(Draft Guide-1425) Draft Regulatory Guide 1.183, Revision 2 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

ACRS Subcommittee Meeting

INTRODUCTION

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December 19, 2024

ML24348A217

Key Messages

- RG 1.183, Rev. 2 would provide guidance for fuel burnup levels up to 80 GWD/MTU and enrichment up to 10 weight percent U-235—in support of the Commission's high priority IE rulemaking—and updated guidance on other matters to be reviewed during this meeting.

- The draft IE proposed rule includes a set of graded, risk-informed and performance-based control room dose design criteria. The implementing guidance for the proposed criteria is provided in RG 1.183, Rev. 2. The graded, risk-informed and performance-based approach provides incentives for safety improvements that is flexible and scalable.

- Staff hosted three public workshops to seek input from external stakeholders for staff consideration on RG 1.183, Rev. 2, discussed selected Rev. 2 guidance in subsequent two public meetings.

- Staff has completed its review of issues identified in the EDO's FitzPatrick DPO Appeal Decision memo and concluded that the plant complies with relevant NRC requirements. Main recommendations included in the memo have been considered and reflected in RG 1.183, Rev. 2.

- ACRS recommendations documented in the 9/20/23 letter regarding RG 1.183, Rev. 1 have been considered and reflected in RG 1.183, Rev. 2.

ACRS Subcommittee Meeting

STAFF PRESENTATIONS

David Garmon, Health Physicist, NRR/DRA
Elijah Dickson, Senior Reliability and Risk Analyst, NRR/DRA
John Parillo, Senior Reactor Engineer, NRR/DRA
Chris Van Wert, Senior Technical Advisor, NRR/DSS

December 19, 2024

Agenda

- Overview of Development of DG-1425
- Follow-up of Selected Topics
- Recap of Increased Enrichment Rulemaking for Control Room Design Criteria
- Purpose and Regulatory Requirements
- Key Proposed RG 1.183, Rev. 2 Changes
  - Risk-informed and Performance-based Framework for Control Room Design Criteria – Guidance for IE RM
  - Pathway-Specific Source Term
  - 10 CFR 50.46 FFRD Source Term and Acceptance Criteria
  - Best Estimate Plus Uncertainty

** Break **

Additional Changes
  - Non-LOCA Gap Release Fractions and Changes to Appendix I
  - Clarification of MHA as DBA
- Discussion/Feedback

Overview of Process Staff Used in Developing DG-1425

- Phased approach, as described to ACRS in 2023
  - Revision 1 – Applied experience from AST implementation since Revision 0
  - Revision 2 – Addresses modern and future needs for AST regulatory framework
- Multidisciplinary NRC staff working group guided by interoffice steering committee of Senior Executives
- Stakeholder interaction
  - 5 public meetings to increase external stakeholder participation in the regulatory process
  - Working group technical discussions and Comment and Issue Resolution Process
- Looking forward
  - Publishing DG-1425 with IE Proposed Rule
  - After comment period and consideration of public comments, finalize and publish RG with the IE Final Rule
  - Staff work continues as it pertains to some technical issues (e.g., EQ and aerosol deposition in containment)

Follow-up of Selected Topics

- Recommendations from 2023 ACRS letter (ML23256A179)
  - Multiple revisions of the RG will remain in effect for the foreseeable future
    - Consistent with NRC's use of regulatory guidance and the "Reliability" Principle of the agency's Principles of Good Regulation
    - Within applicability statements in the Regulatory Guide (i.e., applicability of the source term included in the particular revision being considered)
  - Clarification of the purpose and objectives of the design basis accident
    - Several Revision 2 sections contain additional information on basis/rational for DBA analyses
    - Knowledge Management Appendix was added to Revision 2
- DPO 2021-001 (ML23263A639) EDO Decision Memo
  - Background - DPO 2021-001 raised several questions regarding design basis accident analysis
  - Guidance-Related Questions are addressed in RG 1.183, Rev. 2
    - Several sections of the RG were updated to provide high-level guidance for evaluating alternative options in Regulatory Guidance and SRPs, as well as to enhance focus on the overall intent of regulations related to the DBA analysis.
  - In response to the EDO's direction, staff verified licensee's compliance with 50.67, as addressed in the staff's safety evaluation addendum for FitzPatrick (ML24207A019)

Recap of Increased Enrichment Rulemaking for Control Room Design Criteria

- Consider Commission-directed PRA-related policies, which advocate for 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), which states:**

"… 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."*

Recap of Increased Enrichment Rulemaking for Control Room Design Criteria (Cont.)

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.*

<u>Element 1 (Discussed in detail at 12-18-2024 ACRS Meeting on IE Rulemaking)</u>

Recap of Increased Enrichment Rulemaking for Control Room Design Criteria (Cont.)

Public input guided Rev. 2 changes

Staff recommended Alternative 2 of Regulatory Basis document (88 FR 61986), which reads:

“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”

Regulatory Basis Question #2

Nearly all comments included suggestions to develop a graded, risk-informed approach to the control room design criteria.

Public Engagement

Public considerations

Industry feedback

Research

Assessment of Health Effects (ML23027A059)

White Paper on a Graded, Risk-Informed and Performance-based Framework (ML24212A254)

Others

DPO 2020-002 (ML21067A645)

DPO 2021-001 (ML23263A639)

ACRS Comments on Rev. 1

Purpose and Regulatory Requirements

- NRC staff developed the original RG 1.183 Rev. 0 (NRC, July 2000) in response to the 1985 Severe Accident Policy Statement (50 FR 153) to support implementation of 10 CFR 50.67, “Accident source term,” which provides the regulatory pathway to update the deterministic accident source term of TID-14844 (AEC, 1962) to a more realistic, mechanistic, accident source term (AST), among other licensing basis updates.

- 10 CFR 50.67 maintains consistency with the original 10 CFR Part 100, “Reactor Site Criteria,” which requires applicants to deterministically assume a fission product release to the containment for evaluating containment leak-rates at its “expected demonstrable leak rate” with the implicit assumption that the containment remains intact against the “maximum credible accident.”

- NRC has updated the AST for large LWRs in response to various national and industry needs. Note: applicants for various SMR designs have developed their own design-specific ASTs.

Purpose and Regulatory Requirements

The Regulation

10 CFR 100.11(a)

"*As an aid in evaluating a proposed site, an applicant should assume a fission produce release[1] from the core, the expected demonstrable leak rate from the containment and the meteorological conditions pertinent to his site to derive an exclusion area, a low population zone and population center distance. For the purpose of this analysis, which shall set forth the basis for the numerical values used, the applicant should determine the following…*"

Footnote 1

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

Purpose and Regulatory Requirements (Cont.)
Guidance and Source Term Limits of Applicability

- Each Regulatory Guide vintage:
  - Meets the regulatory requirements.
  - Endorses an accident source term with limits of operational applicability and use.
  - Contains various fission product transport and mitigation models.
  - Assumes various operational and human health parameters.

Regulatory Guide	Max. Rod Average Burnup GWd/MTU	Enrichment w/o U-235	Source Term
1.3, 1.4, 1.195	18-25	3-3.5	TID-14844
1.183, Rev. 0	Up to 60, With NUREG/CR-5009, up to 62	5	NUREG-1465
1.183, Rev. 1	Up to 68	8	SAND-2011-0128
1.183, Rev. 2	Up to 80	10	SAND-2023-01313

History of the in-containment source

TID-14844
(AEC, 1962)

NUREG-1465
(NRC, 1995)

SAND 2011-0128
(SNL, 2011)

SAND 2023-01313
(SNL, 2023)

SAND 2024-10674
(SNL, 2024)

ML24229A044;

RG 1.3 and 1.4
(AEC, 1970-74)

RG 1.183 Rev. 0
(NRC, 2000)

RG 1.183 Rev. 1
(NRC, 2023)

DG-1425
(NRC, 2024)

Purpose and Regulatory Requirements (Cont.)

History of the in-containment source term and why the NRC stops at the end of early in-vessel

- **1950s and 60s – Power Reactor Siting**
  - 10 CFR Part 100.11, *Determination of exclusion area, low population zone, and population center distances* (27 FR 3509, Apr. 12, 1962)
    - Footnote 1 and J.J. DiNunno et al., TID-14844, *Calculation of Distance Factors for Power and Test Reactor Sites*, (AEC, 1962)
    - Implementation guidance 1.3 and 1.4
- 1979 – Three Mile Island Unit 2 Accident
- **1980s – PRA and Research into Severe Accident-based Accident Source Terms**
  - 1985 *Severe Accident Policy Statement* (50 FR 153)
  - SECY 86-76, *Implementation Plan for the Severe Accident Policy Statement and the Regulatory Use of New Source-Term Information* (ML12251A697)
  - SECY-89-341, *Updated Light Water Reactor (LWR) Source Term Methodology and Potential Regulatory Applications* (ML12251A697)
  - SECY-90-016, *Evolutionary Light Water Reactor (LWR) Certification Issues and their Relationship to Current Regulatory Requirements* (ML003707849)
- **1990s – Rebaselining Operating Reactor Accident Source Term from TID-14844 to the AST**
  - SECY-94-302, *Source term-related technical and licensing issues pertaining to evolutionary and passive light-water reactor designs* (ML003708141)
  - NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants" (NRC, 1995)
  - ACRS letter of 3/15/94 (ML20064L817)
  - SECY-96-242, "Use of the NUREG-1465 Source Term at Operating Reactors," dated November 25, 1996
  - SRM-SECY-96-242, "Use of the NUREG-1465 Source Term at Operating Reactors," dated February 12, 1997 (ML003752965)
  - SECY-98-0154, "Results of the Revised (NUREG-1465) Source Term Rebaselining for Operating Reactors," dated June 30, 1998 (ML992880064)
- **1999 / 2000 - AST Final Rule and Guidance**
  - 10 CFR 50.67, *Accident Source Term* (64 FR 71990, Dec 23, 1999)
  - 473rd Advisory Committee on Reactor Safeguards (ACRS) Full-committee meeting held June 7, 2000
  - RG 1.183, Revision 0, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Washington, DC, July 2000 (ML003716792)

Reactor	Environment	Fraction iodine escaping
Windscale (1957)	Air, burning fuel	0.25 (0.123 to environ.)
NRU (1958)	Air, burning fuel	Chemical species not identified
HTRE-3 (1958)	Air, melting fuel	0.027
NRX (1952)	water	Fission products not identified
SL-1 (1961)	water	0.005
TMI-2 (1979)	Water, steam, hydrogen	0.000002

Purpose and Regulatory Requirements (Cont.)
Implications of Updating Source Terms

- The AST changes the prescription of the source term and fission product transport models from those originally described in TID-14844. These changes influence the major areas of dose analysis and can prompt plant modifications, technical specification and procedure modifications.

  - **Timing**: The ASTs are assumed to occur over a period of time (DG-1425 timing is about four to eight hours) as opposed to the TID source term, which assumed the release of the entire source term occurs instantaneously at time zero. Therefore, plant systems originally provided to mitigate an instantaneous source term by very rapid actuation would not be required to perform under such stringent requirements with the revised source term.

  - **Composition**: Each AST vintage has included additional chemical classes and radionuclides, whereas the original TID focused primarily on specific radionuclides with the greatest impact on human health. Thus, some original licensing basis assumptions may need to be revisited due to the harsher radiation environments.

  - **Speciation**: The AST assume up to 100% of the radioiodine to be released as an aerosol, CsI. This is in contrast to the original TID source term, which prescribed nearly the opposite, 95% of the iodine as vapor and 5% as aerosol. Therefore, systems needed to remove iodine vapors are less important under conditions where iodine is an aerosol.

- Since issuance of RG 1.183 Rev. 0, the staff have resolved many "rebaselining" issues and a number of other more subtle differences between the source terms, as well as the impact of improved modeling of fission product processes, with RG 1.183 Rev. 2.

Purpose and Regulatory Requirements (Cont.)

- For several regulatory requirements, compliance is demonstrated, in part, by the evaluation of the radiological consequences of DBAs. The current licensing basis may include, but is not limited to, the following:
  - Original Power Reactor Siting
    - 10 CFR 100.11 for applications before January 10, 1997, and 10 CFR 100.21, "Non-seismic siting criteria," which references criteria in 10 CFR 50.34(a)(1), for subsequent applications
  - Site Safety Assessment
    - 10 CFR 50.67, 10 CFR 50.34, 10 CFR 52.17, 10 CFR 52.47, 10 CFR 52.79, 10 CFR 52.137, 10 CFR 52.157
  - Control Room Habitability
    - 10 CFR 50.67, 10 CFR Part 50, Appendix A, General Design Criteria (GDC) 19, 10 CFR 50.34
  - Equipment Qualification
    - 10 CFR 50.49
  - Post-TMI requirements (NUREG-0737)
    - Items, II.B.2, II.B.3, II.F.1, III.D.1.1, III.A.1.2 and III.D.3.4

Key Proposed RG 1.183, Rev. 2 Changes

Risk-informed and Performance-based Framework
for the Control Room Design Criteria - Guidance for IE RM

DG-1425, Section 4.4, *Acceptance Criteria*

Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM

<u>Element 2 of IE Rulemaking</u>

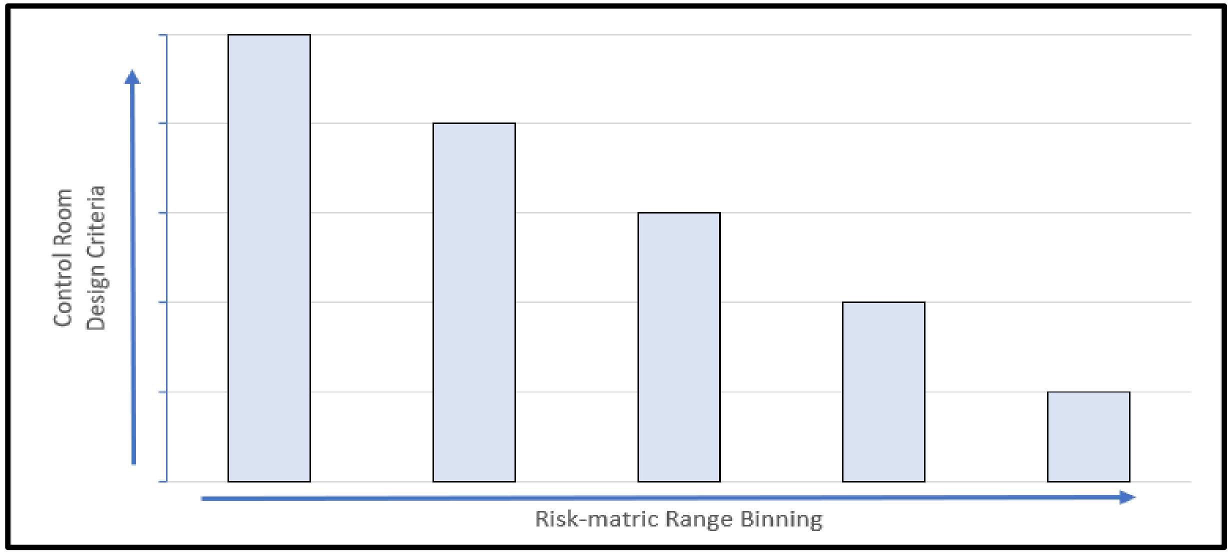
- Establish in guidance, a framework for a graded, risk-informed and performance-based control room design criterion.
  - Enables a performance-based evaluation using traditional deterministic radiological consequence analysis methods within defined risk-informed boundaries.
  - Boundaries are defined 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.
  - Provides flexibility when determining how to meet an established acceptance criterion in a way that encourages and rewards safety of the facility.
  - In practice, the framework leverages the facility's safe design and operations to justify a higher control room design criterion with a lower plant-specific risk metric.

- Considerations:
  - Consistent with Commission policy
  - 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.

Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)

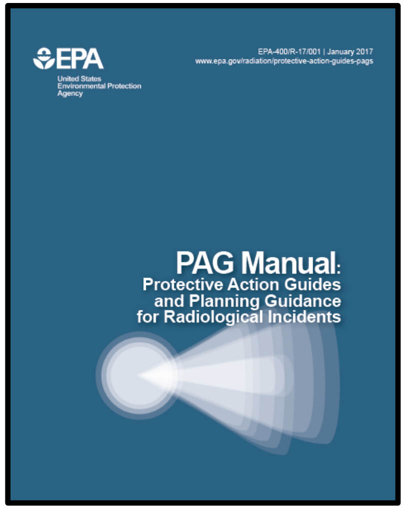
- Method: systematic mapping of a pre-determined range of acceptable dose-based control room *design* values onto a specified risk metric.
- Range of acceptable *design* values would be defined based on an assessment of regulatory precedence and organizations responsible for making radiation protection recommendations.
- Risk metrics would be defined based on an assessment of those commonly used in nuclear risk analysis.

Method for Developing a Graded, Risk-Informed, and Performance-Based Framework

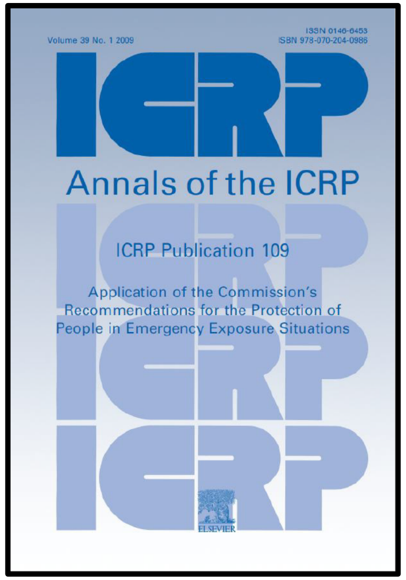


Source: Dickson, E., NRC, internal memorandum to K. Hsueh, “Method for a Graded Risk-Informed Performance-Based Control Room Design Criteria Framework,” Washington, DC, July 2024 (ML24212A254).

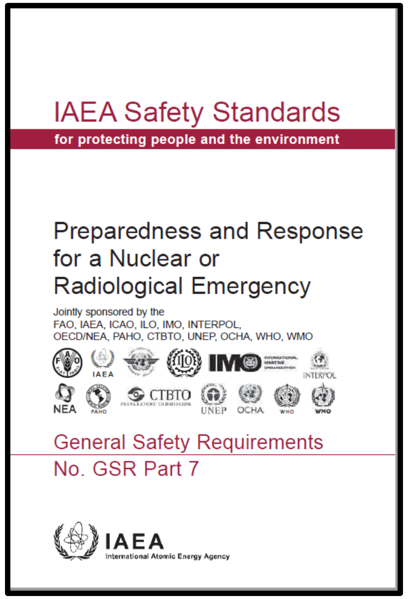
Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)



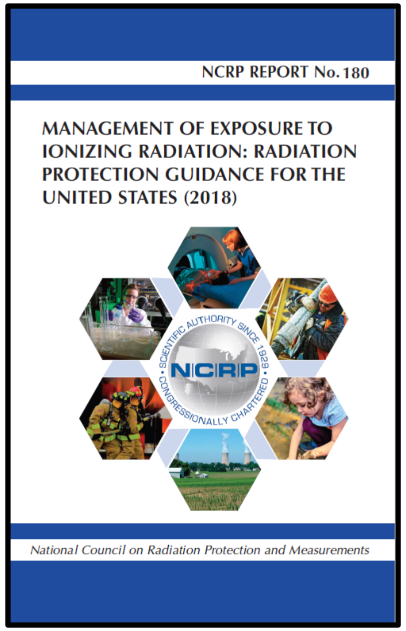
25 rem for lifesaving or protection of large populations



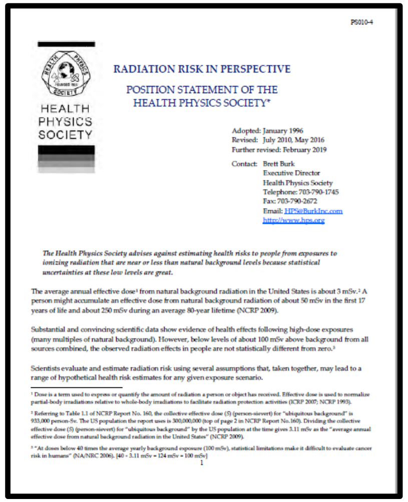
2–10 rem acute for emergency exposure situations



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.

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

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Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)

- Evaluation performed to assess various risk-metrics to understand how the framework would impact the fleet and if it can meet Commission expectations in SRM-SECY-98-144.

- Risk-metric datasets developed from various licensing activities.
  - Metrics: CDF, LERF, rem TEDE, Whole Body, Population Dose risk.
  - Datasets: IPE and IPEEE, risk-informed ISI and ILRT, SAMA, 50.69, TSTF-505.

Evaluation examples provided to examine:

1 – Flexibility that encourages further safety improvements

2 – Impact of modeling fidelity, quality, and inclusion of external hazards

3 – Differences among reactor designs

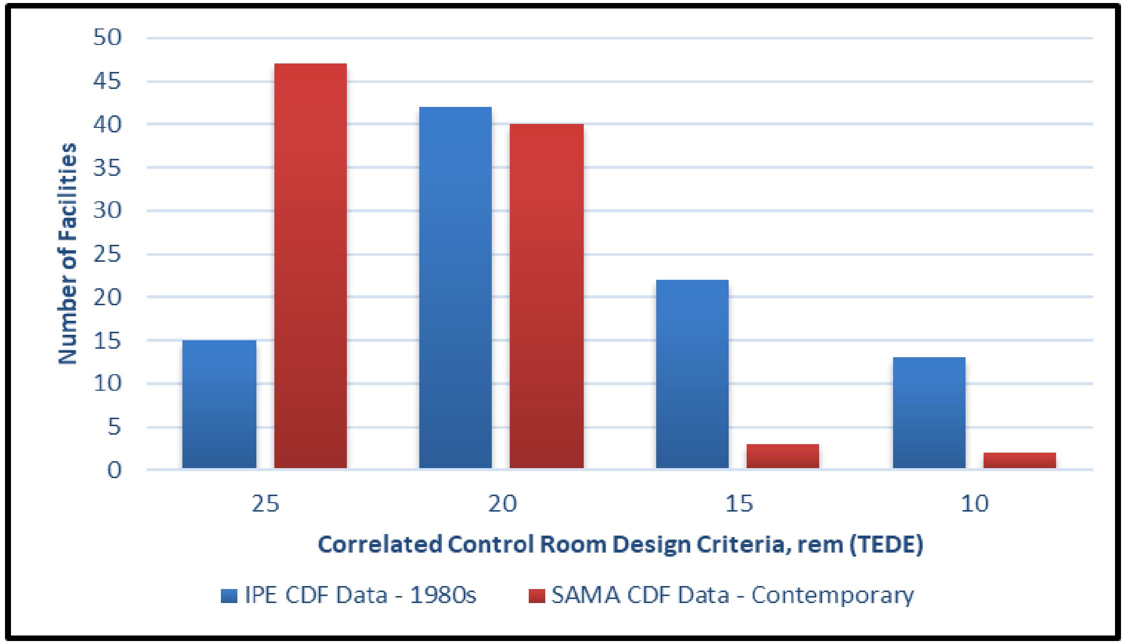
Risk-informed and Performance-based Approach to Provide Incentives for Safety Improvements that is Flexible and Scalable

Example 1—Flexibility that encourages further safety improvements

Purpose: Assess how the operating nuclear reactor fleet would align within each control room design criterion bin over a period of time.

Constraints:

- IPE dataset represent 1980's risk-profiles.
- SAMA results represent major facility changes, improving safety since the IPE submission.
- The PRA quality of the SAMA dataset is consistent with the approved guidance in NEI 05-01, "Severe Accident Mitigation Alternatives (SAMA) Analysis," issued November 2005 (NEI, 2005) and later standards.



Example 1—Analysis Result Histogram, Comparing 1980s to Contemporary CDF Risk Profiles

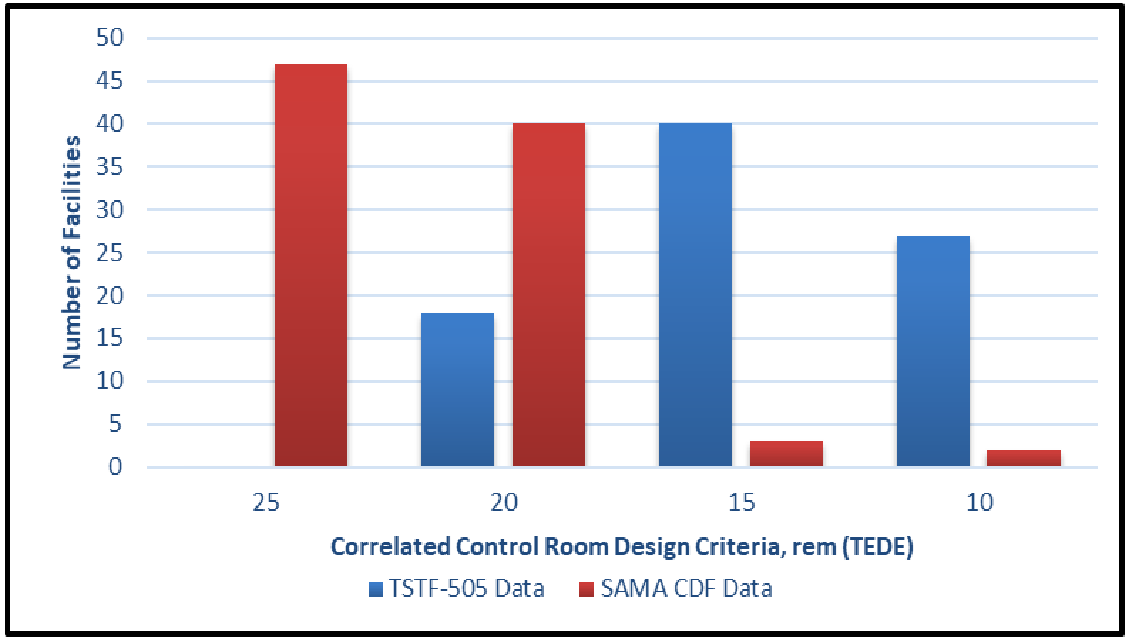
Risk-informed and Performance-based Approach to Provide Incentives for Safety Improvements that is Flexible and Scalable (Cont.)

Example 2—Examining modeling fidelity, quality, and inclusion, or updated, external hazards

Purpose: Assess increased model fidelity, quality, and updated external hazards.

Constraints:

- The TSTF-505 dataset are consistent with RG 1.174 and 1.200, ANS Standard and approved guidance in NEI 06-09-A.



Example 2—Analysis Results Histogram Comparing License Renewal SAMA to TSTF 505 CDF Risk Profiles

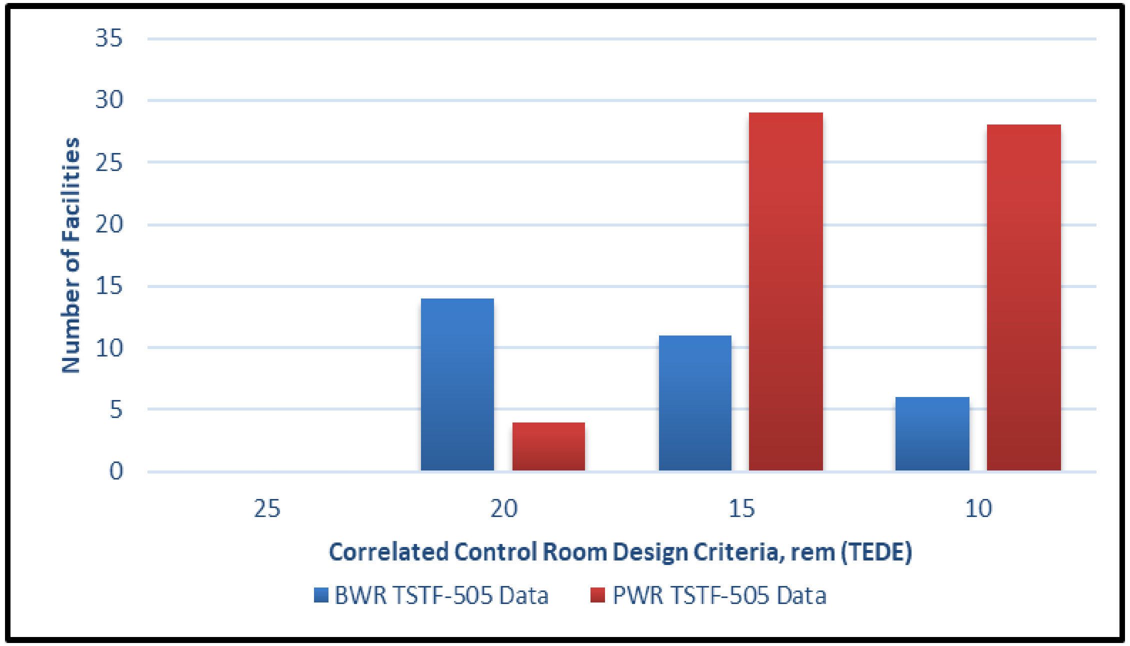
Risk-informed and Performance-based Approach to Provide Incentives for Safety Improvements that is Flexible and Scalable (Cont.)

Example 3—Examining differences among reactor designs

Purpose: Understand how the operating nuclear reactor fleet would align based on reactor design type.

Constraints:

- Use of TSTF-505 data set.



Example 3—Analysis Results Examining Difference Between Reactor Designs

Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)

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 licensing 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

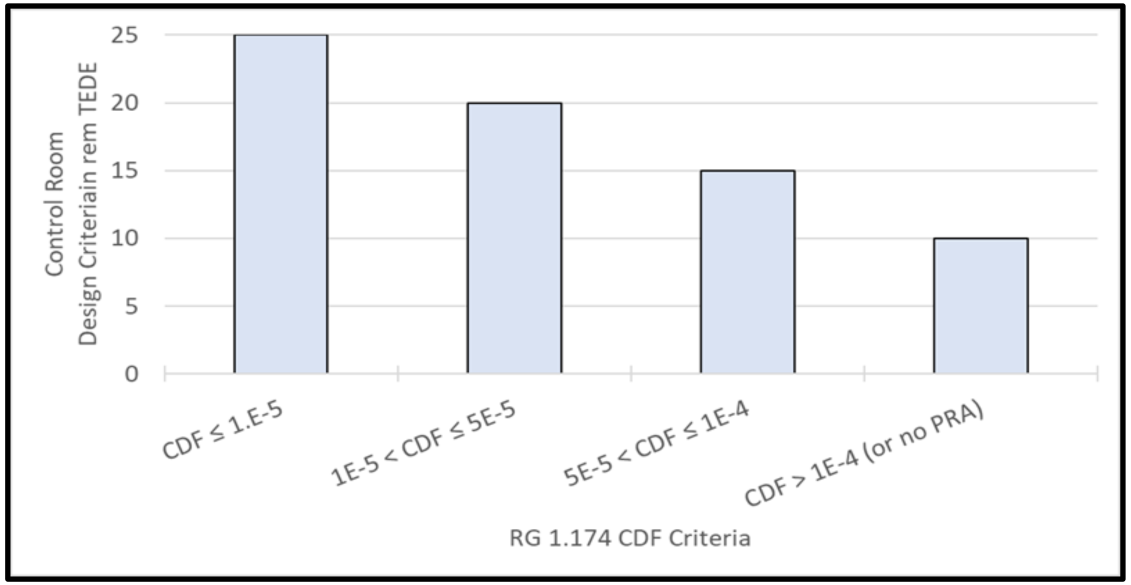
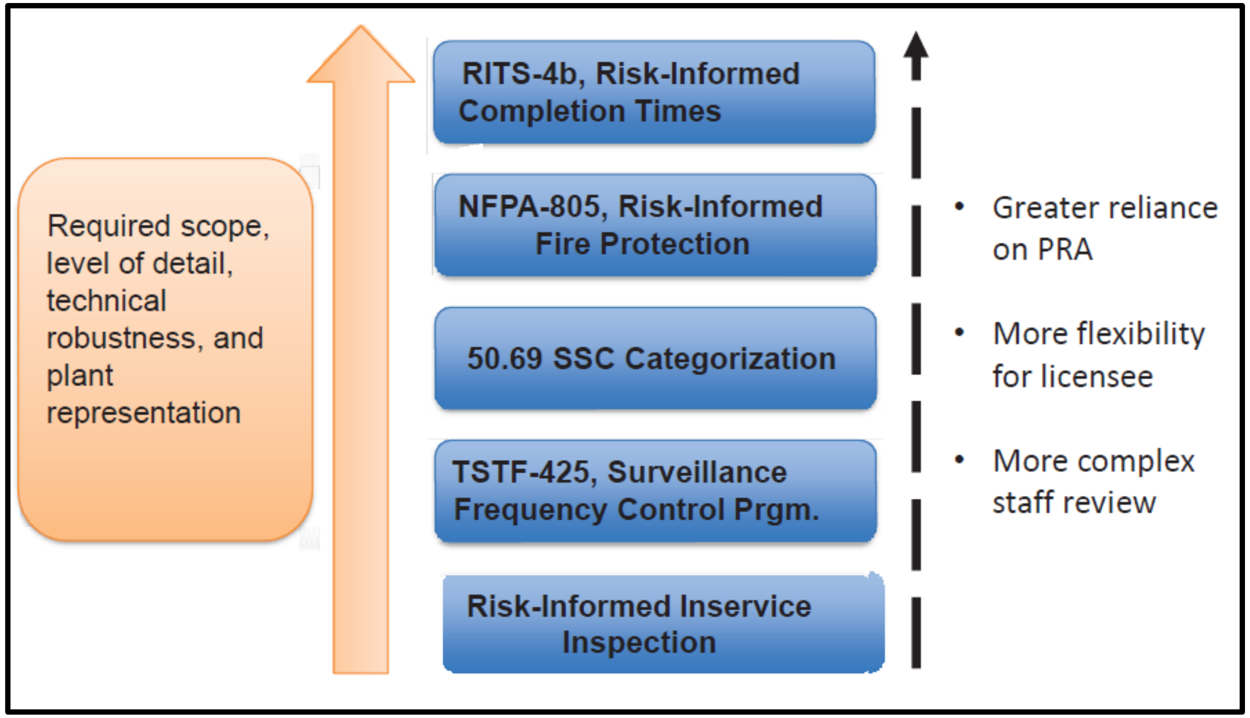


Figure representing DG-1425, Table 8 Graded, Risk-Informed and Performance-Based Framework

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

Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)

- Guidance does not request licensees to perform or update their PRA, only to submit information on their most recent PRA with justification of its acceptability.
- Expectation is that the PRA model is consistent with the philosophy in RG 1.174 and the technical adequacy expectations in RG 1.200.
- PRA model should estimate overall risk for all significant sources of risk, both internal and external to the plant.
- Guidance provides examples of risk-information based on “acceptable” PRAs, but other risk-information would be reviewed on a case-by-case basis.



The NRC’s Concept of PRA Acceptability for Different Licensing Applications

Risk-informed and Performance-based Framework for the Control Room Design Criteria - Guidance for IE RM (Cont.)

<u>Recap</u>:

- 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.

- The proposed draft regulatory guidance executes the Commission's SRM-SECY-98-144 with a framework, which provides flexibility to determine how to meet the established performance criteria in a way that will encourage and reward improved outcomes.

Key Proposed RG 1.183, Rev. 2 Changes

Pathway-Specific Source Term

DG-1425, Section 3.2.1, *Pathway-Specific Release Fractions Using Mechanistic Transport Modeling*

Pathway-Specific Source Term

Regulation:

- Meets the regulatory requirements.

Applicability:

- Applies to MHA-LOCA models for applicants and licensees using zirconium-alloy cladded uranium dioxide (UO_2) fuel rod designs within the following scope of applicability:
  - The guidance applies to reactor core burnups up to a maximum rod-average of 80 GWd/MTU.
  - The guidance applies to fuel enrichments up to 8 weight percent U-235 for PWRs and up to 10 weight percent U-235 for BWRs.
  - The guidance applies to currently approved (as of the issuance date of this RG revision) fuel and cladding materials (e.g., coated zirconium alloy claddings and doped fuels).
  - The guidance also applies to fuels with iron-chromium-aluminum (FeCrAl) alloy cladding and chromium-coated cladding.

Pathway-Specific Source Term (Cont.)

- Increased Enrichment Regulatory Basis document and industry input led to the development of pathway-specific source terms for the BWR design.
- Models replace historical practices endorsed in Rev 0. and Rev. 1, which dated back to the 1960s.
  - Directly accounts for safety-related function to mitigate fission product releases. such as suppression pool scrubbing.
- Modeling approach is consistent with modern severe accident source term analyses and tools, which can mechanistically transport the source term from one location to another, which has been an area of interest since the 1980s.
- Consistent with DPO-2020-002, philosophy of crediting safety-related SSCs to transport and ultimately mitigate fission product releases in radiological consequence analyses.
  - Corrects several Rev. 0 licensing issues.
- Note: Additional research is being performed to develop similar mechanistic models for containment deposition and spray.

Pathway-Specific Source Term (Cont.)

- Containment source term (ST) separated into three parts:
  - Suppression pool
  - Containment atmosphere
  - Main Steam Line (MSL)
- The first two STs are derived from SAND2023-01313 results.
- The MSL ST is developed from new calculations documented in SAND2024-10674.

ML24229A044;

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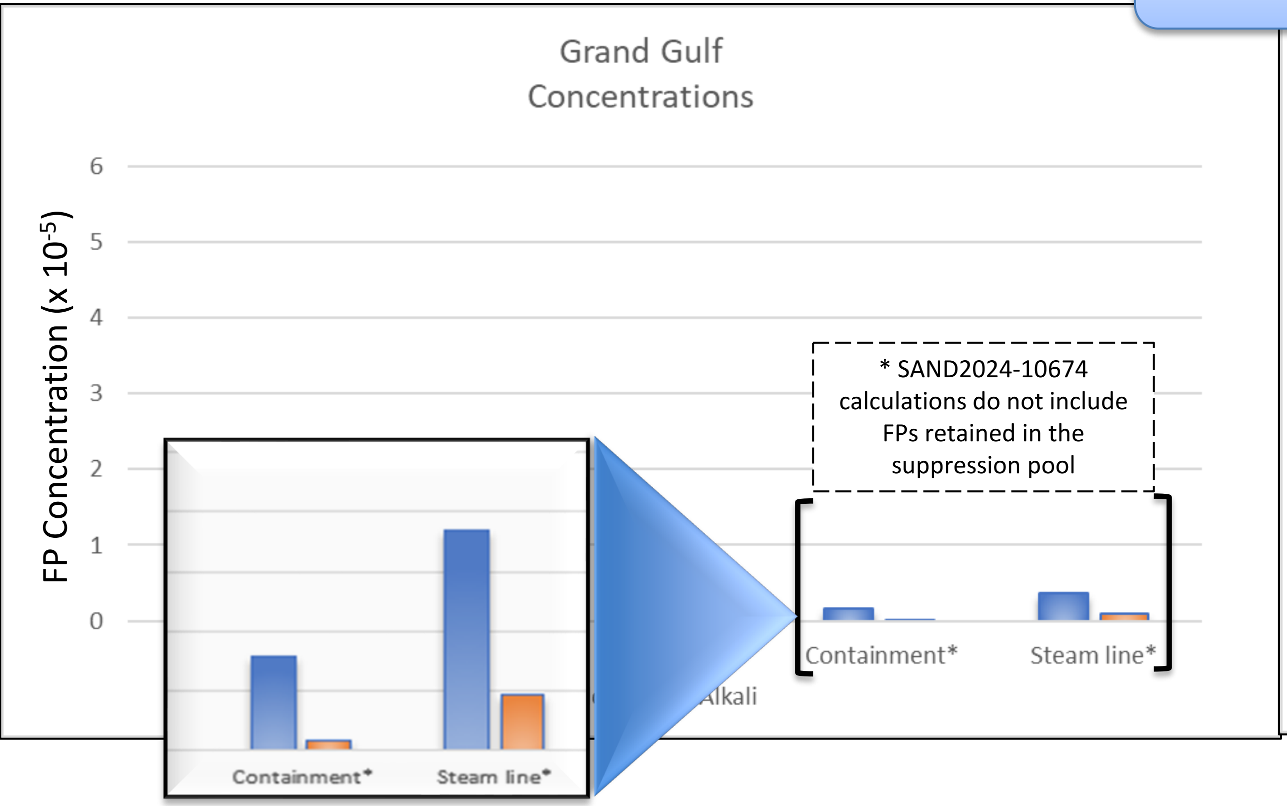
Pathway-Specific Source Term (Cont.)

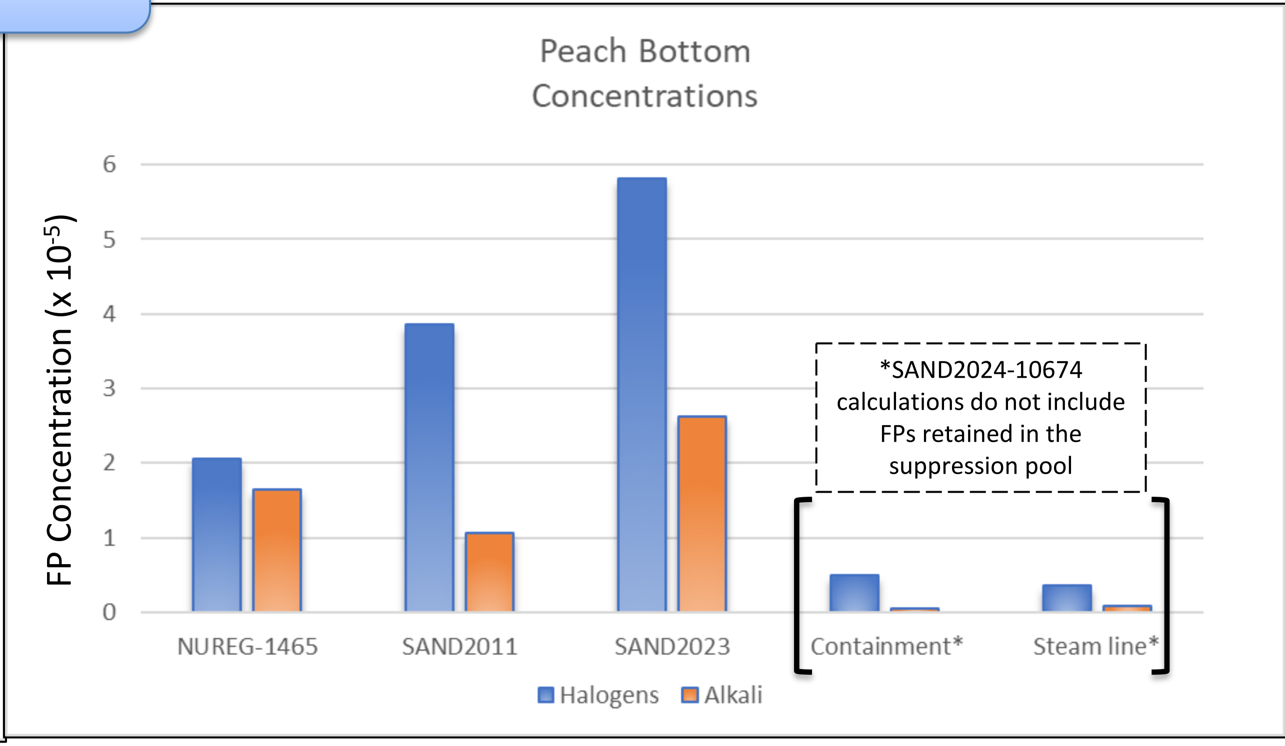
DG-1425 Tables 1.1, 1.2, 1.3 BWR Core Inventory Fractions

	Table 1.1 Containment		Table 1.2 Suppression Pool		Table 1.3 Steamline	
Phase Duration	0.7	7.4	0.7	7.4	0.7	7.4
Group	Gap Release Phase	Early In-Vessel Phase	Gap Release Phase	Early In-Vessel Phase	Gap Release Phase	Early In-Vessel Phase
Noble Gases	1.6×10^{-2}	9.5×10^{-1}	0.0	0.0	2.9×10^{-5}	1.1×10^{-3}
Halogens	1.3×10^{-6}	6.0×10^{-2}	5.0×10^{-3}	6.5×10^{-1}	5.6×10^{-6}	5.1×10^{-5}
Alkali Metals	1.2×10^{-6}	6.0×10^{-3}	5.0×10^{-3}	3.1×10^{-1}	5.1×10^{-6}	1.3×10^{-5}
Tellurium Metals	0	3.8×10^{-2}	3.0×10^{-3}	5.2×10^{-1}	3.2×10^{-6}	2.7×10^{-5}
Barium, Strontium	0	3.0×10^{-4}	6.0×10^{-4}	4.7×10^{-3}	6.1×10^{-7}	2.4×10^{-7}
Noble Metals	0	7.4×10^{-6}	0	6.0×10^{-3}	0	2.4×10^{-7}
Cerium Group	0	0	0	0	0	0
Lanthanides	0	0	0	0	0	0
Molybdenum	0	1.0×10^{-4}	1.9×10^{-5}	1.2×10^{-1}	3.3×10^{-9}	3.0×10^{-6}

BWR Example Fission Product (FP) Concentrations (C_0)

$C_0 = ST / Vol$





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Key Proposed RG 1.183, Rev. 2 Changes

10 CFR 50.46 FFRD Source Term and Acceptance Criteria

DG-1425, Section 3.7, *Assessment of Radiological Consequences due to Fuel Fragmentation Relocation and Dispersal for Analysis of a 10 CFR 50.46 Large-Break LOCA*

10 CFR 50.46 FFRD Source Term and Acceptance Criteria

Alternative 4 of FFRD Regulatory Basis document (88 FR 61986), which reads:

"Develop a Regulatory Framework to Treat Fuel Dispersal Using Traditional Radiological Consequence Analysis Methods and Risk Insights"

DG-1425 note:

This regulatory guide does not provide guidance on demonstrating compliance with the 10 CFR 50.46 requirements or how to estimate the total mass of fuel released because of FFRD.

Purpose:

Treat 50.46 LOCA analyses that predict fuel dispersal as other DBAs with regard to radiological consequences.

Largely consistent with current regulatory framework.

Acceptance Criteria:

Dose-based acceptance criteria of 6.3 rem TEDE at EAB and LPZ and 10 rem TEDE for the Control Room.

Research:

Bounding source term applies a fraction of MHA-LOCA in-containment source term.

(ML21197A069)

Technical Specification:

Establish new technical specification to identify limiting condition for operation as a fission product barrier is challenged for a design basis accident analysis.

Key Proposed RG 1.183, Rev. 2 Changes

Best Estimate Plus Uncertainty

DG-1425, Section 5.1.3, *Assignment of Numerical Input Values*.

Best Estimate Plus Uncertainty

DBA Consequence Analyses

- Mathematically represented by a series of linear, first-order, coupled differential equations.
- Models can be very simplistic to very complex, ranging from a few dozen to thousands of input parameters.
- These input parameters not only model the DESIGN of the system itself, but other aspects as well, such as;
  - meteorological conditions, human performance and health effects to radiation, various physical processes to mitigate the source term, as well as representing testing data, to name a few.
- As such, the uncertainties in these calculations are quite large and results are only presented as point estimates.

Modeling Approach

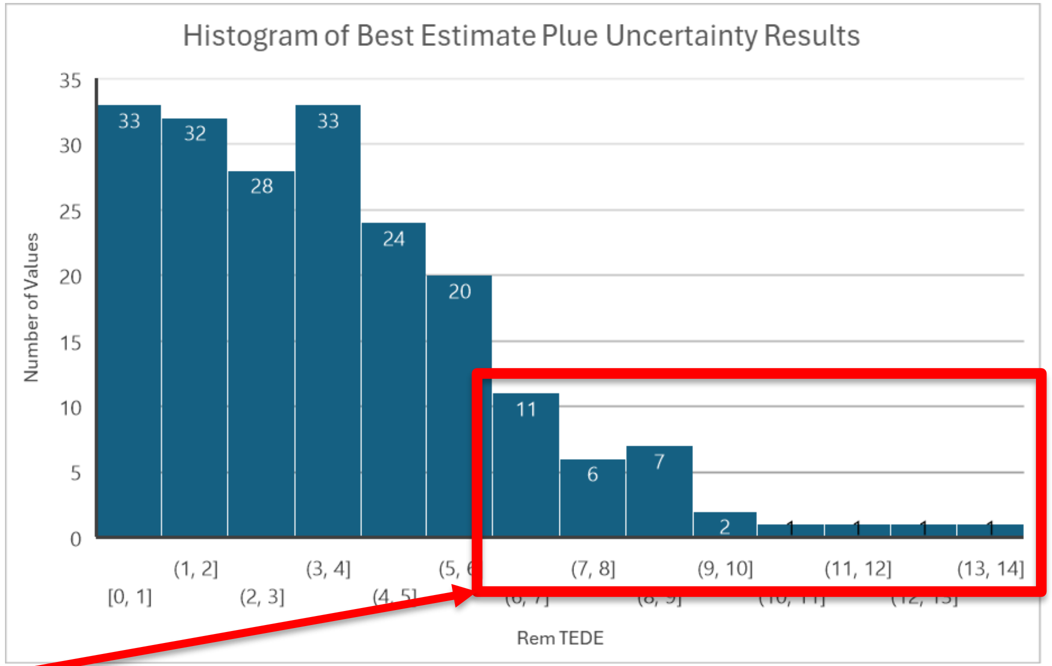
- Staff does not accept design basis accident analyses that credit facility features that:
  - are not safety-related;
  - are not covered by technical specifications;
  - do not meet single-failure criteria,
  - rely on the availability of offsite power, or
  - are best-estimate.
- Modeling practice is to manually adjust individual input parameters in an effort to maximize radiological consequence results.
- Results are presented to the as point estimates with no consideration in uncertainty.

At issue

- Current approach can lead to departures from the facility's design and licensing-bases.
- Burdensome and potentially prone to error.
- Does not holistically assess model uncertainty, which can uncover vulnerabilities during operation.

Best Estimate Plus Uncertainty (Cont.)

- Radiological consequence analyses uncertainty methods have been well-established since the 1980s.
- Computational tools are readily available to perform automated best-estimate plus uncertainty.
  - NRC leverages the Sandia National Laboratory code DAKOTA within the SNAP/RADTRAD code framework.
- Guidance provides high-level modeling approach, which:
  - clarifies the use of safety margin and defense-in-depth,
  - presents quantification methods,
  - facilitates parameter distribution assignment and limitations, and
  - defines acceptable confidence levels.
- Can provide some regulatory relief by modeling systems as designed, while providing important insights for operational improvements.



Key Proposed RG 1.183, Rev. 2 Changes

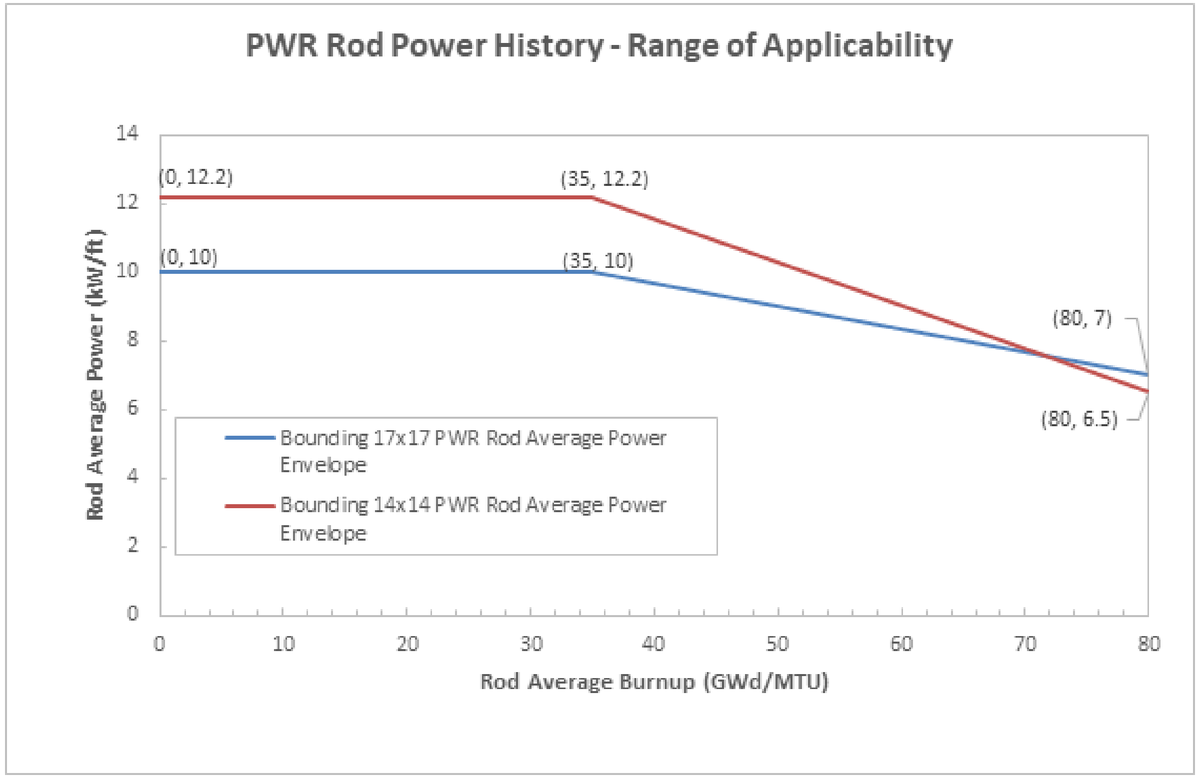
Non-LOCA Gap Release Fractions and Changes to Appendix I

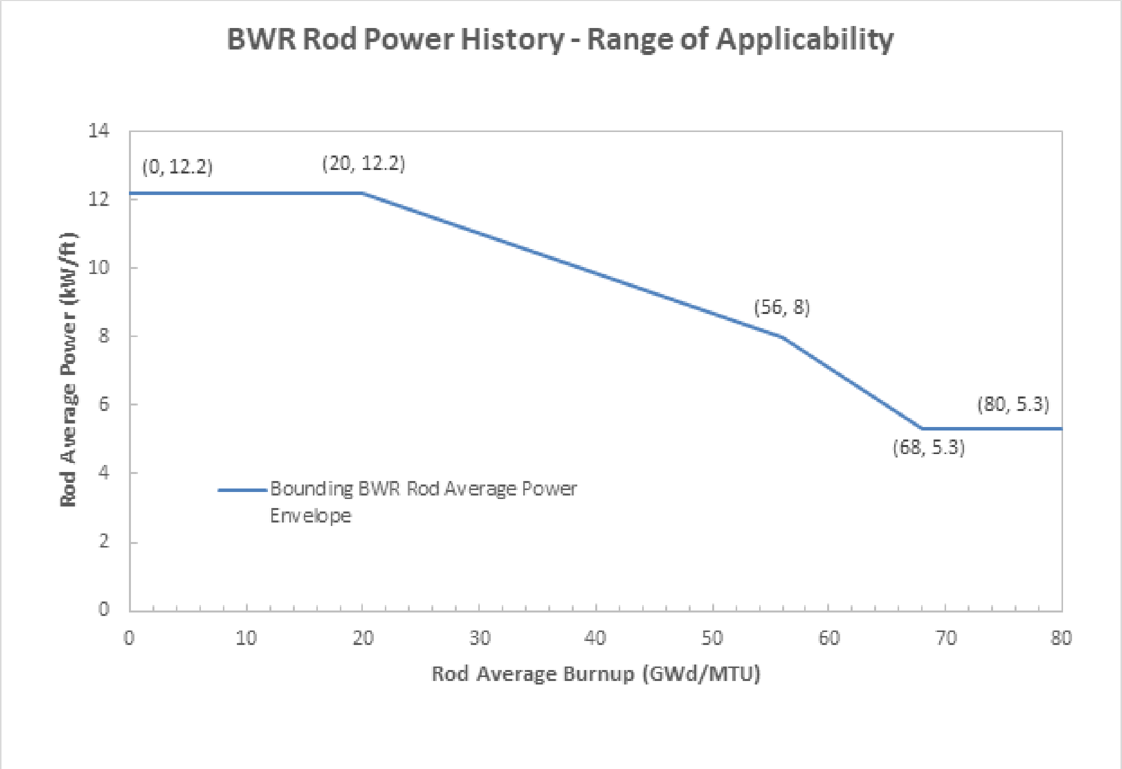
DG-1425, Section 3.2, *Release Fractions*, and Appendix I, *Analytical Technique for Calculating Fuel-Design or Plant-Specific Steady-State Fission Product Release Fractions for Non-Loss-of-Coolant Accident Events*.

Non-LOCA Gap Release Fractions

- The non-LOCA gap release fractions in Tables 3 and 4 have been updated based on the following considerations:
  - A maximum burnup of 80 GWd/MTU
  - A 10 wt% U-235 enrichment
  - Inclusion of values for 17x17 PWR fuel assembly designs

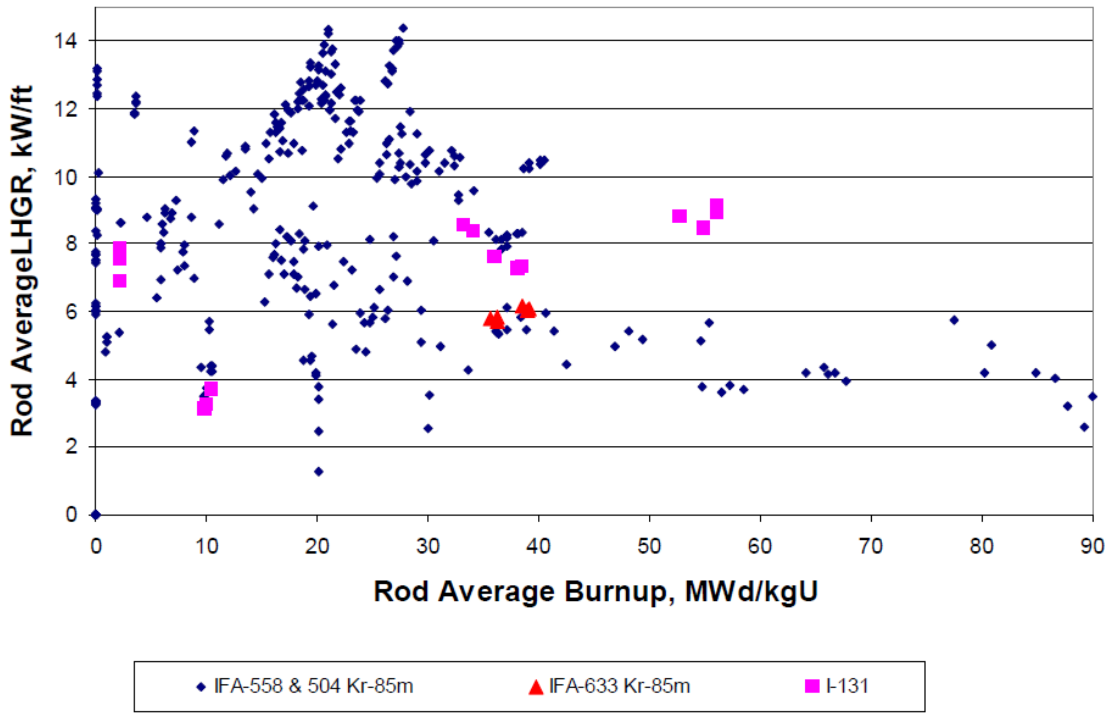
Non-LOCA Gap Release Fractions (Cont.)





Non-LOCA Gap Release Fractions (Cont.)

- The extension of burnups to 80 GWd/MTU required the use of ANS 5.4 standard above the previous limit
  - Database includes some data in this higher burnup range
  - These high burnup rods will be at low power densities



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Non-LOCA Gap Release Fractions (Cont.)

BWR Steady-State Fission Product Release Fractions
Residing in the Fuel Rod Plenum and Gap

Group	Rev. 1 Release Fraction	Rev. 2 Release Fraction
I-131	0.03	0.04
I-132	0.03	0.04
Kr-85	0.32	0.40
Other Noble Gases	0.03	0.04
Other Halogens	0.02	0.03
Alkali Metals	0.16	0.20

Non-LOCA Gap Release Fractions (Cont.)

PWR Steady-State Fission Product Release Fractions
Residing in the Fuel Rod Plenum and Gap

Group	Rev. 1 14x14 Release Fraction	Rev. 2 14 x 14 Release Fraction	Rev. 2 17 x 17 Release Fraction
I-131	0.07	0.09	0.03
I-132	0.07	0.10	0.03
Kr-85	0.40	0.52	0.50
Other Noble Gases	0.06	0.08	0.02
Other Halogens	0.04	0.05	0.02
Alkali Metals	0.20	0.26	0.25

Changes to Appendix I

Updates to Appendix I were generally made to provide clarification based on stakeholder feedback

- Description of how time-dependent radiological consequences could be calculated based on burnup dependent source terms
- Clarification that one of the attributes of an acceptable technique for calculating steady-state gap inventories is that it include an NRC-approved methodology for calculating fission gas release and fuel temperatures

Key Proposed RG 1.183, Rev. 2 Updates

Clarification of MHA as a Design Basis Accident

DG-1425, Section 1.1.3.1, *Various Radiological Source Terms Used by the NRC*, and other locations referring to the MHA-LOCA analysis.

Clarification of the MHA as a DBA

- The term MHA was introduced in guidance to distinguish the DBA dose consequence analysis using the source term footnoted in regulation from the DB LOCA analysis to show § 50.46 compliance. (see DPO 2020-002)
- Hypothetical accident - No need to specify how the source term occurs, e.g.,
    - Postulating a delay in the functioning of safety related equipment,
    - Postulating a pipe rupture, etc.
- For unexplained reasons, substantial core meltdown occurs.
- The purpose of the design basis MHA is to evaluate the effectiveness of the containment and other safety related features of plant design.
- The MHA is evaluated as a DBA; not as a severe accident
    - Credit for demonstrable containment leak rate – fuel melt release to an intact containment
    - Credit for fission product cleanup systems
    - Single failure criteria imposed
    - LOOP, but not SBO

Discussion/Feedback

Acronyms

ACRS – Advisory Committee on Reactor Safeguards

AOR – analysis of record

ATF – accident tolerant fuel

AST – alternative source term

CFR – Code of Federal Regulations

Cs – cesium

DBA – design basis accident

DF – decontamination factor

DG – draft guide

DPO – Differing Professional Opinion

DRA – Division of Risk Assessment

DSS – Division of Safety Systems

EDE – effective dose equivalent

FAST – Fuel Analysis under Steady-State and Transients

FFRD – fuel fragmentation, relocation, and dispersal

FHA – fuel handling accident

FRN – Federal Register Notice

FSAR – final safety analysis report system

GWD/MTU – gigawatt day per metric ton uranium

HBU – high burnup

IE – Increased Enrichment

KW/ft – kilowatt per foot

LAR – license amendment request

LOCA – loss of coolant accident

LWR – light water reactor

MHA – maximum hypothetical accident

MOX – mixed oxide

MSIV – main steam isolation valve

MSL – main steam line

NRR – Office of Nuclear Reactor Regulation

OGC – Office of General Counsel

PWR – pressurized water reactor

RADTRAD – RADionuclide, Transport, Removal, and Dose Estimation

RCS – reactor coolant system

RES – Office of Research

RG – regulatory guide

SBO – Station Blackout

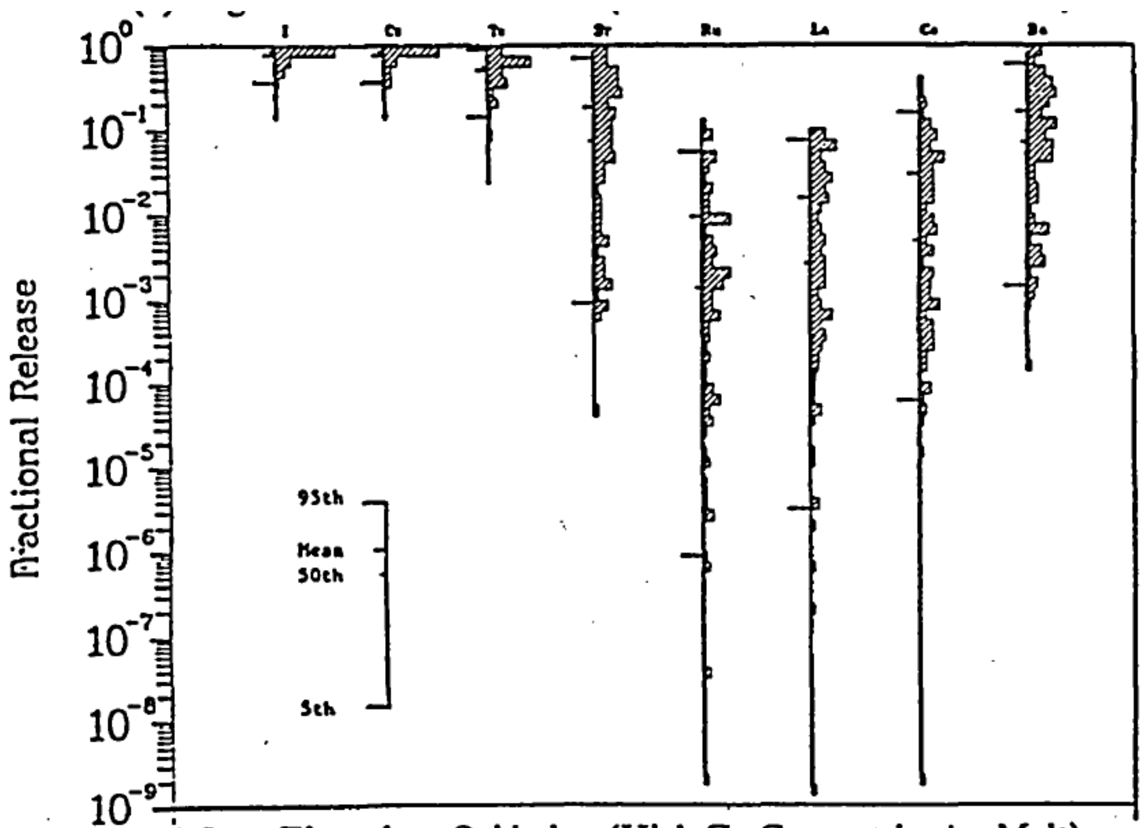
TEDE – total effective dose equivalent

TID – technical information document

TMI – Three Mile Island

Background Slides

Background: In-Containment Fractional Release Uncertainty Distributions



(b) Low Zirconium Oxidation (High Zr Content in the Melt)

Uncertainty Distributions for Total Releases Into Containment PWR, Low RCS Pressure, Limestone Concrete, Dry Cavity, Two Openings After VB, FPART = 1.

Source: NUREG-1465, Appendix A Uncertainty Distributions

Table 4.1 Measures of Low Volatile In-Vessel Release Fractions

Nuclide	Mean	Median	75th percentile
Sr	0.03	0.001	0.006
Ba	0.04	0.003	0.009
La	0.002	0.00003	0.0003
Ce	0.01	0.00006	0.0006

Source: NUREG-1465, Section 4.4, Composition and Magnitude of Release

NUREG-1465

In-Containment Fractional Release Uncertainty Distributions

Table 11. Proposed source term for PWRs using high burnup uranium dioxide fuel. Durations and release fractions for high burnup fuel are shown in bold. Parenthetical quantities are for lower burnup fuel and are included for the purposes of comparison. A dash entry means that a negligible release of the group was predicted to occur during the indicated phase of an accident.

	Gap Release	In-vessel Release	Ex-vessel Release	Late In-vessel Release
Duration (hours)	**0.22 ± 0.04** (0.33 ± 0.12)	**4.5 ± 2.4** (5.3 ± 1.2)	**4.8 ± 1.3** (9 ± 10)	**143 ± 8** (130 ± 20)
Release Fractions of Radionuclide Groups				
Noble Gases (Kr,Xe)	**0.017 ± 0.003** (0.022 ± 0.002)	**0.94 ± 0.01** (0.85 ± 0.05)	**0.011 ± 0.008** (0.08 ± 0.05)	**0.003 ± 0.003** 0.002 ± 0.002
Halogens (Br,I)	**0.004 ± 0.002** (0.007 ± 0.002)	**0.37 ± 0.13** (0.30 ± 0.13)	**0.011 ± 0.008** (0.08 ± 0.03)	0.21 ± 0.16 (0.15 ± 0.11)
Alkali Metals (Rb, Cs)	**0.003 ± 0.001** (0.005 ± 0.002)	**0.23 ± 0.10** (0.23 ± 0.10)	0.02 ± 0.01 (0.03 ± 0.04)	**0.06 ± 0.04** (0.03 ± 0.01)
Alkaline Earths (Sr, Ba)	**0.0006 ± 0.0003** (0.0014 ± 0.0006)	**0.004 ± 0.002** (0.004 ± 0.001)	**0.003 ± 0.002** (0.002 ± 0.001)	-
Tellurium Group (Te, Se, Sb)	**0.004 ± 0.002** (0.007 ± 0.003)	**0.30 ± 0.12** (0.26 ± 0.11)	**0.003 ± 0.002** (0.03 ± 0.01)	**0.10 ± 0.10** (0.10 ± 0.07)
Molybdenum (Mo, Tc, Nb)	-	**0.08 ± 0.03** (0.10 ± 0.02)	**0.01 ± 0.01** (0.10 ± 0.09)	**0.03 ± 0.03** (0.05 ± 0.06)
Noble Metals (Ru, Pd, Rh, etc.)	-	**0.006 ± 0.006** (0.006 ± 0.004)	[0.0025]	-
Lanthanides (Y, La, Sm, Pr, etc.)	-	**1.5 ± 1.2 $\times 10^{-7}$** (1.1 ± 0.9 $\times 10^{-7}$)	**1.3 ± 0.3 $\times 10^{-5}$** (2.6 ± 0.8 $\times 10^{-5}$)	-
Cerium Group (Ce, Pu, Zr, etc.)	-	**1.5 ± 1.2 $\times 10^{-7}$** (1.1 ± 0.9 $\times 10^{-7}$)	**2.4 ± 0.9 $\times 10^{-4}$** (1.0 ± 0.8 $\times 10^{-4}$)	-

SAND 2011-0128

Source: SAND 2011-0128, Table 11

Background: In-Containment Fractional Release Uncertainty Distributions

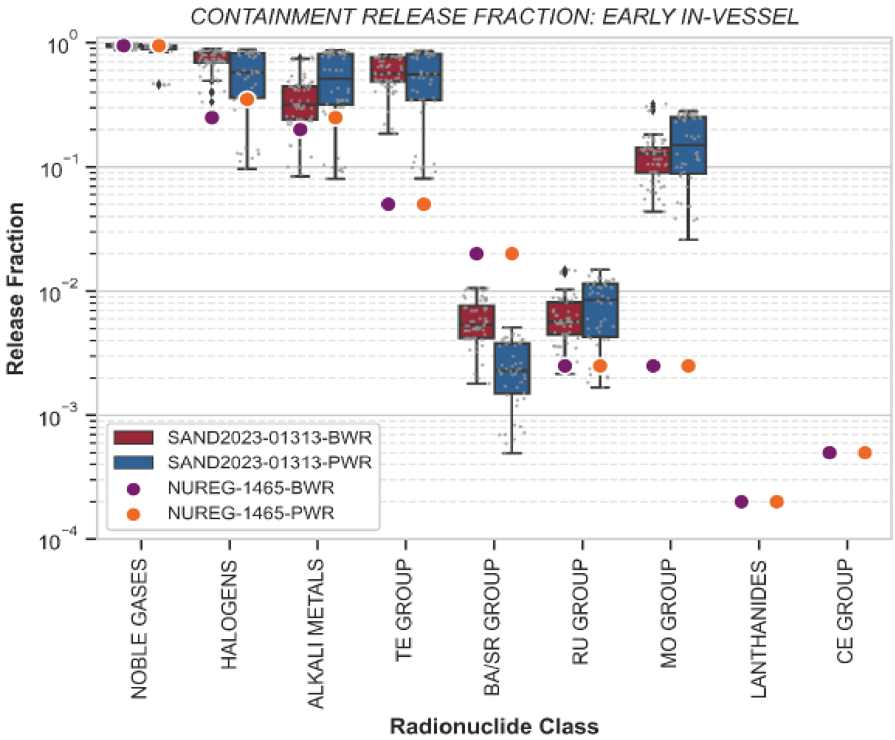


Figure 5-4 **MELCOR calculated fractional release to containment during early in-vessel phase. Releases for noble gases are comparable to NUREG-1465 recommendations. Releases for Ba/Sr group, lanthanides, and Ce group radionuclides are less than NUREG-1465 recommendations. Releases for halogens, alkali metals, Te group, Ru group and Mo group radionuclides are generally greater than NUREG-1465 recommendations.[15]**

SAND 2023-01313

Source: SAND 2023-01313, Section 5, Results