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

February 5, 2025

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 ≤ 5.0 weight percent uranium-235; however, technology developments may require numerous exemptions to utilize fuel enriched above 5.0 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.

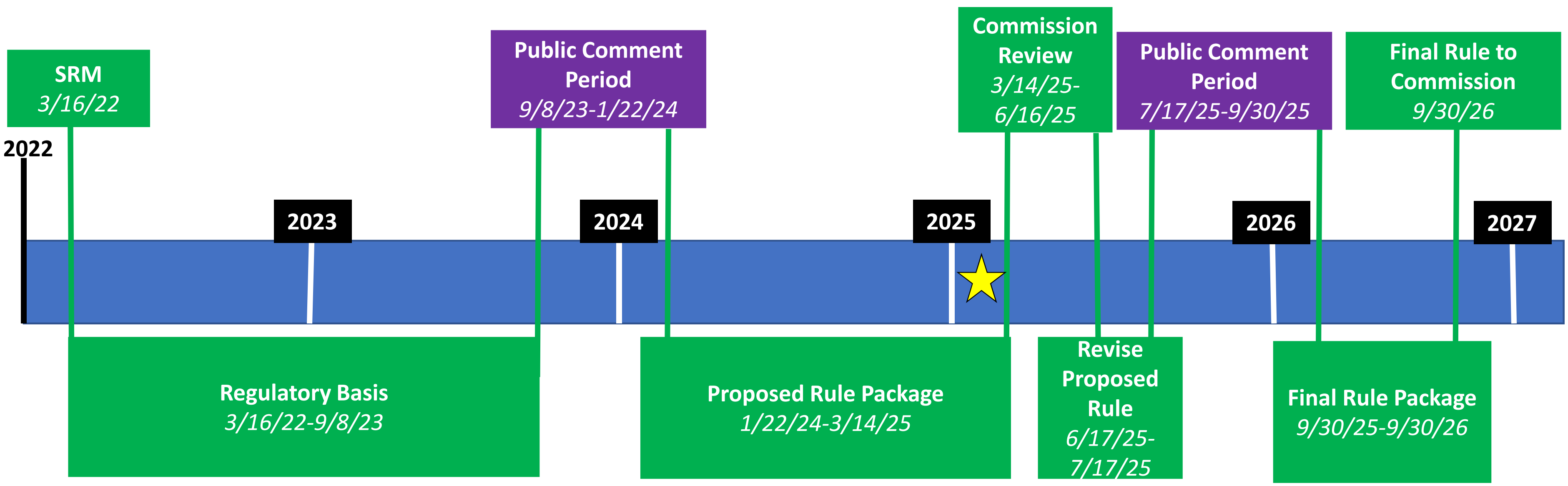
SRM-SECY-21-0109 Overview

- 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)
  - After 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.

NRC Staff Presenters

- **Charley Peabody, NRR:**
  - Criticality Accident Requirements (10 CFR 50.68)
- **Jason Piotter, NMSS:**
  - General Requirements for Fissile Material Packages (10 CFR 71.55)
- **Elijah Dickson, NRR:**
  - Control Room Requirements (10 CFR 50.67 and GDC-19)
- **Joseph Messina, NRR:**
  - Fuel Fragmentation, Relocation, and Dispersal (10 CFR 50.46a)

Criticality Accident Requirements of 10 CFR 50.68

Charley Peabody
Nuclear Systems Performance Branch
NRR

Criticality Accident Requirements of 10 CFR 50.68

- This rulemaking would amend the current 5.0 weight percent U-235 limit in 50.68(b)(7) and allow for an alternative between the existing 5.0 weight percent U-235, or a plant-specific criticality safety limit based on the limit specified in a licensee's or applicant's operating license.

- Licensees would be allowed to increase enriched fuels above 5.0 weight percent as long as this increased enrichment level is approved specifically in their technical specifications design features or equivalent part of the operating license as a part of a fuel transition license amendment request.

Questions?

Packaging Requirements of 10 CFR 71.55

Jason Piotter

New Fuels Team

NMSS

Packaging Requirements of 10 CFR 71.55

- Current transportation regulations are adequate to certify UF6 transportation packages with material enriched up to 20.0 weight percent U-235. (10 CFR 71.55(b), 71.55(c), 71.55(g)).

- 10 CFR 71.55(g), specific to UF6 transportation packages, is an exception to 71.55(b), which requires the consideration of moderator when performing criticality calculations. For UF6 packages with enrichment levels up to 5.0 weight percent U-235 certified under 71.55(g), moderator does not have to be considered in criticality calculations.

Packaging Requirements of 10 CFR 71.55

- This rulemaking would amend 10 CFR 71.55(g) to allow the current exception of UF_6 enriched up to 5.0 weight percent U-235 to expand to 10.0 weight percent U-235. This amended rule would require a defense-in-depth design feature for those packages containing UF_6 enriched between 5.0 and 10.0 weight percent U-235.

Questions?

Control Room Design Criterion of 10 CFR 50.67 and GDC-19

Elijah Dickson
Radiation Protection and Consequence Branch
NRR

Control Room Design Criterion of 10 CFR 50.67 and GDC-19: Summary of Regulatory Issue

- This rulemaking would amend the control room design criteria from the current 5 rem (0.05 Sv) total effective dose equivalent (TEDE) to a revised value of 10 rem (0.10 Sv) TEDE; the value may range up to 25 rem (0.25 Sv) TEDE with consideration of the plant-specific risk profile or risk information.

- The amended rule, and subsequent guidance, would align with Commission direction provided in SRM-SECY-98-144 to take a risk-informed, performance-based approach to regulations and guidance.

Questions?

Fuel Fragmentation, Relocation, and Dispersal

Joseph Messina
Nuclear Methods and Fuel Analysis
NRR

Fuel Fragmentation, Relocation, and Dispersal

- This rulemaking would enable entities to voluntarily recategorize large-break loss-of-coolant accidents (LOCA) as beyond design basis accidents, leveraging the previous 50.46a rulemaking, which was delivered to the Commission as a draft final rule in 2010, but rescinded due to Fukushima and a lack of industry interest.
- This rulemaking would divide the current spectrum of LOCA into two regions delineated by a transition break size (TBS). The smaller region (breaks up to the TBS) would be treated same as all breaks under the current 10 CFR 50.46 emergency core cooling system (ECCS) rules. The larger region (breaks greater than TBS) would be allowed to be analyzed using best-estimate modeling and more realistic assumptions based on their lower likelihood of occurrence.

Fuel Dispersal

- While the wording is not significantly different regarding coolability than 50.46, the NRC staff added a discussion in the Federal Register Notice (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.
- True best-estimate modeling and realistic assumptions are expected to significantly reduce or eliminate the calculated potential for fuel dispersal
  - DG-1434 provides guidance on 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 facilitate option of these alternatives (including a less prescriptive interpretation of core coolability)

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

Changes Made Since January ACRS Subcommittee Meeting

Weld inspections:

50.46a(b)(3) requirement to inspect 10% of similar metal welds on piping larger than the TBS has been replaced with "an NRC-approved sampling of similar metal welds."

Allowing operating reactors to define their own TBS:

Transition break size (TBS) 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, **or a plant-specific alternative break area**. 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.

Clarification on alternative approaches:

Added clarification in the Preamble and SECY paper that the performance-based view of coolability in 50.46a(f) and the fact that fuel dispersal is not necessarily incompatible with coolability can facilitate alternative approaches to addressing FFRD. The staff plans to continue engaging with industry on other approaches (e.g. modified Alternatives 4 and 5) via licensing interactions and workshops.

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

Proposed 50.46a Highlighted Requirements

The primary requirements of 50.46a are:

- Weld inspections for similar metal welds on piping > the TBS
- Evaluation of plant-specific applicability of the TBS
- Evaluation that changes made do not invalidate the TBS
- A risk-informed evaluation process is established to analyze changes enabled by 50.46a
  - Changes must be kept to very small risk increases (i.e., ΔCDF ≤ 1E-6/rx.yr. and ΔLERF ≤ 1E-7/rx.yr.) and the overall risk must remain small
- Principal ECCS criteria
  - Maintain fuel coolability
  - Long-term cooling
- Fuel performance criteria:
  - Address cladding degradation phenomena
  - Maintain fuel coolability
  - Avoid explosive concentration of combustible gas
  - Long-term cooling
- Breaks at or below the TBS must continue to have a *high probability* that the ECCS and fuel performance criteria are met
- Breaks above the TBS must demonstrate that ECCS and fuel performance criteria are met to at least a *best-estimate* level

50.46a Rule High-Level Highlighted Requirements

50.46a(c)(1)(i) and (c)(3)(i): Evaluate applicability of the TBS to the licensee's facility
50.46a(b)(3) and (c)(ii): Inspect an approved sampling of similar metal welds on piping > TBS

50.46a(c)(1)(iv)-(v) and (c)(3)(iii)-(iv): Risk-informed evaluation process for proposed changes made under 50.46a
50.46a(h): Acceptance criteria for changes made under 50.46a

50.46a(f)(1): fuel performance criteria. Must have NRC-approved limits that:

- i. Address cladding degradation phenomena
- ii. Maintain fuel coolability
- iii. Avoid explosive concentration of combustible gas
- iv. Ensure long-term cooling

Guidance

DG-1428, "Plant-Specific Applicability of the Transition Break Size"

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

DG-1263, Rev. 1, "Establishing Analytical Limits for Zirconium-Based Alloy Cladding"

DG-1261, Rev. 1, "Measuring Breakaway Oxidation Behavior"

DG-1262, Rev. 1, "Determining Post-Quench Ductility"

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

TBS development

Historic TBS Technical Basis:

- 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 generalize results

Recent Confirmation of the TBS Technical Basis:

NUREG-1829 Confirmation:

- Internal and External Elicitation
- Impact of Recent Operational Experience

Qualitative

- Probabilistic Fracture Mechanics Study
- International Operational Database Study

Quantitative

NUREG-1903 Confirmation:

- Evaluated three cases: unflawed and flawed piping failure and indirect piping failure by other components and component supports.
- Used most recently updated seismic hazard curves for the assessment
- For unflawed piping, failure probabilities were significantly low compared to the 1E-05 per year frequency used as a basis to establish the TBS.
- Flawed piping and indirect failure frequencies expected to be < 1E-05 per year but more comprehensive, plant-specific analysis needed to confirm.

TBS Confirmation:

- LOCA frequencies and TBS are applicable if plant specific applicability is demonstrated.
- 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.

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

Cladding Testing: DG-1261 through 1263

DG-1261, Rev. 1: Measuring Breakaway Oxidation Behavior

- NRC's LOCA program showed that minor changes in alloy composition or manufacturing processes can have significant impact on breakaway oxidation behavior
- Defines an experimental technique capable of determining the effect of composition changes or manufacturing changes on the breakaway oxidation behavior
- Discusses both initial testing and periodic confirmatory testing

DG-1262, Rev. 1: Determining Post-Quench Ductility

- Defines an experimental technique to measure the ductile-to-brittle transition for the zirconium-alloy cladding material
- 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

DG-1263, Rev. 1: Establishing Analytical Limits for Zirconium-Based Alloy Cladding

- 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.

“True Best-Estimate”

- LOCAs above the TBS must be analyzed to at least a “true best-estimate” level.
- Consistent with what is permitted in other beyond DBAs, such as ATWS and SBO.
- The NRC staff specified in the Preamble of the proposed rule FRN that “true best estimate” analyses are based on nominal inputs, without conservative biases, and without adding uncertainties.
- The NRC staff plans to align with industry on a definition in workshops in the final rule phase.

Questions?

Backup Slides

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 a Risk-Informed Evaluation Process 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”

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