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

Revision of Regulatory Guide 1.183 "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

Micheal Smith, Health Physicist (Project Lead)
NRR/DRA/ARCB
micheal.smith@nrc.gov

November 19, 2020
ML20296A425

Agenda

- Key Messages
- Background
- Regulatory Guide (RG) Update Process
- RG 1.183 Guidance Updates Under Consideration
- Looking Forward
- Feedback/Discussion
- Comments and input from the public

Key Messages

- The NRC staff has restarted efforts to revise RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors."

- The objectives of the revision are to:
  - incorporate lessons learned from recent NRC staff reviews of Alternative Source Term (AST) and Main Steam Line Isolation Valve (MSIV) leakage LARs;
  - incorporate relevant operating experience as well as recent post-Fukushima seismic risk insights and walkdowns;
  - respond to change of regulatory environment (e.g., backfit guidance SRM-SECY-18-0049 & NuScale SRM-SECY-19-0036);
  - make the guidance more useful by considering feedback and comments from licensees;
  - ensure sufficient guidance is in place for licensing advanced light-water reactors (LWRs), accident tolerant fuel (ATF), high-burnup, and increased enrichment fuel; and,
  - incorporate insights from new research activities.

Key Messages (Cont'd)

- RG 1.183 Rev. 0 and Rev. 1 will co-exist as a result of SRM-SECY-18-0049, "Management Directive and Handbook 8.4, "Management of Backfitting, Issue Finality, and Information Collection."

- NRC staff will hold public meetings for external stakeholder engagement on the revision of RG 1.183.

- Publish the draft RG for comment in 4th Quarter CY 2021.

- Final revised RG being issued in 2nd Quarter CY 2022.

Background

Background

- Origin: Footnote to 10 CFR 100.11(a) is a performance-based rule to evaluate the defense-in-depth provided by the containment.
  - TID-14844 Source term provided guidance which assumed the source term is instantaneously available in the containment.

- Radionuclide behavior observed during the TMI accident did not appear at all similar to the TID-14844 source term.
  - NRC initiated research effects in the area of severe accidents which culminate in publication of NUREG-1150.
  - NUREG-1465 source term was derived from the sequences in NUREG-1150.
    - RG 1.183 Rev. 0 adopted the NUREG-1465 early in-vessel fuel melt source term.

Background (cont'd)

- NRC staff developed RG 1.183 Rev. 0 (July 2000) to support implementation of 10 CFR 50.67, "Accident source term"
- RG 1.183 Rev. 0 is applicable to nuclear power reactor applicants and licensees adopting 10 CFR 50.67
  - Limited range of applicability on Non-LOCA release fractions
- RG 1.183 Rev. 0 identified the significant attributes of an acceptable accident AST based on NUREG-1465, "Accident Sources Terms for Light-Water Nuclear Power Plants" (1995)
- RG 1.183 Rev. 0 provides assumptions and methods that are acceptable to the NRC staff for performing design basis radiological analyses using an AST

DG-1199

- In October 2009, the NRC issued for public comment DG-1199 as a proposed Rev. 1 of RG 1.183.
- Staff received 150 public comments
- The reasons for revision of RG 1.183 in DG-1199 were:
  - Providing additional guidance for modeling BWR MSIV leakage,
  - Expand applicability of Non-LOCA release fractions to support modern fuel utilization,
  - Extending the applicability of the proposed RG for use in satisfying the radiological dose analysis requirements contained in 10 CFR Part 52 for advanced LWR design and siting,
  - Providing additional meteorological assumption guidance.

Modern Fuel Utilization

- Since DG-1199 was issued for public comment, NRC issued several license amendments to support modern fuel utilization.
  - Oconee Units 1, 2, and 3 (2019)
  - Shearon Harris (2018)
  - H.B. Robinson (2017)
  - Catawba Units 1 and 2, McGuire Units 1 and 2, Oconee Units 1, 2, and 3 (2016)
  - Diablo Canyon Units 1 and 2 (2015)
- Reinforced need for expanded Non-LOCA release fractions

2019 License Amendment Requests

- In 2019, NRC received several AST LARs requesting increased MSIV leakage
- As a result, work on DG-1199 was postponed to allow NRC staff to incorporate lessons learned, from evaluation of the LARs, into the revised RG 1.183:
  - James A. FitzPatrick Amendment No. 338 for AST, July 21, 2020 (ML20140A070)
  - Quad Cities Nuclear Power Station, Units 1 & 2 – Amendment Nos. 281 and 277 to increase allowable MSIV leakage, June 26, 2020 (ML20150A328)
  - Nine Mile Point Nuclear Station, Unit 2 – Amendment No. 182 to change allowable MSIV leak rates, October 20, 2020 (ML20241A190)
  - Dresden Nuclear Power Station, Units 2 & 3 – Amendments Nos. 272 and 265 to increase allowable MSIV leakage, October 23, 2020 (ML20265A240)

Regulatory Guide Update Process

Regulatory Guide Update Process

- Identify which RGs need to be revised based on:
  - Rulemakings
  - Lessons learned
  - Stakeholder feedback
  - Periodic reviews
- Develop draft RG through internal collaboration
- Draft RG available for public comment (4th Quarter CY 2021)
- Internal staff comment resolution
- Finalize RG package for OGC and ACRS review
- Issue final RG (2nd Quarter CY 2022)

RG 1.183 Guidance Updates Under Consideration

Expected General Updates

- The intent of the NRC staff is for RG 1.183 Rev. 0 and Rev. 1 to co-exist
- With the exception of items discussed later, NRC will consider changes proposed in DG-1199 as modified by public comments.
  - Incorporate updates, new or withdrawn regulatory guidance (i.e., RG 1.194 (meteorology)).
  - Guidance for modern fuel utilization (non-LOCA gap fractions).
  - Changes due to Regulatory Information Summaries (i.e., 06-04, 01-19).
  - Lessons learned from license reviews (i.e., clarify DFs and containment isolation as used in the FHA).
  - Clarify TEDE calculation terminology (i.e., EDEX vs. EDE).
  - Remove environmental qualification guidance from RG and refer to RG 1.89.

ATF, High-Burnup, Extended Enrichment

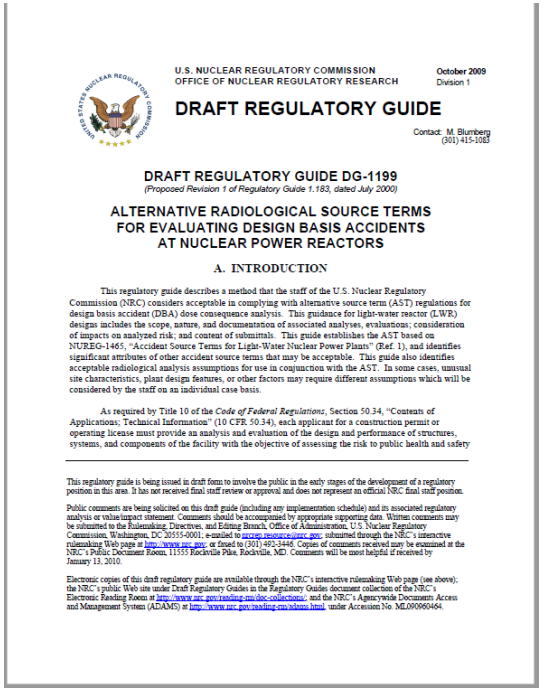
- Applicability of Rev.1 expanded to encompass fuel burnup extension to 68 GWd/MTU (rod average) and ^{235}U enrichments up to 8.0wt%.

- Applicability of Rev.1 to near-term ATF design concepts being considered.
  - Non-LOCA release fractions sensitive to fuel design

- Utilize accident source terms from Sandia National Laboratories report SAND2011-0128, "Accident Source Terms for Light Water Nuclear Power Plants Using High-Burnup of MOX Fuel," and non-loss-of-coolant accident (non-LOCA) source terms based on FAST calculations (similar to those calculated in the proposed update to RG 1.183, Draft Guide 1199).

NRC Memorandum, "Applicability of Source Term for Accident Tolerant Fuel, High Burn Up and Extended Enrichment," dated May 13, 2020, ADAMS Accession Number ML20126G376

Draft Guide DG-1199 Non-LOCA Release Fractions



U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REGULATORY RESEARCH

October 2009
Division 1

DRAFT REGULATORY GUIDE

Contact: M. Blumberg
(301) 415-1083

DRAFT REGULATORY GUIDE DG-1199
(Proposed Revision 1 of Regulatory Guide 1.183, dated July 2000)

ALTERNATIVE RADIOLOGICAL SOURCE TERMS FOR EVALUATING DESIGN BASIS ACCIDENTS AT NUCLEAR POWER REACTORS

A. INTRODUCTION

This regulatory guide describes a method that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable in complying with alternative source term (AST) regulations for design basis accident (DBA) dose consequence analysis. This guidance for light-water reactor (LWR) designs includes the scope, nature, and documentation of associated analyses, evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes the AST based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 1), and identifies significant attributes of other accident source terms that may be acceptable. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the AST. In some cases, unusual site characteristics, plant design features, or other factors may require different assumptions which will be considered by the staff on an individual case basis.

As required by Title 10 of the *Code of Federal Regulations*, Section 50.34, "Contents of Applications; Technical Information" (10 CFR 50.34), each applicant for a construction permit or operating license must provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position.

Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted to the Rulemakings, Directives, and Editing Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, e-mailed to nrcrep.resource@nrc.gov, submitted through the NRC's interactive rulemaking Web page at http://www.nrc.gov; or faxed to (301) 492-3446. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by January 13, 2010.

Electronic copies of this draft regulatory guide are available through the NRC's interactive rulemaking Web page (see above); the NRC's public Web site under Draft Regulatory Guides in the Regulatory Guides document collection of the NRC's Electronic Reading Room at http://www.nrc.gov/reading-rm/doc-collections/; and the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML090980464.

DG-1199 (2011) included the following components:

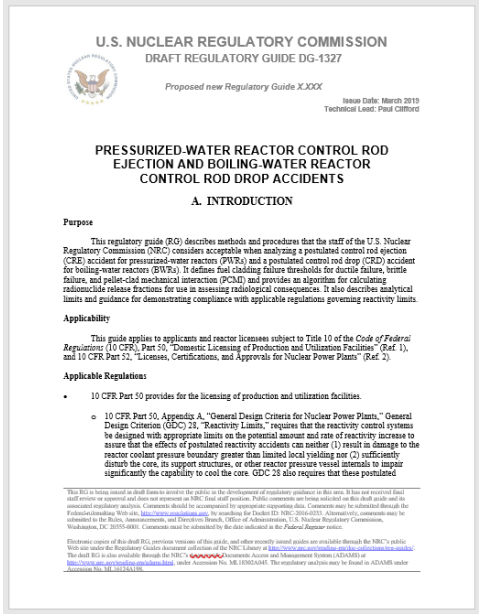
1. Revised Table 3 Non-LOCA release fractions based on expanded power profile
2. New Table 4 RIA transient fission gas release fractions
3. New analytical procedure for revising release fractions

Planned Updates for Non-LOCA Release Fractions

1. Maintain Table 3 release fractions up to 62 GWd/MTU rod average burnup
2. New table for release fractions with expanded applicability up to 68 GWd/MTU rod average burnup
3. Update Table 4 RIA transient fission gas release to include burnup-dependent correlations
4. Update example calculation based on FAST

DG-1327 CRE/CRD Public Comments (2019)

- Many of the planned changes to RG 1.183 were included in draft regulatory guide DG-1327
  - Revised Table 3 Non-LOCA gap fractions using new version of FAST fuel rod thermal-mechanical code
  - Revised Table 4 RIA transient fission gas release with BU-dependent correlations
  - Acceptable analytical procedure for revising Non-LOCA gap releases
- Public comments received on these topics will be reflected in RG 1.183



U.S. NUCLEAR REGULATORY COMMISSION
DRAFT REGULATORY GUIDE DG-1327

Proposed new Regulatory Guide X.XXX

Issue Date: March 2019
Technical Lead: Paul Clifford

PRESSURIZED-WATER REACTOR CONTROL ROD EJECTION AND BOILING-WATER REACTOR CONTROL ROD DROP ACCIDENTS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes methods and procedures that the staff of the U.S. Nuclear Regulatory Commission (NRC) considers acceptable when analyzing a postulated control rod ejection (CRE) accident for pressurized-water reactors (PWRs) and a postulated control rod drop (CRD) accident for boiling-water reactors (BWRs). It defines fuel cladding failure thresholds for ductile failure, brittle failure, and pellet-clad mechanical interaction (PCMI) and provides an algorithm for calculating radionuclide release fractions for use in assessing radiological consequences. It also describes analytical limits and guidance for demonstrating compliance with applicable regulations governing reactivity limits.

Applicability

This guide applies to applicants and reactor licensees subject to Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 1), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2).

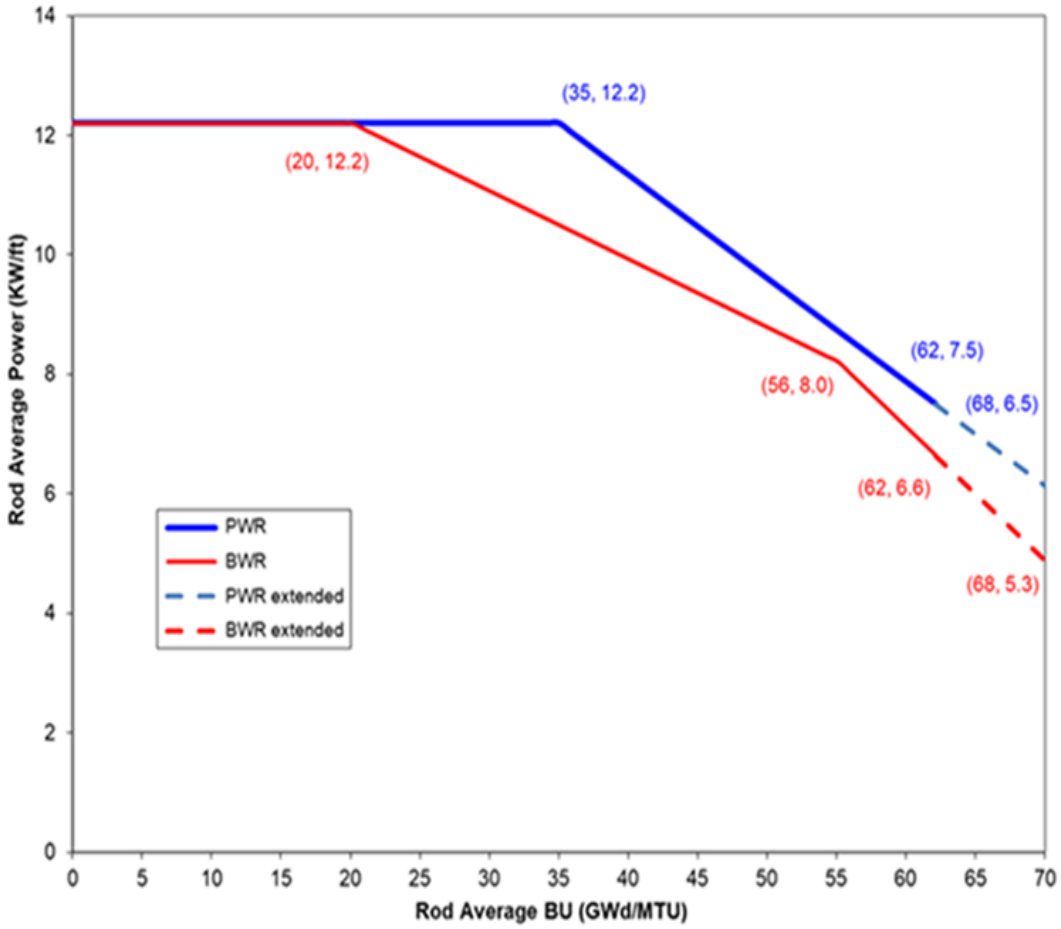
Applicable Regulations

- 10 CFR Part 50 provides for the licensing of production and utilization facilities.
  - 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criterion (GDC) 28, "Reactivity Limits," requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. GDC 28 also requires that these postulated

This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this draft guide and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal eRulemaking Web site, http://www.regulations.gov, by searching for Docket ID: NRC-2016-0233. Alternatively, comments may be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this draft RG, previous versions of this guide, and other recently issued guides are available through the NRC's public Web site under the Regulatory Guides document collection of the NRC Library at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/. The draft RG is also available through the NRC's ADAMS Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML18302A045. The regulatory analysis may be found in ADAMS under Accession No. ML16234A198.

FAST Calculations (1)



- Extended rod average power profiles out to 68 GWd/MTU
- Preliminary calculations show no increase in release fractions
- Axial Power Distribution:
  - Sweeping (3 cycles) AXPDs with the following peak Fz peaking factors
    - PWR Peak Fz = 1.144
    - BWR Peak Fz = 1.228
- **Is this sufficient to support future reloads?**

Table A.1. FRAPCON-3.3 Input Parameters for PWR and BWR Analyses

Description of Design Parameter	PWR 14 x 14	BWR 9 x 9
Pitch (mm, *in*)	14.7, *0.58*	13.0, *0.510*
Cladding OD (mm, *in*)	11.2, *0.44*	10.8, *0.424*
Cladding Thickness (mm, in)	0.737, *0.029*	0.711, *0.028*
Cladding ID (mm, *in*)	9.70, *0.382*	9.35, *0.355*
Diametral Gap Thickness (mm, *in*)	0.191, *0.0075*	0.203, *0.008*
Fuel Pellet Diameter (mm, *in*)	9.51, *0.374*	9.14, *0.360*
Plenum Spring Diameter (mm, *in*)	9.51, *0.374*	9.14., *0.360*
Pellet Length (mm, *in*)	12.7, *0.5*	11.2, *0.44*
Dish Diameter (mm, *in*)	4.57, *0.18*	5.81, *0.2288*
Dish Depth (mm, *in*)	0.305, *0.012*	0.216, *0.0085*
Plenum Length (mm, *in*)	254, *10*	245, *9.64*
Turns in Plenum Spring	30	29
Plenum Spring Wire Diameter (mm, *in*)	1.32, *0.052*	1.4, *0.055*
Helium Fill Gas Pressure (MPa, *psi*)	2.07, *300*	1.03, *150*
Active Fuel Length (m, *in*)	3.66, *144*	3.81, *150*
System Coolant Pressure (MPa, *psi*)	15.5, *2250*	7.27, *1055*
Coolant Inlet Temperature (°C, *°F*)	288, *550*	277, *530*
Coolant Flow Rate ($x10^3$ kg/s-m^2, $x10^6$ lb/hr-ft^2)	3.60, *2.65*	1.67, *1.23*
Enrichment (atom %)	4.95	5
Pellet Density (% TD)	96	96
Limit on Pellet Density Increase (% TD)	0.7	0.7
Fuel Surface Roughness (μm, *in*)	1.6, *$6.3x10^{-5}$*	1.5, *$5.9x10^{-5}$*
Cladding Surface Roughness (μm, *in*)	1.0, *$3.9x10^{-5}$*	0.9, *$3.6x10^{-5}$*
Cladding Material	ZIRLO™	Zircaloy-2
Cold Work (%)	50	0

FAST Calculations (2)

- Generic fuel rod parameters for bounding PWR and BWR designs
- **Should there be separate PWR and BWR tables?**
- **How to address BWR part-length fuel rods?**

Revised Fuel Handling Accident

- Revisited the original studies forming the technical basis for the FHA and incorporate updated information.
- Model improvements established from the current understanding of reactor fuel pin physics and iodine chemistry under the environmental conditions in which fuel handling operations are taking place.
- Concluded that considerable margin exists regarding the scrubbing effects of iodine in the spent fuel or reactor pool and that the current staff DBA FHA fission product transport model can be refined while still maintaining conservatism.
- Reference: Memo from RES to NRR, “Closeout to Research Assistance Request for Independent Review of Regulatory and Technical Basis for Revising the Design-basis Accident Fuel Handling Accident,” November 23, 2019 (ML19270E335)

Additional Method for Aerosol Deposition Models

- Staff is considering an additional method for aerosol deposition models
- Staff is addressing issues in RIS 2006-04, “Experience with Implementation of Alternative Source Terms” (considering reconstitution of AEB-98-03 and reviewing the multigroup method).
- Regulatory position in Rev. 0 continues to be acceptable. As a result, RG 1.183 Rev. 0 and Rev. 1 will co-exist.
- Over the last 10 years no applicant or licensee has adopted the methodology from SAND2008-6601, “Analysis of Main Steam Isolation Valve Leakage in Design Basis Accident Using MELCOR 1.8.6 and RADTRAD.”
- There have been no communications that applicants or licensees intend to adopt the SAND2008-6601 methodology.
- NRC staff plans to consider stakeholder input/feedback to inform the NRC’s decision on what methodology to include in RG 1.183 Rev. 1.

Lessons Learned from Licensing Reviews

- Staff are considering whether to clarify:
  - the expectations for containment spray in BWR drywells/containments (i.e., Rev. 0 Appendix A Assumption 3.3)
  - the expectations for performing and using sensitivity analysis (i.e., Rev. 0, RPs 1.3.3 and 5.1.3)
  - if crediting pathways should be consistent with design requirements for safety (i.e. technical specifications, safety related, Rev. 0, RP 5.1.2)
  - RG wording to assume that a LOOP is coincident with a turbine trip (not with initiation of the accident)(i.e., Rev. 0, App. F & G Assumption 5.4)
  - the expectations for BWR MSIV Leakage LOCA analysis assumptions with respect to pipe breaks

Use of Risk and Engineering Insights

- Update the expectations for use of risk insights as directed in SRM-SECY-19-0036.
- NRC staff has developed a technical assessment on this topic considering 20+ years of operational and seismic risk insights.
  - Assessment will be publicly available via the NRC's Interim Staff Guidance process.
- Four issued safety evaluations are supported by risk and engineering insights.
- Staff is exploring streamlined approach for quantitative credit for hold-up and retention of MSIV leakage within the power conversion system for BWRs.
  - **Is there interest in a streamlined approach?**
  - **What portion(s) of the alternative pathway justification in Rev. 0 are resource intensive (availability of pathway, seismic "robustness" steps, both?)**

Additional Considerations

- Consider revising footnote 7 which provides an incorrect method to convert thyroid dose to TEDE
  - Implies a back-of-the-envelope calculation appropriately converts between ICRP 2 and ICRP 26/30 dosimetry methodologies.
  - There is no simple methodology to convert between these two systems of dosimetry.
  - To correctly calculate the radiological dose consequences for design basis accidents the appropriate dose methodology (and DCFs) must be applied.

Looking Forward

- Consider feedback from stakeholders
- Develop updated draft RG 1.183 Rev. 1
- Hold additional public meeting 1st Quarter CY 2021
- Draft RG 1.183 Rev. 1 issued for public comment (4th Quarter CY 2021)
- Staff review and disposition of public comments
- Update of draft RG 1.183 Rev. 1 as necessary
- ACRS and OGC-review of final draft (1st Quarter CY 2022)
- Issuance of RG 1.183 Rev. 1 (2nd Quarter CY 2022)

Discussion/Feedback

Questions/Comments?

Mark Blumberg, Senior Reactor Engineer (Technical Lead)
NRR/DRA/ARCB
mark.blumberg@nrc.gov

Micheal Smith, Health Physicist (Project Lead)
NRR/DRA/ARCB
micheal.smith@nrc.gov