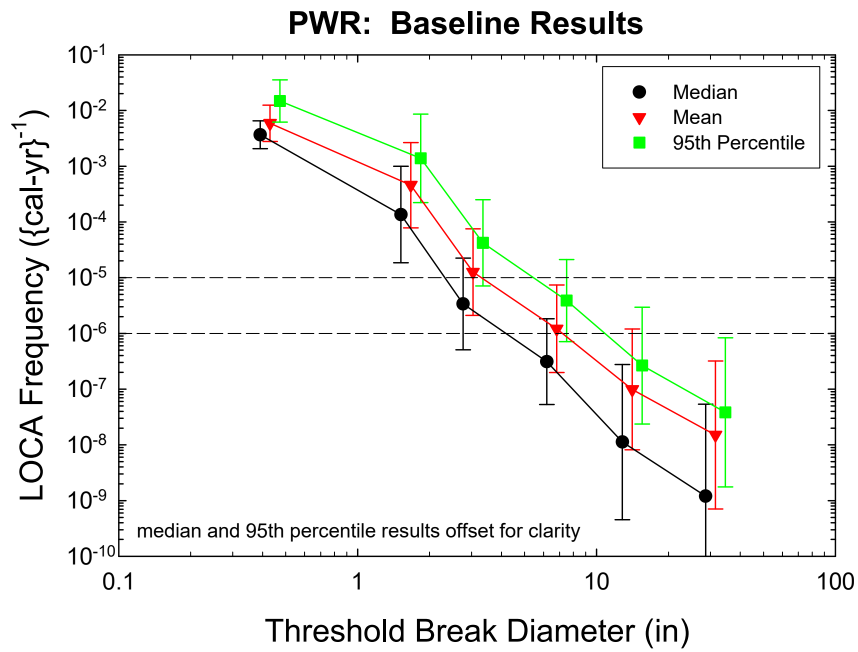
- The variety and complexity of the non-piping failure mechanisms makes the piping base case approach intractable
- Approach
  - Develop general non-piping precursor database
  - Use PFM modeling to develop LOCA frequencies for targeted degradation mechanisms
    - CRDM ejection
    - BWR vessel rupture: normal operating and LTOP
    - PWR vessel rupture: PTS
- Analysis method
  - Choose appropriate base case: non-piping precursor, piping precursor, piping base case, or non-piping base case
  - Determine relative likelihood of each non-piping failure scenario compared to chosen base case

NUREG-1829: Analysis of Elicitation Responses

- Calculate individual estimates for each panelist
  - Total BWR and PWR LOCA estimates
  - Approach is self-consistent and ensures that qualitative rationale and quantitative estimates match
- Aggregate individual estimates: Philosophy
  - Group results more accurate than any single estimate
  - Outliers should not dominate quantitative estimates
- Aggregate individual estimates: Approach
  - Combine parameters (i.e., mean, median, 5th & 95th percentiles) of individual distributions
  - Calculate confidence bounds associated with each parameter estimate
- Perform sensitivity analyses to evaluate calculation approach
- Final LOCA distributions reflect uncertainty and variability
  - Uncertainty: Individual panel member responses
  - Variability: Range of individual responses

NUREG-1829: Total LOCA Frequencies





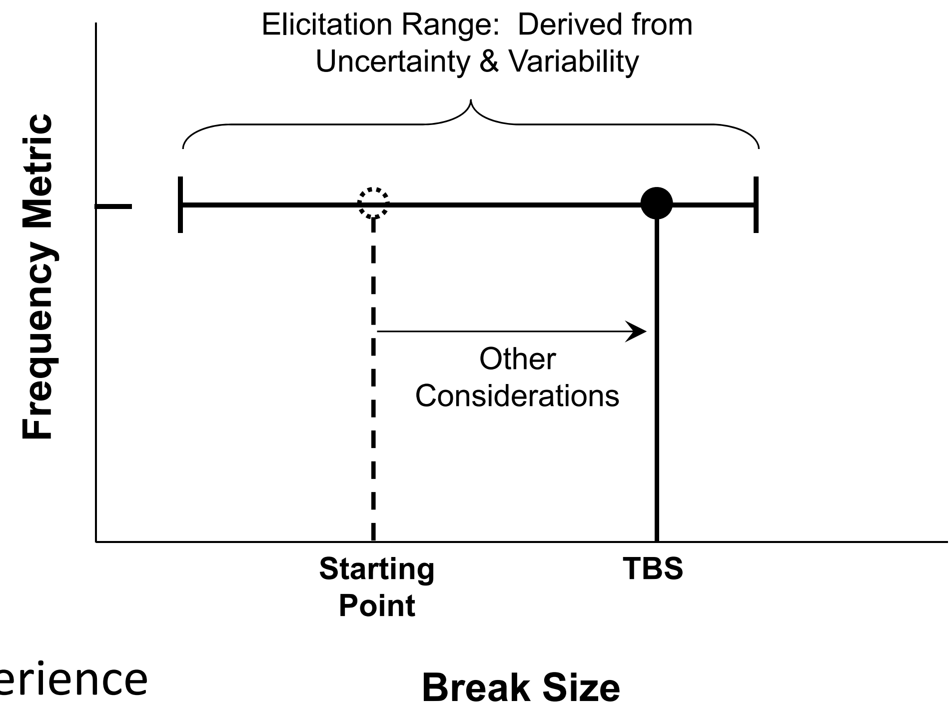
- 95% confidence bounds (i.e., error bars) account for diversity among panelists
- Differences between median and 95th percentiles reflect individual panelist uncertainty

NUREG-1829: Summary

- Formal elicitation process used to estimate generic BWR and PWR passive-system LOCA frequencies associated with material degradation during normal operations
- Piping and non-piping base cases were developed and evaluated for anchoring elicitation responses
- Panelists provided quantitative estimates supported by qualitative rationale in individual elicitations for underlying technical issues
  - Generally good agreement on qualitative LOCA contributing factors
  - Large individual uncertainty and panel variability in quantitative estimates
  - Results are generally comparable to NUREG/CR-5750 estimates
- Group results determined by aggregating individual panelists' estimates
  - Geometric mean aggregated results are consistent with elicitation objectives and results are generally comparable with NUREG/CR-5750 estimates
  - Alternative aggregation schemes can result in higher LOCA frequencies

Selection of Transition Break Size (TBS)

- NUREG-1829 results used as starting point
- Range of pipe sizes correlate to break frequency < 10^{-5}/yr (95th percentile)
  - BWRs: 13 to 20 inches
  - PWRs: 6 to 10 inches
- Selection should accommodate uncertainties
- Other types of LOCAs considered in determining TBS
  - Active LOCAs
  - Load-generated LOCAs (i.e., dropped heavy loads, water hammer)
  - Seismically induced LOCAs
- Actual plant piping design and operating experience considered in final selection



TBS Selection

- TBS is defined as a pipe break that is the size of the cross-sectional flow area of largest pipe attached to the main coolant loop
  – For PWRs, the size of the largest pipe attached to the cold or hot leg main loop piping (≈ 12 inches)
  – For BWRs, the size of the largest pipe in either of the RHR or Feedwater systems inside primary containment (≈ 20 inches)
- Supporting rationale
  – Next larger pipes are significantly less likely to break
  – Piping sizes < TBS have experienced most significant degradation
  – Accommodates uncertainties and provides regulatory stability as variation in future LOCA frequencies estimates not likely to require new TBS definition to maintain acceptable risk

NUREG-1903: Objective and Approach

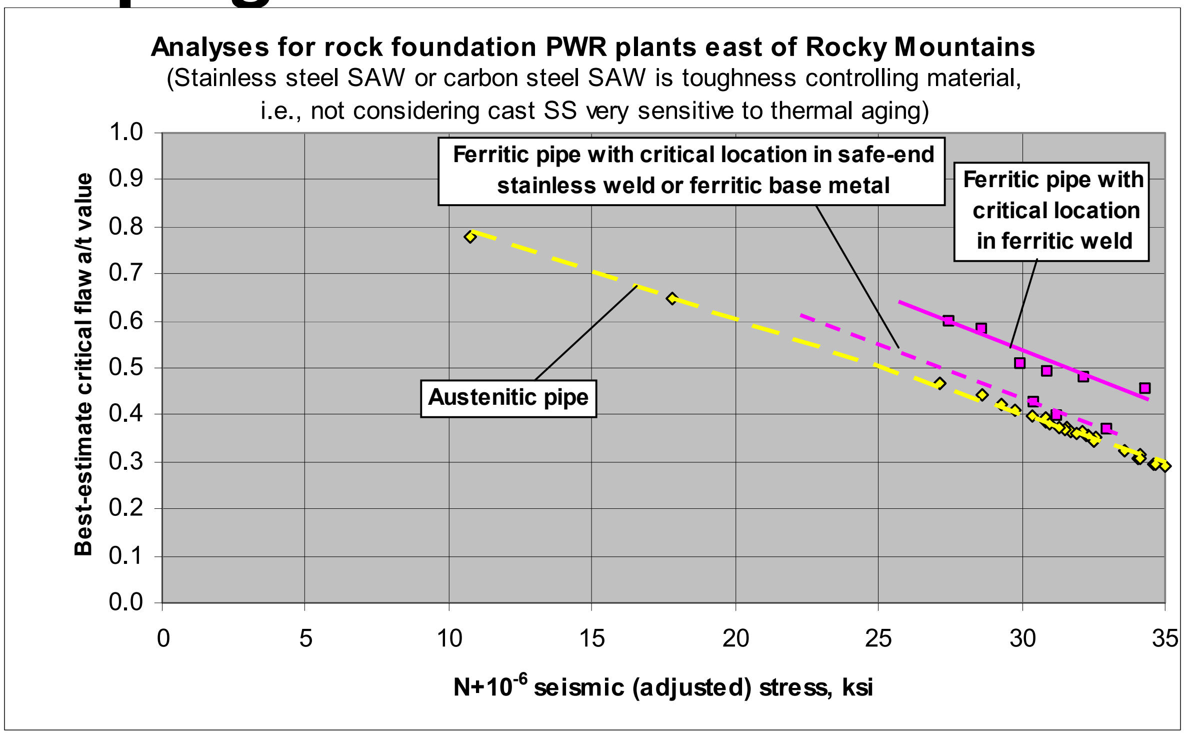
- **Objective**
  - Determine if seismic risk is acceptable for breaks greater than TBS
- **Scope and Approach**
  - Six supporting activities
    - Unflawed piping failure
    - **Flawed piping failure**
    - **Indirect piping failure**
    - Review of past earthquake experience
    - Review of past seismic PRAs
    - Review a mid-80s LLNL study of direct and indirect seismic piping rupture used to support GDC 4 revision
  - Use mix of deterministic and probabilistic approaches

NUREG-1903: Approach, cont.

- Analyzed direct piping failure under rare seismic events
  - Evaluated unflawed and flawed piping systems with diameters > TBS (e.g., hot leg, cold leg, and cross-over leg) using available design information
  - Used most-recent seismic-hazard curves for plants east of the Rocky Mountains
  - Determined stresses for 10^{-5} and 10^{-6} yr^{-1} seismic event by scaling plant specific SSE stresses
  - Apply scale factors to address conservatisms in the design process, material behavior, and extrapolation to rare seismic loading

- Analyzed indirect piping failure under rare seismic events
  - Analyzed large component support failures that may lead to piping failure
  - Assumed that support failure leads directly to piping failure
  - Updated results from prior LLNL study to reflect new hazard and ground motion information
  - Determined mean failure probability of component supports

Direct Piping Failure: Surface Flaw Results



- 26 PWRs analyzed
- Critical flaw depth (a/t) for long flaw ($\theta/\pi \approx 0.8$) under 10^{-6}/yr seismic event

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

Indirect Piping Failure: Case Studies

NUREG/CR-3663 Sample Results

Group A Plants **(Combustion Engineering)**	Confidence Limit (1)		
	10%	50%	90%
Calvert Cliffs	2.3×10^{-8}	6.1×10^{-7}	6.1×10^{-6}
Millstone 2	9.0×10^{-10}	6.6×10^{-8}	1.2×10^{-6}
Palisades	5.0×10^{-7}	6.4×10^{-6}	5.2×10^{-5}
St. Lucie 1	1.2×10^{-8}	3.8×10^{-7}	4.1×10^{-6}
St. Lucie 2	6.6×10^{-8}	1.4×10^{-6}	1.1×10^{-5}
Westinghouse Lowest Capacity Plant	2.3×10^{-7}	3.3×10^{-6}	2.3×10^{-5}

(1) Confidence limit of 90% implies a 90% confidence that annual probability is less than value indicated

- Generic seismic hazard curves used in evaluation
- Group A had highest failure probabilities for CE plants

NUREG-1903

- Only 2 plants evaluated
- Mean result for Calvert Cliffs: 1.7E-6/yr

NUREG-1903: Summary

– Unflawed piping: Failure frequency is much lower than 10^{-5}/yr

– Flawed piping

  – Critical flaws for long, circumferential flaws ($\theta/\pi = 0.8$) are generally large
    - 40% of wall thickness for 10^{-5}/yr seismic event
    - 30% of wall thickness for 10^{-6}/yr seismic event

  – Conditional probability of breaks larger than the TBS should be less than 10^{-5}/yr

– Indirect failures

  – Only two cases analyzed (one W and one CE plant)

  – Piping failure induced by major component support failure has a mean probability of approximately 10^{-6}/yr

Historic TBS Technical Basis Development: Summary

- Passive System LOCA frequencies developed for generic BWR and PWR plants through an expert elicitation process (NUREG-1829)
  - Accounted for panelist uncertainty and variability among responses
  - Used results as the starting point for selecting the transition break size
- Increased TBS to address additional factors and to promote regulatory stability
  - Considered other types of LOCAs
  - Accounted for plant piping design and operating experience
- Performed confirmatory study to determine if risk of LOCAs > TBS due to rare seismic was acceptable (NUREG-1903)
  - Risk due to unflawed and flawed direct piping failures expected to be acceptable for most, if not all, plants
  - Risk due to indirect piping failures acceptable for two cases evaluated
  - Seismic risks, however, are plant-specific, making it difficult to completely generalize results

Recent Confirmation of the Transition Break Size Technical Basis

David Rudland
Senior Technical Advisor for Materials
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulations

Confirmation Study

- Confirmation of the NUREG-1829 LOCA Frequencies
- Confirmation of the NUREG-1903 Results
- Determination of TBS impact

NUREG-1829 Confirmation

- Internal and External Elicitation
- Impact of Recent Operational Experience

Qualitative

- Probabilistic Fracture Mechanics Study
- International Operational Database Study

Quantitative

Details in “White Paper on Continued Applicability of NUREG-1829”
ML24205A015

Internal and External Elicitation

- **Motivation**
  – NUREG-1829 based on formal expert elicitation
    - Pilot elicitation performed initially
    - External elicitation formulated based on lessons-learned from internal pilot
  – Mimic process to evaluate the completeness and continuing viability of the NUREG-1829 and NUREG-1903 results
- **Objectives**
  – Identify possible scenarios either not considered or under-estimated in NUREGs-1829 and 1903
  – Assess likelihood and/or technical or rulemaking gaps associated with each scenario

Approach

- **Select appropriate internal and external panelists**
  - **Internal**: 13 senior staff with collective expertise in all relevant technical areas
  - **External**: Two NUREG-1829 panelists with complementary expertise pertaining to passive system reliability
- **Formulate initial set of questions and topics**
  - Focus on knowledge gained and operating experience since the mid-2000s
  - Consider direct, indirect, and potential common-cause failure scenarios
  - Identify important causal factors
- **Hold a kick-off meeting**
  - Present objectives, background and motivation of the effort
  - Discuss and clarify the elicitation topics and questions
  - Identify initial considerations
- **Develop initial independent responses**
- **Conduct follow-on meetings**
  - Collectively discuss the individual responses
  - Determine the path forward for dispositioning any open issues

Summary of Internal Elicitation Responses

- **Scenarios – 21 identified**
  - **Addressed in applicability studies**: RPV embrittlement, SCC in main loop piping, increased seismic risk since NUREG-1903, evolution of ISI and relief requests
  - **Addressed within 10 CFR 50.46a rulemaking**: Rulemaking motivation, effects of future plant changes, PRA representativeness, indirect failures from small pipe rupture, TBS margin, treatment of LBB piping, degraded supports and snubbers, NUREG-1829 uncertainties, BWR applicability, maintaining mitigative capabilities, definition of a pipe
  - **Addressed within current regulations:** pilot-operated relief valve failure, common-cause maintenance errors, RPV through-wall cracking, water chemistry excursions, impact on plant security, degraded grid stability

Addressed later

Sample of Internal Elicitation Topics

- **Treatment of LBB piping**
  - **Issue**: Consideration of special treatment for plants with LBB approval
  - **Disposition**:
    - No explicit special treatment in rule although NUREG-1829 results reflect LBB margins
    - May be able to leverage approved LBB analysis as part of plant-specific applicability demonstration
- **Potential for degraded grid stability**
  - **Issue:** Higher risk could result if LOOP is not evaluated within LOCA analysis
  - **Disposition:**
    - LOOP event frequencies, while relatively sparse, don't indicate an increasing trend
    - PRA still needs to consider risk associated with such events and continually update data
    - Many plants currently employ load monitoring software to predict offsite power unavailability

Summary of External Elicitation Responses

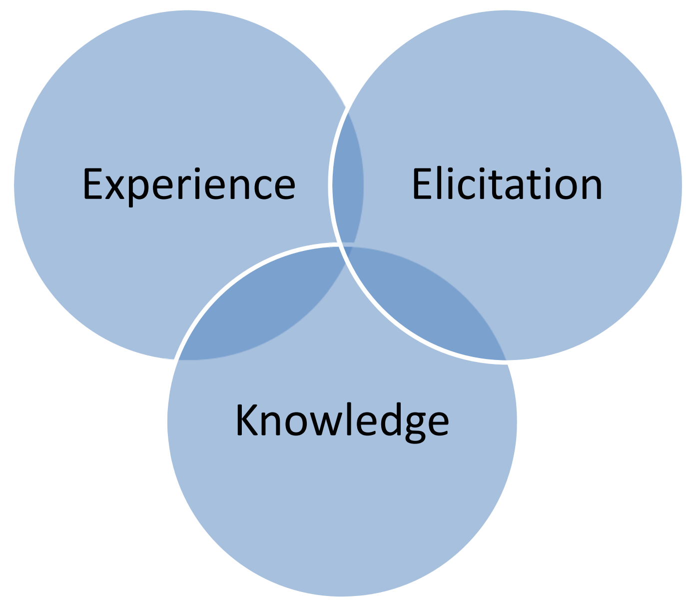
- **Continuing Applicability of NUREG-1829 LOCA frequencies**
  - Frequencies for breaks < TBS are representative (one panelist) or conservative (one panelist)
  - Frequencies for breaks > TBS are conservative
  - Opinions are based on successful mitigation practices and increased knowledge pertaining to the structural integrity of large piping systems
- **Historical TBS remains viable**
- **Possible scenarios leading to breaks > TBS**
  - One panelist: no such credible scenarios envisioned
  - One panelist
    - Should continue to explore thermal aging, cold work, and residual stress effects
    - SCC leading to long, shallow surface flaws especially in either CASS or adjoining weld or cold-worked component are most credible
    - Likelihood of such scenarios is strongly plant-specific

Disposition of External Elicitation Responses

- **Continued research and monitoring**
  - Harvest representatively aged austenitic weld and CASS materials to validate current models, which are based largely on accelerated aging laboratory testing
  - Extend aging studies to represent properties at the end of subsequent license renewal period and beyond
  - Continue to monitor both U.S. and international operating experience relevant to potential for SCC cracking, especially in large diameter piping systems
- **Demonstrate plant-specific applicability of the TBS**
  - Proposed 10 CFR 50.46a rule requires that an entity demonstrate that plant-specific effects do not invalidate the applicability of the TBS for their plant before implementing the rule
  - Additional guidance proposed in DG-1428 providing several methods for demonstrating plant-specific applicability

Impact of Recent Operational Experience (OE)

- Thermal Embrittlement of Piping
- Stress Corrosion Cracking (SCC) of Stainless Steel in PWRs
- Carbon Macrosegregation
- Quasi-laminar Indications
- Small Surface Break Flaws
- Reactor Pressure Vessel Embrittlement
- Inspection Frequency changes
- Secondary-Side Piping Failure



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

Impact of Recent OE - Piping

Thermal Embrittlement of Piping	SCC of Stainless Steel in PWRs
• Issue – Decrease in fracture toughness of cast authentic stainless steel and austenitic stainless welds – Can this decrease impact failure frequencies • Staff Action – Considered experimental testing, development of aging management, lack of active degradation, ongoing inspections – No safety concern	• Issue – Many cracks identified in Safety injection and residual heat removal system in stainless welds in French Fleet – unexpected SCC – Could this occur in US and may it impact the failure frequencies • Staff Action – Conducted Risk-informed analysis (LIC-504) – Reviewed industry actions – Determined reasonable assurance of integrity

Impact of Recent OE - Piping

Inspection Frequency Changes	**Impact on TBS**
• Issue – Ongoing efforts at ASME to optimize inspection frequencies resulting in less inspections for piping and components – Risk-informed ISI is in place, but some categories may change inspection frequency • Staff Action – Continuing inspection of these welds is essential to the basis supporting the transition break size.	• Conclusion – Some issues were analyzed through our LIC-504 risk-informed process – Some issues were analyzed through research or licensing actions – No impact on the TBS • Staff Action – However, the staff recommends that for those reactor coolant pressure boundary piping whose diameter is greater than the TBS, a 10 percent sample of the welds > TBS is needed each interval – Can leverage existing ISI programs

Minority View on Inspection Requirements

- Piping failure resulting in LOCA > TBS is highly unlikely, but possible
  - Most prominent concern is an SCC-like mechanism that causes a long-surface flaw with slow through-wall growth coupled with toughness decrease due to thermal aging
    - Increases likelihood of break before leak
    - Issue identified in internal and external elicitations
    - Characteristic of flaws leading to ruptures in PFM analyses
    - Flaws which such characteristics have been occasionally discovered (e.g.., Duane Arnold and Penly 1)
  - Such a scenario is plant-specific, not generic
- Performance monitoring, through inspection, of piping with inner diameter greater than TBS provides assurance that failure likelihood remains extremely low
  - Rulemaking utilizes classical approach of defining a risk-informed inspection sample and then performing repeat inspections each ISI interval
  - Minority view recommends choosing a new risk-informed inspection sample every ISI interval to ensure that a greater population of such welds is inspected at least one time during operation

Impact of Recent OE - Vessels

Carbon Macrosegregation (CMAC)

- Issue
  - In early 2015 regions of (CMAC) were discovered in European Pressurized Reactor pressure vessel heads manufactured for a plant in Flamanville, Manche, France
  - Higher strength, lower toughness, may be more susceptible to embrittlement
- Staff Action
  - Conducted risk-informed analysis, considered EPRI and ASN analyses
  - Concluded that the safety significance of CMAC to the U.S. fleet is negligible

Quasi-laminar Indications (QLI)

- Issue
  - In July of 2012, ultrasonic inspections of RPV ring forgings at two nuclear power plants in Belgium revealed thousands of sub-surface, nearly-axial indications
  - Do the many flaws impact RPV integrity?
- Staff Action
  - Reviewed Electrabel PFM evaluation
  - Conducted independent risk-informed evaluation
  - Concluded the potential existence of QLI is not expected to affect structural integrity of U.S. RPVs.

Impact of Recent OE - Vessels

Small Surface Break Flaws (SSBF)

- Issue
  - 2016 ORNL analyses suggested that SSBF can produce a greater through-wall crack frequency than a 1/4T flaw
  - What are the impacts on P-T limits and PTS?
- Staff Action
  - PFM analyses conducted and determined that even though there is an increase in conditional probability of failure, the impact on through-wall crack frequency is minimal.
  - Realistic cooldown transient frequencies and their occurrence frequency was considered.

RPV Embrittlement

- Issue
  - The existing RG 1.99 (and 10 CFR 50.61) embrittlement trend curve (ETC) model may underpredict of RPV embrittlement under the high fluences
  - Licensees are allowed to defer surveillance capsule testing that is intended to confirm embrittlement predictions from the ETC model
- Staff Action
  - Staff developed a risk-informed analysis that suggested the staff’s confidence in the integrity of the RPV for certain plants may be impacted
  - Staff proposed a change to the rule in SECY-22-0019
  - Staff waiting on Commission decision

Impact of Recent OE - Vessels

Inspection Frequency Changes

- Issue
  - Through 10 CFR 50.55a(z) many licensees have been granted approval to modify their inspection intervals for RPV, steam generator and pressurizer welds
  - Cumulative effect of these relaxation may impact the TBS
- Staff Action
  - Ensure reasonable assurance of safety
  - Verify appropriate performance monitor occurs within these components

Impact on TBS

- Conclusion
  - Some issues were analyzed through our LIC-504 risk-informed process
  - Some issues were analyzed through research or licensing actions
  - Cumulative effects were considered
- Staff Action
  - Impact of the embrittlement concerns on the TBS be revisited following Commission action on the rulemaking plan.

Impact of Recent OE

Secondary-Side Piping Failure

- Issue
  - Secondary side failure (<TBS), impacts larger piping (>TBS) and increases LOCA frequencies
- Staff Conclusion
  - GDC-4 and SRP 3.6.2 provide reasonable assurance safety is maintained
  - Piping is very flaw tolerant – probability of enough damage to rupture large piping is small
  - Guidance is needed to cover any possible impacts of secondary side failure causing indirect failure of piping greater than the TBS

Probabilistic Fracture Mechanics Study

- Base Cases from NUREG-1829 (2004) re-examined using improved the state of knowledge and PFM modeling capabilities
- LOCA frequencies recalculated for 4 piping systems relevant to transition break size
- Calculations performed using NRC's extremely Low Probability of Rupture (xLPR) PFM code

Case #	Weld location	Pipe Size	Plant Type	Degradation mechanisms
PWR-1	Reactor Vessel Outlet Nozzle	30-inch	PWR	PWSCC
PWR-2	Surge Line	10-inch	PWR	PWSCC
BWR-1a	Recirculation	28-inch	BWR	IGSCC
BWR-1b	Recirculation	12-inch	BWR	IGSCC

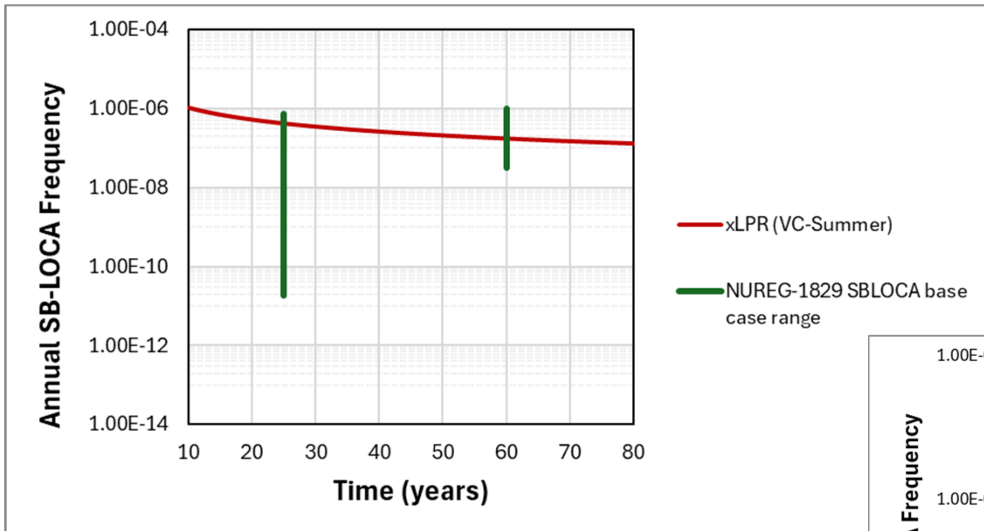
xLPR Approach

- General settings
  - 80 year reactor operation time
  - Sufficient sample size to detect a 10^{-6} probability event
  - Leak rate detection enabled
- Quantities of interest
  - Probability of leakage
  - Probability of rupture
  - Leak rate
- Post-processing converts these quantities to annual component-level LOCA frequencies

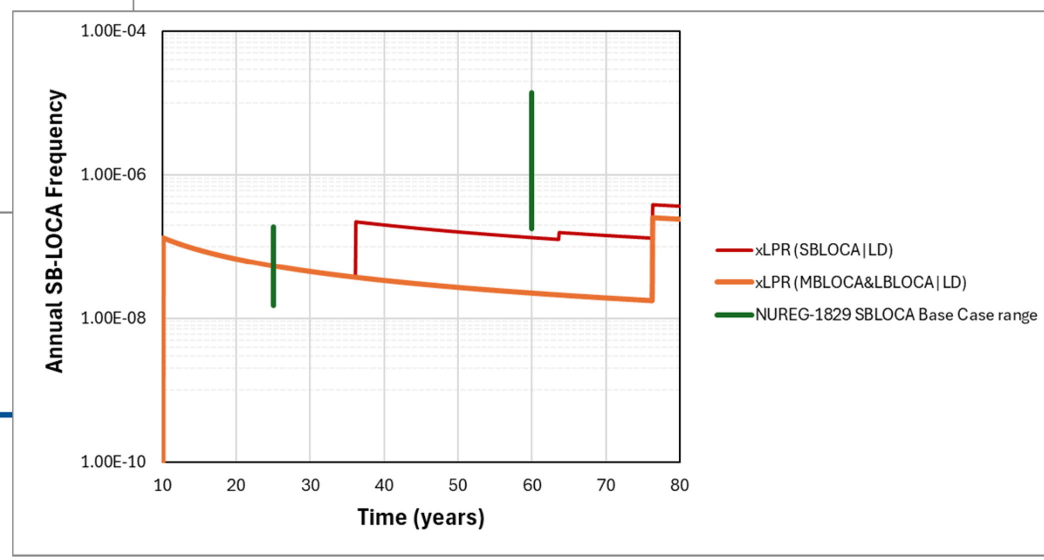
PWR	BWR
PWSCC crack growth	IGSCC crack growth • Generic crack growth model parameters set to match IGSCC model in 2023 ASME Section XI Article Y2310
Weld residual stress • PWR-1: set WRS to mirror profile from VC Summer leak event for conservatism • PWR-2: Generic representative WRS profile	Weld residual stress • Generated using finite element analysis from EPRI data

The PWR probabilistic fracture mechanics analyses were within the range of the NUREG-1829 base case results.

PWR-1: Vessel Outlet Nozzle

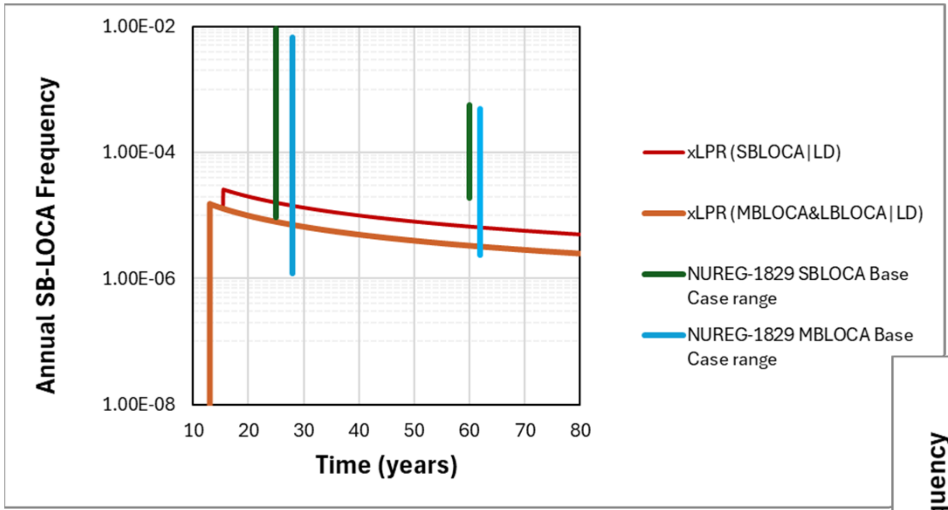


PWR-2: Pressurizer Surge Line

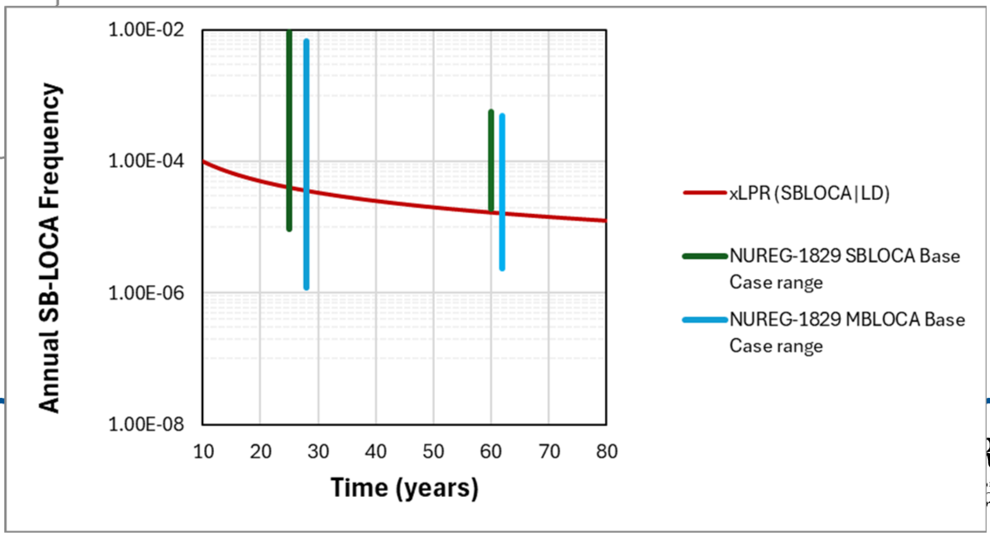


The BWR probabilistic fracture mechanics analyses were also within the range of the NUREG-1829 base case results.

BWR-1: Recirculation System (28-inch)



BWR-1: Recirculation System (12-inch)



International Operational Database Study

- **Motivation**
  - Participants in NUREG-1829 elicitation based their estimates in part from knowledge from operating experience
  - The basis has changed 20 years later
    - More OE knowledge in the later post 25-year lifetime of reactors
    - New mitigation technologies
- **Objectives**
  - Re-evaluate NUREG-1829 LOCA Frequency estimates with knowledge from post-2004 operating experience

Scope of OpE Review	Plant Type	NUREG-1829		Expert Elicitation 2024		ΔEFPY (2024 vs 2004)
		2004		2024		
		ROY	EFPY	ROY	EFPY	
Domestic Plant	BWR	987.8	839.6	1345.9	1144.0	304.4
	PWR	1615.4	1373.1	2735.4	2325.1	952.0

Analysis Procedure

(1) Calculate piping failure precursor frequencies from OE[1]
   (1) Up to 2004 (l_1)
   (2) Up to 2024 (l_2)
(2) Calculate conditional probability of failure distribution
   (1) Extract CFP from NUREG-1829 LOCA frequency estimates using 2004 precursor failure frequency l_1
(3) Calculate updated LOCA frequencies
   (1) Find new LOCA Frequency uncertainty distribution estimates using found CFP and 2024 precursor failure frequency l_2

$$\rho = l \times CFP$$

ρ= LOCA Frequency Distribution

l= Precursor Failure Frequency – (# failures/(Component x Year))

CFP= Conditional Probability of Failure

1. OE database developed under the Nuclear Energy Agency's Component Operational Experience, Degradation and Ageing Programme (CODAP) provided the source for the OE data.

The operating experience analysis results indicate at least an order of magnitude less than the NUREG-1829 results.

LOCA Category	Effective Break Size (inch)	Plant-Level LOCA Frequency (1/Year) – Statistical Mean Values			
		BWR - Piping		PWR - Piping	
		NUREG-1829	2024 Update	NUREG-1829	2024 Update
4	≥ 7	5.9E-06	2.4E-08	7.6E-07	6.0E-08
5	≥ 18	1.0E-06	4.3E-09	1.3E-07	2.6E-08
6	≥ 41	--	--	1.2E-08	4.0E-10

Improved mitigation technologies such as weld overlays attributed with the reduction

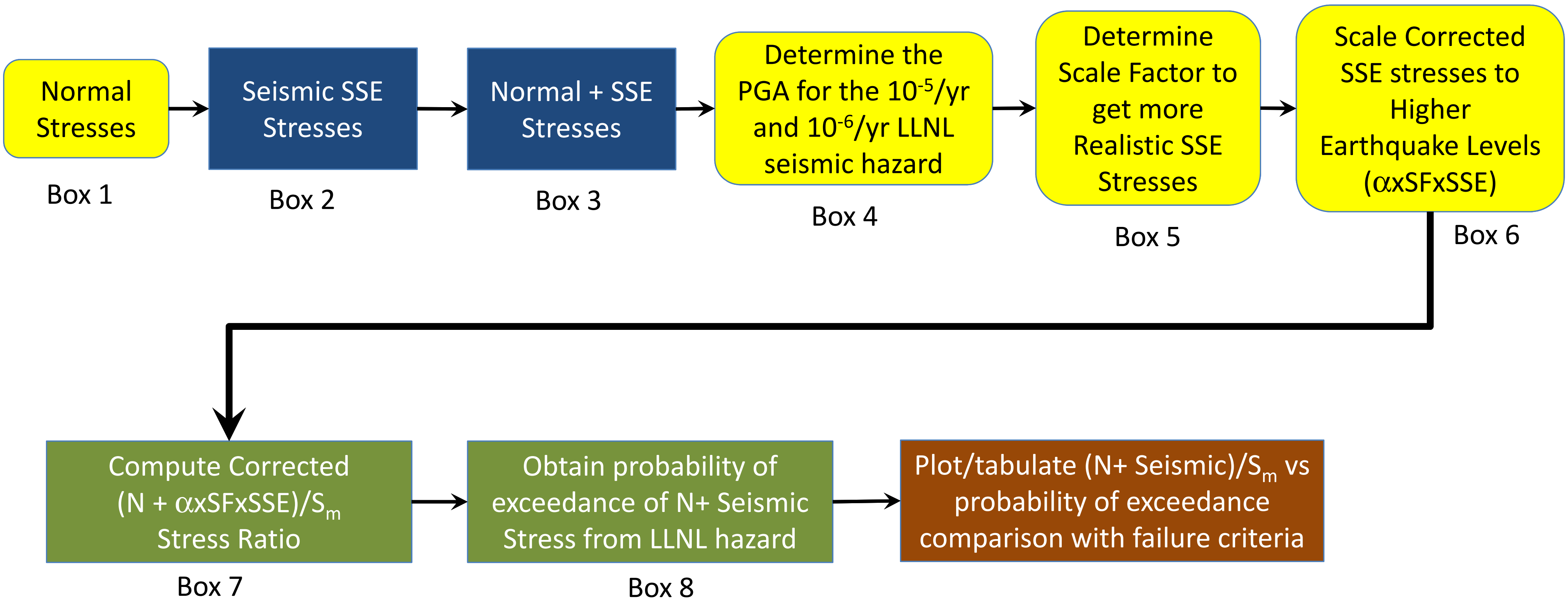
NUREG-1903 Confirmation

- NUREG-1903(ML080880140) addressed potential seismic effects on TBS.
- Evaluated three cases: unflawed and flawed piping failure and indirect piping failure by other components and component supports.
- Used LLNL seismic hazard curves for the assessment.
- For direct unflawed piping, failure probabilities were significantly low compared to the 1E-05 per year frequency used as a basis to establish the TBS.

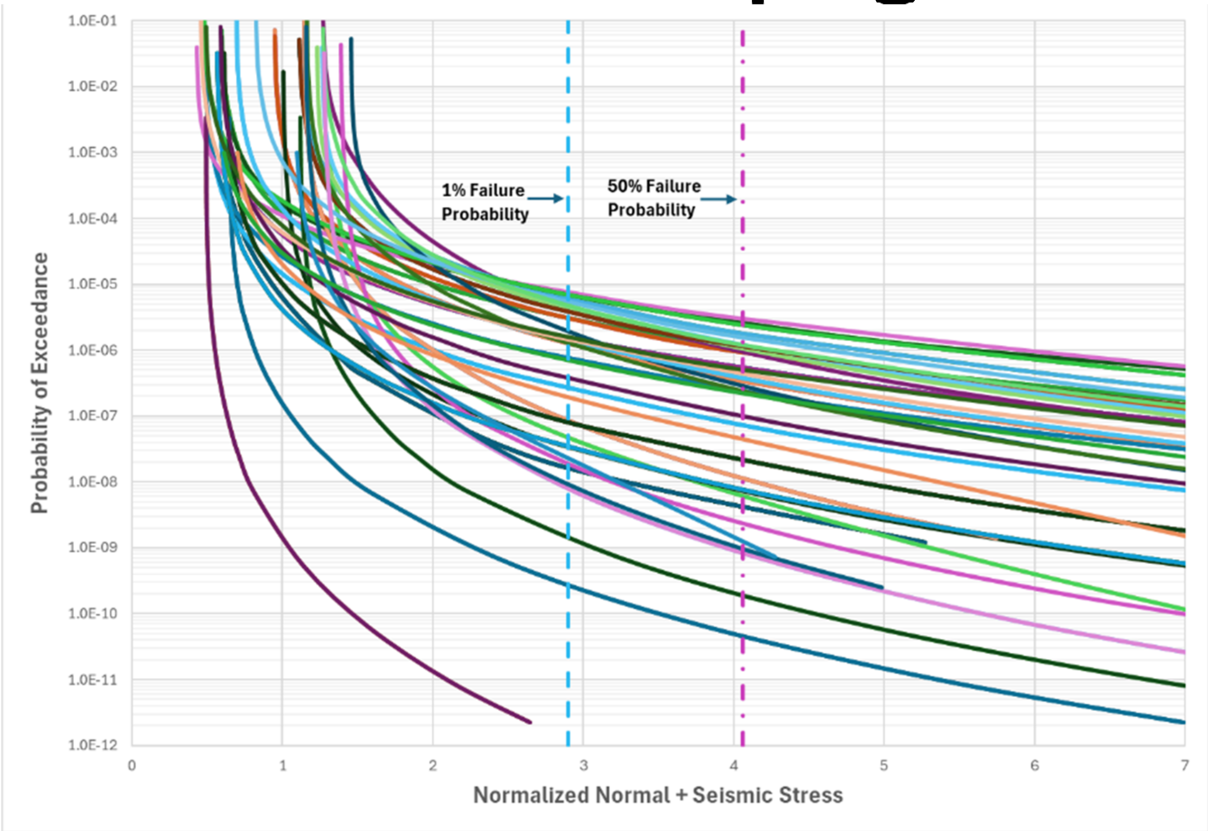
NUREG-1903 Confirmation

- For direct flawed piping, the critical flaws associated with the stresses induced by the 1E-05 and 1E-06 probability of exceedance events were generally large, and the probabilities of pipe breaks larger than the TBS were determined to be less than the TBS frequency criterion.
- For indirect piping failure, the mean probability of failure of the lowest capacity component support was less than 1E-5 for the CE and Westinghouse plants.
- All central and eastern US NPPs recently re-evaluated their seismic hazards (NUREG KM-0017).
- The original assessment results have been updated by using the latest site hazard information for each site (ML24323A205).

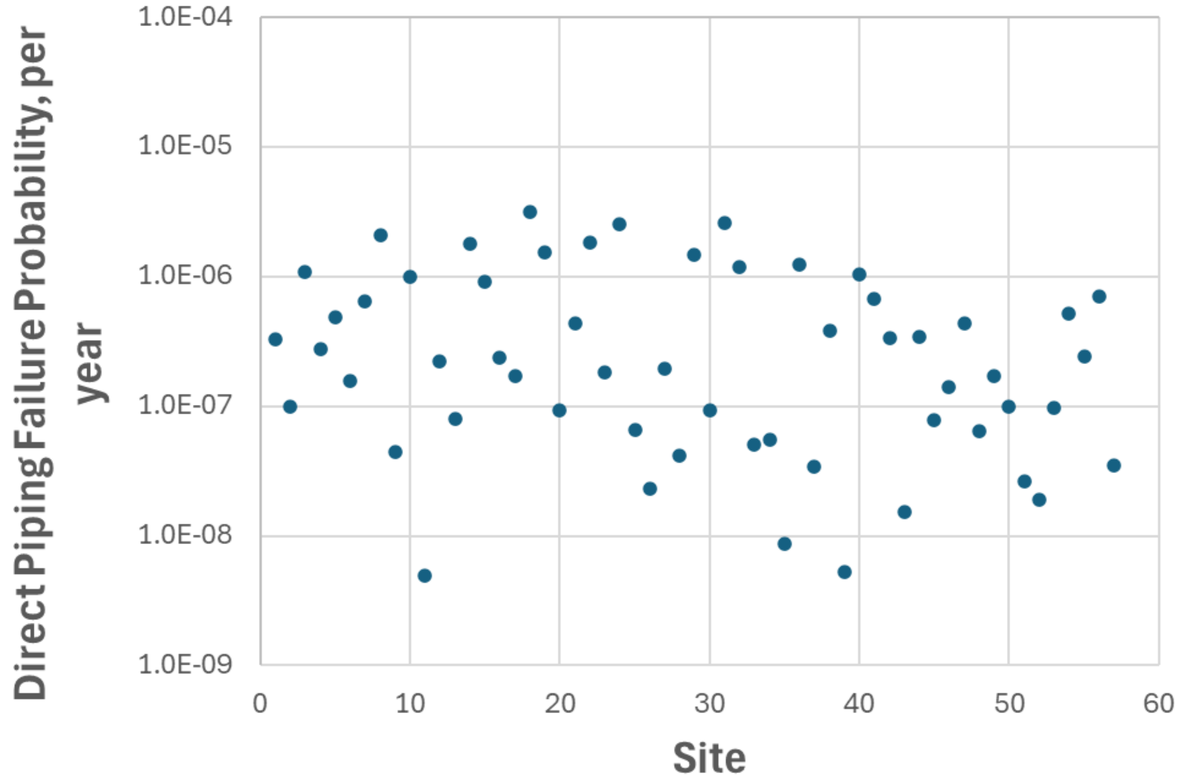
Direct Unflawed Piping Failure



Direct Unflawed Piping Failure



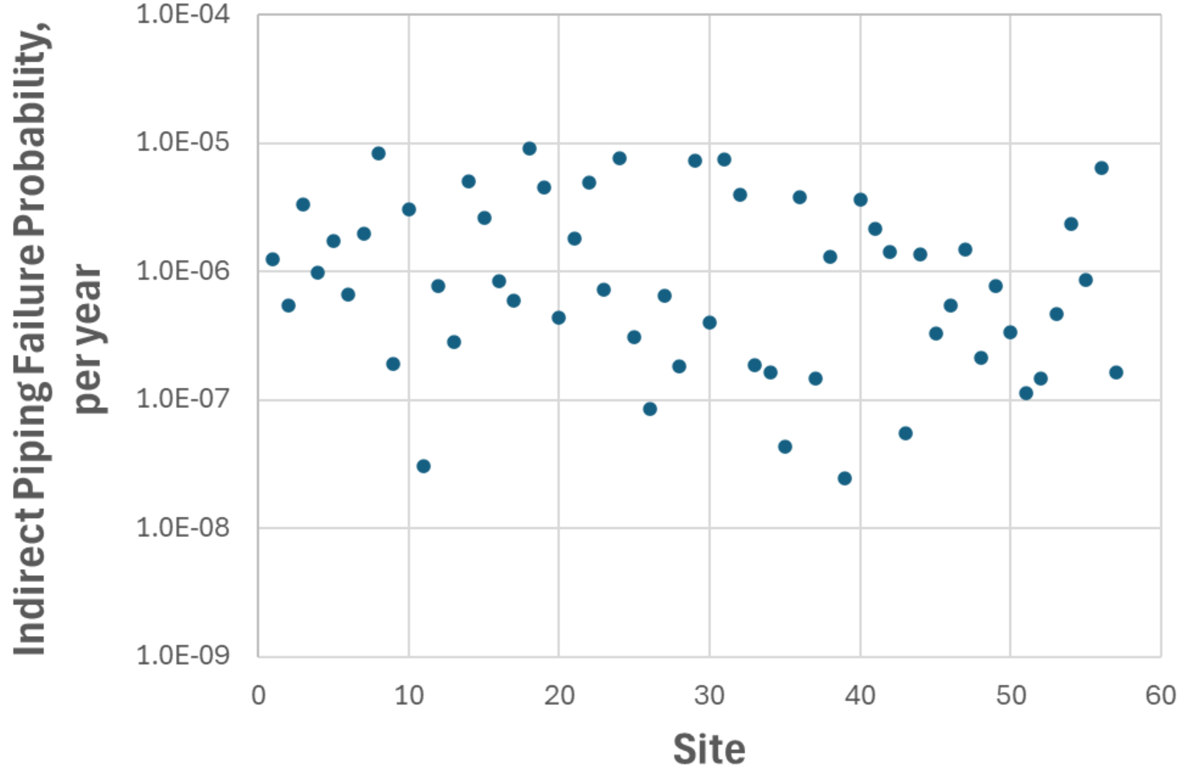
Direct Unflawed Piping Failure Probability



Indirect Piping Failure Probability

- Indirect failures are pipe ruptures caused by failures of major components (e.g., reactor pressure vessel, steam generators, and reactor coolant pumps (RCPs)) or component supports as a result of an earthquake.
- NUREG-1903 indirect piping failure fragility curve
- The results show that mean probabilities of indirect piping failure are all below the TBS frequency criterion of 1E-05.

Indirect Piping Failure Probability



Summary from TBS Confirmation

- LOCA frequencies and TBS are applicable if plant specific applicability is demonstrated.
  - NUREG-1829 and NUREG-1903
  - New designs can develop plant specific TBS
- Inspection of the piping welds with diameters greater than the TBS are needed to ensure LOCA frequencies remain applicable
  - Plants can leverage existing ISI program as needed

Draft Proposed Rule Language for 10 CFR 50.46a

Joseph Messina
Reactor Systems Engineer
Nuclear Methods and Fuel Analysis

Draft Proposed 50.46a Rule Structure

a) Definitions
b) Applicability and scope
c) Application
d) Programmatic requirements
e) ECCS performance
f) Fuel performance criteria
g) Use of NRC-approved fuel in reactor
h) Changes to facility, Technical Specifications, or procedures
i) Authority to impose restrictions on operation
j) Reporting
k) Significant change or error in the ECCS evaluation model

50.46a(a): Definitions

Highlighted definitions:

- Changes enabled by this section: means changes to the facility, Technical Specifications, and procedures that satisfy the alternative ECCS analysis requirements under this section but do not satisfy the ECCS requirements under § 50.46.
- Entity: means an applicant for or a holder of a construction permit, operating license, combined license, standard design approval, or manufacturing license, or an applicant for a standard design certification rule (including such applicant after NRC issuance of a final standard design certification rule).
- Loss-of-coolant accident: means the hypothetical accidents that would result from the loss of reactor coolant, at a rate in excess of the capability of the reactor coolant makeup system, from **breaks in the reactor coolant pressure boundary** up to and including a break equivalent in size to the double-ended rupture of the largest pipe in the reactor coolant system. LOCAs involving breaks at or below the transition break size (TBS) are design basis accidents. **LOCAs involving breaks larger than the TBS are beyond design basis accidents**.
- Transition break size: for reactors licensed under this part before December 31, 2015, is a break area equal to the largest cross sectional flow area of the reactor coolant pressure boundary piping excluding the hot leg, cold leg, or crossover leg piping for a pressurized water reactor, or the largest cross sectional flow area of either the feedwater line or residual heat removal line inside containment for a boiling water reactor. For reactors that are or will be licensed under this part after December 31, 2015, and for light-water reactors (LWRs) that are or will be licensed under part 52 of this chapter, the TBS will be determined on a plant-specific basis.

50.46a(b) Applicability

- OLs issued prior to Dec. 31, 2015
- Entities whose reactor design is demonstrated under 50.46a(c)(2) to be similar to designs of reactors licensed under part 50 before Dec. 31, 2015

- NUREG-1829 and the TBS based on plants licensed before Dec. 31, 2015. LWRs licensed after may have different piping materials, configurations, and operational and service conditions, among other factors, that may impact the piping break frequencies and thus the TBS
- **Paragraph 50.46a(c)(2)** states that for plants licensed after Dec. 31, 2015, an analysis should be submitted that demonstrates why the proposed reactor design is similar to the designs of reactors licensed under part 50 before Dec. 31, 2015, such that the provisions of this 50.46a may properly apply
  - The analysis must include a recommendation for an appropriate TBS and a justification that the recommended TBS is consistent with the technical basis of this section

50.46(b): Applicability (cont'd): Inspections

Paragraph (b)(3): A licensee must inspect, under § 50.55a(g), for those reactor coolant pressure boundary piping whose inner diameter is greater than the TBS, a sampling of at least 10% of the similar metal piping circumferential welds in a PWR and the circumferential welds in a BWR that are classified as Category A welds before implementation of this section and in every subsequent in-service inspection interval (as defined in § 50.55a(y)). The sampling must include those circumferential welds with the highest failure potential. Credit may be taken for welds inspected as part of established inspection programs (e.g., risk-informed inservice inspection programs). The effect on the TBS of any degradation identified during these inspections must be evaluated.

- This requirement is new from the 2010 draft final rule, added in response to public comments, to reduce the burden of demonstrating plant-specific applicability of the TBS, while providing assurance of safety. Allows licensee to leverage ongoing inspection programs to reduce burden.
- Verifies that analyses that predict component failure remain accurate through the time the component is analyzed, and provides a method to identify novel degradation that may impact the analysis and the structural integrity of the component

50.46a(c): Application

Entities electing to adopt 50.46a will need to provide:

- Evaluation of the applicability of the TBS to the facility
- Weld inspection report
- Description of the risk informed evaluation process for changes made under this rule to meet the risk acceptance criteria in paragraph (h)
- Description of the approach, methods, and decision-making process to be used to evaluate the continued applicability of the TBS
- Description of the non-safety systems credited for LOCAs > TBS and they must be placed in Tech Specs
- Evaluation of leak detection program
- For reactors licensed after 12/31/15, an analysis demonstrating that the reactor design is similar to those licensed before 12/31/15 and the appropriate TBS

50.46a(d): Programmatic Requirements

1) ECCS models and analysis methods are maintained per requirements in (e)(1), (2), and (3)
2) Leak detection systems must be available and used to identify, monitor, and quantify leakage
3) Changes made must be evaluated in a risk-informed evaluation
4) ~~Risk assessments must be maintained and upgraded risk assessments at least once every 5 years~~ – Removed from draft proposed rule during concurrence
5) The effect of all planned facility changes must be evaluated and any changes that would invalidate the evaluation demonstrating the applicability of the TBS cannot be implemented
6) During operation, licensees must perform the (b)(3) weld inspections every subsequent inservice inspection interval (as defined in § 50.55a(y)) on the same samples inspected to satisfy paragraph (b)(3) of this section and evaluate any additional degradation

50.46a(e): ECCS Performance

50.46a(e)(1) establishes two principle ECCS acceptance criteria:

- The ECCS must provide sufficient coolant so that the fuel remains in a coolable geometry during and following the LOCA heatup and quench.
- The ECCS must provide sufficient coolant so that decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the fuel.

❑ Maintaining coolability and removing decay heat has been fundamental to LOCA analysis since the origin of 50.46

50.46a(e)(2): ECCS Performance at or below the TBS

- A number of LOCAs must be analyzed such that there is assurance that the most severe LOCAs at or below the TBS are analyzed
- Uncertainty must be accounted for such that there is a high probability that the ECCS and fuel system acceptance criteria are met
- Changes in fuel geometry must be addressed

❑ Analysis requirements for LOCAs at or below the TBS are essentially unchanged and (still require high probability modeling)

50.46a(e)(3): ECCS Performance above the TBS

- A number of LOCAs must be analyzed such that there is assurance that the most severe LOCAs larger than the TBS up to the double-ended guillotine rupture of the largest pipe in the RCS are analyzed
- There must be assurance to at least a **best-estimate** that the ECCS and fuel system acceptance criteria are met
- Changes in fuel geometry must be addressed
- Calculations may take credit for availability of offsite power
- Do not require assumption of a single failure
- Non-safety-related equipment may be credited if supported by plant-specific data or analysis, and provided that onsite power can be readily provided through simple manual actions to equipment that is credited in the analysis.

❑ **LOCAs > TBS would be beyond-design-basis accidents and be analyzed with best-estimate (best-estimate would refer to nominal and unbiased analyses) modeling, as other beyond-design-basis accidents are (e.g., ATWS and SBO)**

50.46a(f): Fuel Performance Criteria

Fuel system designs must have NRC-approved limits that:

i. Address cladding degradation phenomena
ii. Maintain fuel coolability
iii. Avoid explosive concentration of combustible gas
iv. Demonstrate that, after any calculated successful initial operation of the ECCS, the ECCS must provide sufficient coolant to remove decay heat and prevent further cladding failure for the extended period of time required by the long-lived radioactivity remaining in the fuel.

- Thermal effects of crud and oxide layers must be accounted for

- ❑ Fuel-technology neutral requirements
- ❑ Specific criteria for traditional Zr-UO2 fuel is provided in DG-1263, which would state how the SECY-16-0033 embrittlement research findings should be addressed

Fuel Coolability

While the wording is not significantly different in regards to coolability than 50.46, the NRC staff added a discussion in the FRN Preamble (formerly known as Statements of Consideration) that adds clarification on the interpretation of coolability

- The NRC can envision that some amount of dispersed fuel can remain coolable and safe during a LOCA, therefore the NRC finds that if it can be shown to be safe, then it may be acceptable for LOCAs greater than the TBS
  - Departure from precedent
- The NRC outlined 2 scenarios that remain undesirable though:
  - Widespread brittle failure
  - Fuel or cladding melt
- DG-1434 provides guidance for analyzing the consequences of fuel dispersal

50.46a(g): Use of NRC-Approved Fuel

1) *Fuel load*. A licensee that is approved to use this section may not load fuel into a reactor unless the resulting core design satisfies the ECCS performance requirements of paragraph (e) of this section and the fuel system acceptance criteria and modeling requirements in paragraph (f) of this section, or otherwise complies with Technical Specifications governing lead test assemblies in its license.
2) *Operation*. If a licensee that is approved to use this section determines that fuel in the reactor no longer complies with the ECCS performance requirements of paragraph (e) and the fuel system acceptance criteria and modeling requirements in paragraph (f) of this section, then the licensee must take immediate action to come into compliance with paragraph (e) or (f) of this section, as applicable.

- ❑ Clarifies requirement on use of NRC approved fuel designs for which specific ECCS performance requirements have been established.
- ❑ Recognizes importance of LTAs for collecting irradiated data to approve new fuel designs.

50.46a(h): Changes Enabled by 50.46a

- Paragraph (h)(1): Changes enabled by this section (50.46a) without prior NRC approval
- Paragraph (h)(2): Changes enabled by this section not permitted under (h)(1)
- Paragraph (h)(3): Criteria that all changes enabled by this section under this section must meet
- Paragraph (h)(4): PRA requirements
- Paragraph (h)(5): Non-PRA requirements

50.46(h)(1): Changes without Prior NRC Approval

Changes enabled by 50.46a are allowed without prior NRC approval if:

i. Change is permitted under 50.59
ii. The NRC-approved risk-informed evaluation process demonstrates that any increases in estimated risk are minimal and the requirements in (h)(3) are met
iii. There is no significant increase in LOCA frequencies and the evaluation demonstrating or establishing the TBS is not invalidated

50.46a(h)(2): Changes Not Permitted Under (h)(1)

For changes enabled by this section not permitted under (h)(1), entities need to submit:

- Information required under 50.90
- Demonstration from the risk-informed evaluation process that the total increases in CDF and LERF are *very small** and the overall risk remains *small*
- Demonstration that the requirements in (h)(3) are met
- Risk-informed evaluation of the cumulative effect on risk on the plant change and all previous changes made under this section
- Demonstration that the ECCS performance criteria are met
- Demonstration that is no significant increase in LOCA frequencies and the evaluation demonstrating or establishing the TBS is not invalidated

*In SRM-SECY-07-0082, the Commission directed that the staff, in the 10 CFR 50.46a draft final rule, should restrict changes to a plant to very small risk increases. Very small risk increase corresponds to an increase in CDF of 1E-6 per reactor year and an increase in LERF of 1E-7 per reactor year.

50.46a(h)(3): Requirements for All Changes Enabled Under 50.46a

All Changes made under this section must meet the following criteria

- Adequate defense-in-depth is maintained
- Adequate safety margins are retained to account for uncertainties
- Adequate performance-measurement programs are implemented to ensure that the risk-informed evaluation continues to reflect actual plant design

50.46a(h)(4): PRA Requirements

PRAs used for the risk-informed evaluation must:

i. Address initiating events, from sources both internal and external to the plant and for all modes of operation, that would affect the regulatory decision in a substantial manner;

ii. Reasonably represent the current configuration and operating practices at the plant;

iii. Have sufficient technical acceptability (including consideration of uncertainty) and level of detail to provide confidence that the total risk estimates and the change in total risk estimates adequately reflect the plant and the effect of the proposed change on risk; and

iv. Be determined, through peer review, to meet industry standards for PRA quality that have been endorsed or otherwise found acceptable by the NRC.

- Expect that low power and shutdown conditions will be addressed with non-PRA methods

50.46a(h)(5): Non-PRA Requirements

Whenever risk assessment methods other than PRAs are used to develop quantitative or qualitative estimates of changes to risk in the risk-informed evaluation, an integrated and systematic process must be used. All aspects of the analyses must reasonably reflect the current plant configuration and operating practices and applicable plant and industry operating experience.

50.46a(i): Authority to impose restrictions on operation

The Director of the Office of Nuclear Reactor Regulation may impose restrictions on reactor operation if the NRC finds that the submitted evaluations of ECCS cooling performance are not consistent with the requirements of this section.

- Maintains the authority of the director of NRR to impose restrictions on operation if there are problems found in a licensee's ECCS evaluation
- This authority has existed since the origin of 50.46 and in the draft final 50.46a and 50.46c rules

50.46a(j): Reporting

(j)(1) and (j)(2): ECCS Reporting and corrective actions

- Eliminates reporting requirements for changes or errors that do not result in an inability to assure compliance with § 50.46a until an SDA or a DC is referenced in an application for a CP, OL, COL, or ML.
  - Parallels what is proposed in the Part 50/52 Alignment rulemaking (ACRS letter: ML22069A269)
- Otherwise, it simply clarifies existing reporting requirements

(j)(3): Risk Assessment reporting

- ~~As part of the risk assessment maintenance and updating required under 50.46a(d)(4), if the re-evaluation results in exceeding the acceptance criteria, must report and explain the changes in PRA modeling, plant, designs, or plant operation that led to the increase(s) in risk no more than 60 days after completing the PRA re-evaluation~~
- Removed from draft proposed rule during concurrence

50.46a(j): Reporting (Cont'd)

Minimal changes reporting

- Must report the changes made under (h)(1) involving minimal changes in risk and a brief summary of the basis for the changes not invalidating the plant's TBS every 24 months

Welding inspection reporting

- Must submit the weld inspection report within 120 days after completing the outage with details of the results of the inspections and the evaluation of the effects on the TBS of any additional degradation since the previous evaluation.
- Can be combined with the summary report required under 50.55a(b)(2)(xxxii)

50.46a(k): Significant changes/Error in ECCS Evaluation Models

<u>**For LOCAs at or below TBS**</u>, a significant change for UO2 or MOX fuel within cylindrical zirconium-alloy cladding:

i. A calculated peak fuel cladding temperature different by more than 50 °F from the temperature calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective temperature changes is greater than 50 °F; or

ii. A calculated integral time-at-temperature different by more than 1.0 percent equivalent cladding reacted from the oxidation calculated for the limiting transient using the last acceptable evaluation model, or is a cumulation of changes and errors such that the sum of the absolute magnitudes of the respective oxidation changes is greater than 1.0 percent equivalent cladding reacted.

- Maintains threshold for significant change in calculated PCT at 50 °F
- Adds a new threshold for significant change in integral time-at-temperature of 1.0% ECR
  - Matches what was in draft final 50.46c rule

50.46a(k): Significant changes/Error in ECCS Evaluation Models

2) **For LOCAs above the TBS**, a significant change or error in the ECCS evaluation model for uranium oxide and mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding is one that results in a significant reduction in the capability to meet the requirements of paragraphs (e)(1) and (f) of this section.
3) For fuel that does not consist of uranium or mixed uranium-plutonium oxide pellets within cylindrical zirconium-alloy cladding, a significant change in the ECCS evaluation model is one that results in a significant reduction in the capability to meet the requirements of paragraphs (e)(1) and (f)(1) of this section.

- For breaks above the TBS, an entity could define alternative criteria for a significant change. If alternative criteria are not defined, then the same reporting criteria in proposed 10 CFR 50.46(k)(1) would be applied (50°F PCT and 1% ECR)
- A new definition of significant change or error may be necessary for other fuel/cladding materials

Changes to Base 50.46

- Corresponding changes were made to 50.46
  - For example, that either 50.46 or 50.46a could be used along with applicability statements
- The 50.46(b) criteria (e.g., 2200°F PCT and 17% ECR limits) were *not* changed for licensees who do not elect to adopt 50.46 as an attempt to limit the scope of the rule and possible delays
- While the applicability of 50.46 criteria was not expanded from UO2 pellets within Zircaloy or ZIRLO cladding, a statement was added to say that the criteria in 50.46(b) or 50.46a(f)(1) must be met
  - 50.46(3)(i): "The ECCS system must be designed so that its calculated cooling performance following postulated loss-of-coolant accidents (LOCAs) conforms to the criteria set forth in paragraph (b) of this section *or § 50.46a(f)(1)*..."
  - **Therefore entities with that do not adopt 50.46a and have non-Zircaloy or non-ZIRLO fuel can either elect to submit an exemption to 50.46 to use the 50.46(b) criteria or elect to use the fuel-technology-neutral criteria in 50.46a(f)(1) without an exemption**
    - Entities that elect to use the 50.46a(f)(1) criteria would be expected to address the SECY-16-0033 embrittlement research findings

Questions

Draft Guide-1426
An Approach for a Risk-Informed Evaluation Process Supporting Alternative Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Reactors

Kristy Bucholtz
Reliability and Risk Analyst
PRA Oversight Branch

50.46a Risk Approach

- 50.46a(c)(iii) requires a risk-informed evaluation to make any proposed change.
- 50.46a has two risk pathways:

(1) Individual submittals - risk-informed evaluation with a risk assessment of the proposed changes.

  - Multiple submittals are allowed
    - Initial adoption of the rule
    - Each future change enabled by 50.46a.
  - Initial and each enabled change submitted to NRC for review.
  - Paragraph (h)(2) applies → Acceptance guidelines in DG-1426, Section 2.2.3.1 apply.

50.46a Risk Approach

- 50.46a has two risk pathways:

  (2) Risk-Informed Evaluation Process (RIEP) submittals - risk-informed evaluation with risk assessment of the proposed changes, **however, RIEP is also submitted for NRC approval**.

  - Multiple submittals are allowed
    - Initial adoption of the rule which includes the RIEP.
    - Each future change enabled by 50.46a.
  - Each future change enabled by 50.46a is evaluated with the RIEP.
  - Paragraph (h)(1) → if met, the change may be made without NRC approval. → DG-1426, Section 2.2.3.2.
  - Paragraph (h)(2) → if met, submit for NRC approval → DG-1426, Section 2.2.3.1.

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

DG-1426 Structure

- **DG-1426 follows the same structure as RG 1.174**
  - C.1 - Element 1: Define the proposed change
  - C.2 - Element 2: Perform Engineering Analysis
  - C.3 - Element 3: Define Implementation and Monitoring Program
  - C.4 - Element 4: Submit the License Amendment Request
  - C.5 - Quality Assurance
  - C.6 - Documentation

Element 1

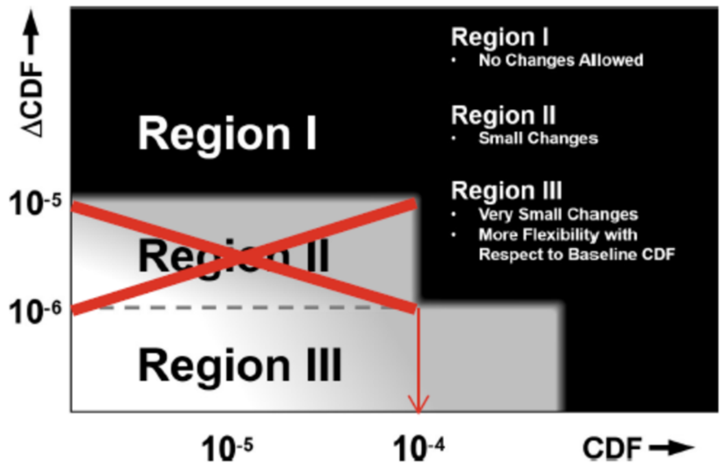
- **C.1 - Element 1: Define the proposed change**
  - DG-1426 follows the structure of RG 1.174, which is written for licensees changing their licensing basis, with a few minor changes, identified below:
    - Modified from “licensee” to “entity.”
    - Modified from “licensing basis changes” to “proposed changes.”
    - Added sentences for entity to identify the aspects that may be affected, such as the licensing basis or entity-controlled documentation, and implementation pathway.
      - NRC review - license amendment via 10 CFR 50.90
      - Without NRC review

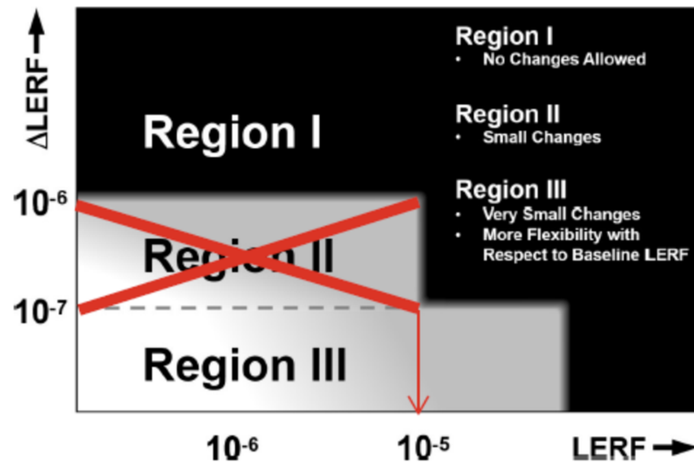
Element 2

- **C.2 - Element 2: Perform Engineering Analysis**
  - Section 2.1 - Risk-Informed Evaluation Process
  - Section 2.2 - Risk Assessment
    - Section 2.2.1 - Probabilistic Risk Assessment (PRA)
      - Addresses scope, level of detail, technical elements, and plant representation.
      - Based on RG 1.200.
    - Section 2.2.2 - Non-PRA risk assessments
    - Section 2.2.3 - Risk Metrics
      - Section 2.2.3.1 - Acceptance guidelines for risk-informed evaluations requiring prior NRC approval
      - Section 2.2.3.2 - Acceptance guidelines for self-approved risk-informed evaluations
    - Section 2.2.4 - Defense in Depth
    - Section 2.2.5 - Safety Margins

Element 2

- Section 2.2.3.1 - Acceptance guidelines for risk-informed evaluations requiring prior NRC approval

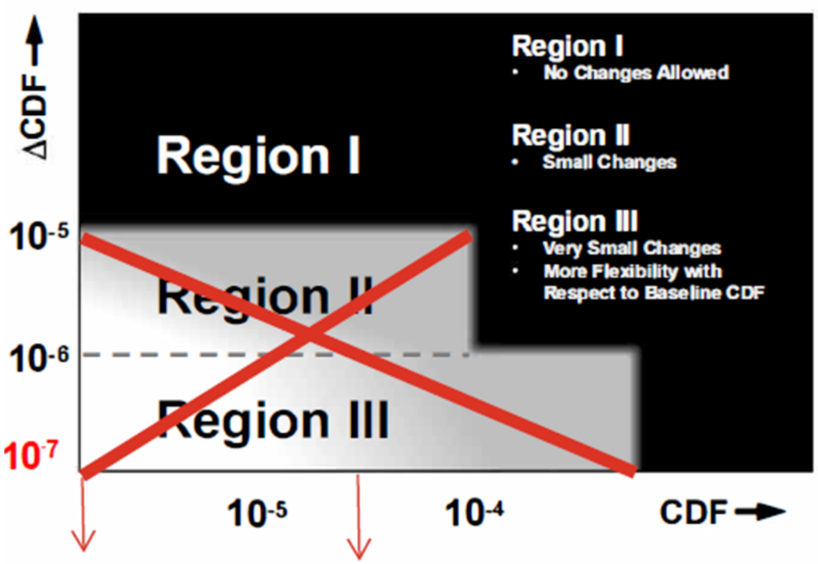


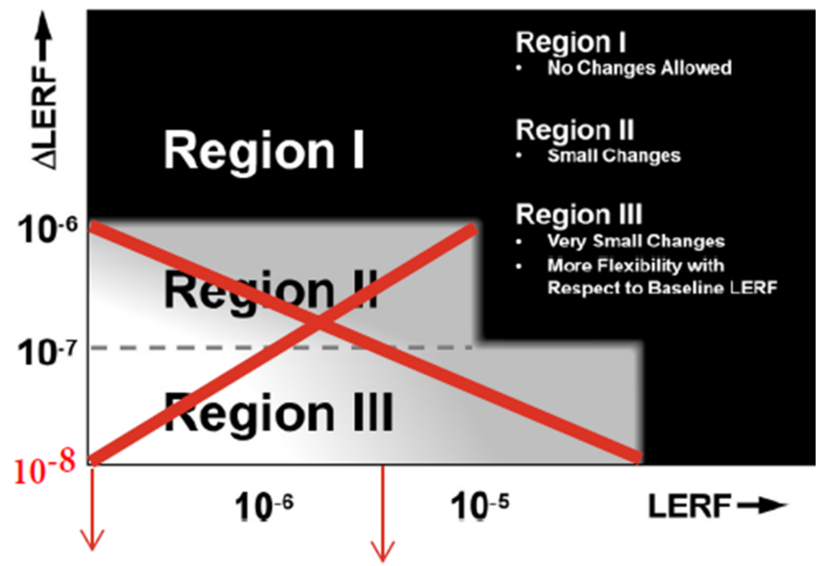


5. In order to more closely follow the approach presented in Regulatory Guide 1.174, the staff should modify the proposed rule to ensure that any changes under this rule be further restricted to very small risk increases, notwithstanding the fact that they would otherwise be permitted under 50.59. Therefore, staff should add the word "very" before the word "small" in section (f)(1)(i) so that it reads "...the total increase in core damage frequency and large early release frequency are very small and the overall risk remains small..." or make other changes as appropriate to achieve the above objective.

Element 2

- Section 2.2.3.2 - Acceptance guidelines for self-approved risk-informed evaluations





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

Element 3

- **C.3 - Element 3: Define Implementation and Monitoring Program**
  - Pointer to the section in RG 1.174 for implementation and monitoring program
    - The periodic updating of the risk assessments was removed from the draft proposed 50.46a rule and DG-1426.
    - DG-1426 no longer includes re-evaluation of the risk assessment.
    - Staff removed the requirement to update the PRA every 5 years from the draft proposed 50.46a rule.
    - The update remains in DG-1426, but has been changed from a "must" to a "should."

*Note: Staff is revising the version of DG-1426 that was originally submitted to the ACRS to address the four changes listed above.

Element 4

- **C.4 - Element 4: Submit the License Amendment Request**
  - Submit a summary of the PRA model and methods used to evaluate the proposed change.
    - which risk methods are used and why they are acceptable,
    - key modeling assumptions and consideration of uncertainty,
    - key operator actions, and
    - changes required to event or fault trees in the PRA model.
  - For RIEP submittals, submit details of the RIEP to be used to support changes without NRC approval.
    - Description of the entity's PRA model and any non-PRA risk assessment methods to be used.
    - Description of the entity's approach, methods, and decisionmaking process to evaluate:
      - Risk criteria
      - Defense in depth
      - Safety margins
      - Performance measurement and monitoring

Quality Assurance and Documentation

- **C.5 - Quality assurance**
  - The same as RG 1.174, with no substantial changes.

- **C.6 - Documentation**
  - Differs from RG 1.174
    - For each plant change, the entity should document the risk-informed evaluation, consistent with section C.4 of RG 1.200.

Questions

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

Proposed Rule: Increased Enrichment of Conventional and Accident Tolerant Fuel Designs for Light-Water Reactors

December 17-18, 2024

Opening Remarks

Theresa Clark
Director
Division of Safety Systems

Draft Regulatory Guides for Zirconium-Alloy Cladding Analytical Limits

James Corson
Senior Reactor Systems Engineer (Fuels Analyst)
Fuel and Source Term Code Development Branch

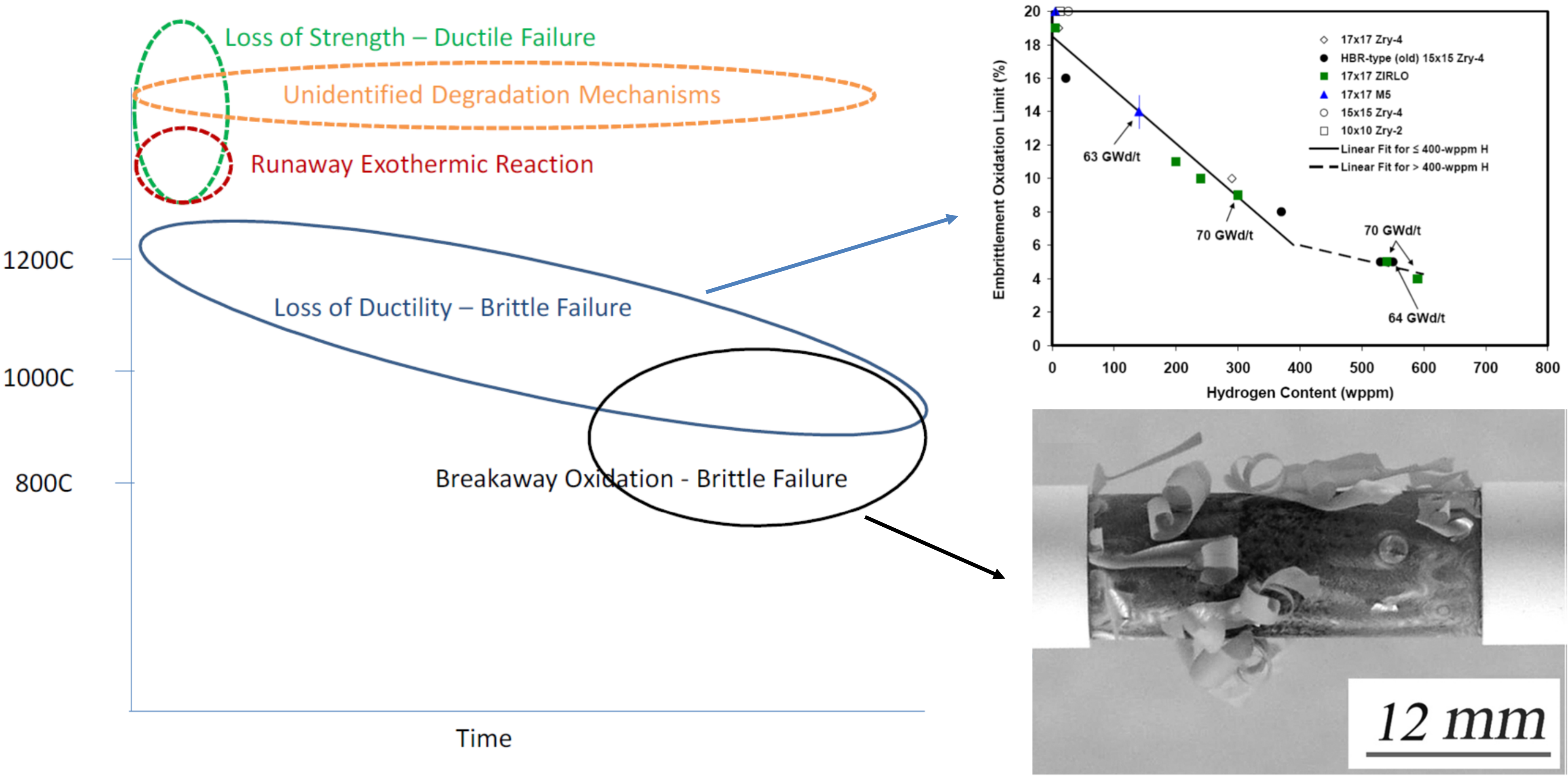
Overview

- As part of the 50.46c rulemaking, NRC staff developed 3 draft regulatory guides to address zirconium-alloy cladding analytical limits
  - DG-1261 (RG-1.222): Measuring Breakaway Oxidation Behavior
  - DG-1262 (RG-1.223): Determining Post-Quench Ductility
  - DG-1263 (RG-1.224): Establishing Analytical Limits for Zirconium-Based Alloy Cladding
- Staff have updated these documents to reflect the 50.46a proposed rule language
  - Updates reflect fact that aspects of the 50.46c rule language have been moved to guidance for 50.46a proposed rule
  - Otherwise, the guides are (mostly) unchanged from the versions included in the 50.46c draft final rule package

Relation of DGs to the Rule Language

- 50.46a(f)(1) *Fuel performance criteria.* Fuel system designs must have NRC-approved limits that:
  i. Address cladding degradation phenomena;
  ii. Maintain fuel coolability
  iii. Avoid explosive concentration of combustible gas; and
  iv. Demonstrate that, after any calculated successful initial operation of the ECCS, the ECCS must provide sufficient coolant to remove decay heat and prevent further cladding failure for the extended period of time required by the long-lived radioactivity remaining in the fuel.
- DGs 1261, 1262, and 1263 primarily address zirconium-alloy cladding embrittlement
  - Thus, they mostly address 50.46a(f)(1)(i), though DG-1263 includes a limit for 50.46a(f)(1)(iii)

Zirconium Alloy Cladding Degradation Mechanisms



Abbreviated History

- DGs 1261, 1262, and 1263 included in the 50.46c proposed rule published March 2014 (ML12283A174)
  - Public comment period ended August 2014
  - Several public meetings held on 50.46c during public comment period
- Several public meetings held between close of public comment period and publication of draft final rule in 2016
  - Public meeting on regulatory guidance in April 2015 (ML15132A743)
  - Overview of preliminary draft changes to the rule and guidance in October 2015 (ML15321A004)
  - ACRS SC on the draft final rule package on November 3, 2015 (ML15320A187)
  - ACRS FC meeting on the draft final rule package on December 3, 2015 (ML15349A717)
- DGs included as RGs 1.222, 1.223, and 1.224 in draft final rule package published March 2016 (ML15238A933)

DG-1261: Measuring Breakaway Oxidation Behavior

– Breakaway oxidation in zirconium alloy cladding associated with change from protective tetragonal oxide to non-protective monoclinic oxide

– Breakaway oxidation characterized by significant increase in oxidation rate and hydrogen pickup, both of which lead to cladding embrittlement

– NRC's LOCA program showed that minor changes in alloy composition or manufacturing processes can have significant impact on breakaway oxidation behavior

DG-1261

- Defines an experimental technique capable of determining the effect of composition changes or manufacturing changes on the breakaway oxidation behavior
- Experimental technique includes flexibility, where possible, to allow variation of equipment and procedures in use at other laboratories
- Discusses both initial testing and periodic confirmatory testing

DG-1261

- Initial testing includes examination of breakaway oxidation behavior at a range of temperatures to identify the critical temperature associated with the shortest time to breakaway oxidation
- Allows adoption of Argonne National Laboratory test data for initial implementation of 50.46a rule

DG-1261

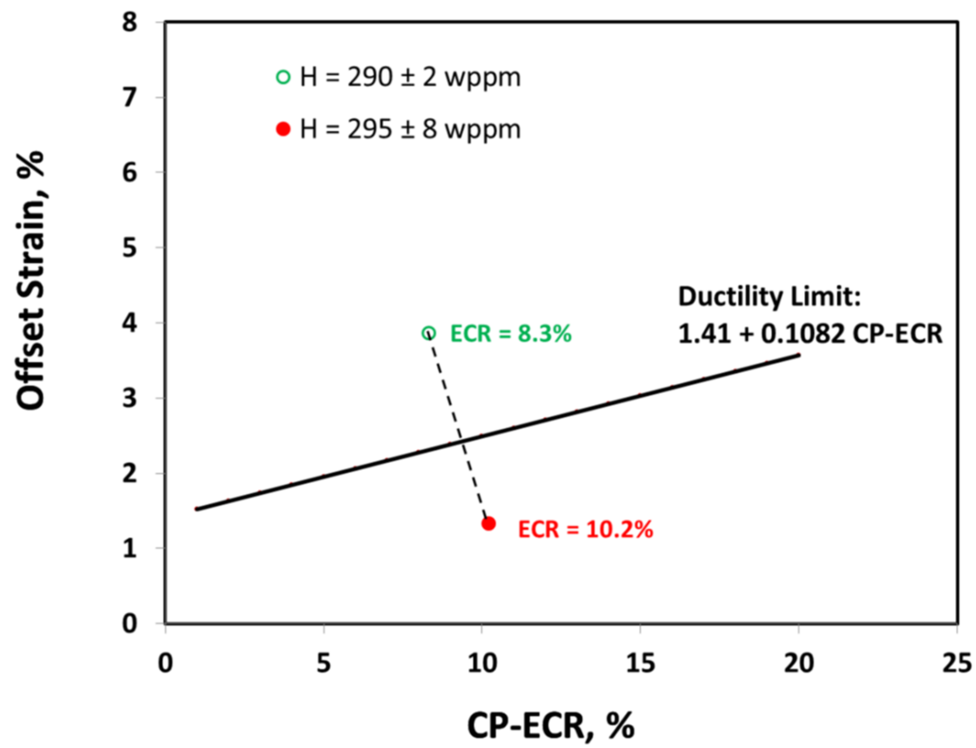
- Periodic testing is used to confirm the initial testing (and associated analytical limit) remains applicable to manufacturing life-cycle
  - Periodic Confirmatory Test Program Plans (PCTPP) would be developed by each cladding vendor and submitted for NRC review and approval
  - Periodic testing is focused only on the critical temperature identified in initial testing
  - Vendors would define periodic testing frequency in the PCTPP; DG-1263 provides an optional default frequency (testing once per ingot) and states that other frequencies could be reviewed and approved
  - Guidance allows for relaxation of test frequency with time
  - Periodic testing results are not submitted but must be available for audit

DG-1262: Determining Post-Quench Ductility

- Defines an experimental technical to measure the ductile-to-brittle transition for the zirconium-alloy cladding material
- Experimental technique includes flexibility, where possible, to allow variation of equipment and procedures in use at other laboratories
- Provides detailed discussion of determining the ductile-to-brittle transition CP-ECR for a given hydrogen level, allows for binning results with similar H content
  - CP-ECR = equivalent cladding reacted calculated using the Cathcart-Pawel correlation

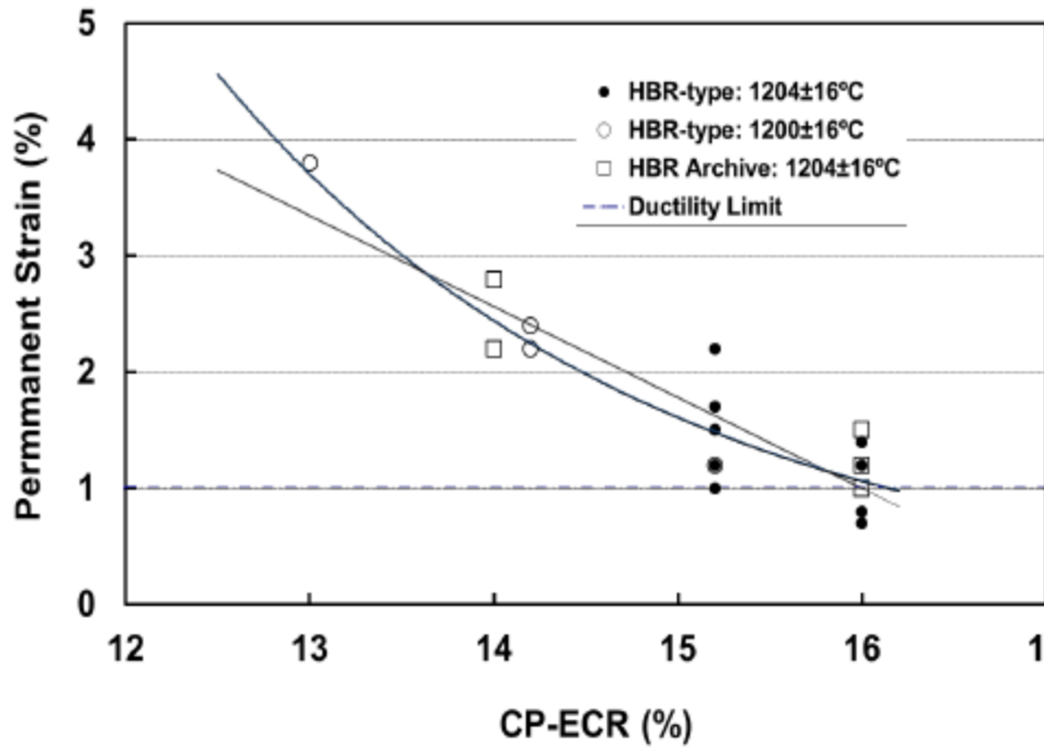
DG-1262

- Two approaches (set and curve-fit) are provided to address expected data scatter in a data "bin" and determine the ductile-to-brittle transition CP-ECR for a given hydrogen level
- Set approach



DG-1262

- Two approaches (set and curve-fit) are provided to address expected data scatter in a data “bin” and determine the ductile-to-brittle transition CP-ECR for a given hydrogen level
- Curve-fit approach



DG-1263: Establishing Analytical Limits for Zirconium-Alloy Cladding Material

- Describes an approach to establish limits to address zirconium-alloy cladding degradation phenomena
  - Analytical limits for post-quench ductility and breakaway oxidation
  - PCT limit to address post-quench ductility also protects against higher-temperature degradation mechanisms
- Provides guidance on how to consider the impact of oxygen diffusion from inside surfaces on cladding degradation
- Provides default cladding hydrogen uptake models for currently approved cladding models
- Provides an analytical limit for combustible gas generation

DG-1263

C.1.A – An acceptable limit for currently deployed alloys

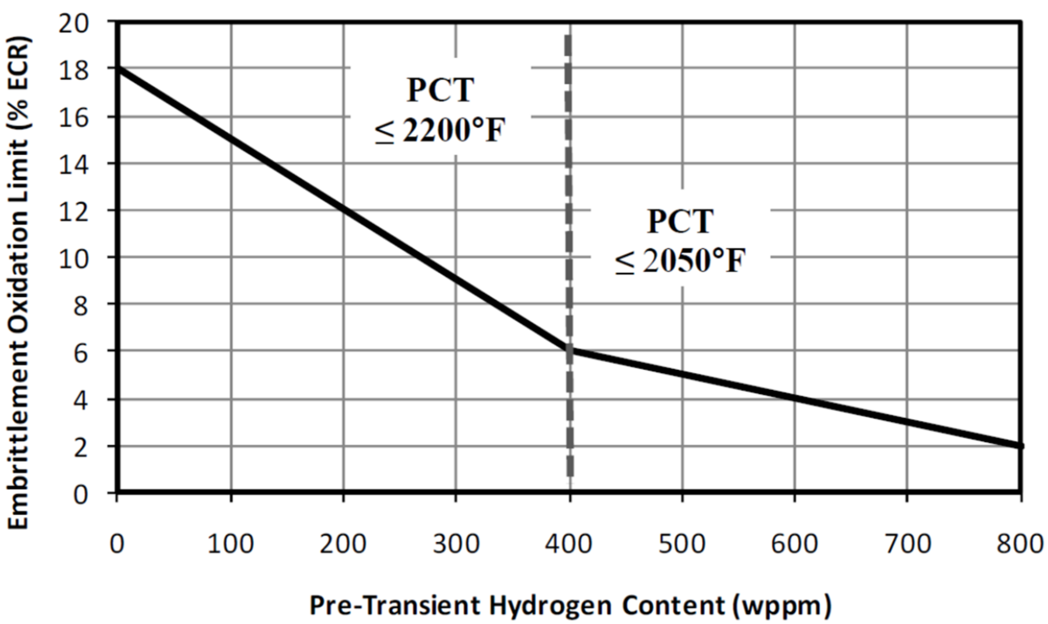


Figure 2 of DG-1263. Acceptable analytical limits for peak cladding temperature and integral time at temperature (as calculated in local oxidation calculations using the CP correlation) for Zircaloy-2, Zircaloy-4, ZIRLO®, M5®, and Optimized ZIRLO™

DG-1263

C.1.B – Adopting Figure 2 for New Alloys

New alloys can adopt Figure 2 by providing the measured ductile-to-brittle transition level for cladding material in the following conditions:

1. As received
2. Unirradiated, pre-hydrided within 100 ppm of the maximum hydrogen content specified at end of life (EOL)
3. Unirradiated, pre-hydrided within 100 ppm of half of the maximum hydrogen content specified at EOL
4. Irradiated (unless the new alloy is “similar” to previously tested alloys)

DG-1263

- New cladding alloys are considered "similar" to alloys tested in NRC's LOCA program (conducted at Argonne National Laboratory) if they:
  - Use the Kroll process
  - Operate less than or equal to the maximum fluence
  - Include only the alloying elements present in the materials tested
  - Have similar alloying content of each element to the materials tested in NRC's LOCA program, whereby each alloy element is defined by less than or equal to 25 percent deviation from the alloying limits defined for the tested alloy

DG-1263

C.1.C & C.1.D – Adopting other post-quench ductility limits

- Analytical limits other than those defined in Figure 2 can be adopted for new and existing alloys to gain margin for superior alloy-specific cladding performance (C.1.C) or for slower embrittlement behavior at lower temperatures (C.1.D)

DG-1263

C.1.E – Hydrogen pickup models

- An alloy-specific cladding hydrogen uptake model should be used in conjunction with the hydrogen-dependent embrittlement threshold provided in Figure 2
- Appendix A of DG-1263 provides acceptable fuel rod cladding hydrogen uptake models for the current commercial zirconium alloys

DG-1263

C.1.F – Demonstrating compliance for PQD

- Identify the limiting conditions and assumptions that maximize predicted PCT and local oxidation
- Demonstrate PCT and max local oxidation are below PQD analytical limit
- Provides allowance for subdividing the ECCS evaluation based on cladding hydrogen content, burnup, fuel rod power, or a combination
- Provides allowance to use Figure 2 for legacy fuel to show compliance with 50.46a(f)(1) requirements

DG-1263

C.2 – Breakaway oxidation

- Provides allowance for legacy fuel to use analytical limit established for the current version of the alloy
- Applicants may elect to establish the analytical time limit for breakaway oxidation with conservatism relative to the measured minimum time (i.e., reduce the time) to the onset of breakaway oxidation
- The total time that the cladding is predicted to remain above the temperature that the zirconium-alloy cladding material has been shown to be susceptible to breakaway oxidation (800^{o}C default) must be less than the analytical limit
  - Applicant may credit operator action to limit the duration at elevated temperatures provided these actions are consistent with existing procedures and the timing of such actions is validated by operator training on the plant simulator or via a job performance measure

DG-1263

C.3 – Hydrogen generation

- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam should not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react
- Same as existing requirement in 50.46(b)(3)

DG-1263

C.4 – Inner diameter oxidation

- ECCS evaluation models should consider oxygen diffusion from the cladding inside surfaces if an oxygen source is present on the inside surface of the cladding at the onset of the LOCA
  - Cladding rupture: calculate 2-sided oxidation using CP correlation, consider the reduced cladding thickness and the rupture mid-plane an apply Figure 2 (same approach used today)
  - Fuel-cladding bond: calculate 2-sided oxidation once the fuel-cladding bond layer is predicted to occur (default threshold is 30 GWd/MTU, but higher limits can be proposed for NRC review and approval)

Conclusions

- Three draft guides have been developed to support performance-based criteria related to zirconium-alloy cladding degradation in the 50.46a proposed rule. The DGs provide guidance to develop material-specific analytical limits on key embrittlement mechanisms.
  - DG-1263 also includes a hydrogen generation limit
- The DGs are based on the guides submitted as part of the 50.46c draft final rule package but have been updated to reflect requirements of the 50.46a proposed rule (e.g., removing specific limits from the rule language)
  - DGs reflect extensive interactions with industry stakeholders during the 50.46c draft final rule public comment period.

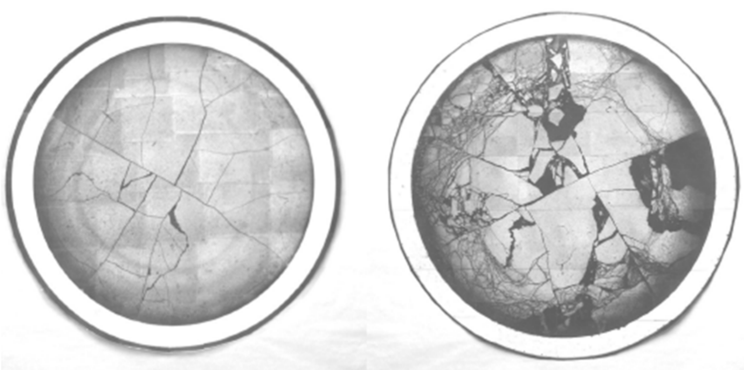
Questions

DG-1434: Addressing the Consequences of Fuel Dispersal in Light-Water Reactor Loss-of-Coolant Accidents

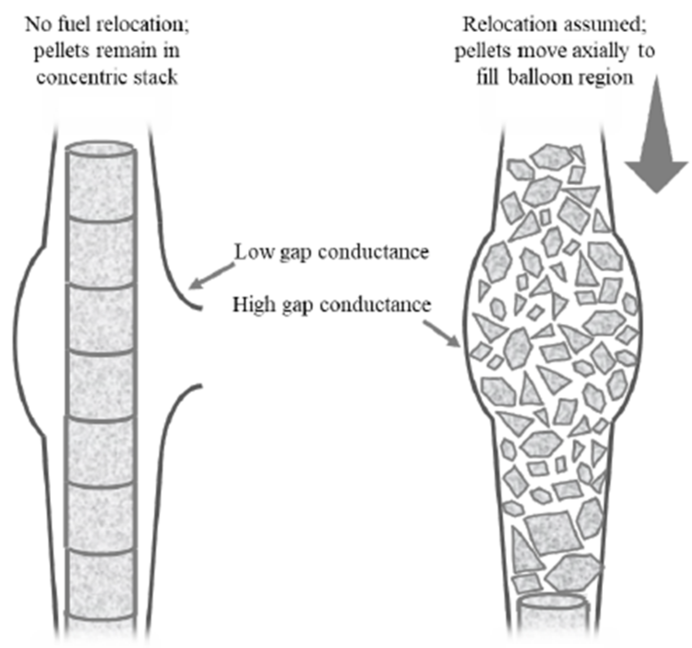
James Corson
Senior Reactor Systems Engineer (Fuels Analyst)
Fuel and Source Term Code Development Branch

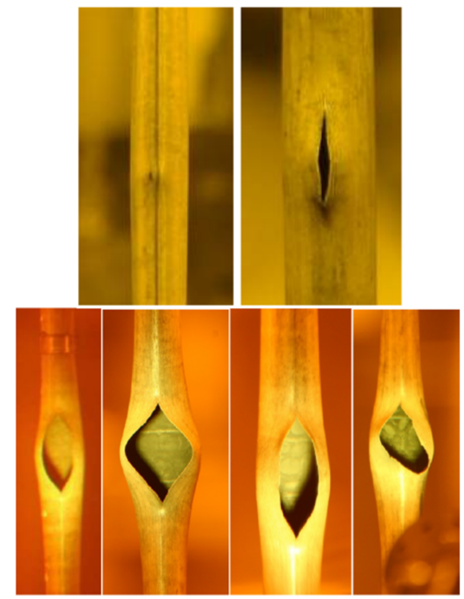
Fuel Fragmentation, Relocation, and Dispersal (FFRD)

- At 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



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





Burst openings from Studsvik LOCA tests (NUREG-2121)

FFRD: History

- 1974 – 10 CFR 50.46 criteria for LOCA’s
- 1980 – FFRD Discovered
- 1984-1995 – GI-92, “Fuel Crumbling During LOCA”
- 2006 – FFRD testing showed gross fuel loss at high BU
- 2008 – RIL-0801
- 2012 – NUREG-2121
- 2015 – SECY-15-0148 – No imminent safety concern
- 2016 – Draft final rule for 50.46c
- 2021 – RIL-2021-13
- 2022 – SRM-SECY-21-0109 – Include FFRD in IE Rulemaking
- 2024 – NUREG/CR-7307

Fuel Dispersal: Background and Regulatory Issue

- The 50.46 acceptance criteria date to 1974 when FFRD were not known phenomena
- Acceptable approaches to demonstrate compliance with the regulations have ensured that catastrophic failure of the fuel rod structure and loss of fuel bundle configuration are precluded
  - Fuel dispersal would be a departure of precedent
- Fuel dispersal is not explicitly addressed within the current regulations
  - Proposed rule language (50.46a) allows for some flexibility regarding fuel dispersal
  - DG-1434 provides guidance for addressing fuel dispersal within the proposed rule

Draft Guidance for Fuel Dispersal

- DG-1434 provides guidance for addressing the impact of fuel dispersal on ECCS performance
  - Includes a model to estimate the mass of dispersed fuel
  - Provides high-level acceptance criteria for fuel dispersal
  - Lists analyses to perform to address consequences of dispersed fuel
- DG-1434 builds on recent research efforts and reflects the current state of knowledge
  - Research Information Letter (RIL) 2021-13, "Interpretation of Research on Fuel Fragmentation, Relocation, and Dispersal at High Burnup" (ML21313A145)
  - NUREG/CR-7307, "PIRTs on High Burnup Fuel Fragmentation, Relocation, Dispersal, and Its Consequences" (ML24155A058)
  - EPRI-sponsored 2024 White Paper, "Assessment of Existing Fuel Fragmentation, Relocation, and Dispersal Data: Best Estimate Interpretation"
  - Several recent publications from researchers at Oak Ridge and Idaho National Laboratories

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 PIRT panelists believe it should be possible to perform simplified analyses to demonstrate coolability so long as the dispersed mass remains low

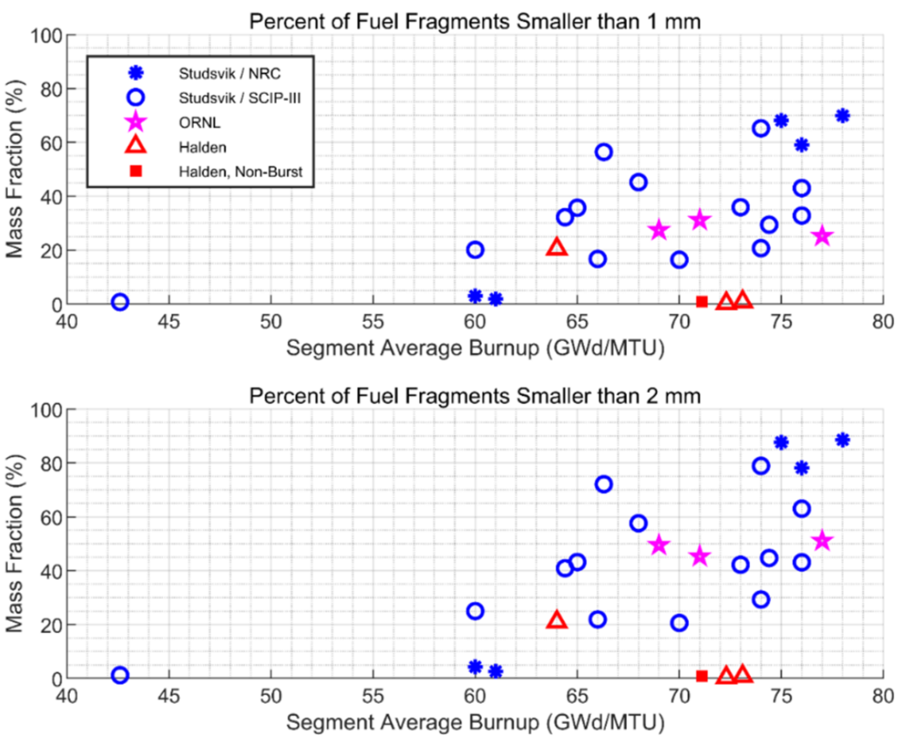
Structure of DG-1434

1. Limits on applicability
2. FFRD thresholds
3. Analytical limits for fuel dispersal
4. Methods for estimating the dispersed fuel mass
5. Impacts of fuel dispersal
    a. Fuel particle transport and deposition
    b. Fuel coolability and long-term cooling
    c. Re-criticality
    d. Radiological consequences and environmental qualification (covered by RG 1.183 Rev. 2)

Limits on Applicability

- FFRD thresholds and methods for estimating the mass of dispersed fuel apply to undoped UO_2 fuel in zirconium-alloy cladding
  - Extension to UO_2 fuel with dopants (e.g., gadolinia, chromia, alumina, and/or silica) or MOX will be considered on case-by-case basis
  - Recently completed SCIP-IV tests and upcoming tests under the Second Framework for Irradiation Experiments (FIDES-II) and SCIP-V could help address this limitation
- Other sections of the guidance are generally applicable to all fuel designs, unless otherwise noted
  - For example, some limits related to recriticality (see upcoming slides)

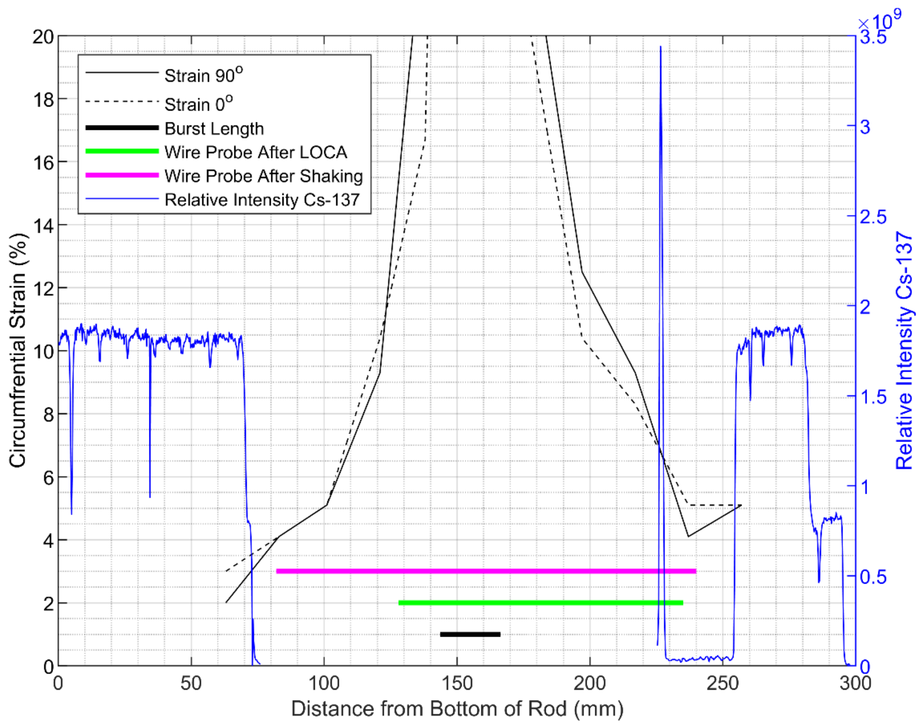
FFRD Thresholds



- NRC staff position is that fine fragmentation begins around a burnup of 55 GWd/MTU
- This is a simplification of complex processes in the fuel
  - Burnup is only a surrogate for microstructural changes
  - Other parameters (e.g., temperature) influence fragmentation behavior

Figure 3 from RIL 2021-13

FFRD Thresholds



- Relocation is limited below a cladding hoop strain threshold of 3%
- Thus, fuel rods with pellet-average burnups above 55 GWd/MTU that balloon and burst during LOCA are susceptible to fuel dispersal

NRC test #	Strain threshold, top (%)	Strain threshold, bottom (%)
189	6.0	3.0
191	6.0	4.0
192	5.0	4.0
193	1.0	4.0
196	3.0	5.0
198	4.5	9.0

Figure 4 and Table 1 from RIL 2021-13

Analytical Limits for Fuel Dispersal

- DG-1434 provides acceptance criteria for the dispersed fuel mass
  - No fuel dispersal for breaks < TBS
    - Can be addressed by showing no ballooning and burst for rods peak pellet burnup > 55 GWd/MTU
  - For breaks > TBS, either show there is no fuel dispersal or show that other criteria are met (see upcoming slides)
- Dispersed fuel mass should be calculated using an approved evaluation model and fuel dispersal models
  - Evaluation model should include the impacts of transient fission gas release

Plausibility of the No Dispersal Criterion

- NRC staff analysis performed around 2013 provided fuel dispersal estimates for 3 plant designs (Westinghouse 4-loop PWR, CE PWR, GE BWR/4) (see ML23086B272)
  - Based on current licensed burnup limits and fuel management practices
  - Using nominal (rather than intentionally conservative) initial conditions
  - No dispersal predicted for CE PWR or GE BWR/4
    - Core wide PCTs of 700°C and 500°C, respectively
- During the IE rulemaking regulatory basis public comment period, NRC staff received comments stating that it may be possible to show no dispersal using more realistic LOCA methods allowed for break sizes > TBS

Methods for Estimating Dispersed Fuel Mass

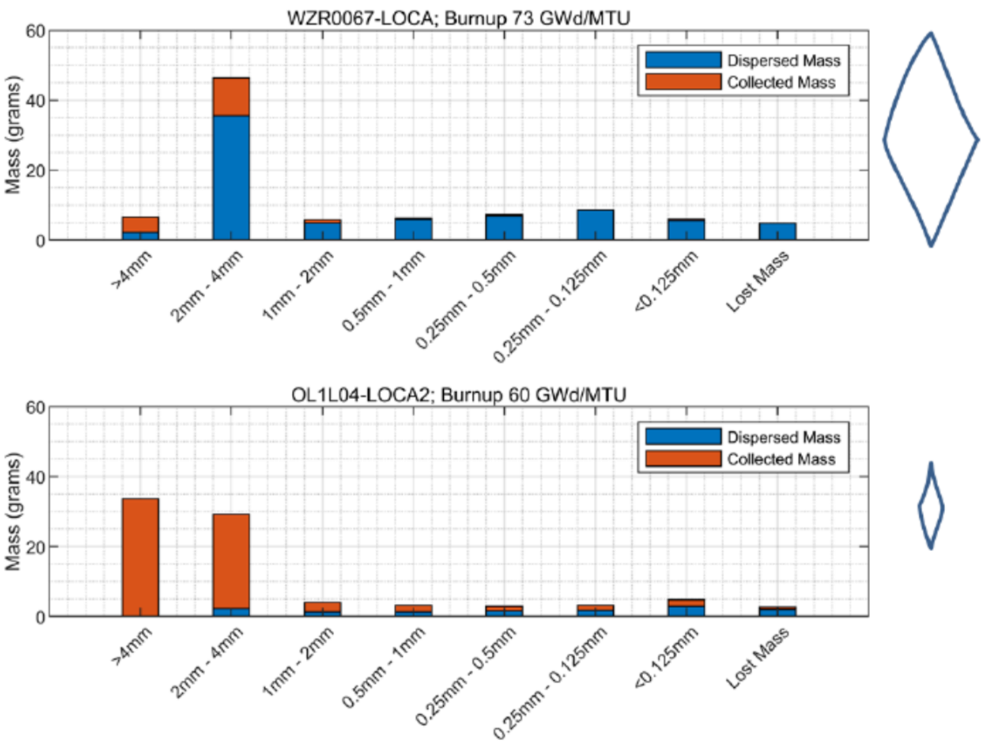


Figure A-4 from RIL 2021-13: Fragment size distribution for two SCIP-III tests, with relative burst opening size

- Fuel dispersal is impacted by many parameters, most of which are highly uncertain
  - Fuel dispersal PIRT identified “burst opening size relative to the fuel fragment size distribution” as high importance / low knowledge level
  - Other parameters like rod internal pressure also impact dispersal
- DG-1434 proposes a surrogate model for fuel dispersal
  - Avoids mechanistically modeling transport of particles through burst opening

Methods for Estimating Dispersed Fuel Mass

Figure A-6 from RIL 2021-13:
Mass fraction of fragments < 1 mm, with RIL Model A for comparison

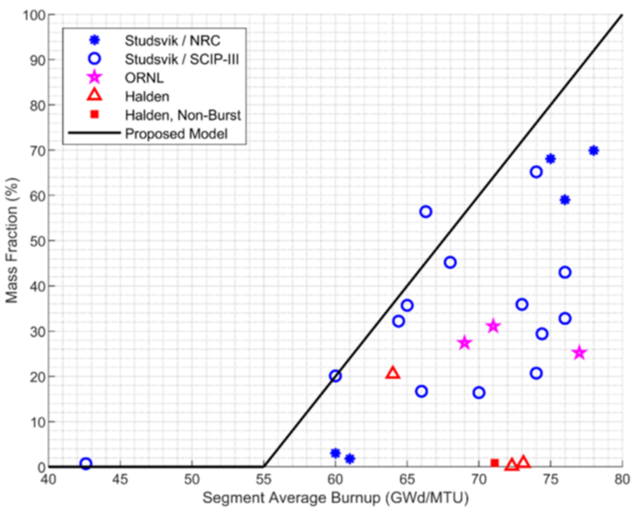
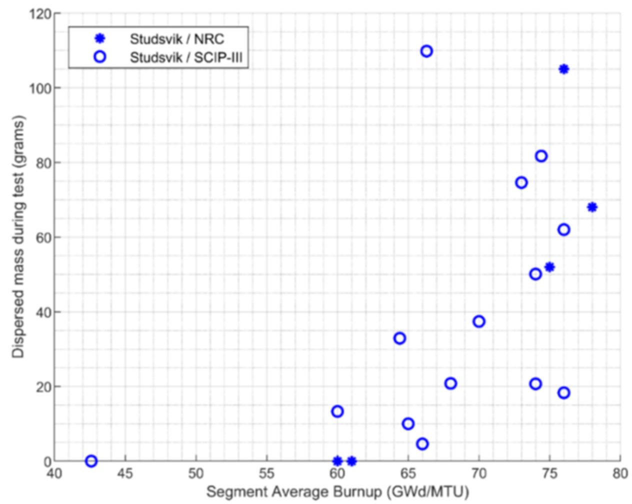


Figure A-1 from RIL 2021-13:
Mass of fuel dispersed during the test



- RIL 2021-13 provided surrogate models for fuel dispersal
  - Model A assumes that the mass of fragments below 1 mm is a good surrogate for dispersed mass
  - However, this does not mean that all dispersed fragments are less than 1 mm, nor does it mean that all 1 mm fragments disperse
  - Still, Model A is consistent with observations that dispersal increases with burnup

Methods for Estimating Dispersed Fuel Mass

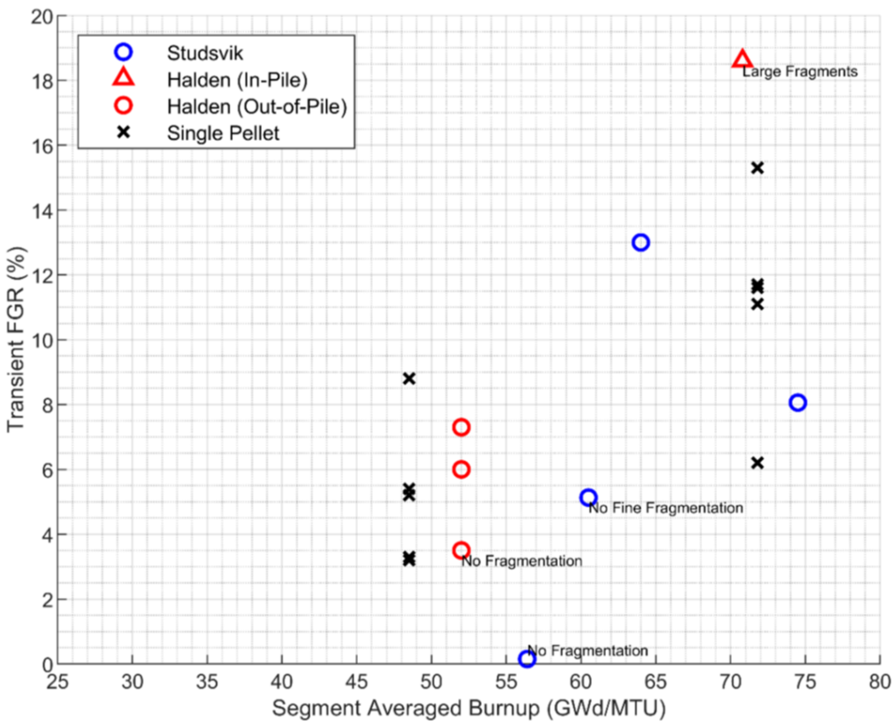
- DG-1434 recommends using RIL Model A to calculate fuel dispersal

$$\text{mass fraction} = \begin{cases} 0, & BU < 55 \\ 0.04\left(BU - 55\right), & 55 < BU < 80 \\ 1, & BU > 80 \end{cases}$$

  - Mass fraction should be multiplied by the mass of fuel in the region with >3% hoop strain
    - Can credit grid spacers to limit axial length of fuel susceptible to dispersal
  - Calculation should be performed for rods predicted to balloon and burst

	Difference between dispersal predicted by the model and dispersal observed in the experiment	
SCIP-III Test	A (mass, g)	A (%)
OL1L04-LOCA-2	29	314%
N05-LOCA	(10)	70%
VUR1-LOCA-1	(26)	76%
WZR0067-LOCA	(18)	75%
VUL2-LOCA1	34	169%
VUL2-LOCA3	142	874%
VUL2-LOCA4	99	259%

Methods for Estimating Dispersed Fuel Mass



- Methods for estimating mass of fuel dispersed should consider impact of transient fission gas release
  - Models should only consider gas release up to the point of ballooning and burst
    - Expected release at time of burst likely less than results shown in RIL 2021-13 due to burst temperatures being lower than test temperatures and impact of rod internal pressure in suppressing gas release
  - DG-1434 does not endorse any models but identifies potential starting points

Figure 7 from RIL 2021-13: Transient fission gas release as a function of burnup; note that peak temperatures for the Studsvik and single-pellet tests were between 1000 and 1200^{o}C

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

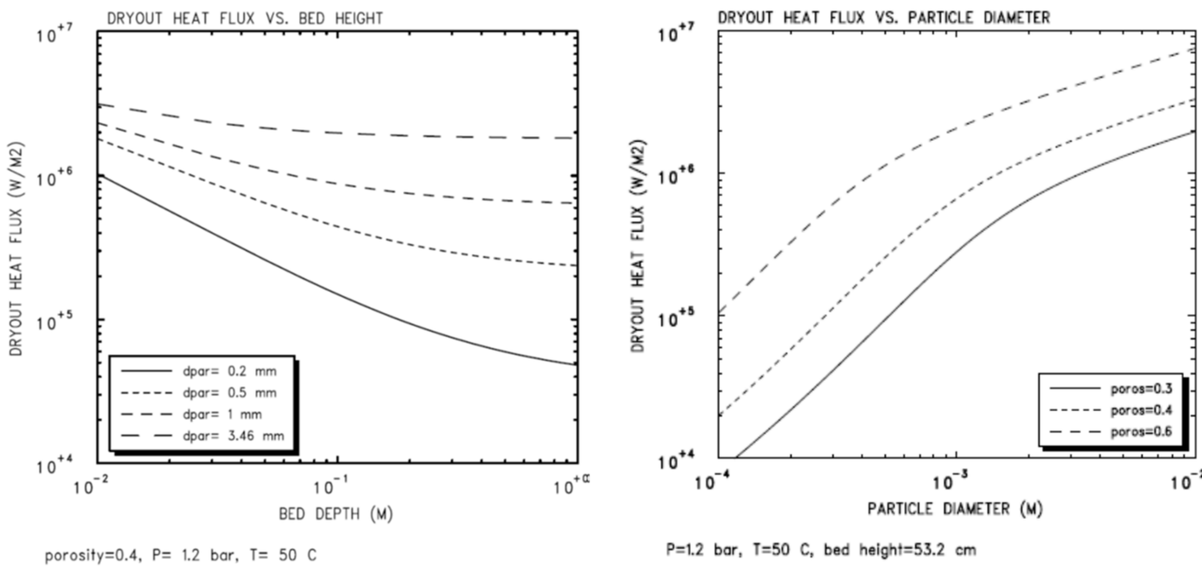
Fuel Particle Transport

- The fuel dispersal PIRT panel noted that transport of irregularly shaped particles in multi-phase flow is poorly understood
  - Panelists suggested addressing impacts of dispersal through simplified, bounding calculations, using engineering judgement about where particles may deposit
- DG-1434 identifies several potential dispersed fuel configurations to use when addressing fuel coolability
  - For example: on spacer grid immediately below cladding burst location, lower plenum, RCS piping

Fuel Coolability

- Analyses should address impact of dispersal on PCT and maximum local oxidation
  - Assuming fuel collects on spacer grid immediately below the burst location
- Analyses should verify that the dryout heat flux is not exceeded for particle beds on spacer grids and for other locations
  - Should use the 0-D Lipinski model to calculate dryout heat flux (i.e., maximum heat that can be removed from surface of particle bed)
- Analyses should perform calculations using range of conditions
  - Fuel particle size: 0.125 – 4 mm
  - Bed porosity: 20% – 40%

Fuel Coolability: Lipinski Model



- OECD/NEA Working Group on Fuel Safety report on FFRD included dryout heat flux calculations
  - Showed dryout heat flux could be exceeded under some conditions (especially for small particle sizes)

Figs. 8.1-1 and 8.1-2 from the OECD/NEA Working Group on Fuel Safety 2016 report on FFRD

Fuel Long-term Cooling

- Analyses should demonstrate that adequate coolant flow is provided to remove decay heat from within the core and from fuel dispersed out of the core
- Analyses should verify that the dryout heat flux is not exceeded for particle beds
  - Should consider potential impact of coagulants or other debris that could reduce bed porosity below 20%

Dispersed Fuel Recriticality

- The fuel dispersal PIRT panel believed that recriticality of dispersed fuel is not a concern
  - 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 DG-1434) and had initial U-235 enrichment of 8 weight percent
  - Did not credit soluble boron

Dispersed Fuel 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)

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

Dispersed Fuel Recriticality

- Staff analysis shows that recriticality is very unlikely
  - This is consistent with analysis performed by the OECD/NEA Working Group on Fuel Safety
  - Staff only performed quantitative analysis for one configuration, but based on engineering arguments recriticality is unlikely for other configurations
- DG-1434 states that 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 calculation
  - At the same time, staff is working on providing stronger basis to resolve recriticality concern for the draft final rule package

Conclusions

- DG-1434 provides guidance for addressing the impact of fuel dispersal on ECCS performance
  - Guidance relies on use of more realistic LOCA methods and less conservative models from RIL 2021-13 to limit dispersed fuel mass
  - Guidance also includes methods to address impact of dispersed fuel on coolability
- Guidance is only one method for meeting regulatory requirement to maintain fuel coolability (50.46a(f)(1)(ii))
  - Industry can propose alternative approaches for NRC review

Questions

Increased Enrichment Rulemaking for GDC-19 – Control Room and 10 CFR 50.67, Accident Source Term

Elijah Dickson, Senior Reliability and Risk Analyst
Dave Garmon, Health Physicist
Division of Risk Assessment
Office and Nuclear Reactor Regulation
December 18, 2024

Topics

- Rulemaking Driver and Goal
- Background and History
- Regulatory Issues
- Rulemaking Proposal
- Rulemaking Approach

Rulemaking Driver and Goal

Legislative Driver:

- January 14, 2019, the President signed into law the Nuclear Energy Innovation and Modernization Act (NEIMA). Section 107, "Commission Report on Accident Tolerant Fuel," of NEIMA defines ATF as a new technology that makes an existing commercial nuclear reactor more resistant to a nuclear incident (as defined in section 11 of the Atomic Energy Act of 1954 (42 U.S.C. 2014)) and lowers the cost of electricity over the licensed lifetime of an existing commercial nuclear reactor.

Purposes:

- Facilitate the use of light-water reactor (LWR) fuel containing uranium enriched to greater than 5.0 weight percent uranium-235 (U-235).
- Developed in response to nuclear power industry interest to use fuel enriched to greater than 5.0 weight percent U-235 and deploy accident tolerant fuels (ATFs).

Staff Response:

- Evaluated areas within the regulatory framework and considered whether the current weight percent limits can be adjusted while maintaining reasonable assurance of adequate protection of public health and safety. Additionally, considered whether this rulemaking would support a more efficient review of licensing actions.

Background and History

Both GDC-19 and 10 CFR 50.67(b)(2)(iii) provide a specific dose-based criterion of 5 rem TEDE for demonstrating the acceptability of the control room design.

Represent a distinct layer of defense-in-depth that assumes a major accident that results in substantial meltdown of the reactor core with subsequent release of appreciable quantities of fission products.

Classic performance-based regulations which require that a licensee or applicant provide a control room habitability design using traditional deterministic radiological consequence analyses methods to judge the acceptability of the design.

Consequence analyses are also used to verify other regulatory requirements, guide maintenance activities, and serve as a guideline for performing 10 CFR 50.59 analyses.

Regulatory Issues

Assess Applicability in Current Environment

- Control room design criteria is limiting between the three acceptance criteria (EAB, LPZ, CR) for current enrichments and burnups.
- Development during the 1960s did not foresee how licensees are currently operating their facilities and managing fuel.
- The history of fuel utilization fleet has seen a gradual progression toward higher fuel discharge burnups and increased enrichments.
- There has been enough margin in the facilities' design- and licensing bases to accommodate the criteria, even for power uprates of up to 120 percent of the originally licensed steady-state thermal power level.
- Impact on Commission's comprehensive radiation protection and emergency planning frameworks.

Considerations of Control Room *Design* Criteria Impact is multifaceted

- Designer margin and operational flexibilities.
- Maturity of the regulated industry and compliance infrastructure.
- Maintenance activities and controlling actual operational exposure.
- 10 CFR 50.59 and low safety-significant licensing actions.

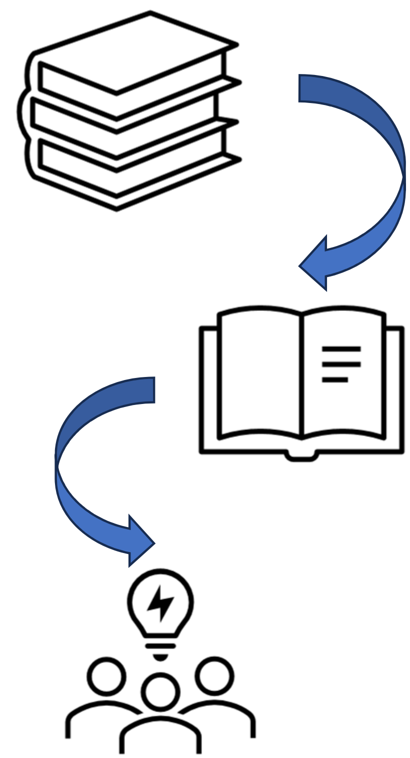
Radiological Risk Communications

- Design Criteria vs. Occupational Dose Limit.
- Radiation Protection and Emergency Response Frameworks.
- Health Physics First Principles and Radiation Epidemiology.
- Insights from Category 9 events.

Regulatory Issues (Cont.)

- The preamble for the 10 CFR 50.67 (64 FR 71990; December 23, 1999), final rule included the Commission's rationale for establishing 5 rem (0.05 Sv) TEDE as the GDC-19 numeric design criterion for licensees using an alternative source term. That rationale comprised the following:

> *"The criteria in GDC 19 were based on a primary occupational exposure limit.*
>
> *The use of 5 rem (0.05 Sv) TEDE as the control room criterion did not imply that this value would be an acceptable exposure during emergency conditions, or that other radiation protection standards of 10 CFR part 20, including individual organ dose limits, might not apply. This criterion was provided only to assess the acceptability of design provisions for protecting control room operators under postulated DBA conditions. The DBA conditions assumed in these analyses, although credible, generally did not represent actual accident sequences but were specified as conservative surrogates to create bounding conditions for assessing the acceptability of engineered safety features."*



Regulatory Basis Alternatives

Regulatory Basis document assessed three Alternatives. (88 FR 61986)

- *Alternative 1: No Action*
- *Alternative 2: Pursue Rulemaking to Amend the Control Room Design Criteria and Update the Current Regulatory Guidance Accordingly with Revised Assumptions and Models and Continue to Maintain Appropriate and Prudent Safety Margins*
- *Alternative 3: Update the Current Regulatory Guidance with Revised Assumptions and Models and Continue to Maintain Appropriate and Prudent Safety Margins*

Public Comments

Regulatory basis document sought comments on the alternatives proposed and asked two questions. (88 FR 61986)

Question 1:

Would the numerical selection of the control room design criteria be better aligned with regulations designed to limit occupational exposures during emergency conditions (e.g., 10 CFR 20.1206, "Planned special exposures," and 10 CFR 50.54(x)), or regulations designed to limit annual occupational radiation exposures during normal operations (e.g., 10 CFR 20.1201, "Occupational dose limits for adults," specifically the requirements in 10 CFR 20.1201 (a)(1)(i))? Please provide a basis for your response.

Question 2:

Would a graded, risk-informed method, to demonstrate compliance with a range of acceptable control room design criterion values instead of a single selected value, such as the current 5 rem (50 mSv) TEDE, provide the necessary flexibilities for current and future nuclear technologies up to but less than 20.0 weight percent U-235 enrichment? Please provide a basis for your response.

Assessment of Regulatory Basis Alternative 2 to perform rulemaking

- <u>**Option 2A**</u>—Amend the codified numerical acceptance value from 5 rem total effective dose equivalent (TEDE) to a new single value of 10 rem TEDE, with conforming changes to guidance.
- **<u>Option 2B</u>—Amend the codified numerical acceptance value from 5 rem TEDE to a range of values from 10 to 25 rem TEDE, with a graded, risk-informed, performance-based framework in guidance.**
- <u>**Option 2C**</u>—Amend the codified numerical acceptance value from 5 rem TEDE to a new single-value 25 rem TEDE, with conforming changes to guidance.

Proposed Rulemaking Language

- High-level Changes
  - Increase from 5 rem to 10 rem TEDE.
  - If additional operational flexibilities are needed beyond 10 rem TEDE, facility-specific risk profile or information can be leveraged to justify a higher numerical value up to 25 rem TEDE with is provided in DG-1425 (RG 1.183 Rev. 2).
  - Clarify the purpose of the control room design criteria and distinguish it from the radiation protection and emergency preparedness frameworks.
  - Consistence with other regulations containing either dose-based design criteria or radiation exposure limits.
  - DG-1425 adopts a method that develops a framework for a graded, risk-informed, and performance-based control room design criterion. Approach is consistent with SECY 98-144.

Proposed Rulemaking Language

Example of 10 CFR 50.67(b)(2)(iii) proposed language:

"(iii) **The necessary design, fabrication, construction, testing, and performance criteria for structures, systems, and components important to safety are** provided to permit occupancy of the control room under accident conditions without calculated radiation exposures in excess of **0.10 Sv (10 rem)** total effective dose equivalent (TEDE) **or a higher design criterion limit established in accordance with paragraph (b)(3) of this section** for the duration of the accident.

(3) The licensee may establish a design criterion limit higher than 0.10 Sv (10 rem) total effective dose equivalent (TEDE) but not greater than 0.25 Sv (25 rem) TEDE for compliance with paragraph (b)(2)(iii) of this section provided the licensee demonstrates that the specified limit is consistent with the plant risk-profile or commensurate with the risk of the plant.

Approach for Rulemaking for the Control Room Design Criteria

Policy and Regulation

- Flexibility within Commission Policy
- Radiation Protection and Emergency Response Framework

Evidence-based justifications

- Scientific Recommendations
- Radiation protection and radiation epidemiology

Ability to Provide Reasonable Regulatory Relief

- Reduce regulatory burden while maintaining safety and compliance
- Risk-informed and performance-based Rulemaking

Policy and Regulation – Flexibility within Commission Policy

- 10 CFR Part 20 puts into practice recommendations from the ICRP and NCRP. (56 FR 23360; May 21, 1991)
    - ICRP Publication 26, Recommendations of the ICRP (ICRP, 1977) subsequent ICRP publications.
    - NCRP Report No. 91, Recommendations on Limits for Exposure to Ionizing Radiation. (NCRP, 1987)

- From ICRP 26
    - Occupational exposure limit set to limit stochastic effects and prevent deterministic effects.
        - 5 rem/yr dose-equivalent to limit stochastic effects to an acceptable level.
        - 50 rem/yr dose-equivalent to all tissues except the lens to prevent deterministic effects.

- Both ICRP and Part 20 provide flexibility for planed special exposures.
    - ICRP proposal would have permitted a 15-rem dose in 1 year.
    - Part 20 condition theoretically possible to get a 10-rem dose in 1 year.
        - Concluded that an infrequent exposure of workers up to twice the occupational dose limit was adequately protective of radiation workers.

Finding: The system of dose limitation, as adopted by the Commission in 10 CFR Part 20, provides flexibility when considering risk-informing the dose-based control room design criteria.

Policy and Regulation – Radiation Protection and Emergency Response Framework

10 CFR Part 20, *Standards for Protection Against Radiation.*

10 CFR Part 50.47, *Emergency plans*.

Appendix E to Part 50—*Emergency Planning and Preparedness for Production and Utilization Facilities.*

10 CFR Part 55, *Operators' Licenses.*

10 CFR Part 50.54, *Conditions of licenses.*

10 CFR Part 50.155, *Mitigation of beyond-design-basis events.*

Finding: a range of regulatory dose-based occupational expose limits and design/siting criteria up to 25 rem TEDE. Understanding of their basis, purpose, and application helped inform IE rulemaking proposal.

Policy and Regulation – Radiation Protection and Emergency Response Framework (Cont.)

Original GDC-19 (1971)	• At the time that GDC-19 was established in 1971, 10 CFR Part 20 limited occupational radiation exposure to 3 rem (0.03 Sv) whole body dose per calendar quarter, provided the total lifetime dose was verified not to exceed 5 rem (0.05 Sv) times the individual's age in years minus 18. Thus, a worker could receive a radiation exposure of up to 12 rem (0.12 Sv) in a given year.
10 CFR Part 20.1201, "Occupational dose limits for adults,"	• An adult worker could receive occupational radiation exposure of up to 10 rem (0.10 Sv) TEDE over a 12-month period straddling two calendar years.
10 CFR Part 20.1206, "Planned special exposures,"	• Permits an adult worker to receive doses in addition to, and accounted for separately from, the doses received under the limits specified in 10 CFR 20.1201 of five times the annual dose limits during the individual's lifetime, not to accumulate faster than 5 rem (0.05 Sv) TEDE in any one year. As such, an adult worker could receive radiation exposure of up to 10 rem (0.10 Sv) TEDE within a single calendar year period.

Policy and Regulation – Radiation Protection and Emergency Response Framework (Cont.)

10 CFR 50.47(b)(11)

- Establish the means for controlling radiological exposures in an emergency and states that the means for controlling radiological exposures must include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protection Action Guides (PAG).
  - EPA PAG Manual, recommends that doses received under emergency conditions should be maintained ALARA and, to the extent practicable, limited to 5 rem (0.05 Sv).
  - The guideline for actions to protect valuable property is 10 rem (0.10 Sv) where a lower dose is not practicable, the guideline for actions to save a life or to protect large populations is 25 rem (0.25 Sv) where a lower dose is not practicable, and exposures greater than 25 rem (0.25 Sv) may be appropriate for lifesaving or protecting large populations if the workers are volunteers who are fully aware of the risks involved.

Policy and Regulation – Radiation Protection and Emergency Response Framework (Cont.)

10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance";

10 CFR 50.34, "Contents of applications; technical information";

10 CFR 50.67; and

10 CFR part 52.

- The upper range of the proposed numerical values would be consistent with the Commission's use of the 25 rem (0.25 Sv) TEDE limit primarily in regulations for power reactor siting to protect the public during emergencies.

- As discussed in the preamble for the final rule updating the NRC's siting criteria (61 FR 65159; December 11, 1996),

  "the Commission's use of 25 rem (0.25 Sv) TEDE does not imply that it considers it to be an acceptable limit for an emergency dose to the public under accident conditions, but only that it represents a reference value to be used for evaluating plant features and site characteristics intended to mitigate the radiological consequences of accidents in order to provide assurance of low risk to the public under postulated accidents."

Evidence-based justification – Scientific Recommendations

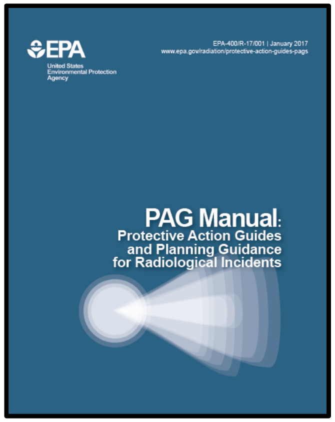
- Reviewed several source materials to understand the current recommendations from national and international organizations responsible for making recommendations for radiation protection standards.

- Purpose of this review was to determine whether reexamining the scientific and technical basis for the numerical value of the control room design criteria would be warranted.

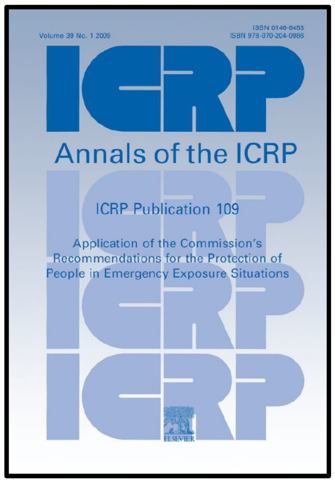
Finding - a range of international and national organization-based recommendations for radiation exposures for radiation workers under normal and emergency conditions up to 25 rem TEDE.

Source: Brock, Et al., NRC, White Paper, "Control Room Design Criteria and Radiological Health Effects (ADAMS ML23027A059)

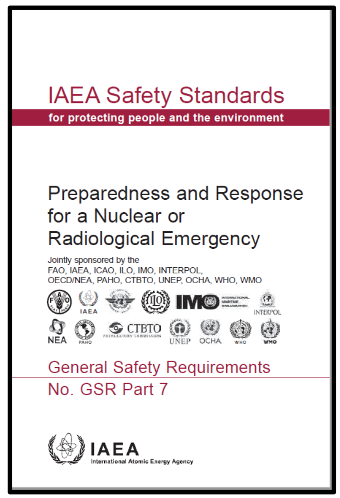
Evidence-based justification – Scientific Recommendations



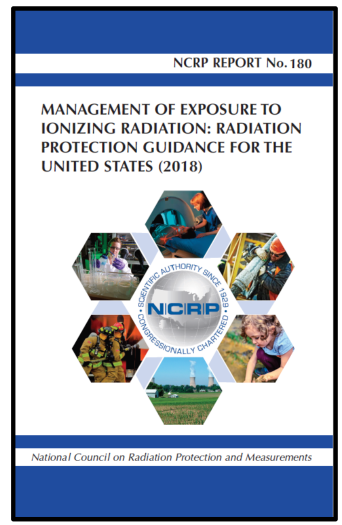
25 rem for lifesaving or protection of large populations



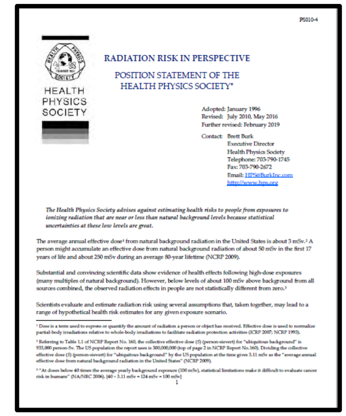
2–10 rem acute for emergency exposure situations



10-50 rem, 100 rad, acute depending on the severity of the actions needed



50 rad to 10 rem, depending on actions needed



Substantial and convincing scientific evidence of health effects following high-dose exposures. However, below levels of about 10 rem above background from all sources combined, the observed radiation effects in people are not statistically different from zero.

Evidence-based justification – Radiation protection and radiation epidemiology

- Proposed rule recommendations are firmly founded on modern radiation protection- and health effects knowledge.

- Deterministic Health Effects (rad)
  - Significantly below the threshold for observable deterministic health affects such as acute radiation syndrome and hematopoietic syndrome which occurs at doses around 70 to 100 rad respectively.
  - Far below the mean lethal dose of ionizing radiation without medical treatment, estimates to be approximately 300 to 500 rad.
  - Part 20 exposure limit set low enough to protect against deterministic effects.

- Stocastic Health Effects (rem)
  - Far below individual estimates of radiation risks for cancer mortality given the relatively short time frame exposures would be incurred.
  - Radiation protection and emergency response programs would actively monitor and manage occupational exposure before the 10 CFR Part 20 limit.
  - Traditional DBA radiological consequence analysis will continue to provide additional defense-in-depth from the 10 CFR Part 20 occupational exposure limits.
  - Continues to ensures a high level of protection is still provided, minizmizing long-term health impacts.

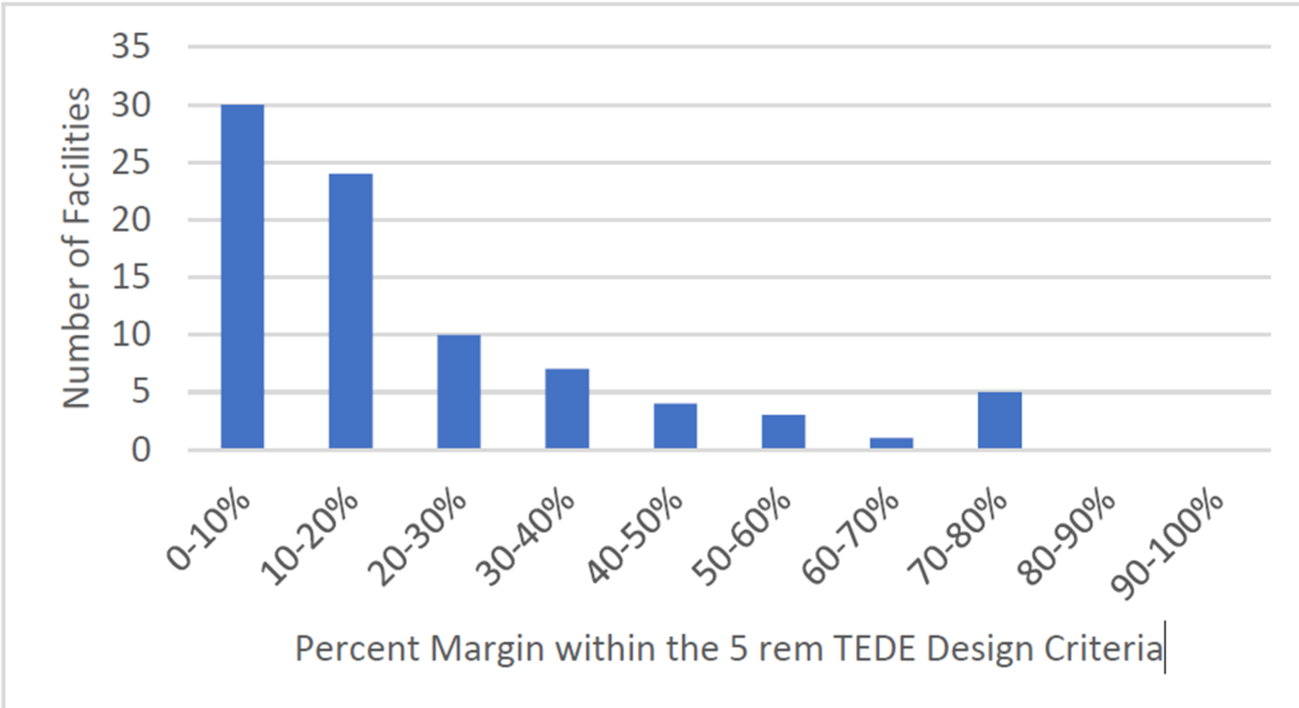
Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance

- A very low design criteria value can result in an excessive amount of maintenance, leading to avoidable occupational exposure and unnecessary operational disturbances. Conversely, a very high value may allow for unacceptable degradation, potentially compromising overall safety and performance over time.

- Considerations of Control Room Design Criteria Impact is multifaceted:
  - Maturity of the regulated industry and compliance infrastructure.
  - Designer margin and operational flexibilities.
  - Maintenance activities and controlling actual operational exposure.
  - 10 CFR 50.59 and low safety-significant licensing actions.

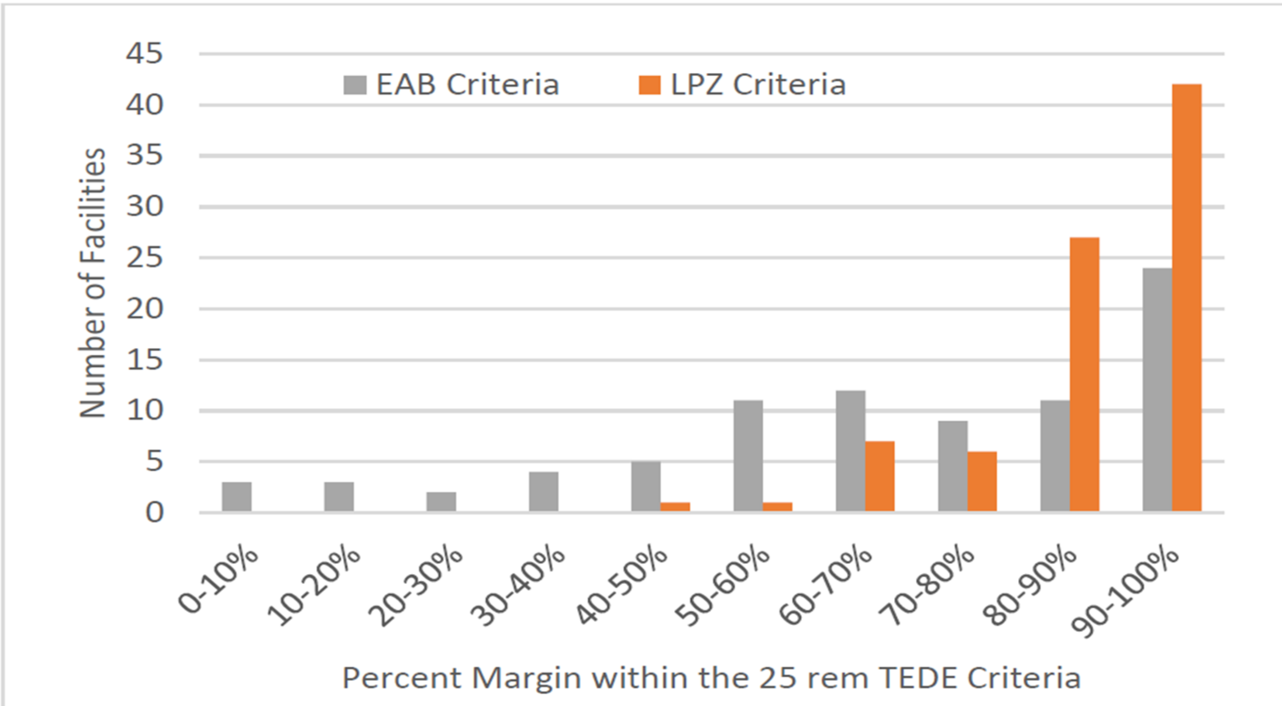
Finding: Adequate protection of public health and safety and occupational radiological safety can still be achieved at a higher, but still safe, control room design criteria performance level while balancing both dose-savings to workers and providing some regulatory relief to maintain operational flexibilities.

Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance



Histogram of Licensees' Analysis of Record DBA Result Percent-Margin to Control Room 5 rem TEDE

Ability to Provide Reasonable Regulatory Relief – Reduce regulatory burden while maintaining safety and compliance



Histogram of Licensees' Analysis of Record DBA Result
Percent-Margin to Control Room 5 rem TEDE

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

- Consider Commission-directed PRA-related policies which advocate certain changes to the development and implementation of its regulations using risk-informed, and ultimately performance based, approaches.
  - 1985 Severe Reactor Accident policy statement (50 FR 32138; August 8, 1985)
  - PRA Policy Statement (60 FR 42622, August 16, 1995)
  - **SRM-SECY-98-144, “Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulations” (ADAMS Accession No. ML003753601)**
- SRM-SECY-98-144 defines the terms and Commission expectations for risk-informed and performance-based regulation. Item 8, “Risk-Informed, Performance-Based Approach,” reads as follows:

  “… Stated succinctly, a risk-informed, performance-based regulation is an approach in which risk insights, engineering analysis and judgment including the principle of defense-in-depth and the incorporation of safety margins, and performance history are used, to (1) focus attention on the most important activities, ***(2) establish objective criteria for evaluating performance, (3) develop measurable or calculable parameters for monitoring system and licensee performance, (4) provide flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes,*** *and (5) focus on the results as the primary basis for regulatory decision-making.”*

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Break SRM-SECY-98-144 into Rulemaking- and Guidance Elements…

Element 1 – **Rulemaking** to design a rule which *establishes objective criteria for evaluating performance with developed measurable or calculable parameters for monitoring system and licensee performance.*

Element 2 – **Guidance** to design a regulatory framework which *provides flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes.*

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Element 1

- Increase control room design criteria by a factor of 2, from 5 rem to 10 rem TEDE and range up to 25 rem TEDE with consideration of the plant-specific risk profile c risk information.
  - Numerical values are risk-informed based:
    - Commission Policy and Regulations for infrequent- and emergency exposures.
    - Recommendations from national and international and organizations responsible for radiation protection standards and guidance.
    - modern radiation protection- and epidemiology.
  - Performance-based aspect of rule retained within guidance which historically requires traditional DBA radiological consequence analyses thereby retaining staff's experience and licensee's licensing basis.
  - Flexibility and scalability incentivizes safety improvements if additional margin is needed beyond 10 rem TEDE where a lower facility-specific risk-metric allows for a higher and still safe performance criteria.

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

Element 2

- Establish in guidance, a framework for a graded, risk-informed and performance-based control room design criterion.
  - Enables a performance-based evaluation using traditional deterministic radiological consequence analysis methods within defined risk informed boundaries.
  - Boundaries are defined by acceptable radiation exposure guidelines for radiation workers during accident and emergency conditions and acceptable contemporary nuclear facility risk profiles using modern probabilistic risk assessment methods.
  - Provide flexibility when determining how to meet an established acceptance criterion in a way that encourages and rewards safety of the facility.
  - In practice, the method produces a framework that leverages in part, its safe design and operations to justify a higher control room design criterion with a lower plant-specific risk metric.
- Considerations:
  - Simple to understand and use.
  - Leverages well-known and understood methods and analyses.
  - Similar to other graded regulatory methods such as RG 1.174, SRP-specific DBA dose-based acceptance criteria, Frequency-consequence curves.

Source: Dickson, E., NRC, internal memorandum to K. Hsueh, “Method for a Graded Risk-Informed Performance-Based Control Room Design Criteria Framework,” Washington, DC, July 2024 (ML24212A254).c

Ability to Provide Reasonable Regulatory Relief – Risk-informed and performance-based Rulemaking

DG-1425 (RG 1.183 Rev. 2) Graded, Risk-informed and Performance-based Framework

- Risk-Metric Range: Regulatory Guide 1.174 CDF Criteria within five-bins.
- Criteria Range: 10 to 25 rem TEDE.
- Like similar licensing actions, approved higher licensing basis criteria is a “snapshot” in time until the licensee needs approval for an amendment (e.g. 50.59 criteria).
- Addressing framework defense-in-depth, safety margin, and uncertainty through license submittals.

Table 8. Guidelines for Control Room Location Based on a Graded, Risk-Informed, and Performance-Based Framework

Overall CDF	Graded Control Room Design Criteria (rem-TEDE)
CDF ≤ 1.E-5	25
1E-5 < CDF ≤ 5E-5	20
5E-5 < CDF ≤ 1E-4	15
CDF > 1E-4; or licensee not adopting the graded framework to determine acceptance criteria	10

Reference: DG-1425, Table 8

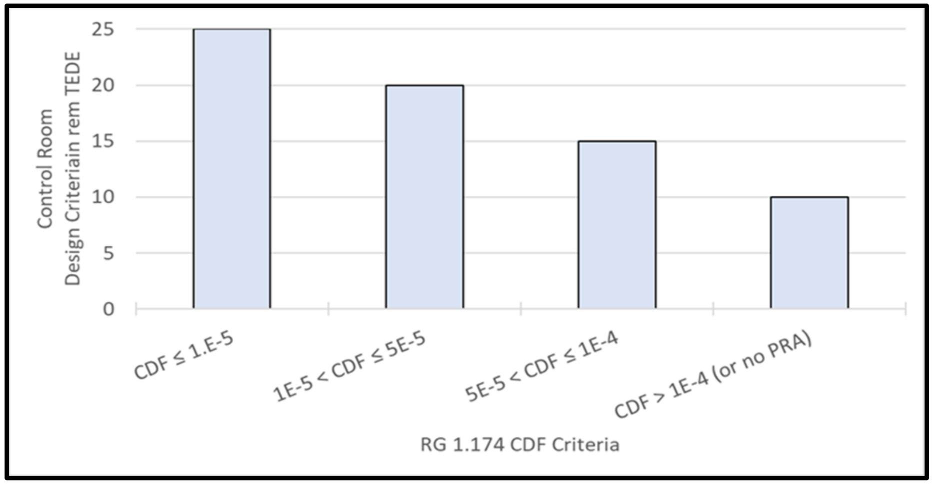


Figure representing DG-1425, Table 8 Graded, Risk-Informed and Performance-Based Framework

Using risk-information as a sliding scale to identify a higher control room design criterion.

Recap: Approach for Rulemaking for the Control Room Design Criteria

Policy and Regulation

- Flexibility within Commission Policy
- Radiation Protection and Emergency Response Framework

Evidence-based justifications

- Scientific Recommendations
- Radiation protection and radiation epidemiology

Ability to Provide Reasonable Regulatory Relief

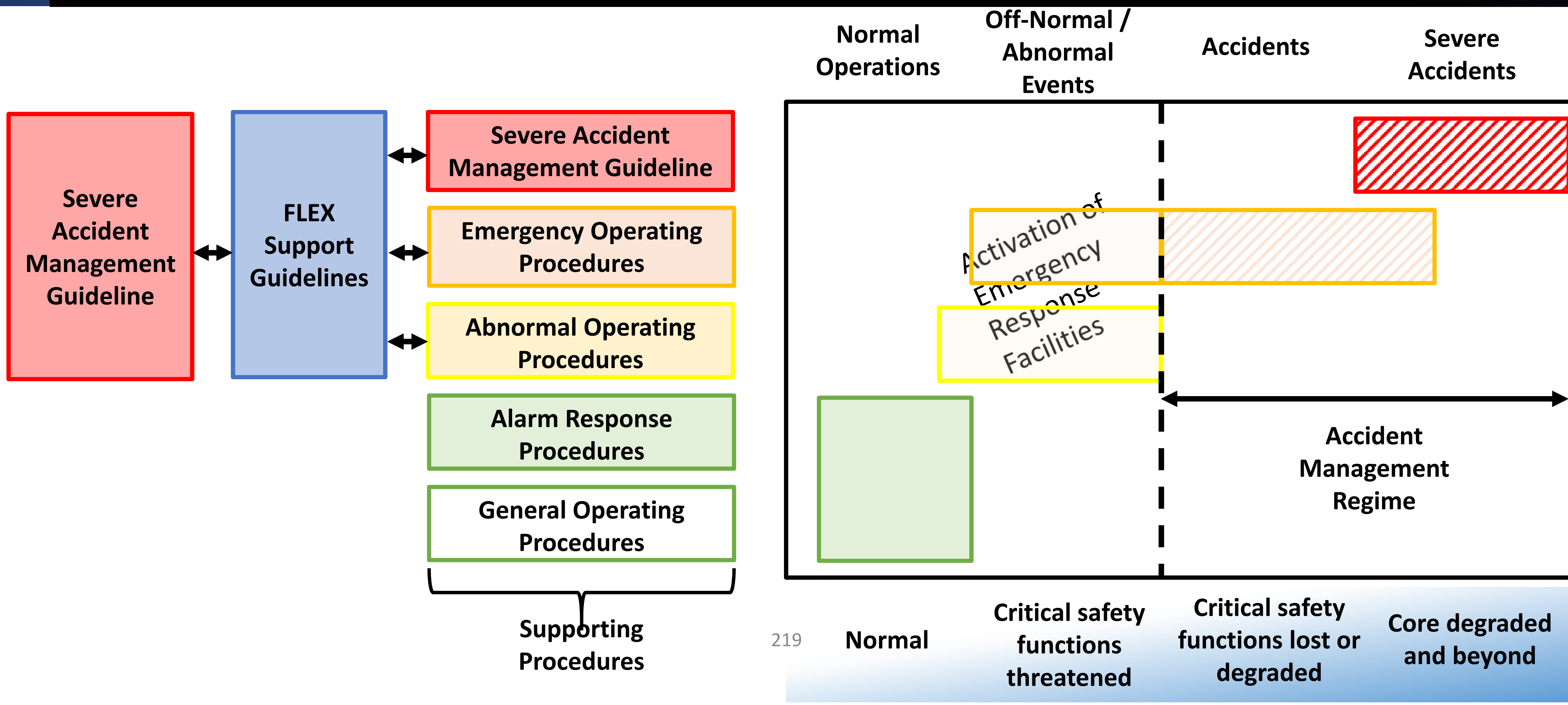
- Reduce regulatory burden while maintaining safety and compliance
- Risk-informed and performance-based Rulemaking

Closing

- The NRC recognizes the challenges that licensees face to retain margin within their licensing bases for the purposes of operational flexibility and the small amount of margin to the control room design criteria itself.
- The key driver behind the proposal to amend the control room design criteria is to facilitate increased regulatory efficiency and consistency while continuing to provide adequate protection of public health and safety.
- Proposed rule and supporting regulatory guidance executes Commission SRM-SECY-98-144 defines the terms and Commission expectations for risk-informed and performance-based regulation.
- This rulemaking effort would also support increased consistency within the Commission's comprehensive radiation protection and emergency planning frameworks.
- Consistency among these regulations, 10 CFR 50.67 and GDC 19 would provide operational flexibilities to further limit actual occupational exposures while also realigning the numerical value as a design criterion with a potential reactor accident of exceedingly low probability.

Backup Slides

Policy and Regulation –
Radiation Protection and Emergency Response Framework (Cont.)



Class 9 Accidents

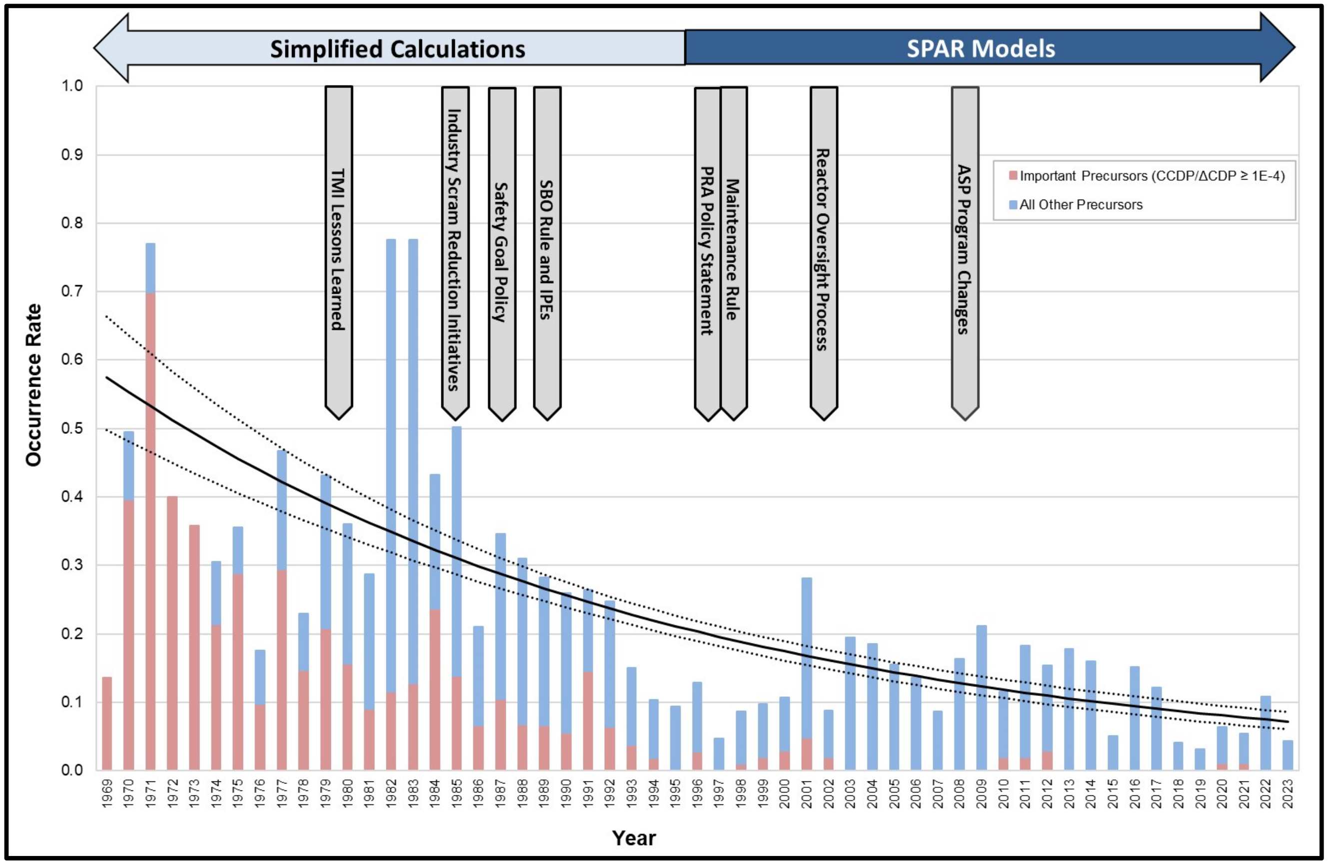
NRC Assessments of Severe Accident Risk

- 10 CFR Part 51, *Environmental Protection Regulations for Domestic Licensing and Regulatory Functions,* implement the National Environmental Policy Act (NEPA).
  - Consideration of the costs and benefits of severe accident mitigation/design alternatives and the bases for not incorporating severe accident mitigation design alternatives.
- 10 CFR Part 54, *Requirements for Renewal of Operating Licenses for Nuclear Power Plants*, requires an application to include a supplement to the environmental report that complies with the requirements of Subpart A of 10 CFR Part 51.

Severe Accident Events of Western Designs

- Three Mile Island
  - Maximum whole body dose received by an actual individual during the TMI accident in March 1979, which involved major core damage, was estimated to be about 0.1 rem. (61 FR 65159; December 11, 1996)
- Fukushima Daiichi
  - Maximum effective dose was 67.9 rem, six received effective doses greater than 25 rem, and 168 workers between 10 to 25 rem. (UNSCEAR, 2021)
  - Internal contamination was attributed to the severe working conditions and the inadequate implementation of protective measures (e.g. improper use of respiratory protection, iodine thyroid blocking measures, actions that resulted in inadvertent ingestion of radionuclides), due primarily to the lack, or ineffectiveness, of training. (IAEA, 2014)

Accident Sequence Precursor Program



Source: *U.S. Nuclear Regulatory Commission Accident Sequence Precursor Program Summary Description*, Appendix B: *Historical Precursor Occurrence Rates* (ML24177A020)