# Public Workshop (3 of 3)

## Update to RG 1.183, Revision 1 – Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors

March 8, 2024

ADAMS Accession No. ML24066A177

# Introductions

Please ensure to identify yourself before speaking throughout the meeting.

# NRC Opening Remarks

Michael Franovich, Director

Division of Risk Assessment
Office of Nuclear Reactor Regulation

# Workshop Agenda

- Workshop Format
  - Breaks: Approximately every 1 – 1.5 hours
  - Discussion: Encouraged throughout the workshop

- Project Schedule

- Generalized Framework for Developing a Graded, Risk-Informed Method for the Control Room Design Criteria

- Non-LOCA Release Fraction Calculations

- External Presentations (as applicable)

**86 FR 14964** – NRC Policy on Enhancing Participation in NRC Public Meetings

Comment-Gathering Meeting – Purpose is for the **NRC to obtain feedback** on regulatory issues and NRC actions – in this case for an update to RG 1.183

Focused on allowing participants **opportunity** to provide opinions, perspectives and feedback

**NRC staff will take notes** during meeting and use these notes to inform decision-making

**Meeting summaries** will include high-level summaries of notes NRC staff has taken

# Workshop Format

# RG 1.183 Update Stakeholder Involvement

- Public Workshops and Information Meetings
  - 3 Workshops
  - 2 Information Meetings
- Contact Project Lead
- Meeting Feedback
- ACRS Meeting – approximately November 2024
- Draft Guide Comment Period – approximately June 2025
  - Relationship with Proposed Increased Enrichment Rulemaking
    - Docket ID NRC-2020-0034, www.regulations.gov
    - Comment period on Proposed Increased Enrichment Rulemaking Regulatory Basis ended on January 22, 2024

# Project Schedule

| Milestone | Tentative Completion |
|---|---|
| Public Workshops* and Information Meetings** | January 2024 – May 2024 |
| DG Internal Review begins | June 2024 |
| DG Internal Review completed | October 2024 |
| Pre-Decisional DG Publicly Available to Support ACRS Briefings | November 2024 |
| ACRS Briefings (Staff to respond as needed) | November 2024 |
| Increased Enrichment Proposed Rule package to Commission (will include DG referenced in SECY paper) | December 2024 |
| DG Available for Public Comment via Proposed IAW Increased Enrichment Rulemaking SECY | June 2025 (Estimated) |

* The term "Workshop" means a Comment-Gathering Meeting as described in the NRC's policy statement on public meetings (see 86 FR 14964)

** The term "Information Meeting" means NRC will inform attendees and allow questions (see 86 FR 14964)

# Generalized Framework for Developing a Graded Risk-informed Method for the Control Room Design Criteria

Elijah Dickson, Ph.D.
Sr. Reliability and Risk Analyst

Division of Risk Assessment
Office of Nuclear Reactor Regulation

# Overview

BACKGROUND

METHOD

ANALYSIS

METHODOLOGY ANALYSIS AND RESULTS

# Background

- Consider Commission-directed PRA-related policies, which provide direction on 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), March 1, 1999
  - SECY-19-0036, “Application of the Single Failure Criterion to NuScale Power LLC's Inadvertent Actuation Block Valves”, July 2, 2019
  - NRC memorandum to the Executive Director for Operations , “Implementing Commission Direction on Applying Risk-Informed Principles in Regulatory Decision Making,” (ADAMS Accession No. ML19319C832), November 18, 2019
- Present a generalized method to develop a framework for a graded risk-informed, performance-based acceptance criteria.
  - Purpose would be to enable a clear deterministic evaluation using traditional deterministic radiological consequence analyses within defined risk-informed boundaries.
- Assess the method through several examples using various risk- and deterministic metrics commonly used in nuclear risk analyses.

# Method

- Method: systematic mapping of a predetermined range of acceptable dose-based control room *design* values onto a specified risk-metric.
- Range of acceptable *design* values would be defined based on an assessment of regulatory precedence and organizations responsible for making radiation protection recommendations.
- Risk-metrics would be defined based on an assessment of those commonly used in nuclear risk analysis.
- In practice, a lower plant-specific risk metric would justify a higher control room design criterion.

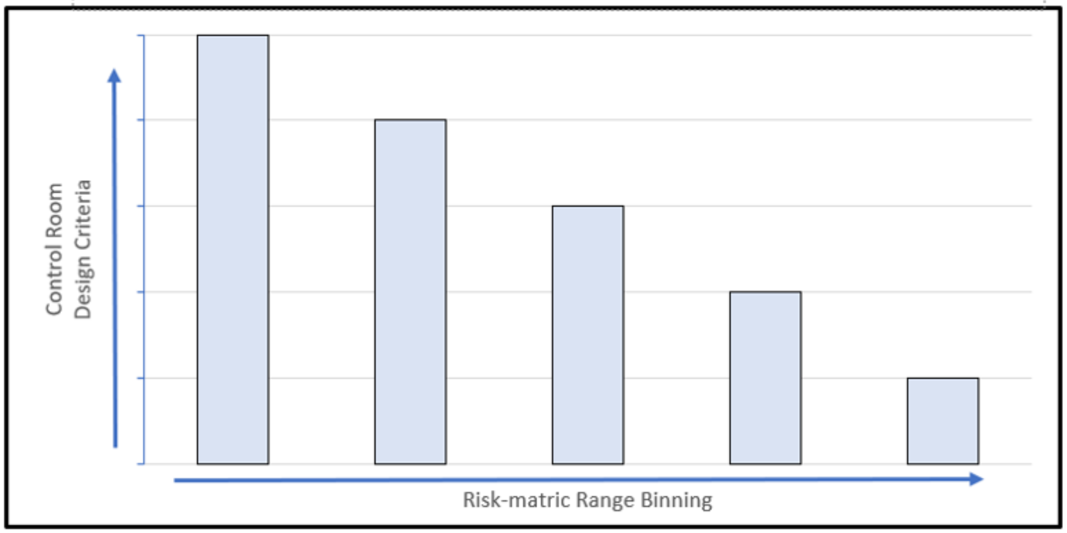


Figure 1: Example of Method Leveraging a Graded Risk-Informed Control Room Design Criteria Framework

# Analysis: Risk- and Deterministic Metrics Used in Nuclear Safety Analyses

Survey and assess several risk- and deterministic metrics applied in nuclear safety analyses.

## Risk Metrics:

- Core Damage Frequency
- Large Early Release
- Develop data sets to define boundary conditions and binning:
  - 1980's CDF dataset extracted from the Individual Plant Examinations (IPEs) results.
  - Contemporary CDF and LERF datasets extracted from the NRC SPAR models and various risk-informed applications, license renewal and subsequent license renewal submittals.

## Deterministic Metrics:

- Exclusion Area Boundary
- Low Population Zone
- Develop data sets to define boundary conditions and binning:
  - EAB and LPZ datasets extracted from licensee's approved analysis of record for their initial adoption of 10 CFR 50.67.

# METHODOLOGY ANALYSIS AND RESULTS

- Developed three examples of a graded control room design criterion with a <u>TEDE criteria range being arbitrarily selected for illustration purpose only</u>.
  - Example 1: Risk-informed model based on RG 1.174 CDF criteria
  - Example 2: Risk-informed model based on IPE CDF data
  - Example 3: Deterministic model based on EAB or LPZ data
- Assess each example to understand how the framework would apply to the operating reactor fleet's plant-specific information and contemporary understanding of radiological health effects and radiation protection recommendations.

# **Example 1:** Risk-informed model based on RG 1.174 that leverages facility-specific risk information (<u>TEDE criteria range is arbitrarily selected for illustration only)</u>

Risk-Metric Range: Regulatory Guide 1.174 CDF Criteria within three-bins.

Criteria Range (**<u>Arbitrary</u>**): 5 to 10 rem TEDE

Scientific Underpinning: not necessarily consistent with HPS PS010-4, ICRP Pub. 109, NCRP Pub. 180

Table 1: Example 1 - Binning of RG 1.174 Criteria and Corresponding Design Criteria

| RG 1.174 CDF Ranges | Design Criteria |
|---|---|
| CDF < 1.E-5 | 10 |
| 1E-4 ≥ CDF ≤ 1E-5 | 7.5 |
| CDF > 1E-4 (or no PRA) | 5 |

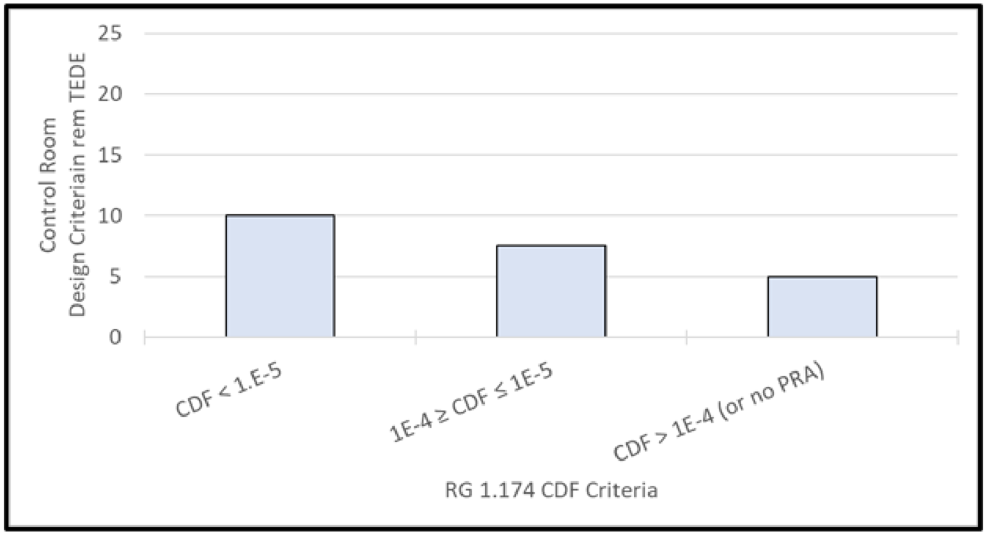



Figure 1: Example 1 - Risk-Informed, Performance-Based Approach for Control Room Design Criteria utilizing Regulatory Guide 1.174 CDF Criteria

# Example 1: Analysis and Results (TEDE criteria range is arbitrarily selected for illustration only)

Data: Develop modern CDF datasets from the NRC SPAR models and various risk-informed applications representing contemporary risk profiles form the 2020's.

Filter 2020's CDFs into each bin.

~ 57% of the facilities have plant-specific CDF results lower than 1.E-5 and would corresponding to a control room design value of 10 rem TEDE.

~ 43% of the facilities have plant specific CDF values between 1E-4 and 1E-5 and would correspond to a control room design criteria value of 7.5 rem TEDE.

Table 2: Example 1 Analysis Results of Binning CDF Data into RG 1.174 Histogram

| RG 1.174 CDF Ranges | Analysis of CDF Data Binning | |
|---|---|---|
| | CDF Data Binned | Percent of Facilities Binned |
| CDF < 1.E-5 | 55 | 59% |
| 1E-4 ≥ CDF ≤ 1E-5 | 39 | 41% |
| CDF > 1E-4 (or no PRA) | 0 | 0% |

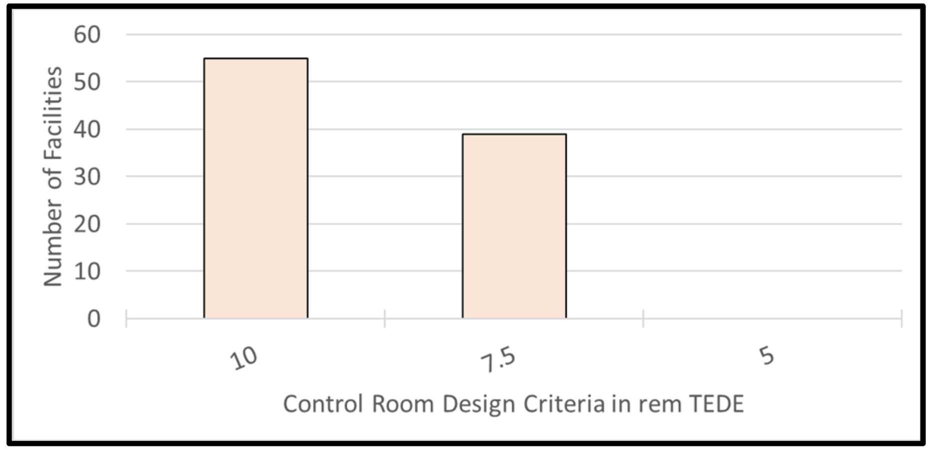



Figure 2: Example 1 Analysis Results of Binning CDF Data into RG 1.174 Histogram

# **Example 2:** Analysis and Results (TEDE criteria range is arbitrarily selected for illustration only)

Data: Develop Contemporary CDF datasets from the NRC SPAR models and various risk-informed applications to assess contemporary risk profiles form the 2020's.

Filter 2020's CDFs into each bin.

Compare 1980's and 2020 CDF data

Improving risk profiles would allow for flexible control room design criteria

~ 93% of the fleet would have a control room design criteria of 25 rem TEDE.

Table 4: Example 2 Analysis Results Binning of 2020's CDF dataset and Corresponding Design Criteria

| Design Criteria | Analysis of CDF Data Binning | | | |
|---|---|---|---|---|
| | IPE CDF Data (1980s) | Percent of Facilities Binned | Updated CDF Data (2020s) | Percent of Facilities Binned |
| 25 | 21 | 24% | 85 | 92% |
| 20 | 26 | 29% | 5 | 5% |
| 15 | 20 | 22% | 2 | 2% |
| 10 | 22 | 25% | 0 | 0% |
| 5 (or no PRA) | 0 | 0% | 0 | 0% |

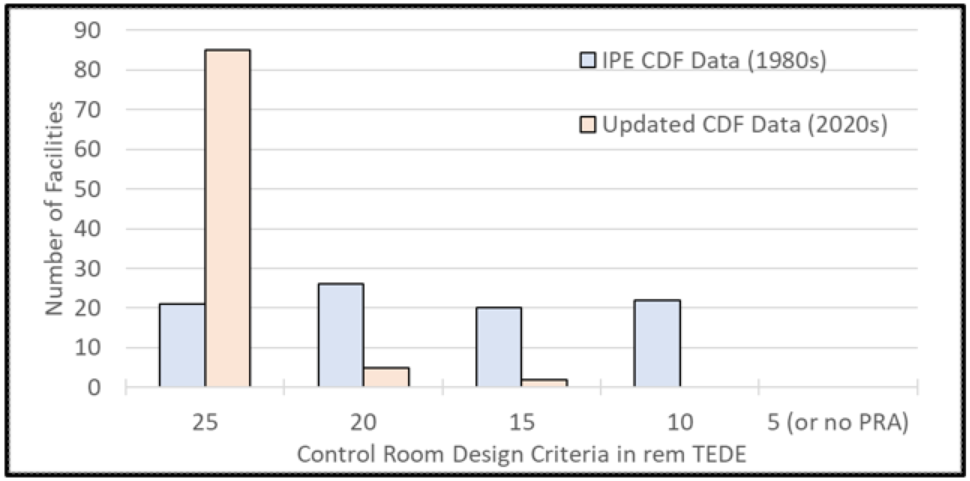



Figure 4: Example 2 Analysis Results Binning of 2020's CDF dataset and Corresponding Design Criteria

# Example 2: Risk-informed model based on IPE CDF data (TEDE criteria range is arbitrarily selected for illustration only)

Risk-Metric Range: A CDF dataset developed from IPEs results representing understood risk-profiles from the 1980's. IPE data was quartiled to define lower and upper boundaries for each of the five bins.

Criteria Range (**Arbitrary**): 5 to 25 rem TEDE

Scientific Underpinning: EPA PAG Manual, Part 100 Site Criteria

Table 3: Example 2 - Binning of 1980's CDF dataset and Corresponding Design Criteria

| **Statistic** | IPE Quantiles (Upper Bin) | CDF Bin Boundaries (IPE Data) |
|---|---|---|
| min | **8.8E-08** | CDF < 2.3E-5 |
| 25th | **2.3E-05** | 2.3E-5 ≤ CDF < 4.4E-5 |
| 50th | **4.4E-05** | 4.4E-5 ≤ CDF < 7.7E-5 |
| 75th | **7.7E-05** | 7.7E-5 ≤ CDF < 3.3E-4 |
| max | **3.3E-04** | CDF ≥ 3.3E-4 (or no PRA) |

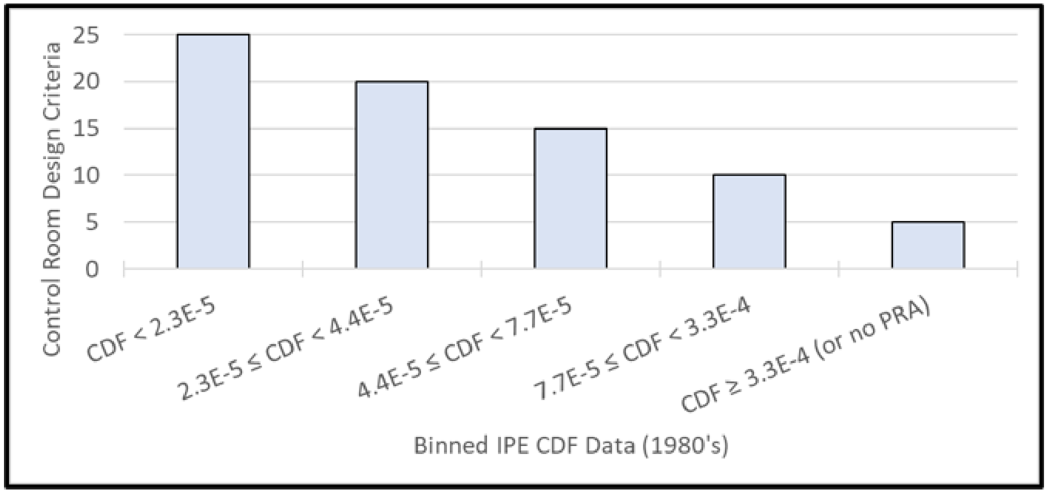



Figure 3: Example 2 - Binning of 1980's CDF dataset and Corresponding Design Criteria

# Example 3: Deterministic model based on EAB or LPZ data (TEDE criteria range is arbitrarily selected for illustration only)

Each bin boundary is defined as multiples of 5, from zero to 25. Each bin is correlated to a corresponding value 25 rem EAB siting value (can also use LPZ value, next slide)

Criteria Range (**Arbitrary**): 5 to 25 rem TEDE

Scientific Underpinning: EPA PAG Manual, Part 100 Site Criteria

Table 5: Example 3 - Performance-based Control Room Design Criteria utilizing Part 100 EAB

| EAB Criteria | Design Criteria |
|---|---|
| EAB ≤ 5 | 25 |
| 5 < EAB ≤ 10 | 20 |
| 10 < EAB ≤ 15 | 15 |
| 15 < EAB ≤ 20 | 10 |
| 20 < EAB | 5 |

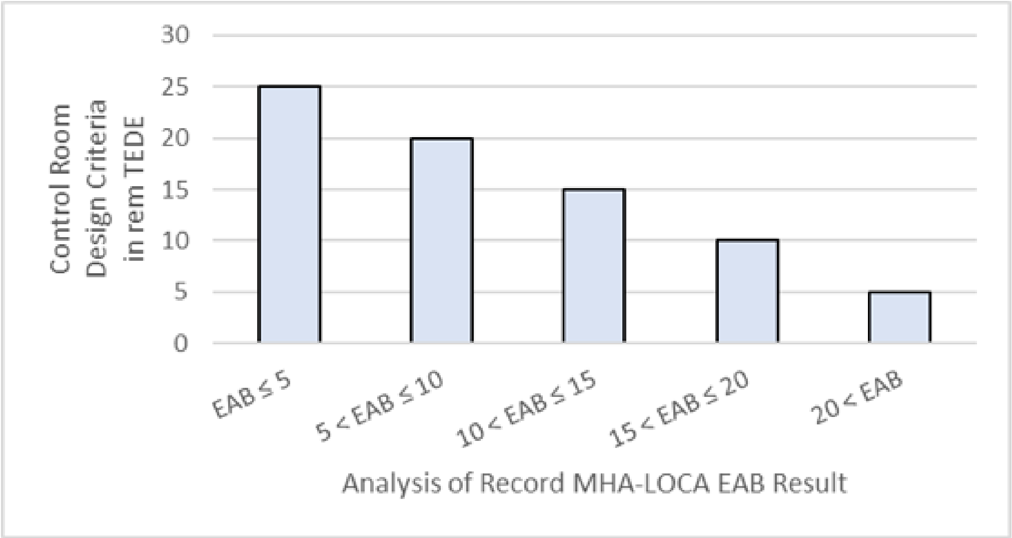



Figure 5: Example 3 - Performance-based Control Room Design Criteria utilizing Part 100 EAB

# Example 3: Analysis and Results (TEDE criteria range is arbitrarily selected for illustration only)

Data: Developed from reactor fleet's analysis of record results for the EAB and LPZ MHA-LOCA results from approved license amendment requests to adopt 10 CFR 50.67.

Filter Facility-specific EAB and LPZ data into each bin.

EAB criteria is more limiting that the LPZ criteria.

~ 42% of the facilities would obtain a control room design criteria of 25 rem with EAB model

~ 82% of the facilities would obtain a control room design criteria of 25 rem with LPZ model

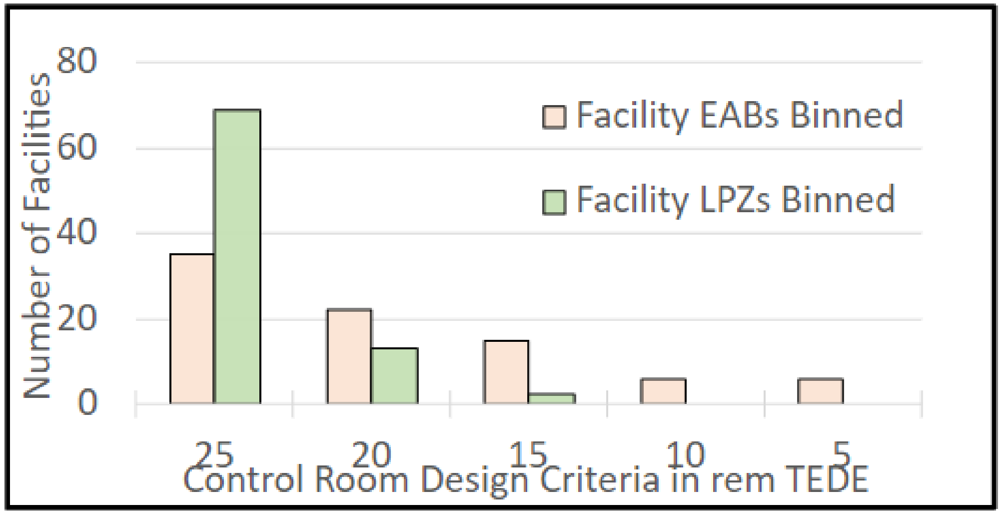



Figure 6: Example 3 Analysis Results of Performance-based Control Room Design Criteria utilizing Part 100

Table 6: Example 3 Analysis Results of Performance-based Control Room Design Criteria utilizing Part 100 EAB and LPZ Site Criteria

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

# Supporting / Background

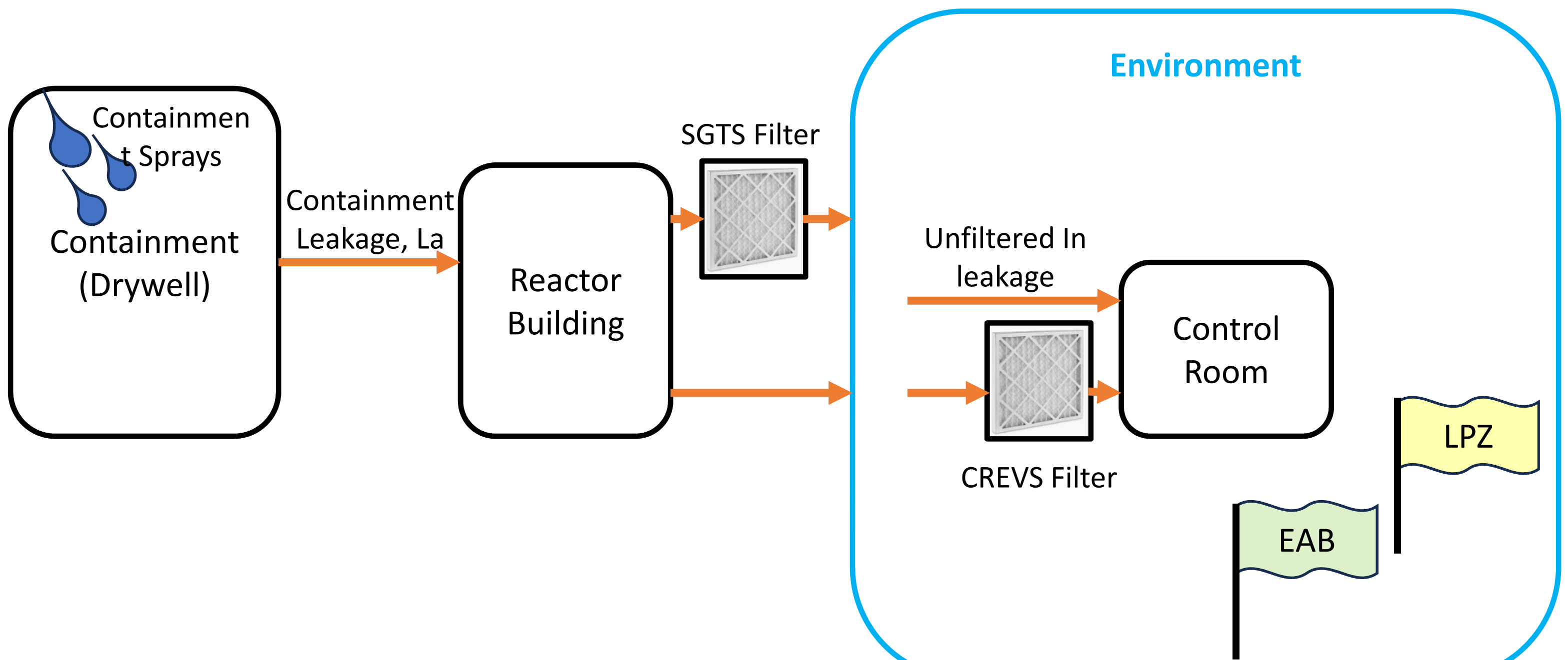
# DBAs vs. Actual Event Sequences

- The DBAs are not intended to be actual event sequences but, rather, are intended to be surrogates to enable a deterministic evaluation of the performance of plant engineered safety features, such as the control room habitability systems. (see the following three slides for illustration)
- These analyses are intentionally conservative in order to address uncertainties in accident progression, fission product transport, and atmospheric dispersion, and to provide desirable defense-in-depth.
- An actual accident sequence may not progress as modeled by a DBA (e.g., may involve multiple failures), resulting in a greater challenge to the control room systems. If the challenge exceeds the design basis, radiation exposures in the control room may exceed those envisioned in the design bases of the control room habitability features. In such an event, the facility radiation protection and emergency response programs implement measures to minimize radiation dose. (see the slide for accident management guidelines)
- The NRC's comprehensive radiation protection and emergency response regulatory framework and the conservatism in the agency's deterministic radiological consequence analysis assumptions ensures reasonable assurance that adequate protective measures can and would be taken in an actual radiological emergency.
- The range of considered acceptance criteria are well-established values based on the probability and severity of the accident being assessed.

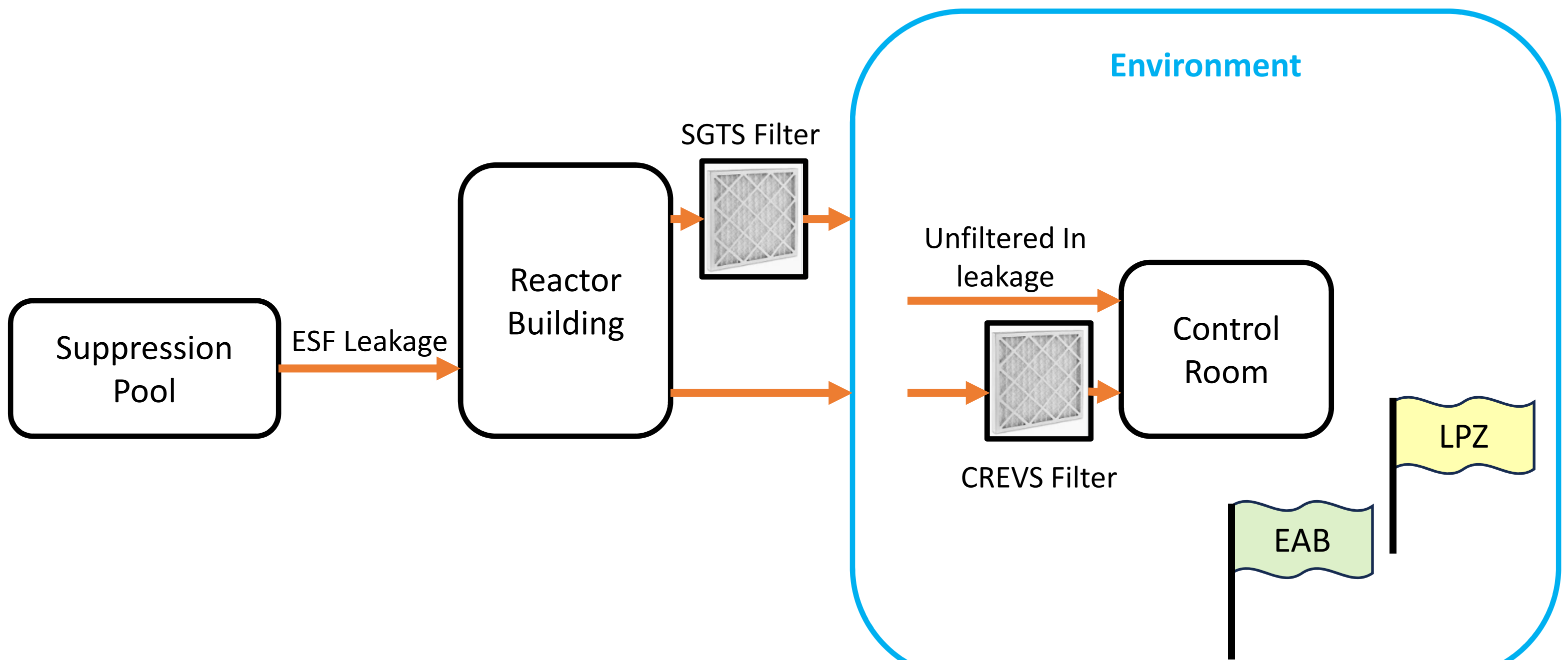
Sources of Health Effects Information:

- RG 8.29, Instruction Concerning Risks from Occupational Radiation Exposure.
- Information Assistance Request NRR-2022-019, "Control Room Design Criteria and Radiological Health Effects," U.S. Nuclear Regulatory Commission, June 2023 (ML23027A059).

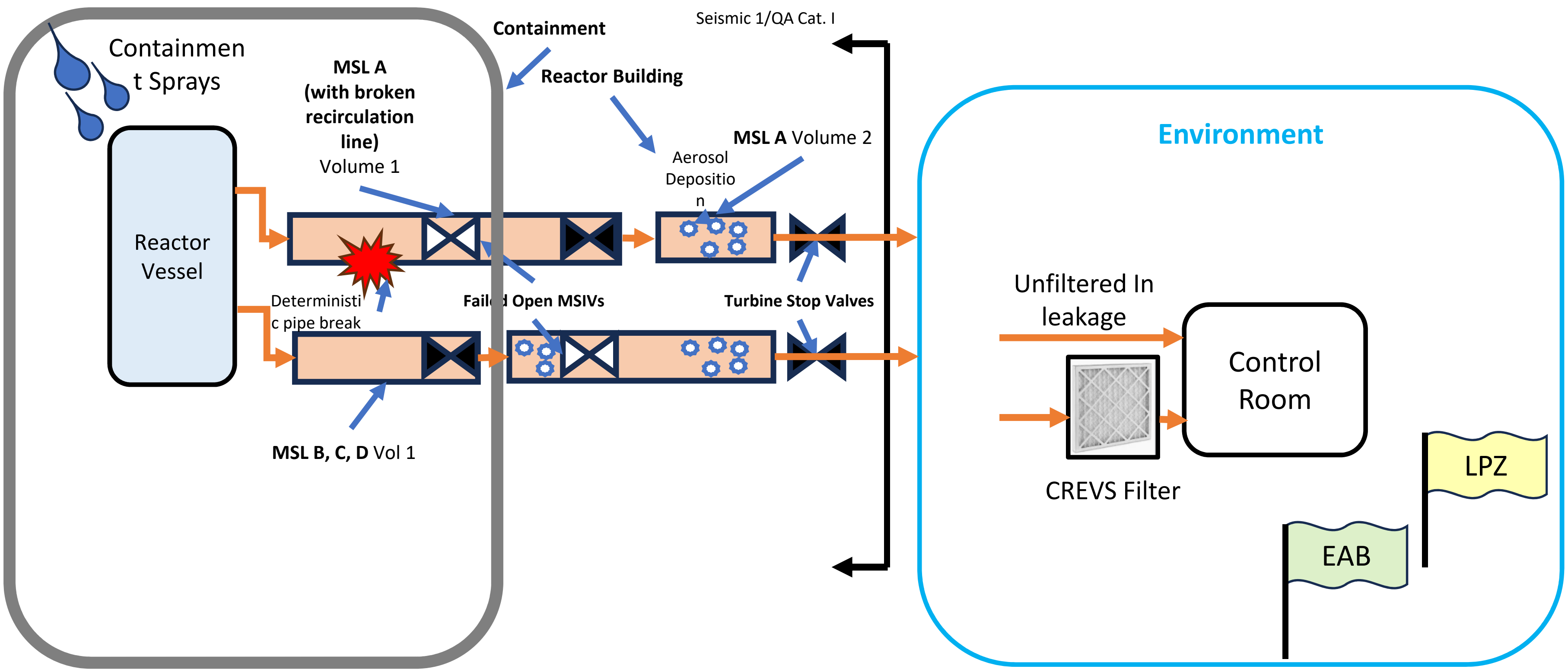
# Containment Leakage Pathway Model



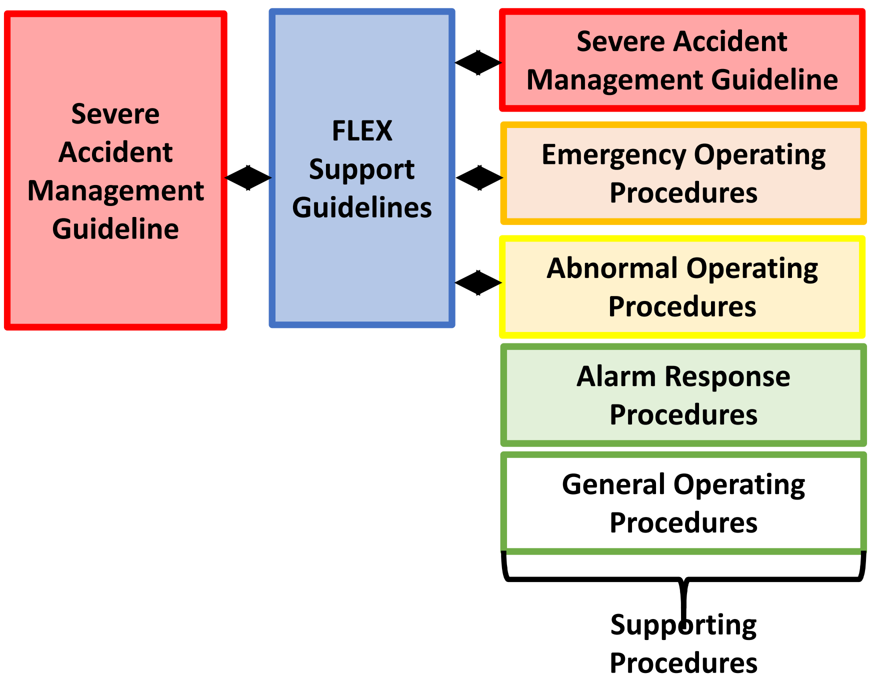
# Engineered Safety Feature Pathway Model

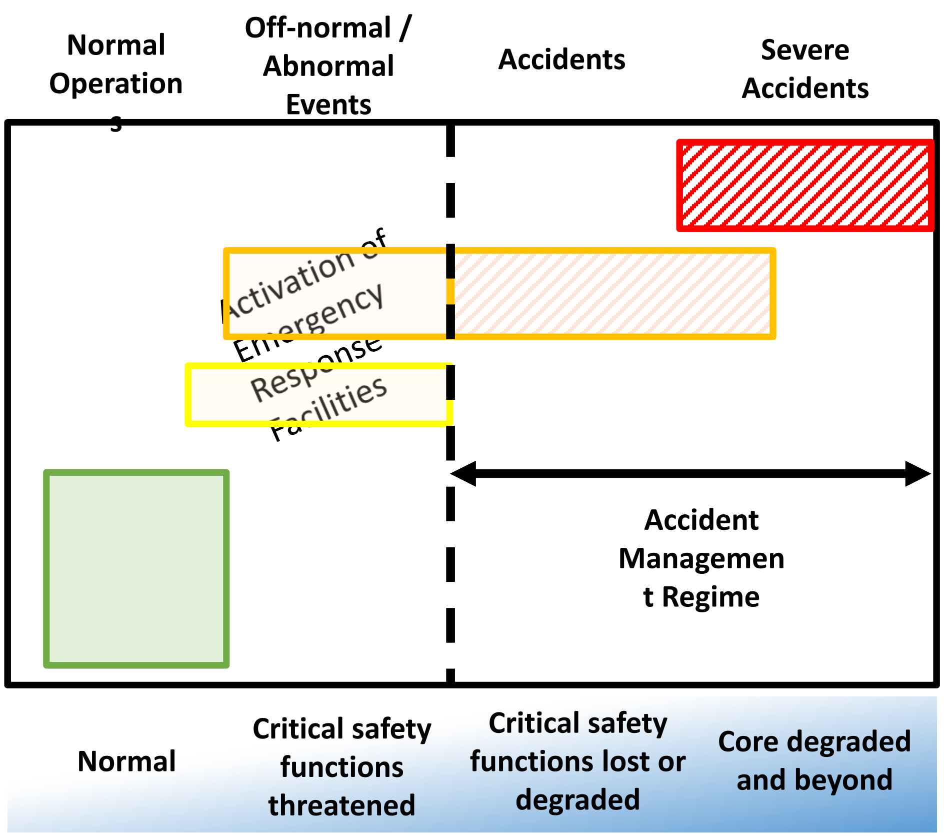


# **Base-Case** Nodalization:
# RG 1.183 Rev. 0 Main Steamline Isolation Valve Leakage



# Control Room Design Criterion of 10 CFR 50.67 and GDC-19: Typical Role of Accident Management Guidelines



# Non-LOCA Gap Release Fractions for RG 1.183 Rev. 2

March 8, 2024 Workshop

Chris Van Wert, Senior Technical Advisor
Joseph Messina, Reactor Systems Engineer

Division of Safety Systems
Office of Nuclear Reactor Regulation

# Overview

- Purpose
- Background
- Assumptions
- Draft PWR Gap Release Fractions
- Draft BWR Gap Release Fractions
- Enrichment Sensitivity Studies

# Purpose

- The NRC is seeking feedback on the assumptions used to generate the draft non-LOCA gap release fractions presented, particularly on the draft power history curves

# Background

- A gap release fractions are the amount of a fission product that will be located in the fuel-cladding gap of a fuel rod relative to the amount produced and therefore be released if rod failure occurs
- The Non-LOCA release fractions in RG 1.183 Rev. 1 are applicable if operation remains below the power history provided in the RG.
    - The power history curve extends to 68 GWd/MTU (rod-average)
- Appendix I provides an analytical procedure to calculate release fractions for other power histories, fuel designs, etc.

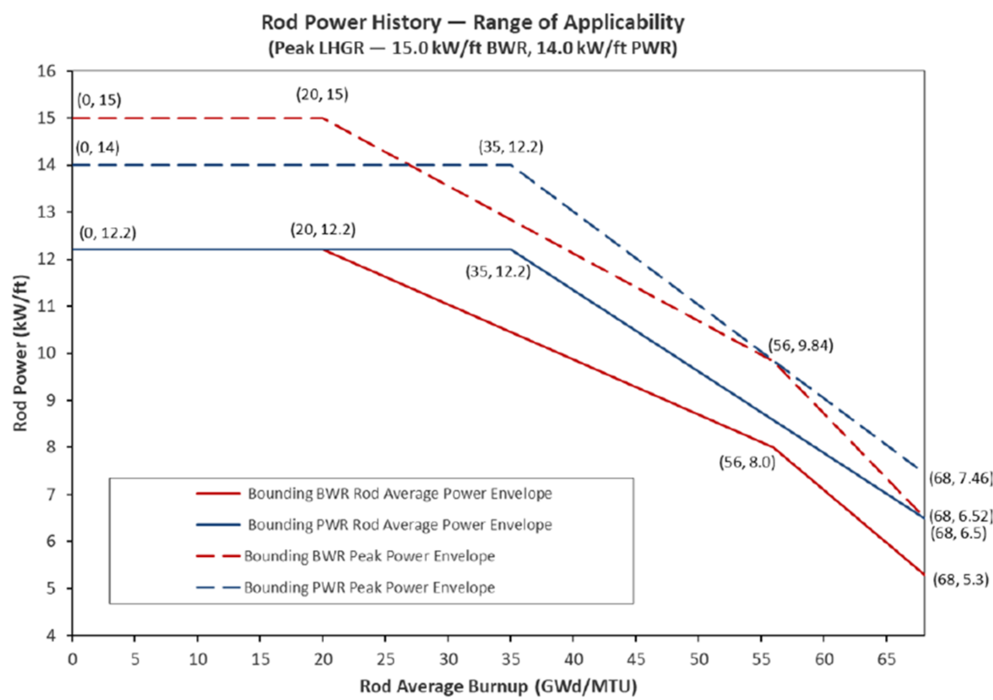



**Figure 1. Maximum allowable power operating envelope for steady-state release fractions**

# Background

- The ANS-5.4 (2011) standard is used to calculate the volatile fission product release fractions
  - Details and derivation of the ANS-5.4 standard is provided in NUREG/CR-7003
  - Database used to develop ANS-5.4 included IE and HBU fuel

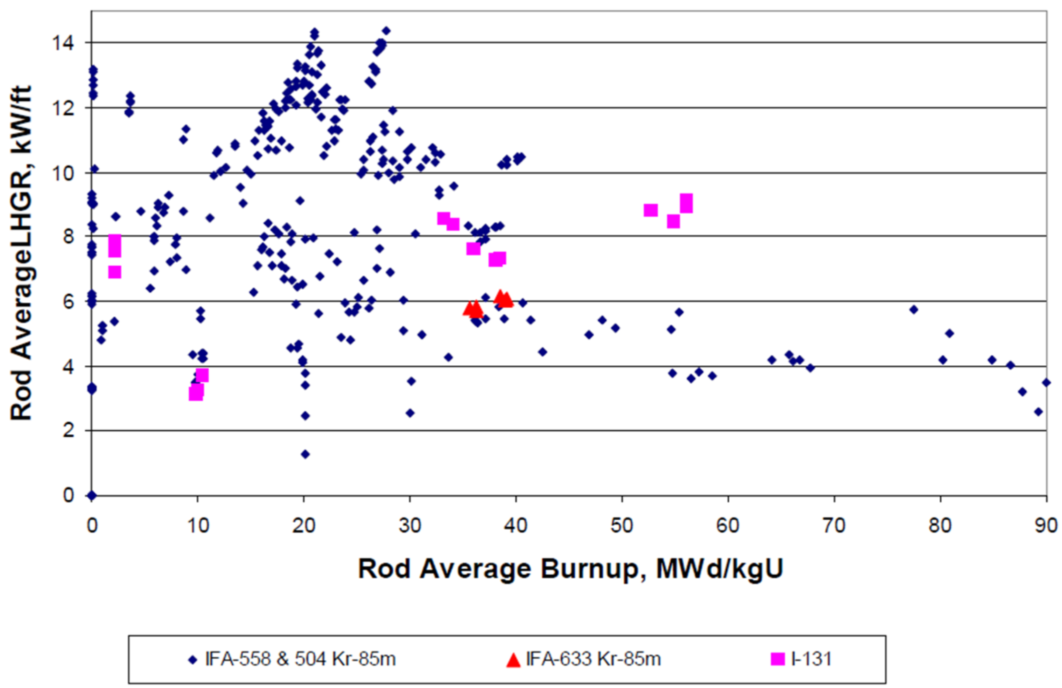
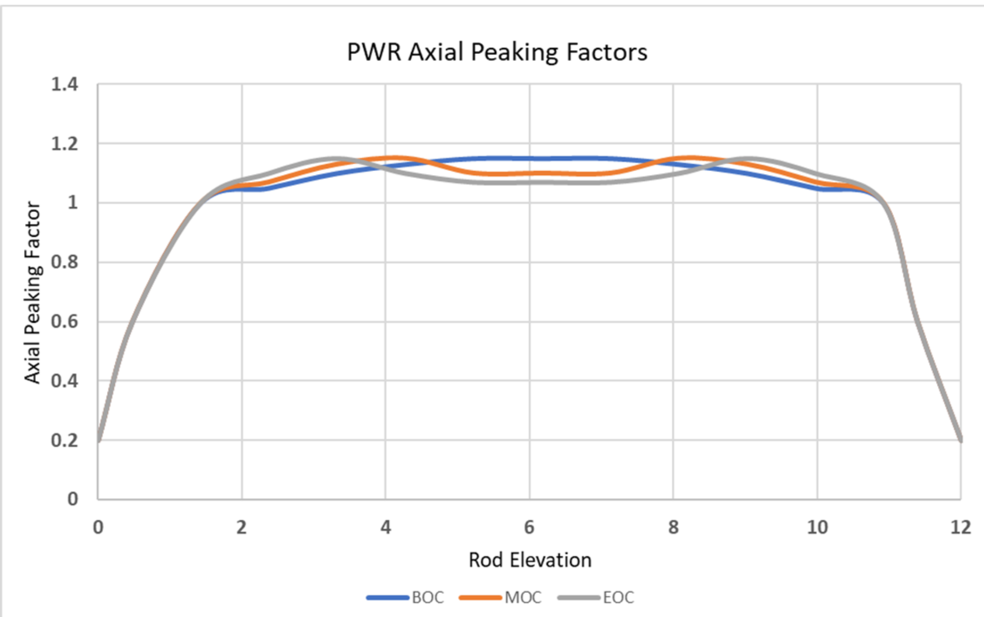
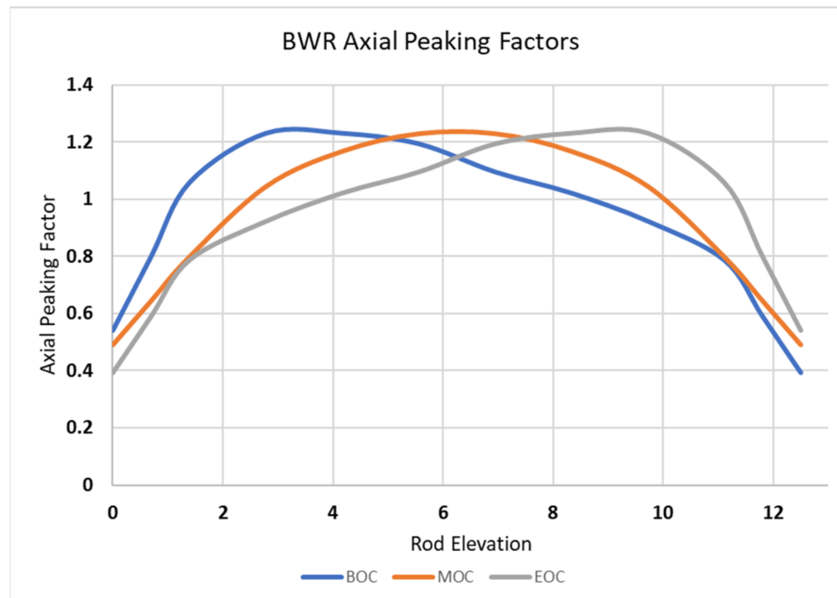



*Figure 1 from NUREG/CR-7003, "Background and Derivation of ANS-5.4 Standard Fission Product Release Model"*

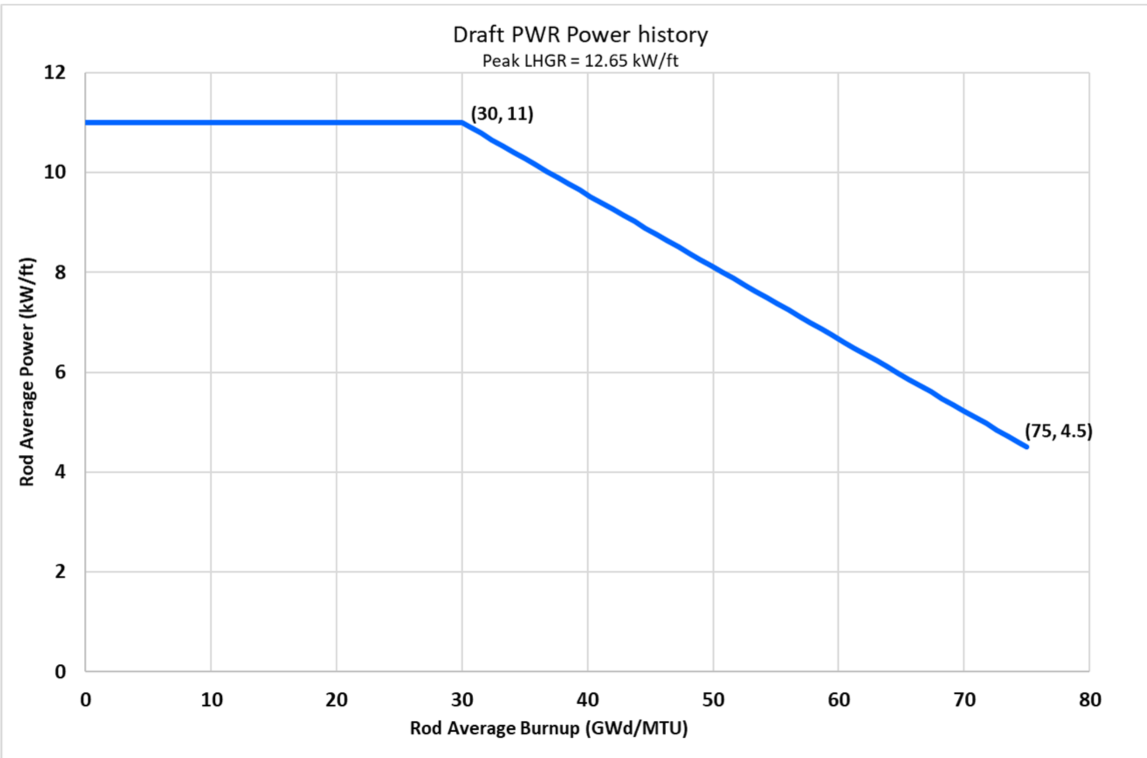
# Assumptions

- Enriched to 8 wt% U-235
- Axial peaking factors:



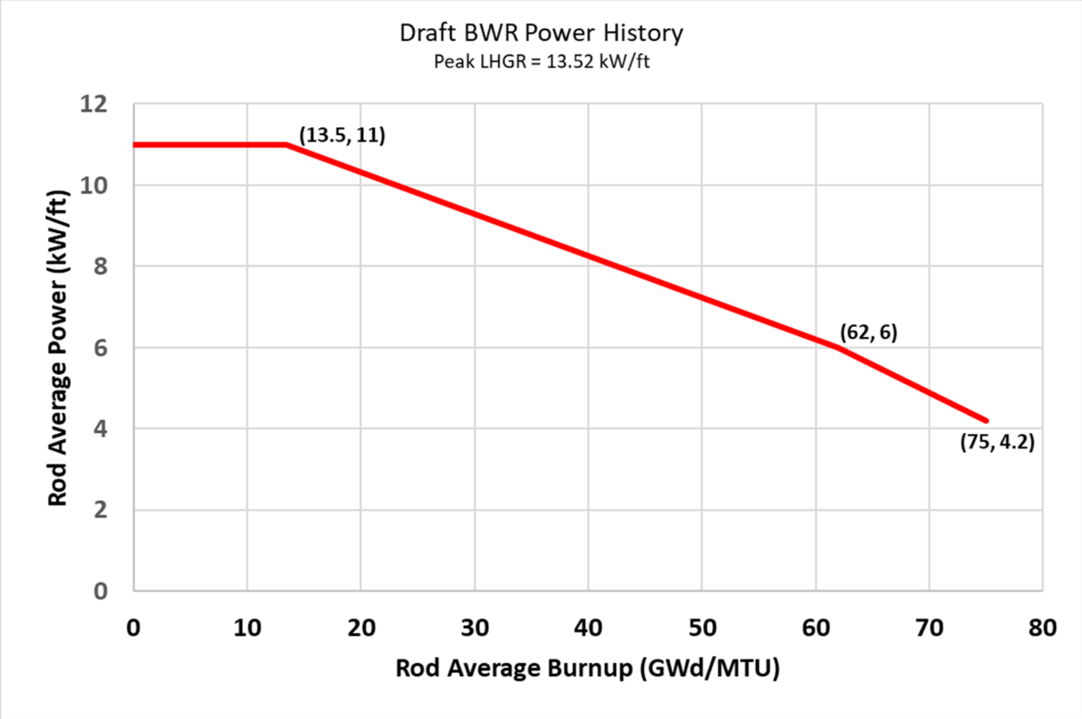



# *Draft* PWR Gap Release Fractions




| Group | Rev 1 Fraction | Draft Rev 2 Fraction (17x17) |
|---|---|---|
| I-131 | 0.07 | 0.04 |
| I-132 | 0.07 | 0.04 |
| Kr-85 | 0.40 | 0.34 |
| Other Noble Gases | 0.06 | 0.03 |
| Other Halogens | 0.04 | 0.02 |
| Alkali Metals | 0.20 | 0.17 |

# *Draft* BWR Gap Release Fractions

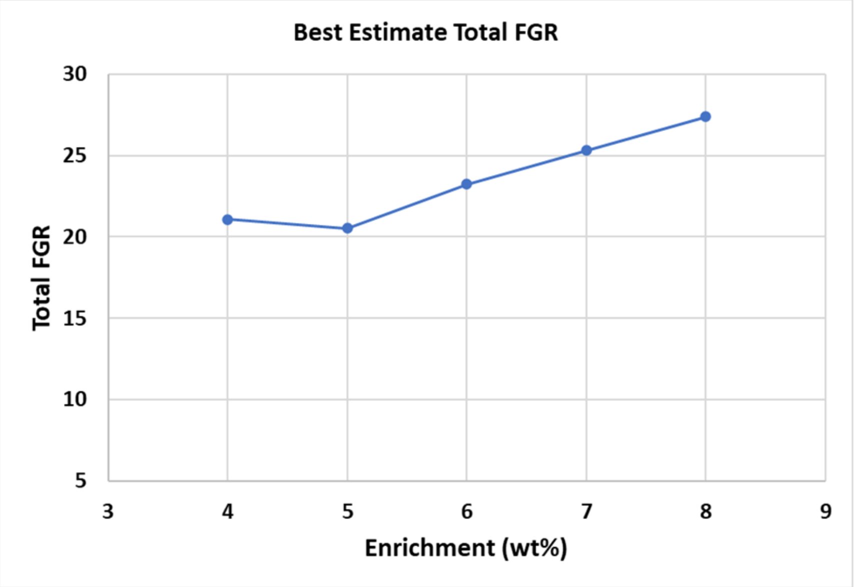



| Group | Rev 1 Fraction | Draft Rev 2 Fraction |
|---|---|---|
| I-131 | 0.03 | 0.01 |
| I-132 | 0.03 | 0.01 |
| Kr-85 | 0.32 | 0.26 |
| Other Noble Gases | 0.03 | 0.01 |
| Other Halogens | 0.02 | 0.01 |
| Alkali Metals | 0.16 | 0.13 |

# Enrichment Sensitivity Study

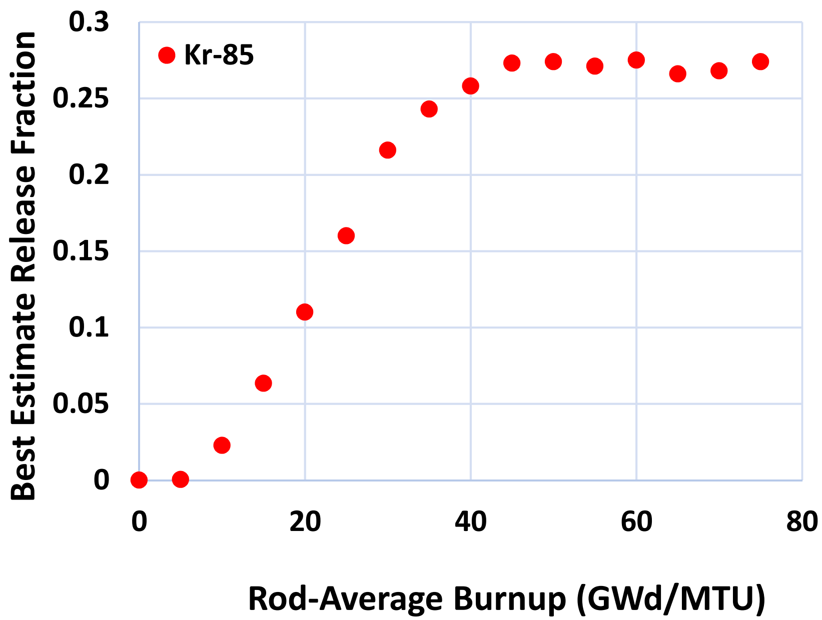
- Non-LOCA releases were found to increase with increasing enrichment

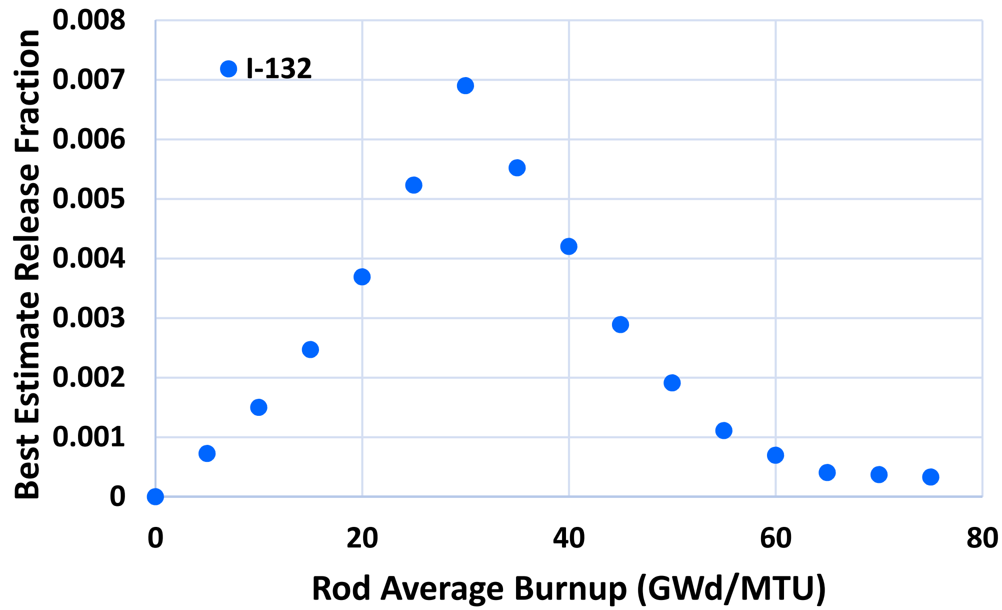
| Enrichment (wt%) | Total FGR (%) | I-132 BE Release Frac | Kr-85m BE Release Frac | Kr-85 BE Release Frac |
|---|---|---|---|---|
| 4 | 21.09 | 4.62E-03 | 1.71E-03 | 2.20E-01 |
| 5 | 20.53 | 5.42E-03 | 2.01E-03 | 2.42E-01 |
| 6 | 23.23 | 6.08E-03 | 2.25E-03 | 2.54E-01 |
| 7 | 25.31 | 6.54E-03 | 2.42E-03 | 2.68E-01 |
| 8 | 27.38 | 6.90E-03 | 2.56E-03 | 2.75E-01 |



# Trends

- Long-lived radionuclides tend to peak late in life
- Short-lived radionuclides tend to peak at the “knee” of the power history curve



# Comments/Questions?

# External Presentations

# Break

# Discussion

# RG 1.183 Update Stakeholder Involvement - Reminder -

- Public Workshops and Information Meetings
  - 3 Workshops
  - 2 Information Meetings
- Contact Project Lead
- Meeting Feedback
- ACRS Meeting – approximately November 2024
- Draft Guide Comment Period – approximately June 2025
  - Relationship with Proposed Increased Enrichment Rulemaking
    - Docket ID NRC-2020-0034, www.regulations.gov
    - Comment period on Proposed Increased Enrichment Rulemaking Regulatory Basis ends on January 22, 2024

# Contacts

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


David Garmon, Project Lead/
Technical Contact

- [david.garmon@nrc.gov](mailto:david.garmon@nrc.gov)
- 301-415-3512

Chris Van Wert, Technical Contact

- [christopher.vanwert@nrc.gov](mailto:christopher.vanwert@nrc.gov)
- 301-415-3217


**How did we do?**

## Public Meeting Schedule: Meeting Details

[ New Search ]

Meeting info

**Purpose**

The NRC is hosting 3 workshops in early 2024 to discuss a revision to RG 1.183, Revision 1 (https://www.nrc.gov/docs/ML2308/ML23082A305.pdf). These workshops will provide opportunities for NRC staff to discuss u extend the applicability of the guidance to higher burnup and increased enrichment fuel applications, in support of the NRC's I Enrichment Rulemaking (https://www.regulations.gov/document/NRC-2020-0034-0005), and to consider other feedback that i to the guidance. The workshops allow for the NRC's external stakeholders to participate in the regulatory process through a c gathering format as described in the NRC's policy statement on public meetings at 86 FR 14964. The staff will collect comme input through note-taking and will consider this information while drafting updates to guidance.

**Meeting Feedback**

**Meeting Dates and Times**

01/09/24
1:00PM - 4:30PM ET

**Meeting Feedback**

Meeting Feedback Form EXIT

**Meeting Location**

Virtual and In-Person at NRC One White Flint North
11555 Rockville Pike
Rockville MD

**Webinar**

Webinar Link:https://events.gcc.teams.microsoft.com/event/a7b85e10-06e8-42d9-b7dc-364a2a63c06c@e8d01475-c3b5-436 5def4c64f52e EXIT
Webinar Meeting Number:233 658 082 952
Webinar Password:JkRaDj

https://www.nrc.gov/pmns/mtg?do=details&Code=20231392